CP&L
A Progress Energy Company

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U. S. Nuclear Regulatory Commission
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BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING REQUEST FOR LICENSE AMENDMENTS - EXTENDED POWER UPRATE (NRC TAC NOS. MB2700 AND MB2701)

Ladies and Gentlemen:

On August 9, 2001 (Serial: BSEP 01-0086), Carolina Power & Light (CP&L) Company requested a revision to the Operating Licenses (OLs) and the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed license amendments increase the maximum power level authorized by Section 2.C.(1) of OLs DPR-71 and DPR-62 from 2558 megawatts thermal (MWt) to 2923 MWt. Subsequently, on November 21, 2001, the NRC provided an electronic version of a Request For Additional Information (RAI) associated with the Emergency Preparedness and Health Physics Section's review of the extended power uprate amendment request. The response to this RAI is enclosed.

Please refer any questions regarding this submittal to Mr. David C. DiCello, Manager - Regulatory Affairs, at (910) 457-2235.

Sincerely,

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Enclosure:

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John S. Keenan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Dean S. Mason

Notary (Seal)

My commission expires: 8-29-04

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ENCLOSURE

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Response to Request For Additional Information (RAI) 9

Background

On August 9, 2001 (Serial: BSEP 01-0086), Carolina Power & Light (CP&L) Company requested a revision to the Operating Licenses (OLs) and the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed license amendments increase the maximum power level authorized by Section 2.C.(1) of OLs DPR-71 and DPR-62 from 2558 megawatts thermal (MWt) to 2923 MWt. Subsequently, on November 21, 2001, the NRC provided an electronic version of a Request For Additional Information (RAI) associated with the Emergency Preparedness and Health Physics Section's review of the extended power uprate (EPU) amendment request. The responses to this RAI follow.

NRC Question 9-1

In the SAR, Section 8.2.1 (Offgas System), the applicant notes that, "the system radiological release rates are administratively controlled to remain within existing limits, and the release limits are not changed with EPU." Aside from limiting power (to the point of shutting down the plant, in the case of significant fuel leakers, etc.), describe how an operator can administratively control gaseous effluents from the main condenser offgas during plant operation?

Response to Question 9-1

Offgas releases are administratively controlled by monitoring, with the effluent monitor setpoints established in accordance with the BSEP Offsite Dose Calculation Manual (ODCM). The setpoints are based on a conservative radionuclide mix (i.e., GALE Code) and established to meet NUREG-0133, "Preparation of Radiological Technical Specifications for Nuclear Power Plants," requirements (i.e., 500 mrem/yr whole body or 3000 mrem/yr skin). Additionally, a conservative fraction is applied to each release pathway to ensure that the site boundary limit is not exceeded during simultaneous releases from several pathways. These limits are administratively controlled variables and are not a function of reactor power.

NRC Question 9-2

Section 8.4.2 (Activated Corrosion Products) of the SAR, states that the production level of activated corrosion products (ACP) in the reactor coolant may increase due to a number of factors. In Section 8.1 (Liquid Waste Management), it is noted that ACP in the liquid waste effluents are expected to increase proportionally to the EPU. Additionally, in Section 11.4.2.14 (Environmental Consequences) of the SAR, it states, "There will be no change in the quantity of radioactivity released to the environment through liquid effluents...". Please explain these apparent inconsistencies.

Response to Question 9-2

Although there are anticipated to be increases in the concentration of activated corrosion products in the waste streams sent to the liquid waste processing system, the standards for release of liquids will be maintained. The plant maintains liquid effluent releases as-low-as-reasonably-achievable (ALARA) via radwaste processing and the establishment of activity limits for release. Liquid release activity is tracked as a plant key performance indicator as part of the liquid effluent reduction program. Therefore, liquids released to the discharge canal will continue to be controlled and processed to the lowest level of radioactivity practical.

NRC Question 9-3

In Section 8.4.3 (Fission Products) of the SAR, it is noted that fission products in the reactor coolant will not change as a result of the EPU. However, in the Supplemental Environmental Report (SEP), Section 8.1.3 a different conclusion is drawn -- it is clearly stated that gaseous effluent releases (principally fission product noble gases) would increase in proportion to power increases. Additionally, in Section 8.6 of the SAR (Normal Operation Off-Site Doses), it is noted that the gaseous activity levels should increase in proportion to the EPU. Please clarify these apparent inconsistencies.

Response to Question 9-3

The referenced statement in Section 8.4.3 of the Power Uprate Safety Analysis Report (PUSAR) refers to the anticipated activity due to normal operation. The PUSAR should have stated: "Releases from the fuel are not expected to increase significantly as a result of the EPU." Omission of the word "significantly" was an administrative error. In fact, some increase in fission product activity in reactor coolant during normal operation is expected. Using the formula in ANSI/ANS 18.1-1976, "Radiological Source Term for Normal Operation for Light Water Reactors," an approximately 14% increase in fission product activity in the reactor coolant is expected. Accounting for this increase, the reactor coolant activity will remain well within the BSEP Technical Specification 3.4.6, "RCS Specific Activity," limit of < 0.2 μCi/gm dose equivalent I-131.

NRC Question 9-4

Section 8.5.2 (Normal Post-Operation) of the SAR notes that the radiation level increases in some areas of the plant could be greater than the percentage increase in power. Provide the specific locations of these areas where higher dose rates are predicted, give the reasons for the expected additional increase in radiation levels in these areas, state the percentage increase in dose rates expected, and state what measures will be put in place in these areas to ensure that dose to plant personnel is maintained ALARA.

Response to Question 9-4

The referenced statement in Section 8.5.2 of the PUSAR should not have been included. The BSEP EPU Normal Post-Operation dose evaluations demonstrate that expected dose levels will not increase by more than the percentage increase in power level.

As part of EPU implementation, CP&L will conduct a startup test program that will include radiation monitoring. This monitoring will be conducted to ensure that (1) personnel exposures are maintained ALARA, (2) radiation maps are accurate, and (3) radiation zones are properly posted. In addition, much of the plant was designed for higher than expected radiation sources. Therefore, the small potential increase in radiation levels resulting from EPU is not expected to significantly affect radiation zoning or shielding. Doses to individual workers will continue to be maintained within acceptable limits by controlling access to radiation areas.

NRC Question 9-5

a. Section 8.5.2 (Post Accident) of the SAR concludes that the post-accident access and occupancy of vital areas as required by NUREG-0737, Item II.B.2 will not be significantly affected by the EPU. The pre-uprate, TID-14844 source term methodology used to calculate post-accident dose rates inside the plant differs significantly from the post-uprate Alternative Source Term (AST) model, in terms of timing issues and nuclide release/retention factors. When using a different source/dose model for the EPU, explain and justify why a simple linear increase in post-accident dose rates in proportion to thermal power provides a valid estimate of dose rates for NUREG-0737, Item 11.B.2 purposes. Specifically address source term timing issues and cesium release fractions.

b. For each of the vital areas, provide the calculated pre-uprate and post-uprate mission doses to an operator performing vital tasks following a LOCA. Verify that the mission doses to personnel in these vital areas, as well as the calculated dose estimates for personnel performing required post-accident duties in the plant's Technical Support Center and Control Room (and EOF if within 10 miles of the plant), are within the dose guidelines of GDC 19 (10CFR Part 50, Appendix A). When evaluating the Control Room for Item II.B.2. purposes, only consider the external radiation (whole body) from contained sources outside the control room envelope (and not operator inhalation doses from airborne sources entering the Control Room, which are covered in the design base accident analyses elsewhere in the EPU review).

Response to Question 9-5a

Mission doses were evaluated for EPU and AST impact as follows.

1. Current direct doses from containment or suppression pool sources were scaled for EPU. As addressed in SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," Figures 4 and 5, an evaluation of BWR post-accident containment atmosphere and suppression pool integrated environmental qualification doses indicates that AST doses (with AST-specific source term timing and cesium release fractions) from these sources are bounded by TID-14844 doses for the first 1000 hours after the accident. Since the vital missions take place well within 1000 hours post-accident, it is appropriate to scale TID-14844 based direct dose components from containment and/or suppression pool samples.

2. Reactor Building "shine" dose components for each mission were evaluated using AST based calculations.

3. Post-accident cloud immersion dose components for each mission were evaluated using AST based calculations.

4. Total mission doses were calculated by adding the contributions from all the above components.

Response to Question 9-5b

The BSEP design post-accident vital area missions are as follows:

- Post-accident Sampling System (PASS) liquid sample mission,
- PASS gas sample mission, and
- Standby Gas Treatment System (SGTS) stack sample mission.

The BSEP post-accident vital mission doses, considering EPU and AST impact are as follows:

Mission	EPU Dose (rem TEDE)
PASS Liquid Sampling	4.45
PASS Gas Sampling	3.00
SGTS Stack Sampling	4.71

These doses are all below the 5 rem regulatory limit. Results of previous estimates of mission doses indicate a maximum dose of 2.25 rem, however, the bases for the pre-EPU maximum dose could not be located. As such, CP&L elected to completely reconstitute the evaluation of mission doses. Calculation BNP-RAD-006, "NUREG-0737 Item II.B.2 - Mission Dose

Assessment For AEP and AST," documents this evaluation. This calculation was submitted to the NRC in a letter dated November 28, 2001 (Serial: BESP 01-0136).

The BSEP Emergency Operations Facility (EOF) and Technical Support Center (TSC) are located in a common onsite facility. The BSEP post-accident Control Room (CR) and TSC/EOF doses, excluding inhalation doses, are as follows:

Dose Component	CR (rem TEDE)	TSC/EOF (rem TEDE)
External Cloud	0.01	0.04
Reactor Building Direct Shine	0.36	0.36
SGTS Filter Direct Shine	0.18	0.18
CR/TSC Filter Direct Shine	0.64	0.09
Emergency Core Cooling System Piping Shine	0.10	Negligible
TOTAL	**1.29**	**0.67**

These CR and TSC/EOF doses are below the 5 rem regulatory limit.

NRC Question 9-6

SAR Sections 8.6 (Normal Operation Offsite-Doses) and 11.3 (Environmental Assessment) discuss offsite exposure pathways. In Section 8.6, radiation skyshine is noted as an insignificant offsite exposure pathway.

a. Section 8.6 noted, "...radiation from shine is not a significant exposure pathway." Does this statement refer to 40 CFR 190? Are you referring to the N-16 gamma dose source from turbine and other major steam components in turbine building? Please provide the calculated whole body dose at the site boundary from skyshine for each of the past 3 years.

b. Relative to the potential impact of increased skyshine doses from EPU on station employees working in nearby, onsite administration buildings, describe the dose monitoring program for the occupants of these buildings. State the estimated collective dose increase from EPU and provide the highest calculated (or monitored) individual dose to date? Describe the status of these employees -- are they designated members of the public under the dose limits of 10 CFR 20.1301 or are they occupational workers?

Response to Question 9-6a

PUSAR Section 8.6 addresses 10 CFR 50 Appendix I and 10 CFR 20, as indicated. Radiation shine to offsite receptors, from N-16 in the turbine and its associated piping, has three components: (a) direct (i.e., unscattered) gamma radiation, (b) air scatter from the mass of air above the turbine building (i.e., skyshine), and (c) concrete scatter off concrete shield walls. The direct and concrete scatter components do not contribute significantly to offsite locations. The most important contribution to the dose rates at ground level is from the skyshine component, and for this source the accumulated activity on the turbine operating floor determines the magnitude of this exposure.

The calculated whole body doses at the site boundary from skyshine for each of the past 3 years are:

Year	Dose (mrem/2000 hours)
1998	9.5
1999	9.3
2000	9.3

Response to Question 9-6b

Individuals working in nearby, onsite administration buildings are issued thermoluminescent dosimeters (TLDs). The highest calculated dose to date based on area TLD data is 264 mrem per 2000 hours in 1998. These individuals are considered occupational workers.

As a result of the EPU, the production rate of coolant activation products will increase by approximately 14% in the reactor water. Due to a corresponding increase in the main steam flow rate, the concentration of activation products carried over in the steam remain approximately constant. However, the increased steam flow also reduces the decay time of the activation products, specifically N-16, in transit through the turbine equipment resulting in elevated activity levels in downstream turbine operating floor components. Thus, doses to onsite personnel should not increase by more than approximately 14%, on average.

A Hydrogen Water Chemistry (HWC) system has been installed at BSEP. The current hydrogen injection rate at BSEP is 40 scfm. Radiation measurements collected in the course of the HWC Mini Test Program in 1998 indicate that there is minimal plant effect from direct shine to the environment outside of the owner controlled plant area. Post-EPU implementation surveys will be performed to assess the impact of HWC in conjunction with the EPU to demonstrate continued compliance with the requirements of 40 CFR 190.

Offsite shine doses from onsite storage of low level radwaste is not expected to be significantly impacted by EPU as any increases in solid waste resulting from processing of the projected increase in liquid waste would be counteracted by increased waste shipments. Design offsite doses for 40 CFR 190 compliance from onsite radwaste facilities would not be impacted as those doses are evaluated at postulated maximum facility capacities. Other EPU effects, such as direct gamma and neutron radiation dose rate increases that result from the increased core and stored fuel source terms, are localized increases in radiation levels inside shielded plant structures and are not expected to contribute significantly to offsite exposures.