William States Lee III Nuclear Station

COL Application

Part 7

Departures and Exemptions Requests

Revision 9

A. STD and WLS Departures

This Departure Report includes deviations in the Lee Nuclear Station COLA FSAR from the Tier 2 information in the applicable Design Control Document (DCD), pursuant to 10 CFR Part 52, Appendix D, Sections VIII and X.B.1.

The following Departures are described and evaluated in detail in this report.

Departure Number	Description
STD DEP 1.1-1	Administrative departure for organization and numbering for the FSAR sections
WLS DEP 1.8-1	Departure to correct regulatory citation error in AP1000 DCD
WLS DEP 2.0-1	Lee Site Foundation Response Spectra
WLS DEP 3.2-1	Addition of downspouts to the condensate return portion of the Passive Core Cooling System
WLS DEP 3.8-1	Lee Passive Earth Pressures
WLS DEP 3.11-1	Revision of "Envir. Zone" numbers for Spent Fuel Pool Level instruments
WLS DEP 6.3-1	Quantification of the term “indefinitely” as used in the DCD for maintenance of safe shutdown conditions using the PRHR HX during non-LOCA accidents.
WLS DEP 8.3-1	Class 1E voltage regulating transformer current limiting features
WLS DEP 18.8-1	Emergency Response Facility locations

A.1 Departures That Can Be Implemented Without Prior NRC Approval

Departure Number	Description
STD DEP 1.1-1	Administrative departure for organization and numbering for the FSAR sections
WLS DEP 1.8-1	Departure to correct regulatory citation error in AP1000 DCD
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WLS DEP 6.3-1	Quantification of the term “indefinitely” as used in the DCD for maintenance of safe shutdown conditions using the PRHR HX during non-LOCA accidents.
WLS DEP 8.3-1	Class 1E voltage regulating transformer current limiting features
WLS DEP 18.8-1	Emergency Response Facility locations

Departure Number: STD DEP 1.1-1

Affected DCD/FSAR Sections: 2.1.1, 2.1.4, 2.2.1, 2.2.4, 2.4.1, 2.4.15, 2.5, 2.5.6, 9.2.11, 9.2.12, 9.2.13, 9.5.1.8, 9.5.1.9, 13.1, 13.1.4, 13.5, 13.5.3, 13.7, 17.5, 17.6, 17.7, 17.8 (Note the affected sections may vary in subsequent COL applications, but the departure is standard)

Summary of Departure:

This FSAR generally follows the AP1000 DCD organization and numbering. Some organization and numbering differences are adopted where necessary to include additional material, such as additional content identified in Regulatory Guide 1.206.

Scope/Extent of Departure:

The renumbered sections associated with this Departure are identified in the FSAR (at the sections identified above).

Departure Justification:

An administrative departure is established to identify instances where the renumbering of FSAR sections is necessary to effectively include content consistent with Regulatory Guide 1.206, as well as NUREG-0800, Standard Review Plan (SRP).

Departure Evaluation:

This Departure is an administrative change that affects only section numbering of the indicated FSAR sections. Accordingly, it does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD;
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the plant-specific DCD;
3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD;
4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD;
5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD;
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD;
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered; or
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This Departure does not affect resolution of a severe accident issue identified in the plant-specific DCD.

Therefore, this Departure has no safety significance.

NRC Approval Requirement:

This departure does not require NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

Departure Number: WLS DEP 1.8-1

Affected DCD/FSAR Sections: DCD Tier 2 Table 1.8-1 (Sheet 6 of 6), COLA Table 1.8-203 Item 13.1 (Sheet 7 of 9)

Summary of Departure:

In Table 1.8-203, Item 13.1, revise the interface description from "Features that may affect plans for coping with emergencies as specified in 10 CFR 50, Appendix O" to read "The information pertaining to design features that affect plans for coping with emergencies in the operation of the reactor facility or a major portion thereof as specified in 10 CFR 52.137(a)(11)."

Scope/Extent of Departure:

This departure is identified in FSAR Table 1.8-203 Item 13.1.

Departure Justification:

Appendix O was transferred from Part 50 to Part 52, effective May of 1989, although the NRC neglected to physically remove the Appendix O text from Part 50. Appendix O text was not physically removed from Part 50 until the reorganization of the regulations was published in August of 2007. In the August 2007 reorganization the content of Appendix O in Part 52 was relocated to the new Subpart E of Part 52. This relocation of the regulation impacts DCD Tier 2 Table 1.8-1 (Sheet 6 of 6). There is no change in requirements, only relocation to another regulation.

Departure Evaluation:

This Departure is a correction to a regulatory citation error in the DCD. The requirements are the same. Accordingly, it does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD;
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the plant-specific DCD;
3. Result in more than a minimal increase in the consequence of an accident previously evaluated in the plant-specific DCD;
4. Result in more than a minimal increase in the consequence of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD;
5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD;
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD;
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered; or
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This Departure does not affect resolution of a severe accident issue identified in the plant-specific DCD.

Therefore, this Departure has no safety significance.

NRC Approval Requirements:

This departure does not require NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

Departure Number: WLS DEP 3.11-1

Affected DCD/FSAR Sections: DCD Table 3.11-1 (Sheet 14 of 51)

Summary of Departure:

DCD Table 3.11-1 (Sheet 14 of 51) "Envir. Zone" numbers for Spent Fuel Pool Level instruments SFS-JE-LT019A, SFS-JE-LT019B, and SFS-JE-LT019C are changed to correct an inconsistency in the DCD. All 3 instruments currently have an Environmental Zone number of "11". SFS-JE-LT019A is changed to Envir. Zone 6, SFS-JE-LT019B is changed to Envir. Zone 7 and SFS-JE-LT019C is changed to Envir. Zone 6 in DCD Table 3.11-1 (Sheet 14 of 51).

Scope/Extent of Departure:

SFS-JE-LT019A is revised to Envir. Zone 6, SFS-JE-LT019B is revised to Envir. Zone 7 and SFS-JE-LT019C is revised to Envir. Zone 6 in DCD Table 3.11-1 (Sheet 14 of 51).

Departure Justification:

The actual location of the Spent Fuel Pool Level instruments is not being changed from the designed location in this departure. The environmental zones the instruments are located in are being revised to be consistent with the designed instrument location. The AP1000 SFP level transmitters are located in rooms outside of the Fuel Handling Area in the Auxiliary Building. Per Westinghouse design documents, Spent Fuel Pool Level channels 019A and 019C are in room 12365 and channel 019B is in room 12341. Room 12365 is in Zone 6 on DCD Table 3D.5-1 (Sheet 2 of 3). Room 12341 is in Zone 7 on DCD Table 3.D.5-1 (Sheet 2 of 3). Based on this information, SFS-JE-LT019A is being changed to Envir. Zone 6, SFS-JE-LT019B is being changed to Envir. Zone 7 and SFS-JE-LT019C is being changed to Envir. Zone 6 in DCD Table 3.11-1 (Sheet 14 of 51).

DCD Table 3.11-1 Environmental Zone numbers for Spent Fuel Pool Level provide a reference to environmental conditions in the associated instrument location correlated to an environmental zone in DCD Table 3D.5-1 for "Normal Operating Environments", DCD Table 3D.5-4 for "Abnormal Operating Environments Outside Containment" and DCD Table 3D.5-5 for "Accident Environments". The environmental qualification of the instrument is consistent with conditions identified for the associated environmental zone. Revising the Spent Fuel Pool Level instruments' environmental zone to accurately reflect their actual location will ensure they are environmentally qualified to function properly during normal, abnormal, and accident conditions.

Departure Evaluation:

This Tier 2 departure revises SFS-JE-LT019A Envir. Zone from 11 to 6, SFS-JE-LT019B Envir. Zone from 11 to 7, and SFS-JE-LT019C Envir. Zone from 11 to 6 in DCD Table 3.11-1 (Sheet 14 of 51). This departure does not result in any adverse affects to the SFP level indication design function and does not change the environmental qualification methodology. Therefore, this departure does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD.
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety and previously evaluated in the plant-specific DCD.
3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD.
4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD.

5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD.
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD.
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered.
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This departure does not affect resolution of a severe accident issue identified in the plant-specific DCD. Therefore, this departure has no safety significance.

NRC Approval Requirement:

This departure does not require NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

Departure Number: WLS DEP 6.3-1

Affected DCD/FSAR Sections: Subsection 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, and 6.3.3.4.1, Subsection 7.4.1.1, Table 9.5.1-1 (Sheet 11), Subsection 15.2.6.1, Table 19.59-18 (Sheet 6), and Subsection 19E.4.10.2

Summary of Departure:

The Passive Residual Heat Removal Heat Exchanger (PRHR HX) has a functional requirement to be able to bring the AP1000 plant to a stable condition for events not involving a loss of coolant (i.e., non-LOCA event), DCD 6.3.1.1.4. The DCD in Subsection 6.3.1.1.1 further states “The PRHR HX in conjunction with the passive containment cooling system, is designed to remove decay heat for an indefinite time in a closed-loop mode of operation.” Additional evaluations have been subsequently performed that have identified that the use of the term “indefinite” does not describe the predicted PRHR HX long term operation properly. The word “indefinite” can be defined as an “unknown” or “unidentified” length of time; “indefinite” does not mean “infinite” which means having no boundaries or limits in time. The word “indefinite” in regards to PRHR HX long term operation needs to be changed with a definitive time period.

Scope/Extent of Departure:

There are additional areas in the DCD that use the term “indefinite” in reference to long term PRHR HX operation that need to be changed in a departure to the DCD to more accurately reflect the PRHR HX long term operation during a non-LOCA event. The changes needed for the DCD departure WLS DEP 6.3-1 to incorporate this information include the following FSAR Sections or Tables:

Section 5.4.14.1
Section 6.3.1.1.1
Section 6.3.1.2
Section 6.3.1.3
Section 6.3.2.1.1
Section 6.3.3.4.1
Section 7.4.1.1
Table 9.5.1-201
Section 15.2.6.1
Table 19.59-201
Section 19E.4.10.2

Departure Justification:

Recent PRHR HX evaluations performed under a variety of operating scenarios identified 14 days would be a conservative replacement time period for “indefinite”. The Westinghouse evaluation of the PRHR HX operation under non-bounding, conservative conditions demonstrates the ability to keep the average RCS temperature in safe shutdown conditions for greater than 14 days under passive conditions (no operator action). The evaluation does indicate that if no action is taken, the average RCS temperature will increase at some point after 15 days but the PRHR HX operation would still keep the average RCS temperature below 420°F for a longer period of time of approximately 20 days (420°F is identified as the RCS temperature objective for safe shutdown). If no action is able to be taken after this period of time and there is adverse trending of RCS conditions that might be indicative of leading to an unstable condition, the operators do still have the ability to initiate Automatic Depressurization System (ADS), go to open loop cooling and retain the plant in a stable condition.

Departure Evaluation:

This Tier 2 departure is associated with defining the term “indefinite” as a conservative but specific duration (greater than 14 days). The departure results in a change to the DCD that does not impact the required design function (i.e., containment cooling condensate return). Accordingly, it does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD.
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety and previously evaluated in the plant-specific DCD.
3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD.
4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD.
5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD.
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD.
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered.
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This departure does not affect resolution of a severe accident issue identified in the plant-specific DCD. Therefore, this departure has no safety significance.

NRC Approval Requirement:

This departure does not require NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

Departure Number: WLS DEP 8.3-1

Affected DCD/FSAR Sections: 8.3.2.2

Summary of Departure:

The DCD states that the Class 1E battery chargers and Class 1E voltage regulating transformers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side. However, the AP1000 voltage regulating transformers do not have active components to limit current.

Scope/Extent of Departure:

This departure is identified in FSAR Subsection 8.3.2.2.

Departure Justification:

DCD section 8.3.2.2 states that the Class 1E voltage regulating transformers have built-in circuit breakers at the input and output sides for protection and isolation. The circuit breakers are coordinated and periodically tested to verify their designed coordination and isolation function. They are qualified as isolation devices between Class 1E and non-Class 1E circuits in accordance with IEEE 384 and Regulatory Guide 1.75. Since the isolation and protection function is provided by the breakers, there is no need for the voltage regulating transformers to have current limiting capability. This departure does not adversely affect any safety-related system, nor does it conflict with applicable regulatory guidance.

Departure Evaluation:

This Tier 2 departure is associated with isolation between Class 1E loads and the non-Class 1E ac power source. The departure results in a change to the DCD that does not impact the required design function (i.e., isolation). Accordingly, it does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD;
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the plant-specific DCD;
3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD;
4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD;
5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD;
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD;
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered; or
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This Tier 2 departure does not affect resolution of an ex-vessel severe accident design feature identified in the plant-specific DCD.

Therefore, this departure has no safety significance.

NRC Approval Requirement:

This departure does not require NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

Departure Number: WLS DEP 18.8-1

Affected DCD/FSAR Sections: 1.2.3, 12.3.1.2, 12.5, 12.5.2.2, 9A.2.1, 18.8.3.5, 18.8.3.6

Summary of Departure:

At Lee Nuclear Station, the Technical Support Center (TSC) is not located in the control support area (CSA) as identified in DCD Subsection 18.8.3.5; the TSC location is as described in the Emergency Plan. Additionally, the Operations Support Center (OSC) is also being moved from the location identified in DCD Subsections 12.5.2.2 and 18.8.3.6 and as identified on DCD Figures 1.2-18, 9A-3 (Sheet 1 of 3), 12.3-2 Sheet 11 of 15), and 12.3-3 (Sheet 11 of 16); the OSC location is as described in the Emergency Plan.

Scope/Extent of Departure:

This departure is identified in FSAR Subsection 18.8.

Departure Justification:

The referenced DCD states "The TSC is located in the control support area (CSA)." This is not the case for Lee Nuclear Station. The TSC location is moved to a central location such that a single TSC can serve both Lee Nuclear Station Units 1 and 2 as identified in the Emergency Plan. The referenced DCD also states "The ALARA briefing and operational support center is located off the main corridor immediately beyond the main entry to the annex building" and indicates that the OSC location is identified on Figure 1.2-18. However, the OSC is being moved to the control support area vacated by the move of the TSC in order to better utilize the now available space.

Departure Evaluation:

This Departure is for a non-safety-related system, and the alternate locations of the TSC and OSC meet applicable requirements. Relocating the TSC and OSC does not adversely affect their function and therefore this Departure does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD;
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the plant-specific DCD;
3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD;
4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD;
5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD;
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD;
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered; or
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This Departure does not affect resolution of a severe accident issue identified in the plant-specific DCD.

Therefore, this Departure has no safety significance.

NRC Approval Requirement:

This departure does not require NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

A.2 Departures That Require NRC Approval Prior to Implementation

Departure Number	Description
WLS DEP 2.0-1	Lee Site Foundation Response Spectra
WLS DEP 3.2-1	Addition of downspouts to the condensate return portion of the Passive Core Cooling System
WLS DEP 3.8-1	Lee Passive Earth Pressures

Departure Number: WLS DEP 2.0-1

Affected DCD/FSAR Sections: FSAR Table 2.0-201, Subsections 3.7.1.1.1, 3.7.2.8.4, 3.7.2.15, Appendix 3I, and 19.55.6.3

Summary of Departure:

The seismic design of the AP1000 standard plant is based on the Certified Seismic Design Response Spectra (CSDRS) as addressed in DCD Subsection 3.7.1.1. The newly released CEUS-SSC model as specified in NUREG-2115, “Central and Eastern United States Seismic Source Characterization for Nuclear Facilities,” is considered part of the Lee Nuclear Station site-specific design requirements. Consideration of the CEUS-SSC model along with local and regional refinements results in updated seismic hazards and updated site-specific foundation response spectra (i.e., ground motion response spectra (GMRS), foundation input response spectra (FIRS), and Nuclear Island FIRS [envelope of GMRS and FIRS]) for Lee Nuclear Station that exceed the AP1000 CSDRS.

As shown on Figure 3.7-201, the horizontal Nuclear Island (NI) FIRS for Lee Nuclear Station exceeds the horizontal AP1000 CSDRS at frequencies above approximately 14 hertz. Peak Ground Acceleration (PGA) at 100 hertz for the horizontal NI FIRS is 0.352 g which exceeds the AP1000 CSDRS PGA of 0.30g. Figure 3.7-201 also shows that the horizontal NI FIRS exceeds the horizontal AP1000 HRHF spectrum at all frequencies above approximately 3 hertz. As shown on Figure 3.7-202, the vertical NI FIRS for Lee Nuclear Station exceeds the vertical AP1000 CSDRS at frequencies above approximately 16 hertz. Figure 3.7-202 also shows that the vertical NI FIRS exceeds the vertical AP1000 HRHF spectrum at frequencies between approximately 3 to 55 hertz and 80 to 100 hertz.

The Lee Nuclear Station site provides uniform hard rock support for the Nuclear Island, but the site characteristic horizontal and vertical spectra (NI FIRS) exceed the horizontal and vertical AP1000 HRHF spectra considered in DCD Appendix 3I. Site-specific analysis of the Nuclear Island and Seismic Category II (SC-II) Adjacent Buildings to demonstrate the adequacy of the design is allowed under the requirements of DCD Table 5.0-1, “Site Parameters for Seismic” (Tier 1), DCD Subsection 2.5.2.1, Paragraph 4b, and DCD Subsection 3.7.2.8.4. The DCD allows a Combined License (COL) applicant to perform a site-specific dynamic analysis of the

AP1000 Nuclear Island and SC-II Adjacent Buildings if the site-specific spectra exceed the AP1000 CSDRS.

Since the site-specific seismic response spectra and accelerations for the Lee Nuclear Station site resulting from the application of the CEUS-SSC model exceed the spectra and accelerations evaluated and documented in the AP1000 DCD, this constitutes a departure from the AP1000 certified design. In order to address these changes in seismic response spectra and accelerations, the following departure from the AP1000 DCD is required.

As indicated above, the Lee Nuclear Station site-specific foundation response spectra and accelerations depart from the DCD in that the horizontal and vertical foundation response spectra (GMRS [Unit 2 FIRS] and Unit 1 FIRS) exceed the AP1000 CSDRS at frequencies above 14 and 16 hertz respectively. The PGA for the horizontal and vertical foundation response spectra exceed the PGA for the horizontal and vertical AP1000 CSDRS. The alternative HRHF response spectra evaluated in DCD Appendix 3I as part of AP1000 design certification are also exceeded by the horizontal and vertical foundation response spectra. Therefore, a reevaluation of the AP1000 standard plant design is required. This departure affects information presented in Table 2.0-201, “Comparison of AP1000 DCD Site Parameters and Lee Nuclear Station Units 1 & 2 Site Characteristics;” Subsection 3.7.1.1.1, “Design Foundation Response Spectra;” Subsection 3.7.2.8.4, “Seismic Modeling and Analysis of Seismic Category II Building Structures,” Subsection 3.7.2.15, “Site-Specific Analysis of Nuclear Island Seismic Category I Structures;” DCD Appendix 3I, “Evaluation For High Frequency Seismic Input;” and Subsection 19.55.6.3, “Site Specific Seismic Margin Analysis.”

Scope / Extent of Departure:

This departure is identified in FSAR Table 2.0-201 and Subsections 3.7.1.1.1, 3.7.2.8.4, 3.7.2.15, Appendix 3I and 19.55.6.3.

Departure Justification:

The Lee Nuclear Station site-specific foundation response spectra and peak ground acceleration depart from the DCD in that they exceed the AP1000 CSDRS. The Lee Nuclear Station site-specific response spectra also exceed the AP1000 generic hard rock high frequency response spectra. Therefore, a site-specific analysis of the AP1000 Nuclear Island was performed, similar to the analysis described in DCD Appendix 3I to demonstrate that dynamic loads from these high frequency spectra exceedances are within the seismic design margin of the AP1000 certified design.

The AP1000 certified design had previously been analyzed for high frequency (HF) exceedances as part of qualifying the AP1000 standard plant design for the HRHF spectra. The Lee Nuclear Station NI FIRS and the AP1000 HRHF spectra are very similar. Therefore, the same general screening criteria documented in DCD Appendix 3I was used to identify a representative sample of structures, components, supports, piping and equipment to evaluate to demonstrate the acceptability of the AP1000 certified design for the Lee Nuclear Station HF motion. A brief summary of the general screening criteria used to identify a representative sample for evaluation follows:

- Based on their importance to safety and the ability to achieve safe shutdown.
- According to location in areas of the plant that are susceptible to large HF seismic inputs.
- Have exhibited significant modal response within the region of HF amplification.
- Possession of significant total stress as compared to allowable, when considering load combinations that include seismic.

The Lee Nuclear Station site-specific analysis determined that the HF exceedances result in only a minimal increase in the evaluated dynamic loads and would not result in any damage to the AP1000 structures, systems or components (SSCs). The site-specific analysis includes evaluations of building structures, reactor pressure vessel internals, primary component supports, primary loop nozzles, piping and electro-mechanical equipment. Both the Lee Nuclear Station site-specific building forces and equipment support forces are enveloped by the CSDRS building forces and equipment support forces.

The site-specific evaluation reviewed the current AP1000 electro-mechanical equipment qualification test methods and requirements, and compared those requirements to the Lee Nuclear Station site-specific requirements. There are a few minor exceedances between the site-specific in-structure response spectra and the comparable AP1000 qualification required response spectra (RRS) envelopes. However, in all cases the test response spectra (TRS) used in actual equipment qualification testing exceed the site-specific demand by a significant margin.

Site-specific evaluation of the Lee Nuclear Station SC-II adjacent buildings concludes the following:

- The Lee Nuclear Station SC-II Turbine Building 1st Bay and Annex Building backfill configuration and backfill properties are uniform and consistent with those evaluated in the AP1000 DCD.
- The bearing capacity of the Lee Nuclear Station SC-II Turbine Building 1st Bay and Annex Building backfill material is greater than the corresponding calculated bearing demand.
- The maximum relative horizontal seismic displacements between the Lee Nuclear Station SC-II Turbine Building 1st Bay and Annex Building and the nuclear island are much less than the 2-inch foundation and 4-inch top gap clearances provided.
- The site-specific Lee Nuclear Station horizontal SC-II foundation input response spectra are very similar to the AP1000 SC-II design envelope foundation spectra, and are of primary importance in assessing potential interactions between nuclear island and the SC-II adjacent structures.
- The site-specific Lee Nuclear Station vertical SC-II foundation input response spectra exceed the corresponding AP1000 SC-II design envelope foundation spectra, but are of less importance for potential interactions between the nuclear island and the SC-II adjacent structures.

Duke Energy will ensure that the SC-II adjacent buildings are designed for the calculated site-specific foundation spectra to satisfy all AP1000 DCD criteria to confirm that they do not interact with the nuclear island.

The site-specific analyses and evaluations conclude that the proposed change does not result in an adverse effect on any plant-specific DCD described design function.

Departure Evaluation:

This Tier 2 departure adds a discussion of the application of the Lee Nuclear Station site-specific response spectra, developed by applying the CEUS-SSC model, for determining the horizontal and vertical seismic demand on the AP1000 Nuclear Island. This departure does not result in any adverse effects on the safety functions of SSCs as demonstrated by the Lee Nuclear Station site-specific evaluation reports (WLG-GW-GLR-815, “Effects of William S. Lee Site-Specific Seismic Requirements on AP1000 SSCs” and WLG-1000-S2R-804, Rev 3, “William S. Lee Site Specific Adjacent Buildings Seismic Evaluation Report”). The site-specific analyses include evaluation of building structures, reactor pressure vessel internals, primary component supports, primary loop nozzles, piping and electro-mechanical equipment.

1. Site-specific analysis confirms that the exceedances in seismic response spectra and peak ground accelerations result in only minimal increase in the evaluated dynamic loads for AP1000 SSCs, and do not result in damage to the analyzed SSCs. Site-specific analysis also confirms that relative building displacements are small and there are no interactions between the Nuclear Island and SC-II adjacent buildings. There is no change associated with or impact to AP1000 SSC design features or functions. This change has no impact on the frequency of occurrence of an accident previously evaluated in the plant-specific DCD. Therefore there is not more than a minimal increase in the frequency of occurrence.
2. This change does not impact the likelihood of a malfunction of an SSC important to safety and previously evaluated in the plant-specific DCD. Evaluation coupled with analysis indicates that the high-frequency exceedances in seismic response result in minimal increases in the evaluated dynamic loads for AP1000 building structures, reactor pressure vessel internals, primary component supports, primary loop nozzles, piping, and electro-mechanical equipment, and do not result in damage to the analyzed SSCs. Also evaluations determined that the small site-specific spectral exceedances are bounded by the actual test response spectra (TRS) used to qualify high frequency sensitive electrical equipment; therefore, no damage will result to electro-mechanical equipment. Evaluation determined that there are no interactions between the Nuclear Island and the SC-II adjacent buildlings.
3. This change has no impact on AP1000 design capacities, source term inventories, or evaluated release rates. Therefore, this change does not increase the consequences of an accident previously evaluated.
4. Site-specific analysis confirms that the exceedances result in minimal increases in the evaluated dynamic loads for AP1000 SSCs, and do not result in damage to the analyzed SSCs. Site-specific analysis also determined that there are no interactions between the Nuclear Island and SC-II adjacent buildings. There is no change associated with the AP1000 SSC design capacities or source term inventories. Therefore, there is no increase in the consequences of a malfunction of an SSC important to safety.
5. There are no changes associated with the AP1000 design that introduce the possibility of new or different types of accidents. Therefore the change does not create a possibility for an accident of a different type than previously evaluated in the plant-specific DCD.
6. There are no changes associated with AP1000 design that introduce the possibility of new or different types of malfunction. Therefore, there will not be a malfunction of an SSC important to safety with a different result than any previously evaluated in the plant-specific DCD, since analysis along with electrical equipment being qualified to a robust test response spectra (TRS) indicates that the HF exceedances in seismic response result in minimal effects on important to safety SSCs and analysis confirms that there are no interactions between the Nuclear Island and SC-II adjacent buildings.
7. Other than the design basis seismic source, there are no changes associated with AP1000 design basis limits. This change does not result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered.

8. The proposed change is based on an evaluation methodology that is described in the plant-specific DCD and NRC requirements, and thus is not a revision or replacement of a plant-specific DCD described evaluation methodology; nevertheless, since the site-specific response spectra are not bounded by the CSDRS, the response to this question is determined to be yes. (The methodology is consistent but the inputs are selected to be site-specific).

The site-specific analyses demonstrate that the minor increase in dynamic loads due to high frequency spectra and peak ground acceleration exceedances are within the seismic design capacity of the AP1000 certified design and that there are no interactions between the Nuclear Island and SC-II adjacent buildings; therefore, the departure does not affect a resolution of an ex-vessel severe accident design feature identified in the DCD. Therefore, this departure has no safety significance.

NRC Approval Requirement:

This departure requires NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

Departure WLS DEP 3.2-1 is a departure from AP1000 Tier 1 information, in addition to Tier 2 information in the DCD; an exemption request and NRC approval is required prior to implementation.

Departure Number: WLS DEP 3.2-1

Affected DCD/FSAR Sections: Tier 1 Table 2.2.3-1 and Table 2.2.3-2, Tier 2 Table 3.2-3 (Sheet 16 of 75), Figure 3.8.2-1 (Sheet 3), Subsections 5.4.11.2 and 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.1.4, 6.3.1.1.6, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, and 6.3.3.2.1.1, Chapter 6, Figure 6.3-1 (Sheets 1 through 3), Figure 6.3-2 (Not Used), Subsection 7.4.1.1, Table 14.3-2 (Sheets 7 and 8 of 17), Subsection 15.0.13, Chapter 16 (TS Bases B 3.3.3 and B 3.5.4), Subsections 19E.4.10.2 and 19E.9, Table 19E.4.10-1, and Figures 19E.4.10-1 through 19E.4.10-4

Summary of Departure:

Modifications to the Polar Crane Girder (PCG), Internal Stiffener, and Passive Core Cooling System (PXS) gutters were made. The fabrication holes at the top surface of the PCG and in the stiffener are blocked, drainage holes in the bottom of the PCG boxes are blocked, and flow communication holes between PCG boxes are added. A downspout piping network is added to collect and transport condensation from the top and interior of the PCG and the stiffener to the PXS Collection Boxes. Eight new PXS downspout screens are added at the entrance of each of the downspouts at the top of the PCG and the stiffener to prevent any larger debris from blocking the downspout piping. Visual inspection requirements to verify that the return flow to the IRWST will not be restricted by debris have been added to Technical Specification Bases.

Scope/Extent of Departure:

Upon actuation of the Passive Residual Heat Removal Heat Exchanger (PRHR HX), a series of air-operated valves are actuated to isolate the normal gutter drain path to the Liquid Radwaste System, and divert condensation to the In-containment Refueling Water Storage Tank (IRWST). It is important that sufficient condensate return is achieved during non-loss of coolant accident (LOCA) PRHR HX operation, since reduction of IRWST level to below the top of the tubes will begin to degrade the heat exchanger performance to the point where safe shutdown (<420 deg F in <36 hours) may not be achieved.

As steaming in the containment begins, following initiation of PRHR HX operation and saturation of the IRWST, there are a number of mechanisms, both thermodynamic and geometric, that can prevent the condensed steam from returning to the IRWST. The mechanisms are as follows:

a. Steam to pressurize the containment
b. Steam condensation on Passive Heat Sinks
c. Raining from the containment roof, Containment ring misalignment
d. Losses at the Polar Crane Girder and Stiffener
e. Losses at support plates attached to the containment vessel
f. Losses at the Equipment Hatch and Personnel Airlock
g. Losses at entry to IRWST gutter

Losses due to pressurization and condensation on heat sinks are quantified with development of two new calculations. Two additional existing calculations have been revised based on the results of the new calculations in order to quantify the PRHR HX performance with the revised value of the condensate return and to ensure that the safe shutdown requirements are met. A full scale section of the containment wall was constructed to test condensate losses.

As a result of the condensate return testing, modifications to the Polar Crane Girder (PCG), Internal Stiffener, and Passive Core Cooling System (PXS) gutter designs are made. The fabrication holes at the top surface of the PCG and in the stiffener are blocked, drainage holes in the bottom of the PCG boxes are blocked, and flow communication holes between PCG boxes are added. A downspout piping network is added to collect and transport condensation from the top and interior of the PCG and the stiffener to the PXS Collection Boxes. Eight new PXS downspout screens are added at the entrance of each of the downspouts at the top of the PCG and the stiffener to prevent any larger debris from blocking the downspout piping. Visual inspection requirements to verify that the return flow to the IRWST will not be restricted by debris have been added to Technical Specification Bases.

Departure Justification:

The proposed change does not involve a significant reduction in the margin of safety. The proposed change does not reduce the redundancy or diversity of any safety-related SSCs. The proposed changes increase the amount of condensate available in the IRWST after the initiation of a design basis event compared to the design described in the AP1000 DCD Revision 19. Though the fraction of condensate returned is smaller than originally assumed, the proposed changes provide sufficient condensate return flow to maintain adequate IRWST water level for those events using the PRHR HX cooling function. While lower condensate return rates result in an earlier transition to PRHR HX uncovery, the long-term shutdown temperature evaluation results show that the PRHR HX would continue to meet its acceptance criteria.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) approval of the change will not be inimical to the common defense and security or to the health and safety of the public.

Departure Evaluation:

This Tier 2 departure performs modifications to the PCG, Internal Stiffener, and PXS gutter designs. The fabrication holes at the top surface of the PCG and in the stiffener are blocked, drainage holes in the bottom of the PCG boxes are blocked, and flow communication holes between PCG boxes are added. A downspout piping network is added to collect and transport condensation from the top and interior of the PCG and the stiffener to the PXS Collection Boxes. Eight new PXS downspout screens are added at the entrance of each of the

downspouts at the top of the PCG and the stiffener to prevent any larger debris from blocking the downspout piping. Visual inspection requirements to verify that the return flow to the IRWST will not be restricted by debris have been added to Technical Specification Bases. The proposed change does not involve a significant reduction in the margin of safety. The proposed change does not reduce the redundancy or diversity of any safety-related SSCs. The proposed changes increase the amount of condensate available in the IRWST after the initiation of a design basis event compared to the original design. Though the fraction of condensate returned is less than assumed in the original design, the proposed design does not result in significantly degraded overall PXS performance, in that the ability to achieve safe shutdown within the required time frame is accomplished. Therefore, this departure does not:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD.
2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety and previously evaluated in the plant-specific DCD.
3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD.
4. Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD.
5. Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD.
6. Create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD.
7. Result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered.
8. Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This Departure does not affect resolution of a severe accident issue identified in the plant-specific DCD. Therefore, this Departure has no safety significance.

NRC Approval Requirement:

This departure requires an exemption from the requirements of 10 CFR Part 52, Appendix D, Section III.B, which requires compliance with Tier 1 requirements of the AP1000 DCD. Therefore, an exemption is requested in Part B of this COL Application Part.

<u>Departure Number: WLS DEP 3.8-1</u>

Affected DCD/FSAR Sections: 3.8.4.4.4, Figure 3.8-201a, Figure 3.8-202a, Figure 3.8-203, and Figure 3.8-204

Summary of Departure:

The Lee Nuclear Station site-specific lateral earth pressures on the nuclear island below-grade walls for Load Combinations (LC) 1 through 9 in DCD Table 3.8.4-2 are evaluated and are compared to the corresponding pressures that were used in the AP1000 standard design. Site-specific lateral pressures on the nuclear island exterior walls below grade are bounded by the AP1000 design pressures for load combinations 1, 2, 3, 4, 5, 6, 8 and 9 in both the east-west (E-W) and north-south (N-S)directions. The site-specific lateral pressure in Load Combination 7 (LC7) for the GW backfill material slightly exceeds the AP1000 LC7 lateral pressure.

LC7 includes the summation of the full passive lateral earth pressure, the static and dynamic lateral surcharges, and the water pressure. The difference in LC7 for the Lee Nuclear Station site compared to the AP1000 generic site is mainly attributed to the site-specific ground water

level, which in its shallowest/closest to ground surface condition, is at 8.0 feet below ground surface (bgs) as compared to the AP1000 ground water level of 2 feet bgs. This difference results in six additional feet of non-buoyant (heavier) soil than considered in the AP1000 standard evaluations, resulting in a corresponding higher passive earth pressure component in the LC7 equation.

Since the site-specific lateral earth pressures for the Lee Nuclear Station site resulting from assuming full passive earth pressures exceed the design pressures evaluated and documented in the AP1000 DCD, this constitutes a departure from the AP1000 certified design. In order to address these changes, the following departure from the AP1000 DCD is required.

As indicated above, the Lee Nuclear Station site-specific lateral pressure in LC7 for the GW backfill material slightly exceeds the AP1000 LC7 lateral pressure. Therefore, a reevaluation is required. This departure affects information presented in Subsection 3.8.4.4.4, “Below Grade Exterior Walls.”

Scope / Extent of Departure:

This departure is identified in FSAR Subsection 3.8.4.4.4 and Figures 3.8-201a, 3.8-202a, 3.8-203, and 3.8-204.

Departure Justification:

The Lee Nuclear Station site-specific lateral earth pressure on below-grade walls (assuming full passive earth pressures) depart from the DCD in that it exceeds the AP1000 design pressure evaluated and documented in the AP1000 DCD. Therefore, a site-specific analysis was performed to demonstrate that actual Lee Nuclear Station site-specific lateral pressures are bounded by the AP1000 certified design.

Passive pressure, which resists the lateral movement of the nuclear island, is not relied upon for resistance to sliding during a seismic event. Further, development of the full passive pressure requires more displacement than the nuclear islands will experience during a seismic event. As described in DCD Subsection 3.8.5.5.5, the maximum lateral displacement at the base of the nuclear island when subjected to the CSDRS is expected to be 0.12 inches neglecting buoyancy of the nuclear island, and 0.19 inches considering buoyancy effects. Since overall site-specific analyses of the nuclear island demonstrate that the effects of the CSDRS bound those of the Lee Nuclear Station NI FIRS, the Lee site-specific lateral displacements will be even less. Small lateral displacements such as these are not capable of developing the full passive earth pressure. Therefore, the site-specific nuclear island below-grade wall pressures resulting from the NI FIRS will be less than those used in the standard AP1000 design for this load combination.

The site-specific analyses and evaluations conclude that the proposed change does not result in an adverse effect on any plant-specific DCD described design function.

Departure Evaluation:

This Tier 2 departure adds a discussion of the application of the Lee Nuclear Station site-specific lateral pressure. This departure does not result in any adverse effects on the safety functions of SSCs as demonstrated by the Lee Nuclear Station site-specific evaluation report (“William S. Lee Site-Specific Assessment of Lateral Earth Pressure Loads Due to 2012/2013 CEUS Ground Motion Seismic Input," WLG-1000-S2R-806 Rev. 2, November, 2013).

1. Site-specific analysis confirms that the Lee Nuclear site will never develop the theoretical full passive earth pressure since the Lee Nuclear Station lateral displacements are small. Therefore, the AP1000 nuclear island walls below grade

are determined to be safe for the Lee Nuclear site considering the site-specific loadings that are expected to occur. There is no change associated with or impact to AP1000 SSC design features or functions. This change has no impact on the frequency of occurrence of an accident previously evaluated in the plant-specific DCD. Therefore there is not more than a minimal increase in the frequency of occurrence.

2. This change does not impact the likelihood of a malfunction of an SSC important to safety and previously evaluated in the plant-specific DCD. Evaluation coupled with analysis indicates that the Lee Nuclear site will never develop the theoretical full passive earth pressure. Therefore, the AP1000 nuclear island walls below grade are determined to be safe for the Lee Nuclear site and maintain its design feature of protecting SSCs important to safety during a design basis event.
3. This change has no impact on AP1000 design capacities, source term inventories, or evaluated release rates. Therefore, this change does not increase the consequences of an accident previously evaluated.
4. Site-specific analysis confirms that the AP1000 nuclear island walls below grade are safe for the Lee Nuclear site considering the site-specific loadings that are expected to occur and do not result in damage to SSCs important to safety. There is no change associated with the AP1000 SSC design capacities or source term inventories. Therefore, there is no increase in the consequences of a malfunction of an SSC important to safety.
5. There are no changes associated with the AP1000 design that introduce the possibility of new or different types of accidents. Therefore the change does not create a possibility for an accident of a different type than previously evaluated in the plant-specific DCD.
6. There are no changes associated with AP1000 design that introduce the possibility of new or different types of malfunction. Therefore, there will not be a malfunction of an SSC important to safety with a different result than any previously evaluated in the plant-specific DCD, since analysis determines that the AP1000 nuclear island walls below grade are safe for the Lee Nuclear site during design basis events.
7. Other than the design basis lateral earth pressures on the AP1000 nuclear island wall below grade, there are no changes associated with AP1000 design basis limits. This change does not result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered.
8. The proposed change is based on an evaluation methodology as described in the plant-specific DCD and NRC requirements, and thus is not a revision or replacement of a plant-specific DCD described evaluation methodology; nevertheless, since the site-specific lateral pressure in LC7 for the Lee Nuclear Station GW backfill material slightly exceed the AP1000 LC7 lateral pressure by about 3 to 8 percent in the E-W and N-S directions, respectively, the response to this question is determined to be yes. (The methodology is consistent but the inputs are selected to be site-specific).

The site-specific analyses demonstrate that expected lateral pressures are within the design capacity of the AP1000 certified design; therefore, the departure does not affect a resolution of an ex-vessel severe accident design feature identified in the DCD. Therefore, this departure has no safety significance.

NRC Approval Requirement:

This departure requires NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

B. Lee Nuclear Station Exemption Requests

Duke requests the following exemption related to:

1) Combined License Application Organization and Numbering
2) Special Nuclear Material (SNM) Material Control and Accounting Program Description
3) Containment Cooling Changes in regard to Passive Core Cooling System Condensate Return

Discussion and justifications for this request are provided in the following pages.

1) Combined License Application Organization and Numbering (Part 52, Appendix D)

Applicable Regulation(s): 10 CFR Part 52, Appendix D, Section IV.A.2.a

Specific wording from which exemption is requested:

> IV. Additional Requirements and Restrictions
> A. An applicant for a combined license that wishes to reference this appendix shall, in addition to complying with the requirements of 10 CFR 52.77, 52.78, and 52.79, comply with the following requirements:
> 1. Incorporate by reference, as part of its application, this appendix.
> 2. Include, as part of its application:
> a. A plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the AP1000 design, as modified and supplemented by the applicant's exemptions and departures;

Pursuant to 10 CFR 52.7 and 52.93 (as amended and promulgated effective Sept. 27, 2007), Duke Energy Carolinas, LLC (Duke) requests an exemption from the requirement of 10 CFR 52, Appendix D, Section IV.A.2.a, to include a plant-specific DCD "containing the same type of information and using the same organization and numbering as the generic DCD for the AP1000 design…." While the William States Lee III Nuclear Station Units 1 and 2 (Lee Nuclear Station) plant-specific DCD (i.e., the final safety analysis report) does contain the same type of information and generally follows the same organization and numbering as the generic DCD for the AP1000 design, some limited subsections of the FSAR (as identified in the departures report as item STD DEP 1.1-1) do not follow the "same organization and numbering as the generic DCD for the AP1000 design." Duke proposes to provide the plant-specific DCD (i.e., FSAR) with some administrative revisions to the organization and numbering of the AP1000 DCD.

Discussion:

The AP1000 Design Control Document (DCD) generally has an organization and numbering format that provides text by subject in general conformance with the Standard Review Plan (SRP) in effect at the time the DCD was written. Generally, Combined License information items are included at the end of a chapter, section, or subsection. In some cases, such as DCD Sections 2.1 and 2.2, the section may consist solely of a short description of topic and the Combined License information item subsection. This organization and numbering does not allow for the detailed discussion of these topics that is to be included in a complete FSAR section. As such, it is necessary to include numerous additional subsections to fully address the topic as identified in the guidance of Regulatory Guide 1.206 and the applicable SRP. In other cases, the organization and numbering must be modified slightly to allow for inclusion of plant-specific discussions within the appropriate section of the FSAR, such as including an additional water system description in Section 9.2. In these cases, the Combined License information item discussions are retained at the end of the DCD corresponding chapter, section, or subsection (to maintain the organization), but the numbering may be different.

These differences are well identified in the FSAR as STD DEP 1.1-1 at each location where the departure is taken and are considered to be purely administrative to support a logical construction of the document. Where the departure from the DCD organization and numbering is taken, the revised organization and numbering generally follows the guidance provided in Regulatory Guide 1.206 and the applicable SRP. As such, there are no significant departures from the expected organization and numbering of a typical FSAR, and the information is readily identifiable to facilitate NRC review.

In view of the above, we believe that it would be less efficient for both Duke and the NRC to comply with the portion of the regulation of 10 CFR Part 52, Appendix D, Section IV.A.2.a, that requires strict adherence to the "same organization and numbering as the generic DCD for the AP1000 design." Accordingly, Duke hereby submits a request for an exemption from the

regulations of 10 CFR 52, Appendix D, Section IV.A.2.a, pursuant to 10 CFR 52.7, “Specific Exemptions,” and 10 CFR 52.93, “Exemptions and Variances.”

Granting this request, which is authorized by law, would facilitate the NRC review of the Lee Nuclear Station COL application. For this and other reasons, granting this exemption request will not present an undue risk to the public health and safety, and is consistent with the common defense and security.

Moreover, compliance with the current rule would cause undue hardship for Duke and would also be inefficient and burdensome for the NRC staff. That approach would require Duke to prepare, and NRC to review, information with an organization and numbering that is unfamiliar and inconsistent with the current guidance for format and content of a combined license application.

For these reasons, Duke requests approval of the requested exemption from current regulations of 10 CFR 52, Appendix D, Section IV.A.2.a, as identified herein and in the application departures report.

2) Special Nuclear Material (SNM) Material Control and Accounting (MC&A) Program Description [Part 70, Subpart D and Part 74, Subparts C, D, and E]

Applicable Regulation(s): 10 CFR §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51

Specific wording from which exemption is requested:

10 CFR 70.22(b), Contents of applications:

(b) Each application for a license to possess special nuclear material, to possess equipment capable of enriching uranium, to operate an uranium enrichment facility, to possess and use at any one time and location special nuclear material in a quantity exceeding one effective kilogram, except for applications for use as sealed sources and for those uses involved in the operation of a **nuclear reactor licensed pursuant to part 50 of this chapter** and those involved in a waste disposal operation, must contain a full description of the applicant's program for control and accounting of such special nuclear material or enrichment equipment that will be in the applicant's possession under license to show how compliance with the requirements of §§ 74.31, 74.33, 74.41, or 74.51 of this chapter, as applicable, will be accomplished.

10 CFR 70.32, Conditions of licenses:

(c) (1) Each license authorizing the possession and use at any one time and location of uranium source material at an uranium enrichment facility or special nuclear material in a quantity exceeding one effective kilogram, except for use as sealed sources and those uses involved in the operation of a **nuclear reactor licensed pursuant to part 50 of this chapter** and those involved in a waste disposal operation, shall contain and be subject to a condition requiring the licensee to maintain and follow:

(i) The program for control and accounting of uranium source material at an uranium enrichment facility and special nuclear material at all applicable facilities as implemented pursuant to § 70.22(b), or §§ 74.31(b), 74.33(b), 74.41(b), or 74.51(c) of this chapter, as appropriate;

(ii) The measurement control program for uranium source material at an uranium enrichment facility and for special nuclear material at all applicable facilities as implemented pursuant to §§ 74.31(b), 74.33(b), 74.45(c), or 74.59(e) of this chapter, as appropriate; and

(iii) Other material control procedures as the Commission determines to be essential for the safeguarding of uranium source material at an uranium enrichment facility or of special nuclear material and providing that the licensee shall make no change that would decrease the effectiveness of the material control and accounting program implemented pursuant to § 70.22(b), or §§ 74.31(b), 74.33(b), 74.41(b), or 74.51(c) of this chapter, and the measurement control program implemented pursuant to §§ 74.31(b), 74.33(b), 74.41(b), or 74.59(e) of this chapter without the prior approval of the Commission. A licensee desiring to make changes that would decrease the effectiveness of its material control and accounting program or its measurement control program shall submit an application for amendment to its license pursuant to § 70.34.

10 CFR 74.31, Nuclear material control and accounting for special nuclear material of low strategic significance:

(a) General performance objectives. Each licensee who is authorized to possess and use more than one effective kilogram of special nuclear material of low strategic significance, excluding sealed sources, at any site or contiguous sites subject to control by the licensee, other than a production or **utilization facility licensed pursuant to part 50** or 70 of this chapter, or operations involved in waste disposal, shall implement and maintain a Commission approved material control and accounting system that will achieve the following objectives:

10 CFR 74.41, Nuclear material control and accounting for special nuclear material of moderate strategic significance:

(a) General performance objectives. Each licensee who is authorized to possess special nuclear material (SNM) of moderate strategic significance or SNM in a quantity exceeding one effective kilogram of strategic special nuclear material in irradiated fuel reprocessing operations other than as sealed sources and to use this material at any site other than a **nuclear reactor licensed pursuant to part 50 of this chapter**; or as reactor irradiated fuels involved in research, development, and evaluation programs in facilities other than irradiated fuel reprocessing plants; or an operation involved with waste disposal, shall establish, implement, and maintain a Commission-approved material control and accounting (MC&A) system that will achieve the following performance objectives:

10 CFR 74.51, Nuclear material control and accounting for strategic special nuclear material:

(a) General performance objectives. Each licensee who is authorized to possess five or more formula kilograms of strategic special nuclear material (SSNM) and to use such material at any site, other than a **nuclear reactor licensed pursuant to part 50 of this chapter**, an irradiated fuel reprocessing plant, an operation involved with waste disposal, or an independent spent fuel storage facility licensed pursuant to part 72 of this chapter shall establish, implement, and maintain a Commission-approved material control and accounting (MC&A) system that will achieve the following objectives:

Discussion:

Duke Energy Carolinas, LLC (Duke) requests an exemption from the requirements of 10 CFR § 70.22(b) and, in turn, §§ 70.32(c), 74.31, 74.41, and 74.51[1]. Section 70.22(b) requires an application for a license for special nuclear material to contain a full description of the applicant's program for material control and accounting (MC&A) of special nuclear material under §§ 74.31, 74.33, 74.41, and 74.511. Section 70.32(c) requires a license authorizing the use of special nuclear material to contain and be subject to a condition requiring the licensee to maintain and follow a special nuclear material control and accounting program, measurement control program, and other material control procedures, including the corresponding records management requirements. However, §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51 contain exceptions for nuclear reactors licensed under 10 CFR Part 50. The regulations applicable to the MC&A of special nuclear material for nuclear reactors licensed under 10 CFR Part 50 are provided in 10 CFR Part 74, Subpart B, §§ 74.11 through 74.19, excluding § 74.17. The

[1] While not containing an explicit exception for Part 50 reactors, § 74.33 applies only to uranium enrichment facilities and thus is not directly implicated in this exemption request.

purpose of this exemption request is to seek a similar exception for this combined license (COL) under 10 CFR Part 52, such that the same regulations will be applied to the special nuclear material MC&A program as nuclear reactors licensed under 10 CFR Part 50.

Nuclear reactors licensed under Part 50 are explicitly excepted from the requirements of §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51. There is no technical or regulatory reason to treat nuclear reactors licensed under Part 52 differently than reactors licensed under Part 50 with respect to the MC&A provisions in 10 CFR Part 74. As indicated in the Statement of Considerations for 10 CFR § 52.0(b) (72 Fed. Reg. 49352, 49372, 49436 (Aug. 28, 2007)), applicants and licensees under Part 52 are subject to all of the applicable requirements in 10 CFR Chapter I, whether or not those provisions explicitly mention a COL under Part 52. This regulation clearly indicates that plants licensed under Part 52 are to be treated no differently than plants licensed under Part 50 with respect to the substantive provisions in 10 CFR Chapter I (which includes Parts 70 and 74). In particular, the exception for nuclear reactors licensed under Part 50, as contained in §§ 70.22(b), 70.32(c), 74.31, 74.41, or 74.51, should also be applied to reactors licensed under Part 52.

An exemption from the requirements of §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51 would not mean that a MC&A program would be unnecessary or that the COL application would be silent regarding MC&A. To the contrary, the MC&A requirements in Subpart B to Part 74 would still be applicable to the COL just as they are to licenses issued under Part 50. Additionally, the COL application will describe the MC&A program for satisfying Subpart B to Part 74.

This exemption request is evaluated under 10 CFR § 52.7, which incorporates the requirements of § 50.12. That section allows the Commission to grant an exemption if 1) the exemption is authorized by law, 2) will not present an undue risk to the public health and safety, 3) is consistent with the common defense and security, and 4) special circumstances are present as specified in 10 CFR § 50.12(a)(2). The criteria in § 50.12 encompass the criteria for an exemption in 10 CFR §§ 70.17(a) and 74.7, the specific exemption requirements for Parts 70 and 74, respectively. Therefore, by demonstrating that the exemption criteria in § 50.12 are satisfied, this request also demonstrates that the exemption criteria in §§ 52.7, 70.17(a) and 74.7 are satisfied.

<u>Evaluation Against Exemption Criteria</u>

1) This exemption is not inconsistent with the Atomic Energy Act or any other statute and is therefore authorized by law.

2) An exemption from the requirements of 10 CFR §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51 would not present an undue risk to public health and safety. The exemption would treat the COL applicant similarly to Part 50 license applicants, who are excepted from the regulations in question. Furthermore, the COL application will contain a description of the applicant's MC&A program under Subpart B to Part 74. Therefore, the exemption from 10 CFR §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51 would not present an undue risk to public health and safety.

3) An exemption from the requirements of 10 CFR §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51 would not be inconsistent with the common defense and security. The exemption would treat the COL applicant similarly to Part 50 license applicants, who are excepted from the regulations in question. Furthermore, the COL application will contain a description of the applicant's MC&A program under Subpart B to Part 74. Therefore, the exemption from

§§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51 is consistent with the common defense and security.

4) The exemption request involves special circumstances under 10 CFR § 50.12(a)(2)(ii). That subsection defines special circumstances as when "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." Since the Commission determined that the requirements in 10 CFR §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51 are unnecessary for Part 50 applicants, those requirements are also unnecessary for Part 52 applicants.

As demonstrated above, the exemption complies with the requirements of 10 CFR §§ 50.12, 52.7, 70.17, and 74.7. For these reasons, approval of the requested exemption is requested from the regulations of 10 CFR §§ 70.22(b), 70.32(c), 74.31, 74.41, and 74.51, as described herein.

3) Containment Cooling Changes in regard to Passive Core Cooling System Condensate Return

Applicable Regulation(s): 10 CFR Part 52 Appendix D, Section III.B

Specific wording from which exemption is requested:

"III. Scope and Contents

B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix, including Tier 1, Tier 2 (including the investment protection short-term availability controls in Section 16.3 of the DCD), and the generic TS except as otherwise provided in this appendix. Conceptual design information in the generic DCD and the evaluation of severe accident mitigation design alternatives in appendix 1B of the generic DCD are not part of this appendix."

Pursuant to 10 CFR §52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is requested for plant-specific Tier 1 material departures from the AP1000 DCD for Tier 1 information. These material departures are contained in Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, and involve the addition of components to the condensate return design to enable the Passive Core Cooling System to more effectively perform its design functions. This exemption request is in accordance with the provisions of 10 CFR §50.12, 10 CFR §52.7, and 10 CFR Part 52, Appendix D.

Discussion:

The changes requested to Tier 1 Table 2.2.3-1 and Table 2.2.3-2 and associated Tier 2 changes to Table 3.2-3, Figure 3.8.2-1, Subsections 5.4.11.2 and 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.1.4, 6.3.1.1.6, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, and 6.3.3.2.1.1 and Figures 6.3-1 and 6.3-2, Subsection 7.4.1.1, Table 14.3-2, Subsection 15.0.3, Technical Specification Bases B 3.3.3 and B 3.5.4, Subsections 19E.4.10.2 and 19E.9, Table 19E.4.10-1, and Figures 19E.4.10-1 through 19E.4.10-4 provide additional equipment and surveillance requirements, provide reasonable assurance that the facility has been constructed and will be operated in conformity with the applicable design criteria, codes and standards, and demonstrate acceptable Passive Core Cooling System (PXS) system performance during design basis scenarios.

Conclusion:

This exemption request is evaluated in accordance with 10 CFR Part 52, Appendix D, Section VIII.A.4, 10 CFR §50.12, 10 CFR §52.7 and 10 CFR §52.63, which state that the NRC may grant exemptions from the requirements of the regulations provided the following six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, Appendix D, VIII.A.1]. The requested exemption satisfies the criteria for granting specific exemptions, as described below.

1) This exemption is authorized by law

   The NRC has authority under 10 CFR §§ 50.12, 52.7, and 52.63 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR §§50.12 and 52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 with proper justification. No law exists that would preclude the changes covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations.

   Accordingly, this requested exemption is "authorized by law," as required by 10 CFR §50.12(a)(1).

2) This exemption will not present an undue risk to the health and safety of the public

   The proposed exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would allow changes to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information. The plant-specific Tier 1 DCD will continue to reflect the approved licensing basis for the applicant, and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. Because the change to the condensate return portion of the passive core cooling system description maintains its design functions, the changed design will ensure the protection of the health and safety of the public. Therefore, no adverse safety impact which would present any additional risk to the health and safety is present. The affected Design Description in the plant-specific Tier 1 DCD will continue to provide the detail necessary to support the performance of the associated ITAAC.

   Therefore, the requested exemption from 10 CFR 52, Appendix D, Section III.B would not present an undue risk to the health and safety of the public.

3) The exemption is consistent with the common defense and security

   The exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would change elements of the plant-specific Tier 1 DCD by departing from the AP1000 certified (Tier 1) design information relating to the condensate return portion of the passive core cooling system. The exemption does not alter the design, function, or operation of any structures or plant equipment that are necessary to maintain a safe and secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

   Therefore, the requested exemption is consistent with the common defense and security.

4) Special circumstances are present

   10 CFR §50.12(a)(2) lists six “special circumstances” for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR §50.12(a)(2)(ii). That Subsection defines special circumstances as when “Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

   The rule under consideration in this request for exemption from Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, is 10 CFR 52, Appendix D, Section III.B, which requires that an applicant referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D)

shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information. The WLS Units 1 and 2 COLA references the AP1000 Design Certification Rule and incorporates by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D to maintain the level of safety in the design.

The proposed changes to the condensate return portion of the passive core cooling system maintain the design margins of the Passive Core Cooling System. This change does not impact the ability of any structures, systems, or components to perform their functions or negatively impact safety. Accordingly, this exemption from the certification information in Tier 1 Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, will enable the applicant to safely construct and operate the AP1000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

5) The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption

   Based on the nature of the changes to the plant-specific Tier 1 information and the understanding that these changes support the design function of the Passive Core Cooling System, it is likely that other AP1000 applicants and licensees will request this exemption. However, if this is not the case, the special circumstances continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the Passive Core Cooling System associated with this request will continue to be maintained. This exemption request and the associated marked-up tables demonstrate that the Passive Core Cooling System function continues to be maintained following implementation of the change from the generic AP1000 DCD, thereby minimizing the safety impact resulting from any reduction in standardization.

   Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. In fact, as described in Condition 6 below, the exemption will result in no reduction in the level of safety.

6) The design change will not result in a significant decrease in the level of safety

   The exemption revises the plant-specific DCD Tier 1 information by adding components to Subsection 2.2.3, Tables 2.2.3-1 and 2.2.3-2, which were added to the condensate return design to enable the Passive Core Cooling System to more effectively perform its design functions. Because these functions are met, there is no reduction in the level of safety.

   Therefore, the design change will not result in a significant decrease in the level of safety.

   As demonstrated above, this exemption request satisfies NRC requirements for an exemption to the design certification rule for the AP1000.