March 2, 2006

Mr. James Scarola, 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 REGARDING MAIN STEAM ISOLATION VALVE LEAKAGE LIMIT (TAC NOS. MC8106 AND MC8107)

Dear Mr. Scarola:

The Commission has issued the enclosed Amendment No. 239 to Facility Operating License No. DPR-71 and Amendment No. 267 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP). The amendments are in response to your application dated August 11, 2005, as supplemented by letters dated October 11, November 16, and December 12, 2005, and February 7, 2006.

Carolina Power & Light Company (CP&L, the licensee) requested changes to the technical specifications (TS) for the Brunswick Steam Electric Plant, Unit Nos. 1 and 2.

The proposed changes to the BSEP TS revise the Surveillance Requirement 3.6.1.3.9 to increase the allowable main steam isolation valve (MSIV) leakage rate. Specifically, the limit is revised from an allowable leakage rate of less than or equal to 11.5 standard cubic feet per hour (scfh) through each MSIV to less than or equal to 100 scfh through each main steam line (MSL) with the combined leakage of the four MSLs being less than or equal to 150 scfh. In support of this change, CP&L will incorporate automatic initiation of the control room emergency ventilation (CREV) system. The auto-start signal will be provided from the Secondary Containment Isolation logic. The auto-start function is designed to initiate the CREV system well within the assumed 2 minutes in the revised analysis (i.e., before fuel gap release initially reaches the external environment).

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

Sincerely,

/RA/

Brenda L. Mozafari, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 239 to
   License No. DPR-71
2. Amendment No. 267 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 biweekly *Federal Register* Notice.

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Brenda L. Mozafari, Senior Project Manager
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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. 239
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 11, 2005, as supplemented by letters dated October 11, November 16, and December 12, 2005, and February 7, 2006, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 239, 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael L. Marshall, Jr., Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 2, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 239

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. Bases page changes provided for information only.

Remove Pages	Insert Pages
3.3 - 63	3.3 - 63
3.3 - 64	3.3 - 64
3.3 - 65	3.3 - 65
3.3 - 66	3.3 - 66
3.3 - 67	3.3 - 67
3.3 - 68	3.3 - 68
3.3 - 69	3.3 - 69
3.3 - 70	3.3 - 70
3.3 - 71	3.3 - 71
3.3 - 72	3.3 - 72
3.3 - 73	3.3 - 73
3.3 - 74	3.3 - 74
3.6 - 13	3.6 - 13

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 267
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 11, 2005, as supplemented by letters dated October 11, November 16, and December 12, 2005, and February 7, 2006, 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-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. 267, 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael L. Marshall, Jr., Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 2, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 267

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
3.3 - 63	3.3 - 63
3.3 - 64	3.3 - 64
3.3 - 65	3.3 - 65
3.3 - 66	3.3 - 66
3.3 - 67	3.3 - 67
3.3 - 68	3.3 - 68
3.3 - 69	3.3 - 69
3.3 - 70	3.3 - 70
3.3 - 71	3.3 - 71
3.3 - 72	3.3 - 72
3.3 - 73	3.3 - 73
3.3 - 74	3.3 - 74
3.6 - 13	3.6 - 13

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 239 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 267 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 11, 2005, as supplemented by letters dated October 11, November 16, and December 12, 2005, and February 7, 2006, the Carolina Power & Light Company (CP&L, the licensee) submitted a request for changes to the Brunswick Steam Electric Plant, Units 1 and 2 (BSEP), Technical Specifications (TS). The proposed changes would revise the TS Surveillance Requirement 3.6.1.3.9 to increase the allowable main steam isolation valve (MSIV) leakage rate. Specifically, the limit is revised from an allowable leakage rate of less than or equal to 11.5 standard cubic feet per hour (scfh) through each MSIV to less than or equal to 100 scfh through each main steam line (MSL) with the combined leakage of the four MSLs being less than or equal to 150 scfh. In support of this change, CP&L will incorporate automatic initiation of the control room emergency ventilation (CREV) system. The auto-start signal will be provided from secondary containment isolation logic. The auto-start function is designed to initiate the CREV system well within the assumed 2 minutes in the revised analysis (i.e., before fuel gap release initially reaches the external environment).

The licensee stated that the refurbishment of an MSIV necessary to meet the current 11.5 scfh leakage criterion is a labor-intensive effort, which results in unnecessary personnel exposure and expenditure of resources. In addition, the licensee stated that the actual leak rate observed for any one MSIV during leakage testing has been demonstrated to be influenced by a number of factors such as the MSL in which the subject MSIV is installed (i.e., the effects of asymmetric flows due to pipe configurations), the actual as-installed valve stem orientation, and the temperature of the valve when the valve is closed and the leak rate testing is performed. Based on a review of long-term MSIV leak rate history, the licensee concluded that a 100-scfh limit for an individual MSL and a maximum combined leakage of 150 scfh for all MSLs would significantly reduce the amount of rework, and so would avoid incurring the exposure attendant with maintaining the current allowable leakage rates. In addition, the licensee stated that the reduction in rework would also result in less wear induced by maintenance activities on the MSIVs.

The letters dated October 11, November 16, and December 12, 2005, and February 7, 2006, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) staff evaluated the radiological consequences of the increase in MSIV leakage on the design-basis loss-of-coolant accident (DB LOCA) as proposed by the licensee against the dose criteria specified in Section 50.67(b)(2) of Title 10 of the *Code of Federal Regulations* (10 CFR); these criteria are 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the postulated fission product release, and 5 rem TEDE in the control room for the duration of the postulated fission product release.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed design-basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in10 CFR 50.67, as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, Standard Review Plan (SRP) 15.0.1, and General Design Criterion (GDC) 19. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards:

10 CFR Part 50.67, "Accident Source Term"

- 10 CFR Part 50, Appendix A, "General Design Criterion for Nuclear Power Plants": GDC 19, "Control Room"
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
- NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems"
- SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term”

3.0 TECHNICAL EVALUATION

As stated in Reference 1, there are four MSLs installed on each BSEP unit. Each MSL penetrates the primary containment and connects the nuclear steam supply to the steam turbine-generator. Each MSL is isolated by a pair of fast-closing MSIVs. These valves serve to isolate the reactor coolant system in the event of steam line breaks outside primary containment, a DB LOCA, or other events requiring containment isolation.

The licensee referred to 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" (Reference 2), as 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 (Reference 3), determined that the BWROG approach of using 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.

On May 30, 2002, the NRC staff issued Amendments Nos. 221 and 246 (Reference 4) to the Facility Operating Licenses for the BSEP, Units 1 and 2, respectively. These amendments revised the BSEP licensing basis to replace the accident source term used in the LOCA, the MSL break (MSLB) accident, and the control rod drop accident design basis analyses with an alternative source term (AST) in accordance with 10 CFR 50.67, "Accident Source Term."

As documented in Section 3.0 of the SE for the BSEP AST license amendments (Reference 4), the NRC staff concluded that the main steam system piping and components which comprise the alternate leakage treatment (ALT) system (i.e., ALT path) are seismically rugged and, thus, able to perform the safety function of an MSIV leakage treatment system. In the present license amendment request (LAR) the licensee has not requested any alteration in the design or function of the ALT path to support the revision of the MSIV leakage criteria. Therefore, this SE does not address the functional design of the ALT path or the capability to establish the ALT path under postaccident conditions.

The licensee stated that the atmospheric dispersion factors that are used in the analyses performed to support the evaluation of increased MSIV leakage are not affected by the proposed change and remain the same as those reviewed and approved for the BSEP AST amendment. Therefore, the affects of atmospheric dispersion are unchanged for the purpose of calculating doses in the control room and technical support center, and at the EAB and LPZ and are not addressed in this SE.

Based on the information provided by the licensee the radiological analysis submitted in conjunction with the request for increased limits on MSIV leakage includes revisions to the following elements of the LOCA analysis:

- Control room direct shine dose contributions
- Initiation of the control room emergency ventilation system
- Control room unfiltered inleakage
- Reactor building positive pressure period
- Radionuclide transport model

All other analysis inputs, assumptions and models are the same as previously found acceptable to the NRC staff in Reference 4.

3.1 Control Room Direct Shine Dose Contributions

The licensee revised the analysis of the control room dose based on recalculated direct shine contributions from the reactor building cloud, the core spray discharge piping, the standby gas treatment system (SGTS) filter radiological loading, and the CREV filter radiological loading.

In the original evaluation, the licensee credited the existing 8-inch concrete wall between the CREV filter and the control room. In the updated CREV filter radiological loading calculation, the licensee assumes an additional 2 inches of steel plate between the CREV filter and the control room. The licensee has made a commitment to install shielding, consistent with the assumptions of this calculation, prior to implementation of the proposed amendment.

In the revised direct shine calculations, the licensee performed more accurate modeling of the actual geometric configurations of the radiological direct shine sources (i.e., SGTS filter, reactor building cloud, and core spray piping) with respect to the control room. The licensee stated that the revised modeling better represents plant structures, equipment layout, and piping configurations. The licensee is crediting an additional 2-foot thick concrete wall for shielding, located in the control room interior, which was not originally included in the analysis. In addition, in the revised model the licensee considers two target points for estimating dose, rather than one as in the prior model. The two points represent the nearest point for an operator positioned in the back panel area of the control room to the accident unit and the nearest point of an operator positioned in the main control panel area to the accident unit, respectively. The occupancy factors applied in the original dose analysis calculation were unchanged for the revised calculations. The licensee combined the dose calculated at each of the two target points such that the total dose value used is the summation of 25 percent of the dose calculated at the back panel position and 100 percent of the dose calculated at the main control panel position.

The licensee stated that this arrangement of dose analysis locations differs from the previous analysis in that the locations chosen represent actual crew work locations, whereas the previous analysis was based on evaluating the highest dose point in the control building. Because of symmetry in the control room layout, the licensee only evaluated one unit in order to address potential accident exposure from either unit. The licensee stated that there are no control functions or activities that take place in the highest dose point area previously analyzed for the control room occupancy 30-day dose. The licensee further stated that the area previously analyzed is not essential to control room functioning during either normal or abnormal operating conditions. In addition, the licensee stated that current emergency plans call for a health physics technician to be posted in the control room to monitor exposure rates.

Notwithstanding the rational discussion of realistic operator receptor locations within the control room, the incorporation of this methodology in the dose analysis prompted the NRC staff to investigate, by way of a request for additional information, what the dose would be at the highest dose point in the control building. Additionally, the NRC staff requested more information regarding the licensee's method of limiting the dose to control room operators during an accident condition.

The licensee provided, by letter dated November 16, 2005 (Reference 6), the results of a detailed assessment using the assumptions outlined in the August 11, 2005, LAR, which determined that the control room occupancy 30-day total dose for the highest dose point in the control building would be 4.7 rem TEDE for the DB LOCA. The licensee's response included a

detailed comparison table providing a summary of the various dose contributions to the 30-day LOCA control room dose for the highest dose location case. In addition, the licensee provided the results from the calculation supporting this LAR, as well as results from the AST calculation previously reviewed and found acceptable by the NRC staff in support of the amendments in Reference 4. As stated by the licensee, and depicted in the table, the only dose contribution that differs in the "highest dose location" case as compared to the case provided in support of this LAR is the Emergency Core Cooling System piping shine contribution. This dose contribution is due to gamma shine from a core spray discharge line located to the west of the control room. The licensee assumed that operators were positioned at the "highest dose location" with respect to this core spray discharge line, which resulted in an increase in this dose contribution. Conservatively, the licensee did not reduce the doses received from other sources of direct shine due to the assumed receptor location chosen to maximize the contribution from core spray discharge line. For example, the dose contribution from the accident related fission product buildup on the CREV filter for the highest dose location case was not reduced to account for the receptor location positioned some distance away from the filters near the exterior wall at either the northwest or southwest corner of the control room. The licensee stated that the "highest dose location" actually refers only to the dose received from the core spray discharge line shine, whereas the underlying assumption for CREV filter dose has always been and remains an individual standing directly beneath (i.e., physically closest to) the filter. In essence, the licensee conservatively maximized the dose consequence for each of the direct shine contributions even though it would be physically impossible for any one operator to be located in the various positions concurrently to accumulate the total dose reported as the “highest dose location.” The licensee stated that this conservative method of dose summation is consistent with previous practice.

Given the conservative approach followed in the calculation of the “highest dose location” and the fact that the summation of the doses from each of the most conservative receptor locations, including shine dose and inhalation dose, results in a calculated dose below the 5 rem TEDE criterion, the NRC staff finds that the control room dose of 4.4 rem TEDE as originally reported in Reference 1, is acceptable.

In the response dated November 16, 2005, the licensee provided additional information regarding the methods of limiting the dose to control room operators during an accident condition. The licensee stated that closeout activities associated with the adoption of increased limits on MSIV leakage will include placing placards in the northwest and southwest corners of the control room to identify the potential elevated dose rates in these areas in the event of a corresponding unit LOCA. The NRC staff recognizes these actions as commendable and is satisfied that these actions will serve to ensure that postaccident control room doses are maintained as low as reasonably achievable. The NRC staff also recognizes that the licensee is not relying on these exposure control activities to meet the regulatory limit of 5 rem TEDE, even assuming the most conservative worst case receptor locations.

3.2 Initiation of the Control Room Emergency Ventilation System

The licensee is proposing to provide an additional automatic initiation signal for the CREV system. The following description of the BSEP CREV system and the discussion of the addition

of an initiation signal from the secondary containment isolation logic is based on information provided to the NRC staff in Reference 1.
The CREV system is designed to provide a radiologically-controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Each of the two independent CREV subsystems is capable of fulfilling the stated safety function. Currently, the CREV system instrumentation has two independent sets of trip logic, either of which can initiate the CREV system; however, the BSEP CREV system design only permits one CREV train to be in service at a time. Each trip system receives input from the two Control Building Air Intake Radiation - High Function channels. The Control Building Air Intake Radiation - High Function is arranged in a one-out-of-two logic for each trip system. The CREV system initiates a filtered pressurization flow of 1500 cubic feet per minute (cfm) into the control room to provide breathing air and to reduce the influx of unfiltered radioactive material entering the control room environment. In addition the CREV system includes a filtered recirculation flow of 400 cfm to minimize the consequences of radioactive material entering the control room environment.

The LOCA analysis performed in support of the BSEP AST license amendments assumed manual operator action to start one CREV filter train at 20 minutes after accident initiation. In support of the MSIV leakage rate change, the revised analysis assumes an automatic start of one CREV filter train within the first 2 minutes from event initiation, which is prior to the time that an assumed initial gap activity release would reach the ambient environment.

The TS for the CREV system instrumentation have been changed to be consistent with the Standard TS, Revision 3 of NUREG-1433, “Standard Technical Specifications, General Electric Plants, BWR/4". In addition, associated changes were made to the TS Bases, consistent with the Standard TS. The addition of an initiation signal from the secondary containment isolation logic will enhance the ability of the CREV to provide protection to control room personnel and is, therefore, acceptable to the NRC staff.

3.3 Control Room Unfiltered Inleakage

As stated in Reference 4, because inleakage had not yet been quantified, the licensee used a bounding value of 10,000 cfm for the control room unfiltered inleakage in the AST submittal. In the revised analysis, the licensee used a control room unfiltered inleakage value of 2000 cfm. The revised value of 2000 cfm is based on tracer gas testing (as documented in Reference 7), performed subsequent to the AST submittal, to quantify the control room unfiltered inleakage. The results of the tracer gas tests that are pertinent to the revised LOCA analysis with increased MSIV leakage are shown in the following table. The revised design basis unfiltered inleakage value of 2000 cfm is well above the measured values in all cases examined and, therefore, the NRC staff finds the revised value acceptable for use in the control room radiological dose analyses.

Applicable Tracer Gas Test Results from Reference 7

Mode	Fan Alignment	Measured Inleakage (cfm)	Design Basis Inleakage (cfm)
A Train Radiation Protection	ID & 2E	800 +/- 125	2,000
B Train Radiation Protection	ID & 2E	708 +/- 118	2,000
A Train Radiation with Cable Spread Room Supply Fans	ID & 2E	869 +/- 146	2,000

3.4 <u>Reactor Building Positive Pressure Period</u>

The analytical model prepared by the licensee in support of this LAR includes an increase in the assumed secondary containment positive pressure period (PPP) from 5 minutes to 10 minutes. This revision results in a corresponding increase in the time period during which the released gap activity is assumed to leak into secondary containment without the benefit of effective reactor building negative pressure control being established. The licensee stated that this revised assumption was implemented as a conservative measure to increase the analytical margin in the required performance of the SGTS.

Increasing the assumed secondary containment PPP from 5 minutes to 10 minutes has an adverse impact on the calculated DBA doses. The increase in the calculated doses is due to the increased time period during which radioactive material that leaks into the reactor building does not have the benefit of filtration by the standby gas treatment system or the more favorable atmospheric dispersion resulting from an elevated release of the filtered effluent. The doses as calculated with the revised PPP of 10 minutes remain below the applicable regulatory limits and, therefore, the NRC staff finds that the increase in the assumed PPP is acceptable for the purpose of providing an increased margin in the required performance of the SGTS.

3.5 <u>Radionuclide Transport Model</u>

The evaluations performed by the licensee in support of the request for increased MSIV leakage limitations have been updated based on new information and approaches, including the adoption of NUREG/CR-6604, Supplement 2, dated October 2002, "RADTRAD: A Simplified Model for Radionuclide Transport and removal and Dose Estimation" (RADTRAD Version 3.03, Reference 5). Previously, the licensee used RADTRAD Version 3.02 for the analysis in support of the BSEP AST amendment request (Reference 4).

3.5.1 Torus Volume Mixing Assumptions

With the adoption of RADTRAD Version 3.03, the licensee’s transport model has been modified to include the incorporation of the torus volume. The revised transport model now includes a recirculation pathway between the suppression chamber air space and the modeled volume representing the drywell. The recirculation pathway is given an artificially high flow in order to ensure homogeneous mixing between the two volumes. Credit for mixing between the drywell and torus volumes is not employed in the model during the initial 2 hours of the accident

condition. The NRC staff finds this assumption acceptable since the DB LOCA assumes that the fission product release is completed at the end of the early in-vessel release phase.

3.5.2 Pressure Profile Assumptions

The licensee used the same pressure profile assumptions in the re-analysis as those applied in the original AST analysis. BSEP uses a containment atmosphere control (CAC) containment atmosphere dilution (CAD) system that gradually pressurizes the primary containment following the initial pressure reduction. Therefore, the licensee did not credit the pressure reduction of the primary containment leakage rate and MSIV leakage rate to the maximum allowed by RG 1.183, Appendix A, Sections 3.7 and 6.2. Maintaining consistency with the AST analysis, the licensee conservatively set the primary containment leakage rate to 80 percent of the initial 24-hour leak rate value for the remainder of the 30-day duration.

The licensee has determined, based on the previous AST analysis, that assuming a steady state MSIV leakage corresponding to the 25 pounds per square inch gauge test pressure bounds the process of initial pressure reduction followed by gradual pressure increase resulting from the operation of the CAC/CAD system. This simplifying and conservative assumption is retained in the revised analysis.

3.5.3. Temperature Profile Assumptions

In the analysis of iodine deposition in the MSLs, deposition is enhanced and resuspension is minimized with lower temperatures. In the AST analysis, CP&L conservatively assumed a constant temperature of 560 °F for the duration of the accident (0-30 days). For the revised analysis, the licensee calculated a cool-down rate based on an analytical simulation model using the GOTHIC thermal hydraulic code. This calculation replaced the conservative assumption that the steam piping remained at 560 °F for the duration of the analysis period with a more realistic model.

In the re-analysis the licensee conservatively referenced the worst case post-LOCA containment temperature profiles as documented in Section 6.2.1.1.3.2, "Primary Containment Response to Pipe Breaks," of the BSEP updated final safety analysis report. Of the four cases used in the licensing basis to analyze the long-term pressure and temperature response of the primary containment during a DB LOCA, the licensee chose the case that develops the highest suppression pool peak temperature. As previously stated the higher temperature profile is more conservative for the determination of the effects of iodine deposition on the radiological source term.

The licensee developed a post-LOCA temperature profile within the MSLs and the primary and the alternate drain pathways to the condenser. The licensee incorporated the most conservative results of this analysis to develop a bounding temperature profile for the calculation of iodine deposition within the main steam piping using the well-mixed model in AEB-98-03 (Reference 9).

In a letter dated February 7, 2006 (Reference 8), the licensee addressed an NRC staff question as to whether the pipe wall temperature profile used in the elemental iodine deposition modeling accounted for the decay heat of the deposited material in the pipe and if not, how this additional source of heat would effect the deposition assumed. The licensee responded by stating that

the pipe wall temperature model used did not specifically address the decay heat of the deposited material. The licensee stated that the conservatism in the MSL temperature profile model would more than offset the small contribution to the pipe wall temperature resulting from the decay heat of the deposited material.

The licensee indicated that the MSL cool-down model used conservative assumptions regarding the assumed ambient temperature and the effectiveness of the MSL pipe insulation, thereby maximizing the calculated temperature profile and as a result minimizing the effective deposition and resuspension. For additional conservatism, no credit was taken in the temperature profile model for the heat loss resulting from branch piping, valve yokes and uninsulated pipe supports attached to the MSLs. The licensee provided an example that demonstrated that the heat loss from a small fraction of the support structures and attachments to the MSL would offset the decay heat generated from the deposited material.

It should be noted that the licensee did not take credit for the reduction in temperature that the model predicts during the first 12 hours after the LOCA. The licensee conservatively held the temperature profile constant for the first 12 hours based on the initial temperature of 550 EF. In addition, even though the licensee's bounding temperature profile indicated a continuing decrease after 20 days, the 20-day value was conservatively held constant for the last 10 days of the evaluation.

3.5.4 Main Steam Line Deposition Model

In the revised analysis, the licensee replaced the Brockmann-Bixler pipe deposition model as incorporated into the RADTRAD computer code with a well-mixed pipe deposition model. For the well-mixed pipe deposition, model, the licensee used a single composite pipe model to evaluate pipe deposition removal efficiency based on the combined total MSIV and secondary containment bypass (SCB) leakage rate through the total active drain network. To conservatively model pipe deposition, the licensee assumed that all of the TS allowable MSIV/SCB leakage occurs in the minimum number of MSLs, thereby limiting the length of pipe in which deposition is assumed to take place. The proposed MSIV leakage limit is less than or equal to 100 scfh through each MSL, with the combined leakage of the four MSLs limited to less than or equal to 150 scfh. To minimize the deposition credit the licensee modeled the total allowable 150-scfh MSIV leakage within two MSLs with the faulted MSL assigned a 100-scfh leakage rate and the intact MSL assigned a 50-scfh leakage rate.

The licensee used a well-mixed pipe deposition model, as described in NRC Accident Evaluation Branch (AEB) Report AEB 98-03 (Reference 9). The licensee conservatively modeled the MSL pathway described above as a single well-mixed volume. Conservatively, the licensee did not take credit for iodine removal from the operation of containment sprays. For the purposes of aerosol deposition, the licensee used the median or 50th percentile settling velocity of 0.00117 meters per second as documented in Table A-1 of Appendix A of AEB 98-03 (Reference 9). The licensee evaluated elemental and organic iodine deposition using the well-mixed model from Reference 9, in conjunction with the deposition rate constants developed in Technical Evaluation Report, "MSIV Leakage Iodine Transport Analysis," March 26, 1991 (Reference 10).

In a letter dated February 7, 2006 (Reference 8) the licensee responded to several NRC staff questions related to the application of the iodine deposition model used in the revised analysis.

Specifically, the NRC staff questioned the application of organic iodine deposition in the MSLs, the expected aerosol size distribution over time, the calculation of the area used for aerosol deposition and the issue of iodine resuspension.

The removal mechanisms for the deposition of organic iodine are not well understood and for this reason the NRC staff is reluctant to accept credit for organic iodine removal as a result of MSL deposition. The licensee provided the results of a sensitivity analysis that evaluates the effect of organic iodine removal at the efficiencies calculated using the above mentioned methods. The results as confirmed by the NRC staff indicate that, in this case, crediting organic iodine removal by deposition has very little effect on the overall dose calculation and the elimination of the credited organic iodine deposition would not change the acceptability of the licensee's results.

Regarding the issue of aerosol size distribution, the licensee stated that the use of a single volume, the output of which flows directly into the condenser, provides a conservative model for iodine deposition that does not permit the simulation of deposition-induced changes to the assumed initial aerosol size distribution as a function of time. To provide additional assurance of the conservatism of the deposition model with respect to changes in aerosol size distribution over time, the licensee provided the results of a sensitivity analysis to evaluate the effect of eliminating the credit for aerosol iodine removal after 24 hours. The results indicate that eliminating the credit for aerosol iodine removal after 24 hours has a small effect on the overall dose calculation and would not change the acceptability of the licensee's results. The licensee also evaluated the effect of changes in the method of calculating the area used for aerosol deposition with similar results.

Regarding the issue of iodine resuspension in the MSLs, the licensee stated that the MSL deposition model did not specifically address the issue of resuspension. However, in the response the licensee pointed out a significant conservatism incorporated in the analysis of the MSIV leakage pathway that was not discussed in the submittal. In the model used for determining iodine deposition in the condenser, the licensee did not take credit for communication between the two condenser shells. As a result of this conservatism, the effective volume of the condenser is reduced significantly. Based on sensitivity analyses performed by the licensee and the NRC staff, the incorporation of this conservative assumption has a significant effect on the calculated doses from MSIV leakage. The licensee stated, and the NRC staff agrees, that the use of the conservative assumption of no communication between the condenser shells more than compensates for any potential nonconservatism related to not addressing the issue of iodine resuspension in the main steam lines.

As mentioned previously, in response to questions from the NRC staff, the licensee provided the results of a sensitivity analysis (Reference 8) in which the effects of various assumptions on the dose analysis were explored. The results of this sensitivity analysis indicate that increases in dose due to the removal of credit for organic iodine deposition, the elimination of aerosol iodine removal after 24 hours, as well as changes in the assumed aerosol deposition area, all combined do not significantly impact the calculated dose. Further, the sensitivity analysis shows that the conservative assumption of no communication between the condenser shells significantly out weighs any potential nonconservatism in the licensee's MSL iodine deposition model.

The major assumptions and results for the licensee's revised LOCA analysis with increased MSIV leakage are shown in Tables 1 through 3. Tables 1 through 3 also include comparisons to the original AST analysis.

Table 1 - Revised Radiological Analysis Assumptions for the BSEP LOCA

Parameter	AST Value	Current Value
Control room isolation	Manual at 20 minutes	Automatic < 2 minutes
Control room unfiltered inleakage	0, 3000, and 10,000 cfm	2,000 cfm
Torus free-air volume	Not included	122,000 ft^3
MSIV leakage (total)	46 scfh	150 scfh
Reactor building positive pressure period (PPP)	5 minutes	10 minutes
Main steam line deposition model	Brockmann-Bixler @ constant pressure and constant temperature	Well mixed volume, AEB-98-03 @ constant pressure and variable temperature

Table 2 - BSEP Control Room Dose Comparisons (rem TEDE)

Dose Contribution	Highest Dose Location	Increased MSIV Leakage Calculation	Original AST Calculation
Primary Leakage - RB	0.6321	0.6321	0.22
Primary Leakage - Stack	0.1128	0.1128	------
ESF Leakage - RB	0.4940	0.4940	0.25
ESF Leakage - Stack	0.0797	0.0797	------
MSIV / SCB Leakage	2.9402	2.9402	1.64
External Cloud	0.0057	0.0057	0.01
RB Direct Shine	0.0572	0.0572	0.36
SGTS Filter Direct Shine	Negligible	Negligible	0.18
CREV Filter Direct Shine	0.0676	0.0676	0.64
ECCS Piping Shine	0.333	0.0001	0.10
Totals	4.7	4.4	3.4
Regulatory Limit	5.0	5.0	5.0

Note: Totals for licensee results are expressed to two significant figures.

Table 3 - BSEP Increased MSIV Leakage LOCA Dose Comparisons (rem TEDE)

Location	Control Room	EAB	LPZ
Original AST Calculation	3.6	0.64	1.4
Increased MSIV Leakage Dose	4.4	2.4	4.1
Regulatory Limit	5.0	25	25

Note: Licensee results are expressed to two significant figures. 3.5.5 Commitments

In conjunction with the acceptance of the license amendment described herein the licensee has made commitments to incorporate additional automatic initiation functions for the CREV system and to install additional shielding between the CREV filter and the control room consistent with the assumptions used in the design basis calculations performed in support of this amendment.

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 Surveillance Requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (70 FR 54087). 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

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 CP&L to NRC regarding a license amendment request to revise the main steam isolation valve leakage limit, dated August 11, 2005.

2. GE Topical Report, NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," September 1993.

3. Letter from F. M. Akstulewicz, NRC, to T. A. Green, BWROG, March 3, 1999, Subject: Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," September 1993.

4. Letter from Brenda L. Mozafari, NRC, to J. S. Keenan, "Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendment Re: Alternative Source Term," May 30, 2002 (ADAMS Accession Number ML021480483).

5. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," April 1998, Supplement 1, June 1999, and Supplement 2, October 2002.

6. Letter from BSEP to NRC, Response to Request for Additional Information, Revised Main Steam Isolation Valve Leakage Limit, SERIAL: BSEP 05-0138, TSC-2005-05, November 16, 2005.

7. Letter from Cornelius J. Gannon to NRC, "Response to Generic Letter 2003-01, Control Room Habitability," (SERIAL: BSEP 04-0093), July 29, 2004 (ADAMS Accession Number ML042170286).

8. Letter from BSEP to NRC, Response to Request for Additional Information, Revised Main Steam Isolation Valve Leakage Limit (SERIAL: BSEP 06-0020, TSC-2005-05), February 7, 2006.

9. AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," Appendix A.

10. Technical Evaluation Report, J. E. Cline, "MSIV Leakage Iodine Transport Analysis," March 26, 1991 (ADAMS Accession Number ML003683718).

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