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March 16, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

10 CFR 52.79

Subject: Duke Energy Carolinas, LLC
William States Lee III Nuclear Station - Docket Nos. 52-018 and 52-019
AP1000 Combined License Application for the William States Lee III Nuclear Station Units 1 and 2 Supplemental Response 2 to Request for Additional Information Letter No. 25 (eRAI 50), RAI 13.03-061, SITE-8, Item J
Ltr#: WLG2015.03-01

References:

1. Letter from Brian Anderson (NRC) to Peter Hastings (Duke Energy), Request for Additional Information Letter No. 25, Related to SRP Section 13.03 – Emergency Planning, dated September 26, 2008 (ML082690889)
2. Letter from Bryan J. Dolan (Duke Energy) to NRC Document Control Desk, Response to Request for Additional Information Letter No. 025 (eRAI 50), Ltr# WLG2008.12-30, dated December 23, 2008 (ML090020175)
3. Letter from Christopher M. Fallon (Duke Energy) to NRC Document Control Desk, Supplemental Information Related to Design Changes to the Lee Units 1 and 2 Physical Locations and Additional Design Enhancements, Ltr# 2013.05-02, dated May 2, 2013 (ML13127A224 and ML131127A225)

This letter provides Duke Energy's supplemental response to the Nuclear Regulatory Commission's request for additional information (RAI) included in Reference 1. Duke Energy's initial response was provided in Reference 2. In the response to RAI 13.03-061, SITE-8, Emergency Facilities and Equipment, Item J, Duke Energy stated the information in the response would be updated to reflect changes incorporated into Westinghouse AP1000 DCD Revision 17. Since that time the AP1000 DCD has been revised to Revision 19. In Addition, (1) design enhancements have been made to the Technical Support Center conceptual design; and (2) the Lee Nuclear Site footprint was relocated and a full two years of on-site meteorological data has been recorded as described in Reference 3. Enclosure 1 of this letter updates the original response to reflect the AP1000 DCD Revision 19, these design enhancements and associated revisions to applicable calculations. This updated response replaces the original SITE-8, Item J response in RAI 13.03-061.



D093
NRO

If you have questions or require additional information, please contact Robert H. Kitchen, Nuclear Development Licensing Director, at (704) 382-4046.

I declare under penalty of perjury that the forgoing is true and correct. Executed March 12, 2015.

Sincerely,

Christopher M. Fallon

Christopher M. Fallon
Vice President
Nuclear Development

Enclosure:

1) Supplemental Information to Lee Nuclear Station Units 1 and 2 Response to Request for Additional Information (RAI) Letter No. 25, SRP Section 13.03-061 (eRAI 50), SITE-8, Item J

xc (w/o enclosure):

Frederick Brown, Deputy Regional Administrator, Region II

Brian Hughes, Senior Project Manager, DNRL

**Enclosure 1**

**Supplemental Information to**

**Lee Nuclear Station Units 1 and 2 Response to Request for Additional Information (RAI)**

**RAI Letter No. 25**

**SRP Section 13.03-061 (eRAI 50), SITE-8, Item J**

**Lee Nuclear Station Supplemental Response 2 to Request for Additional Information (RAI)**

**RAI Letter No. 025**

**NRC Technical Review Branch:** **Licensing and Inspection Branch (NSIR/DPR/LIB (EP))**

**Reference NRC RAI Number(s):** **13.03-061 (eRAI 50), SITE-8, Item J**

**NRC RAI:**
**SITE-8:** Emergency Facilities and Equipment
Basis: 10 CFR 50.47(b)(8); 10 CFR 50, Appendix E.IV.E.2; Appendix E.IV.E.3; Appendix E.IV.E.4; Appendix E.IV.E.8; Appendix E.IV.G; 10 CFR 52.79(a)(17), Three Mile Island Requirements; 10 CFR 50, Appendix E.VI Emergency Response Data System; Appendix E.VI. Maintaining Emergency Response Data System; Appendix E.VI Implementing the Emergency Response Data System Program; NUREG-0654/FEMA-REP-1; Evaluation Criterion H.1; Evaluation Criterion H.4; Evaluation Criterion H.5; Evaluation Criterion H.6; Evaluation Criterion H.8; Evaluation Criterion H.9; Evaluation Criterion H.10; Evaluation Criterion H.11
SRP ACCEPTANCE CRITERIA: Requirements A, B and E; Acceptance Criteria 1, 2, 4, 5, 12, 25, 26, 27, 28, 29

[Items A through I are omitted since they are not being addressed in this response.]

J. In accordance with SRP Chapter 15.0.3, Section II D(3), the staff reviews whether the total calculated radiological consequences in the TSC for the postulated fission product releases fall within the exposure acceptance criteria specified in GDC 19 of 5 rem TEDE (0.05 Sv) for the duration of the design basis accidents (DBAs). Provide the radiological consequence analyses for the Lee TSC for the postulated DBAs. The DBAs are listed and evaluated in Chapter 15 of the certified AP1000 DCD, Revision 15 and in the AP1000 Design Certification Amendment Application (AP1000 DCD, Revision 16). The radiological analyses must include, but not limited to, the following parameters:

   1. TSC ventilation air inlet and recirculation flow rates
   2. HEPA filter and charcoal adsorber fission product removal efficiencies
   3. TSC unfiltered air in-leakage rate
   4. Atmospheric dispersion factors ($\chi$/Q values) at TSC air intake
   5. TSC occupancy factors
   6. TSC free air volume
   7. Occupant breathing rate
   8. Description of the ventilation design

**Duke Energy Supplemental Response:**

J. Standard Review Plan 15.0.3 states that the radiation protection design of the Technical Support Center (TSC) is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the 5 Rem TEDE exposure acceptance criteria specified for the control room for the duration of the accident.

   The radiological consequence calculation for the Lee Units 1 and 2 TSC uses the methodology of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 1) and the RADTRAD (Radionuclide Transport and Removal and Dose Estimation) 3.03 Code (Ref. 2 through 4).

RADTRAD 3.03 calculates fission product transport and removal along with the resulting radiation doses at selected receptors.

The limiting AP1000 offsite radiological consequences are associated with the postulated LOCA with core melt (Ref. 5, Table 15.6.5-3). Therefore a LOCA release from the containment shell is conservatively assumed in the TSC radiological analysis. The RADTRAD 3.03 input parameters used in the Lee TSC radiological analysis are discussed below.

*Core Source Terms and Releases*

For an assumed LOCA with core melt at an AP1000, the release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity. The reactor coolant is assumed to have activity levels consistent with operation at the Technical Specification limits of 280 μCi/gm dose equivalent Xe-133 and 1.0 μCi/gm dose equivalent I-131 (Ref. 5). Based on Regulatory Guide 1.183 (Ref. 1), for a plant using leak-before-break methodology, the release of coolant into the containment can be assumed to last for ten minutes. The AP1000 is a leak-before-break plant (Ref. 5); however, for simplicity, the delay of 10 minutes before reactor coolant system blow down into the containment is conservatively neglected in this analysis.

The release of activity from the fuel takes place in two stages. First is the gap release which is assumed to occur at the end of the primary coolant release phase and to continue over a period of half an hour. The second stage is that of the in-vessel core melt in which the bulk of the activity releases associated with the accident occur. The in-vessel release phase lasts for 1.3 hours.

Core inventories of fission products ~~are from ORIGEN calculations for the AP1000 at end of the fuel cycle at 102-percent power, 3468 MWt~~ are presented in Table 15A-3 of the AP1000 Design Control Document (DCD), Revision 19 (Ref. 5). The source term model applied in the RADTRAD 3.03 calculation is based on Regulatory Guide 1.183 guidance.

The default PWR 60-isotope, 9-element NUREG-1465 nuclide data file was used for 58 of the isotopes decay and daughter data. Several isotopes provided in the AP1000 core source term were not in the RADTRAD 3.03 PWR default inventory (Cs-138, Xe-131m, Xe-133m, Xe-135m and Xe-138). The decay and daughter data for Xe-131m and Xe-133m were obtained from the RADTRAD 3.03 TID14844 default nuclide inventory file. The decay and daughter data for Xe-135m, Xe-138, and Cs-138 were obtained from values in Federal Guidance Report 11 and Federal Guidance Report 12. ~~The RADTRAD 3.03 nuclide inventory is limited to 58 isotopes with decay and daughter data. The default PWR 60-isotope, 9-element NUREG-1465 nuclide data file was used in this analysis. Data from the RADTRAD 3.03 TID14844 default data file for the Xe131m and Xe133m isotopes was added in place of the data for Co58 and Co60 in order to better represent the AP1000 inventory. A normalized core power was assumed and the AP1000 inventory for each nuclide (Ref. 5, Table 15A-3) was substituted for the default inventory.~~

The guidance in Regulatory Guide 1.183, suggests the following chemical forms for the released iodine:

**Species Distribution**

| Form | Fraction (%) |
|---|---|
| CsI as aerosol | 95 |
| Elemental | 4.85 |
| Organic | 0.15 |

Assumptions regarding release fractions applied are consistent with Regulatory Guide 1.183.

**PWR Core Inventory Fraction Released Into Containment**

| Group | Gap Release Phase | Early In-Vessel Phase | Total |
|---|---|---|---|
| Noble Gases | 0.05 | 0.95 | 1.0 |
| Halogens | 0.05 | 0.35 | 0.4 |
| Alkali Metals | 0.05 | 0.25 | 0.3 |
| Tellurium Metals | 0.00 | 0.05 | 0.05 |
| Ba, Sr | 0.00 | 0.02 | 0.02 |
| Noble Metals | 0.00 | 0.0025 | 0.0025 |
| Cerium Group | 0.00 | 0.0005 | 0.0005 |
| Lanthanides | 0.00 | 0.0002 | 0.0002 |

*Containment Sump Iodine Re-evolution*

If the pH is maintained above 7, very little (less than 1%) of the dissolved iodine will be converted to elemental iodine (Ref. 1). The AP1000 passive core cooling system provides sufficient tri-sodium phosphate to the post-LOCA cooling solution to maintain the solution pH at 7.0 or greater following a LOCA (Ref. 5). As such, this analysis did not consider any impact to the TSC due to iodine re-evolution from the containment sump.

*Dose Conversion Factors [DCFs]*

The effective dose conversion factors for the TEDE calculations are based on Federal Guidance Report (FGR) 11 (Ref. 6) and FGR 12 (Ref. 7). In most cases, these DCFs are taken directly from FGR 11 and 12; however, in some cases, the DCFs applied include the DCFs of the isotope's decay products. This is consistent with the RADTRAD 3.03 code manual as noted in NUREG/CR-6604 Table 1.4.3.3-2 (Ref. 2 through 4).

*Atmospheric Dispersion Factors*

Atmospheric dispersion factors ($\chi/Q$) values are a required input to radiological evaluations. The site-specific TSC atmospheric dispersion values determined for William States Lee III Nuclear Station are given in Table 1 below.

**Table 1**
**TSC Atmospheric Dispersion (χ/Q) Factors for Accident Dose Analysis (s/m$^3$)**

| Time Interval | Unit 1 Containment Shell Release | Unit 2 Containment Shell Release |
|---|---|---|
| 0 – 2 hours | ~~1.5~~1.31E-04 | ~~1.3~~1.07E-04 |
| 2 – 8 hours | ~~1.0E-04~~9.58E-05 | ~~1.1E-04~~8.89E-05 |
| 8 – 24 hours | ~~4.2E-05~~3.44E-05 | ~~4.5~~3.77E-05 |
| 1 – 4 days | ~~2.9~~2.78E-05 | ~~3.5~~3.16E-05 |
| 4 – 30 days | ~~2.0~~2.13E-05 | ~~2.3~~2.16E-05 |

Due to the difference in X/Q values for release from Unit 1 and Unit 2, the ~~Radtrad~~ RADTRAD 3.03 calculation was performed for each scenario varying only the X/Q data. Release from Unit 1 resulted in a higher TEDE and is therefore bounding.

~~The most limiting atmospheric dispersion values determined for each time interval for an accident at Unit 1 or Unit 2 are used to determine bounding radiological consequences for the TSC.~~

*Breathing Rate and Occupancy Factors*

The breathing rates applied in the calculation of the inhalation dose were consistent with those reported for the control room in Section 4.2.6 of Regulatory Guide 1.183 (Ref. 1) and are given in the table below.

**Breathing Rates (m$^3$/s)**

| Time Period | Control Room |
|---|---|
| 0 to 8 hours | ~~3.47~~3.5E-04 |
| 8 to 24 hours | ~~3.47~~3.5E-04 |
| 1 to 30 days | ~~3.47~~3.5E-04 |

The TSC occupancy factors are consistent with those reported for the control room in Section 4.2.6 of Regulatory Guide 1.183 and are tabulated below.

**Control Room Occupancy Factors**

| Time Period | Occupancy Factor |
|---|---|
| 0 to 24 hours | 1.0 |
| 1 to 4 days | 0.6 |
| 4 to 30 days | 0.4 |

*In-Containment Activity Removal Processes*

The AP1000 does not include active systems for the removal of activity from the containment atmosphere. However, ~~The~~ the containment atmosphere is depleted of elemental iodine and of particulates as a result of natural processes within the containment. Appendix 15B of Reference 5, the AP1000 Design Control Document (DCD), provides a discussion of the models and assumptions used in calculating the AP1000 natural deposition removal coefficients~~[1]~~. An elemental iodine deposition removal coefficient of $1.7h^{-1}$ is determined. The removal coefficient for particulates is a function of time. The aerosol removal coefficients in the AP1000 containment following a design basis LOCA with core melt are given in Table 15B-1 of Reference 5. Since there is a limit of ten time intervals in the RADTRAD 3.03 input for aerosol removal coefficients, Table 15B-1 was simplified as given in the table below. Removal coefficients were rounded to two decimal places and then conservatively small removal coefficients were selected for ten time intervals ending at 24 hours.

~~[1] This response is based on the content of the Lee COL application which incorporates by reference DCD Revision 16. See discussion at the conclusion of this response for information regarding the change in methodology presented in AP1000 DCD Revision 17.~~

**Aerosol Removal Coefficients Following a Design Basis LOCA with Core Melt**

| Time Interval (hours) | Removal Coefficient ($hr^{-1}$) |
|---|---|
| 0 - 0.631 | 0.84 |
| 0.631 – 0.801 | 0.78 |
| 0.801 – 1.171 | 0.66 |
| 1.171 – 1.475 | 0.55 |
| 1.475 – 1.776 | ~~0.45~~0.46 |
| 1.776 – 2.371 | 0.38 |
| 2.371 – 4.276 | 0.29 |
| 4.276 – 5.362 | 0.35 |
| 5.362 – 24.0 | 0.46 |
| 24 – 720 | 0.0 |

The AP1000 DCD identifies a maximum decontamination factor for elemental iodine of 200. An overall DF of 200 is achieved at 4.276 hours. Consequently, at 4.276 hours the value of the elemental spray removal coefficient, $\lambda_e$, was set to zero.

~~Credit was also assumed for aerosol removal from containment cracks representing assumed containment leakage paths. The aerosol removal efficiency due to this "impaction" removal process is given as 80% in Table 15.6.5-2 of Reference 5.~~

*TSC HVAC System*

~~The preliminary design of the TSC and TSC ventilation system is described in the Technical Support Center Design Description Document (Ref. 8).~~ The TSC is in the basement of the Maintenance Support Building located approximately 705 ft (215 meters) SSE of the Unit 1 containment shell (Ref. 8). A conceptual design for the TSC and the HVAC system is evaluated to confirm acceptability of the proposed Maintenance Support Building location. The ~~volume of the~~ TSC free air volume is modeled as ~~given as a maximum of 35,700~~60,320 ft$^3$. ~~The TSC heating, ventilation and air conditioning (HVAC) system is manually isolated from the normal outdoor air intake when a high gaseous radioactivity concentration is detected in the TSC supply air duct. A maximum normal outside airflow of 1925 cfm is assumed for 30 seconds prior to initiation of supplemental air filtration units. Two trains of filter units are provided in the TSC HVAC system for defense in depth. Only one train is needed during emergency conditions. Each train is designed to provide a total nominal flow (fresh air + recirculation) of 4,000 cfm to the TSC.~~ The TSC heating, ventilation and air conditioning (HVAC) system operates unfiltered until isolation of the TSC following the accident. After isolation of the TSC, the HVAC system operates through a filter train to reduce exposure to airborne radioactivity. The HVAC design uses a "push through" filter train arrangement that mixes the recirculation and ventilation inlet air prior to entering the air supply fan (negative side) which will then pressurize the filter train and all downstream ductwork entering the TSC boundary. This arrangement will result in minimal unfiltered air in-leakage.

The TSC is assumed to be in the normal ventilation mode at the onset of the LOCA. ~~The emergency mode is initiated based on a high radiation signal in the TSC air inlethas a setpoint of 2.0E-06 Ci/m$^3$ dose equivalent I-131. The setpoint is reached within the first minute of the LOCA; however, it~~The emergency filtered mode of operation is initiated based on a high radiation signal in the TSC air inlet or manual action. It is conservatively assumed that 75 minutes are required to manually initiate emergency air filtration, accounting for 15 minutes for notification of responders and 60 minutes for activation of the TSC. The emergency HVAC mode places the filter train in line with the airflow path as described above. Upon isolation, the maximum ~~F~~filtered fresh air intake rate is ~~limited to 860~~ 1860 cfm~~.~~ and the filtered air recirculation rate is 940 cfm. A positive pressure of at least 1/8 inch water gauge is maintained. It is assumed that ~~25~~ 40 cfm is required to maintain this positive pressure. A conservative maximum u~~U~~nfiltered air in-leakage to the TSC is assumed to be ~~given as 90~~ 280 cfm, including 10 cfm for ingress/egress inleakage; therefore the leakage from the TSC to the environment would be ~~925~~ 2100 cfm (~~860~~ 1860 cfm fresh air supply + ~~90~~ 280 unfiltered in-leakage – ~~25~~ 40 cfm pressurization). ~~Each supplemental air filtration unit includes a high efficiency filter bank, an electric heating coil, a charcoal adsorber with upstream HEPA filter bank, a downstream post-filter bank and a fan. Consistent with the main control room HVAC design, each charcoal adsorber has an efficiency of 90% for elemental and organic iodine and 99% for particulates.~~ The TSC ventilation system includes high-efficiency particulate air (HEPA) filters and charcoal filters. Each charcoal adsorber has a minimum efficiency of 90% for elemental, organic and particulate iodine.

*Containment Release Pathways*

The AP1000 containment release pathways to the environment are the containment purge line and containment leakage. During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. No credit is taken for the filters in the purge exhaust line. The containment purge flowrate is 8800 cfm. It requires 30 seconds for isolation of the purge subsequent to an accident. (Ref. 5)

The majority of the postulated AP1000 releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate, 0.1 percent by volume per day, for the first 24 hours and at half that rate for the duration of the accident, 30 days. The volume of the containment is 2.06E+06 $ft^3$. (Ref. 5)

Consistent with the AP1000 DCD, it is assumed that core cooling is accomplished by the passive core cooling system, which does not pass coolant outside of containment. Therefore, no recirculation leakage path is modeled in the TSC radiological consequence analysis.

*Other Sources of Radiation*

The direct radiation and sky-shine doses ~~reported for the control room in the AP1000 DCD, Table 15.6.5-3 (Ref. 5), are conservatively assumed to also be applicable to the TSC. This is a conservative assumption, given the proximity of the TSC to plant structures (i.e., below grade, in the basement of the Maintenance Support Building). (Ref. 8)~~were evaluated for the TSC. The direct radiation from adjacent structures was evaluated using the MicroShield 6.20 code. The sky-shine doses were evaluated using the MicroSkyshine 1.18 code.

In addition, at the time the LOCA occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The control room dose for this scenario given in the AP1000 DCD, Section 15.6.5.3.8.2 (Ref. 5) is conservatively included in the TSC dose consequences.

*TSC Radiological Consequences*

The bounding technical support center (TSC) radiological consequences determined for a postulated LOCA with core melt at either Lee Nuclear Station Units 1 or 2 are given below. The TSC radiological consequences for the postulated accident fall within the ~~GDC 19~~ 5 rem TEDE exposure acceptance ~~criteria~~criterion; therefore, it can be concluded that the radiation protection conceptual design of the TSC at the proposed Maintenance Support Building location is acceptable.

**TSC Radiological Consequences of a LOCA with Core Melt**

| TSC Dose Contributor | TEDE Dose (rem) |
|---|---|
| Airborne Activity Entering the TSC | ~~0.31~~2.72 |
| Direct Radiation from Adjacent Structures | 0.016~~15~~ |
| Sky-shine | 0.05~~1~~ |
| Spent Fuel Pool Boiling* | 0.01 |
| **Total** | ~~0.48~~2.8 |

**Assumed to be bounded by the main control room doses in DCD Table 15.6.5-3 [Reference 5]*

~~*Impact of AP1000 DCD Revision 17*~~

~~AP1000 DCD Revision 17 amends a portion of the methodology described in this response, i.e., by eliminating credit for aerosol impaction. Preliminary analyses indicate a resulting increase in doses at the TSC that nonetheless remains well below the dose criteria discussed above. These analyses will be completed and amended results will be provided when the Lee COL application is updated to reflect AP1000 DCD Revision 17.~~

**References:**


1. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”, July 2000
2. NUREG/CR-6604, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, April 1998
3. NUREG/CR-6604 Supplement 1, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, June 8, 1999
4. NUREG/CR-6604 Supplement 2, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, October 2002
5. “AP1000 Design Control Document”, Westinghouse Electric Company, Revision ~~16~~19.
6. Federal Guidance Report 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion”, Second Printing, 1989
7. Federal Guidance Report 12, “External Exposure to Radionuclides in Air, Water, and Soil”, 1993
8. ~~Lee Nuclear Station 1 & 2, COL Application, Part 5, Emergency Plan, Design Description Document, Technical Support Center, Draft~~Letter from Christopher M. Fallon (Duke Energy) to the Document Control Desk, Supplemental Information Related to Design Changes to the Lee Units 1 and 2 Physical Locations and Additional Design Enhancements, Ltr# WLG2013.05-02, dated May 02, 2013 (ML13127A224 and ML13127A225)


**Associated Revision to the Lee Nuclear Station Final Safety Analysis Report or Emergency Plan:**

None