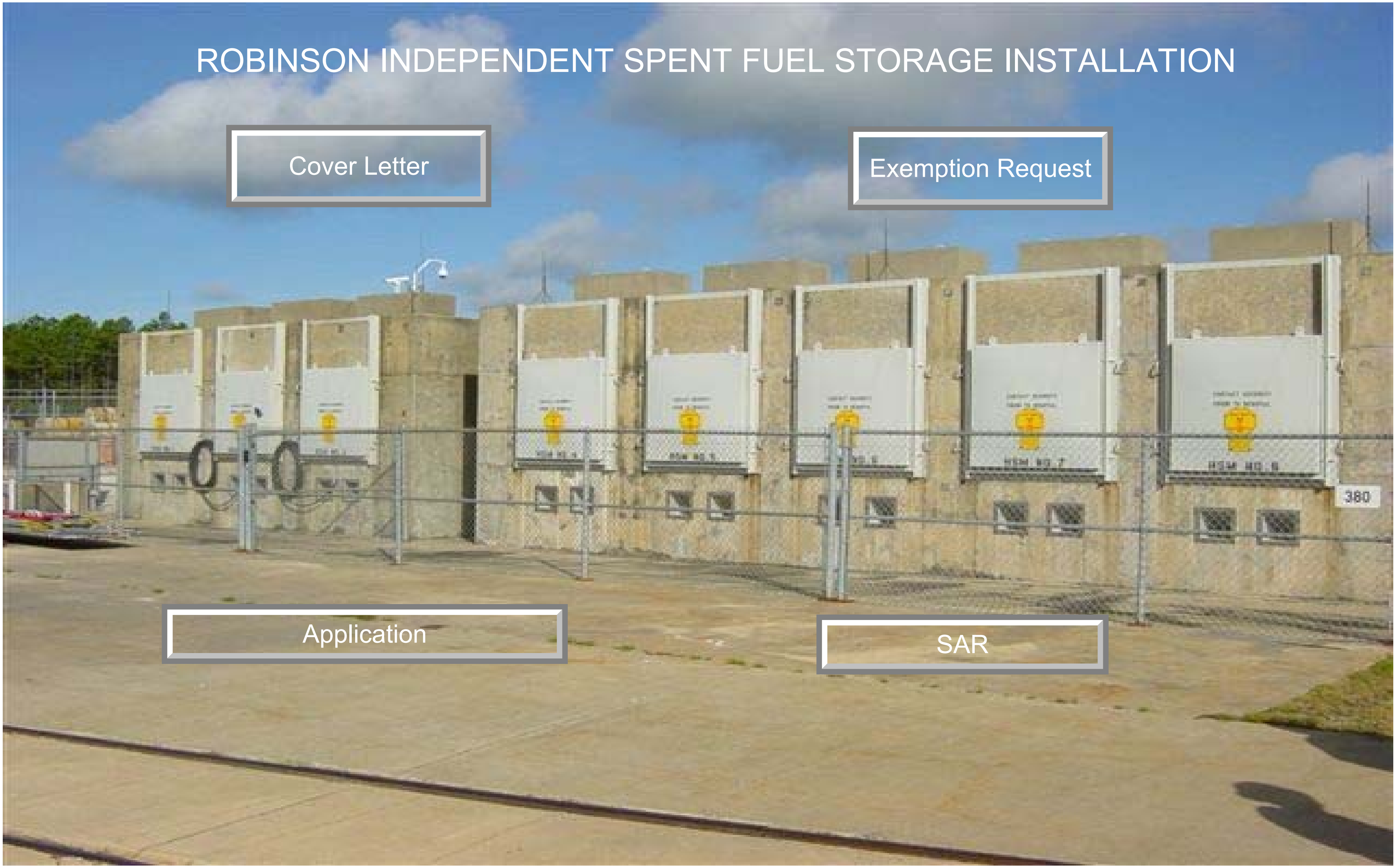
ROBINSON INDEPENDENT SPENT FUEL STORAGE INSTALLATION
Cover Letter
Exemption Request
380
Application
SAR

Progress Energy

Serial: RNP-RA/04-0027

FEB 2 7 2004

ATTN: Document Control Desk
Director, Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
United States Nuclear Regulatory Commission
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
DOCKET NO. 72-3/LICENSE NO. SNM-2502
REQUEST FOR RENEWAL OF
INDEPENDENT SPENT FUEL STORAGE INSTALLATION LICENSE

Ladies and Gentlemen:

Pursuant to 10 CFR 72.42(b) and (c), Carolina Power and Light (CP&L) Company, now doing business as Progress Energy Carolinas (PEC), Inc., hereby submits an application for renewal of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Independent Spent Fuel Storage Installation (ISFSI) license. The current license expires on August 31, 2006. Based on the expected duration of the HBRSEP, Unit No. 2, Operating License and the estimated time needed to remove the storage casks from the site, PEC is requesting a license renewal period of 40 years. An exemption request to support the license renewal period is provided as Attachment II. The application for renewal of the ISFSI license, Attachment III, was prepared in accordance with applicable provisions of 10 CFR 72, Subpart B, and the Preliminary NRC Staff Guidance for 10 CFR Part 72 License Renewal.

The application is being submitted on CD-ROM. It should be noted that the CD-ROM also contains an informational copy of the HBRSEP, Unit No. 2, ISFSI Safety Analysis Report for reference purposes.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom at the letterhead address, phone: 843-857-1253, or e-mail: Chuck.Baucom@pgnmail.com.

Sincerely,

J. F. Lucas

J. F. Lucas
Manager - Support Services - Nuclear

JSK/jsk

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

Attachments:

I. Affirmation
II. Request for Exemption from 10 CFR 72.42(a)
III. Application for Renewed Site-Specific Materials License for the H. B. Robinson Steam Electric Plant, Unit No. 2, Independent Spent Fuel Storage Installation

The enclosed CD-ROM contains a file:
001_RNP_ISFSI_LRA.pdf, approximately 34.5 MB, publicly available

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

AFFIRMATION

The information contained in letter RNP-RA/04-0027 is true and correct to the best of my information, knowledge and belief; and the sources of my information are officers, employees, contractors, and agents of Progress Energy Carolinas, Inc., formerly known as Carolina Power and Light Company. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 27 Feb. 2004

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

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
REQUEST FOR EXEMPTION FROM 10 CFR 72.42(a)

In accordance with the provisions of 10 CFR 72.7, "Specific exemptions," Carolina Power and Light (CP&L) Company, now doing business as Progress Energy Carolinas (PEC), Inc., requests an exemption from certain requirements of 10 CFR 72.42, "Duration of License; Renewal." Specifically, PEC requests exemption from the Independent Spent Fuel Storage Installation (ISFSI) license renewal period in 10 CFR 72.42(a), which the NRC has interpreted in a November 7, 2000, letter from Mr. E. William Brach, NRC, to Mr. W. R. Matthews, Virginia Electric and Power Company, to be 20 years. PEC is requesting a license renewal period of 40 years.

The spent fuel pool at H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, is nearing usable storage capacity. Operation of HBRSEP, Unit No. 2, is now dependent on the continued ability of the ISFSI to store spent fuel, and the construction of an additional ISFSI. Spent fuel storage at the ISFSI is, and will continue to be, necessary since the United States Department of Energy (DOE) has not begun, and is not soon to be able, to take spent nuclear fuel as it was required to do under the provisions of the Nuclear Waste Policy Act. Therefore, the license renewal period for the ISFSI must consider future operation of HBRSEP, Unit No. 2. Absent approval of an extended period for license renewal, PEC will be required to request a second license renewal for the ISFSI over the course of plant operations. This would be an unnecessary diversion of PEC and NRC resources.

This request for an exemption is based on the need for a longer license renewal period and is supported by a technical justification that demonstrates the ability of the ISFSI to safely perform its intended function for a 40 year license renewal period.

Need For 40 Year License Renewal Period

The current Operating License for HBRSEP, Unit No. 2, expires on July 31, 2010. PEC applied for a 20 year renewal of the HBRSEP, Unit No. 2, Operating License on June 14, 2002, and assuming that license renewal is granted, the renewed Operating License would expire on July 31, 2030.

The current ISFSI license expires on August 31, 2006. A license renewal of only 20 years would expire on August 31, 2026, which is approximately four years before the plant Operating License is expected to expire. During that time, PEC anticipates that the ISFSI will be required for spent fuel storage in concert with the HBRSEP, Unit No. 2, spent fuel pool, and the additional ISFSI currently being planned. In addition, it is assumed that the spent fuel pool will be emptied prior to all fuel being removed from the ISFSI. Therefore, the ISFSI license will need to be retained until the last fuel assembly is shipped offsite to a permanent storage facility.

Based on the assumption that the DOE will begin taking spent fuel in 2010, it is unlikely that the HBRSEP, Unit No. 2, spent fuel pool would be emptied before 2026. Further, assuming that the HBRSEP, Unit No. 2, Operating License is renewed, the spent fuel shipments would not be completed until an even later date. PEC is requesting a renewal period of 40 years, which would allow the ISFSI to continue to store spent fuel until 2046.

Technical Justification

The technical justification that the ISFSI will be able to fulfill its safety functions over a license renewal period of 40 years is provided in the application for renewed site-specific license, which is included as Attachment III. The ISFSI license renewal application (LRA) addresses the applicable provisions of 10 CFR 72, Subpart B, as required by 10 CFR 72.42(b). The systems, structures, and components (SSCs) that were within the scope of license renewal, and the required evaluations, were identified. Aging management reviews were performed on these SSCs to determine the materials and environments to which these SSCs are exposed, as well as any aging effects requiring aging management. Time-limited aging analyses were also performed to determine time-limited aging effects on the SSCs within the license renewal scope. Aging management activities were identified that provide reasonable assurance that SSCs within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the renewal period. This is consistent with the request for renewal of the HBRSEP, Unit No. 2, Operating License for a total operational period of 60 years.

Requirements of Section 72.7

The specific requirements for granting an exemption from 10 CFR 72 regulations are set forth in 10 CFR 72.7. Under 10 CFR 72.7, the NRC is authorized to grant an exemption upon demonstration that the exemption: (i) is authorized by law, (ii) will not endanger life or property or the common defense and security, and (iii) is in the public interest. The following addresses each of these requirements and demonstrates that the NRC should grant the exemption request.

A. The Exemption Request is Authorized by Law

The NRC's authority to grant an exemption from 10 CFR 72 is established by law as discussed in 10 CFR 72.7. Therefore, granting an exemption is explicitly authorized by the NRC's regulations.

B. The Exemption Request Will Not Endanger Life or Property or the Common Defense and Security

Continued operation does not endanger life or property, as discussed in the Environmental Report Supplement, which is provided as Appendix E to the ISFSI LRA (Attachment III). A 40 year license renewal period has been evaluated in the ISFSI LRA and it has been determined that new and existing monitoring activities provide reasonable assurance that SSCs within the scope of license renewal will continue to perform their intended functions. The common defense and security of the United States is not endangered by the renewal of the ISFSI license for 40 years.

A 40 year ISFSI license renewal period will support continued operation of HBRSEP, Unit No. 2, until 2030, assuming plant license renewal is granted. Since the spent fuel pool at HBRSEP, Unit No. 2, is nearly at usable storage capacity, continued operation of the plant is dependent on an operational ISFSI. The continued safe operation of nuclear power plants, including HBRSEP, Unit No. 2, enhances the common defense and security of the United States by providing dependable, low-cost electricity.

C. The Exemption is in the Public Interest

The subject exemption would allow HBRSEP, Unit No. 2, to continue operation for the duration of the proposed renewed Operating License (until 2030) without having to repeat the ISFSI license renewal process. The granting of this exemption would conserve both PEC and NRC resources, permitting more focused attention to areas of nuclear safety significance.

Conclusion

The requested exemption from the NRC's accepted 20 year ISFSI license renewal period has no adverse impact on safety. Since operation of HBRSEP, Unit No. 2, is now dependent on the continued ability of the ISFSI to store spent fuel, the ISFSI must be available to store spent fuel for the expected duration of the plant Operating License. In addition, the ISFSI must be available to store spent fuel until the last fuel assembly is removed from the site. The ISFSI is, therefore, required until the DOE is able to accept all of the spent fuel stored at HBRSEP, Unit No. 2.

Since there is a clear need for the ISFSI, subsequent renewal of the license for a third period is an unnecessary use of PEC and NRC resources. Technical justification provided in the ISFSI LRA establishes that new and existing monitoring activities provide reasonable assurance that SSCs within the scope of license renewal will continue to perform their intended functions.

Therefore, because the requested exemption for the ISFSI license renewal period is authorized by law, will not endanger life or property or the common defense and security, is in the public interest, and is requested for good cause, PEC requests that, in accordance with the provisions of 10 CFR 72.7, the NRC grant the requested exemption.

Attachment III

APPLICATION FOR RENEWED SITE-SPECIFIC MATERIALS LICENSE for the

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2, INDEPENDENT SPENT FUEL STORAGE INSTALLATION

ACRONYMS AND ABBREVIATIONS	
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AMA	Aging Management Activity
AMP	Aging Management Program
AMR	Aging Management Review
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CASS	Cast Austenitic Stainless Steel
CLB	Current Licensing Basis
CFR	Code of Federal Regulations
CP&L	Carolina Power & Light Company, a Progress Energy Company
CUF	Cumulative Usage Factor
DOE	U. S. Department of Energy
DSC	Dry Shielded Canister (or Dry Storage Canister/Cask)
DU	Depleted Uranium
EPRI	Electric Power Research Institute
ER	Environmental Report
GE	General Electric
HBRSEP, Unit No. 2 (also HBR2)	H. B. Robinson Steam Electric Plant, Unit No. 2 (also referred to as Robinson Nuclear Plant)
HNP	Shearon Harris Nuclear Plant
HSM	Horizontal Storage Module
IFA	Irradiated Fuel Assembly
IF-300	Series of Duratek, Inc. (formerly General Electric) shipping casks
INPO	Institute of Nuclear Power Operations
ISFSI	Independent Spent Fuel Storage Installation
LR	License Renewal
LRA	License Renewal Application
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NUHOMS	NUTECH, Inc. Horizontal Modular Storage (system)
OE	Operating Experience
PEC	Progress Energy Carolinas, Inc.
ONS	Oconee Nuclear Station
RNP	Robinson Nuclear Plant
SAR	Safety Analysis Report
SFP	Spent Fuel Pit (Pool)
SER	Safety Evaluation Report
SNF	Spent Nuclear Fuel
SCs	Structures and Components
SSC	System, Structure and Component
SPS	Surry Power Station
UFSAR	Updated Final Safety Analysis Report

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APPENDICES:

LIST OF TABLES:

1.0 GENERAL INFORMATION

Carolina Power & Light (CP&L) Company, now doing business as Progress Energy Carolinas, Inc. (PEC), has prepared this application for renewal of the site-specific license for the Independent Spent Fuel Storage Installation (ISFSI) located at the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, also referred to as the Robinson Nuclear Plant (RNP). This application supports license renewal for an additional 40 year period beyond the end of the current license term of Materials License Number SNM-2502 (Docket No. 72-3). The original 20 year license will expire on August 31, 2006. This application includes the applicable general, technical, and environmental supporting information required by 10 CFR 72.42(b).

The information contained in this section includes:

1. Information on the organization of the application (Section 1.1),
2. A general description of the RNP site ISFSI facility (Section 1.2),
3. The administrative information required by 10 CFR 72.22 (Section 1.3),
4. Summary of abbreviations and intended function code definitions (Section 1.4), and
5. A list of the references for Section 1.0, General Information (Section 1.5).

1.1 APPLICATION FORMAT AND CONTENT

The format and content of the application are based on the preliminary guidance for renewal of site-specific 10 CFR 72 licenses and on the precedent of the Surry ISFSI, referred to as "Surry Power Station," license renewal application (TAC Nos. L23455 and L23456), and include:

1. General Information – Section 1.0 has been expanded beyond the general administrative requirements of 10 CFR 72.22 to provide (1) information on the format and content of the application, (2) general facility description, and (3) a summary of abbreviations and intended function code definitions used in the application.
2. Scoping Evaluation - Section 2.0 provides the scoping evaluation for the ISFSI systems, structures, and components (SSCs).
3. Aging Management Reviews – Section 3.0 includes the methodology and results of the aging management reviews (AMRs) performed for ISFSI SSCs that are in the scope of license renewal.

4. Appendices:

   Appendix A: Aging Management Programs

   Appendix B: Time-Limited Aging Analyses (TLAAs)

   Appendix C: Safety Analysis Report (SAR) Supplement and Changes

   Appendix D: Technical Specifications Changes

   Appendix E: Environmental Report Supplement

   Appendix F: Additional Information (training and qualifications, financial assurance for decommissioning and emergency planning)

1.2 FACILITY DESCRIPTION

The ISFSI is located on the H. B. Robinson Steam Electric Plant (RNP) site near Hartsville, South Carolina. PEC owns and operates a 2339 MWt nuclear generating unit (Unit No. 2) and a 185 MWe fossil-fueled generating unit (Unit No. 1) on the Robinson site. The ISFSI is located within the nuclear unit protected area, approximately 600 ft. west of the Unit No. 2 containment building.

The 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 steel dry shielded canister (DSC) with an internal basket which holds the IFAs. Each HSM contains one DSC and each DSC contains seven fuel assemblies. The outer, exposed walls of each module are 3 1/2 feet thick concrete to provide the necessary shielding. A total of eight modules have been built and operated at the RNP site. The initial phase of construction included three modules. An additional five modules were constructed nearby, but separate from the initial three. A complete description of the RNP ISFSI is provided in the Independent Spent Fuel Storage Installation Safety Analysis Report (ISFSI SAR) (Reference 1.5-2).

In addition to these primary components, the ISFSI also required 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 consists of a transfer cask, a hydraulic ram, a tow vehicle, a trailer and a cask skid. The transfer system interfaces with the existing spent fuel pool, the cask crane, the site layout (i.e., roads and topography) and is controlled by procedural requirements.

The H. B. Robinson Steam Electric Plant, Unit No. 2, is operated under a separate license, DPR-23 (Docket 50-561), issued pursuant to the provisions of 10 CFR 50 and is, therefore, not addressed in this application. However, the cessation of the Part 50 license would not obviate meeting the ISFSI license requirements for retrieval, which are presently satisfied by the cask handling systems and structures (e.g., spent fuel pool, decontamination area, cask crane, etc.) per Section 5.1.1.8 of the ISFSI SAR (Reference 1.5-2).

1.3 INFORMATION REQUIRED BY 10 CFR 72.22

1.3.1 NAME OF APPLICANT

Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc. (PEC), a Progress Energy Company

1.3.2 ADDRESS OF APPLICANT

Progress Energy Carolinas, Inc.
411 Fayetteville Street
Raleigh, NC 27601-1748

1.3.3 ADDRESS OF THE RNP ISFSI

Progress Energy Carolinas, Inc.
3581 West Entrance Road
Hartsville, SC 29550

1.3.4 DESCRIPTION OF BUSINESS OR OCCUPATION OF APPLICANT

Progress Energy Carolinas, Inc., a subsidiary of Progress Energy, Inc., is a corporation primarily engaged in the generation, transmission, distribution, and sale of electricity in portions of North and South Carolina. Progress Energy Carolinas, Inc., (the Company) serves more than 1.3 million customers in a territory encompassing over 34,000 square miles including the cities of Raleigh, Wilmington, Fayetteville, and Asheville in North Carolina; and Florence and Sumter in South Carolina.

1.3.5 ORGANIZATION AND MANAGEMENT OF APPLICANT

Progress Energy, Inc., is a registered holding company under the Public Utility Holding Company Act of 1935, as amended. The Company is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The Company makes this application on its own behalf and is not acting as an agent or representative of any other person.

The names and addresses of Progress Energy directors and principal officers are listed below. All persons listed are U. S. citizens.

Directors	Address
Edwin B. Borden	Goldsboro, NC
James E. Bostic, Jr.	Atlanta, GA
David L. Burner	Charlotte, NC
William Cavanaugh III	Raleigh, NC
Charles W. Coker	Hartsville, SC
Richard L. Daugherty	Raleigh, NC
W. D. Frederick, Jr.	Orlando, FL
William O. McCoy	Chapel Hill, NC
E. Marie McKee	Corning, NY
John H. Mullin III	Brookneal, VA
Richard A. Nunis	Orlando, FL
Peter S. Rummel	Jacksonville, FL
Carlos A. Saladrigas	Miami, FL
J. Tylee Wilson	Ponte Vedra Beach, FL
Jean Giles Wittner	St. Petersburg, FL

Principal Officers	Address
William Cavanaugh III Chairman and Chief Executive Officer	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
Robert B. McGehee President and Chief Operating Officer	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748

Principal Officers	Address
William D. Johnson Group President – Energy Delivery	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
Tom D. Kilgore President and Chief Executive Officer, Group President – Progress Ventures	Progress Energy Ventures, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
William S. Orser Group President - Energy Supply	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
Don K. Davis Executive Vice President – Rail and Telecom	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
Peter M. Scott III President and Chief Executive Officer	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748
H. William Habermeyer, Jr. President and Chief Executive Officer	Progress Energy Florida, Inc. 100 Central Avenue St Petersburg, Fl 33701-3324
Fred N. Day IV President and Chief Executive Officer	Progress Energy Carolinas, Inc. 411 Fayetteville Street Raleigh, NC 27601-1748
Geoffrey S. Chatas Executive Vice President and Chief Financial Officer – Financial Services	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748
Brenda F. Castonguay Senior Vice President - Administrative Services	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748
John R. McArthur Senior Vice President and General Counsel – Corporate Relations	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748

Principal Officers	Address
Jeffrey J. Lyash Senior Vice President – Energy Delivery	Progress Energy Florida, Inc. 100 Central Avenue St Petersburg, FL 33701-3324
Bonnie V. Hancock President – Progress Fuels	Progress Energy Ventures, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
C. S. Hinnant Senior Vice President and Chief Nuclear Officer - Nuclear Generation	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
E. Michael Williams Senior Vice President - Power Operations	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748

1.3.6 Financial Qualifications

As required by 10 CFR 72.22(e), Progress Energy Carolinas, Inc., will remain financially qualified to carry out the operation and decommissioning of the HBRSEP, Unit No. 2, ISFSI during the period of the renewed operating license.

Data, including corporate annual reports, to support the conclusion that Progress Energy is financially qualified to operate the ISFSI may be accessed at the following website:

http://www.progress-energy.com/investors/financials/index.asp

1.4 ABBREVIATIONS AND INTENDED FUNCTION CODE DEFINITIONS

1.4.1 Abbreviations

The acronyms and abbreviations that pertain to the administrative and technical information in this application, Appendices A through D, and Appendix F are listed prior to the Table of Contents. The abbreviations that pertain to the environmental information are included in the front of Appendix E, Environmental Report Supplement.

1.4.2 Intended Function Code Definitions

This section contains the meanings for the subcomponent intended function represented by the abbreviations used in subsequent sections of this application, including Table 3.2-1 through Table 3.5-1. Subcomponent intended functions are the specific functions that support the intended function of the structure and component of which they are a part.

Abbreviation	**Definition**
CC	Provides criticality control of spent fuel
HT	Provides heat transfer
PB	Directly or indirectly maintains a pressure boundary (confinement)
SH	Provides radiation shielding
SS	Provides structural support and/or functional support of important to safety equipment (structural integrity)

1.5 REFERENCES (SECTION 1.0, GENERAL INFORMATION)

1.5-1 *10 CFR, Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste*, Code of Federal Regulations, U.S. Nuclear Regulatory Commission, 1988

1.5-2 *Independent Spent Fuel Storage Installation Safety Analysis Report* (ISFSI SAR), H.B. Robinson Steam Electric Plant, through Amendment 18

2.0 SCOPING EVALUATION

2.1 INTRODUCTION

Progress Energy Carolinas, Inc., used the pilot 10 CFR 72 ISFSI license renewal process between Dominion (Virginia Electric and Power Company) and the Nuclear Regulatory Commission (NRC), for the Surry Power Station (SPS) Independent Spent Fuel Storage Installation (ISFSI), to develop the RNP ISFSI license renewal process. A general description of the RNP ISFSI facility is provided in Section 1.2, Facility Description. A more thorough description of the ISFSI facility is contained in the ISFSI SAR (Reference 2.4-4).

RNP's ISFSI license renewal methodology follows the comments on the Preliminary Guidance for License Renewal for Site-Specific Independent Spent Fuel Storage Installations (ISFSIs) (Reference 2.4-1) that were provided to the NRC on June 26, 2001 (Reference 2.4-2) by Dominion (Virginia Electric and Power Company). The proposed Part 72 license renewal process adopts the regulatory philosophy of 10 CFR 54, The License Renewal Rule (Reference 2.4-3). This philosophy is summarized in the two principles of license renewal from the Part 54 Final Rule Statements of Consideration published in the Federal Register, Vol. 60, No. 88, May 8, 1995, page 22464, and re-stated below:

> *"The first principle of license renewal was that, with the exception of age-related degradation unique to license renewal and possibly a few other issues related to safety only during the period of extended operation of nuclear power plants, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provides and maintains an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security. Moreover, consideration of the range of issues relevant only to extended operation led the Commission to conclude that the detrimental effects of aging is probably the only issue generally applicable to all plants. As a result, continuing this regulatory process in the future will ensure that this principle remains valid during any period of extended operation if the regulatory process is modified to address age-related degradation that is of unique relevance to license renewal.*
>
> *The second and equally important principle of license renewal holds that the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term. This principle would be accomplished, in part, through a program of age-related degradation management for systems, structures, and components that are important to license renewal ...."*

Based on these principles, license renewal is not intended to impose requirements beyond those that were met by the facility when it was initially licensed by the NRC. Therefore, the current licensing basis (CLB) for the ISFSI will be carried forward through the renewed license period.

During scoping, the systems, structures, and components (SSCs) of the ISFSI that are within the scope of license renewal, and required evaluation for the effects of aging, were identified. A description of the scoping process is provided in Section 2.2, Scoping Methodology.

2.2 SCOPING METHODOLOGY

The first step in the license renewal process involved the identification of the in-scope ISFSI SSCs. This was done by evaluating the SSCs that comprise the ISFSI against the following scoping criteria provided in the comments on the Preliminary Guidance for License Renewal for Site-Specific Independent Spent Fuel Storage Installations (ISFSIs) (Reference 2.4-2):

Any SSC that meets either of the criteria shall be evaluated further in the aging management review (AMR) process described later. The categories of SSCs are those that are:

1. *Important to safety; as they are relied upon to:*

    a) *Maintain the conditions required to store spent fuel safely,*
    b) *Prevent damage to the spent fuel during handling and storage.*
    c) *Provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public, as identified in the current licensing basis (CLB).*

    These SSCs ensure that these important safety functions are met: (1) criticality, (2) shielding, (3) confinement, (4) heat transfer, and (5) structural integrity.

2. *Classified as not important to safety, but, according to the CLB, whose failure could prevent an important to safety function from being fulfilled or whose failure as a support SSC could prevent an important to safety function from being fulfilled.*

The function performed by an SSC that causes it to be within the scope of license renewal is its intended function.

Also, SSCs which perform ISFSI support functions are generally not within the scope of license renewal. The fuel in storage is considered to be within the scope of license renewal.

Any ISFSI SSC that met either Scoping Criterion 1 or 2 above was determined to be within the scope of license renewal (in-scope), and the function(s) it is required to perform during the extended term was identified. The results of the Scoping evaluation are presented in Section 2.3.

The basic premise of the license renewal scoping process is that the CLB determines which SSCs perform intended functions that meet either Scoping Criterion 1 or 2, as defined above. The following documents comprise the ISFSI CLB:

- Safety Analysis Report (Reference 2.4-4),
- Materials License No. SNM – 2502 (Reference 2.4-6),
- Technical Specifications (Reference 2.4-6), and
- Docketed Licensing Correspondence

The ISFSI SAR provides a description of the ISFSI facility, ISFSI SSCs and their functions, including safety classifications as established by the safety analyses. The Technical Specifications govern the safety of, the receipt, possession, and storage of irradiated nuclear fuel at the ISFSI, and the transfer of such irradiated fuel to and from H. B. Robinson Steam Electric Plant, Unit No. 2, and the ISFSI.

Additionally, the Safety Evaluation Report (Reference 2.4-5), which summarizes the results of the NRC staff's safety review of the original licensing, and the SERs associated with subsequent amendments were used in the license renewal scoping process.

Other design and design basis documents such as Topical Reports (References 2.4-7, 2.4-8, and 2.4-9) were consulted as appropriate to further clarify SSC descriptions, classifications, and intended functions. Each Topical Report contained a corresponding Safety Evaluation Report (SER).

2.3 SCOPING RESULTS

The SSCs comprising the ISFSI are identified in Table 2.3-1, Scoping Results. Those SSCs meeting Scoping Criterion 1 or 2 are identified in the table as being within the scope of license renewal.

As indicated in Table 2.3-1, only the ISFSI horizontal storage modules (including the attached lightning protection system), IF-300 transfer cask, dry shielded canisters, and the spent fuel assemblies stored in the canisters were determined to be within the scope of license renewal and to require further review in the aging management review process. The intended functions performed by the individual subcomponents of these in-scope SSCs are identified in the aging management review summary tables (Tables 3.2-1, 3.3-1, 3.4-1, and 3.5-1, respectively), which are located at the end of Section 3.0, Aging Management Reviews.

Table 2.3-1 Scoping Results

SSC [1]	Criterion 1	Criterion 2	In-Scope
Horizontal Storage Modules (HSMs) [2]	N	Y	Y
Dry Shielded Canisters (DSCs)	Y	N/A	Y
Irradiated Fuel Assemblies (IFAs)	Y	N/A	Y
IF-300 Transfer Cask	Y	N/A	Y
Transfer Components	N	N	N
Instrumentation	N	N	N
Security Fence & Gates	N	N	N
Approach Concrete Slab, Miscellaneous Concrete, and Joint Material	N	N	N
Observation Platform	N	N	N
Lighting	N	N	N

(1) See Tables 3.2-1 through 3.5-1 for subcomponent intended function(s).
(2) Includes the attached lightning protection system.

Y – Yes
N – No
N/A – Not Applicable

2.4 REFERENCES (SECTION 2.0, SCOPING EVALUATION)

2.4-1 Letter from Mr. Steven Baggett, NRC, to Mr. John Moyer, CP&L, Serial No. RRA-01-0054, *Preliminary NRC Staff Guidance for 10 CFR Part 72 License Renewal*, May 17, 2001

2.4-2 Letter from Mr. L. N. Hartz, Dominion, to NRC Document Control Desk, Serial No. 01-367, *Surry Independent Spent Fuel Storage Installation, Comments on NRC Preliminary Guidance,* June 26, 2001

2.4-3 *10 CFR Part 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants*, Code of Federal Regulations, U. S. Nuclear Regulatory Commission, 1995

2.4-4 *Independent Spent Fuel Storage Installation Safety Analysis Report*, H.B. Robinson Steam Electric Plant, through Amendment 18

2.4-5 *Safety Evaluation Report of H.B. Robinson Steam Electric Plant Unit No. 2 Independent Spent Fuel Storage Installation*, U.S. Nuclear Regulatory Commission, Docket 72-3, June 1986

2.4-6 *Independent Spent Fuel Storage Installation Material License No. SNM-2502*, H.B. Robinson Steam Electric Plant, through Amendment 13, December 3, 2003, Appendix A (Technical Specifications)

2.4-7 *Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel*, NUH-001, Pacific Nuclear Fuel Services, Inc., San Jose, California, March 1990, Revision 2

2.4-8 Letter from NRC Spent Fuel Project Office to Duratek, *Certificate of Compliance for Radioactive Materials Package*, Subject: Model No. IF-300 Package, Docket Number 71-9001, December 2, 2002

2.4-9 *IF-300 Shipping Cask Consolidated Safety Analysis Report*, NEDO-10084-5, Duratek, November 2000

3.0 AGING MANAGEMENT REVIEWS

3.1 AGING MANAGEMENT REVIEW METHODOLOGY

The scoping process identified the ISFSI SSCs within the scope of license renewal which require evaluation for the effects of aging in the AMR process.

The purpose of the AMR is to assess the in-scope SSCs with respect to aging effects that could affect the ability of the SSC to perform its intended function during the renewed license period. The AMR process involved the following four (4) major steps:

(1) Identification of in-scope subcomponents requiring AMR (Screening)

(2) Identification of materials and environments

(3) Identification of aging effects requiring management

(4) Determination of the activities/programs required to manage the effects of aging

Each of these steps is discussed in Subsections 3.1.1 through 3.1.4, respectively. Also, the operating experience review for confirmation of the AMR process and the document sources used in the process are discussed in Subsections 3.1.5 and 3.1.6, respectively.

The results of the aging management review for the subcomponents of the ISFSI SSCs that are in the scope of license renewal are provided in the following sections:

- Section 3.2, Aging Management Review Results – HSMs,
- Section 3.3, Aging Management Review Results – DSCs,
- Section 3.4, Aging Management Review Results – IFAs, and
- Section 3.5, Aging Management Review Results – IF-300 Transfer Cask

Corresponding tables that summarize the aging management review for these ISFSI SSCs are located at the end of Section 3.0, Aging Management Reviews.

3.1.1 IDENTIFICATION OF IN-SCOPE SUBCOMPONENTS REQUIRING AMR

The scoping process did not identify the specific subcomponents for the in-scope ISFSI SSCs that require AMR. Therefore, during the first step in the AMR process, the in-scope SSCs were further reviewed to identify and describe the subcomponents that support the SSC intended function. The subcomponents and associated intended functions have been identified by reviewing the documentation sources identified in Subsection 3.1.6.

Subcomponents that perform or support any one of the identified intended functions in a passive manner, without moving parts or a change in configuration or properties, were determined to require an aging management review.

Those subcomponents that either do not support an intended function, or perform an intended function by a change in configuration or properties (active), or have their condition monitored at some established frequency, were excluded from further evaluation in the aging management review with supporting justification.

Tables 3.2-1, 3.3-1, 3.4-1, and 3.5-1, respectively, identify the intended functions for the ISFSI subcomponents that required aging management review. The tables also identify subcomponents that did not support the SSC intended function and are not subject to aging management review.

3.1.2 Identification of Materials and Environments

The second step of the AMR process involved the identification of the materials of construction and the environments to which these materials are exposed, for the ISFSI subcomponents that required an AMR.

The materials of construction have been identified through a review of pertinent design and/or design basis documents, which are discussed in Subsection 3.1.6. A summary of the materials of construction is provided in Subsection 3.2.2 for the HSM subcomponents, Subsection 3.3.2 for the DSC subcomponents, Subsection 3.4.2 for the IFA subcomponents, and Subsection 3.5.2 for the IF-300 Transfer Cask subcomponents. The materials of construction are also reflected in the corresponding aging management review summary tables (Tables 3.2-1, 3.3-1, 3.4-1, and 3.5-1, respectively).

The environments to which components are exposed play a critical role in the determination of potential aging mechanisms and effects. A review of plant documentation, discussed in Subsection 3.1.6, was performed to quantify the environmental conditions to which the ISFSI SSCs are continuously or frequently exposed. The environmental conditions identified during this review include any conditions known to exist on a recurring basis. They are based on operating experience, unless design features have been implemented to preclude those conditions from recurring. Descriptions of the internal and external environments, which have been used in the aging management review, are included in Subsections 3.2.3, 3.3.3, 3.4.3, and 3.5.3, respectively, and are reflected in the corresponding aging management review summary table.

3.1.3 IDENTIFICATION OF AGING EFFECTS REQUIRING MANAGEMENT

The third step in the AMR process involved the identification of the aging effects requiring management. Aging effects requiring management during the renewed license period are those that could cause a loss of passive SSC intended function(s). If degradation of a subcomponent would be insufficient to cause a loss of function, or the relevant conditions do not exist at the RNP ISFSI for the aging effect to occur and propagate, then no aging management is required.

Potential aging effects, presented in terms of material and environment combinations, have been evaluated and those aging effects requiring management determined. Both potential aging effects that theoretically occur, as well as aging effects that have actually occurred based upon industry and RNP operating experience, were considered. The evaluation was applied to subcomponents, regardless of form (i.e., canister body, cover, lid, guide tube, etc.).

As described above, the environments considered in this evaluation are the environments that the subcomponents normally experience. Environmental stressors that are conditions not normally experienced (such as accident conditions), or that may be caused by a design problem, are considered event-driven situations and have not been characterized as sources of aging. Such event-driven situations would be evaluated and corrective actions, if any, implemented at the time of the event.

Aging effects are the manifestation of aging mechanisms. In order to effectively manage an aging effect, it was necessary to determine the aging mechanisms that are potentially at work for a given material and environment application. Therefore, the AMR process addressed both the aging effects and the associated aging mechanisms. Various mechanisms are only applicable at certain conditions, such as high temperature or moisture, for example. Each identified mechanism was characterized by a set of applicable conditions that must be met for the mechanism to occur and/or propagate. Given this evaluation process, each ISFSI subcomponent that was subjected to AMR was evaluated to determine if the potential aging effects/mechanisms were credible considering the material, environment, and conditions of storage.

The compilation of potential aging effects/mechanisms that were considered by PEC for the RNP ISFSI was based largely on experience from the RNP license renewal (10 CFR Part 54) process, the EPRI Dry Cask Characterization Study, and RNP ISFSI operating experience. The majority of potential aging mechanisms were extracted from various industry documents (e.g., EPRI, NRC) for the material/environment combinations applicable to RNP ISFSI subcomponents. The EPRI Dry Cask Characterization Project final report, TR-1002882 (Reference 3.6-6), and associated documents were the primary source for fuel assembly (and DSC internals) related aging mechanism evaluations.

A summary of aging effects requiring management for the subcomponents of the HSMs, DSCs, IFAs, and the IF-300 Transfer Cask is provided in Subsections 3.2.4, 3.3.4, 3.4.4, and 3.5.4, respectively. The aging effects that require management during the renewed license period are also reflected in the corresponding aging management review summary tables.

3.1.4 Determination of the Activities Required to Manage the Effects of Aging

The final step in the AMR process involved the determination of the Aging Management Activities (AMAs) or Aging Management Programs (AMPs) to be credited or developed for managing the effects of aging. To the extent practical, existing ISFSI programs and/or activities were credited for the management of aging effects that could cause a loss of component intended function during the renewed license period.

As indicated in Subsections 3.3.4 and 3.4.4, and reflected in the corresponding aging management review summary tables (Table 3.3-1 and Table 3.4-1), there are no aging effects requiring management during the renewed license period for the subcomponents of the DSCs or IFAs during extended storage. The AMPs for the management of aging of the pertinent HSM and IF-300 Transfer Cask subcomponents are described in Subsections 3.2.5 and 3.5.5, respectively, and are listed in the corresponding aging management review summary tables (Table 3.2-1 and Table 3.5-1).

The demonstration of the effectiveness of the AMPs that were selected for the HSMs and the IF-300 Transfer Cask is discussed in Appendix A, Aging Management Programs (Sections A.2.1 and A.2.2, respectively).

3.1.5 Operating Experience Review for Process Confirmation

As described in Subsection 3.1.3, the potential aging effects for RNP ISFSI material and environment combinations were compiled from common industry and plant operating experience through the use of accepted industry standards and reference materials, including various metallurgical literary references relating specific materials and environments to aging effects and mechanisms. These aging effects/mechanisms were evaluated, as described above, based on the premise that similar materials in similar environments experience similar aging effects and mechanisms.

A further review of industry and plant-specific operating experience for the ISFSI was also performed in order to confirm the applicability of previously identified potential aging effects/mechanisms and to identify any aging effects not previously addressed in aging effect evaluations. In addition, the site-specific operating experience for another Progress Energy site (Harris Nuclear Plant) was reviewed with respect to use of the IF-300 Cask series for offsite shipments of fuel (10 CFR 71).

This ISFSI operating experience review is described in the following subsections. A discussion of operating experience, as it pertains to the effectiveness of AMPs credited with the management of aging, is also contained in the appropriate subsections of Appendix A, Aging Management Programs.

The RNP ISFSI design employs a rugged sealed canister and concrete overpack for dry storage of irradiated fuel, as described in this application. However, for completeness, operating experience for dry storage casks and for shipping casks was also considered for applicability to the RNP ISFSI SSCs.

3.1.5.1 *Cask/Canister Degradation*

Industry operating experience for both shipping and storage casks is primarily focused on loading and closure of the shipping casks and storage casks in, or just removed from, the spent fuel pool environment, rather than in storage. However, cracking of basket welds, degradation/loosening of closure bolts, and leakage through the metal O-ring for the secondary lid of a storage cask have been identified for the type of dry storage casks employed at the Surry Power Station (SPS).

The cracking of basket welds was identified as part of the Dry Cask Storage Characterization project conducted by the Electric Power Research Institute (EPRI). As a result of the testing configurations for the demonstration cask, unusually high compressive stresses that would not be expected during extended storage were experienced. Additionally, EPRI concluded that the cracked welds did not affect the cask safety. Thermal stresses were considered in the design of the DSCs used at the RNP ISFSI. No excessive design or event related stresses were considered in the aging management reviews for the DSCs, which was based on expected temperatures during the extended storage.

Also, the DSCs employed at the RNP are seal-welded rather than bolted closures with O-rings, but the OE does support that wetted surfaces are the most susceptible to degradation. With respect to the IF-300 Cask, degradation of bolting and closures is not considered to be aging related, but event-driven with prompt repairs, except for prolonged or intermittently wetted surfaces.

Furthermore, the only identified instance of the possible degradation of a sealed canister was for the Oconee Nuclear Station. The impact of plastic foreign material in a sealed DSC was evaluated and determined to pose no concern because of the small quantity of material and the inert environment.

3.1.5.2 *Fuel Assembly Degradation*

The conditions and findings of the Dry Cask Storage Characterization Project conducted by EPRI are considered to be representative of the conditions and materials inside a DSC at the RNP ISFSI, as described in Section 3.4, Aging Management Review Results for the IFAs. This project evaluated the condition of fuel rods following approximately 15 years of dry storage, and found no evidence of significant degradation of the cask systems (including internal basket and fuel assemblies) important to safety, no signs of air ingress into the container or signs of cladding failure, no evidence of major crud spallation from the fuel rod surfaces, and that all materials inside the cask, including the assemblies, appeared as they did at initial loading.

Operating experience relative to the degradation of stored fuel assemblies, identified via published sources, was limited to either Westinghouse fuel assemblies, high burn-up fuel assemblies, or degradation as a result of exposure to fluid environments during reactor operation and/or spent fuel pool storage. No Westinghouse or high burn-up fuel assemblies were placed in RNP ISFSI storage. In addition, the fuel assemblies were inspected before being placed in a DSC to preclude the dry storage of damaged fuel.

Lastly, Department of Energy (DOE) research associated with the Yucca Mountain project and with foreign dry storage programs provides further support that fuel failure or degradation is expected to be very small during dry storage and that fuel can be stored for up to one hundred years.

3.1.5.3 *Summary of Operating Experience*

The above review of operating experience did not identify any potential aging effects and associated mechanisms for the ISFSI beyond those previously identified in the Subsection 3.1.3 process. Additionally, the appropriateness of the RNP ISFSI aging management review was confirmed by the operating experience review.

3.1.6 Documentation of Sources Used for the AMR Process

The following Topical Reports, each with a corresponding Safety Evaluation Report, as well as the RNP ISFSI SAR (Reference 3.6-1), were used to determine the intended functions, materials, and environmental conditions for ISFSI subcomponents identified as in-scope for license renewal:

- *Topical Report on the Nutech Horizontal Modular Storage System for Irradiated Fuel*, NUH-001 (Reference 3.6-4)
- *Duratek – IF-300 Shipping Cask - Consolidated Safety Analysis Report,* NEDO-10084-5 (Reference 3.6-5)

Docketed correspondence between Progress Energy Carolinas, Inc., and the NRC has also been utilized. Other plant documents such as drawings, technical reports, vendor manuals, procedures were consulted, as appropriate, to further clarify SSC descriptions and intended functions.

The documents listed at the end of Section 2.2, Scoping Methodology, were also used in the AMR process. Lastly, industry topical reports, reference books, and standards were consulted as appropriate for the description and evaluation of aging effects/mechanisms.

3.2 AGING MANAGEMENT REVIEW RESULTS – HSMs

This section provides the results of the aging management review of the Horizontal Storage Modules (HSMs), referred to as concrete overpack in ISFSI license renewal guidance documents, which were determined to be in the scope of license renewal as identified in Section 2.3, Scoping Results.

A summary of the results of the aging management review for the HSM subcomponents is provided in Table 3.2-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and were, therefore, not subjected to further aging management review.

A description of the HSM subcomponents that support an SSC intended function is provided in Subsection 3.2.1, and a summary of the materials and environments for the HSMs is provided in Subsection 3.2.2 and Subsection 3.2.3, respectively. Subsection 3.2.4 and Subsection 3.2.5, respectively, provide a discussion of the aging effects requiring management for the applicable HSM subcomponents and the aging management activities used to manage the effects of aging.

3.2.1 DESCRIPTION OF HSM SUBCOMPONENTS

The HSMs provide a unitized, low profile, modular storage location for irradiated fuel. Each HSM structure is constructed from reinforced concrete and structural steel. The thick concrete walls and roof of the HSM provide adequate neutron and gamma shielding. Inlets and outlets, and associated pathways, are provided in each HSM for the dissipation of decay heat. A description of each subcomponent is as follows:

<u>Concrete Walls and Roof</u>
Above the foundation, concrete and reinforcing rebar form the roof slab, outer walls, and the between-module walls for the original three HSMs and for the five additional HSMs, with each HSM serving as a storage bay. An air inlet precast slab is included in each of the eight bays. This slab is supported on the front wall and an interior wall in each bay. Removable concrete blocks, which provide shielding for the air outlets, are located on the roof slab of each HSM. The concrete codes for design and construction are included in the ISFSI SAR.

Concrete Foundation

A concrete foundation, including rebar, supports the original three HSMs and the five additional HSMs, and is predominantly underground (below grade), with portions that are visible above ground level (above grade). The concrete codes for design and construction are included in the ISFSI SAR. A water stop installed in the exterior construction joints is included as part of the foundation and is considered part of the concrete function.

Concrete Foundation Mud Slab (Below Grade)

A 4 inch mud slab was placed as a working surface for the foundation. This is non-structural concrete.

Anchorages/Embedments

Anchorages/embedments are the steel members, studs, etc., that are embedded in concrete. These anchors also have an exposed surface above the concrete.

Expansion Anchors

Expansion anchors, wedge anchors, and shell type anchors may be floor, wall, or ceiling mounted, and may be used alone or in combination with a baseplate.

DSC Support Assembly (Rails, Stopping Blocks)

Each HSM bay includes three structural steel support members, two T-rails that extend approximately the full length of the module, and the plates/shims which support the DSC. The surface of the T-rails and support plates are plated with a high phosphorous electroless nickel finish which is blasted and coated with Everlube 823.

Heat Shield

A heat shield is installed inside in each HSM bay. This consists of a galvanized plate assembly mounted by expansion bolts to the walls in the upper portion, approximately 1-1/2” from the roof slab, of each of the bays.

Rear and Front Access Cover Plates

Steel access shielding plates are provided on the front and rear of each HSM. A steel shielding plate is bolted to the rear access of each HSM. The front access shielding plate of each HSM is comprised of a steel plate, a steel frame which encloses a lead shielding plate, and 5% boron-polyethylene shielding. This front access plate is held in place by a steel frame attached to the front wall and secured tight with bolts, as discussed below.

Frame and Cask Tie Downs

A steel frame attached to the front wall outside of each HSM secures the front access cover plate in place.

Inlet/Outlet Screens and Frames

Stainless steel screens are mounted at the air inlet and air outlet openings of each HSM to prevent the entry of debris and birds/rodents. The screens are supported by a stainless steel frame which is attached to the concrete with expansion anchors.

Seismic Retaining Assemblies

A steel tube member that seismically retains the DSC in place is installed at the rear access of each HSM, inside the rear access sleeve. The assembly for each HSM is secured in place by the rear access shield plate discussed above.

Threaded Fasteners

Threaded fasteners are used at the rear access cover plate, the DSC support steel, and the front access cover plate.

Lightning Protection

The ISFSI is provided with a lightning protection system that is attached to HSMs and provides protection for all eight HSMs. The system includes lightning rods (air terminals, bases, and bracing), cable, masonry clamps, ground rods, and connecting clamps. There are four lightning rods on the original three HSMs and six on the additional five HSMs.

Dry Film Lubricant

The lubricant discussed above, for DSC Support Assembly, is used only for reducing friction while sliding a DSC along the support assembly inside an HSM. Once the DSC is in place, the lubricant performs no function.

The following subcomponents were excluded from aging management review because they do not support or impact the intended function of the HSM during the renewed license period:

- Concrete foundation mud slab (below grade),
- Dry film lubricant (Everlube 823),
- Left in place steel forms for the air inlets and outlets,
- Drain pipe embedded in each HSM,
- Alignment plates embedded in each HSM, and
- Embedded conduit, couplings and thermocouples (abandoned-in-place) for two of the HSMs

3.2.2 HSM Materials Evaluated

The materials of construction for HSM subcomponents that are subject to further aging management review are described below and generally include metals, concrete, and polymers. The material type of the individual HSM subcomponents is identified in Table 3.2-1.

Metals:

- Carbon Steel
- Galvanized Steel
- Stainless Steel
- Lead
- Copper/Brass (some nickel-plated)

Concrete (and carbon steel rebar):

- Concrete is 4000 psi

Polymers:

- Polyethylene block injected with 5% boron

3.2.3 ENVIRONMENTS FOR THE HSMS

The environments that are experienced by HSM subcomponents, continuously or on a recurring basis, are all external and are described below.

The HSMs are located outdoors at the RNP ISFSI site, in Darlington County, South Carolina. The ambient outdoor environment for the HSMs is bounded by the extreme ambient temperatures considered in the ISFSI design (-40°F to 125°F), except as noted below. The outdoor air (ambient) environment includes precipitation, humidity, ultraviolet radiation, and wind.

Portions of the HSM subcomponents are below grade and experience the same outdoor conditions with the added exposure to slightly aggressive/acidic ground water due to the pH being less than 5.5, as determined in the RNP license renewal (10 CFR 54) process.

Inside each HSM, considered to be an indoor environment with no air-conditioning, subcomponents are protected from outdoor effects (e.g., precipitation), but do experience higher temperatures and radiation in an air environment. Based on actual results from instrumentation, the normal mode maximum interior concrete temperature is less than 150°F (Reference 3.6-8), therefore, the temperature range used for all concrete (indoor and outdoor), and embedded steel, is -40°F to 149°F. The design temperature of 300°F (Reference 3.6-4) was used for all the structural steel components inside an HSM.

Based on the original shielding analysis for an HSM (Reference 3.6-4), the total gamma dose rate at the most limiting location (HSM air outlet with no shielding) is 4450 mrem/hr. By simple extrapolation, this converts to an integrated gamma dose of 2.3E6 Rads for 60 years. Similarly, the accumulated neutron flux for the HSM is 3.2E13 neutrons/cm^2 in 50 years (Reference 3.6-4), which extrapolates to 3.84E13 neutrons/cm^2 for 60 years.

3.2.4 AGING EFFECTS REQUIRING MANAGEMENT FOR THE HSMS

This section describes the aging effects that could, if left unmanaged, cause degradation of HSM subcomponents and result in loss of the SSC intended function(s) during the renewed license period. The aging management review results for individual HSM subcomponents are reflected in Table 3.2-1.

Based on the HSM material and environment combinations, and consideration of the conditions during extended storage, the following aging effects and associated mechanism(s) were determined to require management for the applicable steel (metallic) subcomponents of the HSMs:

- *Loss of Material Due to General Corrosion* (Outdoor Environment Only) – Carbon and galvanized steel portions of the HSMs, if not inside the concrete bays.

- *Loss of Material Due to Crevice and Pitting Corrosion* (Outdoor Environment Only) – Certain stainless steel and brass subcomponents attached to the external surfaces of the HSMs are conservatively considered to be susceptible.

- *Change in Material Properties Due to Corrosion* (Outdoor Environment Only) – Certain brass subcomponents of the lightning protection system are conservatively considered to be susceptible.

The review of industry and site-specific operating experience discussed in Subsection 3.1.5, supported the management of the above aging effects, but did not identify any other aging effects for an HSM during extended storage.

There are no aging effects that require management during the renewed license period for subcomponents located inside an HSM. Furthermore, this is consistent with the existing Technical Specifications for license SNM-2502. An earlier amendment, accepted by the NRC, eliminated the requirement to inspect the interior of an HSM because degradation of interior concrete or steel was not likely to occur (Reference 3.6-9).

With respect to concrete (and reinforcing steel), no aging effects/mechanisms were identified that require management for the above grade concrete. However, the current NRC position for plant (10 CFR 54) license renewal is to include concrete in an aging management program for the period of extended operation and that loss of material, cracking, and change in material properties are plausible and applicable aging effects for above ground concrete structures. (Reference 3.6-10). To meet the current NRC position, the condition of accessible HSM concrete that is above grade is conservatively considered to require management.

No additional remote inspections of the interior concrete and steel surfaces are planned during the renewal period because of the durable corrosion resistant materials used, the exterior concrete surface is a leading indicator of the interior concrete surface, and the radiation exposures associated with such an inspection.

The following aging effects and associated mechanisms require management during the renewed license period for below grade concrete (and reinforcing steel) due to exposure to aggressive groundwater with pH less than 5.5:

- *Loss of Material Due to Aggressive Chemical Attack*
- *Loss of Material Due to Corrosion of Embedded Steel/Rebar*
- *Change in Material Properties From Aggressive Chemical Attack*

With respect to the boron-polyethylene shielding material, there are no aging effects that require management during the renewed license period since the shielding block is fully encapsulated and temperatures expected inside an HSM, per Section 3.2.3, are insufficient to cause degradation. In addition, degradation due to radiation was considered a Time Limited Aging Analysis (TLAA), which is discussed in the pertinent section of Appendix B. The result of the TLAA confirms that the radiation exposure of the shielding is such that the polyethylene will remain serviceable through the renewed license period.

3.2.5 AGING MANAGEMENT ACTIVITIES FOR THE HSMS

The activities associated with the ISFSI aging management program, when continued in the renewed license period, will manage the aging effects for steel and concrete portions of the HSMs, and will include the conservative evaluation of the condition of accessible concrete for the HSM subcomponents identified in Table 3.2-1.

A description of this aging management program is provided in Appendix A, Aging Management Programs, along with the demonstration that the aging will be effectively managed during the renewed license period.

3.2.6 AMR CONCLUSION FOR THE HSMS

Based on the demonstrations provided in Appendix A, Aging Management Programs, the aging of applicable HSM subcomponents will be adequately managed so that there is reasonable assurance that SSC intended function(s) will be maintained for all current licensing basis conditions during the renewed license (extended storage) period.

3.3 AGING MANAGEMENT REVIEW RESULTS – DSCs

This section provides the results of the aging management review of the Dry Shielded Canisters (DSCs) which were determined to be in the scope of license renewal as identified in Section 2.3, Scoping Results.

A summary of the results of the aging management review for the DSC subcomponents is provided in Table 3.3-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and were, therefore, not subjected to further aging management review.

A description of the DSC subcomponents that support an SSC intended function is provided in Subsection 3.3.1, and a summary of the materials and environments for the DSCs is provided in Subsection 3.3.2 and Subsection 3.3.3, respectively. Subsection 3.3.4 and 3.3.5, respectively, provide a discussion of the aging effects requiring management for the applicable DSC subcomponents, if any, and any aging management activities used to manage the effects of aging.

3.3.1 DESCRIPTION OF DSC SUBCOMPONENTS

Per Sections 1.3.1 and 3.1.2.2 of the ISFSI SAR (Reference 3.6-1), each DSC serves as the confinement vessel during transport of irradiated fuel assemblies (IFAs) to and from a horizontal storage module (HSM), as well as during storage of the IFAs in an HSM. The shielded end plugs provide biological shielding during transport of the IFAs and also provide shielding for the front and rear accesses to the corresponding HSM. A single DSC is sized to hold seven irradiated pressurized water reactor (PWR) fuel assemblies.

A total of eight (8) DSCs were loaded with IFAs and stored in the corresponding HSMs starting in 1989. CP&L (Progress Energy Carolinas, Inc.) does not intend to load more spent fuel into the existing ISFSI (Reference 3.6-11), or to remove and replace any of the stored DSCs, except in the case of a design basis accident or for shipment to an approved federal repository for permanent storage. All of the DSC subcomponents are passive and long-lived, not subject to replacement based on a qualified life or specified time period. Consistent with the RNP ISFSI SAR (Reference 3.6-1), Table 1.3-1 and Figure 1.3-1, each DSC comprises:

<u>Canister Body (Casing)</u>
The main component of construction of each DSC is a stainless steel cylinder. The canister body, or shell, consists of a rolled and welded steel plate. Only one longitudinal weld was permitted in the canister shell itself, a full penetration weld that required many weld passes and was inspected during fabrication to insure weld metal was as sound as the parent metal.

<u>End Plugs</u>
Stainless steel drums at both ends of each DSC are lead-filled for biological shielding when the DSC is in a shipping cask or in an HSM. The plugs also perform a pressure retention function. The bottom end plug was installed during fabrication and consists of an inner pressure plate and outer cover plate, which encapsulate the lead shielding. The bottom end plug also includes a lead-filled shielding framework that encircles the shield plug and is welded to the pressure plate. The upper end plug consists of an inner plate and outer pressure plate, which encapsulate the lead shielding. When welded to the canister body, this outer pressure plate forms the primary pressure boundary at the upper end of each DSC.

<u>Top Cover Plate (End Cover Plate)</u>
The upper end plug and top cover plate were welded in each DSC after the fuel was loaded. The top cover plate provides redundant pressure protection for the upper end plug. The top cover plate has an attached rolled ring for handling purposes. The plate thickness prevents significant deformation and bending stress on the canister body that might occur during handling.

<u>Upper Plug Retaining Ring Assembly</u>
The shield plug support ring assembly at the top of each DSC includes a shield plug key that incorporates two piping penetrations into the canister cavity. The key includes a two plane, dog-leg type offset to prevent streaming. One penetration, the vent tube, ends at the bottom of the retaining ring. The other, the siphon tube, continues to the bottom of the canister cavity. The penetrations terminate in normally closed Swagelok quick-connect fittings. Plugs were welded over these penetrations in each DSC prior to the final closure.

Basket Assembly

A basket enclosed in each DSC serves as the structural support for the irradiated fuel elements and contains neutron poison in the form of boral plates within the guide sleeves enclosing each fuel assembly. Seven spacer discs locate the fuel assemblies in the radial direction and provide support for the fuel assemblies as well as the surrounding sleeves (also referred to as square cells). The spacer discs are aligned by four support rods which run the length of each canister. The boron-loaded aluminum alloy (Boral), which lines the sleeves, provides criticality control in the canister during wet loading (and unloading) operations, but is not used in any structural capacity.

Penetration Assembly (Instrumented DSCs Only)

For research purposes, two of the DSCs installed at the H. B. Robinson facility were designed to accept instrumentation. The permanently installed DSC thermocouples were connected to an external cable by means of a specially designed feed-through penetration assembly through the bottom end plug. This feed-through incorporates the same redundant seal philosophy used in the DSC containment design (References 3.6-1 and 3.6-4).

The body (casing) of a penetration assembly is a stainless steel cylinder filled with a stack of lead disks. Holes were drilled through the disks to allow passage of the stainless steel sheathed, magnesium oxide insulated thermocouple leads. A cylinder (or plug) was double-welded to the bottom of each instrumented DSC while the thermocouple sheaths were brazed to the plug assembly at both ends. A layer of epoxy approximately one-sixteenth of an inch thick was applied to both ends of each thermocouple sheath. The epoxy adheres to the thermocouple wires and the steel sheath to prevent gas leakage out of an instrumented DSC, or air or water vapor in-leakage into an instrumented DSC (References 3.6-12 and 3.6-13).

A penetration assembly passes through the pressure plate of each instrumented DSC bottom end plug. A weld sleeve was used to weld the penetration casing to the other side of the pressure plate, and thus provide a redundant seal. The penetration assembly (casing) continues from the weld sleeve through, and is welded to, a wedge in the DSC shielding framework that encircles the bottom end plug.

The following DSC subcomponents were excluded from further aging management review because they do not support or impact the intended function of the DSC during the renewed license period:

- Dry film lubricant (Everlube 823) on sliding surfaces of each cylinder
- Thermocouples and associated subcomponents, except for the penetration assembly described above

3.3.2 DSC MATERIALS EVALUATED

The materials of construction for DSC subcomponents that are subject to further aging management review include stainless steel, aluminum-boron (Boral), lead, and epoxy-resin (Epoxylite #8611 for two instrumented DSCs only). The material type of individual DSC subcomponents is identified in Table 3.3-1.

3.3.3 ENVIRONMENTS FOR THE DSCS

The environments that affect the subcomponents of each DSC, both externally and internally, are those that are normally (continuously) experienced and are described below.

External
Each DSC is positioned for long-term storage inside an HSM. As such, the external surface of each DSC is exposed to the same environment, including neutron fluence and integrated gamma dose, described in Section 3.2.3 for the interior of an HSM. That is, an indoor, not-air conditioned environment that is protected from precipitation and wetting. The normal operating temperature of the outside DSC surface is highest at the top of the cylinder and was expected to be 284°F (for 70°F ambient air) during the original license term (Reference 3.6-4). This surface temperature is conservatively extended into the license renewal period.

Internal
A design temperature of 400°F was used for DSC internal structures in the aging management review. The heat generated in the fuel regions of the IFAs inside a sealed DSC is transferred towards the canister shell by radiation, convection, and conduction. Helium is present in the canister to facilitate the conduction. A parametric study of temperature versus time has shown that the fuel temperature and, therefore, the helium temperature and DSC internal structure temperature, will decrease over time (Reference 3.6-4). For conditions of 70°F ambient temperatures, the normal expected helium temperature during the original license term was determined to be 389°F (Reference 3.6-4). As such, the use of the higher temperature is conservative for evaluation of the long-term effects of temperature on stainless steel.

After 20 years of dry storage, the fast neutron flux and gamma radiation doses are expected to be on the order of 1E14 neutrons/cm^2 and 1E9 Rads respectively (Reference 3.6-7). A simple extrapolation between 20 and 60 years does not result in a change of more than one order of magnitude for these expected doses. In Table 3.3-1, the helium inside a DSC is listed as an Air and Gas environment.

3.3.4 AGING EFFECTS REQUIRING MANAGEMENT FOR THE DSCs

Based on a review of the DSC materials of construction and the environments (e.g., relevant conditions and stressors) experienced during extended ISFSI storage, there are no aging effects requiring management during the renewed license period for the subject DSC subcomponents.

There are no aging effects requiring management for the stainless steel or lead subcomponents of the DSCs because of the durable construction, double seal-welded closure, and the environments to which each DSC is exposed. Each DSC was sealed during the original license term to contain seven appropriately aged fuel assemblies and an inert (Helium) environment, resulting in sub-critical IFAs and lower neutron fluence. Each sealed DSC was then placed into an HSM and, thereafter, exposed to only a relatively mild ambient environment. In addition, a continued decrease of IFA temperatures and radiation levels are expected over time. None of the DSCs are planned to be removed from the HSMs for inspection, nor are any remote inspections planned to be performed during the renewal period, because of the durable corrosion resistant materials used and the additional radiation exposure associated with such inspections. This also precludes any potential damage during the extra movement of a DSC from an HSM.

The same is true for the aluminum-boron (neutron poison) subcomponents of each DSC. Additionally, a change in material densities at temperature extremes, degradation, and reduced neutron absorption capability may be a concern for neutron poisons (Reference 3.6-11). As discussed in the pertinent section of Appendix B, Time-Limited Aging Analyses (TLAAs), neutron radiation will cause a portion of any neutron shield to be depleted, however, the effects of radiation on neutron shielding and steels is expected to be negligible (e.g., less than 0.3%) and sufficient neutron poison material will remain for criticality control during wet unloading operations, should they occur. The dry fuel is not moderated, assuring subcriticality during the extended storage.

For the epoxy-resin seals associated with the penetration assembly of each instrumented DSC, there are no aging effects requiring management for the outer seals that are exposed to a relatively mild environment and do not experience the temperature and radiation effects on the inside surface of the DSC. The effects of radiation on the inner seal of the penetration assembly has been previously evaluated (References 3.6-12 and 3.6-13). As discussed in the pertinent section of Appendix B, Time-Limited Aging Analyses (TLAAs), degradation of the inner seal due to ionizing radiation (gamma and/or neutron) is expected to be minimal during the renewed license period and the seals are expected to remain intact.

The storage cask related operating experience discussed in Subsection 3.1.5 is also applicable to the DSCs, which together with the HSMs serve as dry storage for the IFAs. However, this experience is primarily focused on contamination, corrosion, and failed leak tests during loading of the canister/cask and not on the age-related degradation of double seal-welded DSCs. Aging management review results for individual DSC subcomponents are shown in Table 3.3-1.

3.3.5 Aging Management Activities for the DSCs

There are no aging management programs or activities required for DSC subcomponents during the renewed license period. Therefore, no aging management activities are credited.

3.3.6 AMR Conclusion for the DSCs

Reasonable assurance is provided that the intended functions of DSC subcomponents will be maintained under all current licensing basis (CLB) conditions during the renewed license (extended storage) period.

Furthermore, each sealed DSC will continue to store its original contents during the renewed license period and will not be reused to store different fuel. Except in the case of a design basis accident, or the shipments of IFAs to an approved permanent federal repository or other interim storage facility, the stored DSCs will not be removed from storage or opened. Therefore, the relevant storage conditions will be maintained, potential damage during movement and repositioning will be precluded, and personnel radiation exposures will not be increased unnecessarily.

3.4 AGING MANAGEMENT REVIEW RESULTS – IFAs

This section provides the results of the aging management review of the Irradiated Fuel Assemblies (IFAs), also referred to as spent nuclear fuel (SNF), which were determined to be in the scope of license renewal as identified in Section 2.3, Scoping Results.

A summary of the results of the aging management review for the IFA subcomponents is provided in Table 3.4-1 which is located at the end of Section 3.0, Aging Management Reviews. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and were, therefore, not subjected to further aging management review.

A description of the IFA subcomponents which support an SSC intended function is provided in Subsection 3.4.1, and a summary of the materials and environments for the IFAs is provided in Subsection 3.4.2 and Subsection 3.4.3, respectively. Subsections 3.4.4 and 3.4.5, respectively, provide a discussion of the aging effects requiring management for the subject IFA subcomponents, if any, and any aging management activities used to manage the effects of aging, respectively.

Also, Idaho National Engineering and Environmental Laboratory and Argonne National Laboratory have completed studies to determine the aging effects on fuel in dry storage casks. The fuel assemblies had been in dry storage casks for over fourteen years. A visual examination and a material analysis were performed on the fuel assemblies. The results of this study have been included in EPRI reports, including the most recent report (Reference 3.6-6). The project identified and examined several questions concerning fuel behavior during dry storage. The characteristics and storage conditions of the IFAs used in this study are similar to, and bound the characteristics and storage conditions of, the IFAs in the RNP ISFSI.

3.4.1 Description of IFA Subcomponents

Each Dry Shielded Canister (DSC) in the ISFSI contains seven pressurized water reactor (PWR) spent fuel assemblies which had a maximum heat generation limit of 1 kilowatt per assembly, or 7 kilowatts per DSC, a maximum average burn-up of 35,000 megawatt-days/metric ton uranium, and that were cooled for at least five years prior to storage (Reference 3.6-1).

The intended functions of the IFAs were conservatively determined during scoping to include criticality control, pressure boundary, structural integrity, and heat transfer. The geometry of the IFAs is a factor in the proper conduction and convection of heat to the DSC surface and in the criticality model. The fuel cladding provides a confinement barrier, and the structural integrity is necessary to maintain a favorable geometry and for retrieval. After fuel loading and DSC drying, the spent fuel assemblies are not moderated, assuring subcriticality during subsequent operations and configurations. Furthermore, a total cladding failure has been evaluated from the perspective of both DSC pressurization and DSC leakage, with the dose consequences determined to be acceptable (Reference 3.6-4). The IFAs principle function during dry storage is to maintain proper geometry and position.

As indicated by the various cross-references in the RNP Updated Final Safety Analysis Report (UFSAR) and RNP ISFSI SAR (References 3.6-14 and 3.6-1), the fuel assemblies described below were originally supplied by Exxon Nuclear Company, Inc. A 15X15 fuel assembly array includes twenty guide tubes, one instrumentation tube, and two-hundred-four fuel rods (Reference 3.6-14). Additionally, seven grid spacers, a lower tie plate, and an upper tie plate with attachments form the structural skeleton of the fuel bundle/assembly, along with the guide tubes (Reference 3.6-14). The following subcomponents of an IFA are described in this section and are listed in Table 3.4-1:

- Fuel Rods (Cladding, End Caps/Plugs)
- Guide Tubes
- Instrumentation Tube
- Spacers (Grid Assemblies)
- Lower Tie Plate (Bottom Nozzle)
- Upper Tie Plate (Top Nozzle, Springs, Capscrews)

Fuel Rods (Cladding, End Caps/Plugs)
The fuel rods consist of slightly enriched UO_2 pellets inserted into Zircaloy-4 tubes (Reference 3.6-14). The fuel cladding is a tube that is cold worked and stress relieved. Plug-type end caps are seal welded to each end. The cladding and end caps contain the fuel pellets and fission gases. A fuel rod is shown in Figure 4.2.2-2 of the RNP UFSAR (Reference 3.6-14). Unlike conventional material cladding, the fuel cladding is not mechanically bonded to the fuel pellets that it protects. Each rod is pressurized with helium, which assists in the prevention of clad creep collapse (Reference 3.6-14).

Guide Tubes
The guide tubes are welded to the grid spacers, and mechanically attached and secured to the upper and lower tie plates (Reference 3.6-14). During reactor operations, the guide tubes provided channels for control rods and a means to align insert components. During ISFSI storage, the guide tubes for each stored IFA provide structural support and alignment and, as such, are a factor in the retrievability of an IFA.

Spacers (Grid Assemblies)
Each IFA contains seven spacers, six of which are located in the active fuel region. All of these are bi-metallic (Reference 3.6-14). The seven spacers (or grid assemblies), which are attached to the guide tubes, provide support for the fuel cladding tubes and maintain correct rod-to-rod spacing. The spacers consist of interlocking rectangular grid strips and spring strips mechanically secured within the grid strips.

Lower Tie Plate (Bottom Nozzle)
The lower tie plate is a square, box-like structure connected to the guide tubes which maintains guide tube array and functions as the bottom structural element of a fuel assembly.

Upper Tie Plate (Top Nozzle, Springs, Cap Screws)
The upper tie plate is a box-like structure which is the upper structural element and is the means to lift an entire fuel assembly. It also forms a plenum space for coolant discharge during reactor operations and maintains the guide tube array.

In addition to the above IFA subcomponents, the following IFA subcomponents, although in the scope of license renewal, were excluded from further aging management review because they do not support or impact the intended function of the DSC during the renewed license (extended storage) period:

- Fuel assembly inserts (typically none were left in the stored assembly),
- Fuel pellets (including all fuel rod internals),
- Instrumentation tube,
- Spacer (grid assembly) spring strips

3.4.2 IFA MATERIALS EVALUATED

The materials of construction for the subcomponents of each IFA that are subject to aging management review are zirconium-based alloy (Zircaloy-4), stainless steel (including some cast austenitic stainless steel - CASS), and nickel-based alloy (Inconel).

3.4.3 ENVIRONMENTS FOR THE IFAS

During reactor service and for a minimum of five years following discharge from the reactor, the IFAs were exposed to a borated water environment in the reactor vessel or spent fuel pool. However, this water, with the exception of trace amounts, was removed following fuel loading into a DSC. Likewise, the exposure to borated/demineralized water during the loading or end-of-canister life unloading are of limited duration and do not require consideration for the effects of aging. As such, the exposure to borated/demineralized water is not a factor in the evaluation of the effects of aging during the renewed license (extended storage) period.

In dry storage, the IFA subcomponents that are subject to aging management review are stored in an air and gas environment with the following considerations. The fuel rods (cladding) and guide tubes are the only IFA subcomponents that have internal and external surface exposures. The guide tubes are open on the end and have the same internal and external environment. The fuel rods were originally supplied with a pressurized fill gas, such that the internal surface and external surface experience roughly the same environment.

Internal

The fuel rod cladding internal environment is helium fill gas with some fission product gases (References 3.6-1 and 3.6-14). It is impossible to accurately partition the amount of fission gas released inside the cladding during in-reactor operation and during dry storage. The initial fill gas pressure in the RNP Exxon fuel rods was 300 psig (Reference 3.6-1).

Following initial cask loading, the temperature inside the fuel cladding was expected to be less than 716°F (380°C) for the hottest fuel rod on the hottest day conditions (References 3.6-3 and 3.6-1). After roughly 12 years of dry storage (17 years after discharge from the reactor), the fuel cladding temperature is expected to be less than 347°F (175°C) (Reference 3.6-4), which is consistent with thermal factors described in the pertinent ASTM Standard, C1562-03 (Reference 3.6-7).

External

Externally, the fuel rods are exposed to the same helium gas as the other subject IFA subcomponents, including the inside of the open guide tubes, in a dry shielded canister (DSC) as described in Subsection 3.3.3. The helium gas temperature external to the fuel rods is a function of fuel cladding temperature and decreases over time.

After 20 years of dry storage, the fast neutron flux and the cumulative gamma radiation doses are expected to be less than approximately 1E14 n/cm^2 and 1E9 Rads, respectively (Reference 3.6-7). A simple extrapolation between 20 and 60 years does not result in a change of more than one order of magnitude for these expected doses.

3.4.4 Aging Effects Requiring Management for the IFAs

Based on a review of the IFA materials of construction against the environments (e.g., relevant conditions and stressors) experienced during extended ISFSI storage, there are no aging effects requiring management during the renewed license period for the zirconium-based alloy, stainless steel (including CASS), or nickel-based alloy IFA subcomponents.

This is consistent with the EPRI Dry Cask Storage Characterization Project Report (Reference 3.6-6), which also did not identify any evidence of aging degradation of fuel assembly subcomponents from the time of initial loading up to the time of testing, and is considered to bound the IFAs stored in the RNP ISFSI due to similar characteristics and conditions.

3.4.5 AGING MANAGEMENT ACTIVITIES FOR THE IFAS

There are no aging effects requiring management for the IFAs. Therefore, no aging management program or activities are credited during the renewed license period for IFA subcomponents.

3.4.6 AMR CONCLUSION FOR THE IFAS

Due to the inert storage environment and decreasing temperatures over the extended storage period, and as supported by operating experience (including findings of a lack of degradation), there are no aging effects requiring management during the renewed license period for the IFA subcomponents stored in the ISFSI. Therefore, reasonable assurance is provided that the intended functions of the ISFSI irradiated fuel assemblies will be maintained under current licensing basis conditions during the renewed license period.

3.5 AGING MANAGEMENT REVIEW RESULTS – IF-300 TRANSFER CASK

This section provides the results of the aging management review of the cask that is used for ISFSI transfers to and from an HSM, also referred to as the IF-300 Transfer Cask. This cask was determined to be in the scope of license renewal as identified in Section 2.3, Scoping Results.

As discussed in Section 1.3.1.3 of the RNP ISFSI SAR (Reference 3.6-1), the transfer cask used with the ISFSI provided shielding during the Dry Shielded Canister (DSC) drying/sealing operation and during transfer to the Horizontal Storage Modules (HSMs). For the ISFSI, the IF-300 (which CP&L owns), licensed under 10 CFR 71 as a transportation cask, was used in a slightly modified configuration.

A summary of the results of the aging management review for the IF-300 Transfer Cask subcomponents are provided in Table 3.5-1. The table provides the following information related to each subcomponent determined to require aging management review: (1) the intended function, (2) the material group, (3) the environment, (4) the aging effects requiring management, and (5) the specific aging management activities that manage those aging effects. The table also identifies subcomponents that did not support, or whose failure would not compromise, the SSC intended function and were, therefore, not subjected to further aging management review.

A description of the IF-300 subcomponents that support an SSC intended function is provided in Subsection 3.5.1, and a summary of the materials and environments for the IF-300 is provided in Subsection 3.5.2 and Subsection 3.5.3, respectively. Subsections 3.5.4 and 3.5.5, respectively, provide a discussion of the aging effects requiring management for the applicable

IF-300 Transfer Cask subcomponents and the aging management activities used to manage the effects of aging.

3.5.1 Description of IF-300 Subcomponents

The major cask portions are divided into sub-components that are included in Table 3.5-1, along with the particular function the individual sub-component performs to support the overall cask intended functions. A summary of those cask subcomponents is provided below:

<u>Cask Body</u>
The cask body is a cylindrical shape enclosed on one end and composed of layers which are as follows:

- The inner cavity is encircled by a thick stainless steel cylinder that is welded to the forged closure flange on one end, and is sealed with a stainless steel end plate on the bottom end, forming the inner cavity shell.
- Surrounding and shrink-fitted to the inner cavity shell is depleted uranium shielding material, and castings that are fitted with overlapping joints to prevent radiation streaming and that serve as the radiological or biological shield for the cask.
- The cask outer shell is a stainless steel cylinder shrink-fitted to the uranium casting, thus forming a composite or laminated vessel. This cylinder is welded to the forged closure flange on one end, and a stainless steel end plate on the bottom end.

<u>Cask Attachments</u>
The IF-300 includes the following external attachments to the cask body:

- Four structural rings provide protection from side impact and also support the neutron shield water jacket sections.
- Radially mounted impact fins (energy absorbing members) are welded in place at the cask bottom, as well as to the closure head and valve boxes.
- A set of lifting blocks located just below the closure flange and pinned to the structural rings.
- Four large valve boxes, each with a cover, on the exterior of the cask outer shell, two for the neutron shield water jacket cavities and two for the cask (inner) cavity. The latter two are nested in the upper and lower pairs of structural rings.

<u>Cask Penetrations</u>
The IF-300 includes the following penetrations of the cask body:

- A thermowell, including external fittings, enters from the bottom of the cask for insertion of a thermocouple for temperature monitoring purposes (not needed for ISFSI transfers).

- A vent line at the top and drain line at the bottom. These lines are sealed by bellows globe valves and valve quick-disconnect couplings or pipe caps/pipe plugs. The vent line is equipped with a rupture disk. All valves are housed in the protected boxes described above.

Cask Neutron Shield Water Jacket

The cylindrical portion of the outer shell is encircled by a thin-walled water jacket that extends axially from just below the closure flange to a point slightly above the cask bottom. The jacket is corrugated for heat transfer and partitioned at cask mid-length to form two independent cavities (compartments). This outer layer serves as containment for a water annulus functioning as a neutron shield.

Each longitudinal compartment of the neutron shield water jacket is equipped with two externally mounted expansion tanks, fill and relief valves, and associated piping. The fill line for each compartment is terminated by a globe valve in a protected box (separate from cavity boxes). The vent (relief) line from each compartment goes to an expansion tank that is provided with a pressure relief valve. Each set of two expansion tanks (and the associated relief valve and piping) is mounted to the structural ring and to the corresponding neutron shield water jacket valve box

Cask Extension Collar and Cask Lid

To meet the cask cavity minimum length requirement to accommodate a DSC, and the criteria for cask collar lid removal in the horizontal position, an addition was required for the IF-300 Cask. The IF-300, for ISFSI purposes, includes an extension collar with the same inside diameter as the cask and a cask collar lid. In this modified configuration, the energy absorbing properties of the cask are significantly reduced. However, there is no credible condition during cask handling and transfer in which the cask could be dropped on its head and the safety features of the ISFSI transfer operation are not affected (Reference 3.6-1).

The collar and liner flange is placed on the closure flange of the cask and bolted into place. The collar lid is provided with two lifting rods and a lifting eye for proper placement and securing to the collar during transport.

Threaded Fasteners

Threaded fasteners are used in the following locations for the IF-300 cask, and all are external to the cask and neutron shield water jacket:

- Cask lid to cask extension collar connections,
- Cask liner flange to cask extension collar connections,
- Cask extension collar to cask closure flange connections,
- Two guide pins provide closure alignment and orientation with the cask closure flange,
- Four guide pins provides closure alignment and orientation for the cask lid to the cask extension collar,
- Valve/piping closure bolting,
- Expansion tank mounting bolting, and
- Valve box cover bolting.

In addition to the above IF-300 subcomponents, some of the cask subcomponents are active or short-lived, or are design features that support 10 CFR 71 shipments, but are not used or credited for 10 CFR 72 ISFSI purposes. Thus, such cask subcomponents do not support dry storage or retrieval intended functions. Also, the degradation or failure of these subcomponents would not prevent an important to safety function from being accomplished. Therefore, the following IF-300 subcomponents were not subjected to aging management review:

- Cask sealing and valve box cover gaskets,
- Rupture disc,
- Closure head and basket,
- Valve internals,
- Flow baffles installed in the water jacket,
- Seal retainer device used during the initial loading,
- Thermocouple, and
- Cask skid/enclosure.

3.5.2 IF-300 MATERIALS EVALUATED

The materials of construction for the individual sub-components of the IF-300 that are subject to aging management review are listed in Table 3.5-1 and include carbon steel, stainless steel (including some CASS), and depleted uranium. Additionally, to prevent the formation of a low melting point alloy of steel and uranium, a minimum 4 mil thick copper diffusion barrier exists at every uranium-steel interface.

3.5.3 ENVIRONMENTS FOR THE IF-300

The cask exterior is exposed to borated water during fuel loading while the cask is in the spent fuel pool, and to demineralized water in the annulus between the DSC and inner cavity wall of the cask. Following fuel loading into the DSC, the cask was removed from the pool. The annulus water is removed following welding of the lid to the DSC body, purging of the water in the DSC, vacuum drying and inerting of the DSC interior. The brief exposure of the cask to the borated and demineralized water does not contribute to the aging of the transfer cask materials during the renewed license period. It is the prolonged or frequently recurring exposure to environmental conditions and stresses that must be evaluated for aging effects, such as those encountered during storage (staging) prior to use for ISFSI transfers.

Also, as described within Section 21 of the Certificate of Compliance for the IF-300 (Reference 3.6-15), the license for the 10 CFR 71 shipment of radioactive materials will currently expire in September of 2005, but may be renewed to September 30, 2008 (Reference 3.6-16). Therefore, the IF-300 cask is limited to approximately two additional years of 10 CFR 71 shipments during the renewed license period

The capabilities of the IF-300 to contain and transport irradiated fuel off-site have been previously evaluated for the more stringent 10 CFR 71 purposes/requirements and are documented in NEDO-10084-5, *IF-300 Shipping Cask Consolidated Safety Analysis Report* (Reference 3.6-5). This, in combination with the annual inspections performed on the cask for 10 CFR 71 compliance purposes, ensures that the conditions of the IF-300 while shipping irradiated fuel assemblies will not be a factor in the aging of the IF-300 materials during the RNP ISFSI renewed license period. The radiation and temperature levels included in the IF-300 design (for shipments meeting both NRC and DOT requirements) are considered to bound the much lower radiation and temperature levels associated with the infrequent on-site ISFSI transfers that may occur in the future.

As such, the following external and internal environments are those that will be experienced by the IF-300 sub-components during storage (staging) prior to and between infrequent use for ISFSI transfers:

<u>External</u>
The IF-300 may be staged outdoors or in a sheltered location prior to and between usage for infrequent ISFSI transfers. As such, the external environment for the cask is bounded by ambient air at temperatures in the range of -5°F to 130°F (Reference 3.6-15). The Outdoor Air environment includes the intermittent effects of precipitation, ultraviolet radiation, ozone, and wind. For sheltered staging, an Indoor, Not-Air Conditioned environment is used, which is the same as the Outdoor environment, except that protection is provided from precipitation, ultraviolet radiation, wind, etc.

<u>Internal</u>
Inside the cavity of the cask and for void spaces in the cask layers, if any, an Ambient Air and Gas definition is used. The pressure of the environment is atmospheric, temperatures are ambient (-5°F to 130°F), and humidity is 10-

100%. Neutron fluence and integrated gamma dose for IF-300 surfaces are residual from spent fuel shipments. The design of the IF-300 with respect to fast neutron radiation for the 10 CFR 71 offsite shipments of irradiated fuels is contained in Sections 8.1.2 and 8.2.2 of Reference 3.6-5. It was assumed for 10 CFR 72 license renewal purposes that the neutron fluence and integrated gamma dose experienced during 10 CFR 71 shipments were within the design of the IF-300 and would bound the fluence experienced during loading and on-site transfer of the eight (8) DSCs during the late 1980s (1989), since the IF-300 is presently being used. Also, no additional neutron fluence or gamma dose for the IF-300 needs to be estimated during the renewed license term since the cask will not receive continuous exposure to radiation during that time. The IF-300 may experience radiation exposure during shipments or transfer and unloading of the eight (8) DSCs.

Additionally, portions of the IF-300 are continuously exposed to a treated water environment in the cask neutron shield water jacket. The annulus fluid (treated water) for this neutron shield is a 50/50 mixture of demineralized water and ethylene glycol at atmospheric pressure (during staging). Because of the ethylene glycol, temperatures in the water jacket may be slightly different than ambient.

3.5.4 AGING EFFECTS REQUIRING MANAGEMENT FOR THE IF-300

Because of the durable steel construction and relatively mild environments to which the IF-300 transfer cask subcomponents are normally exposed during staging prior to and between infrequent ISFSI transfers, only the following require aging management:

- Carbon steel subcomponents
- Surfaces of the IF-300 Transfer Cask that are continuously exposed to the glycol/water mixture in the neutron shield water jacket, or are intermittently exposed to wetting, if located (staged) outside

The relevant conditions could exist on wetted surfaces for the following aging effect to occur for exposed stainless steel surfaces. If left unmanaged, this aging effect could result in loss of component intended function(s), and thus requires management during the renewed license period.

- Loss of Material (due to crevice and/or pitting corrosion)

Although the cask collar and lid are fabricated from durable, “thick” steel plates, these carbon steel subcomponents of the IF-300 Transfer Cask are exposed to weather and other forms of humidity during either outdoor or indoor staging and some loss of material is expected. Conservatively, the following aging effect could result in loss of component intended function(s), and thus requires management during the renewed license period:

- Loss of Material (due to general corrosion)

No aging effects could potentially result in a loss of component intended function(s) for the copper or depleted uranium subcomponents of the IF-300 during the renewed license period.

Operating experience described in Subsection 3.1.5 supports age-related degradation primarily in wetted environments. Table 3.5-1 provides a summary listing of the aging effects requiring management and the activity used to manage the effects.

3.5.5 Aging Management Activities for the IF-300

The Transfer Cask Aging Management Program is credited with managing either the effect of loss of material, or the relevant conditions that could lead to the onset and propagation of a mechanism leading to loss of material during the renewed license (extended storage) period as identified in Table 3.5-1 for:

- Carbon steel subcomponents
- Stainless steel surfaces of the IF-300 (continuously and intermittently wetted)

A description of this aging management activity is provided in Appendix A, Aging Management Programs, along with the demonstration that the identified aging effect will be effectively managed for the renewed license period.

3.5.6 AMR Conclusion for the IF-300

Based on the above discussion and demonstration provided in Appendix A, Aging Management Programs, the following conditions will be adequately managed:

- Loss of material due to crevice or pitting corrosion for stainless steel subcomponents exposed to the glycol/water neutron shield water jacket mixture (e.g., mixture purity) or to intermittent wetting, if located (staged) outside, and
- Loss of material due to general corrosion of carbon steel surfaces

More stringent requirements associated with the current 10 CFR 71 usage of the IF-300 encompass any associated degradation considerations. However, upon expiration of the 10 CFR 71 license in 2005 or 2008, the pertinent portions of these cask maintenance/inspection activities will be incorporated into a Transfer Cask Management Program. These activities, when continued into the renewed license period, will provide reasonable assurance that the component intended function(s) will be maintained under all CLB conditions during the renewed license period.

3.6 REFERENCES (SECTION 3.0, AGING MANAGEMENT REVIEWS)

3.6-1 *Independent Spent Fuel Storage Installation Safety Analysis Report*, H.B. Robinson Steam Electric Plant, through Amendment No. 18

3.6-2 *Safety Evaluation Report of H.B. Robinson Steam Electric Plant Unit No. 2 Independent Spent Fuel Storage Installation*, U.S. Nuclear Regulatory Commission, Docket No. 72-3, June 1986

3.6-3 *Independent Spent Fuel Storage Installation Material License No. SNM-2502*, H.B. Robinson Steam Electric Plant, through Amendment No. 12, November 6, 2000, Appendix A (Technical Specifications)

3.6-4 *Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel*, NUH-001, Pacific Nuclear Fuel Services, Inc., San Jose, California, March 1990, Revision 2 (SER Included)

3.6-5 *IF-300 Shipping Cask Consolidated Safety Analysis Report*, NEDO-10084-5, Duratek, November 2000 (SER Included)

3.6-6 *Dry Cask Storage Characterization Project*, Final Report, Electric Power Research Institute, Palo Alto, CA. September 2002. 1002882.

3.6-7 *Standard Guide for Evaluation of Materials Used in Extended Storage of Interim Spent Nuclear Fuel Dry Storage Systems*, American Society for Testing and Materials, ASTM C1562-03, March 2003

3.6-8 Letter from C.R. Dietz (CP&L) to Office of Nuclear Material Safety and Safeguards (USNRC), *H.B. Robinson Independent Spent Fuel Storage Installation (SNM-2502) Request for License Amendment – Deletion of Requirement for Interior Concrete Inspection*, November 1, 1993

3.6-9 Letter from NRC Office of Nuclear Material Safety and Safeguards to C.R. Dietz (CP&L), *Carolina Power & Light Company—Amendment of Materials License No. SNM-2502, Amendment No. 10*, March 17, 1994 (SER Included)

3.6-10 *Proposed Revision of Chapters II and III of Generic Aging Lessons Learned (GALL) Report on Aging Management of Concrete Elements*, Interim Staff Guidance (ISG-3), US Nuclear Regulatory Commission, November 23, 2001

3.6-11 *Safety Evaluation Report of Carolina Power & Light Company's Response to NRC Bulletin 96-04 (NUHOMS-07P System)*, Docket No. 72-3, US NRC, June 16, 1997

3.6-12 Letter from L.I. Loflin (CP&L) to NRC Office of Nuclear Material Safety and Safeguards, *H.B. Robinson Independent Spent Fuel Storage Installation Response to Request for Information*, April 28, 1989

3.6-13 Letter from L.I. Loflin (CP&L) to NRC Office of Nuclear Material Safety and Safeguards, *H.B. Robinson Independent Spent Fuel Storage Installation Submittal of Corrected Information,* June 2, 1989

3.6-14 *H.B. Robinson Steam Electric Plant Unit 2, Updated Final Safety Analysis Report*, Chapter 4 – Reactor

3.6-15 Letter from NRC Spent Fuel Project Office to Duratek, *Certificate of Compliance for Radioactive Materials Package*, Subject: Model No. IF-300 Package, Docket Number 71-9001, December 2, 2002 (SER Included)

3.6-16 NRC News 03-166, *NRC Revises Safety Standards for Packaging and Transportation of Radioactive Material*, December 30, 2003

AGING MANAGEMENT RESULTS TABLES (Section 3.0, Aging Management Reviews)

Table 3.2-1 Aging Management Review Results for the Horizontal Storage Modules (HSMs)

Subcomponent [1]	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activity
Concrete (Above Grade)	HT, SH, SS	Concrete	Outdoor	Loss of Material [2] Cracking [2] Change in Material Properties [2]	ISFSI Aging Management Program
Concrete (Below Grade)	HT, SS	Concrete	Buried	Loss of Material Change in Material Properties	ISFSI Aging Management Program
Anchorages / Embedments	SS	Carbon Steel	Embedded/ Encased	None Identified	None Required
		Galvanized Carbon Steel	Embedded/ Encased	None Identified	None Required
Anchorages / Embedments (Exposed)	SS	Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
			Indoor, Not Air-Conditioned	None Identified	None Required
		Galvanized Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
Expansion Anchors	SS	Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
		Galvanized Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
			Indoor, Not Air-Conditioned	None Identified	None Required
DSC Support Assembly	SS	Carbon Steel	Indoor, Not Air-Conditioned	None Identified	None Required

Table 3.2-1 Aging Management Review Results for the Horizontal Storage Modules (HSMs)

Subcomponent (1)	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activity
Frame and Cask Tie Downs	SS	Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
Inlet/Outlet Screens and Frames	HT, SS	Stainless Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
Rear Access Cover Plates	SH	Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
Front Access Cover Plates	SH	Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
		Lead	Outdoor	None Identified	None Required
		Boron-Polyethylene	Outdoor	None Identified	None Required
Heat Shield	HT	Galvanized Steel	Indoor, Not Air-Conditioned	None Identified	None Required
Seismic Retaining Assembly	SS	Carbon Steel	Indoor, Not Air-Conditioned	None Identified	None Required
Fasteners	SS	Galvanized Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
		Galvanized Carbon Steel	Indoor, Not Air-Conditioned	None Identified	None Required
		Stainless Steel	Outdoor	Loss of Material	ISFSI Aging Management Program

Table 3.2-1 Aging Management Review Results for the Horizontal Storage Modules (HSMs)

Subcomponent (1)	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activity
Connecting Clamps and Bolts (3)	SS	Brass	Outdoor	None Identified	None Required
			Below Grade	None Identified	None Required
		Stainless Steel	Below Grade	None Identified	None Required
			Outdoor	None Identified	None Required
Cable (3)	SS	Copper	Outdoor	None Identified	None Required
			Below Grade	None Identified	None Required
Ground Rod (3)	SS	Copper	Below Grade	None Identified	None Required
Air Terminal Bracing (3)	SS	Carbon Steel	Outdoor	Loss of Material	ISFSI Aging Management Program
Air Terminal Base (3)	SS	Brass	Outdoor	Loss of Material Change in Material Properties	ISFSI Aging Management Program
Air Terminal (3)	SS	Copper	Outdoor	None Identified	None Required
Cable Support (3)	SS	Brass	Outdoor	Loss of Material Change in Material Properties	ISFSI Aging Management Program
Concrete Foundation Mudslab	None	N/A	N/A	N/A	N/A
Dry Film Lubricant	None	N/A	N/A	N/A	N/A
Air Inlet/Outlet Forms (Embedded)	None	N/A	N/A	N/A	N/A
Drain Sleeves (Embedded)	None	N/A	N/A	N/A	N/A

Table 3.2-1 Aging Management Review Results for the Horizontal Storage Modules (HSMs)

Subcomponent (1)	Intended Function	Material Group	Environment	Aging Effects Requiring Management	Aging Management Activity
Alignment Plates (Embedded)	None	N/A	N/A	N/A	N/A
Thermocouples and Wires (Embedded)	None	N/A	N/A	N/A	N/A

(1) Each individual HSM contains the listed subcomponents, with two of the HSMs instrumented with thermocouples.

(2) Aging effect conservatively included to meet current NRC position for 10 CFR 54 plant license renewal (ISG-3).

(3) Lightning Protection System only.

Table 3.3-1 Aging Management Review Results for the Dry Shielded Canisters (DSCs)

Subcomponent (1)	Intended Function	Material Group	Environment (2)	Aging Effects Requiring Management	Aging Management Activity
Canister Body (Casing)	PB, SH, SS, HT	Stainless Steel	Indoor, Not Air-Conditioned	None Identified	None Required
			Air and Gas	None Identified	None Required
End Plugs - Plates	PB, SS, HT	Stainless Steel	Indoor, Not Air-Conditioned	None Identified	None Required
			Air and Gas	None Identified	None Required
End Plugs - Shielding Framework	SS	Stainless Steel	Indoor, Not Air-Conditioned	None Identified	None Required
End Plugs - Shielding	SH	Common Lead	Air and Gas	None Identified	None Required
Top Cover Plate (Including Rolled Ring)	SS, PB	Stainless Steel	Indoor, Not Air-Conditioned	None Identified	None Required
Upper Retaining Ring Assembly - Support Ring	SS	Stainless Steel	Air and Gas	None Identified	None Required
Upper Retaining Ring Assembly - Key	SS	Stainless Steel	Air and Gas	None Identified	None Required
Upper Retaining Ring Assembly - Swagelok Fittings	PB	Stainless Steel	Air and Gas	None Identified	None Required
			Air and Gas	None Identified	None Required

Table 3.3-1 Aging Management Review Results for the Dry Shielded Canisters (DSCs)

Subcomponent (1)	Intended Function	Material Group	Environment (2)	Aging Effects Requiring Management	Aging Management Activity
Upper Retaining Ring Assembly - Siphon and Vent Tube, Tube Plugs	PB	Stainless Steel	Air and Gas	None Identified	None Required
Basket Assembly - Spacer Disks	SS	Stainless Steel	Air and Gas	None Identified	None Required
Basket Assembly - Support Rods	SS	Stainless Steel	Air and Gas	None Identified	None Required
Basket Assembly - Guide Sleeves (Square Cells)	SS, CC	Stainless Steel	Air and Gas	None Identified	None Required
Basket Assembly - Neutron Poison	CC	Aluminum Boron	Air and Gas	None Identified	None Required
Penetration Assembly - Casing (3)	PB	Stainless Steel	Air and Gas	None Identified	None Required
Penetration Assembly - Shielding Disks (3)	SH	Common Lead	Air and Gas	None Identified	None Required
Penetration Assembly Caps (Including Brazing) (3)	PB	Stainless Steel	Indoor, Not Air-Conditioned	None Identified	None Required
			Air and Gas	None Identified	None Required
Penetration Assembly - Weld Sleeve (3)	PB	Stainless Steel	Indoor, Not Air-Conditioned	None Identified	None Required
Penetration Assembly - Shielding Framework Wedge (3)	SS	Stainless Steel	Indoor, Not Air-Conditioned	None Identified	None Required

Table 3.3-1 Aging Management Review Results for the Dry Shielded Canisters (DSCs)

Subcomponent (1)	Intended Function	Material Group	Environment (2)	Aging Effects Requiring Management	Aging Management Activity
Penetration Assembly - Sealant (Wires to Sheathing) (3)	PB	Epoxy-Resin	Indoor, Not Air-Conditioned	None Identified	None Required
			Air and Gas	None Identified	None Required
Thermocouples (3)	None	N/A	N/A	N/A	N/A
Dry Film Lubricant	None	N/A	N/A	N/A	N/A

(1) Each individual DSC contains the listed subcomponents.

(2) Air and Gas environment is inert helium inside each DSC, with possible trace amounts of fission product gases. Temperature and radiation have been considered as described in Section 3.3.3, Environments for the DSCs.

(3) Applicable to two (2) instrumented DSCs only.

Table 3.4-1 Aging Management Review Results for the Irradiated Fuel Assemblies (IFAs)

Subcomponent (1)	Intended Function	Material Group	Environment(2)	Aging Effects Requiring Management	Aging Management Activity
Fuel Rod Cladding and End Caps	CC, HT, PB	Zircaloy-4	Air and Gas	None Identified	None Required
Fuel Rod Pellets and Other Internal Portions	None	N/A	N/A	N/A	N/A
Guide Tubes	SS	Zircaloy-4	Air and Gas	None Identified	None Required
			Air and Gas	None Identified	None Required
Instrument Tube	None	N/A	N/A	N/A	N/A
Spacers – Grid Strips	CC, HT, SS	Zircaloy-4	Air and Gas	None Identified	None Required
Spacers – Spring Strips	None	N/A	N/A	N/A	N/A
Tie Plate Castings (Upper and Lower)	SS	Stainless Steel (including CASS)	Air and Gas	None Identified	None Required
Tie Plate Clamps (Upper)	SS	Stainless Steel	Air and Gas	None Identified	None Required
Tie Plate Leaf Springs and Cap Screws	SS	Inconel	Air and Gas	None Identified	None Required

(1) Each individual IFA contains the listed subcomponents.

(2) Air and Gas environment is inert helium at atmospheric pressure, except for inside the cladding which is pressurized, and minimal fission product gases may also be present. Temperature and radiation have been considered as described in Section 3.4.3, Environments for the IFAs.

Table 3.5-1 Aging Management Review Results for the IF-300 Transfer Cask

Subcomponent	Intended Function	Material Group	Environment [1]	Aging Effects Requiring Management	Aging Management Activity
Body - Inner Cavity Shell	PB, SS	Stainless Steel	Air and Gas	None Identified	None Required
			Air and Gas	None Identified	None Required
Body - Shielding (Castings)	SH	Depleted Uranium (Unalloyed)	Air and Gas	None Identified	None Required
			Air and Gas	None Identified	None Required
Body – Shielding (Diffusion Barrier)	SS	Copper	None [2]	None Identified	None Required
Body - Outer Shell	SS	Stainless Steel (CASS)	Air and Gas	None Identified	None Required
			Treated Water [3]	Loss of Material	Transfer Cask Aging Management Program
Body - Closure Flange (Forging, Top)	PB, SS	Stainless Steel	Air and Gas	None	None Required
			Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
Attachments - Structural Rings	SS	Stainless Steel	Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
			Treated Water [3]	Loss of Material	Transfer Cask Aging Management Program

Table 3.5-1 Aging Management Review Results for the IF-300 Transfer Cask

Subcomponent	Intended Function	Material Group	Environment [1]	Aging Effects Requiring Management	Aging Management Activity
Attachments - Impact Fins (Bottom and Valve Box)	SS	Stainless Steel	Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
Attachments - Lifting Blocks and Pins	SS	Stainless Steel	Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
Attachments - Valve Boxes and Valve Box Covers	SS	Stainless Steel (and CASS)	Air and Gas	None	None Required
			Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
			Treated Water [3]	Loss of Material	Transfer Cask Aging Management Program
Penetrations - Thermowell	SS, PB	Copper	Air and Gas	None	None Required
			Air and Gas	None	None Required
			Outdoor [4] or Indoor, Not Air-Conditioned	None	None Required

Table 3.5-1 Aging Management Review Results for the IF-300 Transfer Cask

Subcomponent	Intended Function	Material Group	Environment [1]	Aging Effects Requiring Management	Aging Management Activity
Penetrations - Vent and Drain Line Pipe and Fittings	PB	Stainless Steel	Air and Gas	None	None Required
Penetrations - Vent and Drain Line Valves (Body Only)	PB	Stainless Steel	Air and Gas	None	None Required
Neutron Shield Water Jacket - Corrugated Jacket (Barrel)	HT, PB, SH	Stainless Steel	Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
			Treated Water [3]	Loss of Material	Transfer Cask Aging Management Program
Neutron Shield Water Jacket - Expansion Tanks	HT, PB	Stainless Steel	Treated Water [3]	Loss of Material	Transfer Cask Aging Management Program
			Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
Neutron Shield Water Jacket - Valves (Body Only)	HT, PB	Stainless Steel	Treated Water [3]	Loss of Material	Transfer Cask Aging Management Program
			Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required

Table 3.5-1 Aging Management Review Results for the IF-300 Transfer Cask

Subcomponent	Intended Function	Material Group	Environment [1]	Aging Effects Requiring Management	Aging Management Activity
Neutron Shield Water Jacket - Pipe and Fittings	HT, PB	Stainless Steel	Treated Water [3]	Loss of Material due to Crevice / Pitting Corrosion	Transfer Cask Aging Management Program
			Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
Neutron Shield Water Jacket - Misc. Supports	SS	Stainless Steel	Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
Extension Collar (Includes Cask Collar Liner Flange)	PB, SS	Carbon Steel	Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material	Transfer Cask Aging Management Program
Cask Lid	PB, SS	Carbon Steel	Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material	Transfer Cask Aging Management Program
Threaded Fasteners	PB, SS	Stainless Steel	Outdoor [4] or Indoor, Not Air-Conditioned	Loss of Material None Identified	Transfer Cask Aging Management Program None Required
Attachments – Valve Box Gaskets	None	N/A	N/A	N/A	N/A

Table 3.5-1 Aging Management Review Results for the IF-300 Transfer Cask

Subcomponent	Intended Function	Material Group	Environment (1)	Aging Effects Requiring Management	Aging Management Activity
Attachments – Flow Baffles	None	N/A	N/A	N/A	N/A
Penetrations – Vent Line Rupture Disk	None	N/A	N/A	N/A	N/A
Sealing (Gasket)	None	N/A	N/A	N/A	N/A
Closure Heads	None	N/A	N/A	N/A	N/A
Fuel Basket	None	N/A	N/A	N/A	N/A
Seal Retainer Assembly	None	N/A	N/A	N/A	N/A
Skid/Enclosure	None	N/A	N/A	N/A	N/A
Thermocouple	None	N/A	N/A	N/A	N/A

(1) Air and Gas environment represents ambient conditions on the interior of the cask, conservatively including connecting surfaces. Temperature and radiation were considered, as described in Section 3.5.3, Environments for the IF-300.

(2) A flame sprayed copper diffusion barrier is present at all steel-uranium (shielding) interfaces.

(3) A 50/50 mixture of demineralized water and ethylene glycol in a sealed water jacket system.

(4) Exterior surfaces are susceptible to intermittent wetting if located outdoors.

Appendix A

Aging Management Programs

TABLE of CONTENTS (Appendix A)

APPENDIX A: AGING MANAGEMENT PROGRAMS

A1.0 INTRODUCTION

Appendix A summarizes the activities that manage the effects of aging for ISFSI subcomponents that have been identified in the License Renewal Application (LRA) as being subject to aging management review. The following aging management programs (AMPs) have been credited for the RNP ISFSI:

- ISFSI Aging Management Program, and
- Transfer Cask Aging Management Program.

The ISFSI Aging Management Program is discussed in Section A2.1 below. Section A2.2 discusses the Transfer Cask Aging Management Program. These sections provide a description of the AMP which includes an introduction, an evaluation of the AMP in terms of the attributes of an effective aging management program, and a summary.

Section 3.0, Aging Management Reviews, provides tables that summarize the results of the AMRs. These tables identify the programs/activities credited for managing the aging effects for each subcomponent listed in the AMR. The identified aging management program manages the aging effects, or the relevant conditions that could lead to the aging effects; applicable to the subcomponent, and provides reasonable assurance that the integrity of the subcomponent will be maintained under current licensing basis (CLB) conditions during the renewed license period.

A2.0 EXISTING AGING MANAGEMENT PROGRAMS

A2.1 ISFSI AGING MANAGEMENT PROGRAM

The RNP ISFSI provides for long-term dry interim storage for irradiated fuel assemblies until such time that the irradiated fuel assemblies may be shipped off site for final disposition. The fuel assemblies are confined in stainless steel canisters. Each canister is protected and shielded by a concrete horizontal storage module. Each canister rests on a steel support rail assembly that is anchored to the walls of the corresponding storage module and restrained inside the storage module. Other steel components provide heat shielding, screens, and attachments, both inside and outside the modules. The ISFSI Aging Management Program includes the concrete and steel members associated with the horizontal storage modules (HSMs).

The purpose of the ISFSI Aging Management Program is to:

- Ensure that no significant degradation to the horizontal storage modules occurs,
- Maintain the air inlets and outlets free from obstructions

A description of the ISFSI Aging Management Program is provided below using each attribute of an effective aging management program as described in the preliminary guidance for the renewal of site specific Part 72 licenses (Reference A3-2):

Scope
The scope of the RNP ISFSI Aging Management Program involves monitoring the exterior surfaces of the ISFSI. It includes visual inspection of the accessible concrete (including below grade concrete, if exposed during excavation) and exposed steel. It also includes monitoring area radiation levels, and airborne and smearable contamination levels at selected areas of the ISFSI, and ensuring that the inlets and outlets do not become blocked.

Preventive Actions
This is primarily a condition monitoring program. With the exception of daily surveillances to ensure HSM inlets and outlets are not obstructed, no preventive actions are performed.

Maintaining the inlets and outlets free from obstruction ensures temperatures are not elevated for prolonged periods, the concrete is not subject to related damage, and overheating of the components inside an HSM is prevented.

Parameters Monitored or Inspected
Consistent with the current NRC position relative to including concrete in an aging management program, the accessible concrete is visually examined for indication of surface deterioration. Degradation could affect the ability of the concrete to provide support to the DSCs, to provide radiation shielding, or to provide a path for heat transfer from each module. The above grade exterior concrete is accessible. The interior concrete and below grade concrete surfaces are inaccessible. The above grade exterior concrete is a leading indicator for the interior concrete. The below grade inaccessible concrete will be visually examined only if the concrete is exposed due to excavation of the backfill material for other maintenance activities.

Accessible steel, that is steel on the external surface of the HSMs and subject to wetting/moisture, is visually examined for the aging effect of loss of material (corrosion). This aging effect could affect the ability of the miscellaneous structural steel to perform its intended function.

Surveillances of area radiation levels, airborne contamination, and smearable contamination are made and compared to established limits. Levels exceeding limits are investigated for potential degradation of the ISFSI components. Increased levels could indicate a reduction in the ability of the concrete and steel to provide adequate radiation shielding, or could indicate a breach in the containment function of the DSC and/or IFA. Dose rates are measured at predetermined HSM locations, including the module surface, at 1 meter from a module, and outside the ISFSI radiation control area. Contamination levels are also monitored at HSM drains.

Daily surveillances are performed by Operations personnel to ensure the air inlets and outlets are free from obstructions, thereby preventing reduced air flow and potential overheating of the components located inside an HSM.

Existing plant procedures are in place for these inspections and surveillances.

Detection of Aging Effects
The examination method used for the accessible concrete and steel is primarily a visual examination at an established frequency. A baseline inspection was performed, with subsequent examination frequencies determined by the Robinson Engineering Section based on the condition observed. The results of this baseline inspection are discussed for the Operating Experience attribute below.

Radiation level readings and contamination levels are determined at established locations by surveillance testing.

Monitoring and Trending
The visual examinations are performed for the ISFSI as described above. A baseline examination was performed. Subsequent examinations, which are determined based on the condition observed, will be performed at the same frequency as other plant structures in the scope of Maintenance Rule (i.e., 10 CFR 50.65), but will not exceed ten years. In addition, the ISFSI System Engineer completes a System Health Report each refueling cycle that includes an indicator of ISFSI performance, reliability, and materiel condition.

The surveillance tests for monitoring radiation level readings and contamination levels could identify a crack in the shielding or a loss of the containment function. This surveillance is performed yearly. If any of the pre-established limits are exceeded, the Robinson Engineering Section is required to be notified.

Acceptance Criteria
A plant procedure provides a set of inspection attributes and acceptance standards for steel and concrete that is commensurate with industry codes, standards, and guidelines. Components are determined to be Acceptable, Acceptable with Deviations, or Unacceptable. Acceptable signifies that the components are free of significant deficiencies or degradation that could lead

to the loss of structural integrity. Acceptable with Deficiencies signifies that components contain deficiencies or degradation that will remain able to perform within design basis allowable load, stress, deflection, or functional limits until the next inspection. Unacceptable signifies components contain deficiencies or degradation that either prevent (or could prevent prior to the next inspection) the ability to perform within design basis allowable load, stress, deflection, or functional limits.

A plant surveillance test procedure provides radiation and contamination level acceptance limits based on the requirements of the ISFSI SAR.

A plant surveillance test procedure provides the acceptance limits for maintaining the inlets and outlets free from obstruction consistent with the ISFSI Technical Specifications.

Corrective Actions
Corrective actions, including root cause determinations and prevention of recurrence, are performed in accordance with the Corrective Action Program. This may include initiation of a Work Request or Nuclear Condition Report (NCR). Corrective actions are taken in a timely manner in accordance with the significance of the NCR. As such, deficiencies are either promptly corrected or are evaluated to be acceptable through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions.

Confirmation Process
Activities initiated in accordance with the implementing procedures for the ISFSI Aging Management Program, such as corrective actions, are subject to Quality Assurance Program controls. Thus, the effectiveness is monitored using Corrective Action Program procedures, review and approval processes, and administrative controls, which are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. Use of these procedures, processes, and controls ensures that corrective actions are taken and are effective.

Administrative Controls
The ISFSI Aging Management Program is subject to Corporate and RNP Corrective Action and Quality Assurance procedures, review and approval processes, and administrative controls. These are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50, and will continue to be adequate for the renewed license (extended storage) period.

Operating Experience
The RNP ISFSI has been in operation since the late-1980s. Examinations and inspections have been performed in accordance with plant procedures.

A review of the Corrective Action Program indicated that any deficiencies identified for the ISFSI have been administrative and were not related to aging mechanisms and effects. Minor corrosion was noted on some of the exterior carbon steel components, which required touchup painting. Loose concrete was observed around some of the embedded plates during the baseline inspection performed in September 2003. Cosmetic repairs were determined to be acceptable.

Visual inspections of the interior surfaces were performed using fiberoptic technology in 1993 and again in 1998. No concrete degradation was observed during either inspection. As discussed in Section 3.1.5, plant specific and industry operating experience, as well as a review of the system files, did not indicate any aging related deficiencies with the ISFSI components, particularly the structural steel and concrete associated with the HSMs.

Summary
Operating experience to date has not indicated any significant degradation to any of the ISFSI components. Inspections and surveillances continue to be implemented that would identify any deficiencies. A Corrective Action Program is in place to track and correct deficiencies in a timely manner.

Continued implementation of the ISFSI Aging Management Program provides reasonable assurance that the aging effects will be managed, such that the intended functions of the ISFSI components, particularly the structural concrete and steel of the HSMs, will be maintained under current licensing basis conditions for the renewed license period.

A2.2 TRANSFER CASK AGING MANAGEMENT PROGRAM

The IF-300 Transfer Cask is used to transport the dry shielded canisters containing the irradiated fuel assemblies from the corresponding horizontal storage modules to the RNP spent fuel pool. There are no plans to load additional irradiated fuel assemblies into the dry shielded canisters or horizontal storage modules. The transfer cask used is a GE IF-300 series shipping cask with a new cask collar and lid. The IF-300 Transfer Cask subcomponents, materials, environments and aging effects requiring management are described in Section 3.5, Aging Management Review Results – IF-300 Transfer Cask.

The purpose of the Transfer Cask Aging Management Program is to ensure that no significant degradation to the IF-300 Transfer Cask occurs, with the focus being on the continuously and intermittently wetted surfaces, as well as carbon steel surfaces, prior to its use for future retrieval of a DSC from the corresponding HSM.

A description of the Transfer Cask Aging Management Program is provided below using each attribute of an effective aging management program as described in the preliminary guidance for the renewal of site specific Part 72 licenses (Reference A3-2):

Scope

The Transfer Cask Aging Management Program is applicable to the IF-300 Transfer Cask and the pertinent subcomponents. The focus of this aging management program is on the stainless steel subcomponents that have continuously wetted surfaces and, conservatively, those external surfaces exposed to outdoor conditions and intermittent wetting. It also conservatively includes carbon steel subcomponents that are exposed to weather and/or other forms of moisture (e.g., humidity).

The program performs visual inspections of the exterior surfaces and monitors the water chemistry of the cask neutron shield water jacket fluid to prevent the corrosion of exposed surfaces.

If the environmental conditions are changed at a later date, such as by moving the cask to indoor storage and/or draining the neutron shield water jacket fluid, there will be no need to continue visual inspection of the stainless steel external surfaces of the cask and/or monitoring the chemistry of the neutron shield water jacket fluid.

Preventive Actions

The Transfer Cask Aging Management Program includes guidance and direction for maintaining a suitable environment that precludes the onset or propagation of a loss of material due to crevice or pitting corrosion for continuously wetted surfaces.

The Transfer Cask Aging Management Program includes the periodic sampling of the IF-300 neutron shield water jacket fluid for acceptable levels of contamination. By maintaining and periodically verifying fluid purity, the concentrations of corrosive contaminants are minimized, removed, or treatments can be added to negate the corrosive tendencies.

As described for the Scope attribute above, draining of the neutron shield water jacket at a later date would eliminate the need for these preventive actions.

Parameters Monitored or Inspected
The parameter inspected by the Transfer Cask Aging Management Program is visual evidence of degradation of external surfaces of the IF-300 and the condition of the neutron shield water jacket fluid.

Visual inspections of external cask, cask collar, and cask lid surfaces are performed periodically, at a minimum of 1 year prior to moving a DSC (if no other inspection has been performed), to ensure that the intended function of the pertinent cask subcomponents are not compromised. Visual inspections look for signs of deterioration (corrosion).

As described under the Preventive Actions above, periodically monitoring the chemical concentrations of the neutron shield water jacket fluid will preclude the conditions for crevice and/or pitting corrosion to occur. The samples will also provide evidence of corrosion as impurities (corrosion products) in the fluid are detected.

As described for the attributes above, a future change in the conditions, such as moving the IF-300 to indoor storage or draining the neutron shield water jacket, would eliminate the need for parameters to be monitored or inspected.

Detection of Aging Effects
Loss of material for stainless steel subcomponents, due to crevice and/or pitting corrosion in wetted locations, and for carbon steel subcomponents, due to general corrosion in moist atmospheric environments, is an aging effect that is managed by this aging management program.

The Transfer Cask Aging Management Program relies upon a visual inspection to determine the physical condition of the exterior surfaces of the IF-300 Transfer Cask, including cask collar and lid, prior to its use for ISFSI unloading or transfers. These inspections check for loss of material (corrosion).

Sampling of the neutron shield water jacket fluid, as described in the Preventive Actions and Parameters Monitored or Inspected attributes above, provide detection of any corrosion that has occurred, since corrosion products would be evidenced as impurities in the fluid.

Monitoring and Trending
Visual inspections will determine the existence of loss of material on the external surfaces of the IF-300, and observations regarding the materiel condition recorded in accordance with inspection procedures and are corrected or evaluated as satisfactory before use of the Transfer Cask. These inspections are either performed periodically or during the preparations for retrieval of a DSC from the corresponding HSM.

With respect to the neutron shield water jacket, the Transfer Cask Aging Management Program will continue periodic sampling and analysis of the fluid for contaminant concentrations (presently once/year) until the 10 CFR 71 license expires. The results of this sampling provide information on the amount of corrosion that may have taken place and on the potential for these concentrations to result in a loss of component intended function.

For the renewed license period after the 10 CFR 71 license expires, the inspection and sampling interval will be determined by the Robinson Engineering Section, with a minimum requirement being to sample and analyze the fluid within one year of retrieval of a DSC from the corresponding HSM.

Evaluation of this information during the preparations for infrequent DSC retrieval/transfers provides both adequate predictability and allows for corrective action prior to the need for the component intended function to be performed.

Acceptance Criteria
The acceptance criteria for the Transfer Cask Aging Management Program for exterior surfaces is no unacceptable loss of material that could result in a loss of component intended function(s), as determined by the Robinson Engineering Section.

The acceptance criterion for neutron shield water jacket fluid contaminant concentrations is acceptable contamination levels (fluid purity).

Unsatisfactory degradation is entered in the Corrective Action Program for resolution.

Corrective Actions
Corrective actions, including root cause determinations and prevention of recurrence, are performed in accordance with the Corrective Action Program.

This may include initiation of a Work Request or Nuclear Condition Report (NCR). Corrective actions are taken in a timely manner in accordance with the significance of the NCR. As such, deficiencies are either promptly corrected or are evaluated to be acceptable through engineering analysis, which provides reasonable assurance that the intended function is maintained consistent with current licensing basis conditions. Each of the implementing procedures associated with the Transfer Cask Aging Management Program is within the scope of the Corrective Action Program.

Confirmation Process

Activities initiated in accordance with the implementing procedures for the Transfer Cask Aging Management Program, such as corrective actions, are subject to Quality Assurance Program controls. Thus, the effectiveness is monitored using Corrective Action Program procedures, review and approval processes, and administrative controls, which are implemented in accordance with the requirements of 10 CFR 50, Appendix B. Use of the procedures, processes, and controls ensures that corrective actions are taken and are effective.

Administrative Controls

The Transfer Cask Aging Management Program is subject to Corrective Action and Quality Assurance Program procedures, review and approval processes, and administrative controls. These are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50, and will continue to be adequate for the renewed license (extended storage) period.

Operating Experience

The RNP ISFSI has been in operation since the late-1980s. Prior to and during that time (since the mid-to-late 1970s), Progress Energy has owned IF-300 series shipping casks that have been utilized for the off-site shipment of spent nuclear fuel. One of these casks was also used for the onsite loading of the RNP ISFSI in the 1980s. Since the initial loading of the RNP ISFSI, no unloading or transfers have occurred. However, the IF-300 series shipping casks have continued to be utilized for 10 CFR 71 spent nuclear fuel shipments and to satisfy the more stringent NRC and DOT requirements for shipments.

In order to meet 10 CFR 71 requirements, and ensure readiness for off-site shipments, annual examinations and inspections have been performed on these casks. These examinations and inspections are currently performed in accordance with a combination of procedures. The overall effectiveness of these inspections in maintaining the condition and functionality of the casks is confirmed by the continued use of the casks and their continued compliance with the associated Certificate of Compliance (Reference 3.6-15). Any deficiencies identified are promptly corrected prior to shipping fuel. This same process will be followed, as applicable, for moving the DSCs from the HSM back to the RNP spent fuel pool (10 CFR 72).

Independent inspections and assessments have also confirmed the effectiveness of activities to manage Transfer Cask aging. This provides objective evidence that the effects of aging have been and will continue to be adequately managed. A discussion of pertinent operating experience is contained in Section 3.1.5, Operating Experience Review for Process Confirmation. Furthermore, the lack of identification of cask degradation through the existing annual inspections is evidence that Transfer Cask activities have been effective in maintaining the condition and functionality of the IF-300 Transfer Casks.

During annual inspections, rupture disks and pipe caps/plugs have failed or have been found to be damaged. The rupture disk and pipe caps/plugs are regularly inspected and replaced on condition, similar to cask closure gaskets, and therefore are not subject to AMR. In addition, both the upper and lower section neutron shield water jacket fluid have been sampled, analyzed, and found to be acceptable by the RNP Chemistry Unit.

With respect to the effectiveness of activities pertaining to the use of the IF-300 series shipping casks, various inspections have been performed by the NRC. In particular, NRC Inspection Report 50-261/96-10 (Reference A3-6) found that the operations, inspection, and maintenance procedures for the spent fuel shipping casks to be adequate and sufficiently detailed. Inspectors also found that the Nuclear Assessment Section (NAS) assessment of spent fuel shipping program readiness to be thorough and probing, with identified weaknesses being properly resolved and effective enhancements implemented prior to conducting shipment activities.

Operating experience for the Transfer Cask Aging Management Program, including past corrective actions, have resulted in program enhancements. A review of the operating experience provided objective evidence that the effects of aging have, and will continue to be, adequately managed following the expiration of the 10 CFR 71 license.

Summary

The Transfer Cask Aging Management Program is credited for the management of relevant conditions that could lead to degradation of IF-300 Transfer Cask subcomponents from the associated aging effects/mechanisms as shown in Table 3.5-1, and for the management of actual degradation. Based on the above, the continued implementation of the Transfer Cask Aging Management Program activities will provide reasonable assurance that aging effects will be managed, such that the IF-300 Transfer Cask subcomponents within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis throughout the renewed license period.

If the environmental conditions are changed at a later date, e.g., by moving the cask to indoor storage or draining the neutron shield water jacket fluid, there will be no need to continue visual inspection of the stainless steel external surfaces or to monitor the chemistry of the neutron shield water jacket.

A3.0 REFERENCES (AGING MANAGEMENT PROGRAMS)

A3-1 10 CFR Part 71, *Packaging and Transportation of Radioactive Material,* Code of Federal Regulations, U.S. Nuclear Regulatory Commission

A3-2 Letter from Mr. Steven Baggett, NRC, to Mr. John Moyer, Serial No. RRA-01-0054, *Preliminary NRC Staff Guidance for 10 CFR Part 72 License Renewal*, May 17, 2001

A3-3 *Independent Spent Fuel Storage Installation Safety Analysis Report*, H.B. Robinson Steam Electric Plant, through Amendment No. 18

A3-4 *IF-300 Shipping Cask Consolidated Safety Analysis Report*, NEDO-10084-5, Duratek, November 2000 (SER Included)

A3-5 Letter from NRC Spent Fuel Project Office to Duratek, *Certificate of Compliance for Radioactive Materials Package*, Subject: Model No. IF-300 Package, Docket Number 71-9001, December 2, 2002 (SER Included)

A3-6 Letter from M.B. Shymlock, NRC, to C.S. Hinnant, CP&L, Subject: NRC Integrated Inspection Report 50-261/96-10, September 16, 1996

Appendix B

Time-Limited Aging Analyses

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APPENDIX B: TIME-LIMITED AGING ANALYSES (TLAAs)

B1.0 INTRODUCTION

RNP's ISFSI license renewal methodology uses the methodology described in the comments on the Preliminary Guidance for License Renewal for Site-Specific Independent Spent Fuel Storage Installations (ISFSIs) that was provided to the NRC by the Surry Nuclear Station on June 26, 2001 (Reference B3-1).

B2.0 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES (TLAAs)

TLAAs are defined in the comments on the Preliminary Guidance for License Renewal for Site-Specific Independent Spent Fuel Storage Installations (ISFSIs) (Reference B3-1) as those licensee calculations and analyses that meet all of the following criteria:

1. *Involve systems, structures, and components within the scope of license renewal.*

2. *Consider the effects of aging.*

3. *Involve time-limited assumptions defined by the current licensing term (e.g., 20 years).*

   The defined licensing term should be explicit in the analyses. Simply asserting that the SSC is designed for a service life or ISFSI life is not sufficient. The assertions must be supported by a calculation, analyses, or testing that explicitly include a time limit.

4. *Must be pertinent to a specific safety determination that exists in the CLB.*

   Such analyses would have initially provided the basis for the applicant's initial safety determination, and without the analyses, the applicant may have reached a different safety conclusion.

5. *Must provide conclusions or a basis for conclusions regarding the capability of the SSC to perform its intended function.*

   Analyses that do not affect the intended functions of the SSCs are not considered TLAAs, and

6. *Must already be contained or incorporated by reference in the current licensing basis (CLB) for the ISFSI.*

Facility-specific documentation contained or incorporated by reference in the CLB includes SARs, SERs, Technical Specifications, fire protection plan/hazards analyses, correspondence to and from the NRC, QA plan, and topical reports included as references in the SAR. Calculations and analyses that are not in the CLB or not incorporated by reference are not TLAAs.

B2.1 IDENTIFICATION PROCESS AND RESULTS FOR THE TLAAs

Both generic and RNP-specific potential ISFSI TLAAs were considered. The ISFSI License Renewal Application for another utility (e.g., Dominion's for the Surry ISFSI), and the associated Requests for Additional Information (RAIs), were reviewed to identify any generic ISFSI TLAAs.

For RNP-specific TLAAs, RNP and vendor calculations and evaluations that could potentially meet the six criteria, as described above, were identified. Keyword and manual searches of current license basis documents were performed, including the Materials License, Technical Specifications, SAR, docketed licensing correspondence, and vendor topical reports incorporated by reference in the SAR.

For completeness, additional searches were conducted of the 10 CFR 71 licensing basis for the IF-300 Transfer Cask. Documents searched included the vendor Certificate of Compliance, topical report, and docketed licensing correspondence.

Documents that meet the six criteria listed in Section B2.0 are the time-limited aging analyses for the RNP ISFSI.

B2.2 EVALUATIONS AND DISPOSITION OF THE IDENTIFIED TLAAs

Evaluations of the TLAAs identified using the process described in Section B2.1 were performed to demonstrate that each identified TLAA for the RNP ISFSI has been dispositioned using one of three different approaches described below:

i. The analysis will remain valid for the renewed license period.
ii. The analysis has been projected to the end of the renewed license period.
iii. The effects of aging on the intended function(s) will be adequately managed for the renewed license period.

The results of these evaluations are discussed below.

B2.2.1 DSC Shell Cracking Due to Fatigue

A calculation analyzed the shell of the dry storage canisters for fatigue. The fatigue cumulative usage factor (CUF) was calculated to be 0.21 for the first 50 years of service. The additional 10 years of service was estimated to increase the usage factor to 0.25. This value is less than the allowable 1.0. Therefore, the CUF has been reanalyzed and projected to be valid for the extended period of operation, in accordance with approach (ii) defined above.

B2.2.2 DSC Penetration Assembly Epoxylite Seal Change in Material Properties Due to Ionizing Radiation

Two of the first three DSCs installed at the RNP ISFSI contained sheathed thermocouples which feed through a penetration plug assembly in the bottom end plug. The thermocouple wires were sealed to the associated sheathing inside and outside the assembly using a heat-cured epoxy-resin (Epoxylite 8611) material. The integrated exposure, assuming that neutron radiation had negligible effect, for Epoxylite at the inner seal after 40 years was previously calculated to be 2.21E11 ergs/g, and after 50 years to be 2.56E11 ergs/g (Reference B3-3, B3-4, and B3-5).

Extrapolation of this information has projected the integrated exposure after 60 years of service to be less than 2.91E11 ergs/g. This is well below the acceptable level of 1E12 ergs/g for this type of resin. Therefore, the effects of ionizing radiation (gamma dose) on the Epoxylite seal inside each instrumented DSC has been reanalyzed and projected to be valid for the license renewal period in accordance with approach (ii) defined above.

The validity of the above assumption regarding the negligible effects of neutron radiation for the epoxy-resin material for the renewed license period was also confirmed. To determine the neutron fluence at the inner seal of the penetration assembly, the neutron fluence at the inside surface of a DSC end plug, through which the penetration assembly passes, was conservatively projected for 60 years of service in accordance with approach (ii) defined above.

The neutron dose level outside a DSC end plug is listed in Table 7.3-2 of the ISFSI SAR (Reference B3-6). The dose level inside the end plug was determined by working backwards using an increase factor of 4.6 to account for the lead shielding, as well as some margin in the dose level. This dose level inside a DSC end plug was then converted using a fast neutron flux to dose conversion factor of 1E-1 mRem/hr/neutrons/cm^2-sec. In 60 years, the neutron flux at the inner seal of the penetration assembly for each instrumented DSC is projected to be 5.2E12 neutrons/cm^2, which is less than the allowable dose of 4E13 neutrons/cm^2 reported in the initial analysis (Reference B3-4 and B3-5).

B2.2.3 DSC Poison Plate Depletion of Boron

Boron depletion can result in reduced neutron absorption capability for neutron poison materials, such as the aluminum boron metal used in the DSC baskets. The total flux was estimated to be 4.1E8 n/cm^2/sec. After 60 years of service, the total flux (fluence) is estimated to be 7.8E17 n/cm^2. Conservatively assuming that the total flux is thermal, the depletion in Boron B-10 level will be negligible (less than 0.3 %). Therefore, DSC Poison Plate Depletion of Boron has been reanalyzed, boron depletion has been determined to be negligible, and this has been projected to be valid for the license renewal period in accordance with approach (ii) defined above.

B2.2.4 5% Boron-Polyethylene Front Access Cover Plate Cracking and Change in Material Properties Due to Ionizing Radiation

The front access cover plate on each horizontal storage module is a carbon steel plate and frame that encloses a lead plate and a 2-inch thick boron-polyethylene sheet. Degradation of the boron-polyethylene sheet due to ionizing radiation was considered a TLAA. The gamma dose rate on the inside face of the access cover plate was calculated to be 23 mR/hr, which results in a gamma radiation dose of less than the allowable 5E8 Rads over a life of 60 years. The neutron dose on the inside face of the front access cover plate was calculated to be 1.14E12 n/cm^2, which is less than the allowable 2.5E13 n/cm^2 over a life of 60 years. Therefore, degradation of the boron-polyethylene sheets due to ionizing radiation has been reanalyzed and projected to be valid for the license renewal period in accordance with approach (ii) defined above.

B2.2.5 Other Potential TLAA Considerations

No TLAAs were identified for the irradiated fuel assemblies. The potential aging mechanisms of creep, stress corrosion cracking, and delayed hydride cracking were considered during the aging management review process documented in Section 3.4, Aging Management Review Results for the IFAs.

The IF-300 Transfer Cask is not continuously exposed to increased temperatures and radiation, only intermittently during 10 CFR 71 or 10 CFR 72 transfer, and the conditions that could result in a weakened state of the Transfer Cask do not exist under the CLB conditions. No TLAAs were identified for the IF-300 Transfer Cask.

B2.3 CONCLUSIONS

The following TLAAs have been identified and will remain valid for the renewed license period in accordance with approach (ii) defined in Section B2.1:

- DSC Shell Cracking Due to Fatigue
- DSC Penetration Assembly Epoxylite Seal Change in Material Properties Due to Ionizing Radiation
- DSC Poison Plate Depletion of Boron
- 5% Boron-Polyethylene Front Access Cover Plate Cracking and Change in Material Properties Due to Ionizing Radiation

B3.0 REFERENCES (TIME-LIMITED AGING ANALYSES)

B3-1 Letter Serial No. 01-367, *Surry Independent Spent Fuel Storage Installation, Comments on NRC Preliminary Guidance,* L. N. Hartz to NRC Document Control Desk, June 26, 2001

B3-2 Letter Serial No. RRA-01-0054, from Mr. Steven Baggett, NRC, to Mr. John Moyer, *Preliminary NRC Staff Guidance for 10 CFR Part 72 License Renewal*, May 17, 2001

B3-3 Letter from L.I. Loflin (CP&L) to NRC Office of Nuclear Material Safety and Safeguards, *H.B. Robinson Independent Spent Fuel Storage Installation Response to Request for Information*, April 28, 1989

B3-4 Letter from L.I. Loflin (CP&L) to NRC Office of Nuclear Material Safety and Safeguards, *H.B. Robinson Independent Spent Fuel Storage Installation Submittal of Corrected Information,* June 2, 1989

B3-5 Letter from L.C. Rouse (NRC) to L.W. Eury (CP&L), *Re: Carolina Power and Light Company - Amendment of Materials License No. SNM-2502, Amendment No. 7*, June 22, 1989 (includes the SER for the use of an Epoxy Resin to Seal Thermocouple Leads)

B3-6 *Independent Spent Fuel Storage Installation Safety Analysis Report*, H.B. Robinson Steam Electric Plant, through Amendment No. 18

B3-7 *IF-300 Shipping Cask Consolidated Safety Analysis Report*, NEDO-10084-5, Duratek, November 2000 (SER Included)

B3-8 Letter from NRC Spent Fuel Project Office to Duratek, *Certificate of Compliance for Radioactive Materials Package*, Subject: Model No. IF-300 Package, Docket Number 71-9001, December 2, 2002 (SER Included)

Appendix C

Safety Analysis Report Supplement and Changes

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APPENDIX C: SAFETY ANALYSIS REPORT SUPPLEMENT AND CHANGES

C1.0 INTRODUCTION

This appendix provides a supplement and identifies pertinent changes to the RNP ISFSI Safety Analysis Report (SAR). Section C2.0 of this appendix contains proposed changes to the existing RNP ISFSI SAR. Section C3.0 of this appendix contains a proposed new section for the ISFSI SAR to be added under Section 9, Conduct of Operations. The new section 9.7, Aging Management, provides a summarized description of the activities for managing the effects of aging of ISFSI SSCs. This proposed new SAR section will also present the evaluations of time-limited aging analyses (TLAAs) for the renewed license period.

C2.0 CHANGES TO EXISTING RNP ISFSI SAR INFORMATION

C2.1 ISFSI SAR SECTION 1.0 CHANGES

Section 1.1:

- Change reference to 20 years in last paragraph to 60 years, and delete "under the initial license," as follows:

The fuel assemblies to be stored in the ISFSI are currently located in the HBR2 spent fuel pool and were irradiated in the HBR2 reactor. Seven fuel assemblies are stored in each dry shielded canister. One dry shielded canister is stored in each concrete module. The license application by CP&L requests a license to construct and operate a total of eight modules. CP&L initially intends to construct three modules. Construction of the first three modules will take approximately one year. In accordance with the DOE agreement, the first year of operation of the three module ISFSI will be part of a test program. Normal operation of the facility will continue past the first year for up to ~~20~~ 60 years ~~under the initial license~~ until a permanent federal repository is available to store the spent fuel. An additional five modules have been constructed adjacent to the original three.

Section 1.2.2:

- Add "and back, if necessary" to the first sentence, as follows. This allows for retrieval of the DSC from the HSM.

The major operating systems of the ISFSI are those required for fuel handling and transport of the fuel from the spent fuel pool to the ISFSI and back, if necessary. General operations are outlined in Table 1.2-2 and the primary design parameters of the required systems are listed in Table 1.2-3. The fuel handling operations involving the cask (i.e., fuel loading, drying, trailer loading, etc.) and the remaining operations (cask-HSM alignment and DSC transfer) are unique to the ISFSI. Procedures for these activities were developed using H. B. Robinson fuel shipment procedures and experience gained during testing of the ISFSI.

Table 1.2-1:

- Revise Table 1.2.1 to change Design Life to 60 years and Footnote 2 to explain license renewal to 60 years, as follows:

TABLE 1.2-1

DESIGN PARAMETERS FOR THE HBR ISFSI

Category	Criterion or Parameter	Value
Fuel Acceptance Criteria	Fissile Content	3.5% Fissile (U-235 Equivalent)
	Radiation Source	
	Gamma	5.73×10^{15} photons/sec/assembly[1]
	Neutron	1.67×10^{8} neutrons/sec/assembly
	Heat Load	1 KW/Assembly
Dry Shielded Canister	Capacity per Canister	7 PWR Fuel Assemblies
	Size	
	Length (typical)	4.56m (179.5 in)
	Diameter	0.94m (36.9 in)
	Temperature (max. fuel rod clad)	380°C (716°F)
	Cooling	Natural Convection
	Design Life	60 ~~50~~ Years[2]
	Material	304 Stainless Steel with Lead End-Shields
	Internal Helium Pressure	0.0 psig +/- 0.5 psig
Horizontal Storage Module	Capacity	1 Dry Shielded Canister per Module
	Unit Size	3 modules per Unit
	Length	6.71m (22.00 ft)
	Height	3.81m (12.50 ft)
	Width	7.54m (24.75 ft)
	Unit Size	5 modules per Unit
	Length	6.71m (22.00 ft)
	Height	3.81m (12.50 ft)
	Width	11.86m (38.92 ft)
	Surface Radiation Dose Rate (average on contact)	20 mrem/hr
	Material	Reinforced Concrete
	Design Life	60 ~~50~~ years[2]

[Footnotes continued from previous page:]

[1] Actual design limits are for seven assemblies in the DSC with source rates of 1.17×10^{9} neutrons/sec/DSC and 4.01×10^{16} photons/sec/DSC.

[2] Expected life is much longer (hundreds of years); however, initial license application was ~~is~~ for 20 years only. Renewed license is for 60 years. ~~Future amendments may seek to extend the life~~

Section 1.3.1.3:

- Revise the first sentence to add the words "and from" to allow for transfer of the DSC to and from the HSM.

The transfer cask used with the ISFSI provides shielding during the DSC drying operation and during the transfer to and from the HSM. For the HBR ISFSI, the IF-300 cask (which CP&L owns) licensed under 10 CFR 71 as a transportation cask will be used (Reference 1.3).

C2.2 ISFSI SAR SECTION 3.0 CHANGES

Section 3.3.3.2:

- Change the second sentence in the first paragraph to state that two of the DSCs and HSMs "were instrumented" rather than "will be instrumented."

The HBR ISFSI is designed to be totally passive and therefore, no safety related instrumentation is required for operation of the facility. However, two of the DSCs and HSMs were ~~will be~~ instrumented for experimental purposes only for the one year test period (Agreements with DOE and EPRI).

Section 3.5:

- Delete the second sentence of the second paragraph. The Transfer Cask licensed to 10 CFR 71 currently expires in 2005. The first sentence adequately explains the current shipping philosophy.

Shipping cask design and transportation requirements will depend on the regulations in effect at the time when the federal repository begins receiving spent fuel. ~~In the absence of new regulations, the existing GE IF-300 shipping cask owned by Carolina Power & Light Company would be used to transport the canisters.~~

C2.3 ISFSI SAR SECTION 5.0 CHANGES

Section 5.1.1.8:

- Delete the last sentence and add a sentence to provide for flexibility for disposition of the fuel, depending on the shipping and storage regulations in effect at the time.

The fuel is in a safe configuration in the DSC within the IF-300 shipping cask. During the one year demonstration phase, provisions will be made for the DSC to be returned to the spent fuel pool, if necessary. Possible conditions upon which the DSC would be returned include exceeding the design limits shown in section 10.2 and 10.3. When returned, the DSC could be shipped offsite, the fuel could be unloaded and shipped offsite by using alternate shipping casks, or stored on site depending on the shipping and storage regulations in effect at the time. ~~Additionally, if shipping were required, the DSC would be returned to the decontamination area or the fuel pool for removal of the cask collar and placement of the BWR head on the IF-300.~~

C2.4 ISFSI SAR SECTION 8.0 CHANGES

Section 8.0:

- In the fourth paragraph, delete reference to the three module unit, because RNP installed eight units, which is consistent with the generic design.

As discussed in Section 3.2 of this Safety Analysis Report (SAR), some design features of the HBR ISFSI are unique and differ from those of the NUHOMS generic concept. In particular, the HSM has ~~is a three module unit with~~ a rear access penetration, whereas the generic concept ~~is an eight module unit~~ is without any rear access. However, as discussed earlier the methodology of the structural evaluation of the HSM under the above categories of design events as utilized by the referenced report is such that it will conservatively envelop any modular stacking arrangement including the three unit concept of the HBR site. Hence, the stress evaluation and the analytical results presented in Chapter 8 of the referenced report for the NUHOMS modules are fully applicable to the site specific HSM.

C3.0 NEW RNP ISFSI SAR SECTION

The following information will be integrated into the ISFSI SAR Section 9.7 to document aging management programs credited in the RNP ISFSI license renewal review, and time-limited aging analyses evaluated to demonstrate acceptability during the period of extended operation. The following information will be numbered sequentially within the new ISFSI SAR Section 9.7, Aging Management.

C3.1 AGING MANAGEMENT PROGRAMS

An assessment of the RNP ISFSI and Transfer Cask inspection and monitoring activities identified existing activities necessary to provide reasonable assurance that 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.

C3.1.1 ISFSI Aging Management Program

The RNP ISFSI Aging Management Program involves monitoring the exterior surface of the horizontal storage module. It includes visual inspection of the accessible concrete (including below grade concrete, if exposed during excavation) and any exposed steel embedments and attachments. It also includes monitoring the area radiation levels, airborne contamination, and smearable contamination at selected areas of the ISFSI. This is a primarily a condition monitoring program, however, preventive actions include a daily surveillance to ensure horizontal storage module air inlets and outlets are not obstructed.

C3.1.2 Transfer Cask Aging Management Program

The Transfer Cask Aging Management Program performs visual inspections of the exterior surfaces of the IF-300 Transfer Cask, cask collar, and lid. The program also monitors the water chemistry of the cask neutron shield water jacket to prevent the corrosion of exposed internal surfaces.

Note: If the environmental conditions are changed at a later date, such as by moving the cask to indoor storage or draining the neutron shield water jacket fluid, there will be no need to continue visual inspections of the external stainless steel surfaces or to continue monitoring the chemistry of the neutron shield water jacket.

C3.2 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.

C3.2.1 DSC Shell Cracking Due to Fatigue

The Dry Shielded Canister shell fatigue cumulative usage factor (CUF) was calculated to be 0.21 for the first 50 years of service. The additional 10 years of service was estimated to increase the usage factor to 0.25. This value is less than the allowable 1.0. Therefore, the CUF has been reanalyzed and projected to be valid for the renewed license period.

C3.2.2 DSC PENETRATION ASSEMBLY EPOXYLITE SEAL CHANGE IN MATERIAL PROPERTIES DUE TO IONIZING RADIATION

Two of the dry shielded canisters contain sheathed thermocouple feeds through a penetration plug assembly in the bottom end plug, which were sealed (wire to sheathing) using a heat-cured epoxy-resin material. The integrated exposure for epoxy-resin after 40 years was calculated to be 2.21E11 ergs/g, and after 50 years calculated to be 2.56E11 ergs/g. The integrated exposure after 60 years of service was projected to be less than 2.91E11 ergs/g. This is well below the acceptable level of 1E12 ergs/g for this type of resin. In addition, the validity of the assumed negligible effects of neutron radiation was confirmed for the renewed license period. Therefore, the effects of ionizing radiation on the epoxy-resin has been reanalyzed and projected to be valid for the renewed license period.

C3.2.3 DSC POISON PLATE DEPLETION OF BORON

Boron depletion can result in reduced neutron absorption capability for neutron poison materials, such as the aluminum boron metal used in the DSC baskets. The total flux was estimated to be 4.1E8 n/cm^2/sec. After 60 years of service, the total flux (fluence) is estimated to be 7.8E17 n/cm^2. Conservatively assuming that the total flux is thermal, the depletion in Boron B-10 level will be negligible (less than 0.3 %). Therefore, DSC Poison Plate Depletion of Boron has been reanalyzed, boron depletion has been determined to be negligible, and this has been projected to be valid for the renewed license period.

C3.2.4 5% BORON-POLYETHYLENE FRONT ACCESS COVER PLATE CRACKING AND CHANGE IN MATERIAL PROPERTIES DUE TO IONIZING RADIATION

The front access cover plate on each horizontal storage module is a carbon steel plate and frame that encloses a lead plate and a 2-inch thick boron-polyethylene sheet. Degradation of the boron-polyethylene sheet due to ionizing radiation may result in reduced shielding capability. The gamma dose rate on the inside face of the access cover plate was calculated to be 23 mR/hr, which results in a gamma radiation dose of less than the allowable 5E8 Rads over a life of 60 years. The neutron dose on the inside face of the front access cover plate was calculated to be 1.14E12 n/cm^2, which is less than the allowable 2.5E13 n/cm^2 over a life of 60 years. Therefore, degradation of the boron-polyethylene sheets due to ionizing radiation has been reanalyzed and projected to be valid for the renewed license period.

C4.0 REFERENCES (ISFSI SAR SUPPLEMENT AND CHANGES)

None

Appendix D

Technical Specifications Changes

APPENDIX D: TECHNICAL SPECIFICATIONS CHANGES

Section 72.42 of 10 CFR 72 provides the requirements for renewal of an independent spent fuel storage installation license. The preliminary guidance for license renewal for site-specific ISFSIs requires that an application for license renewal include any Technical Specifications change, or addition, that are necessary to manage the effects of aging during the renewal period. Review of the information provided in the RNP license renewal application and in the ISFSI Technical Specifications has confirmed that no changes to the ISFSI Technical Specifications are needed.

Appendix E

Environmental Report Supplement

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ACRONYMS AND ABBREVIATIONS

ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
CP&L	Carolina Power & Light
DOE	U.S. Department of Energy
DSC	Dry shielded canister
EA	Environmental Assessment
EIS	Environmental Impact Statement
FES	Final Environmental Statement
FSAR	Final Safety Analysis Report
HSM	Horizontal storage module
FWS	U.S. Fish and Wildlife Service
GEIS	Generic Environmental Impact Statement
IFA	Irradiated fuel assembly
INPO	Institute of Nuclear Power Operations
ISAR	ISFSI Safety Analysis Report
ISFSI	Independent Spent Fuel Storage Installation
MWd/MT	Megawatt days per metric tons (Uranium)
NEPA	National Environmental Policy Act
NMSS	Nuclear Material Safety and Safeguards
NRC	U.S. Nuclear Regulatory Commission
PEC	Progress Energy Carolinas
psig	pounds per square inch, gauge
PWR	Pressurized water reactor
RNP	Robinson Nuclear Plant
SAR	Safety Analysis Report
SCDNR	South Carolina Department of Natural Resources
SHPO	State Historic Preservation Officer
SSR	Secondary state road
TLD	Thermoluminescent dosimeter
TMI	Three Mile Island
TIGER	Technology Integrated Geographic Encoding and Referencing
UFSAR	Updated Final Safety Analysis Report

E1.0 INTRODUCTION

E1.1 PURPOSE AND NEED FOR THE PROPOSED ACTION

The U.S. Nuclear Regulatory Commission (NRC) licenses the operation of independent spent fuel storage installations (ISFSIs) for storing power reactor spent fuel and associated radioactive materials in accordance with the Atomic Energy Act of 1954 (42 USC 2011, et seq.) and NRC implementing regulations. Carolina Power and Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc. (PEC), operates the H. B. Robinson ISFSI pursuant to NRC license SNM-2502 (NRC 1986a). This site-specific license will expire August 31, 2006. PEC has prepared this supplemental environmental report in conjunction with its application to NRC to renew the license, as provided by the following NRC regulations:

- Title 10, Energy, Code of Federal Regulations (10 CFR), Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, Sections 72.42, Duration of License; Renewal, and 72.34, Environmental Report, and
- 10 CFR 51, Environmental Protection Requirements for Domestic Licensing and Regulatory Functions, Section 51.60, Environmental Report - Materials License

The purpose and need for the proposed action, renewal of the H. B. Robinson ISFSI license, is to provide an option that allows for interim spent fuel storage beyond the term of the current ISFSI license. The storage is interim pending the availability of a federal repository for permanent disposal.

The renewed ISFSI license would permit 40 additional years of storage beyond the currently licensed period. The additional time would give PEC approximately 16 years beyond the current station operating license to arrange for shipment to the federal repository[1].

E1.2 ENVIRONMENTAL BACKGROUND

NRC has extensive experience evaluating environmental impacts from ISFSIs in accordance with the National Environmental Policy Act (NEPA). This experience includes the following:

- Preparing an environmental impact statement in conjunction with establishing the ISFSI regulation (10 CFR 72) and two environmental assessments for substantive revisions to the regulation

[1] Assumes NRC renewal of the station operating license, the subject of NRC review of a June 14, 2002, application.

- Preparing an environmental impact statement for an ISFSI at the Idaho National Engineering and Environmental Laboratory for Three Mile Island Unit 2 spent fuel
- Preparing a generic environmental impact statement for station license renewal
- Preparing environmental assessments for site-specific ISFSI licenses at nine nuclear power plant sites
- Approving ISFSI operation under general license provisions at 16 nuclear power plant sites
- Preparing an environmental impact statement for a private, commercial ISFSI (Skull Valley)
- Issuing, and twice updating, its waste confidence decision (codified as 10 CFR 23) that considers, among other things, operation of spent fuel storage for 30 years beyond the term of a renewed reactor operating license.

Table E1-1 identifies each NEPA evaluation and summarizes its conclusion. In the 22 years represented by these evaluations, NRC has not identified any significant environmental impact associated with ISFSI operation.

The U.S. Department of Energy (DOE) has also analyzed ISFSI environmental impacts. As part of its evaluation of the impact of constructing a national repository for spent nuclear fuel, DOE analyzed environmental impacts from a no-action alternative that included leaving spent nuclear fuel in power plant ISFSIs (DOE 2002). The analysis accounted for the fuel at all operating nuclear power plants, including H. B. Robinson Steam Electric Plant, Unit No. 2, also known as the Robinson Nuclear Plant (RNP). DOE concluded that environmental impacts would be small for at least 100 years and, with appropriate institutional controls, could continue to be small for thousands of years.

Table E1-1.
NRC Environmental Reviews of Spent Fuel Storage in ISFSI[a]

Date	Subject	Conclusion
1979	Establishment of regulation 10 CFR 72 authorizing spent fuel storage at an ISFSI (NRC 1979)	Regulations in place will assure protection of the environment
1984	Final Waste Confidence Decision (NRC 1984a)	Spent fuel generated in any reactor can be stored without significant impacts for at least 30 years beyond expiration of that reactor's operating license at that reactor's spent fuel storage basin or at an onsite or offsite ISFSI
1984	Revision of regulation 10 CFR 72 to authorize offsite ISFSI (monitored retrievable storage) (NRC1984b)[b]	Environmental consequences of long-term storage not significant
1985	Surry ISFSI EA (NRC 1985)	No significant environmental impact
1986	Robinson ISFSI EA (NRC 1986b)	No significant environmental impact; see also (CP&L no date)
1988	Revision of regulation 10 CFR 72 to authorize ISFSI general license (NRC 1988a)	No significant environmental impact
1988	Oconee ISFSI EA (NRC 1988b)	No significant environmental impact
1990	Review and Final Revision of Waste Confidence Decision (NRC 1990)	Spent fuel generated in any reactor can be stored without significant impacts for at least 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor at its spent fuel storage basin or at an onsite or offsite ISFSI
1991	Fort St. Vrain ISFSI EA (NRC 1991a)	No significant environmental impact
1991	Calvert Cliffs ISFSI EA (NRC 1991b)	No significant environmental impact
Undated	Prairie Island ISFSI EA (NRC undated)	No significant environmental impact
1994	Rancho Seco ISFSI EA (NRC 1994)	No significant environmental impact
1996	Trojan ISFSI EA (NRC 1996a)	No significant environmental impact
1996	Nuclear plant license renewal (NRC 1996b)	No significant environmental impact from spent fuel storage
1997	North Anna ISFSI EA (NRC 1997)	No significant environmental impact
1998	TMI 2 ISFSI at Idaho National Engineering and Environmental Laboratory EIS (NRC 1998)	Small and acceptable effects on the environment
1999	Waste Confidence Decision Review Status (NRC 1999)	No significant and unexpected events have occurred that would cast doubt on NRC's waste confidence findings
2001	Skull Valley ISFSI EIS (NRC 2001)	Environmental impacts from operation would be small

a. In addition to the site-specific reviews listed, NRC prepares an environmental assessment for each dry storage cask listed in 10 CFR 72.214. Currently, 13 casks are listed.

b. EA completed in 1984; rule revised in 1988.

EA = Environmental Assessment
EIS = Environmental Impact Statement
ISFSI = Independent Spent Fuel Storage Installation
TMI = Three Mile Island

E1.3 ENVIRONMENTAL REPORT SCOPE AND METHODOLOGY

NRC regulation 10 CFR 72.42 provides for ISFSI license renewal, and regulation 72.34 requires an application to include an environmental report that meets the requirements of 10 CFR 51 Subpart A. In Subpart A, 10 CFR 51.60 requires that the environmental report be a separate document entitled "Supplement to Applicant's Environmental Report" and specifies environmental report contents. The regulation focuses on presenting any significant environmental change from the previously submitted environmental report. Progress Energy has prepared Table E1-2 to verify conformance with the regulatory requirements. For each requirement of 10 CFR 51.60, including 10 CFR 51.45 as adopted by reference, Table E1-2 indicates which environmental report section provides responsive information.

Table E1-2.
Sections of this Environmental Report that Respond to License Renewal Environmental Regulatory Requirements of 10 CFR 51

Regulatory Requirement	Responsive Environmental Report Section	
10 CFR 51.60(a)		Entire Document
10 CFR 51.45(a) description of proposed action	3.0	Proposed Action
10 CFR 51.45(a) statement of purposes	1.1	Purpose and Need for the Proposed Action
10 CFR 51.45(a) affected environment	2.0	Site and Environmental Interfaces
10 CFR 51.45(b)(1)	4.0	Environmental Consequences and Mitigating Actions
10 CFR 51.45(b)(2)	4.0	Environmental Consequences and Mitigating Actions
	6.3	Unavoidable Adverse Impacts
10 CFR 51.45(b)(3)	7.0	Alternatives
	8.0	Comparison of the Impacts of License Renewal with the Alternatives
10 CFR 51.45(b)(4)	6.5	Short-Term Use Versus Long-Term Productivity of the Environment
10 CFR 51.45(b)(5)	6.4	Irreversible and Irretrievable Resource Commitments
10 CFR 51.45(c) alternatives for reducing or avoiding effects	4.0	Environmental Consequences and Mitigating Actions
	6.2	Mitigation
10 CFR 51.45(d)	9.0	Status of Compliance
10 CFR 51.53(c)(3)(iv)	5.0	Assessment of New and Significant Information

CFR = Code of Federal Regulations

In determining the appropriate scope for the ISFSI license renewal environmental report, PEC had to determine an appropriate license renewal term. The ISFSI license authorizes CP&L to store 56 spent fuel assemblies removed from the RNP.

PEC and NRC intend for the storage at the ISFSI to be interim pending availability of a federal repository. However, there is uncertainty regarding when a repository will be available and the schedule under which it will accept spent fuel shipments. The repository schedule drives the ISFSI schedule; the longer it takes for the repository to begin accepting spent fuel shipments, the longer the ISFSI must store spent fuel. The earliest that DOE anticipates accepting shipments at the Yucca Mountain repository is the year 2010 (DOE 2003). PEC believes it is prudent to plan for the possibility that shipments to a federal repository will be delayed until later in the first quarter of the 21st century, consistent with the NRC's finding in its Waste Confidence Rule (NRC 1984a). Based on the inventory of spent fuel that Progress Energy will have at its four nuclear facilities when a repository becomes available, and the time that Progress Energy will need to eliminate the spent fuel inventory from all its facilities, PEC proposes the year 2046 as the end of the period of extended ISFSI operation. This is 16 years after expiration of the proposed renewed operating license for RNP. This environmental report analyzes renewal of the ISFSI license assuming that shipments will be according to plans and consistent with the NRC's Waste Confidence Rule.

As mentioned previously, NRC has prepared a generic environmental impact statement (GEIS) for station license renewal (NRC 1996b). The GEIS considers spent fuel storage during the license renewal of an ISFSI license as being an inherent part of reactor license renewal. The GEIS describes spent fuel generation and storage during current station license terms and during station license renewal terms. This discussion includes the RNP spent fuel (NRC 1996b, Section 6.4.6). The GEIS generically discusses land use and terrestrial resources; water use and aquatic resources; radiological impacts of normal operations, off-normal events and accidents; off-site dose; occupational dose; other effects; and, resources committed.

ISFSIs located at nuclear plant sites, such as RNP, share many attributes such as affected environment, monitoring and reporting programs, and staffing with the power plant. The CP&L application to renew the RNP operating license includes an environmental report (CP&L 2002). Because the GEIS addresses ISFSI operations during a station license renewal term, and because CP&L recently prepared an environmental report for station license renewal, CP&L has adopted by reference [per 10 CFR 51.53(a) and 51.60] in the ISFSI license renewal environmental report appropriate material from the GEIS and the station license renewal environmental report. The ISFSI license renewal is for a period 16 years longer than the operating license renewal term. For instances where the analysis in the plant operating license environmental report did not adequately address impacts that might occur during the additional 16 years, this environmental report performs those analyses.

The ISFSI license renewal environmental report is comprised of nine chapters. This chapter describes the purpose and need for the proposed action, i.e., renewal of the ISFSI operating license. Chapter 2.0 describes the environs affected by ISFSI operations, and Chapter 3.0 describes pertinent aspects of the installation. Chapter 4.0 provides results of the analyses of impacts on the environment from ISFSI license renewal. Chapter 5.0 describes the process Progress Energy used to identify any new and significant information regarding environmental impacts. Chapter 6.0 summarizes the impacts of license renewal and mitigating actions. Chapter 7.0 describes feasible alternatives to the proposed action and their environmental impacts. Chapter 8.0 compares the impacts of license renewal with those alternatives. Chapter 9.0 discusses ISFSI compliance with regulatory requirements.

E1.4 REFERENCES

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Robinson Independent Spent Fuel Storage Installation. March. Washington, D.C.

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E2.0 SITE AND ENVIRONMENTAL INTERFACES

E2.1 LOCATION AND FEATURES

The H. B. Robinson Steam Electric Plant is located in northeastern South Carolina, approximately five miles west-northwest of Hartsville. The nearest large city is Columbia, South Carolina, approximately 55 miles west-southwest. The plant site is located approximately 30 miles south of the North Carolina border and 90 miles from the Atlantic Ocean. Figures E2-1 and E2-2 are plant 50-mile and 6-mile vicinity maps, respectively.

The Robinson Nuclear Plant (RNP) site encompasses more than 5,000 acres of property in northwestern Darlington and southwestern Chesterfield Counties, including the 2,250-acre cooling water impoundment, Lake Robinson. The plant is comprised of Unit 1, a coal-fired power plant, and Unit 2, a nuclear power plant known as Robinson Nuclear Plant (RNP). The Darlington County Internal Combustion Turbine Electric Plant is also on the CP&L property, slightly more than one mile north of Units 1 and 2. The photograph below shows the H. B. Robinson Steam Electric Plant and Lake Robinson, looking north. The ISFSI is approximately 600 feet to the west (left) of the Unit No. 2 containment structure.

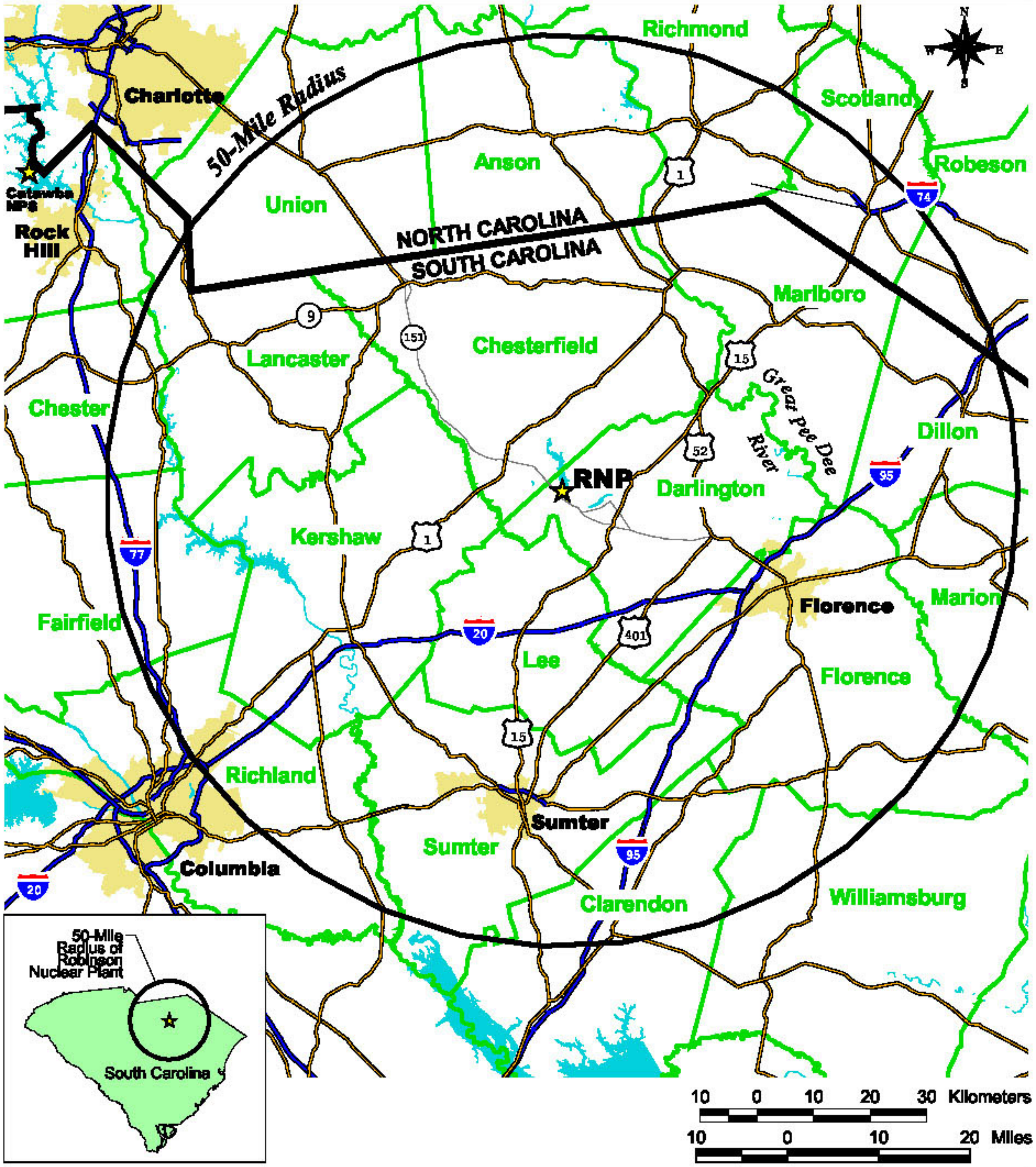
H. B. Robinson Steam Electric Plant and Lake Robinson

The RNP exclusion distance and low population distances are 1,400 feet and 4.5 miles, respectively, from the containment building. Other site features include auxiliary buildings, the intake structure, liquid storage tanks, the turbine structure, the radwaste facility, the fuel handling building, waste retention basins, the switchyard, associated transmission lines, and all facilities related to the coal-fired power plant.

CP&L owns land around the impoundment, but leases to adjacent property owners for access to the impoundment. As a result, the eastern side of Lake Robinson is developed with homes, recreational areas, a marina, and public access points. CP&L leases the northern portion of its property to the State of South Carolina, which manages it in conjunction with its adjacent Sandhills State Forest. CP&L manages the balance of the undeveloped property for timber production.

The region within six miles of the site includes Darlington, Chesterfield, and a portion of Lee Counties, and the City of Hartsville. Topography consists of rolling sand hills interspersed with watercourses. The area surrounding the site is predominantly rural, consisting of farmlands and woodlands with intermittent industrial sites. The site surface soil is sandy and surface water drains to the impoundment.

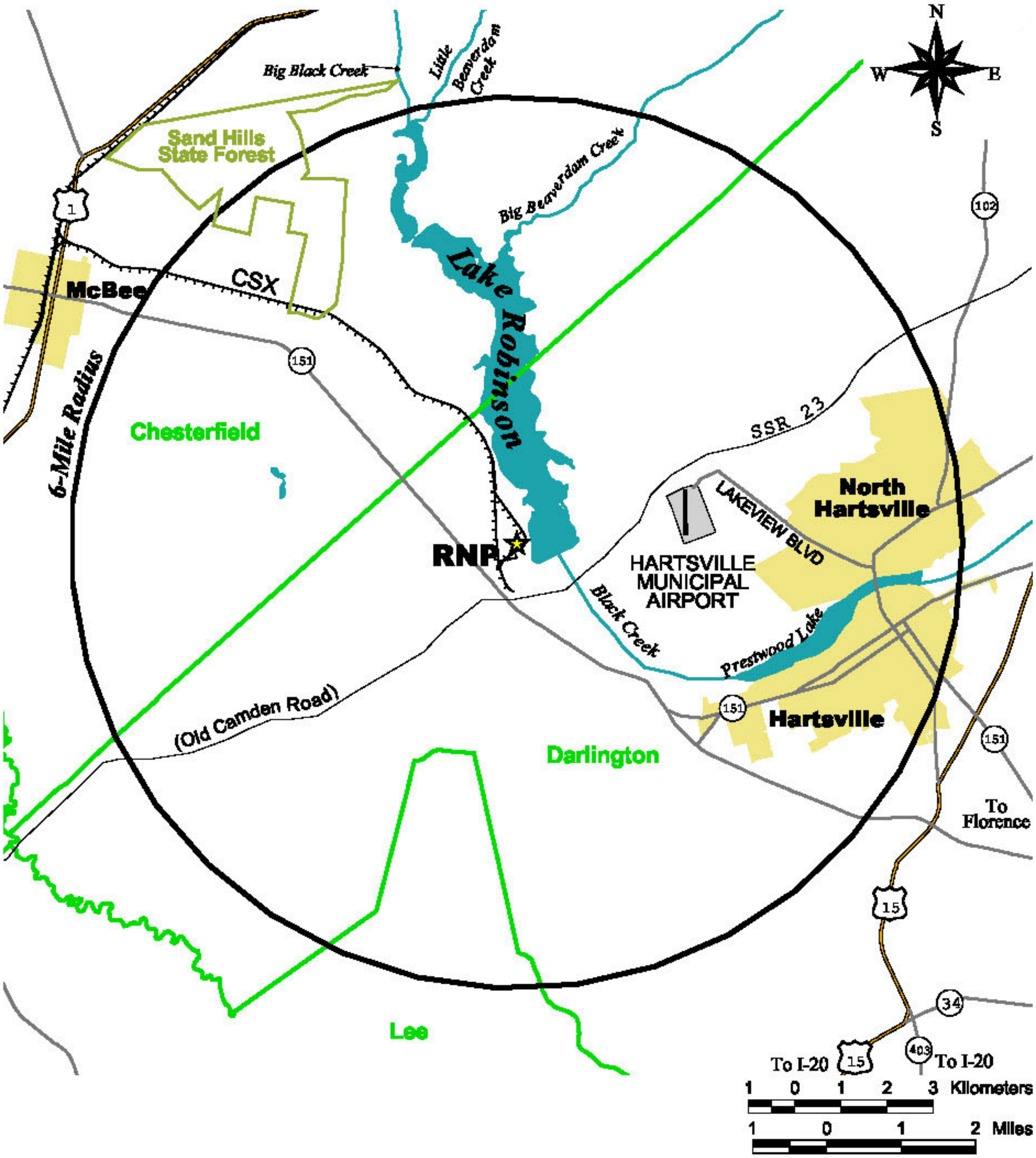
Section E3.1 describes key features of the ISFSI.



LEGEND

- County Boundaries
- Lakes and Rivers
- Urban

FIGURE E2-1.
50-Mile Vicinity Map



LEGEND

Railroad
County Boundaries
Lakes and Rivers
Cities
SSR = Secondary State Route

FIGURE E2-2.
6-Mile Vicinity Map

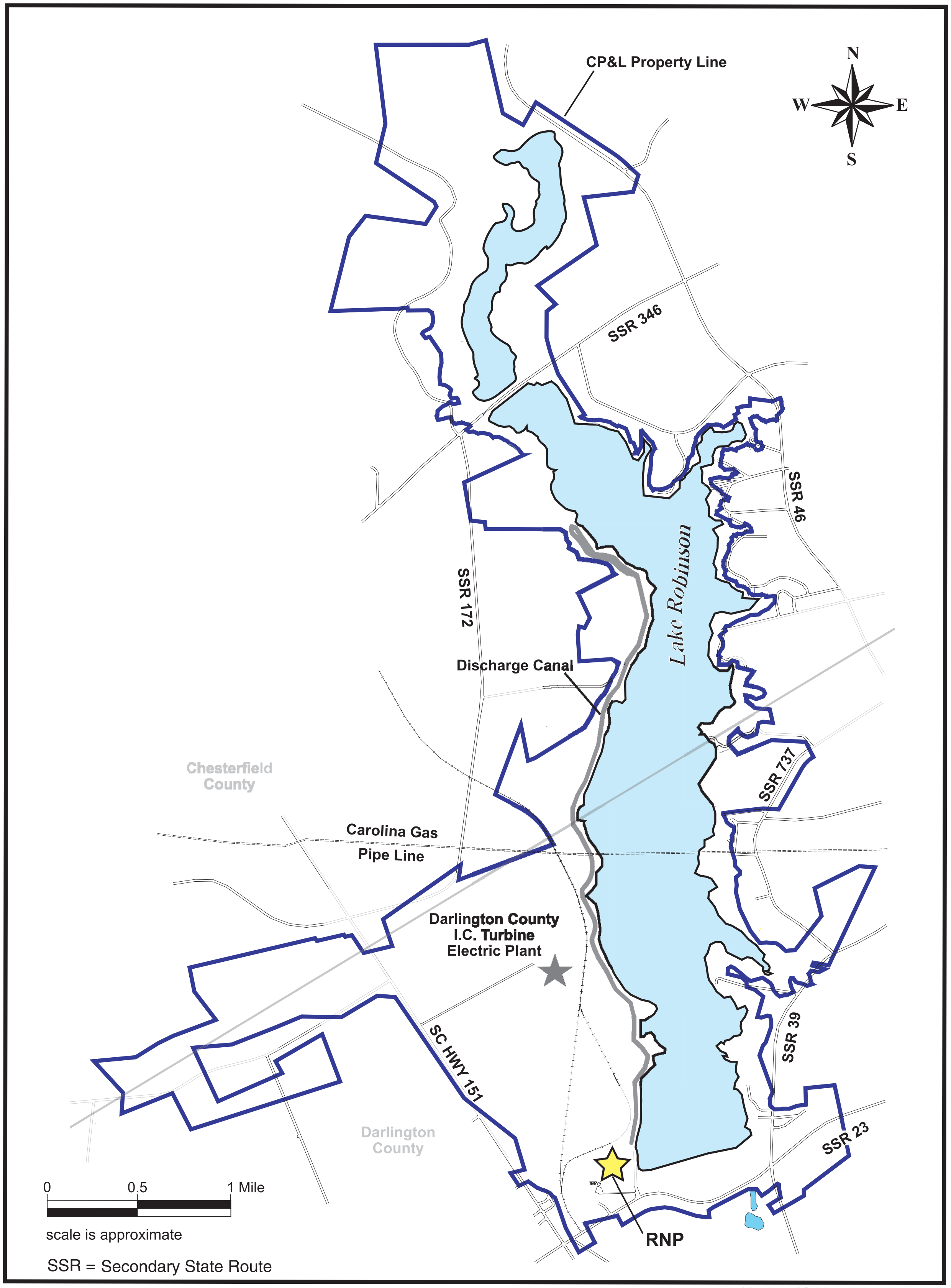


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FIGURE E2-3. CP&L Lake Robinson Property

E2.2 GEOLOGY, SOILS, SURFACE WATER, AND GROUNDWATER

Piedmont crystalline basement rock overlain with approximately 460 feet of unconsolidated coastal plain sediment is beneath the site. The sediments comprise about 30 feet of surface alluvium over 430 feet of the Middendorf formation. The Middendorf formation is made up of sands, silty and sandy clay, sandstone, and siltstone interbedded with impure clay and kaolin. Compressional wave velocities are 17,500 feet per second (fps) in the basement rock, 7,200 fps in the Middendorf formation, and 1,500 fps in the top 30 feet of alluvium. Analysis of the site prior to its selection for the ISFSI indicated that subsidence was not a problem (CP&L no date).

Groundwater in the area of the site is derived from the Middendorf (also known as the Tuscaloosa) aquifer. The Middendorf kaolin occurs in lenticular bodies extending laterally for several miles and has a thickness of 30 to 40 feet. The kaolin layer is responsible for a perched water condition where it is present. Both water table and artesian conditions exist at the site. Surficial groundwater at the site discharges to Lake Robinson and to Black Creek. In the artesian aquifers, flow is generally to the southeast. Data indicate that the static head of groundwater underlying the site should be at an elevation of approximately 300 feet mean sea level, or approximately 80 feet above the normal Lake Robinson level, thereby generally precluding leakage from the lake to the aquifer. Domestic wells in the area are within the artesian zone of the Middendorf aquifer.

E2.3 THREATENED OR ENDANGERED SPECIES

No areas designated by the U.S. Fish and Wildlife Service (FWS) as “critical habitat” for endangered species exist at the plant or in the plant vicinity. The primary terrestrial plant community in the vicinity of the site is the pine-turkey oak-wire grass community typical of the Sandhills (Barry 1980). This community is characterized by longleaf pine (*Pinus palustris*) and loblolly pine (*P. taeda*) with a midstory of oaks, chiefly turkey oak (Q*uercus laevis*) along with blackjack oak (*Q. marilandica*), upland willow oak (*Q. incana*), and post oak (*Q. stellata*). Most of the upland CP&L property west of Lake Robinson and south of Secondary State Route (SSR) 346 consists of forest from which timber has been harvested in recent years. After timber is removed, areas are replanted with tree species appropriate to the terrain, soils, and drainage characteristics of the site.

Lake Robinson, an impoundment of Black Creek, provides limited marsh habitat in shallow backwaters at the north (upstream) end of the lake. These marshes and adjacent shallows are used by various waterfowl. Bottomland forest habitat occurs along Black Creek and is characterized by cypress (*Taxodium distichum*), white cedar (*Chamaecyparis thyoids*), red maple (*Acer rubrum*), water oak (*Q. nigra*), red bay (*Persea borbonia*), sweet bay (*Magnolia virginiana*), and black willow (*Salix nigra*) (NRC 1975).

In preparation for the RNP license renewal, CP&L reviewed the South Carolina Department of Natural Resources (SCDNR) Heritage Trust Program database for records of state- or federally-listed species occurring at or near the site. The NRC prepared a biological assessment and consulted with the FWS, the National Marine Fisheries Service, and the SCDNR in conjunction with its review of the CP&L application (NRC 2003a). CP&L also reviewed the Heritage Trust database in preparation for this license renewal application (SCDNR 2003). The following paragraphs describe the results of CP&L's and NRC's reviews.

The bald eagle (*Haliaeetus leucocephalus*) is the only federally-listed terrestrial species known from the Robinson site. Other federally-listed species with potential habitat at the site include the red-cockaded woodpecker (*Picoides borealis*), chaffseed (*Schwalbea americana*), rough-leaved loosestrife (*Lysimachia asperulifolia*), and Canby's dropwort (*Oxypolis canby*).

Bald eagles are occasionally observed at Lake Robinson (CP&L 1998), but there are no known eagle nests in the vicinity of the impoundment (SCDNR 2003). The bald eagle is federally-listed as threatened and state-listed as endangered. NRC concluded that RNP license renewal would have no effect on bald eagles.

Red-cockaded woodpeckers, federally- and state-listed as endangered, are known to occur in Darlington and Chesterfield Counties (SCDNR 2001). An active red-cockaded woodpecker colony is located in Sandhills State Forest, approximately 5.2 miles northwest of RNP (SCDNR 2003). Two abandoned red-cockaded woodpecker cavity trees are located on the RNP site near the Darlington County Plant. Both of these cavity trees have been abandoned for many years. CP&L conducted a field survey for the red-cockaded woodpecker in 1999 throughout the RNP site; the survey identified no active cavity trees and no foraging habitat for this species. In accordance with a Safe Harbor Agreement with the State, CP&L manages the site to maintain and enhance habitat for red-cockaded woodpeckers (CP&L 1999). NRC concluded that RNP may affect, but is not likely to adversely affect, the red-cockaded woodpecker.

Chaffseed is federally- and state-listed as endangered. Habitat for this perennial herb consists of open, moist flatwoods, fire-maintained savannahs, ecotones between peaty wetlands and xeric sandy soils, and other open grass-sedge systems. Factors such as fire, mowing, or fluctuating water tables are necessary to maintain the open to partly open conditions that chaffseed requires. Chaffseed has been recorded in adjacent counties but not at the site. Because habitat for this species may exist on site, and may even be maintained by CP&L activities, NRC determined that RNP license renewal may affect, but is not likely to adversely affect, the chaffseed.

Rough-leaved loosestrife is federally and state-listed as endangered. Habitat for this perennial herb consists of Carolina bays and the ecotones between longleaf pine uplands and pond pine pocosins, an upland swamp community type. The species has been recorded in an adjacent county, but not at the site. Because habitat for this species may exist on site, and may even be maintained by CP&L activities, NRC determined that RNP license renewal may affect, but is not likely to adversely affect, the rough-leaved loosestrife.

Canby's dropwort is federally- and state-listed as endangered. This perennial plant is known to occur in adjacent counties, but not the plant site. This coastal plain species grows in wet meadows, wet pineland savannahs, ditches, sloughs, and along edges of cypress-pine ponds. Because habitats for this species may exist on the site, NRC determined that RNP license renewal may affect, but is not likely to adversely affect, the Canby's dropwort.

Two state-listed species, the endangered Rafinesque's big-eared bat (*Corynorhinus rafinesquii*) and the threatened pine barrens tree frog (*Hyla andersonii*), frequent habitats that may occur on the plant site, but there are no recorded occurrences.

No aquatic species that are federally- or state-listed as endangered or threatened have been recorded in the site vicinity. The federally-listed endangered shortnose sturgeon (*Acipenser brevirostrum*) occurs in the Pee Dee River system, but the presence of two dams on Black Creek is presumed to prevent sturgeon from accessing Lake Robinson. The Carolina heelsplitter (*Lasmigona decorata*), a federally-listed endangered mussel, occurs in the Pee Dee River system, but not in Black Creek. NRC concluded that RNP license renewal would not affect the shortnose sturgeon or the Carolina heelsplitter.

CP&L is unaware of any species proposed for listing by the FWS that occur on the RNP site.

E2.4 REGIONAL DEMOGRAPHY

E2.4.1 GENERAL POPULATION

The generic environmental impact statement for license renewal of nuclear power plants (GEIS) presents a population characterization method that is based on two factors: "sparseness" and "proximity" (NRC 1996, Section C.1.4). "Sparseness" measures population density and city size within 20 miles of a site. "Proximity" measures population density and city size within 50 miles.

CP&L used 2000 census data (U.S. Census Bureau 2000a) and geographic information system software (ArcView®) to determine demographic characteristics in the RNP vicinity. The Census Bureau provides updated annual projections, in addition to decennial data, for selected portions of its demographic information.

As derived from 2000 Census Bureau information, 90,408 people lived within 20 miles of RNP. This is a population density of 71 persons per square mile within 20 miles and, applying the GEIS sparseness measures, RNP falls into a less sparse category, Category 3 (having 60 to 120 persons per square mile or less than 60 persons per square mile with at least one community with 25,000 or more persons within 20 miles) (PE 2002).

As estimated from 2000 Census Bureau information, 809,852 people lived within 50 miles of RNP. This equates to a population density of 103 persons per square mile within 50 miles. Applying the GEIS proximity measures, RNP is classified as Category 3 (having one or more cities with 100,000 or more persons and less than 190 persons per square mile within 50 miles). According to the GEIS sparseness and proximity matrix, the RNP ranks of sparseness Category 3 and proximity Category 3 result in the conclusion that RNP is located in a medium population area (PE 2002).

Table E2-1 depicts historical, current, and projected population figures for Darlington and Florence Counties, the counties in which 80 percent of the RNP workers live.

Table E2-1.
Robinson ISFSI Regional Population and Growth Rates

	Darlington County		Florence County	
Year	**Population**	**Average Annual Growth Rate (Percent)**	**Population**	**Average Annual Growth Rate (Percent)**
1980[a]	62,717	1.7	110,163	2.3
1990[a]	61,851	-0.1	114,344	0.4
2000[b]	67,394	0.9	125,761	1.0
2010[c]	67,800	0.1	134,200	0.7
2020[c]	69,900	0.3	142,800	0.7
2030[d]	72,027	0.3	150,993	0.6
2040[d]	74,058	0.3	159,506	0.6
2050[d]	76,090	0.3	168,019	0.5

a. U.S. Census Bureau 1995.
b. U.S. Census Bureau 2000b.
c. South Carolina Budget and Control Board 2000.
d. Linear extrapolation.

E2.4.2 MINORITY AND LOW-INCOME POPULATIONS

For RNP license renewal, CP&L and NRC used a 50-mile radius as the area that could contain environmental impact sites and the state as the geographic area for comparative analysis (NRC 2001). PEC believes that this analysis conservatively bounds the expected limit of minority and low-income population impacts from Robinson ISFSI license renewal, the 4-mile radius from the center of the ISFSI (NRC 2003b).

CP&L used ArcView® geographic information system software to combine U.S. Census Bureau TIGER line data with Census Bureau 2000 census data to determine the minority and low-income population characteristics on a block-group level. CP&L included a block group if any of its area lay within 50 miles of RNP. The 50-mile radius includes 670 block groups. CP&L defined the geographic area for RNP as the states of North and South Carolina. CP&L analyzed block groups in each state separately against their state's data.

E2.4.2.1 MINORITY POPULATIONS

The NRC Office of Nuclear Material Safety and Safeguards (NMSS) environmental justice procedures defines a "minority" population as: American Indian or Alaskan Native; Asian; Native Hawaiian and other Pacific Islander; Black races; other; multi-racial; the aggregate of all minority races; or, Hispanic ethnicity (NRC 2003b, Appendix C). The guidance indicates that a minority population exists if either of the following conditions exists:

> The minority population of the census block or environmental impact site exceeds 50 percent; or
>
> The minority population percentage of the environmental impact area is significantly greater (typically at least 20 percentage points) than the minority population percentage in the geographic area chosen for comparative analysis.

The NRC guidance calls for use of the most recent U.S. Census Bureau decennial census data. CP&L used 2000 census data from the U.S. Census Bureau website (U.S. Census Bureau 2000c) in determining the percentage of the total population within South Carolina and North Carolina for each minority category, and in identifying minority populations within 50 miles of RNP.

CP&L divided Census Bureau population numbers for each minority population within each block group by the total population for that block group to obtain the percent of the block group's population represented by each minority. For each of the 670 block groups within 50 miles of RNP, CP&L compared the percent of the population in each minority category to the corresponding geographic area's minority category threshold percentage to determine if that block group constituted a minority population. CP&L defined the geographic area for RNP as the State of North Carolina when the block group was in North Carolina, and the State of South Carolina when the block group was in South Carolina. Census Bureau data (U.S. Census Bureau 2000c) for North Carolina characterizes 1.24 percent of the population as American Indian or Alaskan Native, 1.41 percent as Asian, 0.05 percent as Native Hawaiian or other Pacific Islander, 21.59 as percent Black races, 2.32 percent as all other single minorities, 1.28 percent as multi-racial, 27.89 percent as aggregate of minority races, and 4.71 percent as Hispanic ethnicity.

U.S. Census Bureau data (U.S. Census Bureau 2000c) for South Carolina characterizes 0.34 percent of the population as American Indian or Alaskan Native, 0.90 percent as Asian, 0.04 percent as Native Hawaiian or other Pacific Islander, 29.54 as percent Black races, 1.00 percent as All Other Single Minorities, 1.00 percent as multi-racial, 32.81 percent as aggregate of minority races, and 2.37 percent as Hispanic ethnicity.

Based on the “more than 20 percent” or the “exceeds 50 percent” criteria, no Asian, Native Hawaiian or Pacific Islander, and no multi-racial minorities exist in the geographic area. Table E2-2 presents the numbers of block groups within each county in North and South Carolina that exceed the threshold for determining the presence of minority populations.

Based on the “more than 20 percent” criterion, American Indian or Alaskan Native minority populations exist in five block groups (Table E2-2). Three of these block groups are found in Robeson County, North Carolina. The other two block groups are located adjacent to the others in Scotland County, North Carolina. The American Indian or Alaskan Native minority block group locations are displayed in Figure E2-4 and are at the perimeter of the 50-mile geographic area.

Based on the “more than 20 percent” criterion, the Black races minority populations exist in 237 block groups (Table E2-2). Figure E2-5 displays the locations of these minority block groups, while Table E2-2 displays the minority block group distributions among the counties in the geographic area.

Based on the “more than 20 percent” criterion, the All Other Single Minorities populations exist in a single block group (Table E2-2). Figure E2-6 displays the location of this minority block group in Union County, North Carolina.

Based on the “exceeds 50 percent” criterion, the Aggregate of Minority Races populations exist in 254 block groups (Table E2-2). Figure E2-7 displays the locations of these block groups, while Table E2-2 displays the minority block group distributions among the counties in the geographic area.

Based on the "more than 20 percent" criterion, the Hispanic ethnicity minority populations exist in 5 block groups (Table E2-2). Figure E2-8 displays the locations of these minority block groups in Union County, North Carolina.

NMSS guidance is that for a rural location, such as at Robinson, a radius of approximately 4 miles should be used as the area of potential environmental impact. Review of the RNP 50-mile demographic data shows that 10 of the Black races minority block groups are within 10 miles of RNP (Table E2-3). The same 10 block groups also have aggregate minority populations because of the black minority populations.

E2.4.2.2 LOW-INCOME POPULATIONS

NRC guidance defines "low-income" by using U.S. Census Bureau statistical poverty thresholds (NRC 2003b, Appendix C). CP&L divided U.S. Census Bureau low-income household numbers for each block group by the total households for that block group to obtain the percentage of low-income households per block group. U.S. Census Bureau data (U.S. Census Bureau 2000d) characterize 12.3 percent of North Carolina households as low-income, and 14.1 percent of South Carolina households as low-income. A low-income population is considered to be present if:

- The low-income population of the census block or environmental impact site exceeds 50 percent, or
- The percentage of households below the poverty level in an environmental impact area is significantly greater (typically at least 20 percent) than the low income population percentage in the geographic area chosen for comparative analysis.

Based on the "more than 20 percent" criterion, 61 block groups within 50 miles of RNP contain a low-income population. Figure E2-9 displays the locations of low-income household block groups, and Table E2-2 displays the low-income household block group distributions among the counties in the geographic area.

NMSS guidance is that for a rural location, such as at Robinson, a radius of approximately 4 miles should be used as the area of potential environmental impact. Review of the RNP 50-mile demographic data identified three low-income household block groups within 10 miles of the Robinson ISFSI.

Table E2-2.
Minority and Low-Income Population Block Groups

County	State	2000 Block Groups	American Indian or Alaskan Native	Asian	Native Hawaiian or Other Pacific Islander	Black Races	All Other Single Minorities	Multi-Racial Minorities	Aggregate of Minority Races	Hispanic Ethnicity	Low Income
Chester	SC	8	0	0	0	1	0	0	1	0	0
Chesterfield	SC	37	0	0	0	6	0	0	6	0	3
Clarendon	SC	19	0	0	0	12	0	0	12	0	2
Darlington	SC	59	0	0	0	23	0	0	23	0	9
Dillon	SC	27	0	0	0	14	0	0	14	0	5
Fairfield	SC	10	0	0	0	6	0	0	7	0	0
Florence	SC	110	0	0	0	37	0	0	41	0	7
Kershow	SC	41	0	0	0	5	0	0	6	0	2
Lancaster	SC	44	0	0	0	7	0	0	7	0	3
Lee	SC	17	0	0	0	13	0	0	14	0	2
Marion	SC	16	0	0	0	10	0	0	10	0	2
Marlboro	SC	29	0	0	0	14	0	0	16	0	4
Richland	SC	40	0	0	0	12	0	0	16	0	0
Sumter	SC	63	0	0	0	29	0	0	30	0	6
Williamsburg	SC	8	0	0	0	5	0	0	5	0	2
Anson	NC	21	0	0	0	14	0	0	12	0	2
Richmond	NC	42	0	0	0	10	0	0	9	0	6
Robeson	NC	6	3	0	0	4	0	0	6	0	1
Scotland	NC	25	2	0	0	9	0	0	10	0	4
Union	NC	48	0	0	0	6	1	0	9	5	1
TOTAL		670	5	0	0	237	1	0	254	5	61

State			American Indian or Alaskan Native	Asia	Native Hawaiian or Other Pacific Islander	Black Races	All Other Single Minorities	Multi-Racial Minorities	Aggregate of Minority Races	Hispanic Ethnicity	Low Income
South Carolina			0.34%	0.90%	0.04%	29.54%	1.00%	1.00%	32.81%	2.37%	6.93%
North Carolina			1.24%	1.41%	0.05%	21.59%	2.32%	1.28%	27.89%	4.71%	6.20%

Table E2-3.
Minority and Low-Income Block Groups within 10 miles of Robinson ISFSI

Block Group Identity	Low-Income Population[a]	Black Races Population[b]	Aggregate of Minority Races Population[c]
Lee County			
450619801001		√	√
450619801002		√	√
Darlington County			
450310103001		√	√
450310109001		√	√
450310106003		√	√
450310106002	√		
450310107001	√	√	√
450310107002		√	√
450310107003	√	√	√
450310107004		√	√
450310108001		√	√

a. 34.1% or more of the population in the block group.

b. 49.6% or more of the population in the block group.

c. 52.8% or more of the population in the block group. All the aggregate block groups are the result of black populations in the block group.

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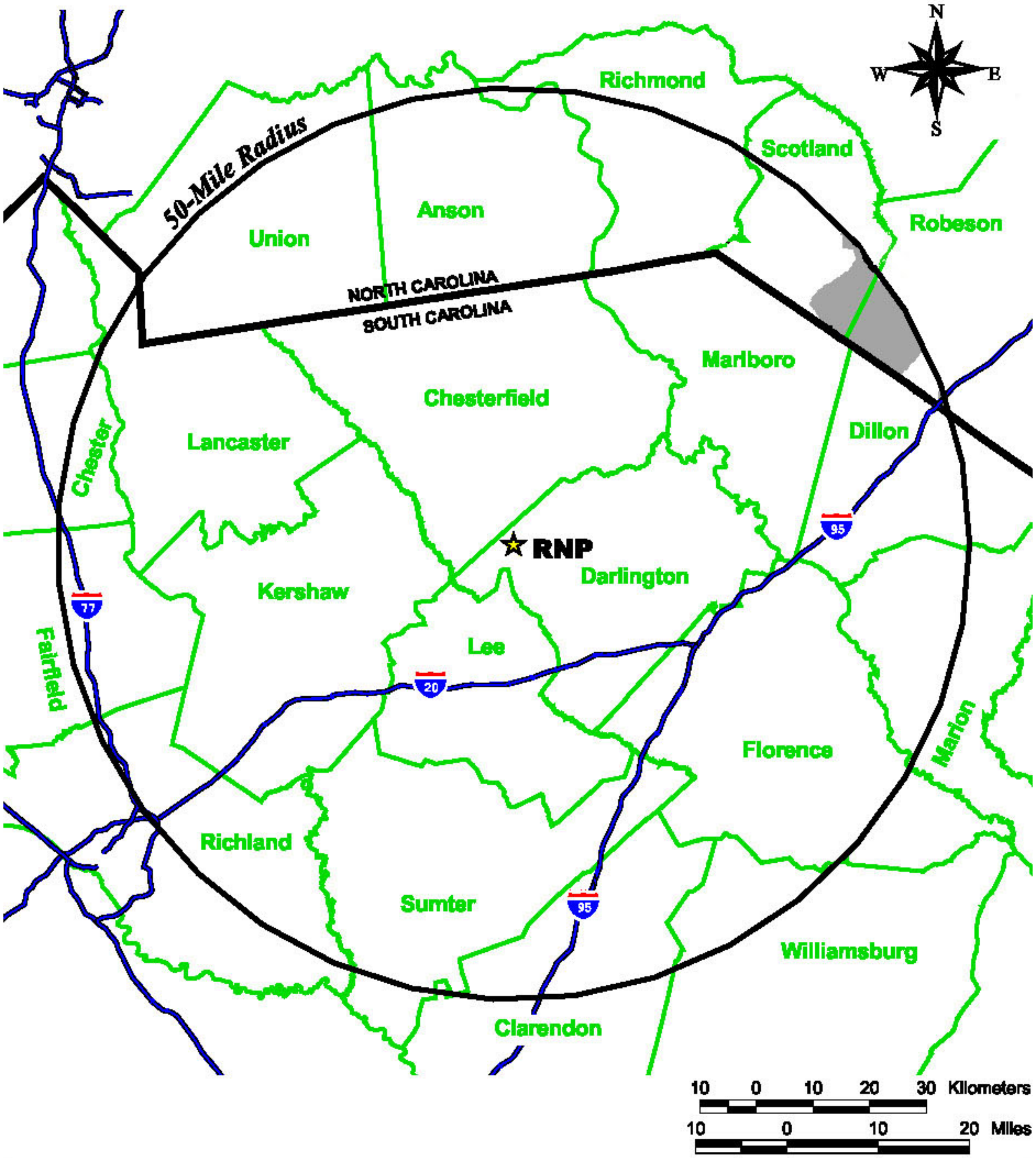
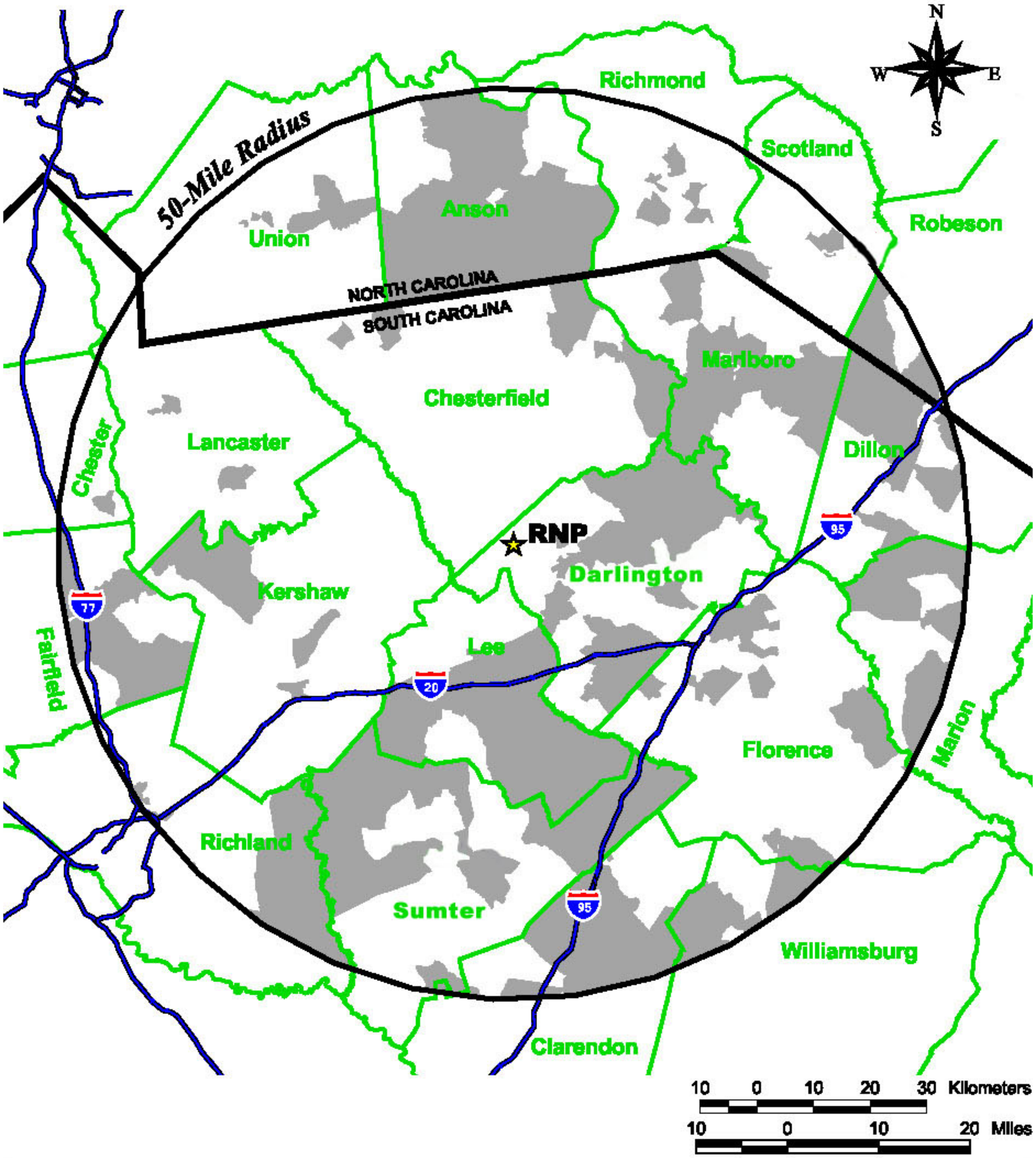


LEGEND

County Boundaries
American Indian or Alaskan Native Minority Populations

**FIGURE E2-4.
American Indian or Alaskan Native Minority Populations**



LEGEND

County Boundaries
Black Races Minority Populations

FIGURE E2-5.
Black Races Minority Populations

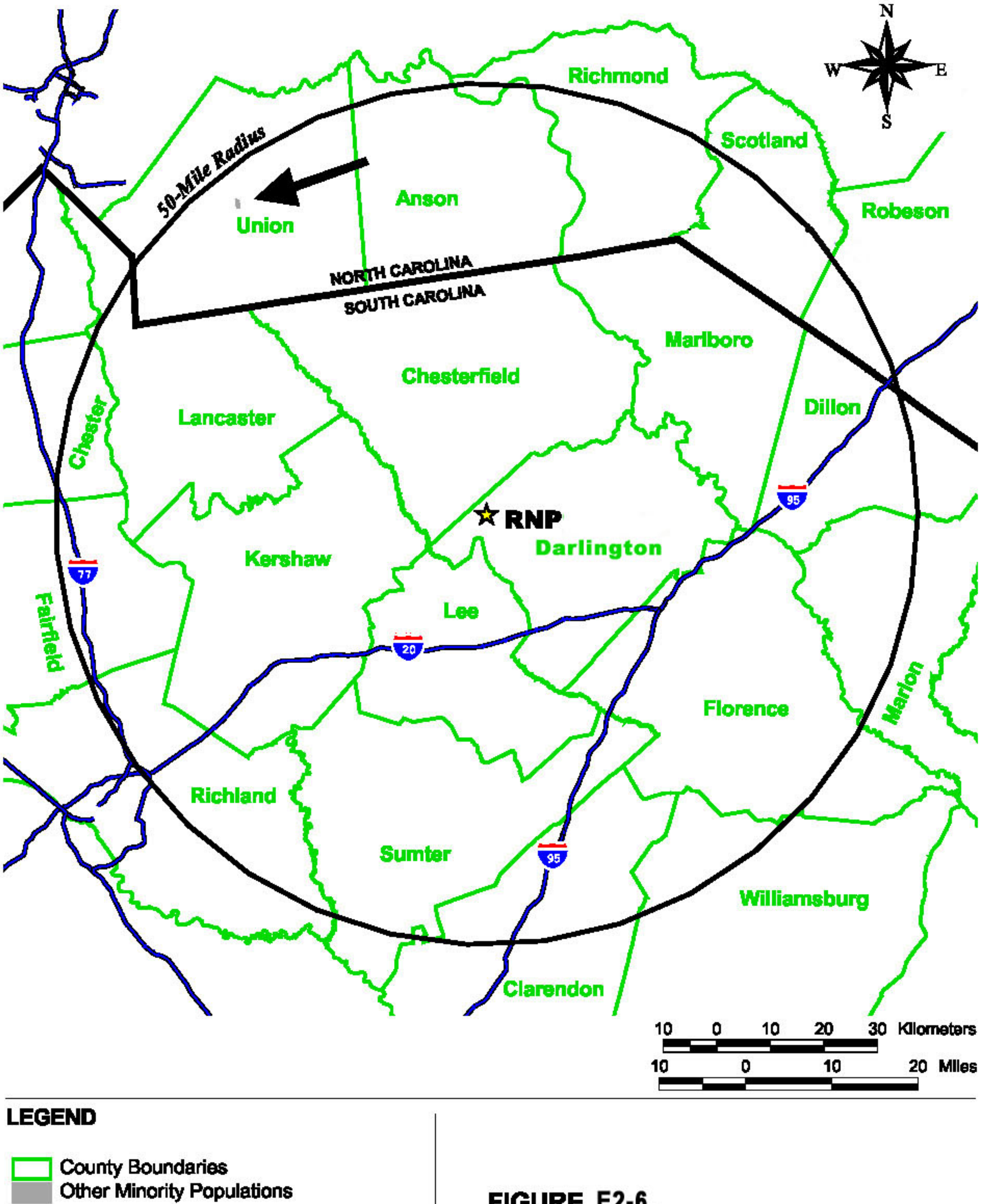
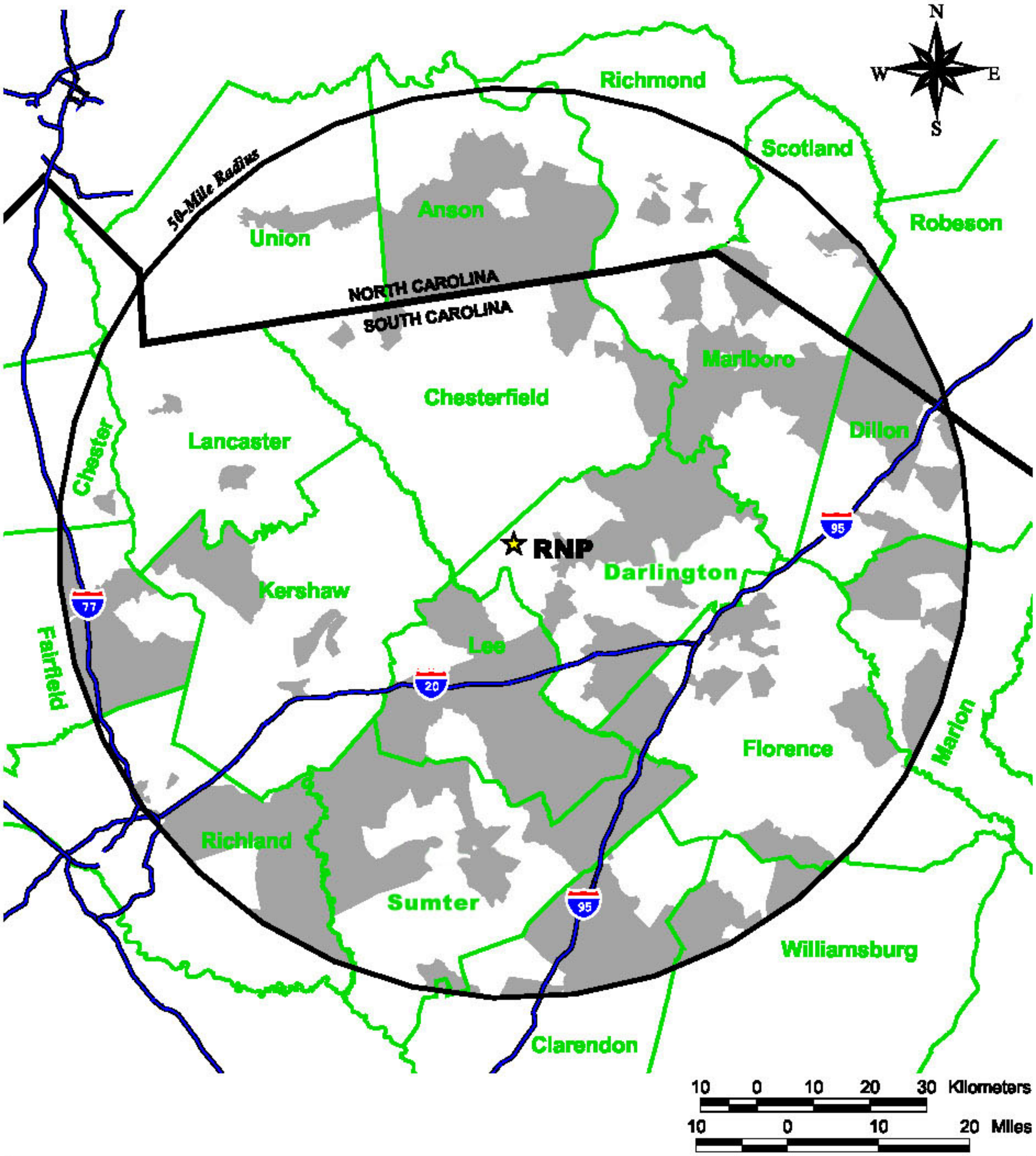


FIGURE E2-6.
Other Minority Populations



LEGEND

County Boundaries
Aggregate Minority Populations

FIGURE E2-7.
Aggregate Minority Populations

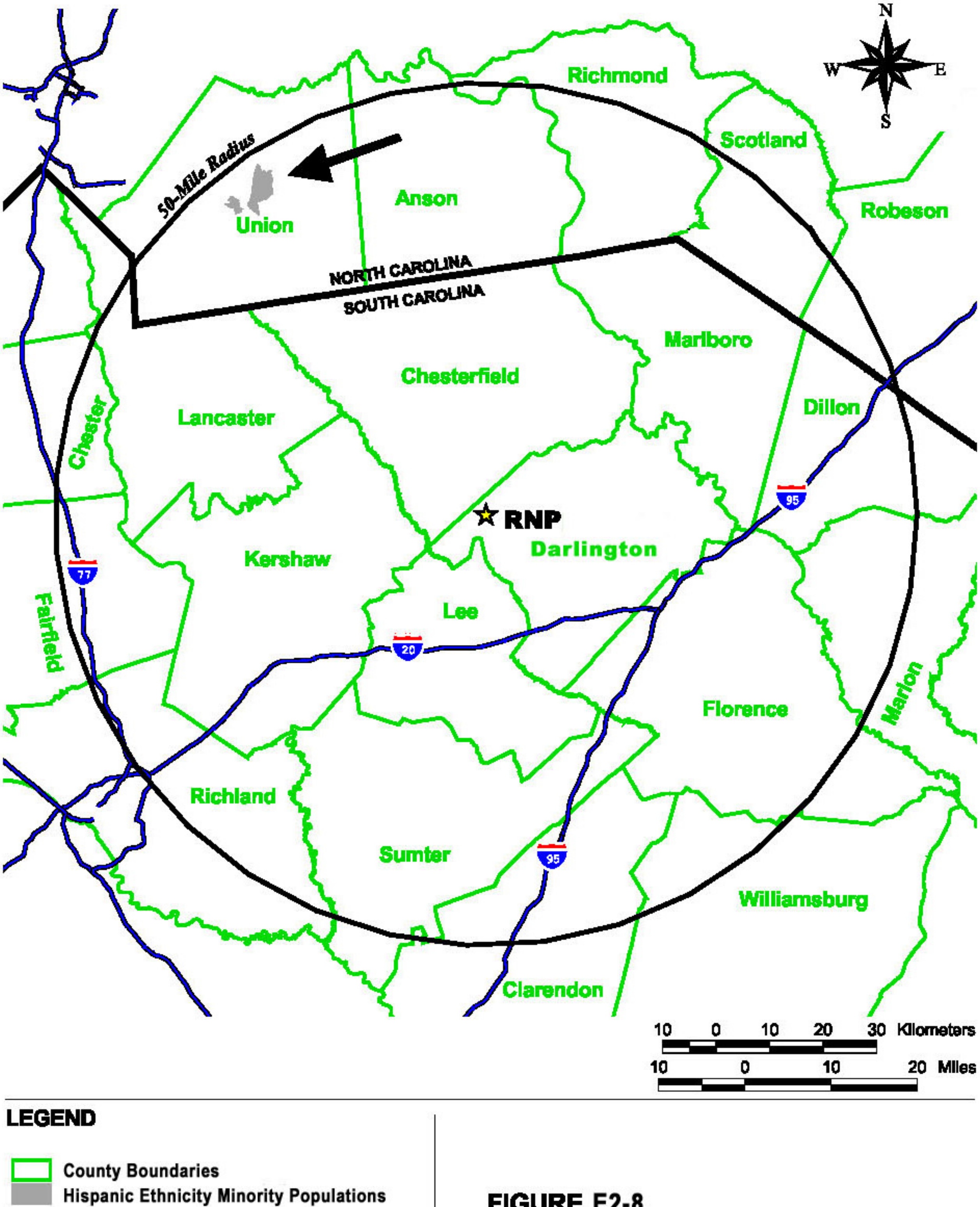


FIGURE E2-8.
Hispanic Ethnicity Minority Populations

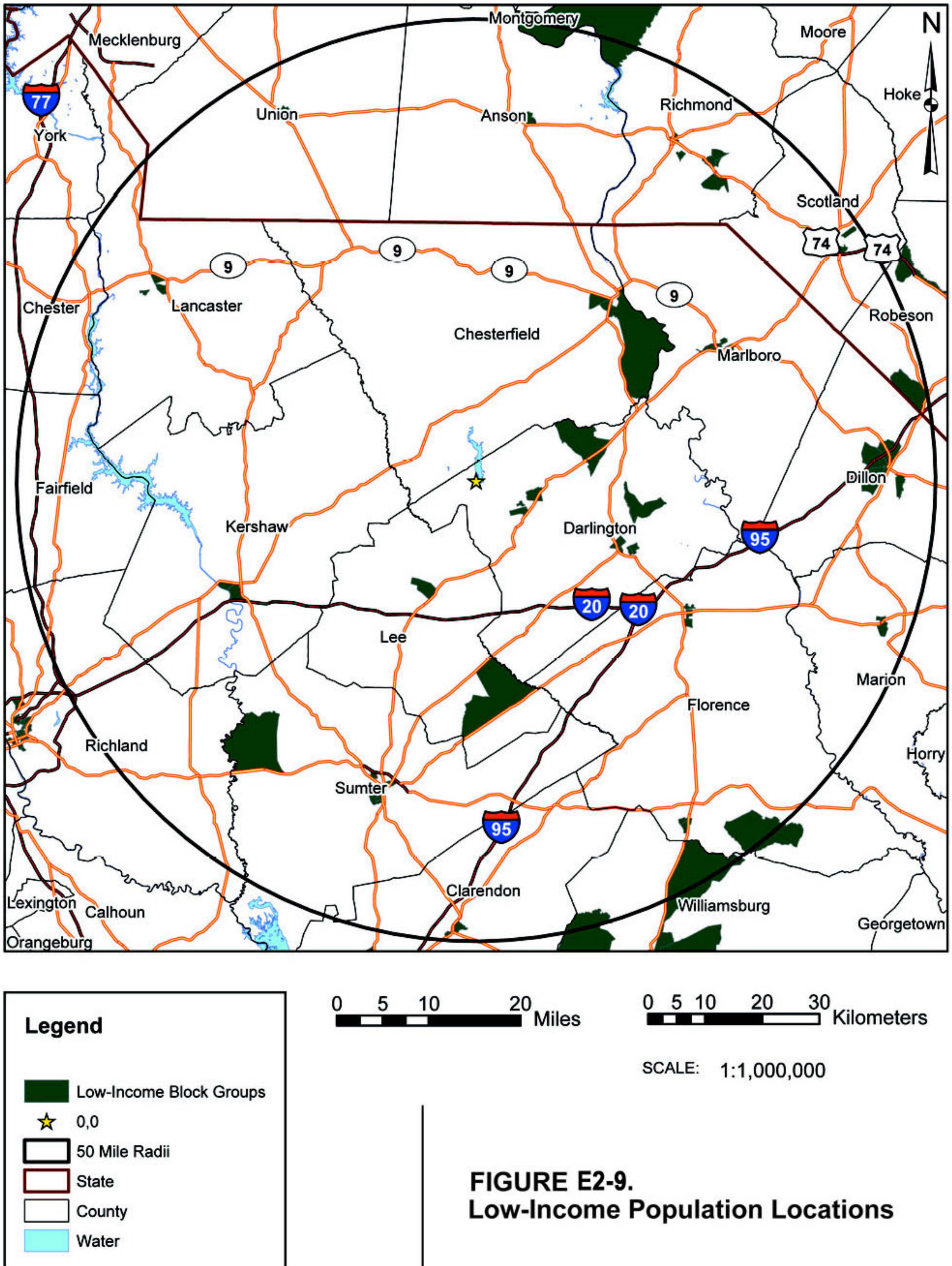


FIGURE E2-9.
Low-Income Population Locations

E2.5 TAXES

CP&L pays annual property taxes for RNP to Darlington and Chesterfield Counties, South Carolina. The payments include, but do not differentiate, the Robinson ISFSI. Property tax revenues fund Darlington County operations, school systems, the County General Fund, fire districts, libraries, the Emergency Management System, and various environmental services (Copeland 2001). Chesterfield County property tax revenues fund the school districts, the County General Fund, local technical colleges, road maintenance, libraries, County office maintenance, hospitals, and prisons (Sowell 2001a and Sowell 2001b). For the years 1995 to 2001, RNP's property taxes provided an approximate average of 19 percent of Darlington County's total property tax revenues. Table E2-4 compares CP&L's RNP tax payments to Darlington County tax revenues. CP&L pays approximately $6,000 of the $2.5-3.5 million in revenues collected annually by Chesterfield County (less than one-half of one percent).

The South Carolina legislature is studying the issue of electric power industry deregulation. The effects of deregulation are not yet fully known, but could affect utilities' tax payments to the counties. Any changes to RNP tax rates due to deregulation would, however, be independent of ISFSI (or RNP) license renewal.

Table E2-4.
Tax Information, 1995 - 2001

Year	Total Darlington County Property Tax Revenues[a]	Property Tax Paid for RNP	Percent of Total Property Taxes
1995	$25,668,652	$6,202,683	24
1996	$26,699,800	$6,486,468	24
1997	$31,538,858	$6,124,758	19
1998	$33,845,257	$6,482,958	19
1999	$33,468,691	$5,323,630	16
2000	$38,077,751	$6,105,886	16
2001	$39,396,122	$5,665,144	14

a. Copeland 2001.

E2.6 LAND USE PLANNING

This discussion summarizes a more lengthy discussion of land use in the RNP license renewal environmental report (PE 2002).

CP&L employs a nuclear-related permanent workforce of approximately 435 employees and an additional 120 contract and matrixed employees at RNP. This section focuses on Darlington and Florence Counties, because the majority (approximately 80 percent) of the permanent RNP workforce lives in these counties and CP&L pays property taxes in Darlington County. Both counties have experienced growth over the last several decades and their respective comprehensive land use plans reflect planning efforts and public involvement in the planning process. Land use planning tools, such as zoning, guide future growth and development. Both plans share the goals of encouraging growth and development in areas where public facilities, such as water and sewer systems, are planned, and discouraging strip development along County roads and highways. The Darlington County plan does not include land-use zoning applicable to unincorporated areas of the county, such the Robinson ISFSI area.

During the last 30 years Darlington and Florence Counties have experienced little growth. Florence County's population increased by 2.3 percent during the 1970s, 0.4 percent during the 1980s, and 1.0 percent during the 1990s. Darlington County's population increased by 1.7 percent during the 1970s, declined by 0.1 percent during the 1980s, and increased by 0.9 percent during the 1990s.

Although development continues to slowly spread throughout the counties, the majority of the land in the counties is rural in nature, either vacant, forested, or in agricultural production.

E2.7 SOCIAL SERVICES AND PUBLIC FACILITIES

As described in the RNP license renewal environmental report (PE 2002, Section 2.9.1) the public water systems in the locations where most RNP employees reside provide sufficient water to the populations in their service areas. Some of the systems are near their capacities, but other systems have excess capacity available for their use, and the utilities are investigating increasing capacity.

As described in Section 2.9.2 of H. B. Robinson Steam Electric Plant, Unit No. 2, Environmental Report (PE 2002), the transportation system in the vicinity of RNP is adequate to support the level of traffic it receives.

E2.8 HISTORIC AND ARCHAEOLOGICAL RESOURCES

The Final Environmental Statement (FES) for operation of RNP reported that there were no known historic or archeological sites on the site or in the transmission line corridors that were cleared for Unit 2 (NRC 1975). This was confirmed in a letter dated June 5, 1973, written by Charles E. Lee, the South Carolina State Historic Preservation Officer (NRC 1975). At that time, it was determined that "the staff is satisfied that no known historical sites or archeological remains were disturbed in the construction of this plant" (NRC 1975). There was no evidence of an interdisciplinary (historic/archeological) investigation of the development area prior to or during construction of the impoundment, Unit 2, or the transmission facilities. Lake Robinson, however, was constructed in 1958, well before the requirements of the National Historic Preservation Act of 1966 were established (NRC 1975).

For RNP license renewal, CP&L and NRC made reasonable efforts to identify historic and archaeological resources in the area of potential effects. These efforts included examination of archaeological and historic site files at the South Carolina Department of Archives and History and the South Carolina Institute of Archaeology and Anthropology, and NRC consulted with eight Native American tribes or groups. The nearest recorded archaeological sites are located along South Carolina Highway 151, running north-south to the west of the plant site (Figure E2-3). Twenty-six above-ground (historic) sites within 6 miles of RNP are on the National Register of Historical Places (Table E2-5).

NRC concluded in its review of the RNP license renewal application that:

> Areas with no potential for historic or archaeological resources include areas where past disturbances related to construction of the power station and appurtenant facilities have taken place to such an extent that any cultural resources that once existed are no longer present. No further archaeological investigations are recommended for these areas. (NRC 2003a, Section 4.4.5)

The ISFSI is located within the protected area of RNP, a location that had been previously disturbed during plant construction. CP&L concludes that there are no historic or archaeological resources at the ISFSI.

**Table E2-5.
Sites on the National Register of Historic Places
Within 6 Miles of RNP**

Site Name	Location	City
	Darlington County	
Arcade Hotel	204 N. Fifth St.	Hartsville
W. E. Cannon House and Store	612 W. Home Ave.	Hartsville
Coker Experimental Farms	West of Hartsville on S.C. 151	Hartsville
J. L. Coker Company Building	5th St. and Carolina Ave.	Hartsville
James L. Coker III House	620 W. Home Ave.	Hartsville
Robert R. Coker House	1318 W. Carolina Ave.	Hartsville
S. Pressly Coker House	402 W. Home Ave.	Hartsville
Davidson Hall, Coker College	College Ave.	Hartsville
C. K. Dunlap House	1346 W. Carolina Ave.	Hartsville
East Home Avenue Historic District	Roughly, E. Home Ave. from N. Fifth St. to just east of First Ave.	Hartsville
J. B. Gilbert House	200 Fairfield Terr.	Hartsville
John L. Hart House	East of C.R. 133	Hartsville
Thomas E. Hart House and Kalmia Gardens	624 W. Carolina Ave.	Hartsville
Hartsville Armory	539 W. Carolina Ave.	Hartsville
Hartsville Community Center--Hartsville Community Market	Fifth St. between College and Home Aves. and 106 W. College Ave.	Hartsville
Hartsville Passenger Station	114 S. 4th St.	Hartsville
Hartsville Post Office	Jct. of Home Ave. and Fifth St.	Hartsville
Wade Hampton Hicks House	313 W. Home Ave.	Hartsville
Jacob Kelley House	West of Hartsville on S.C. S-16-12	Hartsville
Lawton Park and Pavilion	Prestwood Dr. at jct. with Lanier Dr.	Hartsville
Magnolia Cemetery	S. Cedar Ln.	Hartsville
A. M. McNair House	153 W. Home Ave.	Hartsville
Memorial Hall	2nd St. between Home and Carolina Aves.	Hartsville
Paul H. Rogers House	628 W. Home Ave.	Hartsville
West College Avenue Historic District	W. College Ave. from Sixth to west of Eighth Ave.	Hartsville
	Chesterfield County	
Evy Kirkley Site	Address Restricted	McBee

Source: U.S. Department of the Interior, 2003.

E2.9 OTHER RELATED ACTIVITIES

CP&L is planning construction and operation of a new ISFSI under the 10 CFR 72 general license. The facility is planned to be about 100 feet northeast of the existing ISFSI. The new ISFSI is designed to accommodate spent fuel assemblies during the RNP extended operating license term, i.e., until 2030. The new facility is planned to begin operating in 2005.

E2.10 REFERENCES

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E3.0 PROPOSED ACTION

The proposed action is to renew the operating license of the RNP Independent Spent Fuel Storage Installation (ISFSI) for an additional 40 years beyond the current license term. The ISFSI comprises eight HSMs, each containing one DSC with seven IFAs. Three modules are on one foundation, and five are on a second foundation. CP&L has no plans to expand this ISFSI to accommodate more fuel during the license renewal term.

Much of the information presented in this chapter was taken from the RNP ISFSI Safety Analysis Report. The information is consistent with the construction environmental report (CP&L no date b) and the environmental assessment (NRC 1986).

E3.1 GENERAL INSTALLATION INFORMATION

Location
Chapter 4 of the ISFSI SAR provides descriptive information on the RNP ISFSI structures, systems, and components. The ISFSI is approximately 600 feet west of the RNP containment structure and is within the protected area (see Figure E3-1).

Surrounding Areas and Boundaries
The RNP ISFSI is approximately 0.15 acre. The controlled area for the ISFSI is within the exclusion area, and the protected area is within the 1,400-foot exclusion zone. The nearest residence is located approximately 1,350 feet from the ISFSI. The emergency planning zone for the ISFSI is the RNP exclusion zone.

Layout
The facility stores 56 irradiated fuel assemblies (seven assemblies per canister) in eight dry shielded canisters (DSC) inside eight horizontal storage modules (HSM). Each HSM is about 19 feet long by 12 feet high by 6 feet wide. Three interconnected HSMs are constructed on one foundation and five are constructed on a second adjacent foundation. The foundations are approximately 3-foot thick reinforced concrete with a loading pad in front of the modules and a pad for the hydraulic ram at the back of the modules. The outer, exposed walls are 3.5 feet thick to provide the necessary shielding.

Auxiliary Systems
No electric power is supplied to the ISFSI.

FIGURE WITHHELD UNDER 10 CFR 2.390

Utility\Robinson\ISFSI\3-1 Rob w_ISFSI.ai

FIGURE E3-1. Layout of H.B. Robinson Steam Electric Plant Showing Location of ISFSI

Decay heat cooling is by natural circulation convection. Because the irradiated fuel assemblies (IFAs) are confined within the double weld sealed dry shielded canisters (DSCs) and contamination on the external surface of the DSC results in no radioactive contaminant releases, no contamination filtration system is required. Ventilation and offgas systems are not required for the ISFSI, and none are provided.

Because there are no airborne contaminants associated with the ISFSI, neither compressed air nor breathing air supply systems are required or provided. Air sampling systems are not required.

Steam and water are not required, and none are provided.

No communication or alarm system is required, and none is provided.

No flammable or combustible substances are stored at the ISFSI or in the immediate vicinity. The ISFSI is constructed of non-combustible heat-resistant materials. Therefore, no fire extinguishing system is needed or provided. The fire-fighting equipment and personnel at RNP would be available, if needed.

No cold chemical operations are required for the ISFSI.

Neither sanitary nor chemical sewage treatment is required, and none is provided.

The ISFSI requires no maintenance other than periodic inspection of the air inlets and outlets, and possible debris removal.

There have been no operational accidents during the period of ISFSI operation to date.

E3.2 INVENTORY

The characteristics of the irradiated fuel assemblies (IFAs) placed inside the dry shielded canisters (DSCs) and stored inside the horizontal storage modules (HSMs) are described in the RNP ISFSI Safety Analysis Report (SAR) (CP&L no date a). This information is consistent with the construction environmental report (CP&L no date b) and the environmental assessment (NRC 1986).

Physical Characteristics
The mechanical and structural designs of the DSCs are based on the physical characteristics of the pressurized water reactor (PWR) IFAs to be stored within the DSC. The physical characteristics of these IFAs are presented in Table 3.1-1 of the SAR. The ISFSI SAR (CP&L no date a) noted that additional information on the physical characteristics of these fuel assemblies is contained in Section

4.2 of the H.B. Robinson Steam Electric Plant, Unit No. 2, Updated Final Safety Analysis Report (UFSAR).

Thermal Characteristics
The heat generation of 15x15 PWR IFAs was limited to one kilowatt per assembly. This resulted in a maximum of seven kilowatts per DSC immediately following initial loading. All 56 IFAs stored in the ISFSI were irradiated to less than 35,000 megawatt days per metric ton uranium (MWd/MT) and cooled at least five years prior to loading to meet the thermal criteria specified in the SAR. Heat generation is due to radioactive decay; as the fuel ages beyond the initial five-year minimum cooling period, the amount of heat generated is further reduced.

Radiological Characteristics
The principal design criteria for acceptable radiological characteristics are shown in Table 3.1-2 of the ISFSI SAR. The decay heat generation and the radiological characteristics described above bound fuel with original enrichment equal to or less than 3.5 weight percent Uranium-235 that has been irradiated up to 35,000 MWd/MT and cooled for at least five years. This is considered to be the maximum radiation source fuel. The radiation source from the fuel stored at the ISFSI was less than or equal to that described in the NUHOMS Topical Report referenced in the ISFSI SAR.

All eight HSMs have been fully loaded with a total of 56 IFAs since 1989, and no removal or exchange of these IFAs is planned until the time of off-site transportation. Therefore, the only changes in ISFSI inventory characteristics since the HSMs were first loaded would be due to the normal decay process of the radioactive material contained in the IFAs. The SAR mentions, but does not take credit for this process; however, continuous decay results in lower inventories of the radionuclides over time, leading to a gradual reduction in the gamma and neutron radiation energy emitted from the stored spent nuclear fuel.

E3.3 CONSTRUCTION

No construction or refurbishment is planned for the ISFSI during the license renewal term.

E3.4 AGING MANAGEMENT ACTIVITIES

The HSMs stored at the Robinson ISFSI will be subject to aging management activities to ensure their integrity for the duration of the renewal period.

Aging management activities for the HSMs are summarized in the ISFSI license renewal application.

E3.5 EMPLOYMENT

The workers involved in routine ISFSI operations are drawn from the general population of employees at RNP. Procedures require a daily visual inspection of the ISFSI. Table 7.4-2 of the ISFSI SAR estimates the time required for these tasks to be about 60 person-hours per year. These operations would be required for as long as the ISFSI contains IFAs.

ISFSI operations require much less than one full-time-equivalent employee from a nuclear work force of approximately 555 employees. ISFSI operations will continue to be performed by RNP employees who have other responsibilities at the nuclear plant. Employment at RNP will not be affected by continued ISFSI operations.

Decommissioning the ISFSI may require that the concrete storage modules be manually decontaminated, removal of the modules and foundation slabs, and the site restored to pre-ISFSI conditions. Restoration would be limited in large part to the removal and disposal of the concrete slabs in a construction debris landfill, followed by backfilling, grading, and landscaping. Although detailed decommissioning plans have not been developed, it is expected that such activities would not require a workforce greater than that used to construct the ISFSI.

E3.6 DECOMMISSIONING

The following discussion of decommissioning considerations is based on information in Section 3.5 of the RNP ISFSI Safety Analysis Report (SAR) (CP&L no date a).

The DSCs are intended to be transferred to a federal repository, when such a facility is operational. The concrete HSMs are designed so that a DSC can be safely returned to a shipping cask and transported offsite to the federal repository. Shipping cask design and transportation requirements will depend on the regulations in effect at the time that the federal repository begins receiving spent fuel.

Contamination on the DSC exterior surfaces will be very small. Contamination of the module internals and air passages is also expected to be minimal. This level of contamination may be removed by manual methods, so that the reinforced concrete module can be broken-up and removed using conventional methods.

The canister itself may be contaminated internally and may be slightly activated by spontaneous neutron emissions from the irradiated fuel. The canister is designed to be used in the repository for final disposal; however, if the fuel is removed from the canister, the canister could be disposed of as low-level waste.

The exact decommissioning plan to be applied will depend on the status of the U.S. waste repository program at the time of decommissioning.

E3.7 REFERENCES

CP&L (Carolina Power & Light). no date a. H. B. Robinson Independent Spent Fuel Storage Installation Safety Analysis Report. Revision 19

CP&L (Carolina Power & Light). no date b. H.B. Robinson Steam Electric Plant Independent Spent Fuel Storage Installation Environmental Report

NRC (U.S. Nuclear Regulatory Commission). 1986. Environmental Assessment Related to the Construction and Operation of the H. B. Robinson Independent Spent Fuel Storage Installation. Docket No. 72-3 Carolina Power and Light Company. Office of Nuclear Material Safety and Safeguards. March. Washington, D.C.

E4.0 ENVIRONMENTAL CONSEQUENCES AND MITIGATING ACTIONS

E4.1 NRC REVIEWS

The U.S. Nuclear Regulatory Commission (NRC) has reviewed the environmental impacts of dry storage of spent nuclear fuel many times (Section E1.2). As noted in Table E1-1, each analysis concluded that the activity would have no significant impacts on the affected environment.

E4.2 IMPACTS FROM REFURBISHMENT AND CONSTRUCTION

No refurbishment to or additional construction of the ISFSI is planned during the 40-year license renewal period. Therefore, there would be no impacts from refurbishment or construction.

E4.3 IMPACTS FROM OPERATIONS

E4.3.1 OCCUPATIONAL AND PUBLIC HEALTH

Radiological protection and doses from ISFSI operations are discussed in Chapter 7 of the RNP ISFSI Safety Analysis Report (SAR) (CP&L no date). The major aspects of the radiological protection program are summarized in the following sections. There are no other potential health impacts other than the hazards associated with moving heavy objects and equipment during cask transfer operations.

Policy Considerations
CP&L corporate and facility health physics policies are applicable to the Independent Spent Fuel Storage Installation (ISFSI). CP&L is committed to a program of keeping occupational radiation exposure as low as reasonably achievable (ALARA). The company follows Regulatory Guides 1.8, 8.8, 8.10, and publications that deal with ALARA concepts and practices, including 10 CFR 20.

The goals and objectives of the health physics programs are to maintain as low as reasonably achievable (ALARA) both the annual dose to individual facility personnel and the annual integrated dose to facility personnel, i.e., the sum of annual doses (expressed in person-rem) to all facility personnel. The health physics programs identify the organizations participating in the programs, the positions involved, and the responsibilities and functions of the various positions in conducting the programs.

Adequately trained personnel are provided to develop and conduct the health physics programs. Health physics personnel receive INPO (Institute of Nuclear Power Operations)-certified training and obtain process experience to carry out the health physics programs in an efficient manner to assure that company and regulatory requirements are met.

NRC-mandated training programs in the fundamentals of radiation protection and facility exposure control procedures are established to provide instructions to all facility personnel, including contractors, whose duties require working in radiation areas. Training programs for health physics personnel are provided to improve their performance in the health physics programs.

Design Considerations

The designs of the dry shielded canister (DSC) and horizontal storage module (HSM) comply with 10 CFR 72.3 concerning ALARA considerations. Specific considerations that are directed toward ensuring ALARA are:

a) Thick concrete walls on the HSM to reduce the surface dose to an average of less than 20 millirem per hour (mrem/hr).
b) Lead shield plugs on the ends of the canister to reduce the dose to workers performing the drying and sealing operations.
c) Use of a shielded transfer cask for handling and transportation operations involving loaded DSCs.
d) Fuel loading procedures that follow accepted practice and build on existing experience.
e) A recess in the front of the HSM for cask docking to reduce scattered radiation during transfer.
f) Double seal welds on each end of DSC to provide redundant radioactive material containment.
g) Placing demineralized water in the cask annulus and DSC, and sealing the DSC-cask gap to minimize contamination of the DSC exterior during loading.
h) Placing external shielding blocks over the HSM air outlets to reduce direct and streaming radiation.
i) Passive system design that requires minimum maintenance.
j) Insertion of internal shielding blocks around air inlets to reduce direct and streaming radiation.
k) Development of shipping procedures based upon previously used procedures and experience to control contamination during handling and transfer of fuel.
l) Use of additional shielding in front access cover plates.

Items b, d, and g involve cask loading considerations that no longer apply to the emplaced DSCs; they would only apply to the unlikely event of a DSC that requires transfer to the spent fuel pool for opening and removal of the IFAs.

Operational Considerations

Operational considerations at RNP that promote the ALARA philosophy include the determination of the origins of radiation exposures, the proper training of personnel, the preparation of radiation protection procedures, implementing these procedures, and the formation of a review system to assess the effectiveness of the ALARA philosophy.

Operational radiation protection objectives deal with access to radiation areas, exposure to personnel, and decontamination. Working at or near highly radioactive components requires planning, special methods, and criteria directed toward keeping occupational radiation exposure ALARA. Job training and debriefing following selected high exposure jobs contribute toward reduced exposures. Decontamination also helps to reduce exposure. Procedures and techniques are based upon operational criteria and experience that have worked to keep radiation exposure ALARA.

Procedures for the ISFSI are integrated into the current RNP plant operating manual and incorporate the same ALARA philosophy.

The ISFSI is considered a radiation control area. Therefore, a locked fence prevents direct access to the ISFSI. The RNP Radiation Control Unit controls the key to this area.

Sources of Radiation

Neutron and gamma radiation emanating from the spent fuel assemblies stored in the DSC and penetrating the HSM are the primary sources of radiation exposure. The radiation source strength associated with the RNP fuels stored in the ISFSI are bounded by those of the NUTECH Topical Report (NUTECH Engineering, Inc., "Topical Report for the NUTECH Horizontal Storage Module Storage System for Irradiated Nuclear Fuel," NUH-001, Revison 1, November 1985). Due to the lower initial enrichment and the longer burnup, the RNP fuel has slightly different neutron and gamma sources. The neutron source is slightly larger (17.2%) and the gamma source is slightly smaller (11.4%). However, the shielding of the DSC and the HSM are sufficient to ensure that dose limits are not exceeded.

Occupational Dose

Estimated doses for the fuel loading, drying, sealing, and transfer are provided in the NUHOMS Topical Report and in the ISFSI SAR. However, doses from such operations have already been accrued by site personnel, and only the doses from continuing storage and final retrieval of the DSCs for off-site transport will be addressed here. In the ISFSI SAR, the radiation dose rate estimates are based on storage of the fuel in the first three HSMs. Since all eight modules have been built and filled, the calculated dose rates are 8/3 times the reported values.

The resulting dose for a person located at the ISFSI fence for eight hours a day for 250 days per year would be approximately 10 mrem. For a person located at the nearest offices, the yearly dose would be less than 1 mrem (the analysis in the SAR does not take credit for the fact that the actual dose inside the office is less due to shielding from the building).

In estimating occupational exposures during the storage phase, direct neutron and gamma flux, as well as the air-scattered radiation from the module surfaces, were considered. Because of the very rapid decrease of dose rate with distance from the storage facility, a maximum distance of 600 feet was used in the analyses.

Air-scattered dose rates were determined with the computer code SKYSHINE-II. Initial loading of three modules with the Topical Report design basis five-year post-irradiated fuel was assumed. Using conservative assumptions that do not take credit for radioactive decay over time or shielding of personnel by buildings, the collective dose rate was estimated to be 6.0 person-rem per year of storage. Since a total of eight modules were built, the calculated collective dose rate is 8/3 times this value, or 16 person-rem per year. One quarter of this collective dose is incurred by the person(s) performing daily visual inspections. The greatest contribution to collective dose, about one third of the total, is incurred by personnel located in the Operations and Maintenance Building; however, this dose is distributed among an estimated 225 workers. Because of radioactive decay over the 40 additional years of storage beyond the currently licensed period, this annual dose would continue, but at reduced levels.

The collective dose from retrieval and off-site shipment operations was not estimated in the ISFSI SAR. However, this dose can be conservatively estimated assuming that it will be the same as the dose incurred by personnel transferring each DSC from the spent fuel pool to the HSM for emplacement. This operation was estimated to accrue 0.27 person-rem per DSC, or a total of 2.2 person-rem for all eight DSCs. Due to the time that will have passed between emplacement and retrieval operations, a considerable amount of radioactive decay would result in significantly lower dose rates for the retrieval operations.

Dose to the Public

The ISFSI is located within the protected area of RNP. The RNP environmental monitoring program includes environmental thermoluminescent dosimeters (TLDs) maintained at air sampling sites adjacent to the plant boundary. These are located at 170 degrees, 830 feet, and 150 degrees, 1500 feet from the ISFSI. These TLDs are changed quarterly.

Because the ISFSI provides containment yielding essentially no radioactive gaseous or liquid effluents, assessment of offsite collective dose is limited to one of direct and scattered radiation to the nearest residence. The nearest residence is located at 160 degrees, 1350 feet from the facility.

The SAR estimated the ISFSI would contribute approximately 4.3 mrem per year (with all eight modules full) to the dose of this off-site resident. This estimated exposure rate would decrease over time due to radioactive decay of the primary gamma-emitting radionuclides (Cs-137 and Co-60) in the IFAs. The actual exposure from all uranium fuel cycle activities, including the ISFSI contribution, is within the regulatory limits set forth in 40 CFR 190, and would remain so for the duration of the ISFSI storage and retrieval phases.

The cumulative dose to the population (due to the ISFSI) integrated out to 50 miles from the ISFSI, was estimated in the ISFSI SAR to be less than 2.0 person-rem per year. Given that the dose contributions from the ISFSI drop rapidly with increasing distance from the facility, most of the collective dose is incurred by a few people residing within the low population area (up to 4.5 miles from the ISFSI). The past and projected populations in Darlington and Chesterfield Counties, in which most of these affected residents live, are reported in Table E2-1 of this report. While the Darlington County population is estimated to grow slowly through 2050 (from 63,000 in 1980 to 76,000 in 2030), the Chesterfield County population growth rate is estimated to be significantly higher (from 110,000 in 1980 to 170,000 in 2050). Thus, the projected collective dose rate, in the absence of any reduction in the source term due radioactive decay, would be no more than 40% higher at the time of IFA removal. However, given that the half-life of Cs-137 and Co-60 are approximately 30 years and 5 years, respectively, more than one full half-life will have elapsed during this period, reducing the projected dose rates in 2050 to less than 50% of the original estimate. Therefore, the estimated dose rate of less than 2.0 person-rem per year reported in the ISFSI SAR is bounding for the projected duration of the IFA storage and retrieval period.

E4.3.2 OTHER IMPACTS

The continued operation of the RNP ISFSI during the 40-year license renewal term would have no impacts on the following resources:

- Geology or soils
- Hydrology
- Air quality
- Aquatic resources
- Socioeconomics
- Social services or public utilities
- Land use
- Threatened or endangered species
- Cultural or historic resources
- Aesthetics

The RNP ISFSI is a totally passive installation to provide shielding and containment of irradiated fuel. The ISFSI is within the plant protected area. There are no residences or agricultural activities within the 1,400-foot radial exclusion zone.

There are no liquid discharges from the ISFSI, so no geologic, water, or aquatic resources would be affected. There are no air emissions from the ISFSI, so air resources would not be affected.

As described in Section E3.5, Employment, maintenance and surveillance activities at the ISFSI would be performed by RNP employees. No additional employees would be required to maintain or monitor the ISFSI. Therefore, CP&L concludes that the continued operation of the RNP ISFSI would not affect regional socioeconomics, social services, or public utilities.

As described in Section E2.6, Land Use Planning, the presence of RNP has had little impact on land use in Darlington and Florence Counties. The continued operation of the ISFSI would not affect land use patterns in the region.

As described in Section E2.3, Threatened or Endangered Species, bald eagles have been observed at Lake Robinson immediately east of RNP, and an active red-cockaded woodpecker colony is approximately 5 miles northwest of RNP. CP&L is aware of no activities related to continued ISFSI operations during the license renewal term that would adversely affect threatened or endangered species, and NRC concluded that RNP license renewal would be unlikely to adversely affect any such species.

As described in Section E2.8, Historic and Archaeological Resources, the ISFSI is located on the type of disturbed land that NRC has concluded has no potential for unknown cultural resources. Continued operations for the 40-year license extension would have no adverse effect on historic, archaeological, cultural, scenic, or aesthetic resources.

E4.4 IMPACTS FROM POTENTIAL ACCIDENTS

Chapter 8 of the RNP ISFSI SAR (CP&L no date) describes the potential radiological dose impacts from design basis events occurring during normal operations, defined as events that are expected to occur one or more times during the operation of the ISFSI (Category I); off-normal operations, defined as events that can be expected to occur with moderate frequency (Category II); and, accidents, defined as serious occurrences that are expected to happen on an extremely infrequent basis, if ever, during the lifetime of the facility (Category III). The ISFSI SAR also describes maximum hypothetical accidents that are not considered credible (Category IV).

E4.4.1 CATEGORY I ANALYSIS

The ISFSI SAR describes five types of design loads for the ISFSI and compares the system response to these loads with the generic assumptions reported in the NUHOMS Topical Report. These loads include dead weight loads, design basis internal pressure loads, design basis operating temperature loads, operation handling loads, and design basis live loads. The structural components affected by these loads are the DSC, DSC internals, HSM, DSC support assembly, and the foundation.

Based on the analyses in the ISFSI SAR, the Topical Report assumptions are bounding for design load conditions, except for certain assumptions related to the design basis internal pressure loads. An increase in the effective thickness of the cover plates from 1.5 inch to 1.75 inch required a supplemental analysis of bending stress to be performed, with results that were within American Society of Mechanical Engineers Code requirements as reported in the ISFSI SAR. The maximum DSC internal pressure under accident conditions was calculated to be the same as that specified in the Topical Report.

E4.4.2 CATEGORY II EVENT ANALYSIS

The SAR defines two types of off-normal scenarios: transportation/handling incidents and air flow blockage. None of these incidents have on- or off-site radiological consequences beyond those described in Section E4.3 of this report.

Off-normal events associated with the transport operation of the DSC could occur due to malfunctioning of the auxiliary components (i.e., crane, transporter ram, etc.), or by misalignment of the DSC and the HSM. Because the DSCs have already been emplaced, the misalignment incidents are not of concern during removal operations and could occur only if a DSC required interim maintenance and re-emplacement. A postulated malfunction of the crane used to handle the DSCs, or more specifically a yoke failure during this operation, is considered as part of the cask drop accident, which is reported in Section E4.4.3 of this report.

Another off-normal event that could occur is an air inlet blockage. Because the air inlets are close to the ground, they could become blocked with blowing paper, dirt, snow, or other debris. Due to the height of the air outlets and the distance between them, it is less likely that both of the outlets would become blocked. Furthermore, blockage of one outlet alone would not be as severe as blockage of both inlets. Therefore, this off-normal event is defined as complete blockage of the HSM inlets.

An analysis of the blockage of the air inlets indicates that any rise in temperature and pressure in various components of the storage system would be well within the acceptance limits. The blockage of the air inlets would be discovered during the normal surveillance of the modules. As the analysis shows, excessive temperatures could not be reached and, hence, if the blockage were to occur just after an inspection and not be discovered until 24 hours later, no threat to the public health and safety would result. Once detected, the air inlets would be cleared of the blockage.

E4.4.3 CATEGORY III AND IV EVENT ANALYSIS

The SAR defines the following Category III or IV design events:

- Loss of air outlet shielding blocks
- Tornado and tornado generated missiles
- Earthquake
- Eight-foot drop
- Lightning
- Blockage of air inlets and outlets
- Accident pressurization of the DSC
- Fire
- Leakage of the DSC
- Load combination
- Train derailment

Most of these accidents do not have off-site dose consequences, as the integrity of the sealed DSC would not be breached. Only a postulated DSC leakage event would have off-site dose consequences, but these impacts are further mitigated relative to the event postulated in the ISFSI SAR due to ongoing radioactive decay of the major dose-contributing radionuclide (Kr-85) since the DSCs were emplaced in 1989.

Loss of Air Outlet Shielding Blocks
This postulated accident assumes the loss of both air outlet shielding blocks from the top of the horizontal storage module. Other components of the ISFSI are assumed to be in normal condition. The air outlet shielding blocks are designed to remain in place and functional during postulated accidents, except tornado-generated missiles. The ISFSI SAR indicates that there would be no structural or thermal consequences to the ISFSI as a result of the loss of the shielding blocks; the resulting increase in air-scattered (sky shine) doses or direct radiation would result in radiological consequences that are within 10 CFR 100 dose limits.

Doses would also be incurred during recovery from this event. To replace a lost or damaged shielding block, one of the spare blocks would be transferred to the

HSM. After the shield block was transferred to the HSM, a crane would lift the block into position and the block bolted in place. The entire remounting operation should take less than 30 minutes, during which a mechanic would be on the HSM roof for approximately 15 minutes. During this time the mechanic would receive less than 50 mrem. The doses to the crane operator and the mechanic on the ground would be approximately 20 mrem each (assuming an average distance of 15 feet from the center of the module roof).

Tornado and Tornado-Generated Missiles

The ISFSI SAR postulates the most severe tornado wind loadings as specified by the NRC, and the applicable design parameters of the design-basis tornado are the same as those specified in the NUHOMS Topical Report. Given the fact that the HSM method of structural analysis as utilized by the NUHOMS report conservatively envelopes any stacking arrangement of the modules, including the three modular concept at the RNP ISFSI, the maximum moment and shear for the design basis wind pressure and missiles are also bounded by the results in the NUHOMS report. Furthermore, the walls of the HSMs are anchored into the concrete foundation and, as such, there is no possibility of overturning or sliding of the modules due to the impact of a massive high kinetic energy missile.

The result of this accident analysis indicates that components of the ISFSI are capable of withstanding the tornado wind loads and tornado-generated missiles, with the exception of the air outlet shielding blocks. The consequences from a loss of the shielding blocks were addressed earlier in this report.

Earthquake

Section 3.2 of the ISFSI SAR indicates that the maximum ground horizontal acceleration is 0.20g and the maximum ground vertical acceleration is 0.133g. The NUHOMS Topical Report assumes a value of 0.25g for maximum horizontal acceleration and 0.17g for maximum vertical acceleration. In the Topical Report, for the seismic stress analysis of various components, a multiplier of 2 is used to account for multimode excitations. Because the values of the vertical and horizontal accelerations in the referenced report are higher than the maximum RNP site accelerations, the seismic analysis for the HSM and the DSC support assembly are fully applicable, and the results of these analyses envelop the site-specific design.

Based on the results of the seismic analyses described in Section 8.2.3 of the ISFSI SAR, the major components of the RNP ISFSI were designed and analyzed to withstand the forces generated by the safe shutdown earthquake; hence, there are no dose consequences.

Eight-Foot Drop

Section 8.2.4 of the ISFSI SAR postulates two types of drop accidents: a DSC drop during a transfer operation in either the horizontal or vertical orientation.

A drop accident could occur at the one time during the transfer operation that the Transfer Cask is operating without its redundant yoke. This occurs when the cask is lowered into the cradle of the skid assembly. The maximum height that the cask is raised during this operation is 8 feet. Hence, the maximum height of a postulated drop accident is limited to 8 feet. Furthermore, because the cask is always lifted from the trunnions located at the upper regions of the cask, the postulated failure of the single yoke could only cause a cask bottom end or a corner drop. Consequently, if the yoke failed during the tilting operation, the cask would either land on the bottom end fins or on the side steel rings located near the upper and lower regions of the cask outer shell.

Therefore, an 8-foot drop criterion in either the horizontal or vertical orientation bounds any possible drop orientation during a transfer operation, including a corner drop orientation. The skid assembly and the cask/skid/trailer tie-down systems are designed to withstand the inertia forces associated with the transportation shock loads, and as such, there is no possibility of a cask drop during a transport. Even if such an unlikely event occurred or the cask/skid/trailer tipped over as a unit, the height of this drop condition is enveloped by the 8-foot drop height criterion.

The results of the horizontal and vertical drop analysis for the IF-300 Transfer Cask indicate that the stresses in all components of the DSC and its internals are within the ASME acceptance limits and are capable of withstanding inertia forces associated with the 8-foot drop accident conditions. A corner drop accident was also considered. However, the deceleration values as established by the IF-300 SAR are significantly lower than the values of either the horizontal or the vertical deceleration components. Therefore, the stresses for corner drop analysis are bounded by the analyses for the horizontal and vertical drops.

Lightning

Because the ISFSI is outdoors, there is a likelihood of a lightning strike. In order to protect the ISFSI from any damage that could be caused by a lightning discharge, a lightning protection system is installed on the ISFSI. The lightning protection system is designed to prevent any damage to the HSMs and their internals.

Lightning protection systems have proven to be an effective means of protecting a structure and its contents from the effects of a lightning discharge. The lightning protection system does not prevent the occurrence of a lightning discharge; however, the system does intercept the lightning discharge before it can strike the HSM and provides a continuous path for the discharge to the earth. In the event of lightning striking the HSM, the air terminal located on the HSM roof slab would intercept the lightning discharge. The current would follow the low impedance path of the air terminal, conductors, and ground terminals to the earth. Since the system diverts the current, the HSM and its contents would not be damaged by the heat or mechanical forces generated by the current passing through the HSM. In addition, since the ISFSI requires no electrical system for its continuous operation, the resulting current discharge would have no effect on the operation of the ISFSI.

Therefore, lightning striking the HSM and causing an off-normal condition is not a credible accident.

Blockage of Air Inlets and Outlets
This accident assumes the complete and total blockage of the air inlets and outlets of the horizontal storage module. Since the ISFSI is located outdoors, it can be postulated that the module is totally covered by debris from such an unlikely event as a tornado. The ISFSI's design features, such as a perimeter fence and separation of air inlets and outlets, minimize the probability of such an accident occurring under normal conditions. Nevertheless, such an accident was postulated and analyzed.

There are no structural consequences for this event. The thermal consequences of this accident would result from heating of the DSC and HSM due to the blockage of air flow. The Topical Report addresses this accident condition, and the results of the analysis indicate that there would be no structural or dose consequence if the air inlets and outlets are cleared within 48 hours. This 48 hour time limit for clearing the air inlets and outlets is specified in the ISFSI operation and limits criteria.

Accidental Pressurization of the DSC
Section 8.2.7 of the ISFSI SAR describes how internal pressurization of the DSC would result from fuel cladding failure and the subsequent release of fuel rod fill gas and free fission gas. To establish the maximum accident pressurization, all fuel rods in the DSC were assumed to rupture, 25% of the fission gases were released, and the original fuel rod fill pressure was 500 pounds per square inch, gauge (psig), exceeding the actual fill pressure of 300 psig. The resulting internal pressures at the maximum ambient temperature of 105^{o}F and at the minimum ambient temperature of -5^{o}F are below the accident pressures reported in the NUHOMS Topical Report (for temperature extremes of 125^{o}F and -40^{o}F). The limiting accident for DSC pressurization is the blockage of air flow to the DSC. Under these conditions, the gas temperatures in the DSC will rise to 413^{o}C (775^{o}F) producing a DSC internal gauge pressure of 39.7 psig. The DSC shell stresses due to accident pressurization are enveloped by those reported in the Topical Report.

The DSC has a safety margin of greater than three (3) under this accident condition, and as such, there are no dose consequences.

Fire

No flammable or combustible substances are stored within the ISFSI or within the ISFSI's radiation control area. Furthermore, the ISFSI is constructed of non-flammable heat-resistant materials (concrete and steel). The only credible accident that could expose the ISFSI to a flammable substance would be the accidental spillage of a flammable liquid, either through human error or equipment malfunction, at the perimeter of the ISFSI. However, the sandy soil between the ISFSI's perimeter fence and the HSMs is highly porous. Most of the flammable liquid would be absorbed by the soil, greatly reducing the probability of ignition or the intensity and duration of the fire.

The only other time in which a component of the ISFSI would be exposed to a potential fire hazard would be during the DSC transport operation. Throughout this operation, the DSC is located within the cavity of the IF-300 Transfer Cask.

Based on the above discussion, exposure of the ISFSI to a long or intense fire is not considered a credible accident.

Leakage of the DSC

The DSC is designed for no leakage under any normal or credible accident conditions. The accident analyses in previous sections show that none of the events could breach the DSC body. However, to show the ultimate safety of the ISFSI, a total and instantaneous leak was postulated. The postulated accident assumes that one DSC ruptured and all fuel rod claddings failed simultaneously, such that 25% of all fission gases in the irradiated fuel assemblies (mainly Kr-85) are instantaneously released to the atmosphere. The dose consequences from the leaking DSC were evaluated in the NUHOMS Topical Report, and the resulting accident dose was found to be well below the 10 CFR 72.68 acceptable limit of 5 rem. The dose consequences would be further reduced due to the elapsed time since DSC emplacement, which is approaching twice the half-life of Kr-85 (10.76 years).

Load Combination

Normal operating and postulated accident loads associated with various components of the ISFSI are either the same as or are enveloped by those reported in the NUHOMS Topical Report, except for the DSC and the foundation. Hence, the combined effect of various accident and normal operating loads for the DSC support assembly and the HSM are enveloped by the load combination results presented in the Topical Report. The methodology used in combining normal operating and accident loads and their associated overload factors for various components of the ISFSI, with the exception of the foundation, is presented in the Topical Report. The DSC analysis load combination in the ISFSI SAR utilized the same methodology as the Topical Report. The generic DSC fatigue analyses due to normal operating pressure loads, accident pressure loads, seismic loads, seasonal temperature loads, and daily temperature cycling envelop the RNP ISFSI site-specific analysis. This is because the extreme ambient temperatures selected for generic design of the DSC (-40^{o}F to 125^{o}F) envelop the ambient temperature range (-5^{o}F to 105^{o}F), and the RNP site has a lower seismic acceleration.

Train Derailment

This postulated accident is described in Section 8.5 of the ISFSI SAR. The closest rail line is approximately 33 feet north of the ISFSI foundation. This line is used for temporary holding of empty coal cars and as a run-through track. The maximum speed limit for trains on this track is 5 mph. The soil between the rail line and the ISFSI is very porous, which would tend to impede the motion of a derailed car, even though the ISFSI site is at a somewhat lower elevation than the track. In view of the foregoing and the fact that there are no switches within 500 feet of the ISFSI location, damage to the ISFSI from a postulated train derailment is not considered credible.

E4.5 REFERENCES

CP&L (Carolina Power & Light). no date. H. B. Robinson Independent Spent Fuel Storage Installation Safety Analysis Report. Revision 19

CP&L (Carolina Power & Light). 2000. Radiological Environmental Monitoring Report. 1999. RNP-RA/00-0087. Robinson Nuclear Plant. Hartsville, S.C.

E5.0 ASSESSMENT OF NEW AND SIGNIFICANT INFORMATION

E5.1 DISCUSSION

In assessing whether there has been any significant environmental change since the NRC evaluated RNP ISFSI construction and operation, CP&L reviewed the following documents:

- The generic environmental impact statement (GEIS) that the U.S. Nuclear Regulatory Commission (NRC) prepared for station license renewal (NRC 1996)
- The description of the new and significant information identification process that CP&L undertook for RNP license renewal (CP&L 2002)

Each of the above-referenced documents represents a structured approach to identifying and evaluating the significance of environmental impacts. The scope of each is station license renewal, and also addresses spent fuel storage, including the RNP Independent Spent Fuel Storage Installation (ISFSI). The GEIS covers the time from construction and start of operation to publication in 1996. The RNP license renewal application evaluates whether there have been any significant environmental issues not covered in the GEIS, and includes the time from GEIS publication to the present. In combination, therefore, the documents span the time from the RNP ISFSI final environmental statement (NRC 1986) to the present, and serve as a mechanism for identifying any significant environmental changes since the NRC evaluated RNP ISFSI construction and operation.

The following is a description of the process used during the RNP license renewal investigations. CP&L used a qualified investigative team comprised of corporate and plant personnel. These individuals form a group knowledgeable about plant systems, the site environment, and plant environmental issues. In addition, CP&L contracted with an environmental consulting firm with expertise in the NRC license renewal environmental review process, the National Environmental Policy Act (NEPA), and the scientific disciplines involved in preparing a license renewal environmental report. The team and consultants (1) interviewed CP&L subject matter experts regarding specifics of plant operations, including management of discharges and emissions, (2) reviewed environmental documentation, (3) consulted with state and federal agencies to determine if the agencies were concerned about plant operations, and (4) reviewed internal procedures for reporting to the NRC events that could have environmental impacts. The subject matter experts interviewed during the plant license renewal process included persons responsible for RNP spent fuel management and ISFSI operations. The results of those interviews identified no new significant information.

Documentation and procedures reviewed for this application covered ISFSI impacts, and regulatory agencies were afforded the opportunity to address operational issues. While preparing this environmental report, CP&L reviewed the original environmental report for the ISFSI, the environmental assessment for the construction and operation of the ISFSI, the ISFSI Safety Analysis Report, and applicable station monitoring reports. RNP and corporate personnel familiar with the operation of the ISFSI provided input to, and commented on, the information provided in this environmental report.

The assessments performed for the RNP license renewal and for the ISFSI were thorough and comprehensive, and would have identified any new and significant issues related to the ISFSI. CP&L is aware of no new and significant information regarding the environmental impacts of the RNP ISFSI license renewal.

E5.2 REFERENCES

CP&L (Carolina Power and Light Company). 2002. Applicant's Environmental Report – Operating License Renewal Stage, H. B. Robinson Steam Electric Plant, Unit No. 2. Docket No. 50-261, License No. DPR-23. January.

NRC (U.S. Nuclear Regulatory Commission). 1986. Environmental Assessment Related to the Construction and Operation of the H. B. Robinson Independent Spent Fuel Storage Installation. March. Washington, D.C.

NRC (U.S. Nuclear Regulatory Commission). 1996. Generic Environmental Impact Statement for License Renewal of Nuclear Plants. NUREG-1437. May. Washington, D.C.

E6.0 SUMMARY OF LICENSE RENEWAL IMPACTS AND MITIGATING ACTIONS

E6.1 LICENSE RENEWAL IMPACTS

This environmental report describes the proposed action, renewing the license of the RNP Independent Spent Fuel Storage Installation (ISFSI), and any associated impacts. All impacts would be small and not significant. Table E6-1 identifies the impacts license renewal would have on the environmental resources. Based on this evaluation, renewal of the license for the RNP ISFSI would involve no significant environmental impact.

Table E6-1.
Environmental Impacts Related to the License Renewal of the Robinson ISFSI

Issue	Environmental Impact
Geology or Soils	None
Hydrology	None
Air Quality	None
Aquatic Resources	None
Socioeconomics	None
Social Services or Public Utilities	None
Land Use	None
Threatened or Endangered Species	None
Historic or Cultural Resources	None
Occupational Doses from Normal Operations	Small. ISFSI workers would come from the RNP workforce. The conservative occupational collective dose rate is estimated to be no more than 16 person-rem per year. The nearest office workers would receive less than 1 mrem annually.
Other Occupational Health Effects	Small. Any other health effects would be the result of hazards associated with moving heavy objects and equipment during cask emplacement, which is not expected to occur.
Doses to the Public from Normal Operations	Small. The annual maximum dose to the nearest offsite resident is estimated to be 4.3 mrem.
Occupational Doses from Accidents	Small. The largest dose from a credible accident would be 50 mrem incurred during air outlet repairs.
Doses to the Public from Accidents	None. No credible accident would cause a dose to the public.

E6.2 MITIGATION

The impacts of license renewal are small and would not require mitigation. Current operations include mitigation activities that would continue during the term of the renewed license. CP&L performs routine monitoring activities and any associated mitigation to ensure the safety of workers, the public, and the environment. These monitoring activities include the radiological environmental monitoring program conducted for RNP and the RNP ISFSI, periodic monitoring of the HSMs and preventative maintenance as necessary, and monitoring and maintenance of the perimeter and security fences.

E6.3 UNAVOIDABLE ADVERSE IMPACTS

Renewing the RNP ISFSI license would incur no unavoidable adverse impacts as a result of normal operations. No credible accidents would increase the dose to the public, and occupational doses from a credible accident would be 50 mrem or less.

E6.4 IRREVERSIBLE AND IRRETRIEVABLE RESOURCE COMMITMENTS

The continued operation of the RNP ISFSI for the license renewal term will result in no additional irreversible and irretrievable resource commitments beyond those materials committed during the initial licensing of the ISFSI that cannot be recovered or recycled, or that are consumed or reduced to unrecoverable forms.

E6.5 SHORT-TERM USE VERSUS LONG-TERM PRODUCTIVITY OF THE ENVIRONMENT

The RNP ISFSI occupies approximately 0.15 acres of land. However, the ISFSI is a temporary storage facility. Once the IFAs are moved to a permanent repository, the HSMs, concrete pads, and fencing, could be removed and the land used for another purpose.

E7.0 ALTERNATIVES

Preparatory to construction, Carolina Power and Light (CP&L) Company and NRC evaluated the following alternatives to the RNP ISFSI (CP&L no date, NRC 1986):

- Ship spent fuel to a permanent federal repository
- Ship spent fuel to CP&L's Shearon Harris plant
- Construct a new independent spent fuel storage pool at the RNP site
- Provide dry cask storage of spent fuel at the RNP site
- No action

As Table E7-1 illustrates, these alternatives are a subset of alternatives that NRC and license applicants have evaluated for all ISFSIs. Recent evaluations have addressed technologies that were not available at the time of the RNP ISFSI original licensing. The following paragraphs supplement the original RNP ISFSI alternatives evaluation, and address other technologies.

Table E7-1.
Alternatives Analyzed in Independent
Spent Fuel Storage Installation License Applications

	Alternatives											
	Ship To											
ISFSI	Permanent federal repository	Interim federal repository	Same utility, different reactor site	Other utility's reactor site	Reprocessing center	Increase existing spent fuel pool storage capacity	Construct new spent fuel storage pool at the site	Improve fuel storage	Operate reactors at reduced power	Construct an ISFSI at a remote location	Store fuel in dry casks	No action
Surry	✔	✔	✔	✔	✔	✔	✔	✔	✔	✔		✔
Robinson[2]	✔		✔				✔				✔	✔
Oconee[3]	✔		✔			✔						
Fort. St. Vrain	✔	✔		✔ [4]	✔		✔ [5]			✔		
Calvert Cliffs[6]	✔	✔		✔	✔	✔	✔					✔
Prairie Island[7]	✔	✔	✔	✔	✔	✔	✔	✔				✔
Rancho Seco[8]	✔			✔	✔							
Trojan	✔			✔	✔							✔
North Anna[9]			✔			✔	✔	✔				✔
TMI						✔	✔			✔		✔
Skull Valley[10]			✔	✔						✔		✔

ISFSI - Independent Spent Fuel Storage Installation
TMI - Three Mile Island

1. Deleted.
2. Robinson also considered other dry storage technologies.
3. Oconee also considered full-scale rod consolidation.
4. Fort St. Vrain evaluated storing fuel at Idaho National Engineering and Environmental Laboratory, as well as commercial reactor sites.
5. Fort St. Vrain evaluated storing fuel in fuel storage wells, not spent fuel pools.
6. Calvert Cliffs also considered full-scale rod consolidation and other dry storage technologies.
7. Prairie Island also considered other dry storage technologies.
8. Rancho Seco also evaluated other dry storage technologies and maintaining the fuel in the existing fuel pool.
9. North Anna also considered other dry storage technologies.
10. Skull Valley also evaluated dry and wet storage technologies, and alternatives that would eliminate the need for the proposed storage facility.

E7.1 ALTERNATIVES CONSIDERED IN THE ORIGINAL ISFSI LICENSING ANALYSIS

E7.1.1 SHIP SPENT FUEL TO A PERMANENT FEDERAL REPOSITORY

NRC noted that shipping spent fuel to a permanent federal repository would be CP&L's preferred alternative, but that the repository was not likely to be ready to receive spent fuel in time to meet the RNP spent fuel storage needs. This remains true today. The U.S. Department of Energy (DOE) currently expects the Yucca Mountain repository to begin receiving spent fuel no sooner than 2010 (DOE 2003), four years after the expiration date for the current RNP ISFSI license. In addition, DOE has imposed on commercial nuclear reactor licensees, such as CP&L, limits on the quantity of spent nuclear fuel that licensees can ship to Yucca Mountain annually. CP&L took this shipping schedule into account in proposing the expiration date for the RNP ISFSI license renewal term. Because a federal repository will not be available in time, shipping spent fuel to a permanent federal repository is not a reasonable alternative to the proposed action, i.e., renewing the RNP ISFSI license.

E7.1.2 SHIP SPENT FUEL TO SHEARON HARRIS

NRC reviewed shipping RNP spent fuel to the Shearon Harris plant spent fuel pool and concluded that the action would not be viable until Shearon Harris received its operating license, which it did not have at the time the EA for the original ISFSI was prepared. Additionally, transshipment would not further the demonstration of dry storage modules, as contemplated by the National Waste Policy Act. After Shearon Harris was licensed, CP&L transshipped spent fuel from RNP and Brunswick Steam Electric Plant to the Shearon Harris fuel pool, which had excess capacity. However, the Shearon Harris fuel pool cannot support storing additional spent fuel from offsite reactors during the license renewal terms of RNP and Brunswick Steam Electric Plant.

E7.1.3 CONSTRUCT A NEW INDEPENDENT SPENT FUEL STORAGE POOL AT THE RNP SITE

Expansion of the RNP spent fuel storage pool would necessitate construction of an independent pool facility (CP&L no date, Section 9.0) and transferring assemblies from existing RNP ISFSI storage modules into the pool. CP&L would later have to remove these assemblies from the pool for packaging into casks for shipment to Yucca Mountain, and would have to decontaminate the expanded pool space. Occupational doses from these activities would be in addition to those from the proposed action. Constructing new fuel pool storage capacity would be more costly than continued dry storage. It would also require an Operating License amendment. Given the increased environmental impacts from construction, increased worker dose, necessary modifications to plant systems, and increased facility maintenance requirements with no identifiable reduction in significant environmental impact, CP&L concludes that constructing a new spent fuel storage pool at RNP offers no environmental advantages over ISFSI license renewal.

E7.1.4 Dry Cask Storage

Dry cask storage of spent fuel at the RNP site was considered by NRC in the EA for the original constuction and operation. However, DOE already had an agreement with Virginia Power to demonstrate dry cask spent fuel storage technology, so a second demonstration project was not needed.

CP&L is planning to construct an additional ISFSI that will store spent fuel in dry casks. The environmental impacts of dry cask storage are discussed in Section E7.2.5

E7.2 ALTERNATIVES NOT CONSIDERED IN THE ORIGINAL ISFSI LICENSING ANALYSIS

E7.2.1 Increase the Storage Capacity of the Existing Spent Fuel Pool

As CP&L indicated in the construction environmental report, the RNP spent fuel pool was at capacity and CP&L had determined that it could not store more spent fuel in the pool. This determination remains valid; increasing pool capacity is not a reasonable alternative to license renewal.

E7.2.2 Ship Spent Fuel to a Reprocessing Facility

No commercial reprocessing facility exists in the United States and there is no prospect for one in the foreseeable future. Therefore, reprocessing is not a reasonable alternative.

E7.2.3 Ship Spent Fuel to a Federal Interim Storage Facility

NRC has evaluated impacts of storing spent fuel at a federal interim storage facility, and some planning has been done for constructing such a facility, most recently at the Yucca Mountain repository site. However, no federal interim storage facility has been built and there appears to be no prospect for one in time to eliminate the need for RNP ISFSI license renewal. Therefore, shipping to a federal interim storage facility is not a reasonable alternative.

E7.2.4 SHIP SPENT FUEL TO OTHER UTILITY COMPANIES' REACTORS FOR STORAGE

CP&L uses inter-plant transfers of spent fuel as a means to manage its spent fuel inventory, and those transfers were feasible only because of a unique circumstance; the receiving plant, Shearon Harris, constructed spent fuel pool storage capacity for more units than were constructed at the site. However, for reasons described in Section E7.1.2, CP&L is evaluating alternatives to continuing to transship fuel. An alternative would be to ship the RNP fuel to another utility's fuel pool or ISFSI. No other utilities ship fuel among plants, or to other utility storage facilities. CP&L concludes that shipping RNP spent fuel to any other commercial nuclear plant for storage is not a reasonable alternative to RNP ISFSI license renewal.

E7.2.5 CONSTRUCT A NEW ISFSI AT THE RNP SITE

CP&L will construct a new ISFSI at RNP that will provide dry cask storage for additional spent fuel. As Section E1.2 discusses, NRC has evaluated numerous dry storage technologies, including different cask designs and storage concepts, and found all to have small environmental impacts. NRC has certified 14 cask designs (10 CFR 72.214) for use under a general license (10 CFR 72, Subpart K). CP&L could repackage RNP ISFSI spent fuel into certified-design casks and store the spent fuel in the new ISFSI under its license to operate RNP. This alternative would obviate the need to renew the original ISFSI license. The alternative would, however, add costs and doses associated with repackaging. CP&L concludes that repackaging and moving the fuel to the general license ISFSI does not offer net environmental benefits over renewing the existing RNP ISFSI license.

E7.2.6 CONSTRUCT AN ISFSI AWAY FROM THE RNP SITE

NRC evaluated the environmental impacts of constructing and operating a private ISFSI at the Skull Valley Goshutes Indian Reservation and concluded that the proposed ISFSI would reduce the already small environmental effects of spent fuel storage at reactor sites (NRC 2001). The Skull Valley ISFSI is being sized to accommodate spent fuel stored at all of the nation's commercial nuclear plants (Skull Valley Goshutes no date) and, if constructed and operated as planned, would be available for RNP spent fuel storage. Given the small nature of ISFSI impacts at either location, CP&L would probably base a decision to use the Skull Valley ISFSI on economic factors. Until Skull Valley is operational, however, CP&L believes that prudent management dictates maintaining the option of continuing RNP ISFSI usage through license renewal.

E7.3 NO ACTION

Under the no action alternative, NRC would not renew the RNP ISFSI license. CP&L could not lawfully store spent fuel at the ISFSI after August 31, 2006, and would have to remove all spent fuel that it currently stores there. Other sections of Chapter 7 address potential alternatives for storing RNP ISFSI spent fuel.

NRC has prepared a generic environmental impact statement on decommissioning of nuclear facilities, including ISFSIs (NRC 1988). NRC evaluated ISFSI decommissioning alternatives, radiation safety, cost, waste disposal, and socioeconomic effects. NRC identified no prohibitory technical or environmental issues.

The RNP ISFSI no action alternative, like the proposed action and all other alternatives, would involve eventual decommissioning of the RNP ISFSI. Decommissioning activities and their impacts are not discriminators between the proposed action and any other alternative, including no action. CP&L will have to decommission the RNP ISFSI regardless of the NRC decision on license renewal; license renewal would only postpone decommissioning. CP&L adopts by reference the NRC discussion of ISFSI decommissioning effects.

CP&L concludes that the no action alternative provides no environmental advantages over license renewal.

E7.4 REFERENCES

CP&L (Carolina Power and Light Company). No date. Independent Spent Fuel Storage Installation. Environmental Report. H. B. Robinson Steam Electric Plant.

DOE (U.S. Department of Energy). 2003. Repository Licensing Overview Fact Sheet. Available at: www.ocrwm.doe.gov/factsheets/doeymp0111.html. Accessed October 2, 2003.

NRC (U.S. Nuclear Regulatory Commission). 1986. Environmental Assessment Related to the Construction and Operation of the H. B. Robinson Independent Spent Fuel Storage Installation. March. Washington, D.C.

NRC (U.S. Nuclear Regulatory Commission). 1988. Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities. NUREG-0586. August.

NRC (U.S. Nuclear Regulatory Commission). 2001. Final Environmental Impact Statement for the Construction and Operation of an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians and the Related Transportation Facility in Tooele County, Utah. NUREG-1714. December. Washington, D.C.

Skull Valley Goshutes. No date. Skull Valley Facility. Available at http://www.skullvalley goshutes.org/. Accessed October 24, 2003.

E8.0 COMPARISON OF THE IMPACTS OF LICENSE RENEWAL WITH THE ALTERNATIVES

Table E8-1 compares the environmental impacts of renewing the license of the RNP Independent Spent Fuel Storage Installation with the alternatives.

E8-1.
Comparison of Robinson ISFSI License Renewal with the Alternatives

	Alternatives										
Impacts	**License Renewal**	**No Action**	**Ship to permanent repository**	**Ship to Shearon Harris**	**Increase RNP pool capacity**	**Construct new pool at RNP**	**Ship to reprocessor**	**Ship to interim storage facility**	**Ship to another utility's storage**	**Construct a second ISFSI**	**Other technologies**
		No environmental advantage. Requires transfer of fuel from the ISFSI to another storage facility	Not a reasonable alternative, because repository will not be available until after expiration of current license	Not a reasonable alternative, because there is insufficient storage space in Shearon Harris spent fuel storage pool	Not a reasonable alternative, because past modification has maximized the capacity of the existing pool		Not a reasonable alternative, because there are no domestic reprocessors	Not a reasonable alternative, because there is no federal interim storage facility	Not a reasonable alternative, because excess capacity is not available at other plants	No environmental advantage. Requires transfer of fuel from the ISFSI to another storage facility	No environmental advantage. Requires transfer of fuel from the ISFSI to another storage facility
Geology/Soils	None					Small				Small	None
Hydrology	None					Small				Small	None
Air Quality	None					Small				Small	None
Aquatic Resources	None					None				None	None
Socioeconomics	None					Small				Small	None
Land Use	None					Small				Small	None
Threatened or Endangered Species	None					None				None	None
Historic/Cultural Resources	Small					Small				Small	Small
Normal Operations											
Occupational Dose	Small					Small, but greater than license renewal				Small, but greater than license renewal	Small, but greater than license renewal
Dose to Public	Small					Small				Small	Small
Accidents											
Occupational Dose	Small					Small				Small	Small
Dose to Public	Small					Small				Small	Small

E9.0 STATUS OF COMPLIANCE

E9.1 PROPOSED ACTION

E9.1.1 GENERAL

The H. B. Robinson Steam Electric Plant, Unit No. 2, Environmental Report for license renewal provides a list of all authorizations for current RNP operations. The RNP ISFSI does not require any additional permits, licenses (other than SNM-2502), or approvals to operate.

Table E9-1 below lists the authorizations and consultations that are precedent to the Nuclear Regulatory Commission renewing the ISFSI Operating License. This section discusses each of these in more detail.

Table E9-1.
Environmental Authorizations for RNP ISFSI License Renewal[a]

Agency	Authority	Requirement	Remarks
U.S. Nuclear Regulatory Commission	Atomic Energy Act (42 USC 2011 et seq.)	ISFSI License Renewal	Environmental Report Supplement submitted in support of ISFSI license renewal application.
U.S. Fish and Wildlife Service	Endangered Species Act, Section 7 (16 USC 1531)	Consultation	Requires federal agency issuing a license to consult with FWS.
South Carolina Department of Archives and History	National Historic Preservation Act, Section 106 (16 USC 470)	Consultation	Requires federal agency issuing a license to consider cultural impacts and consult with State Historic Preservation Officer.

a. No renewal-related requirements identified for local or other agencies.
FWS = U.S. Fish and Wildlife Service

E9.1.2 THREATENED AND ENDANGERED SPECIES CONSULTATIONS

Section 7 of the Endangered Species Act (16 USC 1531 et seq.) requires federal agencies to ensure that agency action is not likely to jeopardize any species that is listed or proposed for listing as endangered or threatened. The Act addresses consultation with the U.S. Fish and Wildlife Service (FWS) regarding effects on non-marine species. FWS and the National Marine Fisheries Service (which has jurisdiction over marine species) have issued joint procedural regulations in 50 CFR 402, Subpart B, which address consultation, and FWS maintains the joint list of threatened and endangered species in 50 CFR 17.

As discussed in Section E4.3.2, threatened or endangered species are known from the vicinity of RNP. Therefore, under Section 7 of the Endangered Species Act, the NRC may consult with FWS to ensure the proposed action will not jeopardize the continued existence of any threatened or endangered species.

Although not required of an applicant by federal law, CP&L has chosen to invite comment from federal and state agencies regarding potential effects that the RNP ISFSI license renewal may have. Appendix A includes copies of CP&L correspondence with FWS and the South Carolina Department of Natural Resources.

E9.1.3 NATIONAL HISTORIC PRESERVATION ACT

Section 106 of the National Historic Preservation Act (16 USC 470 et seq.) requires federal agencies having the authority to license any undertaking, prior to issuing the license, to take into account the effect of the undertaking on historic properties, and to afford the Advisory Committee on Historic Preservation the opportunity to comment on the undertaking. Council regulations provide for establishing an agreement with any State Historic Preservation Officer (SHPO) to substitute State review for Council review (35 CFR 800.7). Therefore, the NRC may request comments from the South Carolina SHPO prior to renewing the ISFSI license.

Although not required of an applicant by federal law, CP&L has chosen to invite comment from the SHPO regarding potential effects that the RNP ISFSI license renewal may have. Appendix B includes copies of CP&L correspondence with the South Carolina SHPO.

APPENDIX A

Special Status Species Correspondence

Progress Energy

Serial: RNP-RA/03-0135

OCT 2 9 2003

Mr. Roger Banks
Field Supervisor
U.S. Fish and Wildlife Service
176 Croghan Spur Road
Suite 200
Charleston, S.C. 29407

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
REQUEST FOR INFORMATION REGARDING THREATENED OR ENDANGERED SPECIES

Dear Mr. Banks:

Carolina Power & Light (CP&L) Company, now doing business as Progress Energy Carolinas, Inc. (PEC), is preparing an application to the United States Nuclear Regulatory Commission (NRC) to renew the license of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Independent Spent Fuel Storage Installation (ISFSI) in Darlington County, South Carolina. As part of the license renewal process, the NRC expects applicants to identify any adverse impacts to threatened or endangered species.

The ISFSI is co-located with HBRSEP, Unit No. 2. NRC is reviewing a PEC application to renew the HBRSEP, Unit 2, operating license and has provided your office with a biological assessment concluding that the HBRSEP, Unit 2, license renewal might affect, but is unlikely to adversely affect, four endangered species.

The ISFSI is composed of eight 12-foot high reinforced concrete storage modules on two concrete pads supporting one three module unit and one five module unit. The ISFSI occupies a small area inside the HBRSEP, Unit No. 2, protected area and is not visible from offsite. A layout of the site, with the ISFSI identified, is included as an attachment to this letter.

In the late 1980s, PEC placed dry shielded canisters containing spent nuclear fuel from HBRSEP, Unit No. 2, into the modules. Operational activities are limited to routine inspection and monitoring. PEC has no plans to add more fuel, modify the ISFSI, or change current ISFSI operations. PEC will remove the fuel and dismantle the modules once the U. S. Department of Energy completes the federal geologic repository for spent fuel and PEC can ship spent fuel to the repository.

PEC has concluded that the operation of the ISFSI has no effect on any threatened or endangered species and that license renewal would not alter this conclusion. PEC plans to submit its ISFSI license renewal application to NRC in the first quarter of 2004. As part of its review, NRC may

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

U.S. Fish and Wildlife Service
Serial: RNP-RA/03-0135
Page 2 of 2

consult with your agency regarding this license renewal application in accordance with Section 7 of the Endangered Species Act (16 USC 1536).

By contacting you early in the ISFSI license renewal application process, any questions or concerns may be addressed early in order to facilitate an expeditious NRC consultation. After your review, PEC would appreciate receiving a letter from your agency detailing any concerns or confirming PEC's conclusion that operation of the ISFSI over the license renewal term would not adversely affect any threatened or endangered species. Your agency's response will be included in the environmental report that will be submitted to the NRC as part of the license renewal application.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom at (843) 857-1253.

Sincerely,

J F Lucas

J. F. Lucas
Manager - Support Services - Nuclear

JSK/jsk

Attachment

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. H. J. Porter, Division of Radioactive Waste Management (SC)
Ms. Julie Holling, S.C. Heritage Trust Program, SCDNR

FIGURE WITHHELD UNDER 10 CFR 2.390

South Carolina Department of
Natural Resources

NOV 06 2003

Paul A. Sandifer, Ph.D.
Director
William S. McTeer
Deputy Director for
Wildlife and Freshwater Fisheries

October 31, 2003

Mr. J.F. Lucas, Manager – Support Services - Nuclear
Progress Energy Carolinas, Inc.
3581 West Entrance Rd.
Hartsville, SC 29550

RRA-03-0113

RE: H. B. Robinson Steam Electric Plant, Unit No. 2
Independent Spent Fuel Storage Installation
Request for Information Regarding Threatened or Endangered Species

Dear Mr. Lucas,

I can only verify that there are no occurrences in the vicinity of your project. I can't approve or disapprove of your project. I have checked our database, and there are no known occurrences of any federally or state listed threatened or endangered species within one mile of the project site. Please understand that our database does not represent a comprehensive biological inventory of the state. There may be occurrences of species in the vicinity of your project area that have not been reported to us. Fieldwork remains the responsibility of the investigator. As an indication of other potential occurrences in the area, I have enclosed a list of rare, threatened, and endangered species for Darlington and Chesterfield counties. The highlighted ones are of legal significance. The remaining species on the list are of concern in the state. As a professional courtesy, we ask that you acknowledge S.C. Heritage Trust as a source of information whenever you use this data in reports.

If you need additional assistance, please contact me by phone at 803/734-3917 or by e-mail at JulieH@scdnr.state.sc.us.

Sincerely,

Julie Holling

Julie Holling, Data Manager
SC Department of Natural Resources
Heritage Trust Program

Encl.

Rembert C. Dennis Building • 1000 Assembly St • P.O. Box 167 • Columbia, S.C. 29202 • Telephone: 803/734-3886
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RARE, THREATENED, AND ENDANGERED SPECIES OF DARLINGTON COUNTY

STATUS	GRANK	SRANK	SCIENTIFIC NAME	COMMON NAME
ANIMALS:				
SC	G5	S3?	CONDYLURA CRISTATA	STAR-NOSED MOLE
SE	G3G4	S2?	CORYNORHINUS RAFINESQUII	RAFINESQUE'S BIG-EARED BAT
SC	G5	S?	KINOSTERNON BAURII	STRIPED MUD TURTLE
FE/SE	G3	S2	PICOIDES BOREALIS	RED-COCKADED WOODPECKER
SC	G5	S?	RANA PALUSTRIS	PICKEREL FROG
PLANTS:				
SC	G3T2	S?	AMORPHA GEORGIANA VAR GEORGIANA	GEORGIA LEADPLANT
SC	G5?Q	S1	ARABIS MISSOURIENSIS	MISSOURI ROCK-CRESS
SC	G5	S?	ARISTOLOCHIA TOMENTOSA	WOOLLY DUTCHMAN'S-PIPE
SC	G3	S?	ASTRAGALUS MICHAUXII	SANDHILLS MILKVETCH
SC	G2G3	S?	BALDUINA ATROPURPUREA	PURPLE BALDUINA
SC	G4	S?	BALDUINA UNIFLORA	ONE-FLOWER BALDUINA
SC	G4?	S?	CALOPOGON BARBATUS	BEARDED GRASS-PINK
SC	G4	S1	CAREX COLLINSII	COLLINS' SEDGE
SC	G5T5	S1	CIRCAEA LUTETIANA SSP CANADENSIS	ENCHANTER'S NIGHTSHADE
SC	G5	S1	CLADIUM MARISCOIDES	TWIG RUSH
SC	G5	S?	DRABA REPTANS	CAROLINA WHITLOW-GRASS
SC	G5	S?	DRYOPTERIS SPINULOSA	SPINULOSE WOOD-FERN
SC	G4	S?	HELENIUM PINNATIFIDUM	SOUTHEASTERN SNEEZEWEED
SC	G4	S3	ILEX AMELANCHIER	SARVIS HOLLY
NC	G3	S1	KALMIA CUNEATA	WHITE-WICKY
SC	G4G5	S?	LEPUROPETALON SPATHULATUM	SOUTHERN LEPUROPETALON
SC	G4	S1S2	LYGODIUM PALMATUM	CLIMBING FERN
FE/SE	G3	S1	LYSIMACHIA ASPERULIFOLIA	ROUGH-LEAVED LOOSESTRIFE
SC	G2G3	S?	MACBRIDEA CAROLINIANA	CAROLINA BIRD-IN-A-NEST
SC	G5	S?	MENISPERMUM CANADENSE	CANADA MOONSEED
SC	G5	S?	OPHIOGLOSSUM VULGATUM	ADDER'S-TONGUE
SC	G4	S?	PLEEA TENUIFOLIA	RUSH FALSE-ASPHODEL
SC	G5	S1S2	PSILOTUM NUDUM	WHISK FERN
NC	G4	S1	PYXIDANTHERA BARBULATA	FLOWERING PIXIE-MOSS
NC	G2Q	S2	PYXIDANTHERA BREVIFOLIA	WELL'S PIXIE-MOSS
SC	G3	S2	RHEXIA ARISTOSA	AWNED MEADOWBEAUTY
SC	G5T3T4	S?	RUELLIA CAROLINIENSIS SSP CILIOSA	A PETUNIA
SC	G3	S1	SARRACENIA RUBRA	SWEET PITCHER-PLANT
SC	G4G5	S?	SCIRPUS SUBTERMINALIS	WATER BULRUSH
SC	G4	S?	SCUTELLARIA PARVULA	SMALL SKULLCAP
SC	G5	S1	SOLIDAGO BICOLOR	WHITE GOLDENROD
NC	G3	S1	SOLIDAGO VERNA	SPRING-FLOWERING GOLDENROD
SC	G3	S?	TOFIELDIA GLABRA	WHITE FALSE-ASPHODEL

RARE, THREATENED, AND ENDANGERED SPECIES OF CHESTERFIELD COUNTY

STATUS	GRANK	SRANK	SCIENTIFIC NAME	COMMON NAME
ANIMALS:				
SC	G5	S3?	CONDYLURA CRISTATA	STAR-NOSED MOLE
SC	G4	S?	ELLIPTIO CONGARAEA	CAROLINA SLABSHELL
SC	G2G3	S?	ELLIPTIO LANCEOLATA	YELLOW LANCE
SC	G5	S1	ETHEOSTOMA FLABELLARE	FANTAIL DARTER
SC	G2	S?	HETERODON SIMUS	SOUTHERN HOGNOSE SNAKE
ST	G4	S2S3	HYLA ANDERSONII	PINE BARRENS TREEFROG
FE/SE	G1	S1	LASMIGONA DECORATA	CAROLINA HEELSPLITTER
SC	G4	S4	LIMNOTHLYPIS SWAINSONII	SWAINSON'S WARBLER
SC	G4	S1?	NOTROPIS CHILITICUS	REDLIP SHINER
FE/SE	G3	S2	PICOIDES BOREALIS	RED-COCKADED WOODPECKER
SC	G4	S3S4	PITUOPHIS MELANOLEUCUS	PINE OR GOPHER SNAKE
SC	G5	S?	PYGANODON CATARACTA	EASTERN FLOATER
SC	G3	S2	SEMOTILUS LUMBEE	SANDHILLS CHUB
SC	G3	S?	VILLOSA CONSTRICTA	NOTCHED RAINBOW
SC	G4	S?	VILLOSA DELUMBIS	EASTERN CREEKSHELL
PLANTS:				
SC	G4?	S?	ANDROPOGON MOHRII	BROOMSEDGE
SC	G5T3T4	S1	ANDROPOGON PERANGUSTATUS	NARROW LEAVED BLUESTEM
RC	G4	S1	ASPLENIUM BRADLEYI	BRADLEY'S SPLEENWORT
SC	G3	S?	ASTRAGALUS MICHAUXII	SANDHILLS MILKVETCH
SC	G4G5	S?	BURMANNIA BIFLORA	NORTHERN BURMANNIA
NC	G4	S?	CALAMOVILFA BREVIPILIS	PINE-BARRENS REED-GRASS
SC	G4	S1	CAREX COLLINSII	COLLINS' SEDGE
SC	G4G5	S1S2	CHRYSOMA PAUCIFLOSCULOSA	WOODY GOLDENROD
SC	G3?	S?	DANTHONIA EPILIS	BOG OAT-GRASS
SC	G4	S?	ERIOCAULON TEXENSE	PIPEWORT
SC	G3	S2	GENTIANA AUTUMNALIS	PINE BARREN GENTIAN
RC	G4	S1	HUDSONIA ERICOIDES	GOLDEN-HEATHER
NC	G3	S1	KALMIA CUNEATA	WHITE-WICKY
SC	G2	S1	LILIUM IRIDOLLAE	PANHANDLE LILY
RC	G3	S2	MYRIOPHYLLUM LAXUM	PIEDMONT WATER-MILFOIL
SC	G4	S2	NESTRONIA UMBELLULA	NESTRONIA
SC	G5	S?	PASPALUM BIFIDUM	BEAD-GRASS
SC	G4	S1	POTAMOGETON CONFERVOIDES	ALGAE-LIKE PONDWEED
NC	G4	S1	PYXIDANTHERA BARBULATA	FLOWERING PIXIE-MOSS
SC	G4T4	S?	PYXIDANTHERA BARBULATA VAR BARBULATA	WELL'S PYXIE MOSS
NC	G2Q	S2	PYXIDANTHERA BREVIFOLIA	WELL'S PIXIE-MOSS
SC	G5	S1	RHYNCHOSPORA ALBA	WHITE BEAKRUSH
SC	G4	S?	RHYNCHOSPORA OLIGANTHA	FEW-FLOWERED BEAKED-RUSH
SC	G4	S?	RHYNCHOSPORA STENOPHYLLA	CHAPMAN BEAKRUSH
SC	G3	S1	SARRACENIA RUBRA	SWEET PITCHER-PLANT
SC	G3G4	S?	SCIRPUS ETUBERCULATUS	CANBY BULRUSH
SC	G3	S?	SOLIDAGO PULCHRA	CAROLINA GOLDENROD
NC	G3	S1	SOLIDAGO VERNA	SPRING-FLOWERING GOLDENROD
SC	G3	SR	SPOROBOLUS PINETORUM	CAROLINA DROPSEED
NC	G2?	S1	SPOROBOLUS TERETIFOLIUS	WIRE-LEAVED DROPSEED
RC	G5	S1	SYNGONANTHUS FLAVIDULUS	YELLOW PIPEWORT
SC	G3	S?	TOFIELDIA GLABRA	WHITE FALSE-ASPHODEL

STATUS..	GRANK....	SRANK..	SCIENTIFIC NAME......................	COMMON NAME...............
SC	G3	S?	TRIDENS CAROLINIANUS	CAROLINA FLUFF GRASS
SC	G4	S1	UTRICULARIA OLIVACEA	PIEDMONT BLADDERWORT
SC	G3	S?	XYRIS CHAPMANII	CHAPMAN'S YELLOW-EYED GRASS
SC	G3	S?	XYRIS SCABRIFOLIA	HARPER'S YELLOW-EYED GRASS

APPENDIX B

Cultural Resources Correspondence

Progress Energy

Serial: RNP-RA/03-0134
OCT 2 9 2003

Dr. Rodger E. Stroup
State Historic Preservation Officer
South Carolina Department of Archives and History
Archives and History Center
8301 Parklane Road
Columbia, S.C. 29233

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
REQUEST FOR INFORMATION REGARDING CULTURAL AND HISTORIC RESOURCES

Dear Dr. Stroup:

Carolina Power & Light (CP&L) Company, now doing business as Progress Energy Carolinas, Inc. (PEC), is preparing an application to the United States Nuclear Regulatory Commission (NRC) to renew the license of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Independent Spent Fuel Storage Installation (ISFSI) in Darlington County, South Carolina. As part of the license renewal process, the NRC requires that an applicant "assess whether any historic or archaeological properties will be affected by the proposed project" (10 CFR 51.53(c)(3)(ii)(K)).

The ISFSI is co-located with HBRSEP, Unit No. 2. NRC is reviewing a PEC application to renew the HBRSEP, Unit No. 2, operating license, and Ms. Marta Matthews of your office has been in communication with the NRC regarding that review.

The ISFSI is composed of eight 12-foot high reinforced concrete storage modules on two concrete pads supporting one three module unit and one five module unit. The ISFSI occupies a small area inside the HBRSEP, Unit No. 2, protected area and is not visible from offsite. A layout of the site, with the ISFSI identified, is included as an attachment to this letter.

In the late 1980s, PEC placed dry shielded canisters containing spent nuclear fuel from HBRSEP, Unit No. 2, into the modules. Operational activities are limited to routine inspection and monitoring. PEC has no plans to add more fuel, modify the ISFSI, or change current ISFSI operations. PEC will remove the fuel and dismantle the modules once the U. S. Department of Energy completes the federal geologic repository for spent fuel and PEC can ship spent fuel to the repository.

PEC has concluded that the operation of the ISFSI has no adverse impacts on any historic or archaeological resources and that license renewal would not alter this conclusion. PEC plans to submit its ISFSI license renewal application to NRC in the first quarter of 2004. As part of its review, NRC may request an informal consultation with your agency regarding this license

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

State Historic Preservation Officer
Serial: RNP-RA/03-0134
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renewal application in accordance with Section 106 of the National Historic Preservation Act of 1966, as amended (16 USC 470), and the Federal Advisory Council on Historic Preservation Regulations (36 CFR 800).

By contacting you early in the ISFSI license renewal application process, any questions or concerns may be addressed early in order to facilitate an expeditious NRC consultation. After your review, PEC would appreciate receiving a letter from your agency detailing any concerns or confirming PEC's conclusion that operation of the ISFSI over the license renewal term would not adversely affect any historic or archaeological resources. Your agency's response will be included in the environmental report that will be submitted to the NRC as part of the license renewal application.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom at (843) 857-1253.

Sincerely,

J. F. Lucas

J. F. Lucas
Manager - Support Services - Nuclear

JSK/jsk

Attachment

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. H. J. Porter, Division of Radioactive Waste Management (SC)

FIGURE WITHHELD UNDER 10 CFR 2.390

Appendix F

Additional Information

APPENDIX F: ADDITIONAL INFORMATION

F1.0 ADDITIONAL INFORMATION

F1.1 TRAINING AND QUALIFICATIONS (10 CFR 72.28)

F1.1.1 TECHNICAL QUALIFICATIONS

Carolina Power and Light (CP&L) Company has been involved in the nuclear energy field since the 1960's. CP&L's first commercial nuclear power plant, H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, also referred to as the Robinson Nuclear Plant (RNP), started operation in 1970. CP&L additionally operates Brunswick Steam Electric Plant, Units 1 and 2, and the Shearon Harris Nuclear Power Plant, Unit No. 1. Reactor operations have provided extensive experience in the receipt, handling, storage, and shipping of nuclear fuel. Operations activities of the RNP Independent Spent Fuel Storage Installation (ISFSI) are conducted by the operating staff of RNP. Additional experience in fuel storage has been obtained by operation of the RNP ISFSI since initial fuel loading in 1989.

A discussion of technical qualifications was submitted with the initial license application for the RNP ISFSI. Additionally, discussion of personnel qualifications is available in the Section 9.1.3 of the ISFSI Safety Analysis Report (SAR) and in Section 13.1.3 of the RNP Updated Final Safety Analysis Report (UFSAR). Both of these documents are updated in accordance with regulatory requirements.

A commitment to staff the project (i.e., the ISFSI) with an adequate cadre of personnel possessing the required skills throughout all phases of the project was contained in Chapter 2 of the initial license application for the RNP ISFSI. The commitment remains throughout the renewed license period to provide continued assurance of the safety of the public and operating personnel.

F1.1.2 PERSONNEL TRAINING

A discussion of operator training was submitted with the initial license application for the RNP ISFSI. Additionally, discussion of the training programs is provided in Section 9.3 of the ISFSI Safety Analysis Report and in Section 13.2 of the RNP UFSAR. Both of these documents are updated in accordance with regulatory requirements.

F1.1.3 OPERATING ORGANIZATION

Operation of the RNP ISFSI is integrated with the operation of RNP. A description of the operating organization of RNP is contained in the RNP

UFSAR, Sections 13.1.1 and 13.1.2. This document is updated periodically in accordance with regulatory requirements.

F1.2 FINANCIAL ASSURANCE FOR DECOMMISSIONING (10 CFR 72.30)

A decommissioning plan was submitted with the original license application. The basic elements of the plan, i.e., shipping of the fuel to an off-site repository, and removal and disposal of the horizontal storage modules, remain unchanged. The actual activities at the time of decommissioning will be dependent upon the regulations and practices in effect at that time. Discussion of decommissioning of the ISFSI is contained in Section 3.5 and 9.6 of the ISFSI Safety Analysis Report, which is periodically updated in accordance with regulatory requirements.

Decommissioning costs were estimated to be a small fraction of the decommissioning costs of the operating nuclear unit. A recent estimate of this cost is $1.67 million for decontamination and removal of the ISFSI. Additional discussion of decommissioning funding and financial qualifications are contained in CP&L letter dated September 15, 1999.

F1.3 EMERGENCY PLANNING (10 CFR 72.32)

A description of emergency planning was submitted with the initial license application for the RNP ISFSI.

Emergency planning for the ISFSI is integrated into the emergency plan for RNP. As such, the emergency plan and implementing procedures are updated in accordance with 10 CFR 50, Appendix E; 10 CFR 50.54(q); and, 10 CFR 72.44(f). Emergency plan changes made pursuant to 10 CFR 72.44(f) are submitted to the NRC in accordance with 10 CFR 72.44(f).

The emergency plan is discussed in Section 9.5 of the ISFSI Safety Analysis Report, which is periodically updated in accordance with regulatory requirements.

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H. B. Robinson
Steam Electric Plant

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

1.1 INTRODUCTION

Carolina Power & Light Company (CP&L) has entered into an agreement with the U. S. Department of Energy (DOE) to conduct a licensed at-reactor dry storage demonstration program for spent nuclear fuel to be located at the H. B. Robinson Steam Electric Plant Unit No. 2 (HBR2). This document provides the safety analysis report (SAR) required as part of the license application under 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)." This SAR is organized in accordance with the guidelines contained in Regulatory Guide 3.48. Table 1.1-1 lists the acronyms and Table 1.1-2 lists the abbreviations used throughout this document.

The Nuclear Waste Policy Act of 1982 (NWPA) established 1998 as an operational date for a spent fuel/high level radioactive waste repository. To assist utilities in providing for spent fuel storage until a repository is operational, the Nuclear Waste Policy Act requires DOE to ". . . establish a demonstration program in cooperation with the private sector, for the dry storage of spent nuclear fuel at civilian nuclear power reactor sites, with the objective of establishing one or more technologies that the Commission [NRC] may by rule approve for use at the sites of civilian nuclear power reactors without, to the maximum extent practicable, the need for additional site-specific approvals by the Commission." Accordingly, on May 9, 1983, the DOE issued its Solicitation for Cooperation Agreement Proposal (#DE-SC06-83RL10432) for a Licensed, At-Reactor, Dry Storage Demonstration Program. Carolina Power & Light Company, in response to the DOE solicitation, submitted a proposal on August 23, 1983 to DOE to demonstrate the NUTECH Engineers, Inc. (NUTECH) Horizontal Modular Storage (NUHOMS) system at the site of the H. B. Robinson Steam Electric Plant Unit No. 2. DOE accepted CP&L's proposal in October 1983, and the contract was signed in March 1984 (Reference 1.1). The Electric Power Research Institute (EPRI) has also entered into the program as a participant. EPRI, in particular, is involved in the research and development aspects of the program.

The NUHOMS system is the dry storage design used for the H. B. Robinson (HBR) ISFSI. In addition to this SAR, Revision 1 of the generic Topical Report for the NUHOMS system, submitted by NUTECH in November 1985 (Reference 1.2), provides the details of the system to be utilized at HBR2. Figure 1.1-1 shows the primary components of the HBR ISFSI. The location of the ISFSI on the Robinson site is shown on Figure 1.1-2.

The NUHOMS 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 seven irradiated fuel assemblies is transferred from the reactor 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 system is a totally passive installation that is designed by analysis to provide shielding and safe confinement of irradiated fuel. The dry shielded canister and horizontal storage module have been designed to function during and withstand certain accidents as described in this SAR.

The fuel assemblies to be stored in the ISFSI are currently located in the HBR2 spent fuel pool and were irradiated in the HBR2 reactor. Seven fuel assemblies are stored in each dry shielded canister. One dry shielded canister is stored in each concrete module. The license application by CP&L requests a license to construct and operate a total of eight modules. CP&L initially intends to construct three modules. Construction of the first three modules will take approximately one year. In accordance with the DOE agreement, the first year of operation of the three module ISFSI will be part of a test program. Normal operation of the facility will continue past the first year for up to 20 years under the initial license until a permanent federal repository is available to store the spent fuel. An additional five modules have been constructed adjacent to the original three.

TABLE 1.1-1

ACRONYMS

ACI	American Concrete Institute
A/E	architect/engineer
AIF	Atomic Industrial Forum
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASTM	American Society of Testing and Materials
CP&L	Carolina Power & Light Company
CQAD	Corporate Quality Assurance Department
CVCS	Chemical and Volume Control System
DBT	Design Basis Tornado
DOE	U. S. Department of Energy
DSC	dry shielded canister
E	East
EEI	Edison Electric Institute
ENE	East Northeast
EOC	Emergency Operations Center
EPRI	Electric Power Research Institute
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report
GE	General Electric Company
HBR	H. B. Robinson Steam Electric Plant
HBR2	H. B. Robinson Steam Electric Plant Unit No. 2
HSM	horizontal storage module
IC	internal combustion
IFA	irradiated fuel assembly
ISFSI	independent spent fuel storage installation
N	North
NC	North Carolina
NE	Northeast
NED	Nuclear Engineering Department
NFS	Nuclear Fuel Section
NNE	North Northeast
NNW	North Northwest
NRC	Nuclear Regulatory Commission
NUHOMS	NUTECH Horizontal Modular Storage
NUTECH	NUTECH Engineers, Inc.
NW	Northwest
NWPA	Nuclear Waste Policy Act of 1982
ONRR	Office of Nuclear Reactor Regulation
PWR	pressurized water reactor
QA	quality assurance
QC	quality control
RG	Regulatory Guide
SAR	Safety Analysis Report
SC	South Carolina
SE	Southeast
SSE	safe shutdown earthquake

SSE	South Southeast

TABLE 1.1-1 (Cont'd)

SSW	South Southwest
SW	Southwest
US	United States
UTM	Universal Transverse Mercator
WDS	Waste Disposal System
WNW	West Northwest
10CFR	Code of Federal Regulations, Title 10

TABLE 1.1-2

ABBREVIATIONS

atm	atmosphere
bar	bar
cm	centimeter
^{o}C	degrees Centigrade
^{o}F	degrees Farenheit
fps	feet per second
ft/s	feet per second
ft	foot
ft-lb	foot pounds
He	helium
kg	kilogram
kw	kilowatt
k-in	kip inch
ksi	kips per square inch
Kr-85	Krypton 85
MWd/MT	megawatt days per metric ton
MWe	megawatts electric
MWt	megawatts thermal
Hg	Mercury
m	meter
☞Ci/cm^2	microcuries per square centimeter
mph	miles per hour
mm	millimeter
mrem/hr	millirem per hour
mR/hr	milliroentgen per hour
k_{eff}	neutron multiplication factor, effective
N	Newton
Pu-239	Plutonium-239
Pu-241	Plutonium-241
lb	pound
lb_f	pounds-force
psf	pounds per square foot
psi	pounds per square inch
psia	pounds per square inch, atmospheric
psig	pounds per square inch, guage
sec	second
sq. mi.	square mile
kips	thousand pounds
ton	ton
U-235	Uranium 235

1.2 GENERAL DESCRIPTION OF INSTALLATION

1.2.1 GENERAL DESCRIPTION

The 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 steel dry shielded canister (DSC) with an internal basket which holds the IFAs.

Each HSM contains one DSC and each DSC contains seven fuel assemblies. The modules are to be constructed on a common foundation and are interconnected. The outer, exposed walls are 3 1/2 feet thick concrete to provide the necessary shielding. The initial phase of construction includes three modules. A total of eight modules have been built and operated at the Robinson site. The second foundation for the additional five modules is constructed nearby, but separate from the initial three. It has been determined that construction of the additional five modules would not have any impact on continued operation of the initial three modules. Figure 1.2-1 shows the configuration of the HBR ISFSI.

In addition to these primary components, the HBR ISFSI also requires transfer equipment to move the DSCs from the irradiated fuel pool (where they are loaded with the IFAs) to the HSMs where they are stored. This transfer system consists of a transfer cask, a hydraulic ram, a tow vehicle, a trailer and a cask skid. This transfer system will interface with the existing HBR2 irradiated fuel pool, the cask crane, the site layout (i.e., roads and topography) and other procedural requirements.

1.2.2 PRINCIPAL SITE CHARACTERISTICS

The ISFSI is located on the H. B. Robinson Steam Electric Plant site near Hartsville, South Carolina. Carolina Power & Light Company owns and operates a 2339 MWt nuclear generating unit (Unit 2) and a 185 MWe fossil-fueled generating unit (Unit 1) on the Robinson site. The ISFSI is located within the Unit 2 protected area approximately 600 ft. west of the Unit 2 containment building.

1.2.3 PRINCIPAL DESIGN CRITERIA

The principal design criteria and parameters for the HBR ISFSI are shown in Table 1.2-1. The radiation sources are for the maximum burnup fuel. For the fuel to be stored, the radiation sources shall be less than or equal to the sources described in Table 1.2-1.

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 operational procedures for the transfer of the cask from fuel pool to the module site are such that the maximum height at which a credible cask drop can occur is limited to 2.44 m (8 ft). The detail description of the cask handling procedure and the transfer operation is presented in Sections 1.3.1.7 and 8.2.4 of this report. The canister body, the double containment welds on each end, and the DSC internals are designed to provide their intended safety functions after a 2.44 m (8 ft) drop. In fact, the original design of the DSC and its internals, as presented in the NUHOMS topical report, has been modified to withstand decelerations associated with drop heights significantly higher than 2.44 m (8 ft) limit. However, the 2.44 m (8 ft) drop is the minimum requirement used as a design criteria, since it envelopes any actual drop accident that could occur at the Robinson site. The details of the cask drop accident are contained in Section 8.2.4 of this report.

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., 51.7°C (125°F) inlet and 98.9°C (210°F) outlet).

1.2.4 OPERATING AND FUEL HANDLING SYSTEMS

The major operating systems of the 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 1.2-2 and the primary design parameters of the required systems are listed in Table 1.2-3. The fuel handling operations involving the cask (i.e., fuel loading, drying, trailer loading, etc.) and the remaining operations (cask-HSM alignment and DSC transfer) are unique to the ISFSI. Procedures for these activities were developed using H. B. Robinson fuel shipment procedures and experience gained during testing of the ISFSI.

1.2.5 SAFETY FEATURES

The principal safety feature of the ISFSI is the containment provided by the DSC and the concrete shielding of the HSM. 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 mrem/hr. Additional ISFSI safety features include:

a) Filling the DSC and cask annulus with demineralized water and providing a seal prior to lowering them into the spent fuel pool - Minimizes contamination of the DSC exterior by pool water.

b) Internal shield blocks inside the HSM - Reduces scatter dose out of the air inlet.

c) External shield blocks on the HSM - Reduces scatter dose out of the air outlet.

d) Shield plugs on the DSC - Reduces dose during DSC drying, helium filling and seal welding.

e) Double containment closure welds on each end of the DSC - Prevent 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 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 HBR2 waste systems handle the fuel pool water and air and inert gas which are vented from the DSC and cask during drying. Auxiliary handling systems (such as hydraulic pressure control, alignment, crane, etc.) are also required during the loading and transfer operation.

TABLE 1.2-1

DESIGN PARAMETERS FOR THE HBR ISFSI

Category	Criterion or Parameter	Value
Fuel Acceptance Criteria	Fissile Content	3.5% Fissile (U-235 Equivalent)
	Radiation Source Gamma Neutron Heat Load	 5.73×10^{15} photons/sec/assembly[1] 1.67×10^{8} neutron/sec/assembly 1 KW/Assembly
Dry Shielded Canister	Capacity per Canister	7 PWR Fuel Assemblies
	Size Length (typical) Diameter	 4.56m (179.5 in) 0.94m (36.9 in)
	Temperature (max. fuel rod clad)	 380°C (716°F)
	Cooling	Natural Convection
	Design Life	50 Years[2]
	Material	304 Stainless Steel with Lead End-Shields
	Internal Helium Pressure	0.0 psig " 0.5 psig
Horizontal Storage Module	Capacity	1 Dry Shielded Canister per Module
	Unit Size Length Height Width	3 modules per Unit 6.71m (22.00 ft) 3.81m (12.50 ft) 7.54m (24.75 ft)
	Unit Size Length Height Width	5 modules per Unit 6.71m (22.00 ft) 3.81m (12.50 ft) 11.86m (38.92 ft)
	Surface Radiation Dose Rate (average on contact)	20 mrem/hr
	Material	Reinforced Concrete
	Design Life	50 years[2]

1 Actual design limits are for seven assemblies in the DSC with source rates of 1.17×10^{9} neutrons/sec/DSC and 4.01×10^{16} photons/sec/DCS.

2 Expected life is much longer (hundreds of years); however, initial license application is for 20 years only. Future amendments may seek to extend the life.

TABLE 1.2-2

SUMMARY OF
ISFSI FUEL HANDLING OPERATIONS

1. Clean the DSC and load it into the transfer cask.
2. Fill the cask annulus and DSC with demineralized water.
3. Place the top lead plug on the DSC.
4. Lift the cask containing the DSC into the spent fuel pool.
5. Remove the top lead plug.
6. Load the fuel into the DSC.
7. Place the top lead plug on the DSC.
8. Install the cask collar lid.
9. Lift the cask containing the filled DSC out of the spent fuel pool and move it to the cask decontamination facility.
10. Drain water from the cask and DSC to a level approximately two inches below the top surface of the lead plug (approximately 15 gallons).
11. Seal weld the top lead plug onto the DSC body.
12. Perform liquid penetrant examination of top lead plug seal weld.
13. Drain the water from the cask annulus.
14. Drain, evacuate, and dry the DSC interior.
15. Backfill the DSC with helium.
16. Perform helium leak test.
17. Seal weld plugs in the drain and vent lines of the DSC.
18. Perform liquid penetrant examination of drain and vent line plug welds.
19. Perform helium leak test of drain and vent line plug welds.
20. Place and seal weld the top cover plate.
21. Perform liquid penetrant test on top cover plate weld.
22. Install the cask collar lid.
23. Lift the cask onto the trailer and lower it into the horizontal position.
24. Tow the trailer to the HSM.
25. Remove the HSM front access cover.
26. Remove the cask collar lid.
27. Align the cask and the HSM.
28. Insert the hydraulic ram.
29. Pull the DSC into the HSM.
30. Move the cask and the trailer away from the front of the HSM.
31. Replace and tack weld the HSM front access cover.
32. Disassemble the hydraulic ram from the rear of the HSM.
33. Install the seismic retainer and rear cover plate.

TABLE 1.2-3

PRIMARY DESIGN PARAMETERS FOR THE
ISFSI OPERATING SYSTEMS

System	Parameters	Value
Cask	Cavity Diameter	0.953m (37.5 in.)
	Cavity Length	4.572m (180 in.)
	Payload Capacity	9524 kg (21,000 lb)
	Heat Rating	$\geq$ 7kw
	Shielding (Surface Dose)	200 mrem/hr
Cask Movement	Liftable by Crane	N/A
	Rotatable by Crane from Vertical to Horizontal	N/A
Cask Lid	Removable in Horizontal Position	N/A
Trailer and Skid	Truck Transportable	N/A
	Cask Lid Must Protrude Past End of Trailer and Skid	15.25cm (6 in.)
	Capacity (Trailer)	73,000kg (80 tons)
	(Skid)	100,000kg (110 tons)
	Positioning Capability	6 in. Vertically 5 in. Towards Module 3 in. Parallel to Module

1.3 GENERAL SYSTEMS DESCRIPTIONS

The major systems, subsystems, and components of the HBR ISFSI are shown in Table 1.3-1. The following subsections briefly describe the principal systems and components and their operation.

1.3.1 SYSTEMS DESCRIPTIONS

1.3.1.1 Canister Design

Figure 1.3-1 shows the dry shielded canister. The DSC is sized to hold seven irradiated pressurized water reactor (PWR) fuel assemblies. The main component of construction is a stainless steel cylinder with a 0.5 inch wall and 0.94m (36.9 in.) outside diameter. The overall length is 179.5 in. The general system description of the canister design is available in the NUTECH Topical Report.

1.3.1.2 Horizontal Storage Module

An isometric view of a unit of three HSMs is shown in Figure 1.2-1. The HSM provides a unitized modular storage location for irradiated fuel. The HSM is constructed from reinforced concrete and structural steel. The modules will be constructed in place at the storage location. The thick concrete walls and roof of the HSM provide adequate neutron and gamma shielding. The general systems description of the HSM is provided in the NUTECH Topical Report.

The HSMs are placed in service on a load bearing foundation. Certain civil work is required to prepare the storage site for a level foundation and access area. This work includes the relocation of any existing underground utilities, excavation, backfill, compaction and leveling. Also, a 4 inch mud slab will be placed on the leveled subgrade to provide smooth working surface for the placement of the foundation.

1.3.1.3 Transfer Cask

The transfer cask used with the ISFSI provides shielding during the DSC drying operation and during the transfer to the HSM. For the HBR ISFSI, the IF-300 cask (which CP&L owns) licensed under 10CFR71 as a transportation cask will be used (Reference 1.3). In order to meet the cask cavity minimum length requirement and the criteria for cask collar lid removal in the horizontal position, the IF-300 cask requires an addition. The addition includes an extension collar (12.5 inches long and approximately 6 inches thick) with the inside diameter the same as that of the cask, and a 1 inch thick cask collar lid. In this modified configuration the energy absorbing properties of the cask is significantly reduced. However, as described in detail in Section 8.2.4 of this report there is no credible condition during the cask handling and transfer operation in which the cask could be dropped on its head. Hence, the removal of the cask head with its radial impact limiting fins does not affect the safety features of the ISFSI transfer operation.

1.3.1.4 Transporter

The transporter consists of a trailer with a capacity of 80 tons. The trailer carries the cask skid and the loaded transfer cask. The trailer is designed to ride as low to the ground as possible to minimize the HSM height. Four hydraulic jacks are placed under the trailer to provide vertical movement for alignment of the cask and HSM. The trailer is pulled by a conventional tractor.

1.3.1.5 Skid

The cask positioning skid is a welded wideflange frame assembly, which houses the cask cradle support and the cask saddle assemblies. These components are welded to the skid frame. The cask cradle and its support provide the rotational capabilities required to orient the cask from vertical to horizontal position. Once in the horizontal position the cask will be seated on the saddle and will be prevented from further movement. The skid is seated on 75 ton (minimum) capacity guided Hilman Rollers at each corner. There are four hydraulic positioner cylinders mounted on the skid frame and the trailer bed. The rollers and the cylinders will be used for the final alignment of the cask and the HSM. The entire skid assembly is seated on the trailer bed. During towing of the trailer, the skid is secured to the trailer bed by means of tie down brackets.

The skid and its various components are designed to withstand the inertia forces associated with transportation shocks. The features of the skid described above are shown in Figure 1.3-3.

1.3.1.6 Horizontal Hydraulic Ram

The horizontal hydraulic ram is a telescopic, hydraulic boom with a minimum capacity of 22,000 lb_f and a reach of 25 ft. The ram will be mounted on the concrete foundation and wall of the HSM on the opposite side from the loading position. Figure 1.3-4 shows the hydraulic ram.

1.3.1.7 System Operation

The primary operations (in sequence of occurrence) for the HBR system are shown schematically in Figure 1.3-5 and are described below:

a) Cask Preparation - Cask preparation includes exterior washdown and interior decontamination. These operations are done in the HBR2 cask decontamination facility outside the spent fuel pool area. The operations are standard cask operations and have been previously performed by CP&L personnel. Detailed procedures for these operations are described in Chapter 5.

b) Canister Preparation - The canister exterior is wiped down with demineralized water. The interior is cleaned of any debris as required. This ensures that the newly fabricated canister will meet existing HBR2-specific criteria for placement in the spent fuel pool.

c) Cask-Canister Loading - The empty canister is inserted into the cask. Proper alignment is assured by visual inspection.

d) Cask Lifting and Placement in the Pool - The cask annulus and DSC inside the cask are filled with demineralized water. A seal is then installed. This prevents an inrush of pool water when the cask with the DSC is placed in the spent fuel pool. This will also reduce (if not prevent) contamination of the DSC outer surface by the pool water. The cask with the DSC inside is then lifted into the fuel pool.

e) Canister-Assembly Loading - Seven irradiated fuel assemblies are placed into the canister basket. This operation is performed using approved ISFSI procedures.

f) Cask Collar Lid Placement - The cask collar lid placement operation consists of placing the DSC upper end-shield plug inside the DSC using the overhead crane. The cask collar lid is then placed on the cask using temporary guide pins and it is raised to the surface where the cask collar lid bolts are installed. The cask collar lid firmly holds the DSC in place while in the cask.

g) Cask Lifting out of the Pool - The filled and closed cask is lifted out of the spent fuel pool and placed (in the vertical position) on the drying pad inside the decontamination area. This operation is performed using ISFSI procedures. During this operation the overhead crane is equipped with a redundant yoke and as such is operating in a single failure proof mode. The use of the redundant yoke eliminates the possibility of any drop accident at this stage of operation.

h) Canister Sealing - The seal is removed from the cask/DSC annulus. Using the cask drain valve, a sufficient amount of water is drained from the annulus so as to enable a contamination survey to be performed approximately 12 inches below the top of the DSC shell. Should the contamination values from the annulus survey be higher than acceptable, the annulus may be flushed using demineralized water to reduce the contamination levels in this area. Using a pump, the water level in the canister is reduced to approximately 2 inches below the bottom surface of the top lead shield plug. A seal weld is applied to the interface of the DSC shell and the lead shield plug. This provides the primary seal for the DSC. After completion of this weld, the remaining water is removed from the DSC interior in preparation of backfilling with helium to perform a helium leak test of the weld.

i) Cask-Canister Drying - Initial removal of water from the DSC is accomplished by connecting a hose from the instrument air supply to the DSC vent connection. A second hose is then connected to the DSC siphon connection and routed to the drain manifold at CD-7. The instrument air valve is opened and the air flow recalculated. When the DSC is drained, the instrument air valve is closed. Following evacuation of the water from the DSC using compressed air, the vacuum drying system is connected to the DSC and activated to facilitate DSC drying until the water content meets the design criteria.

j) Helium Filling - In order to ensure that no fuel and/or cladding oxidation occurs during storage, the canister is filled with helium (He). To accomplish this, a portable He gas bottle is connected to the piping/tubing of the vacuum drying system.

The canister is filled with He gas (by activating valves) to a minimum pressure of 25 psig and a helium leak test is performed. The pressure is then reduced to 0.0 psig. After the canister is filled with the inert gas, the vent and drain lines are removed and the DSC vent and drain line connectors are plugged and welded closed. These welds are then checked for inert gas leakage. When the steel cover plate is then welded in place, the integrity of the penetrations is assured.

k) Final Canister Sealing - After the inert gas filling, the steel 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 were made and tested during fabrication. This operation provides the double seal integrity of the DSC.

l) Transporter-Skid Loading - The cask collar lid is positioned and bolted in place to close the cask for transport to the HSM. The cask is then lifted from the decontamination area by the overhead crane with its redundant yoke attached, and is placed on a concrete pad adjacent to the skid/trailer. Once the cask is on the concrete pad the lower half of the redundant yoke is removed to allow the cask to fit into the cask cradle. The maximum height required to raise the cask above the cradle is 2.44 m (8 ft). The cask is then raised just above the cradle atop the skid assembly which is held in the horizontal position. At this juncture the crane is operating without the redundant yoke. The cask is then lowered into the cradle until it is firmly seated. Next, the cask is tilted from the vertical to the horizontal position until the top region of the cask is firmly positioned on the saddle atop the skid. The lifting and tilting procedures used are identical to those used for loading the IF-300 shipping cask onto its rail car skid (Reference 1.5). The crane yoke is then removed and the cask is secured to the skid frame. At no time during the tilting operation is the bottom of the cask more than 2.44 m (8 ft) off the ground. Furthermore, since the cask is always lifted from the cask trunnions located on the upper region of the cask, the failure of the yoke will not cause the cask to drop on its head; hence, no possibility of a cask top end drop. If the yoke fails during tilting operation, the cask would land on its steel ring fins located near the top of the cask's outer shell. The cask drop accident analysis and further discussion on the postulated drop heights and orientations are presented in Section 8.2.4.

m) Transfer - Once loaded and secured, the transfer trailer is towed to the HSM. This movement is completely within the HBR2 plant site and protected area. The skid assembly is designed such that none of the cask redundant tie downs and support mechanism can fail due to the inertia forces associated with the transportation shocks and vibrations. Additionally, the skid is secured to the trailer bed by means of tie down brackets, designed to withstand the same forces. The possibility of cask dropping from the skid, or the cask/skid/trailer tipping or rolling over is extremely remote. The impact decelerations generated by such unlikely events are enveloped by the 2.44 m (8 ft) horizontal and vertical drop criteria due to the fact that the center of gravity of the cask is less than 2.44 m (8 ft) from the ground level during the transport operation.

n) Cask-Module Preparation - At the HSM storage area, the transfer trailer is backed into position and the HSM front access cover is raised and removed. Next, the cask collar lid is removed. The rear access cover plate is also removed. An alignment system and the hydraulic skid positioners are used for the final alignment of the canister and module.

o) Module Loading - After final alignment, the canister is pulled into the HSM by the hydraulic ram (located at the rear of the HSM).

p) Storage - After the DSC is inside the HSM, the hydraulic ram is released from the DSC. The transfer trailer is pulled away and the HSM front access cover plate is closed. The seismic retainer is installed in the rear opening. The rear access cover plate is also installed and secured in place. The DSC is now in storage within the HSM.

q) Retrieval- For retrieval, the cask is positioned as previously described and the hydraulic ram is used to push the DSC into the 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 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 canister to the HBR2 spent fuel pool if necessary.

TABLE 1.3-1

MAJOR SYSTEMS, SUBSYSTEMS AND COMPONENTS
OF THE H. B. ROBINSON ISFSI

Dry Shielded Canister

- Canister Basket
    - Square Cells
    - Spacer Disk
    - Support Rods
- Canister Body
- Shielded End Plugs
- Top Cover Steel Plate

Horizontal Storage Module

- Concrete Module
- Precast Outlet Shielding Blocks
- Dry Shielded Canister Support Assembly
- DSC Seismic Retaining Assembly
- Alignment System
- Front Access Cover Plate
- Rear Access Cover Plate
- Air Flow Penetrations

Trailer

- Cask Positioning Skid
- Skid Positioning System (Vertical and Horizontal)

Transfer Cask

- Cask Body
- Cask Lids
- Cask Drains
- Cask Extension Collar

See Figures 1.1-1, 1.2-1, and 1.3-1.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The prime contractor for design and analysis of the HBR ISFSI is NUTECH, Inc. of San Jose, California. The construction is the responsibility of the CP&L onsite construction organization. Carolina Power & Light Company Nuclear Engineering Department, along with the Modification Implementation Section, is responsible for material procurement as well as contract administration.

1.5 MATERIAL INCORPORATED BY REFERENCE

The Topical Report for the NUTECH Horizontal Modular Storage (NUHOMS) System for Irradiated Nuclear Fuel (NUH-001, Revision 1, ADV001.0100) submitted to the Nuclear Regulatory Commission by NUTECH Engineers, Inc. in November 1985 is hereby incorporated into this SAR by reference. The NUHOMS topical is referenced in Chapters 1, 3, 4, 5, 7, 8, and 11.

The General Electric Company Safety Analysis Report for the IF-300 Shipping Container (NEDO-10048-2) is also incorporated by reference. The IF-300 SAR is referenced in Chapters 3, 5, and 8.

REFERENCES: CHAPTER 1

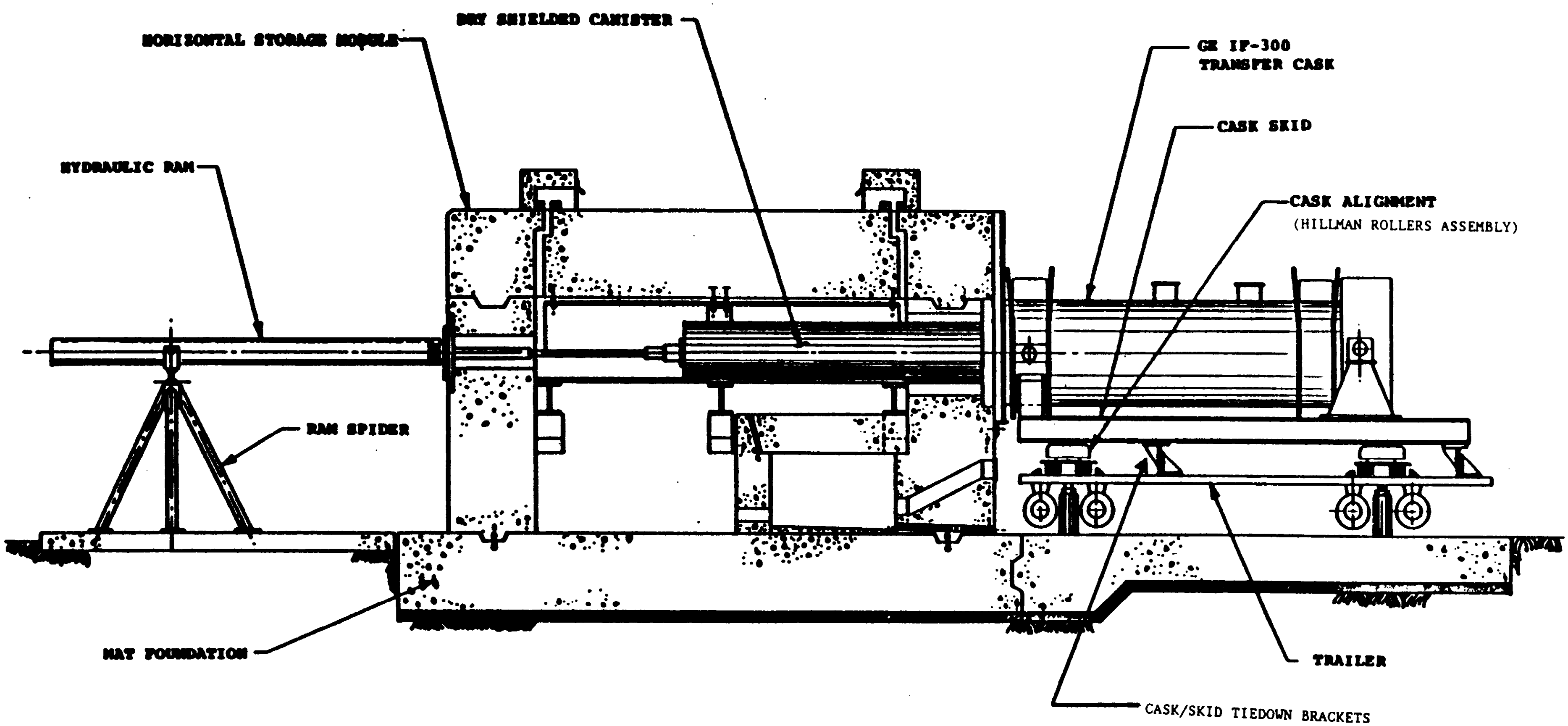
1.1 CP&L/DOE Licensed At-Reactor Dry Storage Demonstration Program, Cooperative Agreement No. DE-FC06-84RL10532, Amendment No. M005, October 1988.

1.2 NUTECH Engineers, Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-001, Revision 1, November 1985.

1.3 Docket Number 71-9001, Certificate of Compliance Number 9001 for General Electric Model No. IF-300 Shipping Container, Package Identification No. USA/9001/B( )F.

1.4 Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

1.5 Carolina Power and Light Company, H. B. Robinson Steam Electric Plant, Plant Operating Manual, "Refueling Instruction Spent Fuel Cask Handling Instructions for Loading and Shipping of Power Fuel," FHP-034.

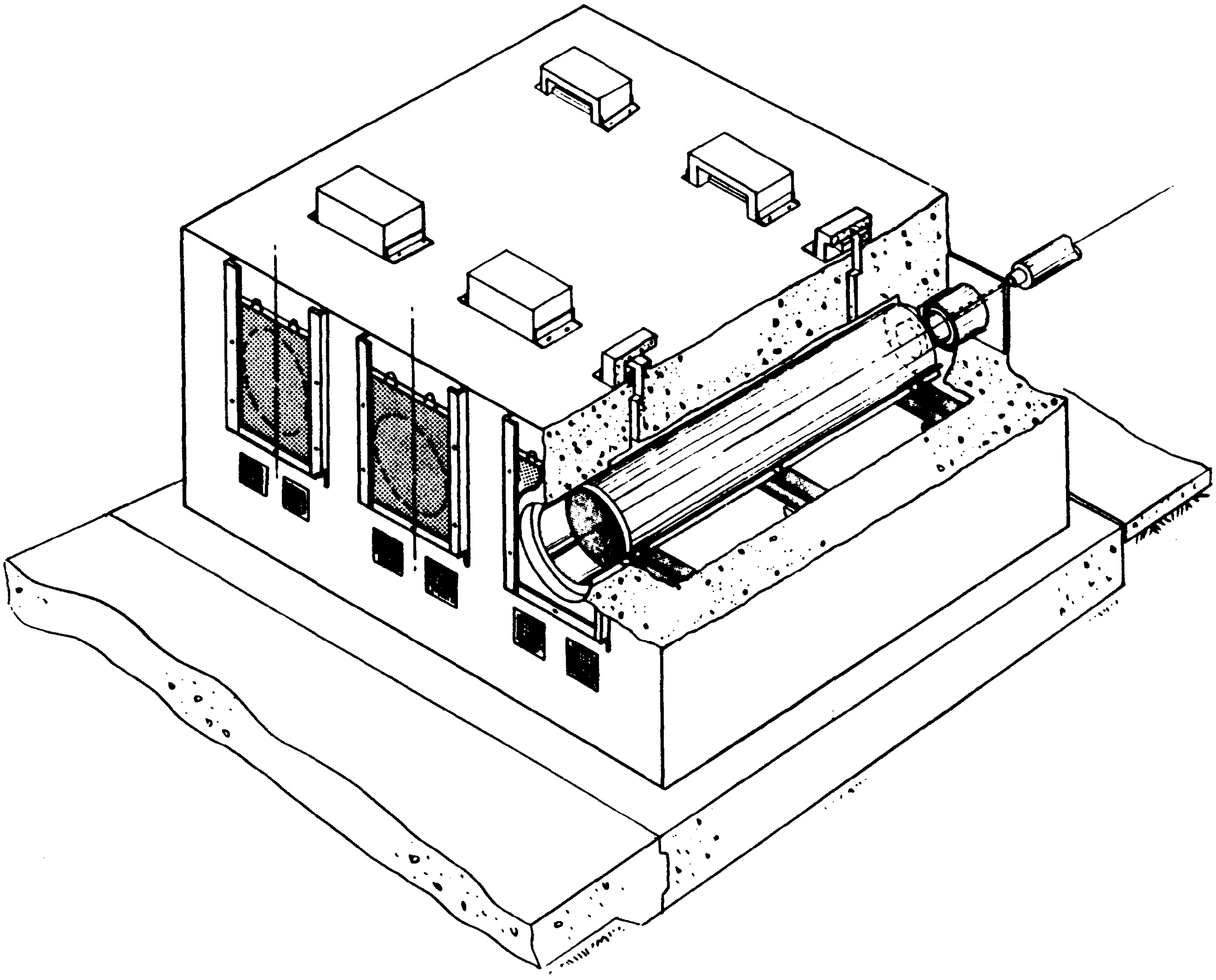


AMENDMENT NO. 5

H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

PRIMARY COMPONENTS OF THE
ISFSI
Figure 1.1-1

FIGURE WITHHHELD UNDER 10 CFR 2.390

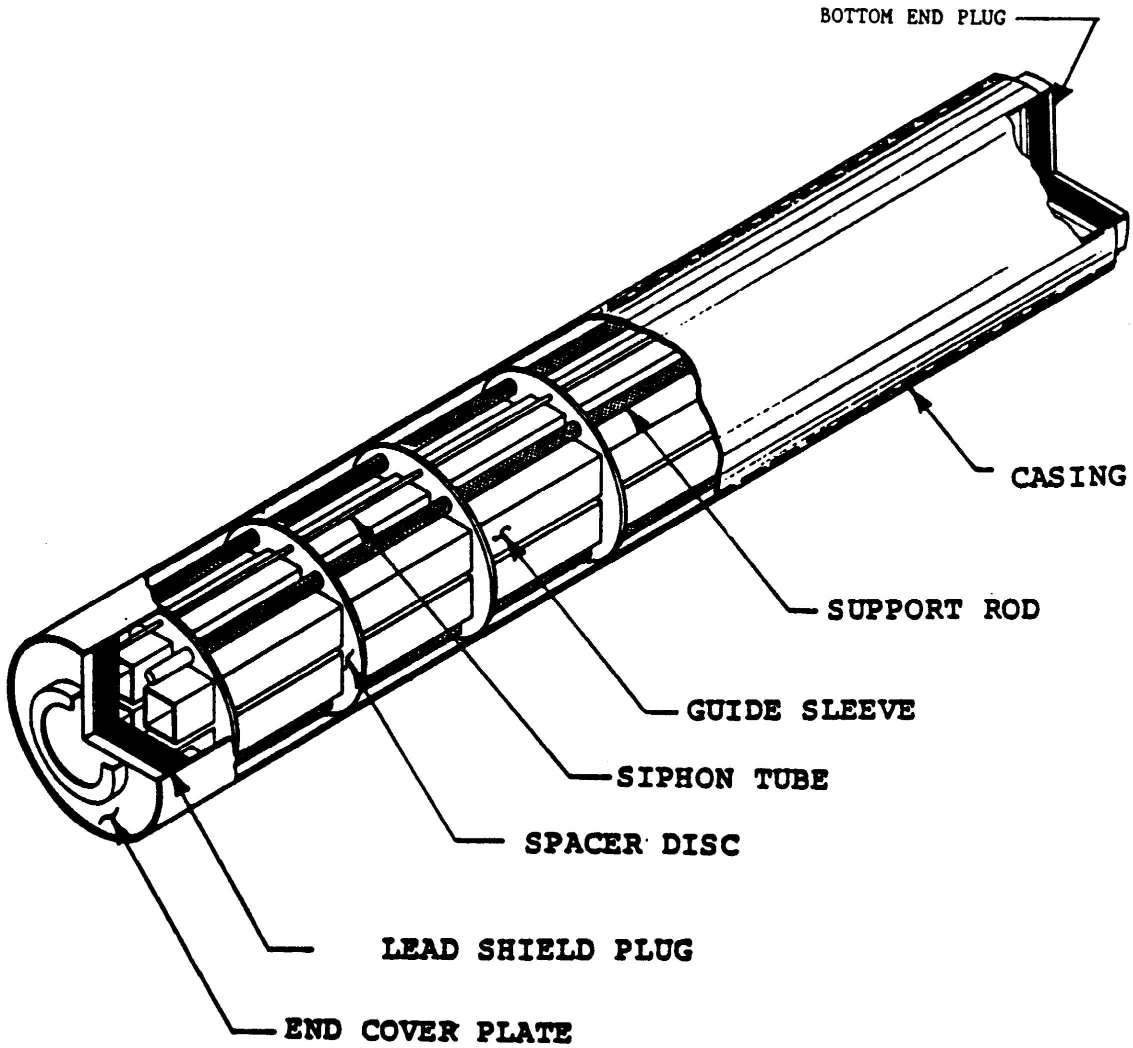


AMENDMENT 1

H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

HORIZONTAL STORAGE MODULE

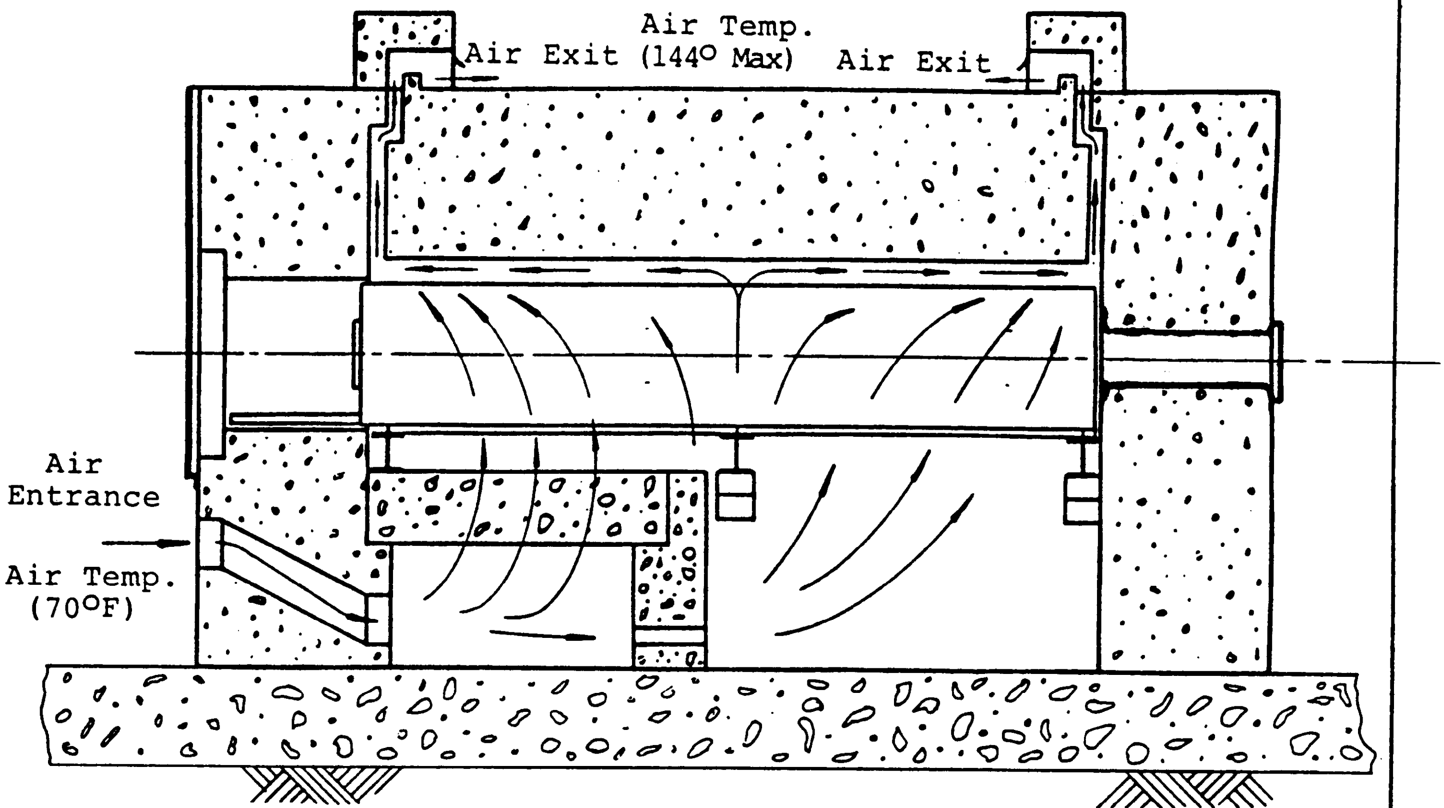
Figure 1.2-1



AMENDMENT NO. 5

H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

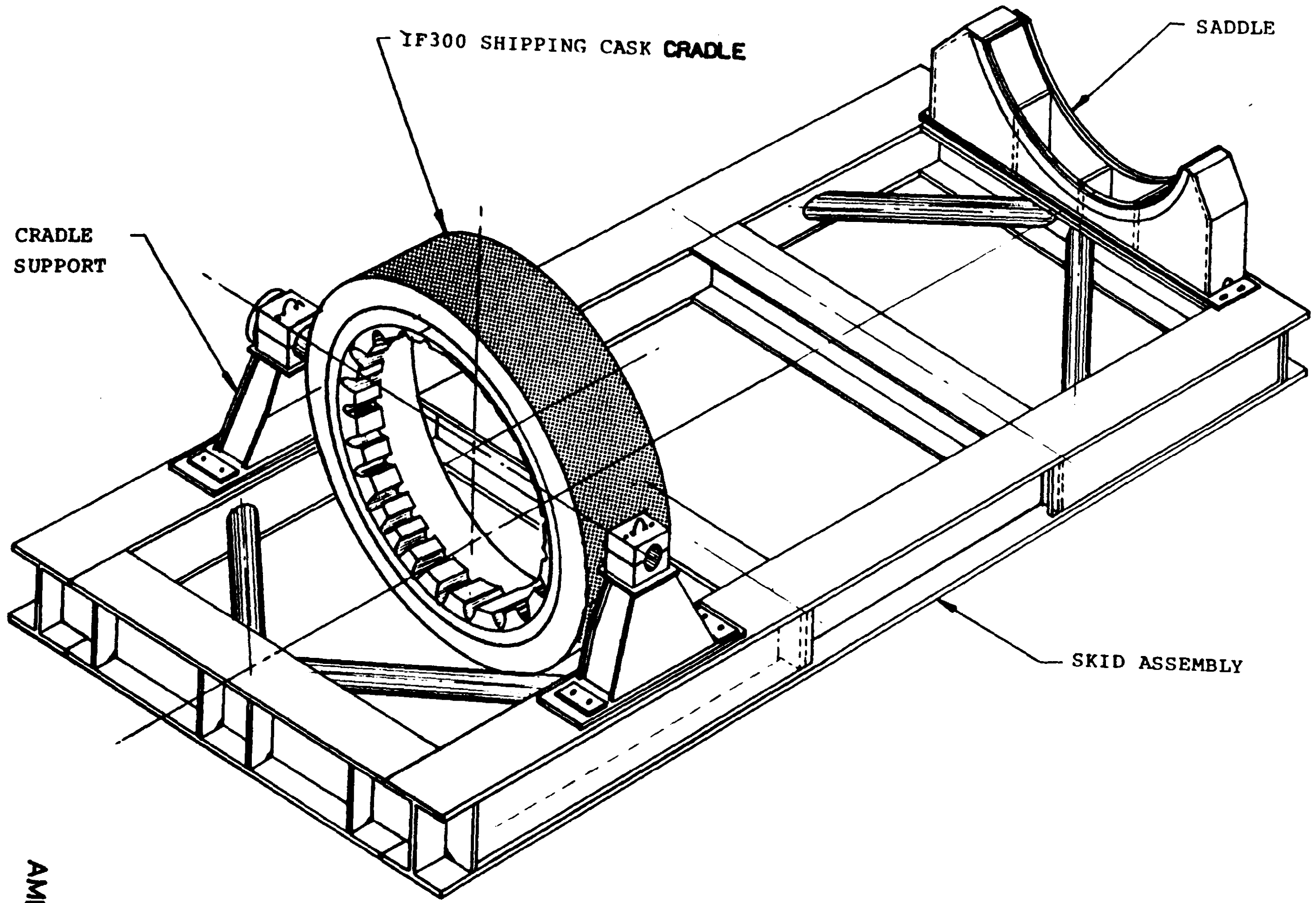
DRY SHIELDED CANISTER AND
INTERNAL BASKET
Figure 1.3-1



Amendment No. 1

H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

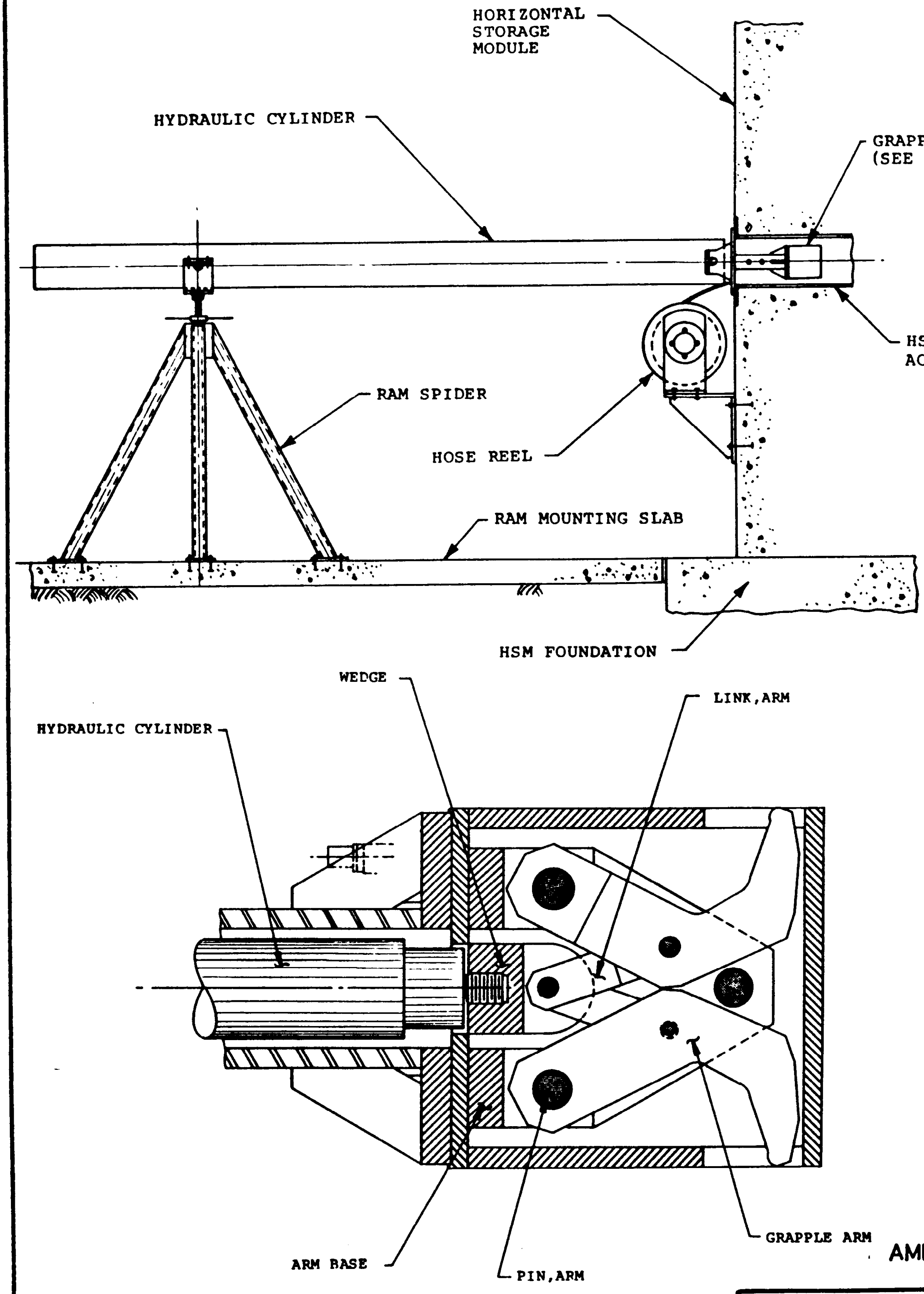
HSM AIR FLOW DIAGRAM
Figure 1.3-2



AMENDMENT 1

H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

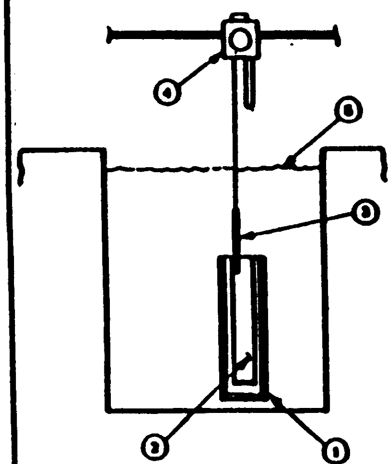
SKID FEATURES
Figure 1.3-3



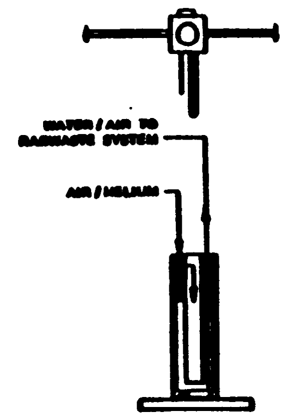
AMENDMENT 1

H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

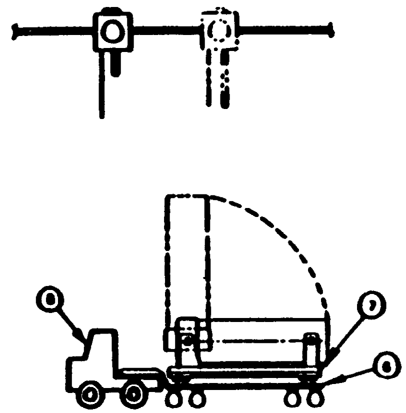
HYDRAULIC RAM
Figure 1.3-4



CASK LOADING

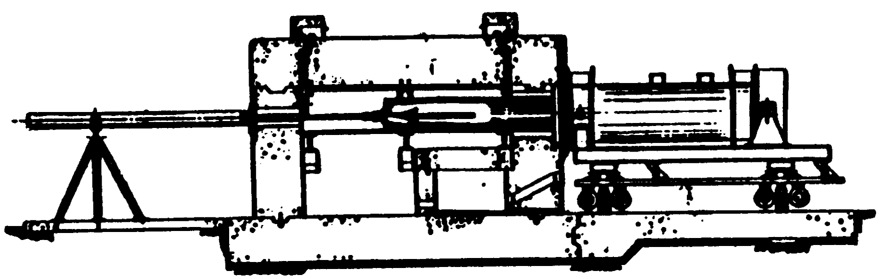


CASK / CANISTER DRYING AND SEALING

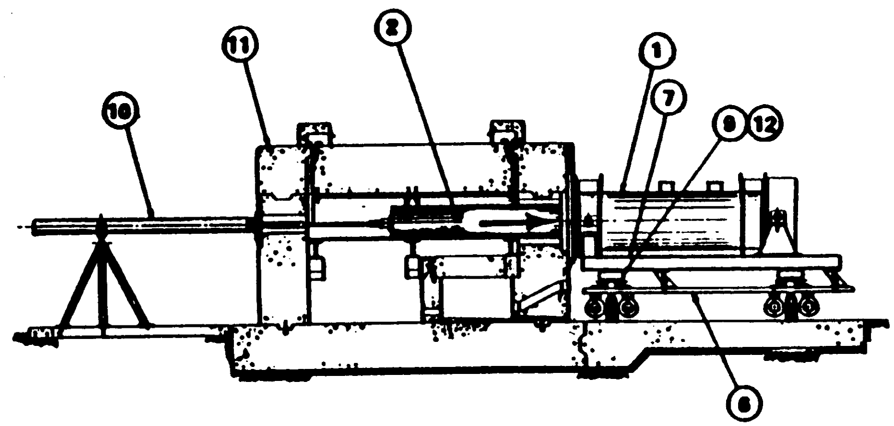


SKID LOADING

1. TRANSPORT CASK
2. DRY SHIELDED CANISTER
3. IRRADIATED FUEL ASSEMBLY
4. OVERHEAD CRANE
5. IRRADIATED FUEL STORAGE POOL
6. TRANSFER TRAILER
7. SKID
8. TOW VEHICLE
9. HYDRAULIC POSITIONERS
10. HYDRAULIC RAM
11. HORIZONTAL STORAGE MODULE
12. HORIZONTAL ROLLERS



MODULE LOADING



MODULE UNLOADING (RETRIEVAL)

AMENDMENT NO. 5

H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

PRIMARY CANISTER HANDLING OPERATIONS
Figure 1.3-5

CHAPTER 2

SITE CHARACTERISTICS

2.0 SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 SITE LOCATION

The Independent Spent Fuel Storage Installation (ISFSI) is located on the H. B. Robinson (HBR) Plant site. The site is located in northwest Darlington County, South Carolina, approximately 3 miles west-northwest of Hartsville, South Carolina; 24 miles northwest of Florence, South Carolina; 34 miles north-northeast of Sumter, South Carolina; and 54 miles east-northeast of Columbia, South Carolina. The North Carolina border is 28 miles north of the site and the Atlantic Ocean is about 88 miles southeast (Figure 2.1-1).

The site is on the southwest shore of Lake Robinson, a cooling impoundment of Black Creek. Coordinates are 34°24' 12" north latitude and 80°09' 30" west longitude. Universal Transverse Mercator (UTM) coordinates are 3,806,800 north and 577,500 east.

The site is located in the Coastal Plain physiographic province, approximately 15 miles southeast of the Piedmont province. Topography of the region (Figure 2.1.1-2 of Reference 2.1) is characterized by rolling sand hills interspersed with water courses.

2.1.2 SITE DESCRIPTION

A map of the site showing property lines (site boundary) is provided in Figure 2.1-2. The site currently covers approximately 2,500 acres of land and surrounds Lake Robinson, a 2280-acre impoundment. Carolina Power & Light (CP&L) Company owns all land below the 230 ft. contour surrounding the lake.

Since the original purchase of the land, CP&L sold lots on the lake's east shore. In addition to these lots, CP&L sold 4.4 acres of land to the New Market United Methodist Church for expansion of church facilities. In all cases, CP&L retained ownership of land below the 230 ft. contour. However, property owners were allowed to lease land between the 230 ft. contour and the lake (220 ft. contour normal operating level) and to construct piers, boathouses, and ramps. Provisions were also made for means of access to the lake by the general public. As a result, three privately owned recreational areas, one private sailboat club, and numerous access points throughout the lake allow for the use of the lake by the local population.

2.1.2.1 Other Activities Within the Site Boundary

Carolina Power & Light Company owns and operates a 2339 MWt nuclear generating plant (Unit 2) on the HBR Plant site. The ISFSI is located within the protected area for the nuclear unit. Unit 2 received an operating license from the U. S. Atomic Energy Commission in 1970 (Docket No. 50-261/License No. DPR-23).

Carolina Power & Light Company also owns and operates a 185 MWe fossil-fueled generating plant (Unit 1) adjacent to the nuclear unit (Unit 2). Unit 1 was placed in service in 1960, prior to the construction of Unit 2. Additionally, CP&L leases the Darlington IC Plant, a 572 MWe internal combustion oil plant, from General Electric. The Darlington Plant is located approximately 1 1/3 miles NNW of the Unit No. 2 plant on land originally included as part of the HBR Plant site property.

Approximately 1000 ft. south of the Unit 2 reactor, CP&L operates a nuclear visitors' center. As part of the center's facilities, there is a covered picnic pavilion, located on the southwest shore of Lake Robinson.

A spur track of a commercial railroad branches from a mainline at McBee, South Carolina, and passes 1600 ft. west of the plant. An extension of this spur enters the immediate plant area north of the plant and allows for delivery of coal to Unit 1.

The maximum speed for these tracks, with the exception of the southernmost tracks, is observed as 10 mph. The southernmost track is used as a "run through" track by larger locomotives, which are limited to a maximum speed of 5 mph. This track is also used as a storage area for coal cars,which are moved by a smaller locomotive which has a maximum speed of 10 mph on run through tracks. However, since the southernmost track is not used as a run through track by the higher speed locomotive, the maximum speed for locomotives on the track closest to the ISFSI is 5 mph.

Figure 1.1-2 shows two supporting facilities constructed on plant property. The Chemical Barrel Storage Building is located southwest of the HSMs and the oil storage facility is located near the path that was used to transport the DSCs from the fuel handling area to the HSMs.

The Waste Oil Storage System provides temporary storage facility of two classes of fluids: low-level radioactively contaminated lube oil and contaminated solvents. The facility is located on the area north of the Operations and Maintenance Building.

The Contaminated Waste Oil System is designed to store radioactively contaminated waste liquids. This system does not tie into the existing waste disposal systems.

The system consists of one: 75-gallon oil fill tank, strainer, 20 gpm fill/recirculation pump, 150 gpm transfer/ recirculation pump, 10,000-gallon storage tank for oil, 400-gallon storage tank for solvent, and 40 gpm transfer/recirculation pump for solvents.

The Waste Oil Storage System is designed for future disposal and is sized to store fluids until a permanent disposal system is implemented.

A 2'-6" dike (774 ft^2) surrounds the contaminated system. The total capacity of the dike is 14,474 gallons which is adequate to prevent the release of radioactive liquids in the event of a fire or a major spill.

The Chemical/Barrel Storage Facility is a 2,250 square foot, pre-engineered building located west of the Unit 2 protected area boundary and north of the Chemical/Bulk Receiving Warehouse. The facility was designed for storage of paints, lab chemicals to support the on-site chemistry lab, and provide storage of bottled gas.

The building has no floor drains. It does have a 4" high perimeter curb to contain any spills. The spills will be dry treated and cleaned up rather than drained to the Storm Sewer System.

Ventilation is accomplished by the power roof ventilators and filtered wall louvers.

Fire protection is provided by dry pipe sprinklers and portable fire extinguishers which are located in each area. Full-height, fire-rated Concrete Masonry Unit (CMU) walls divide the building. All equipment and fixtures are explosion proof suitable for Class I, Division 2, Group D atmospheres, per NFPA 70, NEC Article 500. Bottled gases are separated into groups of compatible gases by CMU partitions.

2.1.2.2 Boundaries for Establishing Effluent Release Limits

The exclusion zone is defined as the 1400 ft. radial area surrounding the plant. There are no residences or agricultural activities inside of the exclusion zone.

The 1400 ft. exclusion zone encompasses land and a portion of Lake Robinson. All land included in this area is owned in fee simple, without reservations, by CP&L. As such, all mineral rights are owned by CP&L. The public is allowed access to that part of Lake Robinson which is included in the exclusion zone.

The controlled area for the ISFSI is defined as being contained within the HBR Unit 2 exclusion area. The protective area is within the Plant's exclusion zone.

A spur track of the Seaboard Coastline Railroad branches from a mainline in McBee, South Carolina, and passes 1600 ft. west of the plant. An extension of this spur is operated by CP&L and intersects the exclusion zone north of the plant. Approximately three coal deliveries are made to Unit 1 each week. There is no passenger traffic on either the spur extension or the spur track.

An easement has been granted to Southern Bell Telephone and Telegraph Company for maintenance of telephone lines within the exclusion zone.

2.1.3 POPULATION DISTRIBUTION AND TRENDS

2.1.3.1 Population Within 10 Miles

The 1980 estimated resident population between zero and ten miles of the Robinson site is presented in Section 2.1 of Reference 2.1 as well as estimates of future population.

The area between zero and ten miles includes parts of four counties: Darlington, Chesterfield, Kershaw, and Lee, and the total 1980 resident population was approximately 31,000. The majority of these residents live in or around the city of Hartsville, 3 miles SSE (7631 city/11,529 suburban). One other small concentration of resident population was indicated for the city of McBee, 7 miles NW (774 city). Other population within the area is generally considered to be rural.

2.1.3.2 Population Between 10 and 50 Miles

The 1980 estimated resident population between ten and fifty miles is presented in Section 2.1 of Reference 2.1 along with estimates of future population.

The area is generally rural and is characterized by population concentrations in and around Florence, SC, 24 miles SE (30,062 city/9951 suburban); and Sumter, SC, 34 miles SSW (24,890 city/10,485 suburban). Cities with area populations over 10,000 include Laurenburg, NC, 44 miles ENE (11,480 city/536 suburban); Monroe, NC, 44 miles NNW (12,639 city/none suburban); Rockingham, NC, 42 miles NNE (8300 city/7203 suburban); and Lancaster, SC, 40 miles WNW (9603 city/6082 suburban).

2.1.3.3 Transient Population

The transient population within 10 miles of the HBR Plant site is composed of four major components: the industrial labor force, seasonal population variation, school population, and hospital/nursing home populations. These specific effects are discussed in Section 2.1 of Reference 2.1.

2.1.4 USES OF NEARBY LAND AND WATERS

Uses of land and water and respective populations in the 10-mile area surrounding the site is discussed in Section 2.1 of Reference 2.1 and Section 2.2 of this report.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 LOCATIONS AND ROUTES

The Independent Spent Fuel Storage Installation is located on the H. B. Robinson (HBR) Plant site. The HBR Plant is located in an area which is generally rural and lightly developed. Immediately west of the site and along the eastern shore of Lake Robinson, residential development has occurred, as well as the establishment of various public and private recreational areas.

The ISFSI is located within the controlled area of the 2339 MWt nuclear unit. A coal fired electric generating plant is located east and adjacent to the nuclear unit. The Darlington IC Plant is located approximately 1 1/3 miles NNW of the nuclear unit. Immediately north of the Darlington IC Plant is a gas pipeline. Other industrial development within 5 miles is limited to the areas in and surrounding Hartsville (3 miles SSE).

Agricultural development has occurred within the five-mile area, especially in areas north and west of the plant.

Principal transportation routes or facilities include highways, a railroad line (1600 ft. W), and a small airport (2 1/2 miles E).

There are no military bases within the five mile area.

2.2.2 DESCRIPTIONS

Residential development along the shores of Lake Robinson is conentrated along the shores of the lake and in the direction of Hartsville. Since 1960, numerous permanent and vacation homes have been built above the 230 ft. contour. Below the 230 ft. contour, property owners have constructed small private piers, boat docks, and ramps; access is provided by lease agreements between landowners and Carolina Power & Light (CP&L) Company.

Public recreational areas include Easterling's Landing, 1.7 miles NNE (a beach, picnic and paved boat launch area); Atkinson's Landing, 1.2 miles NNE (a beach and boat launch area); and Johnson's Landing, 4300 ft. E (a beach, paved boat launch, and boat gasoline facility). A small private sailboat club is also located on the lake, 2 miles NNE. All facilities are on the eastern lake shore. Several other areas provide recreational access to the lake, but this use is limited compared to that of other facilities.

East and adjacent to the nuclear unit (Unit 2), CP&L owns and operates a 185 MWe coal fired electric generating plant (Unit 1). Unit 1 was placed in service in 1960.

The Darlington IC Plant (1 1/3 miles NNW) is a 572 MWe internal combustion electric generating plant. The plant is owned by Westinghouse, Inc., but is leased and operated by CP&L.

South Carolina Electric & Gas Company transports natural gas via an underground pipeline (2 miles N). The pipeline transects Lake Robinson in an east/west direction. That part of the pipeline which crosses the discharge canal extends above ground.

Other industrial development within 10 miles of the plant is not extensive, and includes seven firms which employ more than 100 people. Principal products are paper products, textiles, fertilizer, seeds, bearings, and metal works (Reference 2.2). All of these firms are located in or near Hartsville (3 miles SSE).

Agricultural development has occurred within the five-mile area especially in areas north and west of the plant. Acreage to the north includes numerous peach orchards. Associated with the peach orchards is a fruit processing firm which processes and distributes local peaches, as well as other non-local produce.

Principal transportation routes include SC 151 (1/2 mile W), a highway running north and south, and numerous state maintained secondary roads.

A small private airport is located 2 1/2 miles east of the plant. Only small aircraft use the runway.

2.3 METEOROLOGY

2.3.1 REGIONAL CLIMATOLOGY

The H. B. Robinson site lies in the transition zone delineating the Piedmont and Coastal Plain of South Carolina. The climatology of South Carolina largely depends on elevation above sea level and distance from the Atlantic Ocean and the Appalachian Mountain chain. At an elevation of about 225 feet above mean sea level and a distance of about 160 miles from the Appalachian Mountains, the site has a temperate climate.

Long summers are prevalent with warm weather usually lasting from May into September. In summer, the Bermuda high is the greatest single weather factor influencing the area. This permanent high more or less blocks the entry of cold fronts so that many stall before reaching central South Carolina. Also, the southwestern flow around the offshore Bermuda high pressure supplies moisture for the many summer thunderstorms. There are relatively few breaks in the heat during the midsummer. The typical summer has about six days with temperatures of 100^{o}F or more. Thundershower activity usually shows a decided increase during June, decreasing about the first of September. Summer is the rainiest season of the year contributing about 33 percent of the annual total. The summer rains are largely in the form of local thundershowers. About once or twice a year, affects of a passing tropical storm are felt in the form of strong winds and heavy rains. The incidence of these storms is greatest in September, although they represent a possible threat from midsummer to late fall. Damage from tropical storms is usually minor in the HBR area.

Fall is the most pleasant time of the year. Rainfall during the late fall is at an annual minimum, while the sunshine is at a relative maximum. About 20 percent of the annual rainfall is recorded during the fall. Winters are mild with the cold weather usually lasting from late November to mid-March. However, only about one-third of the days in this period have minimum temperatures below freezing. The winter weather around the HBR site is largely made up of polar air outbreaks that reach this area in a much modified form. On rare occasions in winter, Arctic air masses push southward as far as central South Carolina and cause some of the coldest temperatures. Disruption of activities from snowfall is unusual; in fact, more than three days of sustained snow cover is rare. A day or more with snowfall is probable during nine out of eleven winters. A day with more than one inch of snowfall is likely to occur in one out of five winters. The average winter has five days with temperatures of 20^{o}F or below. Temperatures below 10^{o}F are rare; occurrences average about one per four winters. There are two to five cold waves during the winter. The winter rainfall is about 22 percent of the annual total.

Spring is the most changeable season of the year. The temperature varies from an occasional cold snap in March to generally warm and pleasant in May. While tornadoes are infrequent, they occur most often in the spring. Hailstorms are not frequent, with the annual incidence at a maximum in spring and early summer. The spring rainfall represents 25 percent of the annual total.

The average date of the last spring freeze is March 30, and the average date of the first fall freeze is November 3, or a growing period of 218 days.

Temperatures of 32°F or below have occurred as late as April 21 and as early as October 4. The average date of the last spring occurrence of 24°F is February 25, while the average date of the first fall occurrence of 24°F is November 26. More than 1,530 hours below 45°F per winter are probable once in ten years, while more than 930 hours may occur nine out of ten winters.

Specific details on regional temperatures, precipitation, winds, and other meteorological conditions are documented in Section 2.3 of the HBR2 Updated Final Safety Analysis Report (Reference 2.1).

2.3.2 LOCAL METEOROLOGY

2.3.2.1 Data Sources

The local meteorology is based upon HBR onsite data collected from January 1, 1976 through December 31, 1981, and offsite data from Charlotte, NC; Greensboro, NC; Raleigh-Durham, NC; Florence, SC; and Columbia, SC. Normal and mean data are based on the 1941-1970 recording period.

The onsite lower level (12.5 m) mean wind speed based on 1976-1981 data is 6.2 mph. The maximum site area one-minute average wind of 60 mph from the west was recorded in March 1954. As would be expected, the intensities of wind and precipitation produced by hurricanes at the plant site are generally no greater than those produced by severe thunderstorms in the area.

The site area diurnal temperature range spans from about 20°F in the winter and summer seasons to around 25°F in the transitional autumn and spring months. The lowest temperature recorded was -5°F in February 1899 in Charlotte and the highest recorded temperature was 107°F at Columbia in June 1954.

The onsite average dewpoint of 48.0°F compares well to the 52°F average dewpoint observed at Columbia, SC. Onsite winter and summer dewpoint temperatures are slightly lower.

Precipitation is rather uniformly distributed on an annual basis in the site region. Climatologically, July has a tendency to be the wettest month, October the driest, but the variance is small such that the region does not possess a "wet" and "dry" season. The maximum 24-hour precipitation total is 4.76 in. on September 4, 1979, with the passage of the remains of Hurricane David.

For the period 1948-1980, heavy fog (visibility $\leq$ 1/4 mile) occurred at Columbia on an average of 27 days per year, with the fall and winter months showing the greater number of days of nearly 3 per month. The most common type of fog occurring in the HBR area is ground fog as a result of nighttime radiational cooling. Ordinarily, ground fog occurs more frequently in the early morning hours near sunrise when the daily minimum surface temperature is reached. It is usually shallow and disappears shortly after sunrise.

Temporal variations of frequencies within the individual Pasquill stability classes are small. Almost 56 percent of all hours fall into either neutral (D) or slightly stable (E) stability categories. Nearly 9 percent of all hours fall into the extremely stable (G) stability category. Extremely unstable (A), moderately unstable (B), and slightly unstable (C) stability categories combined occur only approximately 35 percent of the total hours.

2.3.2.2 Topography

The topography of the site can be seen in Figure 2.1.1-2 of Reference 2.1.

2.3.3 ONSITE METEOROLOGICAL MEASUREMENTS PROGRAM

2.3.3.1 Onsite Operational Program

Collections of HBR onsite meteorological data began in April, 1974. A guyed, open latticed tower supports the lower and upper levels of instrumentation. Wind direction, wind speed, and wind variance (sigma theta) are recorded at both levels. Ambient temperature and relative humidity are measured at the lower level. The differential temperature between the upper and lower levels is measured by twin, redundant delta temperature systems operating simultaneously. Solar radiation, barometric pressure, and precipitation are collected near ground level. The wind sensors are mounted on 12-foot booms oriented perpendicular to the general NE-SW prevailing wind flow to minimize tower shadow effects. The temperature probes and relative humidity sensor are housed in aspirated shields mounted on 8-foot booms.

The meteorological tower is located about 0.53 miles north of the HBR Unit 2 (HBR2) Containment Building. The base of the tower is at the plant grade level of about 225 feet above mean sea level.

A data logger at the HBR 2 meteorological site acquires data from the sensors and converts the signals to engineering units. Fifteen-minute averaged data is stored within the data logger and the converted signals can be acquired, via modem, by plant computer systems and offsite contract meteorologist, as well as other systems or locations, as required. The data logger also sends specific converted signals to a recorder as analog input signals. The data logger information can also be displayed locally using a computer.

The HBR 2 ERFIS computer system accesses the data logging system every 15-minutes to acquire the latest 15-minute averaged data. This information is stored in the ERFIS System and displayed on demand from any ERFIS terminal (such as in the Control Room).

The recorder also provides local display of the meteorological parameters sent to it from the data logger. The trend charts are changed as required. They are used as backup data to provide checks on the system and provide consistency of data. A communication port allows remote retrieval of data via standard telephone lines to an approved contractor.

Available computer outputs from the data collection system include:

a) Monthly data summaries listing maximum temperature, minimum temperature, average temperature, barometric pressure, precipitation, solar radiation, and lower level dewpoint temperature as a daily average and monthly average.

b) Hourly averages of precipitation, barometric pressure, ambient temperature, differential temperature, lower level dewpoint, upper and lower level wind direction and wind speed, upper and lower level wind direction variance (sigma theta), Pasquill stability classes (as outlined in Regulatory Guide (RG) 1.23) computed from the average of the two delta temperature systems, and accumulated solar radiation (langleys/minute).

c) The 15-minute averages of both upper level and lower level wind direction, speed, and sigma theta, barometric pressure, and accumulated solar radiation.

d) Joint wind frequency distributions by direction (as outlined in RG 1.23) for both upper and lower levels, showing average wind speeds and number of unrecovered data hours.

An onsite maintenance and calibration program was initiated in January 1976. Regulatory Guide 1.23 data recovery requirements are met by performing scheduled calibrations.

Further details about the onsite meteorological measurement program can be found in Section 2.3 of Reference 2.1.

2.3.3.2 Onsite Data

Westinghouse System onsite joint wind percentage frequency distributions (compiled per RG 1.23) for both upper and lower sensor elevations for the period January 1976 through December 1981 is presented in Section 2.3 of Reference 2.1. Average onsite windspeeds for the total six-year period at the lower and upper levels are 5.2 mph and 9.6 mph, respectively.

2.3.4 DIFFUSION ESTIMATES

The short-term (accident) and long-term (routine) diffusion estimates are detailed in Section 2.3.4 of Reference 2.1. The diffusion estimates for HBR2 are also applicable for the ISFSI.

2.4 SURFACE HYDROLOGY

2.4.1 HYDROLOGIC DESCRIPTION

The main surface water feature in the vicinity of the site is Lake Robinson. The primary purpose for which Lake Robinson was constructed by Carolina Power & Light Company is industrial cooling. Secondary uses such as recreation are not restricted by CP&L as long as the primary function is not impaired by those activities. Lake Robinson is about 4000 ft. wide at the plant site and about 7 1/2 miles long at its maximum water elevation of 222 ft. MSL.

Downstream from Lake Robinson, on the northern edge of Hartsville, is another smaller impoundment of Black Creek called Prestwood Lake. It serves the Sonoco Products Company, located adjacent to the lake. Sonoco manufactures paper products.

Cooling water canals and reservoirs for HBR use are discussed in Section 2.4.4 of Reference 2.1.

2.4.1.1 Site and Facilities

Flooding at the ISFSI will not occur. Facility grade is above the maximum lake level which can be maintained by the dam and appurtenant structures.

Figure 2.1.1-2 of Reference 2.1 is a topographic map of the site. The ISFSI will not affect any natural drainage features.

2.4.1.2 Hydrosphere

Water inflow to and outflow from Lake Robinson has been measured and recorded since 1959 and 1960, respectively. Values for this data and a topographic map of the Lake Robinson area prior to impoundment of Black Creek are provided in Section 2.4.1 of the Reference 2.1.

2.4.2 FLOODS

Flooding of the HBR site, on which the ISFSI is located, is addressed in Section 2.4.4 of the Reference 2.1. Flooding at the ISFSI will not occur because the facility grade is above the maximum lake level which can be maintained by the dam and appurtenant structures. Two different peak flows were calculated using the design unit hydrograph for the drainage area above the Lake Robinson Dam. Using these two peak flows, the lake level would not exceed 222 ft. during high flow conditions. Grade elevation at the location of the ISFSI is approximately 235 ft.

2.4.3 PROBABLE MAXIMUM FLOOD ON STREAMS AND RIVERS

Information on the probable maximum flood and probable maximum precipitation at the site is presented in Section 2.4.4 of Reference 2.1. Flooding at the ISFSI will not occur because the facility grade is above the maximum water level at high flow conditions.

2.4.4 POTENTIAL DAM FAILURES

The dams for both Lake Robinson and Prestwood Lake are downstream of the ISFSI; therefore, a dam failure would not flood the site, and since the ISFSI requires no water from the lakes for operation, a potential dam failure would not affect operation of the ISFSI.

2.4.5 PROBABLE MAXIMUM SURGE AND SEICHE FLOODING

The meteorology of the area is discussed in Section 2.3. Lake Robinson is a long, narrow lake and meteorological and astronomical events do not cause significant effects. The maximum expected lake level during high flow conditions is 222 ft. The ISFSI is approximately 1400 ft. away from the lake and the grade elevation is 235 ft. Therefore, surge and seiche flooding at the ISFSI will not occur.

2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

The ISFSI is not adjacent to coastal areas; therefore, tsunami flooding is not a credible event.

2.4.7 ICE FLOODING

Due to the regional meteorological conditions (see Section 2.3) and the characteristics of the area water bodies (see Sections 2.4.1 and 2.4.2), ice flooding will not occur at the ISFSI.

2.4.8 FLOODING PROTECTION REQUIREMENTS

Flooding of the ISFSI is not a credible event; therefore, no flood protection requirements are necessary.

2.4.9 ENVIRONMENTAL ACCEPTANCE OF EFFLUENTS

The only liquid used for the ISFSI is during preparation of the spent fuel assemblies for loading into the GE IF-300 shipping canister is water. No liquids are used during the actual operation of the ISFSI.

Dispersion, dilution, and travel times of accidental releases of liquid effluents in surface water from the Robinson Plant, discussed in Section 2.4.6 of Reference 2.1, will remain unaffected by the ISFSI.

2.5 SUBSURFACE HYDROLOGY

The Independent Spent Fuel Storage Installation 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 Robinson site.

2.5.1 GROUNDWATER USAGE

Municipal and industrial sources of potable water within a 20 mile radius of the Robinson site are obtained from groundwater sources, primarily from artesian wells. All domestic water usage in the vicinity of the plant is artesian in origin. Details of the groundwater usage is provided in Section 2.4.7 of Reference 2.1.

2.5.2 SITE CHARACTERISTICS

The drainage area of the Black Creek above the dam site is underlain and bounded by the Middendorf Formation, a sequence of unconsolidated and semi-consolidated cross-bedded, micaceous, feldspathic quartz sand and gravel beds. A detailed description of the site groundwater characteristics is provided in Section 2.4.7 of Reference 2.1.

2.5.3 CONTAINMENT TRANSPORT ANALYSIS

Section 2.4.7 of Reference 2.1 describes the test program used to evaluate the groundwater conditions in the vicinity of the Robinson site. This program's area of investigation included the H. B. Robinson plant site, the ash disposal pond, and the area in between. Its objective was to establish the velocity of groundwater movement. The shortest route of groundwater travel from the ash pond to the lake is a distance of 400 ft. The time required for the groundwater to travel this distance is approximately 45 days. For the longer, rather circuitous route of 3550 ft. from the ash pond to the discharge canal, the ground water travel time is approximately 9 years and 10 months. The ISFSI, however, stores spent fuel in a dry condition and does not utilize any liquids for operation. Therefore, the
groundwater system will not be affected.

2.6 GEOLOGY AND SEISMOLOGY

Specific soil testing has been performed at the designated location for the ISFSI. The data obtained from this testing is utilized in the foundation design for the ISFSI. As part of the foundation analysis/design, the subject of soil liquefaction is addressed. The following sections discuss the Robinson site geology and seismology. The foundation analysis is presented in Section 8.3 of this report.

2.6.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

The Robinson site is located in the Coastal Plain physiographic province approximately 15 miles southeast of the Piedmont province. In South Carolina, the Coastal Plain is composed of largely unconsolidated sediments which overlie a slightly sloping surface of crystalline rock. The Coastal Plain sediments in the area of the site were formed at the same time as the Tuscaloosa Formation, but locally are known as the Middendorf Formation.

The surficial materials at the Robinson site are recent sands or soils developed from the Middendorf.

Because of the high quartz content of the sands and the climatic environment, the surficial soils may not weather sufficiently to differ considerably from the parent material. Thus, it is nearly impossible to distinguish the recent alluvial soils from the parent Middendorf sand since both the alluvial and weathered soils are derived from the Middendorf. Only their manner of placement would be different. From an engineering standpoint, the difference is minor.

The subsurface materials encountered in the test holes drilled at the site are completely consistent with recent alluvium and Middendorf Formations encountered throughout the vicinity. Discontinuities within the strata are sedimentary and no structural deformation is apparent in the Middendorf Formation in the site area.

The Middendorf is about 400 ft. thick and overlies an eroded, slightly sloping surface of Piedmont crystallines that may be somewhat weathered near the surface.

Triassic basins are known in the area; however, it is believed that the likelihood of a Triassic basin at the site is quite small. The basement rock at the site is considered to be Piedmont crystalline since the results of the seismic surveys indicate a high velocity material at a depth consistent with the depth of Piedmont crystallines encountered in wells in the area.

In general, the upper alluvial sands and gravels are moderately compact. Layers of compressible material occur in the upper 30 to 50 ft. Because of the quantity of fines in the sand and gravel, it could not be considered free-draining material. The underlying Middendorf contains generally compact relatively incompressible sands and firm to hard clayey soils. Several strata of cemented sandstone were encountered in the borings at depths of roughly 90 to 100 ft

From a geological standpoint, the Middendorf is considered to be an unconsolidated formation. From an engineering point of view, however, the materials are firm and compact and would provide good foundation support for the proposed construction. The materials range in texture from a hard or compact soil to a soft rock.

Further details of the regional geology are discussed in Section 2.5.1 of Reference 2.1.

2.6.2 VIBRATORY GROUND MOTION

A seismological study for the Robinson site has been performed to determine the design and hypothetical earthquakes for the site and the ground motion spectra associated with them. Details are discussed in Section 2.5.2 of Reference 2.1.

2.6.2.1 Earthquake History

The largest earthquake in this region occurred at Charleston, South Carolina in August 1886. This shock had an intensity of about Modified Mercalli IX at the epicenter and it is estimated that this shock had a magnitude of 6 1/2 to 7 with epicentral acceleration of 0.25g to 0.30g. However, damage was confined to a relatively small area.

Only one earthquake of intensity V or greater has ever been recorded within 50 miles of the Robinson site. This shock occurred on October 26, 1959, near McBee, Chesterfield County, South Carolina, with an intensity of Modified Mercalli VI. The epicenter was located about 15 miles from the site. The estimated intensity at the site was about V.

Except for a trend of epicenters paralleling the Blue Ridge, there is no apparent trend of other epicenters in the region. Most of the smaller historical shocks were reported in scattered population centers. Further details of area seismicity are described in Section 2.5.2 of Reference 2.1.

The site is located in Zone 1 of the U. S. Coast and Geodetic Survey Map and Equal Seismic Probability. Zone 1 is characterized as a zone of light earthquake activity which would result in minor damage. Therefore, on an historical basis, it would appear that the site will not experience damaging earthquake motion during the life of the planned facility.

2.6.2.2 Earthquake Probabilities

On the basis of historical data, it is expected that the site area could experience a shock on the order of the 1959 McBee shock once during the life of the facility. This shock could be as far distant as in 1959 (15 miles), or perhaps closer.

2.6.2.3 Design Earthquake

Comparison between the Robinson site and certain areas in California indicate a similarity in the depth and type of overburden material. For this reason, Dr. G. W. Housner of the California Institute of Technology recommended the use of his average California response spectra to define the earthquakes (see

Appendix 2.5E of Reference 2.1). Dr. Housner's specific recommendations are: design for maximum horizontal ground acceleration of 0.1g with a vertical component of 2/3 of the horizontal acceleration, and hypothetical earthquake maximum horizontal ground acceleration of 0.20g.

The safe shutdown earthquake and operating basis earthquake for HBR2 are 0.2g and 0.1g, respectively. It is important to note that even if an earthquake comparable to the Charleston shock were to occur 35 miles from the site, the ground acceleration would not exceed 0.2g. These values are identical to those reported in Reference 2.1.

2.6.3 SURFACE FAULTING

A study of the possibility of the existence of faults in the area indicated that no active faulting was apparent. The sediments underlying the site are quite thick and apparently undisturbed. The surface of the buried crystallines is an ancient eroded one, and active faults are unknown in the vicinity of the site.

No faulting is apparent in the unconsolidated sediments of the Coastal Plain. The underlying basement rocks are effectively masked by more than 400 ft. of sediments at the site and cannot be directly observed below the Fall Zone. However, faulting in the basement complex is known from exposures above the Fall Zone and cores from scatterd borings drilled through the Coastal Plain sediments.

2.6.4 STABILITY OF SUBSURFACE MATERIALS

2.6.4.1 Geologic Features

The test boring program, refraction surveys, and laboratory tests, when combined, present the following picture of the subsurface and geologic site conditions: The piedmont crystalline basement rock at the site is overlain with approximately 460 ft. of unconsolidated coastal plain sediment. These sediments are comprised of about 30 ft. of surface alluvium over 430 ft. of the Middendorf formation. The Middendorf is made up of sands, silty and sandy clay, sandstone, and siltstone. Compressional wave velocities are 17,500 fps in the basement rock, 7,200 fps in the Middendorf, and 1,500 fps in the top 30 ft. of alluvium.

2.6.4.2 Properties of Subsurface Materials

In order to evaluate changes in the properties of the soils which underly HBR2, a series of static and dynamic triaxial compression tests and confined compression tests were performed at the time HBR2 was designed.

a) Dynamic Confined Compression Tests: The test results indicated that the compressibility characteristics of the silty and clayey soils encountered at the site are not appreciably affected by dynamic loading. The settlement of the sandy soils increased somewhat when the sample was subjected to an oscillating load of short duration.

b) Dynamic Triaxial Compression Tests: The test results indicated that the available shearing strength of the soils at the site are generally slightly reduced when subjected to dynamic influence. The upper moderately firm clayey silt is not appreciably affected, but the stiff, silty clay, and dense sandy soils apparently experience some minor strength reduction.

2.6.4.3 ISFSI Foundation

A specific soil testing and foundation evaluations has been performed at the designated location for the ISFSI. The data obtained from this testing is being utilized in the foundation design for the ISFSI. The subject of soil liquefaction has been addressed as follows:

The analysis of the subsurface profile at the site of the ISFSI confirms previous findings made from other subsurface investigations carried out on the HBR2 site. The analysis of the data obtained from the ISFSI site borings used the procedures recommended by Seed, Idriss, and Arango for estimating liquefaction susceptibility using standard penetration resistances (N-values).

Data collected from the borings indicates potential liquefaction; however, this sample represents the soil conditions at a depth of approximately 100 feet. This loose soil zone is less than 5 feet thick, is located beneath approximately 23 feet of the very hard silty clay of the Middendorf Formation and is surrounded by approximately 25 feet of firm to dense sand. Hard silty clay continues below the sand stratum at a depth of approximately 112 feet.

Based on a review of the previous explorations performed at the H. B. Robinson site, there is no indication of a similar zone of loose sand located at the depths encountered by this boring. The 5-foot zone of loose sand is believed to be an isolated pocket or lens not representative of an area-wide layer in the Middendorf Formation. Based on this analysis, the ISFSI site profile is considered to have a very low to no likelihood of liquefaction during a design basis earthquake.

Because liquefaction of the loose sand pocket at 100 feet cannot be completely discounted, the effects of liquefaction were examined. In the case of a confined layer of potentially liquefiable saturated sands at large depths, as in this boring for the ISFSI, the paramount problem is not one of bearing capacity or landslide susceptibility as in surficial sands, but one of surface settlement following complete liquefaction. It can be expected that the hard silty clay and dense sands surrounding the loose sand layer would bridge any effects from the liquefaction of the loose sands and no surface settlements would occur.

2.6.5 SLOPE STABILITY

The failure of any slopes at the Robinson site will not adversely effect the ISFSI. A discussion of slope stability of the earth dam and appurtenances at the Robinson site is provided in Section 2.5.5 of Reference 2.1.

2.7 SUMMARY OF SITE CONDITIONS AFFECTING CONSTRUCTION AND OPERATING REQUIREMENTS

The HBR ISFSI is a totally passive installation designed by analysis to provide shielding and containment of irradiated fuel. The ISFSI is located within the protected area of the nuclear unit.

There are no residences or agricultural activities inside of the 1400 ft. radial area exclusion zone. The plant site has a temporate climate with the ISFSI facility grade above the maximum lake level which can be maintained by the dam and appurtenant structures.

The ISFSI will not affect any natural drainage features. A dam failure will also have no affect on the ISFSI as the dams for Lake Robinson and Lake Prestwood are downstream of the ISFSI.

On the basis of historical data, it is expected that the plant site area could experience a shock on the order of the 1959 McBee shock once during the life of the ISFSI facility. The design basis for the ISFSI is consistent with the HBR2 earthquake design basis of 0.2g horizontal acceleration.

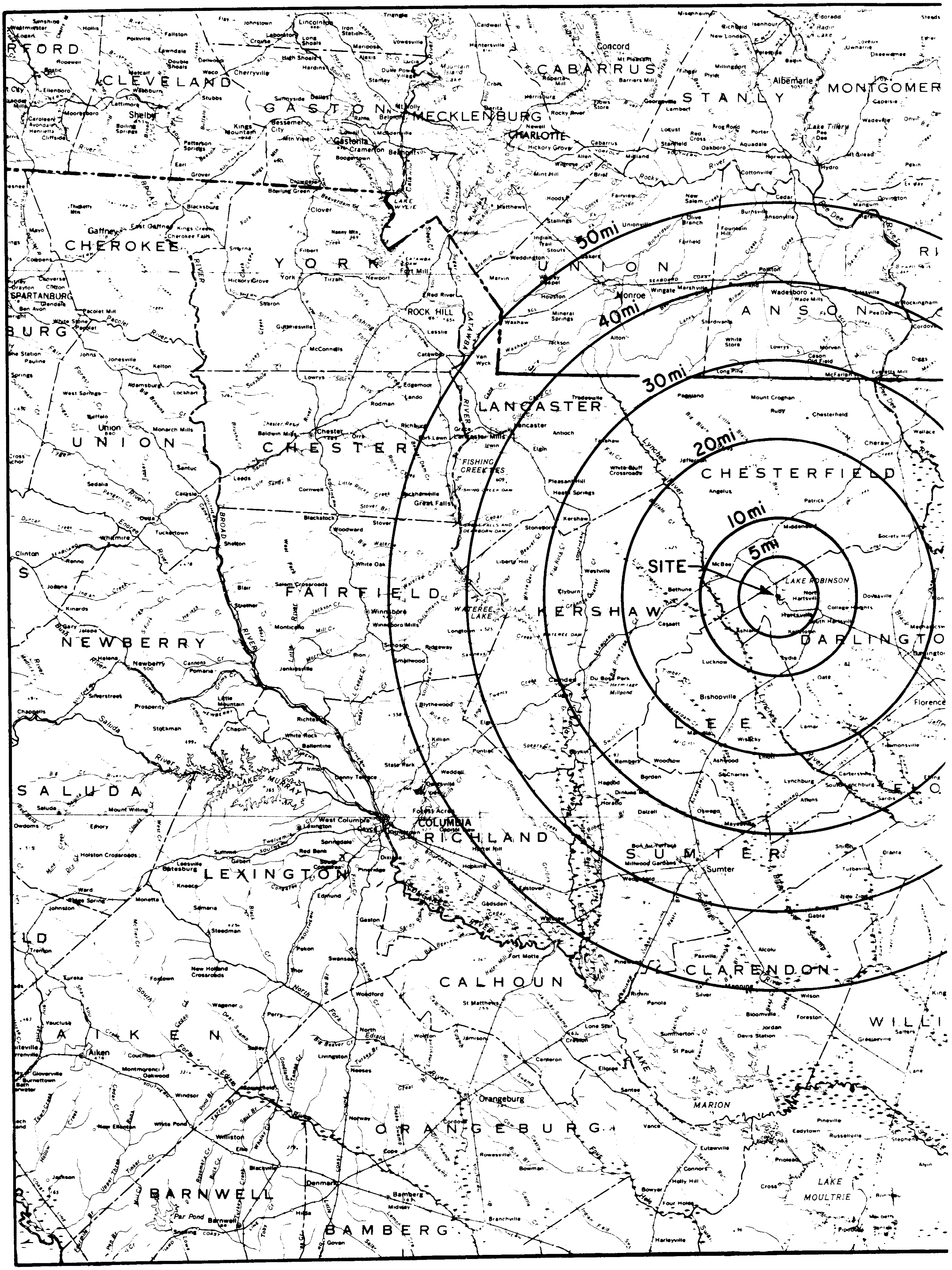
A study of the possibility of the existence of faults in the area indicate that no active faulting was apparent. The sediments underlying the site are quite thick and apparently undisturbed. The surface of the buried crystallines is an ancient eroded one and active faulting is unknown in the vicinity of the site.

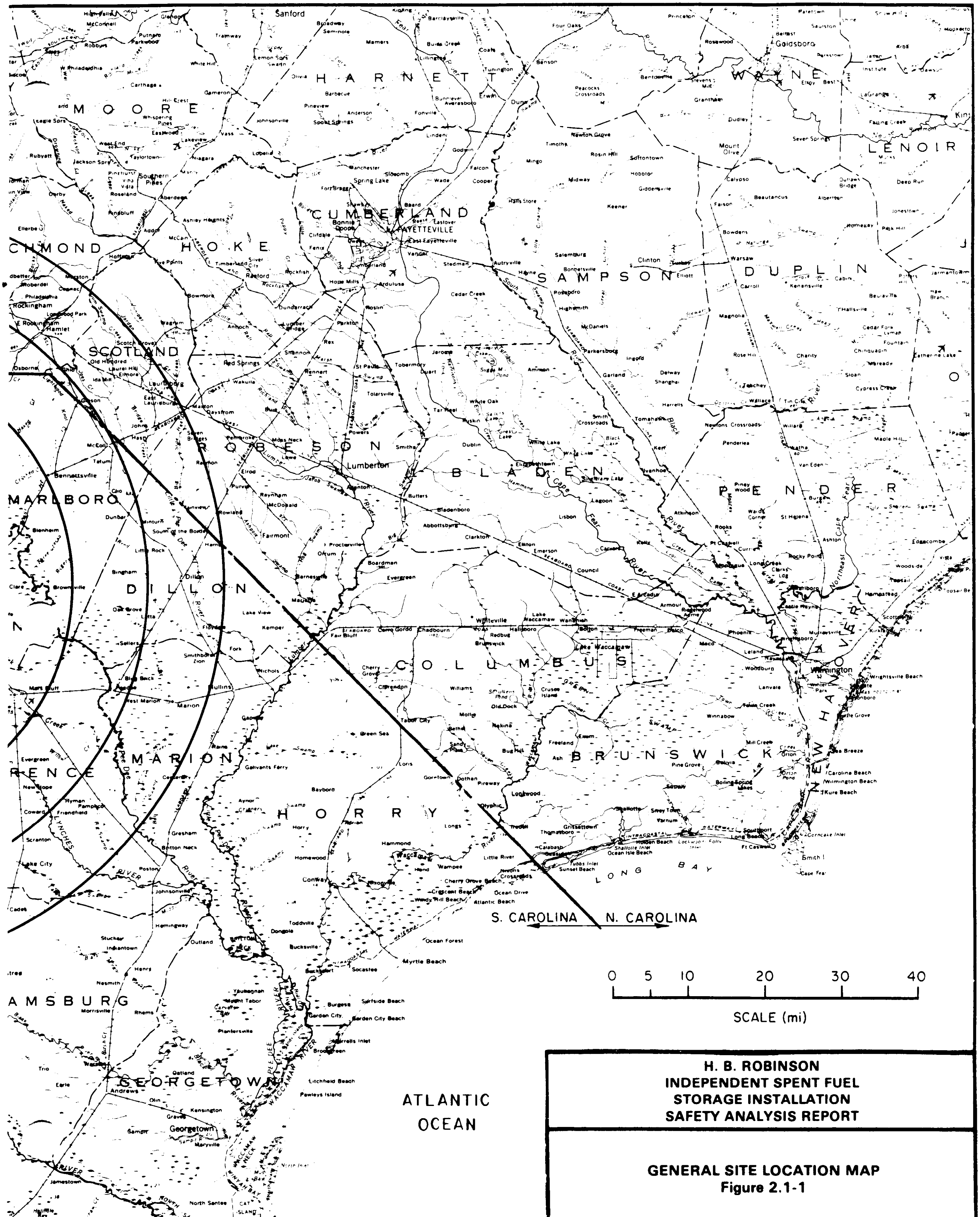
The failure of any slopes at the HBR site will not adversely affect the ISFSI.

REFERENCES: CHAPTER 2

2.1 Carolina Power & Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

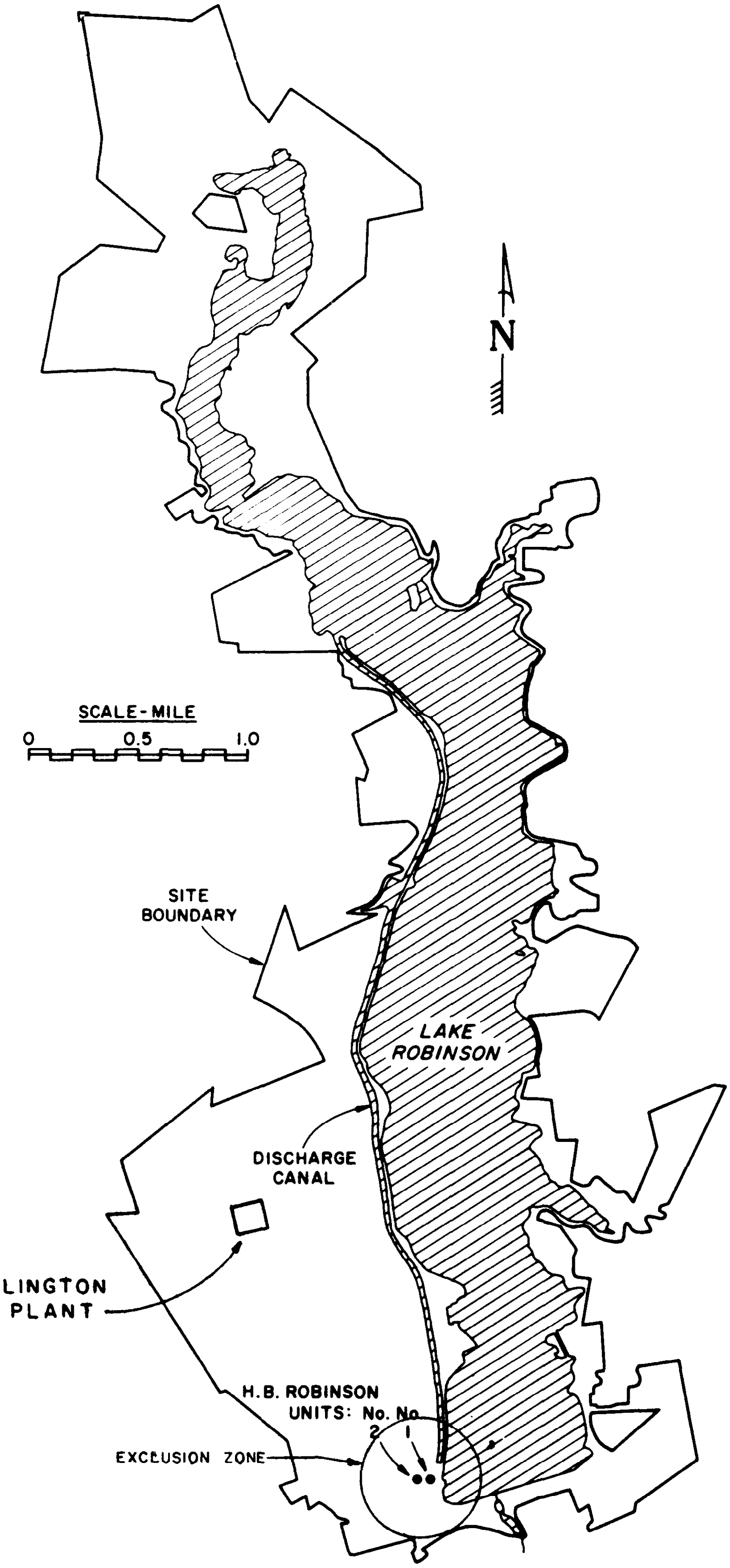
2.2 SC Development Board, "Industrial Directory of South Carolina," Columbia, SC, 1982.


SITE
LAKE ROBINSON
5mi
10mi
20mi
30mi
40mi
50mi
CLEVELAND
GASTON
MECKLENBURG
CABARRUS
STANLY
MONTGOMER
CHEROKEE
YORK
UNION
ANSON
LANCASTER
CHESTER
CHESTERFIELD
FAIRFIELD
KERSHAW
DARLINGTO
NEWBERRY
LEE
SALUDA
RICHLAND
SUMTER
LEXINGTON
CALHOUN
CLARENDON
AIKEN
ORANGEBURG
BARNWELL
BAMBERG
CHARLOTTE
Gastonia
Concord
Albemarle
Monroe
ROCK HILL
Chester
Lancaster
Camden
Bishopville
Hartsville
Chesterfield
Cheraw
Florence
Sumter
COLUMBIA
Orangeburg
Aiken
Newberry
Union
Gaffney
SPARTANBURG
LAKE MURRAY
WATEREE LAKE
FISHING CREEK RES
LAKE WYLIE
MARION
LAKE MOULTRIE



H. B. ROBINSON
INDEPENDENT SPENT FUEL
STORAGE INSTALLATION
SAFETY ANALYSIS REPORT

GENERAL SITE LOCATION MAP
Figure 2.1-1



H. B. ROBINSON INDEPENDENT SPENT FUEL STORAGE INSTALLATION SAFETY ANALYSIS REPORT	ROBINSON SITE BOUNDARY AND EXCLUSION ZONE	Figure 2.1-2

CHAPTER 3

PRINCIPAL DESIGN CRITERIA

3.0 PRINCIPAL DESIGN CRITERIA

3.1 PURPOSE OF THE INSTALLATION

The purpose of the H. B. Robinson (HBR) Independent Spent Fuel Storage Installation (ISFSI) is to provide the long-term storage of irradiated fuel assemblies (IFAs) in a dry environment. The HBR ISFSI is based on the NUTECH Horizontal Modular Storage (NUHOMS) System and is composed of a series of reinforced concrete horizontal storage modules (HSM). Each HSM will house a steel, helium filled, dry shielded canister (DSC) containing seven IFAs. The double weld sealed DSC serves as the confinement vessel for the IFAs while the HSM provides the biological shielding as well as a passive heat removal system for the decay heat of the IFAs.

3.1.1 MATERIAL TO BE STORED

Each modular unit of the ISFSI is capable of storing up to seven pressurized water reactor (PWR) fuel assemblies as specified in Section 3.1 of the NUHOMS Topical Report (Reference 3.1). The following subsections describe the physical, thermal, and radiological parameters of the fuel to be stored at the HBR ISFSI.

3.1.1.1 Physical Characteristics

The mechanical and structural design of the DSC is based on the physical characteristics of the PWR IFAs to be stored within the DSC. The physical characteristics of these IFAs are presented in Table 3.1-1. The overall length of the IFAs in Table 3.1-1 includes an allowance for 3/4 inch irradiation growth. Additional information on the physical characteristics of these fuel assemblies is contained in Section 4.2 of the Robinson Updated Final Safety Analysis Report (FSAR) (Reference 3.2).

3.1.1.2 Thermal Characteristics

The heat generation per fuel assembly is limited to one kilowatt per assembly. This results in a maximum of seven kilowatts per DSC. Fuel irradiated to less than 35,000 MWd/MT and cooled for five years will meet this criteria (based on ORIGEN2 (Reference 3.3) calculations). Other combinations of burnup and cooling time may also be acceptable upon further analysis.

3.1.1.3 Radiological Characteristics

The principal design criteria for acceptable radiological characteristics are shown in Table 3.1-2. The decay heat generation and the radiological characteristics described above bound fuel that has been irradiated to 35,000 MWd/MT and cooled for five years. This is considered to be the maximum radiation source fuel. The radiation source from the majority of the fuel to be stored will be less than or equal to that described in the NUHOMS Topical Report. ORIGEN2 calculations were used to determine this maximum burnup and minimum cooling time. Other combinations of decay time and burnup may also be acceptable if additional calculations are performed. However, for fuel which can be shown to have a burnup of less than or equal to 35,000 MWd/MT and been cooled for at least five years, no further fuel specific analysis is necessary.

3.1.2 GENERAL OPERATING FUNCTIONS

3.1.2.1 Overall Functions of the Facility

The ISFSI is designed to be totally passive, requiring no utilities or waste processing system and to utilize HBR2's existing cask handling, fuel handling and associated auxiliary equipment in preparing the IFAs for dry storage. The cask to be used for the onsite transfer operation is the GE IF-300 shipping cask which is fully documented in Reference 3.5.

The DSC will be placed into the cavity of the GE IF-300 shipping cask. The shipping cask will then be lowered into the existing HBR2 spent fuel pool where seven fuel assemblies will be placed into the DSC. Once the top lead shield plug is placed onto the loaded DSC, the cask containing the loaded DSC will be raised out of the pool and the water drained partially from the DSC cavity. At this juncture, the top lead shield plug is welded into place. The DSC cavity will then be drained and vacuum dried of all water and backfilled with helium through the vent and siphon tube penetrations. After backfilling the DSC with helium, the penetrations will be seal welded and the top cover plate will be welded into place. With the sealed DSC still within the confines of the shipping cask, the shipping cask will be transported to the HSM and aligned with the front access of the HSM. A hydraulic ram will then extend from the rear access of the HSM through the HSM and attach onto the DSC grappling plate. The hydraulic ram will be retracted through the HSM, pulling the DSC into the HSM. Once the DSC is properly positioned within the HSM, the front and rear accesses of the HSM will be closed.

The HSMs which house the DSCs are located on a level, reinforced concrete, load bearing slab. The slab is designed for normal and postulated accident conditions. The HSM is also designed to maintain its dimensional and structural integrity during postulated environmental and geological events.

Safe storage in the HSM is provided by: (1) a natural convection heat removal path, (2) the concrete radiation shielding, and (3) the double closure welds of the DSC. The operation of the HSMs and DSCs is totally passive. No active systems are required.

Since the first three units of the ISFSI facility will be used as part of the demonstration program, two of the HSMs and the DSCs will be instrumented for the purpose of collecting data. The instrumentation is limited to placement of a number of thermocouples in these components. The placement of thermocouples inside the HSMs does not effect its structural and mechanical integrity. The instrumentation of the DSC, however, requires a feed through penetration at its bottom cover plate. This penetration is designed such that the confinement integrity of the DSC is not compromised under both normal operating and accident conditions. Furthermore the instrumentation of these components does not change the total passive nature of the system. Details of the instrument penetration analysis are provided in Chapter 8 of this report.

A more detailed description of each component's functions is located in the following subsections.

3.1.2.2 Handling and Transfer Equipment

The ISFSI is designed to utilize the existing HBR2 fuel handling equipment and the GE IF-300 shipping cask. The function of the various HBR2 fuel handling equipment which are employed in loading and handling the IF-300 shipping cask are described in the plant's existing procedures and Section 9.1.4 of the HBR2 Updated FSAR (Reference 3.2).

The systems and equipment which are unique to the ISFSI for handling and transfer of IFAs are the DSC, the docking collar, the transfer trailer, the cask skid, the hydraulic ram, and the HSM. The function of each transfer and handling system or piece of equipment along with the waste processing system are briefly described in the following paragraphs.

a) DSC - The DSC will serve as the confinement vessel during transport of the IFAs to and from the HSM as well as during storage of the IFAs in the HSM. The shielded end plugs will provide biological shielding during transport of the fuel assemblies and also provide shielding for the front and rear accesses of the HSM.

b) Cask - The GE IF-300 shipping cask is used to transport the DSC to and from the HSM. The function of the shipping cask is to provide biological shielding along the axial length of the IFAs and a means for removing a sufficient quantity of decay heat so that the mechanical integrity of the DSC or IFAs is not jeopardized. The GE IF-300 is fully documented in Reference 3.5. As stated in Section 1.3.1.3 a new cask collar and lid are used on the IF-300 cask. This addition will provide the minimum cask cavity length requirement.

c) Cask Positioning Skid - The function of the cask positioning skid is to provide a means by which the final alignment of the cask with respect to the HSM can be achieved, and to restrain the shipping cask in the horizontal position during the transfer of the DSC to and from the HSM. The skid and the cask will be transported from the fuel building to the HSMs by a trailer. Both the skid and the DSC are designed to withstand the inertia forces associated with the transportation shock loads.

d) Hydraulic Ram - The hydraulic ram is used to move the DSC between the cask and the HSM. The hydraulic ram is a long hydraulic cylinder with a grapple device mounted at the cylinder's head. The hydraulic ram will be positioned at the rear access of the HSM. The ram will then be extended through the rear access and through the entire length of the HSM. Once the grapple is seated within the grappling plate of the DSC, the arms of the grapple will be extended so that it is securely in place between the DSC and the grapple plate. The ram will then be retracted, pulling the DSC out of the cask cavity and into the HSM.

e) Horizontal Storage Module- The function of the HSM is to provide protection for the DSC against geological and environmental events as specified in Section 3.2, and serve as the principle biological shield for irradiated fuel during storage. The HSM contains shielded air ducts near the bottom of the structure to admit ambient air around the DSC for cooling purposes. The air, warmed by the canister, is exhausted through shielded vents at the top of the HSM by natural draft convection. The HSM also provides support for the DSC. The DSC rests on a support rail assembly which is anchored to the walls of the HSM. The rear end of the DSC support rails are equipped with stopping blocks. The purpose of these stopping blocks is to establish the final axial position of the DSC inside the HSM. The final positioning is achieved when the DSC, being pulled into the HSM by the ram, makes contact with these stopping blocks. After the DSC insertion into the HSM is completed, a seismic retaining assembly will be attached to the grappling plate of the DSC top cover plate. This will prevent any possible axial movement of the DSC during any postulated accident such as an earthquake. The front access of the HSM is covered by a plate. The air inlets and outlets are covered with stainless steel wire bird screens to prevent foreign objects from entering the HSM.

f) Waste Processing - During the normal storage of IFA's in the ISFSI, no waste will be generated. However, contaminated water and possibly contaminated gases will be removed from the DSC cavity during the cask drying operation. The cask drying operation will take place in the HBR2 decontamination facility. The HBR2 utilities and radioactive waste processing system are described in Section 9.1.4. and Chapter 11.0 of the HBR2 Updated FSAR (Reference 3.2).

TABLE 3.1-1

PHYSICAL CHARACTERISTICS OF PWR
FUEL ASSEMBLIES[1] BASED ON NOMINAL DESIGN

Array	15 x 15
Envelope (in)	8.426
Overall Length[2] (in)	162.05
Weight (lbs)	1466
Fuel Rod Number	204
Fuel Rod Length (in)	152.0
Active Fuel Length (in)	144.0
Maximum Distance Between Grid Straps (in)	26.19

1 See Reference 3.4.

2 Additional 3/4 inch added to overall length to allow for irradiation growth.

TABLE 3.1-2

ACCEPTABLE RADIOLOGICAL CRITERIA FOR
STORAGE OF MATERIAL IN THE HBR ISFSI

CRITERIA	VALUE
GAMMA SOURCE PER CANISTER (total)	1.48×10^{16} Mev/sec
Fractional Breakdown*	
Above 1.3 Mev	0.004
Between 1.3 Mev and 0.8 Mev	0.114
Between 0.8 Mev and 0.4 Mev	0.808
Below 0.4 Mev	0.074
NEUTRON SOURCE PER CANISTER (total)**	1.17×10^{9} n/sec
Fractional Breakdown	
Above 5 Mev	5.40×10^{7} n/sec = 5.41%
Between 2.5 and 5 Mev	2.43×10^{8} n/sec = 24.32%
Between 1 and 2.5 Mev	4.56×10^{8} n/sec = 45.67%
Below 1 Mev	2.45×10^{8} n/sec = 24.53%

* Fractional breakdown based on isotopic composition and resulting gamma spectrum calculated by ORIGEN2 analysis.

** Spectrum from U-235 fission, total number of neutrons per second from ORIGEN2 analysis.

3.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

The H. B. Robinson Independent Spent Fuel Storage Installation is designed to perform its intended function under extreme environmental and geological hazards as specified in 10CFR, Part 72.122(a). The HSMs are installed on a monolithically placed, reinforced concrete mat foundation. The mat foundation is also designed to resist forces generated by extreme environmental and geological conditions. Specifics of the foundation design are reported in Section 8.3.

The environmental features at the ISFSI site, which are used to define the normal operating design basis for the DSCs and HSMs, are such that they are the same as or enveloped by those specified in Table 3.2-1 of the NUHOMS Topical Report (Reference 3.1). Specifically , the highest recorded ambient temperature of 107°F, and the lowest recorded temperature of -5°F, are well within the extreme ambient range of 125°F to -40°F specified in the NUHOMS Topical Report. The maximum diurnal temperature range for the ISFSI site is 25°F. This range is also lower than the 45°F diurnal temperature range specified in the above referenced report. The site area design basis solar radiation value of 188 Btu/hr-(ft^2) is the same as that reported in the above referenced report.

In general, the structural and mechanical safety criteria of the ISFSI are the same as or enveloped by the criteria specified in Section 3.2 of the NUHOMS Topical Report.

As stated earlier in this report, some design features of the H. B. Robinson ISFSI differ from those of the NUHOMS generic concept. In particular, the NUHOMS Topical Report addresses an eight module unit, whereas the H. B. Robinson ISFSI consists of a three-module unit and a five-module unit. Each unit will be a monolithically poured in place unit, with rear access penetrations for each HSM which allows for the hydraulic ram to be operated from the back of the modules. These unique features of the ISFSI reinforced concrete modules have no impact on the structural evaluation presented throughout chapter 8 of the NUHOMS Topical Report. This is due to the fact that the NUHOMS structural analysis of the HSM utilizes the frame action of the roof slab and the walls of only a single module in the transverse direction. This single module approach conservatively envelopes the structural response of any multi-module units including the three or five-module concept which is the subject of this report. The rear access penetration, however, require additional shielding evaluation which is presented in chapter 7 of this report.

Some of the design features of the H. B. Robinson DSCs also differ from those of the NUHOMS Topical Report. In particular, the bottom region of the DSC has been redesigned to fit into the IF-300 cask. Furthermore, both the top and bottom regions, the spacer disk and the support rods of the DSC have been redesigned to withstand the inertia forces associated with cask drop accidents in which the drop heights are significantly greater than the minimum 8 foot drop criteria established earlier in this report. This was done for compatibility with future shipping options. Another unique feature of the H. B. Robinson DSCs is the instrument penetration at the bottom region of two of the initial three DSC assemblies. The instrumentation of the DSC is for the purpose of collecting temperature data during the first year of storage.

The penetration assembly is designed to maintain the confinement integrity of the DSC during both normal operating and accident conditions. The structural evaluation of the instrument penetration under various normal operating and accident conditions is addressed in chapter 8 of this report.

3.2.1 TORNADO AND WIND LOADINGS

3.2.1.1 Applicable Design Parameters

The ISFSI will be constructed within the existing boundaries of the H. B. Robinson Steam Electric Plant which is located within Region I of the NRC Regulatory Guide 1.76 regionalization, and as such, the intensity of the design basis tornado for this region is the same as that assumed in the NUHOMS Topical Report (Reference 3.1).

The design basis tornado (DBT) intensities were obtained from NRC Regulatory Guide 1.76. Region I intensities were considered since it has the most severe parameters. For this region, 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 or 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. The maximum transit time based on the specified 5 miles per hour minimum translational speed was not used since the transit time is conservatively assumed to be infinite.

3.2.1.2 Determination of Forces on the Structures

The forces due to the design basis tornado and tornado generated missiles are enveloped by those reported in the NUHOMS Topical Report.

The method of analysis for overall and local damage prediction due to a design basis tornado and tornado generated missiles is discussed in Section 3.2.1 of the NUHOMS Topical Report, and is fully applicable to the Robinson site specific analysis.

3.2.1.3 Ability of Structures to Perform

The ISFSI is designed to withstand the design basis tornado wind loads. Furthermore, all components of the ISFSI with the exception of the air outlet shielding block are designed to withstand the tornado generated missile forces. The loss of an air outlet shielding block is addressed in Section 8.2.1 of this report.

Since the ISFSI is not housed in any storage building, there is no possibility of any roof collapse on the facility. However, 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.

3.2.2 WATER LEVEL (FLOOD) DESIGN

The maximum flood water level at the Robinson site is 222 ft. elevation (see Section 2.4). The grade level of the ISFSI foundation is at the 234 ft. elevation. Therefore, there is no possibility of flooding within the ISFSI.

3.2.3 SEISMIC DESIGN

3.2.3.1 Input Criteria

The maximum horizontal ground acceleration specified for HBR2 is 0.20g for safe shutdown earthquake (SSE) (see Section 2.6.2.3 and Reference 3.2). The maximum vertical ground acceleration is specified at two thirds of the horizontal component, or 0.133g. These horizontal and vertical component values are less than the values of 0.25g and 0.17g, respectively, specified in the NUHOMS Topical Report (reference 3.1). Hence, the seismic evaluation contained in Section 8.2.3 of the referenced report is fully applicable to those components of the H. B. Robinson ISFSI which have design features similar to those of the NUHOMS generic concept. It is also applicable to those that can be conservatively enveloped by the NUHOMS generic assumptions, such as the single module approach to HSM evaluation discussed earlier in this section. In this manner, the seismic response of the HSM and the DSC support assemblies of the H. B. Robinson ISFSI are enveloped by the responses reported in the NUHOMS Topical Report for these components. The DSC, however, which has some unique features, is analyzed for the site specific seismic event, utililizing the methodology and the analytical approach of the NUHOMS Topical Report. The seismic evaluation of the DSC is contained in Section 8.2 of this SAR.

3.2.3.2 Seismic-System Analysis

The stresses in the HSM and the DSC support assembly due to the 0.20g horizontal and 0.133g vertical acceleration are enveloped by the results of the generic seismic analysis reported in the NUHOMS Topical Report. The stresses in the DSC due to the horizontal and vertical seismic acceleration specified above are evaluated and reported along with the DSC rollover evaluation in Section 8.2 of this report.

The ISFSI foundation and the HSMs tie down system are also designed to withstand the forces generated by the SSE. The details of the foundation design are provided in Section 8.3 of this report.

3.2.4 SNOW AND ICE LOADS

The NUHOMS Topical Report specified a postulated live load of 200 pounds per square foot which conservatively envelopes the maximum snow loads for the Robinson site (see Section 2.3 for meteorology of the site area).

3.2.5 COMBINED LOAD CRITERIA

Load combination criteria established in the NUHOMS Topical Report (Reference 3.1) for the HSM, DSC and DSC support assembly are also applicable to the HBR ISFSI. The specific load combination evaluation of the DSC, utilizing the NUHOMS criteria, are reported in Section 8.2 of this report.

The facility's mat foundation is designed to meet the requirements of ACI 349-80 (3.6). The ultimate strength method of analysis was utilized with appropriate strength reduction factors. The load combination procedure of Section 9.2.1 of the ACI 349.80 was used in combining normal operating loads (i.e. dead loads and live loads) with severe and extreme loads (i.e., seismic and tornado loads). The details of the load combination procedure are described in the NUHOMS Topical Report. Specific foundation analyses including load combination and anchorage analysis are presented in Section 8.3 of this report.

3.3 SAFETY PROTECTION SYSTEM

3.3.1 GENERAL

The HBR Independent Spent Fuel Storage Installation is designed for safe and secure, long-term containment and storage of IFAs. The equipment which must be designed to assure that the safety objectives are met are shown in Table 3.3-1.

The major features which require special design consideration are:

a) Double Closure Seal Welds on DSC Upper End

b) Radiation Exposure During DSC Drying and Closure

c) Design of DSC Body and Internals for a Cask Drop Event During the Transfer Operation

d) Minimization of Contamination of the DSC Exterior by the Spent Fuel Pool Water

e) Minimization of Radiation Shine During Transfer of the DSC from the 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 ISFSI relies on a system of multiple confinement barriers during all handling and storage operations. Table 3.3-2 from the NUHOMS Topical Report (Reference 3.1) has been included and summarizes the radioactive confinement barriers and systems employed in the design of the ISFSI.

During transport and storage operations, the IFAs are confined within the DSC. The DSC consists of a cylindrical shell and multiple end plates. Each end plate will be seal welded to the canister in order to provide redundant seals for the DSC. These redundant seals minimize the likelihood of an uncontrollable release of radioactivity. Detailed discussion of the DSC confinement integrity, including the discussion on helium confinement, is presented in Section 3.3.2.1 of the NUHOMS Topical Report and is fully applicable here.

The criteria for protection against any postulated internal or external natural phenomena are discussed in Section 3.2 and Chapter 8 of this report.

3.3.2.2 Ventilation - Offgas

During the normal storage operations of the ISFSI, there will be essentially no release of radioactive material. Additionally, as discussed in Chapter 8 of the NUHOMS Topical Report (Reference 3.1), there are no credible accidents which could cause a release of radioactivity. Therefore, the HBR ISFSI does not require an offgas system.

During the cask drying operation, water and gas will be removed from the cavity of the DSC. This operation will take place in the HBR2 decontamination facility and the water and gas will be routed through Unit 2's existing radioactive waste processing system.

3.3.3 PROTECTION BY EQUIPMENT AND INSTRUMENTATION SELECTION

3.3.3.1 Equipment

The DSC and the GE IF-300 shipping cask are the only equipment that specifically provide protection during normal and off-normal operations of the ISFSI. The design criteria for the DSC are provided in Section 3.2 of this SAR. The design criteria for the cask are listed in the GE IF-300 Safety Analysis Report (Reference 3.5).

3.3.3.2 Instrumentation

The HBR ISFSI is designed to be totally passive and therefore, no safety related instrumentation is required for operation of the facility. However, two of the DSCs and HSMs will be instrumented for experimental purposes only for the one year test period (Agreements with DOE and EPRI).

The instrumentation is limited to placement of a number of thermocouples within these components. Instrumentation of the HSM does not effect its structural and mechanical properties. The placement of thermocouples in the DSC, however, requires a feed-through penetration at the DSC bottom region. This feed-through incorporates the same redundant seal philosophy used in the DSC design. The penetration is also designed such that confinement integrity of the DSC is not compromised under both normal operating and accident conditions. The instrument penetration analysis is provided in Section 8.4 of this report.

3.3.4 NUCLEAR CRITICALITY SAFETY

The DSC internals are designed to provide nuclear criticality safety during wet loading operations. A combination of administrative procedures, materials properties, geometry, and neutron poisons are used to assure that subcritical conditions exist at all times.

The DSC internals were analyzed for criticality safety using the KENO-IV Monte Carlo criticality code. The calculational procedures, cross-section sets, biasing techniques, and computer hardware used to perform the criticality analyses were identical to the methods used in the NUHOMS generic design (refer to the NUHOMS Topical Report - Reference 3.1).

Since the DSC for the HBR ISFSI design uses a different boron content in the aluminum boron alloy material on the fuel guide sleeves, additional calculations were performed to assure criticality safety. The 70E H_2O case

(Case No. 1, Table 3.3-4 of the NUHOMS Topical Report) was executed using the minimum specified boron content (4.5%). Twenty-one thousand neutron histories were executed to obtain the maximum k_{eff}.

$$\begin{aligned} \text{Max } K_{eff} &= K_{eff} + 2 \text{ sigma} + K_{bias} \\ &= 0.91960 + 2(0.00537) + 0.0174 \\ &= 0.94774 \end{aligned}$$

For details of the analysis methodology, and a discussion of the computational bias, see Section 3.3.4 of the NUHOMS Topical Report (Reference 3.1).

3.3.5 RADIOLOGICAL PROTECTION

3.3.5.1 Access Control

A locked fence will be placed partially around the ISFSI for the purpose of designating the area a radiation control area. The key will be controlled by the HBR2 Radiation Control unit. Access to the ISFSI will on an as needed basis.

3.3.5.2 Shielding

An estimate of collective onsite and offsite doses during operations and around the ISFSI are presented in Chapter 7.

3.3.5.3 Radiological Alarm System

There are no radiological alarms required.

3.3.6 FIRE AND EXPLOSION PROTECTION

3.3.6.1 Fire Protection

The degree of fire protection a structure requires is based on a number of factors: type and location of combustible materials and their proximity to or location within the ISFSI, type of construction and its fire resistance characteristics, fire barriers, and the ability of the plant's fire brigade to reach and effectively extinguish a credible fire.

No combustible materials are stored within the ISFSI or within the ISFSI's boundaries. There is no fixed fire suppression system within the boundaries of the ISFSI. The facility is, however, located outside the confines of any building and is directly accessible to HBR2's fire brigade. The fire brigade has access to HBR2's existing portable fire suppression equipment or the site's water fire protection system, as described in Section 9.5.1 of the HBR2 FSAR (Reference 3.2).

3.3.6.2 Explosion Protection

The DSC and HSM contain no volatile materials and therefore, no credible internal explosion is possible. Internal explosions are not considered as part of the design criteria. The design basis for explosions away from the HSM is bounded by the design basis tornado missile described in Section 3.2 of this report and of the NUHOMS Topical Report (Reference 3.1).

3.3.7 MATERIALS HANDLING AND STORAGE

3.3.7.1 Irradiated Fuel Handling and Storage

The fuel handling systems used in loading the IFAs into the DSC are presented in Section 9.1.4 of the HBR2 Upated FSAR (Reference 3.2). Irradiated fuel handling outside the spent fuel storage pool will be done with the fuel assemblies enclosed in the canister. Criticality safety during handling and storage is discussed in Section 3.3.4. The criterion for safe configuration is an effective mean plus two-sigma neutron multiplication factor (k_{eff}) of 0.95. Calculations have shown that the expected k_{eff} value is well below this limit.

The basic criterion for the cooling of irradiated fuel during storage is a maximum cladding temperature of 380°C (716°F). Higher temperatures may be sustained for brief periods without endangering cladding integrity. During canister drying and other normal and abnormal transients, the criterion is a cladding temperature of 570°C for 48 hours. For further details, see Section 3.3.7.1 of the NUHOMS Topical Report (Reference 3.1).

The canister external contamination limits are listed below:

Beta/Gamma Emitters	10^{-3})Ci/Cm^2 or 220,000 dis/min/100 cm^2
Alpha Emitters	10^{-5})Ci/Cm^2 or 2,200 dis/min/100 cm^2

The canister is sealed by double welds prior to storage so that any contamination of the canister interior or its contents will remain confined during transfer and storage.

3.3.7.2 Radioactive Waste Treatment

The contaminated water removed from the cask annulus and cavity of the loaded DSC will be handled by HBR2's radioactive waste treatment system. The site's radioactive waste treatment system is discussed in Chapter 6 of this report and is described in Chapter 11 of the HBR2 Updated FSAR (Reference 3.2).

3.3.7.3 Waste Storage Facilities

No radioactive wastes will be generated during the life of the ISFSI. The contaminated water removed from the cask annulus and cavity of the loaded DSC will be handled by HBR2's radioactive waste treatment system. The waste storage facility associated with the HBR2 radioactive waste treatment system is described in Chapter 11 of the HBR2 Updated FSAR (Reference 3.2).

3.3.8 INDUSTRIAL AND CHEMICAL SAFETY

No hazardous or volatile chemicals or chemical reactions are involved in the operation of the ISFSI and therefore, were not considered in any of the facility's design criteria.

TABLE 3.3-1

H. B. ROBINSON
ISFSI IMPORTANT TO SAFETY (SAFETY RELATED) FEATURES

I. Transfer Cask (IF-300) (Q-List)	Safety Related[1]
II. Dry Storage Canister (DSC) (Q-List)	Safety Related[2]
II.a. Basket	
II.b. Stiffener Plates	
II.c. Support Rods	
II.d. Lead Plug/Support	
II.e. Canister Body	
II.f. End Closure Plates	
III. Horizontal Storage Module (HSM)	Non-Safety Related[3]
III.a. Concrete Shielding	
III.b. DSC Support Assembly	
IV. Foundation Non-Safety Related[3]	
V. Transfer Components	Non-Safety Related
V.a. Transport Vehicle	
V.b. Hydraulic Ram	
VI. Instrumentation Non-Safety Related	

Notes:

1. As defined by 10CFR71 for a licensed transportation cask.

2. For the purposes of this license application, CP&L considers the Dry Storage Canister Safety Related as defined by 10CFR50-Appendix B.

3. CP&L applied a "Radwaste Related" QA program to these components for the procurement, construction and testing phases of the project. The 10 CFR 50 Appendix B program as described in RNP UFSAR Section 17.3 shall be applied to the operational phase of the project.

TABLE 3.3-2

RADIOACTIVITY CONFINEMENT BARRIERS
AND SYSTEMS OF THE ISFSI

Radioactivity	Confinement Barriers and Systems
Contaminated Spent Fuel Pool Water	1. Demineralized Water in Cask
	2. Cask/DSC Annulus Seal
Irradiated Fuel Assemblies	1. Fuel Cladding
	2. DSC Body
	3. Seal Welded Primary Closure
	4. Seal Welded Secondary Closure

3.4 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

A classification of the ISFSI's various structures, components, and systems is listed in Table 3.3-1.

3.5 DECOMMISSIONING CONSIDERATIONS

The dry storage canisters are intended to be transferred to a federal repository when such a facility is operational. The concrete module is designed so that the canister can be safely returned to a shipping cask and transported offsite to the federal repository.

Shipping cask design and transportation requirements will depend on the regulations in effect at the time when the federal repository begins receiving spent fuel. In the absence of new regulations, the existing GE IF-300 shipping cask owned by Carolina Power & Light Company would be used to transport the canisters.

Contamination on the canister exterior will be very small (see Section 3.3.7 of the NUHOMS Topical Report (Reference 3.1)). The resulting contamination of the module internals and air passages will also be minimal. This level of contamination may be removed by manual methods so that the reinforced concrete module can be broken-up and removed using conventional methods.

The canister itself may be contaminated internally by crud from the irradiated fuel and will be slightly activated by spontaneous neutron emissions from the irradiated fuel. The canister is designed to be used in the respository for final disposal; however, if the fuel is removed from the canister, the canister could be disposed of as low-level waste. The exact decommissioning plan to be applied will depend on the status of the U.S. waste repository program at the time of decommissioning.

REFERENCES: CHAPTER 3

3.1 NUTECH Engineers, Inc. "Topical Report for the NUTECH Horizontal Modular Storage System For Irradiated Nuclear Fuel," NUH-001, Revision 1, November 1985.

3.2 Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

3.3 Oak Ridge National Laboratory, "ORIGEN2 - Isotope Generation and Depletion Code - Matrix Exponential Method," CCC-371, 1982.

3.4 Exxon Nuclear Company, Inc., Drawing NX-302.517 (NP), Exxon Nuclear Company, Inc., Richland, Washington, 1975.

3.5 General Electric Co., "IF-300 Shipping Cask Consolidated Safety Analysis Report," NEDO-10048-2, Nuclear Fuel and Special Products Division, March l983.

3.6 American Concrete Institute, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary, ACI 349-80 and ACI 349R-80, American Concrete Institute, Detroit, Michigan, 1980.

CHAPTER 4

INSTALLATION DESIGN

4.0 INSTALLATION DESIGN

4.1 SUMMARY DESCRIPTION

4.1.1 LOCATION AND LAYOUT OF INSTALLATION

The H. B. Robinson (HBR) Independent Spent Fuel Storage Installation (ISFSI) is located within the existing site boundary of the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR2). The design of the ISFSI is based on the NUTECH Horizontal Modular Storage (NUHOMS) System for irradiated nuclear fuel. The ISFSI is composed of a series of reinforced concrete horizontal storage modules (HSM). Each HSM contains one dry shielded canister (DSC) which serves as the confinement barrier for the irradiated fuel assemblies (IFAs). The following sections describe the ISFSI, its components and their interrelationship with HBR2's existing facilities. Figure 1.1-2 shows the location of the ISFSI at HBR2.

4.1.2 PRINCIPLE FEATURES

4.1.2.1 Site Boundary

The site boundary is shown on Figure 2.1-2.

4.1.2.2 Controlled Area

The controlled area is shown on Figure 1.1-2.

4.1.2.3 Emergency Planning Zone

The exclusion zone is shown on Figure 2.1-2.

4.1.2.4 Site Utility Supplies and Systems

No utility supplies and systems are required for the ISFSI.

4.1.2.5 Storage Facilities

H. B. Robinson Unit 2 facilities are shown on Figure 1.1-2.

4.1.2.6 Stack

The ISFSI has no stack.

4.2 STORAGE STRUCTURES

4.2.1 STRUCTURAL SPECIFICATIONS

4.2.1.1 Design Basis

The ISFSI and all its components are designed in accordance with the requirements of 10CFR Part 72. The components are designed to maintain their dimensional and structural integrity and safely perform their intended functions under normal and off-normal operating conditions and during postulated geological or environmental events. The loading conditions associated with normal and accident events are specified in Section 3.2 and Chapter 8 of this report.

4.2.1.2 Construction, Fabrication, and Inspection

a) Horizontal Storage Module - The HSM is constructed of reinforced concrete. The HSM is designed in accordance with the requirements of the ACI 349-80. The applicable American Society of Testing and Material (ASTM) standards referenced in Section 3.8 of the ACI code was used as part of the fabrication and construction requirements of the HSM. Construction of the HSM is in accordance with ACI 301-84, Specification for Structural Concrete for Buildings.

The concrete materials used for construction of the HSM and the foundation consist of: Type II cement conforming to ASTM C150, fine and coarse aggregates conforming to ASTM C33, concrete air-entraining admixture conforming to ASTM C-260, and reinforcing bars conforming to ASTM A615 Grade 60. The concrete compressive strength specified is 4000 psi at 28 days. Minimum specified density of concrete is 145 pounds per cubic foot. The HSM walls are tied to the foundation by reinforcing dowels to prevent possible overturning or sliding during any accident condition specified in Section 8.2 of this report.

b) Dry Shielded Canister - The DSC is designed and fabricated in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, 1983 Edition. Material selection, welding, and inspection of the DSC was per requirement of this code and the applicable ASTM, ANSI, and other codes and standards invoked by the ASME code.

Details of the material and construction of the DSC and its internals are contained in Section 4.2.3.1 of the NUHOMS Topical Report (Reference 4.1).

4.2.2 INSTALLATION LAYOUT

4.2.2.1 Building Plans

There is no building associated with the HBR2 ISFSI other than the horizontal storage module which is discussed in Section 1.3.1.2.

4.2.2.2 Confinement Features

The confinement features for each unit of the ISFSI are provided in Section 3.3.2.1.

4.2.3 INDIVIDUAL UNIT DESCRIPTION

Each unit of the ISFSI is composed of two components; a dry shielded canister and a horizontal storage module. The DSC is a weld-sealed stainless steel container that provides confinement of contaminants associated with the irradiated fuel assemblies, encloses the fuel basket in an inert atmosphere and provides biological shielding at the ends of the canister. The HSM is a reinforced concrete structure that serves as the primary biological shield for the irradiated fuel assemblies along with providing protection for the DSC against environmental and geological hazards. A detailed discussion and description of the function, design basis, and safety assurance considerations of each component are provided in Section 4.2.3.1 of the NUHOMS Topical Report (Reference 4.1). The engineering drawings of the DSC and the HSM, showing plan views, sections, and elevations of these components are provided in Figures 4.2-1 and 4.2-2.

Two DSCs and their corresponding HSMs will be instrumented for the purpose of collecting data during the first year of storage. The instrumentation is limited to placement of thermocouples in the DSC and the HSM. The HSM thermocouples are cast in place or attached to the concrete surfaces, and as such, have no impact on the integrity of the structure. The DSC thermocouples will be connected to an external cable by means of a specially designed feed-through. This feed-through penetration incorporates the same redundant seal philosophy used in the DSC containment design. After the penetration plug assembly has been welded to the bottom of the DSC cover plate, a sleeve will be welded over the plug, forming a redundant seal. Thermocouple sheaths will likewise be brazed to the plug assembly at inner and outer surfaces of the penetrations. To ensure that possible leakage through the aluminum oxide insulation is precluded, each end of the brazed thermocouple will be sealed with an environmentally qualified resin.

4.3 AUXILIARY SYSTEMS

The design of the HBR ISFSI is based on the NUHOMS system for irradiated fuel. Each unit of this dry storage system is totally passive and self contained, requiring no auxiliary systems other than a transfer cask for transport operation.

4.3.1 VENTILATION AND OFFGAS SYSTEM

4.3.1.1 Ventilation System

The decay heat rejection system for each module is based on natural circulation convection cooling. The IFAs are confined within a double weld sealed DSC. The decay heat from the IFAs is transferred by radiation, conduction and convection to the surface of the canister. Air inlets near the bottom and air outlets at the top of each HSM allow air to circulate around the DSC. Decay heat is removed by convection and radiation from the surface of the DSC. The driving force for circulation of the air is thermal buoyancy. An analysis of the HSM ventilation system is described in Section 8.1.3 of the NUHOMS Topical Report (Reference 4.1). Since the IFAs are confined within the double weld sealed DSC and the contamination on the external surface of the DSC results in essentially no release of radioactive material, no filtration system for contamination is required.

4.3.1.2 Offgas System

The offgas system used during the DSC drying and backfilling operations is the same as that described in Chapter 6 of this report and Chapter 11 of the HBR2 Updated Final Safety Analysis Report (Reference 4.2).

4.3.2 ELECTRICAL SYSTEM

The ISFSI is totally passive and requires no electrical system. However, two of the first three DSCs will contain thermocouples for research purposes only. The instrumentation will utilize HBR2's existing power supplies. The power supply systems are described in Chapter 8 of the HBR2 Updated FSAR (Reference 4.2). Any instruments which may be used are for experimental and data collection purposes only and therefore require no emergency power source or means of ensuring an uninterruptible power source.

The HBR2 electrical system associated with the fuel handling area is utilized during DSC drying and backfill operations.

4.3.3 AIR SUPPLY SYSTEM

4.3.3.1 Compressed Air

The ISFSI requires no compressed air supply system. The HBR2 compressed air supply system will be utilized during the DSC drying operation.

4.3.3.2 Breathing Air

The ISFSI is located in an outside environment and requires no breathing air supply. Provisions for breathing air supply during an emergency situation exist at HBR2.

4.3.4 STEAM SUPPLY AND DISTRIBUTION SYSTEM

The ISFSI requires no steam supply.

4.3.5 WATER SUPPLY SYSTEM

The ISFSI requires no water supply system. The HBR2 water supply system associated with the fuel handling area will be used during cask preparation and decontamination operations.

4.3.5.1 Major Components and Operating Characteristics

The existing water supply system of HBR2 will be utilized during the cask preparation and decontamination operations and for fire protection purposes.

4.3.5.2 Safety Consideration and Controls

A loss of water supply would have no effect on the operation of the ISFSI. The water is used for removing contamination from the surface of the cask and DSC. Any loss of water during the cask wash down and loading operations would only extend the period of time which the DSC would be contained within the IF-300 shipping cask.

4.3.6 SEWAGE TREATMENT SYSTEM

4.3.6.1 Sanitary Sewage

The ISFSI requires no sanitary sewage system.

4.3.6.2 Chemical Sewage

The ISFSI requires no chemical sewage treatment system.

4.3.7 COMMUNICATIONS AND ALARM SYSTEM

The ISFSI is totally passive and requires no communication or alarm systems.

4.3.8 FIRE PROTECTION SYSTEM

No flammable or combustible substances are stored within the ISFSI or in its immediate vicinity. The ISFSI is constructed of noncombustible heat-resistant materials (concrete and steel). Since no combustible materials are stored within the ISFSI, the ISFSI requires no fire extinguishing system. In the unlikely event of a fire near the HSMs, the existing HBR2 fire extinguishing system will be utilized. This system and procedures for fire extinguishing are described in Section 9.5.1 of the HBR2 Updated FSAR (Reference 4.2).

4.3.9 MAINTENANCE SYSTEMS

The ISFSI is totally passive and requires no maintenance other than periodic inspection of the air inlets and outlets and possible removal of debris from the air inlets (see Chapter 10).

4.3.10 COLD CHEMICAL SYSTEMS

The ISFSI has no cold chemical system.

4.3.11 AIR SAMPLING SYSTEM

No air samples are required at any time during operation. Therefore, the ISFSI has no air sampling system.

4.4 DECONTAMINATION SYSTEMS

The ISFSI has no decontamination system; however, the existing HBR2 decontamination facility will be used to decontaminate the spent fuel cask.

The decontamination facility has been abandoned in place. Portable equipment is used to aid in the decontamination process.

4.5 SHIPPING CASK REPAIR AND MAINTENANCE

Existing HBR2 procedures are followed for maintenance and repair of CP&L's GE IF-300 shipping container and its auxiliary systems. Routine inspections of the cask system and components are accomplished in accordance with existing procedures. Where applicable, General Electric's recommended inspection intervals are followed. Corrective maintenance is planned and accomplished in a controlled manner as needed. Minor repairs can be accomplished onsite with the cask in the decontamination stand using existing HBR2 facilities (see Figure 4.5-1). However, if major repairs should be required on any large or heavy portion of the cask system, it may require offsite work. Such work can be accomplished for example at General Electric's Morris Operations facility located in Morris, Illinois.

During all phases of repair or maintenance on the cask, existing HBR2 health physics procedures are followed. These procedures address contamination control and emphasize occupational exposure reduction.

All necessary records generated during the maintenance, repair, or operation of the cask system will be retained in a controlled manner for the time period required by appropriate QA procedures.

4.6 CATHODIC PROTECTION

The ISFSI is dry and above ground so that cathodic protection in the form of impressed current is not required. The normal operating environment for all metallic components is well above ambient air temperatures so that there is no opportunity for condensation on those surfaces.

The austenitic canister body requires no corrosion protection for any forseeable event. The DSC support assembly components are Type A36 carbon steel. The top surface of the DSC rail is to be machined and plated with a high-phosphorus electroless nickel finish having a minimum thickness of .001 inch. The remaining surfaces of the rail and other components of the support assembly shall be painted with Carbo-Zinc 11 (Reference 4.3). Consequently, the A36 structural steel is protected against corrosion.

4.7 FUEL HANDLING OPERATION SYSTEM

The HBR ISFSI is based on the NUHOMS system for storage of irradiated nuclear fuel. The basis and engineering design for the various fuel handling systems used during the operation of the ISFSI are described in the following sections.

4.7.1 STRUCTURAL SPECIFICATIONS

The bases and engineering design of HBR2's fuel handling systems are described in Section 9.1.4 of the HBR2 Updated FSAR (Reference 4.2). Other handling and transport operation equipment (cask positioning skid, hydraulic ram, and trailer) are designed to meet the criteria established in Chapter 3 of the NUHOMS Topical Report (Reference 4.2). This equipment is designed to safely perform its intended functions under both normal and off-normal operating conditions. However, this equipment does not impact the safety features of the ISFSI facility and as such is not safety related.

4.7.2 INSTALLATION LAYOUT

4.7.2.1 Building Plans

Fuel handling operations will occur within the existing HBR2 Fuel Handling Building (see Figure 4.5-1).

4.7.2.2 Confinement Features

The irradiated fuel assemblies will only be handled while in the spent fuel pool or in the confines of the DSC which is placed inside the cavity of the shipping cask or the HSM. The confinement features of the HBR2 spent fuel pool are provided in Section 9.1 of the HBR2 Updated FSAR (Reference 4.2) and the confinement features of the NUHOMS system are discussed in Section 3.3.2 of the NUHOMS Topical Report.

4.7.3 INDIVIDUAL UNIT DESCRIPTION

4.7.3.1 Shipping Cask Preparation

During the preparation, the DSC will be placed into the shipping cask cavity. This operation will take place in the HBR2 decontamination facility (see Figure 4.5-1). After loading, the cask annulus and the DSC will be filled with demineralized water and then lifted into the spent fuel pool.

The following components will be used for this operation:

a) Spent Fuel Cask Handling Crane - The spent fuel cask handling crane is used to place the DSC into the shipping cask cavity. The design basis and safety assurance features of the crane are discussed in Section 9.1.4 of the HBR2 Updated FSAR (Reference 4.2).

b) Spent Fuel Cask Lifting Yoke - The spent fuel cask lifting yoke is used in transporting and handling the shipping cask while in the plant and loading the shipping cask onto the transport skid. The design basis and safety features of the lifting yoke are provided in Section 9.1.4 of the HBR2 Updated FSAR (Reference 4.2).

4.7.3.2 Spent Fuel Loading

Loading of the IFAs into the DSC takes place in the existing spent fuel pool. The components which are used are described below.

a) Spent Fuel Cask Handling Crane - The spent fuel cask handling crane is used to transport the shipping cask to and from the spent fuel pool. The design basis and safety assurance features of the crane are discussed in Section 9.1.4 of the HBR2 Updated FSAR (Reference 4.2).

b) Spent Fuel Pit Bridge - The spent fuel pit bridge is used to move the fuel assemblies within the spent fuel pool. The design basis and safety features are described in Section 9.1.4 of the HBR2 Updated FSAR (Reference 4.2).

4.7.3.3 DSC Drying, Backfilling and Sealing

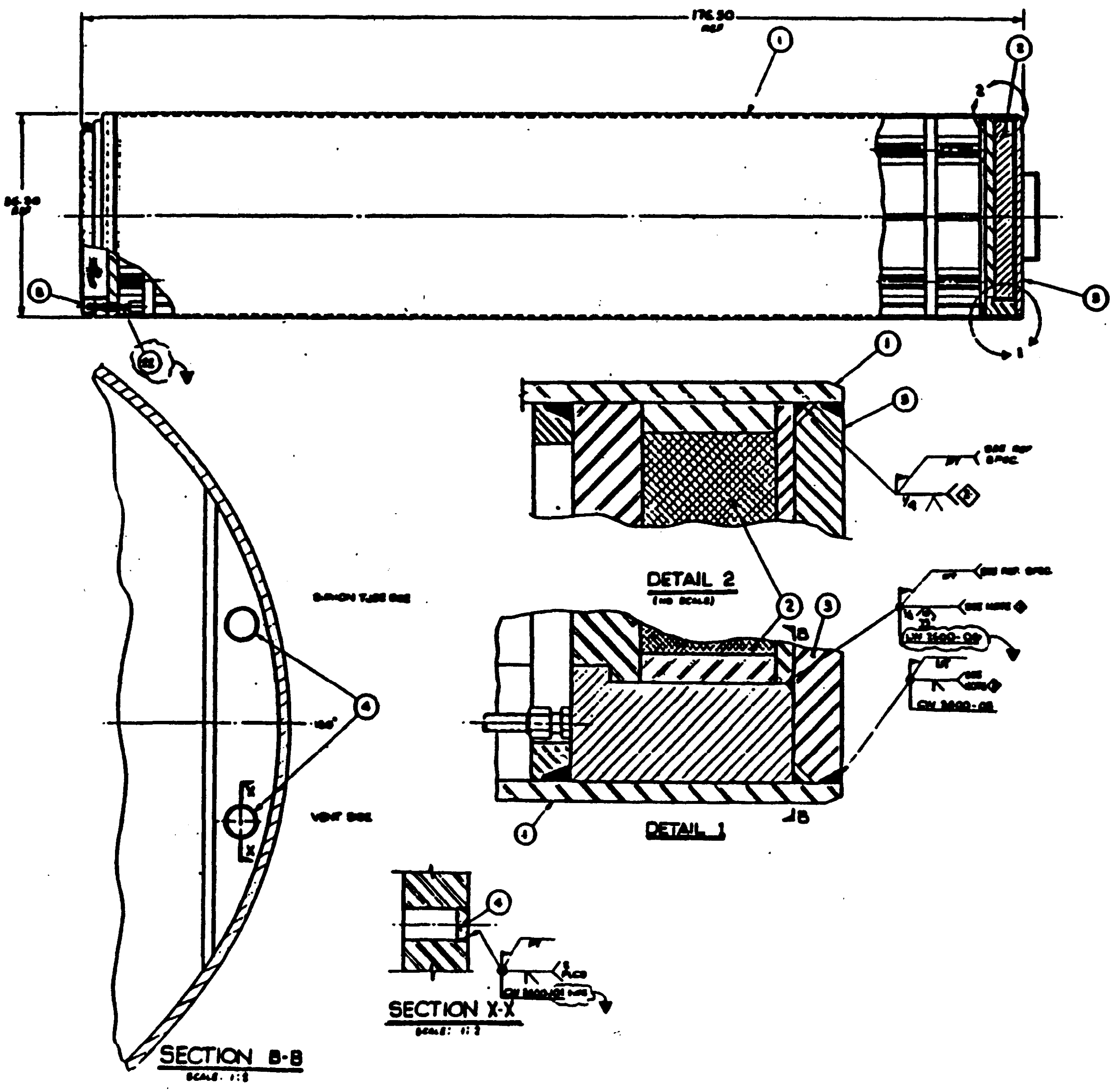
Once the IFAs have been placed into the DSC, the top lead shield plug is placed on the DSC. The cask collar cover plate is then installed. Using the spent fuel cask handling crane, the loaded DSC, sitting in the spent fuel cask cavity, will be removed from the spent fuel pool and moved to the decontamination facility where the cask will be drained. The DSC will be drained, vacuum dried, and backfilled with helium. The top lead shield plug and cover plates will be seal welded to the DSC body. After these operations, the cask will be placed onto the transport skid and taken to the ISFSI site where the DSC will be loaded into the HSM. Throughout all of the above operations, the fuel will be confined within the DSC and the DSC will be seated within the shipping cask. The design basis and safety assurance features of the DSC are discussed in Sections 3.2 and 3.3 of the NUHOMS Topical Report (Reference 4.1). More details on the drying and sealing operations are provided in Chapters 4 and 5 of Reference 4.1.

REFERENCES: CHAPTER 4

4.1 NUTECH Engineers, Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-001, Revision 1, November 1985.

4.2 Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

4.3 Carboline Company, St. Louis, MO.



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DRY SHIELDED CANISTER
Figure 4.2-1
SHEET 1 OF 2

FIGURE WITHHELD UNDER 10 CFR 2.390

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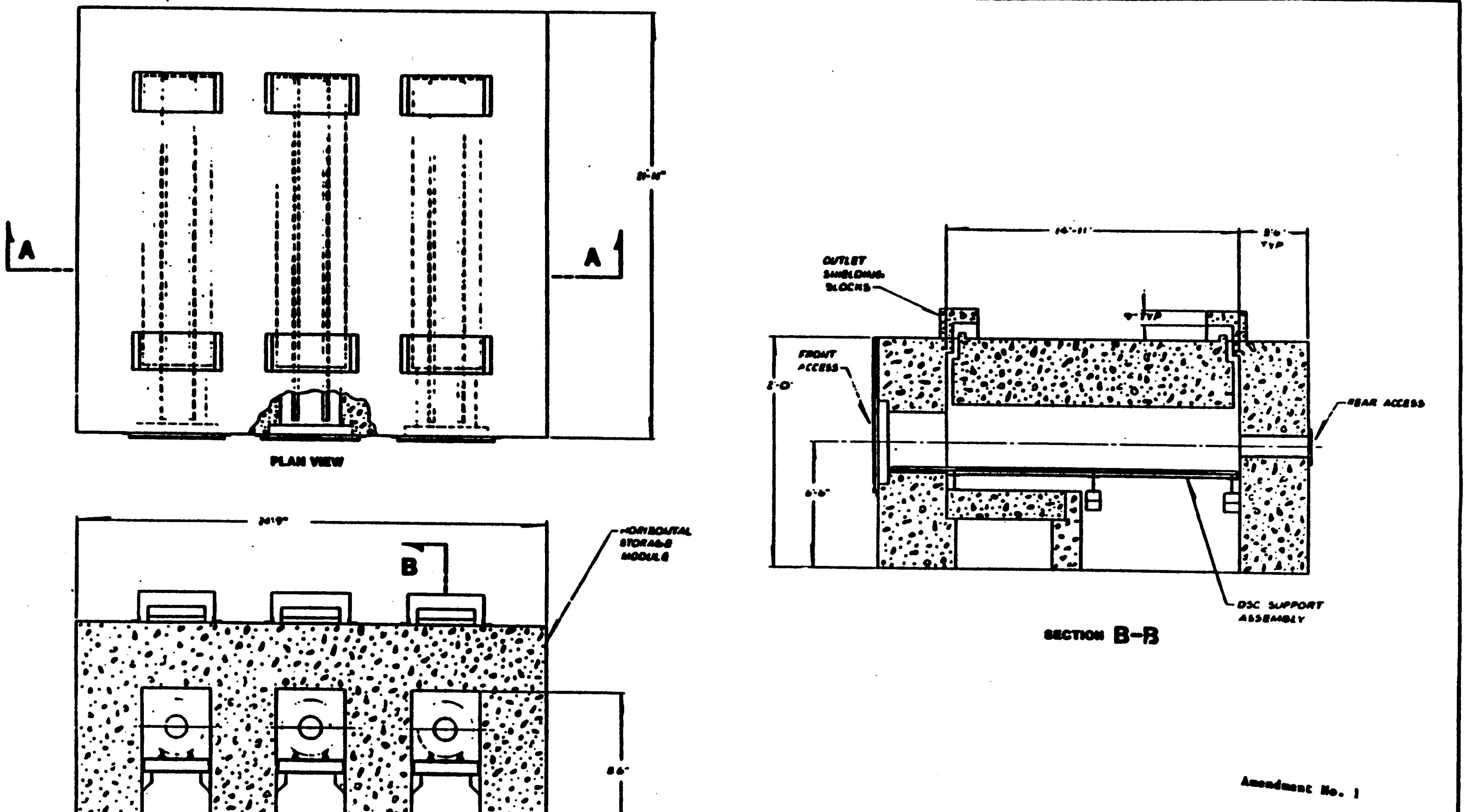
DRY SHIELDED CANISTER
FIGURE 4.2 - 1
SHEET 2 OF 2



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HORIZONTAL STORAGE MODULE
Figure 4.2-2

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE WITHHELD UNDER 10 CFR 2.390

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GENERAL ARRANGEMENT - HBR2 FUEL
HANDLING BUILDING
Figure 4.5-1

Chapter 5

Operation Systems

5.0 OPERATION SYSTEMS

5.1 OPERATION DESCRIPTION

The following sections describe the operating procedures, which are unique to the operations of the H. B. Robinson (HBR) Independent Spent Fuel Storage Installation (ISFSI) such as loading, unloading, and surveillance. Existing fuel and cask handling operations, which are currently being employed at fuel and cask handling operations, which are currently being employed at H.B. Robinson Steam Electric Plant Unit No. 2 (HBR2) will be incorporated with these procedures.

5.1.1 NARRATIVE DESCRIPTION

The following steps describe the operating procedures for the ISFSI.

5.1.1.1 Preparation of the Transfer Cask and Canister

a) Prior to the start of the operation, the fuel assemblies to be place in dry storage will be visually examined (e.g., by television cameras, binoculars, or other means, etc.) to insure that no visible defects exist and that the assembly structure is intact. The assemblies will also be checked (by analysis or by examination of appropriate records) to verify that they meet the physical, thermal and radiological criteria described in Chapter 3. Measures will be taken to ensure that no known failed fuel will be placed in dry storage. This process will be independently verified to ensure that fuel assemblies meeting the fuel specifications (see chapter 10) are selected for storage.

b) Place the cask in the vertical position in the decontamination facility.

c) Prior to the loading of fuel, the fuel basket used during the transportation operation will be removed from the GE IF-300 shipping cask and the cask cavity will be cleaned or decontaminated as necessary.

d) Using the spent fuel cask handling crane, lower the docking collar (Figure 5.1-1) onto the cask. Once the docking collar (cask extension) is properly oriented onto the transportation cask, bolt the docking collar into place and tighten.

e) Using the crane, lower the dry shielded canister (DSC) into the cask cavity.

f) Fill the DSC and the cask-canister annulus with clean, demineralized water.

g) Seal the top of the gap between the DSC exterior and cask interior.

h) Place the lid on the cask and lift the cask into the spent fuel pool.

5.1.1.2 Fuel Loading

a) Remove the cask lid and lead plug and place the irradiated fuel assemblies (IFAs) in the DSC (which is inside the shipping cask) using the existing HBR2 fuel handling equipment and procedures.

b) When all seven of the IFAs have been loaded into the DSC and the lead plug and cask lid have been secured, the cask will be moved to the decontamination facility.

5.1.1.3 Cask Drying Process

a) Place the cask in a vertical position in the decontamination stand.

b) Remove the cask lid.

c) Lower the water level about 2 in. in the cask canister annulus by removing water from the cask. Lower the water level in the DSC by removing approximately 15 gallons from the DSC.

d) Seal weld the upper steel cladding plates of the top lead plug to the canister body.

e) Connect a compressed air supply to the vent tube and another hose from the siphon connection to the decontamination facility's radioactive waste system. Activate the air supply forcing the remaining water out of the DSC cavity. (See figure 4.7-1 of the NUHOMS Topical Report (Reference 5.1) for a schematic of the piping system).

f) Once the water stops flowing from the DSC, remove the siphon hose from the DSC.

g) Connect the vent tube piping system to the intake of the vacuum pump. A hose should be connected from the discharge outlet of the vacuum system to the site radioactive waste system.

h) Start the vacuum system and draw a vacuum of 3 mm Hg within the DSC cavity.

i) Once a vacuum of 3mm Hg has developed in the DSC cavity, disengage the vacuum pump, connect the helium source, and backfill the DSC with 1.5 atm (22 psig) of helium. Verify with pressure gauge that pressure is holding and helium leak test the entire length of the primary end plug closure weld.

j) Seal weld the prefabricated plug over the vent and siphon tube connection.

k) Seal weld the prefabrication plug over the vent tube connection.

l) Perform helium leak test to ensure weld tightness.

5.1.1.4 DSC Sealing Operations

a) Place the top cover plant onto the DSC and weld the cover plate to the body of the DSC.

b) Perform dye penetrant test on seal weld.

c) Drain water from annulus and place the cask lid onto the cask and bolt the lid into place.

5.1.1.5 Transportation of the Cask to the Horizontal Storage Module (HSM)

a) Use redundant yoke to lift cask out of decontamination pit.

b) Disengage lower portion of redundant yoke and position cask in tilting cradle.

c) Lower cask to horizontal position on trailer and secure.

5.1.1.6 Loading of the Canister into the HSM

a) 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. Leave the front access of the HSM open.

b) Using the optical alignment system, align the ram with the cask centerline.

c) Position the cask so that the docking collar is within 1 foot of the HSM.

d) Remove the cask collar lid.

e) Using an optical alignment system and targets on the cask, skid, and HSM as necessary, adjust the position of the cask until the cask us properly positioned with respect to the HSM.

f) Extend the hydraulic ram and activate the grapple to grab the canister.

g) Move the cask against the HSM, so that the docking collar is positioned in the HSM recess.

h) Retract the piston of the hydraulic ram. If the ram fails to retract when the load on the hydraulic system exceeds 6,200 lbs, stop the ram. Check the orientation of the cask with respect to the HSM and reorient the cask if necessary. Continue this step until the DSC contacts the stopping blocks mounted on the top of the DSC support rails at the rear end. These blocks will assure correct axial positioning of the DSC within the HSM.

i) Collapse the grapple arms and withdraw the hydraulic ram.

j) Pull the skid away from the HSM.

k) Install the plate over the front access of the HSM.

l) Insert through the rear access opening the seismic retainer assembly until it couples with the DSC grapple assembly. Bolt down the retainer assembly to the rear access cover plate. Bolt down the cover plate to the rear access embedded plate.

5.1.1.7 Monitoring Operations

On a daily basis, site personnel will walk around the perimeter of the HSMs to visually inspect the air inlets and outlets to ensure that they remain unblocked and the integrity of the screens remains intact. Any debris present will be removed. If damage is evident, appropriate remedial action will be taken. The level of action will be responsive to the observation and will be sufficient to ensure safe operation of the facility. For example: if superficial damage to a screen has occurred, an effort will be made to determine the cause and prevent its reoccurrence, if the screen has been punctured, it will be replaced, the air passage it covers will be examined using a boroscope and any obstructing material removed. If the boroscopic examination indicates a possibility of further blockage, that too will be checked.

Response to any finding potentially affecting the safe operation of the facility will be appropriate to the finding. If a generic problem is indicated, the remaining modules will be inspected and appropriate remedial action taken. It is anticipated that the many conservatisms and safeguards inherent in this design will ensure safe spent fuel storage over the lifetime of the facility.

5.1.1.8 Unloading the DSC from the HSM

a) Using the optical alignment system, align the cask and ram with respect to the HSM.

b) Remove front access cover and rear cover plate and seismic retainer of the HSM and install the ram and grapple.

c) Align the ram with the cask centerline. Extend the ram through the rear access of the HSM until it contacts the DSC.

d) Activate the grapple of the ram.

e) Activate the ram and push the DSC into the cask.

f) Retract the ram piston out of the cask and HSM.

g) Slowly pull the cask forward 1 foot away from the HSM.

h) Place the cask lid onto the cask and bolt the lid into place.

i) Close the front and rear accesses to the HSM.

The fuel is in a safe configuration in the DSC within the IF-300 shipping cask. During the one year demonstration phase, provisions will be made for the DSC to be returned to the spent fuel pool, if necessary. Possible conditions upon which the DSC would be returned include exceeding the design limits shown in section 10.2 and 10.3. Additionally, if shipping were required, the DSC would be returned to the decontamination area or the fuel pool for removal of the cask collar and placement of the BWR head on the IF-300.

5.1.2 FLOW SHEET

A flow sheet for the handling operations is presented in Figure 5.1-3.

5.1.3 IDENTIFICATION OF SUBJECTS FOR SAFETY ANALYSIS

5.1.3.1 Criticality Prevention

Criticality is prevented by geometrical separation of the guide sleeves and by boron poison contained in the boral guide sleeves of the canister basket. All DSC baskets will include seven boral guide sleeves.

5.1.3.2 Chemical Safety

There are no chemicals used during the operation of the ISFSI that require special precautions.

5.1.3.3 Operation Shutdown Modes

The ISFSI is a totally passive system and therefore this section is not applicable.

5.1.3.4 Instrumentation

The ISFSI is a totally passive system requiring no instrumentation. However, some of the units will be instrumented for experimental purposes only. The description of the instrumentation is provided in section 4 of this report.

5.1.3.5 Maintenance Techniques

The ISFSI is a totally passive system and therefore will not require maintenance. However, to insure that the airflow is not interrupted, the module will be periodically inspected to insure that no debris is in the airflow inlet or outlet. This inspection will be performed daily (see chapter 10).

5.2 FUEL HANDLING SYSTEMS

5.2.1 SPENT FUEL HANDLING AND TRANSFER

The ISFSI is a modular storage system which provides for the dry storage of irradiated fuel in a horizontal position with natural draft cooling of the dry storage canister. The ISFSI is located within the HBR2 protected area and utilizes HBR2's existing system for handling the irradiated fuel and irradiated fuel cask. The DSC is designed to be used for transporting the spent fuel to a federal repository and can be removed from the HSM as described in Section 5.1.1.8.

5.2.1.1 Functional Description

Figure 5.1-3 presents the flow diagrams for the transfer, loading and retrieval operations. The transfer system is composed of the HBR2 fuel handling system, the GE IF-300 irradiated fuel cask, a transport skid, an optical alignment system, the hydraulic ram, and the HSM T-section guides. Table 5.2-l lists these major systems and their important subsystems.

a) HBR2 Fuel Handling System - The ISFSI is designed to utilize the existing HBR2 fuel handling system. The major components of this system that will be employed during the cask loading operation are the spent fuel pit bridge, the spent fuel cask handling crane, and the spent fuel cask lifting yoke. A description of these components is provided in Section 9.1.4 of the HBR2 Updated FSAR (Reference 5.2).

b) HBR2 Decontamination Facility - The HBR2 cask decontamination facility will be used to decontaminate the irradiated fuel cask. It also provides the location for preparing the DSC.

c) Irradiated Fuel Cask - The General Electric IF-300 Shipping 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 the transfer, loading, and retrieval operations. A description of the cask's cooling and shielding capabilities is provided in the IF-300 Shipping Cask Safety Analysis Report (Reference 5.3).

   The DSC surfaces will be treated with a lubricant that is compatible with the spent fuel pool chemistry. The cask docking collar is a circular ring of steel which is bolted to the top of the cask. The top six inches of the cask docking collar are seated inside of the HSM walls. The cask, HSM, and docking collar serve as the shield for radiation during the transfer operation.

   The cask auxiliary components described above aid in the horizontal transfer of the DSC and were previously described in sections of this document and the NUHOMS Topical Report (Reference 5.1).

d) Cask Positioning Skid - The purpose of the skid is to transport the cask in a horizontal position to the HSM and to maintain the cask in the properly aligned position during the loading and retrieval operations.

e) Optical Alignment System - Once the loaded skid has been positioned at the HSM front access, the cask will be aligned with the HSM. The alignment system consists of a precision transit and targets installed on the cask, skid, and HSM as required. Once the cask is aligned with the HSM, the jack system and cask clamping system will insure that the alignment is maintained throughout the transfer or retrieval operation.

The cask position control system physically moves the cask into precise alignment with the HSM. It consists of a group of hydraulic jacks to adjust vertical position and a set of hydraulic cylinders to control horizontal position.

f) Ram and Grappling Apparatus - The ram is a telescopic hydraulic cylinder which extends from the back of the HSM through the length of the HSM. The grappling apparatus is mounted on the front of the piston. Figures 5.2-l and 5.2-2 show drawings of the hydraulic ram and the grappling apparatus, respectively. The hydraulics for the grappling apparatus are activated and the arms move out between the cover plate and grappling plate. Once the arms are extended, they are locked into position, the ram is retracted, pulling the DSC out of the cask and into the HSM. For retrieval of the DSC, the process is reversed.

g) HSM T-Section Guide - During the transfer operation, the DSC will slide out of the cask and onto the T-section guides, which are within the cavity of the HSM. The T-section guides serve as both the sliding surfaces during the transfer operation as well as supports during storage of the DSC.

5.2.1.2 Safety Features

Except for the transfer of the DSC from the cask to the HSM, the loaded DSC will always be seated inside the cask cavity until it is inside the HSM. The safety features of the HBR2 fuel handling systems are described in Section 9.1.4 of the HBR2 Updated FSAR (Reference 5.2). The safety features of the shipping cask are described in the GE IF-300 SAR (Reference 5.3).

To ensure that the minimum amount of force is applied to the DSC during the transfer operation, the surfaces of the DSC may be treated with a solid film lubricant. A low coefficient of friction will minimize the amount of force applied to the DSC, thus minimizing the possibility of damage to the DSC.

The maximum force which may be exerted by the ram is 22,000 lbs. All components of the DSC, ram, grappling assembly and DSC supports are designed to withstand this force. Materials and lubricants are specified so that an operating force of 6200 pounds should easily accelerate the DSC from rest and move it into the HSM. A pressure limitation device in the hydraulic pump will limit the ram force to less than 6200 pounds. The operator can increase the ram force to the 22,000 pound maximum design pressure only by stopping the ram and resetting the limit. Operating procedures will establish the methods for resetting or adjusting the ram speed during travel on extend and retract. It should be noted that it is not expected that a 22,000 pound force will ever be required. However, the system was designed to take such a force if it is ever needed.

5.2.2 SPENT FUEL STORAGE

A description of the operations involved in the transfer and retrieval of the DSC to and from the HSM are presented in Section 5.1. During storage, the ISFSI area will be patrolled and the HSM will be visually inspected once per day. The removal of the DSC from storage was described in Section 5.1.1.8 of this document and the NUHOMS Topical Report (Reference 5.1).

5.2.2.1 Safety Features

The features, systems and special techniques, which provide for the safe loading and retrieval operations are described in Section 5.2.1.2.

TABLE 5.2-l

TRANSFER SYSTEM COMPONENT LIST

HBR2 Fuel Handling System
HBR2 Cask Handling System
HBR2 Decontamination Facility
Irradiated Fuel Shipping Cask
- Cask Docking Collar
- Cask Lid

Cask Positioning Skid
- Tilting Cradle
- Skid Body (with rollers)
- Transport Trailer

Optical Alignment System
- Precision Transit
- Optical Targets
- Cask Position Control System
- Hydraulic Jacks
- Hydraulic Cylinders
- Control Unit
- Cask Clamping System

Hydraulic Ram
- Grappling Device
- Control Unit

5.3 OTHER OPERATING SYSTEMS

The ISFSI is a totally passive storage system, which requires no additional operating systems other than those systems associated with the loading and retrieval of the DSC.

5.3.1 OPERATING SYSTEM

No operating systems are required other than those used in transferring the DSC to and from the HSM.

5.3.2 COMPONENTS/EQUIPMENT SPARES

The only component postulated to be damaged during the life of the installation is the air outlet shielding block. As described in Section 8.2.1, a tornado induced missile could damage or knock off the shielding blocks. Consequently, two additional shielding blocks will be precast during construction and maintained as spares at the site. The screens on the air inlets and outlets will be inspected periodically for damage or blockage by debris. If the screens appear to be damaged they will be replaced. Additional or alternate responses to any event affecting the integrity of the screens will be appropriate to the level of damage or disturbance observed. For example, if a tree branch is seen to penetrate a screen, the screen and branch would be removed, the air passage borescoped, any blockage removed, and the screen replaced.

5.4 OPERATION SUPPORT SYSTEM

The ISFSI is a self-contained system and requires no instrumentation and control systems to monitor any of the safety-related variables. For research purposes, however, some of the DSCs and the HSMs to be installed at the H. B. Robinson facility have been designed to accept instrumentation. Instrumentation was included as part of an agreement between CP&L, EPRI and the DOE to augment the U.S. data base on LWR fuel rods in dry storage.

The instrumentation of these components is limited to placement of the thermocouples. The DSC thermocouples will be connected to an external cable by means of a specially designed feed-through. This feed-through incorporates the same redundant seal philosophy used in the DSC containment design. Details of the feed-through are shown in Figure 5.4-l. After the penetration plug assembly has been welded to the bottom of the DSC cover plate, a sleeve will be welded over the plug, forming a redundant seal. Thermocouple sheaths will likewise be brazed to the plug assembly at inner and outer surfaces of the penetrations. To ensure that possible leakage through the aluminum oxide insulation is precluded, each end of the sheathed thermocouple is sealed with an environmentally qualified resin.

HSM instrumentation will consist of thermocouples cast in place in the concrete and others attached to the surface and at various locations on the heat shield.

5.5 CONTROL ROOM AND/OR CONTROL AREAS

The ISFSI requires no control room or control area to safely operate under both normal and off-normal conditions.

5.6 ANALYTICAL SAMPLING

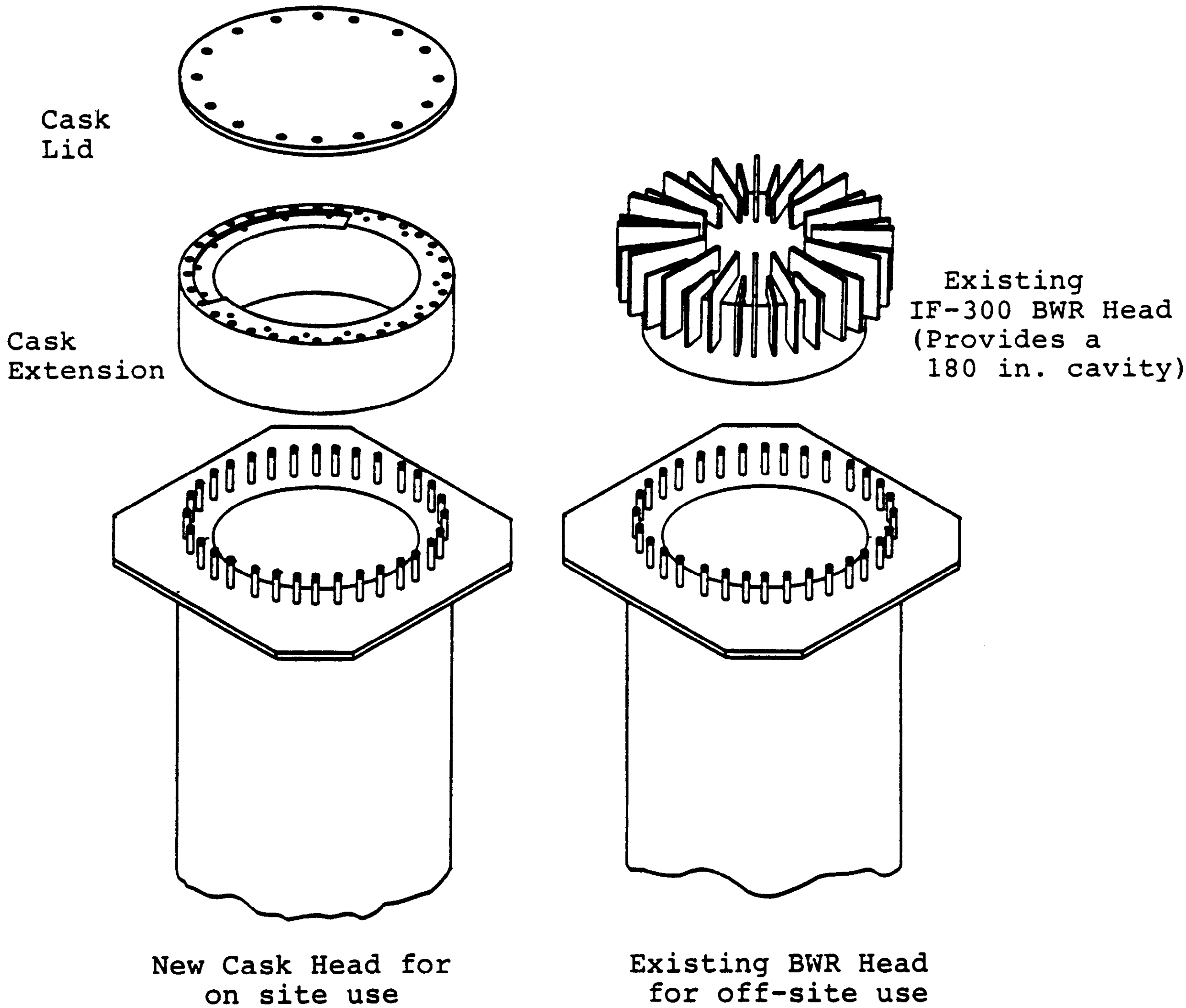
The ISFSI requires no analytical sampling. Any analytical sampling such as DSC surface contamination levels will utilize the existing HBR2 analytical equipment.

REFERENCES: CHAPTER 5

5.1 NUTECH Engineers, Inc., “Topical Report for the NUTECH Horizontal Modular Storage System For Irradiated Nuclear Fuel,” NUH-001, Revision 1, November 1985.

5.2 Carolina Power and Light Company, “H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report,” Docket No. 50-261, License No. DPR-23.

5.3 General Electric Co., “IF-300 Shipping Cask Consolidated Safety Analysis Report, "NEDO-10048-2, Nuclear Fuel and Special Products Division, March 1983.



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CASK EXTENSION FOR THE
GE IF-300 CASK
Figure 5.1-1

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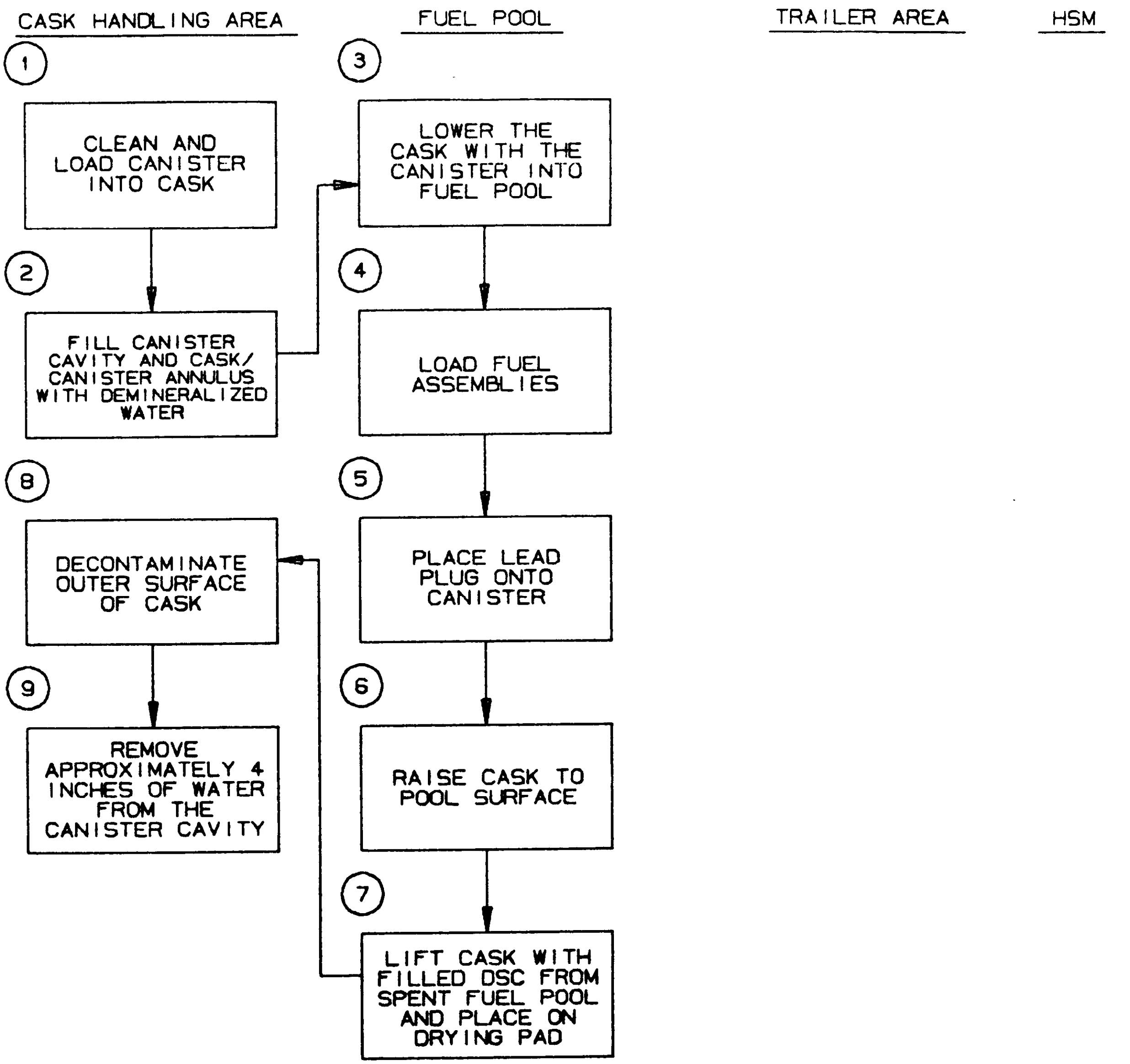
CASK LINER FOR THE
GE IF-300 CASK
Figure 5.1-2



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HANDLING OPERATIONS FLOW SHEET
FIGURE 5.1-3

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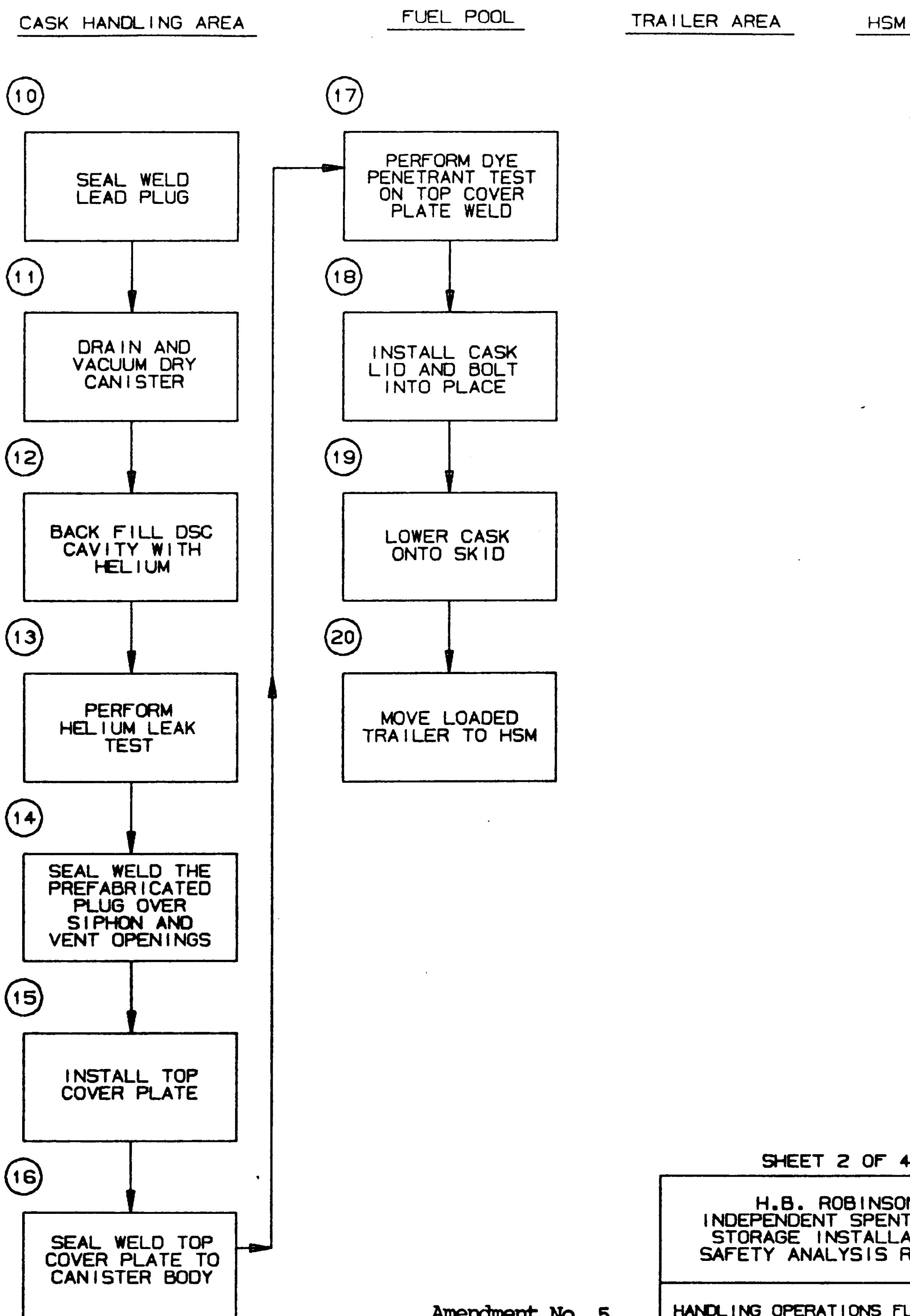

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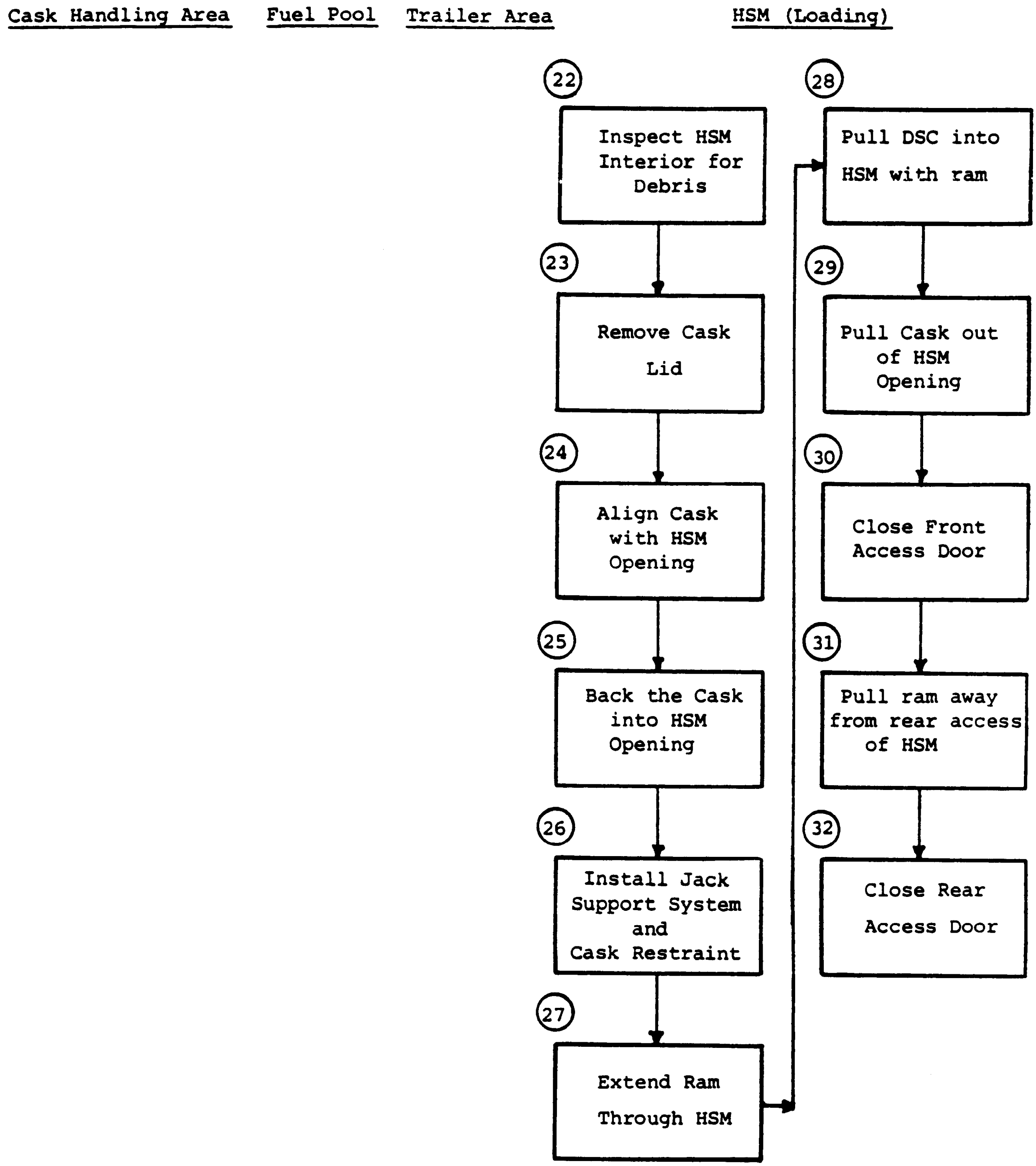
HANDLING OPERATIONS FLOW SHEET
FIGURE 5.1-3



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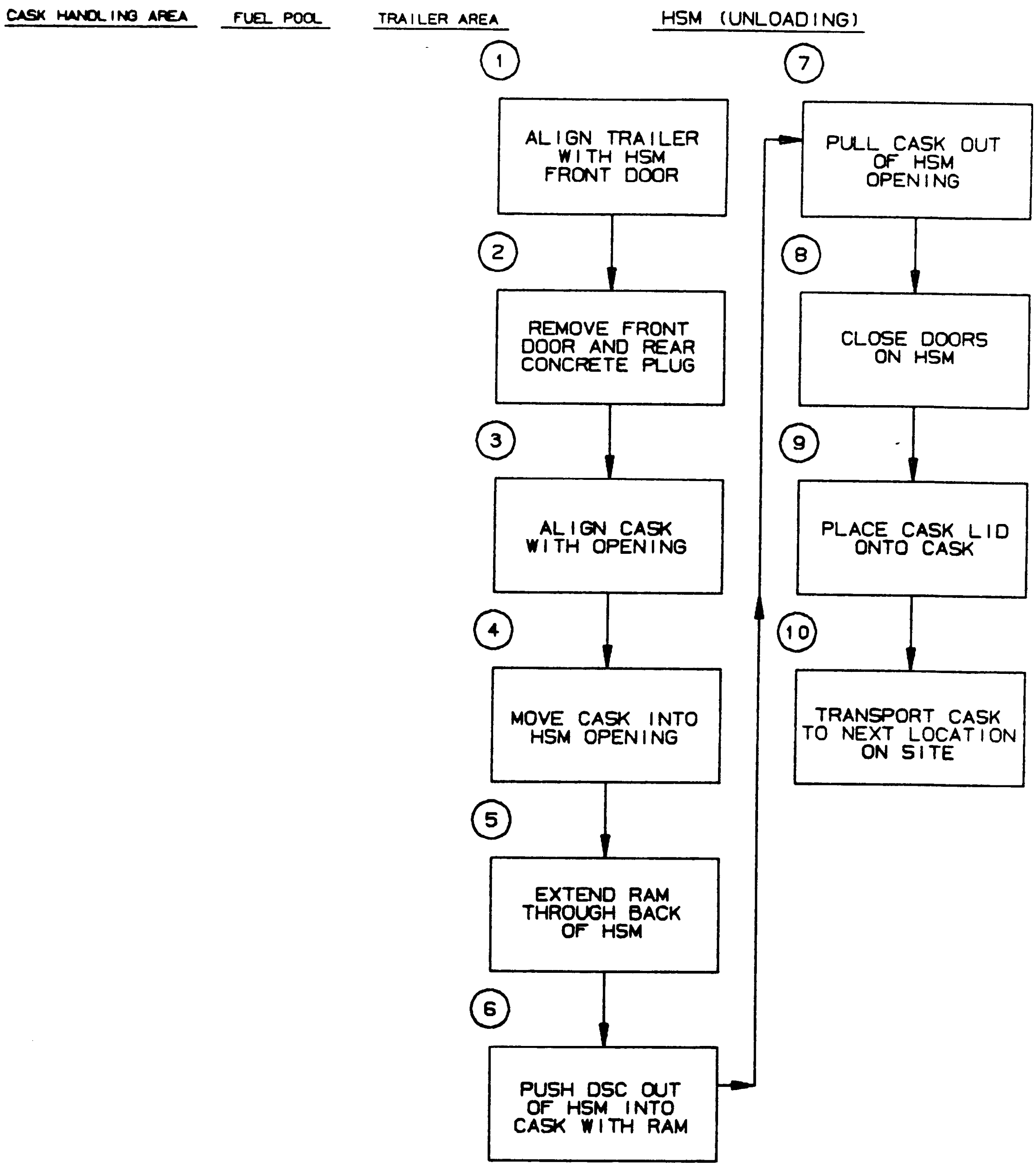
HANDLING OPERATIONS FLOW SHEET
Figure 5.1-3



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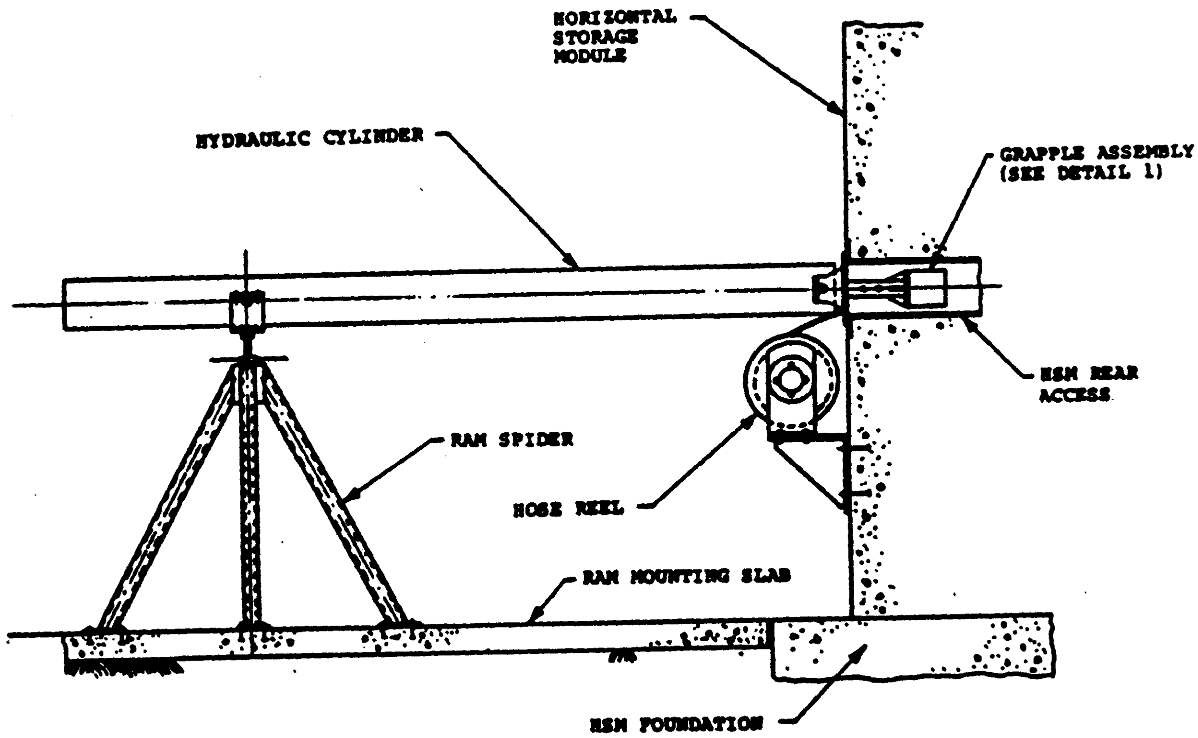
HANDLING OPERATIONS FLOW SHEET
FIGURE 5.1-3



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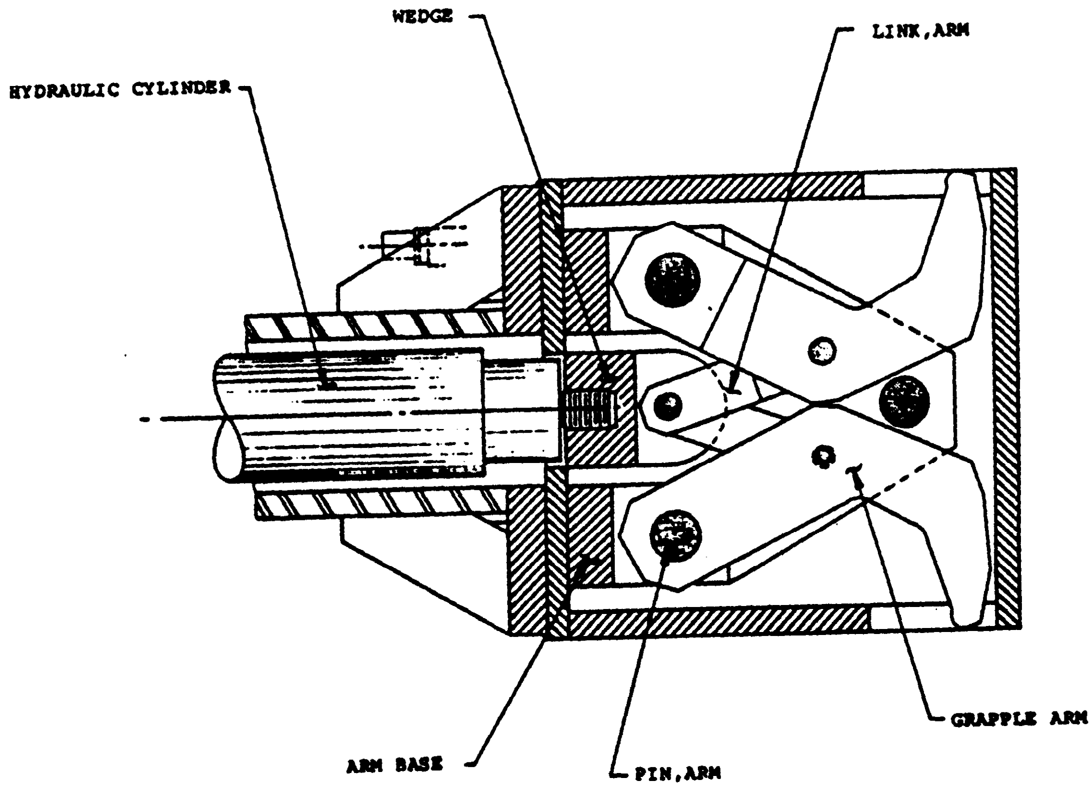
HYDRAULIC RAM SYSTEM

Figure 5.2-1



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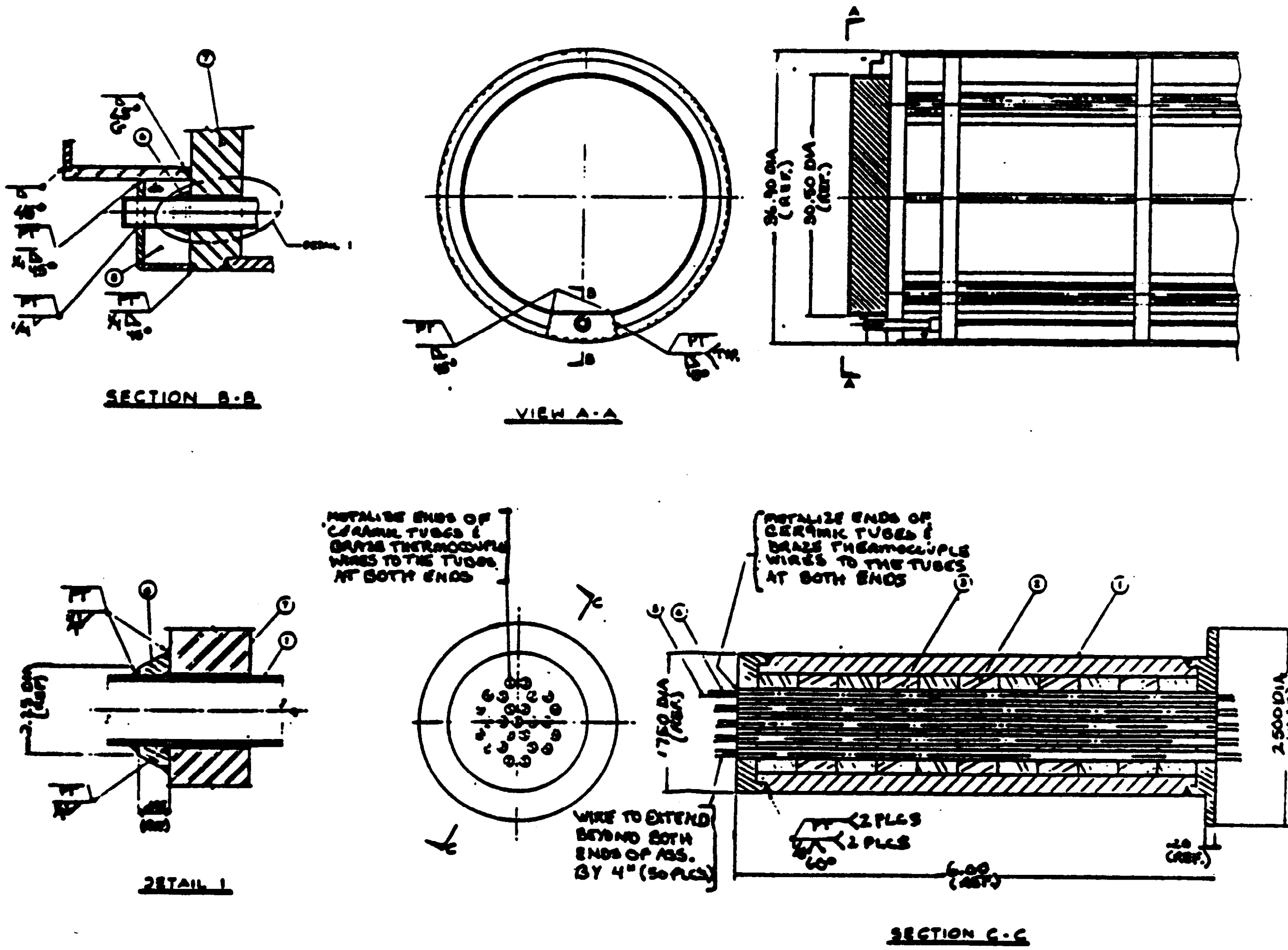
DSC GRAPPELING SYSTEM

Figure 5.2-2



AMENDMENT 1

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Figure 8.4-1

DSC INSTRUMENT LOCATIONS

CHAPTER 6

WASTE CONFINEMENT AND MANAGEMENT

6.0 WASTE CONFINEMENT AND MANAGEMENT

6.1 WASTE SOURCES

No contaminated wastes are generated during the storage of irradiated fuel in the H. B. Robinson (HBR) Independent Spent Fuel Storage Installation (ISFSI). The contaminated waste generated as a result of loading the spent fuel into the dry shielded canister (DSC) and into the IF-300 shipping container (see Chapter 5 for discussion of operations) is handled using existing H. B. Robinson Steam Electric Plant Unit No. 2 (HBR2) systems and procedures. The small amounts of contaminated waste which may be generated consist of the spent fuel pool water and air and inert gas which are vented from the DSC and the cask during the drying operation. The handling of spent fuel and loading into the canisters occurs in the existing cask handling area within the Fuel Handling Building of HBR2.

6.2 OFFGAS TREATMENT AND VENTILATION

Small amounts of slightly contaminated air and inert gas may be generated during loading of the spent fuel into the dry shielded canister and then the cask. Since the spent fuel will be handled as if it is being prepared for offsite shipment using the IF-300, no new procedures are required. The HBR2 gaseous waste management systems will be used, as necessary, to control, collect, process, store, and dispose of any gaseous wastes generated.

6.2.1 HBR2 WASTE GAS SYSTEM

During HBR2 operations, gaseous wastes originate from:

a) Degassing reactor coolant discharged to the Chemical and Volume Control System (CVCS)

b) Displacement of cover gases as liquids accumulate in various tanks

c) Miscellaneous equipment vents and relief valves

d) Sampling operations and gas analyzer operations

The amount of waste gas generated by the ISFSI is small compared to that generated by normal plant operations and can be handled by HBR2 systems.

6.2.1.1 System Description

Radioactive gases at HBR2 are collected at a slight positive pressure in a vent header. From there, they are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gases in the nitrogen blanketing system are reused to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent. The system is provided with holdup capacity and discharge controls for gaseous wastes such that plant operations will not be limited by environmental conditions.

A detailed description of the HBR2 gaseous waste management system is located in Section 11.3 of Reference 6.1.

6.2.1.2 Component Description

a) Gas Decay Tanks - Four gas decay tanks are provided at HBR2 to store waste gas. Controls are provided on the waste disposal-boron recycle panel to operate most of the air operated valves associated with the gas decay tanks.

Overpressure protection is provided by a relief valve that discharges to the plant vent stack.

A system interconnection to HBR2's nitrogen system can be used to purge the gas decay tanks. This may be necessary due to excessive hydrogen or in the event that maintenance is required on the tank or piping.

b) Waste Gas Compressors - There are two Nash waste gas compressors at HBR2. They are essentially centrifugal with an eliptical casing. A seal water system furnishes the seal between the rotating and stationary components in the compressor. The component cooling water system supplies makeup and cooling water to the compressor.

c) Gas Analyzer - A gas analyzer is provided at HBR2 to sample from various points in the waste gas and related systems. The gas analyzer can be set up to automatically sample any point that is not bypassed. Controls are provided to sample any point desired in the manual mode and to zero the analyzer using nitrogen. The concentration of the oxygen and hydrogen will be printed on recorders for each point sampled.

6.3 LIQUID WASTE TREATMENT AND RETENTION

The only liquid waste generated from the ISFSI results from loading the spent fuel assemblies into the dry shielded canister and into the IF-300 shipping container. This liquid waste is handled using existing HBR2 systems and procedures.

6.3.1 HBR2 LIQUID WASTE MANAGEMENT SYSTEM

During normal plant operation the HBR2 Waste Disposal System (WDS) processes liquids from the following sources:

a) Equipment drains and leakoffs

b) Radioactive chemical laboratory drains

c) Radioactive shower drains

d) Decontamination area drains

e) Demineralizer regeneration

The system also collects and transfers liquids from the following sources directly to the CVCS for processing:

a) Reactor coolant loop drains

b) Pressurizer relief tank

c) Reactor coolant pump secondary seals

d) Excess letdown during startup

e) Accumulators

f) Valve and reactor vessel flange leakoffs

The amount of liquid waste generated by the ISFSI is small compared to that generated by normal HBR2 operations and can be handled by HBR2 systems.

6.3.1.1 Design Objectives

The HBR2 system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10CFR20.

The HBR2 design includes those means necessary to maintain control over the plant liquid radioactive effluents. Appropriate holdup capacity is provided for retention of liquid effluents, particularly where unfavorable environmental conditions are expected to require operational limitations upon the release of radioactive effluents to the environment.

The system is capable of processing all wastes generated during continuous operation of the HBR2 primary system assuming that fission products escape from one percent of the fuel into the reactor coolant.

At least two valves must be manually opened to permit discharge of liquid waste from the WDS. One of these valves is normally locked closed. The control valve will trip closed on a high effluent radioactivity level signal.

6.3.1.2 System Description

Radioactive liquid wastes are first collected by various sump and collection tanks throughout the HBR2 auxiliary building and containment vessel. These wastes are received from either leakage from various radioactive systems or from being pumped from another source. From there, the wastes are pumped to the waste holdup tank for storage until they can be processed. Liquids from the waste holdup tank are processed using the Waste Water Demineralization System (WWDS). This system contains filters and demineralizer vessels for removing particulate and dissolved contaminates. The process water is stored in the Waste Condensate Tanks for ultimate offsite release.

A detailed description of the HBR2 liquid waste management system is contained in Section 11.2 of Reference 6.1.

6.4 SOLID WASTES

The ISFSI is not expected to generate any solid waste. Should any solid waste be generated, however, the existing HBR2 procedures will be used to process and dispose of the waste. Section 11.4.2 of Reference 6.1 describes the HBR2 solid waste processing system, the design basis, and an identification of the normal types of solid waste generated by HBR2 operations.

6.5 RADIOLOGICAL IMPACT OF NORMAL OPERATIONS - SUMMARY

The H. B. Robinson ISFSI is a passive and independent system which provides for the dry storage of irradiated fuel. The ISFSI will not cause a significant radiological impact; the ISFSI has no requirement for radwaste or auxiliary systems for normal operation. Liquid and gaseous effluents utilized in the loading and handling of the Dry Shielded Canisters will be handled using the plant's existing process systems. There will be essentially no solid, liquid, or gaseous wastes generated during normal operation of the ISFSI.

The existing HBR2 health physics program and procedures are followed to maintain exposures ALARA during routine surveillances of the ISFSI. The existing health physics procedures will be followed should there be a request to obtain any data during the testing phase of the ISFSI.

REFERENCE: CHAPTER 6

6.1 Carolina Power & Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

CHAPTER 7

RADIATION PROTECTION

7.0 RADIATION PROTECTION

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE ALARA

7.1.1 POLICY CONSIDERATIONS

The Carolina Power & Light Company (CP&L) corporate and facility (H. B. Robinson) health physics policies, described in Section 12.1.1 of Reference 7.1, are applicable to the Independent Spent Fuel Storage Installation (ISFSI). Carolina Power & Light Company is committed to a program of keeping occupational radiation exposure as low as reasonably achievable (ALARA). The Company follows the general guidance of Regulatory Guides 1.8, 8.8, 8.10, and publications which deal with ALARA concepts and practices, including Title 10, Code of Federal Regulations, Part 20.

The goals and objectives of the health physics programs are to maintain the annual dose to individual facility personnel to as low as reasonably achievable and to maintain the annual integrated dose to facility personnel; i.e., the sum of annual doses (expressed in man-rem) to all facility personnel, as low as reasonably achievable. The health physics programs identify the organizations participating in the programs, the positions involved, and the responsibilities and functions of the various positions in conducting the programs.

Adequate trained personnel are provided to develop and conduct all necessary health physics programs. The health physics personnel possess the necessary training and expertise to carry out the health physics programs in an efficient manner to assure that Company and regulatory requirements are met.

Appropriate training programs in the fundamentals of radiation protection and facility exposure control procedures are established to provide instructions to all facility personnel including contractors whose duties require working in radiation areas. Training programs for health physics personnel are provided to improve their performance in the health physics programs.

7.1.2 DESIGN CONSIDERATIONS

The designs of the dry shielded canister (DSC) and horizontal storage module (HSM) comply with 10 CFR 72.3 concerning ALARA considerations. Specific considerations that are directed toward ensuring ALARA are:

a) Thick concrete walls on the HSM to reduce the surface dose to an average of less than 20 mrem/hr.

b) Lead shield plugs on the ends of the canister to reduce the dose to workers performing the drying and sealing operations.

c) Use of a shielded transfer cask for handling and transportation operations involving loaded DSCs.

d) Fuel loading procedures which follow accepted practice and build on existing experience.

e) Recess in HSM front for cask docking to reduce scattered radiation during transfer.

f) Double seal welds on each end of DSC to provide redundant radioactive material containment.

g) Placing demineralized water in the cask annulus and DSC and sealing the DSC-cask gap to minimize contamination of the DSC exterior during loading.

h) Placing external shielding blocks over the HSM air outlets to reduce direct and streaming doses.

i) Passive system design that requires minimum maintenance.

j) Insertion of internal shielding blocks around air inlets to reduce direct and streaming doses.

k) Development of shipping procedures based upon previously used procedures and experience to control contamination during handling and transfer of fuel.

l) Use of additional shielding in front access cover plates.

7.1.3 OPERATIONAL CONSIDERATIONS

Operational considerations at HBR2 that promote the ALARA philosophy include the determination of the origins of radiation exposures, the proper training of personnel, the preparation of radiation protection procedures, the development of conditions for implementing these procedures, and the formation of a review system to assess the effectiveness of the ALARA philosophy.

Operational radiation protection objectives deal with access to radiation areas, exposure to personnel, and decontamination. Working at or near highly radioactive components requires planning, special methods, and criteria directed toward keeping occupational radiation exposure ALARA. Job training and debriefing following selected high exposure jobs contribute toward reduced exposures. Decontamination also helps to reduce exposure. Procedures and techniques are based upon operational criteria and experience that have worked to keep radiation exposure ALARA.

Procedures for the ISFSI will be integrated into current HBR2 procedures and will incorporate the same ALARA philosophy.

The ISFSI is considered a radiation control area. Therefore, a locked fence is located partially around the ISFSI. The HBR2 Radiation Control unit controls the key to this area.

7.2 RADIATION SOURCES

7.2.1 CHARACTERIZATION OF SOURCES

Table 3.1-2 lists the radiation sources to be stored at the Robinson ISFSI. The radiation source strength associated with the Robinson fuels to be stored are bounded by those of the NUTECH Topical Report (Reference 7.2). Due to the lower initial enrichment and the longer burnup, the Robinson fuel has slightly different neutron and gamma sources. The neutron source is slightly larger and the gamma source is slightly smaller. However, the shielding of the DSC and the HSM are sufficient to ensure that the dose limits are not exceeded. Therefore, the gamma and neutron shielding features described in the Topical Report are conservative for the HBR ISFSI.

7.2.2 AIRBORNE SOURCES

Loading the irradiated fuel into the DSC is carried out under water and is identical to the existing HBR2 loading procedures for the IF-300 cask. Potential airborne releases from irradiated fuel assemblies during handling operations prior to sealing the canisters are addressed in Section 15.7.5 of Reference 7.1.

The sealing and drying of the DSC are performed under procedures to prevent any gaseous leakage. All vent lines (liquid and gaseous) are routed to the existing HBR2 radioactive waste treatment facilities. Once the DSC is dried and seal welded, there are no design basis accidents which can cause breaching of the DSC and the airborne release of radioactivity. However, to demonstrate ultimate safety, a postulated leak is analyzed in Section 8.2.8 of Reference 7.2.

At the storage location, the only potential for airborne radioactivity is from any potential contamination of the DSC exterior. However, as described in Section 5.1 of the NUHOMS Topical Report, procedures are used to minimize this contamination and should limit any possible DSC surface contamination to below the limits specified in the Technical Specifications. There is essentially no onsite release of radioactivity from the DSC exterior and therefore does not present any health hazard.

7.3 RADIATION PROTECTION DESIGN FEATURES

7.3.1 INSTALLATION DESIGN FEATURES

The ISFSI is a passive outdoor storage system. Each HSM is capable of providing sufficient ventilation to ensure adequate cooling of the DSC and its contents. The convective cooling system is completely passive and requires no filtration system.

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 ALARA.

The DSC body is a section of 0.5 inch thick, 36 inch inside diameter stainless steel pipe. Two lead-filled end plugs and three steel plates provide shielding at the ends of the DSC. During handling operations, shielding in the radial direction is provided by the IF-300 shipping cask.

Two penetrations in the top lead 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. Table 7.3-1 lists relevant dimensions of the shielding materials present at the ends of the canister.

The HSM provides shielding in both the radial and axial directions during the storage phase. Forty-two inch thick, portland-cement concrete walls and roofs provide the shielding. The module's front access is covered by a two-inch thick composite plate which has additional shielding.

Four penetrations in the module allow convective air cooling of the DSC and module internals. Two identical intake vents at the bottom of the front HSM wall draw air into a shielded box inside the module. The exit vents are placed at both ends of the module roof. Openings to the HSM interior are placed above the end shield regions and not directly over the active fuel region. Sharp duct bends and precast concrete shielding caps over the exhaust exits assure that radiation streaming is reduced to a minimum. Figure 4.2-2 shows details of the module penetrations.

Further details of the radiation shielding design features are presented in Section 7.3.2 of Reference 7.2.

7.3.2.2 Shielding Analysis

The shielding analysis methodology for the NUHOMS generic design (Section 7.3 of Reference 7.2) is applicable to the HBR ISFSI as described in this section. However, due to the increased burnup and decreased enrichment of the HBR fuel, the neutron and gamma source terms are slightly different from those used for the generic design. Source terms are 11.4% lower for gamma rays and 17.2% higher for neutrons. This causes the dose rates calculated in Reference 7.2 to require scaling for use in this SAR. Also, because of the revised structure criteria and the necessity of fitting into an existing shipping cask, some steel and lead plate thickness on the DSC have changed. The HSM also has a rear ram access. These design changes required complete analyses of some of the HBR ISFSI shielding.

Figure 7.3-1 shows the locations at which dose rates are presented. Table 7.3-2 shows the resulting surface doses for HBR fuel. HBR ISFSI-unique shielding calculations have been performed for points shown in Figure 7.3-1. Dose rates at other points were scaled from the values in the Topical Report. The following paragraphs provide a brief description of the analysis at each point that was reanalyzed.

Due to the different design of the bottom shield plug on the DSC the shielding analysis of the HSM front air outlet was redone (points 2.1 and 3.1). This analysis used the DOT code to calculate the neutron and gamma dose rates at point 3.1. Hand calculation was then used to calculate the attenuation by the shield cap of the radiation beam coming out of the air outlet slot. The results are shown in Table 7.3-2.

Points 4, 5, and 7 provide estimates of the dose rate at the front of the module with the door open or closed. The dose rate at point 5 was obtained with the DOT analysis of the front half of the module (same analysis used for point 3.1). Dose rates of point 7 (4.5 feet away from the open door) and point 4 (surface of closed door) were obtained by hand calculations.

Because the dose at point 6 is primarily due to radiation leaving the canister longitudinal surface (and the design for the topical and the HBR canisters are identical for the canister body), the dose rate here was obtained by scaling the topical dose rates by the factors of the source strengths. The dose rates at points 8 and 9.1 were obtained by using the dose rate on the top cover plate surface of the DSC from the DOT analysis and a hand calculation to determine the attenuation due to the 3 1/2 feet distance to the outside HSM surface and the attenuation of the above plate (point 8). The dose rate at points 8 and 9.2 were obtained from a DOT analysis of the rear of the HSM with the canister half way in the HSM (a scoping study was done by hand albedo methods to determine that this was the case for the worse dose at the uncovered rear ram exit.

The dose rates for all points on the DSC top and transfer cask were calculated using the QADMOD (gamma only), DOT and the MORSE (Monte Carlo) programs. The three different codes were used to assure that the radiation streaming through the cask-canister gap was adequately modeled and estimated. The dose rates from all three methods yielded similar results (reduction factors of 10^{-3}). The QAD and DOT analysis methodologies were described in the Topical Report. The MORSE calculations are described below.

MORSE, a three-dimensional Monte Carlo shielding code (Reference 7.5), was used to assess the severity of the neutron and gamma ray streaming which occurs when the loaded DSC is inside the transport cask. The shielding calculations were performed in 22 neutron groups and 18 gamma ray groups with a P_3 expansion of the angular distributions using a coupled cross section set. Thus, secondary gamma rays are included when primary neutrons are taken as the source.

The MORSE model was constructed in three dimensions using the MARS geometry package. The PICTURE code was used to verify the model. One octant of the cask/canister system was modeled with reflective boundaries at symmetric planes. Fuel assemblies were modeled discretely as two-zone regions containing an outer layer of (homogenized) 0.125 inch thick Boral and a homogenized interior composed of irradiated fuel and cladding. The upper two spacer disks were modeled discretely due to their significant effect of local dose rates in the area of interest. Additional spacer disks were omitted to reduce computational time and ensure conservatism. The lead region in the top shield plug was modeled as a disk with reduced diameter to account for the steel siphon line region. The cask/canister system was modeled with the nominal 0.25 inch annular gap. A boundary crossing routine was employed to determine the average dose rate in the annular gap containing air only. The choice of boundary crossing, rather than point detector collision estimating, was made in order to allow octant modeling and to improve the statistical deviation.

The source terms were obtained in a fashion similar to the NUHOMS Topical Report and will not be discussed here. Russian Roulette, path length stretching, and source energy biasing were all used to minimize statistical deviation in the area of interest. Russian Roulette weighting parameters were established based on the number of mean free paths from significant sources to the cask/canister gap. Path length stretching was in the forward direction. Adjoint XSDRNPM-S (Reference 7.5) calculations were used to determine source energy biasing parameters.

Dose rates were calculated from the fluxes by using the Snyder-Neufeld factors for neutrons and the Henderson factors for gamma rays (Reference 7.8).

The application of MORSE to the cask/canister streaming problem represents a refinement in the albedo technique used in the NUHOMS Topical Report. Sixty-four thousand, eight hundred neutron histories and 1,920,000 primary gamma ray histories were executed to obtain streaming dose rates. The neutron dose rate (which includes a negligible contribution due to secondary gamma rays) was calculated to a one-sigma deviation of 6.6%. The primary gamma ray dose rate was calculated to a one-sigma deviation of 11.9%

The dose rates reported for the DSC in the cask include numerous combinations of the presence of water in the DSC and DSC-cask gap (the gap hereafter). These combinations are present due to the operational procedures. The various cases are described below.

Point 1.1, lead plug on DSC with water in the DSC and gap is the condition during the welding of the lead plug assembly to the DSC. After welding the DSC will be drained, dried and backfilled with He. The calculations for point 1.2 reflect the estimates for the dose rate after the DSC is drained. However, it should be noted that personnel will not be required to be directly over the canister during this time.

All operations will be done from the side of the cask and only the forearms and hands of the personnel will be over the cask for a short time during the connecting and disconnecting (with "quick connect" Swagelok fittings) of water, air and He lines. The dose rate in the gap during this initial welding and drying is shown for point 3.1. Point 2 gives the dose rates on the top cover plate during the welding of that plate to the DSC body. The dose rate in the gap is given for point 3.2.

After the top cover plate is welded on the DSC, the cask lid will be lowered into position, bolted on, and then the water drained from the gaps. Point 5 gives the estimated dose rate for the top of cask lid.

Point 4 gives the dose rate at the cask side where the operating personnel will be working.

Point 6 gives the dose rate at the cask collar side during the insertion of the DSC. While the DSC is being seal welded, the lead plug is next to the inner surface of the cask collar and, hence, the dose at the outside of the collar is small. The large dose rate shown for point 6 is only present during the loading of the DSC into the HSM. This dose rate is present over a 6-inch wide ring of the collar which is not inserted inside the HSM cask docking recess. Although no personnel will be within 20 feet of this area during loading, it would be prudent, from an ALARA standpoint, to use portable lead/ polyethelyne shielding to reduce the dose rate in the vicinity of the HSM/cask interface.

7.3.3 VENTILATION

The HSM has a ventilation system to provide for natural draft cooling of the DSC. However, no off-gas or filtered treatment system is required due to the low exterior contamination level of the DSC.

The ISFSI is designed for essentially no release of radioactive material during normal storage of the DSC in the HSM. Therefore, additional design work and equipment would not result in a reduction of radioactive materials released. Furthermore, no credible site accident will result in a radioactive leak because of the integrity of the double seal welds at each end of the DSC, the passive nature of the system, the operational limits and controls used during handling and the integrity of the stainless steel body of the canister.

7.3.4 RADIATION MONITORING INSTRUMENTATION

The operation of the ISFSI will be monitored under the HBR2 radioactivity monitoring program. No additional radiation monitoring instrumentation is required.

TABLE 7.3-1

DSC END SHIELDING MATERIAL THICKNESSES

DSC Top End[1]

Shields[1]	
Lead Shield Plug	0.50 in. Steel + 3.5 in. Lead + 1.75 in. Steel
Cover Plate	1.25 in. Steel

DSC Bottom End

Shields[1]	
Pressure Plate	2.00 in. Steel
Lead Shield Plug	0.25 in. Steel + 4.75 in. Lead

1 "Top" and "Bottom" refer to the top and bottom ends of the irradiated fuel assemblies.

TABLE 7.3-2

SHIELDING ANALYSIS RESULTS

ISFSI Location	Nominal Surface Dose Rates (mrem/hr) Neutron	Gamma	Total
DSC in HSM			
1. HSM Wall or Roof	0.03	2.5	2.5
2. HSM Air Outlet Shielding Cap	0.03	10	10
2.1 Front HSM Shield Cap	0.06	103	103
2.2 Rear HSM Shield Cap	0.06	10	10
3. HSM Air Outlet (no shielding cap)			
3.1 Front HSM Air Outlet (no shielding cap)	3.0	4450	4450
3.2 Rear HSM Air Outlet (no shielding cap)	1.6	440	440
4. Center of Door	35	81	116
5. Center of Door Opening	50	378	428
6. Center of Air Inlets	27	29	56
7. 4.5 Ft. from HSM Door	7.1	17	24
8. Ram Opening with Access Plate (fully inserted DSC)	2.9	0.37	3.3
9. Ram Opening Without Access Plate			
9.1 Fully Inserted DSC	3.8	0.92	4.7
9.2 Half Inserted DSC	1.2	605	606
DSC in Cask			
1. Centerline Top of DSC Plug			
1.1 No Water in DSC, Water in Gap	296	92	390
1.2 Water in DSC and Gap	0.83	36	37
2. Centerline Top of DSC Cover Plate (no water in DSC, water in gap)	252	62	314
3. Cask/Canister Annular Gap			
3.1 Water in Gap and DSC	0.34	390	390
3.2 Water in Gap (no water in DSC)	190	365	555
4. Transfer Cask Side Surface (no water in DSC)	3.3	2.8	6.1

TABLE 7.3-2 (Continued)

5. Transfer Cask Top Cover Surface	194	24	218
6. Cask Collar During DSC Transfer from Cask to HSM	37	620	657

Note:

1. These values for worst case situation where no water is present in the gap. For ALARA purposes, water should be present in the gap which will reduce streaming exposures by approximately 1/20.

7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENT

Estimated doses for the fuel loading, drying, sealing and transfer are provided in Section 9.1.2 of the HBR2 Updated FSAR (Reference 7.1) and Section 7.4.1 of the NUHOMS Topical Report (Reference 7.2). Onsite radiation dose rates due to the storage of the fuel in the first three HSMs are shown in Figure 7.4-1. If all eight modules are built and filled, the dose rate will be 8/3 times the values listed in Figure 7.4-1. The resulting dose for a person located at the ISFSI fence for eight hours a day for 250 days per year would be approximately 10 mrem. For a person located at the offices, the yearly dose would be less than 1 mrem. (Actual dose inside the office is less due to shielding from the building). If an additional 176 modules are built and filled, the onsite radiation dose rates due to the storage of fuel in the added modules are shown in Figure 7.4-1a.

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 ISFSI module. Chapter 5 describes in detail the ISFSI operational procedures, a number of which involve radiation exposure to personnel.

A summary of the operational procedures which result in radiation exposure to personnel is given in Table 7.4-1. The cumulative dose is calculated by a four step process. First, the number of personnel required to perform the task is estimated. Then the time required to perform the task is estimated based on the operational guidelines presented in Chapter 1, engineering judgement, and previous experience with similar or identical operations. An ambient dose rate is obtained for the operation. It is based on an estimated distance from the individual's trunk to the most significant radiation source. The dose rate is conservatively calculated by modeling radiation sources as line, cylinder, or plane sources, as is appropriate to the particular geometry of the operation. With the number of personnel, the time required, and the local dose rate, individual and collective exposures may be calculated.

7.4.2 STORAGE TERM DOSE ASSESSMENT

Figure 7.4-2 is a graph of the dose rate versus distance from the face of a filled storage array of three ISFSI modules. Direct neutron and gamma flux, as well as the air-scattered radiation from the module surfaces, are considered. Air-scattered dose rates are determined with the computer code SKYSHINE-II (Reference 7.3). Initial loading of all modules with the NUHOMS Topical Report design basis five-year post irradiated fuel is assumed. If five additional modules are added, the dose rate will be 8/3 times that shown in Figure 7.4-2.

Estimates of cumulative doses to site personnel from three filled ISFSI modules are given in Table 7.4-2. The dose rates are based on data shown in Figure 7.4-2. Occupancy information for number of personnel, location, and time is estimated based on the five-year plan for facilities' layout at the HBR2 site. Because of the very rapid decrease of dose rate with distance from the storage facility, a maximum distance of 600 feet was used in these analyses. No credit is taken for shielding of personnel by buildings, or for radioactive decay.

TABLE 7.4-1

SUMMARY OF ESTIMATED ONSITE DOSES DURING FUEL HANDLING OPERATIONS

	Number of	Time	Average Distance From Source Surface	Dose Rate	Dose Per Person	Total Personnel Dose

Operation	Personnel	(Hours)	(Feet)	(mrem/hr)	(mrem)	(mrem)
Location: Fuel Pool						
Load fuel into canister	2	8.0	-	5	40	80
Bolt lid assembly onto cask	2	0.5	1.5	22.3	11.2	22.3
Location: Cask Handling Area						
Decontaminate outer surface of cask	3	8.0	-	6.1[(2)]	48.8	146.4
Place scaffolding around cask	4	0.8	4.0	9.5	7.6	30.4
Unbolt lid, remove lid and spacer	2	0.75	1.5	22.3	16.73	33.5
Remove approximately 15 gal. of water from DSC and lower water level in canister cask gap	2	0.5	1.5	22.3	11.2	22.3
Weld lead plug to canister	2	3.0	4.0	3.5	10.5	21.0
Hydrotest canister	2	2.0	1.5	22.3	44.6	89.2
Remove water from canister[(1)] cavity	2	2.3	4.0	29.8	68.5	137.0
Seal weld prefabricated plug to siphon tube connection	2	0.5	1.5	210	105	210
Vacuum dry canister and[(1)] backfill with Helium	2	4.0	4.0	29.5	118.0	236
Helium leak test weld	2	1.0	1.5	210	210	420
Seal weld prefabricated plug to vent tube	2	0.5	1.5	210	105	210
Perform NDE (PT)	1	1.0	1.5	210	210	210
Install end cap	2	0.5	1.5	210	105	210
Weld end cap to canister	2	2.3	4.0	29.5	67.9	135.8
Install cask lid and bolt into place	2	0.5	1.5	118.3	59.2	118.3

TABLE 7.4-1 (Continued)

Remove scaffolding from around cask	4	0.8	4.0	9.5	7.6	30.4
Transport cask to skid and trailer	2	0.5	-	2	1	2
Location: Trailer						
Attach skid tie down to cask	2	0.5	1.0	6.1[(2)]	3.1	6.1
Transport cask to HSM	5	0.5	5	6.1[(2)]	3.1	15.5
Remove cask lid	2	0.5	1.5	118.3	59.2	118.3
Install cask jacking system and align cask with HSM	4	1.5	3.0	6.1[(2)]	9.3	37.2
Transfer canister from cask to HSM	4	0.5	3.0	6.1[(2)]	3.1	12.4
Install steel plate over front access of HSM	2	0.5	3.0	2.5[(2)]	1.3	2.6
Tack weld front access door	2	0.5	1.5	70.2	35.1	70.2
Install seismic retainer assembly	2	0.5	1.0	4.7[(2)]	2.4	4.8
Install cover plate to rear access	2	0.5	1.5	2.5[(2)]	1.65	3.3
TOTAL		42.95			1366	2635

(1) Monitoring operation - personnel could leave radiation field.
(2) Conservatively assumed to be surface dose rate.

TABLE 7.4-2

ESTIMATED ANNUAL ONSITE DOSES DURING STORAGE PHASE

Area	Number of Personnel	Time (Hours/yr)	Average Distance From Facility (Feet)	Dose Rate (mrem/hr)	Dose Per Person (mrem/yr)	Total Personnel Dose (mrem/yr)
E&RC Building	30[1]	2080	560	1.8×10^{-3}	3.7	110
Operations and Maintenance (O&M) Building		225[1]	2080	360	4.3×10^{-3}	8.9 2000
Yard Work Area[12]	13[1]	2080	120	4.9×10^{-2}	100.0	1300
Yard Work Area[2]	12[1]	2080	350	4.4×10^{-3}	9.2	110
TSC/EOF Building	25[1]	2080	580	1.6×10^{-3}	3.3	83
Primary Access Point	10[2]	8760	290	7.5×10^{-3}	66.0	660
Bulk Storage Warehouse	25[1]	2080	360	4.3×10^{-3}	8.9	220
Daily Visual Inspection	1	60	4	26	1560.0	1560
HSM Interior Inspection	1	0.5	1.5	4.0	2.0	6045

[1]One 8-hour shift per day, 5 days per week.

[2]Continuous occupancy (44 people at one 8-hour shift per day, 5 days per week, 50 weeks per year).

7.5 HEALTH PHYSICS PROGRAM

Appropriate health physics programs are established for all Company operations which deal with radiation. The programs are consistent with the corporate health physics policy and all applicable regulations. The Nuclear Assessment Section periodically evaluates the various health physics programs and other Company activities which have impacts on the programs and reports to senior management regarding the effectiveness and adequacy of the programs. The Nuclear Assessment Section makes recommendations to senior management, as necessary, to maintain effective overall health physics programs.

The existing HBR2 health physics program is applicable to the ISFSI. The HBR2 program has been established to provide an effective means of radiation protection for plant personnel, visitors, and the general public. Section 12.5.1 of Reference 7.1 describes the HBR2 health physics program.

7.5.1 ORGANIZATION

As discussed in Section 9.1.2, the HBR2 organization provides the Radiation Control Supervisor access to the General Manager through the Manager - Environmental and Radiation Control. This organization allows the Plant General Manager to be involved in the review and approval of specific ALARA goals and objectives as well as review of data and dissemination of information related to the ALARA program.

The organization also provides the ALARA Analyst, who is normally free from routine health physics activities, to implement the plant's ALARA program. This individual is primarily responsible for coordination of plant ALARA activities and routinely interfaces with first line supervision in radiation work planning and post-job review.

7.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

The ISFSI will utilize the equipment, instrumentation, and facilities of HBR2 as necessary. Section 12.5.2 of Reference 7.1 describes the HBR2 equipment, instrumentation, and facilities.

7.5.3 PROCEDURES

The ISFSI will utilize existing HBR2 health physics procedures. These procedures are discussed in Section 12.5.3 of Reference 7.1. Health physics procedures specific to the ISFSI will be incorporated into the existing HBR2 procedures. In particular, radiation surveys of the ISFSI will be conducted on an annual basis. Additional health physics/radiation protection procedures required for use during the ISFSI test program will be developed as needed and will comply with the existing HBR2 program.

7.6 ESTIMATED OFFSITE COLLECTIVE DOSE ASSESSMENT

Because the ISFSI provides containment yielding essentially no radioactive gaseous or liquid effluents, assessment of offsite collective dose is limited to one of direct and reflected radiation to the nearest residence.

7.6.1 EFFLUENT AND ENVIRONMENTAL MONITORING PROGRAM

The ISFSI is located within the protected area of HBR2. The HBR2 environmental program is described in the Offsite Dose Calculation Manual, Section 4.0 (Reference 7.4). In addition, environmental TLDs are maintained at air sampling sites adjacent to the plant boundary. These are located at 170 degrees, 830 feet and 150 degrees, 1500 feet from the ISFSI. The nearest residence is located at 160 degrees, 1350 feet from the facility. These TLDs are changed quarterly.

7.6.2 ANALYSIS OF MULTIPLE CONTRIBUTION

An analysis of multiple contribution was performed in order to determine the radiological impact the ISFSI will impose on the population surrounding the H. B. Robinson plant. This impact added to contributions made by other uranium cycle facilities were compared to the natural background radiation and the regulatory requirements of 40CFR190.

The maximally exposed member of the public would receive approximately 1.6 mrem per year from an ISFSI made up of a three-unit HSM (reference Figure 7.6.1). An ISFSI consisting of an eight-unit HSM would contribute approximately 4.3 mrem per year. This is a result of external radiation only; there are no gaseous, particulate, or liquid effluents associated with the normal operation of the ISFSI. It can be concluded that the actual exposure contribution from the ISFSI along with the total of all other uranium fuel cycle activities is within the regulatory limits set forth in 40CFR190.

7.6.3 ESTIMATED DOSE EQUIVALENTS

During the normal operation of the ISFSI, there are essentially no effluents released.

The cumulative dose for the population (due to the ISFSI) integrated over 10 regions out to 50 miles radially from the ISFSI, is less than 2.0 man-rem per year. The currently accepted national average background radiation level is approximately 300 mrem/year per person which includes 200 mrem/year per person for radon.

REFERENCES: CHAPTER 7

7.1 Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

7.2 NUTECH Engineers, Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-001, Revision 1, November 1985.

7.3 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.

7.4 Carolina Power & Light Company, H. B. Robinson Steam Electric Plant, Unit No. 2 Offsite Dose Calculation Manual (ODCM),@ Docket No. 50-261.

7.5 Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," ORNL/NUREG/CSD-3.

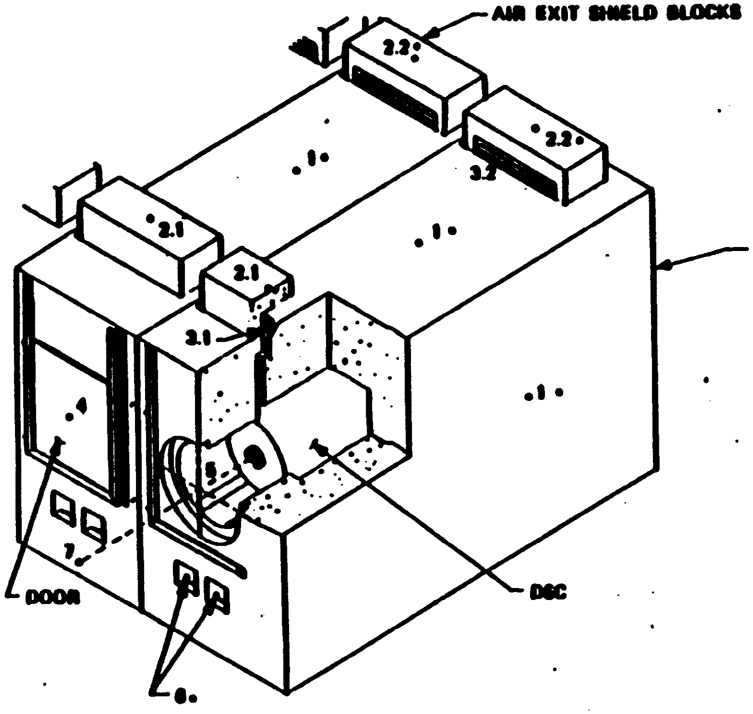
7.6 Oak Ridge National Laboratory, "ANISN-ORNL Multigroup One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," RSIC CCC-254, 1973.

7.7 J. H. Price, W.G.M. Blattner, "Utilization Instructions for QADMOD-G," RRA-N7914, RSIC CCC-396, 1979.

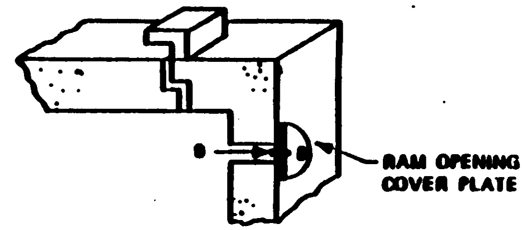
7.8 Oak Ridge National Laboratory, "Cask 40 Group Coupled Neutron and Gamma-Ray Cross-Section Data," RSIC DLC-23, 1978.

7.9 Grove Engineering, "MICRO SKYSHINE User's Manual," July 15, 1987, Grove Engineering, Inc., Rockville, MD.

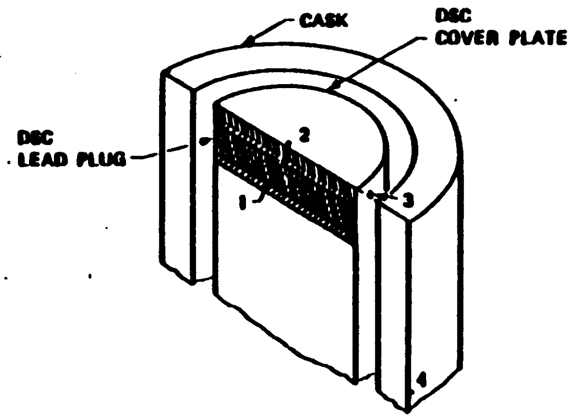
7.10 Grove Engineering, "MICROSHIELD User's Manual," Version 3, August 1988, Grove Engineering, Inc., Rockville, MD.



HSM DOSE RATE LOCATIONS



RAM OPENING DOSE RATE LOCATIONS



DSC-CASK DOSE RATE LOCATIONS

* INDICATES LOCATION WHERE TOPICAL REPORT VALVES HAVE BEEN SCALED TO PRODUCE REPORTED RESULTS.

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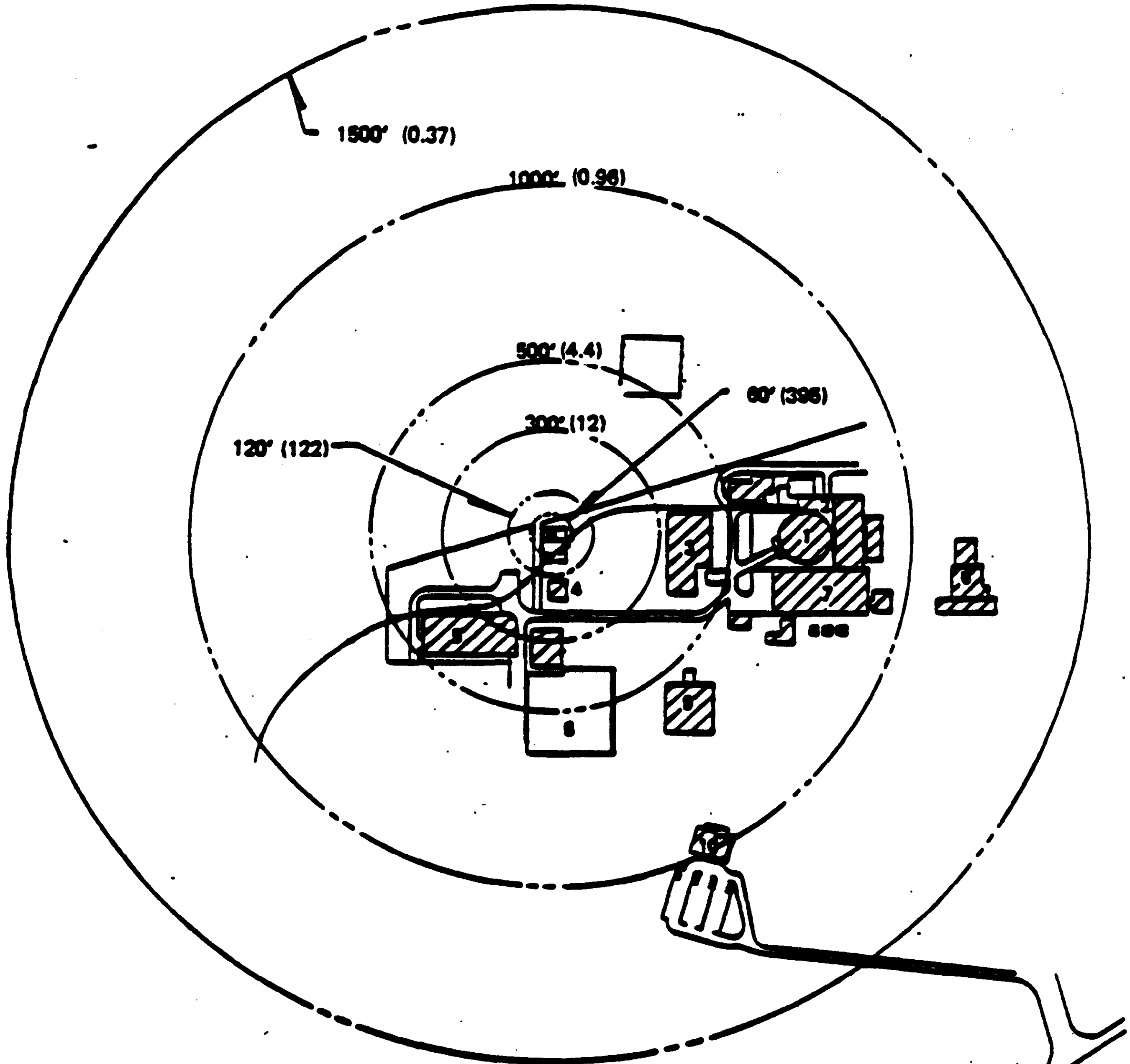
LOCATIONS OF REPORTED
DOSE RATES (TABLE 7.3-2)

Figure 7.3-1



1. REACTOR CONTAINMENT BLDG
2. FUEL BLDG
3. OPERATIONS & MAINTAINANCE BLDG
4. STEAM GENERATOR STORAGE BLDG
5. BULK WAREHOUSE
6. UNIT ONE COAL PLANT
7. TURBINE BAY
8. PARKING
9. NEW TEC/EOF BLDG
10. VISITORS CENTER

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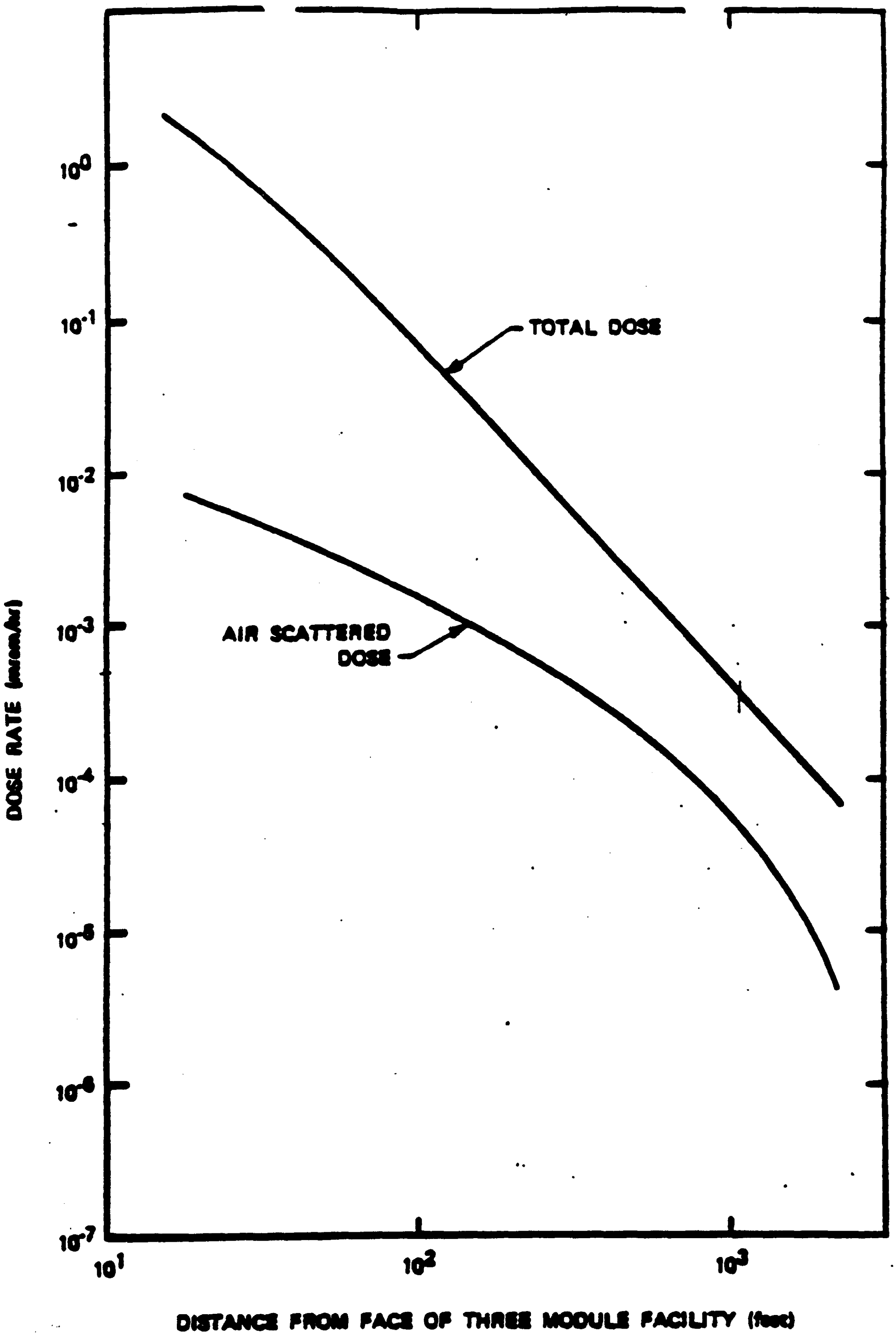
ANNUAL DOSE (mrem/yr) FROM
3 HSMs (Assuming 2080 hours/yr)
Figure 7.4-1



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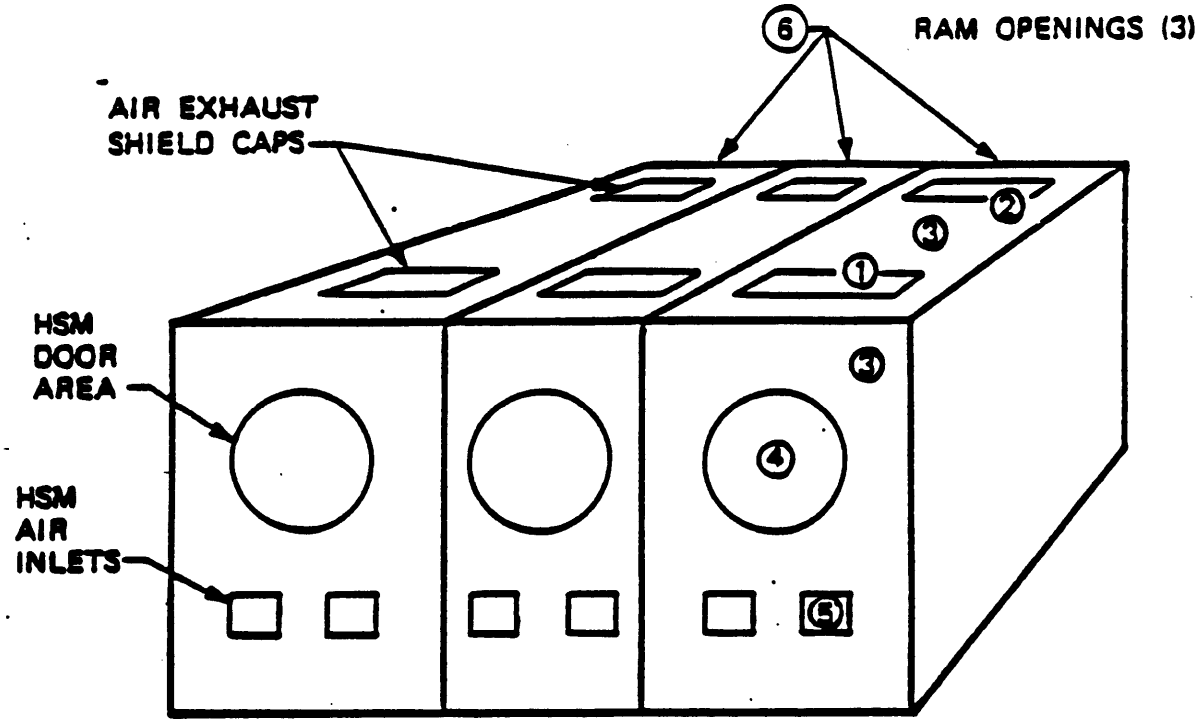
DOSE RATE VS. DISTANCE FROM
SURFACE OF HSM (Assuming 3 Modules)
Figure 7.4-2



LOCATION		AREA (Ft^2)	DOSE RATE (mrem/hr) NEUTRON	GAMMA
ROOF	1	21.5	0.06	103
	2	21.5	0.06	10
	3	501.5	0.03	2.5
		AREA WEIGHTED AVE.	0.032	6.76
FRONT	3	278.5	0.03	2.5
	4	24.9	35	81
	5	6.0	27	29
	6	3.2	2.9	0.37
		AREA WEIGHTED AVE.	3.37	9.33

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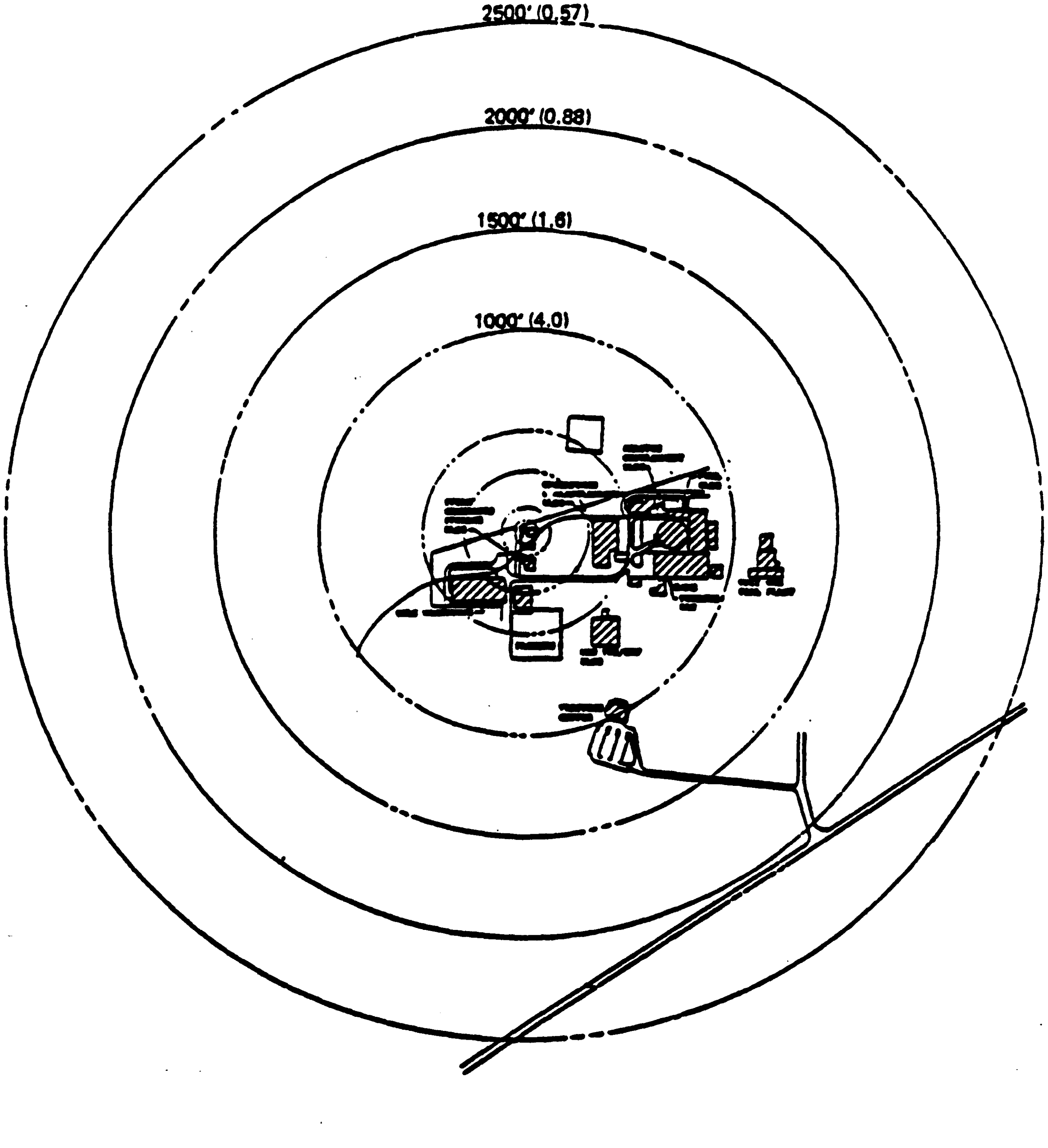
RADIATION ZONE MAP OF
MODULE SURFACE DOSE RATES

Figure 7.4-3



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ANNUAL OFFSITE DOSE (mrem per year) FROM 3 HSMS (based on 24 hours a day, 365 days a year)
Figure 7.6-1

CHAPTER 8

ACCIDENT ANALYSES

8.0 ANALYSIS OF DESIGN EVENTS

The purpose of this section is to evaluate the safety of the H. B. Robinson (HBR) Independent Spent Fuel Storage Installation (ISFSI). The safety evaluation is accomplished by analyzing the response of the various components of the ISFSI to normal and off-normal conditions and a range of credible and hypothetical accident conditions.

In accordance with NRC Regulatory Guide 3.48, design events identified by ANSI/ANS 57.7-1981, are used in the safety evaluation of the ISFSI. In ANSI/ANS 57.7-1981, four categories of design events are defined. Design events of the first and second type are addressed in Section 8.1, and design events of the third and fourth type are addressed in Section 8.2 of this report.

Many of the design events in the above four categories have been addressed in the NUTECH Horizontal Modular Storage (NUHOMS) System Topical Report (Reference 8.1) using enveloping criteria. Whenever the site specific load is enveloped by that of the NUHOMS Topical Report, it will be noted and will reference the appropriate section of the Topical Report. Additional site specific analysis which has not been covered in the NUHOMS Topical Report will be discussed in detail in the following sections.

As discussed in Section 3.2 of this Safety Analysis Report (SAR), some design features of the HBR ISFSI are unique and differ from those of the NUHOMS generic concept. In particular, the HSM is a three module unit with a rear access penetration, whereas the generic concept is an eight module unit without any rear access. However, as discussed earlier the methodology of the structural evaluation of the HSM under the above categories of design events as utilized by the referenced report is such that it will conservatively envelop any modular stacking arrangement including the three unit concept of the HBR site. Hence, the stress evaluation and the analytical results presented in Chapter 8 of the referenced report for the NUHOMS modules are fully applicable to the site specific HSM.

Some design features of the HBR DSC are also different than those of the NUHOMS generic concept. Specifically the DSC is designed to withstand inertia forces associated with cask drop accidents in which the drop height is significantly higher than the soft drop criteria established earlier in this report. Because of these design features, additional structural evaluation of the DSC is required. The method of analysis, however, for many of the design event cases is the same as the methodology utilized in the NUHOMS Topical Report. For these cases the appropriate sections of the referenced report containing the applicable methodology will be referenced. In other cases where a new methodology is utilized, such as the drop accident case, the analytical approach will be presented. In either case the resulting DSC stress evaluation will be tabulated and reported throughout this chapter.

The design of the DSC support assembly for the HBR ISFSI is identical to the NUHOMS generic concept and as such the stress evaluation presented in the referenced report is fully applicable to this component.

Since a foundation design was not included in the NUHOMS Topical Report, Section 8.3 is included in this Safety Analysis Report (SAR) to describe the foundation design and analysis using the four categories described above.

As described earlier in this report two of the DSCs will be instrumented for the purpose of collecting data. Section 8.4 of this report addresses the safety features of the instrument penetration.

8.1 NORMAL AND OFF-NORMAL OPERATIONS

Design events of the first type consist of a set of events that occur regularly in the course of normal operation of the ISFSI. These events are addressed in Section 8.1.1 of this report. 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). These off-normal events are addressed in Section 8.1.2 of this report.

8.1.1 NORMAL OPERATION ANALYSIS

The loads associated with the normal operating condition of the ISFSI are as follows: dead weight loads, design basis internal pressure loads, design basis operating temperature loads, operation handling loads, and design basis live loads. The structural components effected by these loads are the dry shielded canister (DSC), DSC internals, horizontal storage module (HSM), DSC support assembly and the foundation. The following paragraphs discuss these loads and compare them to the generic assumptions reported in Section 8.1.1 of the NUHOMS Topical Report (Reference 8.1).

a) Dead Weight Loads - Dead weight analysis contained in Chapter 8 of the NUHOMS Topical Report for the HSM and the DSC support assembly envelops the Robinson ISFSI analysis. Hence, the analysis of dead weight in the Topical Report is applicable to the HBR ISFSI analysis for these components.

The DSC component weights are tabulated in Table 8.1-1. The dead weight analysis of the DSC shell is based on the same analytical approach specified in Section 8.1.1.2, Page 8.1-17 of the referenced report. Furthermore, since the total weight of the DSC is approximately the same as that of the NUHOMS generic DSC, the resulting DSC shell stresses are the same. For the dead weight analysis of the spacer disk the results of the finite element analysis reported in Section 8.1.1.3, page 8.1-32 of the referenced report can be directly ratioed for the effect of the weight redistribution and the change in the spacer disk thickness. The site specific spacer disks are 2 inch thick compared to the 1.25 inch of the NUHOMS spacers. Also, the maximum total weight distributed on one spacer is 2034 pounds compared to the 1834 pounds for the NUHOMS. Based on these differences the maximum resulting stress reported in Table 8.1-7 of the referenced report can be ratioed by the relation:

$$S_{SP} = (S_{NU}) \quad \frac{(W_{SP})}{(W_{NU})} \quad \frac{(t_{NU})}{(t_{SP})}$$

Where:

S_{sp} = ksi, the site specific spacer disk membrane stress

S_{NU} = 1.58 ksi, the NUHOMS spacer disk membrane stress

W_{sp} = 2,034 lb, weight per site specif. spacer disk

W_{NU} = 1,823 lb, weight per NUHOMS spacer disk

t_{NU} = 1.25 in, NUHOMS spacer disk thickness

t_{sp} = 2.0 in, site specific spacer disk thickness

Therefore

S_{sp} = 1.10 ksi

The results of the above analyses are tabulated in Table 8.1-2 of this report. The stresses caused by the weight of other components of the DSC and its internals are insignificant and do not warrant extra analysis.

b) Design Basis Internal Pressure Loads - The HBR DSCs are operated with 0.0 psig pressure. However, the DSC is designed for 25.0 psig operating pressure at off-normal conditions. This pressure is the same as that specified in the referenced report. Since the HBR DSC has the same shell thickness as the NUHOMS, the resulting primary membrane stress will remain uneffected. However, for the secondary stresses at the discontinuities the analysis reported in Section 8.1.1.2, Pages 8.1-21 through 8.1-24 of the referenced report is reworked to incorporate the change in the effective thickness of the cover plates. The NUHOMS analysis is based on an effective thickness of 1.5 inch, whereas the minimum available cover plate thickness of the site specific DSC is 1.75 inch. With all other conditions and assumptions being identical, the analysis yields a maximum secondary membrane plus bending stress of 7.64 ksi.

For the bending stress on the cover plate itself, the result of the analysis contained in Pages 8.1-24 and 8.1-25 of the referenced report is multiplied by the square of the ratio of the thicknesses. In this manner, the maximum bending stress on the 1.75 inch thick cover plate is 3.27 ksi.

The results of the above pressure analysis of the DSC and comparison against code allowables are contained in Table 8.1-2 of this report. The maximum DSC internal pressure under accident conditions is 39.7 psig, which is the same as that specified in the NUHOMS Topical Report.

c) Design Basis Operating Temperature Loads - The extreme range of ambient temperature at the Robinson site is -5^{o}F to 105^{o}F. For the NUHOMS Topical Report design, a range of -40^{o}F to 125^{o}F was assumed. Consequently the thermal analyses of the HSM and the DSC support assembly reported in Sections 8.1.4 through 8.1.5 of the NUHOMS Topical Report conservatively envelopes those of the HBR ISFSI.

The DSC thermal analysis contained in Section 8.1.1.2 of the referenced report conservatively envelopes the site specific DSC thermal analysis. This is due to the fact that the maximum shell bending stress reported in that report is based on the generic assumption that no gaps exist between the spacer disk and the inside cavity of the DSC (Section 8.1.1.2, Pages 8.1-26 through 8.1-28). The HBR DSC, however allows for a nominal radial gap of 0.13 inch. This amount of gap is larger than the differential thermal expansion of the disk. Other thermal stress evaluations of the NUHOMS canister, such as the shell stress evaluation due to temperature variation in circumferential direction, and due to dissimilar material, indicated stresses far below the 20.9 ksi obtained for the case discussed above. Hence, for the sake of conservatism and in order to envelope the actual state of thermal stress in the HBR DSC, the thermal stress obtained from the differential expansion of the spacer disk will be reported herein and is tabulated in Table 8.1.2 of this report.

For the thermal expansion evaluation of the DSC internals, the evaluation reported in Section 8.1.1.3, Pages 8.1-33 and 8.1-34 of the referenced report is also fully applicable. This is due to the fact that the gap existing between the top of the fuel region and the bottom of the HBR DSC lead plug is the same as that reported in the referenced report.

d) Operation Handling Loads - The handling loads on the DSC, DSC support assembly, and the HSM are based on the maximum capacity of the hydraulic ram of 22000 pounds. This capacity is the same as that specified in the NUHOMS Topical Report. Therefore, the handling load analysis of the Topical Report, Section 8.1.1, covers the site specific design. Since the ram mounting plate assembly at the rear access of the HSM is site specific, the loading from the ram on this assembly was investigated. The ram loads are transferred to the wall through the embedded pipe and plate which have welded stud anchors. The 22000 pound loading was found to have a negligible effect on the HSM rear wall. The net effect of the tornado-generated missile impact considered in the topical report is to load the side wall with over 1000 kips. The much narrower end wall, during operational loading, is easily enveloped by the previous analysis. The results of the operational handling load analysis of 22000 pounds are tabulated in Table 8.1-2 of this report.

e) Design Basis Live Loads - The maximum snow load (or other live loads) for the Robinson site as derived from the Updated FSAR (Reference 8.2) is bounded by the NUHOMS Topical Report which assumes a live load of 200 psf.

8.1.2 OFF-NORMAL OPERATION ANALYSIS

This section describes the design basis off-normal events associated with the operation of the HBR ISFSI. The events which are considered here are expected to occur on a moderate frequency.

8.1.2.1 Transport

Off-normal events associated with the transport operation of the DSC may occur due to malfunctioning of the auxiliary components (i.e., crane, transporter ram, etc.), or by misalignment of the DSC with respect to the HSM. Malfunctioning of the auxiliary components does not relate to the safe functioning of the DSC and can be rectified without any impact to the operation of the system. As described in Section 1.3.1.7 of this report the only time the cask crane is operating without the redundant yoke is during the cask lowering on the skid assembly. A postulated malfunction or more specifically a yoke failure during this operation is considered as part of the cask drop accident which is reported in Section 8.2.4 of this report. The DSC and the ram grappling assembly are designed to the maximum ram capacity loading of 22,000 pounds. Hence, any off-normal event such as misalignment or ram malfunctioning will not cause any damage to any component of the ISFSI. The misalignment of the DSC may also cause jamming or binding of the canister casing. The analysis of the DSC under assumed jamming and binding conditions is covered in Section 8.1.2 of the NUHOMS Topical Report (Reference 8.1), and is applicable to Robinson ISFSI operation.

All auxiliary components used during the transport operation (i.e., the cask positioning skid, the cask tie-down system, the cradle support, the saddle and the transporter) are designed to withstand the inertia forces associated with transport shock loadings. The DSC and the cask are designed for the postulated drop accident. The inertia forces of a drop accident is significantly greater than the transport shock forces and, hence, inertia forces associated with transportation shock for these components are enveloped by the 8 ft drop accident.

8.1.2.2 Air Flow Blockage

Another off-normal event that may occur is the possibility of air inlet blockage. Because the air inlets are close to the ground, there is a chance that they could become blocked with blowing paper, dirt, snow or other debris. Due to the height of the air outlets, their separation and since hot air is blowing out of the exits, it is less likely that both the exits would become blocked. Furthermore, blockage of one exit alone would not be as severe as blockage of both inlets. Therefore, this off-normal event is defined as complete blockage of the HSM inlets. Blockage of all inlets and outlets is considered highly unlikely and is presented in Section 8.2 of this SAR and in Section 8.2 of the NUHOMS Topical Report.

The blockage of the air inlets has been addressed in the NUHOMS Topical Report in Section 8.1.2. The results of this analysis indicate that the rise in temperature and pressure in various components of the storage system is well within the acceptance limits. The blockage of the air inlets would be discovered during the normal surveillance of the modules. As the analysis shows, excessive temperatures are not reached and, hence, if the blockage were to occur just after one inspection and not be discovered until 24 hours later, no threat to the public health and safety would result. Once detected, the air inlets will be cleared of the blockage.

8.1.3 RADIOLOGICAL IMPACT FROM OFF-NORMAL OPERATIONS

Based on the off-normal operations described in Section 8.1.2, there is no additional radiological impact from the ISFSI beyond what is described in Chapter 7.

TABLE 8.1-1

DRY SHIELDED CANISTER AND HORIZONTAL STORAGE MODULE COMPONENT WEIGHTS

COMPONENT DESCRIPTION	CALCULATED WEIGHT (Pounds)
1. Dry Shielded Canister:	
Casing	2849
Top Grapple Assembly	49
Top Cover Plate	357
Top Lead Casing	617
Top Lead Plug	1307
Top Ring Plate	60
Bottom Cover Plate	605
Bottom Lead Casing	152
Bottom Lead Plug	1555
Total	7551
2. Canister Internals:	
Spacer Disks	1591
4 x 2 1/2" Ø Support Rods	897
7 x Boral Tubes	918
Total	3406
3. 15 x 15 PWR Spent Fuel Assembly	9975
Total Three Loaded Canisters Weight	62796
4. 3 Canister Support Assemblies	4725
5. 3-Bay Reinforced Concrete Module	800770
6. 3 x 2" Steel Door	12528
7. 6 x Shielding Blocks	10556
Total (3 Bay HSM Weight Loaded)	892236

TABLE 8.1-2

MAXIMUM DRY STORAGE CANISTER
SHELL STRESSES FOR NORMAL OPERATING LOADS

DCS COMPONENTS	LOAD TYPE / STRESS TYPE	STRESS (ksi) (1)				ASME CODE ALLOWABLES (ksi) (2)
		DEAD WEIGHT	DESIGN BASIS PRESSURE	DESIGN BASIS TEMPERATURE	OPERATION HANDLING (3)	
Canister Shell	Primary Membrane	0.21	0.91	7.4	0.38	18.7
	Local Primary Membrane	N/A	1.38	7.4	N/A	28.05
	Primary Membrane + Secondary Bending	11.55	7.64	20.90	10.99	56.10
Cover Plate	Primary Membrane	N/A	N/A	N/A	N/A	18.70
	Primary Membrane + Bending	N/A	3.27	0.45	13.36	28.05
Spacer Disk	Primary Membrane	1.10	N/A	N/A	N/A	18.70

Notes:

1. Values shown are maximums irrespective of location.
2. Allowable stresses are conservatively taken at 400EF.
3. Values are based on ram capacity load of 22,000 lb.

8.2 ACCIDENT ANALYSIS

This section addresses design events of the third and fourth types specified by ANSI/ANS 57.7-1981 and any other credible accident that could affect safe operation of the H. B. Robinson ISFSI. The postulated accidents are:

- Loss of air outlet shielding blocks
- Tornado and tornado generated missiles
- Earthquake
- Eight foot drop
- Lightning
- Blockage of air inlets and outlets
- Accident pressurization of the DSC
- Fire
- Leakage of the DSC
- Load Combination
- Train Derailment

In the following paragraphs, the accident analyses for various components of the ISFSI are described. When the accident loads or conditions are the same as (or enveloped by) those addressed in the NUHOMS Topical Report (Reference 8.1), reference will be made to the appropriate section of that report.

8.2.1 LOSS OF AIR OUTLET SHIELDING

This postulated accident assumes the loss of both air outlet shielding blocks from the top of the horizontal storage module. All other components of the ISFSI are assumed to be in normal condition. The air outlet shielding blocks are designed to remain in place and remain completely functional for all postulated accidents except tornado generated missiles. There are no structural or thermal consequences to the ISFSI as a result of the loss of the shielding blocks; however, there are radiological consequences which have been addressed and analyzed in the NUHOMS Topical Report, Section 8.2.1. The resulting increase in air scattered (sky shine) doses or direct radiation as reported in the Topical Report are within 10CFR100 dose limits.

Recovery

To recover from a lost or damaged shielding block caused by a tornado projectile, one of the spare blocks is transferred to the HSM. After the shield block is transferred to the HSM, a 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, during which a mechanic will be on the HSM roof for approximately 15 minutes. During this time he will receive less than 50 mrem. The dose to the crane operator and the mechanic on the ground while putting the shield block in place will be approximately 20 mrem each (assuming an average distance of 15 ft from the center of the module roof). Note: The times listed are only to provide estimates of radiation dose to workers. There are no commitments to ensure the shield blocks would be replaced within 30 minutes.

8.2.2 TORNADO/TORNADO GENERATED MISSILE

The most severe tornado wind loadings as specified by NUREG 0800 (Reference 8.3) and NRC Regulatory Guide 1.76 (1974) are selected as a design basis for this accident condition. The applicable design parameters of the design basis tornado (DBT) are the same as those specified in Section 3.2.1 and 3.2.2 of the NUHOMS Topical Report. The accident analysis of the ISFSI under the DBT is covered by the analysis presented in Section 8.2.2 of the referenced report. Given the fact that the HSM method of structural analysis as utilized by the referenced report conservatively envelopes any stacking arrangement of the modules, including the three modular concept at the H. B. Robinson ISFSI, the maximum moment and shear for the design basis wind pressure and missiles are also enveloped by the values given in Table 8.2-3 of this referenced report. Furthermore, the walls of the horizontal storage modules are anchored into the concrete foundation and as such, there is no possibility of overturning or sliding of the modules due to the impact of a massive high kinetic energy missile. The uplift forces generated by the impact of the massive missile and tornado wind loads are included in the foundation design presented in Section 8.3 of this report. The design and analysis of the anchorage system is also presented in Section 8.3.

The result of this accident analysis indicates that all components of the ISFSI are capable of withstanding the tornado wind loads and tornado generated missiles with the exception of the air outlet shielding blocks. The loss of the shielding blocks is addressed in Section 8.2.1 of this report.

8.2.3 EARTHQUAKE

8.2.3.1 Accident Analysis

As specified in Section 3.2 of this report, the maximum ground horizontal acceleration is 0.20g and the maximum ground vertical acceleration is 0.133g. The NUHOMS Topical Report assumes a value of 0.25g for maximum horizontal acceleration and 0.17g for maximum vertical acceleration. In the Topical Report, for the seismic stress analysis of various components, a multiplier of 2 is used to account for multimode excitations. Since the values of the vertical and horizontal acceleration of the referenced report are higher than the H. B. Robinson site accelerations, the seismic analysis for the HSM and the DSC support assembly presented in Section 8.2.3 of this referenced report is fully applicable and the results of these analyses envelop the site specific design. To establish the actual seismic response of the HBR DSC additional analysis is performed. However, the methodology is the same as that reported in Section 8.2.3.2, Pages 8.2-15 through 8.2-19 of the referenced report. Since the site specific design is different than that of the Topical Report, the longitudinal horizontal seismic loading on the HSM was reviewed. First of all, the DSC loads are transferred to the seismic retainer during a seismic event. The retainer is connected to the ram mounting assembly plate through tiedown bolts in the two inch cover plate. Consequently the loading is transferred to the embedded pipe and plate which are anchored into the HSM rear wall. The maximum loading of 8800 pounds generated by this event was found to have a negligible effect on the HSM rear wall.

In the referenced report, Section 8.2.3.2, the DSC shell ovaling mode was found to yield the lowest natural frequency. Since the HBR DSC shell parameters (i.e., the thickness and nominal diameter) have not been changed the lowest natural frequency remains the same at 37.2 Hz. The stresses induced on the canister casing and the basket due to the 0.20g horizontal and 0.133g vertical seismic accelerations are calculated on the basis of equivalent static method. The static stresses obtained are increased by a factor of 2.0 to account for multimode excitation. To obtain the DSC stresses due to the vertical component of the seismic load, the bending stresses calculated for the dead weight analysis can be factored directly by 0.266. The maximum stress obtained in this manner is 3.07 ksi. For the horizontal seismic analysis both the longitudinal and the transverse directions are considered. For the horizontal acceleration in the transverse direction, the method of analysis presented in Page 8.2-16 of the referenced report was employed and a bending stress intensity of 7.5 ksi was obtained. The stresses in the DSC shell and outer top plate due to the restraining action of the seismic restraint assembly under the longitudinal seismic loading was also investigated and found to have negligible effect. The shell stresses obtained for the vertical and horizontal cases were summed absolutely and a combined stress of 9.32 ksi was obtained.

Additionally, using the same methodology as that presented in Section 8.2.3.2, Page 8.2-17 of the referenced report, a margin of safety against a DSC roll over during a seismic event was established. A value of 2.5 was obtained for this margin of safety against the DSC roll over.

In summary, the ISFSI seismic analysis using site specific accelerations is enveloped by that reported in the NUHOMS Topical Report. Furthermore, the HSMs are anchored to the foundation and as such, no overturning or sliding of the modules is possible. However, the overturning effects on the foundation are included in the foundation design which is presented in Section 8.3 of this report. The anchorage design is also presented in Section 8.3 of this report.

8.2.3.2 Accident Dose Calculation

The major components of the HBR ISFSI are designed and analyzed to withstand the forces generated by the safe shutdown earthquake, hence there are no dose consequences.

8.2.4 DROP ACCIDENT

8.2.4.1 Postulated Cause of Events

As described in Section 1.3.1.7 of this report, the only time during the transfer operation that the IF-300 cask is operating without its redundant yoke is during the cask lowering into the cradle of the skid assembly. As shown in Figure 8.2-1 the maximum height that the cask is raised during this operation is 8.0 feet. Hence, the maximum height of a postulated drop accident is limited to this value. Furthermore, since the cask is always lifted from the trunnions located at the upper regions of the cask, the postulate failure of the single yoke can only cause a cask bottom end or a corner drop. Consequently, if the yoke fails during the tilting operation the cask will either land on the bottom end fins or on the side steel rings located near the upper and lower regions of the cask outer shell.

Based on the above discussion, an 8-foot drop criteria in either horizontal or vertical bottom end orientation will bound any possible drop orientation during the transfer operation, including a corner drop orientation. The skid assembly and the cask/skid/trailer tie down systems are designed to withstand the inertia forces associated with the transportation shock loads, and as such there is no possibility of a cask drop during the transport operation from the decon area to the HSM site. Even if such unlikely event occurs or the cask/skid/trailer tip over as a unit, the height of this drop condition is enveloped by the 8 foot drop height criteria.

8.2.4.2 Drop Accident Analysis

As stated earlier in Section 1.3.1.3 of this report, the IF-300 cask requires an additional extension collar and a new cask lid, in order to meet the cask cavity minimum length requirement and meet the criteria for cask lid removal in horizontal orientation. In this modified configuration the cask's impact limiters which are the radial fins attached to the cask's original head are removed. Hence, the energy absorbing properties of the cask is significantly reduced at upper regions. However, as discussed earlier the cask is always handled in upright position and no postulated failure mechanism can produce a top end drop. Additionally, the 8 feet rise of the cask is not sufficient for the cask to rotate 180 degrees in mid air to land on its head or upper corner. The remaining part of the cask impact limiters, i.e., the bottom radial fins and the ring and both ends are not altered and will provide the energy absorption mechanism needed for the vertical bottom end and the horizontal drop.

The IF-300 cask energy absorbing properties are contained in the cask Safety Analysis Report (Reference 8.4). This SAR contains extensive data concerning a 30-foot drop accident.

The latest deceleration time history development work of the IF-300 cask is contained in Appendix V-1 of the above referenced document. These particular impact time histories contain peak deceleration values, at early time of impact. These peak acceleration values are associated with the dynamic yield stress characteristic of the stainless steel fins (strain rate dependency). These time histories which envelop the previous histories reported in the referenced document, include 3 horizontal and 2 vertical drop orientations. These selected time histories were modified to reflect the 8 ft drop criteria described earlier. Since the overall geometry and the weight of the loaded cask are not significantly changed, these deceleration time histories were linearly scaled to reflect the 8 ft drop criteria. Figure 8.2-2 shows the modified deceleration time histories used in the DSC drop analyses.

Horizontal Drop

Principle structures effected by the horizontal drop are the spacer disk and the boral tubes. The boral tubes serve only as a guide for the fuel assemblies and are not considered load bearing members, except for their own weight. In the NUHOMS Topical Report, Section 8.2.9, Page 8.2-35, the stresses in the boral tube under the inertia forces of a 34g drop criteria were evaluated by a finite analysis technique. Since the boral tube design is not changed, the result of this analysis can be directly ratioed for the higher deceleration value of 54.4g. In this manner a maximum stress of 4.24 ksi is obtained.

The DSC basket is designed such that the locations of the spacer disk coincide with the fuel assembly grid strap. Therefore the weight of the fuel assemblies is directly transmitted to the disk. For the analysis of this member, the finite elements analysis reported in Section 8.2.4, pages 8.2-31 through 8.2-34 of the referenced report can be utilized directly. This is due to the fact that the overall configuration of this member has not been changed from that of the NUHOMS generic concept, with the exception of thicker disks, and the analysis is linear elastic. Consequently, the results of the referenced analysis can be factored to include the effect of the mass, thickness, length and deceleration value changes. Additionally, a factor was added to include the additional weight of the support rods in relation to the mass used in the STARDYNE Model.

$$S_{sp} = S_{nu} \left(\frac{M_{sp}}{M_{nu}}\right) \left(\frac{t_{nu}}{t_{sp}}\right) \left(\frac{g_{sp}}{g_{nu}}\right) \left(\frac{l_a}{l_u}\right) \left(\frac{M_{nsr}}{M_m}\right)$$

where:

S_{sp} = Maximum Stress, ksi

S_{nu} = 38.52 ksi (from NUHOMS)
M_{sp} = 2034 lb Mass, CP&L without rods
M_{nu} = 1818 lb Mass, NUHOMS without rods
g_{sp} = 54.4g, site deceleration value
g_{nu} = 34g, NUHOMS drop value
l_a = 26.19 in, actual cell length
l_u = 26.00 in, NUHOMS length
M_{nsr} = 1962.1 lbs Mass, NUHOMS with lids
M_m = 2133.1 lbs Mass, STARDYNE Model

therefore:

S_p = 39.93 ksi

It must be noted that this stress intensity is mainly due to the shear force developed near the imposed artificial support boundary, and as such is not representative of the actual stress of the disk. A more critical stress location of the disk is at the spacer beams adjacent to the fuel assemblies. The maximum membrane stress intensities is 23.5 ksi which is obtained by the same ratioing technique discussed above. The results of the horizontal drop analysis are contained in Table 8.2-1 of this report.

Vertical Bottom End Drop

The components of the DSC that are critically effected during a vertical bottom end drop are the DSC shell, the top and bottom DSC regions, the support rods and related DSC welds. The vertical drop analyses utilize both hand calculations and finite element technique. For the DSC shell and the end regions ANSYS program was employed for its axisymmetric and linear or nonliner features. Other components of the DSC are analyzed by hand calculation techniques.

As stated earlier in this report, the HBR DSC configuration is different from that of the NUHOMS generic concept. The DSC has been redesigned to fit into the IF300 cask, and also is designed to withstand a drop accident in which the height of the drop is significantly greater than the 8-foot criteria. This is done for compatibility with future shipping options.

For the DSC bottom region analysis a model consisting of 131 elements and 183 nodes was developed. The model is shown in Figure 8.2-3. Both lead and steel are modeled as 2-D, 4-node isoparametric axisymmetric finite elements (STIF42). The interface of the lead and steel is modeled with coincident nodes which are coupled in vertical direction only. In this manner, only normal forces are transmitted between the two surfaces, and the shear and friction forces are conservatively released. Both physical and symmetrical boundary conditions are imposed at appropriate locations. The material properties are conservatively taken at 400^{o}F to envelop peak temperatures of the DSC shell. The entire weight of the basket and the fuel assemblies are included as added mass elements along the top surface of the 2 inch cover plate. The weight of the top region of the DSC and that portion of the DSC shell that is not included in the model, is also included as added mass at the appropriate location on the shell. The response of the DSC bottom region under the drop impact time history was conservatively approximated by an equivalent static analysis. The impact time history has a very short duration and essentially behaves like a very short triangular impulse. Frequency analyses performed on various DSC components indicated that the longest natural period was much greater than the duration of the impulse. Thus, the dynamic impact loads cannot produce a response that exceeds the static response, and as such the dynamic amplification factor is less than unity. Therefore, the static analysis performed is more conservative than a dynamic analysis. An acceleration value of 76.5g was imposed statically on the model. Both membrane and extreme fiber stress intensities at critical locations including the weld elements are reported in Table 8.2-1 of this report.

For the top region of the DSC, another ANSYS finite element model as shown in Figure 8.2-4 was developed. This model consists of 298 isoparametric STIF42 elements and 409 nodes. Similar assumptions and modeling technique as discussed for the bottom region model were employed. Static acceleration of 76.5g was applied to the model to obtain the membrane and bending stresses for various components and welds. The results of this analysis is also included in Table 8.2-1 of this report along with the ASME code allowables.

The 2 1/2 inch diameter steel support rods were analyzed under the postulated vertical drop. These rods extend the entire length inside cavity of the DSC. The main function of these rods is to provide resistance to axial loads for the spacer disks.

Each of the seven spacer disks is welded to these rods by means of fillet welds. One inch clearance is provided between the support rods and the top lead plug of the DSC. This clearance is provided so that thermal expansion of the components and deflection of components during accident loading conditions, such as a drop accident, will not cause interference.

The 2 1/2 inch diameter support rods are designed so that they will resist the weight of the spacer disks under the postulated drop. The most critical segment of the support rod is between the two bottom spacer disks. For this analysis, the weight imposed on a single rod at this critical location was the weight of six spacer disks divided by 4 plus the self weight of 1 rod. The axial stress at this 26 inch segment of the rod was found from the following relationship:

$S_{mx} = W \times a / A$

where: S_{mx} = Axial stress
W = 651.7 lb, total weight imposed on the rod
a = 76.5g, peak vertical deceleration
A = 4.91 in^2, cross sectional area of the rod

therefore:

S_{mx} = 10.15 ksi

The support rod material has been changed from SA304 stainless steel to SA-479 Type XM19 material which has a yield stress of 40.8 ksi at 400°F. The allowable compressive stress for this material is established by the rules of the ASME Appendix XVII, and Appendix F, which include the effect of slenderness ratio.

The results of the support rod analysis along with the compressive stress allowable are tabulated in Table 8.2-1 of this report.

The results of the horizontal and vertical drop analysis as shown in Table 8.2-1 indicate that the stresses in all components of the DSC and its internals are within the ASME acceptance limits and are capable to withstand inertia forces associated with the 8 foot drop accident condition. A corner drop accident was also considered. However the deceleration values as established by the IF-300 cask SAR are significantly lower than the values of either the horizontal or the vertical deceleration components. Therefore the stresses for corner drop analysis are bounded by the analyses presented above.

8.2.5 LIGHTNING

8.2.5.1 Postulated Cause of Events

Since the ISFSI is outdoors, there is a likelihood that lightning could strike the ISFSI. Section 2.3.1 of the HBR2 Updated FSAR (Reference 8.2) provides information on the frequency of cloud-to-ground lightning strokes (the only type of lightning stroke which poses a hazard to the ISFSI) at the site. In order to protect the ISFSI from any damage which could be caused by a lightning discharge, a lightning protection system is installed on the ISFSI. The lightning protection system is designed in accordance with NFPA No. 78-1979 Lightning Protection Code (Reference 8.5). This system will prevent any damage to the HSM and its internals. Therefore, lightning striking the HSM and causing an off-normal condition is not a credible accident.

8.2.5.2 Analysis of Effects and Consequences

Lightning protection systems have proven to be an effective means of protecting a structure and its contents from the effects of a lightning discharge. The lightning protection system does not prevent the occurrence of a lightning discharge; however, the system does intercept the lightning discharge before it can strike the HSM and provides a continuous path for the discharge to the earth. In the event of lightning striking the HSM, the air terminal located on the HSM roof slab would intercept the lightning discharge. The current will follow the low impedance path of the air terminal, conductors, and ground terminals to the earth. Since the system diverts the current, the HSM and its contents will not be damaged by the heat or mechanical forces generated by the current passing through the HSM. In addition, since the ISFSI requires no electrical system for its continuous operation, the resulting current discharge will have no effect on the operation of the ISFSI.

8.2.6 BLOCKAGE OF AIR INLETS AND OUTLETS

This accident is the complete and total blockage of the air inlets and outlets of the horizontal storage module. Since the ISFSI is located outdoors, it can be postulated that the module is totally covered by debris from such an unlikely event as a tornado. The ISFSI's design features, such as a perimeter fence and separation of air inlets and outlets, minimize the probability of such an accident occurring under normal conditions. Nevertheless, such an accident is postulated and analyzed.

There are no structural consequences under this event. The thermal consequence of this accident results from heating of the DSC and HSM due to the blockage of air flow. Section 8.2.7 of the NUHOMS Topical Report addresses this accident condition. The results of the analysis indicate that there is no structural or dose consequence if the air inlets and outlets are cleared within 48 hours. This 48 hour time limit for clearing the air inlets and outlets is specified in the HBR2 ISFSI operation and limits criteria (See Chapter 10).

8.2.7 ACCIDENT PRESSURIZATION OF DSC

Internal pressurization of the DSC results from fuel cladding failure and the subsequent release of fuel rod fill gas and free fission gas. To establish the maximum accident pressurization, it is assumed that all fuel rods in the DSC are ruptured and that the fission gas release fraction is 25%, and the original fuel rod fill pressure is 500 psig. (HBR fuel actually has a fill pressure of 300 psig.) The resulting internal pressures at HBR's maximum ambient temperature of 105°F and at the minimum ambient temperature of -5°F are below the accident pressures reported in Section 8.2.9 of the NUHOMS Topical Report (for temperature extremes of 125°F and -40°F). The limiting accident for canister pressurization is the blockage of air flow to the DSC. Under these conditions, the gas temperatures in the DSC will rise to 413°C (775°F) producing a DSC internal gauge pressure of 2.76 bar (39.7 psig). The canister shell stresses due to accident pressurization are enveloped by those reported in the Topical Report.

The DSC has a safety margin of greater than 3 under this accident condition and as such, there are no dose consequences.

8.2.8 FIRE

No flammable or combustible substances are stored within the ISFSI or within the ISFSI's radiation control area. Additionally, the ISFSI is constructed of non-flammable heat-resistant materials (concrete and steel). The only credible accident which could expose the ISFSI to a flammable substance would be the accidental spillage of a flammable liquid, either through human error or equipment malfunction, at the perimeter of the ISFSI. However, the sandy soil between the sides of the ISFSI's perimeter fence and the HSM, is highly porous. Most of the flammable liquid would be absorbed by the soil, greatly reducing the intensity or duration of the fire.

The only other time in which a component of the ISFSI would be exposed to a potential fire hazard would be during the DSC drying and transport operations. Throughout these operations, the DSC is located within the cavity of the GE IF-300 shipping cask.

Based on the above discussion, exposure of the ISFSI to a long or intense fire is not considered a credible accident.

8.2.9 DRY STORAGE CANISTER LEAKAGE

The DSC is designed for no leakage under any normal or credible accident conditions. The accident analyses in previous sections show that none of the events could breach the canister body. However, to show the ultimate safety of the ISFSI, a total and instantaneous leak was postulated. The postulated accident assumes that one DSC ruptured and all fuel rod claddings failed simultaneously such that 25% of all fission gases in the irradiated fuel assemblies (mainly Kr-85) are instantaneously released to the atmosphere. The dose consequences from the leaking DSC are evaluated in the NUHOMS Topical Report, Section 8.2.8, and the resulting accident dose is found to be well below the 10 CFR Part 72.68 acceptable limit of 5.0 rem.

8.2.10 LOAD COMBINATION

Normal operating and postulated accident loads associated with various components of the ISFSI are either the same as or are enveloped by those reported in the NUHOMS Topical Report, except for the DSC and the foundation. Hence, the combined effect of various accident and normal operating loads for the DSC support assembly and the HSM are enveloped by the load combination results presented in Section 8.2.10 of the Topical Report. The methodology used in combining normal operating and accident loads and their associated over load factors for various components of the ISFSI, with the exception of the foundation, is presented in the aforementioned report. Load combination procedures for the foundation are addressed in Section 8.3 of this report. The DSC analysis load combination utilizes the same methodology as in the Topical Report, but due to design differences the results are changed slightly. The results of the DSC load combination for the worst case, i.e., drop accident, are contained in Table 8.2-2 of this report. Furthermore, the DSC fatigue analysis due to normal operating pressure loads, accident pressure loads, seismic loads, seasonal temperature loads, and daily temperature cycling as presented in Section 8.2.10 of the Topical Report, envelops the HBR site specific analysis. This is because the extreme ambient temperature selected for generic design of the DSC (-40^{o}F to 125^{o}F) envelops the HBR ambient temperature range (-5^{o}F to 105^{o}F) and the HBR has a lower seismic acceleration.

Table 8.2-1

MAXIMUM DSC STRESSES FOR 8-FOOT
BOTTOM END DROP ACCIDENT

DSC COMPONENTS(2)	STRESS TYPE	STRESS (ksi)	
		CALCULATED	ALLOWABLE(1)
Canister Shell	Primary Membrane	10.11	44.88
	Primary Membrane + Bending	16.56	64.40
Bottom Cover Plate	Primary Membrane	6.09	44.88
	Primary Membrane + Bending	13.40	64.40
Top Cover Plates	Primary Membrane	2.77	44.88
	Primary Membrane + Bending	5.47	64.40
Support Ring For Top Lead Plug	Primary Membrane	1.71	44.88
	Primary Membrane + Bending	5.43	64.40
Lead	Compressive	1.65	6.80
Lead Casing	Primary Membrane	17.23	44.88
Spacer Disk	Primary Membrane	39.93	44.88
Boral Tubes	Primary Membrane + Bending	4.24	64.40
2 ½ in. diam. Support Rods	Compression	10.16	21.13(3)
¼ in. Fillet Weld	Primary	11.64	22.44
J-Weld	Primary	6.16	29.20

Notes:

1. Allowable stresses shown correspond to service Level D limits, unless noted otherwise.
2. Material properties taken at 400°F design temperature.
3. Compressive stress allowable of the support rods is based on Appendices XVII and F rules and for Level A limits.

Table 8.2-2

DSC ENVELOPING LOAD COMBINATION[1]

DSC COMPONENTS[4]	STRESS TYPE[5]	STRESS (ksi)	
		COMBINED[2]	ALLOWABLE[3]
Canister Shell	Primary Membrane	11.23	44.88
	Primary Membrane + Bending	35.75	64.40
Bottom Cover Plate	Primary Membrane	6.09	44.88
	Primary Membrane + Bending	15.91	64.40
Top Cover Plates	Primary Membrane	2.77	44.88
	Primary Membrane + Bending	8.74	64.40
Support Ring For Top Lead Plug	Primary Membrane	1.71	44.88
	Primary Membrane + Bending	5.44	64.40
Lead	Compressive	1.86	6.80
Lead Casing	Primary Membrane	17.29	44.88
Spacer Disk	Primary Membrane	41.03	44.88
Boral Tubes	Primary Membrane + Bending	4.36	64.40
2 ½ in. diam. Support Rods	Compression	10.19	21.13[6]
¼ in. Fillet Weld	Primary	11.68	22.44
J-Weld	Primary	6.18	29.20

Notes:

1. When applicable, stresses due to the drop accidents are combined with that of pressure and dead weight.
2. Stresses for each DSC components are conservatively combined irrespective of location.
3. Allowable stresses shown correspond to service Level D limits, unless noted otherwise.
4. Material properties taken at 400°F design temperature.
5. Thermal stresses need not be included under service Level D limits.
6. Compressive stress allowable of the support rods is based on Appendices XVII and F rules for Level A limits.

8.3 FOUNDATION DESIGN

To provide a means of transmitting the reaction loads of the ISFSI modules to the ground, a rectangular, flat plate type, mat foundation was selected. The mat foundation is ideally suited for the ISFSI since it spreads out the loadings and consequently reduces the soil bearing pressure and at the same time minimizes the differential settlements.

To accommodate the ISFSI modules, the front cask unloading area and the hydraulic ram area behind the modules, an overall foundation size of 28'-9" by 60'-0" was selected. The HSM foundation slab is 3 feet thick. A construction joint connects this slab to the cask unloading slab which is 2 feet thick starting from a point 5 feet from the module front. The ram mounting slab at the rear of the modules is 8 inches thick and connects to the 3 foot foundation by an expansion joint. The foundation concrete is 4000 psi normal weight concrete poured on a 4 inch mud slab. The HSM foundation and the cask unloading slab are interlaced with continuous two-way reinforcing top and bottom. Number 9 bars are used for tensile reinforcement and as dowels to anchor the HSM walls to the foundation. The ram mounting slab has a number 5 bar continuous two-way reinforcing at the bottom only. Welded wire fabric is placed at the top of the 8 inch slab.

For analysis purposes, a STARDYNE rectangular plate finite element model as shown in Figure 8.3-1 was developed. The model consists of 255 nodes and 224 plate elements. At each node, a ground support spring was added to simulate the soil elastic properties. The elastic soil spring is obtained by modifying the experimental modulus of subgrade reaction by an appropriate size factor of the foundation. The modified modulus of subgrade reaction is then multiplied by the tributary area associated with each node. The resulting values of the spring stiffnesses were used as input to the finite element model as a ground stiffness matrix. The method for finding the stiffness K is shown below (Reference 8.7):

$$K = K_V \left(\frac{B+1}{2B}\right)^2 A$$

Where:

K_v = experimental modulus of subgrade reaction
= $100^{\#}/in^3$ (for granular soil)
B = foundation width = 28.75 feet
A = nodal tributary area (varies)

Five separate load cases were considered in the foundation design:

1) Center module loading
2) Outside module loading
3) Dead Weight + Live Load
4) Dead Weight + Tornado Wind/Impact (lengthwise)
5) Dead Weight + Tornado Wind/Impact (widthwise)

Since cask unloading and ram mounting slabs are cast in place after HSM construction, the differential settlement due to HSM dead weight will not be experienced by the cask unloading slab. Consequently, for load cases 1 and 2 the dead weight was not included. For load cases 1 and 2 the total trailer loading of 175 k, which includes the saddle, canister, cask, skid, trailer, rollers, trunnion, and cradle is applied as concentrated loads at nodal locations in the unloading areas. Since only one loading or unloading operation will occur at a time, the two load cases were evaluated independently. Load case 3 consists of the dead weight of the three modules containing the DSCs. This total 3 bay module weight of 800.8 k is added to the weight of three DSCs. The total loading is divided by the surface area in contact with the foundation to get an equivalent pressure load. Additionally, a live load of 200 psf is postulated for the HSM roof. This total load is also divided by the contact area to get a pressure loading on the foundation. These loads are applied as pressure loads on the appropriate plate elements of the STARDYNE model. Load cases 4 and 5 are the maximum uplift load combinations caused by tornado loadings in the two horizontal directions. Using a conservative wind pressure of 400 psf applied on the module walls and roof plus the reaction load of 458.2 k caused by a 3967 lb. automobile traveling at 184.8 ft./s applied to the top of the module in the same direction combined with the module dead weight, the maximum uplift forces were calculated. Live loads are excluded since they would reduce uplift loads. A simple frame model was used to calculate uplift forces as shown in Figure 8.3-2. The maximum uplift force of 38.80 k is converted to a pressure using the contact area of one wall. This yields a maximum uplift pressure of 3.1 psi which is used in the analysis. Comparison of tornado loads and HSM seismic loadings shows that tornado loads are much more severe. Consequently, seismic loads will not be included in the foundation analysis. Once the uplift forces are calculated they are applied as negative pressures on the appropriate plate elements corresponding to the HSM/foundation connection surface. Tornado wind and impact loads are evaluated for both directions.

Since an uplift force is created by the tornado loads the foundation itself will have a negative bearing or uplift along the edge of the module. The soil itself does not resist uplift. Results from load cases 4 and 5 were reviewed for the effects of the uplift along the foundation edge. Minimal uplift was experienced in low stressed areas. Therefore, results from load cases 4 and 5 are not significantly affected by the uplift since the high stressed areas are not in that vicinity.

The maximum calculated bearing stress is shown in Table 8.3-1. For sandy soils present at the bearing level of the mat foundation, allowable soil bearing pressures in the range of 3000 to 4000 pounds per square foot are recommended by the Southern Standard Building Code per the geotechnical exploration performed at the Robinson site by Law Engineering Testing Company. Since the maximum soil contact pressure produced by the HSM foundation analysis is 2210 psf the bearing strength is sufficient. For normal dead weight and live loads the bearing pressure is only 1605 psf.

The reinforcement design was based on the element bending moment results from the finite element analysis. Using the ultimate design method, the reinforcement was designed to withstand all postulated load combination bending moments with a conservative load factor of 1.7 applied to envelope all load combination factors specified in Section 9.2 of ACI 349-80. A tabulation of the results for the HSM Foundation and the cask unloading slab is presented in Table 8.3-2.

A license amendment issued on March 23, 1989, authorized construction of a five module foundation. Supporting information regarding the five module foundation is found in CP&L letters dated January 7, 1989, and February 1, 1989, and in the NRC letter dated March 23, 1989.

TABLE 8.3-1

FOUNDATION BEARING STRESS

LOAD CASE	LOAD DESCRIPTION	BEARING STRESS (KSF)
1	Center Module Loading	0.247
2	Outside Module Loading	0.463
3	Dead Weight + Live Load	1.605
4	Tornado Wind + Impact (Widthwise)	2.210
5	Tornado Wind + Impact (Lengthwise)	0.671

Note:

1. Dead weight of module not included in load cases 1 and 2.

TABLE 8.3-2

FOUNDATION SLAB
MAXIMUM BENDING MOMENTS

LOAD CASE	LOAD DESCRIPTION	SLAB THICKNESS	MAXIMUM MOMENT (K-in./in.)	ALLOWABLE MOMENT (K-in./in.)
1	Center Module Loading	2'-0"	13.8	87.0
		3'-0"	28.9	186.0
2	Outside Module Loading	2'-0"	13.1	87.0
		3'-0"	24.5	186.0
3	Dead Weight + Live Load	2'-0"	N/A	87.0
		3'-0"	46.4	186.0
4	Tornado Wind + Impact (Widthwise)	2'-0"	N/A	87.0
		3'-0"	80.1	186.0
5	Tornado Wind + Impact (Lengthwise)	2'-0"	57.6	87.0
		3'-0"	155.8	186.0

Notes:

1. Dead weight of module not included in load cases 1 and 2.

2. All moments conservatively factored by 1.7 to envelop all ACI 349 load combination factors.

The moment capacity of the sections are calculated per methods identical to the NUHOMS Topical Section 8.1.1.5, Equation 8-1-32. The 3'-0" slab with number 9 bars at 9 inches yields an ultimate strength of 186 k.in per inch section. The 2'-0" slab with number 9 bars at 12 inches yields an ultimate strength of 87 k.in per inch section. Therefore all bending moments experienced by the foundation are below ultimate capacity.

As calculated before, the maximum uplift pressure exerted by the module wall is 3.1 psi for load case 5. For a 1'-0" section of module the resulting uplift is 1.56 k/ft. section. The dowel area required can be calculated by:

$$A = \frac{(\text{UPLIFT})\,(1.7)}{(\emptyset)\,(f_y)}$$

Where:

UPLIFT = 1.56 K
θ = 0.9 = Factor for Tension
f_y = 60 ksi

Therefore, Area A = 0.05 in^2. Conservatively 2 number 6 bars with an area of 0.88 in^2 will be used every 12 inches to prevent uplift. Keyways will be used between the module foundation interface to prevent sliding. Assuming the maximum horizontal tornado loads are shared by the walls perpendicular to load yields a maximum shear force of 24 k/ft including the load factor of 1.7. The nominal shear strength of the keyway and dowels can be found from:

$$V_n = (V_s + V_c)\ \emptyset$$

Where:

V_c = 2 F'_c b_w = concrete shear strength (k)
f'_c = 4000 psi
b_w = 9 inches
d = 12 inches
V_s = $(A_v)\ (f_y)$ = steel reinforcement shear strength (k)
A_v = .88 in^2
f_y = 60 ksi
$\emptyset$ = .85 = shear factor

Consequently, V_n = 56.49 k which exceeds the maximum factored shear force of 24 k. Thus, the module will neither slide nor overturn. Table 8.3-3 presents foundation anchor loads and capacities for load cases 4 and 5.

The 8 inch ram mounting slab was designed by hand calculations suggested by Teng (Reference 8.7) and Bowles (Reference 8.8). By applying the maximum factored spider leg loadings from the hydraulic ram a simple span is approximated by treating the soil as a uniform load and the spider leg as reaction points. A maximum factored moment of 32.3 k-in/ft is calculated. Using the ultimate strength method with number 5 bars at 12 inches, the ultimate strength of the 8 inch slab is 64.2 k-in/ft. Welded wire fabric was placed at the top of the slab as shrinkage and temperature reinforcement. Additionally all punching shear from the ram supports were found to be negligible.

Furthermore, the cask unloading slab was analyzed for bearing and punching shear due to the hydraulic cylinder. A maximum bearing stress of 2.34 ksi was calculated which is less than the allowable of 4.76 ksi calculated from ACI 349-80 Section 10.16. The maximum punching shear was also found to be under code allowables.

TABLE 8.3-3

FOUNDATION ANCHOR LOADS

LOAD CASE	DESCRIPTION	LOADING TYPE	LOAD (K/FT)	CAPACITY (K/FT)
4	Tornado Wind + Impact (Widthwise)	Shear	13.00	56.5
		Uplift	0	28.0
5	Tornado Wind + Impact (Lengthwise)	Shear	24.00	56.5
		Uplift	1.56	28.0

Notes:

1. All Shear and Uplift loads factored by 1.7 to envelop all ACI 349 load combination factors.

2. Shear capacity based on concrete keyway plus embedded dowels.

3. Uplift capacity calculated from embedded dowel area.

8.4 DSC INSTRUMENTATION PENETRATION DESIGN

Instrumentation is not required to support the operation of the ISFSI. However, for research purposes two of the DSCs to be installed at the H. B. Robinson facility have been designed to accept instrumentation. Instrumentation was included as part of an agreement between CP&L, EPRI and DOE to augment the U.S. database on LWR fuel rods in dry storage.

The DSC thermocouples will be connected to an external cable by means of a specially designed feed-through. This feed-through incorporates the same redudant seal philosophy used in the DSC containment design. After the penetration plug assembly has been welded to the bottom of the DSC cover plate, a sleeve will be welded over the plug, forming a redundant seal. Thermocouple sheaths will likewise be brazed to the plug assembly at inner and outer penetrations. To preclude possible leakage through the aluminum oxide insulation, each end of the sheathed thermocouples will be sealed with an environmentally qualified resin.

The instrumentation penetration described above was analyzed for the maximum of the three load combinations described below:

1) DW + Accident Pressure + Thermal + 8 Ft. Vertical Drop

2) DW + Accident Pressure + Thermal + 8 Ft. Horizontal Drop

3) DW + Accident Pressure + Thermal + Seismic

The dead weight of the instrumentation penetration including the sleeve, the lead, the junction box and miscellaneous fittings was approximately 11 pounds. A 1g acceleration was added on top of the peak deceleration for the 8 foot drop to account for dead weight stresses.

An accident pressure of 39.7 psi was applied to the external surface of the stainless steel tubing sleeve. Using the formula from Roark for a thick-walled vessel (Reference 8.9, Table 32, Case 1d) the maximum stresses were calculated for the accident pressure load as shown below.

$$s_1 = \frac{-qa^2}{a^2 - b^2}$$

$$s_2 = \frac{-2qa^2}{a^2 - b^2}$$

$$t_{MAX} = \frac{s_2}{2}$$

where:

S_1= longitudinal stress, psi
S_2 = circumferential stress, psi
t_{MAX} = maximum shear stress, psi
a = .8125 in., outside radius
b = .625 in., inside radius
q = 39.7 psi, accident pressure
S_1 = 97 psi
S_2 = 195 psi
t_{MAX} = 97 psi

Consequently the maximum stress intensity for the accident pressure case is 0.25 ksi.

The thermal expansion of the tubing sleeve between the 2 in. plate and the outer 1/4 in. lead casing plate was examined. After comparison of the tubing axial stiffness in relation to the 1/4 in. plate stiffness it was concluded that the sleeve is essentially free to grow since the plate is approximately 72 times as flexible as the tubing. Consequently thermal stresses are considered negligible for the penetration analysis.

The maximum seismic ground accelerations in the horizontal and vertical directions are .2g and .133g, respectively. They are enveloped by the 8 foot drop peak decelerations.

For the 8 foot drop analysis both vertical and horizontal drops were considered. The peak deceleration for the horizontal drop of 55.4g (1g added for dead weight) was applied to the junction box and tubing extending internally from the weld on the two inch plate. Assuming a cantilever type beam as shown in Figure 8.4-1, the maximum stress intensity of the tubing is 8.19 ksi which is located near the weld. Applying the horizontal drop load to the weld shows a maximum stress of 0.89 ksi. Additionally, the effect of the 2 1/4 in. lead plug pressure loading on the outside surface of the tubing was checked. Results from this analysis indicated a stress much less than the 8.19 ksi previously calculated. Therefore this was not a critical area.

The peak vertical bottom end drop deceleration of 77.5g (1g added for dead weight) was applied to the tubing penetration in the axial direction. The corresponding axial stress was 1.01 ksi, which is considerably less than the horizontal drop case. The weld stress for the vertical drop was calculated as 0.17 ksi. Consequently, the 8 foot horizontal drop load case will be used as the governing load combination.

Combining the dead weight, accident pressure and horizontal drop stresses absolutely for the tubing penetration indicates a maximum stress intensity of 8.44 ksi. The maximum calculated weld stress was 0.89 ksi. These stresses are far below allowable stresses for both components. Clearly then, the confinement integrity of the instrumented DSC will not be jeopardized.

The method for sealing the thermocouple sheaths has been changed from the originally planned sealing by metalizing the aluminum oxide insulation to the use of an environmentally qualified resin. The thermocouple system with the resin sealant has been analyzed for drop (up to 15 inches) and cooling-related accidents. In addition, the integrity of the epoxy seal was reviewed against effects of the following:

- environmental conditions (thermal, radiation) inside and outside of the DSC;

- potential changes in epoxy characteristics over time under expected environmental conditions;

- evaluation of permeation rates; and

- responses to accidents including overpressurization

Analyses have concluded that under normal operating conditions seal leakage will be insignificantly small. Also, the structural capability of the material is such that possible accidents will not compromise performance. A detailed discussion of the analyses may be found in References 8.10, 8.11, 8.12, and 8.13.

8.5 TRAIN DERAILMENT

This postulated accident was analyzed in Section 2.1.2 of Reference 8.10. The closest rail line of those lying north of the ISFSI foundation pad is approximately 33 feet distant. This line is used for temporary holding of empty coal cars and as a run through track. The maximum speed limit for trains on this track is 5 mph. The soil between the rail line and the ISFSI is very porous, which would tend to impede the motion of a derailed car even though the ISFSI site is at a somewhat lower elevation than the track. In view of the foregoing and the fact that there are no switches within 500 feet of the ISFSI location, damage to the ISFSI from train derailment is not considered credible.

REFERENCES: CHAPTER 8

8.1 NUTECH Engineers, Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-001, Revision 1, November 1985.

8.2 Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

8.3 U.S. Nuclear Regulatory Commission, "Missiles Generated by Natural Phenomena," Standard Review Plan NUREG-0800, 3.5.1.4, Revision 2, July 1981.

8.4 General Electric Company, "IF-300 Shipping Cask Consolidated Safety Analysis Report," NEDO-10048-2, Nuclear Fuel and Special Products Division.

8.5 National Fire Protection Association, National Fire Codes, No. 78, 1979 Edition.

8.6 Cybernet Services, STARDYNE User Information Manual, Control Data Corporation, Minneapolis, Minnesota, Revision C, April 1980.

8.7 W. C. Teng, "Foundation Design," Prentice-Hall, Inc., Englewood Cliffs, N.J., 1962.

8.8 J. E. Bowles, "Foundation Analysis and Design," McGraw-Hill, New York, N.Y., 1977.

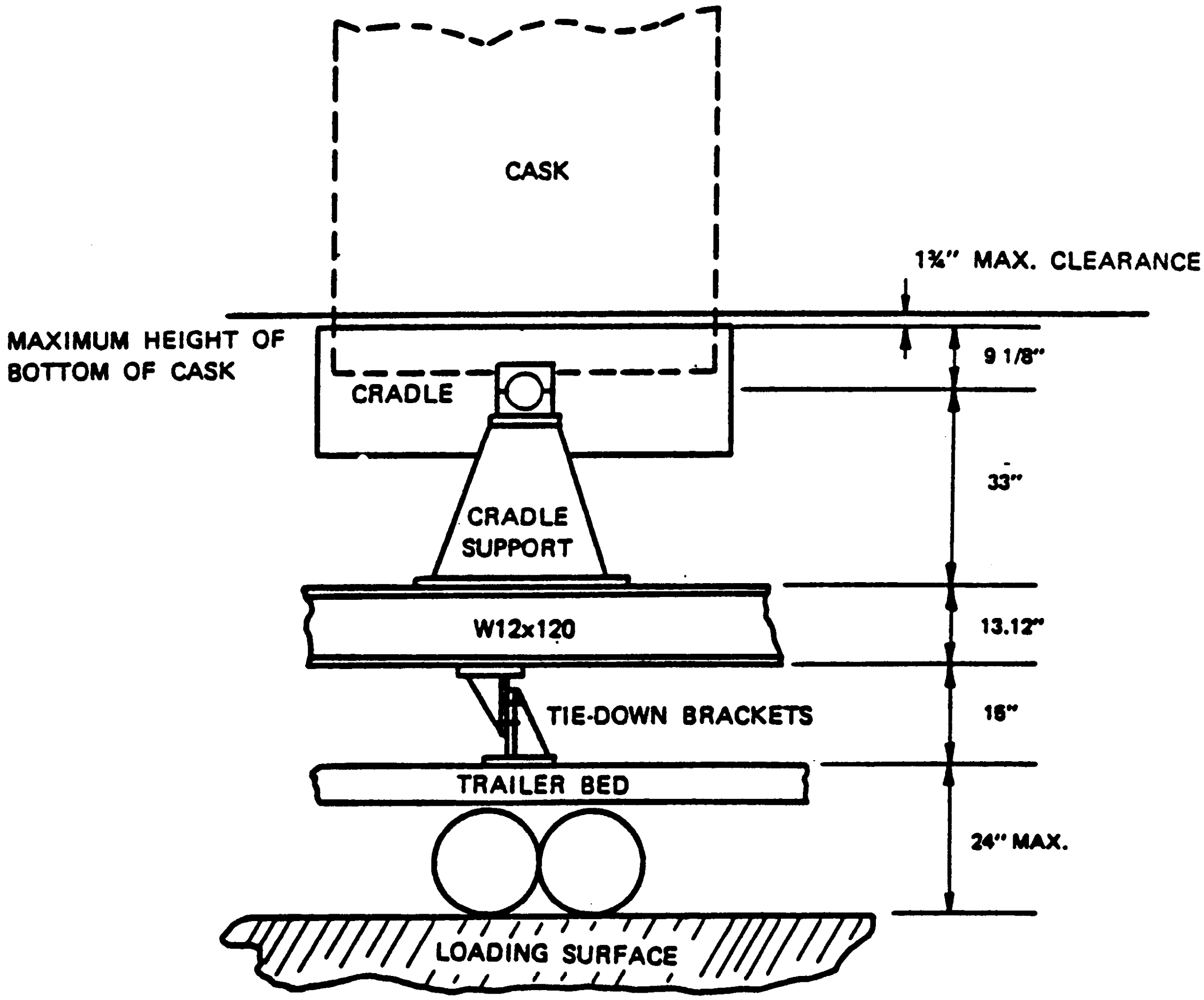
8.9 R. J. Roark and W. C. Young, "Formulas for Stress and Strain," Fifth Edition, McGraw-Hill, New York, N.Y., 1975.

8.10 Letter M. A. McDuffie, CP&L to NRC dated January 11, 1989, NLS-89-002.

8.11 Letter L. I. Loflin, CP&L to NRC dated April 28, 1989, NLS-89-117.

8.12 Letter L. I. Loflin, CP&L to NRC dated June 2, 1989, NLS-89-164.

8.13 Letter L. C. Rouse, NRC to CP&L dated June 22, 1989, Amendment to Materials License No. SNM 2502, Amendment No. 7



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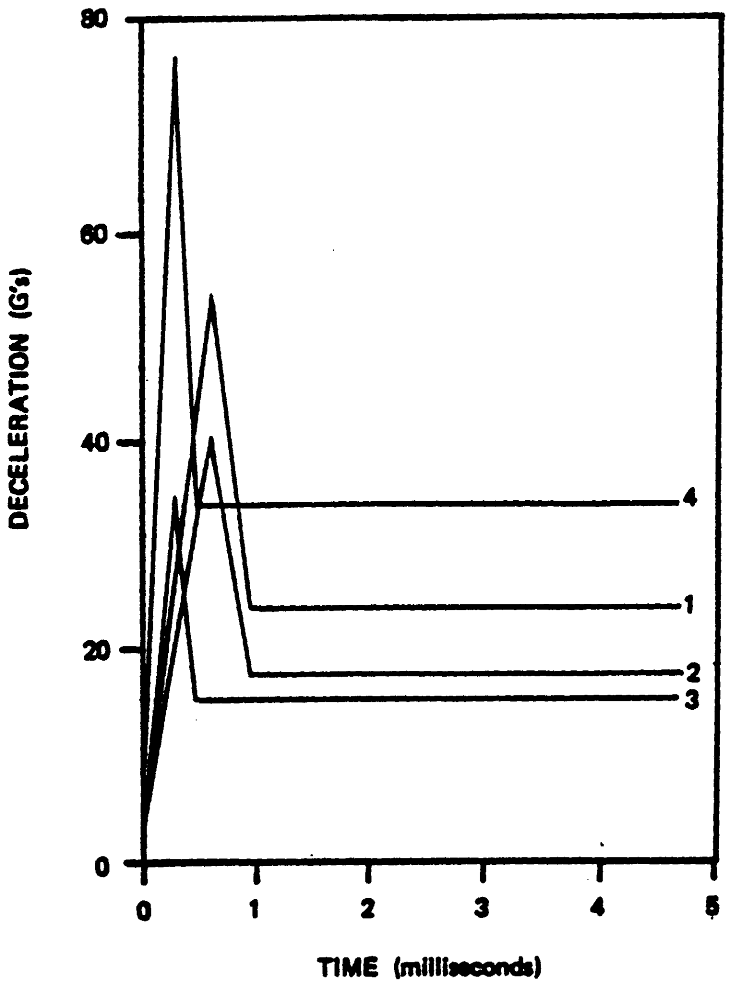
CASK DROP HEIGHT CRITERIA

Figure 8.2-1

NO.	CASK ORIENTATION	PEAK G's	PLATEAU G's
1	0 DEGREES	54.4	24.3
2	45 DEGREES	40.8	18.1
3	90 DEGREES	35.5	15.7
4	BOTTOM END	76.5	33.9



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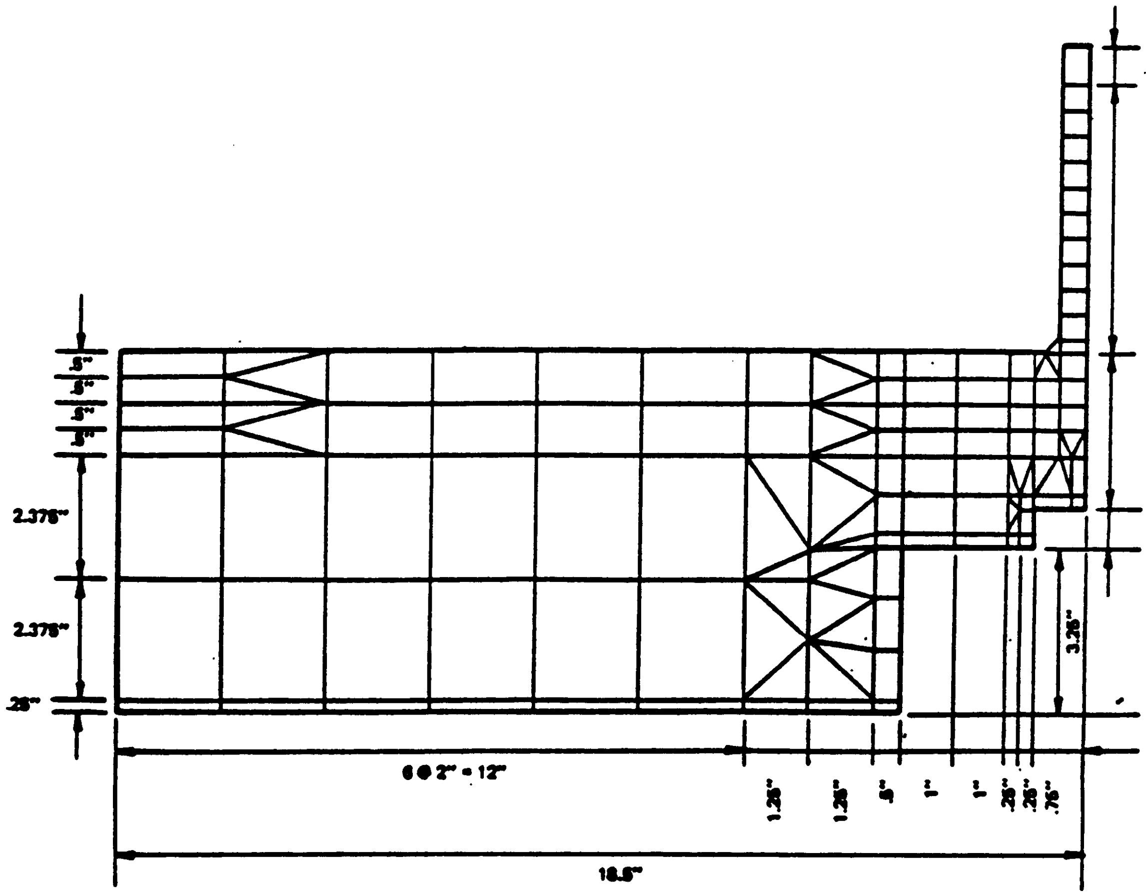
CASK DECELERATION vs. TIME
8 FOOT DROP
Figure 8.2-2



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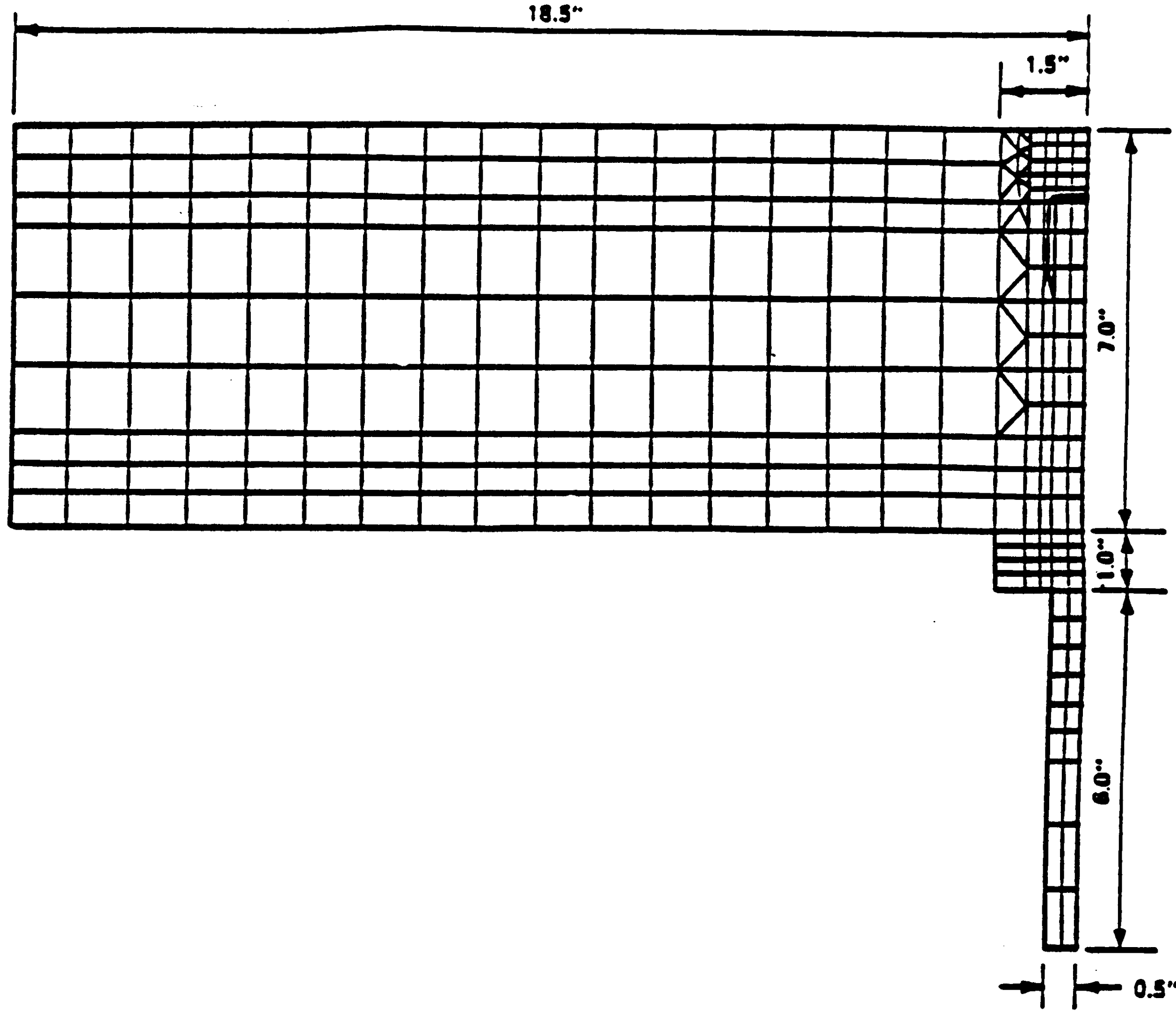
DSC BOTTOM REGION ANSYS MODEL

Figure 8.2-3



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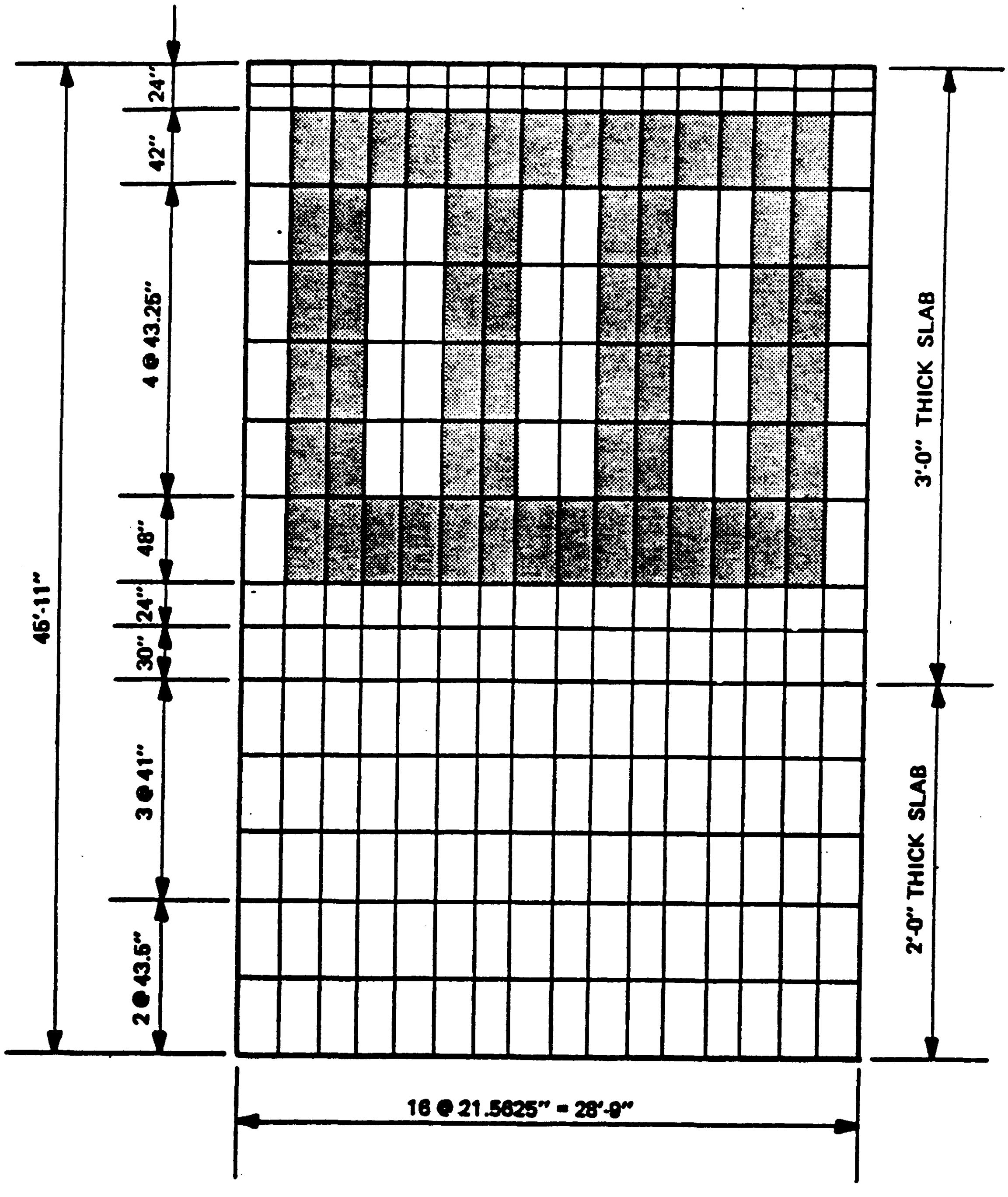
DSC TOP REGION ANSYS MODEL

Figure 8.2-4



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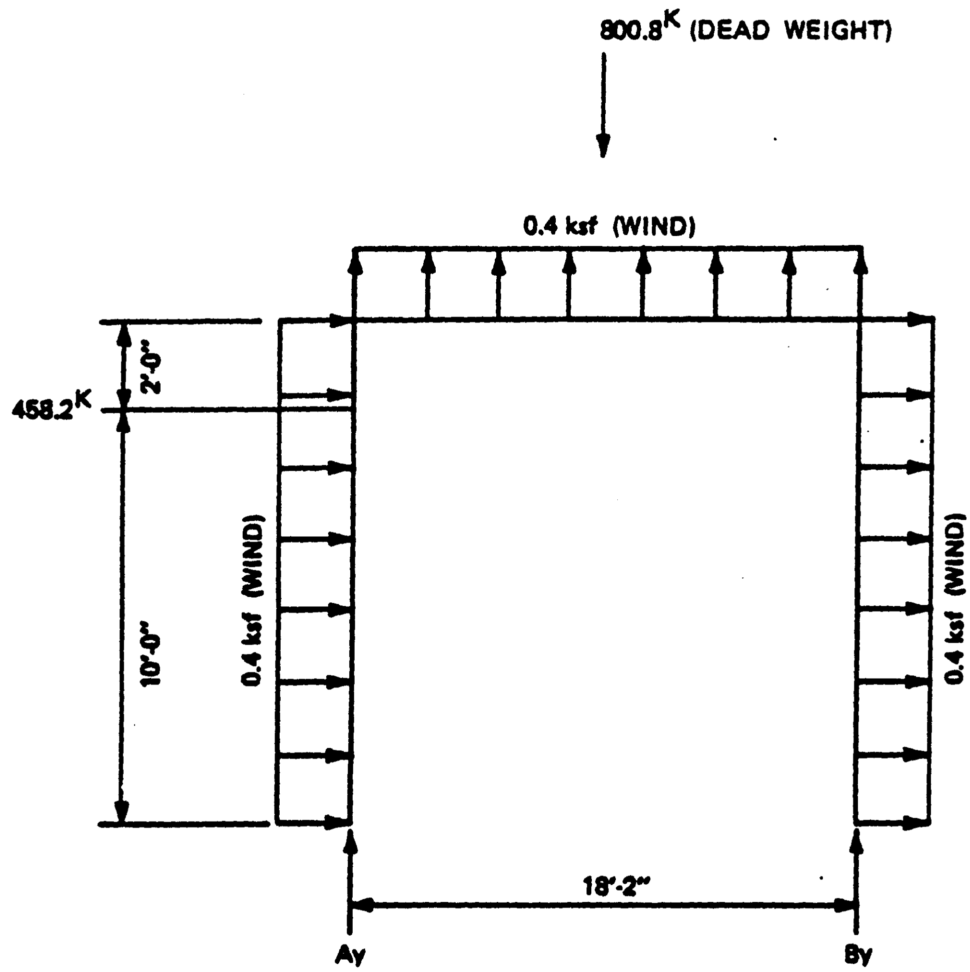
MAT FOUNDATION STARDYNE MODEL
Figure 8.3-1



Ay = 38.80K (UPLIFT)
By = 622.59K

NOTE
1) MODEL IS 301.875" WIDE

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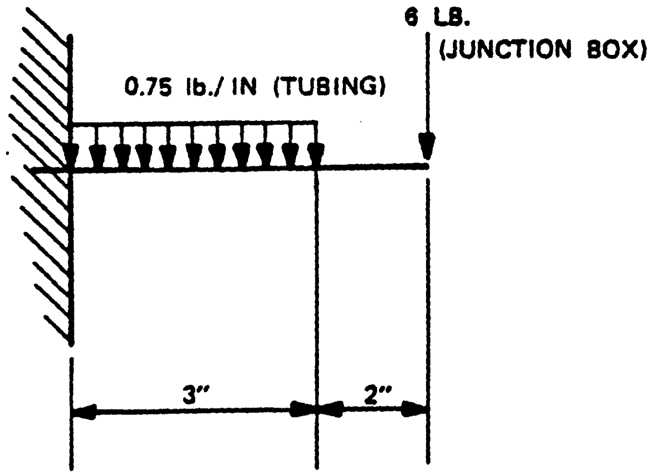
FOUNDATION UPLIFT MODEL
Figure 8.3-2



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PENETRATION MODEL FOR 8-FOOT
HORIZONTAL DROP ANALYSIS

Figure 8.4-1

CHAPTER 9

CONDUCT OF OPERATIONS

9.0 CONDUCT OF OPERATIONS

9.1 ORGANIZATIONAL STRUCTURE

The development of the Independent Spent Fuel Storage Installation (ISFSI) was managed by Carolina Power & Light (CP&L) Company with support from NUTECH Engineers, Inc. (NUTECH), the U. S. Department of Energy (DOE), and the Electric Power Research Institute (EPRI). Final responsibility for construction, preoperational testing, startup, and operation of the ISFSI remains with CP&L. Therefore, CP&L's organization and its interfaces with outside support organizations are described below.

9.1.1 CORPORATE ORGANIZATION

CP&L is engaged in the production, transmission, distribution, and sale of electric energy to residential, commercial, and industrial customers spread over a service area of 30,000 sq. mi. in North and South Carolina. The Company has extensive experience in the design, construction, startup, testing, operating, and staffing of modern generating facilities. The Company's management of nuclear activities is described in Reference 9.1.

9.1.1.1 Corporate Functions, Responsibilities, and Authorities

CP&L's corporate staff was responsible for installation engineering and design, quality assurance, and acceptance of testing results for the ISFSI. The Nuclear Fuels Section (NFS) and the Nuclear Engineering Department (NED) were jointly responsible for the engineering and design of the ISFSI and the acceptance of test results prior to operation.

An overall Program Manager within the NED was responsible for execution of the program until completion of fuel loading.

9.1.1.2 Applicant's In-House Organization

The corporate organization is shown in Reference 9.2, Figure 13.1.1-1. Ultimate responsibility for operation of the ISFSI rests with the Senior Vice President and Chief Nuclear Officer – Nuclear Generation Group reporting to Executive Vice President - Energy Supply. The details of the responsibilities of the CP&L Groups and Departments are described in Section 13.1 of the HBR2 Updated Final Safety Analysis Report (UFSAR) (Reference 9.2).

9.1.1.3 Interrelationships with Contractors and Suppliers

The development of the ISFSI was managed by CP&L with support from DOE and EPRI. NUTECH, CP&L's subcontractor, provides certain engineering, technical support, and other services for the program relating primarily to the NUTECH Horizontal Modular Storage (NUHOMS) system design. CP&L and other subcontractors provide the remaining engineering, technical support, and services.

9.1.1.4 Applicant's Technical Staff

The Corporate technical staff supporting the ISFSI is described in Section 13.1.1 of Reference 9.2.

9.1.2 OPERATING ORGANIZATION, MANAGEMENT, AND ADMINISTRATIVE CONTROLS SYSTEM

9.1.2.1 Onsite Organization

Operation of the Facility is the responsibility of CP&L, primarily by the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR2), Operations Unit, with support by the Robinson Engineering Section. The Nuclear Assessment Section (NAS) is responsible for performing periodic assessments of ISFSI activities. The on-site organization is described in Reference 9.2, Chapter 13.1.2.

9.1.3 PERSONNEL QUALIFICATION REQUIREMENTS

9.1.3.1 Minimum Qualification Requirements

Minimum qualification requirements are discussed in Reference 9.2 Section 13.1.3.

9.1.4 LIAISON WITH OUTSIDE ORGANIZATIONS

The development of the ISFSI was managed by CP&L with support from NUTECH, DOE, and EPRI. NUTECH, CP&L's subcontractor, provided certain engineering, technical support, and other services for the program relating primarily to the NUHOMS system design. The overall program was conducted under a Program Management Plan which had been approved under the DOE/CP&L Cooperative Agreement (Reference 9.4). This Program Management Plan provided for an oversight committee which includes representatives of CP&L, DOE, and EPRI. The oversight committee was responsible for input to, and approval of, the schedular, technical, procedural, and administrative details of the program. EPRI, acting through and in cooperation with CP&L (Reference 9.5), participated in the formulation and formal definition of the research and development aspects of the program. DOE participated in the program through financial and/or in-kind services (DOE/CP&L Cooperative Agreement). Program review and guidance was provided to CP&L by all participants. The design, procurement, construction, and operation of the HBR ISFSI are controlled through existing CP&L procedures and programs. Operation of the Facility is the sole responsibility of CP&L.

Pages 9.1-3 through 9.1-21
deleted by Amendment No. 6

Figure 9.1-1 and
Figure 9.1-2 deleted
by Amendment No. 6

9.2 PRE-OPERATIONAL TESTING AND OPERATION

Two types of tests are planned prior to loading of spent fuel. The first is thermal testing of the DSC/HSM system to verify its heat removal capacity; the second is testing of the DSC transfer system to ensure a safe, smooth DSC insertion into the HSM and back to the cask.

It will be necessary to perform the thermal tests only upon the first HSM and DSC (or prototype) constructed. Subsequent units built to the same requirements may be safely presumed to operate in a similar manner. Likewise, handling tests need only be performed once, since all future operations will use equipment proven during pre-operational testing or equipment fabricated from those plans.

9.2.1 ADMINISTRATIVE PROCEDURES FOR CONDUCTING TEST PROGRAM

Existing HBR2 modification procedures will be followed for the ISFSI test program.

9.2.2 CP&L TEST PROGRAM DESCRIPTION

Thermal testing will be performed using one of the completed HSMs. The DSC need not be an exact replica of the actual stainless steel canister, but may be simply an appropriately sized closed-end pipe. Resistance-type electric heating elements within the DSC replica will simulate decay heat. The purpose of this test is to verify the thermal-buoyancy driven convective air flow through the HSM.

Handling testing will be performed using the actual HSM, a DSC replica and the associated transfer handling equipment. The purpose of this series of tests is to ensure that all DSC handling operations will be performed safely. The tests will simulate, as nearly as possible, actual handling operations with spent fuel including mock-ups for performing the DSC welding operations. Off-normal handling conditions need not be addressed, since the DSC can remain in the cask indefinitely or be returned to the spent fuel pool, if proper alignment cannot be achieved.

9.2.3 TEST DISCUSSION

9.2.3.1 Physical Facilities Testing (Thermal Testing of HSM and DSC)

The DSC replica will be placed in the HSM and internally heated by electric resistance heaters. Heat flux will simulate the decay heat value of seven assemblies (7 kw). The purpose of the test is to verify the passive heat removal capacity of the HSM/DSC. The test will measure the difference between entering and exiting air temperatures. The expected response is that exit air will be 100EF above entering air temperture. The air flow volume will be measured to allow computation of heat flow by convection through the module.

This test will be repeated with partial and complete blockage of the HSM air passages.

If the results of the tests show that insufficient convective cooling occurs, resulting in elevated concrete and fuel cladding temperatures, corrective measures will be taken and the test repeated.

9.2.3.2 Operations Testing (Handling Tests)

The purpose of the handling tests is to verify that the DSC handling system is adequate to achieve alignment of the cask with the HSM and insertion of the DSC into the HSM.

The expected response of the system testing is that alignment and insertion may be achieved. A successful test is one in which alignment and insertion is achieved without damage to system components.

Corrective action, in the event of an unsuccessful test, will be to modify system components as appropriate.

Handling tests will also be conducted to ensure that the DSC can be removed from the HSM, returned to the IF-300 shipping cask, and returned to the HBR2 spent fuel pool.

9.3 TRAINING PROGRAM

9.3.1 PLANT STAFF TRAINING PROGRAM

The existing training program for the H. B. Robinson Steam Electric Plant Unit 2 (HBR2) has been modified to incorporate the operation of the Independent Spent Fuel Storage Installation. Since the ISFSI is essentially a passive structure with no equipment or instrumentation required for normal operation, the operator training program will concentrate on the accident training. In addition to the operator training, the radiation protection and fire brigade training programs have been expanded to cover the ISFSI. These training programs are expected to concentrate on fuel handling, canister loading, canister welding, canister/module interactions and retrieval.

The current HBR2 plant staff training program is described in Section 13.2.1 of the HBR2 Updated FSAR (Reference 9.2).

9.3.2 REPLACEMENT AND RETRAINING PROGRAM

The HBR2 replacement training and annual retraining programs have been updated to include the ISFSI.

The current HBR2 replacement and retraining programs are described in Section 13.2.2 of the HBR2 Updated FSAR (Reference 9.2).

9.4 NORMAL OPERATIONS

The H. B. Robinson ISFSI provides an independent and passive system for the dry storage of irradiated fuel. No radioactive waste or auxiliary systems are required during normal storage operations.

9.4.1 PROCEDURES

Although not considered normal operations, the processes for transfer of the fuel from the spent fuel pool to the horizontal storage modules (HSMs) are briefly discussed below. These processes will be conducted only when loading and unloading the ISFSI. Specific procedures will be developed for transporting, loading into the HSM, and retrieving operations involving the dry shielded canister (DSC). The existing HBR2 procedures for handling irradiated fuel in the spent fuel pool, loading of the GE IF-300 shipping cask, and handling by the spent fuel crane will be used for these ISFSI operations. As discussed in Chapter 6, some waste and auxiliary systems are required during the DSC loading, drying and transfer into the module. The waste systems handle the spent fuel pool water, air, and inert gas which are vented from the DSC and cask during drying. Auxiliary handling systems (such as hydraulic pressure, control, alignment, crane, etc.) are also required during the loading and transfer operation.

9.4.2 RECORDS

The ISFSI records will be maintained in accordance with existing HBR2 procedures.

9.5 EMERGENCY PLANNING

The Emergency Program for H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR2), has been determined to be adequate for events which might occur involving the Independent Spent Fuel Storage Installation. The Emergency Program for HBR2 consists of the Robinson Emergency Plan and its implementing Procedures. Also included are related radiological emergency plans and procedures of state and local organizations. The purpose of these programs 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 HBR2 Plant. The combined emergency preparedness programs have the following objectives:

a) Effective coordination of emergency activities among all organizations having a response role.

b) Early warning and clear instructions to the population-at-risk in the event of a serious radiological emergency.

c) Continued assessment of actual or potential consequences both on-site and off-site.

d) Effective and timely implementation of emergency measures.

e) Continued maintenance of an adequate state of emergency preparedness.

The HBR2 Emergency Plan has been prepared in accordance with Section 50.47 and Appendix E of 10 CFR 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 Plan. Other types of emergencies that do not have a potential for involving radioactive materials are not included in the Plan.

The activities and responsibilities of outside agencies providing an emergency response role at the HBR Plant site are summarized in the Plan (for completeness) and detailed in the State Emergency Plans.

The Plan provides the basis for performing advance planning and for defining specific requirements and commitments to be implemented by other documents and procedures. The HBR2 Plant site procedures provide the detailed actions and instructions that will be required to implement the Plan in the event of an emergency.

The emergency response resources available to respond to an emergency consist of the personnel at Corporate Headquarters, at other CP&L facilities, and, in the longer term, at organizations involved in the nuclear industry. 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 HBR2

staff are assigned defined emergency response roles that are to be assumed whenever an emergency is declared. The overall management of the emergency is normally performed by plant management. Onsite personnel have preassigned roles to support the Site Emergency Coordinator/Emergency Response Manager and to implement their directives. These roles, for the purpose of emergency planning, are cast in terms of emergency teams and assignments, each having a designated person assigned to it.

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 Robinson Plant Control Room, Operational Support Center, Technical Support Center, Emergency Operations Facility, Joint Information Center, Harris Energy & Environmental Center, and Forward Emergency Operations Center (FEOC). Descriptions of these facilities as well as the South Carolina EOC, the Darlington County EOC, the Lee County EOC, and the Chesterfield County EOC, are described in the Plan.

Emergency plan implementing procedures define the specific (i.e., step-by-step) 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:

a) Specific instructions to the plant operating staff for the implementation of the Plan.

b) Specific authorities and responsibilities of plant operating personnel.

c) A source of pertinent information, forms, and data to ensure prompt actions are taken and that proper notifications and communications are carried out.

d) A record of the completed actions.

e) The mechanism by which emergency preparedness will be maintained at all times.

9.6 DECOMMISSIONING PLAN

The HBR ISFSI makes provisions for the dry storage canister to be transferred to a federal repository when such a facility becomes operational. The concrete storage module is designed so that the canister can be safely returned to a shipping cask and transported offsite to the federal facility.

The shipping cask design and transportation requirements will depend on the regulations in effect at the time when the federal facility begins receiving spent fuel. In the absence of new regulations, the existing GE IF-300 shipping cask would be used to transport the canisters.

Upon removal of the canisters, the level of contamination within the concrete module is expected to be extremely low and would be removed so that the concrete could be broken-up and removed by conventional methods.

Section 3.5 of this report discusses decommissioning considerations for the HBR ISFSI.

REFERENCES: CHAPTER 9

9.1 Carolina Power & Light Company, Harris Nuclear Project, "Management Capability Report," letter from E. E. Utley (CP&L) to H. R. Denton (NRC) dated January 10, 1984.

9.2 Carolina Power & Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

9.3 Carolina Power & Light Company, "Technical Specifications and Bases for H. B. Robinson Unit No. 2," Appendix A to Facility Operating License DPR-23, Docket No. 50-261, Darlington County, SC.

9.4 CP&L/DOE Licensed At-Reactor Dry Storage Demonstration Program, Cooperative Agreement No. DE-FC06-84RL10532, Amendment No. A000, March 1984.

9.5 Associate Party Agreement Between CP&L and EPRI, Inc., RP2566-1, July 1984.

Figure 9.1-1 and
Figure 9.1-2 deleted
by Amendment No. 6

CHAPTER 10

OPERATING CONTROLS AND LIMITS

10.0 OPERATING CONTROLS AND LIMITS

The H. B. Robinson (HBR) Independent Spent Fuel Storage Installation (ISFSI) is a totally passive system which requires minimum operating controls. In the following sections, the operating controls and limits that are pertinent to the ISFSI are specified. The conditions and other items to be controlled are based on the safety assessments for normal and postulated accident conditions provided in Chapter 8 of this report.

The following operating controls and limits are specified:

10.1 Fuel Specifications

10.2 Limits for the Surface Dose Rate of the HSM While the DSC is in Storage

10.3 Limits for the Maximum Air Temperature Rise After Storage

10.4 Surveillance of the HSM Air Inlets

10.5 Surveillance of the HSM Inside Cavity

10.1 FUEL SPECIFICATIONS

1.1	Title:	Fuel Specifications	
1.2	Specifications: Type	15 x 15 PWR Fuel	
		Burnup	≤ 35,000 MWd/MT
		Initial (Beginning of Life) Enrichment	≤ 3.5% U-235
		Post Irradiation Time	≥ 5 years
		Weight Per Distance Between Adjacent Spacers Per Assembly	≤ 106.56 kg
		Distance Between Spacers	≤ 0.65 m

Any fuel not specifically filling the above requirements may still be stored in the ISFSI if all the following requirements are met:

Decay Power	≤ 1 kw/assembly
Neutron Source	1.67×10^8 n/sec/assembly
Gamma Source	5.73×10^{15} photons/sec/canister With spectrum bounded by that shown in Table 10.1-1
End of Life Fissile Content	0.8% U-235 0.5% Pu-239 0.1% Pu-241

1.3 Applicability: This specification is applicable to all fuel to be stored in the ISFSI.

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

1.5 Action: If this specification is not met, additional analysis and/or data must be presented before the fuel can be placed in the DSC.

1.6 Surveillance Requirements: The fuel selected for storage must have the parameter values specified in 1.2 above verified prior to fuel loading. No other surveillance is required.

1.7 Basis: The fuel parameters specified in this operating control and limit were selected to bound the types of PWR fuel which were in use at the time the HBR ISFSI was installed and loaded with spent fuel.

TABLE 10.1-1

ACCEPTABLE RADIOLOGICAL CRITERIA FOR
STORAGE OF MATERIAL IN THE HBR ISFSI

CRITERION	VALUE
GAMMA SOURCE PER CANISTER (total)	1.48×10^{16} Mev/sec
Fractional Breakdown[1]	
Above 1.3 Mev	0.004
Between 1.3 Mev and 0.8 Mev	0.114
Between 0.8 Mev and 0.4 Mev	0.808
Below 0.4 Mev	0.074
NEUTRON SOURCE PER CANISTER (total)[2]	1.17×10^{9} n/sec
Fractional Breakdown	
Above 5 Mev	5.40×10^{7} n/sec = 5.41%
Between 2.5 and 5 Mev	2.43×10^{8} n/sec = 24.32%
Between 1 and 2.5 Mev	4.56×10^{8} n/sec = 45.67%
Below 1 Mev	2.45×10^{8} n/sec = 24.53%

1 Fractional breakdown is based on isotopic composition and resulting gamma spectrum calculated by ORIGEN2 analysis.

2 Spectrum from U-235 fission, total number of neutrons per second from ORIGEN2 analysis.

10.2 LIMITS FOR THE SURFACE DOSE RATE OF THE HSM WHILE THE DSC IS IN STORAGE

2.1 Title: Surface Dose Rates on the HSM While the DSC is in Storage

2.2 Specification: Surface dose rates at the following locations

1) Outside of HSM door on centerline of DSC — 200 mrem/hr
2) Center of air inlets — 200 mrem/hr
3) Center of air outlets — 200 mrem/hr

Average Dose rates for the following surfaces

1) Roof — 50 mrem/hr
2) Front/Back — 50 mrem/hr
3) Side — 50 mrem/hr

Dose rates one meter from the center of the following surfaces of a unit of modules

1) Front/Back — 20 mrem/hr
2) Side — 20 mrem/hr

2.3 Applicability: This specification is applicable to the ISFSI.

2.4 Objective: The objective of this specification is to maintain as-low-as-reasonably-achievable dose rates on the modules.

2.5 Action: If the dose rates are exceeded, temporary shielding must be placed so as to reduce the dose rates to the specified levels. When temporary shielding is used, the outlet air temperature must be measured after the shielding is installed to verify that the air flow has not been restricted.

2.6 Surveillance: The HSM shall be monitored to verify that this specification has been met immediately after the DSC is placed in storage and the HSM front and rear accesses are closed.

2.7 Basis: The dose rates stated in this specification were selected to maintain as-low-as-reasonably-achievable exposures to personnel performing air duct clearing on the HSM. These dose rates are within industry accepted standards for contact handling, operation and maintenance of radioactive material. Maintenance personnel will be required to remove any potential air blockage. At 200 mrem/hr the dose for a one hour job of unblocking the air inlets (or outlets) would be less than 200 mrem (whole body) and hence would be only 4% of the total yearly burden. Furthermore, analysis provided in Chapter 7 of the HBR ISFSI SAR shows that the expected dose rates around the HSM surface will be well below the specifications listed above.

10.3 LIMITS FOR THE MAXIMUM AIR TEMPERATURE RISE AFTER STORAGE

3.1	Title:	Maximum Air Temperature Rise from HSM Inlet to Outlet
3.2	Specification:	Maximum air temperature rise l00°F (55.6°C)
3.3	Applicability:	This specification is applicable to the ISFSI.
3.4	Objective:	To limit the maximum air temperature around the DSC.
3.5	Action:	If the temperature rise is greater than l00°F (55.6°C), the air inlets and exits should be checked for blockage. If the blockage is cleared and the temperature still exceeds this limit, the DSC must be removed from the module or additional information and analysis shall be provided that will prove the existing condition does not represent an unsafe condition.
3.6	Surveillance:	The temperature rise shall be checked at the time the DSC is stored in the HSM, again 24 hours later, and again after 7 days.
3.7	Basis:	The l00°F (55.6°C) temperature rise was selected to limit the hottest rod in the DSC to below 716°F (380°C). If this temperature rise is maintained, then the hottest rod will be below the 716°F (380°C) limit even on the hottest day conditions of 125°F (5l.7°C). The expected temperature rise is less than 100EF (i.e., 82EF (45.5EC) per NUHOMS Topical Report (Reference 8.1), Section 8.1.3) and hence, the current design provides adequate margin for this specification. If the temperature rise is within the specifications, then the HSM and DSC are performing as designed and no further temperature measurements are required during normal surveillance.

10.4 SURVEILLANCE OF THE HSM AIR INLETS AND OUTLETS

4.1 Title: Surveillance of the HSM Air Inlets and Outlets

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

4.3 Applicability: This specification is applicable to the ISFSI.

4.4 Objective: To assure that no HSM air inlets or outlets are plugged for more than 48 hours and to assure that complete blockage of all inlets and exits due to an accident will be removed in less than 48 hours.

4.5 Action: If the air inlets or outlets are plugged, they should be cleared. If the screen is damaged, it should be replaced.

4.6 Surveillance: The HSM shall be inspected to verify that the air inlets are free from obstructions.

4.7 Basis: Analysis in Chapter 8 of the HBR ISFSI SAR showed that no temperature limits are exceeded if a module is completely plugged for 48 hours. Furthermore, analysis showed that blockage of the air inlet alone did not result in unacceptable temperatures. Therefore, for normal operations, an inspection of the inlets once per day will assure that any local obstructions can be removed. Likewise, after an accident (such as those described in Chapter 8) the HSMs should be examined within 24 hours to assure that if a module is completely buried, flow can be restored within another 24 hours.

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CHAPTER 11

QUALITY ASSURANCE

11.0 QUALITY ASSURANCE

The activities associated with the Independent Spent Fuel Storage Installation (ISFSI) are performed under the guidance of applicable portions of the CP&L Quality Assurance (QA) Program delineated in Reference 11.1. Controls are established for the applicable activities which CP&L performs as well as for those appropriate activities which subcontractors perform.

Nuclear Assessment personnel in the H. B. Robinson Nuclear Assessment Section perform oversight to assure that appropriate quality requirements are being met in the normal implementation of technical programs. Work associated with this project will be performed in the same way as other nuclear plant related activities, and will, therefore, be covered by the normal assessment process.

11.1 QUALITY ASSURANCE

It is the policy of CP&L to design, construct, and operate nuclear power plants without jeopardy to its employees or to the public health and safety. The QA programs shall be developed, implemented, and updated as necessary to assure that the Company's nuclear facilities will be managed such that all systems used to produce, convey, or use nuclear generated steam and all systems used to treat, store, or convey waste produced by the generation of nuclear steam will be designed, constructed, and operated in a safe manner. Deviations from this program shall be permitted only upon written authority from the corporate management position originally approving the program or implementing procedures.

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

The operation of the Company's ISFSI shall be in accordance with the terms and conditions of the ISFSI material license issued by the NRC (10 CFR Part 72). Changes in operating procedures, experiments at the ISFSI, modifications to the ISFSI hardware or systems, shall be made in accordance with the terms and conditions of the ISFSI material license.

The Dry Storage Canister and Transfer Cask are considered safety-related and are subject to a QA program in conformance with the requirements of 10 CFR 50, Appendix B and the QA Program described in Reference 11.1.

This QA program will be applied to those structures, systems, and components of the Horizontal Storage Module (HSM) and foundation that are important to safety. The program will apply for the continued inspection, testing, operation, maintenance, repair, and modification of these HSM structures, systems, and components which are important to safety.

A Radwaste QA Program was applied to the HSM and foundation during the procurement, construction, and testing phases of the project. After these phases were completed and the initial loading of the Dry Shielded Canisters (DSCs) was completed, the H. B. Robinson ISFSI deemed "operational." Keeping in mind that the NUHOMS system being utilized for the H. B. Robinson ISFSI is a totally passive installation, there are no pieces of equipment which require operation nor a requirement for data collection and reporting.

However, as stated in Section 5.1.1.7 of this document, a daily visual inspection of air inlets and outlets will be made to insure that they remain unblocked and the integrity of the screens remains intact. In addition, it is stated in Section 7.3.4 of this document that "The operation of the ISFSI will be monitored under the H. B. Robinson Unit 2 Radioactivity Monitoring Program. No additional radiation monitoring instrumentation is required." Operation of the H. B. Robinson ISFSI will be as established by the requirements of the Technical Specifications contained in the Safety Analysis Report and will be carried out in accordance with the QA program.

Assessments will be performed by the Nuclear Assessment Section at least every two years. Assessments and other management oversight activities shall be performed in accordance with the QA Program for safety-related activities.

11.2 H. B. ROBINSON QUALITY ASSURANCE PROGRAM

The H. B. Robinson (HBR) QA Program is controlled by the policies and requirements of the QA Program. These policies and requirements are implemented through the Plant Operating Manual and other approved procedures. The program is designed to ensure compliance with the applicable NRC Regulatory Guides and ANSI Standards.

Details of the HBR QA program are contained in Reference 11.1.

11.3 NUTECH QUALITY ASSURANCE

The HBR ISFSI is designed by NUTECH Engineers, Inc. The quality assurance followed by NUTECH is described in Reference 11.2 and has been approved by CP&L Quality Assurance.

REFERENCES: CHAPTER 11

11.1 Carolina Power & Light Company, "H. B. Robinson Steam Electric Plant Unit No. 2 Updated Final Safety Analysis Report," Docket No. 50-261, License No. DPR-23.

11.2 NUTECH Engineers, Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-001, Revision 1, November 1985.