Progress Energy

10 CFR 50.90

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

SUPPLEMENT TO AMENDMENT REQUEST REGARDING FULL IMPLEMENTATION OF THE ALTERNATIVE SOURCE TERM (TAC NO. MB5105)

Ladies and Gentlemen:

By letter dated May 10, 2002, Carolina Power and Light Company, now doing business as Progress Energy Carolinas (PEC), Inc., submitted a license amendment request regarding full implementation of the Alternative Source Term (AST) for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2.

A supplement to the May 10, 2002 license amendment request was submitted by letter dated March 12, 2003. That supplement provided revised atmospheric dispersion factors, revised accident dose consequence results, and responded to NRC staff reviewer questions. The supplement also provided a revised page to the proposed Technical Specifications (TS) changes requested in the submittal dated May 10, 2002, related to the definition of Dose Equivalent Iodine-131. Additionally, the supplement proposed a change to Appendix B, "Additional Conditions," of the Operating License to delete the 504 Effective Full Power Day (EFPD) cycle length restriction as part of the approval of the Alternative Source Term license amendment. This restriction had been added by Amendment No. 196 to the Operating License and TS that was issued by the NRC on November 5, 2002.

In a letter dated December 3, 2003, supplemented by letters dated January 14, 2004 and February 6, 2004, PEC submitted a license amendment request to delete the Appendix B, 504 EFPD cycle length restriction, independent of approval of the AST amendment. That license amendment request is currently under review by the NRC.

The purpose of this supplement is to:

1. Respond to additional NRC questions related to the AST analyses. The NRC questions were received during 2003 via electronic mail, phone conversations, and during a public meeting at NRC headquarters on October 30, 2003. Responses to the NRC questions are provided in Attachment II.

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2. Provide revised input assumptions and dose analysis results for the Loss of Coolant Accident (LOCA) AST evaluation. These revised input assumptions and dose results are provided in Attachment II, in the response to NRC Question 8.

3. Revise the proposed change to Technical Specifications page 1.1-2 related to the definition of Dose Equivalent Iodine-131. The revised definition adds reference to the "effective dose factor" to ensure that the appropriate dose factors are used when performing Technical Specifications surveillances. The proposed revision is provided in Attachments III and IV.

4. Withdraw the proposed changes to Technical Specifications page 5.0-21 that were submitted in the May 10, 2002 letter. The proposed changes were related to the dose acceptance criterion for assumed failures of the Waste Gas Decay Tank (WGDT). With this withdrawal, PEC also withdraws the submittal of the revised WGDT rupture dose analysis as provided in the May 10, 2002 letter. The source term used for the WGDT rupture is not related to the AST source term, but relies on a curie inventory limit specified in the Technical Requirements Manual. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," does not address this specific accident. Therefore, full implementation of the AST does not require reanalysis of this accident and there is no need for revision and further NRC review.

5. Withdraw the proposed change to Appendix B, "Additional Conditions," related to the cycle length restriction. This withdrawal is based on the expectation that this restriction will be removed with NRC approval of the amendment requested by the HBRSEP, Unit No. 2, letter dated December 3, 2003.

6. Provide a revised No Significant Hazards Consideration and Environmental Impact Consideration. The revised No Significant Hazards Consideration and Environmental Impact Consideration provided in the March 12, 2003 letter made reference to the WGDT accident analysis and the specific definition of Dose Equivalent Iodine-131 and therefore requires revision based on the changes noted above. There are no changes to the conclusions of the No Significant Hazards Consideration and Environmental Impact Consideration. The revised versions are provided in Attachment V.

Attachment I provides an Affirmation pursuant to 10 CFR 50.30(b).

In accordance with 10 CFR 50.91(b), the State of South Carolina is being provided a copy of this letter.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

J. F. Lucas
Manager - Support Services - Nuclear

Attachments:

I. Affirmation
II. Responses to NRC Questions
III. Markup Version of Revised Technical Specifications Page
IV. Retyped Version of Revised Technical Specifications Page
V. Revised No Significant Hazards Consideration and Environmental Impact Consideration

RAC/rac

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspectors, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/04-0014 is true and correct to the best of my information, knowledge, and belief; and the sources of my information are officers, employees, contractors, and agents of Progress Energy Carolinas, Inc., formerly known as Carolina Power and Light Company. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 5 March 2004

J. W. Moyer
Vice President, HBRSEP, Unit No. 2

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

RESPONSES TO NRC QUESTIONS

NRC Question 1

Provide the containment sump volumes used in the calculation of the doses from a Loss of Coolant Accident (LOCA) due to Emergency Core Cooling System (ECCS) leakage.

Response 1

The following are the containment sump volumes used in the LOCA analyses:

Time 0 to 21 minutes: NA – No ECCS sump recirculation, hence not a release pathway.
Time 21 minutes to 40 minutes: 35,850 ft^3
Time 40 minutes to 51.5 minutes: 40,889 ft^3
Time 51.5 minutes to 720 hours: 43,939 ft^3

NRC Question 2

In the analysis of the Single Rod Control Cluster Assembly (RCCA) Withdrawal, since this event is not discussed in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," the licensee states that this event most closely resembles the reactor coolant pump (RCP) locked rotor event, which is discussed in Regulatory Guide 1.183, Appendix G, and therefore the licensee uses the release assumptions described in Appendix G. Provide additional justification for the selection of the RCP locked rotor event, as opposed to the rod ejection accident as described in Appendix H, for the modeling of the RCCA Withdrawal event.

Response 2

As described in Regulatory Guide 1.183, Appendix H, there are two separate release cases considered for the rod ejection accident. One case assumes the rod ejection also resulted in a loss of the reactor coolant system (RCS) barrier and 100% of the activity released from the fuel is released into the containment. In this case, the containment would be pressurized well above normal operating conditions due to the small to medium size LOCA. Therefore, Regulatory Guide 1.183 specifies that releases due to containment leakage are to be modeled. In the second rod ejection accident case, Regulatory Guide 1.183 specifies that all activity will remain in the RCS and hence the only analyzed release path is via the secondary system. For this case, no releases from the containment are assumed. Each case is analyzed separately and there is no summation of the dose results.

As described in Regulatory Guide 1.183, Appendix G, for the RCP locked rotor event, all activity released from the fuel is assumed to remain in the RCS and the only pathway considered

is a release via the secondary system. Unlike the first case defined by Regulatory Guide 1.183 for the rod ejection accident, there is no loss of RCS integrity. Although minor leakage (within Technical Specifications RCS leakage limits) from the RCS to the containment is possible, such leakage does not pressurize the containment and hence any releases from the containment would be insignificant. Therefore, Regulatory Guide 1.183, Appendix G, does not require consideration of releases from the containment.

In regard to RCS and containment conditions, the RCCA Withdrawal accident being analyzed for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, is similar to the case defined by Regulatory Guide 1.183 for the RCP locked rotor event. There is no loss of the RCS barrier and no pressurization of the containment. The activity released from the fuel is assumed to remain in the RCS and hence the potential release path that needs to be considered is a release via the secondary system. Similar to the RCP locked rotor event, RCS leakage into the containment that is within allowed limits would be an insignificant contributor to the dose due to the lack of containment pressurization. The RCCA Withdrawal does not result in a LOCA, which is assumed for the rod ejection accident, and hence the containment release pathway is not applicable. In regard to the secondary system release pathway modeling, the guidelines in Appendix G (RCP Locked Rotor) and Appendix H (Rod Ejection) are the same.

NRC Question 3

In the AST submittal letter of May 10, 2002, in item 15 on page 22 of Attachment II, regarding Steam Generator Tube Rupture (SGTR) analysis assumptions, the licensee states: "For the accident-induced iodine spike case, a similar assumption is made with one exception. The primary coolant iodine activity increases during the first eight hours of the transient as a result of the release from the defective fuel at a rate 335 times the iodine equilibrium appearance rates consistent with an initial dose-equivalent (DE) I-131 concentration **twice** (emphasis added) the value of the proposed TS 3.4.16 limits." Based on this statement, it appears that the iodine spiking appearance rates for the SGTR analysis are based on the iodine equilibrium appearance rates corresponding to an equilibrium RCS DE I-131 concentration of 0.5 μCi/gm, which is twice the proposed Technical Specifications limit of 0.25 μCi/gm. However, based on the licensee's dose results, it appears that this may not be the assumption actually incorporated into the dose analysis. Please clarify the assumptions used in the SGTR dose analysis in regard to iodine spiking.

Response 3

The statement from the May 10, 2002 submittal is incorrect. The statement should read: "For the accident-induced iodine spike case, a similar assumption is made with one exception. The primary coolant iodine activity increases during the first eight hours of the transient as a result of the release from the defective fuel at a rate 335 times the iodine equilibrium appearance rates consistent with an initial DE I-131 concentration at the proposed Technical Specifications 3.4.16 limits."

The RCS activity assumptions used in the SGTR accident-induced spike case analysis are:

- The initial RCS DE I-131 concentration is 0.25 μCi/gm, which is consistent with the proposed Technical Specifications limit.
- The release rate from defective fuel is equal to 335 times the equilibrium appearance rates based on an equilibrium DE I-131 concentration of 0.25 μCi/gm. The concentration of 0.25 μCi/gm is consistent with the proposed Technical Specifications. The appearance rates corresponding to this assumption are those presented in Table 3 of Attachment XI to the May 10, 2002 letter.
- Consistent with the guidelines of Regulatory Guide 1.183, Appendix F, only iodine spiking was assumed. No increased appearance rates from the fuel for cesium nuclides were assumed in the analysis. See the response to NRC Question 4 for more details.

NRC Question 4

In the AST submittal letter of May 10, 2002, in item 14 on page 22 of Attachment II, regarding Steam Generator Tube Rupture (SGTR) analysis assumptions, the licensee states: "The ratio of radioiodines to other radionuclides provided in the UFSAR, Table 11.1.1-2, is assumed to be a constant." Based on this statement, it could be assumed that for the pre-accident spike case, where the DE I-131 concentration is assumed to have spiked to 60 μCi/gm, that a comparable pre-accident spike has occurred for other radionuclides such as cesium. However, based on the licensee's dose results, it appears that spiking of nuclides other than iodine was not assumed. Please clarify the assumptions used in the SGTR pre-accident spike case dose analysis in regard to RCS activity and spiking, and the intent of item 14 on page 22 of Attachment II.

Response 4

The statement in item 14 on page 22 of Attachment II was only intended to apply to the ratio of nuclides when adjusting the RCS concentration from those in Updated Final Safety Analysis Report (UFSAR) Table 11.1.1-2 to concentrations corresponding to an RCS Technical Specifications activity limit of 0.25 μCi/gm DE I-131. Once a pre-accident iodine spike is assumed, this statement no longer applies, as iodine is the only nuclide assumed to experience spiking. The RCS concentration of the cesium nuclides for the SGTR pre-accident spike case remains at the concentrations specified in Table 4 of Attachment VI to the May 10, 2002 letter.

The RCS activity assumptions used in the SGTR pre-accident spike case analysis are:

- The initial RCS DE I-131 concentration is 60 μCi/gm.
- Consistent with Regulatory Guide 1.183, Appendix F, only iodine spiking was assumed. The RCS activity concentration for noble gases and cesium are as given in Table 4 of Attachment VI to the May 10, 2002 letter.

A realistic analysis of transient induced RCS activity spiking would indicate that some cesium spiking also occurs. This realistic evaluation would also conclude that the predominant form of the iodine and cesium released from the fuel is in the particulate form of CsI. As a particulate, very little of this activity would be released from the secondary side of the steam generators,

especially if the steam generator tubes remain covered with water, which is the case for the HBRSEP, Unit No. 2, SGTR analysis. Regulatory Guide 1.183 specifies that the iodine being released from the steam generators is assumed to be 97% elemental and 3% organic. No particulate iodine is assumed to be released. Therefore, the particulate cesium is expected to be an insignificant contributor to the dose. Regulatory Guide 1.183, Appendix F, and the other appendices that are related to RCS activity spiking, specify that iodine spiking should be assumed, but specify no requirements related to spiking of other nuclides.

NRC Question 5

The existing radiological consequence analysis of a Steam Generator Tube Rupture (SGTR) accident assumes that the leak flow from the reactor coolant system (RCS) to the secondary side of the steam generator (SG) is terminated in 30 minutes following the event initiation. The analysis uses an equilibrium break flow that continues at a constant rate for 30 minutes. The resulting break flow mass transfer is then used to calculate the radiological consequences of the SGTR. You have considered that this assumption will result in a conservative calculation, since the reduction in the break flow over the 30 minutes time is ignored. Inherent in this evaluation is the assumption that the operator can terminate the break flow in 30 minutes. Plants with similar design to HBRSEP, Unit No. 2, have reported to the NRC that, in simulator exercises, the operators demonstrated that the time to terminate the break flow exceeded the 30 minute assumption. Should the operator significantly exceed the 30 minute termination criteria, this event could lead to an increase in radiological releases from that assumed when the leak flow was terminated within 30 minutes. Please provide the necessary information to confirm that the assumption of termination of the break flow in 30 minutes following a design basis SGTR event is valid for HBRSEP, Unit No. 2, at the current licensed power level and will indeed lead to a bounding calculation regarding the radiological consequences of the event.

Response 5

The HBRSEP, Unit No. 2, assumption is that the ruptured SG is isolated from the environment by operator action within 30 minutes, not that the primary to secondary break flow is terminated in 30 minutes. The assumptions on the ruptured SG isolation time of 30 minutes and mass releases are based on the approved current licensing basis (CLB) for HBRSEP, Unit No. 2. The original plant licensing basis established the 30 minute SG isolation time for the SGTR event, and this basis was re-affirmed to be acceptable through the NRC review and approval of the steam generator replacement and power uprate licensing changes in the early 1980's [see NRC Safety Evaluation Report (SER) for License Amendment No. 87, dated November 7, 1984]. HBRSEP, Unit No. 2, has recently undergone an NRC inspection of the control of the design bases for the SGTR event, and no findings were identified.

As described in UFSAR Section 15.6.3, the mass flows (primary to secondary and secondary to atmosphere) are based on a scenario that assumes a loss of offsite power and a stuck open SG power operated relief valve (PORV) on the ruptured SG. The scenario assumes that the ruptured SG is isolated from the environment by operator action within 30 minutes. It does not assume that primary to secondary leakage or break flow is terminated at 30 minutes. Although break flow may

continue beyond 30 minutes, this additional flow into the secondary side of the affected SG will not be released to the environment, as the affected SG secondary side is isolated from the environment within 30 minutes. Based on simulator experience, the continuing break flow does not result in the opening of the main steam safety valves on the affected SG, nor does it result in any SG overfill conditions.

Isolation of the ruptured SG from the environment within 30 minutes is consistent with the current Emergency Operating Procedures (EOPs). This would include the ability to manually isolate an assumed stuck open SG PORV, as it would only take a few minutes to isolate the air supply to the valve, which would result in valve closure. Consistent with the requirements of Regulatory Guide 1.183, releases from the secondary system to the environment are assumed to continue for 53 hours from the SG PORVs on the unaffected SGs, at which time the residual heat removal (RHR) system is used for cooldown.

The dose evaluation methods of the Alternative Source Term (AST) guidelines in Regulatory Guide 1.183 do not require a change in the means by which the mass releases to the environment are calculated. Section 5.3 of Appendix F of Regulatory Guide 1.183 specifies that primary to secondary leakage should continue until certain conditions are satisfied. The HBRSEP, Unit No. 2, AST analysis is consistent with the requirements of Regulatory Guide 1.183 for leakage of primary fluid containing activity to the secondary system. The isolation assumptions for the affected SG and mass transfer from the primary to the secondary are consistent with the values presented in the UFSAR, which have been previously reviewed by the NRC. The steam releases from the secondary side of the steam generators were conservatively determined for a time period consistent with the requirements of Regulatory Guide 1.183

NRC Question 6

In regard to assumed control room envelope unfiltered inleakage, different values were assumed in different accident analyses. For example, in the LOCA analysis, the assumed inleakage after one hour is 100 cfm, however, for other accidents the assumed inleakage is 300 cfm. Explain why different values were used and how the design basis inleakage is defined.

Response 6

It was initially planned that the same assumed value for control room unfiltered inleakage would be used for each accident analysis. However, the LOCA analysis was performed following completion of the other accident analyses, and the inleakage values assumed for these other accidents did not result in acceptable LOCA dose consequences for the control room. Therefore, the assumed inleakage values were reduced for the LOCA analysis. There was no need to reanalyze the previously completed analyses, because acceptable results were obtained with higher inleakage rates.

Once the Alternative Source Term amendment is approved, the design basis inleakage values for the control room envelope will be the most limiting values established by the dose analyses, and hence will be the values used in the LOCA analysis. That is, the design basis requirement for

inleakage will be 170 cfm during the ventilation mode where the Hagan room area outside the control room is pressurized, and 100 cfm when the Hagan room is not pressurized. These values will be used as the basis of the acceptance criterion for any subsequent inleakage testing.

NRC Question 7

For the LOCA dose analysis, confirm that a sliding two hour window was used for establishing the worst 2 hour dose for the Exclusion Area Boundary (EAB).

Response 7

The LOCA dose analysis used the sliding 2 hour window to determine the worst 2 hour dose for the Exclusion Area Boundary. The 0 to 2 hour atmospheric dispersion factor for the EAB was entered into the RADTRAD code for the entire 30 day time period. RADTRAD then determined the worst 2 hour dose, which occurred from 0.5 to 2.5 hours.

NRC Question 8

In the LOCA dose analysis, the licensee has used a transfer rate of 130,000 cfm from the sprayed region of the containment to the unsprayed region of the containment, and likewise a 130,000 cfm flow from the unsprayed region to the sprayed region. This transfer rate is based on forced recirculation flow from two 65,000 cfm containment cooling fans. Provide additional justification for the use of the 130,000 cfm transfer rate between the two containment zones.

Response 8

A review of the ventilation patterns in the containment has determined that only a fraction of the total forced flow of 130,000 cfm can be justified as a mixing rate between the sprayed and unsprayed regions. Additionally, while reviewing the analysis, a non-conservative error was discovered in the modeling of the containment leak rate. In order to compensate for changes in these two factors that would increase the calculated dose consequences, other inputs to the LOCA dose analysis were re-evaluated and revised. The following provides the revised assumptions and the basis for the revisions:

1. Sprayed and Unsprayed Volumes

   The calculation of the sprayed volume used in the previously submitted analysis was conservative. For example, the analysis considered only gravity affecting the spray coverage patterns, neglecting any additional spray coverage that would occur due to air movement patterns (a "still air" assumption). Using these conservative assumptions, the calculated spray volume was only 52% of the containment volume. The spray volume calculation was revised to be more representative of the expected sprayed volume, based in part upon air movement patterns described in NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings." The following are the results from the revised spray volume calculation:

Spray Train A: fraction of the containment free air volume = 0.829
Spray Train B: fraction of the containment free air volume = 0.815

(Note – because the two spray trains have different characteristics, such as spray coverage, spray flow, and fall height, it could not be determined which train would yield the highest calculated doses without running an analysis for each train separately. Hence the input parameters for each train are provided. The results show that Train B provides the highest doses.)

Using the total containment free air volume of 1,958,526 ft^3 (not changed from previous analysis), the following sprayed and unsprayed volumes are obtained for input into the RADTRAD code:

For Spray Train A, the free air volume of the sprayed volume is:

1,958,526 ft^3 * 0.829 = 1,623,618.1 ft^3

The free air volume of the unsprayed volume is:

1,958,526 ft^3 * 0.171 = 334,907.9 ft^3

For Spray Train B, the free air volume of the sprayed volume is:

1,958,526 ft^3 * 0.815 = 1,596,198.7 ft^3

The free air volume of the unsprayed volume is:

1,958,526 ft^3 * 0.185 = 362,327.3 ft^3

These revised sprayed/unsprayed volumes will also impact the RADTRAD input parameter for the fraction of the source term that is released into the sprayed and unsprayed volumes. The fractions used will be equivalent to the fractions of the containment free air volume given above.

2. Source of Containment Leakage

In the original calculation it was conservatively assumed that all of the containment leakage was coming from the unsprayed volume (it was later discovered that this assumption was implemented into the model incorrectly). In this revised analysis, the containment leakage rate is assumed to originate from both of the two regions, such that the total leak rate (0.1 weight %/day for the first day, and 0.05 weight %/day for days two through 30) is split between the sprayed and unsprayed regions in proportion to their total volumes.

3. Aerosol (Particulate) Spray Removal Modeling

In the original calculation, a 'user identified' particulate spray removal coefficient, consistent with modeling specified in Standard Review Plan Section 6.5.2, was employed. In this revised analysis, the Powers spray removal model, as incorporated into the RADTRAD code, was used instead of a 'user identified' coefficient.

The calculation used the 50 percentile Powers Model, which represents the best estimate or mean removal rate values expected from mechanistic models for any given set of containment spray parameter inputs. To ensure that the calculation has maximized the dose consequences from the design basis LOCA accident evaluation, as required by Regulatory Guide 1.183, Footnote 1 of Appendix A, conservative inputs to the Powers Model determination of spray removal rates were chosen as follows:

a) The flow area used to establish the spray flux input to the Powers Model was chosen in accordance with the RADTRAD code documentation manual recommendations to balance the conservative and non-conservative effects of changing spray conditions at different containment elevations.

b) Spray flow is minimized by using a maximum expected degraded pump performance curve, and by neglecting the significant beneficial effects of operation in piggy-back mode after switchover to sump recirculation.

c) Spray flow is minimized by evaluating flow against the maximum containment design pressure, rather than allowing spray flow to increase during times where containment pressure has decreased.

d) To ensure that the limiting (least coverage/least flow) containment spray train is conservatively chosen for the design basis dose consequence evaluation, each single train of spray removal was evaluated.

e) Consistent with the LOCA single failure evaluation, only one train of spray flow is credited for dose evaluation purposes.

f) Spray nozzle performance is determined from worst case containment design pressure, rather than considering the time-dependent effects of the changing containment pressure and temperature on the nozzle spray pattern and droplet sizes.

g) The spray timing is modeled to actuate after maximum delays, and to maximize the period of no-spray during switchover to sump recirculation.

h) Using the NUREG/CR-4102 evaluation of air and spray droplet cross-flow velocities expected during post-LOCA spray and HVAC operation, the Powers Model basis described in NUREG/CR-5966 (that the sprayed areas be essentially

fully sprayed) has been shown to be applicable to a large fraction of the containment.

i) The spray fall height was limited to the distance from the average location of the spray ring to the highest flood level in containment.

The following are the Powers Model input parameters used in the RADTRAD analyses:

Train A: spray flux of 1.265E-02 ft^3/ft^2-min, spray height of 165.5 ft
Train B: spray flux of 1.238E-02 ft^3/ft^2-min, spray height of 154.1 ft

These values were used for the time periods when the sprays are assumed to be on. These time periods are unchanged from the original HBRSEP, Unit No. 2, submittal, and are provided below for particulate spray removal parameters:

Time Period	Spray Status	Comments
0 to 0.05 hrs	Off	Not yet fully activated
0.05 to 1.2833 hrs	On	
1.2833 to 1.6667 hrs	Off	Switching from injection to sump recirculation
1.6667 to 2.7833 hrs	On	
2.7833 to 720 hrs	Off	Secured due to reduced containment temperature and pressure

4. Natural Deposition Removal Cut Off Time

In the original calculation, it was conservatively assumed that natural deposition removal in the containment terminated when the decontamination factor (DF) was 1000. In the revised analysis, natural deposition is assumed to continue to the end of the accident, consistent with methodology referenced in Regulatory Guide 1.183.

5. Mixing Rate Between the Sprayed and Unsprayed Region

In the original calculation, it was assumed that the forced ventilation flow of 130,000 cfm provided by two recirculation fans provided a flow of 130,000 cfm from the sprayed region to the unsprayed region, and also from the unsprayed region to the sprayed region. To resolve NRC staff questions on this approach, a detailed analysis of the ventilation flow patterns within the containment was performed. Based on that evaluation, it was determined that the initially assumed mixing rate flows were a non-conservative simplification.

The revised evaluation uses a mixing rate flow of 65,000 cfm from the sprayed to the unsprayed region, and 65,000 cfm from the unsprayed region to the sprayed region. The

following discussion is taken from the revised LOCA dose analysis and provides the justification for the assumed 65,000 cfm mixing rate:

The intake of the 4 HVH units is located near the walls of the containment above the operating deck. After the air is suctioned through the intakes it is routed to a common ring header situated approximately 20 feet above the operating deck. The common ring header then discharges the flow through ductwork into the lower compartments located below the operating deck. The discharge of the HVAC fan units splits roughly 90%/10% between an area below operating deck which directs air flow radially inwards, and a small amount of above operating deck discharge. The below deck HVAC discharge is into an area that is only about 10% spray covered by openings and gratings. This area freely communicates with 2 other comparably sized below deck areas that are between 19% and 41% spray covered, using the prior, "still air" operating deck coverage assumptions, and depending on which spray train is in operation. For Train "A," the average coverage is about 21%. For Train "B," the spray coverage in these areas averages about 37%. To determine the amount of forced HVAC flow that begins as sprayed air intake, and travels entirely through sprayed area flow paths, these below deck spray coverage areas will be used.

All HVAC intake flow is sprayed, since the intakes are confirmed to be near the operating deck. As described in the evaluation of air flow patterns in NUREG/CR-4102, there are significant radial air and droplet horizontal velocities that promote adequate mixing between any sources of sprayed and unsprayed air. The evaluation of the total containment spray coverage has shown that there will be full spray coverage at the operating deck, including the annulus next to the containment walls that existed in the prior, "still air assumption" evaluation of spray coverage. The intake of the HVAC fan is no longer in or near a boundary between sprayed and unsprayed regions. The discrete, two points of HVAC intake do not support any credible mechanism for a significant amount of distributed below deck unsprayed air to bypass the strong mixing velocities associated with the air and spray droplet movement near the operating deck shown in NUREG/CR-4102.

About 10% of the HVAC discharge flow will be diverted into operating deck HVAC discharge ductwork leading to various locations around the containment. The original 130,000 cfm flow available for mixing with unsprayed air is thus immediately reduced by 10% (13,000 cfm) to 117,000 cfm.

The discharge of this remaining 117,000 cfm will be distributed uniformly between sprayed and unsprayed regions of the steam generator and pump compartment (approximately 116,000 ft^3 total in this compartment), in proportion to the volume of those sub-regions. Besides the 10% discharge at the operating deck level described above, this is the only discharge location for the post-LOCA HVAC system. Only about 10% of this SG and pump compartment is sprayed. Thus 10% of the remaining 117,000 cfm (11,700 cfm) may not mix with unsprayed air in this compartment, and

will be conservatively assumed to be removed from any further mixing flow by diffusion upwards through the operating deck grating. This leaves 105,300 cfm of flow into the unsprayed volume of this compartment. This would lead to the conclusion that the 105,300 ÷ 130,000 or 81% of the nominal HVAC flow can be credited for circulation between a sprayed and an unsprayed region.

Conservatively requiring that all unsprayed regions below the operating deck be considered when establishing a simple sprayed/unsprayed model of containment would lead to the following extension of the above reasoning. The remaining 105,300 cfm may be assumed to either move into and mix with the other below deck regions, or to diffuse upwards to the operating deck and be lost to further mixing.
The split between diffusion loss upwards and transfer to other compartments will be based on below deck compartment sizes. The combined volume of the other areas represents 229,400 ft^3, or approximately 2/3 of the total below deck volume. Therefore, only about 1/3 of the 105,300 cfm will be conservatively assumed to be immediately lost directly to the sprayed areas above. Before returning to the operating deck (sprayed) level, the remaining 70,200 cfm may again be uniformly distributed between the remaining sprayed and unsprayed regions below the operating deck. The most spray coverage condition (Train "B") averages about 37% spray coverage for these remaining regions. Reducing the 70,200 cfm by 37% (or about 26,000 cfm) leaves only about 44,200 cfm to pass through the remaining unsprayed regions before reaching the operating deck through the gratings. This is 44,200 ÷ 130,000 or 34% of the original forced HVAC flow. Use of this value in a 2 volume mixing exchange rate would be overly conservative, since it neglects the benefits of the larger mixing flow rate in the original SG and pump bay discharge point.

A composite or "effective" HVAC flow rate for evaluating spray removal can be determined for the entire below deck volume as follows. A below deck volume weighted average flow rate can be determined since 1/3 of this volume has a mixing flow rate of 105,300 cfm, and 2/3 of the total volume below the operating deck has a lower flow rate of 44,200 cfm. The volume weighted mixing flow is:
(105,300 + 2×44,200) ÷ 3, or about 65,000 cfm (to the nearest 1000 cfm). This is 50% of the total 130,000 cfm forced air flow rate.

The upper containment dome unsprayed regions (above the spray rings) are not affected by the forced HVAC flow, so they are not considered in establishing the minimum forced HVAC volumetric flow. These regions will be adequately mixed by convective exchange between the sprayed and unsprayed regions, so neglecting to explicitly model this exchange of air volumes for this region is an acceptable simplification.

On this basis, RNP conservatively credits only 50% of the forced HVAC flow as actual mixing (or exchange) flow between the sprayed and unsprayed regions in the

2-volume model of containment. This treatment assures that a conservative volumetric flow rate is used for the below deck mixing regions.

To aid the understanding of the above description, a diagram is provided at the end of this attachment to depict the air flow pathways being discussed.

6. Spray Removal Cut Off Times

In the original calculation, the elemental iodine spray removal was terminated at a time of 2.46 hours, based on a determination of the time at which a DF of 200 was achieved, which is the maximum spray DF allowed by Regulatory Guide 1.183. With the above-described changes in parameters, such as the spray volume and mixing rate, the time at which a DF of 200 will be achieved will change. Special RADTRAD runs were performed to establish the time at which a DF of 200 is achieved based on the revised assumptions. The results of these evaluations are that elemental iodine spray removal will terminate at 2.08 hours for Spray Train A and 2.11 hours for Spray Train B.

The original calculation also established a time for the particulate iodine at which a DF of 50 would be achieved. The calculated time was 20.8 hours. Regulatory Guide 1.183 specifies that the removal coefficient should be reduced by a factor of 10 at this time. In the revised analysis, the Powers time dependent removal model is employed. With this model, it is not necessary to apply this factor of 10 reduction when a DF of 50 is achieved. In both analyses, this consideration of a reduced removal rate is not used, as the analyses assume that the operator secures the spray flow at 2.7833 hours, well before a DF of 50 is achieved.

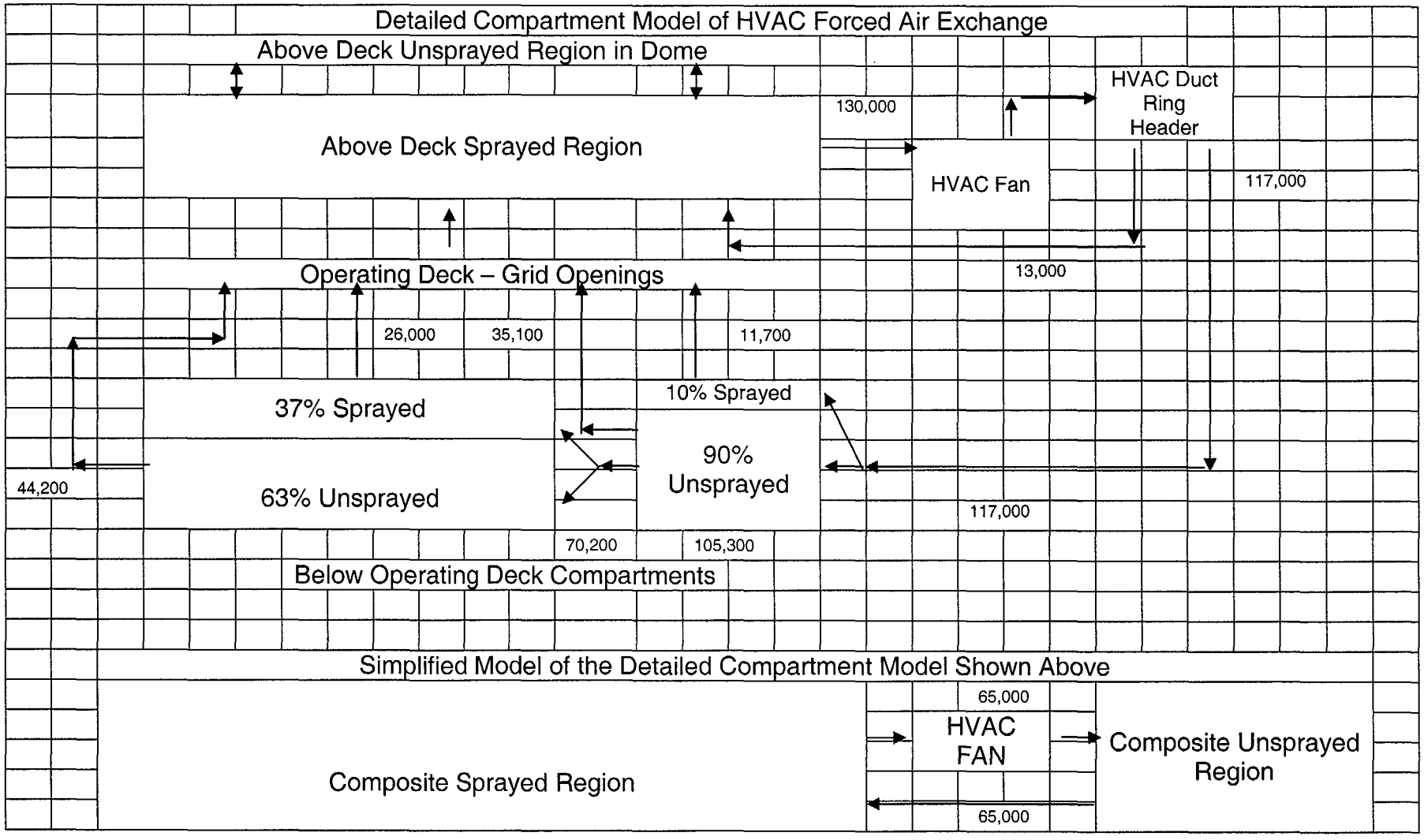
The assumption changes described above are related to the modeling of the containment and removal mechanisms in the containment. There were no changes to the modeling of the dose contribution from Engineered Safety Features (ESF) leakage outside containment. There were no changes to the control room or the Technical Support Center (TSC) design inputs, and there were no changes to other assumptions such as atmospheric dispersion factors or core inventory source terms. Changes in the containment modeling did impact the amount of activity in the containment and released to the environment, which does impact the calculated direct dose to the control room and TSC from containment shine and plume shine. These direct doses were re-evaluated.

Based on the input assumption changes described above, the following table provides the revised LOCA dose consequence results. The results are from the most limiting spray train, which is Spray Train B. These dose results are within the applicable acceptance criteria.

LOCA Dose Consequences – TEDE Dose (REM)

Dose Pathway	EAB	LPZ	Control Room	TSC
Containment Release	2.28E+01	1.31E+00	2.39E+00	1.86E+00
ESF Leakage	1.05E+00	2.27E-01	1.94E+00	1.73E-01
Containment Shine	NA	NA	3.00E-02	3.00E-02
Plume Shine	NA	NA	2.99E-02	2.35E-02
Total	2.39E+01	1.54E+00	4.39E+00	2.09E+00

HBRSEP, Unit No. 2 - LOCA Containment HVAC Flow Modeling



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

MARKUP VERSION OF REVISED TECHNICAL SPECIFICATIONS PAGE

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed under the "Effective" column of Table 2.1 of Federal Guidance Report 11.
Ē-AVERAGE DISINTEGRATION ENERGY	Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than

Deleted: thyroid

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United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/04-0014
2 pages including cover page

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

RETYPED VERSION OF REVISED TECHNICAL SPECIFICATIONS PAGE

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
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$\bar{E}$-AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than

(continued)

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

REVISED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND ENVIRONMENTAL IMPACT CONSIDERATION

The following provides a revised No Significant Hazards Consideration Determination and Environmental Impact Consideration. The only differences from the determinations submitted in the March 12, 2003 letter are in the definition of Dose Equivalent I-131 in the description of the proposed Technical Specifications changes; the deletion of references to the operating cycle restriction from Appendix B of the Operating License; and, deletion of reference to the Waste Gas Decay Tank (WGDT) accident and the withdrawn Technical Specifications page related to WGDT accident dose acceptance criterion. These changes do not impact the basis or conclusions of the original determinations.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Progress Energy Carolinas, Inc., formerly known as Carolina Power and Light Company, is proposing changes to Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. These changes revise the licensing basis for HBRSEP, Unit No. 2, to implement the Alternative Source Term (AST) described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The AST was used in evaluating the radiological consequences of the following Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents:

- Main Steam Line Break,
- Reactor Coolant Pump Shaft Seizure (Locked Rotor),
- Single Rod Control Cluster Assembly (RCCA) Withdrawal,
- Steam Generator Tube Rupture (SGTR),
- Large Break Loss of Coolant Accident (LBLOCA)

In addition, revised atmospheric dispersion factors for onsite and offsite dose consequences have been calculated and incorporated in the reanalysis of these events. As part of the full 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. A request for selective implementation of an AST in the HBRSEP, Unit No. 2, Fuel Handling Accident analysis was previously submitted by letter dated March 13, 2002, and approved by NRC letter dated October 4, 2002.

Full implementation of the AST is supported by the following TS changes:

The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988, as the source of dose conversion factors and to specify effective dose factors in lieu of thyroid dose factors.

The reactor coolant system (RCS) operational leakage limits stated in Limiting Condition for Operation (LCO) 3.4.13 for total primary to secondary leakage through all steam generators is reduced from 1 gpm to 0.3 gpm. In addition, the limit specified for primary to secondary leakage through any one steam generator is reduced from 500 gallons per day to 150 gallons per day.

The Dose Equivalent I-131 requirements of TS 3.4.16, "RCS Specific Activity," are reduced from 1.0 μCi/gm to 0.25 μCi/gm in Condition A and in Surveillance Requirement (SR) 3.4.16.2. In addition, Figure 3.4.16-1, "Reactor Coolant Dose Equivalent I-131 Specific Activity Limit Versus Percent of Rated Thermal Power," is deleted. Required Action A.1 is revised to replace the reference to the acceptable region of Figure 3.4.16-1 with a limit of ≤ 60.0 μCi/gm. The second entry condition of Condition C is revised to replace the reference to the unacceptable region of Figure 3.4.16-1 with reference to > 60 μCi/gm.

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

   Implementation of the Alternative Source Term does not affect the design or operation of HBRSEP, Unit No. 2. Rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequences of the postulated accident. The implementation of the Alternative Source Term has been evaluated in revisions to limiting design basis accidents at HBRSEP, Unit No. 2. Based on the results of these analyses, it has been demonstrated that, with the requested changes to the Technical Specifications, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC. This guidance is presented in 10 CFR 50.67 and Regulatory Guide 1.183. The proposed Technical Specifications changes result in more restrictive requirements and support the revisions to the limiting design basis accident analyses.

   Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

   The proposed changes do not affect plant structures, systems, or components. The Alternative Source Term and those plant systems affected by implementing the proposed changes do not initiate design basis accidents.

   Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

   The proposed changes are associated with the implementation of a new licensing basis for HBRSEP, Unit No. 2. The new licensing basis implements an Alternative Source Term in accordance with 10 CFR 50.67 and the associated Regulatory Guide 1.183. The results of the revised limiting design basis analyses are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies in accordance with the regulatory guidance. The dose consequences of the limiting design basis events are within the acceptance criteria found in the regulatory guidance associated with Alternative Source Terms.

   The proposed changes continue to ensure that doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits. Specifically, the margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits, which are conservatively set below the 10 CFR 50.67 limits. With respect to control room personnel doses, the margin of safety (the difference between the 10 CFR 50.67 limits and the regulatory limits defined by 10 CFR 50, Appendix A, Criterion 19 (GDC-19)) continues to be satisfied.

   Therefore, this change does not involve a significant reduction in a margin of safety.

Based on the above discussion, Progress Energy Carolinas, Inc., formerly known as Carolina Power and Light Company, has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT CONSIDERATION

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. Progress Energy Carolinas, Inc., formerly known as Carolina Power and Light Company, has reviewed this request and determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required in connection with the issuance of the amendment. The basis for this determination follows:

Progress Energy Carolinas, Inc., formerly known as Carolina Power and Light Company, is proposing changes to Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. These changes revise the licensing basis for HBRSEP, Unit No. 2, to implement the Alternative Source Term (AST) described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The AST was used in evaluating the radiological consequences of the following Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents:

- Main Steam Line Break,
- Reactor Coolant Pump Shaft Seizure (Locked Rotor),
- Single Rod Control Cluster Assembly (RCCA) Withdrawal,
- Steam Generator Tube Rupture (SGTR),
- Large Break Loss of Coolant Accident (LBLOCA)

In addition, revised atmospheric dispersion factors for onsite and offsite dose consequences have been calculated and incorporated in the reanalysis of these events. As part of the full 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.

Full implementation of the AST is supported by the following TS changes:

The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988, as the source of dose conversion factors and to specify effective dose factors in lieu of thyroid dose factors.

The reactor coolant system (RCS) operational leakage limits stated in Limiting Condition for Operation (LCO) 3.4.13 for total primary to secondary leakage through all steam generators is reduced from 1 gpm to 0.3 gpm. In addition, the limit specified for primary to secondary leakage through any one steam generator is reduced from 500 gallons per day to 150 gallons per day.

The Dose Equivalent I-131 requirements of TS 3.4.16, "RCS Specific Activity," are reduced from 1.0 μCi/gm to 0.25 μCi/gm in Condition A and in Surveillance Requirement (SR) 3.4.16.2. In addition, Figure 3.4.16-1, "Reactor Coolant Dose Equivalent I-131 Specific Activity Limit Versus Percent of Rated Thermal Power," is deleted. Required Action A.1 is revised to replace the reference to the acceptable region of Figure 3.4.16-1 with a limit of ≤ 60.0 μCi/gm. The second entry condition of Condition C is revised to replace the reference to the unacceptable region of Figure 3.4.16-1 with reference to > 60 μCi/gm.

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not involve a significant hazards consideration.

2. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not result in a significant increase in the consequences of an accident previously evaluated and does not result in the possibility of a new or different kind of accident. Therefore, the proposed change does not result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

3. The Alternative Source Term does not affect the design or operation of the facility. Rather, once the occurrence of an accident has been postulated, the Alternative Source Term is an input to evaluate the consequences. The implementation of the Alternative Source Term has been evaluated in revisions to the analyses of the limiting design basis accidents at HBRSEP, Unit No. 2. Based on the results of these analyses, it has been demonstrated that, with the requested Technical Specifications changes, the dose consequences of the limiting event are within the regulatory guidance provided by the NRC for use with the Alternative Source Term. Therefore, the proposed change does not result in a significant increase in either individual or cumulative occupational radiation exposures.