CP&L
A Progress Energy Company

John S. Keenan
Vice President
Brunswick Nuclear Plant

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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 - REQUEST FOR LICENSE AMENDMENTS TO ADOPT ALTERNATIVE RADIOLOGICAL SOURCE TERM (NRC TAC NOS. MB2570 AND MB2571)

Ladies and Gentlemen:

On August 1, 2001 (Serial: BSEP 01-0063), Carolina Power & Light (CP&L) Company submitted a license amendment application to allow a full-scope implementation of an Alternative Radiological Source Term (AST) for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. On November 15, 2001, a meeting was held between the NRC, CP&L, and Applied Analysis Corporation (AAC) to discuss proprietary calculations developed in support of the AST radiological consequence analyses. At the conclusion of the meeting, CP&L was requested to submit a written summary of CP&L's responses to the NRC questions posed during the course of the meeting. The responses to the meeting questions are enclosed.

Please refer any questions regarding this submittal to Mr. Leonard R. Beller, Manager - Regulatory Affairs, at (910) 457-2073.

Sincerely,

John S. Keenan

John S. Keenan

P.O. Box 10429
Southport, NC 28461

T> 910.457.2496
F> 910.457.2803

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WRM/wrm

Enclosures:

1. Responses to November 15, 2001, Meeting Questions
2. CD-ROM Containing Data Files for ARCON-96 and PAVAN Computer Codes

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.

Notary (Seal)

My commission expires: 8/29/04

cc (with Enclosure 1 only):

U. S. Nuclear Regulatory Commission, Region II
ATTN: Dr. Bruce S. Mallett, Regional Administrator
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303-8931

U. S. Nuclear Regulatory Commission
ATTN: Mr. Theodore A. Easlick, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

Ms. Jo A. Sanford
Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

Mr. Mel Fry
Director - Division of Radiation Protection
North Carolina Department of Environment and Natural Resources
3825 Barrett Drive
Raleigh, NC 27609-7221

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Responses to November 15, 2001, Meeting Questions

NRC Question 1-1:

Provide a general summary of the submittal and technical approach used in the Brunswick license amendment application.

CP&L Response:

By letter dated August 9, 2001, Carolina Power & Light (CP&L) Company requested a revision to the Operating Licenses and Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, to increase the maximum licensed power level from 2558 megawatts thermal (MWt) to 2923 MWt (i.e., referred to hereafter as extended power uprate). The extended power uprate required development of new design basis accident dose analyses; therefore, CP&L decided to use Alternative Radiological Source Term (AST) to be able to take advantage of the new source term assumptions and to generally update BSEP dose consequence calculations to incorporate current methodologies and codes.

The BSEP submittal is similar to an AST submittal requested by and approved for the Duane Arnold Energy Center. The AST analyses were performed in accordance with Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," dated July 2000. The RADTRAD, PAVAN, and ARCON96 computer codes were used for these analyses. In order to support the extended power uprate of BSEP, Units 1 and 2, the AST analyses were performed at 102 percent of the uprated power level (i.e., 2981 Mwt). New atmospheric dispersion factors (χ/Q calculations) were also performed to support the dose calculations. Copies of the dose consequence calculations and χ/Q calculations were provided by CP&L's letter dated November 28, 2001 (Serial: BSEP 01-0136). The BSEP AST submittal also includes changes for relaxing the operability requirements for secondary containment which have been generically accepted by the NRC through Technical Specification Task Force (TSTF) Item 51, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations."

NRC Question 1-2:

Provide a general overview describing the atmospheric dispersion models used for the Brunswick AST analyses. Provide a copy of the meteorological input data files used in the Brunswick AST analyses.

CP&L Response:

For the BSEP AST submittal, the atmospheric dispersion models used and key modeling assumptions employed, in general, were as follows:

- The ARCON-96 computer code was used to assess onsite χ/Q values.
- The PAVAN computer code was used to assess offsite χ/Q values.
- Conservative minimum distances from each radiological release point to the onsite/offsite receptor were used.
- Except for release pathways through the plant stack, release locations were modeled as ground releases (i.e., no diffuse source models used).
- For ground level releases, building wake effects were modeled, but conservatively minimized.
- For offsite elevated releases, site terrain effects were conservatively not considered.
- No analysis credit was included for vertical velocity or plume rise at the plant stack.

In lieu of a hard copy printout of the meteorological input data files, CP&L is providing a CD-ROM containing meteorological input data files used in the BSEP AST analyses. The CD-ROM includes the following data files, formatted for use with the ARCON-96 computer code.

- arcon96.met
- arcon97.met
- arcon 98.met
- arcon 99.met
- arconall.met

In addition, the CD-ROM includes the following data files formatted for use with the PAVAN computer code.

- calm.dat
- jfdcnd.dat

NRC Question 1-3:

Please describe how the inleakage values were selected for the Control Room sensitivity analyses. In addition, provide a description of the model for the Control Room analyses.

CP&L Response:

The BSEP AST Control Room analyses were performed assuming the BSEP design inleakage value of 3,000 cfm, plus bounding minimum and maximum values of 0 cfm and 10,000 cfm.

The 3,000 cfm value was established based on pressurization testing; this inleakage value was submitted to the NRC in 1985 as part of the evaluation of Control Room habitability (i.e., CP&L letter NLS-85-311 dated August 30, 1985). The NRC accepted the 3,000 cfm value as part of the basis for Control Room habitability acceptability in 1989 (i.e., NRC letter dated February 16, 1989). The basis for the 3,000 cfm inleakage value was a Control Room pressurization test, with extrapolation of inleakage that would correspond to 1/8-inch water gauge pressure. Continued use of 3,000 cfm is considered to be conservative in view of Control Room envelope boundary integrity improvements which have been made in recent years. These improvements include sealing of heating, ventilation, and air conditioning (HVAC) duct joints, sealing of electrical conduit penetrations, and sealing of all control panel cable penetrations with new flowable elastomeric coating.

For the AST dose consequence calculations, sensitivity runs were performed to consider 0 cfm and 10,000 cfm inleakage. It was noted that dose consequences were limiting for the main steam line break (MSLB) accident with an assumed value of 0 cfm. However, the consequences for the loss-of-coolant accident (LOCA), control rod drop accident (CRDA), and fuel handling accident (FHA) asymptotically approach a limiting value with the assumed 10,000 cfm inleakage. For purposes of reporting consequences in terms of bounding values of inleakage, the calculated doses quoted in the BSEP license amendment applications are based on 0 cfm for the MSLB and 10,000 cfm for the LOCA, CRDA and FHA.

The RADTRAD model for the BSEP Control Room is described in calculation BNP-RAD-007, Section 6.1, and shown schematically in Figure 1 (i.e., BNP-RAD-007, Attachment 3, page 3-2).

- The Control Room free volume is 298,650 cubic feet. It is a dual unit control room with common ventilation. The ventilation system duct work is an extension of the Control Room envelope; fans and associated duct work are located in an HVAC equipment room located above the Control Room.
- The Control Room HVAC flow rates used are 2,100 cfm normal mode unfiltered outside air intake plus 3,000 cfm unfiltered air inleakage (i.e., alternatively, inleakage sensitivity values of 0 cfm and 10,000 cfm are used).

- Switchover to Control Room HVAC emergency (i.e., radiation) mode is assumed to occur following operator action at 20 minutes, at which time the filtered outside air intake used is 1,500 cfm, plus 3,000 cfm unfiltered air inleakage.
- An additional 400 cfm of Control Room air is recirculated and filtered.
- The flow rates are periodically verified by surveillance testing.
- The filter removal efficiencies used (i.e., for both outside intake and recirculated air) are 95 percent for particulates and 90 percent for elemental and organic iodines.

NRC Question 1-4:

For the postulated fuel handling accident, the license amendment application assumes manual isolation of the Control Room 20 minutes following the initiation of the accident. Justify a 20 minute isolation. Discuss the impact of a loss of communications between the refueling bridge and the Control Room. Discuss the human error probability associated with a 20 minute action under stress with multiple actions required. Why is a manual operator action acceptable under these conditions?

CP&L Response:

With respect to the 20 minute time period for isolation of the Control Room, the action to manually isolate the Control Room is based on the recognition and response to indications, rather than being based on the diagnosis of conditions; therefore, the time required to decide that action is required will be extremely short. The total time to recognize the indication and to accomplish Control Room isolation is estimated to be substantially less than two minutes, based on operator experience. For additional discussion, refer to CP&L's response to NRC Question 5-1 submitted by letter dated January 24, 2002 (i.e., Serial: BSEP 02-0012).

BSEP Technical Requirements Manual Specification 3.24 requires that direct communications be maintained between the Control Room and refueling platform personnel during core alterations (i.e., except for normal control rod movement). If direct communications are lost during refueling operations, these operations would be immediately secured. Manual isolation of the Control Room would not be initiated upon loss of direct communications.

On December 20, 2001, the NRC Equipment and Human Performance Branch provided an electronic request for additional information regarding the use of the manual operator action, instead of automatic action, for initiation of Control Room isolation. The responses to these questions address the specific human actions required, the training received by the personnel performing the actions, the time involved in performing the actions, and the operator's ability to recover from credible errors in performing the actions and are provided in BSEP 02-0012.

NRC Question 1-5:

The submittal states that Engineered Safety Feature (ESF) leakage was assumed to occur during post-LOCA conditions, and that a leakage rate of 20 gallons per minute was assumed in the AST analyses. Provide a description of the models, sources, percentage flashing, and percentage of iodine evolution used in the Brunswick AST analyses, and compare the modeling and assumptions to Regulatory Guide 1.183.

CP&L Response:

The BSEP AST RADTRAD model is described in calculation BNP-RAD-007, Sections 6.9 and 6.10 and shown in Figures 9 and 10 (i.e., BNP-RAD-007, Attachment 3, pages 3-10 and 3-11).

- Iodines and particulates released post-LOCA are assumed to be diluted in the minimum volume of torus water, but only the iodines are released at the ESF leakage rate discussed above for the 30-day LOCA duration.
- Since the post-LOCA torus liquid temperature remains below 212°F, the activity assumed to flash is 10 percent of the total iodine activity associated with the ESF leakage fluid.
- Consistent with Regulatory Guide 1.183, the flashed iodine is composed of 97 percent elemental iodines and 3 percent organic iodines.
- The modeling characteristics specified in Appendix A of Regulatory Guide 1.183 are incorporated into the BSEP AST RADTRAD model.

Rather than calculating the leakage from ESF leakage sources, the sum of the leakage from all ESF leakage sources (i.e., from the High Pressure Coolant Injection System, Reactor Core Isolation System, Core Spray System, and Residual Heat Removal System) into the Reactor Building was established as 1 gpm based on the BSEP standard for allowable floor drain inleakage within the Reactor Building,. Then, to evaluate ESF leakage in the BSEP AST analysis, 20 gpm of ESF leakage was conservatively assumed (i.e., 10 times the Regulatory Guide 1.183 specified value of twice the sum of the leakage from all ESF system components) as the input value in the RADTRAD computer model.

NRC Question 1-6:

Justify the use of an aerosol filter efficiency of 99.99 percent for the Standby Gas Treatment System.

CP&L Response:

Each BSEP Standby Gas Treatment System train contains two, redundant high efficiency particulate absorber (HEPA) filters; these redundant filter systems are in series. Each of these redundant filter systems is separately tested in accordance with NRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria For Atmosphere Cleanup System, Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Reactors" and, thus, was credited with a 99 percent efficiency. Since each redundant filter system was credited for a maximum bypass and/or penetration of 1 percent, the combined effect of these two filter systems in series would be a maximum penetration and/or bypass of no more than 0.01 x 0.01 = 0.0001 (i.e., 0.01 percent), which corresponds to a minimum efficiency of 99.99 percent.

During the November 15, 2001, meeting, the NRC provided further insights into the bases for the removal efficiency criterion established in NRC Regulatory Guide 1.52 (i.e., crediting a maximum of 99 percent efficiency for removal of particulates by ESF air filtration systems). As a result of these discussions, CP&L is revising the AST radiological consequence calculations to use an assumed efficiency of 99 percent, rather 99.99 percent, for each Standby Gas Treatment System filter train. The assumption of a particulate filter efficiency of 99 percent is bounded by the existing BSEP Technical Specification (TS) requirements (i.e., TS 5.5.7a); therefore, no Technical Specification revision is necessary to reflect this analysis input change. Preliminary calculation results indicate a slight increase in offsite and control room operator doses; however, the calculated doses remain well below the limits of Regulatory Guide 1.183.

NRC Question 1-7:

Justify and describe the basis for using the NEDC-31858P-A methodology for the main steam line deposition models.

CP&L Response:

The Brockmann-Bixler pipe deposition model in the RADTRAD computer model was used to assess post-LOCA radionuclide deposition within the main steam lines. The RADTRAD Brockmann-Bixler pipe deposition model is based on plug flow within the main steam piping. Based on the implementation of NEDC-31858P-A, "BWROG Report For Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," methodology for condenser holdup and deposition, the use of a plug flow deposition model versus the NRC Regulatory Guide 1.183 recommended well-mixed model is deemed adequate since the bulk of the activity removal in this release path is due to deposition and decay in the condenser. This same assessment is documented in Appendix A of AEB-98-03 (i.e., "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," dated December 9, 1998) in which the NRC notes, regarding the Perry plant:

Given the availability of the condenser and piping downstream of the third main steam valve for deposition in the earlier cases (i.e., referring to NEDC-31858P qualified plants), it was likely to be less important whether the flow was modeled as plug flow or well-mixed.

Additionally, model conservatisms include:

- 10^{th} percentile Powers model conservatively used for natural deposition within the Primary Containment. AEB-98-03 combines the well mixed flow model with a 40^{th} to 50^{th} percentile Powers model for natural deposition.
- The assumed failure of valve F021, the main steam drain orifice bypass valve in the primary pathway for main steam line isolation valve (MSIV) leakage, which results in releases via the less effective alternate pathway.
- The assumed failure of a single inboard MSIV. In this main steam line, no credit is assumed for pipe deposition upstream of the outboard MSIV or holdup in the volume between MSIVs.
- The assumed constant pressure and temperature of 4.33 atmospheres and 560°F, respectively, for the duration of the accident.
- The assumed constant MSIV leakage rate.

Model simplifications maximize the flow rates within the main steam lines. The higher flow rates in the main steam lines result in less deposition within the piping.

NRC Question 1-8:

Justify and describe the basis for using the NEDC-31858P-A methodology for the main condenser deposition models. Provide the basis for the filter efficiency used for the condenser.

CP&L Response:

According to NRC Regulatory Guide 1.183, Appendix A, paragraph 6.5, an acceptable model for accounting for the reduction of MSIV releases due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser is provided in NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Rates Limits and Elimination of Leakage Control Systems." CP&L evaluated the seismic ruggedness of the condenser and the alternate leakage treatment path as described in NEDC-31858P-A, Revision 2. CP&L's letter dated September 27, 2001 (i.e., Serial: BSEP 01-0112) provides additional information regarding the seismic ruggedness of the alternate leakage treatment path, including CP&L's schedule for completing of seismic verification walkdowns and resolution of outliers. BSEP has adopted the recommendations for use of NEDC-31858P-A and, as indicated in the AST license

amendment application, is fully implementing all recommendations for compliance with this methodology.

The BSEP condenser filter efficiencies used were evaluated based on the methodology of NEDC-31858P-A, using BSEP-specific input values. The NEDC-31858P-A methodology does not consider scrubbing, deposition, or retention of radionuclides within the condenser liquid volume. This methodology considers only the plate-out, dilution, holdup, and decay of radionuclides which would occur within the condenser volume and turbine above the entrance elevation of the alternate leakage treatment pathways. The methodology has been demonstrated conservative when compared to experimental data.

NRC Question 1-9:

Justify and describe whether there are any unmonitored release paths for the Reactor Building. Can the Reactor Building ventilation system maintain negative pressure with respect to outside air with the Reactor Building railroad doors open? Justify how GDC 64 is met with the proposed revised containment requirements during handling of irradiated fuel and core alterations.

CP&L Response:

TSTF-51, Revision 2, provides relaxation of TS requirements for secondary containment integrity during the handling of irradiated fuel in the secondary containment based on the recognition that after reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. TSTF-51, Revision 2, requires licensees adopting the secondary containment system relaxations to commit to the establishment of "prompt methods" for closing open secondary containment openings if a FHA should occur. Consistent with TSTF-51, Revision 2, CP&L has committed, in the August 1, 2001, letter (i.e., Serial: BSEP 01-0063), to establish these administrative controls. The prompt closure of secondary containment openings, and restoration of Standby Gas Treatment System functionality (i.e., if required), will enable ventilation systems to draw the release from a FHA in the proper direction such that it can be treated and released. Thus, the provisions of GDC-64, which requires means for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from a postulated accident, will continue to be satisfied. Following implementation of TSTF-51 relaxations, the Reactor Building vent radiation monitors will continue to function to monitor releases. If a post-FHA condition were to occur that could potentially result in an unmonitored release, field surveys would be performed in accordance with plant procedures.

With respect to handing irradiated fuel in conjunction with the relaxed secondary containment requirements, if the Reactor Building railroad doors are opened, CP&L believes the Reactor Building will be maintained at a negative pressure with respect to outside air. When operating, the Reactor Building Ventilation System attempts to maintain the Reactor Building between 0.25 and 0.3 inches of negative pressure with respect to the outside air. An annunciator informs the

Control Room operators if Reactor Building pressure is approaching a non-negative condition. These fans will, under normal operational conditions, exhaust more air from the building than the supply fans will draw in. The Reactor Building Ventilation System uses constant volume exhaust fans, so building pressure is maintained by automatic modulation of the supply fan vortex dampers. If the Reactor Building railroad doors are opened, as would be permitted by the TSTF-51 relaxations contained in the AST license amendment application, the Reactor Building Ventilation System controller will continue to function as designed and will modulate (i.e., move toward the closed position) the ventilation system supply fan dampers, thereby directing the system to maintain a negative pressure with respect to the outside air.

NRC Question 1-10:

Justify and describe the basis for the change reducing the primary containment leakage from 0.5 percent per day to 0.4 percent per day.

CP&L Response:

The BSEP design leak rate from primary containment is 0.5 percent weight per day. Consistent with Regulatory Guide 1.183, Appendix A, Section 3.7, which allows up to a 50 percent reduction in the design primary containment leak rate based on reduced primary containment pressure 24 hours post-LOCA, reduction from the design leakage rate after 24 hours is credited at BSEP.

The BSEP post-LOCA primary containment pressure response shows a substantial reduction from the BSEP design pressure (i.e., P_a) of 49 psig by 24 hours post-LOCA. However, this post-LOCA primary containment pressure will gradually increase over the 30-day LOCA event as the result of nitrogen inerting for combustible gas control.

Conservatively, the reduction in primary containment leakage after 24 hours is based on the 30-day end-point combustible gas control pressure of 31 psig. Primary containment leakage for the post-LOCA period between 24 hours and 30 days is evaluated based on the assumption that the primary containment pressure is constant at 31 psig throughout this time period.

The reduced primary containment leakage, based on the relationship that flow rate is proportional to the square root of the differential pressure, is then calculated to be:

$$L_r = \left[\frac{(31 psig - 0 psig)^{0.5}}{(49 psig - 0 psig)^{0.5}} \right] = 0.8$$

Thus, the reduced BSEP primary containment leakage rate after 24 hours post-LOCA is 0.8 times (i.e., 0.5 percent/day) or 0.4 percent/day.

NRC Question 1-11:

The Extended Power Uprate license amendment application states that NUREG-0737 accident mission doses have been determined by proportionally scaling based on the increase in power level. The AST license amendment applications appears to change the basis for determining the accident mission doses. Please summarize the basis for determination of accident mission doses.

CP&L Response:

In order to assess the NUREG-0737 accident mission doses, CP&L reviewed the mission time responses, with the responsible on-site groups, to validate the time basis for each mission described in the Updated Final Safety Analysis Report. The time basis was then used to determine mission doses by applying the dose rates from the original NUREG-0737 guidance. These doses were then scaled for the new background/shine dose rates established in the AST evaluation to assess the new accident mission doses. The results are documented in calculation BNP-RAD-006. For additional discussion, refer to CP&L's response to NRC Question 9-5a, submitted by letter dated December 17, 2001 (i.e., Serial: BSEP 01-0162).

ENCLOSURE 2

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