APPENDIX A

DUKE ENERGY CAROLINAS, LLC
OCONEE
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
TECHNICAL SPECIFICATIONS FOR
MATERIAL LICENSE SNM-2503

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1 INTRODUCTION

These Technical Specifications govern the safety of the receipt, possession, and storage of irradiated nuclear fuel at the Oconee Independent Spent Fuel Storage Installation and the transfer of such irradiated nuclear fuel to and from Units 1, 2, and 3 of the Oconee Nuclear Station and the Oconee Independent Spent Fuel Storage Installation.

1.1 Definitions

The following definitions apply for the purpose of these Technical Specifications.

A. Administrative Controls: Provisions relating to organization and management procedures, recordkeeping, review and audit, and reporting necessary to assure that the operations involved in the storage of spent fuel at the Oconee ISFSI are performed in a safe manner.

B. Design Features: Features if the facility associated with the basic design such as materials of construction, geometric arrangements, dimensions, etc., which, if altered or modified, could have a significant effect on safety.

C. Functional and Operating Limits: Limits on fuel handling and storage conditions necessary to protect the integrity of the stored fuel, to protect employees against occupational exposures, and to guard against the uncontrolled release of radioactive materials.

D. Fuel Assembly: The unit of nuclear fuel in the form that is charged or discharged from the core of a light-water reactor (LWR). Normally, will consist of a rectangular arrangement of fuel rods held together by end fittings, spacers, and tie rods.

E. Limiting Conditions: The minimum or maximum functional capabilities or performance levels of equipment required for safe operation of the facility.

F. Surveillance Requirements: Surveillance requirements include: (i) inspection, test, and calibration activities to ensure that the necessary integrity of required systems, components, and the spent fuel in storage is maintained; (ii) confirmation that operation of the installation is within the required functional and operating limits; and (iii) a confirmation that the limiting conditions required for safe storage are met.

G. Tonne (Te): One metric ton, equivalent to 1,000 kg or 2,204.6 lb. Fuel quantity is expressed in terms of the heavy metal content of the fuel measured in metric tons and written TeU.

H. Loading Operations: Loading Operations include all cask preparation steps prior to cask transport from the fuel building area.

1.2 Preoperational License Conditions

The license issued under Part 72 shall not allow the loading of spent nuclear fuel until such time as the following preoperational license conditions are satisfied:

1 A training exercise (Dry Run) of all dry shielded canister (DSC), transfer cask (TC) and horizontal storage module (HSM) loading and handling activities shall be held which shall include but not be limited to those listed and which need not be performed in the order listed.

a. Loading DSC in cask.

b. DSC (length may be truncated) drying, welding, and cover gas backfilling operations.

c. Moving cask to and aligning and docking with HSM on the storage pad.

d. Insertion of DSC in HSM.

e. Withdrawal of DSC from HSM.

f. Returning the cask to the decontamination pit.

g. Removing the cask lid and cutting open the DSC (length may be truncated) assuming fuel cladding failure.

h. Removing the DSC from the cask.

i. All dry run activities shall be done using written procedures.

j. The activities listed above shall be performed or modified and performed to show that each activity can be successfully executed prior to actual fuel loading.

2. The Oconee Nuclear Station Emergency Plan shall be reviewed and modified as required to include the ISFSI. (Abnormal event notifications will have to be updated for ISFSI events.)

3. As required by Subpart H, a Physical Security Plan shall be established to implement a physical protection program for the ISFSI. Further, the Oconee Nuclear Station Safeguards Contingency Plan and the Guard Training and Qualification Plan shall be modified as necessary to incorporate commitments to support the ISFSI.

4. A training module shall be developed for the Oconee Nuclear Station Training Program establishing an ISFSI Training and Certification Program which will include the following:

    a. DSC, TC and HSM Design (overview)

    b. ISFSI Facility Design (overview)

    c. ISFSI Safety Analysis (overview)

    d. Fuel loading and DSC and TC handling procedures and abnormal procedures

    e. ISFSI License (overview)

5. The Oconee Nuclear Station health physics procedures shall be reviewed and modified as required to include the ISFSI.

6. The Oconee Nuclear Station Administrative Procedures shall be reviewed and modified as required to include the ISFSI.

7. A procedure shall be developed for the documentation of the characterizations performed to select spent fuel to be stored in the canisters and modules. Such procedure shall include independent verification of fuel assembly selection by an individual other than the original individual making the selection.

8. Written operating and abnormal/emergency procedures shall be prepared.

1.3 General License Conditions

1.3.1 Quality Assurance

The design, construction and operation of the ISFSI shall be accomplished in accordance with the U.S. Nuclear Regulatory Commission (NRC) Regulations specified in Title 10 of the U.S. Code of Federal Regulations. All commitments to the applicable NRC Regulatory guides and to engineering and construction codes shall be carried out.

1.3.2 Fuel and Cask Handling Activities

Fuel and TC movement and handling activities which are to be performed in the Oconee Nuclear Station Fuel handling Building will be governed by the requirements of the Oconee Nuclear Station Facility Operating Licenses (DRP-38, -47 and -55) and associated Technical Specifications.

1.3.3 Administrative Controls

The Oconee ISFSI is located on the Oconee Nuclear Station site and will be managed and operated by the Duke Energy Carolinas, LLC staff. The administrative controls shall be in accordance with the requirements of the Oconee Nuclear Station facility Operating Licenses (DRP-38, -47 and -55) and associated Technical Specifications.

2 FUNCTIONAL AND OPERATING LIMITS

2.1 Fuel to be Stored at ISFSI

2.1.1 Specification

The spent nuclear fuel to be received and stored at the Oconee ISFSI shall meet the following requirements:

(1) Only fuel irradiated at the Oconee Units Nos. 1, 2, or 3 may be used.

(2) Maximum initial enrichment shall not exceed 4.0 weight percent U-235.

(3) Maximum assembly average burnup shall not exceed 40,000 megawatt-days per metric ton uranium (MWD/MTU) or shall meet the alternative specifications set forth in Section 4.3.1 of these Technical Specifications.

(4) Maximum heat generation rate shall not exceed 0.66 kilowatt (kW) per fuel assembly.

(5) Fuel shall have cooled a minimum of 10 years after reactor discharge and prior to storage in the Oconee ISFSI or shall meet the alternative specification set forth in Section 4.3.1 of the Technical Specifications.

(6) Fuel shall be intact unconsolidated fuel.

(7) Maximum assembly mass including control components shall not exceed 763 kilograms.

(8) The nominal load per spacer disk shall not exceed 109 kilograms per assembly per disk.

(9) Fuel assemblies known or suspected to have structural defects sufficiently severe to adversely affect fuel handling and transfer capability unless canned shall note be loaded into the DSC for storage.

(10) Immediately prior to insertion of a spent fuel assembly into a DSC, the identity of the assembly shall be independently verified by two individuals.

(11) Prior to insertion of a spent fuel assembly into a DSC, a dissolved boron level in water in the reactor pool and introduced into the DSC cavity shall be independently verified as being $\geq$ 1810 ppm by two individuals.

2.1.2 Basis

The design criteria and subsequent safety analysis of the Oconee ISFSI assumed certain characteristics and limitations for the fuels that are to be received and stored. Specification 2.1.1 assures that these bases remain valid by defining the source of the spent fuel, maximum initial enrichment, irradiation history, maximum thermal heat generation, and minimum post irradiation cooling time.

The radiological analyses are based on a radiation spectrum for 4.0 weight percent U-235 fuel at 40,000 MWD/MTU burnup. Compliance with the enrichment and burnup limits will ensure that the Dry Storage Casks design criteria are not exceeded. In addition, design criteria will not be exceeded for fuel not specifically meeting the above requirements for burnup and post irradiation time if the alternative requirements set forth in Section 4.3.1 of these Technical Specifications are met.

2.2 Dry Shielded Canister (DSC)

2.2.1 Specification

The DSCs used to store spent nuclear fuel in HSMs at the Oconee ISFSI shall have the operating limits shown in Table 2-1.

2.2.2 Basis

The design criteria and subsequent safety analysis of the DSC assumed certain characteristics and operating limits for the use of the DSC. This specification assures that those design criteria are not exceeded.

TABLE 2-1

OPERATING LIMITS

	Operating Limit
Max. Lifting height with a Non-Redundant Lifting Device for Transfer Cask Outside the Fuel Storage Buildings	80 inches
Dose Rate	
Surface of HSM	≤ 200 mrem/hr
(These limits conform to transportation cask dose rate limits. Actual does rates for most surface locations on the loaded HSM will be significantly less.)	
DSC Tightness	
(Standard He-Leak Rate)	
Top Shield Plug Closure Weld	$\leq 10^{-4}$ atm^{-}cc/s
Siphon and Vent Port Cover Welds	$\leq 10^{-4}$ atm^{-}cc/s
Top Cover Plate Weld	Dye Penetrant Test (ASME B&PV Code Section III, Division 1, Subsection NB-5350 (1983) Liquid Penetrant Acceptance Standards)
Max. Specific Power of One Fuel Assembly*	0.66 kW
Helium Pressure Limit (DSC cavity)	2.5 psig ± 2.5 psig
Pressure during Canister Drying (DSC Cavity)	≤ 3 torr (for not less than 30 min)
DSC Water Moderator during Loading and Unloading of Fuel Assemblies	≥ 1810 ppm (Boron concentration)
Time Limit to Complete DSC Draining after Removal from Spent Fuel Pool	≤ 50 hours

* This limit may be analytically determined.

2.3 Dry Shield Canister Internal Cover Gas

2.3.1 Specification

The DSC shall be backfilled with helium.

2.3.2 Basis

The thermal analysis performed for the DSC assumes the use of helium as a cover gas. In addition, the use of an inert gas (helium) is to ensure long-term maintenance of fuel clad integrity.

2.4 Dry Shielded Canister Exterior Surface Contamination

2.4.1 Specification

Removable contamination on the DSC exterior shall be less than 22,000 dpm/100 cm^2 from beta, gamma, and 2,200 dpm/100 cm^2 from alpha sources. Surveillance requirement 4.4.1 ensures that this requirement will be met.

2.4.2 Basis

Compliance with this limit ensures that the offsite dose limits in 10 CFR Part 20, 10 CFR Part 50 – Appendix I, 10 CFR Part 72, and 40 CFR Part 190 are met.

2.5 Dry Shielded Canister Moderator

2.5.1 Specification

The DSC cavity shall be moderated only by supplied water with a boron concentration greater than or equal to 1,810 ppm.

2.5.2 Basis

The specification assures subcriticality of the DSC during fuel loading and unloading.

2.6 Dry Shielded Canister Draining

2.6.1 Specification

The time to complete draining of the DSC cavity of water moderator shall not exceed 50 hours after removal of the DSC from a spent fuel pool.

2.6.2 Basis

This specification assures subcriticality of the DSC after fuel loading.
FUNCTIONAL AND OPERATING LIMITS

3 LIMITING CONDITIONS

3.1 Limiting Condition – Handling Height

3.1.1 Specification

This specification applies to handling of a cask being used for spent fuel storage outside of the Fuel Handling Building and its cask decontamination area.

a. The TC shall not be handled at a height of greater than 80 inches.

b. In the event of a cask drop from a height greater than 15 inches, fuel in a DSC in the cask shall be removed and inserted in a replacement DSC or, if damaged, returned to the spent fuel pool. The damaged DSC shall be decontaminated, removed from service and disposed of, as may be appropriate.

3.1.2 Basis

The drop analyses performed for the cask drop incidents for a DSC loaded in a TC confirm that drops up to 80 inches can be sustained without unacceptable damage to the cask and DSC. This limiting condition ensures that the handling height limits will not be exceeded at the storage pad or in transit to and from the spent fuel pool. Design of the DSC is to ASME B & PV Code Section III, Division 1, Subsection NB for Class 1 components, Service Level D requirements.

4 SURVEILLANCE REQUIREMENTS

Requirements for surveillance of cask internal pressure, contamination levels, DSC weld leak rates, and fuel related parameters are contained in this section. These requirements are summarized in Table 4-1 from details contained in Sections 4.1 through 4.4.

TABLE 4 – 1

SURVEILLANCE REQUIREMENTS SUMMARY

Section	Quantity or Item	Period
4.1.1	Surveillance of the HSM Air Inlets and Outlets	D
4.2.1	Limits for Maximum Air Temperature Rise after Storage	S
4.3.1	Fuel Parameters	P
4.4.1	DSC and Cask Contamination	B
4.5.1	DSC Weld Testing	L
4.6.1	HSM Inspection	F
4.7.1	DSC Pressure	L

D – daily on a normal basis, within 24 hours after an accident

S – At initial storage, 24 hours later, 7 days later

P – Prior to cask loading

B – Before and after cask loading and unloading

L – During loading operations

F – Five years and 10 years after initial storage

4.1 Surveillance of the HSM Air Inlets and Outlets

The HSM shall be inspected to verify that the air inlets and outlets are free from obstructions.

4.1.1 Specifications

Normal visual inspection frequency	Daily
Accident visual inspection frequency	Within 24 hours after an accident

4.1.2 Basis

To assure that no HSM air inlets or outlets are plugged for more than 48 hours and to assure that blockage of inlets and outlets due to an accident will be removed in less than 48 hours. Analysis in Chapter 8 of the Oconee ISFSI SAR showed that no temperature limits are exceeded if a module is completely plugged for 48 hours. Therefore, for normal operations, an inspection of the inlets and outlets once per day will assure that any local obstructions can be removed. Likewise, after an accident the HSMs should be examined within 24 hours to assure that air flow can be restored within 48 hours after the accident.

4.2 Limits for the Maximum Air Temperature Rise

4.2.1 Specification

Maximum air temperature rise shall not exceed 60ºF (33.3ºC) The maximum air temperature rise from HSM inet to outlet shall be checked at the time the DSC is stored in the HSM, again 24 hours later, and again after 7 days.

4.2.2 Basis

The 60ºF (33.3ºC) temperature rise was selected to limit the hottest rod in the DSC to below 644ºF (340ºC) at 70ºF (21ºC) ambient air inlet temperature. The expected temperature rise is less than 60ºF (i.e., 49ºF (27.2ºC); see Table 8.1 – 2 of the NUHOMS – 24P topical report) and hence, the current design provides adequate margin for this specification. If the temperature rise is within the specifications, then the HSM and DSC are performing as designed and no further temperature measurements are required during normal surveillance.

4.3 Fuel Parameters

4.3.1 Specifications

Type	15 x 15 PWR Fuel
Burnup	≤ 40,000 MWD/MTU
Initial (Beginning of Life) Enrichment	≤ 4.0 wt% U-235
Heat generation	≤ 0.66 kW/fuel assembly
Fuel cooling period	≥ 10 years
Total fuel assembly mass including control components	≤ 763 kg
Fuel assembly mass per space disk per assembly	≤ 109 kg

Any fuel not specifically filling the above requirements for burnup and post irradiation time may still be stored if all the following requirements are met:

Decay Power Per Assembly	≤ 0.66 kw
Neutron Source Per DSC	≤ 3.715 x 10^9 n/sec/DSC, with spectrum bounded by Table 3.1 – 4 of the NUHOMS-24P Topical Report
Gamma Source Per DSC	≤ 3.85 x 10^{16} MeV/sec/DSC with spectrum bounded by that shown in Table 3.1 – 4 of the NUHOMS-24P Topical Report

This specification is applicable to all fuel to be stored in the ISFSI. This information shall be documented for each fuel assembly to be loaded in a DSC.

4.3.2 Basis

This specification was derived to insure that the peak fuel rod temperatures, surface doses, nuclear subcriticality and mass are below the design values.

4.4 DSC and Cask Contamination

4.4.1 Specification

4.4.1.1

Prior to loading, the top 6 inches of the cask interior shall be smeared to ensure that removal contamination levels on the interior surfaces of the cask, excluding the drain and vent lines are less than 22,000 dpm/100 cm^2 from beta, gamma sources and 2,200 dpm/100 cm^2 from alpha sources.

4.4.1.2

After cask loading, but prior to moving the cask to the HSM, the top of the sealed DSC shall be smeared to ensure that removable contamination levels are less than 22,000 dpm/100 cm^2 from beta, gamma sources and 2,200 dpm/100 cm^2 from alpha sources. This will ensure that the limits in 2.4.1 are met.

4.4.1.3

After cask unloading, the interior surfaces of the cask shall be smeared to ensure that removable contamination levels on the interior surfaces of the cask, excluding the drain and vent lines are less than 2,200 dpm/100 cm^2 from beta, gamma sources and 2,200 dpm/100 cm^2 from alpha sources. This will ensure that the limits in 2.4.1 are met.

4.4.2 Basis

This surveillance requirement will ensure compliance with the DSC surface contamination limits of 2.4.1.

4.5 DSC Weld Testing

4.5.1 Specification

During DSC loading operations, the top shield plug closure and the siphon and vent port cover welds shall be tested using a helium leak detector to ensure that, for each, leak tightness is less than or equal to 10^{-4} atm-cc/s. The DSC top cover plate weld shall be dye penetrant tested.

4.5.2 Basis

The safety analysis of leak tightness of the DSC as discussed is based on a weld being leak tight to 10^{-4} atm-cc/s. This check is done to ensure compliance with this design criterion.

4.6 HSM Inspection

4.6.1 Specification

At intervals of 5 and 10 years after initial storage there shall be a visual inspection of the interior concrete surfaces of the first HSM loaded, in particular the top surface above the heat shield, to assure that no significant deterioration of the concrete has occurred. Only the first HSM at the site need be inspected.

4.6.2 Basis

This surveillance provides added assurance that the structural properties of the HSM concrete will not be impaired as a result of high concrete surface temperatures during early years of fuel storage to the point that the shielding or structural integrity provided by the HSM is significantly affected.

4.7 DSC Pressure

4.7.1 Specification

The helium backfill pressure in the DSC cavity shall be 2.5 psig ± 2.5 psig.

4.7.2 Basis

The value of 2.5 psig was selected to assure that the pressure within the DSC is within pressure design limits during any expected normal operating condition.

5 DESIGN FEATURES

The Oconee ISFSI design approval was based upon review of specific design drawings, some of which have been deemed appropriate for inclusion in the Oconee ISFSI Safety Evaluation Report (SER). Drawings listed in Appendix B of the Oconee ISFSI SER have been reviewed and approved by the NRC. These drawings may be revised under the provisions of 10 CFR 72.48 as appropriate.

5.1 Site

5.1.1 Specification

The Oconee ISFSI is located on the Oconee Nuclear Station site as described in Section 2.1.2 of the Oconee ISFSI SAR.

5.2 Cask Design

5.2.1 Specification

The TC used in the Oconee ISFSI to transfer the DSC to the HSM is described in Section 1.3.1.3 of the Oconee ISFSI SAR.

5.3 DSC Design

5.3.1 Specification

The DSC is described in Section 1.3.1.1 of the Oconee ISFSI SAR.

All components comprising the DSC pressure boundary shall be provided from ASME SA 240, Type 304 stainless steel or its equivalent.

5.4 HSM Design

5.4.1 Specification

The HSM is described in Section 1.3.1.2 of the Oconee ISFSI SAR

The HSM shall be constructed of concrete with a compressive strength greater than or equal to 5000 psi (cured for 28 days; 90 percent of all specimens tested) and a minimum unit weight of 140 pounds per cubic foot. The concrete shall be composed of Type II Portland cement meeting the requirements of ASTMC150. The aggregate shall meet the specifications of ASTMC33.

5.5 Storage Pads

5.5.1 Specification

ISFSI storage pads are reinforced concrete pads nominally 3-feet thick. Design criteria of the storage pads are discussed in Section 2.5.5 of the Oconee ISFSI SAR.

5.6 Total Storage Capacity

5.6.1 Specification

The total storage capacity of the Oconee ISFSI is 996.86 TeU. This corresponds to 88 modules and canisters each containing 24 fuel assemblies.

APPENDIX B

DUKE ENERGY CAROLINAS, LLC
OCONEE
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
SAFEGUARDS LICENSE CONDITIONS
MATERIALS LICENSE SNM-2503

1.1 PHYSICAL PROTECTION REQUIREMENTS FOR SPENT NUCLEAR FUEL IN DRY STORAGE

1.1.A The licensee shall establish and maintain a physical protection program in accordance with the provisions of his physical security plan entitled, “Oconee Nuclear Station, Independent Spent Fuel Storage installation (ISFSI) Security Program, Revision 4,” submitted by letter dated January 10, 1991, (which includes a safeguards contingency plan as Chapter 3.0); and as it may further be revised under the provisions of 10 CFR 72.44(e) and 72.186.

1.1.B The licensee’s physical protection program shall be supported by a security organization, with personnel trained and qualified in accordance with the provisions of 10 CFR Part 73, Appendix B, as outlined in the plan published under the title, “Oconee Nuclear Station Training and Qualification Plan, Revision 9,” submitted by letter dated June 25, 1990; and as it may further be revised under the provisions of 10 CFR 72.44(e) and 72.186.

APPENDIX C

DUKE ENERGY CAROLINAS, LLC
OCONEE
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
TECHNICAL SPECIFICATIONS FOR ENVIRONMENTAL PROTECTION
MATERIALS LICENSE SNM-2503

1 INTRODUCTION

These technical specifications govern the protection of the environment during the receipt, possession, storage, and transfer of spent fuel at the Oconee ISFSI.

1.1 Radioactive Material Releases

1.1.1 Specification – (pursuant to 10 CFR 72.44 (d))

Not applicable.

1.1.2 Basis

Specifications are required pursuant to 10 CFR 72.44 (d), stating limits on the release of radioactive materials for compliance with limits of 10 CFR Part 20 and "as low as is reasonably achievable objective" for effluents. DSC surface contamination within the limits of 2.4.1 ensures that the offsite dose will be inconsequential. In addition, there are no normal or off-normal releases or effluents expected from the double-sealed storage canisters of the ISFSI.

1.2 Effluent Control and Waste Treatment

1.2.1 Specification – (pursuant to 10 CFR 72.44 (d)(1))

Not applicable.

1.2.2 Basis

Specifications are required pursuant to 10 CFR 72.44(d)(1) for operating procedures, for control of effluents, and for the maintenance and use of equipment in radioactive waste treatment systems to meet the requirements of 10 CFR 72.104. However, there are, by the design of the sealed storage canisters at the ISFSI, no effluent releases, and all Oconee site TC loading and unloading operations and waste treatment therefrom will occur at the Oconee Nuclear Station under the specifications of its operating licenses.

1.3 Environmental Monitoring Program

1.3.1 Specification

The licensee shall include the Oconee ISFSI in the environmental monitoring for the Oconee Nuclear Station.

1.3.2 Basis

An environmental monitoring program is required pursuant to § 72.44(d)(2).

1.4 Annual Environmental Report

1.4.1 Specification

The semi-annual radioactive effluent release reports under 10 CFR Part 50 license requirements for the Oconee Nuclear Station shall specify the quantity, if any, of each of the principal radionuclides released to the environment in liquid and in gaseous effluents during the ISFSI operation and such other information as may be required by the Commission to estimate maximum potential radiation dose commitment to the public resulting from effluent release. Copies of these reports shall be submitted to the NRC Region II office and to the Director, Office of Nuclear Material Safety and Safeguards.

1.4.2 Basis

The report of Specification 1.4.1 is required pursuant to 10 CFR 72.44(d)(3).