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Duke Energy Company
Independent Spent Fuel Storage Facility
Oconee Nuclear Site
UPDATED FINAL SAFETY ANALYSIS REPORT

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List of Abbreviations

ACI	AMERICAN CONCRETE INSTITUTE
APM	ADMINISTRATIVE POLICY MANUAL
AFR	AWAY-FROM-REACTOR
AISC	AMERICAN INSTITUTE OF STEEL CONSTRUCTION
ALARA	AS LOW AS REASONABLY ACHIEVABLE
ANSI	AMERICAN NATIONAL STANDARDS INSTITUTE
AWS	AMERICAN WELDING STANDARDS
CFR	CODE OF FEDERAL REGULATIONS
DBT	DESIGN BASIS TORNADO
DOE	DEPARTMENT OF ENERGY
DPC	DUKE POWER COMPANY
DSC[1]	DRY STORAGE CANISTER
EPRI	ELECTRIC POWER RESEARCH INSTITUTE
EPZ	EMERGENCY PLANNING ZONE
ESF	ENGINEERED SAFETY FEATURE
ETQS	EMPLOYEE TRAINING AND QUALIFICATION SYSTEM
FEMA	FEDERAL EMERGENCY MANAGEMENT ADMINISTRATION
HSM	HORIZONTAL STORAGE MODULE
HRS	HYDRAULIC RAM SYSTEM
IFA	IRRADIATED FUEL ASSEMBLY
ISFSI	INDEPENDENT SPENT FUEL STORAGE INSTALLATION
LWM	LIQUID WASTE MANAGEMENT
MHE	MAXIMUM HYPOTHETICAL EARTHQUAKE
MRS	MONITORED RETRIEVABLE STORAGE
NDE	NONDESTRUCTIVE EXAMINATION
NEPA	NATIONAL ENVIRONMENTAL POLICY ACT
NRC	NUCLEAR REGULATORY COMMISSION
NUHOMS®-24P	NUTECH ENGINEERS, INC. HORIZONTAL MODULAR STORAGE
NUREG	NUCLEAR REGULATORY GUIDE
POR	PRUDENT OPERATING RESERVE
PWR	PRESSURIZED WATER REACTOR
ONS	OCONEE NUCLEAR STATION
SPS	SKID POSITIONING SYSTEM
UFSAR	UPDATED FINAL SAFETY ANALYSIS REPORT
VA	VENTILATION AIR SYSTEM
VR	STATION VOLUME REDUCTION SYSTEM
NWPA	WASTE POLICY ACT OF 1982, AS AMENDED

Note:

1. The term Dry Storage Canister (DSC) in this report refers to the same item termed dry shielded canister in the Nutech Topical Report referenced in this UFSAR.

1.0 Introduction and General Description of Storage System

1.1 Introduction

Duke Energy Carolinas LLC (Duke) began commercial operation of the Oconee Nuclear Station, Units 1, 2, and 3 on July 15, 1973, September 9, 1974 and December 16, 1974 respectively. When the original application was submitted for the Oconee Site-Specific Independent Spent Fuel Storage Installation (ISFSI), these three 2568 MWt units had generated millions of KWH in a safe and reliable manner. In so doing, these units had discharged a total of approximately 2300 spent fuel assemblies. These spent fuel assemblies were being stored in two onsite pools and in the McGuire Nuclear Station spent fuel pools. The need to provide additional onsite storage facilities to permit continued operation was discussed in Sections 9, 10, and 11 of the Environmental Report (Reference 1) which was submitted as part of the Oconee Site-Specific ISFSI license application.

To provide storage until the Department of Energy (DOE) begins to accept title to spent fuel under the requirements of the Nuclear Waste Policy Act of 1982, as amended in 1987, Duke received a license to build and operate an ISFSI in compliance with 10 CFR 72. Duke chose the NUHOMS®-24P dry storage system designed by Transnuclear (formerly VECTRA Technologies, formerly Pacific Nuclear Inc., formerly NUTECH Engineers, Inc.) to be used for the Oconee Site-Specific ISFSI. The NUHOMS®-24P system is more fully described in Revision 1A of the Topical Report for the NUHOMS®-24P system submitted in July 1988 and accepted by the NRC on April 21, 1989. The location of the Site-Specific ISFSI on the Oconee site is shown on Figure B-1.

The NUHOMS®-24P system provides long-term interim storage for irradiated fuel assemblies. The fuel assemblies are confined in a helium atmosphere by a stainless steel canister. The canister is protected and shielded by a massive concrete module. Decay heat is removed by thermal radiation, conduction and convection from the canister to an air plenum inside the concrete module. Air flows through this internal plenum by natural draft convection.

The canister containing twenty-four irradiated fuel assemblies is transferred from the spent fuel pool to the concrete module in a transfer cask. The cask is precisely aligned and the canister is inserted into the module by means of a hydraulic ram.

The NUHOMS®-24P system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The dry storage canister and horizontal storage module have been designed to withstand certain accidents as described in Chapter 8 of this UFSAR.

The license (SNM-2503) authorizes a total of eighty-eight modules (2112 assemblies) to be built incrementally, as needed, to match the requirements for additional storage. The initial construction of 20 horizontal storage modules (HSMs) in a 2 x 10 array was designated as Phase I. The construction of the next 20 HSMs was designated as Phase II.

If the ONS-Site-Specific ISFSI is expanded, Duke will perform a survey of the affected area for any federally listed rare, threatened, and endangered species and any species of concern prior to the start of any ground disturbing activities. (This action is a regulatory commitment as stated in Attachment 2 and Attachment 4 of Reference 5.)

In 1997 Oconee installed a new and separate General License (GL) ISFSI, in addition to maintaining the existing Site-Specific ISFSI. The GL ISFSI uses the Standardized NUHOMS®-24P storage system which was approved by the U.S. Nuclear Regulatory Commission (NRC) for use under a GL. The two compatible systems simply utilize the same fuel handling/transport equipment and general site location. The GL system has a separate set of licensing documents including a separate UFSAR and Certificate of

Compliance (CoC), and is not covered in this UFSAR. Any changes, tests, or experiments involving dry storage activities should be reviewed against all applicable licensing documents.

Operation of the Oconee Site-Specific ISFSI will continue past the first year for up to 20 years under the initial license and continue under license renewal as necessary until the fuel can be shipped to a permanent repository, or for a maximum of 40 years under the renewed license period (60 years, total). During this service life, while any given HSM could be unloaded and later reloaded with a new DSC, reloading a given DSC following removal of the original fuel assemblies is not anticipated due to the potential destructive nature of the top end shield plug removal process. However, enhanced techniques may be developed which prevent DSC damage during plug removal. If such circumstances were to arise, a rigorous re-qualification would be performed to ensure such DSCs would meet the CLB and to ensure that no aging effects requiring management are introduced. (This action is a regulatory commitment as stated in Enclosure 3, Appendix C and Enclosure 4 of Reference 6.) Eventual reuse of the HSMs will depend upon the schedule and restrictions for spent fuel deliveries to DOE under the NWPA.

1.2 General Description of Installation

1.2.1 General Description

The Oconee Site-Specific ISFSI provides for the horizontal, dry storage of irradiated fuel assemblies (IFAs) in a concrete module. The principal components are a concrete horizontal storage module (HSM) and a stainless steel dry storage canister (DSC) with an internal basket which holds the IFAs. Each HSM contains one DSC and each DSC contains twenty-four fuel assemblies.

Deleted paragraph(s) per 2007 update

The initial phase of construction including twenty HSMs was completed in May 1990. A second phase of twenty HSMs was completed in January 1992. In addition to these primary components, the Oconee Site-Specific ISFSI also requires transfer equipment to move the DSCs from the spent fuel pool (where they are loaded with the IFAs) to the HSMs where they are stored. This transfer system includes a transfer cask, a hydraulic ram, a trailer and a cask skid. This transfer system interfaces with the existing Oconee spent fuel pool, the cask crane, the site layout (i.e., roads and topography) and other procedural requirements.

1.2.2 Principal Site Characteristics

The Oconee Site-Specific ISFSI is located on the Oconee Nuclear Station site near Seneca, South Carolina. Duke owns and operates three 2568 MWt nuclear generating units on the Oconee site. The ISFSI is located inside the protected area approximately 100 ft. west of the Station's condenser cooling water intake structure (Figure B-1).

1.2.3 Principal Design Criteria

The principal design criteria and parameters for the Oconee Site-Specific ISFSI are shown in (Table A-1). The radiation sources are for the reference fuel assembly. For the majority of the fuel to be stored, the radiation sources will be less than or equal to the sources described in the NUHOMS®-24P Topical Report (Reference 2). For radiation sources larger than the sources described in Reference 2 restrictive measures will be used to ensure surface dose rates that are ALARA and below design basis limits.

1.2.3.1 Structural Features

The HSM is a low profile reinforced concrete structure designed to withstand normal operating loads, the abnormal loads created by seismic activity, tornados and other natural events and the postulated accidental loads which may occur during operation.

The structural features of the DSC are defined, to a large extent, by the cask drop accident. The DSC body, the double seal welds on each end, and the DSC internals are designed to provide for fuel retrieval after a postulated maximum credible drop.

1.2.3.2 Decay Heat Dissipation

The decay heat of the IFAs is removed from the DSC by natural draft convection. Air enters the lower part of the HSM, rises around the DSC and exits through the top shielding slab. The flow cross-sectional area is designed to provide adequate air flow from the draft height of the HSM and the inlet and outlet air temperature differences for the hottest day conditions (i.e., 46.7°C or 116°F).

1.2.4 Operating and Fuel Handling Systems

The major operating systems of the Oconee Site-Specific ISFSI are those required for fuel handling and transport of the fuel from the spent fuel pool to the ISFSI. General operations are outlined in Table A-2 and the primary design parameters of the required systems are listed in Table A-3. The majority of the fuel handling operations involving the transfer cask (i.e., fuel loading, drying, trailer loading, etc.) utilize standard techniques at Oconee for spent fuel shipment. The remaining operations (seal welding, transfer cask-HSM alignment, and DSC transfer) are unique to the Oconee Site-Specific ISFSI.

1.2.5 Safety Features

The principal safety features of the Oconee Site-Specific ISFSI are the containment provided by the DSC and the concrete shielding of the HSM. In addition to its structural and missile protection functions, this shielding reduces the gamma and neutron flux emanating from the IFAs inside a DSC so that the average outside surface dose rate on the HSM is less than 20 mr/hr. Additional Oconee Site-Specific ISFSI features include:

1. Filling the annulus between the DSC and transfer cask with demineralized water and sealing it prior to lowering them into the spent fuel pool - Prevents contamination of the DSC exterior by pool water.
2. Internal shield blocks inside the HSM which comprise the shielded ventilation plenum - Reduces scatter dose out of the air inlet.
3. External shield blocks on the HSM roof - Reduces scatter dose out of the air outlet.
4. Shield plugs on the DSC - Reduces dose during DSC drying, helium filling and seal welding.
5. Double seal welds on each end of the DSC - Prevents leakage of radioactive gases or particulates if the fuel rods should fail.

1.2.6 Radioactive Waste and Auxiliary Systems

Because of the passive nature of the Oconee Site-Specific ISFSI, there are no radioactive waste or auxiliary systems required during normal storage operations. There are, however, some waste and auxiliary systems required during DSC loading, drying and transfer into the module. The Oconee waste systems handle the fuel pool water and air which are vented from the DSC during drying. Auxiliary handling systems (such as hydraulic pressure control, alignment, crane, etc.) are also required during the loading and transfer operation.

1.3 General Systems Descriptions

The major systems, subsystems, and components of the Oconee Site-Specific ISFSI are listed in Table A-4. The following subsections briefly describe the principal systems and components and their operation.

1.3.1 Systems Descriptions

1.3.1.1 DSC Design

The NUHOMS®-24P DSC is shown in Figure 1.3-1 of the Topical Report for the NUHOMS®-24P System (Reference 2). The DSC is sized to hold twenty-four irradiated pressurized water reactor (PWR) fuel assemblies. The main component of construction is a stainless steel cylinder with a nominal 67 inch outside diameter. The nominal overall length is 187 inches, excluding grapple ring.

The components of the internal basket of the DSC are described in Table A-4 and are also shown on Figure 1.3-1 of Reference 2. The basket is comprised of twenty-four square cells. The structural component of the cells is type 304 stainless steel.

Structural support is provided by circular stainless-steel spacer disks. Longitudinal support is provided by the four support rods which run the length of the DSC.

The DSC is equipped with two shielded end plugs so that when the DSC is inside the transfer cask or the HSM, the radiation dose at the ends is limited. The end shield plugs are constructed of lead surrounded by a steel body.

The DSC has redundant seal welds at the top and bottom. The bottom cover plates are welded to the DSC body during fabrication and the top cover plates after fuel loading. Also, all connections (drain and vent ports) are redundantly sealed. This assures that no single failure of the DSC end plates will breach the DSC. Furthermore, there are no credible accidents which would breach the main body of the DSC.

Criticality safety during wet loading operations is assured by 1) the design of the basket structure which maintains a minimum separation between fuel assemblies, 2) technical specifications which require a minimum boron concentration in excess of 1810 ppm to be maintained within the DSC storage cavity during wet loading and unloading operations, and 3) procedures which limit the reactivity of fuel assemblies loaded into the DSC to an established maximum through verification of initial enrichment and exposure history.

Design changes and enhancements were made to the DSCs beginning with dry storage transfer number 22 at Oconee. For this load and subsequent transfers, the DSCs utilized will be of two different types designated “long cavity” or “short cavity” canisters. The long cavity DSC design is used to accommodate spent fuel assemblies with control components. The short cavity DSC is utilized to store spent fuel without control components. The following list describes major changes from the original DSC design that are common to both the new long and short cavity DSCs.

1. The spacer discs material has been changed to carbon steel with aluminum flame spray coating versus 304 stainless steel. There is also an option to use carbon steel support rods.
2. The DSC shell has been modified such that the bottom shield plug assembly fits inside the continuous cylindrical shell rather than attaching to it.
3. The grapple ring support is now attached to the outer bottom cover plate and doesn't penetrate the bottom shield plug.
4. The new top shield plug design utilizes a separate stainless steel inner top cover plate to provide for better weld joint detail. The shield plug is no longer a pressure boundary.

The basket, shell, and final assembly design for the new short and long cavity DSCs are essentially the same, except for the shield plugs and support rods. The short cavity DSC design utilizes thicker top and bottom shield plugs made of solid carbon steel instead of encased lead and the support rods, which provide longitudinal support, are shorter to accommodate the reduced cavity length.

The above changes to the DSC reflect an improved design that will ease fabrication, reduce costs, and improve the welding/closure process. These changes were evaluated in References 3 and 4.

1.3.1.2 Horizontal Storage Module

An isometric view of a unit of four HSMs is shown in Figure 1.3-1A of Reference 2. The HSMs are typically built in units of 20 in a 2 x 10 array. The first construction of 20 HSMs at Oconee is designated as Phase I, with the second grouping of 20 designated as Phase II. Subsequent constructions are permitted, with up to a maximum of 88 HSMs possible under the existing license. The HSM is fabricated from reinforced concrete and structural steel which is constructed in place at the storage location. The thick concrete top, front, and sides of the HSM provide adequate neutron and gamma shielding to achieve an average 20 mr/hr surface dose. Nominal closure door surface doses are less than 100 mr/hr. The transfer cask surface has an average dose rate of less than 200 mr/hr for the locations where workers must perform loading and unloading operations.

Thick shield walls (3.0 ft. thick) are provided on the outside walls of the modules at the end of the unit to provide shielding on the sides. Sufficient (2.0 ft. thick) shielding between modules (to prevent scatter in adjacent modules during loading and retrieval) is provided by the interior module walls.

The HSM provides fuel cooling by a combination of radiation, conduction and convection. The air enters at the bottom of the HSM and passes around the DSC and exits through the flow channels in the top shield slab. Heat is conducted out of the DSC into the natural convection air flow. Heat is also radiated from the DSC to the HSM walls where the natural convection air flow removes the heat. Figure 1.3-2 of Reference 2 shows the flow path and typical conditions. The passive cooling system of the HSM was designed to assure that peak cladding temperatures are less than 340°C (644°F) during long term storage for average normal ambient temperatures of 70°F. The fuel can withstand short term temperatures of up to 570°C (1,058°F) during operational and accidental transients with no anticipated adverse effects. However, calculations show that temperatures remain well below 570°C at any time during normal operation or any postulated accident.

The HSMs are independent, passive systems for the dry storage of irradiated fuel assemblies. Therefore, the HSMs are designed to ensure that normal operation and credible site hazards do not impair their function. To this end, the HSMs are designed to the following loads:

1. Winds and Tornado (includes missile) - Oconee UFSAR, Chapter 3.
2. Seismic - Oconee UFSAR, Chapter 3.
3. Flood - Oconee UFSAR, Chapter 2.
4. Snow and Ice - ANSI A58.1-1982.
5. Combined Load (dead weight, live loads, temperature) - ACI 349-85.

The HSMs are placed in service on a load bearing foundation. Earth work is required to prepare the storage site for a level foundation and access area.

1.3.1.3 Transfer Cask

The transfer cask used with the Oconee Site-Specific ISFSI provides radiological shielding during the DSC drying operation and during the transfer to the HSM. Both neutron (Bisco NS-3, a cementatious

material) and gamma (lead) shielding are incorporated into the cask design. For the Oconee Site-Specific ISFSI, the transfer cask has a nominal 188 inch long internal cavity with a nominal 68 inch internal diameter. Figure 1.3-2A of Reference 2 shows the major components of the transfer cask.

1.3.1.4 Transfer Trailer

The transfer trailer has a capacity of 120 tons. The transfer trailer carries the transfer cask skid and the loaded transfer cask. The transfer trailer is designed to ride as low to the ground as possible to minimize the HSM height. Four hydraulic jacks are incorporated into the transfer trailer design to provide vertical movement for alignment of the cask and HSM. The transfer trailer is pulled by a conventional tractor. Figure 1.3-3 of Reference 2 shows a typical transfer trailer arrangement. Also, as discussed in Section 8.2.5 of Reference 2, the design basis drop height for the NUHOMS®-24P Transfer cask is 80 inches. This analysis bounds the Oconee transport conditions.

The original transfer trailer was replaced by a newer design in 2006 which is similar to the original except that it has an integral 3-stage hydraulic ram mounted to the transfer cask skid.

1.3.1.5 Transfer Cask Skid

The transfer cask skid is similar in design and operation to existing transport skids. The major differences are:

1. No equipment interferes with access to the top of the transfer cask when in the horizontal position.
2. The skid is mounted on a smooth bearing surface and hydraulic positioners provide horizontal alignment with the HSM. A restraining bolt system is provided to prevent movement during trailer towing.
3. The entire skid is mounted on a trailer.

The above features are shown on Figure 1.3-4 of Reference 2.

1.3.1.6 Horizontal Hydraulic Ram

The horizontal hydraulic ram is a 3-Stage design with a capacity of 80,000 lb. and a reach of 6.55m (21.5 ft.). The hydraulic ram is an integral portion of the transfer cask skid.

1.3.1.7 System Operation

The primary operations (in sequence of occurrence) for the Oconee system are shown schematically in Figure 1.3-6 of Reference 2 and are described below:

1. <u>Transfer Cask Preparation</u> - Cask preparation includes taking smears of the cask interior to ensure that the DSC exterior will remain radiologically clean. These operations are done in the decontamination area inside the spent fuel pool area. Detailed procedures for these operations are described in Chapter 5.
2. <u>DSC Preparation</u> - The internals and externals of the DSC are verified to be clean. This ensures that the newly fabricated DSC will meet existing Oconee specific criteria for placement in the spent fuel pool.
3. <u>Placement of DSC in Transfer Cask</u> - The empty DSC is inserted into the transfer cask. Proper alignment is assured through the use of alignment marks on the cask and each DSC.
4. <u>Transfer Cask Lifting and Placement in the Spent Fuel Pool</u> - The DSC/transfer cask annulus is filled with clean demineralized water. The DSC cavity is also filled with borated water from either the spent fuel pool or an equivalent source of borated water. This prevents an inrush of pool water when they

are placed in the spent fuel pool. This will also prevent contamination of the DSC outer surface by the pool water. The DSC/transfer cask annular region is then sealed with an inflatable seal at the top to prevent mixing. The water-filled transfer cask with the DSC inside is then placed into the spent fuel pool.

Polished stainless steel reflectors attached to the cask pit support stand are used as a visual guide to ensure proper seating of the transfer cask in the SFP depression.

5. DSC Loading - Twenty-four spent fuel assemblies are placed into the DSC basket. These spent fuel assemblies will be preselected to control reactivity and decay heat using the administrative controls on burnup, initial enrichment, and decay time detailed in Section 10.2.5.

6. DSC Top End Shield Plug Placement - The DSC top end shield plug is placed inside the DSC using the overhead crane with transfer cask lifting yoke attached. The top end shield plug is suspended from the transfer cask lifting yoke by cables and is emplaced as the transfer cask lifting yoke is re-engaged to the transfer cask trunnions.

7. Transfer Cask Lifting out of the Pool - The loaded transfer cask is lifted out of the spent fuel pool and placed in the decontamination pit.

8. DSC Sealing - The water level in the DSC/transfer cask annulus is then lowered approximately 5-10 inches. Swipes are taken over the DSC exterior at the DSC upper surface and around the circumference. The water level in the DSC is lowered away from the inside surface of the top end shield plug. Then a seal weld is applied to the outer surface of the top end shield plug. This provides the primary seal for the DSC.

   The new design DSCs (beginning with dry storage load 22) have a top shield plug design that utilizes a separate 0.75” thick stainless steel inner top cover plate (ITCP) to provide for better weld joint detail. Since there is a separate ITCP, the shield plug itself is no longer a pressure boundary. The seal weld is applied to the top of the ITCP and DSC shell, not directly to the shield plug. The use of a separate ITCP reduces the welding heat input, consequent deformation of the DSC lip during seal welding, and eliminates the potential for lead intrusion into the seal weld observed in some of the earlier DSCs. This arrangement also precludes the need to decontaminate the top of shield plug before welding can begin, as the seal weld is applied to the ITCP which does not have to be placed in the spent fuel pool water.

9. DSC Drying - A pressure line is connected to the DSC and the water inside the DSC is forced out by helium pressure. The water, which is removed from the transfer cask and the DSC, is returned to the spent fuel pool or routed to the Oconee radioactive waste processing equipment. The pressure line is then used to draw a vacuum to facilitate drying until the water content meets the design criteria.

10. Helium Filling - In order to ensure that no fuel and/or cladding oxidation occurs during storage, the DSC is filled with helium (He). To accomplish this, a portable helium gas bottle is connected.

    The DSC is then filled with He gas. After the DSC is filled with the inert gas, the filling line is removed and the DSC ports are plugged and welded closed.

    A strong-back device supplied by the vendor is used with the new DSC designs to prevent overstressing the ITCP during helium pressurization.

11. Final DSC Sealing - The outer top cover plate is positioned and seal welded. This provides a redundant seal at the upper end of the DSC. The lower end also has redundant seal welds, which are placed and tested during fabrication. This operation provides the double seal integrity of the DSC.

12. Transfer Trailer Loading - After helium filling and seal welding, the transfer cask lid is positioned and bolted in place. The transfer cask is then lifted onto the transfer cask skid mounted on the transfer trailer and secured.

13. Transfer - Once loaded and secured, the transfer trailer is towed to the HSM. This movement takes place completely within the Oconee plant protected area.
14. Transfer Cask/HSM Preparation - At the Oconee Site-Specific ISFSI the transfer trailer is backed into position and the HSM front access cover is raised and removed. Next, the transfer cask lid is removed. An optical alignment system and the hydraulic skid positioners are used for the final alignment of the transfer cask and HSM.
15. Cask Docking - After alignment, the cask is docked to the HSM and secured in place.
16. HSM Loading - After final alignment the DSC is then pushed into the HSM by the hydraulic ram.
17. Storage - After the DSC is positioned inside the HSM, the hydraulic ram is released from the DSC and retracted. The transfer trailer is pulled away and the HSM front access door is closed and tack welded. The DSC is now in storage within the HSM.
18. Retrieval - For retrieval, the HSM access door is raised and removed and the transfer cask is positioned as previously described and the hydraulic ram is used to pull the DSC into the transfer cask. All coupling, attachment, alignment, and closure operations are done in the same manner as previously described, but in reverse order. Once back in the transfer cask, the DSC and its cargo of irradiated fuel assemblies are ready for shipment to a permanent repository or other storage location. Provisions will be made to return the DSC to the Oconee spent fuel pool if necessary.

1.4 Identification of Agents and Contractors

The prime contractor for design and analysis of the Oconee Site-Specific ISFSI was Pacific Nuclear Fuel Systems, Inc. of San Jose, California (now Transnuclear, Inc. of Columbia, Maryland). HSM construction was the responsibility of Duke. Fabrication of transfer equipment and DSCs furnished to date was also the responsibility of Pacific Nuclear Fuel Systems, Inc. (now Transnuclear, Inc. of Columbia, Maryland).

1.5 Material Incorporated by Reference

The Topical Report for the Nutech Horizontal Modular Storage (NUHOMS®-24P) System for Irradiated Nuclear Fuel, originally submitted to the Nuclear Regulatory Commission by Nutech Engineers, Inc. (now Transnuclear, Inc.) on February 26, 1988 and approved on April 21, 1989 is hereby incorporated into this UFSAR by reference.

1.6 References

1. Duke Power Company, Oconee Nuclear Station, Independent Spent Fuel Storage Installation, Environmental Report.
2. Topical Report for the Nutech Horizontal Modular Storage (NUHOMS®-24P) System for Irradiated Nuclear Fuel, NUH-002, Revision 1A, July 1989.
3. DPC Engineering Calculation OSC-3485, “10 CFR 72.48 Evaluations for Revisions to the Oconee ISFSI”, Rev. 9, dated 2-15-93.
4. DPC Engineering Calculation OSC-3485, “10 CFR 72.48 Evaluations for Revisions to the Oconee ISFSI”, Rev. 10, dated 9-13-93.
5. LAR 2007-06, D.A. Baxter to U.S. Nuclear Regulatory Commission, “Site-Specific Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application,” January 30, 2008.
6. RAI response, Dave Baxter to U.S. Nuclear Regulatory Commission, “Response to Requests for Additional Information License Amendment Request No. 2007-06,” January 30, 2009.

7. NRC letter of May 29, 2009, 'Issuance of Renewed Materials License No. SNM-2503, Oconee Independent Spent Fuel Storage Installation (ISFSI) (Tac Nos. L24184 and L25206)"; Enclosure 2, Safety Evaluation Report.

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2.0 Site Characteristics

2.1 Geography and Demography

2.1.1 Site Location

The Oconee Site-Specific Independent Spent Fuel Storage Installation (ISFSI) is located on the Oconee Nuclear Station plant site. The site is located in Oconee County, South Carolina, approximately 8 miles northeast of Seneca, South Carolina at latitude 34°-47'-38.2" N and longitude 82°-53'-55.4" W. Lake Keowee is located to the north and west of the site. The Corps of Engineers' Hartwell Reservoir is south of the site and Duke's Lake Jocassee lies approximately 11 miles to the north. Figure B-2 (based on Figure 2-1 of Oconee UFSAR) shows the site location with respect to neighboring states and counties within 50 miles.

2.1.2 Site Description

Figure B-3 (based on Figure 2-4 of Oconee UFSAR) shows the site, property line, exclusion area, site structures and general features of the area. Figure B-4 is a detailed site layout showing the Oconee Site-Specific ISFSI location in relation to major site features. There are no industrial, commercial, institutional or recreational structures within the site boundary. Located within 1 mile of the station center are a visitors center, the Keowee Hydroelectric Station, the Mosquito Control Facility, the Clemson Operations Center, and the Crescent Resources (Keowee Division) office complex and appurtenances. All of these facilities are Duke properties. Duke does not own the vacated Old Pickens Church and Cemetery, a small, historic property located east of the station which is not currently being used.

The topography immediately surrounding the ISFSI (Figure B-4) consists of relatively flat terrain which has been grassed or graveled over and is routinely maintained by the station. Routine maintenance of the immediate site vicinity assures that erosion will be minimal and that fire hazards due to burning vegetation are also minimized.

2.1.2.1 Legal Responsibilities for Site

All the property within the l mile radius exclusion area including mineral rights is owned by Duke except for the small vacant rural church plot belonging to Old Pickens Church, rights-of-way for existing highways and approximately 9.8 acres of U. S. Government property involved with Hartwell Reservoir.

The Hartwell property is either a portion of the Hartwell Reservoir or subject to flooding and not suitable for other uses. Duke has obtained from the owners of the church plot and from the United States the right to restrict activities on these properties and to evacuate them of all persons at any time without prior notice if, in its opinion, such evacuation is necessary or desirable in the interest of public health and safety.

The property which is within the exclusion area and which is not owned by Duke is shown on Figure B-3.

2.1.2.2 Other Activities Within the Site Boundary

Duke owns and operates the Oconee Nuclear Station and the Keowee Hydroelectric Station. The Oconee Site-Specific ISFSI is located within the owner controlled area of the nuclear plant. ISFSI operations have been considered for impacts upon the Oconee station's facility operating licenses. Pursuant to 10CFR Part 50, the licenses for the three Oconee Units were amended to permit Duke to operate the Site-Specific ISFSI. The amendment concluded that with certain minor modifications all aspects of Site-Specific ISFSI

operation which are conducted within the existing Oconee station can be conducted safely while meeting the criteria for a “no significant hazards” finding.

All ISFSI operations are performed by the existing Oconee workforce. Only the transfer equipment used for the storage system is dedicated exclusively to ISFSI operations. No individual or group is dedicated exclusively to the ISFSI. Operational control of the ISFSI includes procedures for the spent fuel pool loading steps and the subsequent transfer into the ISFSI.

ISFSI operations required the following fuel building modifications:

1. Enlarging the opening of the cask decontamination pit covers.
2. Shortening the projection from the spent fuel pool wall of the cooling system intake pipe. This is needed to provide clearance for the transfer cask in the spent fuel pool cask pit.
3. The addition of a microdrive to the fuel crane positioning system to aid in the precision placement of the transfer cask.

The following auxiliary equipment is used exclusively for DSC/transfer cask operations within the fuel building:

1. Transfer cask lift yoke and extension member.
2. Vacuum drying equipment.
3. Automatic welding equipment.
4. Slings for the transfer cask lid.
5. Cask pit depression platform.

Additional description of the ISFSI and Fuel Building systems and facility is included in Section 4.4.

Other non-plant related activities are limited to the highways through the Exclusion Area, Duke's Visitors Center, recreation on the lakes, the Mosquito Control Facility, and the Old Pickens Church and Cemetery which are historical landmarks and will not be used for regular services.

2.1.2.3 Arrangements for Traffic Control

Arrangements have been made with the South Carolina State Highway Department to control and limit traffic on public highways in the Exclusion Area should it become necessary in the interest of public health and safety.

2.1.3 Population Distribution and Trends

The population distribution is based on the 2000 census (Reference 10). Table A-5 gives the population distribution within the three county area surrounding Oconee. The majority of citizens live in the cities of Walhala, Seneca, Clemson, Central, and Anderson, S.C. The area is largely rural and sparsely populated.

As derived from 2000 Census Bureau information, 187,679 people lived within 20 miles of ONS. This is a population density of 149 persons per square mile within 20 miles and, applying the GEIS sparseness measures, ONS falls into a least sparse category, Category 4 (greater than or equal to 120 persons per square mile within 20 miles).

As estimated from 2000 Census Bureau information, 1,219,121 people lived within 50 miles of ONS. This equates to a population density of 155 persons per square mile within 50 miles. Applying the GEIS proximity measures, ONS is classified as Category 2 (no city with 100,000 or more persons and between 50 and 190 persons per square mile within 50 miles). According to the GEIS sparseness and proximity

matrix, the ONS ranks of sparseness Category 4 and proximity Category 2 results in the conclusion that ONS is located in a medium population area (Reference 11).

The population projections for the three county area around Oconee are given for 2010 through 2050 in Table A-6. These projections are based on the 1980-2000 census. The population within the three county area surrounding Oconee is projected to increase by approximately 55% by the year 2050.

2.1.3.1 Transient Population

It was expected in the late 1980s that Lake Keowee's 300 mile shoreline would be fully developed by the early 1990's at which time the estimated transient population would be 36,000. This estimate was based on development of lakeside lots, public access areas, and expanded commercial activities to take advantage of expanded recreational opportunities. As of 2007, the shoreline of Lake Keowee is not yet fully developed. There will not be any cottages within the Exclusion Area.

The visitors center, located on Duke Property just north of the plant and within the Exclusion Area, hosts approximately 20,000 visitors per year.

There are no large industries within 5 miles of the ISFSI. Duke identified no facilities whose air emissions or other activities may have negative impacts on the ONS ISFSI.

2.1.4 Uses of Nearby Land and Waters

Residential development of Lake Keowee's shoreline is expected to be the major use of the nearby land. Commercial development is anticipated to increase in response to the residential development. The waters of Lake Keowee are used for fishing, boating and swimming by the public through various public and private recreational areas.

The following description of land use and localized populations in Pickens and Oconee Counties in the 10-mile EPZ of the Oconee Nuclear Station is based on the Oconee Nuclear Station Emergency Plan as of August 1, 1988.

Pickens County lies within the 10-mile EPZ of the Oconee Nuclear Station. Involved are approximately 157.08 square miles of county territory and approximately 30,000 people. Also included are approximately 300 dairy cattle, 10 milk-producing goats, 243 head of swine, 2,938 head of beef cattle and 15 head of meat-producing goats.

Also, involved in the 10-mile EPZ are approximately 256 acres of vegetables, 47 acres of apples, and a large number of residential vegetable gardens.

This area has approximately 1,297 acres of hay crops and 4,670 acres of pasture grass.

A large portion of Oconee County lies within the 10-mile EPZ of the Oconee Nuclear Station. Included in this zone are approximately 165.498 square miles of land and approximately 26,000 people, with the largest concentration in Seneca. Oconee County's 654 square miles are divided into 22,665 acres of cropland, 285,605 acres of woodlands, and approximately 127,333 acres that fall into a general category of “all other”. There are a total of 13 dairies in the 10-mile EPZ.

The largest portion of land is devoted to crops such as soybeans, cotton, hay, wheat, small grain, and corn, apples, forestry, poultry, beef or dairying.

Production of meat, agricultural crops and milk for the 5-mile radius of Oconee Nuclear Station for 1980 was as follows:

1. meat = 118 tons
2. crops = 310 tons

3. milk = 86,300 gallons

This data has not significantly changed since it became available in 1980.

There are two schools located within the 5-mile radius: Six Mile Elementary School - 522 students, and Keowee Elementary School - 299 students. Two special care institutions are located within the 5-mile radius: Harvey's Love and CareHome and Six Mile Retirement Center nursing homes have a total of 80 patients.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 Industrial and Military Facilities

There are no large industrial or military facilities or activities within 5 miles of Oconee. No other nuclear facilities including university research reactors are presently located within a 50-mile radius of Oconee Nuclear Station.

2.2.2 Transportation Routes

Figure B-3 shows the major transportation routes within 1 mile of Oconee. There are no oil or gas pipelines within 5 miles of the site. The nearest railroad line or spur is located at Newry, SC which is outside the 5-mile radius from the plant.

The nearest airport is the Clemson-Oconee Airport located approximately 9 miles to the south of the plant. The runway is oriented ENE-WSW. Pickens County Airport is located approximately 10 miles to the east of Oconee Nuclear Station. The runway is oriented in a NE-SW direction. Anderson County Airport is located approximately 23 miles SSE of the plant. It has two runways oriented as follows: NE-SW and NNW-SSE. The orientation of the NNW-SSE runway is not in a straight line toward Oconee Nuclear Station. The above information is based on the "Atlanta Sectional Aeronautical Chart Scale 1:500,000" 37th Edition, September 25, 1986, published by the U.S. Department of Commerce. No structures which could cause damage as described in Reg. Guide 3.48, para. 2.2 are located near the plant.

2.2.2.1 Description of Products and Materials

The highways passing through the 1 mile radius exclusion area are SC Routes 130 and 183 which carry local traffic only with infrequent trucking of hazardous chemicals and explosives since the general area is nonindustrial.

Only small amounts of chlorine are stored on-site since chlorine is not used for condenser cleaning at Oconee. No individual container contains more than 150 lbs. of chlorine. The chlorine is used for disinfection of raw water, with one 150 lb. container typically being in service. The maximum total number of containers on hand at any time is four.

2.3 Meteorology

2.3.1 Regional Climatology

Western South Carolina is far south of major storm tracks but experiences higher precipitation amounts than the east coast due to its location in the lee of the Appalachian Mountains. A semi-permanent belt of high pressure usually influences the regional climate. During the fall season, the area has a high

probability of experiencing atmospheric stagnation during which the dilution rate for effluents is low due to low wind speeds.

The Oconee plant site is situated on Lake Keowee which was established to provide cooling for the three existing Oconee Nuclear units and future steam generating units as well as storage for Jocassee (pumped storage) and Keowee (conventional) hydroelectric stations. The topography in the vicinity of the site is moderately rolling and the local air flow is influenced to some extent by the contour of the lake. The prevailing winds are divided between the southwest and northeast quadrants due to the lake orientation and large scale pressure effects.

A complete description of regional and local wind data, including normal and extreme parameters can be found in Section 2.3 of the UFSAR.

2.3.2 Local Meteorology

2.3.2.1 Data Sources

The accident analysis meteorological data base, discussed in the Oconee SER Section 3.2.4, Units 2 and 3, is for the period March 15, 1970 - March 14, 1972. Joint frequency tables of wind direction, wind speed and atmospheric stability are shown in Table A-7.

2.3.2.2 Topography

Figure B-5 shows the detailed topography within 5 miles of the storage site.

2.3.3 Onsite Meteorological Measurement Program

Meteorological measurements include wind direction and speed, horizontal wind direction fluctuation, temperature, vertical temperature gradient, and rainfall. The relative position of instruments with respect to station yard is noted in Figure B-6. Relative elevations of both surface levels and instrument levels are depicted in Figure B-7.

Wind measurements are made with the Packard Bell Model W/S 1O1B series wind direction-speed system with starting thresholds of 0.7 and 0.6 miles per hour for direction and speed, respectively. Wind direction and speed are recorded in the control room on Esterline Angus Model A 601 C strip chart recorders with a system accuracy of ± 5.4 degrees for direction and ± 0.45 miles per hour for speed. Temperature and delta temperature measurements are made with the Leeds and Northrup 8100 Series 100 ohm resistance temperature device with Packard Bell Model 327 thermal radiation shields. Temperature and delta temperature are recorded on the Leeds and Northrup Speedomax W recorder with a system accuracy of ± 1°F for temperature (at 10 m level) and ± 0.5°F for delta temperature (46 m level referenced to the 10 m level). For data prior to February 24, 1977, delta temperature was measured at the 46 m level and the 1.5 m level. Rainfall is measured near the meteorological tower with the Belfort Weighing Rain Gauge Model 5-780 with an accuracy of ± 0.03 in. and ± 0.06 in. for zero to five and five to ten inch totals respectively.

Operational measurements consist of near real-time digital outputs in addition to the previously described analog system. An entirely new set of instrumentation has been installed including the measurement of dew point (at 10 m level); a supplemental low-level wind system (at 10 m level) provides input for emergency dose assessment (see Figure B-6 and Figure B-7). The rain gauge has been relocated near the supplemental wind system.

Instrument specifications for operational measurements are:

1. Wind Direction

a. Manufacturer Teledyne Geotech

b. Time - averaged digital accuracy ± 3° of azimuth

c. Time - averaged analog accuracy ± 6° of azimuth

d. Starting threshold 0.3 m/sec at 10° initial deflection

e. Damping ratio 0.4 at 10° initial deflection

f. Distance constant 1.1 m

2. Wind Speed

    a. Manufacturer Teledyne Geotech

    b. Time - averaged digital accuracy ± 0.27 m/sec for speeds less than 27 m/sec

    c. Time - averaged analog accuracy ± 0.40 m/sec for speeds less than 27 m/sec

    d. Starting threshold 0.45 m/sec

    e. Distance constant 1.5 m

3. Temperature

    a. Manufacturer Teledyne Geotech

    b. Time - averaged digital accuracy ± 0.3°C

    c. Time - averaged analog accuracy ± 0.5°C

4. Delta Temperature

    a. Manufacturer Teledyne Geotech

    b. Time - averaged digital accuracy ± 0.10°C

    c. Time - averaged analog accuracy + 0.15°C

5. Precipitation

    a. Manufacturer Teledyne Geotech

    b. Digital accuracy ± 6% of total accumulation at 15 cm/hr

    c. Analog accuracy ± 9% of total accumulation at 15 cm/hr

    d. Resolution 0.25 mm

Near real-time digital outputs of meteorological measurements are summarized for end-to-end 15 min. periods for use in a near real-time puff-advention model to calculate offsite dose during potential radiological emergencies. The Operator Aid Computer (OAC) system computes the 15 min. quantities from a sampling integral of 60 sec. It calculates 15 min. average values for high and supplemental low level wind direction and speed; 15 min. averages are also calculated for delta temperature, ambient temperature and dew point temperature. Total water equivalence is computed for precipitation. All 15 min. values are stored with a 24 hr. recall. Permanent archiving of data from the digital system is made by combining the 15 min. quantities into one hour values.

2.3.4 Diffusion Estimates

2.3.4.1 Basis

The design two-hour X/Q at the Exclusion Area Boundary (EAB) for accidental releases is 4.5E-4 (sec/m^3).

2.3.4.2 Calculations

The calculation of a two-hour X/Q value to estimate radiological doses from potential accidental releases from the storage site (See Figure B-1) is based on a plant design condition of Pasquill Type F stability with a wind speed of 1m/sec as proposed in the Oconee Safety Evaluation Report, Section 3.2.4, Units 2 and 3. The equivalent design condition [95 percentile hourly average X/Q] for the ISFSI is a Pasquill Type F stability with a wind speed of 0.65 m/sec. The calculation assumes a gaussian material distribution from a ground level release with essentially a point source geometry.

$$X/Q = \left[\bar{u}\,\pi\,\sigma_y\sigma_z\right]^{-1} = 4.5E-4(\mathrm{sec}/\,m^3)$$ [1]

Where

$\bar{u}$ = mean wind speed at 10 m = 0.65(m/sec)

σ_y(1.0 mi.) = crosswind concentration distribution standard deviation = 57 m

σ_z(1.0 mi.) = vertical concentration distribution standard deviation = 19 m

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

2.4.1.1 Site and Facilities

The location and description of Oconee presented in Chapters 1 and 2 include reference to figures showing the general arrangement, layout and relevant elevations of the station. Station yard grade is 796 ft. mean sea level (msl). The mezzanine floor elevation in the Turbine, Auxiliary, and Service Buildings is 796.5 ft. (msl). Exterior accesses to these buildings are at elevation 796.5 ft. (msl).

All of the man-made dikes and dams forming the Keowee Reservoir rise to an elevation of 815 ft. msl including the intake channel dike. The crest of the submerged weir in the intake canal is at elevation 770 ft. msl.

Flooding at the ISFSI will not occur. Figure B-3 shows the location of the ISFSI at the Oconee site, and Figure B-12 shows the relative location and topography of the ISFSI yard at Elevation 825.0 and the surrounding terrain features, including the Keowee dam and dikes. The Probable Maximum Flood level for Lake Keowee, as defined in Section 2.4.2.2, is Elevation 808.0, which is seventeen feet below the ISFSI site yard level of Elevation 825.0. Also, the peak flood level due to a postulated failure of the

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[1] Slade, D. H. (ed.) 1968: Meteorology and Atomic Energy 1968, TID-24190, National Technical Information Service, Springfield, Va.

upstream Jocassee Dam is Elevation 813.12, as discussed in Section 2.4.5.1, Thus, since all of the man-made dams and dikes forming Lake Keowee are constructed to an elevation of 815.0 and since the ISFSI site elevation of 825.0 is above the maximum lake level which can be maintained, there is no potential for the reservoir level reaching the ISFSI site by overtopping. Therefore, flooding of the ISFSI will not occur.

The ISFSI yard is surrounded by drainage intercept ditches sized to prevent local overland flow from reaching the ISFSI site. In addition, stormwater drainage is provided in the paved areas of the ISFSI site.

Therefore, flooding of the ISFSI site cannot occur either due to reservoir overflow or local intense precipitation.

2.4.1.2 Hydrosphere

The main hydrologic features influencing the Oconee plant site are the Jocassee and Keowee Reservoirs. Lake Jocassee was created in 1973 with the construction of the Jocassee Dam on the Keowee River. The lake provides pump storage capacity to the reversible turbine-generators of the Jocassee Hydroelectric Station, located approximately 11 miles north of the plant. At full pond, elevation 1110 ft. msl, Lake Jocassee has a surface area of 7565 Ac, a shoreline of approximately 75 mi, a volume of 1,160,298 Ac-ft., and a total drainage area of about 148 sq mi.

Lake Keowee was created in 1971 with the construction of the Keowee Dam on the Keowee River and the Little River Dam on the Little River. Its primary purpose is to provide cooling water for the plant and water to turn the turbines of the Keowee Hydroelectric Station. At full pond, elevation 800 ft. msl, Lake Keowee has a surface area of 18,372 Ac, a shoreline of approximately 300 mi, a volume of 955,586 Ac-ft., and a total drainage area of about 439 sq mi. The Jocassee and Keowee Reservoirs and the hydroelectric stations located at these reservoirs are owned and operated by Duke.

The area presently provides for a few raw water users. The City of Greenville and the Town of Seneca take their raw water supplies from Lake Keowee. The Town of Anderson, the Town of Clemson, the Town of Pendleton, Clemson University, and several industrial plants take their raw water supplies from Hartwell Reservoir, downstream of Lake Keowee.

Greenville's raw water intake is located approximately 2 miles north of the plant on Lake Keowee. Seneca's raw water intake is located approximately 7 miles south of the plant on the Little River Arm of Lake Keowee. Anderson raw water intake is located approximately 40 river miles downstream of the Keowee tailrace.

The existing raw water intakes for Greenville, Seneca, and Anderson are shown and located relative to the site on Figure 2.4.1 in the Oconee Site-Specific ISFSI Environmental Report.

2.4.2 Floods

2.4.2.1 Flood History

Since Oconee is located near the ridgeline between the Keowee and Little River valleys, or more than 100 ft. above the maximum known flood in either valley, the records of past floods are not directly applicable to siting considerations.

2.4.2.2 Flood Design Consideration

In accordance with sound engineering practice, records of past floods as well as meteorological records and statistical procedures have been applied in studies of floods routed through the Keowee and Jocassee Reservoirs as a basis for spillway and freeboard design.

The spillway capacities for Lake Keowee and Jocassee were selected in accordance with the empirical expression for design discharge:

$$Q = C\sqrt{DA}$$

Where

Q = peak discharge in cfs

DA = drainage area in square miles

C = 5000, a runoff constant judged to be characteristic of the drainage area

The following tabulation gives pertinent data on this design flood flow:

Lake Keowee[1]	Lake Jocassee	
439	148	Drainage area at damsite, sq mi
25,200	21,000	Maximum recorded flow at nearby USGS gages efs DA
(Newry Gage, DA 455 sq mi)	(Jocassee Gage, DA 148 sq mi	
8-13-40	10-4-64	Date of maximum flow
1939-1961	1950-1965	Period of record
105,000	61,000	Spillway design discharge, cfs
800	1,110	Full Pond elevation
815	1,125	Crest of dam elevaton
0	0	Surcharge on full pond for design discharge
4	2	Number of spillway gates
38 ft. x 35 ft.	40 ft. x 32 ft.	Size of spillway gates Discharge capacity, cfs
107,200	45,700	Spillway
-	16,500	(2 units Dependable flood flow of 4) through units
107,200	62,200	Total discharge capacity, cfs

Note:

1. Little River and Keowee River Arms

The above discharge capacities assume no surcharge above normal full pond level. Statistical analyses have shown design reservoir inflows for both Lake Keowee and Lake Jocassee equal to respective design discharge capacities outlined above to have recurrence intervals less frequent than once in 10,000 years.

The maximum wave height and wave run-up have been calculated for Lake Keowee and Lake Jocassee by the Sverdrup-Munk formulae. The results of these calculations are as follows:

Wave Height	Wave Run-Up	Maximum Fetch	Lake
3.70 ft.	7.85.ft	8 miles	Keowee (Keowee River Arm)
3.02 ft.	6.42 ft.	4 miles	Jocassee
3.02 ft.	6.42 ft.	4 miles	Keowee (Little River Arm)

The wave height and wave run-up figures are vertical measurements above full pond elevations as tabulated above.

Studies were also made to evaluate effects on reservoirs and spillways of maximum hypothetical precipitation occurring over the entire respective drainage areas. This rainfall was estimated to be 26.6 inches within a 48 hour period. Unit hydrographs were prepared based on a distribution in time of the storms of October 4-6, 1964, for Jocassee and August 13-15, 1940, for Keowee. Results are summarized as follows:

Keowee	Jocassee	
147,800	70,500	Maximum spillway discharge, cfs
808.0	1114.6	Maximum reservoir elevation
7.0 ft.	10.4 ft.	Free board below toe of dam

While spillway capacities at Keowee and Jocassee have been designed to pass the design flood with no surcharge on full pond, the dams and other hydraulic structures have been designed with adequate freeboard and structural safety factors to safely accommodate the effects of maximum hypothetical precipitation. Because of the time-lag characteristics of the runoff hydrograph after a storm, it is not considered credible that the maximum reservoir elevation due to maximum hypothetical precipitation would occur simultaneously with winds causing maximum wave heights and run-ups.

The maximum Keowee tailwater level during hydro operation has been calculated to be elevation 672.0 ft. (msl), which is 124 ft. below the nuclear station yard elevation 796.0 ft. (msl) and 153 ft. below the ISFSI yard elevation of 825 ft. (msl)

The maximum discharge calculated, due to hydro operating, is expected to be 19,800 cfs. The minimum discharge calculated with no units operating, is expected to be 30 cfs.

In summary, the above results of flood studies show that Lakes Keowee and Jocassee are designed with adequate margins to contain and control floods which pose no risk to the ISFSI site.

2.4.3 Probable Maximum Flood on Streams and Rivers

2.4.3.1 Probable Maximum Precipitation

See Section 2.4.2.2.

2.4.3.2 Runoff and Stream Course Models

See Section 2.4.2.2.

2.4.3.3 Probable Maximum Flood Flow

See Section 2.4.2.2.

2.4.3.4 Coincident Wind Wave Activity

See Section 2.4.2.2.

2.4.4 Potential Dam Failures, Seismically Induced

Duke has designed the Keowee Dam, Little River Dam, Jocassee Dam, Oconee Intake Canal Dike, and the Intake Canal Submerged Weir based on sound Civil Engineering methods and criteria. These designs have been reviewed by a board of consultants and reviewed and approved by the Federal Energy Regulatory Commission in accordance with the license issued by that agency. The Keowee Dam, Little River Dam, Jocassee Dam, Intake Canal Dike, and the Intake Canal Submerged Weir have also been designed to have an adequate factor of safety under the same conditions of seismic loading as used for design of Oconee.

The construction, maintenance, and inspection of the dams are consistent with their functions as major hydro projects. The safety of such structures is the major objective of Duke's designers and builders, with or without the presence of the nuclear station or ISFSI.

2.4.5 Flooding Protection Requirements

2.4.5.1 Flood Protection Measures for Oconee Station Seismic Class 1 Structures

The Oconee Station plant yard elevation is 796.0 ft. msl and the Oconee Site-Specific ISFSI yard elevation is 825 ft. (msl). All of the man-made dikes and dams forming the Keowee Reservoir are constructed to an elevation of 815.0 ft. msl with a full pond elevation of 800.0 ft. msl. However, Class 1 structures and components at the station are not subject to flooding since the Probable Maximum Flood (PMF) would be contained by the Keowee Reservoir. The minimum external access elevation for the Auxiliary, Turbine, and Service Buildings is 796.5 ft. msl which provides a 6 in. water sill. Also, the plant site is provided with a surface water drainage system that protects the plants facilities from local precipitation.

In the Oconee PRA study, a postulated failure of the upstream Jocassee Dam resulted in a peak flood elevation at Keowee Dam of Elev. 813.12, which gives 1.9 feet available freeboard. Although the connecting canal between the two arms of Lake Keowee would lengthen the travel time of the flood wave, it is conservatively assumed that the water level resulting at Oconee Intake Dike would be the same as for Keowee Dam.

2.4.5.2 Flood Protection Measures for ISFSI Site

The site for the Oconee Site-Specific ISFSI is elevated well above the nominal plant yard grade at El. 825.0. Flooding of the ISFSI is not a credible event; therefore, no flood protection prevention measures are necessary.

2.4.6 Environmental Acceptance of Effulents

The only liquid used for the Oconee Site-Specific ISFSI is during preparation of the DSC and transfer cask within the confines of the plant Auxiliary Building. No liquids are used during the actual operation of the ISFSI.

2.4.7 Subsurface Hydrology

The Oconee Site-Specific ISFSI provides for the storage of spent nuclear fuel in a dry condition. Therefore, there will be no consumption of groundwater or impact to the groundwater system as a result of installing the ISFSI at the Oconee Station.

2.4.7.1 Groundwater Usage

The completed field survey of approximately 30 wells performed in the late 1960's determined that groundwater usage is almost entirely from the permeable zones within the saprolite with only minor amounts obtained from the underlying fractured bedrock. Yields from these shallow wells are low, generally less than 5 gpm, and are used to supply domestic water for homes and irrigation of lawns, gardens, and limited amounts for livestock. With only a few exceptions, the wells are hand dug, equipped with bucket lift and/or jet pump, and 40 to 60 ft. deep. At present, there is no industrial demand for groundwater within the area. The only appreciable groundwater draft observed is being supplied by eight wells for Keowee Elementary School, located four miles west of the site.

2.4.7.2 Regional Groundwater Conditions

The Oconee Station lies within the drainage area of the Little and Keowee Rivers which flow southerly into the Seneca River and subsequently discharge into the main drainage course of the Savannah River. The average annual rainfall at the site area is approximately 53 in.

The deposits of the Little and Keowee drainage basin are generally of low permeability which result in nearly total runoff to the two rivers and their numerous tributary creeks. Runoff occurs soon after precipitation, particularly during the spring and summer months when the soil percolation rates are exceeded by the short term but higher yielding rainfall periods. The area is characterized by youthful narrow streams and creeks which discharge into the mature Little and Keowee Rivers.

Throughout the area, groundwater occurs at shallow depths within the saprolite (residual soil which is a weathering product of the underlying parent rock) soil mantle overlying the metamorphic and igneous rock complex (Reference 1). Refer to Section 2.5. This saprolite soil, which ranges in thickness from a few feet to over 100 ft., is the aquifer for most of the groundwater supply. Wells are shallow and few exceed a total depth of 100 ft. Depths to water commonly range from 5 to 40 ft. below the land surface. Seasonal fluctuation is wholly dependent of the rainfall and the magnitude of change may vary considerably from well to well due to the limited areas of available recharge. Average fluctuation is about 3 to 5 ft. Both surface water and groundwater in this area are of low mineral content and generally of good quality for all uses.

To determine the general groundwater environment surrounding the plant area, groundwater levels were established in numerous domestic wells and exploratory drill holes during the original program in the late 1960's within a four-mile radius. Additional data was obtained from interviews with local residents regarding specific wells and discussions with State and Federal personnel. The results of the groundwater level survey are shown on Figure B-8. The results demonstrate that local subsurface drainage generally travels down the topographic slopes within the more permeable saprolite soil zones toward the nearby surface creek or stream. Gross drainage is southward to the Little and Keowee Rivers which act as a base for the gradient.

Because the topography and thickness of the residual soil, overlying bedrock control the hydraulic gradient throughout the area, and further, the relief is highly variable within short distances, it is not possible to assign a meaningful average gradient for the 15 square mile area surveyed. In all small areas studied within the four-mile radius, the groundwater hydraulic gradient is steep and conforms to the topographic slope. Water released on the surface will percolate downward and move toward the main drainage channels at an estimated rate of 150 to 250 ft. per year.

The gradient throughout the area represents the upper surface of unconfined groundwater and therefore is subject to atmospheric conditions. Confined groundwater occurs only locally as evidenced by the existence of isolated springs and a few exploratory drill holes which encountered artesian conditions. These examples do not reflect general conditions covering large areas but merely represent isolated local strata within the saprolite soil which contain water under a semi-perched condition and/or permeable strata overlain by impermeable clay lenses which have been breached by erosion at its exit and recharged short distances upslope by vertical percolation.

The plant area is on a moderately sloping, northwest trending topographic ridge which forms a drainage divide between the Little and Keowee Rivers located approximately 0.5 mile to the west and east, respectively. Groundwater levels at the plant site, measured during the 1966 drilling program and subsequently in four piezometer holes drilled for pre-construction monitoring purposes, ranged from elevation 792 ft. (msl) to 696 ft. (msl). The slope of this apparently free water surface is predominantly southeasterly toward the Keowee River and its tributary drainage channels. An average hydraulic gradient to the southeast of approximately 8.0 percent was plotted along a line of measured wells. This closely conforms to the existing topography as expected. Refer to Figure B-9 for measured water levels and typical water table profile.

2.4.7.3 Groundwater Quality

The surface water and groundwater of the area is generally of good quality (Reference 2). Of the wells surveyed, none were noted where water treatment is being conducted. Temperature of well water measured ranged from a low of 46 to a high of 59 degrees. The majority of readings were from 50 to 53 degrees Fahrenheit.

Water contains different kinds and amounts of mineral constituents. Temperature, pressure and length of time water is in contact with various rock types and soils determine the type and amount of mineral constituents present. Because ground waters are in intimate contact with the host rocks for longer periods of time, they have a more uniform and concentrated mineral content than surface waters. The mineral content of natural surface waters in the Piedmont Province is low due to the relative insolubility of the granitic, gneissic, and schistose host rocks and the reduced contact time caused by rapid runoff in the mountainous areas.

Tabulated below are the surface water constituents reported in parts per million from the Keowee River near Jocassee, South Carolina. The water sample was taken and analyzed by the U.S. Geological Survey, Water Resources Division in June 1965.

Silica (SiO_2)	7.8	Carbonate (CO_3)	0.0
Iron (Fe)	0.01	Bicarbonate (HCO_3)	7.0
Calcium (Ca)	1.0	Sulfate (SO_4)	1.0
Magnesium (Mg)	0.1	Chloride (Cl)	0.6
Sodium (Na)	1.2	Fluoride (F)	0.1
Potassium (K)	0.4	Nitrate (NO_3)	0.1
Dissolved Solids	150	Phosphate (PO_4)	0.0
Hardness as $CaCO_3$	3.0	PH	6.6
Specific Conductance	13.0		

Soil surveys conducted by the U.S. Department of Agriculture in cooperation with the South Carolina Agricultural Experiment Station assign pH values of between 5.0 and 6.0 for the Hayesville and Cecil soil

series which are present at the site area (Reference 3). Surface water samples taken from the Keowee River within one mile of the site have a pH of 6.5 to 7.0. It is expected groundwater at the site has a pH ranging between 5.5 and 6.0.

The cation exchange potential can be evaluated by knowing the SAR (Sodium Absorption Ratio), saturation extract values, and the pH of the soil. Two samples of saprolite soil were obtained from drill holes used in determining field permeability values and tested for Sodium Absorption Ratio (SAR). The results are tabulated as follows:

		Saturation Extract Values Milligram – equivalent per 100 grains of soil				
Sample No.	**pH**	**Cond. (mhos)**	**Calcium**	**Magnesium**	**Sodium**	**SAR**
1	5.8	5	0.015	0.000	0.0108	0.122
2	5.7	7	0.010	0.000	0.0166	0.235

Considering the amount of soil that is available is so great, it is evident that many times the amount of strontium and/or cesium contained in the waste could be absorbed. Further, the distribution coefficient for ion exchange of radionuclides with the sediments is dependent on the pH of the water in the formation (Reference 4). The distribution coefficient is a ratio of the reaction of these radionuclides that are absorbed on the soil and the fraction remaining in solution. It is expected that the soils surrounding Oconee have a ratio in the range of 80 to 150, and consequently a substantially lower average velocity for any radionuclide to that of natural water will result.

The estimated maximum rate of movement of water through the soils is about 0.75 feet per day. Using this rate in relation with the above distribution coefficient, bulk density and porosity of the soil, and ratio of the weight of soil to volume of groundwater it indicates the radionuclide velocity will be about .0015 that of groundwater. Using a safety factor of five for variance in flow and competition for exchangeable sodium ions, it would require more than 1000 years for strontium or cesium ions to migrate a distance of one-half mile. In summary, the movement would be so extremely slow that the saprolite soil is an effective natural barrier to the migration of radionuclides.

2.4.7.4 Program of Investigation

Permeability tests were performed in borings in the late 60's as part of the original site investigation program to determine permeabilities of the soil underlying the site. The tests were run according to the Bureau of Reclamations Field Permeability Tests, Designation E-19. Figure B-10 shows the arrangement of the field test equipment along with a brief description of the procedure used in determining the soil permeability test results. Test results are from 5 borings as presented in Table A-8. The formulae used in the calculations of the k values are shown in Figure B-11. Field permeability tests conducted within the saprolite soil yielded values ranging from 100 to 250 ft./yr. The permeability tests were performed in holes of varying depths to determine if the zoned typed weathering of the saprolite soil affects vertical permeability. Based on the test results, inspection of nearby road cuts, and a study of the exploratory drill logs, it is concluded that the surficial saprolite possesses lower permeability values than that found in the deeper strata. This correlates with the general profile of the saprolite in that the later stages of weathering produce a soil having a higher clay content than the more coarse-grained silty sand sediments below. This natural process of weathering results in the formation of a partial barrier to downward movement of the surface water.

2.4.7.5 Groundwater Conditions Due to Keowee Reservoir

As previously discussed, the groundwater levels at the plant range from elevation 792 ft. (msl) to below elevation 696 ft. (msl). The Keowee Reservoir operates with a maximum pool elevation of 800 ft. (msl). This results in raising the surface water elevation to that datum on the northern and western portions of land adjoining Oconee. It also raises the existing groundwater table for those local areas bordering the reservoir where formerly the groundwater surface was below elevation 800.0 ft (msl). The reservoir materially contributes in establishing a potentially larger recharge area and where it effects the groundwater results in a more stable hydraulic gradient with less seasonal fluctuation than formerly existed.

Preliminary studies indicate that Keowee Reservoir has created the following groundwater conditions at Oconee.

1. Groundwater continues to migrate downslope through the saprolite soil on a slightly steeper gradient in a southeasterly direction toward the Keowee River base datum.
2. There are two topographic divides which separate the nuclear station from the nearby reservoir: (1) a one-half mile wide north-south stretch of terrain west of the site, and (2) a narrow 500 ft. wide ridge north of the site. Original groundwater measurements in drill hole K-12, located atop the northern ridge, show water table conditions exist at about elevation 810 ft. (msl).
3. There should be no reversal of groundwater movement at the site, and all water percolates downward and away from the plant area.
4. The construction of Keowee Dam and Reservoir has not created adverse groundwater conditions at the plant site.
5. Infiltration of domestic wells, located beyond the plant one-mile exclusion radius, by surface water from the site is not possible under the groundwater conditions imposed by Keowee Reservoir.

2.5 Geology and Seismology

Specific soil testing has been performed at the designated location for the Oconee Site-Specific ISFSI. The data obtained from this testing is utilized in the foundation design of the ISFSI (See Section 2.5.5). It should be noted that foundation conditions at the ISFSI site are typical of those encountered in the general station area. The following sections discuss the Oconee site geology and seismology.

2.5.1 Basic Geologic and Seismic Information

Geologic and seismic investigative studies for Oconee Nuclear Station include the following:

1. a review of the available geological and seismological literature pertaining to the region;
2. a geological reconnaissance of the site, performed primarily for the purpose of evaluating the possibility of active faulting in the area;
3. geophysical explorations and laboratory tests to provide parameters for evaluating the response of foundation materials to earthquake ground motion;
4. an evaluation of the seismic history to aid in the selection of the design earthquake that the station might experience; and
5. the development and recommendation of seismic design parameters for the proposed structures.

The geologic field work at the site was performed concurrently with the drilling for the original plant site. The site reconnaissance is a continuation of the geologic field work done for the Keowee Dam. Local outcrops, though scarce, are examined and the rock types, joint and foliation orientation noted.

The original plant structures are founded on normal Piedmont granite gneisses. The construction characteristics of the residual soils overlying the rock that form the foundation for the ISFSI are known and present no problems in design or construction. The rock underlying the site, below surface weathering, is hard and structurally sound and contains no defects which would influence the design of heavy structures.

The southeastern Piedmont rocks are highly stable seismologically, and the Oconee Nuclear Site should be one of the nation's most inactive areas with respect to earthquake activity.

2.5.1.1 Regional Geology

The regional structure is typical of the southern Piedmont and Blue Ridge. The region was subjected to compression in the northwest-southeast direction which produced a complex assortment of more or less parallel folds whose axes lie in a northeast-southwest direction. The Blue Ridge uplift was the climax of the folding, and it was accompanied by major faulting, along a line stretching northeast through Atlanta and Gainesville, Georgia and across South Carolina, 11 miles northwest of the site. This has been termed the Brevard Fault.

The age of these uplifts has not been agreed on by geologists. The consensus of geologic opinion seems to require a period of severe deformation followed by at least one additional period of less severity. Probably all occurred during the Paleozoic Era, but it has been suggested that the last major uplift was as late as the Triassic (180 million years ago) when the Coastal Plain to the east was downwarped. A number of investigators have maintained that the major deformative movements occurred at least 225 million years ago. However, all the resulting stresses have not yet been fully dissipated.

There is no evidence of any displacement along these faults during either historic times or during the Geologic Recent Era as indicated in displacements in the residual soils that blanket the region. While the well known Brevard Fault passes 11 miles northwest of the site, there is no indication of a major fault in the immediate vicinity of the site. Furthermore, the major faults of the region are ancient and dormant, except for minor adjustments at considerable depth. Therefore, there is no indication of any structural hazard to foundations.

The site is underlain by crystalline rocks which are a part of the southeastern Piedmont physiographic province. This northeastward - trending belt of ancient metamorphic rocks extends northward from Alabama east of the Appalachians, and in South Carolina crosses the state from the Fall Line on the east to the Blue Ridge and Appalachian Mountains on the west. These rocks are generally recognized as being divided into four northeast-southwest trending belts in the Carolinas. From southeast to northwest they are the Carolina slate belt, Charlotte belt, Kings Mountain belt, and Inner Piedmont belt. The Oconee Nuclear Site is in the western, or Inner Piedmont Belt.

The Piedmont metamorphic rocks of the site were formed under many different combinations of pressure and temperature, and represent a complex succession of geologic events. The formerly accepted concept that the Piedmont consists only of the deep, worn-down roots of ancient mountains now seems untenable. The older theory that the rocks were exclusively of igneous origin is being replaced by the proposition that they represent highly metamorphosed sediments which have been folded, faulted, and injected to result in one of the most complex geologic environments in the world. It can be said with certainty, however, that these rocks represent some of the oldest on the continent. The new techniques of dating by radioactive decay have placed the age of the metamorphic episodes that produced these rocks as occurring from 1,100 my (million years) to 260 my ago. The successive northeastward trending bands of rocks vary greatly in lithology from granitic types to highly basic classifications, with gneisses and schists being the predominant classifications petrographically. In summary, the regional geology of the Oconee Nuclear Site can be accepted as typical of the southeastern Piedmont - narrow belts of metamorphic rocks trending northeast, with the foliation dipping generally to the southeast.

2.5.1.2 Site Geology

2.5.1.2.1 Geologic, History, Physiography, and Lithography

The rock present at this site is metamorphic. It is believed to be Precambrian in age; thus, it was formed over 600 million years ago. The complete history of this region is quite complex and has not been fully unravelled. However, it is the consensus of the geologic opinion that the formation consisted of thick strata of sedimentary rocks which were later downwarped and altered by heat and pressure. This first rock formed is termed the country rock.

More than one episode of regional metamorphism transformed the rock into metasediments with accompanying injection and mobilization by plastic flow.

Since the formation of the country rock, most of the mass has been altered or replaced by injection of granite gneiss, biotite hornblende gneiss, and one or possibly more pegmatite dikes.

It is not definite which is the younger: the granite gneiss injection or the biotite hornblende gneiss injection. The limited evidence points to the granite gneiss as the younger of the two.

The pegmatite dikes are the youngest rock known at this site. One such dike is exposed in the road cut on the east side of the state highway passing through the site. It clearly shows the pegmatite cutting through the older rocks, and thus, demonstrates that it is the youngest.

Regional metamorphism, folding, and some minor faulting occurred concurrently much of this early time.

This site is located within the Inner Piedmont Belt, at this locality the westernmost component of the Piedmont Physiographic Province. The topography of the area is undulating to rolling; the surface elevations ranging from about 700 ft. to 900 ft. The region is moderately well dissected with rounded hilltops, representing a mature regional development. The area is well drained by several intermittent streams flowing away from the center of the site in a radial pattern.

The local geology of the Oconee Nuclear Site is typical of the southeastern Inner Piedmont Belt. The foundation rock is biotite and hornblende gneiss striking generally northeast, with the foliation dipping southeast. The rock is overlain by residual soils, which vary from silty clays at the surface, where the rock decomposition has completed its cycle, to partially weathered rock, and finally to sound rock.

The strike of the foliation planes or bands of mineral segregation is north 6 degrees to 15 degrees east with an average dip of 22 degrees to 28 degrees to the southeast. However, due to the local folding or warping at this site, minor variations in the strike and dip of the foliation will occur within the site.

There have been periods of erosion and perhaps even continuous erosion since the close of the Paleozoic Era. The rock now encountered at this site represents the deeper portions of the original metamorphic complex.

The rock encountered at this site is of three main types; light to medium gray granite gneiss, light gray to black biotite hornblende gneiss and white quartz pegmatite with local concentrations of mica, both muscovite and biotite varieties.

The dominate rock type at this site is the light to medium gray granite gneiss. This rock type is generally moderately hard and hard below the initial soft layers encountered in the rock surface. Joints in this rock are brown iron stained in the upper softer layers, but in the deeper harder rock, the joints are not stained. This helps illustrate that the jointing at this site does not control the weathering or decomposition of the rock.

The second most abundant rock type is the biotite hornblende gneiss. The rock is generally weathered or softer to a greater depth than the granite gneiss. This is probably due to the higher percentage of biotite mica. Biotite mica is a potassium magnesium-iron aluminum silicate. The iron content of the biotite mica causes the rate of decomposition to accelerate. However, generally at the deeper portions of the original

plant borings, the biotite hornblende gneiss hardness increases to moderately hard or harder. Only a few thin soft layers were noted in this rock in the deeper portion of the original plant borings but not in the ISFSI site boring logs which are presented and discussed in Section 2.5.4.

A few layers of hard quartz pegmatite with local concentrations of mica were recorded. The thickness of the pegmatite layers are generally less than three feet. These pegmatite layers are dikes. A dike is a sheetlike body of igneous rock that fills a fissure in the older rock which is encountered while in a molten condition. There is an exposure of mica-quartz pegmatite dike on the east side of the state road cut passing through this project. This dike exposure is about 3.5 ft. wide, but due to the lack of knowledge of orientation of the dike, the exact width cannot be computed. The quartz pegmatite encountered in the original station borings probably represent other smaller dikes of the same material. These dikes are of hard, sound and durable material and should cause no concern to construction or foundation requirements.

2.5.1.2.2 Rock Weathering

Rock weathering at the Oconee Nuclear Site is about normal for Piedmont biotite gneisses. The range of depth before sound rock is reached is 0 to 35 ft. for the ISFSI foundation. Yard grade is nominally at elevation 825.0 msl. with the bottom of the foundation at elevation 822.0 msl. The resulting residual materials - clays, silts, and weathered rock - are structurally strong, and are used in situ for the foundation of this structure.

2.5.1.2.3 Jointing

The rock at the Oconee site is moderately jointed. All of the visible rock outcrops were studied in attempting to determine the correct orientation of the joint patterns.

Some moderately good rock outcrops were found and several joint pattern orientations measured. While studying and logging the original site rock cores, all of the joint dips were recorded. The dips of the joint patterns recorded in the rock cores were associated with the dips measured in the rock outcrops.

The rock has apparently not been subjected to stresses causing high concentrations of joints. The core borings indicate that jointing is widely spaced, and has not influenced the weathering pattern. Joints are about equally divided between strike and dip joints, with occasional oblique joints.

2.5.1.2.4 Ground Water

Subsurface water is typical of Piedmont area. The top of the zone of saturation, or water table, follows the topography, but is deeper in the uplands and more shallow in valley bottoms. It migrates through the pores of the weathered rock, where the feldspars have disintegrated and left interstitial spaces between the quartz grains. Additional water is contained in the deeper fractures and joints below the sound rock line. The water table is not stationary, but fluctuates continually as a reflection seasonal precipitation. Additional information on ground water is included in Section 2.4.7. Groundwater elevations encountered during the ISFSI site borings are noted on the boring logs, Section 2.4.4.

2.5.2 Vibratory Ground Motion

A seismological study for the Oconee Nuclear Site has been performed to determine the design and hypothetical earthquakes for the site and the ground motion associated with them. Details are discussed in Section 2.5.2 of the Oconee UFSAR.

2.5.2.1 Earthquake History

The largest earthquakes close to the site occurred near Charleston in August, 1886, some 200 miles from the site. Two shocks occurring closely in time, had an intensity estimated to be about Modified Mercalli IX at the epicenter and were perceptible over an area of greater than two million square miles.

Aftershocks of the main earthquake had intensities ranging up to Modified Mercalli VII. These shocks may be associated with a downfaulted Triassic basin under the coastal plain.

There have been two moderate earthquakes in the immediate vicinity of the plant since construction began.

In 1971, an earthquake occurred near Seneca, South Carolina. The descriptions of this event which occurred at 07:42 (EST) on July 13, 1971 have been examined from various sources. A MM intensity VI was assigned to the event by USGS based primarily on the report of a cracked chimney near Newry, about 10 km south of the present epicentral area. A detailed examination of the buildings and chimneys by Sowers and Fogle (1978) convinced them that the chimney in question had been broken and in a state of disrepair before the shock. They assigned an intensity IV (MM) to the shaking at Newry.

The July 13, 1971 event at 07:42 AM EDT was preceded by a felt shock at about 4:15 AM EDT and followed by at least one felt aftershock at 7:45 AM (Sowers and Fogle, 1978).

On August 25, 1979 (9:31 PM EDST, Aug. 26) a magnitude 3.7 earthquake occurred in the vicinity of Lake Jocassee, South Carolina. This MM intensity VI event was felt in an area of about 15,000 sq. km and was recorded locally on the three station Lake Jocassee seismographic network, and regionally on seismic stations in South Carolina, North Carolina, Georgia, Tennessee, and Virginia. During the period (August 26, 1979 September 15, 1979) 26 aftershocks were recorded and they ranged in magnitude from -60 to 2.0.

A list of earthquakes in the region is provided in Table A-9.

2.5.2.2 Geologic Structures and Tectonic Activity

The region (defined as North Carolina and South Carolina, and parts of Georgia, Alabama, Tennessee, and Virginia) is comprised of three large northeast-southwest trending tectonic zones: The coastal plain, the crystalline-metamorphic zone and the overthrust zone.

The site is located nearly in the center of the crystalline-metamorphic zone, which consists of six generally recognized metamorphic belts. From southeast to northwest these are: The Carolina slate belt, Charlotte belt, Kings Mountain belt, Inner Piedmont belt, Brevard belt, and Blue Ridge belt. The site location is within the Inner Piedmont belt. The rocks in the belts consist of metamorphosed sediments and volcanics that have been folded, faulted, and intruded with igneous rocks. These belts are delineated by differing degrees of metamorphism. Generally, the degree of metamorphism becomes progressively less from the northwest to the southeast.

The oldest metamorphic rocks are located in the Blue Ridge belt. The more easterly belts of younger rocks have undergone progressively less metamorphism.

To the north and west are found a series of fault systems. Since these faults are both numerous and extensive, they can be grouped together and referred to as the overthrust zone. These faults no doubt resulted from the formation of the Appalachians.

The great system of thrust faults in the overthrust zone and most of the known faulting within the crystalline-metamorphic zone apparently occurred during the last period of metamorphism (260 million years ago).

During the Triassic Period (180 to 225 million years ago), sediments were deposited over parts of the exposed metamorphic belts. These deposits and the older metamorphics were intruded by a system of

northwest-trending diabase dikes and were faulted by northeast-trending normal faults in the late Triassic Time (200 million years ago). Some of the older faults within the crystalline-metamorphic zone may have been active at this time.

From the late Triassic time until the present, the coastal plain has accumulated a sedimentary cover over its crystalline-metamorphic bedrock. These sediments overlap the bedrock and thicken toward the southeast, effectively masking any ancient faulting.

It is considered possible that igneous activity has occurred in the region after the Triassic because volcanic bentonitic clays of Eocene (approximately 50 million years ago) and possible Miocene age (12 million years ago) have been mapped in the sediments of the coastal plain in South Carolina. The source of this volcanic activity is presently unknown.

Faulting: The names, distances and directions from the proposed site, and the probable age of the known faulting in the region are as follows:

Name	Distance-Direction Form Site	Probable Age Millions of Years
Brevard Fault	11 Miles NW	260
Dahlonega Fault	40 Miles W	260
Whitestone Fault	47 Miles NW	260
Towaliga Fault	90 Miles S	260
Carterville Fault	104 Miles W	260
Gold Hill Gault	115 Miles E	260
Goat Rock Fault	140 Miles SW	260
Triassic, Deep River Basin, N.C. and S.C.	140 Miles E	200
Triassic, Danville Basin, N.C.	145 Miles NE	200
Crisp and Dooly Counties, GA.	190 Miles SW	12 to 70
Probable Triassic Basin Charleston, S.C.	200 Miles SE	200

The first seven faults are all associated with the last metamorphic period. The Brevard, Whitestone, Dahlonega, and Cartersville faults apparently form an interrelated system. This system separates the eastern metamorphic belts from the Blue Ridge metamorphic belt and the overthrust zone on the west.

The Towaliga, Goat Rock, and Gold Hill Faults, and the Kings Mountain belt apparently form another interrelated alignment within the eastern metamorphic belts. The Kings Mountain belt is not considered a fault. Its association and alignment in relation to the three known faults mentioned and the location of earthquake epicenters within the area bounded by these features, lead to the conclusion that these features form an interrelated alignment.

There is no surface indication that any of these three faults have been active since the Triassic Period (200 million years).

Two fault locations in the region have been thoroughly investigated by borings. These are the Cartersville fault near the Allatoona Dam, and the Oconee-Conasauga fault in Georgia. These faults were found to be completely healed and not to have moved in many millions of years.

The Triassic basins of the Carolinas and further north may be due to the release of the compressional forces which formed the Appalachians. These basins are down-faulted grabens which are filled with

Triassic sediments. Two earthquakes in the vicinity of McBee, South Carolina, may be related to an extension of a Triassic basin which has been inferred in the Chesterfield-Durham area.

Some faulting within the tertiary sediments in Dooly, Crisp, and Clay Counties, Georgia, has been mapped. The true aerial extent of this faulting is unknown. This faulting apparently ranges from Cretaceous to possibly Miocene in age (70 to 12 million years).

The earthquake activity near Charleston, South Carolina, may indicate an active fault in that region. However, no evidence of surface faulting has been found.

2.5.2.3 Correlation of Earthquake Activity with Geologic Structures or Tectonic Provinces

The region surrounding the Oconee Station site can be divided into three major areas on the basis of the regional tectonics and the seismic history. These major seismic areas are:

1. the overthrust zone and Blue Ridge metamorphic belt;
2. the crystalline-metamorphic zone, exclusive of the Blue Ridge belt; and
3. the coastal plain.

The greatest number of recorded shocks have occurred within the overthrust zone and the Blue Ridge metamorphic belt northwest of the Brevard, Whitestone, Dahlonega, and Cartersville fault system. The epicenters in this area are generally widely scattered.

There have been a small number of earthquakes within the crystalline-metamorphic zone, exclusive of the Blue Ridge metamorphic belt. These earthquakes, extending from central Georgia to North Carolina, may be associated with the Towaliga, Goat Rock, Gold Hill, Kings Mountain alignment.

The coastal plain has experienced few earthquakes outside of the Charleston area. Four shocks, at Wilmington, North Carolina and Savannah, Georgia, have occurred but are unrelated to any known faulting, although the Wilmington shocks were adjacent to the Cape Fear Arch.

The only earthquake which does not closely fit this system of seismic areas is the 1924 shock in Pickens County, South Carolina (MM V Intensity). However, it is likely that this earthquake is associated with the overthrust-Blue Ridge seismic area.

2.5.2.4 Maximum Earthquake Potential

The assignment of probable future earthquake activity can only be based upon the previous record and the known geology of the area. Although the seismic history of the region is fairly short, a reasonable picture of the seismicity of the area becomes apparent from a study of the epicenter locations and the regional tectonics.

There are three significant zones of seismic activity in the general vicinity of the site; the Brevard and related faults zone, the overthrust zone, and the Towaliga, Goat Rock, Gold Hill, Kings Mountain alignment.

An evaluation of the earthquake activity and the regional geology can result in the selection of a series of maximum-sized shocks which are likely to occur in these various areas. Conservatively, it can be assumed that the previous maximum-sized shock on a particular fault zone can occur during the economic life of the power station and Oconee Site-Specific ISFSI at perhaps the nearest approach of the particular fault system to the site.

Zone	Location	(MM) Intensity at Epicenter	Estimated Magnitude (Richter)
Brevard Fault Zone	11 Miles NW	VI	Less than 4 ½ to 5
Overthrust	75 Miles NW	VIII	Less than 5 ½ to 6
Towaliga, Goat Rock Gold Hill, Kings Mountain Alignment	30 Miles SE	VII – VIII	Less than 5 ½ to 6

2.5.2.5 Seismic Wave Transmission Characteristics of the Site

Static and dynamic engineering properties of the soil and rock materials that underlie the general plant site area are discussed in Section 2.5.4 of the Oconee UFSAR. Design response spectra that include considerations of the thickness and distribution of these materials are discussed in Section 2.5.2.8 of the Oconee UFSAR.

2.5.2.6 Maximum Hypothetical Earthquake (MHE)

The MHE acceleration value is 0.15 g for structures founded on overburden. The design response spectra are covered in Section 2.5.2.8.

2.5.2.7 Design Base Earthquake

It is considered likely that the shocks listed in Section 2.5.2.4 could occur no closer than the indicated distances from the site during the life of the planned facilities. Since the magnitudes of these shocks are fairly small, the distance from the epicenter becomes extremely important. Ground accelerations would diminish rapidly with the distance from the epicenter. Although larger earthquakes occur within other fault zones, the highest ground accelerations at the site would be experienced from an earthquake along the Brevard fault zone. The assumption of a shock of less than Richter Magnitude five occurring along the Brevard fault zone at its closest location to the site (11 miles), would give ground motions on the order of five percent of gravity at the site. Vertical ground accelerations, as contrasted to the horizontal accelerations, would be only slightly less than five percent of the gravity in the competent rock at the site.

2.5.2.8 Design Response Spectra

The Oconee UFSAR provides that the maximum ground acceleration for structures founded on over burden (MHE) is .15g (Section 2.5.2 and Figure 2-51 of the Oconee UFSAR). The accelerations considered and used for the design of the NUHOMS®-24P system envelope the MHE acceleration (Reference 6).

2.5.3 Surface Faulting

This information is discussed in Sections 2.5.1 and 2.5.2.

2.5.4 Subsurface Materials

2.5.4.1 Exploration

A grid pattern of borings was established to provide the maximum amount of information for determining the foundation and soil conditions and permit flexibility in final Oconee Site-Specific ISFSI layout, alignment, and elevation.

The general site area is shown on Figure B-12 and the site and boring layout is shown on the Boring Plan, Figure B-13.

The drilling, sampling, and rock coring were performed in accordance with methods specified by the American Society for Testing and Materials:

1. "Penetration Testing and Split Barrel Sampling of Soils" - D-1586-64T
2. "Diamond Core Drilling for Site Investigation" - D-2311-62T
3. "Thin Walled Tube Sampling of Soils" - D-1587-63T

Boring logs are given in Figure B-14 through Figure B-27.

2.5.4.2 Groundwater Conditions

Section 2.4.7 provides a discussion of the existing groundwater conditions at the Oconee site.

It is believed that the removal of the overburden due to construction of the Oconee Site-Specific ISFSI has had little, if any, effect on the water table. If the water table elevation did change, it is believed that it would be a slight drop. The elevation of the water table at the ISFSI site when constructed varied from elevation 797 feet at the south end to elevation 822 feet at the north end.

Hydrostatic uplift will not occur during the life of the ISFSI because the foundation of the HSMs and associated pavement is at or above the water table. There may be some seepage through the cut into the hillside; however, adequate drainage is provided around the ISFSI site to carry away seepage.

At the south end of the ISFSI site, the elevation of the water table is far below the foundation of the HSMs. At the north end, the foundation of the HSMs will be near the water table elevation. However, the HSM structure at the north end of the ISFSI site is partially founded on rock. Therefore, there will be no reduction of shear resistance due to potential seepage along the bedding.

2.5.5 ISFSI Foundation

A specific soil testing (results and locations presented in Section 2.5.4) and foundation evaluation has been performed at the Oconee Site-Specific ISFSI site to assist in the development of the insitu static soil bearing pressure. Fourteen (14) soil borings were taken in and around the ISFSI site. The location of these borings is shown in Figure B-13. A line of boring was taken along the length of the future foundation of the HSMs. From these borings several undisturbed samples were taken. Several tests, including the triaxial shear test, were performed on selected undisturbed samples. The results from the triaxial shear test provided essential information used to determine the ultimate and allowable bearing capacity. (The triaxial shear tests were performed in accordance with the Corps of Engineers Manual EM10-2-1906, Appendix 10).

After inspection of the boring logs, soil samples, and tests, the worst case soil data were selected and used in the Meyerhoff bearing capacity equation to determine the ultimate soil bearing capacity, which is approximately 12.0 kips/square foot. To obtain the allowable static soil bearing capacity, a factor of safety of 3.0 was applied to the ultimate capacity, which yields the allowable bearing pressure of 4.0 kips/square foot (Reference 9).

The largest applied static bearing pressure was calculated by first determining the dead weight of the HSM with a fully loaded DSC and then dividing by the area of the foundation. This maximum applied static bearing pressure was computed to be 3.3 kips/square foot, which is less than the allowable soil bearing pressure of 4.0 kips/square foot.

As shown by the boring logs, the HSM foundation will to a large degree rest entirely on either firm soil or partially weathered rock with penetration blow counts ranging from n = 12 to refusal. A conservative

analysis was performed to determine the worst-case settlement of an HSM array. Both a 2x3 array and a 2x10 HSM array were considered. This analysis indicates that the worst-case differential settlement will cause the 2x10 HSM array to experience a differential settlement of about 3.0 inches along the North-South axis. Differential settlement in the East-West direction will be negligible.

These settlements are accounted for in the foundation design. The foundation was analyzed as a finite module using the computer code McAuto STRUDL (Reference 7).

This computer code models settlements by the use of calculated soil springs which provide consideration for the settlements. Considering the small magnitude of this settlement, the integrity and radiological shielding of the HSM will not be adversely impacted. The foundation structure consists of a 3 ft. reinforced concrete mat. Typical HSM reinforcement is shown in Figure 8.1-9 of Reference 6.

The limiting calculated maximum stresses and allowable stresses for loadings as defined by Reference 6 envelope the site foundation stresses for the Oconee ISFSI site. These forces are for the accident condition assuming blocked vents and bound all other loading combinations.

2.5.6 Liquefaction

Potential liquefaction of soils under the Oconee Site-Specific ISFSI foundation area is not a concern because all of the foundation materials are non-liquefiable. The three foot thick concrete mat bears entirely on either firm soil or partially weathered rock having Standard Penetration Test blowcounts ranging from N = 12 to refusal. Figure B-28 shows the longitudinal profile of the ISFSI foundation level in relation to both the original ground and to partially weathered rock, based on site borings.

2.5.7 Slope Stability

The Oconee Site-Specific ISFSI site includes cut slopes along both sides of the ISFSI site access road, and along the west, north, and northeastern sides of the ISFSI site as shown in Figure B-12. Fill slopes are located along the southeastern and south sides of the ISFSI site. The maximum vertical cut is approximately fifty feet and the maximum vertical fill is approximately ten feet. The maximum ISFSI slope is two horizontally to one vertically.

The stability of slopes associated with the ISFSI site was modeled by a program that utilizes the circular arc analysis method of slices. The program postulates a failure arc through the soil embankment or foundation, computes the soil mass driving moment and the soil mass resisting moments associated with the postulated failure arc, and then determines the resulting safety factor by dividing the total resisting moment by the total driving moment. The computer program allows the computation of a large number of safety factors associated with many postulated failure arcs (Reference 8).

The slope stability analyses were performed using the maximum ISFSI site slope of two horizontal to one vertical. Actual site soil engineering parameters, based on laboratory testing of soil samples, were determined. (Reference the site boring records presented in Figure B-14 through Figure B-27). The Seismic Design Input Criteria specified in Section 3.2.3.1 were used as input in determining the seismic behavior of the ISFSI site slopes.

The minimum safety factors calculated for any postulated failure arc of the vertical cut and fill slopes of the ISFSI site are as follows:

Slope Loading Condition	Minimum Calculated Safety Factor
55 feet vertical cut slope, static	1.62
55 feet vertical cut slope dynamic	1.22
10 feet vertical fill slope static	2.06
10 feet vertical fill slope dynamic	2.03

Therefore, the stability of the ISFSI site slopes is ensured since the minimum safety factor is greater than 1.0 for all slopes for all analyzed conditions.

2.6 References

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3.0 Principal Design Criteria

3.1 Purpose of the Oconee ISFSI

The purpose of the Oconee Site-Specific ISFSI is to insure the uninterrupted operation of the three unit Oconee Nuclear Station by providing additional long-term spent fuel storage capacity. Prior to the storage of spent fuel in the ISFSI, the existing storage system consisted of two separate wet spent fuel pools that were rapidly approaching a maximum operating inventory. The Oconee Site-Specific ISFSI utilizes the NUHOMS®-24P System. NUHOMS®-24P is comprised of a series of reinforced concrete HSMs which will each house a stainless steel, helium filled DSC containing 24 qualified spent fuel assemblies. The DSC top inner and outer top cover plates are both independently seal welded to provide total confinement of the irradiated fuel. A shielded transfer cask is used to transfer the DSC to the HSM from the spent fuel pool. During storage, the HSM provides radiation shielding and passive decay heat removal from the DSC.

3.1.1 Material to be Stored

Each DSC is capable of storing 24 PWR assemblies. The following subsections will address the physical, reactivity, thermal and radiological characteristics of spent fuel to be stored in the DSC.

3.1.1.1 Physical Characteristics

The physical characteristics of the reference 15 x 15 fuel are listed in Table A-10. Additional information may be found in the Oconee UFSAR, Chapter 4.

3.1.1.2 Reactivity Characteristics

The reactivity of the spent fuel assemblies must be limited for criticality control purposes. Reactivity is a function of both the initial enrichment and the discharge burnup. Reactivity equivalence curves which show the acceptable combinations of initial enrichment and discharge burnup are given in Figure B-48. For criticality control, the spent fuel assemblies must fall into the acceptable range above the initial enrichment/burnup curve in order to qualify for storage in the DSC (see Section 10.2.5.1). Despite the multiple verification steps and extensive administrative controls used to assure selection of qualified irradiated fuel assemblies, criticality control for a misloaded array of unirradiated fuel is maintained by assuring that the DSC is filled with borated water (≥ 1810 ppm boron) and submerged in a borated water spent fuel pool (≥ 1810 ppm boron) during loading and unloading operations.

In the event that unqualified IFAs or other unirradiated fuel assemblies are erroneously placed in the DSC, the double contingency principle is applied such that the negative reactivity worth of 1810 ppm soluble boron in the spent fuel pool water (from which the DSC cavity will be filled initially) is sufficient to maintain k-eff below 0.95 (0.98 under optimum moderator conditions) even for 24 fresh fuel assemblies enriched to 4.0% wt. U-235. Further margin is available since the Oconee spent fuel pools are maintained at approximately 2000 ppm, or greater, and the DSC cavity is filled with water from the spent fuel pool prior to fuel loading.

3.1.1.3 Thermal Characteristics

The heat generation is limited to 0.66 kw per fuel assembly. This value is based on storage of 24 assemblies per DSC with a nominal burnup of 40,000 MWD/MTU, an initial enrichment of 4.0 wt % U-235 and a nominal decay period of ten years. Other combinations of burnup, initial enrichment and cooling times may also be acceptable upon further analysis demonstrating acceptable decay heat levels.

3.1.1.4 Radiological Characteristics

The DSC is designed for a maximum dose rate of 200 mr/hr at the surface of the top (with temporary neutron shielding if necessary during welding operations) and bottom end shield plugs. The HSM is designed for an average dose rate of 20 mr/hr at the surface of the module dropping down to a negligible level at the site boundary. Fuel with a maximum burnup of 40,000 MWD/MTU, an initial enrichment of 4.0 w/o U-235 and a decay of ten years will not exceed these dose values. Other combinations of burnup, initial enrichment and cooling times may also be acceptable upon further analysis demonstrating acceptable radiation dose rate levels.

3.1.2 General Operating Functions

3.1.2.1 Overall Functions of the Facility

The Oconee Site-Specific ISFSI is designed to maximize the use of existing site features and equipment and minimize the need to add or modify equipment. The storage facility is located within a fenced area inside the ONS plant protected area. The only services required from the station during the ongoing storage mode is through security surveillance equipment tied in with the plant security center. The storage facility is included in a routine daily surveillance. Power supply to the storage facility is retail. Support services from the plant are necessary only during loading (and unloading) operations.

Following periodic delivery of the individual DSCs and construction of the HSMs, the DSC is loaded into the transfer cask and the two are lowered into the spent fuel pool. The DSC/transfer cask is loaded with 24 spent fuel assemblies previously selected per criteria given in Section 10.3. Once fuel loading is complete, the DSC is fitted with its top end shield plug and pulled out of the pool. The water level in the DSC is then lowered slightly and the top end shield plug is welded into place. This is followed by further draining and eventual vacuum drying of the DSC cavity. The cavity is then back-filled with helium followed by further seal welding of both penetrations. An additional cover plate is welded over the top end shield plug, the cask lid bolted in place, and the transfer cask is then lowered to the transfer trailer and rotated to the horizontal position. Transfer from the spent fuel pool receiving area to the ISFSI is done with the use of a separate towing vehicle. The transfer trailer is then carefully aligned with the opening in the HSM, and the cask is docked to the HSM and secured in place. The hydraulic ram system then is used to push the DSC out of the transfer cask and into the HSM. This method utilizes a small penetration at the bottom of the transfer cask to allow access to the DSC through the transfer cask bottom. A large access door is then lowered and tack welded in place to close off the HSM access.

The HSMs are constructed on a level, reinforced concrete slab designed for normal transfer and storage conditions and postulated accidents.

Once loaded and secured, the passive design of the HSM provides for sufficient radiation shielding, tornado missile protection, and decay heat removal capabilities for the stored spent fuel. The double seal welded DSC closure system together with multi-pass welding procedures provide a multiple barrier against releases of radioactive material.

A more detailed description of each NUHOMS®-24P system component is provided in the following subsections.

3.1.2.2 Handling and Transfer Equipment

All components of the NUHOMS®-24P system are designed to interface where necessary with all existing Oconee fuel handling/storage equipment and facilities. This includes fuel pool receiving areas, radwaste systems, overhead cranes, yoke and yoke extension, fuel handling bridge and mast, auxiliary hoists, water, power and gas supplies, and clearance restrictions.

The additional equipment required to support the operation of the NUHOMS®-24P system includes the DSC, the transfer cask, the transfer trailer with hydraulic alignment mechanisms, the hydraulic ram assembly, the HSM and various miscellaneous tools, lids, gauges, hoses. Other equipment necessary to operate the system includes a towing vehicle to be used for moving the transfer trailer to and from the ISFSI, and a mobile yard crane for raising and lowering the HSM front access door. This equipment is further described as follows:

1. DSC - The DSC serves as the confinement vessel for the 24 fuel assemblies during both the storage mode and transfer operations. Seal welds on the inner and outer top cover plates provide redundant containment of all radioactive products within or on the surface of the spent fuel assemblies. The top and bottom shield plugs also provide for biological shielding during DSC welding, drying, and backfilling, operations, transport of the fuel assemblies, and during all operations performed at the front end of the HSM.
2. Transfer Cask - The transfer cask provides for dry transfer of the DSC from the Oconee fuel storage pool to the storage facility. The transfer cask utilizes a lead gamma shield and a solid neutron shield to maintain acceptable surface dose levels during transfer operations. A removable access plate at the bottom of the cask provides access to the DSC by the hydraulic ram during transfer of the DSC into or out of the HSM. The cask has a bolt on closure lid to keep the DSC in place during cask movement. Lift trunnions are provided at the top end of the transfer cask to interface with a lift yoke which will in turn interface with the spent fuel pool overhead crane. These top lift trunnions together with bottom trunnions provide cask support on the trailer during transfer operations.
3. Transfer Trailer - The transfer trailer allows for movement of the entire DSC/transfer cask assembly to the ISFSI. It is designed with a positioning mechanism that moves the cask in the horizontal and vertical directions to ensure alignment with the HSM. Final alignment accuracy is verified by an optical alignment system.
4. Hydraulic Ram - The hydraulic ram assembly is mounted on the transfer trailer. The ram is aligned with the bottom access portal of the horizontally positioned cask and engaged to slowly push the DSC from the cask into the HSM. A grappling ring on the bottom of the DSC and grappling arms on the hydraulic ram allow for eventual retrieval of the DSC using the same operations in reverse.
5. Horizontal Storage Module (HSM) - The HSM provides protection for the DSC during the storage mode and provides sufficient biological shielding from the stored spent fuel. Passive decay heat removal results from air entering shielded air ducts near the bottom of the structure, passing up and around the DSC and picking up heat before being exhausted through shielded vents at the top of the HSM. The HSM design includes a front access fitted with a carbon steel door and the coupling system for mating with the transfer cask. The HSM is fitted with a set of rails which serve as a bearing surface for movement of the DSC into and out of the module and as the primary support structure for the DSC during storage.

A more detailed description of these primary NUHOMS® components, including design criteria, is provided in Chapter 4.

3.2 Design Criteria for Environmental Conditions and Natural Phenomena

The Oconee Site-Specific ISFSI is designed to perform its intended safety function under normal and extreme environmental conditions. In general, the structural and mechanical safety criteria of the ISFSI are the same as or enveloped by the criteria specified in the NUHOMS®-24P Topical Report.

Details of the HSM lightning protection are contained in Section 8.2. Oconee foundation conditions are described in Section 2.5

3.2.1 Tornado and Wind Loadings

3.2.1.1 Applicable Design Parameters

The Oconee Site-Specific ISFSI was constructed within the existing boundaries of the Oconee Nuclear Station. As stated in Section 3.2.1 of Reference 1 the most severe tornado and wind loadings specified by NRC Regulatory Guide 1.76 and NUREG-0800, Section 3.5.1.4, were selected for design consideration. Therefore, both the HSM and the transfer cask are designed in accordance with NRC Regulatory Guide 1.76 and NUREG-0800, Section 3.5.1.4.

As stated in Section 3.2.1.1 of Reference 1, "... the maximum wind speed is 360 miles per hour, the rotational speed is 290 miles per hour, the maximum translational speed is 70 miles per hour, the radius of the maximum rotational speed is 150 feet, the pressure drop across the tornado is 3.0 psi, and the rate of pressure drop is 2.0 psi per second."

3.2.1.2 Determination of Forces on Structures

The tornado loading combination used for design of the HSM is:

$y = 1/\varnothing\ (1.0D + 1.0L + 1.0T_o + 1.0W_t + 1.0P_j)$

Where

Y	= required yield strength of the structure
$\varnothing$	= concrete capacity reduction factor
$\varnothing$	= 0.90 for concrete flexure
$\varnothing$	= 0.85 for shear in concrete
$\varnothing$	= 0.90 for axial tension in concrete
$\varnothing$	= 0.70 for tied compression members
$\varnothing$	= 0.90 for fabricated structural steel
T_o	= normal operating temperature
L	= live loads on structure
D	= dead loads of structures and equipment
W_t	= stress induced by design tornado wind velocity (drag, lift, and torsion)
P_I	= stress due to differential pressure

Shape factors will be applied in accordance with ANSI A 58.1 - 1982.

3.2.1.3 Tornado Generated Missiles

As described in Section 3.2.1.2 of Reference 1, the determination of impact forces created by Design Basis Tornado (DBT) generated missiles for the HSM was based on the criteria provided by NUREG-0800, Section 3.5.1.4, III.4. Accordingly, three types of missiles were postulated. The velocity of the missiles was conservatively assumed to be 35 percent of the combined translational and rotational velocity for the DBT or (0.35)(360), which is 126 miles per hour. For the massive high kinetic energy deformable missile specified in NUREG-0800, a 3,967 pound automobile with a 20 square foot frontal area impacting at normal incidence was assumed. For the rigid penetration-resistant missile specified, a 276 pound, eight-inch diameter blunt-nosed armor piercing artillery shell, impacting at normal incidence

was assumed. For the protective barrier impingement missile specified, a one-inch diameter solid steel sphere was assumed.

The possibility of a tornado damaging a transfer cask/DSC in transit to the HSM is a low probability event. The probability of a tornado occurring at the Oconee site and generating a missile that impacts the cask is less than $1x10^{-7}$ per transfer trip. This is based on site-specific tornado frequencies derived from 35 years of National Severe Storm Forecasting Center data and assumes a conservative exposure time to DBT effects of 24 hours. However, the transfer cask has been evaluated for the tornado wind speed and missiles specified for the HSM. The maximum DBT tornado wind speed of 360 mph produces a design pressure of 304 psf. The 3,967 pound automobile and 276 pound eight inch diameter shell missiles are also considered. The one inch diameter spherical missile effects are enveloped by the eight inch shell missile.

3.2.1.4 Ability of Structures to Perform

The Oconee Site-Specific ISFSI is designed to withstand the design basis tornado wind loads. All components of the ISFSI with the exception of the air outlet shielding blocks of the HSM are designed to withstand the tornado generated missile forces as described in Section 8.2.2 of Reference 1. The loss of the air outlet shielding blocks is discussed in Section 8 of the NUHOMS®-24P Topical Report. The HSM is anchored to the foundation slab to mitigate overturning and sliding effects using dowel rods of a size and spacing consistent with the HSM wall vertical reinforcement.

The possibility of total air inlet and outlet blockage by foreign objects or burial under debris during a tornado event is considered. The effect of facility burial under debris is presented in Section 8.2 of Reference 1.

The transfer cask analysis for tornado wind speed and missile effects was performed for the cask secured in the horizontal position on the support skid and transfer trailer. The following criteria were used to evaluate the adequacy of the transfer cask for the loads described in Section 3.2.1.3.

1. Stability
2. Penetration Resistance
3. Stresses

The main components of the transfer cask considered in this analysis were the structural shell, and the top and bottom cover plates. Since the primary purpose of the solid neutron shield is biological shielding and since it is located on the cask exterior, it was conservatively assumed that the neutron shield will be ruptured by a DBT missile strike and therefore was not considered in the structural analysis. A brief description of the analysis follows.

1. Stability Analysis

   A stability analysis for the transfer cask mounted on the skid/trailer assembly was performed for the wind pressure loads and the massive missile impact.

   For the wind pressure loads, the overturning moment was compared to the stabilizing moment to determine the factor of safety against overturning. A factor of safety of 3.1 was calculated.

   For the massive missile impact, it was conservatively assumed that the missile impacts the uppermost part of the cask. The angle of rotation (θ) of the cask/skid/trailer arrangement at impact was calculated as 3.0 degrees assuming a rigid pavement. This calculation was based on the conservation of angular momentum, and also compared to the angle (θ_{tip}) necessary for the cask/skid/trailer to tip over. Tip-over occurs when the center of gravity of the cask is directly above the point of rotation. This was calculated as 32.7 degrees. Since $\theta < \theta_{tip}$, tip-over does not occur and the stability of the cask/skid trailer arrangement is maintained.

2. Penetration Analysis

Penetration due to the 276 pound rigid missile was calculated using two formulas obtained from the literature. The added energy absorbing affect of the neutron shield material was omitted from this calculation to give a more conservative result. The first approach, suggested by Nelms (Reference 4) is for a lead-backed shell:

$$T = \left[\frac{KE}{2.4\, S_u D^{1.6}} \right]^{0.71} = 0.50 \text{ inches}$$

Where:

T	= Minimum required steel plate or shell thickness to resist penetration
KE	= Kinetic energy = 1/2 mV^2
M	= Mass of missile = 276/g
	= 0.714 lb. sec^2/in.
V	= Velocity of missile
	= 2,218 in./sec.
S^u	= Ultimate strength of cask structural shell = 70,000 psi
D	= Diameter of missile = 8.0 inches

The second formula used was developed by the Ballistic Research Laboratory (Reference 5):

$$T = \frac{KE^{2/3}}{672D} = 0.52 \text{inches}$$

Where:

KE	= Kinetic energy = 1/2 mV^2
M	= Mass of missile
	= 8.57 lb. $sec.^2$/ft.
V	= Velocity of missile
	= 184.8 ft./sec.
D	= Diameter of missile = 8.0 inches

Both methods produce a consistent result which shows a predicted penetration of 0.5 inches compared to the minimum structural shell thickness of 1.5 inches. Therefore the DBT missile will not penetrate the cask and the DSC will remain intact.

3. Stress Analysis

Conservative hand calculations were performed to determine the peak stresses in the cask shell, and the top and bottom cover plates due to DBT loads. A summary of the stress results is provided in the attached Table A-11. The analytical method for each of the load cases shown in this table is briefly described below.

a. Wind Pressure Loads: A uniform line load of 2.18 Kips/ft. was applied to the full length of the cask. The correlation of Roark and Young (Reference 6) Table 31, Case 9c was conservatively used to calculate membrane and bending stresses. The analyses of the three inch top and two inch bottom cover plates were performed using Case 10, Table 24 of Roark and Young. The top cover plate was assumed pinned at the edges while fixed edge supports were assumed for the bottom cover plate.

b. Massive Missile Impact: Based on the conservation of angular momentum, the total force on impact was calculated to be 257 kips. This force was applied as a line load to the cask shell and as a pressure load to the top and bottom cover plates. The analysis method followed that described above for the wind pressure loads.

c. Penetration Resistance Missile: The impact force due to the eight inch diameter, 276 pound missile was calculated from the conservation of momentum as 63.4 kips. Case 9a, Table 31 of Reference 6 was used to calculate the membrane and bending stress for the cask shell while Cases 16 and 17, Table 24 of Reference 6 were used to calculate the stresses in the top and bottom cover plates respectively.

3.2.2 Water Level (Flood) Design

The grade level of the Oconee Site-Specific ISFSI is El 825.0. This elevation is 11.9 ft. higher than the calculated maximum flood water elevation at Oconee due to a postulated breach of the upstream Jocassee Dam (See Section 2.4.5.1).

3.2.3 Seismic Design

3.2.3.1 Input Criteria

The maximum horizontal and vertical ground acceleration (Maximum Hypothetical Earthquake - MHE) specified for the Oconee site is .15g (Section 2.5.2.8 of the Oconee UFSAR). The Oconee site accelerations are less than the analyzed values of .17g vertical and .25g horizontal used for NUHOMS® components (Reference 1) and thus are enveloped by the generic NUHOMS® analysis.

The Oconee HSMs were designed using the seismic criteria from Reference 1. As stated in Section 3.2.3 of Reference 1, "The maximum horizontal ground acceleration component selected for design of the NUHOMS®-24P was 0.25g. The maximum vertical acceleration component selected was two-thirds of the horizontal component which is 0.17g. In order to establish the amplification factor associated with the generic design basis response spectra, various frequency analyses were performed for the different NUHOMS®-24P components and structures. The results of these analyses indicated that the dominant lateral frequency for the reinforced concrete HSM was 25 Hertz. The corresponding horizontal seismic acceleration used for design of the HSM was 0.32g. Conservatively assuming that the dominant HSM vertical frequency is also 25 Hz. produces a vertical seismic design acceleration of 0.22g."

The effects of a seismic event occurring during the transport of a loaded DSC resting inside the NUHOMS®-24P transfer cask and secured to the transport skid/trailer was postulated. This load case is conservatively enveloped by the postulated normal transport load accelerations of ±0.5g acting in the vertical, axial, and transverse directions, applied simultaneously at the center of gravity of the transfer cask, as specified in Section 8 of Reference 1. These accelerations envelope those which would result from a seismic event in the highly unlikely event that a design basis earthquake would occur during transport of the loaded DSC to or from the HSM.

3.2.3.2 Seismic System Analysis

The stresses in the Oconee Site-Specific HSMs and the DSCs due to the .15g horizontal and vertical motion for the MHE are enveloped by the results of the generic seismic analysis reported in the NUHOMS®-24P Topical Report (Reference 1). The maximum HSM reinforced concrete bending moments and shear forces in Table 8.2-3 of Reference 1 envelope the seismic loads at Oconee.

The foundation of the HSM is also designed to withstand the forces generated by the MHE (See Section 2.5.5).

3.2.4 Snow and Ice Loads

The NUHOMS®-24P Topical Report specified a postulated live load of 200 pounds/ft^2 which conservatively envelopes the maximum snow and ice loads for the Oconee site.

3.2.5 Combined Load Criteria

Load combination criteria established in the NUHOMS®-24P Topical Report for the HSM, DSC and DSC support assembly meet or envelop the load combinations required by the Oconee UFSAR Section 3.8.

The HSM analyses summarized in Reference 1 considered combinations of HSMs ranging from a single stand-alone module up to the maximum array size of 2x10. The finite element models used in the analyses are applicable to both side-by-side or back-to-back arrangements. Different DSC loading patterns were analyzed for each size of array to establish the worst case design loadings.

The analyses showed that the single HSM provides the governing case for load combinations containing tornado wind and missile loads, seismic loads and flooding conditions. The postulated failure mode for each of these cases is sliding or overturning of the HSM unit. The analyses also showed that the thermal loads for a 2x10 array control the reinforcement requirements for the walls, roof and foundation members for all intermediate array sizes.

Therefore, Reference 1 presents a design configuration which envelopes the loads from a single HSM to a 2x10 array.

3.3 Safety Protection System

3.3.1 General

The Oconee Site-Specific ISFSI is designed for safe and secure, long-term containment and storage of IFAs. The major components which assure that the safety objectives are met are listed in Table A-12. The major procedures which require special design consideration are:

1. Double Closure Seal Welds on DSC Ends
2. Radiation Exposure During DSC Closure and Drying Operations
3. Minimization of Contamination of DSC Exterior by Pool Water
4. Minimization of Radiation Exposure During Transfer of the DSC from the Transfer Cask to the HSM

These items are addressed in the following subsections.

3.3.2 Protection By Multiple Confinement Barriers and Systems

3.3.2.1 Confinement Barriers and Systems

The Oconee Site-Specific ISFSI design incorporates multiple confinement barriers to ensure there will be no release of airborne radioactive effluent to the environment. The radioactivity which must be confined is from the IFAs themselves and DSC exterior contamination from IFA loading operations in the spent fuel storage pool. ISFSI multiple radioactivity confinement barriers are listed in Table A-13.

DSC exterior contamination is minimized by preventing spent fuel storage pool water from contacting the DSC exterior. DSC loading procedures (See Section 4.4.1) require that the annulus between the transfer cask and DSC be filled with demineralized water and sealed prior to immersion in the spent fuel pool. Annulus sealing is accomplished by an inflatable seal between the transfer cask and DSC. The combination of the above operations provides assurance that the DSC exterior surface has less residual contamination than required for shipping cask externals (i.e., 10CFR 71.87(i)(1)). A surface swipe of the DSC exterior is taken while it is in the cask decontamination pit to assure this level of contamination is not exceeded.

The annulus seal is an inflatable fabric reinforced elastomeric tube. An automotive-type valve stem permits inflation to approximately 25 psig. This design can accommodate the maximum variation in the annulus width (.5 to 1.5" at the cask flange). The seal is placed in the annulus and inflated prior to immersion in the fuel pool. It remains in place at least through the completion of top end shield plug decontamination. The seal may remain in place until just prior to DSC seal welding. The seal is then stored for future use, or discarded if it has become damaged.

The function of the annulus seal is to minimize the potential for DSC and cask contamination during fuel loading. It is not intended to be a “safety protection system” for the NUHOMS® system. The seal provides an added assurance that minimizes the potential spread of contamination and therefore reduces personnel radiation exposures. The NUHOMS® system will safely function with or without the seal, and as such, its correct placement and operation are not critical to the safety of the system. In the event that the seal should fail, the water filled annulus will minimize the spread of contamination below the top of the DSC. Should the DSC surface become contaminated, clean demineralized water will be flushed through the DSC/transfer cask annulus until surface smears show that the contamination levels meet Technical Specification limits.

Transfer cask external contamination will also be controlled to minimize personnel radiation exposure and potential off-site radiological releases during cask handling operations outside the spent fuel pool. 49CFR 173.443(d), which governs contamination levels for off-site shipment in a closed, exclusive use vehicle, will be used as a guideline for establishing cask release limits.

Containment of radioactive material associated with IFAs is provided by fuel cladding, the stainless steel DSC body, and double seal welded primary and secondary closures. These multiple confinement barriers assure that any accidental radioactive releases from stored IFAs to the environment will be ALARA.

3.3.2.2 Ventilation - Offgas

The design of the Oconee Site-Specific ISFSI limits the temperature history of stored fuel rods, such that no fuel damage will occur under design basis conditions. Decay heat dissipation is discussed in Section 1.2.3.2 of this UFSAR. ISFSI response to abnormal cooling conditions (i.e. convective air flow blockage conditions) is provided in Chapter 8, of this UFSAR. There are no radioactive effluent releases during normal operations. Additionally, there are no credible accidents which cause releases of radioactive effluent from the DSC. Therefore, there is no offgas system or radiological effluent release monitoring requirement for the ISFSI.

The only offgas concern results from the DSC purge and drying operations. During this operation, the gases purged from the DSC internals are directed to the spent fuel pool HVAC system upstream of the Engineered Safety Feature (ESF) HEPA, and carbon filter units. The purged air and helium are ultimately released from the station unit vent. Potential radiological effluent releases are monitored by both spent fuel storage facility HVAC and unit vent monitors prior to release. This is the same method and system currently utilized for spent fuel shipping cask operations.

3.3.3 Protection By Equipment and Instrumentation Selection

3.3.3.1 Equipment

The transfer cask and DSC are the only equipment considered safety related during normal and off-normal operations. The HSM is not safety related. However, the functions of the HSM are considered important to the safe operation of the Oconee Site-Specific ISFSI. Based on the function, design, and construction, the HSM is classified as a QA Condition 4 structure. The design criteria for all equipment comprising the ISFSI that is classified to be important to safety are summarized in Section 3.4 of this UFSAR. Design code standards for ISFSI components are summarized in Table A-14.

The design criteria for the NUHOMS® reinforced concrete HSM including its foundation and DSC support structure, the DSC and its internal basket assembly, and the transfer cask are provided in Section 3.2 and summarized in Tables 3.2-1 through 3.2-9 of Reference 1.

The Oconee transfer cask lifting yoke and lift extension used for movement of the transfer cask within the fuel building are designed and procured as components important to safety. The lifting yoke and lift extension used in that part of the operation are controlled by 10CFR Part 50 and NUREG-0612 and are designed to ANSI 14.6-1986 criteria for nonredundant yokes.

The vacuum drying system described in Section 4.7.3 of Reference 1 is not safety related. Failure of any part of this system can only result in a delay of operations, and not in a hazardous situation to the public or operating personnel. The welding materials required to make the closure welds on the DSC top end shield plug and top cover plate are purchased to the same ASME Code criteria as the DSC (Section NB Class 2). The actual equipment used for making the closure welds is purchased in accordance with standard industry codes such as ANSI, AWS and AISC.

As noted in Section 4.5.5 of this UFSAR, all other components of the NUHOMS® system, including the transfer cask skid, skid positioning system, and hydraulic ram system are required to perform their function to successfully transfer a DSC to and from the HSM. These systems are described in Reference 1 with the design requirements further delineated in Chapter 4, of this UFSAR. However, the failure of any of this equipment may cause additional operational effort but will not endanger the health and safety of the public or plant personnel. Therefore, these transfer components are not considered to be important to safety and are therefore designed, constructed, and tested in accordance with accepted industry standards.

In addition, the transfer cask, HSM, and DSC have been designed to meet very conservative design criteria including postulated conditions which envelop those which may result from the mechanistic failure of the transfer system equipment. Design conditions such as cask drop accident and jammed DSC have been included even though there is no plausible way for these worst case events to occur. Conservative bounding analyses for these conditions have been performed using minimum material yield strengths, strength reduction factors, and factors of safety in accordance with the stringent requirements of the ASME and ACI Codes. Even when applying these conservative criteria, considerable design margin for these components and structures remains as evidenced by the analysis results comparisons with acceptance criteria contained in Reference 1. Further, these components and structures are fabricated and constructed to the rigorous standards and methods of the ASME and ACI Codes under a 10CFR 50 Appendix B Quality Assurance Program. These include material qualification, welding and

nondestructive examination, and strict surveillance and quality control inspection. The resulting high integrity of the Transfer Cask, DSC, and HSM provide more than adequate assurance that the health and safety of the public and plant personnel are protected.

3.3.3.2 Instrumentation

The Oconee Site-Specific ISFSI is designed to maintain a safe and secure, long-term containment and storage environment for IFAs using only totally passive components. Therefore, no safety related instrumentation is required for operation of the facility.

3.3.4 Nuclear Criticality Safety

3.3.4.1 Control Methods for Prevention of Criticality

A combination of DSC fuel basket design and station administrative procedures assure subcritical conditions exist at all times. DSC fuel basket material properties and geometry are established to assure subcriticality assuming a full loading of IFAs with a specified minimum burnup that encompasses the majority of the available spent fuel inventory at Oconee. Oconee administrative procedures assure that only qualified IFAs are loaded for storage in a DSC and that a minimum soluble boron concentration of 1810 ppm is maintained within the DSC basket cavity during underwater loading/unloading operations. IFA qualification for storage in a DSC is determined based on initial enrichment, burnup history and post-irradiation cooling time as governed by Oconee Site-Specific ISFSI Technical Specifications.

IFA qualification requirements are provided in Section 10.2.5 on Administrative Controls. Using special nuclear materials control and accountability (SNMCA) records and the burnup results from the Oconee Operator Aided Computer, the specific data needed to characterize any given spent fuel assembly can be gathered. This includes the initial enrichment, discharge burnup, known cladding defects (if any), current storage location, and cumulative cooling time since reactor discharge. After verifying that all the spent fuel specifications of Reference 2 are met, documentation of individual fuel qualifications will be transmitted to fuel handling personnel. Oconee administrative procedures will require receipt of this qualification documentation, and independent verification of fuel assembly identification numbers prior to loading a given assembly into the DSC. In addition, all assembly serial numbers will be checked following the complete loading of 24 assemblies into the DSC.

The Oconee Site-Specific ISFSI Technical Specifications which govern IFA qualification for storage are given in Reference 2. The administrative procedures outlined above will be used to ensure that the requirements for fuel qualification are met.

IFA qualfication criteria do not include a specification on axial burnup distribution. The axial burnup profile used in analyzing the nonuniform axial burnup reactivity effects on fully loaded DSC spent fuel storage arrays is considered worst-case based on a comprehensive review of axial burnup profiles generated by the EPRI-NODE computer program (reference Section 3.3.4.3 of Reference 1). Although some individual IFAs may not be enveloped by the worst-case axial burnup profile considered, the conservative treatment of nonuniform axial burnup in the Reference 1 analysis and the averaging effects of loading up to 24 qualified IFAs per DSC provide adequate assurance that the K_{eff} of any loaded DSC configuration will not exceed the worst-case value presented in Reference 1.

To satisfy the requirements of 10CFR72.124, ANSI N16.1, and ANSI/ANS-57.9 (that at least two independent and unlikely changes in conditions must occur before criticality could be possible) the DSC is designed to remain subcritical for each of two independent and unlikely events-accidental deboration of the pool water during loading, and accidental loading of unqualified fuel assemblies. In the case of accidental deboration of the pool water, the qualifying burnup of the IFAs assures subcriticality. In the case of accidental loading of unqualified fuel assemblies, the dissolved boron concentration of 1810 ppm

assures subcriticality even with 24 fresh fuel assemblies with enrichment of 4.0 weight percent U-235 combined with the effect of water at the optimum density for reactivity.

3.3.4.2 Error Contingency Criteria

The design basis for preventing criticality in Oconee Site-Specific ISFSI spent fuel storage operations is taken from American National Standard Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, ANSI/ANS-57.2-1983. ANSI/ANS-57.2-1983 requires a demonstrated margin of subcriticality of ≥0.05 ΔK under all credible conditions except under certain extreme off-normal conditions where a ≥0.02 ΔK subcritical margin may be justified. Additionally, ANSI/ANS-57.2-1983 requires all uncertainties be included in the final calculated K_{eff} value at 95/95 tolerance limits. See Section 3.3.4 of Reference 1 for details on how these criteria are applied in demonstrating ISFSI criticality safety.

3.3.4.3 Verification Analysis

Two criticality analysis methods are used for the two types of storage. The SCALE-3 system of codes is the basis for Single Region Storage while the CASMO-3/SIMULATE-3 package is used for Mixed Region Storage.

3.3.4.3.1 Single Region Storage

The analysis method which ensures criticality safety for the Oconee Site-Specific ISFSI uses the Criticality Analysis Sequence No. 2 (CSAS2) and the 123GROUPGMTH master cross-section library included in the SCALE-3 system of codes (Reference 2). CSAS2 consists of two cross-section processing codes (NITAWL and BONAMI), a 1-D transport code for cell-weighting cross-section data (XSDRNPM), and a 3-D monte-carlo code (KENO-IV) for calculating the effective multiplication factor for a system.

In CSAS2 calculations involving the zero burnup intercept point, cross section processing and cell weighting of cross sections was performed assuming fresh fuel. For CSAS2 calculations involving irradiated fuel, cross section processing and cell weighting of cross sections was performed assuming irradiated fuel actinide and fission product isotopics.

Irradiated fuel fissile nuclide number density data was obtained from CASMO-2 (Reference 4) calculations and input to the CSAS2 criticality code sequence (reference Section 3.3.4.2 of Reference 1). The CASMO-2 lattice physics code has been used extensively in reactor physics calculations. Its ability to accurately predict fissile nuclide depletion and generation as well as neutron multiplication is well established in benchmark calculations (References 5 and 6) and through its successful application in numerous licensed reactor physics and core reload design calculations.

The Shielding Analysis Sequence No. 2 (SAS2) included in the SCALE-3 package of codes was used to develop irradiated fuel fission product number density data for input to CSAS2. SAS2 is an industry recognized code which employs ORIGEN-S to perform fuel burnup, depletion and decay calculations.

A set of 40 critical experiments have been analyzed using the CSAS2/123GROUPGMTH reactivity calculation method to demonstrate its applicability to criticality analysis and to establish method bias and variability. The experiments analyzed represent a diverse group of water moderated, heterogeneous oxide fuel arrays separated by various materials (stainless steel, Boral, water, etc.) that are representative of LWR shipping and storage conditions. The method bias and uncertainty applied in the calculation of the final K_{eff} result is based on CSAS2/123GROUPMTH calculated results for the set of 40 critical experiments summarized in Table 3.3-6 of Reference 1. All 40 critical experiments included in the method benchmark are similar to zero burnup/nominal case flooded DSC conditions in that all are water moderated, low enrichment heterogeneous U02 systems. Additional benchmark calculations were

performed to demonstrate that the irradiated fuel criticality/equivalence method used is conservative when compared to the method bias basis U02 benchmark results (i.e., Reference 1 Table 3.3-6 results). CSAS2/123GROUPGMTH benchmark results for systems containing Pu02-U02 mixed oxide fuel pins are provided in Table 3.3-7 of Reference 1. Benchmark data representative of irradiated fuel assemblies was obtained from the results of CASMO-2 infinite lattice criticality calculations; the results of benchmark comparisons between CASMO-2 and CSAS2/CASMO2/SAS2 calculated K_{inf} values are provided in Table 3.3-8 of Reference 1. Inspection of the benchmark results provided in Reference 1 Table 3.3-7 and 3.3-8 demonstrates that the criticality/equivalence method used conservatively overpredicts K_{eff} for systems containing plutonium or irradiated fuel of the type proposed for Oconee ISFSI storage.

Further details on the analysis method and the ISFSI verification analysis are provided in Section 3.3.4 of Reference 1.

3.3.4.3.2 Mixed Region Storage

The criticality analysis performed for the mixed region canister uses the CASMO-3/TABLES-3/SIMULATE-3 code system. CASMO-3 is a multigroup two-dimensional transport theory code for burnup calculations on LWR fuel. The 70-group CASMO-3 cross section library is used for the mixed region criticality analysis. TABLES-3 is a linking code which formats data from CASMO-3 for use in SIMULATE-3. SIMULATE-3 is an advanced two-group nodal code utilizing the QPANDA neutronics model.

Fuel assemblies are first depleted using CASMO-3 modeling normal reactor operating conditions. Restarts are then performed using CASMO-3 to determine the reactivity of the depleted assemblies in the dry storage canister geometry. Since CASMO-3 is a lattice code, its calculations are for single assemblies in an infinite array, which is not representative of the canister geometry. Therefore, CASMO-3 cases are also run to model the exterior of the canister as a reflector region. TABLES-3 is used to manipulate the CASMO-3 assembly and reflector data into a library to allow the nodal code to analyze a wide range of assembly types in the canister. SIMULATE-3 is then executed to calculate k_{eff} of the canister with a variety of different fuel configurations within the canister.

The criticality analysis performed for the mixed region canister uses the reactivity equivalencing technique to ensure sufficient criticality margin. The reactivity of a fully loaded single region canister at the minimum burnup requirements is calculated using SIMULATE-3. This represents a safe and acceptable reactivity level for the canister. Hence, any arrangement of fuel assemblies in the canister whose reactivity is less than or equal to that of the single region canister is also acceptable. The reactivity of the mixed region canister is adjusted by independently varying the burnup and enrichment of both regions until the reactivity of the mixed region canister is equivalent to the single region canister. Several sets of restricted and filler fuel burnup and enrichment requirements are defined by this process. A single set of burnup and enrichment curves for the restricted and filler fuel is selected based on the inventory of fuel in the Oconee spent fuel pools.

3.3.5 Radiological Protection

The Oconee Site-Specific ISFSI is designed to maintain onsite and offsite doses ALARA during loading operations and long-term storage conditions. ISFSI loading procedures, shielding design, and access controls provide the necessary radiological protection to assure radiological exposures to station personnel and the public is maintained ALARA. Further details on collective onsite and offsite doses resulting from ISFSI operations and the ISFSI ALARA evaluation are provided in Chapter 7, of this UFSAR.

Access to the spent fuel assemblies stored in the ISFSI is restricted by a radiological controlled area fence inside the Oconee protected area, and the thick walls and heavy door of the Horizontal Storage Module. Since there are no active systems in the storage module, there is no need for continuous monitoring of conditions. Appropriate monitoring is performed prior to loading or unloading Dry Storage Canisters inside the ISFSI fence. Appropriate monitors are in place inside the station to provide warning of radiation hazards while DSC loading and cask handling operations are performed in the fuel building and loading area. During transport, the transfer cask will be monitored to assure no danger to the health of the public or station personnel.

3.3.6 Fire and Explosion Protection

The Oconee Site-Specific ISFSI HSMs and DSCs contain no flammable material and the concrete and steel used for their fabrication can withstand any credible fire hazard. There is no fixed fire suppression system within the boundaries of the ISFSI; however, portable suppression equipment is provided within the fenced boundary. Also, the facility is located such that the station fire brigade can respond to any fire emergency using portable fire suppression equipment or the site's Fire Protection System, as described in Section 9.5.1 of the Oconee UFSAR.

ISFSI initiated explosions are not considered credible since no explosive materials are present in the fission product or cover gases. Externally initiated explosions are considered to be bounded by the design basis tornado generated missile load analysis presented in Section 3.2 of this UFSAR and Reference 1.

3.3.7 Materials Handling and Storage

Materials handling and storage at the Oconee Site-Specific ISFSI includes irradiated fuel and radioactive waste handling and storage. No hazardous chemicals or chemical reactions are involved in the normal ISFSI loading and storage processes.

All irradiated fuel handling outside the fuel storage pool is performed with the fuel assemblies enclosed in a DSC. DSC handling equipment and handling procedures are described in detail in Chapter 4 and Chapter 5 of this UFSAR, respectively.

Radioactive waste generation, treatment and disposal are addressed in Chapter 6 of this UFSAR.

The design criteria for handling spent fuel outside the pool area is to keep the fuel enclosed in the DSC and the Transfer Cask or HSM at all times. There is no waste generation outside the fuel building for normal DSC transfer operations. Waste generated in loading and decontaminating the cask is handled by existing Oconee waste systems in the pool and decontamination areas.

The DSC/transfer cask design is such that fuel handling in the pool area is consistent with routine fuel handling processes. Specific criteria for selecting fuel assemblies for storage are detailed in Reference 2. IFAs are selected to meet design criticality, cooling and radiation protection parameters. Once the assemblies are loaded into the DSC, there is no individual assembly handling. Thus, the only fuel handling procedures are those already in existence for the pool and the administrative criteria for selecting assemblies for storage. Damaged fuel assemblies are not normally considered for storage and would be handled according to existing pool procedures in the event damage occurred during DSC loading or unloading in the pool. (Fuel damage in the context of this discussion represents gross clad or structural failure and does not include pin-hole clad leaks.) Fuel handling operations will be monitored with existing pool area monitors.

All radioactive waste generation is from cask decontamination and consists of liquid waste which is input into the cask decontamination pit drain and thus into the existing plant liquid radwaste system and solid waste which is collected for disposal via the existing plant radwaste facility.

3.4 Summary of Storage System Design Criteria

1. REFERENCE SPENT FUEL CHARACTERISTICS -
   a. 15x15 PWR Assemblies (24 Per Module/DSC)
   b. Decay Heat = .66 KW Per Assembly
   c. Nominal Burnup = 40,000 GWD/MTU
   d. Initial Enrichment = 4.0 weight % U-235
   e. Equivalent Zero Burnup Enrichment ≤1.45 weight % U-235 (criticality)
2. COMPONENT FUNCTIONS
   a. DSC Provides Fuel Support, End Shielding, Heat Transfer, Criticality Control, and Confinement of cover gas and Radioactive Material.
   b. Transfer Cask Provides Shielding, DSC Loading, Handling and Transfer Mechanism, HSM Docking Functions, and Tornado Wind and Missile Protection.
   c. HSM Provides Shielding, Passive Decay Heat Removal, Structural/Seismic DSC Support and Environmental Protection, including Tornado Wind and Missile Protection.
   d. Hydraulic Ram System Provides Mechanism for DSC Transfer From Transfer Cask to HSM and eventual removal of DSC from HSM.
3. ENVIRONMENTAL CONDITIONS
   a. Maximum Tornado:
      1) wind speed = 360 miles per hour
      2) rotational speed = 290 miles per hour
      3) translational speed = 70 miles per hour
      4) pressure drop across the tornado = 3.0 psi
      5) rate of pressure drop = 2.0 psi per second
   b. Tornado Missiles @ 35% of the Combined translational and rotational DBT velocity = 126 miles per hour.
      1) 3967 pound automobile with a 20 square foot frontal area
      2) 276 pound, eight inch diameter blunt-nosed armor piercing artillery shell
      3) one-inch diameter solid steel sphere
   c. Flood Design: Not Applicable
   d. Seismic Forces = .17g Vertical, .25g Horizontal (NUHOMS® components)
      = .15g Vertical, .15g Horizontal (Oconee site conditions)
   e. Snow Ice Loads = 200 Pounds Per Square Foot
4. SAFETY PROTECTION
   a. Normal Operating Clad Temperature ≤ 340°C
   b. Material Confinement - Multiple Barrier Concept

c. Purged-gasses - Passed Through Spent Fuel Pool Vertilation System During Fuel - Loading Otherwise Not Applicable.

d. Criticality Control Through Burnup Credit and 1810 ppm Soluble Boron Credit - $K_{eff} < 0.95$, $K_{eff} < 0.98$ (off-normal)

3.5 References

1. Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel (NUHOMS®-24P), Rev. 1A, dated July, 1989, NUH-002.

2. "SCALE-3: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation," NUREG/CR-0200, ORNL, Revision 3, December 1984.

3. Oconee Nuclear Station Updated Final Safety Analysis Report (UFSAR).

4. "Structural Analysis of Shipping Cask, Effects of Jacket Physical Properties and Curvature and Puncture Resistance", H. A. Nelms, Vol. 3, ORNL TRM-1312, Oak Ridge National Laboratory, Oak Ridge Tennessee, June, 1968.

5. "Design of Structures for Missile Impact," R. B. Linderman, J. V. Rotz, and G. C. K. Yeh, Topical Report BC-TOF-9-A, Bechtel Power Corporation, Revision 2, September 1974.

6. "Formulas for Stress and Strain," R. J. Roark and W. C. Young, Fifth Edition, McGraw-Hill, New York, New York, 1975.

7. "CASMO-2 A Fuel Assembly Burnup Program," Malte Edenius, et. al., STUDSVIK/NR-81/3, March 1981.

8. "CASMO Benchmark Report," M. Edenius, et. al., RF-78/6293, STUDSVIK, March 1978.

9. "CASMO Benchmarking Against Experiments in Rack Geometries," M. Edenius, et. al., NR-81/61 STUDSVIK, November 1981.

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4.0 Storage System

The Oconee Site-Specific ISFSI is located within the Protected Area on the Oconee Nuclear Station site. The storage area is located west of the existing intake structure. Figure B-29 depicts the site layout in relation to other plant features and defines the onsite route that the transfer cask and trailer will travel in moving loaded storage canisters from the Fuel Buildings to the ISFSI.

The Oconee ISFSI utilizes the NUHOMS®-24P storage system which provides for the horizontal, dry storage of irradiated nuclear fuel assemblies. The fuel assemblies are contained in a DSC made of stainless steel which is placed inside a reinforced concrete HSM for long term storage.

In addition to the DSC and HSM, the NUHOMS®-24P system utilizes handling and transfer equipment to load the DSC with fuel, to seal the DSC, to move the loaded DSC inside a heavily shielded transfer cask from the Fuel Building to the HSM and to insert the DSC into the HSM. The DSC is designed to hold 24 PWR fuel assemblies. The margins of safety in the structural design of the HSM, DSC, and transfer cask are more fully described in Section 8, Tables 8.1 and 8.2 of Reference 1. Additional information for the handling and transfer equipment is presented in Section 4.5.5.

The fuel assemblies to be stored are described in Section 3.1.1. The dose to the general public from the operation of the ISFSI is far below the allowable dose limits as set by regulation. Estimates of the annual dose rates are provided in Section 7.7.

It should be noted that the Oconee ISFSI is licensed for the storage of as many as 2112 assemblies; this storage capacity will be added incrementally as needed to support the actual refueling schedules. HSMs and foundation have been designed to be built in any array size no smaller than 2x3 (three modules side by side and back to back with three additional modules) and no larger than 2x10 (Ten modules side by side and back to back with ten additional modules).

The ISFSI system is designed to interface with existing plant equipment and systems. Roadways, buried pipes, trenches, and positioning aprons were verified to be acceptable for the wheel loadings of the transfer equipment. Oconee Nuclear Station asphalt roadways were verified as meeting the design minimum thickness requirements of the American Association of State Highway and Transportation Officials as specified for loading comparable to the ISFSI transfer equipment. Approximately 64 buried pipes and over 26 drain lines were analyzed and verified acceptable according to the applicable codes for each piping material. All interfacing trench systems were analyzed for transfer vehicle loadings. These include the 115 KV, Interim Radwaste, and the Standby Shutdown Facility trenches.

The size and weight of the transfer cask, DSC, and transfer cask lifting yoke/lift extension are acceptable within the current design limits of the crane, cask handling area, and transfer cask positioning aprons of the spent fuel pools. Design features employed to withstand environmental and accident forces are detailed in Chapter 3 and Chapter 8 of this UFSAR. The DSC and transfer cask are important to safety and are designed, constructed, and tested per Duke's QA 1 program that is more fully described in Chapter 11. The HSM is designed, constructed and inspected in accordance with Duke's QA Condition 2 program. Based on the function, design, and construction, the HSM is classified as a QA Condition 4 structure.

The HSM is designed in accordance with the requirements of ACI 349-85 as discussed in Section 3.2.5.1 of Reference 1. The HSM is constructed following the guidelines of ACI 318-83 as discussed in Section 4.2.1 of Reference 1. The DSC and transfer cask are designed and built in accordance with the ASME Code, 1983 edition through Winter 1985 addenda. In addition this equipment complies with the following: ANSI N 14.6-1978, American National Standard for Special Lifting Device for Shipping Containers Weighting 10,000 lbs. or More for Nuclear Materials; ANSI/ANS 57.9-1984, Design Criteria for An Independent Spent Fuel Storage Installation (Dry Storage Type); ASTM E499-73, Standard Methods of Testing for Leaks Using the Mass Spectrometer Leak Detector in the Detector Probe Mode.

4.1 Location and Layout

The location and layout of the Oconee Site-Specific ISFSI with respect to other site structures is shown in Figure B-29. This figure also denotes the transportation route for movement of the DSCs from the spent fuel pool to the HSMs.

If, during the transfer of a DSC from the fuel building to a HSM, an event requiring return to the fuel handling building occured, until the point where it reaches the ramp leading up to the ISFSI, the tow vehicle and trailer has sufficient space to turn around as needed and can return using the approved path.

Once it is on the access ramp leading to the ISFSI, the tow vehicle and trailer would have to continue to the ISFSI site in order to turn around.

The transport route has been reviewed and found to be within the design basis of the cask drop analysis discussed in Section 8.2 of Reference 1. The potential causes for cask and DSC drop accidents are described in Section 8.2.5.1 of Reference 1. The enveloping postulated drop events assumed for design are:

1. A horizontal side drop or slap down from a height of 80 inches.
2. A vertical end drop from a height of 80 inches onto the top or bottom of the transfer cask.
3. A corner drop from a height of 80 inches at an angle of 30° to the horizontal, onto the top or bottom corner of the transfer cask.

These drop scenarios were used to define an equivalent static deceleration load of 75g for cases (a) and (b) and 25g for the corner drop (case (c)). As described in Reference 1, these deceleration values were developed for assumed surface conditions which will envelop all Oconee site conditions which may be encountered. Specifically, these decelerations are based on the work contained in EPRI report NP-4830 and are applicable to impacted surfaces with target hardness numbers up to 400,000. The maximum target hardness along the Oconee transfer route is 2750 for an edge drop scenario.

The transfer cask route from the fuel buildings to the HSM was evaluated to ensure that the maximum transfer cask drop height of 80 inches is not exceeded. Therefore, since the Oconee target hardness and maximum potential cask drop height are less than the values presented in Reference 1, the deceleration values presented in Reference 1 envelop all Oconee site conditions.

The site area will be sloped appropriately to permit surface drainage to collection ditches for channeling the water away from the site. As noted in Section 2.4, the site is 11.9 feet above the probable maximum flood elevation. Local intense rainfall is not a problem since the inlet air opening is 24 in. above yard grade. There is a small drainage pipe passing through the HSM front wall into the plenum area. The base slab of the plenum area is sloped towards this drainage pipe. Additionally, the concrete approach apron is sloped away from the HSM front wall. During a local intense rain, it is remotely possible that some rainwater may backup into the HSM plenum area temporarily, but this water will drain out of the HSM soon after the intense rain subsides. Therefore, due to surface drainage, rain water will not collect to a depth of any concern.

4.2 Storage Site

The design bases covering the analysis and design procedures for the appropriate loadings are specified in Chapter 3 of this report and in Reference 1 for the HSM, DSC, transfer cask and transfer trailer. The foundation design includes allowances for seismic loads. The ground accelerations are from the site ground motions specified in Chapter 2. Liquefaction potential for the ISFSI site is discussed in Section 2.5. Based on the soil investigations and using an equivalent static methodology to account for dynamic effects, spring stiffeners are determined representing the force-deflection relationship of the underlying soil. This information is utilized as input to the structural model described in Reference 1 to determine

settlement effects. See Section 2.5.4 for details of the foundation analysis. Temporary loadings from the extreme environmental cases (Chapter 3) and accident conditions (Chapter 8) have been reviewed and are acceptable.

4.2.1 Structures

The HSM design bases, materials of construction, codes and standards, etc. are fully described in Reference 1. The HSM foundation requirements are discussed in Section 2.5.5. The concrete approach aprons will not be attached to the HSM but will be separated by an expansion joint. Differential settlement between the slab and the HSM is not anticipated to be a problem.

The approach aprons are sized for bearing pressures using the same allowable and ultimate pressures as used for the HSM as discussed in Section 2.5.5. Settlement of the approach aprons will be minimal since they are normally unloaded. In addition, the transfer trailer has jacks used in vertically positioning the cask for DSC insertion into the HSM. The trailer leveling procedure will compensate for any differential settlement that may occur between the HSM and the concrete approach aprons. Outlying areas are concrete or asphalt to provide the space required for transfer trailer maneuvers.

4.2.2 Storage Site Layout

Figure B-30 depicts the Oconee Site-Specific layout and its functional features.

4.2.3 HSM Description

The HSM is constructed of reinforced concrete and structural steel. The HSMs are placed in service on a load bearing foundation which is within a fenced, controlled access area.

The HSM provides the structural support for the DSC as well as protection against tornado missiles plus neutron and gamma shielding. The exterior walls form an array of modules and the front and roof of the modules are sufficiently thick to provide average surface doses that are below 20 mr/hr.

The HSM provides fuel cooling by a combination of radiation, conduction and convection. Natural circulation air flow enters at the bottom of the HSM and passes around the DSC and exits through the flow channels in the top shield slab.

The design of the HSM system includes consideration of both normal and off-normal operating conditions including a range of credible and hypothetical accidents. The HSM design and analysis were performed in accordance with Chapter 3 and Chapter 8 of this UFSAR and Reference 1.

The design criteria for the operational and accident conditions fall into three main categories; structural, nuclear and thermal-hydraulic. Reference 1 describes in detail the analysis of these accidents. The bounding structural loading combinations include thermal, earthquake, tornado and cask drop accidents. For the nuclear analyses, shielding of the DSC end shield plugs, the HSM walls and penetrations, and the criticality analyses were primary considerations. The thermal-hydraulic criteria assures adequate air flow inside the module, acceptable air and concrete temperatures as well as DSC and fuel clad temperatures.

4.2.4 Instrumentation System Description

The Oconee Site-Specific ISFSI is designed to maintain a safe and secure, long-term containment and storage environment for IFAs using only totally passive components. Therefore, no safety related instrumentation is required for operation of the facility.

Instrumentation is necessary to perform the DSC/transfer cask draining purging, and drying operations. This instrumentation consists of commercial grade pressure gauges. The functions served by pressure instrumentation in the DSC loading procedure are discussed in Chapter 5 of this UFSAR.

Radiation monitoring is provided by existing station area, and process effluent monitors. Oconee Site-Specific ISFSI radiation monitoring is provided by environmental TLDs.

4.3 Transfer System

4.3.1 Function

The function of the transfer system is to transfer the DSC containing irradiated fuel assemblies to and from the HSM.

4.3.2 Components

The transport system consists of the transfer cask, DSC, transfer cask skid, transfer trailer, skid positioning system and hydraulic ram.

4.3.2.1 Transfer Cask

The transfer cask is used to transfer the loaded DSC to and from the HSM. The cask provides shielding along the axial length of the fuel during transfer, loading and retrieval operations. A description of the transfer cask is provided in Reference 1. For Oconee, two (2) hard surfaced rails were added to the enhance cask sliding characteristics and the liquid neutron shield described in Reference 1, has been replaced by a solid neutron shield comprised of Bisco NS-3 which is a cementatious material cast in place in the neutron shield jacket. The drain and fill ports, as well as the expansion tank, which are needed for the liquid neutron shield have been deleted. To ensure that off-gassing or vapor expansion will not result in over-pressurization of the bottom neutron shield and the exterior neutron shielding jacket, pressure relief valves set at 20 and 50 psig respectively have been added. This change results in a more passive neutron shield in that operational and maintenance considerations are reduced. Also, the possibility of a complete loss of neutron shielding as a result of an accident is eliminated, although it is still assumed that substantial degradation may occur in some localized area.

This substitution satisfies the requirements of 10CFR 72 because:

1. The surface dose rates still satisfy the requirements established in Reference 1.
2. The temperature of the fuel cladding does not exceed the criteria established in Reference 1.
3. The material characteristics are suitable for the service environment.
4. The consequences of postulated accidents are enveloped by the criteria established in Reference 1.
5. The structural integrity of the transfer cask and DSC is not compromised.

4.3.2.2 Dry Storage Canister (DSC)

The DSC provides the primary confinement for up to 24 irradiated fuel assemblies. The DSC provides shielding at the ends and also maintains the fuel array subcritical under the worst case conditions. The DSC fits inside the transfer cask for safe movement from the spent fuel pool to the ISFSI site.

4.3.2.3 Transfer Cask Skid

The purpose of the transfer cask skid is to provide a support base for rotating the transfer cask to a horizontal position and to maintain the transfer cask in the properly aligned position during transport, loading and retrieval operations.

The basic dimensions and layout for the transfer cask skid are presented in Section 1.3.1.5 and Figure 1.3-4 of Reference 1.

Deleted paragraph(s) per 2007 Update

The skid is bolted to the transfer cask trailer during transfer operations.

The transfer cask skid is a non-safety related item which is designed in accordance with the requirements of the AISC code, eighth edition using linear elastic analytical methods and normal allowables for the bounding design basis loading. The design loads for the transfer skid and attachments are the same as the transfer cask trunnion loads presented in Section C.1-4 and Figure C.1-2 of Appendix C of Reference 1.

The design basis loads for the transfer cask skid were conservatively established to envelope all applied loads including downending of the cask, rotational loads, and transport loads during transfer to the ISFSI site. The transfer skid design loads envelope the postulated off-normal and accident loads discussed in Section 8.2 of Reference 1 such as earthquake and tornado wind loads. Along with the basic Code allowable stresses used in the design analysis, this conservative design basis assures that the skid will adequately support the NUHOMS®-24P transfer cask for all postulated events.

4.3.2.4 Transfer Trailer

The transfer trailer has a capacity of 120 tons. The transfer trailer is designed to ride as low to the ground as possible to minimize the HSM height. Four hydraulic jacks are incorporated into the trailer design to provide vertical movement for alignment of the transfer cask with the HSM. The trailer is pulled to the ISFSI by an appropriate tow vehicle.

The trailer is configured as a 4x2 mechanically or hydraulically steered dolly. Eight hydraulic suspensions carry four pneumatic tires each and are located two wide, in four axle lines. There are a total of 32 tires.

The steering mechanism connects the individual axles such that they have a common turning radius, thus minimizing tire scuffing and the resulting damage to pavement and tires. The suspensions allow other advantages such as adjustable deck height, lockout or repair of failed suspensions or tires, and compensation for road surface irregularities.

The trailer is pulled by an appropriate tow vehicle via a drawbar unit. The drawbar unit also provides motive force for steering of the trailer.

Additional features and accessories for the trailer include: diesel power pack and hydraulic control valves, hand held remote control unit, all-wheel braking, and provisions for mounting four bearing pads, hydraulic alignment system hardware, and four hydraulic lifting jacks to the trailer frame.

The trailer is a commercial grade item of the type commonly used to move heavy loads, such as the space shuttle. The design parameters for a typical trailer are provided in Table A-15. It is constructed in accordance with the manufacturer's standard QA program.

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4.3.2.5 Skid Positioning System

The Skid Positioning System (SPS) includes the following items which either actively or passively position the skid during storage, transport, and alignment operations: low friction bearing plates, skid tie down bolting, hydraulic lifting jacks, hydraulic x-y-theta positioning cylinders, and all associated instrumentation and controls. Controls for the SPS are located on the front of the trailer and on a remote control pendant.

The loaded cask is supported by a steel skid structure. The skid's weight is supported by a set of four low friction bearing plates. The bearing material offers a coefficient of friction of 5% or less with negligible

breakaway friction. The skid is restrained from lateral motion during transport and storage by a set of tie down bolts attached to the trailer frame.

Four support plates for hydraulic lifting jacks are located on the trailer frame. Although the trailer's hydraulic suspensions could be used to perform trailer deck height adjustments, the jacks will more firmly support the load than pneumatic tires. The jacks provide elevation adjustment, plus adjustment of pitch and roll of the trailer frame relative to the concrete HSM pad. The jacks are also used in the fuel building during cask loading. There, the front pair of jacks carries most of the load during the cask setdown and downending.

A system of hydraulic actuators are oriented in the transverse and longitudinal directions on the trailer deck. These cylinders are used to align the cask correctly relative to the HSM after the deck is leveled at the appropriate height.

4.3.2.6 Hydraulic Ram System (HRS) Description

Reference 1 includes a system description of the hydraulic ram in Section 1.3.1.6, a system operation description of loading and retrieval of the DSC in Section 1.3.1.7 and a functional description in Section 5.2.1.1. Figure 1.3-5 shows a typical design for the hydraulic ram system. Figure 1.3-6 shows the primary operations for the NUHOMS® system.

The operations system for loading and unloading of the DSC into and out of the HSM is discussed in Sections 5.1.1.6 and 5.1.1.8 of Reference 1. Figure 5.1-4 of the same reference shows the NUHOMS® System retrieval operations flow chart. Safety features of the ram system are presented in Section 5.2.1.2 of Reference 1.

The HRS consists of the following main components: a double-acting hydraulic cylinder (ram); a ram support frame assembly for support and alignment of the ram hydraulic cylinder (integral to the cask skid); one grapple assembly; one hydraulic power unit with controls: and hydraulic tubing, hoses, hose reel and accessories as required for system operation.

4.3.3 Design Bases and Safety Assurance

4.3.3.1 Transfer Cask

The design bases of the transfer cask are given in Section 1.2.2 of Reference 1. These are based primarily on radiological and structural considerations.

As discussed in Section 4.3.2.1, the solid neutron shield will be permanently filled with Bisco NS-3 - a neutron absorbing cementatious material cast in place in the neutron shield jacket. Pressure relief valves are designed to relieve pressure in the event that any off-gassing were to create excessive internal pressure.

4.3.3.2 Transfer Cask Skid

The transfer cask skid supports the transfer cask in a horizontal position on the trailer deck during the on site road transportation to the ISFSI site. The transfer cask skid is designed to support a transfer cask weighing 110 tons and to allow rotation of the transfer cask between the horizontal and vertical positions. The transfer cask skid is secured to the transfer trailer during movement and is restrained by a securing system to resist the peak loads anticipated under normal conditions of transport between the fuel buildings and the ISFSI.

4.3.3.3 Transfer Trailer

The design parameters for the transfer trailer are presented in Table A-15. Also, as shown in Section 8.2.5 of Reference 1 the design basis drop height for the NUHOMS®-24P transfer cask is 80 inches. This analysis bounds the Oconee transport conditions.

4.3.3.4 Skid Positioning System (SPS)

The SPS is designed to compensate for the following variance in true alignment between the cask and HSM, in any combination.

Pure Vertical Translation	3"
Pure Sideways Translation	3"
Pitch	1/4" / ft
Yaw	3 degrees

In addition to the above corrections, the SPS must move the cask and skid from the transport position, in which the payload's center of gravity lies directly over the centroid of the trailer, to a position where the cask slightly overhangs the rear of the trailer. The required actuator strokes to achieve the design basis compensations are (in terms of pure directional motion) approximately:

Vertical Travel	6"
Transverse Travel	10"
Longitudinal Travel	39"

The SPS components which restrain the cask and skid during cask setdown and transport are designed to withstand the loads described for the cask trunnions in Appendix C.1 of Reference 1. The design basis weights for use in sizing SPS actuators and hardware are, in U.S. tons:

Empty Cask	56 tons
Loaded DSC	38 tons
Skid	6 tons
Trailer	20 tons

The SPS is non-QA and is designed and built to the standard industrial requirements.

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The SPS is designed with several safety features to avoid unnecessary delays in the transfer process. The trailer lifting jacks have mechanical locking collars which preclude settling of the trailer deck, due to loss of hydraulic pressure to the jack cylinders. Operation of the trailer jacks, transverse, and longitudinal cylinders are mutually exclusive; it is impossible to operate more than one sub-system at a time. During alignment and fuel transfer, the skid tie down brackets are unbolted, and the x-y travel of the skid is limited by both the hydraulic cylinder travel and by mechanical restraints on the low friction bearing travel. A tie down between the HSM and cask provides additional restraint during fuel transfer.

4.3.3.5 Hydraulic Ram

HYDRAULIC RAM SYSTEM (HRS) PERFORMANCE REQUIREMENTS:

1. Ram Force - 20,000 lb., Push and pull (Nominal)

80,000 lb., Push and Pull (Maximum)

2. Ram Piston Speed - 36 in/min (max)
3. Stroke - Approximately 21 feet

CODES AND STANDARDS:

The HRS components are not safety related and are designed to conform to standard industrial requirements.

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DESIGN LOAD CONDITIONS:

All system load bearing components of the HRS are designed to withstand the stresses associated with a maximum column load of 80,000 pounds at full extension centered 4.5 inches above the longitudinal axis of the ram cylinder. The system load bearing components include the ram hydraulic cylinder, grapple assembly and ram support frame assembly.

The trailer-mounted tripod support for the ram hydraulic cylinder is designed per American Institute of Steel Construction (AISC) 8th Ed. Manual of Steel Construction Standards.

ENVIRONMENTAL CONDITIONS:

The HRS is designed to withstand the following environmental conditions:

1. Ambient Storage Temperature Range: -30°F to 116°F
2. Ambient Humidity Range: 10 to 100% (coincident with outdoor temperature range)
3. Radiation Dose Rate (Section 10.3.4.1 of Reference 1): 250 mrem/hr
4. Ram force is limited to 20,000 pounds during the extension and retraction strokes for normal operation.
5. Ram force is limited to no more than 80,000 pounds under all conditions.
6. Ram extension and retraction speeds are variable from 0 to 36 inches per minutes.

INSTRUMENTATION AND CONTROLS:

The HRS is designed to allow the operator to extend and retract the ram hydraulic cylinder using hand-operated devices. The control system includes safety features to prevent the inadvertent operation of the HRS and to regulate speeds and forces of the ram to within the design criteria limitations.

COMPONENT DESCRIPTIONS:

Ram Hydraulic Cylinder - The ram hydraulic cylinder is a three stage, double-acting, horizontal design. The maximum extension or retraction force is 80,000 pounds at the maximum extension and retraction fluid pressures. The maximum piston speed for extension or retraction is 36 inches per minute (ipm). The cylinder is mounted to the transfer trailer.

Ram Hydraulic Pump - The hydraulic pump is a positive displacement type pump.

Grapple Cylinder and Pump - The grapple is operated with a small, double acting hydraulic cylinder. The cylinder is operated from the Hydraulic Power Unit (HPU).

Reservoir - The HRS includes a 150 gallon reservoir sized to meet the system capacity requirements.

Instrumentation and Controls - The control system is designed to allow the operator to extend or retract the ram hydraulic cylinder using manually actuated pressure and flow control devices.

<u>Grapple Assembly</u> - The grapple assembly is depicted by Revision 1 of the NUHOMS®-24P Topical Report Figure 1.3-5.

The power for the hydraulic system is provided from a retail 24 KV Oconee support line through a 75 KVA, 3 phase transformer.

Although this system does not have a backup power source, the retail power provided is considered very reliable. However, in the event of a power failure - whether momentary or extended - all efforts to transfer the DSC into the HSM will be halted until power is restored. In the interim period, the hydraulic system will be secured in the "off" position and all personnel will leave the immediate area of the cask. At any point in the transfer process, the HSM, the transfer cask, or a combination of both will provide sufficient shielding to maintain dose rates at acceptable levels during such a loss of power.

4.3.3.6 Other Equipment

All equipment other than the HSM, DSC, and transfer cask used in the transfer operations is classified as non-safety related. The failure of any non-safety related piece of transfer equipment described in Section 1.3 of Reference 1 will not increase human radiation exposures by any significant amount. As described, the transfer trailer has 32 wheels. The route from the fuel building to the ISFSI site is approximately 1/2 mile. The trailer is carefully inspected prior to use, and the probability of a breakdown is small. In the event of a component failure, the trailer can be configured to overcome failure of a wheel or suspension unit and off-loading can be completed prior to repairs. A failure in the system hydraulics could be repaired or the trailer pulled to the HSM site and the DSC off-loaded. Because of this design simplicity, failure of the hydraulic ram will be limited to the hydraulic control system.

4.3.3.7 Qualification of Components

Qualification of the hydraulic ram system (HRS) and skid positioning system (SPS) was done per standard administrative procedures and check out testing for operation of non-safety related equipment. The qualification tests consisted of pre-operational system checkout tests. All phases of the HRS and SPS operation were tested to verify the operability of the system. Normal operation and off-normal events and the respective recovery procedures were confirmed. All system performance criteria were verified to be met.

The HRS and SPS have simple, reliable designs which require minimal maintenance on active components and negligible maintenance on passive components. Primary maintenance requirements consist of periodic inspections of the hydraulic power units, ram hydraulic cylinder, grapple assembly, SPS actuator assemblies, and manual controls. In addition, the hydraulic fluid requires periodic testing to ensure that no water, dirt or other deleterious materials have contaminated the system.

4.3.3.8 Maintenance of HRS and SPS

Maintenance requirements for the HRS and SPS are minimized by corrosion protection provided by component design. All components are manufactured from corrosion resistant materials, or coated with corrosion resistant paints, and/or stored and operated with a grease or oil surface protectant. All controls and instrumentation which are subject to corrosion are housed in a weatherproof enclosure. The ram hydraulic cylinder is stored in its retracted position, filled with hydraulic fluid.

Operating procedures, maintenance procedures and storage procedures will ensure that all HRS and SPS components are kept in operable condition throughout the system design life.

4.4 Operating Systems

4.4.1 Loading and Unloading System

Loading and unloading of IFAs from the DSC and transfer cask requires use of the following equipment:

1. 100 ton spent fuel cask handling crane
2. spent fuel pool manipulator crane auxiliary hoist
3. spent fuel handling tool
4. transfer cask lifting yoke
5. DSC lifting rig
6. transfer cask lift extension
7. SFP cask platform
8. cask pit support stand

4.4.1.1 Preparation for Fuel Loading

Following receipt inspection and acceptance, a DSC is placed in the transfer cask. The orientation of the DSC in the cask is controlled by permanent alignment marks on each DSC and the transfer cask. The DSC is filled with borated water with a minimum concentration of 1810 ppm boron. The transfer cask is then positioned in the decontamination pit. The DSC/transfer cask annulus is filled with demineralized water and sealed with an inflatable seal. The transfer cask is then placed on the SFP cask platform in the spent fuel pool.

The following components are used for this operation:

1. 100 Ton Crane - the 100 ton spent fuel cask handling crane is used to place the DSC into the transfer cask and to move the DSC/transfer cask to the spent fuel pool. The 100 ton crane is described in Section 9.1.4.2.2 of the Oconee UFSAR.
2. DSC Lift Rig - The DSC lift rig is bolted to two of the four lifting lugs attached to the support ring for the top shield plug inside the top of the DSC. It is used for upending the DSC prior to loading into the transfer cask.
3. Transfer Cask Lifting Yoke - The transfer cask lifting yoke adapts the transfer cask to the 100 ton crane hook. It is used during upending and transport of the transfer cask within the fuel building. The transfer cask lifting yoke is designed, built, and maintained in accordance with the criteria of ANSI N14.6. The lifting yoke is a passive, open hook design with two parallel lifting beams. It is fabricated from high strength carbon steel plate and is protected by a decomtaminable coating. Figure B-31 depicts the transfer cask lifting yoke.
4. Transfer Cask Lift Extension - After the DSC/transfer cask is placed on the cask pit platform, the transfer cask lift extension is attached between the 100 ton crane hook and the transfer/cask lifting yoke. The transfer cask lift extension is designed to prevent wetting the 100 ton crane hook and block when the DSC/transfer cask is lowered from the SFP cask platform into the cask pit. The transfer cask lift extension is designed, built, and maintained in accordance with the criteria of ANSI N14.6. Like the lifting yoke, it is fabricated from high strength carbon steel plate and is protected by a decontaminable coating. The lift extension has an elongated pin hole (48 inches) and a screw actuator at the lifting yoke end. For lifting the transfer cask, the lift extension is in the elongated configuration with the lifting yoke pin supported by the bottom of the pin hole. When disengaged from the cask, the lift extension may be placed in the retracted configuration by means of the screw actuator, which

provides support for the lifting yoke while in this configuration. The retracted configuration is required for the combined lift extension and lifting yoke to clear the spent fuel pool operating deck. Figure B-32 depicts the transfer cask lift extension.

5. Cask Pit Support Stand - A removable platform approximately 18 inches in height is placed in the spent fuel pool cask pit. Its functions are to allow release of the transfer cask at an elevation that prevents the 100 ton crane block from contacting spent fuel pool water and to position the cask so that spent fuel can be loaded into the DSC.

   Four polished stainless steel reflectors (targets) are affixed to the corners of the cask pit support stand to serve as a visual guide that the transfer cask is properly centered when lowering into the SFP.

4.4.1.2 Spent Fuel Selection

A description of the administrative procedures which are followed in spent fuel identification is presented in Section 10.2.5. Using special nuclear material control and accountability records, the initial enrichment and burnup for each candidate spent fuel assembly are compared against the acceptable region in Figure B-48. Fuel assemblies falling in the acceptable regions will have qualifying reactivity and decay heat characteristics for safe storage in the NUHOMS®-24P System. If all requirements for spent fuel qualification are met, then documentation of this fact for a given assembly is transmitted to fuel handling personnel prior to assembly retrieval and placement in the DSC. Based upon station maps and special nuclear materials accountability records which indicate the current location of these assemblies, fuel handling personnel visually verify the assembly identification numbers and transfer these assemblies into the DSC. An independent visual verification (using binoculars or CCTV) of the assembly serial number by two different persons is performed prior to assembly retrieval from the spent fuel pool. After all assemblies have been loaded into the DSC, the assembly identification numbers are again checked. In the event that these assemblies must subsequently be retrieved in the future from the HSM/DSC and inserted back into the spent fuel pool, similar accountability/verification procedures will be used.

No fuel is stored in the Site-Specific ISFSI which is known to have any gross structural damage. Duke's damaged fuel assembly and component database contains a record of confirmed and suspect fuel assembly cladding and other structural failures. Assemblies which are suspected of having cladding failure are further examined visually (using cameras) to determine the extent of the damage. Of these assemblies, only those showing gross cladding or structural damage are excluded. This inspection is performed after verification of the assembly identification number. Fuel assemblies which have no record of cladding damage are not inspected in detail; they are observed during the routine fuel handling transfer operation to ensure that the structural integrity of the assembly is maintained.

No fuel assembly cleaning or crud removal operations are planned on initial loadings or retrieval. These operations are not necessary for storage in the NUHOMS®-24P system and would likely increase personnel exposures during fuel loading. The DSC will provide full containment of all radioactive crud materials which are dislodged during the handling and/or storage operations.

4.4.1.3 Spent Fuel Loading

The layout of the spent fuel pool area is shown in Figure B-33 through Figure B-35. After the DSC/transfer cask is lowered from the cask pit platform onto the cask loading pit insert, IFAs which have been qualified are loaded into the DSC. The components and equipment used for this operation are described below:

1. Spent Fuel Pool Manipulator Crane - The spent fuel pool manipulator crane mast or its auxiliary hoist is used to extract IFAs from their pool storage cells and to lower them into the DSC. The spent fuel pool manipulator crane is described in Section 9.1.4.2.2 of the Oconee UFSAR. If the auxiliary hoist is utilized, it is in conjunction with a manual spent fuel handling tool.

2. Spent Fuel Handling Tool - The spent fuel handling tool consists of a pneumatically actuated gripper and suspension cables. Its purpose is to provide remote underwater engagement and disengagement of IFAs. This tool has been used at Oconee for loading IFAs into spent fuel shipping casks, and it required no alteration for use with the DSC.

4.4.1.4 DSC Drying, Backfilling, and Sealing

After the IFAs are loaded into the DSC, the top end shield plug is replaced on the DSC. The DSC top end shield plug is suspended by cables from the transfer cask lifting yoke. The 100 ton spent fuel cask handling crane allows fine adjustment of bridge and trolley positions, hook height, as well as rotation of the crane hook. The bottom of the top shield plug is chamfered to allow a degree of self-centering by the plug. Two separate paths exist for displacement of DSC cavity water as the shield plug is lowered. A gap exists between the shield plug and the DSC shell, and the DSC vent port is open during installation of the top shield plug, both of which provide for displacement of some of the fluid from the DSC. Placement of the shield plug is recognized as a critical step requiring close attention and gradual movements to assure no misalignment or damage to components. The DSC/transfer cask is raised to the SFP cask platform by use of the 100 ton crane with the transfer cask lifting yoke and lift extension. As the DSC approaches the surface of the spent fuel pool, the correct placement of the top lead shield plug is verified visually and through dose rate monitoring. On the SFP cask platform the transfer cask lift extension is removed and the crane hook is attached directly to the lifting yoke in order to provide sufficient lift height during transport of the DSC/transfer cask over the pool deck and into the decontamination pit.

In the decontamination pit the inner top cover plate is seal welded, and the water is purged from the DSC. The DSC is then vacuum dried and backfilled with helium. Helium leak tests are performed on the top shield plug seal weld and the vent and siphon port seal welds. Finally, the outer top cover plate is seal welded into place. These operations are described in detail in Chapter 5 of Reference 1

During the above operations, the IFAs are confined within the DSC with the top end shield plug in place, and the DSC remains seated in the transfer cask. Following these operations, the transfer cask lid is placed, the annular water is drained, and the transfer cask is placed on the transfer trailer for transport to the ISFSI.

The design basis and safety assurance features of the DSC are described in Sections 3.2 and 3.3 of Reference 1. The design basis and safety assurance features of the transfer cask are described in Section 1.3.1.3 of Reference 1. The DSC drying and sealing equipment and operations make use of industry-standard equipment and procedures commonly used for such operations.

4.4.1.5 DSC Unloading

The equipment discussed in Sections 4.4.1.1 and 4.4.1.2 is used for DSC unloading operations. Appropriate DSC cutting equipment and procedures as discussed in Section 5.1.1.9 of Reference 1 will be used to open the DSC which is contained within the transfer cask.

4.4.2 Decontamination System

No decontamination facilities are needed at the ISFSI.

Decontamination of the transfer cask is performed in the decontamination pit. The transfer cask exterior is decontaminated manually before removal from the fuel building by use of detergents and wiping cloths.

It is not anticipated that either the exterior of the DSC or the inside of the transfer cask will become contaminated. The DSC/transfer cask annulus is filled with demineralized water and sealed with an inflatable seal. However, in the event that such contamination occurs, the DSC/transfer cask annulus will be flushed with demineralized water until an acceptable level is achieved.

4.4.3 DSC Repair and Maintenance

No maintenance is required for the DSC for its design life.

4.4.4 Transfer Cask Repair and Maintenance

The function of the transfer cask is to ensure integrity of the DSC during applicable design basis accidents and to provide radiological shielding for the operators during handling and transfer operation. Confinement of radioactive materials is provided by the DSC. Accordingly, a periodic maintenance program has been established to ensure the proper operation of the cask valves, bolts, washers, o-rings and neutron shield pressure relief valves. The lifting surfaces of the cask trunnions are periodically inspected for surface deterioration.

4.4.5 Utility Supplies and Systems

The design of the Oconee Site-Specific ISFSI is based on the NUHOMS®-24P system for storage of irradiated fuel. Each module is a self-contained, passive system requiring no support systems during storage.

However, the ISFSI is provided with a 480/208/120 VAC power supply for operation of the transfer trailer hydraulic positioners, hydraulic ram site security equipment and lighting. Some security equipment at the ISFSI is powered from Oconee plant SSF sources.

Other electrical connections required for ISFSI physical security are described in the Duke Power Company Physical Security Plan (Reference 3).

4.4.6 Other Systems

4.4.6.1 Communications and Alarm System

Details of the communication and alarm system are provided in the Physical Security Plan (Reference 3).

4.4.6.2 Fire Protection System

No flammable or combustible materials are stored within ISFSI or in its immediate vicinity and the ISFSI is constructed of noncombustible heat-resistant materials (concrete and steel). Therefore, no fixed fire extinguishing system is required; however, portable suppression equipment will be provided within the fenced boundary. In the unlikely event of a fire at the ISFSI, the fire brigade will be dispatched from the Oconee Station. The ONS Pre-Fire Plan (Reference 4) contains fire protection requirements for the ISFSI.

4.4.6.3 Maintenance System

The Oconee Site-Specific ISFSI requires no maintenance other than periodic inspection of the HSM air inlets and outlets and removal of debris, if needed.

4.5 Classification of Structures, Systems, and Components

Table A-16 provides a list of major Oconee Site-Specific ISFSI components and their classification. Classification of major components as “Safety Related” or “Radwaste Related” is based on the specific need for component performance under accident conditions. However, designation of specific components as “Safety Related” or “Radwaste Related” is not the only basis for establishing whether any part is important to the safe operation of the ISFSI facilities.

As listed in Revision 1 of the NUHOMS®-24P Topical Report, Section 3.2, Reference 1, the NUHOMS® reinforced concrete HSM including its foundation and DSC support structure, the DSC and its internal basket assembly, and the transfer cask are components considered important to safety. The design criteria for these components are provided in Section 3.2 and summarized in Tables 3.2-1 through 3.2-9 of Reference 1.

The Oconee fuel building overhead crane is non safety-related. The lifting yoke and lift extension used for movement of the transfer cask within the fuel building are designed and procured as components important to safety. The lifting beams used in that part of the operation are controlled by 10CFR Part 50 and NUREG-0612 and are designed to ANSI 14.6-1986 criteria for non-redundant yokes.

As noted in Section 4.5.5, all other components of the NUHOMS® system, including the transfer trailer and cask support skid, skid positioning system, and hydraulic ram system are required to perform their function to successfully transfer a DSC to and from the HSM. These systems are described in Reference 1 with design requirements further delineated in Section 4.5.5 of this UFSAR.

In addition, the transfer cask, HSM, and DSC have been designed to meet very conservative design criteria including postulated conditions which envelop those which may result from the mechanistic failure of the transfer system equipment. Design conditions such as cask drop accident and jammed DSC have been included even though there is no plausible way for these worst case events to occur. Conservative bounding analysis for these conditions have been performed using minimum material yield strengths, strength reduction factors, and factors of safety in accordance with the stringent requirements of the ASME and ACI Codes. Even when applying this conservative criteria considerable design margin for these components and structures remains as evidenced by the analysis results comparisons with acceptance criteria contained in Reference 1. Further, these components and structures are fabricated and constructed to the rigorous standards and methods of the ASME and ACI Codes under a 10CFR 50 Appendix B Quality Assurance Program. These include material qualification, welding and nondestructive examination, and strict surveillance and quality control inspection. The resulting high integrity of the Transfer Cask, DSC, and HSM provide more than adequate assurance that the health and safety of the public and plant personnel are protected.

4.5.1 Transfer Cask

The transfer cask is considered Nuclear Safety Related (QA Condition 1) since it performs primary DSC protection functions under certain transport accident conditions. The transfer cask proposed for use in DSC transfer operations is described in Section 1.3.1.3 of Reference 1, with the exception that the inner cavity length has been modified to 187.43 inches per ONOE-14622.

4.5.2 Dry Storage Canister

The DSC is considered Nuclear Safety Related (QA Condition 1) since it performs criticality control and primary IFA support functions as well as serving as the primary storage containment for the IFAs. It is designed to remain intact under all accident conditions identified in Chapter 8 of this UFSAR with no loss of function. DSC components are designed, constructed, and tested in accordance with Nuclear Safety Related requirements as defined by 10CFR 50, Appendix B and the DPC QA-1 Quality Assurance Program.

4.5.3 Horizontal Storage Module

The HSM functions include shielding, heat removal, DSC support, and DSC tornado missile protection. The HSM is not considered Nuclear Safety Related since it performs no primary IFA containment or criticality control functions. However, HSM functions are considered important to the safe operation of the ISFSI and appropriate levels of documentation and control are applied. The concrete HSM is designed

in accordance with ACI 349-85 and the level of testing, inspection and documentation provided during construction is in accordance with the DPC QA-2 Quality Assurance Program. Based on a review of the function, design, and construction, the HSM and foundation have been appropriately reclassified to the QA-4 designation.

As shown by Table 8.1-12 of Reference 1, the maximum HSM concrete temperature for the long term 70°F ambient storage temperature case in 144°F. This is less than the maximum permissible concrete temperature of 150°F specified by ACI 349-85. The effect of extreme ambient temperatures will be to increase the maximum concrete temperature. However, the methods used to calculate these temperatures assumed that the extreme ambient temperatures would remain constant until steady state conditions can be established within the HSM. In reality, the extreme ambient temperatures will occur infrequently and will last for a short duration insufficient to cause steady state conditions. Therefore, the long term maximum temperatures for the HSM concrete easily meet the ACI 349 temperature limitations.

Coupled with the conservative reductions in concrete material strength used in the HSM design calculations, the design criteria utilized in Reference 1 are adequate to ensure that the HSM will perform its intended safety function for all design conditions.

Typical reinforcing steel design for the HSM basemat, walls, and roof is shown in Figure 8.1-9 of Reference 1. The HSM reinforcing designs are in accordance with the ACI 349-85 Code and are comparable to those previously reviewed and approved by the NRC for the NUHOMS®-07P design.

4.5.4 Foundation

The Oconee Site-Specific ISFSI foundation is designed, constructed and tested to the same design criteria and quality assurance requirements as the HSM.

4.5.5 Transfer Components

The remaining DSC transfer components (i.e. transfer cask trailer and skid, skid positioning system, hydraulic ram system) are necessary for the successful loading of the DSC into the HSM. As discussed in Section 4.3, failure of these components would not endanger the health and safety of the public or plant personnel. Therefore, transfer components are not considered Nuclear Safety Related and are designed, constructed and tested in accordance with good industry practices.

4.5.6 Instrumentation

The Oconee Site-Specific ISFSI is designed to maintain a safe and secure, long-term containment and storage environment for IFAs using only totally passive components. Therefore, no Nuclear Safety Related instrumentation is required for operation of the facility. Instrumentation necessary to perform DSC/transfer cask draining, purging and drying operations consists of industrial grade pressure gauges.

4.6 Decommissioning Plan

Decommissioning of the Oconee Site-Specific ISFSI will be performed consistent with decommissioning of the Oconee Nuclear Station. This is predicated on the ability of the federal government to accept spent fuel at the rates and dates specified in the Nuclear Waste Policy Act of 1982, as amended. It is anticipated that the DSCs will be transported to a federal repository when such a facility is operational. However, should the storage facility not accept the DSCs intact, the NUHOMS®-24P system allows the DSCs to be brought back into the spent fuel pool and the fuel repositioned into the racks for loading into transport casks to be provided by the Department of Energy.

All components of the NUHOMS®-24P system are manufactured of similar materials found in the existing Oconee Station (i.e., reinforced concrete, stainless steel, lead). These components will be

decommissioned by the same methods in place to handle similar materials within the plant. Any of these components that may be contaminated will be cleaned and/or disposed of consistent with the decommissioning technology available at the time of decommissioning.

Although operation of the ISFSI will likely need to continue well beyond decommissioning of the Oconee Nuclear Station, the costs of decommissioning the ISFSI are expected to represent a small and negligible fraction of the cost of the decommissioning the Oconee Nuclear Station. Reference 5 submitted a schedule and justification for a decommission plan which will encompass decommissioning of both Oconee Nuclear Station and the Oconee ISFSI in accordance with 10CFR 50.75 and 10CFR 72.30. The financial options for this plan were submitted on July 24, 1990 for the NRC review and approval, Reference 6 and a clarification was submitted on December 4, 1990, Reference 7.

The radiological impacts due to postulated accidents or operation of the ISFSI are bounding for the conditions when the ISFSI is fully operational. The collective dose to residents within one to two miles of the ISFSI is based on capacity loading of 2112 spent fuel assemblies in 88 storage modules. The occupational dose to site workers assumes radiation from an array of 2 x 10 modules loaded with dry shielded casks each containing 24 spent fuel assemblies. The consequences from accidents are based on failure of 24 spent fuel assemblies contained in a dry shielded cask. The expected radiological impact due to operation of the ISFSI is much less than the regulatory limits specified in 10CFR 72.104 and 10CFR 106(b) and the EPA Protection Action Guides.

4.7 References

1. Topical Report for the Nutech Horizontal Modular Storage System, for Irradiated Nuclear Fuel NUH-002, Rev. 1A, dated July 1989
2. Oconee Nuclear Station Final Safety Analysis Report
3. Duke Power Company Physical Security Plan
4. Oconee Nuclear Station Pre-Fire Plan
5. Letter from H. B. Tucker to U.S. NRC, Document Control Desk dated May 9, 1989
6. Letter from D. L. Hauser to U.S. NRC, Document Control Desk dated July 24, 1990
7. Letter from D. L. Hauser to U.S. NRC, Document Control Desk dated December 4, 1990

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5.0 Storage System Operations

5.1 Operation Description

As a supplement to Sections 4.3 and 4.4 of this report and Section 5.1 of Reference 1 which describe the transport and fuel loading systems and their operation, this chapter describes the actual operations which occur at the ISFSI site after transfer of the DSC from the fuel building.

5.1.1 Narrative Description

The following steps describe the operating procedures which occur after the DSC has been loaded with irradiated fuel assemblies and transferred to the ISFSI site. A more detailed description of HSM loading steps is provided in Section 5.1.1.6 of Reference 1.

5.1.1.1 Loading of the DSC into the HSM

1. Inspect all air inlets and outlets on the HSM to ensure that they are clear of debris. Inspect all screens on the air inlets and outlets for damage. Replace screens if necessary. Using an available yard crane, completely remove the front access door of the HSM. Inspect the interior of the HSM and the DSC support rail surfaces for obstructions or debris.
2. Using an appropriate towing vehicle, position the transfer cask/trailer assembly inside the gross alignment marks on the HSM pad and move it slowly, toward the HSM until the docking collar is at the minimum distance from the HSM opening to allow for cask lid removal.
3. Using the optical alignment system, the targets on the transfer cask and HSM, and the skid positioning system, adjust the position of the cask until the cask is properly aligned with the HSM.
4. Unbolt and remove the cask lid and the cask bottom access plate.
5. Move the cask against the HSM so that the docking collar is completely seated in the HSM recess.
6. Secure the cask to the HSM using the cask restraint system and the anchors on the front wall of the HSM.
7. Align hydraulic ram with transfer cask. Recheck alignment of the HSM, transfer cask, and ram.
8. Extend the hydraulic ram toward the cask and activate the grapple to engage the DSC.
9. Continue extension of the hydraulic ram to move the DSC into the HSM. If the ram fails to extend when the load on the hydraulic system is increased beyond 20,000 lbs., or, if a sudden, large increase in hydraulic pressure is observed, the DSC may be jammed or bound. If jamming or binding is suspected, corrective actions as described in Section 8.1.1.4 will be applied.
10. When the DSC is in the HSM, release the grapple from the DSC and retract the hydraulic ram arm from the transfer cask.
11. Deleted per 2007 update.
12. Lower the HSM front access door back into the door frame to within a few feet of the closed position.
13. Install seismic restraint.
14. Lower to the closed position and tack weld the steel HSM front access door.
15. Measure the change in temperature between the HSM inlet and outlet air vents.
16. Return all equipment to storage locations pending delivery/loading of next DSC.

5.1.1.2 Monitoring Operations

On a 24 hour frequency site personnel will visually inspect all air inlets and outlets of each loaded HSM for both obstructions and screen damage. Obstructions and/or damage will be removed/repaired immediately. The presence of nesting material (e.g. sticks, leaves, paper, insulation, trash, etc.) on or within the HSM structure should be reported to Environmental Health and Safety no later than the next dayshift. (This action is a regulatory commitment as stated in Attachment 2 and Attachment 4 of Reference 2.)

5.1.1.3 Fuel Identification and Accountability

In compliance with NRC regulations, accountability records for all fuel assemblies transferred to, stored in or removed from the ISFSI will be maintained.

The asymmetrical design features of the DSC allow for easy identification of specific assembly storage locations within the DSC. No visible physical labels are necessary for the individual storage locations. Unique storage location symbols will be administratively assigned to each of the 24 DSC storage cells. This is similar to the method which is currently used to track assembly locations within the spent fuel pools. Unique identifications will be assigned to the HSMs, and will be labeled on the HSM exterior. This visible physical identification in combination with the administrative assignment of cell storage locations within the DSC, and a unique serial number stamped on each DSC, will allow for the positive identification of the locations of all ISFSI spent fuel assemblies.

Once a DSC has been inserted into a HSM, the door will be lowered and tack welded into place. These tack welds will sufficiently indicate any attempts at tampering as required in ANSI 57.9-84.

Unique identification of the transfer cask will not be required since only one transfer cask is to be used. This eliminates the possible mixup of transfer casks which might occur with multiple casks being used for concurrent transport operations. Accountability and control of special nuclear materials will be maintained at all times during the loading, transport, and storage of spent fuel assemblies.

5.1.1.4 Unloading the DSC from the HSM

1. Inspect the front access components of the HSM and cut tack welds on HSM access door. Remove cask lid.
2. Position the cask/trailer assembly so that the docking collar is at the minimum distance from the HSM to allow for opening of the HSM front access door.
3. Using the optical alignment system, the targets on the transfer cask and HSM, and the skid positioning system, adjust the position of the transfer cask until the transfer cask is properly positioned with respect to the HSM.
4. Using an available yard crane, raise the front access door of the HSM just high enough to access the seismic restraint.
5. Remove seismic restraint.
6. Remove the HSM access door from the support rails.
7. Move the transfer cask against the HSM so that the docking collar is completely seated in the HSM recess.
8. Secure the transfer cask to the HSM, using the cask restraint system and the anchors on the front wall of the HSM.
9. Align hydraulic ram assembly with transfer cask/trailer and secure in place. Recheck alignment of HSM, cask and ram assembly.

10. Recheck the cask and ram alignment to ensure it is properly positioned with respect to the HSM.
11. Extend the hydraulic ram through the transfer cask into the HSM and activate the grapple to engage the DSC.
12. Retract the hydraulic ram to move the DSC out of the HSM and into the transfer cask. If the ram fails to retract when the load on the hydraulic system is increased beyond 20,000 lbs., or if a sudden large increase in hydraulic pressure is observed, the DSC may be jammed or bound. If jamming or binding is suspected, corrective actions as described in Section 8.1.1.4 will be applied.
13. When the DSC is in the transfer cask, release the grapple from the DSC and retract the hydraulic ram arm from the transfer cask.
14. Replace the top lid and bottom access plate on the transfer cask to allow for transfer cask trailer movement to appropriate location for DSC removal or offsite shipment.

5.1.2 Flow Sheet

Loading and unloading operations are illustrated in Figure B-36.

5.1.3 Identification of Subjects for Safety Analysis

5.1.3.1 Criticality Prevention

Criticality in the NUHOMS®-24P DSC is prevented through a combination of geometrical separation of the fuel cells, neutron absorption in the cell walls and administrative controls on fuel pool soluble boron concentration and the selection of fuel to be stored in the DSC. The DSC basket makes use of two material thicknesses in the cell walls as well as some over-sleeves at the top and bottom of interior cells to accommodate sufficient neutron absorption with qualified fuel assemblies. While the DSC design features will be essentially fixed, the selection of fuel for storage will be a variable. Administrative control of fuel selection will be incorporated into plant procedures. Further discussion of these controls and procedures are provided in Sections 4.4 and 10.2. The criticality analysis for the NUHOMS®-24P System can be found in Section 3.3.4 of Reference 1.

5.1.3.2 Instrumentation

The proposed ISFSI is a system requiring no instrumentation for radiation, temperature or criticality considerations.

5.1.3.3 Maintenance Techniques

Due to the passive nature of the proposed ISFSI, the only maintenance on the HSM will be periodic surveillance of the air inlet and outlet vents to insure continued air flow. Routine maintenance on the transfer cask will also be performed to maintain integrity of top lid, bottom access plate and trunnions.

5.1.3.4 Administrative Controls to Limit DBT Effects

Administrative controls for limiting transfer operations due to potential tornado weather conditions will not be required. The transfer cask in transit has been evaluated for tornado wind speeds and DBT effects in accordance with 10CFR Part 72 and was found to be enveloped by the evaluation for a design basis cask drop accident.

5.2 Control Room and Control Areas

This is a passive system and there is no need for annunicators or other systems to indicate off-normal conditions.

Surveillance for such conditions will utilize visual inspection techniques. Security surveillance will be tied into the main central alarm station and/or secondary alarm station at the Oconee Nuclear Station.

5.3 References

1. Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUH-002, Revision 1A, dated July 1989.
2. LAR 2007-06, RAI response, Dave Baxter to U.S. Nuclear Regulatory Commission, "Response to Requests for Additional Information License Amendment Request No. 2007-06," January 30, 2009
3. NRC letter or May 29, 2009, 'Issuance of Renewed Materials License No. SNM-2503, Oconee Independent Spent Fuel Storage Installation (ISFSI) (Tac Nos. L24184 and L25206)"; Enclosure 2, Safety Evaluation Report.

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6.0 Waste Management

No radioactive wastes are generated during the storage life of DSCs. Radioactive wastes generated during loading operations are treated using existing station facilities and procedures.

Contaminated pool water removed from loaded DSCs is normally drained back into the spent fuel pool with no additional processing. A small amount (<15 CF/DSC) of liquid waste results from transfer cask decontamination. The decontamination procedure results in a small amount of a detergent/demineralized water mixture being collected in the Cask Decontamination Pit. Liquid wastes collected in the Cask Decontamination Pit are directed to the Station Liquid Waste Management System (LWM) for processing.

Potentially contaminated air and helium purged from the DSC following DSC loading and seal welding operations are directed to the Auxiliary Building Ventilation Air System (VA) at a point upstream of the Fuel Building HVAC filter units and radioactive effluent monitor. Purged gases processed with the Fuel Building HVAC filter units are released from the unit vent and will meet station release requirements. This is the same procedure currently utilized for shipping cask operations.

A small quantity (<5 CF/DSC) of low level solid waste is generated as a result of DSC loading operations and transfer cask decontamination. The solid waste generated is processed by compaction using the Volume Reduction (VR) System or incineration using appropriate facilities. This low level waste consists of disposable Anti-C garments, tape, blotter paper, rags, etc.

Descriptions of the LWM, VA and VR Systems are provided in Chapter 11 of the Oconee UFSAR.

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7.0 Radiation Protection

7.1 Ensuring That Occupational Radiation Exposures Are ALARA

7.1.1 Policy & Organizational Considerations

Duke Radiation Protection and ALARA policies are described in Chapter 12 of the Oconee UFSAR. These policies will be applied to the Independent Spent Fuel Storage Facility since it is located on the Oconee Nuclear Station site within the Owner Controlled Area and radiologically supported and controlled by the station Radiation Protection Group.

7.1.2 Design Considerations

The design of the DSC and HSM comply with 10CFR 72 concerning ALARA considerations. Specific considerations that are directed toward ensuring ALARA are:

1. Thick concrete walls on the HSM to reduce the surface dose to below an average of 20 mr/hr. The 20 millirem per hour dose rate was the approved maximum for HSM wall dose rates in the NUHOMS®-07P Topical Report. Actual calculated HSM wall surface dose rates are below 10 millirem per hour except at vent and door openings. The HSM shielding design was deemed ALARA considering construction costs, heat dissipation, and access requirements. Also, refer to Section 7.1.2 of Reference 2 for the basis of the average 20 millirem per hour HSM contact dose rate. Additional shielding analysis is included in Section 7.3.2.2 and Table 7.3-2 of Reference 2.
2. Lead/carbon steel shield plug on the ends of the DSC to reduce the dose to workers performing drying, sealing, and loading operations.
3. Use of a shielded transfer cask for handling and transportation operations of loaded DSCs.
4. Fuel loading procedures which follow accepted practice and build on existing experience.
5. Recess in the HSM front for the transfer cask to fit into so as to reduce scattered radiation during transfer.
6. Double seal welds on each end of DSC to provide redundant radioactive material containment.
7. Placing clean water in the transfer cask and DSC and sealing the DSC/transfer cask annulus to prevent contamination of DSC exterior during loading.
8. Placing external shielding blocks over HSM air outlets to reduce direct and streaming doses.
9. Passive system design that requires minimum maintenance.
10. Insertion of internal shielding blocks around air inlets to reduce direct and streaming doses.
11. Use of portable shielding during DSC drying/welding operations to limit streaming from top end shield plug/DSC annulus. The portable shield used during DSC closure operation to limit streaming from top end shield plug/DSC annulus consists of 2.0 inches of Bisco NS-3, or equivalent, as shown in Figure A.2, Appendix A of Reference 2. The portable shield will be put in place to minimize doses during direct-access operations such as top shield plug automatic welding setup, draining and drying operations, and setup of automatic welding equipment for the top cover plate. The portable shield will incorporate provisions to facilitate access to the drain and fill ports but may not be necessary during automated welding operations.
12. To minimize scatter at the HSM door during DSC loading, the top of the transfer cask docks into a recess in the HSM access door opening.

13. Use of approved procedures to control contamination during handling and transfer of fuel.
14. Leaving water in the DSC cavity and DSC/transfer cask annulus during welding operations as long as possible to reduce streaming through the gap. The water level in the DSC/transfer cask annulus is lowered to approximately 5 to 10 inches below the top of the DSC shell. The water level in the DSC cavity is lowered to approximately 4 inches below the bottom surface of the top end shield plug. These levels are maintained during shield plug welding operations. The remaining water in the annulus is not drained until after the cask cover plate is bolted into place.
15. Providing a large control area around the ISFSI and locating the facility well away from normally occupied areas.
16. Operation of the ISFSI will be performed under the Radiation Protection program of the station as described in Section 7.1.1.
17. Lead blanket screens may be employed to further reduce dose during decontamination and transfer operations. These and other ALARA measures precautions may be employed as needed based on experience gained from preoperational testing and early fuel loading efforts.

7.1.3 ALARA Operational Considerations

Consistent with Duke's overall commitment to keep occupational radiation exposures as low as reasonably achievable, (ALARA), specific plans and procedures are followed by station personnel to assure that ALARA goals are achieved. Operational ALARA policy statements are formulated at the corporate staff level through the issuance of the System Radiation Protection Manual and the ALARA Manual and are implemented at each nuclear plant by means of procedures. These statements and procedures are consistent with the intent of Section C.1 of Regulatory Guides 8.8 and 8.10.

Since the ISFSI is a passive system, no maintenance is expected on a normal basis in the facility. Maintenance operations on the transfer cask, transfer trailer and other ancillary equipment is performed in a very low dose environment when fuel movement is not occurring.

Maintenance activities that could involve significant radiation exposure of personnel are carefully planned. They utilize any previous operating experience, and are carried out using well trained personnel and proper equipment. Radiation Work Permits (RWPs) for non-routine operations, or Standing Radiation Work Permits (SRWPs) for routine operations are issued for each job, listing Radiation Protection requirements that shall be followed by all personnel working in the Radiation Control Area (RCA). Where applicable, specific radiation exposure reduction techniques, such as those set out in Regulatory Guide 8.8, are evaluated and used.

The station ALARA Committee carefully reviews operations and maintenance activities involving the major plant systems to further assure that occupational exposures are kept ALARA.

7.2 Radiation Sources

7.2.1 Characterization of Sources

This section describes the design basis radiation sources and source geometries used for the ISFSI shielding calculations.

Neutron and gamma sources are developed based on the reference irradiated fuel assembly described in Table A-1. The reference fuel assembly is assumed to be irradiated to a burnup of 40,000 mwd/mtu and cooled to a decay heat rate of less than or equal to 0.66Kw before being stored in the DSC. The initial enrichment considered is 4.0 weight percent U-235. The source terms include the irradiated fuel, activated portions of the fuel assemblies and deposited activity from corrosion products in the reactor coolant. All

primary sources are considered to be originating in the fuel with secondary gammas generated in the shielding considered by the shielding codes used.

The detailed calculation of gamma ray group fractions provided in Table 7.2-2 of Reference 2 is summarized in Table A-17.

The fuel region is modeled as a homogeneous cylinder for shielding calculations as shown in the model geometry descriptions. The homogenous source over the active fuel region includes fission product, actinide and light element activation product sources. The burnup distribution is assumed flat along the axial and radial extent of source. This modeling technique is used in all shielding calculations except supplementary calculations performed subsequent to ISFSI operation. The supplementary calculations use a heterogeneous source distribution to demonstrate the effect IFA end fitting and plenum region light element activation and reduced self-shielding have on localized TC surface dose rates.

Additional details of the radiation source terms and dose conversion factors used in ISFSI shielding analysis are provided in Section 7.2.1 of Reference 2.

7.2.2 Airborne Radioactive Material Sources

The DSC is double seal welded to prevent any gaseous release of material during storage. The possibility of release during fuel handling in the spent fuel pool is covered in the accident analysis. The other possible source of airborne radioactive material is the outside surface of the DSC. This surface is protected from contamination while the DSC is in the fuel pool by filling the annulus between the DSC and the transfer cask with demineralized water and sealing the annulus to prevent pool water from coming in contact with the outside surface of the DSC. This prevents any significant accumulation of potential airborne sources on the canister. The outside surface of the transfer cask is considered to be contaminated upon removal from the fuel pool and will be cleaned and contamination measurements made to ensure no unacceptable contamination remains before leaving the fuel building.

Cask venting releases are directed to the fuel pool HVAC units upstream of the HEPA and carbon filter units. The filtered gas is ultimately released through the unit vent after it is monitored by both the spent fuel pool storage area HVAC monitor and unit vent monitor.

7.3 Radiation Protection Design Features

7.3.1 Installation Design Features

The design considerations listed in Section 7.1.2 ensure that occupational exposures to radiation are ALARA and that a high degree of integrity is obtained for the confinement of radioactive materials. The ISFSI will be hand monitored as needed for construction, loading and unloading operations. Since the storage facility contains no active systems, no continuous monitoring systems other than fence-mounted dosimetry are needed. Applicable portions of the guidance given in Regulatory Position 2 of Regulatory Guide 8.8 have been followed: 1) Access control of radiation areas is addressed in Sections 7.1.3 and 10.2.5. 2) Radiation shields substantially reduce exposure of personnel during operations and storage; radiation streaming has been reduced by providing labyrinth-type shield penetrations. 3) NUHOMS®-24P is a passive storage system; no process instrumentation or controls are necessary during storage. 4) Airborne contaminants and gaseous radiation sources are controlled by the integrity of the double seal welded DSC assembly. 5) No crud is produced by the NUHOMS®-24P system. 6) The necessity for decontamination is reduced by maintaining the cleanliness of the DSC during operations (see Section 5.1); the DSC surfaces are smooth, nonporous, and free of crevices, cracks, and sharp corners. 7) No radiation monitoring system is required during storage. 8) No resin or sludge is produced by the NUHOMS®-24P system.

Radiation sources are contained within DSCs which are stored in concrete HSMs. The radioactive sources are described in detail in Section 7.2.1 of Reference 2.

7.3.2 Shielding

7.3.2.1 Radiation Shielding Design Features

Radiation shielding is an integral part of both the DSC and HSM designs. The features described in this section assure that doses to personnel and the public are "as low as is reasonably achievable" (ALARA).

The DSC body is a rolled stainless steel plate. Details of the DSC and HSM and relevant dimensions can be found in the drawings in the proprietary supplement of Reference 2. The lead/carbon steel shield plugs provide gamma shielding at both ends of the DSC. During handling operations, shielding in the radial direction is provided by the NUHOMS®-24P transfer cask.

Two penetrations in the top shielded end plug allow water draining, vacuum drying and helium backfilling of the DSC. The penetrations are located away from fuel assemblies and contain sharp, non-coplanar bends to reduce radiation streaming. Figure B-42 shows the physical arrangements of the DSC end-shields and location of doses reported in Table A-18. These dose rates assume the water levels in the DSC cavity and DSC/Cask annulus are lowered to the levels specified in Section 1.3.1.7.

The transfer cask provides radiological shielding during the DSC drying operation and during the transfer to the HSM. Both neutron (solid Bisco NS-3) and gamma (lead) shielding are incorporated into the cask design. The NS-3 neutron shield is 3" thick (nominal) and has a density of 1.76 gm/cc. A 10% hydrogen content loss is assumed in the shielding analysis due to anticipated degassing of the NS-3 induced by elevated temperatures. The as-built transfer cask lead gamma shield thickness is verified through radiographic examination to be 3.38" thick (nominal), but varies in thickness from approximately 3.15" to 3.5". Areas where the lead thickness falls below 3.38" are covered by an additional 1/4" thickness of stainless steel neutron shield jacket to compensate for the reduced gamma shielding effectiveness of the lead.

The HSM provides shielding in both the radial and axial directions during the storage phase. Thirty six inch thick, portland cement, concrete walls and roofs provide neutron and gamma shielding. The module's front end opening is covered by a steel door with a neutron shield.

Openings to the HSM interior are placed above the end shield regions and not directly over the active fuel region. Sharp duct bends and concrete shielding caps over the exhaust exits assure that radiation streaming is reduced to a minimum. Figure B-42 shows details of the module penetrations and locations of doses reported in Table A-18.

Portable shielding during handling operations may be applied during specific handling operations. However, Section 7.4 provides an assessment of design basis on-site doses without the use of portable shielding.

7.3.2.2 Shielding Analysis

This section describes the radiation shielding analytical methods used in calculating relevant NUHOMS®-24P system dose rates during the handling and storage phases. The dose rates were calculated at the locations listed in Table A-18. Figure B-42 shows these locations on the HSM, DSC and transfer cask. The three computer codes used for analysis are described below.

Computer Codes ANISN (Reference 3), a one-dimensional discrete ordinates transport computer code, was used to obtain neutron and gamma dose rates at the outer HSM wall, centerline of DSC end plug, and outside the loaded transfer cask. The CASK (Reference 6) cross section library, which contains 22 neutron energy groups and 18 gamma energy groups, was applied in an $S_8 P_3$ or $S_{16} P_3$ approximation.

Calculated radiation fluxes were multiplied by flux-to-dose conversion factors to obtain final dose rates. The ANISN calculations used coupled neutron and gamma libraries. Therefore, both primary and secondary gammas are calculated in each run.

QAD-CG (Reference 4), a three-dimensional point-kernel code, was used for direct gamma shielding analysis of the HSM door, the DSC and transfer cask end sections, the DSC/transfer cask annulus, and the HSM air vent penetrations. Mass and buildup were all obtained from QAD-CG's internal library for eight energy groups. The gamma energy spectrum was determined in the same manner as the

Shielding analysis results are summarized in Table A-18. Additional details regarding methods, models ANISN analysis.and assumptions used in ISFSI shielding analyses are provided in Section 7.3 of Reference 2.

A similar version, QAD-CGGP, is used in supplementary calculations performed to determine the level of localized gamma dose rate peaking which may occur over areas of the TC surface corresponding to IFA end fitting and fuel pin plenum axial elevations. Gamma sources and spectra for the various IFA source regions modeled (i.e., active fuel, upper and lower end fittings, and upper and lower fuel pin plenums) are determined in the same manner as in ISFSI shielding calculations described in Section 7.3 of Reference 2. Supplementary ANISN shielding calculations were performed to verify the adequacy of the Phase I and II HSMs assuming a minimum concrete density of 140 lb/ft^3.

7.4 Estimated On-Site Collective Dose Assessment

7.4.1 Operational Dose Assessment

This section establishes the expected cumulative dose delivered to site personnel during the fuel handling and transfer activities associated with one NUHOMS®-24P module. Chapter 5 describes in detail the ISFSI operational procedures, a number of which involve radiation exposure to personnel.

The ISFSI is a radiologically controlled area. Access to the storage modules is restricted such that for normal operation, no access closer than 50 feet is allowed except for security and surveillance purposes. Except during periods of additional module construction, there is no adjacent work area close by, so very little dose is received from fuel in storage. Access is primarily needed to load new canisters into storage modules and dose from previously stored fuel will be received during these operations. The occupational exposure received during DSC transfer operations is included in the operational dose assessment summarized in Table A-19. The occupational dose estimates provided in Table A-19 were calculated using reference fuel assembly characteristics (see Table A-1) and other site-specific parameters. Dose contributions from hidden module scatter effects and self shielding for an 88 loaded module array are included in the Table A-19 results for DSC transfer operations. The dose received for other operations performed within the HSM storage facility secured area is negligible.

The phased construction of modules up to the licensed capacity of 88 will be undertaken on an as-needed basis considering required lead time, station operation and construction schedules. Increments of additional module construction are flexible and can continue until the ultimate licensed capacity of 88 HSMs is reached. Construction work performed subsequent to the loading of any HSM with spent fuel will result in worker exposures from direct and sky shine radiation in the vicinity of the loaded HSMs. In the event that additional Site-Specific HSMs are constructed, construction materials will be staged away from the adjacent loaded HSMs. The construction area will be surveyed prior to beginning work to ascertain actual dose rates and temporary shielding may be provided if needed to lower any unacceptable dose rates. The most significant dose rate contributors to the construction area are the inlet and exhaust vent openings. These dose rates may be reduced using temporary shielding screens around the vents near the construction area. After the concrete is placed for the additional modules, the additional shielding will further reduce dose rates.

The dose estimate for additional construction was based on labor cost estimates for a 2 x 10 module array. It was assumed that 60 percent of the labor hours are expended in the radiation area and the prefabrication work would be done in low or no dose areas. Table A-20 summarizes expected construction doses by task.

The maximum dose received from the loading, construction, and maintenance of Horizontal Storage Modules is 7.5 Rem per year for the expected loading rates. This is approximately 3.5% of normal station dose. The total includes fuel handling and canister loading operations, additional module construction and general maintenance of the facility. These dose estimates are based on operational history from RWP records.

7.4.2 Storage Term Dose Assessment

No firm construction schedule for module addition has been developed at this time and thus the array sizes mentioned in Section 7.4.1 are representative of possible additional increments. Additional increments of HSMs will be constructed as required to balance the off-loading of Oconee's fuel from the storage pools and transhipment to the federal repository.

Figure B-43 is a graph of the dose rate (mr/hr) versus distance from the face of a 2 x 3 array of NUHOMS®-24P HSMs. Figure B-44 and Figure B-45 show the dose rate versus distance from the front or side of the array for various other HSM array sizes. These curves were constructed from the shielding analysis described in the previous sections and are for the dose rate in the worst case direction from the modules (perpendicular to the doors). The bounding conditions may be obtained by simply scaling the results from these curves. Direct neutron and gamma flux, as well as the air-scattered radiation from the module surfaces are considered. The surface radiation sources used for the direct and air scattered dose calculations are shown in Figure B-46. Neutron and gamma flux spectra for the surface of the HSM are provided in Table A-21. The HSM surface spectra are obtained from normalized ANISN model flux data. The ANISN HSM model as well as the CASK cross section library are described in Section 7.3 of Reference 2. The CASK cross section library is made up of 40 energy groups (groups 1-22 are neutron and groups 23-40 are gamma). Air-scattered dose rates are determined with the computer code SKYSHINE-II (Reference 5); direct dose rates are calculated using the computer code MICROSHIELD (Reference 7). The direct flux from the "hidden" row of modules is considered completely shielded by the front row. All HSMs are assumed loaded with sufficiently cooled (≤0.66Kw per assembly) spent fuel.

7.5 Radiation Protection Program

The ISFSI is located adjacent to the Oconee Nuclear Station within the plant protected area. The Oconee Nuclear Station Radiation Protection Manager has responsibility for Radiation Protection activities at the ISFSI.

The Radiation Protection and ALARA programs are discussed in the Oconee Nuclear Station UFSAR, Chapter 12. Detailed discussions of Radiation Protection and ALARA are contained in Duke's Radiation Protection and ALARA program manuals.

Radiation protection requirements for all radiological work at the Oconee Nuclear Station and ISFSI is governed by approved procedures and directives. These procedures and directives include, but are not limited to, the following:

1. Procedure for personnel dosimetry issue.
2. Issuance, revision, and termination of radiation work permits and standing radiation work permits.
3. Procedure for roping off, barricading, and posting of radiation control zones.
4. Decontamination procedure for equipment and areas.

5. Smear swab sampling, counting, and calculation.
6. Procedure for quantifying airborne radioactivity.
7. Radiation Protection ALARA preplanning work.

In addition, the Radiation Work Permits for the maintenance and fuel handling tasks associated with DSC operations incorporate radiological hold points and precautions, where necessary, to ensure these activities are performed in a radiologically safe manner and are ALARA.

Deleted paragraph(s) per 2007 update.

7.6 Environmental Monitoring Program

The current radiological environmental monitoring program for Oconee Nuclear Station will also serve as the operational program for the ISFSI.

No liquid or airborne effluents are anticipated from the HSM. Therefore, the dose to any offsite point will only be from direct and scattered gamma radiation. Several environmental sampling locations are presently located at the Oconee site boundary surrounding the ISFSI. The closest of these is less than 0.3 miles from the ISFSI, well within the 1-mile exclusion area boundary. In addition, the dose rates at the ISFSI will be monitored periodically with fence-mounted dosimetry as part of the Oconee routine radiological monitoring program. This will be used in part to control occupational exposures and will also augment the environmental program.

As a result, no changes to the environmental program are anticipated.

7.7 Estimated Off-Site Collective Doses

Doses to any offsite point are only from direct and scatter gamma radiation from the storage module. The estimated dose from the modules to any dose point beyond the site boundary is well below regulatory limits even when combined with station doses for both airborne and direct gamma dose.

The ISFSI is situated approximately 1 mile from the exclusion area boundary. The estimated maximum dose rate in any direction at 5000 feet for up to an 88 module array of HSM's as provided by Figure B-43 through Figure B-45 is less than 1.0 x 10^{-6} mr/hr. The estimated annual dose to the public is conservatively calculated as 7 person-millirem per year. The maximum dose to the nearest potential future resident from the ISFSI is 7.5E-2 millirem per year.

Note that construction of the site specific HSMs was suspended at module number 40. Subsequent modules have been of the General License (GL) design. An integrated dose analysis of the combined contributions by both the Site-Specific and General License storage systems are contained in the written evaluations prepared for the GL system pursuant to 10 CFR 72.212(b)(2)(i).

7.8 References

1. Oconee Nuclear Station Final Safety Analysis Report
2. Topical Report for the Nutech Horizontal Modular Storage (NUHOMS®-24P) System for Irradiated Nuclear Fuel, NUH-002, Revision 1A, July 1989
3. Oak Ridge National Laboratory, “ANISN - Multigroup One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering” CCC-254, Oak Ridge National Laboratory (1977)
4. Oak Ridge National Laboratory, “QAD-CGGP, Point-Kernal Gamma Ray Shielding Code,” CCC-396, Oak Ridge National Laboratory (1979)

5. C. M. Lampley, “The SKYSHINE-II Procedure: Calculation of the Effects of Structure Design on Neutron, Primary Gamma-Ray and Secondary Gamma-Ray Dose Rates in Air” NUREG/CR-0781, RRA-T7901, USNRC (1979)
6. Radiation Shielding Information Center, “CASK: 40 Group Neutron and Gamma Ray Cross Section Data,” DLC-23, September 1978
7. Grove Engineering, Inc., “Microshield User's Manual, A Program for Analyzing Gamma Radiation Shielding,” Version 2.0, 1985
8. Pacific Nuclear Calculation DUK003.0320, “Shielding Evaluation of 140 lb/ft^3 Density Concrete HSM”, dated 7-2-92
9. Pacific Nuclear Calculation DUK003.0321, “Shielding Evaluation for Oconee Phase II HSMs”, dated 6-26-92

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8.0 Accident Analyses

In previous chapters, features important to safety have been identified and discussed. The purpose of this chapter is to identify and analyze a range of credible accident occurrences (from minor accidents to the design basis accidents) and their causes and consequences. For each situation, reference is made to the appropriate chapter and section describing the considerations to prevent or mitigate the accident.

ANSI/ANS-57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," defines four categories or design events that provide a means of establishing design requirements to satisfy operational and safety criteria. The first design event is associated with normal operation. The second and third design events apply to events that are expected to occur during the life of the installation. The fourth design event is concerned with severe natural phenomena or low probability events. The second design event is addressed in Section 8.1 and the third and fourth design events are discussed in Section 8.2. The first design event is addressed in Chapter 4 and need not be addressed here.)

8.1 Off-Normal Operations

In this section, design events of the second type as defined by ANSI/ANS-57.9-1984 are addressed. Design events of the second type consist of events that might occur with moderate frequency on the order of once during any calendar year of operation.

The limiting off-normal event is defined as a jammed DSC during loading or unloading at the ambient temperature extremes of -40°F and +125°F as described in Reference 1 (Section 8.1). This postulated event results in the limiting structural loads on the DSC and thermal loads on the DSC and HSM for all identified off-normal events. The ambient extremes for the Oconee site are bounded by the assumed values.

8.1.1 Jammed DSC During Loading or Unloading

8.1.1.1 Postulated Cause of Jammed DSC

If the transfer cask is not accurately aligned with the HSM, the DSC might become bound or jammed during the transfer operation. The maximum tolerable misalignment for the Oconee Site-Specific ISFSI transfer operation is discussed in Section 5.1 of Reference 1.

8.1.1.2 Detection of Jammed DSC

When DSC jamming occurs, the hydraulic pressure in the ram will increase above normal insertion pressures. When this occurs, the DSC will be presumed to be jammed. The pushing and pulling forces are limited to 20,000 lbs., with override control available to the operator.

8.1.1.3 Analysis of Effects and Consequences

The analysis of the DSC under assumed jamming and binding conditions is covered in Section 8.1.2.1 of Reference 1. In both jammed DSC scenarios considered, the stress on the DSC body is shown to be much less than the ASME code allowable stress. Therefore, plastic deformation of the DSC body will not occur and there is no potential for rupture. The analysis presented in Reference 1 is applicable to Oconee Site-Specific ISFSI operation.

The ram extension and retraction force is limited to 80,000 pounds by the Hydraulic Power Unit (HPU) PLC.

8.1.1.4 Corrective Actions

In cases of DSC jamming or binding, the required corrective action is to reverse the direction of applied force on the DSC, and return the DSC to its previous position. Since no plastic deformation has occurred, the return of the DSC to its previous position will be unimpeded. The transfer cask alignment is then rechecked and the transfer cask repositioned as necessary before reinsertion is renewed.

8.1.2 Radiological Impact of Off-Normal Operations

Based on the off-normal operation analysis results presented, there is no additional radiological impact due to off-normal operations beyond what is presented in Chapter 7 of this UFSAR.

8.2 Accidents

This section addresses design events of the third and fourth types as defined by ANSI/ANS-57.9-1984, and other credible accidents which could impact the safe operation of the Oconee Site-Specific ISFSI. The postulated events addressed are:

1. Loss of Air Outlet Shielding
2. Tornado/Tornado Missile
3. Earthquake
4. Transfer Cask Drop
5. Transfer Cask Loss of Neutron Shield
6. Lightning
7. Blockage of Air Inlets and Outlets
8. DSC Leakage
9. Accidental Pressurization of DSC
10. Load Combinations
11. Floods
12. Explosions

The postulated accidents listed above include all events identified as potentially resulting in offsite doses in excess of 25 mrem.

8.2.1 Loss of Air Outlet Shielding

This postulated accident involves the loss of both air outlet shielding blocks from the top of the HSM. All other components of the Oconee Site-Specific ISFSI are assumed to be in their normal conditions.

8.2.1.1 Cause of Accident

The air outlet shielding blocks are designed to remain in place and completely functional for all events except tornado missiles. To demonstrate the safety of the Oconee Site-Specific ISFSI design, this accident assumes that both shielding blocks are completely lost.

The air outlet shield blocks are attached to the HSM by welding to an embedded plate in the HSM roof. In the highly unlikely event of a recovery situation, the damaged shield block would be removed from the HSM and temporary shielding would be placed around the outlet opening in such a way that a worker could perform the necessary recovery techniques with a minimal radiation exposure. All Duke ALARA

procedures, such as pre-staging construction activities in a no-dose area, would be followed throughout the entire recovery process.

8.2.1.2 Accident Analysis

There are no structural or thermal consequences to the Oconee Site-Specific ISFSI resulting from the loss of the air outlet shielding blocks. The air flow resistance is less without the shield blocks and, hence, the air flow will increase (slightly) and provide more cooling of the DSC. Radiological consequences of this accident are described in the next section.

8.2.1.3 Accident Dose Consequences

Offsite radiological consequences result from an increase in air scattered (skyshine) dose due to the loss of the shield blocks. Onsite radiological consequences result from an increase in direct (during recovery operations on the HSM roof) and skyshine radiation. The calculation of these doses during normal conditions is described in Section 7.4. Removal of the shield blocks results in local surface dose increase of 3551 mr/hr at the vent opening. This increased surface dose was used in the models described in Section 7.4 to calculate the direct and scattered doses as a function of distance from the HSM. Table A-22 shows comparisons of the increased dose rate as a function of distance due to loss of the shielding blocks. The dose increase to a person located 100 (meters) away from the Oconee Site-Specific ISFSI installation for eight hours a day for seven days (recovery time) would be 30 mr. Subsequent shielding calculations for the Phase II modules predict removal of the shield blocks will yield a local surface doserate of 3827 mr/hr. As a result, the projected 30 mr/hr dose received by a person 100 meters from the ISFSI during the seven day recovery period will increase slightly less than 10%. This increase is considered insignificant based on the 10CFR72.106 limit of 5 rem and the short duration of the accident. The increased dose to an offsite person for 24 hours a day for seven days located 5000 feet away would be minimal.

To recover from the loss of shielding blocks, a new block is transferred to the HSM. After the shield block is transferred to the HSM, a yard crane is used to lift the block into position. The block is then bolted in place. The entire remounting operation should take less than 30 minutes, of which a mechanic will be on the HSM roof for approximately 15 minutes. During this time, the mechanic will receive less than 50 mr. An additional dose to the mechanic and to the crane operator on the ground while putting the shield block in place will be 10 mr each (assuming an average distance of 10 ft. from the center of the HSM front wall).

8.2.2 Tornado/Tornado Missile

8.2.2.1 Cause of Accident

The most severe tornado wind loadings specified by NUREG-0800, NRC Regulatory Guide 1.76 and the Oconee UFSAR are used as the design basis for this accident condition.

8.2.2.2 Accident Analysis

The applicable design parameters of the design basis tornado (DBT) are specified in Section 3.2.1 of this UFSAR. The DBT design parameters specified in Section 3.2.1 are identical to those used in the reference Topical Report in the determination of forces on structures for this accident. The analysis of the HSM and Transfer Cask response to DBT loadings is covered by the analysis presented in Section 8.2.2 of Reference 1.

8.2.2.3 Accident Dose Consequences

The only component of the Oconee Site-Specific ISFSI which is not capable of withstanding tornado generated missiles are the precast air outlet shielding blocks. The consequences of losing the shielding blocks during this accident is presented in Section 8.2.1.3 of this UFSAR.

8.2.3 Earthquake

8.2.3.1 Cause of Accident

As specified in Section 3.2.3, the Oconee Site-Specific ISFSI MHE acceleration value is 0.15g for both vertical and horizontal ground acceleration.

8.2.3.2 Accident Analysis

The reference Topical Report analysis of earthquake loads assumes a value of 0.25g and 0.17g for maximum horizontal and vertical acceleration, respectively. Reference 1 seismic stress analysis also used a multiplier of 1.5.

Since the value of the seismic accelerations for the Oconee Site-Specific ISFSI site are lower than that assumed in Reference 1, the stress analysis envelopes the site specific criteria.

In summary, the Oconee ISFSI seismic analysis using site specific criteria is enveloped by the analysis in Reference 1.

8.2.3.3 Accident Dose Consequences

Major components of the Oconee Site-Specific ISFSI are designed and evaluated to withstand the forces generated by the MHE. Hence, there are no dose consequences.

8.2.4 Cask Drop

8.2.4.1 Cause of Accident

This section addresses the structural integrity of the DSC and its internals under a postulated transport cask accident condition. It is postulated that the transfer cask described in Section 4.3 with the DSC inside is dropped 80 inches onto a thick concrete slab. Due to the design of the transfer trailer and cask skid, an actual drop event is not considered credible. Cask drop target parameters are given in Table A-23.

8.2.4.2 Accident Analysis

The Oconee Site-Specific ISFSI transfer cask is analyzed for an 80 inch drop accident using the method of analysis presented in Section 8.2.5. of Reference 1, as modified by Reference 6.

The analysis presented in Reference 1 assumes an 80 inch cask drop using Oconee ISFSI transfer cask parameters. Hence, the Reference 1 analysis covers the Oconee accident analysis. Therefore, the stress on the various structural components of the DSC and its internals are the same as those reported in Table 8.2-7 of Reference 1, as modified by Reference 6.

8.2.4.3 Accident Dose Consequences

Since the stress analysis has shown that all components important to safety of the DSC and its internal basket will perform their intended function under this accident condition, there are no dose consequences.

8.2.5 Transfer Cask Loss of Neutron Shield

8.2.5.1 Postulated Cause of Solid Shield Loss

The neutron shield jacket is designed, fabricated, tested, and inspected as ASME Section III, Division 1 Class 2 vessels. The associated ASME quality assurance program will assure that there are no poor joints, or other substandard components in the transfer cask. The Bisco NS-3 neutron shield material is a rigid solid when cured and will not flow freely through openings in the jacket. Therefore, a loss of shield material will only occur in cases of external damage to the shield jacket and concurrent displacement of NS-3 material.

8.2.5.2 Detection of Shield Material Loss

Damage to the neutron shield jacket and material would be visually obvious. Anticipated loss of hydrogen from the NS-3 material resulting from degassing at evaluated temperatures is accounted for in the shielding analysis (see Section 7.3.2).

8.2.5.3 Analysis of Effects and Consequences

For the purpose of this analysis, it is assumed that the transfer cask neutron shield will be breached as a result of postulated drop accident, and the shielding effect of the NS-3 will be lost. The effect of this will increase the cask surface contact dose from 180 mrem/hour to 837 mrem/hour. The only potential off-site dose consequences would be additional direct and air scattered radiation if the accident were to occur sufficiently close to the site boundary. It is assumed that eight hours would be required to either recover the neutron shield or to add temporary shielding while arranging recovery operations. As result, it is estimated that on-site workers at an average distance of fifteen feet would receive an additional dose rate of 80 mrem/hr.

Off-site individuals at a distance of 2000 feet would receive an additional dose of 5.7E-4 mrem for the assumed eight hour exposure. This increase is well within the limits of 10CFR 72 for an accident condition. Also, this does not preclude handling operations for recovery of the cask and its contents. Water bags or other neutron absorbing material could be wrapped around the cask to reduce the surface dose to an acceptable limit for recovery operations thus minimizing exposure of personnel in the vicinity. The actual local and off-site dose rates, recovery time and operations needed to retrieve the cask, and the required actions to be performed following the event will depend upon the severity of the event and the resultant cask and trailer/skid damage.

8.2.6 Lightning

8.2.6.1 Cause of Accident

The likelihood of lightning striking the Oconee Site-Specific ISFSI and causing an off-normal operating condition is not considered a credible accident given the ISFSI lightning protection provided. The lightning protection system for the ISFSI is designed in accordance with NFPA NO. 78-1979 Lightning Protection Code. This system precludes any damage to the HSM or its internals due to lightning.

8.2.6.2 Accident Analysis

8.2.6.2.1 HSM

Should lightning strike the Oconee Site-Specific ISFSI, the normal operation of the HSM will not be affected. The current discharged by the lightning will follow the low impedance path offered by the lightning protection system. Therefore, the HSM is not damaged by the heat or mechanical forces

generated by current passing through the higher impedance concrete. Since the HSM requires no equipment for its continued operation, the resulting current surge from the lightning will not affect the normal operation of the HSM.

8.2.6.2.2 Power Supplies

The hydraulic power supplies for the transfer trailer hydraulic positioners and the hydraulic ram are independent systems. Each of these systems have manually operated pumps which could be used in case of a power failure. Electrical power supplies to the Oconee Site-Specific ISFSI site serve no safety related functions, since their loss would not adversely affect the NUHOMS®-24P safety related components or the health and safety of plant personnel or the public. Some security equipment at the ISFSI is powered from Oconee plant SSF sources. These sources also do not serve a safety function.

The electrical power distribution system and associated equipment are electrically bonded to the lightning protection and grounding system for the ISFSI. The retail power transformer is installed with lightning protection features in accordance with National Electric Safety code requirements which were current at the time of construction.

The lightning protection design meets the requirements of NEPA-78, Lightning Protection Code: 1986 Edition and IEEE Standard 665.

8.2.6.2.3 Welding of DSC to Support Structure

Movement of the DSC from the transfer cask to the fully inserted position in the HSM takes less than 20 minutes. Transfer operations will not be attempted during a major thunderstorm when there is potential danger to plant personnel or costly damage to equipment. Therefore, the possibility of the DSC becoming welded to the support structure by a lightning strike is extremely unlikely. In addition, there is contact between the transfer cask and HSM mating collar, such that the anchorage of the transfer cask to the HSM shown in Topical Report Figure 4.2-6 provides a grounding path to the HSM. To complete this path, the attachment plates are grounded to the HSM reinforcing which will provide additional assurance that this event will not occur. Lightning would likely strike the highest nearby structure, which is a light pole.

The HSM rails are bonded to the HSM grounding system by means of exothermically welding a bare copper conductor to the embedded steel support plates and the HSM grounding system. Additionally, the trailer mounted ram assembly tripod is bonded to the HSM grounding system during cask positioning operations.

8.2.6.3 Accident Dose Consequences

Since no off-normal operating condition will develop as a result of lightning striking the ISFSI, there are no radiological consequences.

8.2.7 Blockage of Air Inlets and Outlets

This accident involves the complete and total blockage of all HSM air inlets and outlets.

8.2.7.1 Cause of Accident

Since the HSMs are located outdoors, the air inlets and outlets could potentially be blocked by debris from such unlikely events as tornados. Oconee Site-Specific ISFSI design features such as a perimeter fence and separation of air inlets and outlets reduce the potential for this accident.

8.2.7.2 Accident Analysis

The structural consequences due to the weight of debris blocking the air openings are bounded by the structural consequences of other accidents described in this section (i.e., tornado and earthquake analyses). The thermal consequences of this accident result from heating of the DSC and HSM due to the loss of natural convection cooling. An analysis of this condition is provided by Section 8.2.7 of Reference 1.

8.2.7.3 Accident Dose Consequences

There are no offsite dose consequences as a result of this accident. The only dose increase is related to the recovery operation where the onsite worker will receive an additional 700 mr during an estimated 8 hour debris removal period.

8.2.8 Dry Storage Canister Leakage

The DSC is designed for no leakage and analysis of normal and accident conditions have shown that no credible conditions could breach the canister body or fail the double seal welds at each end of the DSC. However, to show the ultimate safety of the Oconee Site-Specific ISFSI system, a total and complete instantaneous leak is postulated.

This postulated accident is the instantaneous release directly to the environment of 30% of all fission gasses mainly Kr_{85} and I_{129} contained in all the fuel rods in all 24 PWR fuel assemblies. This accident assumes that all fuel rods are ruptured and that concurrent DSC leakage occurs. All other components of the ISFSI system remain intact.

8.2.8.1 Cause of Accident

Due to the passive nature of the Oconee Site-Specific ISFSI system and the various design features, there is no credible event that could result in the rupture of all fuel rods concurrent with DSC leakage. However, to demonstrate the safety of the ISFSI design, this accident assumes that the fuel rods and the canister are ruptured due to an event of unspecified origin.

8.2.8.2 Accident Analysis

In the postulated Dry Storage Canister Leakage Accident, it is assumed that one DSC is breeched and fuel fails simultaneously releasing 30% of all fission gasses contained in 24 fuel assemblies. Following long-term wet storage (>7.5 years) the gaseous fission products which can be released are Kr_{85} and I_{129}. The total DSC inventories assumed for Kr_{85} and I_{129} are 2.75E+03 and 1.87E-02 Curies, respectively; these inventories are based on ORIGEN-S computer code (Reference 2) analysis for 24 B&W 15x15 fuel assemblies irradiated for 40,000 MWD/MTU and decayed for 7.5 years.

Whole body and maximum organ doses are calculated for a hypothetical maximum individual assumed to be present at the nearest site boundary location (a distance of approximately 1 mile) for the duration of the event. A meteorological dispersion parameter (X/Q) of 4.5E-04 sec/m^3 is used in calculating the maximum potential offsite doses; this X/Q value is consistent with the value referenced in the Oconee SER, Section 3.2.4, Units 2 and 3. Dose conversion factors used are obtained from NRC Regulatory Guide 1.109 and a breathing rate of 3.47E-04 m^3/sec is used in calculating inhalation dose.

There are no structural or thermal consequences resulting from the DSC leakage accident described above. The radiological consequences of this accident are presented in Section 8.2.7.3.

8.2.8.3 Accident Dose Consequences

This postulated accident involves the rupture of one DSC. All fuel rods contained in the ruptured DSC are assumed to fail simultaneously such that 30% of all the fission gasses in the irradiated fuel assemblies are instantaneously released to the atmosphere. Whole body and maximum organ doses are calculated for a hypothetical individual assumed to be present at the Oconee Nuclear Station exclusion zone for the duration of the event. A meteorological dispersion parameter of 4.5E-4s/m^3 is used in calculating the maximum potential offsite doses. The resulting calculated doses are 7 and 200 mr for the maximum offsite whole body and thyroid doses, respectively. These accident doses are well within the 10CFR 72 limit of 5000 mr whole body dose equivalent.

8.2.9 Accidental Pressurization of DSC

This accident addresses the consequences of accidental pressurization of the DSC.

8.2.9.1 Cause of Accident

Internal pressurization of the DSC could result from fuel cladding failure which would release fuel rod fill gas and free fission gas.

8.2.9.2 Accident Analysis

The maximum DSC accident pressurization is calculated assuming that the fuel rod fission gas release fraction is 30%, and that the original fuel rod fill pressure is 480 psig (Oconee fuel actually has a maximum initial fill pressure of 465 psig). The resulting internal DSC pressures at Oconee's maximum ambient temperature of 116°F and at the minimum ambient temperature of -30°F are below the accident pressures reported in Section 8.2.9 of Reference 1 (for temperature extremes of 125°F and -40°F). The limiting accident for DSC pressurization is the loss of transfer cask neutron shield. Under these conditions, the gas temperatures in the DSC will rise to 600°F producing a DSC internal pressure of 49.1 psig. The DSC shell stresses due to accident pressurization are enveloped by those reported in Reference 1.

During DSC opening, appropriate radiation protection techniques will be employed for respiratory protection of the workers and for preventing any uncontrolled releases to the environment. During cutting operations these techniques may include installation of exhaust hoods which discharge to the fuel building ventilation system upstream of the HEPA and carbon filter units and supplied air to the workers. During filling and venting, the vented gases will, also, be routed to the fuel building ventilation system. This is a routine precaution taken for opening of spent fuel shipping casks, and it would provide protection from respirable radioactive particles and, also, from the unlikely presence of a significant amount of escaped fission gases.

8.2.9.3 Accident Dose Calculations

Since the accidental pressurization is within the design basis limits of the DSC, there are no dose consequences.

8.2.10 Load Combinations

The load categories associated with normal operating conditions and accident conditions have been described and analyzed in previous chapters of this report. The load combination evaluation of various Oconee Site-Specific ISFSI safety related components is addressed in this section.

8.2.10.1 Cause of Accident

The simultaneous loading of major Oconee Site-Specific ISFSI components by combined accident and normal loads would result in the load combinations analyzed.

8.2.10.2 Accident Analysis

The methodology used in combining normal operating and accident loads and their associated overload factors for various Oconee Site-Specific ISFSI components is presented in Section 8.2.10 of Reference 1. The Reference 1 analysis envelopes the Oconee ISFSI. The load combination and fatigue analysis in Reference 1 indicates major ISFSI components can withstand severe load combination and thermal cycling without failure.

8.2.10.3 Accident Dose Consequences

There are no dose consequences for postulated load combination events.

8.2.11 Flooding

The elevation of the Oconee Site-Specific ISFSI yard at Elevation 825.0 is more than eleven feet higher than the maximum flood level postulated for Lake Keowee, and therefore, flooding of the ISFSI will not occur.

8.2.12 Explosions

8.2.12.1 Cause of Accident

The explosion on S.C. Highways 130 or 183 of a tanker containing 8,500 gallons of gasoline would subject the Oconee Site-Specific ISFSI to a surface overpressure.

8.2.12.2 Accident Analysis

According to the NRC Regulatory Guide 1.91 “Evaluations of Explosives Postulated on Transportation Routes Near Nuclear Power Plants,” the explosion of 8,500 gallons of gasoline 1,100 feet from the Oconee Site-Specific ISFSI on S. C. Highway 130 or 183, would result in a peak overpressure of 1 psi about 1,900 feet from the point of explosion and therefore an overpressure of 2.3 psi at the ISFSI. The HSM has been designed to withstand a maximum tornado wind pressure of 2.75 psi on the HSM leading wall, and -2.48 psi on the HSM roof. Therefore, the HSM overpressure from the explosion of a gasoline tanker on either S. C. Highway 130 or 183 is enveloped by the wind pressure analysis and design for a DBT.

8.2.12.3 Accident Dose Consequences

There are no dose consequences for postulated explosions.

8.3 Site Characteristics Affecting Safety Analysis

All site characteristics affecting safety analyses presented in this UFSAR are noted where they apply.

8.4 References

1. Topical Report for the Nutech Horizontal Modular Storage (NUHOMS®-24P) System for Irradiated Nuclear Fuel, NUH-002, Revision 1A, dated July 1989

2. "SCALE-3: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation," NUREG/CR-0200, ORNL, Revision 3, December 1984
3. Pacific Nuclear Calculation DUK003.0320, "Shielding Evaluation of 140 lb/ft3 Density Concrete HSM", dated 7-2-92.
4. Pacific Nuclear Calculation DUK003.0321, "Shielding Evaluation for Oconee Phase II HSMs", dated 6-26-92.
5. 10CFR 72.48 Evaluation for Revisions to Oconee ISFSI, OSC-3485, Rev. 15, dated October 3, 1995.
6. Memo to file, dated July 9, 2001 documenting Vendor evaluation of ISFSI Phase I & II loaded canisters. ONS Master File No. OS-262.

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9.0 Conduct of Operations

9.1 Organizational Structure

9.1.1 Corporate Organization

Duke is responsible for development of the Oconee Site-Specific ISFSI including design, construction, quality assurance, testing and operation of the facility. The corporate organization is fully described in Chapter 13 of the Oconee UFSAR.

9.1.1.1 Corporate Functions, Responsibilities and Authorities

The corporate organization provides line responsibility for operation of the Company. Various departments within the Company have responsibility for design, construction, quality assurance, testing and operation of the Oconee Nuclear Station as well as the Oconee Site-Specific ISFSI. Duke's corporate functions, responsibilities and authorities for quality assurance addressed in Topical Report DUKE-1-A, as described in Chapter 11 of this report, are applicable for appropriate portions of the ISFSI.

9.1.1.2 Applicant's In-House Organization

Duke's Nuclear Generation Department, headed by the Senior Vice President, Nuclear Generation, has corporate responsibility for overall nuclear safety, as established by Technical Specifications. Reporting to the Senior Vice President is a Vice President for each nuclear site, and the four Nuclear General Office Location Managers.

The Nuclear Generation Department Organization is described in Section 13.1.2 of the Oconee UFSAR.

9.1.1.3 Interrelationship with Contractors and Suppliers

The development of the ISFSI including design, construction, testing and operation are managed and conducted by Duke. Technical support and other services for the program relating to the Nutech Engineers, Inc. supplied NUHOMS®-24P are provided by Nutech Engineers, Inc. (now Transnuclear, Inc.).

9.1.1.4 Applicant's Technical Staff

The Corporate technical staff supporting the Oconee Site-Specific ISFSI is described in Section 13.1.1 of the Oconee UFSAR.

9.1.2 Operating Organization, Management, and Administrative Control System

9.1.2.1 Onsite Organization

The onsite organization of the Oconee Nuclear Station is responsible for operation of the Oconee Site-Specific ISFSI. The organization for Oconee Nuclear Station is fully described in Section 13.1.2 of the Oconee UFSAR.

9.1.2.2 Personnel Functions, Responsibilities and Authorities

The functions, responsibilities and authorities of major personnel positions, including discussions of specific succession of responsibility for overall operation of the Oconee Nuclear Station including the Oconee Site-Specific ISFSI are described in Section 13.1.2.2.2 of the Oconee UFSAR.

9.1.3 Personnel Qualification Requirements

The qualifications of personnel in the operating staff are in accordance with Section 4 of ANSI 3.1-1978, "Selection and Training of Nuclear Power Plant Personnel," and are in accordance with Regulatory Guide 1.8 (Rev. 1). Section 13.1.3 of the Oconee UFSAR provides more details on personnel qualification requirements.

9.1.3.1 Minimum Qualification Requirements

The minimum qualification requirements for major operating, technical, and maintenance supervisory personnel are described in Section 13.1.3.1 of the Oconee UFSAR.

9.1.3.2 Qualifications of Personnel

The qualification of personnel assigned to the managerial and technical positions are available for inspection on site.

9.1.4 Liaison with Other Organizations

All aspects of the Oconee Site-Specific ISFSI development including design, procurement, construction, and operation have been managed and conducted by Duke. Nutech Engineers, Inc. (now Transnuclear, Inc.), Duke's subcontractor provides certain engineering, technical support, and other services for the ISFSI project relating primarily to the NUHOMS®-24P dry storage cask system design.

9.2 Preoperational Testing and Operation

Prior to operation of the Oconee Site-Specific ISFSI, complete functional tests of the in-plant operations, transfer operations, and HSM loading and retrieval were performed. These tests verified that the storage system components (e.g. DSC, transfer cask, transfer trailer, etc.) could be operated safely and effectively.

9.2.1 Administrative Procedures for Conducting Test Program

Pre-operational testing procedures were written in accordance with existing Oconee procedure controls as governed by Duke's QA Program.

9.2.2 Test Program Description

The testing program required the use of a DSC mock-up, transfer cask and associated handling equipment, transfer trailer, hydraulic ram and an HSM. The tests simulated, as nearly as possible, the actual operations involved in preparing a DSC for storage and ensured that they could be performed safely during actual emplacement of IFAs in the Oconee Site-Specific ISFSI. Shielding verification, which was not completely achievable during dry runs, took place during the initial IFA loadings.

9.2.2.1 Operations

9.2.2.1.1 DSC and Associated Equipment

An actual DSC and a part-length mock-up of a DSC were obtained for pre-operational testing. The DSC was loaded into the transfer cask to verify fit and suitability of the DSC lift rig. Additionally, the DSC was used in operational testing of the transfer equipment and HSM.

The part-length mock-up was similar to the top end of the DSC with lead shield plug facsimile. The mock-up was welded by the automated welding equipment. Emphasis was placed on acceptability of the weld, as well as compliance with approved ALARA practices. The mockup was also used for verification of vacuum drying, helium backfilling, and cutting open operations.

9.2.2.1.2 Transfer Cask and Handling Equipment

Functional testing was performed with the transfer cask, lift yoke, lift extension, and remote actuation equipment associated with the lift yoke. These tests ensured that the transfer cask could be safely transported from the ONS truck bay to the decontamination pit. From there, the DSC/transfer cask was placed into the spent fuel pool cask pit to verify clearances and travel path and proper operation of the annulus seal.

9.2.2.1.3 Off-Normal Testing of the DSC and Transfer Cask

In the unlikely event that a problem arises during loading of IFAs into the DSC, seal welding/evacuation/drying, transport of the DSC, or emplacement of a DSC into an HSM, no immediate action would be required. Operations in the spent fuel pool could be suspended indefinitely with IFA cladding temperatures well below the average long-term storage temperature limit of 340°C. During the other operations the IFA cladding temperature remains well below 570°C - an acceptable temperature for short-term operational and accident conditions. The DSC/transfer cask could be returned to the spent fuel pool if these other operations could not be completed in a timely manner. As stated in Section 9.2.2.1.1, the ability to open a sealed DSC was demonstrated by cutting open the DSC mockup.

9.2.2.1.4 Transfer Trailer and HSM

The DSC/transfer cask was loaded with test weights to simulate loaded fuel and placed on the transfer trailer. It was then transported to the ISFSI and aligned with an HSM. Compatibility of the transfer trailer with the transfer cask, negotiation of the travel path to the Oconee Site-Specific ISFSI, and maneuverability within the confines of the ISFSI were verified. Additionally, it was verified that the 80 inch design basis height for a postulated cask drop could not be exceeded.

The transfer trailer was aligned and docked to the HSM. The hydraulic ram was used to emplace a DSC loaded with test weights in the HSM and remove it. Loading of the DSC into the HSM verified that the transfer skid alignment system, hydraulic positioners, and ram grapple assembly could operate safely for both emplacement of a DSC into an HSM, and removal of a DSC from an HSM.

9.2.2.1.5 Off-Normal Testing of the Transfer Trailer and HSM

In the unlikely event that a problem should occur that prevents loading the DSC into the HSM, no immediate remedial action will be required. IFAs may be stored in the transfer cask while corrective action is taken.

The most severe condition would occur if a failure of the hydraulic ram, after partial insertion of a DSC into an HSM, were to prevent complete emplacement of the DSC. (Radiological shielding and decay heat removal are not compromised by this condition, but the transfer trailer may not be moved away until the

DSC is completely within the confines of either the transfer cask or the HSM.) Pre-operational testing verified that reversal of DSC movement could be completed by the operator of the hydraulic ram.

9.2.3 Test Discussion

1. The purpose of the pre-operational tests was to ensure that a DSC could be properly and safely placed in the spent fuel pool, loaded with IFAs, transported to the Oconee Site-Specific ISFSI, emplaced in the HSM, and removed from the HSM. Proper operation of the DSC, transfer cask, and transfer trailer, as well as the associated handling equipment (e.g. lifting yoke and lift extension, welding equipment, vacuum equipment) provided this assurance.
2. Pre-operational test procedures were developed as stated in Section 9.2.1. Specific detailed procedures were developed and implemented by ONS personnel who were responsible for ensuring that the test requirements were satisfied. Changes made to the pre-operational procedures were incorporated into the appropriate loading procedure.
3. The result of the pre-operational tests was the successful completion of the following without damage to any component associated piece of equipment; loading of a DSC into the transfer cask, seal welding, drying, backfilling, and cutting open of the mockup DSC, placement of the transfer cask into and out of the ONS spent fuel pool, transporting the transfer cask loaded with a DSC to the ISFSI, and emplacement in an HSM and removal from an HSM.

9.3 Training Program

The existing training program for ONS was modified to incorporate the training needed for operation of the Oconee Site-Specific ISFSI, in accordance with the Duke Employee Training and Qualification System (ETQS) Standards Manual. ETQS provides a systematic approach to training as described in the ONS UFSAR, Section 13.1 of the Oconee UFSAR

9.3.1 Training for Operations Personnel

Since the Oconee Site-Specific ISFSI is a passive storage system, generalized training is provided in the areas of cooling, radiological shielding, and structural characteristics of the DSC/HSM.

Detailed operator training is provided for DSC preparation and handling, fuel loading, transfer cask preparation and handling, and transfer trailer loading. Although operations personnel may not be directly involved in transport or HSM loading, detailed training is provided to permit oversight of these operations by fuel handling personnel.

Additionally, Fire Brigade training has been expanded to include the ISFSI in the Oconee Nuclear Station Pre-Fire Plan.

9.3.2 Training for Maintenance Personnel

Maintenance personnel, involved with the Oconee Site-Specific ISFSI operations, receive generalized training in the NUHOMS®-24P storage system. Specific training is provided for use of the automated seal welding equipment for the top end shield plug; operation of the transfer trailer; alignment of the cask skid with the HSM; alignment of the hydraulic ram assembly; and normal and off-normal operation of the hydraulic ram. Specific training is also being provided for cleaning of the HSM air inlets and outlets.

9.3.3 Training for Radiation Protection Personnel

Radiation Protection personnel receive generalized training in the NUHOMS®-24P system. Specific training has been provided in radiological shielding design of the system, particularly the top end shield

plug, DSC/transfer cask, the shielding issue associated with transfer of the DSC into the HSM, and the HSM itself.

9.3.4 Training for Security Personnel

Details of the training program for security personnel are provided in the Guard Training Plan contained in a separate enclosure which is withheld from public disclosure in accordance with 10CFR 2.790(d) and 10CFR 73.21.

9.4 Normal Operations

Under normal operations, the Oconee Site-Specific ISFSI provides independent storage of Oconee spent fuel away from the Oconee plant facilities. With the exception of some limited physical and continuous electronic security surveillance, the ISFSI functions as a passive system. Loading of fuel assemblies into the ISFSI, which occurs periodically, requires specific procedures that are separate from those of normal plant operations.

9.4.1 Procedures

Operating, testing, and maintenance procedures are prepared, revised, reviewed, and approved in accordance with the Duke Nuclear Generation Department "Nuclear Policy Manual" (NPM). (The NPM sets forth the specific requirements of the Duke QA Topical Report, DUKE-1-A, which has been approved by the NRC as meeting the requirements of 10CFR 50 Appendix B.)

9.4.2 Records

The Oconee Site-Specific ISFSI records are maintained in accordance with existing Oconee Nuclear Station procedures.

9.5 Emergency Planning

The Emergency Program for Oconee Nuclear Station has been determined to be adequate to manage the consequences of events which might occur involving the Oconee Site-Specific ISFSI. Appropriate reviews were made of the existing emergency plan initiating conditions and it was determined that no changes were necessary. The Emergency Program consists of the Oconee Nuclear Site Emergency Plan and related implementing procedures. Also included are related radiological emergency plans and procedures of state and local governments. The purpose of these plans is to provide protection of plant personnel and the general public and to prevent or mitigate property damage that could result from an emergency at the Oconee Nuclear Site. The combined emergency preparedness programs have the following objectives:

1. Effective coordination of emergency activities among all organizations having a response role.
2. Early warning and clear instructions to the population-at-risk in the event of a serious radiological emergency.
3. Continued assessment of actual or potential consequences both on-site and off-site.
4. Effective and timely implementation of emergency measures.
5. Continued maintenance of an adequate state of emergency preparedness.

The Emergency Plan has been prepared in accordance with Section 50.47 and Appendix E of 10CFR Part 50. The plan shall be implemented whenever an emergency situation is indicated. Radiological emergencies can vary in severity from the occurrence of an abnormal event, such as a minor fire with no

radiological health consequences, to nuclear accidents having substantial onsite and/or offsite consequences. In addition to emergencies involving a release of radioactive materials, events such as security threats or breaches, fires, electrical system disturbances, and natural phenomena that have the potential for involving radioactive materials are included in the plans. The plan contains adequate flexibility for dealing with any type of emergency that might occur.

The activities and responsibilities of outside agencies providing an emergency response role are detailed in the State of South Carolina emergency plans and the emergency plans for Oconee and Pickens Counties.

The emergency response resources available to respond to an emergency consist of the following: 1. ONS Site Personnel, 2. Duke corporate headquarters personnel, 3. Other Duke nuclear station personnel, and, in the longer term, federal emergency response organizations (e.g. NRC, DOE, FEMA). The first line of defense in responding to an emergency lies with the normal operating shift on duty when the emergency begins. Therefore, members of the Oconee staff are assigned emergency response roles that are to be assumed whenever an emergency is declared. The overall management of the emergency is initially performed by the shift supervisor until he/she is relieved by the Station Manager. In the event of an emergency, he serves as the Emergency Coordinator. Onsite personnel have preassigned roles to support the Emergency Coordinator and to implement his directives.

Special provisions have been made to assure that ample space and proper equipment are available to effectively respond to the full range of possible emergencies. The emergency facilities available include the Oconee Control Room, Operational Support Center, Technical Support Center, Joint Information Center, and the Emergency Operations Facility. These facilities are described in the site emergency plan.

Emergency plan implementing procedures define the specific actions to be followed in order to recognize, assess, and correct an emergency condition and to mitigate its consequences. Procedures to implement the Plan provide the following information:

1. Specific instructions to the plant operating staff for the implementation of the Plan.
2. Specific authorities and responsibilities of plant operating personnel.
3. A source of pertinent information, forms, and data to ensure prompt actions are taken and that proper notifications and communications are carried out.
4. A record of the completed actions.
5. The mechanism by which emergency preparedness will be maintained at all times

9.6 Physical Security Plan

The purpose of the security program for the Oconee Nuclear Station is to establish and maintain a physical security program that has the capabilities for the protection of spent fuel stored in the NUHOMS®-24P system.

Information regarding the security program for the Oconee Site-Specific ISFSI is withheld from public disclosure in accordance with 10CFR 2.390(d) and 10CFR 73.21.

9.7 Aging Management

This Section includes the Aging Management information required for the renewed license period of the ONS Site-Specific ISFSI. This information was included in Enclosure 3, Appendix C of the license renewal application (Reference 2).

9.7.1 Aging Management Programs

An assessment of the ONS Site-Specific ISFSI, Transfer Cask, Transfer Cask Lifting Yoke, Transfer Cask Lift Extension, and Cask Pit Support Stand inspection and monitoring activities identified existing activities necessary to provide reasonable assurance that Site-Specific ISFSI and Transfer Cask components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the renewal period. This section describes these aging management programs.

9.7.1.1 Site-Specific ISFSI Aging Management Program

The ONS Site-Specific ISFSI Aging Management Program credits the ONS Part 50 "Inspection Program for Civil Engineering Structures and Components," as described in Chapter 18 of the ONS UFSAR (Reference 1). This program involves monitoring the interior and exterior surfaces of the HSMs, including visual inspection of the accessible concrete; any exposed steel subcomponents, embedments, and attachments; and the lightning protection system. Interior inspections are conducted on a sampling basis (minimum of one HSM) on a 10 year frequency. Exterior inspections are conducted on a 5 year frequency.

Monitored conditions include the following:

1. Concrete – spalling, cracking, delaminations, honey combs, leaching, discoloration, loss of material, or any other property that would be noted by visual inspection
2. Structural Steel – corrosion, peeling paint, deflection, lost or missing anchors/fasteners, missing or degraded grout under base plates, twisted beams, cracked welds
3. Equipment Foundations – settlement, cracked concrete
4. Equipment Supports – cracked concrete, loose connections, corroded steel
5. Roof Systems – structural integrity, deteriorated penetrations (i.e. drains, vents, etc.), signs of water infiltration, cracks, ponding and flashing degradation
6. Seismic Gaps – gaps or loss of joint filler material
7. Lightning Protection System (above grade) - corrosion

9.7.1.2 Transfer Cask Aging Management Program

The Transfer Cask Aging Management Program credits the Transfer Cask annual maintenance procedure. This procedure includes visual inspections of the carbon steel subcomponents. Monitored conditions include corrosion.

9.7.1.3 Transfer Cask Lifting Yoke and Lift Extension Aging Management Program

The Transfer Cask Lifting Yoke and Lift Extension Aging Management Program credits the Transfer Cask lifting equipment annual inspection procedure. This procedure includes visual inspections of the Transfer Cask Lifting Yoke and Lift Extension carbon steel subcomponents. Monitored conditions include corrosion.

9.7.1.4 Cask Pit Support Stand Aging Management Program

The Cask Pit Support Stand Aging Management Program credits the ONS Part 50 "Oconee Chemistry Control Program," as described in Chapter 18 of the ONS UFSAR (Reference 1). Loss of material and cracking are prevented through control of specified limits on chloride in the spent fuel pool water. The spent fuel pool water is sampled on a monthly basis.

9.7.2 Time-Limited Aging Analysis

This section discusses the results for each of the time-limited aging analyses (TLAAs) evaluated for license renewal. The evaluations have demonstrated that the analyses have been projected to the end of the renewed license period.

9.7.2.1 DSC Shell Cracking Due to Fatigue

The original fatigue analysis of the DSC was conducted for a service life of 50 years. The fatigue effects were addressed using the criteria contained in Section III NB-3222.4 of the ASME Code. The analysis evaluated the DSC under the six criteria and concluded that the DSC and other components satisfy the criteria and no consideration of fatigue is required for the 50 year service life .

A new analysis evaluates the DSC against the six criteria for a service life of 60 years. The analysis concludes that all six criteria are satisfied for a service life of 60 years.

Thus, DSC shell cracking due to fatigue has been reanalyzed and has been determined not to be a concern for the renewed license period.

9.7.2.2 BISCO NS-3 and Concrete Radiation Exposure

HSMs

The Phase 1 HSM doors use BISCO NS-3 as shielding material. The gamma dose rate in the door cavity of the HSM was calculated to be 330 mrem/hr. This results in a gamma dose of approximately 1.8×10^5 Rads for a service life of 60 years. This is well below the service limit of 1.5×10^{10} Rads.

The intergrated neutron fluence in the HSM concrete for 50 years was calculated to be 1.2×10^{14} neutrons/cm^2. This results in an integrated fluence of approximately 1.44×10^{14} neutrons/cm^2 for a service life of 60 years. This is well below the service limits for the material for fast and thermal neutron exposure, 1.6×10^{17} neutrons/cm^2 and 1.5×10^{19} neutrons/cm^2, respectively.

Thus, the effects of radiation on the HSM concrete and BISCO NS-3 have been reanalyzed and are projected not to be a concern for the renewed license period.

Transfer Cask

The Transfer Cask contains BISCO NS-3 neutron shielding between the cask outer shell and neutron shield jacket. The bounding estimated gamma and neutron dose rates at the inner surface of NS-3 in the cask are 250 mrem/hr and 959 mrem/hr, respectively. The gamma exposure for a 60 year service life is bounded by the Phase 1 HSM door evaluation, above.

The neutron dose rate for the Transfer Cask neutron shielding is lower than the rate on the HSM interior concrete surface. The integrated fluence is estimated to be approximately 1.44×10^{14} neutrons/cm^2 for a service life of 60 years for the HSM concrete as documented in the HSM evaluation, above. The fluence is well below the service limits for the material for fast and thermal neutron exposure – 1.6×10^{17} neutrons/cm^2 and 1.5×10^{19} neutrons/cm^2, respectively.

Thus, the effects of radiation on the HSM concrete and BISCO NS-3 has been reanalyzed and has been determined not to be a concern for the renewed license period.

9.7.2.3 Transfer Cask Trunnions and Lift Equipment Fatigue

The Transfer Cask, Transfer Cask Lifting Yoke, and Lift Extension are used for the General License version of the NUHOMS® systems that is currently being used at ONS for new loadings. If it is conservatively assumed that this equipment will be used for all future dry storage loadings until the end of reactor operations at ONS (i.e., 2034), including removal of IFAs from the spent fuel pool approximately

300 loadings will be required. If it is further assumed this equipment will be required for returning the loaded DSCs to the spent fuel pools, 300 unloading cycles would be required. With ten lifts required for each loading cycle, six lifts required for each unloading cycle, and another 100 miscellaneous lifts (for dry runs, annual maintenance, etc.), it is estimated that a total of 4,900 lifts will be required for the 60 year service life of the equipment.

Transfer Cask Trunnions

The fatigue analysis for the Transfer Cask trunnions shows the limiting number of lifts is 4, 859 for the insert plate where the trunnions attach to the cask structural shell.

Since this limit is less than the 4,900 lifts estimated for a 60 year service life, corrective action will be required to ensure the lift limit is not exceeded. To ensure timely corrective action is taken, the fatigue calculation will be periodically reviewed throughout the service life of the Transfer Cask.

Transfer Cask Lifting Yoke

The fatigue analysis for the Lifting Yoke shows the limiting number of lifts is 1.3×10^5. This is well above the conservative estimate for number of lifts required for a 60 year service life.

Transfer Cask Lift Extention

The fatigue analysis for the Transfer Cask Lift Extention shows the limiting number of fatigue cycles is 2.3×10^5. This is well above the conservative estimate for number of lifts required for a 60 year service life.

Thus, fatigue of the Transfer Cask Trunnions, Transfer Cask Lifting Yoke, and Transfer Cask Lift Extension has been reanalyzed and has been determined not to be a concern for the renewed license period.

9.7.2.4 HSM Concrete Thermal Cycling

The original thermal cycling analysis of the HSM concrete was conducted for a service life of 50 years. The number of thermal cycles was calculated as 18,250 for the 50 year service life. Prorating this value to a 60 year service life yields a value of 21,900 cycles. This is well below the thermal cycling limit of 1×10^7 cycles for reinforced concrete.

Thus, thermal cycling of HSM concrete has been reanalyzed and has been determined not to be a concern for the renewed license period.

9.7.2.5 DSC Support Structure Thermal Fatigue

The original thermal fatigue analysis of the DSC support structure was performed for a 50 year service life. The number of thermal cycles was conservatively calculated as 18,250 (one per day). The analysis concludes that thermal fatigue need not be considered based on Section III, NF-3331.1 of the ASME code which requires such analysis for components with greater than 20,000 cycles.

The fatigue analysis was revised to reflect 10,950 thermal cycles for a 60 year service life, using a less conservative assumption of 182.5 cycles per year. The revised assumption is still conservative based on actual stress conditions and historical daily temperature variations at ONS. This is well below the ASME code requirement for consideration of thermal fatigue for components with greater than 20,000 cycles.

Thus, fatigue of the DSC support structure has been reanalyzed and has been determined not to be a concern for the renewed license period.

9.8 References

1. Oconee Nuclear Station Updated Final Safety Analysis Report (UFSAR)
2. LAR 2007-06, D.A. Baxter to U.S. Nuclear Regulatory Commission, "Site-Specific Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application, "January 30, 2008
3. NRC letter or May 29, 2009, 'Issuance of Renewed Materials License No. SNM-2503, Oconee Independent Spent Fuel Storage Installation (ISFSI) (Tac Nos. L24184 and L25206)"; Enclosure 2, Safety Evaluation Report.

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10.0 Operating Controls and Limits

The Oconee Site-Specific ISFSI operates as a passive system requiring minimal surveillance. However, there are some operating controls and limits that apply. These controls and limits which are listed below are discussed in detail in the following corresponding sections of this chapter. Other items which must be controlled such as those related to fuel movement and loading are based on normal operation and postulated accidents as discussed in Chapter 4 and Chapter 8, respectively, of this UFSAR.

10.1 Operating Controls and Limits

Operating limits and controls are included in Reference 2.

10.2 Development of Operating Controls and Limits

This section provides an overview and general bases for the operating controls and limits specified in Reference 2, which provides the specifications associated with the operation of the Oconee Site-Specific ISFSI to ensure the protection of the public's health and safety.

10.2.1 Functional and Operating Limits, Monitoring Instruments and Limiting Control Settings

The Oconee Site-Specific ISFSI utilizes the NUHOMS®-24P system which is a passive design. Therefore, with the exception of the limit placed on the translational force exerted on the DSC by the hydraulic ram, no monitoring instruments or limiting control settings are utilized at the ISFSI facility. Long term operating variables such as HSM storage temperatures and confinement integrity will be controlled through observance of the operational control and limit specifications described in Reference 2.

Another control which falls under the Oconee Station's 10CFR50 operating license is a restriction on minimum cooling time for fuel stored in certain locations of the spent fuel pools during cask handling operations. These restrictions ensure that any radioactivity releases remain below regulatory guidelines in the event of an in-pool cask drop accident.

10.2.2 Limiting Conditions for Operation

10.2.2.1 Equipment

Limiting conditions for the Oconee Site-Specific ISFSI equipment are specified in Reference 2. In addition, the ram hydraulic system will be pre-set to insure that translational loads on a DSC during movement into the HSM are automatically limited to a maximum of 20,000 lbs. (Override control will be available to hydraulic ram operator for use during off-normal remedial action if needed.)

10.2.2.2 Technical Conditions And Characteristics

The following technical conditions and characteristics are required for the NUHOMS®-24P system:

1. Boron Concentration in DSC Moderator
2. DSC Vacuum Pressure During Drying
3. DSC Helium Backfill Pressure
4. DSC Helium Leak Rate
5. DSC Dye Penetrant Test of Closure Welds

6. Fuel Assembly Retrieval and Inspection
7. DSC Surface Contamination
8. DSC Draining Requirements

A description of the bases for selecting the above conditions and characteristics is detailed in Reference 2. The overall technical and operational considerations are further described in Section 10.2.2.2 of Reference 1.

10.2.3 Surveillance Requirements

Surveillance Requirements for the Oconee Site-Specific ISFSI are specified in Reference 2.

10.2.4 Design Features

Changes to site specific design features important to safety are not anticipated for the Oconee Site-Specific ISFSI. Design features of the NUHOMS®-24P system important to safe operation are outlined in Section 10.2.4 of Reference 1 and in Reference 2. Changes to any of these design features will be implemented only after appropriate regulatory review and approval.

10.2.5 Administrative Controls

Use of existing and proposed Duke organizational and administrative systems and procedures, record keeping, review, audit and reporting requirements (i.e. Duke NGD Nuclear Policy Manual, Oconee Nuclear Site Directives, Operating Procedures, etc.) will be used to ensure that the operations involved in the storage of spent fuel at the Oconee Site-Specific ISFSI are performed in a safe manner. This includes both the selection of assemblies qualified for ISFSI storage, and the verification of assembly identification numbers prior to and after placement into individual storage canisters.

10.2.5.1 Qualification of Spent Fuel

Fuel assembly qualification is based on the requirements for criticality control, decay heat removal, structural integrity, and radiological protection.

For the NUHOMS®-24P subcriticality is assured for fuel assemblies meeting the 4.0 wt% initial enrichment limit of Reference 2 when the DSC is filled with water borated to at least 1810 ppm (as required by Reference 2) or when the DSC is drained.

To ensure subcriticality in the postulated event that the DSC is filled with demineralized, unborated water, the burnup requirements of Figure B-48 are specified for any permissible initial enrichment. Three curves are specified. For Single Region Storage, burnup of the IFAs in each of the 24 locations in the DSC must meet or exceed the curve for “Unrestricted” fuel. For Mixed Region Storage, the burnup of the IFAs in the 4 center DSC locations must meet or exceed the curve for “Filler” fuel. Burnup of the IFAs in the remaining 20 DSC locations must meet or exceed the curve for “Restricted” fuel.

Procedures currently in place for special nuclear materials accountability and record keeping are used to verify initial fuel assembly enrichment and burnup levels at discharge. New fuel enrichments and initial uranium isotopics are recorded from the DOE/NRC Form 741's and stored in both a database file and on duplicate paper copies of the Form 741's. Individual fuel assembly burnups are also stored in the special nuclear materials database. These values are generated by the Oconee Operator Aided Computer utilizing thermal energy production data determined by in-core flux mapping. Burnup and initial enrichment values from special nuclear material accountability records are compared to Figure B-48 to verify that the reactivity level is acceptable for DSC loading and storage of each irradiated fuel assembly. Actual qualification procedures may utilize a tabular version of the enrichment-burnup curves which will allow

for each linear interpolation between a number of data points. While this enrichment vs. burnup method for reactivity verification is routinely used as required by procedures, Duke reserves the right to rely on other NRC approved analytical methods to qualify fuel assemblies in special cases.

For decay heat control, only those irradiated assemblies which do not exceed a decay heat level of 0.66 kw qualify for loading into the DSC. Decay heat loadings at or below this level ensure that peak pin clad temperatures are maintained within acceptable levels. Since individual fuel assembly decay heat levels are a function of both the discharge burnup and the decay time, procedural controls are used to verify these parameters prior to fuel assembly loading.

For the Oconee fuel design and routine operating histories, the decay time necessary to achieve a .66 kw decay heat level is generally 7.5 years. The variation in required cooling time is a very strong function of discharge burnup and a very weak function of initial enrichment. It is acceptable to store fuel assemblies cooled less than 10 years provided that decay heat production is no more than 0.66 Kw for each fuel assembly and that neutron and gamma source terms for the DSC are verified not to exceed certain values specified in Reference 2.

As mentioned previously, special nuclear materials accountability records are used to verify fuel assembly burnup. These records are also used to verify spent fuel decay time. The individual assembly burnup and decay time is then compared to Figure B-48 for DSC loading qualification purposes.

To ensure the structural integrity of the spent fuel loaded into the current Site-Specific ISFSI, station records of all damaged assemblies are reviewed. A damaged fuel assembly and component database has been compiled which incorporates previous sipping, ultrasonic (UT) testing, and visual observation. This database is examined as a part of the dry storage qualification process to verify that assemblies with gross structural or gross cladding damage are not included.

If the reactivity, decay heat, cooling time structural integrity, and dose limits criteria are all met, then approval for dry storage for a given assembly will be documented. This documentation will subsequently be referenced through procedures at the station prior to loading fuel into the DSC.

10.2.5.2 Spent Fuel Identification

Administrative controls are utilized to avoid fuel misplacement. Information on fuel assembly qualification for dry storage is documented and transmitted to fuel handling personnel. Prior to any transfer of a fuel assembly in the DSC, specific DSC loading procedures require a review of assembly documentation. This is followed by an independent visual verification of the assembly identification number by two individuals. These procedures ensure that the correct (approved) fuel assembly is being accessed and loaded into the DSC. As a final check, all assembly identification numbers are checked after the DSC has been fully loaded with 24 assemblies.

10.3 Operational Control and Limit Specification

Functional and Operating Limits, Monitoring Instruments and Limiting Control Settings; Limiting Conditions for Operations; and Surveillance Requirements are specified in Reference 2.

10.4 References

1. Topical Report for the Nutech Horizontal Modular Storage (NUHOMS®-24P) System for Irradiated Nuclear Fuel, NUH-002, Revision 1A, July 1989
2. Special Nuclear Materials License SNM-2503, Docket No. 72-4 for the Oconee Independent Spent Fuel Storage Installation

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11.0 Quality Assurance

Duke maintains full responsibility for assuring that its nuclear power plants are designed, constructed, tested and operated in conformance with good engineering practices, applicable regulatory requirements and specified design bases and in a manner to protect the public health and safety. To this end Duke has established and implemented a quality assurance program which conforms to the criteria established in Appendix B to 10CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" and to approved industry standards such as ANSI N45.2-1971 and ANSI N18.7-1976 and corresponding daughter standards, or to equivalent alternatives.

The activities associated with the Independent Spent Fuel Storage Installation (ISFSI) will be governed by the applicable portions of Duke's Quality Assurance Program. This Quality Assurance Program is described in the Topical Report, DUKE-1-A. The Topical Report provides the current quality assurance program description for Oconee, McGuire, and Catawba Nuclear Stations, Docket Nos. 50-269, 50-270, 50-287, 50-369, 50-370, 50-413, and 50-414.

The Topical Report describes the Quality Assurance Program for those systems, components, items, and services which have been determined to be safety related. In addition, Duke's Quality Assurance Program provides a method of applying a graded Quality Assurance Program to certain non-safety related systems, components, items, and services. This method involves defining a Quality Assurance "Condition" for each level of quality assurance required. These will be designated as "QA Condition _________." The following conditions have been defined.

QA Condition 1 covers those systems and their attendant components, items, and services which have been determined to be safety related. These systems are detailed in the Safety Analysis Report applicable to each nuclear station. The Topical Report applies in its entirety to systems, components, items, and services identified as QA Condition 1.

QA Condition 2 covers those systems and their attendant components, items, and structures important to the management and containment of liquid, gaseous, and solid radioactive waste.

QA Condition 3 covers those systems, components, items, and services which are important to fire protection as defined in the Hazards Analysis for each station. The Hazards Analysis is in response to Appendix A of NRC Branch Technical Position APCSB 9.5-1.

QA Condition 4 covers those seismically designed/restrained systems, components, and structures whose continued functions are not required during and after the seismic event. The general scope of these systems, components, and structures, identified as Seismic Category 11 (SC11) are defined in Regulatory Guide 1.29, Seismic Design Classification.

QA Condition 5 covers those systems, components, items, and services which are important to the mitigation of design basis and other selected events as defined in applicable procedures and directives. QA Condition 5 only applies to Oconee Nuclear Station.

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