April 1, 2020
DPG 20-045

ATTN: Document Control Desk
Director, Division of Fuel Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Rancho Seco Independent Spent Fuel Storage Installation
Materials License No. SNM-2510
Docket 72-11

RANCHO SECO ISFSI FINAL SAFETY ANALYSIS REPORT UPDATE

Attention: Andrea Kock

This letter serves as the update of the Final Safety Analysis Report (FSAR) for the Rancho Seco Independent Spent Fuel Storage Installation (ISFSI) as required by renewed License SNM-2510 Condition 20.

On March 9, 2020, the NRC renewed SNM-2510 for a 40 year period of extended operation. (ISSUANCE OF RENEWED MATERIALS LICENSE NO. SNM-2510 FOR THE RANCHO SECO INDEPENDENT SPENT FUEL STORAGE INSTALLATION (CAC/EPID NOS. 001028/L-2018-RNW-0005 AND 000993/L-2018-LNE-0004)(ADAMS ML20065N277)

Condition 20 of the renewed license states "Within 90 days after issuance of the renewed license, SMUD shall submit an updated final safety analysis report (FSAR) to the U.S. Nuclear Regulatory Commission (NRC), in accordance with 10 CFR 72.4 and continue to update the FSAR pursuant to the requirements in 10 CFR 72.70(b) and (c). The updated FSAR shall reflect the information provided in Appendix C of the Rancho Seco ISFSI License Renewal Application, Revision 3, dated July 12, 2019 (Agency wide Documents Access and Management System (ADAMS) Accession No. ML19204A248). The licensee may make changes to the updated FSAR, consistent with 10 CFR 72.48(c)"

The Rancho Seco Independent Spent Fuel Storage Installation Final Safety Analysis Report, Revision 8, reflects the information provided in Appendix C of the Rancho Seco ISFSI License Renewal Application, Revision 3, dated July 12, 2019 (ADAMS Accession No. ML19204A248), as required. This update is reflected in the list of effective pages as included in the enclosure. The changes to the Final Safety Analysis Report resulting from license renewal did not require revision of Rancho Seco

Independent Spent Fuel Storage Installation Final Safety Analysis Report Volumes II, III, or the Appendices.

Safety evaluations performed in accordance with 10 CFR 72.48 during the license renewal process, which identified the need to update the ISFSI FSAR since the last biennial update of 2018, have not been incorporated into Revision 8, but will be incorporated into Revision 9, which will be submitted with SMUD's 2020 biennial update, as required by 10 CFR Part 72.70(c)(6).

This submittal is provided as an electronic submission, using the Electronic Information Exchange (EIE) provision from Section 3.1 of the NRC Guidance for Electronic Submissions to the NRC.

As required by 10 CFR 72.70(c)(2), this submittal includes an updated list of effective pages indicating the pages revised by Revision 8.

As required by 10 CFR 72.70(c)(3), each changed (or new) page includes change bars in the left margin, as well as updated footers indicating "Rev. 8, April 2020."

By signature below, the Sacramento Municipal Utility District certifies that the above is true and correct. If you or members of your staff have questions requiring additional information or clarification, you may contact me at (916) 732-4893.

Sincerely,

Dan A. Tallman
Manager, Rancho Seco Assets

cc:

William Allen, NRC (w/o enclosures)
NRC, Region IV (w/o enclosures)
RIC 1F.099

Enclosure:
1. Rancho Seco Independent Spent Fuel Storage Installation Final Safety Analysis Report, Revision 8.

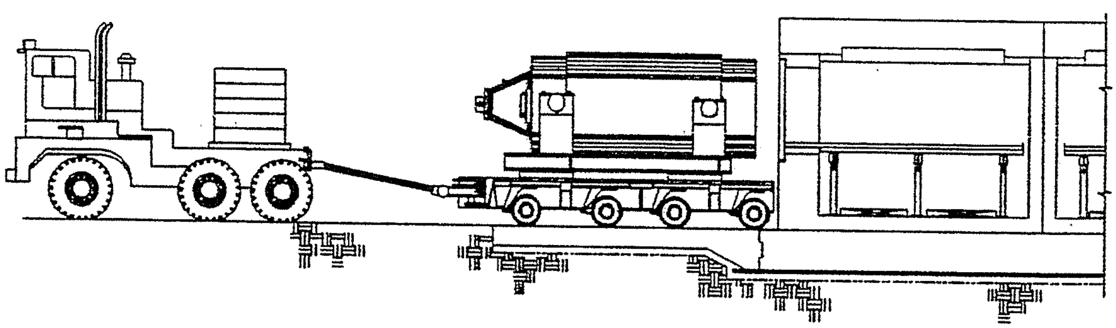
Rancho Seco

Independent Spent Fuel Storage Installation

Final Safety Analysis Report

Volume I

ISFSI System



SMUD

Sacramento Municipal Utility District

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8.2-21	Revision 4A
8.2-22	Revision 4A
8.2-23	Revision 4A
8.2-24	Revision 4A
8.2-25	Revision 4A
8.2-26	Revision 4A
8.2-27	Revision 4A
8.2-28	Revision 4A
8.2-29	Revision 4A
8.2-30	Revision 4A
8.2-31	Revision 4A
8.2-32	Revision 4A
8.2-33	Revision 4A
8.2-34	Revision 4A
8.2-35	Revision 4A
8.2-36	Revision 4A
8.2-37	Revision 4A
8.2-38	Revision 4A
8.2-39	Revision 4A
8.2-40	Revision 4A
8.2-41	Revision 4A
8.2-42	Revision 4A
8.2-43	Revision 4A
8.2-44	Revision 4A
8.2-45	Revision 4A

APPENDIX C

1-1	Revision 0
1-2	Revision 0
1-3	Revision 0
2-1	Revision 0
3-1	Revision 0

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LIST OF ACRONYMS

ACI	American Concrete Institute
AGM	Assistant General Manager
AISC	American Institute of Steel Construction
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&W	Babcock & Wilcox
DOE	U.S. Department of Energy
DSAR	RSNGS Defueled Safety Analysis Report
DSC	Dry Shielded Canister
FC-DSC	Fuel/Control Component DSC
FF-DSC	"Failed" Fuel DSC
FO-DSC	Fuel Only DSC
GM	General Manager
HSM	Horizontal Storage Module
IOSB	Interim Onsite Storage Building
ISFSI	Independent Spent Fuel Storage Installation
MP187	NUHOMS® Multi Purpose (Transfer and Transportation) Cask
NDRC	National Defense Research Committee
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NUHOMS	NUTECH Horizontal Modular Storage
NUREG	Nuclear Regulatory Guide
OSHA	Occupational Safety and Health Administration
PWR	Pressurized Water Reactor
RSNGS	Rancho Seco Nuclear Generating Station
SAR	Safety Analysis Report
SFA	Spent Fuel Assembly
SMUD	Sacramento Municipal Utility District
VDS	Vacuum Drying System

1. INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

1.1 Introduction

The Nuclear Regulatory Commission (NRC) issued an operating license, DPR-54, for Rancho Seco Nuclear Generating Station (RSNGS) in August 1974, and the plant began commercial operation in April 1975. However, as a result of a public referendum of Sacramento Municipal Utility District (SMUD) voters on June 6, 1989, RSNGS has ceased operation, and the reactor has been permanently defueled. Accordingly, on May 20, 1991, SMUD submitted its Proposed Decommissioning Plan [1.6.1] to the NRC discussing the method to be used to decommission RSNGS. This plan was approved by an NRC order dated March 20, 1995. SMUD subsequently revised the decommissioning plan to a Post Shutdown Decommissioning Activities Report (PSDAR) to meet the requirements of the revised regulations (10 CFR 50.82) regarding decommissioning.

Consistent with Rancho Seco PSDAR, the Independent Spent Fuel Storage Installation (ISFSI) is intended to provide dry storage capacity for Rancho Seco spent nuclear fuel and Greater than Class C (GTCC) radioactive waste. The storage system was designed for 50-year service, and initially licensed for 20 years in accordance with 10 CFR 72. On March 9, 2020, the NRC approved renewal of the ISFSI license (SNM-2510) for an additional 40 years. The aging management activities associated with this renewal applies to Amendment 4. Any future amendments will include an aging management review (AMR) and any associated required aging management activities. The current aging management results are detailed in Chapter 9, Section 9.8.

The original ISFSI FSAR chapters indicate design life and service life values of 50 years[1]. The new design life is 60 years. Time-limited aging-analyses (TLAAs) to assess SSCs that have a time-dependent operating life to demonstrate that the existing licensing basis remains valid and that the intended functions of the SSCs in scope of renewal are maintained during the period of extended operation (PEO) to 60 years are detailed in Chapter 9, Section 9.8.4.

Construction of the Rancho Seco ISFSI was completed during 1996 and the initial license was received on June 30, 2000. All fuel was in dry storage at the ISFSI in August 2002 and the single GTCC waste canister was loaded at the ISFSI in August 2006.

1.1.1 Principal Function of the Installation

The Rancho Seco ISFSI design provides temporary dry storage for 100% of the Rancho Seco spent fuel assemblies (SFAs) and GTCC waste in order to complete full plant dismantlement. It is designed with safety features that eliminate the need for an operable spent fuel pool to recover from certain unlikely accident scenarios. The spent fuel will be stored in this manner until it is accepted by the Department of Energy (DOE).

1.1.2 Location of the ISFSI

The Rancho Seco ISFSI is located within the Owner Controlled Area of the Rancho Seco site which is owned and operated by SMUD. The Rancho Seco site comprises approximately 2,480 acres in Sacramento County, California. It is characterized by isolation from population centers, a sound foundation for structures, and favorable conditions of meteorology, seismology, and hydrology.

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[1] *The terms design life and service life are equivalent and interchangeable.*

The location of the ISFSI site within the Rancho Seco site is approximately 600 feet west of the Interim Onsite Storage Building (IOSB). The Owner Controlled Area boundary lies approximately 1200 feet to the west of the ISFSI and 1500 feet to the north. Figure 1-1 shows a general layout of the site.

1.2 General Description of the Installation

The installation exists on a concrete slab approximately 225 feet long, 170 feet wide, and 2 feet thick at the location of the HSMs. The primary mode of storage is within the Horizontal Storage Modules (HSMs) that will be located on the ISFSI pad. The slab is surrounded by a security fence. Figure 1-2 shows a general view of the Rancho Seco ISFSI layout.

The principal ISFSI design criteria are provided in Chapter 3, but are summarized as follows:

Installation Capacity	All RSNGS Fuel (493 SFAs)
Fuel Parameters	Bound RSNGS SFAs
Design Life[1]	60 Years
Earliest Operation[2]	6/96
Maximum Crane Load	130 tons
Environmental Conditions	Bound RSNGS Conditions
ISFSI Fence Dose Rates	≤2 mrem/hr
Site Boundary Dose Rates	≤25 mrem/yr
Criticality Factor	≤0.95

Details of the design criteria and design descriptions, for the ISFSI components used for storage are provided in Chapters 3 and 4 of Volume I (for the Dry Shielded Canisters (DSCs)), Volume II (for the HSMs), and Volume III (for the cask). A summary overview of these components follows.

Note:

Initially, the MP187 cask was intended to be licensed under 10 CFR 72 for storage of a DSC if required to recover from an off-normal event at the ISFSI. Accordingly, much of the original analysis addressed vertical storage of a loaded DSC in the cask at the ISFSI. Although the cask is no longer being licensed for storage under 10 CFR 72, many of the calculational results remain bounding and are still relevant to this SAR revision.

1.2.1 Horizontal Storage Module

The HSM is a low profile, reinforced concrete structure designed to withstand all normal condition loads as well as the abnormal condition loads created by earthquakes, tornadoes, flooding, and other natural phenomena. The HSM is also designed to withstand abnormal condition loadings postulated to occur during design basis accident conditions such as a complete loss of ventilation.

1. The expected life is much greater (hundreds of years), however 60 years is assumed for service conditions.

2. This date is used to determine radiological sources and heat loads.

The Rancho Seco HSM design is similar to the Standardized NUHOMS®-24P HSM design [1.6.2]. The general features of the HSM are shown in Figure 1-3.

Quantity	22[3]
Capacity Each	1 Dry Shielded Canister (DSC)
Arrangement	2x11 Array
Size	15'-0" H, 9'-8" W, 19'-0" L
Approximate Weight, Empty	242,000 lbs

1.2.2 Dry Shielded Canister (DSC)

The DSC is a high integrity stainless steel, welded pressure vessel that provides confinement of radioactive materials, encapsulates the fuel in an inert atmosphere, and provides biological shielding (in the axial direction) during DSC closure, transfer and storage. It provides full canisterization for the fuel prior to storage in either the HSMs or cask.

Since the Rancho Seco ISFSI must provide 100% storage for RSNGS fuel and control components, three types of DSCs are required. The design requirements and design descriptions for each of the three DSCs are provided in Chapters 3 and 4 of this Volume. A general overview of the DSC is shown in Figure 1-4.

The Rancho Seco DSC designs are based on the Standardized NUHOMS®-24P DSC design [1.6.2], except that they include fixed neutron absorbers in order to expedite the licensing of the NUHOMS®-MP187 for offsite transport (10 CFR 71). In addition, modifications have been made to the cavity, basket, and spacer disc designs to qualify the Rancho Seco DSCs for offsite transport. These modifications are described in more detail in Section 4.2.5.2.

Quantity	21
All 3 DSC Types:	
External Size	67.25"ϕ x 186"
Shell Thickness (nominal)	0.625"
Approximate Weight, Loaded	81,000 lbs
Internal Atmosphere	Helium

[3] The Rancho Seco ISFSI uses 21 HSMs for storing spent nuclear fuel and the 22nd HSM for storing GTCC waste.

Fuel/Control Components (FC) DSC:	
Capacity Each	24 SFAs + CCs
Internal Cavity Length	173"
Neutron Absorber	Borated Panels
Quantity	18
Fuel-Only (FO) DSC:	
Capacity Each	24 SFAs
Internal Cavity Length	167"
Neutron Absorber	Borated Panels
Quantity	2
"Failed"-Fuel (FF) DSC:	
Capacity Each	13 SFAs
Internal Cavity Length	167"
Neutron Absorber (criticality control by geometry)	N/A
Quantity	1

1.2.3 NUHOMS®-MP187 Cask

The design of the Rancho Seco ISFSI includes one NUHOMS®-MP187 cask which is licensed under 10 CFR 71. The cask can be used for on-site transfer and offsite transport of a DSC without the need for additional fuel handling. This eliminates any need to return a DSC to the RSNGS spent fuel pool and allows abandonment of this pool as part of RSNGS' decommissioning.

The cask transfer mode is functionally identical to that described in the Standardized NUHOMS® SAR [1.6.2]. It provides the biological shielding and structural support necessary to carry a DSC through the various phases of drying, sealing, and transfer to the HSM for storage.

The cask has a transport mode which, although not the subject of this license application, is worth noting because the offsite transport requirements form the basis for many of the NUHOMS®-MP187 Cask's design features.

A general overview of the cask is shown in Figure 1-5. The cask design criteria and description are provided in Volume I, Chapters 3 and 4, and are summarized as follows:

Quantity	1
Capacity Each	1 Dry Shielded Canister
Size	91.5"ϕ x 201.5"
Approximate Weight, Empty	160,000 lbs
Gamma Shielding	Lead
Neutron Shielding	Castable Hydrogenous Solid Material

After the on-site fuel transfer campaign is completed, SMUD may make the cask available for use offsite.

1.3 General Systems Description

1.3.1 Storage Systems

The Rancho Seco ISFSI Storage System is comprised of the system elements described in Section 1.2 above. Figure 1-6 is a diagram of the Storage System which indicates the major consumables and waste streams for each phase of operation.

Other than the primary storage system (DSCs, HSMs, and the cask), there are no additional systems required for the safe storage of Rancho Seco fuel and control hardware. The following ancillary systems are present at the storage site: lighting, security systems including CCTV and intrusion detection, temperature monitoring, and lightning protection.

Since there is no waste generated during the storage phase, there are no gaseous, liquid, or solid radioactive waste treatment systems associated with the storage system. Likewise, heat removal is totally passive in the HSMs and no cooling system is required.

1.3.2 Transfer System

The Transfer System is designed to safely move loaded DSCs from the Fuel Storage Building to storage, or to retrieve loaded DSCs from storage in preparation for shipping. The Transfer System components are a prime mover (modified semi-tractor) and dedicated trailer, a cask skid, a skid positioning system (integral with the trailer), and a hydraulic ram system for inserting and withdrawing loaded DSCs from the HSMs.

The general arrangement of the transfer system is shown in Figure 1-7. Further description of the Transfer System components is provided in Section 4.0 of this Volume. The operation of the Transfer System is described in Section 5.0 (Volume II for HSM storage).

1.3.3 Auxiliary Systems

Five auxiliary systems are required for DSC drying and sealing operations.

1.3.3.1 The Vacuum Drying System (VDS)

The VDS provides a means for removing liquid water and water vapor from the DSC, and backfilling it with helium. Once all the water has been forced out of the DSC cavity with compressed air, nitrogen or helium, the remaining moisture contained within the cavity is removed with a vacuum drying system. The vacuum drying system evacuates the DSC cavity and lowers the moisture content to an acceptable level.

The suction line of the vacuum drying system is connected to the DSC vent and siphon ports. A particle filter is located on the suction side of the vacuum drying system to keep debris out of the unit. A drain in the vacuum suction line allows any liquid water remaining in the DSC cavity to be removed. The vacuum drying system is operated such that all radioactive material is confined within a controlled system.

1.3.3.2 The Welding System

DSCs are seal welded using an automatic welding system.

The canister Automatic Welding System consists of two major components, the welding machine and the control panel/power supply. The control panel and power supply, along with the purge gas bottle, can be located at any convenient position for the operator within the range of the umbilical cables. The use of an automatic welding machine is essential for ALARA operations in routine use. Manual welding of any of the closure welds is permissible but is recommended only for purposes of weld repair or as a recovery procedure if the machine becomes non-operational during the closure process. Small weldments such as the vent and siphon port plug seals are made manually as part of routine operations because the weld joint is not suitable for automatic welding.

1.3.3.3 The Waste Processing System

VDS exhaust and general cask decontamination waste are generated during DSC drying and sealing operations. Decontamination waste will be managed in accordance with the requirements of the RSNGS 10 CFR 50 license.

1.3.3.4 The Security System

Intrusion detection is provided at the ISFSI as described in the Rancho Seco ISFSI Physical Protection Plan.

1.3.3.5 The Temperature Monitoring System

Instrumentation is provided for monitoring HSM temperature. The signals will be incorporated into the RSNGS Plant Integrated Computer System (PICS). Eventually, the signals will be transmitted to SMUD headquarters in Sacramento. Local readout is also available in the ISFSI Electrical Building.

1.4 Identification of Agents and Contractors

SMUD is responsible for the engineering, design, licensing, and construction of the Rancho Seco ISFSI site. SMUD has also participated in a demonstration project with DOE to provide information to the nuclear utility industry regarding the use of a transportable storage system.

Transnuclear West (TNW) is the prime contractor for the design and fabrication of the HSMs, DSCs, and associated auxiliary systems. TNW is also the prime contractor for the cask supplier and is responsible for cask transportation licensing, fabrication, testing, delivery to the site, and delineation of any cask specific requirements.

SMUD has used various contractors for site preparation and construction, as necessary.

1.5 Material Incorporated by Reference

The Standardized NUHOMS®-24P SAR [1.6.2] and several other documents related to the licensing of RSNGS under 10 CFR 50 are already on file or docketed with the NRC and are referenced throughout this SAR.

The NUHOMS®-MP187 Transportation Cask Safety Analysis Report [1.6.3] was submitted to the NRC Transportation Branch in parallel with Revision 1 of this application. It contains descriptions and analysis of the cask for transportation conditions and was written for review under 10 CFR 71. The NUHOMS®-MP187 Transportation Cask Safety Analysis Report is referenced in this SAR in instances where transportation requirements bound those imposed by 10 CFR 72. The NRC issued the transportation Certificate of Compliance for the MP187 cask on September 10, 1998.

1.6 References

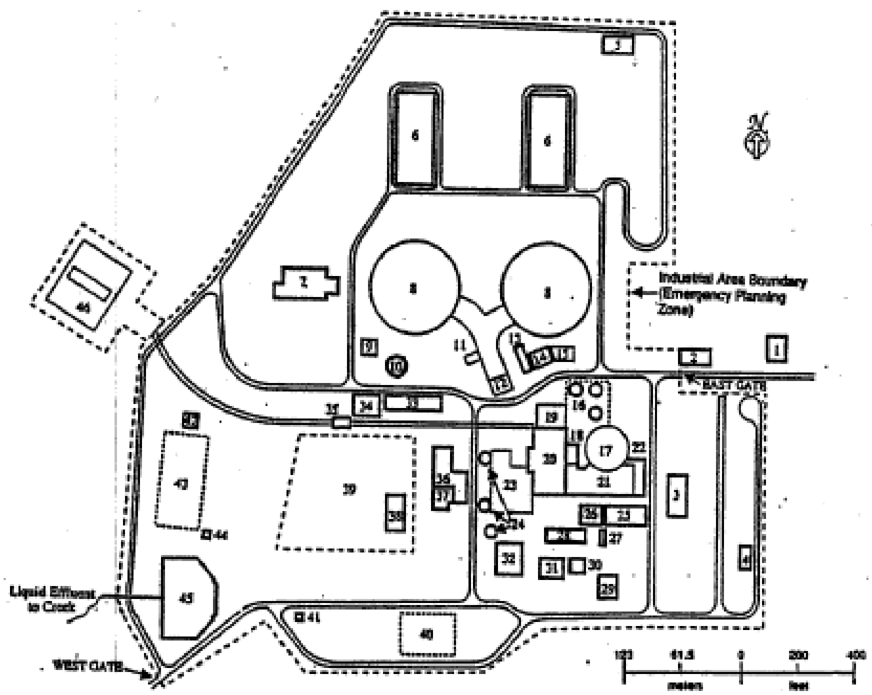
1.6.1 "Rancho Seco Decommissioning Plan" as approved by NRC Order dated March 20, 1995 (TAC No. M80518), USNRC Docket No. 50-312.

1.6.2 "Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.

1.6.3 "Safety Analysis Report for the NUHOMS®-MP187 Multi-Purpose Cask," NUH-05-151, Revision 9, Docket 71-9255, VECTRA Technologies, Inc., September 1998.

Figure 1-1

Rancho Seco ISFSI Location



LEGEND

1. Receiving Warehouse
2. Personnel Access Portal (PAP) EAST GATE
3. Administration Building
4. L & D Building
5. Quonset Building
6. Spray Ponds *
7. Interim Onsite Storage Building
8. Cooling Towers *
9. Microwave Building
10. Diesel Oil Storage Tank *
11. Plant Cooling Water Intake
12. Intake Structure
13. Chlorine Building *
14. Water Treatment Plant
15. Sewage Treatment Area
16. Tank Farm
17. Reactor Building
18. Fuel Storage Building
19. Turbine Building Gantry Crane Access Area
20. Turbine Building
21. Auxiliary Building
22. Radwaste Solidification Building
23. Transformer Yard
24. Regenerative Holdup Tanks
25. Training and Records (T&R) Building
26. Nuclear Service Electrical Building
27. Old CAS Building
28. Tool Room
29. Cask Support facility
30. Electrical Fab Shop
31. CRS Warehouse
32. Fab Shop
33. Warehouse
34. Fab Shop
35. Old Turbine Storage Area
36. "A" Warehouse
37. Machine Shop
38. Switchyard Control Building
39. Switchyard
40. Material Storage Area
41. Reservoir Filter Area
42. Sewage Treatment
43. Sewage Treatment Aeration Pond
44. Chlorine Building
45. Retention Basin
46. ISFSI

* Out of Service

Figure 1-2

ISFSI Layout

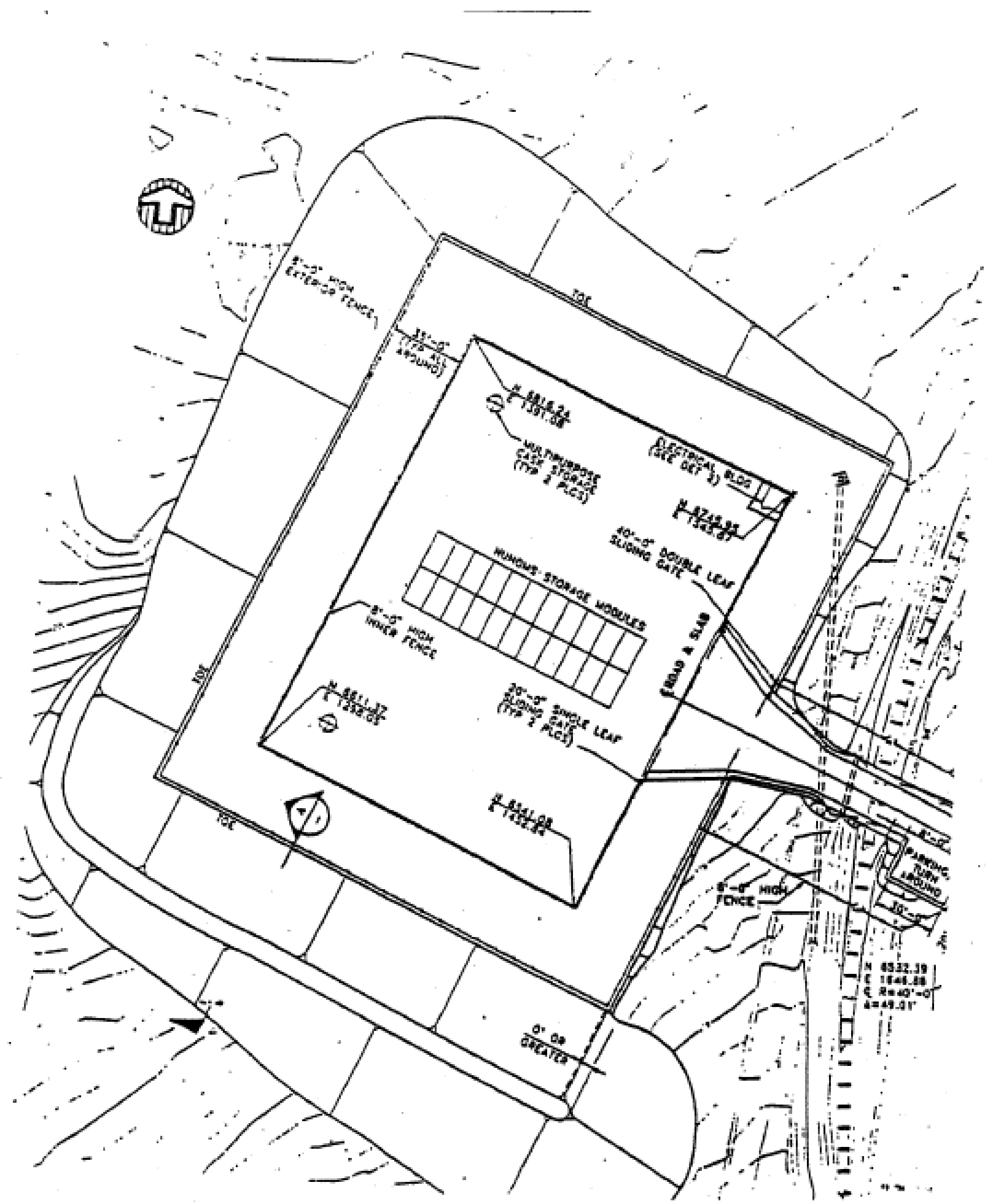


Figure 1-3

Overview of the Horizontal Storage Module

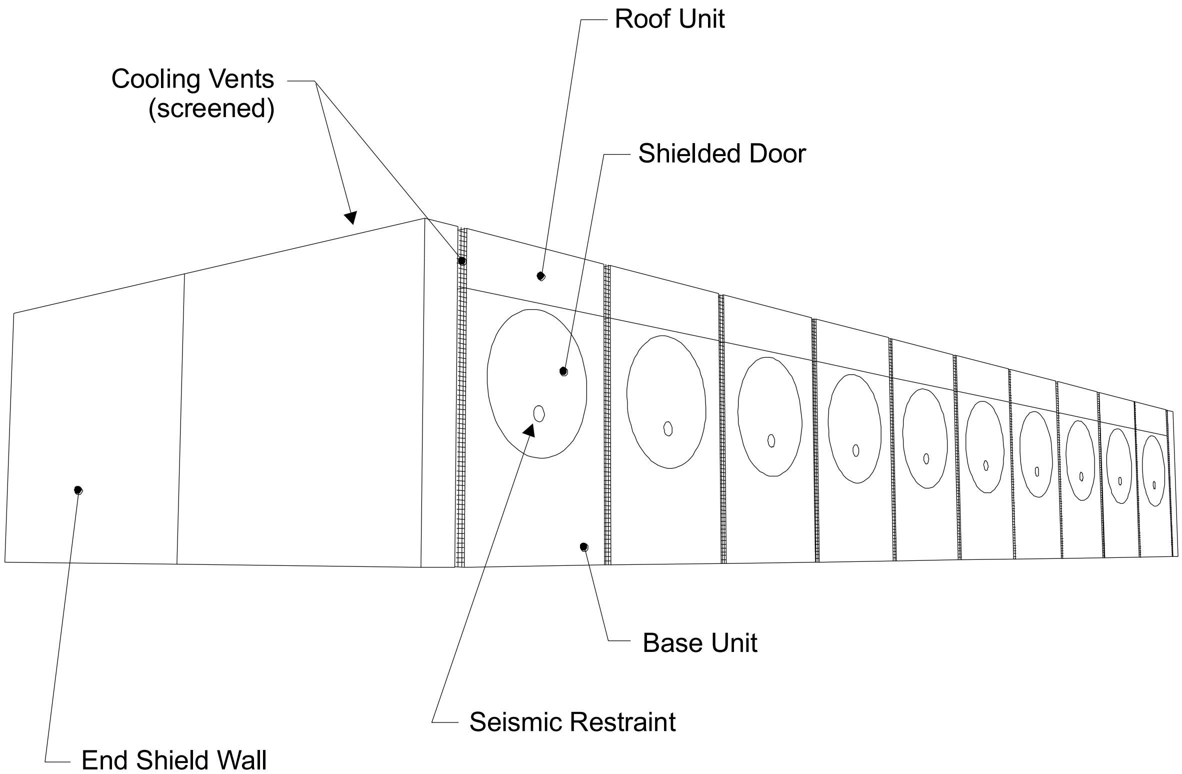


Figure 1-4

Overview of the Dry Shielded Canister

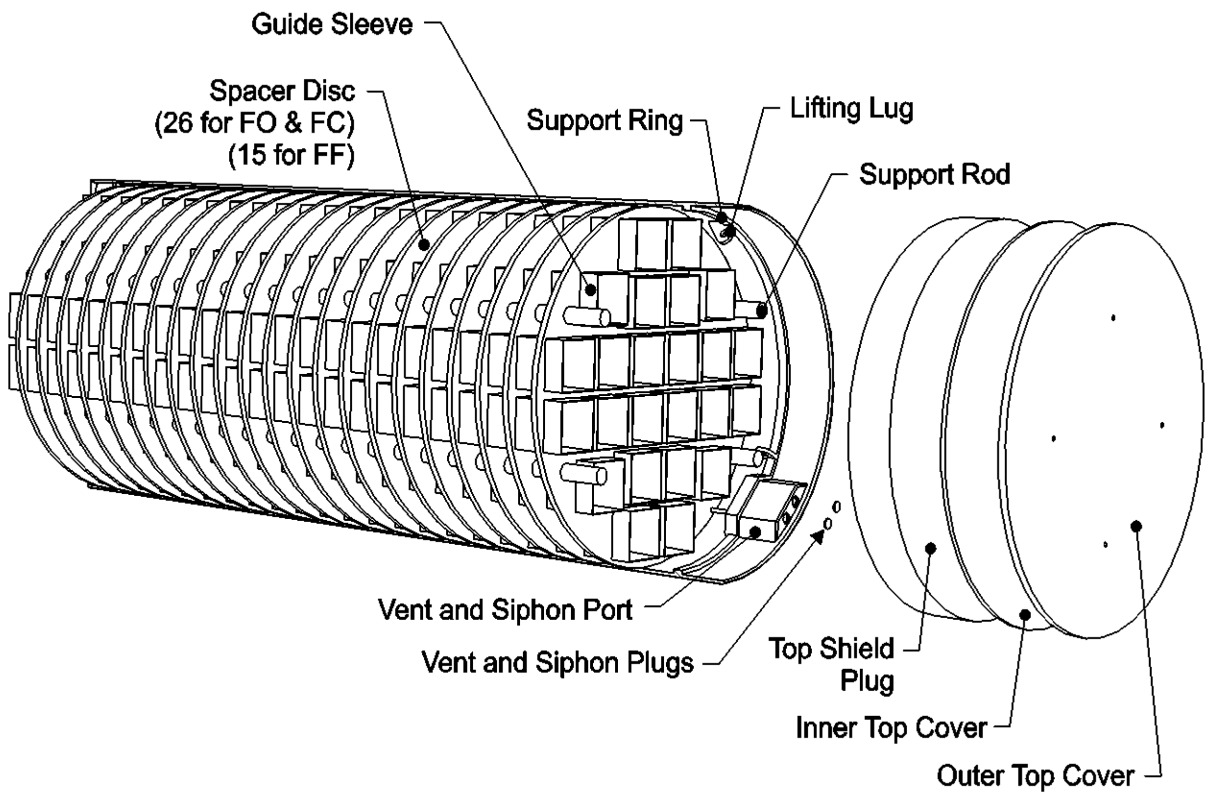


Figure 1-5

Overview of the Cask

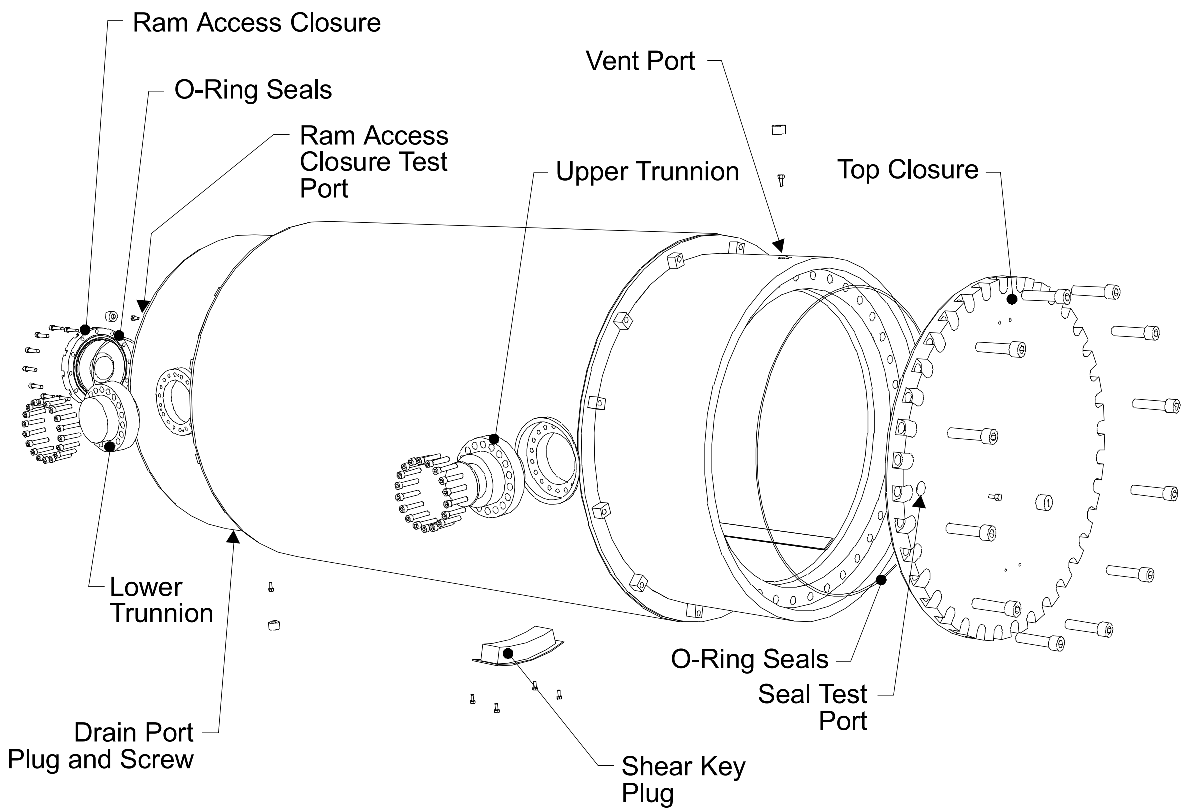


Figure 1-6

Rancho Seco ISFSI Storage System Flowchart

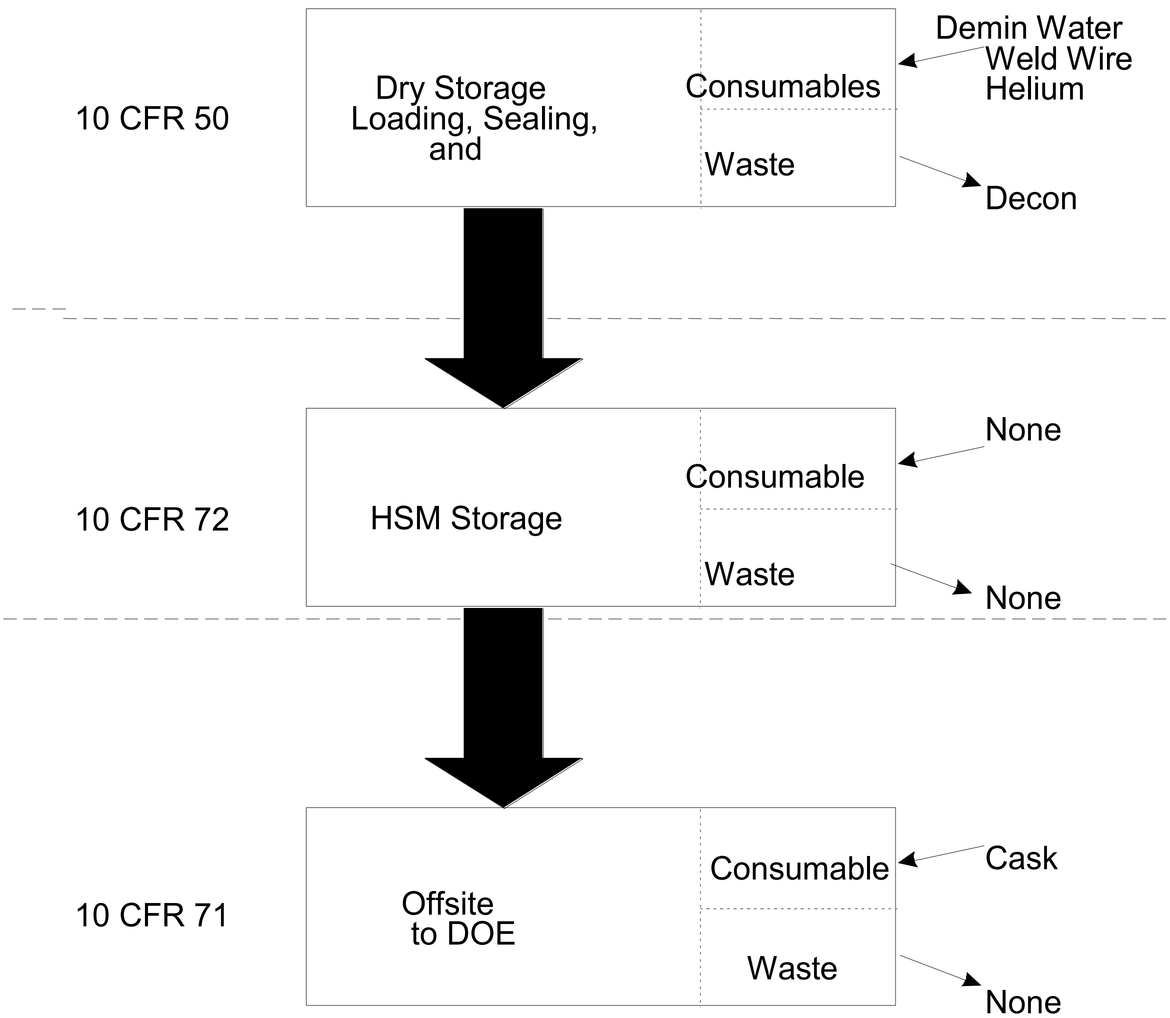
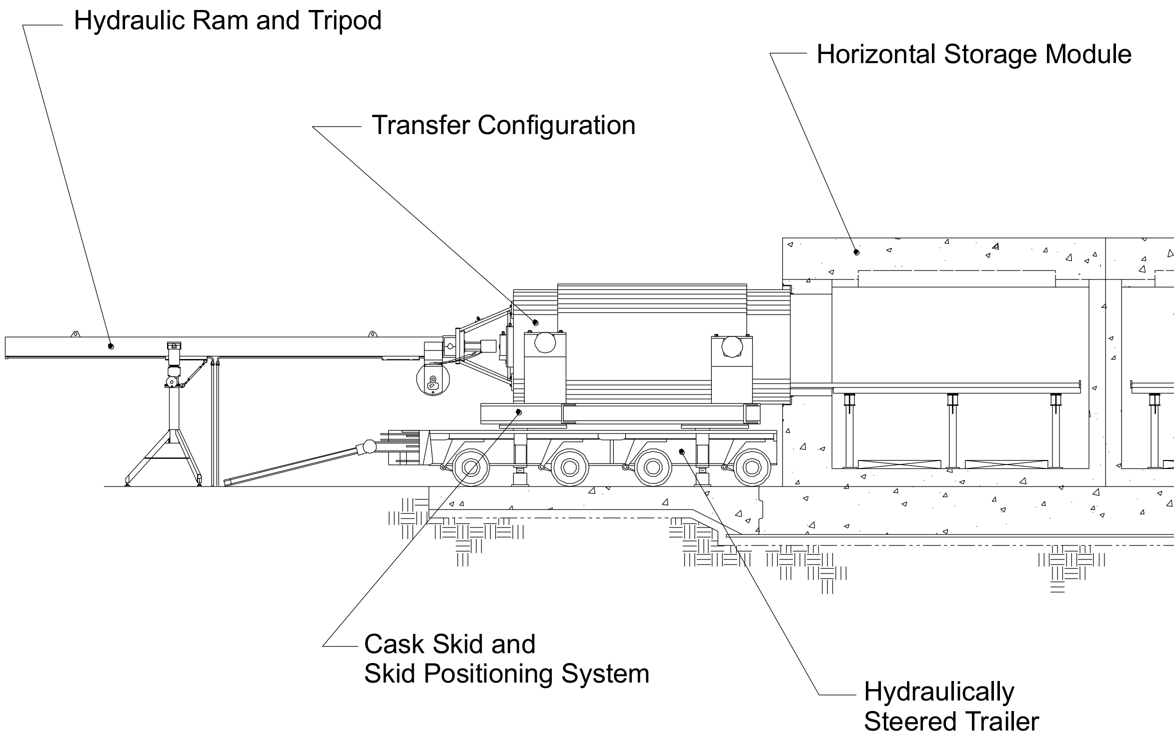


Figure 1-7

General Arrangement of the Transfer System



2. SITE AND ENVIRONMENT

2.1 Geography and Demography

2.1.1 Site Location

The Rancho Seco site is located in the southeast part of Sacramento County, California. It occupies all or parts of Sections 27, 28, 29, 32, 33, and 34 of Township 6 North, Range 8 East. The site is approximately 26 miles north-northeast of Stockton and 25 miles southeast of Sacramento, as shown in Figure 2-1. The Rancho Seco Nuclear Generating Station and Rancho Seco ISFSI are shown in Figure 2-2.

More generally, the site is located between the Sierra Nevadas to the east and the Coast Range along the Pacific Ocean to the west in an area of flat to lightly rolling terrain at an elevation of approximately 200 above feet mean sea level. To the east of the site the land becomes more rolling, rising to an elevation of 600 feet at a distance of about seven miles, and increasing in elevation thereafter approaching the Sierra Nevada foothills.

The approximate coordinates of the site are 38°-20'-40" North Latitude and 121°-07'-10" West longitude, or 4245500 Mn and 664400 Me Universal Transverse Mercator coordinates.

As shown in Figure 2-3, the Rancho Seco ISFSI is located west of the site's Industrial Area, approximately 600 feet west of the Interim On-site Storage Building. The Rancho Seco ISFSI is approximately 225 feet X 170 feet in size.

2.1.2 Site Description

The entire Rancho Seco site is approximately 2480 acres with all acreage being owned by SMUD The nearest population center of 25,000 or more is Lodi, about 17 miles southwest of the site. The area around the site is almost exclusively agricultural, or is used as grazing land.

The climatology of the Rancho Seco site is typical of the Great Central Valley of California. Cloudless skies prevail during summer and much of the spring and fall seasons due to the pacific anticyclone off the California coast which prevents Pacific storms from entering inland. The rainy season usually extends from October through May. Atmospheric dispersion factors for the site are considered favorable.

Groundwater in the site area occurs under free or semi-confined conditions. It is stored chiefly in the alluvium, the older alluvial type deposits, and the Mehrten Formation. Groundwater movement in the area is to the southwest with a slope of about ten feet/mile.

There is no indication of faulting beneath the site. The nearest fault system, the Foothill Fault System, is about ten miles east of the site and has been inactive since the Jurassic

Period, some 135 million years ago. Ground accelerations of no greater than 0.05g are anticipated at the site during the life of the plant.

The soils at the Rancho Seco site are sufficiently strong to safely support the Rancho Seco ISFSI structure and appurtenant facilities. These soils can be categorized as hard to very hard silts and silty clays with dense to very dense sands and gravels.

2.1.2.1 Other Activities Within the Site Boundary

The Rancho Seco ISFSI lies wholly within the 2,480 acre Rancho Seco Nuclear Generating Station site. This site is owned and controlled by SMUD, who has full authority to determine all activities within the site including the exclusion and removal of individuals and property. The Rancho Seco ISFSI Protected Area is approximately 225 feet X 170 feet in size. The Protected Area is located within licensed boundary denoted by the 100 meter fence surrounding the Protected Area. Also within the licensed boundary of the ISFSI lies the Fuel Transfer Equipment Storage Building (FTESB), a 40 foot X 100 foot enclosure to store contaminated fuel handling and transportation support equipment while the spent nuclear fuel remains in storage.

SMUD has completed construction of a 500-MW natural gas fired power plant located approximately ½ mile south of the Industrial Area boundary.

Access for transmission lines and water lines is from the west and south sides of the property.

2.1.2.2 Boundaries for Establishing Effluent Release Limits

There are no radioactive effluent releases associated with the Rancho Seco ISFSI. The boundaries for effluent releases from the Rancho Seco site are described in the Offsite Dose Calculation Manual (ODCM) [2.2.1].

2.1.3 Population Distribution and Trends

The land surrounding the site is presently undeveloped and is used primarily for grazing beef cattle and other agricultural activities. The most recent population distribution estimates are contained in the "Evacuation Time Estimate for the Rancho Seco Plume Exposure Pathway Emergency Planning Zone" [2.2.2].

There are five counties (Amador, San Joaquin, Sacramento, El Dorado, and Calaveras) within a 15-mile radius of Rancho Seco. Only very small portions of El Dorado and Calaveras counties are within the 15-mile radius of Rancho Seco. There is no significant projected growth within these portions of these two counties. The projected development within Amador, Sacramento, and San Joaquin counties is discussed in Section 4.2 of the Rancho

Seco ISFSI Environmental Report, Revision 1 [2.2.3]. A five-mile radius area surrounding the Rancho Seco facility is defined as the low population zone. This area is primarily farm land and vineyards, with few tourist attractions and little seasonal variation in the population.

The Rancho Seco Reservoir and Recreation Area (Rancho Seco Park) attracts a number of day visitors to the area. The average annual number of visitor days at the park for the last four years is 114,860 visitor days.

Additionally, a wildlife sanctuary has been built at Rancho Seco Park. It is estimated that an additional 625 cars could visit the park during special functions at this facility.

A survey of the area beyond the 5-mile radius shows that the nearest population concentration is approximately 6.5 miles from the plant site. The nearest population center of 25,000 or more is Lodi, 17 miles south-southwest of the site. Other population centers of greater than 25,000 people include Sacramento at 25 miles, Stockton at 26 miles, and Modesto at 50 miles.

There are 16 special facilities in Amador and Sacramento Counties within a 10½ mile radius of Rancho Seco. They consist of five public schools (one high school and four elementary schools), one private elementary school, one treatment center for TB and alcoholic patients, four residential care homes, an adult training center for developmentally disabled, a California Department of Forestry Fire Academy, the Preston School of Industry, a nudist ranch, and Mule Creek State Prison. A summary of these facilities is shown in Table 2.2-3 of Rancho Seco Defueled Safety Analysis Report (DSAR), Amendment 4 [2.2.4].

2.1.4 Uses of Nearby Land and Waters

2.1.4.1 Land Use

The site area is almost exclusively agricultural. DSAR, Amendment 4 Figure 2.2-4 provides a description of agriculture and residential activities within a 5-mile radius of the site. There are no commercial dairy farms within this 5-mile radius.

There are at present three large-scale commercial dairies in the vicinity, each with over 200 cows. The closest dairy is approximately 8 miles northwest of the site. A ranch 1 mile east of the site has dairy cows for domestic use only.

Proposed land use for the southeast section of Sacramento County as adopted by the Sacramento Planning Department is predominantly (70 percent) agricultural and is expected to remain agricultural. Approximately 2000 acres of vineyards are being developed on land in proximity to the Rancho Seco site.

2.1.4.2 Access and Egress

As shown in Figure 2-2, State Route 104 runs just north of the site in a general east-west direction and connects with State Route 99 to the west and State Route 88 to the east.

The Twin Cities Access Road, identified in Figure 2-2, is the main access road to the plant and to nearby recreational facilities. The access road to the plant is not a through road and is designed to handle heavy construction vehicles.

Rail access to the site is available via a rail spur from the existing Southern Pacific Railroad line that runs roughly parallel to State Route 104 adjacent to the site. The routing of the rail spur is shown in Figure 2-2.

2.1.4.3 Water Supply

Potable water for the Rancho Seco site is obtained from the site well. Water for RSNGS is from the Folsom South Canal. The Bureau of Reclamation constructed the canal as part of the Central Valley Project. A pipeline and pumping station are located between the plant and the Folsom South Canal.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Industrial

Some mining facilities within 10-miles east of the Rancho Seco site use explosives. These facilities receive their explosives via California State Highway 16 from the east, not via route 104 which runs just north of the site.

Regulatory Guide 1.91 "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants" [2.2.5] describes methods acceptable to the NRC staff for determining whether the risk of damage due to an explosion on a nearby transportation route is sufficiently high to warrant a detailed investigation. The guide establishes a method for determining safe distances at which no significant damage would be expected. The NRC has conservatively established 1 psi overpressure from the explosion as an acceptable level. The guide conservatively defines the relationship

$$R \geq K w^{1/3}$$

where R is the distance in feet from an exploding charge of w pounds of TNT and K is a proportional constant (K= 45). Given that the Rancho Seco ISFSI is approximately 2000 feet from route 104, a truck could carry up to 87,700 pounds of TNT equivalent, explode at the closest distance to the Rancho Seco ISFSI, and have an overpressure of less than 1 psi. Since the maximum probable hazardous solid cargo for a single highway truck is 50,000 pounds, an overpressure greater than 1 psi at the Rancho Seco ISFSI due to the explosion of a truck carrying explosives along route 104 is not a credible threat to the Rancho Seco ISFSI.

2.2.2 Transportation

As shown in Figure 2-2, State Route 104 runs just north of the site in a general east-west direction and connects with State Route 99 to the west and State Route 88 to the east. There are no public highways that traverse the Rancho Seco ISFSI site. Route 104 is not an approved, designated route for transporting explosives. This road is used primarily by local traffic.

There is a Union Pacific railroad line north of the site that, at one point, comes within 1/2 mile of the Rancho Seco ISFSI. The track runs roughly parallel to State Route 104, and was laid several decades ago to supply mining communities in the foothills to the east of the site. The track is now used to haul commodities. The track has not been used to haul any explosives within the last five years, and there are no plans to use the track for this purpose. There are no customers in the foothills that would have a need to use the railroad line for the supply of explosives. The needs of the mining facilities are supplied by trucks travelling on Highway 16, as noted above.

Rail access to the site is available via a rail spur from the existing Union Pacific Railroad line described above.

2.2.3 Military

The nearest major airfield was Mather Air Force Base, 18 miles northwest of the Rancho Seco site. Mather Air Force Base was closed on October 1, 1993; however, it is still used as a commercial air facility. The nearest defensive missile site is more than 45 miles from the site; however, this site is de-activated and no longer operational.

2.3 Meteorology

2.3.1 Regional Climatology

2.3.1.1 General Climate

The climate of the Rancho Seco site is generally that of the Great Central Valley of California. Summers are hot and cloudless and the winters are mild. The rainy season occurs between October and May with more than two-thirds of the annual rainfall occurring in December through March. Heavy fog occurs in mid-winter, primarily in December and January, and may last for several days.

Tornadoes and thunderstorms are infrequent. Tornadoes occurred only 22 times in California between 1953 and 1962. Thunderstorms occurred an average of three times per year in Stockton and five times per year in Sacramento. This is consistent with occurrences at the Rancho Seco site.

The most important controlling geographical influence on the climate results from the mountains which surround the valley to the west, north, and east. During the winter, storms which pass through the area are moderated by the mountains which collect much of the precipitation. The rains that occur in the valley are usually accompanied by south to southeast winds. The cold north and northwest winds pass over the mountains to the north where the air is warmed dynamically by descent into the valley resulting in comparatively warm, dry winds. A similar condition occurs infrequently in the summer when a steep northerly pressure gradient develops, producing a pronounced heat wave.

The Central Valley warms greatly during the day resulting in a marked thermal contrast between the valley and the air over the Pacific. The Coast Range separates the marine air from the valley air except for a gap through the range formed by the Sacramento and San Joaquin Rivers. The heavy marine air flows through this gap and splits into a northerly flow into the San Joaquin Valley and a southerly flow into the Sacramento Valley.

The divergence zone between the two flows usually lies between Stockton and Sacramento near the Rancho Seco site. The divergence zone is generally north of Rancho Seco during the day resulting in north to northwest winds. As the air in the valley cools, the flow decreases and calms may set in. If the drainage from the Sierra Nevada is sufficient, the winds may shift to southeasterly and increase in speed at Rancho Seco. Typical wind trajectories at the Rancho Seco site are shown in Figure 2-5 and Figure 2-6. The effect of this divergence zone upon the climate of Rancho Seco is discussed in more detail in the Rancho Seco Nuclear Generating Station Updated Safety Analysis Report [2.2.6] Appendix 2B.

During the hottest mid-summer months, light westerly winds may persist all night. During the winter, the synoptic gradients prevail much of the time and the wind trajectories over the Sacramento-Stockton-Rancho Seco area are reasonably uniform.

2.3.1.2 Severe Weather

2.3.1.2.1 Extreme Winds

Wind data from Sacramento Executive Airport for the period 1951 to 1971 were used to construct an extreme wind probability distribution appropriate to the Rancho Seco site. The following presents the highest expected wind speed that will be exceeded for the indicated recurrence intervals.

EXPECTED EXTREME WIND SPEEDS

Return Period (yrs)	Wind Speed (mph)
50	90
100	101
1,000	149
10,000	169

The fastest one-minute average winds for Sacramento from July 1877 through December 1989 are presented in Table 2-2.

2.3.1.2.2 Tornadoes

Tornadoes have been reported in California but with a frequency of only two per year (National Climatic Summary, 1969). They are generally not severe, and in many cases amount to little more than a whirlwind that may cause damage to trees and light buildings. As discussed in USAR Appendix 2B, an examination of newspaper accounts of nine tornadoes in California indicated that only one could have been accompanied by winds exceeding 100 mph.

The location of a possible tornado strike can be approximated by a geometrical point. The probability of a tornado occurring at a specific point can be found by the principle of geometric probability. If two tornadoes per year for California are used, the return period for Rancho Seco is 27,855 years. Because the intensity of California tornadoes is much less than the "classical mid-western types," winds in only one of five of these tornadoes would be expected to exceed 100 mph.

2.3.1.2.3 Tropical Storms and Hurricanes

The possibility of severe storms in the area can be limited to thunderstorms and tornadoes. A discussion of tropical storms and hurricanes is not applicable for Rancho Seco.

2.3.1.2.4 Precipitation Extremes

The precipitation climatology of the Great Central Valley is characterized by a dry season from June through September and a rainy season from October to May. No precipitation records were taken at Rancho Seco, but because precipitation is associated with large-scale synoptic systems, the data shown in Table 2-3 should be representative of the Rancho Seco site. The annual rainfall occurs almost totally in the winter months. A representative frequency of occurrence of a given precipitation intensity for Sacramento is presented in Table 2-4.

2.3.1.2.5 Snow and Ice Storms

The possibility of severe storms in the area can be limited to thunderstorms and tornadoes. Snow in the Sacramento area is extremely rare. Most of the snow that has been observed in the Sacramento area occurs in January. Given the lack of significant snow at the Rancho Seco ISFSI, a discussion of snow and ice storms is not applicable.

2.3.1.2.6 Thunderstorms

Thunderstorms, and associated lightning strikes, occur infrequently in the area. The mean number of days during which thunderstorms over a 21-year interval for Sacramento and a 22-year interval for Stockton are listed in Table 2-5.

2.3.1.2.7 Restrictive Dilution Conditions

Inversions occur in the Great Central Valley as a result of cold air advection near the ground or radiational cooling of the earth causing a cooling of the air near the ground. Radiational cooling occurs at night when there are no low clouds. Both types occur at Rancho Seco with the advection type usually associated with the westerly wind bringing in cool air from the Pacific Ocean.

Temperature inversions at the ground can be expected to occur every night during the summer upwards to several hundred feet. These temperature inversion are the result of the flow of cool maritime air into the area during late afternoon and evening hours. During the winter, shallow (a few hundred feet) but intense surface inversions can be expected occasionally during nighttime hours under light wind conditions.

2.3.2 Local Meteorology

2.3.2.1 Data Sources

The meteorological data acquisition system used to collect data from September 1969, through March 1973, for the Rancho Seco site consisted of a meteorological tower installed at the site, instrumentation, a digital recorder, and software. The main purpose of the system was to measure and compile the meteorological data necessary to define the atmospheric diffusion at the site. The system was designed to continue operation indefinitely so that a broad statistical base for meteorological conditions at the site could be assembled.

SMUD erected a 200-foot meteorological tower on the site, and in June 1969, recorded the first analog measurements. The meteorological tower replaced the temporary mechanical weather station that had been operating continuously since April 1967, and had supplied onsite data used in the Rancho Seco Nuclear Generating Station Preliminary Safety Analysis Report (PSAR) [2.2.7].

On September 8, 1969, data obtained from the tower were for the first time recorded in digital form. The first year of digital system data is presented in USAR Appendix 2B, Attachment 1. A compilation of 2 years of site data is presented in USAR Appendix 2B, Attachment 3. A detailed description of the meteorological tower including instrumentation location and performance specifications, data analysis, measurements taken, and revisions to the data collection system can be found in USAR Appendix 2B. This instrumentation has been taken out of service, and in 1998, the meteorological site was decommissioned.

2.3.3 On-Site Meteorological Measurements Program

The Rancho Seco Permanently Defueled Technical Specifications no longer require any meteorological monitoring instrumentation. The meteorological monitoring instrumentation was intended to provide data that could be used to estimate potential radiological doses to the public resulting from the routine or accidental release of radioactive material to the atmosphere.

In lieu of using actual meteorological data, SMUD will use conservative default relative concentration (χ/Q) values. In the event that real-time meteorological data is needed, the Sacramento National Weather Service can provide the required data.

2.3.4 Diffusion Estimates

During an accidental release of gaseous radioactive material, the magnitude of the offsite doses is dictated primarily by the source term and the atmospheric dispersion coefficient, χ/Q. The total number of Curies of noble gases and iodines released is directly proportional to the offsite dose. With the exception of krypton-85, these isotopes have half-lives of only a few days and, therefore, have essentially decayed away since the reactor was shutdown on June 7, 1989. Krypton-85 is now the predominant isotope in the gaseous source term.

Because of the extremely small source term that exists in the defueled condition, the NRC agreed that it is expedient and conservative to use a default χ/Q value in calculations involving the accidental release of radioactive gaseous effluent, instead of relying on meteorological monitoring instrumentation to provide the data needed to calculate actual χ/Q values. During an accidental release of radioactive material, the default χ/Q value is 4.24E-2 sec/m^3 at a distance of 383 feet from the nearest module.

The original atmospheric dispersion factors calculated for the Rancho Seco site were based on site data collected from the 200-foot meteorological tower during the two-year period November 1969 through October 1971. A detailed description of the calculation methodology is included in USAR Appendix 2B.

2.4 Hydrology

2.4.1 Characteristics of Streams and Lakes in Vicinity

USAR Table 2.4-l provides a summary of reservoirs and lakes in the vicinity of RSNGS. Each reservoir and lake is coded by number in USAR Table 2.4-l for easy location on the location map, USAR Figure 2.4-l.

2.4.2 Topography

The site is gently rolling and is not intersected by any streams, but is bounded by well-defined drainage courses that intercept surface runoff from the higher site topography. Plant grade at approximately 165 feet elevation above sea level permits excellent drainage at all times without danger of flooding. Plant areas are graded to provide natural drainage to lower ground. The rolling terrain of the site affords excellent drainage along natural gullies at gradients varying from 2 to 6 percent. Elevations vary from 130 feet to 280 feet above sea level.

2.4.3 Terminal Disposal of Stream Runoff

The site is bounded on the north by Hadselville Creek, which intercepts all drainage from the site and empties into Laguna Creek to the west. Flow is continued westerly by Laguna Creek South, a tributary of the Consumnes River, and into the Mokelumne River. The Mokelumne is a tributary of the southerly flowing Sacramento River and enters the Sacramento River approximately 20 miles south of the city of Sacramento.

Storm water runoff at the Rancho Seco site is controlled primarily by surface ditches. Generally, overland flows will be intercepted by the ditches and diverted around the plant to natural stream channels. When this is not possible, runoff will be diverted down cut slopes in culvert pipes and discharged to the plant drainage ditch system. The drainage system was designed to accommodate the 25-year recurrence storm with a minimum of six inches freeboard and the 100-year recurrence storm with zero freeboard.

2.4.4 Historical Flooding

Within recent historical times, no flooding or inundation from storms or runoff has occurred within the site boundaries. It is unlikely that the site can be inundated or flooded, even with abnormal rainfall intensities.

To provide criteria for the design of an adequate spillway to safeguard the Rancho Seco lake dam embankment from any danger of overtopping, SMUD conducted a hydrologic study of storms which could produce critical floods. There are two types of storms that could produce the critical outflow flood for spillway design. The frontal winter storm would produce the greatest amount of total rainfall, but would be relatively low intensity. A summer thunderstorm, on the other hand, would be of short duration but with very intense rainfall.

The rainfall intensities and their associated time distributions used in the critical flood study are shown in USAR Table 2.4-2.

2.4.4.1 Floods From Frontal Storms

As discussed in USAR Section 2.4.4, the probable maximum frontal storm of 72-hour duration was calculated in accordance with U.S. Weather Bureau Hydrometeorological Report No. 36 "Interim Report, Probable Maximum Precipitation in California," October 1961. This storm was distributed in accordance with recommended procedures for maximizing the hydrograph peak.

The peak of this flood was 2,600 cfs (USAR Figure 2.4-3) and its volume was approximately 650 acre-feet.

2.4.4.2 Floods From Thunderstorms

The thunderstorm probable maximum precipitation (PMP) was calculated in accordance with procedures recommended by the Sacramento District Office of the U.S. Army Corps of Engineers. These procedures are outlined in an unpublished document of the Corps "Basis for Thunderstorm PMP Estimates for Southwest States," December 1968.

A 15-minute unit hydrograph was computed from the same S-curve procedures used for the frontal storm analysis. Loss rates and base flow were considered the same as for the frontal storm. The peak from this storm was 4,270 cfs (USAR Figure 2.4-4) and its volume was approximately 410 acre-feet.

2.4.4.3 Spillway Capacity

The relatively large area of Rancho Seco Lake with respect to the effective drainage area and the maximum flood volumes calculated makes possible the storage of a large percentage of the inflow flood in the reservoir. For example, the entire winter frontal storm can be stored in approximately 4 feet of the reservoir above elevation 240. For this reason, the criteria for spillway design was that the spillway allow for the evacuation of reservoir storage after one storm to permit storage of a subsequent flood without excessive encroachment on freeboard.

Meteorological studies have indicated that a second major frontal storm would take three or four days to build up to its maximum intensity (USAR Figure 2.4-5). As thunderstorm conditions may develop quickly, it was considered that a subsequent thunderstorm may develop in 24 hours. It was considered possible that the probable maximum storm could be preceded or followed by a storm of half its magnitude of precipitation in the periods of time discussed (USAR Figure 2.4-6).

A reservoir routing study with initial reservoir level at the spillway crest showed that with an 8-foot wide crest at elevation 240.5, no combination of the storms described above will surcharge the spillway more than 3 1/2 feet, leaving a minimum freeboard on the dam (crest elevation 248) of 4 feet.

A storm of half the magnitude of the probable maximum thunderstorm occurring by itself would surcharge the spillway about one foot, leaving a freeboard on the dam of 6 1/2 feet.

2.4.5 Prediction of Land Urbanization

SMUD has constructed a solar photovoltaic generating plant adjacent to the site. Other land adjoining the site should remain primarily for agricultural and grazing use. The rainfall runoff factors should remain constant, and not cause any difference in hydrological properties.

2.4.6 Groundwater

Initial pumping tests conducted in exploratory holes indicated the presence of groundwater underlying the site approximately 150 feet below the original ground surface. The water table has receded over recent years, and is expected to recede further due to the grape vineyards now being developed adjacent to the site. The water is of good quality and is readily extracted by wells.

2.4.6.1 Occurrence and Movement

Groundwater in this area occurs under free or semi-confined conditions as a part of the Sacramento Valley Groundwater Basin. The storage capacity of the basin is very large, but in the vicinity of the site, water levels are steadily dropping, as shown by the hydrograph of USAR Figure 2.4-13. The water is stored chiefly in the Mehrten Formation. The sand and gravel zones of that formation yield water readily to wells.

Galt and Lodi are the closest communities with public groundwater supplies to the south and west. Their spatial relationship to the project site is shown on USAR Figure 2.4-14. They are supplied by the City of Galt Water System, the Lodi Municipal Water Works, and the North San Joaquin Water Conservation District (Lodi area). The Galt Irrigation District and the Clay Irrigation District buy Rancho Seco discharge water for irrigation.

The wells supplying Galt and Lodi penetrate a number of aquifers. The Lodi wells draw water from recent alluvium, the Victor Formation, the Laguna Formation, and probably the Mehrten Formation. The Galt wells tap the Laguna Formation and probably the Mehrten Formation. The approximate time required for groundwater moving through the Mehrten Formation aquifer from the Rancho Seco site to the Galt area is discussed in USAR Section 2.4.6.1.

As discussed in USAR Section 2.4.6.1, the estimate for the movement of groundwater from the Rancho Seco site to the Galt area is thought to be conservative. Retarding factors such as dispersion, adsorption by ion-exchange, and lower velocities of ionic species with respect to water are not considered in the computations. These retarding factors plus the low vertical permeabilities of the finer-grained materials above the Mehrten Formation aquifer at the Rancho Seco site would effectively prevent any significant concentration of contaminants resulting from an inadvertent release of radioactive liquids from ever reaching the Galt area through the aquifers.

The 71 exploratory borings made during investigations of the Rancho Seco site reveal that, in the upper 200 feet, the rocks are mainly low permeability siltstone, claystone, and silty sandstone containing lenses and layers of sandstone. From about 200 to 350 feet there are thick interbedded siltstone, claystone, and sandstone. The permeable sandstones in this interval constitute the major local aquifers. Below this are claystone and siltstone.

Infiltration tests made in the upper few feet of alluvium sand and silty sand indicate permeabilities of 2,000 feet/year to l0,000 feet/year. Laboratory permeability tests made on samples of sandy siltstone from bore hole DH-23 taken at l0-foot and 30-foot depths indicate permeabilities of 6 feet/year and 0.6 foot/year, respectively. From these tests and analyses of the lithologies, estimates of horizontal and vertical permeabilities have been assigned to the foundation rocks. The rocks have been grouped into four types as listed below:

1. Sandstone is moderately permeable with assigned estimates of horizontal permeability of l0,000 feet/year and vertical permeability of 2,000 feet/year.

2. Silty-sandstone is less permeable with estimates of horizontal permeability of 2,000 feet/year and vertical permeability of 200 feet/year.

3. The low permeability of siltstone is estimated at horizontal permeability of l0 feet/year and vertical permeability of 0.5 feet/year.

4. Claystone is essentially impermeable with horizontal permeability estimated at less than 0.5 feet/year, and vertical permeability of 0.005 feet/year.

In addition, pumping tests have shown that the permeable aquifer zones below 200 feet (Mehrten Formation) are estimated at a horizontal permeability of l0,000 feet/year and a vertical permeability of 2,000 feet/year.

2.4.6.2 <u>Water Supply</u>

A water well was drilled in May 1969, (USAR Figure 2C-9) to provide a water supply for the construction of RSNGS. The well is 12 inches in diameter, 410 feet deep, with a screened interval from a depth of 156 feet to a depth of 400 feet. A deep-well submersible pump has been set at a depth of 250 feet. Since startup, plant domestic water has been obtained from the well.

2.4.6.3 Quality

The groundwater is of good quality, and is well within U.S. Public Health Department standard limits. It is a sodium bicarbonate-type water with low total dissolved solids, less than 200 ppm. It is a very soft water, less than 50 ppm total hardness ($CaCO_3$). The iron and manganese concentrations do not exceed the recommended 0.3 ppm.

2.4.7 Wells

A survey of well data available for the area within a two-mile radius of the Reactor Building was performed during the design of RSNGS. This survey identified approximately 40 wells within the two-mile radius. The locations of the wells are indicated in USAR Figure 2.4-15. USAR Table 2.4.3 summarizes the information which was available for the identified wells.

2.5 Geology and Seismology

2.5.1 Geology

The Rancho Seco site is about 25 miles southeast of Sacramento in the low hills at the edge of the Sierra Nevada Mountains. The site is founded on the Pliocene Laguna Formation and is underlain by an estimated l,500 to 2,000 feet of Tertiary or older sediments deposited on a basement complex of granitic to metamorphic rocks.

Explorations at the site included field mapping, l,552 feet of bucket auger holes logged in detail, a 602-foot core hole visually and geophysically logged, 2,016 feet of small-hole borings that were logged and from which soil samples were taken for laboratory testing, and approximately 11,500 feet of geophysical refraction profiles. The data obtained indicated the unfaulted nature of the sediments and their suitability as a foundation upon which RSNGS was constructed.

A detailed account of the conditions at the site can be found in USAR Appendix 2C (Geology and Seismology).

2.5.2 Seismology

There is no indication of faulting beneath the site. The nearest fault system, the Foothill Fault System, is about 10 miles to the east of the site. It has been inactive since the Jurassic Period, some 135 million years ago. The nearest active faulting along which historic large earthquake shocks have originated are the Hayward and San Andreas Faults, some 70 and 89 miles to the west, respectively, and the faults over 80 miles to the east beyond the Sierra Nevada Range.

There is no reason to anticipate fault propagation in the site area. Earthquake shaking will occur as the result of shocks along distant faults, but because of their distant origin and the nature of the foundation material beneath the site, ground accelerations greater than 0.05g should not occur during the life of the Rancho Seco ISFSI. Conservative values of 0.25g horizontal and 0.17g vertical were used for the Design Basis Earthquake (DBE) for the Rancho Seco ISFSI.

Further discussion of the site seismicity may be found in the Seismic Report in USAR Appendix 2D and supplements. Earthquake design criteria for the site can be found in USAR Appendix 5B.

2.6 Soils

2.6.1 Rancho Seco Site

The soil and foundation investigation program was performed (USAR Appendix 2E) to determine the suitability and the engineering properties of the soil and foundation at the Rancho Seco site. The investigation was carried out concurrently with the geologic and geophysical investigation. Soil borings, test trenches, and bucket auger holes were drilled and samples were obtained for laboratory testing.

Additional drilling and sampling was performed to determine the design requirements for major structures that were not formally established during the prior investigation. Static strength testing was performed on representative soil samples and dynamic triaxial tests were also performed on selected soil samples to evaluate the dynamic modulus and damping ratios of the foundation soils at various strain levels. Standard testing procedures and techniques were used throughout the program.

Results of the drilling, sampling, and laboratory testing provided the basic technical data from which the foundation and engineering properties of the soils were analyzed. It was concluded that the soils at the Rancho Seco site are sufficiently strong to safely support the nuclear containment structure and appurtenant facilities. These soils can be categorized as hard-to-very-hard silts, and silty clays with dense-to-very-dense sands and gravels.

Construction controls, including visual inspection and materials testing, were performed to assure that design soil conditions were obtained.

An allowable bearing value of 9,000 pounds per square foot was recommended for the Rancho Seco containment structure and those portions of the nuclear steam supply system critical to nuclear safety, based on maximum tolerable settlement criteria. Settlement monitoring of the Class I structures indicate that actual settlements will be less than those predicted.

2.6.2 Rancho Seco ISFSI Site

The ISFSI site is located on the west side of the existing station facilities. Before construction, the site was covered with grass and sloped downward from west to east with an average slope of 12:1 (horizontal:vertical). Construction required a cut of approximately 40 feet in the northwest corner and a minimum cut of approximately 2 feet in the southeastern corner. An earthen berm, with a maximum height of approximately 32 feet, has been constructed along portions of the southern edge of the ISFSI site.

The existing Rancho Seco soils investigations described in the USAR were supplemented by a study performed by Environmental Geotechnical Consultants, Inc. [2.2.8]. This study included boring two holes, 62 and 75 feet deep, at the east and west ends of the location of the prefabricated modules as shown in Figure 2-7. The conditions encountered in the two borings are summarized on the individual bore logs presented in Figure 2-8 and Figure 2-9. Based on a finished subgrade elevation of 173 feet, the bottom of the ISFSI mat was founded on a 3 to 7 foot thick compacted sand layer. This is underlain by a mixture of very dense clay and silt soils which will provide good support for the ISFSI foundations without the need for additional excavation and/or compaction.

As part of the Rancho Seco ISFSI site selection process, SMUD contracted Environmental Geotechnical Consultants, Inc.to analyze borings from the proposed Rancho Seco ISFSI sites. Based on the results of the boring analyses, SMUD performed appropriate remedial measures (e.g., recompaction and/or replacement of soil) to ensure adequate structural support for the Rancho Seco ISFSI [2.2.8].

2.7 References

2.1 Rancho Seco Nuclear Generating Station Offsite Dose Calculation Manual.

2.2 HMM Associates, Inc., Evacuation Time Estimate for the Rancho Seco Plume Exposure Pathway Emergency Planning Zone, December 1989.

2.3 Rancho Seco ISFSI Environmental Report, Revision 1, June 1993.

2.4 Rancho Seco Nuclear Generating Station Defueled Safety Analysis Report, Docket No. 50-312.

2.5 Regulatory Guide 1.91 "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," Revision 1, February 1978.

2.6 Rancho Seco Nuclear Generating Station Updated Safety Analysis Report, Docket No.50-312.

2.7 Rancho Seco Nuclear Generating Station Preliminary Safety Analysis Report, Docket No. 50-312

2.8 Environmental Geotechnical Consultants, Inc., "Geotechnical Study for Proposed Independent Spent Fuel Storage Installation Rancho Seco Nuclear Generating Station Sacramento County California," EE-519/E306-01, June 1, 1993.

Table 2-1

Permanent Population Distribution Within 13 Miles of RSNGS

Deleted

Table 2-2

Highest One-Minute Average Windspeeds

Month	Direction	Speed (mph)	Year
January	SE	60	1954
February	SE	58	1938
March	S	66	1952
April	SW	45	1955
May	SW	40	1912
June	SW	47	1950
July	SW	36	1956
August	SW	38	1954
September	SW	42	1965
October	SE	68	1950
November	SE	70	1953
December	SE	70	1952

Table 2-3

Precipitation Climatology

Averages (inches)

Month	Sacramento	Stockton
January	3.18	2.55
February	2.99	2.46
March	2.36	2.05
April	1.40	1.14
May	0.59	0.44
June	0.10	0.07
July	0.01	0.01
August	0.02	0.01
September	0.19	0.19
October	0.77	0.63
November	1.45	1.17
December	3.24	2.66
Total	16.29	13.37

Table 2-4

Precipitation Intensity

Year	Inches/Hour			
	0.01-0.09	0.10-0.24	0.25-0.49	0.50-0.99
1961	79.5%	17.7%	2.3%	0.5%
1962	81.8%	17.0%	0.8%	0.4%
1963	80.0%	17.8%	2.2%	0.0%
1964	86.2%	11.3%	2.2%	0.3%
1965	89.0%	10.0%	1.0%	0.0%
Average	83.5%	14.6%	1.7%	0.2%

Table 2-5

Mean Number of Days of Thunderstorms

Month	Sacramento	Stockton
January		
February		
March	1	
April	1	1
May	1	
June		
July		
August		
September	1	1
October		
November		
December		
Year	5	3

Figure 2-1

Regional Map of RSNGS

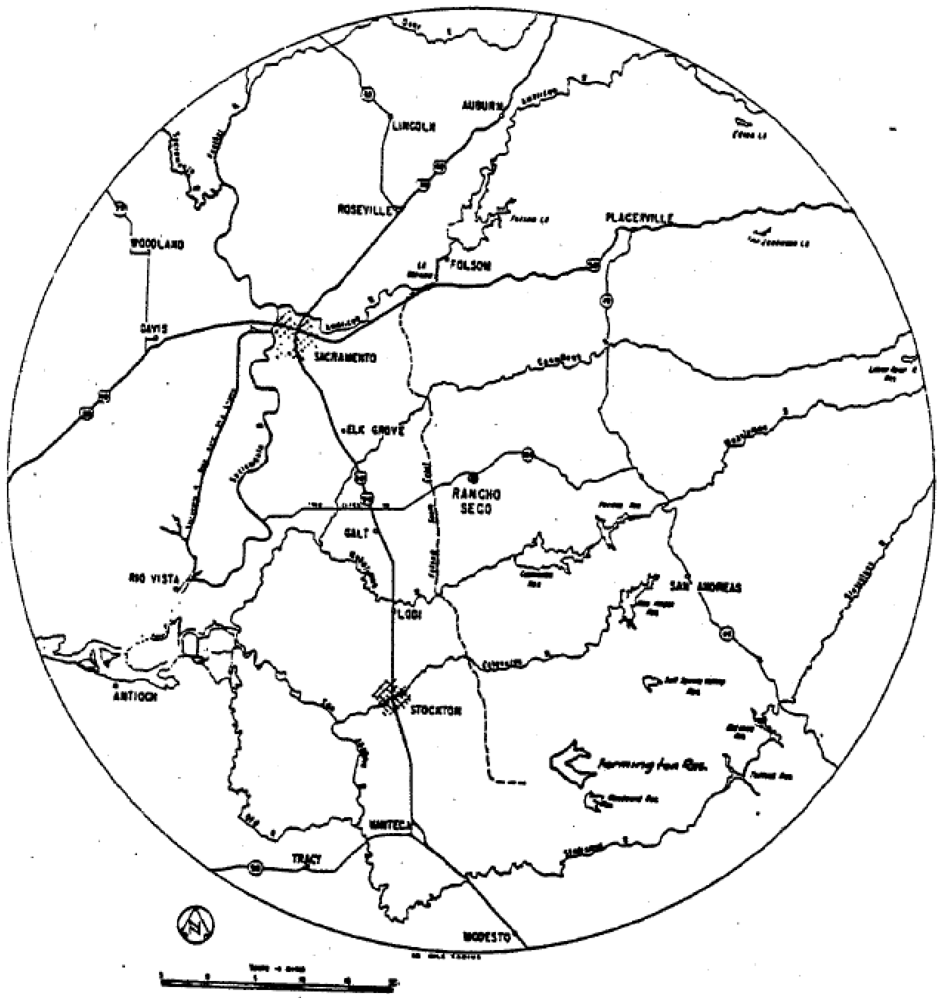


Figure 2-2

RSNGS Site

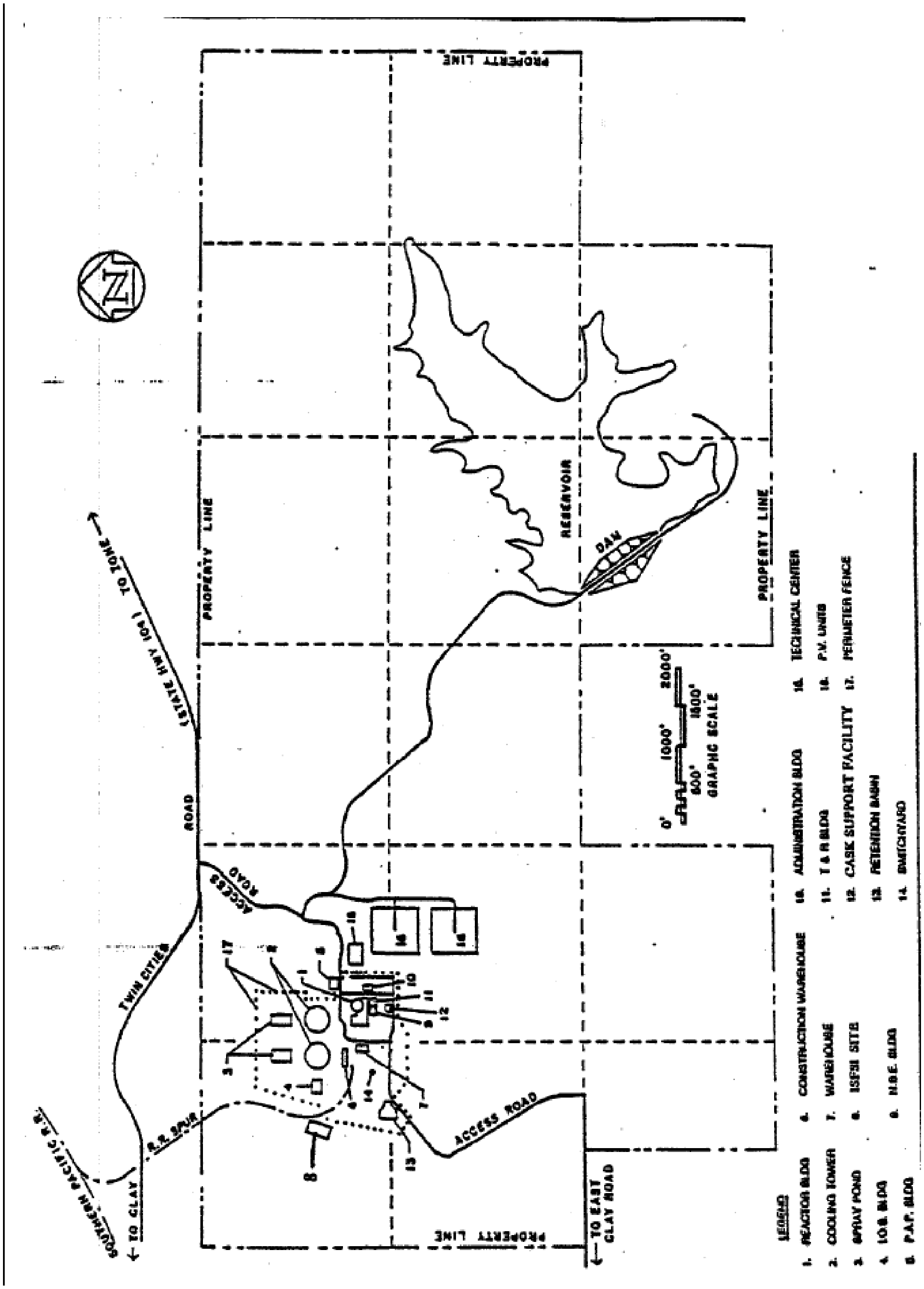


Figure 2-3

Rancho Seco ISFSI Site

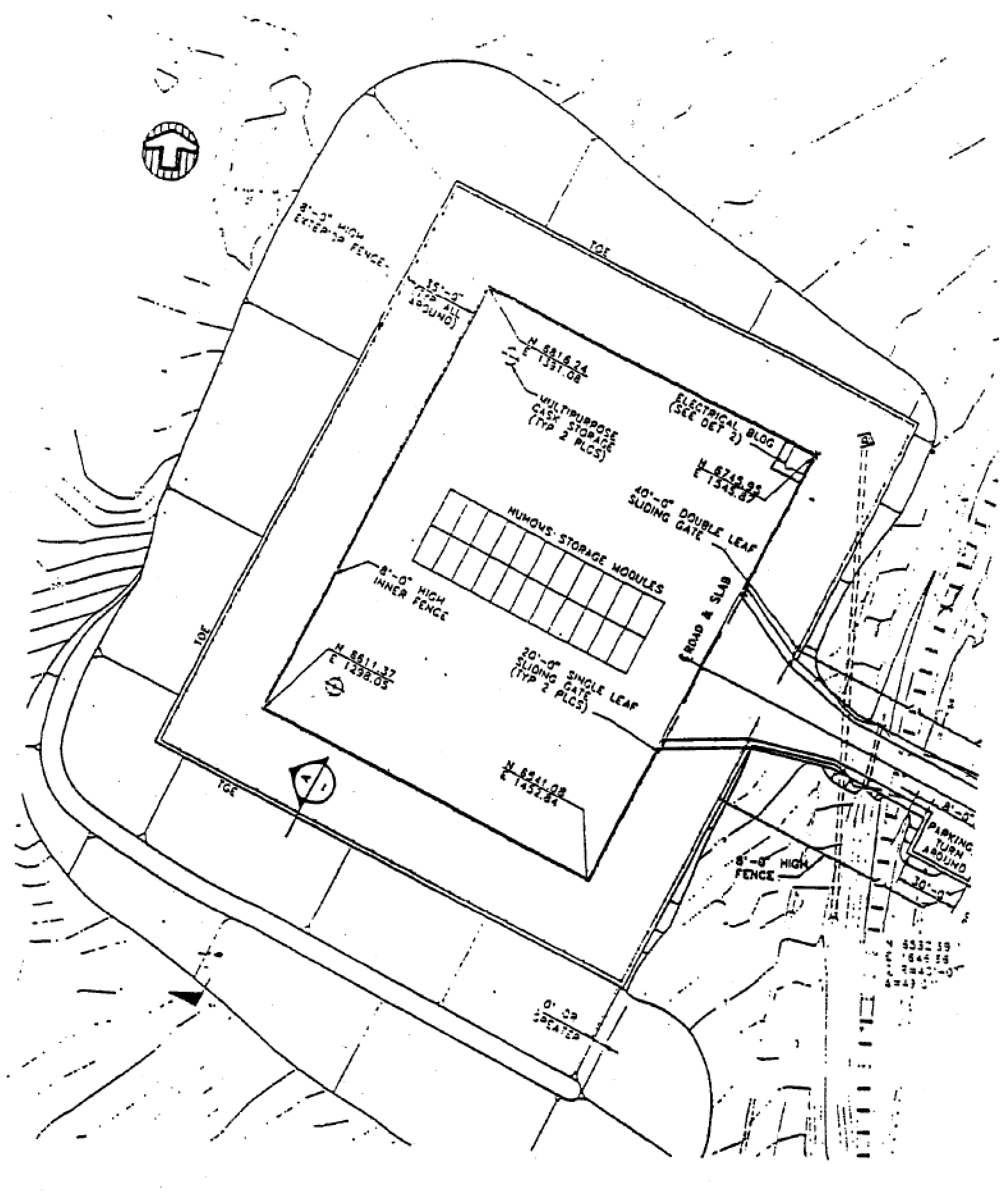


Figure 2-4

Permanent Population Surrounding RSNGS

Deleted

Figure 2-5

Wind Trajectories for RSNGS

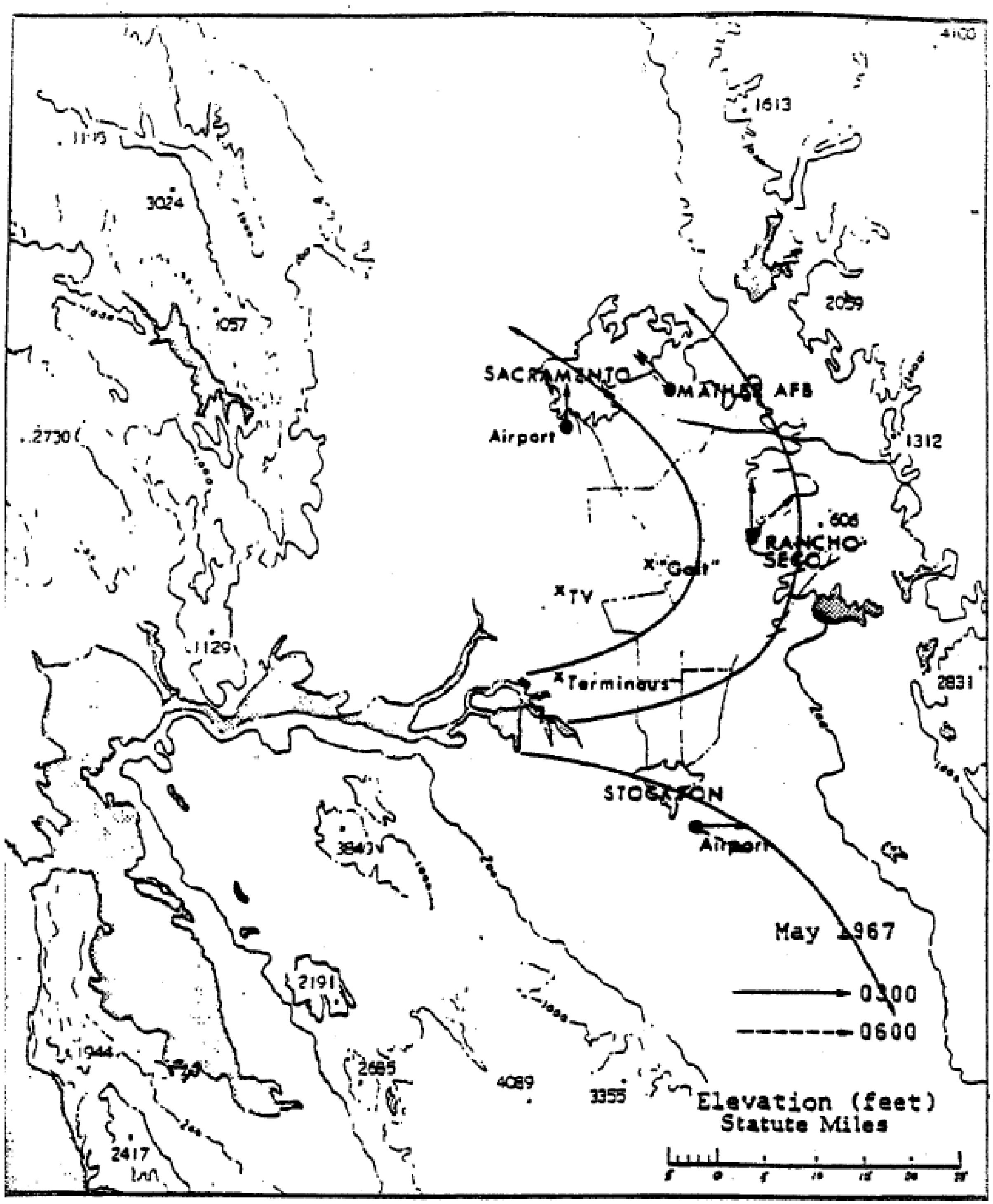


Figure 2-6

Wind Trajectories at RSNGS

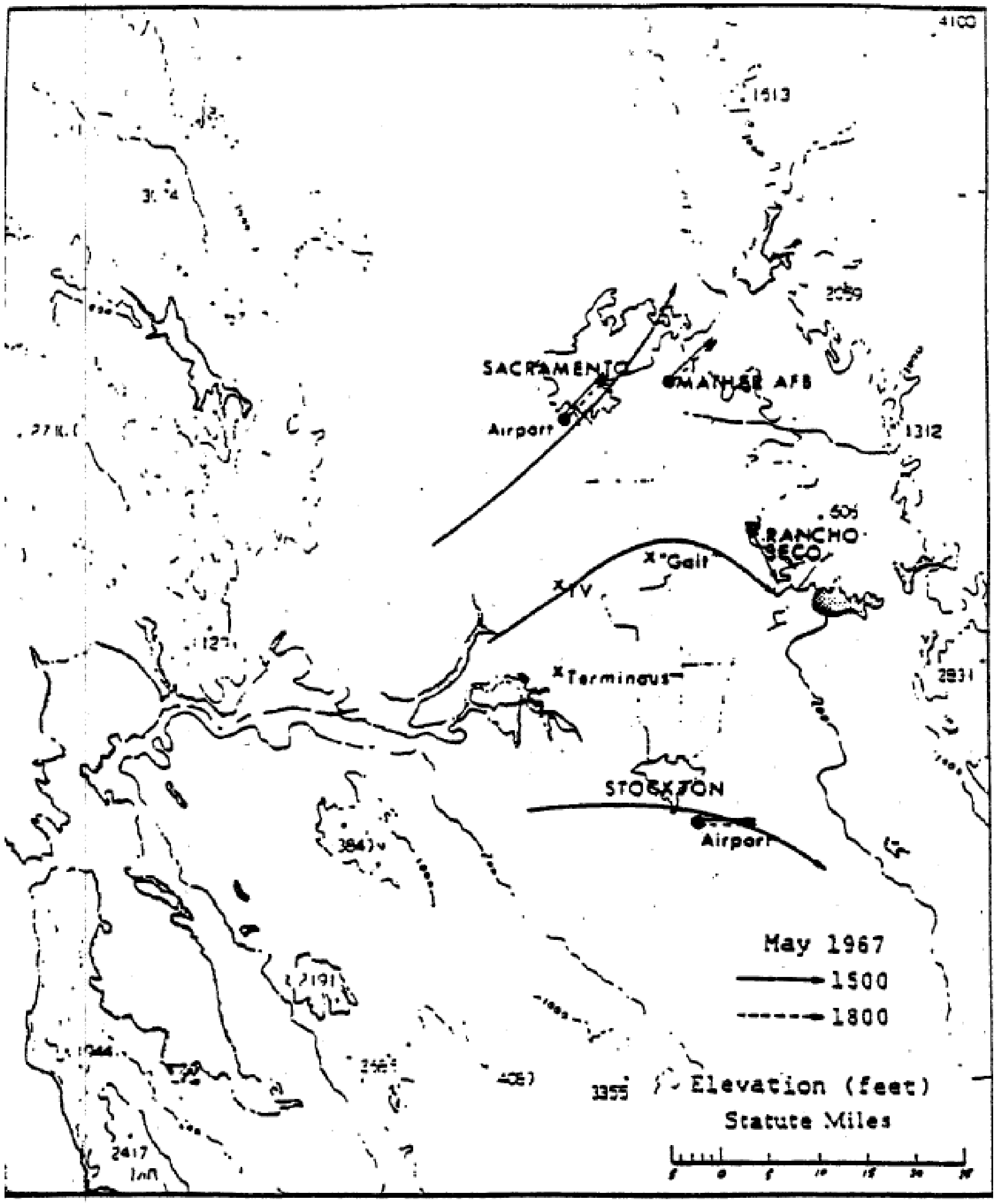
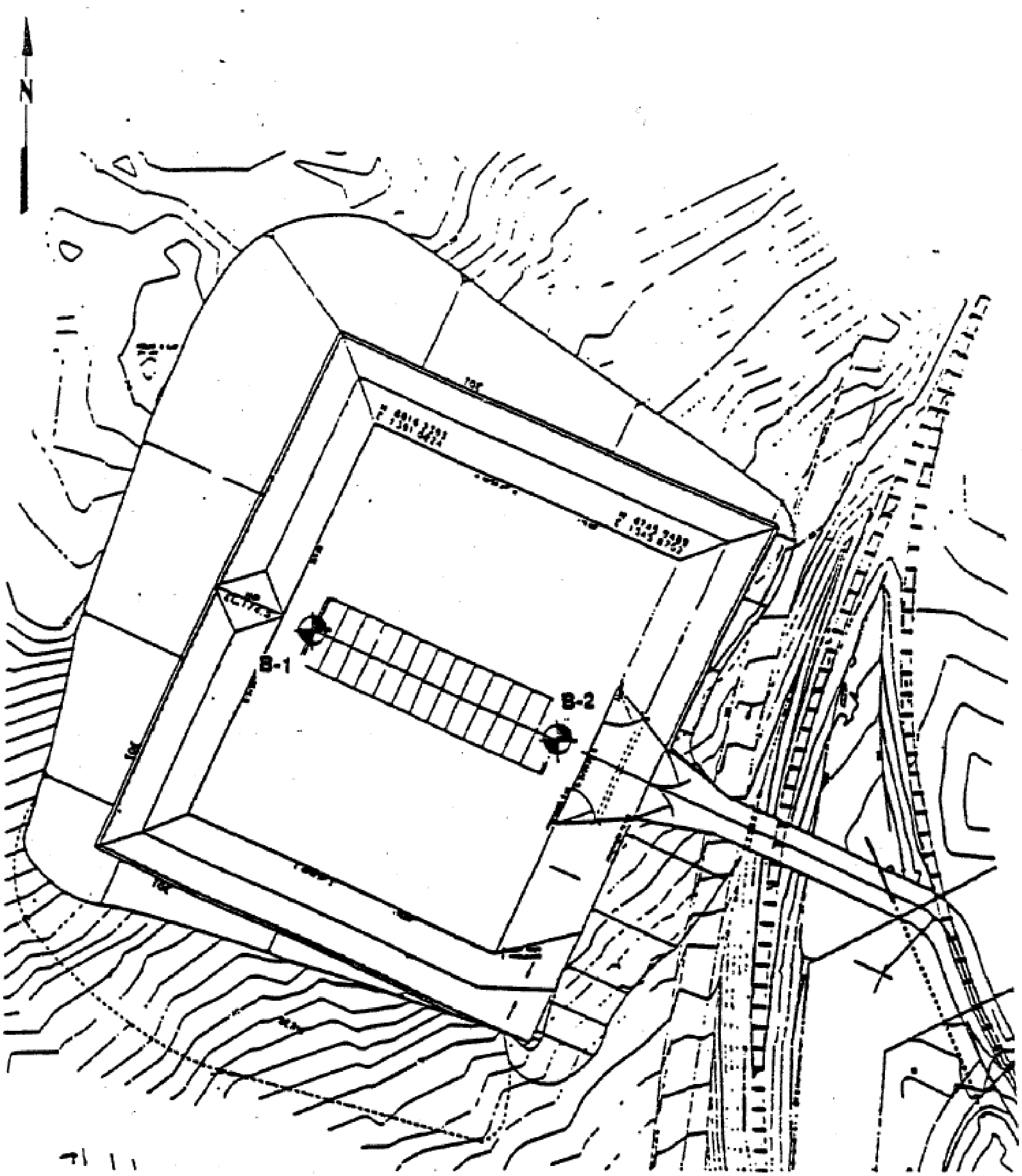


Figure 2-7

Boring Location Map



NOT TO SCALE

Figure 2-8

Subsurface Exploration Log B-1

EXPLORATORY BORING LOG

CLIENT: Sacramento Municipal Utility District LOCATION: Rancho Seco Nuclear Generating Station Sacramento County, California	JOB NO: E306-01 DATE: 4/27/93	BORING B-1
DRILL RIG: Mobile B-61 DRILLER: B&F WT. OF HAMMER/DROP: 140 pounds/30 inches	BORING ELEV.: 195 feet BORING DIAM.: 8-inch LOGGED BY: JMP	PAGE 1 OF 3

MATERIAL DESCRIPTION AND REMARKS	CONSISTENCY	USCS	DEPTH (ft.)	SMPL	N blows per ft.	DRY DENSITY (pcf)	WATER CONTENT (%)	PLASTICITY INDEX (%)	UNCONFINED COMPRESSIVE STRENGTH (psf)
6-inch root zone Sandy SILT, brown, low plasticity, moist	STIFF	ML	1						
			6	BS #1				10.6	
Hard Pan, Cemented Silty SAND, trace gravel, light brown, dry	HARD	SP	11	1	50/5"				9000+
Sandy CLAT, trace gravel, brown, low plasticity, dry	STIFF	CL	15	BS #2					
Sandy SILT, trace gravel, brown, low plasticity, dry	HARD	ML	19	2	50/5"	95	14.9		9000+
			21	3	106	92	14.0		9000+
SAND, fine to medium, brown, dry	VERY DENSE	SP	23	4	63	93	7.5		
			25	5	62				

ENVIRONMENTAL GEOTECHNICAL CONSULTANTS, INC.	SUBSURFACE EXPLORATION LOG B-1 Rancho Seco Nuclear Generating Station Sacramento County, California Sacramento Municipal Utility District	FIGURE NO. 3

Figure 2-8 (continued)

Subsurface Exploration Log B-1

EXPLORATORY BORING LOG

CLIENT: Sacramento Municipal Utility District LOCATION: Rancho Seco Nuclear Generating Station Sacramento County, California	JOB NO: E306-01 DATE: 4/27/93	BORING B-1
DRILL RIG: Mobile B-61 DRILLER: B&F WT. OF HAMMER/DROP: 140 pounds/30 inches	BORING ELEV.: 195 feet BORING DIAM.: 8-inch LOGGED BY: JMP	PAGE 2 OF 3

MATERIAL DESCRIPTION AND REMARKS	CONSISTENCY	USCS	DEPTH (ft.)	SMPL	N blows per ft.	DRY DENSITY (pcf)	WATER CONTENT (%)	PLASTICITY INDEX (%)	UNCONFINED COMPRESSIVE STRENGTH (psf)
Gravel at 26.0 feet			26	5	62				
			27						
			28						
SILT, orange brown, low plasticity, dry	HARD	ML	29						
				6	135				9000+
			30						
			31						
			32						
			33						
			34						
			35						
			36	7	35/1/2"				9000+
Gravel layer @ 37.0 feet			37						
			38						
			39						
			40						
			41	8	60/6"				9000+
			42						
			43						
			44						
Sandy Gravel, grey, dry	VERY DENSE	GW	45						
			46	9	150/6"				
			47						
			48						
SILT, orange brown, low plasticity, dry	HARD	ML	49						
			50	10	50/5"				9000+

ENVIRONMENTAL GEOTECHNICAL CONSULTANTS, INC.	SUBSURFACE EXPLORATION LOG B-1 Rancho Seco Nuclear Generating Station Sacramento County, California Sacramento Municipal Utility District	FIGURE NO. 3

Figure 2-8 (concluded)

Subsurface Exploration Log B-1

EXPLORATORY BORING LOG

CLIENT: Sacramento Municipal Utility District LOCATION: Rancho Seco Nuclear Generating Station Sacramento County, California	JOB NO: E306-01 DATE: 4/27/93	BORING B-1
DRILL RIG: Mobile B-61 DRILLER: B&F WT. OF HAMMER/DROP: 140 pounds/30 inches	BORING ELEV.: 195 feet BORING DIAM.: 8-inch LOGGED BY: JMP	PAGE 3 OF 3

MATERIAL DESCRIPTION AND REMARKS	CONSISTENCY	USCS	DEPTH (ft.)	SMPL	N blows per ft.	DRY DENSITY (pcf)	WATER CONTENT (%)	PLASTICITY INDEX (%)	UNCONFINED COMPRESSIVE STRENGTH (psf)
SILT, orange brown, low plasticity, dry	HARD	ML	51	10	50/5"				9000+
			52						
			53						
			54						
Sandy CLAY, brown, low plasticity, trace gravel, dry	HARD	CL	55						
			56	11	50/5"				9000+
			57						
			58						
			59						
			60						
		CL	61	12	50/6"				9000+
			62						
End of boring @ 62.0 feet. Groundwater not encountered.			63						
			64						
			65						
			66						
Notes: BS = bulk sample			67						
			68						
			69						
			70						
			71						
			72						
			73						
			74						
			75						

ENVIRONMENTAL GEOTECHNICAL CONSULTANTS, INC.	SUBSURFACE EXPLORATION LOG B-1 Rancho Seco Nuclear Generating Station Sacramento County, California Sacramento Municipal Utility District	FIGURE NO. 3

Figure 2-9

Subsurface Exploration Log B-2

EXPLORATORY BORING LOG

CLIENT: Sacramento Municipal Utility District
LOCATION: Rancho Seco Nuclear Generating Station
Sacramento County, California

JOB NO: E306-01
DATE: 4/27/93

BORING B-2

DRILL RIG: Mobile B-61
DRILLER: B&F
WT. OF HAMMER/DROP: 140 pounds/30 inches

BORING ELEV.: 183 Feet
BORING DIAM.: 8-inch
LOGGED BY: JMP

PAGE 1 OF 3

MATERIAL DESCRIPTION AND REMARKS	CONSISTENCY	USCS	DEPTH (ft.)	SMPL	# blows per ft.	DRY DENSITY (pcf)	WATER CONTENT (%)	PLASTICITY INDEX (%)	UNCONFINED COMPRESSIVE STRENGTH (psf)
6-inch root zone Sandy CLAY, brown, low plasticity, moist	STIFF	CL	1						
			6	1	16				3000
SAND, grayish brown, wet	MED DENSE	SP	11	2	25				
Sandy SILT, orange brown, low plasticity, dry	HARD	ML	13	3	100/5"	87	17.9		9000+
			16	4	68	77	12.2		9000+
			19	5	57	81	16.8		
Sandy GRAVEL, grayish brown, slightly moist	VERY DENSE	GW	25	6	50/4"				

ENVIRONMENTAL GEOTECHNICAL CONSULTANTS, INC.

SUBSURFACE EXPLORATION LOG B-2
Rancho Seco Nuclear Generating Station
Sacramento County, California
Sacramento Municipal Utility District

FIGURE NO. 4

Figure 2-9 (continued)

Subsurface Exploration Log B-2

EXPLORATORY BORING LOG

CLIENT: Sacramento Municipal Utility District LOCATION: Rancho Seco Nuclear Generating Station Sacramento County, California	JOB NO: E306-01 DATE: 4/27/93	BORING B - 2
DRILL RIG: Mobile B-61 DRILLER: B&F WT. OF HAMMER/DROP: 140 pounds/30 inches	BORING ELEV.: 183 Feet BORING DIAM.: 8-inch LOGGED BY: JNP	PAGE 2 OF 3

MATERIAL DESCRIPTION AND REMARKS	CONSISTENCY	USCS	DEPTH (ft.)	SMPL	N blows per ft.	DRY DENSITY (pcf)	WATER CONTENT (%)	PLASTICITY INDEX (%)	UNCONFINED COMPRESSIVE STRENGTH (psf)
Sandy GRAVEL, grayish brown, slightly moist	VERY DENSE	GW	26	6	50/4"				
			31	7	50/4"				
Sandy SILT, orange brown, low plasticity, slightly moist	HARD	ML	36	8	50/4"				9000+
Light brown below @ 40 feet			41	9	50/5"				9000+
			46	10	50/6"				9000+
			50	11	50/5"				9000+

ENVIRONMENTAL GEOTECHNICAL CONSULTANTS, INC.	SUBSURFACE EXPLORATION LOG B-2 Rancho Seco Nuclear Generating Station Sacramento County, California Sacramento Municipal Utility District	FIGURE NO. 4

Figure 2-9 (concluded)

Subsurface Exploration Log B-2

EXPLORATORY BORING LOG

CLIENT: Sacramento Municipal Utility District LOCATION: Rancho Seco Nuclear Generating Station Sacramento County, California	JOB NO: 8306-01 DATE: 4/27/93	BORING B-2
DRILL RIG: Mobile B-61 DRILLER: B&F WT. OF HAMMER/DROP: 140 pounds/30 inches	BORING ELEV.: 183 feet BORING DIAM.: 8-inch LOGGED BY: JMP	PAGE 3 OF 3

MATERIAL DESCRIPTION AND REMARKS	CONSISTENCY	USCS	DEPTH (ft.)	SMPL	N blows per ft.	DRY DENSITY (pcf)	WATER CONTENT (%)	PLASTICITY INDEX (%)	UNCONFINED COMPRESSIVE STRENGTH (psf)
Sandy SILT, orange brown, low plasticity, slightly moist	HARD	ML	51	11	50/5"				9000+
Silty SAND, brown, dry	VERY DENSE	SM	56	12	62				
SILT, brown, low plasticity, dry	HARD	ML	66	13	80				9000+
End of boring @ 75 feet. Groundwater not encountered.			75	14	50/5"				9000+

ENVIRONMENTAL GEOTECHNICAL CONSULTANTS, INC.	SUBSURFACE EXPLORATION LOG B-2 Rancho Seco Nuclear Generating Station Sacramento County, California Sacramento Municipal Utility District	FIGURE NO. 4

3. PRINCIPAL DESIGN CRITERIA

This Section establishes the design criteria for the Rancho Seco ISFSI. These include environmental parameters which the facility must withstand, fuel clad temperature limits, DSC design criteria, etc. Design criteria unique to HSM storage are addressed separately in Volume II.

3.1 Purpose of Installation

The Rancho Seco ISFSI is designed to provide interim storage for 100% of the RSNGS spent fuel assemblies and control components. The facility must store 100% of the spent fuel assemblies and control components since the RSNGS spent fuel pool will be decommissioned as a part of the overall plant decommissioning effort.

3.1.1 Material to be Stored

RSNGS fuel is Babcock & Wilcox 15X15 Mark B PWR fuel. The fuel will be stored as non-consolidated fuel assemblies both with and without non-fuel hardware/control components. Since this is a 100% fuel storage campaign, provisions are made to store assemblies with cladding degradation in a specifically designated DSC.

The total amount of uranium to be stored at the ISFSI is approximately 220.32 metric tons of intact and damaged fuel assemblies.

The Rancho Seco ISFSI is also designed to store Rancho Seco GTCC radioactive waste. Appendix C discusses the storage of GTCC waste.

3.1.1.1 Physical Characteristics

The physical characteristics of the fuel to be stored are described in detail in Section 3.2 of DSAR, Amendment 4 [3.3.1] and are summarized in Table 3-1. The characteristics of the control components are also described in detail in Section 3.2 of DSAR, Amendment 4 and are summarized in Table 3-2.

3.1.1.2 Thermal Characteristics

Since Rancho Seco is in a permanently defueled configuration, the heat load for all 493 fuel assemblies has been quantified prior to ISFSI design and operation.

The ISFSI is designed to store the hottest 24 (or 13 in the case of the FF-DSC) RSNGS fuel assemblies in any single DSC assuming storage campaign initiation after June 1996. Actual heat loads should be much less since many RSNGS fuel assemblies have only a fraction of the design basis thermal power.

The maximum single assembly decay heat power, including control components, is less than 0.679 + 0.085 = 0.764 kW where 0.679 kW is the bounding decay heat from the fuel assembly only and 0.085 kW the bounding decay heat from the control component. For the cask thermal analysis, the total decay heat power in the cask inner cavity of 13.5 kW is used to be consistent with the 10 CFR 71 application for certification of the MP187 package. Therefore, the combined heat load for the 24 fuel assemblies is 13.5 kW per FO- or FC-DSC and 9.93 kW for the 13 failed fuel assemblies in the FF-DSC. These heat loads are considerably less than the DSC heat load for the Standardized NUHOMS®-24P system (24 kW) [3.3.2]. Heat loads were calculated using the computer code ORIGEN2. The calculations are fully described in Volume IV, Calculation 2069.0401.

3.1.1.3 Radiological Characteristics

Since Rancho Seco is in a permanently defueled configuration, the radiological sources for all 493 fuel assemblies have been quantified prior to ISFSI design and operation.

The worst case neutron and gamma-ray source terms were determined assuming a fuel loading date after June 1996. The fuel assembly with the largest neutron source term is a 3.18 weight percent U_{235} initial enrichment, 38,268 MWd/MTU burnup assembly cooled for 13 years. The fuel assembly with the largest gamma-ray source term is a 3.21 weight percent U_{235} initial enrichment, 34,143 MWd/MTU burnup assembly cooled for 7 years. The control component with the largest gamma-ray source is an axial power shaping rod assembly.

The maximum neutron and gamma-ray source terms were combined to form a composite design basis assembly for use in all shielding calculations. The neutron and gamma-ray source strengths and spectra are given in Chapter 7, Tables 7-1 and 7-2, respectively. Radiological source terms were calculated using the computer code ORIGEN2. The calculations are fully described in Volume IV, Calculation 2069.0500.

There are two primary Regenerative Neutron Source assemblies that will be stored inside fuel assemblies. These sources have a sixty-day half-life, and have been removed from the reactor over 20 years. The neutron dose from these sources has essentially decayed to zero. There are also two californium neutron sources that will be stored in fuel assemblies at the ISFSI. These sources are such that they do not provide any significant contribution to the calculated neutron source for the shielding dose analysis. The control components do not contain any fissionable nuclides; therefore, no fission products or fission gases are generated in the control components. The control components contain activation products after irradiation; however, no gaseous effluents will be available for release.

There will also be 26 retainer clips that will be inserted as part of the fuel control components. The retainer clips are made of stainless steel and an Inconel spring, and are relatively small and light weight (4.8 pounds each). The inclusion of these clips will not result in a significant addition to the neutron or gamma sources for the shielding dose analyses.

3.1.2 General Operating Functions

The Rancho Seco ISFSI is similar to the Standardized NUHOMS® System. Each aspect of criticality control, radiation protection, containment, and heat rejection is accomplished through passive means. Additional discussions concerning the general operating functions pertaining to the ISFSI are discussed in detail in Section 3.1.2 of the Standardized NUHOMS® SAR [3.3.2].

See Appendix B for Standardized SAR, Section 3.1.2 (pages 3.1-3 to 3.1-6).

3.1.2.1 Handling and Transfer Equipment

The handling and transfer equipment of the Rancho Seco ISFSI are similar to those discussed in Section 3.1.2.1 of the Standardized NUHOMS® SAR [3.3.2].

See Appendix B for Standardized SAR, Section 3.1.2.1 (pages 3.1-3 to 3.1-6).

3.1.2.2 Waste Processing, Packaging and Storage Areas

There are three types of contaminated waste produced as a result of dry storage activities. These are the contaminated water drained from the DSC cavity, the potentially contaminated air and helium evacuated from the DSC, and the wet and dry active waste from the loading, drying, sealing, and decontamination of the DSC. All contaminated water is returned to either the spent fuel pool or the plant's liquid radioactive waste system. All potentially contaminated air and helium will be filtered, monitored, and discharged through the RSNGS Auxiliary Building Stack, or filtered in the event a cask is loaded at the ISFSI. The dry active waste will be disposed of in accordance with existing RSNGS radioactive waste handling procedures.

3.2 Structural and Mechanical Safety Criteria

The Rancho Seco ISFSI system components which are important to safety include the reinforced concrete HSM and its DSC support structure, the FO-DSC, FC-DSC, FF-DSC, the cask, and lifting yoke including any extensions. Since the cask will not be lifted over 80 inches once placed on the transfer trailer, the lifting yoke and lifting yoke extensions are not important to safety for storage purposes; however, they are given this classification to satisfy the evaluation of lifting the cask onto the transfer trailer under the Rancho Seco 10 CFR 50 license. Since the Rancho Seco ISFSI is an independent, passive system, no other components or systems contribute to its safe operation. Many of the components are similar to the Standardized NUHOMS® system components analyzed in the Standardized NUHOMS® SAR [3.3.2].

The extreme environmental and natural phenomena design criteria for the Rancho Seco ISFSI components are discussed below.

The design criteria for the Rancho Seco ISFSI components which are important to safety for HSM storage are discussed in Volume II, Section 3.2.

The design criteria for the Rancho Seco ISFSI components which are important to safety for cask handling are discussed in the following sections. The DSC and cask loading conditions for cask handling operations are summarized in Table 3-3 and Table 3-4, .

3.2.1 Tornado and Wind Loadings

3.2.1.1 Applicable Design Parameters

The Standardized NUHOMS® SAR [3.3.2] design parameters were used to define the design basis tornado (DBT) and wind loadings for the Rancho Seco ISFSI. The Standardized NUHOMS® system was designed to operate anywhere within the 48 contiguous states, and therefore its design conditions bound Rancho Seco site-specific conditions.

3.2.1.2 Determination of Forces on Structures

The design basis forces on the cask and reinforced concrete HSM are described in Section 3.2.1.2 of the Standardized NUHOMS® SAR [3.3.2]. The maximum DBT design pressure loads of 397 psf on the windward side and -357 psf (suction) on the leeward side of the HSM are assumed to act as uniform pressure loads on the HSM walls. The effects of the DBT wind loads on sliding and overturning stability of the HSM are considered in addition to the resulting concrete forces and moments. The HSM tornado winds analysis results are contained in Volume II, Section 8.3.1.

See Appendix B for Standardized SAR, Section 3.2.1.2 (pages 3.2-2 to 3.2-3).

3.2.1.3 Ability of Structures to Perform Despite Failure of Structures Not Designed for Tornado Loads

The HSM protects the DSC from adverse environmental effects and is the principal ISFSI structure exposed to tornado wind and missile loads. Furthermore, all components of the HSM (regardless of their safety classification) are designed to withstand tornadoes and tornado-based missiles. The cask protects the DSC during transit to the ISFSI from adverse environmental effects such as tornado winds and missiles.

Since the HSMs are located outdoors away from other RSNGS structures, there is no possibility of an adjacent building collapsing on an HSM. The possibility of blocking the ventilation air openings by a foreign object during a tornado event, however, is considered. The effects of ventilation opening blockage are presented in Volume II, Section 8.3.5.

3.2.1.4 Tornado Missiles

The HSM and cask are evaluated for the effects of three types of tornado-driven missiles. The tornado missile parameters are identical to those used for the Standardized NUHOMS® HSM and transfer cask and are described in detail in Section 3.2.1.2 of the Standardized NUHOMS® SAR [3.3.2]. The HSM and cask sliding and overturning stability is considered for tornado missile impacts in addition to the NUHOMS® component stresses. The HSM and cask tornado missile analysis results are contained in Volume II, Section 8.3.1 and Volume III, Section 8.3.1, respectively.

See Appendix B for Standardized SAR, Section 3.2.1.2 (pages 3.2-2 to 3.2-3).

3.2.2 Water Level (Flood) Design

The Rancho Seco ISFSI is located at an elevation of approximately 177 feet above sea level on a plant grade that permits excellent drainage. The drainage system at the Rancho Seco site is designed to prevent flooding for a 100-year recurrence storm. Therefore, the probability of flooding at the ISFSI is minimal. The HSMs, DSCs, and cask are conservatively designed for the enveloping design basis flood defined in Section 3.2.2 of the Standardized NUHOMS® SAR [3.3.2], consisting of a 15 m (50 foot) flood height and water velocity of 4.6 m/sec (15 fps). The loads on the HSMs and DSCs resulting from the postulated flood event are discussed in Volume II, Section 3.2.2.

Flooding of the ISFSI greater than 0.46 m (1'-6") above grade results in blockage of the HSM inlet vents. Flooding of the ISFSI greater than 1.7 m (5'-8") above grade results in wetting of the DSC. Greater flood heights result in submersion of the DSC and blockage of the HSM outlet vents.

The DSC and HSM are conservatively designed for an enveloping design basis flood, postulated to result from natural phenomena such as a tsunami, and seiches, as specified by 10 CFR 72.122(b). For the purpose of this bounding generic evaluation, a 15 m (50 foot) flood height and water velocity of 4.6 m/sec (15 fps) impinging upon the side of a submerged HSM. The DSC is subjected to an external pressure equivalent to a 15 m (50 foot) head of water. These evaluations are presented in Section 8.2.4. The effects of water reflection on DSC criticality safety are addressed in Section 3.3.4. Due to its short term infrequent use, the cask is not explicitly evaluated for flood effects. ISFSI procedures should ensure that the cask is not used for DSC transfer during flood conditions.

The calculated effects of the enveloping design basis flood are included in the load combinations and reported stresses presented in Section 8.2.10

3.2.2.1 Flood Elevations

The flood elevations used in the design of the HSMs, DSCs, and cask for buoyancy and static water force effects is 15 m (50 ft) above the ground level at the ISFSI.

3.2.2.2 Phenomena Considered in Design Load Calculations

The phenomena considered in the flood design loading are identical to those described in the Standardized NUHOMS® SAR [3.3.2].

3.2.2.3 Flood Force Application

The flood forces applied to the HSMs, cask, and DSCs are identical to those described in Section 8.2.4 of the Standardized NUHOMS® SAR [3.3.2]. The analysis of the Rancho Seco ISFSI component for the flood loading are presented in Volume II, Section 8.3.3 and Volume III, Section 8.3.3.

See Appendix B for Standardized SAR, Section 8.2.4 (pages 8.2-23 to 8.2-26).

3.2.2.4 Flood Protection

Since the Rancho Seco ISFSI is an independent passive system, no other components or systems contribute to its safe operation. Therefore, no additional flood protection measures for storage structures are necessary.

3.2.3 Seismic Design

The design basis response spectra of NRC Regulatory Guide (R.G.) 1.60 [3.3.3] is selected for the design earthquake as defined in 10 CFR 72.102(a)(2). Since the DSC can be considered to act as a large diameter pipe for the purpose of evaluating seismic effects, the "Equipment and Large Diameter Piping System" category in the NRC Regulatory Guide 1.61, Table 1 [3.3.9] is assumed to be applicable. Hence, a damping value of three percent of critical damping for the design bases safe shutdown earthquake is used. Similarly, from the same R.G. table, a damping value of seven percent of critical damping is used for the reinforced concrete HSM. The horizontal and vertical components of the design response spectra (in Figures 1 and 2, respectively, of the NRC Regulatory Guide 1.60) correspond to a maximum horizontal and vertical ground acceleration of 1.0g. The maximum ground displacement is taken to be proportional to the maximum ground acceleration, and is set at 36 inches for a ground acceleration of 1.0g.

NRC Regulatory Guide 1.60 also states that for sites with different acceleration values specified for the design basis earthquake, the response spectra used for design should be linearly scaled from R.G. Figures 1 and 2 in proportion to the maximum specified horizontal ground acceleration. The maximum horizontal ground acceleration component selected for design of the ISFSI is 0.25g. The maximum vertical acceleration component selected is two-thirds of the horizontal component, which is 0.17g. These ground acceleration values comply with the requirements of 10 CFR 72.102(a)(2) for sites underlaid by rock east of the Rocky Mountain front, except in the areas of known seismic activity.

In order to establish the amplification factor associated with the generic design basis response spectra, various frequency analyses are performed for the system components. The results of these analyses indicate that the dominant lateral frequency for the reinforced concrete HSM is 38.1 Hertz. The dominant frequency of the DSC and the support structure is calculated to be 17.4 Hertz. The corresponding horizontal seismic acceleration used for design of the HSM is 0.25g. The dominant HSM vertical frequency exceeds 33 Hertz, which produces a vertical seismic design acceleration of 0.17g. The resulting seismic design accelerations used for the DSC are 0.37g horizontally and 0.17g vertically. The seismic analyses of the HSM and DSC are discussed further in Section 8.2.3 of the Standardized NUHOMS® SAR [3.3.2].

Seismic-System Analyses

Seismic Analysis Methods. The seismic analysis methods used to evaluate the Rancho Seco ISFSI components for the HSM storage mode are identical to those described in Section 8.2.3.2 of the Standardized NUHOMS® SAR [3.3.2]. The cask and the DSC inside the cask are evaluated for the seismic loading using equivalent static loading. Seismic evaluations are performed for the cask and DSC while in the horizontal transfer mode. See Appendix B for Standardized SAR, Section 8.2.3.2 (pages 8.2-14 to 8.2-23).

Natural Frequencies and Response Loads. The dominant natural frequencies of the HSM and DSC in the HSM are evaluated in Section 8.2.3.2 of the Standardized NUHOMS® SAR [3.3.2]. The dominant structural frequencies calculated for a loaded HSM in the lateral direction are 17.4 Hz and 38.1 Hz for the DSC and HSM, respectively. Equivalent horizontal and vertical static loads of 0.37g and 0.25g, respectively, are calculated using the appropriate amplification factors in accordance with the requirements of NRC Regulatory Guide 1.60 [3.3.3].

See Appendix B for Standardized SAR, Section 8.2.3.2 (pages 8.2-14 to 8.2-23).

Hand calculations are used to determine the dominant natural frequencies of the DSC. The natural frequency for the shell ovalling mode is 13.8 Hz and that for the beam bending mode is 62.8 Hz for the FO-DSC. These result in spectral accelerations of 1.0g in the horizontal direction and 0.68g in the vertical direction [3.3.2].

The dominant natural frequencies of the cask are determined using hand calculations. The dominant structural frequencies of the cask are 17.9 Hz and 83 Hz for the cask shell ovalling and beam bending modes. Based on the cask structural frequencies, an amplification factor of 2.5, determined in accordance with NRC Regulatory Guide 1.60 [3.3.3], is applied to both the peak horizontal and vertical acceleration. A factor of 1.5 is applied to the seismic acceleration loads to account for the effects of possible multimode excitation. Therefore, the resulting equivalent static horizontal and vertical acceleration loads for this cask are 0.95g and 0.65g, respectively.

Methods to Determine Overturning Moments. The HSM overturning moments and the DSC lift-off moment from the HSM DSC support rails are calculated using conservative static methods identical to those used in the Standardized NUHOMS® SAR [3.3.2]. The overturning moment is conservatively calculated for a single free-standing HSM assuming the peak horizontal and vertical seismic accelerations act simultaneously. The HSM and DSC seismic stability analysis results for the HSM storage mode are presented in Volume II, Section 8.3.2.

3.2.4 Snow and Ice Loadings

The Standardized NUHOMS® HSM snow and ice loads, as discussed in Section 3.2.4 of the Standardized NUHOMS® SAR [3.3.2], are used as the design basis Rancho Seco ISFSI loads. This is quite conservative since snow and ice conditions in California's Central Valley are significantly bounded by the Standardized NUHOMS® environmental conditions.

Snow and ice loads for the HSM are conservatively derived from ANSI A58.1-1982. The maximum 100 year roof snow load, specified for most areas of the continental United States for an unheated structure, of 5.27 kN/m^2 (110 psf) is assumed. For the purpose of this conservative generic evaluation, a total live load of 9.58 Kn/m^2 (200 pounds per square foot) is used in the HSM analysis to envelope all postulated live loadings, including snow and ice. Snow and ice loads for the on-site transfer cask with a loaded DSC are negligible due to the smooth curved surface of the cask, the heat rejection of the SFAs, and the infrequent short term use of the cask.

3.2.5 Load Combination Criteria

3.2.5.1 Horizontal Storage Module

The design approach, design criteria and loading combinations for the reinforced concrete HSM and its DSC support structure are discussed in Volume II, Section 3.2.5.1.

3.2.5.2 Dry Shielded Canister

The FO-DSC, FC-DSC, and FF-DSC design approach, design criteria and load combinations for HSM storage are discussed in Volume II, Section 3.2.5.2. Table 3-5 and Table 3-6 provide a summarization.

The FO-DSC, FC-DSC, and FF-DSCs are designed for the cask handling mode using a similar design approach, design criteria, and load combinations as specified for the standardized NUHOMS DSC in Section 3.2.5.2 of the Standardized NUHOMS SAR [3.3.2]. The FO-DSC, FC-DSC, and FF-DSC load combination results are presented in Section 8.3.1. The effects of fatigue on the FO, FC, and FF-DSCs due to thermal cycling are addressed in Section 8.1.1.7. The Dry Shielded Canister code of construction is described below, and structural design criteria are summarized in Table 3-7.

The DSC is designed by analysis to meet the stress intensity allowables of the ASME Boiler and Pressure Vessel Code (1992 Code, 1993 Addendum) Section III, Division I, Subsection NB, NF, and NG for Class I components and supports. The DSC is conservatively designed by using linear elastic or non-linear elastic-plastic analysis methods. The load combinations considered for the DSC normal, off-normal, and postulated accident loadings are shown in Table 3-6. ASME Code Service Levels A and B allowables are conservatively used for normal and off-normal operating conditions. Service Levels C and D allowables are used for accident conditions such as a postulated cask drop accident.

Using this acceptance criteria ensures that in the event of a design basis drop accident, the DSC containment pressure boundary is not breached. As indicated by the results of the analysis of Section 8.2.5 of the Standardized NUHOMS® SAR [3.3.2], the amount of deformation sustained by the spacer disks does not inhibit retrieval of the fuel assemblies. The maximum shear stress theory is used to calculate principal stresses. Normal operational stresses are combined with the appropriate off-normal and accident stresses. It is assumed that only one postulated accident condition occurs at any one time. The accident analyses are documented in Section 8.2. The structural design criteria for the DSC are summarized in Table 3-7. The effects of fatigue on the DSC due to thermal and pressure cycling are addressed in Section 8.2.10 of the Standardized NUHOMS® SAR [3.3.2].

3.2.5.3 NUHOMS®-MP187 Cask

The cask components which serve as the pressure retaining boundary in the postulated storage mode of operation are designed by analysis to meet the stress allowables of the ASME Code [3.3.4] Subsection NB for Class 1 components. All other cask structural components are designed by analysis to meet the stress allowables of the ASME Code Subsection NB for structural or shell components or NF for the neutron shield jacket assembly.

The cask is conservatively designed by utilizing linear elastic analysis methods. The top cover closure bolts are evaluated against NUREG/CR-6007 [3.3.11]. The analyses for all other cask load conditions are presented in Chapter 8. The effects of fatigue on the transfer cask due to thermal cycling are addressed in Sections 8.1.1.8 and 8.1.1.9.

The load combinations considered for the transfer cask normal, off-normal, and postulated accident loadings for cask handling are shown in Table 3-8. Service Levels A and B allowables are used for all normal operating and off-normal loadings. Service Levels C and D allowables are used for load combinations which include postulated accident loadings. Allowable stress limits for the lifting trunnions are conservatively developed to meet the requirements of ANSI N14.6-1993 [3.3.5] for critical loads. The transfer cask structural design criteria are summarized in Table 3-9 and Table 3-10.

3.3 Safety Protection System

3.3.1 General

The Rancho Seco ISFSI is designed for safe containment during dry storage of SFAs. The components, structures, and equipment which are designed to assure that this safety objective is met are summarized in Table 3-11. The key elements of the ISFSI and its operation which require special design consideration are:

1. Minimizing the contamination of the DSC exterior by fuel pool water.
2. The double closure seal welds on the DSC shell to form a pressure retaining containment boundary and to maintain a helium atmosphere.
3. Minimizing personnel radiation exposure during DSC loading, closure, and transfer operations.
4. Design of the cask and DSC for postulated accidents.
5. Design of the HSM passive ventilation system for effective decay heat removal to ensure the integrity of the fuel cladding.
6. Design of the DSC basket assembly to ensure subcriticality.

3.3.2 Protection by Multiple Confinement Barriers & Systems

3.3.2.1 Confinement Barriers and Systems

Section 3.3.2.1 of the Standardized NUHOMS® SAR [3.3.2] describes the Rancho Seco ISFSI confinement barriers and systems.

See Appendix B for Standardized SAR, Section 3.3.2.1 (pages 3.3-1 to 3.3-2).

3.3.2.2 Ventilation - Offgas

The Rancho Seco ISFSI is a passive system. The ventilation of the HSMs is driven by natural convection heat transfer. There is no requirement for an offgas or monitoring system due to the DSC design.

The system relies on natural convection through the air space in the HSM to cool the DSC. This passive convective ventilation system is driven by the pressure difference due to the stack effect (ΔP_s) provided by the height difference between the bottom of the DSC and the HSM air outlet, which is larger than the flow pressure drop (ΔP_f) at the design air inlet and outlet temperatures. The details of the ventilation system design are provided in Chapters 4 and 8.

There are no radioactive releases of effluents during normal and off-normal storage operations. Also, there are no credible accidents which cause significant releases of radioactive effluents from the DSC. Therefore, there are no off-gas or monitoring system requirements for the HSM. During DSC drying or reflood operations, the spent fuel pool or radwaste system is used to process any offgas from the DSC.

3.3.3 Protection by Equipment and Instrumentation Selection

3.3.3.1 Equipment

The Rancho Seco HSMs, DSCs, and transfer cask are designated important to safety. The cask lifting yoke and extensions are designated as important to safety only for their intended use during the fuel loading operations. Other important to safety equipment is required for handling operations within the RSNGS fuel building. These operations are performed under the RSNGS 10 CFR 50 operating license.

3.3.3.2 Instrumentation

To provide a positive means to identify off-normal thermal conditions, the HSM roof concrete temperatures will be monitored. This monitoring system will include a non-safety remote readout. The temperature indications will also be accessible in the ISFSI Electrical Building.

3.3.4 Nuclear Criticality Safety

The design criteria for the Rancho Seco ISFSI requires that the DSCs be designed to remain subcritical under normal, off-normal, and accident conditions. The design of the DSC is such that, under all credible conditions, the highest effective neutron multiplication factor (k_{eff}) remains less than 0.95.

The NUHOMS®-MP187 Cask criticality analysis performed for offsite shipment of Rancho Seco fuel [3.6] bounds the conditions for onsite storage because 1) there is no credible event which would result in the flooding of a DSC in HSM storage and 2) there are no events which could occur during DSC fuel loading procedures which would result in k_{eff} exceeding the worst case 10 CFR 71 transportation conditions. The NUHOMS®-MP187 Transportation SAR [3.6] was submitted to the NRC Transportation Branch in a simultaneous license application. The NRC issued Certificate of Compliance number 71-9255 in September 1998 for the NUHOMS®-MP187 transportation package. Specific information on the criticality safety analysis which bounds the Rancho Seco ISFSI is discussed in this section.

3.3.4.1 Control Methods for the Prevention of Criticality

Subcriticality is maintained during all phases of operations and storage by a combination of mechanical and neutronic separation of the fuel assemblies. Administrative controls are not required for criticality control.

The Rancho Seco FO- and FC-DSCs include the use of fixed neutron absorbing material in the DSC basket. No credit is taken for the presence of dissolved boron in the pool during fuel loading. Therefore the resulting design offers a substantial margin of criticality safety.

The FF-DSC does not require borated materials for criticality control. Because it has fewer fuel assemblies, the FF-DSC basket has larger flux traps and thicker guide sleeves to provide a sufficient degree of neutron attenuation to assure $k_{eff} < 0.95$.

3.3.4.1.1 Fuel-Only (FO) DSC Design Features

The principal performance features of the FO-DSC as they relate to criticality control are:

A. The package is designed such that it would be subcritical if unborated water were to fill the canister. No credit is taken for the borated water in the fuel pool.

B. The criticality analyses have been performed with consideration for the most reactive credible configuration consistent with the chemical and physical form of the material.

The FO-DSC basket support structure is composed of four axially oriented support rods and twenty-six spacer discs. This basket assembly provides positive location for twenty-four fuel assemblies under normal operating conditions (NOC), off-normal operating conditions and accident conditions. The basket assembly uses fixed neutron absorbers that isolate each fuel assembly. Guide sleeves are designed to permit unrestricted flooding and draining of fuel cells.

The neutron absorber panel material was chosen due to its desirable neutron attenuation, low density, and minimal thickness. It has been used for applications and in environments comparable to those found in spent fuel storage and transportation since the early 1950s (the U.S. Atomic Energy Commission's AE-6 Water-Boiler Reactor). In the 1960s, it was used as a poison material to ship irradiated fuel rods from Canada's Chalk River laboratories to Savannah River. More than 12,000 British Nuclear Fuels, Ltd. (BNFL) flasks containing the material have been used to transport fuel to BNFL's reprocessing plant in Sellafield.

The neutron absorber panels are composed of boron carbide and 1100 alloy aluminum. Boron carbide provides the necessary content of the neutron absorbing B10 isotope in a chemically inert, heat resistant, highly crystalline and extremely hard form. Boron carbide contained in the panels does not react under these conditions. The boron carbide core is tightly held within an 1100 aluminum alloy matrix and further protected by solid 1100 aluminum alloy cladding plates.

The efficacy of the neutron absorber panels throughout the 60-year design life of the DSC is demonstrated in the following manner:

1. Acceptance tests are performed on the panels during fabrication as described below and in compliance with Section 8.1.8 of the NUHOMS®-MP187 transportation SAR [3.6]. The neutron absorber plates are verified to have the minimum total B^{10} per unit area (areal density) of the sandwiched material as specified on the drawings in Volume IV. Samples from each sheet of the neutron absorber are retained for testing and record purposes. The minimum areal B^{10} content and the uniformity of dispersion within a panel are verified by wet chemical analysis and/or neutron attenuation testing. All material certifications, lot control records, and test records are maintained to assure material traceability.

2. Depletion of the poison material over the storage period is negligible. Using the results of the canister shielding models described in Volume III, Chapter 7 and in Reference Calculation 2069.0503, the maximum neutron flux in the DSC during storage is 2.6×10^5 neutrons/sec/cm^2. A new calculation was performed using the Monte Carlo N-Particle computer code. The maximum B10 depletion ratio calculated is **[** Note 1 **]** This represents a **[** Note 1 **]** depletion over a period of 100 years, which envelops the 60-yr total storage period with the additional PEO.

3. The structural integrity of the material and the potential for material degradation are acceptable throughout the design life of the DSC. No credit is taken in the design for any structural strength of the absorber material. The neutron absorber panels are fully supported by the DSC guide sleeves and wrappers as shown on the drawings. The absorber sheets are maintained within their recommended temperature limits throughout the storage life, and are maintained in an inert environment. There is, therefore, no mechanism by which degradation of the material is possible.

3.3.4.1.2 Fuel-Control Components (FC) DSC Design Features

The FC-DSC is designed with a longer internal cavity length to accommodate fuel assemblies with control components. No credit is taken for the presence of control hardware, thus the FC-DSC is identical to the FO-DSC for the purpose of criticality analysis. This is conservative because no credit is taken for the dissolved boron in the water during loading operations. Therefore since the maximum calculated k_{eff} for the FO-DSC is calculated at the optimum moderator (fresh water) density, replacing moderator in the guide tubes with control components by definition reduces the reactivity of the system. Further references to the FO-DSC apply to this canister design also.

Note 1 – The values stated on page C-18 of Reference [3.24] are incorporated by reference into this IFSAR

3.3.4.1.3 Failed Fuel (FF) DSC Design Features

The FF-DSC is different from the FO-DSC in its capacity, function, and design. The FF-DSC's capacity is thirteen fuel assemblies and is intended to package fuel with cladding defects identified during a previous visual inspection of the spent fuel assemblies. Fuel assemblies to be stored were visually inspected to document that cladding damage is limited to no more than 15 fuel pins with known or suspected cladding damage greater than hairline cracks and pinhole leaks. Missing cladding and/or crack size in the fuel pins is limited such that a fuel pellet is not able to pass through the gap created by the cladding opening during normal handling. The fuel must not have damage that would preclude it from being handled in the ordinary manner. Each assembly is placed in a separate, removable can with a fixed mesh screen on the bottom and similarly screened lid on top. These cans have slightly larger interior dimensions than the FO-DSCs (9.00 in. vs. 8.90 in.) to accommodate bowed or twisted fuel. Due to its smaller payload and the relatively massive nature of the FF-DSC cans, the FF-DSC does not require borated neutron absorbers. The fuel cans are designed to permit unrestricted flooding and draining of fuel cells.

The FF-DSC is analyzed using the same criteria as the FO-DSC, plus additional considerations arising from mechanical uncertainties of failed fuel after transport or hypothetical accident conditions.

3.3.4.2 Spent Fuel Loading

The Rancho Seco ISFSI is designed to accommodate any of the three DSCs as described above. The design basis fuel is B&W 15x15 Mark B fuel with a maximum fuel enrichment of 3.43 w/o U235. The fuel loading parameters as they relate to criticality are summarized in Table 3-13. The design properties of the reference fuel are given in Table 3-14.

3.3.4.3 Model Specification

3.3.4.3.1 Description of Calculational Model

The criticality calculations were done using full-transfer cask (NUHOMS®-MP187) KENO5A-PC [3.12] models. They are described in detail below. Input files are available in Appendix 6.6.2 of the MP187 Part 71 SAR [3.6].

The safety requirements of ANSI/ANS-8.17 [3.13] prescribe that all applicable biases and uncertainties must be investigated and statistically attached to the nominal case k_{eff}. Rather than a statistical approach, this criticality analysis models the system with all the important parameters concurrently in their worst-case state:

- Maximum fabrication thickness and minimum boron content for all the neutron absorber plates (this combination is the worst case since aluminum displaces moderator and is not a strong absorber),
- Minimum fabrication width for all the neutron absorber plates,

- Minimum fabrication thickness for all steel guide tubes and steel absorber wrappers,
- Only 75% credit taken for the boron in neutron absorber plates,
- Worst-case fuel assembly position (includes DSC fabrication tolerances and an allowance for fuel assembly bow and twist),
- Maximum enrichment (3.43 w/o U235) B&W 15x15 Mark B fuel.

3.3.4.3.2 Transfer Cask (NUHOMS®-MP187)/FO-DSC Model

The KENO models consist of 560 axial layers stacked into an array. The layers consist of partial spacer disc and partial moderator regions inside and outside of the active fuel region. The very top and bottom of the model are the DSC steel cylinder. The length of the active fuel layers is equivalent to the greatest common denominator of the spacer disc and moderator region axial lengths. For example, five 0.25 in. layers of the spacer disc are stacked to make an equivalent 1.25 in. spacer disc region. The center to center spacing of the spacer disc intervals varies over a range starting at 0.0 and ending at 6.75 inches. However, some of these intervals occur in non-fuel areas. This axially finite arrangement is shown in Figure 3-1. By specifying specular reflection on the $\pm$x and $\pm$y directions of these array layers, the model represents an infinite array of casks.

Figure 3-2 shows the KENO model in an exploded view. UNIT 33 is a slice through the cask at the DSC spacer disc level. UNIT 34 is a similar slice, but in between the spacer discs.

Figure 3-3 shows the structure of UNITs 33 and 34: the cask slices. Note that the difference between the two UNITs is that UNIT 33 is a spacer disc (steel surrounding fuel assemblies) and UNIT 34 has steel support rods only (water surrounding fuel assemblies). Also, for the accident condition cases, there is no guide sleeve deformation within the spacer disc (Unit 33) region. The fuel assemblies are identified in Figure 3-3 by the position numbers (1-24) used to refer to their unique locations. UNIT numbers 1-8 represent the active fuel assemblies in the spacer disc region and UNIT numbers 82-89 represent the active fuel assemblies in the moderator region. The fuel assemblies are inserted into the model using KENO's HOLE capability.

A detail of the guide sleeve assembly is shown in the enlarged section of Figure 3-3. These models include all major components of the guide sleeve assembly: the square tube, absorber sheets (4 per tube), and the over sleeves which hold the sheets in place. Note that the guide sleeves on the outer periphery of the basket (12 total) only have two absorber sheets per tube.

Figure 3-4 shows more closely the way in which UNITs 1-8 are constructed. Each HOLE is identified by UNIT number and its own particular coordinate origin.

UNIT 32 is a cross section of the design basis B&W 15x15 Mark B fuel assembly. It is illustrated in Figure 3-5 , which also shows the locations of the fuel assembly guide tubes, instrumentation tube, and the UNIT origin for insertion as a HOLE. The theoretical half width of the fuel (fifteen times half the rod pitch) is 4.26 in. (10.8204 cm).

3.3.4.3.3 Transfer Cask (NUHOMS®-MP187)/FF-DSC Model

The transfer cask (NUHOMS®-MP187)/FF-DSC KENO model is constructed in the same "slice of the cask" style as the MP187/FO-DSC model. The major differences are:

- 13 storage locations
- stainless steel fuel cans, no absorber panels/guide tubes
- different spacer disc pitch
- different support rod orientation (note that the support plates are modeled as an equivalent cylinder)

3.3.4.3.4 Cask Regional Densities

Table 3-15 and Table 3-16 summarize the calculated atom densities used in the KENO models. Note that, when using the Hansen-Roach working library, resonance nuclides are specified by their σ_{peff}, thus the U235 and U238 atom density specifications in Table 3-15 are unique to each moderator state.

3.3.4.4 Criticality Calculation

3.3.4.4.1 Calculational Method

Criticality calculations for the Rancho Seco ISFSI are performed using the microcomputer application KENO5A-PC [3.12] and the Hansen-Roach 16-group (HR-16) cross section working library. In order to use the HR-16 library, σ_{peff}, the effective resonance cross section, must be calculated for each resonance nuclide of interest (for this work, U235 and U238). σ_{peff} includes both resonance self shielding and heterogeneous effects. The proper working library nuclide, or more generally nuclides, must be selected from the HR-16 library based on σ_{peff}.

Corrections for resonance and heterogeneous effects are performed using the Transnuclear West proprietary program PN-HET. PN-HET was developed during TNW's validation of KENO5A-PC as a means to streamline and unify the analytical approach used to calculate σ_{peff}. The calculational procedure is to:

A. Calculate σ_{peff} for U235 and U238 in the fuel rods.

B. Select H-R library nuclides with σ_{peff} above and below the calculated value.

C. Perform a weighted average to accurately represent the resonance nuclide using a mixture of the two selected nuclides.

The major assumptions made in the KENO modeling are:

A. Unirradiated fuel - no credit taken for fissile depletion or fission product poisoning.

B. No credit taken for fuel control components (applies to FC-DSC only).

C. Fuel is intact with no gross damage or missing rods (applies to FO/FC-DSCs only).

D. The fuel enrichment is modeled as uniform throughout the assembly. The maximum pellet enrichment is assumed to exist everywhere.

E. Fuel and cask are modeled as having finite length (water reflection is specified top and bottom) in all models.

F. Only 75% credit is taken for boron in neutron absorber panels.

G. All fuel rods are assumed to be filled with 100% moderator in the fuel cladding gap.

3.3.4.4.2 Fuel Loading Optimization

The Rancho Seco ISFSI criticality analysis is performed for 3.43 w/o enriched, Rancho Seco B&W 15x15 Mark B PWR fuel. The KENO models were specified with either 100% specular albedo, or infinite water conditions on all four sides. All void regions of the package have been modeled with optimum moderation, including the fuel pellet-clad gaps. Further discussion regarding the models can be found in Section 0.

3.3.4.4.2.1 Fuel Loading Optimization- Failed Fuel Considerations

As mentioned in Section 0, the FF-DSC has been analyzed for additional considerations arising from mechanical uncertainties of failed fuel after a hypothetical accident. In the event of a severe transportation accident, rod breakage may be postulated to occur in rods with known pre-existing gross cladding failure. This may result in a more reactive configuration than undamaged fuel, therefore a specification limiting the number of known rods with gross cladding damage per fuel assembly is established in Table 3-13. The maximum number of permissible rods with gross cladding damage was determined by a series of KENO models of a design basis fuel assembly. These models were constructed to evaluate the effects of radial movement of fuel rod pieces (the result of "single-ended" breaks), and axial movement (the result of "double-ended" breaks). Loose fuel pellets or shards may become dislodged if a rod becomes severed, but this will not result in a more reactive state than the cases described below because the fuel assembly is undermoderated by design. As shown by Figure 3-8, it is larger segments of rods that form the limiting case for criticality. The models used to study these limiting breaks are described below.

Single breaks

"Free ends" caused by break were assumed to move towards or away from the rest of the assembly. Increasing the rod spacing of the broken rods was found to increase k_{eff}. Conversely, k_{eff} decreases for local decreases in rod pitch. Rods on the exterior of the fuel assembly were displaced in the models and the assembly was assumed to be pressed in the corner of the fuel cell, thus maximizing the potential rod displacement. Since internal rods can not move as far as rods on the outside of the assembly, they are not limiting. For modeling simplicity, an entire face of 15 rods was assumed to evenly move away from the remainder of an assembly, as shown in Figure 3-7. This overpredicts the effect of single rod breaks since the fuel's grid spacers will limit radial rod displacement over most of the rod's length. The results of the evaluation are presented in Table 3-21. The file names reflect the distance the 1 x 15 row of dislocated rods is shifted from the edge of the remaining 14 x 15 array. (i.e. FFSS020 indicates the 1 x 15 row is 0.20 inches to the right of the 14 x 15 array. The internal moderator is maintained at 100% for all cases. Note: the remaining 14x15 assembly array is "pushed" up against the upper left hand corner of the guide sleeve to provide the most room to move the 1x15 array toward and away from the remaining assembly.

Double breaks

The effects of pieces of fuel rod migrating axially was investigated by axially moving sections of rods in the models. Again, the fuel assembly was assumed to be in the worst case position: pressed in the corner of the fuel cell as shown in Figure 3-7. A study was performed to determine what length of broken rod was the worst case. The results, along with a sketch of the model configuration, are shown in Figure 3-8. The results show that fuel assemblies that have longer sections of broken rod are the worst case.

The limiting case was found to be the double-ended break. The double-ended break models presented in Table 3-21 were run with all assemblies in the worst case configuration shown in Figure 3-7. The NOC double-ended break models presented in Table 3-21 were run assuming the assemblies were intact, but in the worst case location (all pressed inward toward the center of the FF-DSC).

3.3.4.4.3 Criticality Results

The calculated maximum k_{eff} for the Rancho Seco ISFSI is 0.94968 including all biases and uncertainties applicable to the calculation methodology and the design.

Reactivity calculations were performed in six sets of parametric studies for the transfer cask/FO-DSC (assumed to bound the transfer cask/FC-DSC) and five sets for the transfer cask/FF-DSC. The parametric studies were designed to meet the range of conditions summarized in Table 3-17.

The parametric studies for the FO-DSC consist of the following. The first parametric study shows how the guide sleeve deformation affects the multiplication factor for a finite model of the transfer cask. It is very important to note that the deformation is modeled in the moderator regions only of the active fuel. The spacer discs support the fuel assemblies so the assemblies will deform between the spacer discs. No deformation is modeled in the NOC cases. The deformation study was made for the worst moderator condition case, which is the accident condition case with 70wt% external moderator density.

The following four parametric studies consist of finite 3D geometric models of the FO-DSC in the transfer cask. They include NOC, off-normal and accident condition studies with both internal and external moderator variation (independent of each other). A unique PN-HET run must be made for each internal moderator run, due to the fact that the fuel unit cell is changing. However, this is unnecessary for the external moderator cases. The spacer disc "cutouts" are explicitly modeled. The tolerances of the cutout center locations as well as the cutout size tolerances were considered in the worst case configuration in order to be as conservative as possible. However, the minimum allowable ligament size was the controlling factor and was always maintained. The ligament represents the steel region between cutouts. Finally, the deformation of the guide sleeve was labeled as the 6 o'clock direction for the accident conditions cases. Although the guide sleeve deformation was in the 6 o'clock direction, the fuel assemblies were all positioned toward the center of the cask. Thus, the deformation appears to be in the direction away from the center of the cask.

The final parametric study takes a look at the effect of removing cask layers for the worst case and replacing them one at a time with water from the outside towards the inside of the cask. This parametric study was also made for the FF-DSC worst case. The final parametric study determines the effect of close fuel reflection of the containment system by water on all sides.

The results of the studies are shown in graphic and tabular form at the end of this Section.

3.3.4.4.3.1 FO-DSC Summary

The highest calculated k_{eff} was for the Hypothetical Accident Condition, Cask Layer Removal Study (cask structural shell vanished) with 0.70 g/cc moderator interspersed between an infinite array of packages. The reactivity was 0.94015 ± 0.00148. With a 95% confidence (2σ), the maximum k_{eff} is 0.94311. The KENO5A-PC/HR-16/PN-HET calculational bias is zero.

3.3.4.4.3.2 Transfer Cask (NUHOMS®–MP187)/FF-DSC Summary

The highest calculated k_{eff} was for the accident condition (neutron shield vanished) with double-ended shear, one row of half length rods failed, 1.0 g/cc internal moderator, 0.80 g/cc external moderator in a single package with specular reflection at all boundaries. The hypothetical accident calculations were performed assuming the fuel will be in the most reactive credible condition as described above. The reactivity was 0.94598 ± 0.00185. With a 95% confidence (2σ), the maximum k_{eff} is 0.94968. The KENO5A-PC/HR-16/PN-HET calculational bias is zero.

3.3.4.5 Error Contingency Criteria

The Rancho Seco ISFSI components are designed in accordance with the "double contingency" philosophy of ANSI/ANS-57.9-1984 [3.7]. As stated before the Rancho Seco ISFSI design does not take credit for soluble boron in the water during loading. In addition no credit is taken for the burnup of the fuel assemblies. This substantially increases the margin of criticality safety.

3.3.4.6 Verification Analysis

The criticality computer code and cross section data were verified in accordance with ANSI/ANS-8.1-1983 [3.8] using a suite of critical and subcritical benchmark experiments simulating LWR fuel pins in water.

3.3.4.6.1 Benchmark Experiments and Applicability

A suite of 150 critical and subcritical LWR fuel benchmark cases was run by Transnuclear West to validate KENO5A-PC, the Hansen-Roach 16 group working cross section library, and PN-HET [3.12] (a TNW proprietary code for performing nuclear resonance/heterogeneous effects calculations). The large number of cases was choosen to evaluate parameter dependencies, such as fuel enrichment, fuel rod pitch, absorber material, absorber thickness, absorber to cluster distance, reflector material, reflector to cluster distance, and critical cluster separation.

The benchmark problems are representative of critical or subcritical arrays of commercial light water reactor (LWR) fuels with the following characteristics:

A. water moderation

B. neutron absorbers:

- no special neutron absorbers,
- neutron absorption by fixed sheets,
- neutron absorption by aqueous solutions

C. unirradiated light water reactor type fuel (no fission products or "burnup credit") near room temperature (vs. reactor operating temperature)

D. close reflection:

- no specific reflector,
- steel,

- lead, and
- depleted uranium

A statistical analysis of the largest statistical population of benchmark cases was performed to determine if the KENO5A-PC/HR-16/PN-HET methodology produces any bias due to fuel enrichment, fuel rod pitch, absorber material, absorber thickness, absorber to cluster distance, reflector material, reflector to cluster distance, critical cluster separation, or other parameters. This population consisted of 134 benchmark experiments performed on critical arrays of fuel rods, References [3.16] through [3.22]. Of the 150 cases originally run by Transnuclear West, 14 B&W critical experiments (Reference 3.6) and two subcritical experiments (Reference 3.15) were not included with the 134 cases because they contained experimental or empirical uncertainties not related to the benchmark bias.

A subset of the 134 cases was chosen to be most representative of the three DSC designs for the purpose of establishing a calculational bias. The criterion used to select the subset of cases was the neutron absorber material since that parameter most strongly influences the behavior of the system. From the set of 134 cases, those with cadmium, copper, copper/cadmium, unborated aluminum, zircalloy, Boroflex, and no neutron absorbers were discarded. There were six benchmark cases with borated absorber panels (similar to the FO- and FC-DSC absorber panels) and 13 cases with stainless steel neutron absorbing panels (thick stainless steel guide tubes are used in the FF-DSC design, thin stainless sheets are used for guide tubes in the FO- and FC-DSCs).

3.3.4.6.2 Details of Benchmark Calculations

The KENO5A-PC code and HR-16 library were used to model the critical configurations. The modeling technique incorporated a rod-by-rod representation of the fuel assemblies with explicit models of the material interspersed between assemblies. The cross section library identifiers for resonance materials were selected using PN-HET. All pertinent data for each critical configuration are documented in References [3.16] through [3.22] to permit use of these data for validating calculational methods in accordance with ANSI N16.9-1975 [3.14].

3.3.4.6.3 Results of Benchmark Calculations

Statistical analysis of the 134 critical benchmark cases showed that there are no systematic biases for fuel enrichment, fuel rod pitch, absorber material, absorber thickness, absorber to cluster distance, reflector material, reflector to cluster distance, and critical cluster separation. One dependency was noted on reflector to cluster distance for depleted uranium (DU) reflected benchmarks. The source of this bias could not be determined, but since the Rancho Seco ISFSI does not use a DU shielding for loading, transfer or storage, no corrections were made to the criticality results and the DU criticals were not used to calculate the final calculational bias for the KENO5A-PC/HR-16/PN-HET methodology. Figure 3-11 shows the results of the benchmark calculations.

Once the conclusion was drawn that the KENO5A-PC/HR-16/PN-HET methodology produces no systematic biases that would affect the criticality calculations, a subset of cases most like the Rancho Seco ISFSI were chosen as described above for the purpose of calculating the final calculational bias. The results of the nineteen most applicable benchmark critical cases are shown in bold face type in Table 3-23. The results are summarized below:

	Absorber Plates	
	Borated	Stainless
Cases	6	13
Maximum k_{eff}	1.01064	1.01405
Average k_{eff}	1.00819	1.00897
Minimum k_{eff}	1.00499	1.00254
Standard Deviation	0.00197	0.00372

The calculational bias is the maximum difference between any applicable calculated critical benchmark k_{eff} and unity, excluding any cases where the calculated k_{eff} was greater than unity. The calculated k_{eff}, without its associated uncertainty, is used for determining the bias. The group of applicable critical benchmark experiments is the nineteen cases described above.

Since all cases had a calculated k_{eff} greater than unity, the calculational bias is zero.

3.3.5 Radiological Protection

The Rancho Seco ISFSI is designed to maintain on-site and offsite doses ALARA during transfer operations and long term storage conditions. ISFSI operating procedures, shielding design, and access controls provide the necessary radiological protection to assure radiological exposures to station personnel and the public are ALARA. Further description of on-site and offsite doses resulting from ISFSI operations and the ISFSI ALARA evaluation are provided in Chapter 7.

3.3.5.1 Access Control

The Rancho Seco ISFSI is located within the owner controlled area of the Rancho Seco site. A separate secured area consisting of a double fenced, double gated, lighted area is installed around the ISFSI facility. Access is controlled by locked gates, and guards will be stationed whenever the ISFSI gates are open. Remote sensing devices are employed to detect unauthorized access to the facility.

3.3.5.2 Shielding

The Rancho Seco ISFSI is designed to satisfy the applicable dose rate limits of 10 CFR 72, 10 CFR 20, and 40 CFR 190. These limits are listed in Table 3-12. The DSC and HSM surface dose rates are bounded by those listed in the standardized NUHOMS® SAR [3.3.2]. The shielding design criteria for the casks are defined in Volume III, Section 3.3.5. An assessment of the collective operational exposure for the facility is included in Section 7.4.

3.3.5.3 Radiological Alarm Systems

There are no credible events that could result in unacceptable releases of radioactive products or increases in direct radiation levels. In addition, the potential releases postulated as the result of hypothetical accidents are negligible. Therefore, radiological alarm systems are not required.

3.3.6 Fire and Explosion Protection

The Rancho Seco ISFSI contains no permanent flammable material other than electrical and electronic components within the ISFSI Electrical Building. The other ISFSI materials of construction, concrete and steel, can withstand any credible fire hazard. Flammable materials that may be brought into the ISFSI on a temporary basis include fuel for necessary vehicles and construction materials. Use of non-flammable consumable materials will be emphasized. All wood scaffolding and cribbing will be treated with fire retardant paint. Any fuel spill within the ISFSI boundary following HSM loading will involve only diesel fuel (the contents of the fuel tanks on the tow vehicle, the crane and a few other small vehicles), which has a flash point of over 120° F. Vehicles other than electric or diesel fuel vehicles will not be permitted within the ISFSI boundary following HSM loading.

Due to the positive drainage of the ISFSI approach slabs, a spill large enough to cause puddling would also tend to drain toward the site storm drainage system and thus away from the HSMs. This drainage, coupled with the expected rapid detection of any fire by the fuel transfer personnel, will tend to limit the spread and severity of any fire. In addition, offsite fire fighting assistance is available if required. The damage caused by any fire will be negligible given the massive nature of the cask. A spill too small to cause puddling would be very difficult to ignite due to the relatively high flash point of diesel fuel and in any case such a small fire would not pose a credible threat to the ISFSI.

There is no fixed fire suppression system within the boundaries of the ISFSI; however, there is a fire detection system in the ISFSI electrical building which was installed to protect the investment in equipment but not to satisfy or imply any regulatory requirement for fire protection. During Custodial-SAFSTOR, the plant incipient fire brigade can respond to fire using portable fire suppression equipment. Offsite fire support can also be relied upon.

ISFSI initiated explosions are not considered credible since no explosive materials are present. The effects of externally initiated explosions are bounded by the design basis tornado generated missile load analysis presented in Volume II, Section 8.3.1.

3.3.7 Materials Handling and Storage

3.3.7.1 Spent Fuel Handling and Storage

The handling of intact and damaged spent fuel assemblies within the RSNGS is addressed as part of the facility license under 10 CFR 50. This includes handling DSCs and casks using the Turbine Building Gantry Crane (inside and outside of the Fuel Storage Building), and loading the DSCs with irradiated SFAs using the fuel handling bridge.

The DSC heat removal, onsite criticality control during transport, and contamination control requirements for the Rancho Seco ISFSI are as discussed in Section 3.3.7 of the Standardized NUHOMS®-24P System SAR [3.3.2].

See Appendix B for Standardized SAR, Section 3.3.7 (pages 3.3-31 to 3.3-33).

3.3.7.2 Radioactive Waste Treatment

The Rancho Seco ISFSI does not generate radioactive waste. Any secondary waste generated during cask loading and decontamination operations in the Fuel Storage Building will be disposed of in accordance with existing RSNGS radioactive waste handling procedures under the 10 CFR 50 license.

3.3.7.3 Waste Storage Facilities

Waste storage facilities are neither required nor provided for at the Rancho Seco ISFSI. The requirements for on-site waste storage are satisfied by existing RSNGS facilities for handling and storage of waste from the spent fuel pool and dry active wastes as described in Chapter 6.

3.3.8 Industrial and Chemical Safety

No hazardous chemicals or chemical reactions are involved in the operation of the Rancho Seco ISFSI. Industrial safety relating to handling of the cask and DSC are addressed by procedures which meet Occupational Safety and Health Administration (OSHA) requirements.

3.4 Classification of Structures, Components, and Systems

3.4.1 Major ISFSI Components

The classifications of the Rancho Seco ISFSI structures, systems, and components are similar to those of the Standardized NUHOMS®-24P System and are discussed in Section 3.4 of the Standardized NUHOMS®-24P SAR [3.3.2]. These classifications are summarized in Table 3-11 for convenience.

See Appendix B for Standardized SAR, Section 3.4 (pages 3.4-1 to 3.4-4).

3.4.2 Geological and Seismological Characteristics

3.4.2.1 Soil Characteristics at the ISFSI Pad

The HSM and apron slabs were analyzed in accordance with the Uniform Building Code (1991). The soil characteristics such as allowable bearing pressure and vertical subgrade modulus were used in various ways. The bearing pressure is used as a maximum value of pressure the soil is allowed to take from the structure due to vertical loads or overturning loads. A calculated bearing pressure comes from a finite element model analysis and is compared to the allowable bearing pressure. The allowable bearing pressure is given for load applications that are not wind or seismic related; however, a one third stress increase is allowed for those particular cases per the soils report. The vertical modulus of subgrade reaction is used to establish a spring constant representing the soil in the finite element model.

The slab analysis was performed using two finite element models; one for the HSM slab and one for the apron slab. The HSM slab supports only the HSMs and is not subject to transporter loads or crane loads. The design of the HSM slab considered the dead load and seismic loads associated with all HSMs being in place as well as the potential case where only a few of the modules were in place. The intent of these loading cases being to identify the maximum moments reasonably possible in the slab. The design of the apron slab took into account the movements of the transporter load over various parts of the apron slab, the cask load, and the crane load associated with movement of the cask.

Soil properties were modeled as springs and variations of properties were not considered other than those embedded in the allowable bearing pressure and vertical modulus of subgrade reaction. This is consistent with relatively simple designs using the Uniform Building Code as its design basis.

The soils report (Reference 2.8 in SAR Vol. 1) indicated that settlement could be expected. The total settlement was given at 1.5" and differential settlement was given as ½ to $^{1}/_{3}$ of the total settlement. The differential settlement, therefore, would amount to ¾" to 1" over the length or width of the slabs. The smallest dimension of a slab is the HSM slab with a width of 38 feet. A maximum differential settlement of 1" over this width is acceptable and will not adversely affect the design.

The calculated bearing pressures are compared to the allowable bearing pressures in various parts of the calculation. The following presents an overview of the bearing pressure comparisons:

Load Case	Calculated Bearing Pressure	Allowable Bearing Pressure
HSM:		
DL+LL	2.8 ksf	4.0 ksf
D+E (long.)	3.11 ksf	5.3 ksf
D+E (trans)	3.94 ksf	5.3 ksf

Bearing pressures for the apron slab were determined to be less than the HSM slab and so did not control bearing pressure evaluations.

3.4.2.2 Soil Liquefaction

Liquefaction occurs most often where groundwater is within 30 feet of the surface, but it may occur in areas where ground water is up to 50 feet beneath the surface. High pore pressures that build up in sediments during repeated seismic vibrations cause the soil to behave as a liquid. The excess pore pressures are often pushed upward through fissures and soil cracks causing a water-slurry to bubble onto the ground surface. The resulting features are called sand boils, sand blows, or "sand volcanoes." The reduction in soil volume due to densification or extrusion causes settlement, which may result in failure of structural foundations.

For liquefaction to occur, three primary conditions must occur:

1. A moderate to strong earthquake that generates strong ground shaking;
2. Shallow groundwater, within 50 feet of the ground surface;
3. Laterally extensive layers of loose, fine to medium-grained sandy soils within the saturated zone

For the Rancho Seco site, a moderate to strong earthquake that generates strong ground shaking is possible; however, the other two attributes do not exist.

The geotechnical study (Reference 2.8 in SAR Vol. 1) estimated the groundwater table to be approximately 150 feet below the surface well. This exceeds the 50 feet usually attributed to liquefaction potential.

The soil at the site varies at different levels with sand, silt, clay, gravel, and sand all present. The soils report indicates that for the clay and silt layers the consistencies are typically hard while for the sand and gravel they are usually very dense with one location being medium dense. These soils do not meet the requirement of a loose (unconsolidated) soil for liquefaction to occur.

Based on the above it is highly unlikely that liquefaction could occur at the Rancho Seco ISFSI site. This conclusion is consistent with the Updated Safety Analysis Report (USAR) which does not identify liquefaction as a hazard at this site.

3.4.2.3 Soil Amplification due to Soil-Structure Interaction

Duke Engineering Services Calculation 00079.02.0002.ST02 "SSI Effect on ISFSI Slab Acceleration," Revision 1 [3.23] discusses the issue of soil amplification due to soil-structure interaction (SSI) at the ISFSI. As shown in the calculation, the amplification of the acceleration response of the ISFSI slab and HSMs due to SSI is negligibly small when compared to the existing design margins of the slab and HSMs.

3.5 Decommissioning Considerations

Rancho Seco ISFSI decommissioning considerations are similar to those of the Standardized NUHOMS®-24P System. Refer to Section 3.5 of the Standardized NUHOMS® SAR [3.3.2] and to Section 9.6.

The DSC is licensed for offsite transportation in an MP-187 cask. SMUD intends to ship the loaded DSCs to a DOE facility when DOE is ready to take title to the fuel. Because of the minimal contamination of the outer surface of the DSC, no contamination is expected on the internal passages of the HSM. The HSMs may become slightly radioactive due to neutron activation. If necessary, the HSMs will remain at the ISFSI until they can be dismantled and disposed of using commercial demolition and disposal techniques. Alternatively, the HSMs may be refurbished and reused at another site for storage of intact DSCs.

3.6 Summary of ISFSI Design Criteria

Table 3-12 lists the major design criteria for the Rancho Seco ISFSI. Table 3-5 lists the major design criteria for the DSCs. For design requirements specific to HSM storage, see Volume II, Section 3.2.

3.7 References

3.1 Rancho Seco Nuclear Generating Station Decommissioning Safety Analysis Report, Docket No.50-312.

3.2 “Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel,” NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.

3.3 USNRC Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 1, December, 1973.

3.4 American Society of Mechanical Engineers, “ASME Boiler and Pressure Vessel Code,” Section III, Division 1, 1992 Edition with Addenda through 1993.

3.5 American National Standard for Radioactive Materials, “Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More,” ANSI N14.6-1993, June 1993.

3.6 “Safety Analysis Report for the NUHOMS®-MP187 Multi-Purpose Cask,” NUH-005, Revision 9, Docket 71-9255, Transnuclear West Inc., September 1998.

3.7 ANSI/ANS-57.9-1984, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).”

3.8 ANSI/ANS-8.1-1983, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.”

3.9 USNRC Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants.”

3.10 ANSI A58.1-1982, “Minimum Design Loads for Buildings and Other Structures.”

3.11 NUREG/CR-6007, “Stress Analysis of Closure Bolts for Shipping Casks,” Lawrence Livermore National Laboratory, April 1992.

3.12 "KENO5A-PC, Monte Carlo Criticality Program with Supergrouping,” CCC-548, Oak Ridge National Laboratory, June, 1990.

3.13 ANSI/ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors."

3.14 ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety."

3.15 Thomas, B. D., "QA Category 2 Computer Code Verification Document - KENO5A, Pacific Nuclear Version 1.2.0," Revision 2, TNW Proprietary.

3.16 Burn, Reed R., "Boral Accelerated Radiation Aging Tests," Nuclear Reactor Laboratory, University of Michigan, Ann Arbor, Michigan, May 9, 1990.

3.17 Bierman, S.R., et al., "Criticality Experiments with Subcritical Clusters of 2.35 Wt% and 4.31 Wt% ^{235}U Enriched UO_2 Rods in Water With Steel Reflecting Walls," Battelle Pacific Northwest Laboratories, NUREG/CR-1784, April, 1981.

3.18 Bierman, S.R., et al., "Critical Separation Between Subcritical Clusters of 4.29 Wt% ^{235}U Enriched UO_2 Rods in Water With Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories, NUREG/CR-0073, May 1978.

3.19 Bierman, S.R., et al., "Criticality Experiments with Subcritical Clusters of 2.35 Wt% and 4.31 Wt% ^{235}U Enriched UO_2 Rods in Water With Uranium or Lead Reflecting Walls," Battelle Pacific Northwest Laboratories, NUREG/CR-0796, August, 1981

3.20 Bierman, S.R., et al., "Critical Separation Between Subcritical Clusters of 2.35 Wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories, PNL-2438, October, 1977.

3.21 Baldwin, M.N., et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, Babcock & Wilcox Company, Lynchburg, Virginia, July, 1979.

3.22 Bierman, S.R., "Reactivity Measurements on an Experimental Assembly of 4.31 Wt% ^{235}U Enriched UO_2 Fuel Rods Arranged in a Shipping Cask Geometry," Battelle Pacific Northwest Laboratories, PNL-6838, October, 1989.

3.23 Duke Engineering Services Calculation 00079.02.0002.ST02 “SSI Effect on ISFSI Slab Acceleration,” Revision 1

3.24 Enclosure 4 to Letter from Dan Tallman to NRC Document Control Desk, “RESPONSE TO REQUEST FOR CLARIFICATION OF RESPONSE TO ADDITIONAL INFORMATION FOR THE TECHNICAL REVIEW OF THE APPLICATION FOR RENEWAL OF THE RANCHO SECO INDEPENDENT SPENT FUEL STORAGE INSTALLATION LICENSE NO. SNM-2510 (CAC/EPID NOS. 001028/L-2018- RNW-0005; 000993/L-2018-LNE-0004),” dated July 12, 2019 (ADAMS ML19204A239)

Table 3-1

Rancho Seco Fuel Characteristics

Parameter	Value
Fuel Design	B&W 15X15 Mark B
Rods per Assembly	208
Control Rod Guide Tubes per Assembly	16
In-Core Instrument Position	1
Assembly Cross Section	8.536 in.
Fuel Rod Outside Diameter	0.430 in.
Cladding Thickness	0.0265 in.
Fuel Rod Pitch	0.568 in.
Active Fuel Length (nominal)	141.8 in.
Assembly Length (600°F, 40 GWd/MTU)	166.893 in.
Total Assembly Only Weight	1530 lb
Non-Fuel Component Weight	135 lb (max)
Maximum Enrichment	3.43%
Maximum Burnup	38,268 MWd/MTU
Cladding Material	Zircaloy-4

Table 3-2

Rancho Seco Control Element Characteristics

Parameter	Value
Axial Power Shaping Rod Assemblies	
Number of Rods per Assembly	16
Outside Diameter	0.440 in.
Cladding Thickness	0.024 in.
Cladding Material	Type 302 SS
Plug Material	Type 304 SS
Poison Material (gray absorber)	Inconel
Spider Material	SS, Grade CF3M
Female Coupling Material	Type 304 SS
Length of Poison Section	63 in.
Burnable Poison Rod Assemblies	
Number of Rods per Assembly	Up to 16
Outside Diameter	0.430 in.
Cladding Thickness	0.035 in.
Cladding Material	Zircaloy-4
End Plug Material	Zircaloy-4
Poison Material	B_4C in Al_2O_3
Length of Poison Section	126 in.
Spider Material	SS, Grade CF3M
Coupling Mechanism Material	Type 304 SS
Orifice Rod Assembly	
Number of Rods per Assembly	16
Outside Diameter	0.480 in.
Orifice Rod Material	Type 304 SS
Spider Material	SS, Grade CF3M
Coupling Mechanism Material	Type 304 SS and 17-4 pH H 1000

Table 3-3

DSC Loading Summary for Cask Handling Conditions

Design Load Type	Section	Design Parameter
Flood	3.2.2	Maximum water height: 50 ft.
Seismic	3.2.3	Peak Ground Accelerations: Horizontal: 0.25g (both directions) Vertical: 0.17g
Dead Loads	8.1.1.1	Weight of loaded DSC
Normal and Off-Normal Pressure	8.1.1.2	Maximum Internal Pressure: Normal Conditions: 10 psig Off-Normal Conditions: 10 psig
Test Pressure	8.1.1.2	Enveloping internal pressure of 20.0 psig[(1)] applied w/o DSC outer top cover plate and w/ strongback.
Normal and Off-Normal Operating Temperature	8.1.1.3	DSC with spent fuel rejecting 13.5 kW (FC and FO) or 9.93 kW (FF) of decay heat. Ambient air temperature range of 0°F to 101°F (normal) and -20° to 117°F (off-normal).
Normal Handling Loads	8.1.1.4	Deadweight ± 1.0g in vertical direction Deadweight ± 1.0g in radial direction Deadweight ± 1.0g in axial direction Deadweight ± 0.5g simultaneously in vertical, radial and axial directions Hydraulic ram load of 60,000 lb.
Off-Normal Handling Loads	8.1.1.5	Hydraulic ram load of 80,000 lb.
Accidental Cask Drop Loads[(2)]	8.2.1	Equivalent static decelerations: Vertical end drop: 75g Horizontal side drop: 75g Oblique corner drop: 25g
Accident Internal Pressure	8.2.3	Enveloping internal pressure of 50 psig based on 100% fuel cladding rupture and fill gas release, 30% fission gas release, ambient air temperature of 117°F, and blocked HSM vents.

Notes:

1. Envelops the following pressures:
    a) 11 +3/-0 psig test pressure, applied to the shell only, during fabrication.
    b) 20 psig blowdown pressure to evacuate water after fuel loading, prior to installation of the outer top cover plate.
2. These decelerations bound the Rancho Seco DSAR evaluations.

Table 3-4

Cask Loading Summary for Cask Handling Conditions

Design Load Type	Section	Design Parameter
Tornado Wind	3.2.1	Maximum wind pressure: 397 psf
Tornado Missile	3.2.1	Automobile: Weight = 3967 lbs. Area = 20 ft^2 Velocity = 126 mph Penetration Resistant Missile Weight = 276 lbs. Diameter = 8.0 in. Velocity = 126 mph Barrier Impingement Missile (solid steel sphere) Diameter = 1.0 in. Velocity = 126 mph
Flood	3.2.2	Maximum water height: 50 feet Maximum water velocity: 15 fps
Seismic	3.2.3	Peak Ground Accelerations: Horizontal: 0.25g (both directions) Vertical: 0.17g
Snow and Ice	3.2.4	Maximum Load: 110 psf (included in live loads)
Dead Weight	8.1.1.1	Dead weight including loaded DSC
Normal and Off-normal Operating Temperatures	8.1.1.3	DSC with spent fuel rejecting 13.5 kW (FC and FO) or 9.93 kW (FF) of decay heat. Ambient air temperature range of 0°F to 101°F (normal) and -20°F to 117°F (off-normal).
Normal Handling Loads	8.1.1.4	Critical lift conditions Deadweight ± 1.0g in vertical direction Deadweight ± 1.0g in radial direction Deadweight ± 1.0g in axial direction Deadweight ± 0.5g simultaneously in vertical, radial and axial directions Hydraulic ram load of 60,000 lb.
Off-normal Handling Loads	8.1.1.5	Hydraulic ram load of 80,000 lbs.
Accidental Cask Drop Loads	8.2.1	Equivalent static decelerations: Vertical end drop: 75g Horizontal side drop: 75g Oblique corner drop: 25g
Fire and Explosion	3.3.6	Enveloped by other design basis events
Internal Pressure	8.1.1.2	Maximum internal pressure: Normal conditions: 10 psig Off-normal conditions: 10 psig Accident condition: 50 psig (Level D)

Table 3-5

Summary of Dry Shielded Canister Capacity and Internal Pressure Design Criteria

Capacity	FO-DSCs: 24 Rancho Seco SFAs FC-DSCs: 24 Rancho Seco SFAs w/control components FF-DSCs: 13 Rancho Seco SFAs
DSC maximum internal pressure (accident condition)	50 psig (100% fill gas release and 30% fission gas release from all fuel rods in the DSC) and blocked vents
DSC minimum internal pressure (vacuum drying)	≤ 3 Torr for not less than 30 minutes
DSC helium backfill pressure and leakage rate	0 - 2.5 psig ≤ 1×10^{-5} std-cc/sec
DSC inner cover plate design pressure	10 psig while leak testing

Table 3-6

DSC Load Combinations and Service Levels for Cask Transfer and Handling Modes

LOADS	Test	Normal				Off-Normal			Accident									
DEAD WEIGHT																		
Empty DSC	X	X																
Vertical Full DSC			X		X	X		X						X	X	X		
Horizontal Full DSC				X		(or) X	X		X	X	X	X	X		(or) X	(or) X	X	X
THERMAL																		
0 to 101	X	X	X					X	X	X			X	X		X	X	
-20 to 117						X	X				X	X			X			X
PRESSURE																		
Normal Internal			X						X	X								
Off-Normal Internal					X	X	X	X			X					X		X
Accident Internal												X	X		X		X	
External					X													
Hydrostatic					X			X						X				
Test	X																	
HANDLING				X														
Normal																		
Off-Normal							X											
Accident											X		X				X	X
CASK DROP																X		
SEISMIC									X					X				
FLOODING										X								
ASME Code Service Level		A				B			C						D			
Load Combination No.	T	A1	A2	A3	A4	B1	B2	B3	C1	C2	C3	C4	C5	C6	D1	D2	D3	D4

Note:

1. The stress limits of NB-3226 apply.

Table 3-7

Structural Design Criteria for DSC

Item	Stress Type	Stress Values[1] Service Levels A & B	Service Level C	Service Level D
DSC[2]	Primary Membrane	S_m	Greater of 1.2 S_m or S_y	Smaller of 2.4 S_m or 0.7 S_u
	Primary Membrane + Bending	1.5 S_m	Greater of 1.8 S_m or 1.5 S_y	Smaller of 3.6 S_m or S_u
	Primary + Secondary	3.0 S_m	N/A	N/A
DSC[2] Internals	Primary Membrane	S_m	1.5S_m	Smaller of 2.4 S_m or 0.7 S_u
	Primary Membrane + Bending	1.5 S_m	2.25S_m	Smaller of 3.6 S_m or S_u
	Primary + Secondary	3.0 S_m	N/A	N/A
	Average Bearing	S_y	1.5S_y	N/A
	Average Shear	0.6 S_m	0.9S_m	Smaller of 0.42 S_u or 1.2 S_m
	Buckling	Equivalent static loads shall not exceed 2/3 of the limit analysis collapse load		
DSC Partial Penetration Welds[3]	Primary	0.50 S_m E(S_m)	Greater of E(1.2S_m) or E(S_y)	Smaller of E(2.4S_m) or E(0.7S_u)
	Primary Membrane + Bending	E(1.5S_m)	Greater of E(1.8S_m) or E(1.5S_y)	N/A
	Pure Shear	E(0.6 S_m)	Greater of E(0.6 S_y) or E(1.2 x 0.6 S_m)	Smaller of E(2.4 x 0.6 S_m) or E(0.7 x 0.6 S_u)

Notes:

1. Values of S_y, S_m, and S_u versus temperature are given in Table 8.1-3 of the Standardized NUHOMS® SAR [3.3.2], and are identical to those in [3.3.4]. For materials not listed, refer to [3.3 4].

   See Appendix B for Standardized SAR, Table 8.1-3 (pages 8.1-53 to 8.1-57).

2. Includes full penetration volumetrically inspected welds.
3. The joint efficiency factor, E, is 0.8 for the shell assembly end plate inspected welds or is per ASME Table NG-3352-1 (quality factor, n) for basket assembly welds
4. For austenitic base metal, the allowable load for Level D elastic/plastic analysis is 0.6 x [greater of 0.7 S_u or S_y + 1/3(S_u-S_y)].

Table 3-8

Cask Load Combinations and Service Levels for Cask Handling Modes

Load Case	Normal Conditions						Off-Normal Conditions		Accident Conditions				
Dead Load/Live Load	X	X	X	X	X	X	X	X	X	X	X	X	X
Thermal w/DSC													
0° to 101°F Ambient	X	X	X	X	X	X	X		X	X	X	X	X
-20° to 117°F Ambient								X					
Internal Pressure													
Normal/Off-Normal	X	X	X	X	X	X	X	X				X	
Accident									X	X	X		
Handling (Critical Lifts)													
Vertical	X												
Downending		X											
Horizontal			X										
45° Tilt				X									
Handling (Non-Critical)													
Transport					X				X				
Normal Transfer						X				X			
Off-normal Transfer							X	X					
Seismic									X	X			
Tornado Wind/Tornado Missile											X		
Flooding												X	
Drop (end, side or corner)													X
ASME Code Service Level	A	A	A	A	A	A	B	B	C	C	C	C	D
Load Combination No.	A1	A2	A3	A4	A5	A6	B1	B2	C1	C2	C3	C4[1]	D1

Note:

1. This combination is hypothetical for a postulated cask storage mode only. However, this may bound other load combinations for certain cask components and has been used.

Table 3-9

Structural Design Criteria for On-Site Transfer Cask

Item	Stress Type	Stress Values[1]		
		Service Levels A & B	Service Level C	Service Level D
Cask Structural Shell	Primary Membrane	S_m	Greater of 1.2 S_m or S_y	Smaller of 2.4 S_m or 0.7 S_u
	Primary Membrane + Bending	1.5 S_m	Greater of 1.8 S_m or 1.5 S_y	Smaller of 3.6 S_m or S_u
	Primary + Secondary	3.0 S_m	N/A	N/A
Trunnions[2]	Tensile / Bending	Smaller of S_y/6 or S_u/10 [3]	N/A	N/A
	Shear	Smaller of 0.6 S_y/6 or 0.6 S_u/10	N/A	N/A
Cask Components	Bearing Stress	S_y	S_y	N/A [4]
	Pure Shear Stress	0.6 S_m	0.6 S_m	0.42 S_u

Notes:

1. Values of S_y, S_m, and S_u versus temperature are given in Table 8.1-3 of the Standardized NUHOMS® SAR [3.3.2], and are identical to those in [3.3.4]. Refer to [3.3.4] for materials not listed.

   See Appendix B for Standardized SAR, Table 8.1-3 (pages 8.1-53 to 8.1-57).

2. These allowables apply to the upper lifting trunnions for critical lifts governed by ANSI N14.6 [3.3.5]. The lower support trunnions and the upper lifting trunnions for all remaining loads are governed by the same ASME Code criteria applied to the cask structural shell for Service Levels A and B.
3. Stress factors for other states of stress are to be established and justified by the designer per ANSI N14.6 [3.3.5].
4. The allowable bearing stress on seal surfaces is S_y.

Table 3-10

Structural Design Criteria for Bolts

Service Levels A, B, and C	
Average Tension Stress	2/3 S_y
Average Shear Stress	0.6 (2/3 S_y)
Tension Plus Shear	$R_t^2 + R_s^2 \leq 1$ where R_t = stress ratio for average tensile stress R_s = stress ratio for average shear stress
Tension Plus Shear Plus Bending Stress Intensity	1.35 (2/3 S_y)
Service Level D	
Average Tension Stress	Smaller of 0.7 S_u and S_y
Average Shear Stress	Smaller of 0.42 S_u and 0.6 S_y
Tension Plus Shear	$R_t^2 + R_s^2 \leq 1$

Table 3-11

Rancho Seco ISFSI Major Components and Safety Classifications

Component	TNW Classification	Rancho Seco QA Classification
Dry Shielded Canister (DSC)	Important to Safety(1)	Class 1
Horizontal Storage Module (HSM)	Important to Safety(1)	Class 2(2)
ISFSI Basemat and Approach Slabs	Not Important to Safety	Class 2
Transfer Equipment		
Cask	Important to Safety(1)	Class 1
Cask Lifting Yoke	Important to Safety(1)	Class 1(3)
Lifting Yoke Extensions	Important to Safety(1)	Class 1(3)
Transport Trailer/Skid	Not Important to Safety	Class 2
Ram Assembly	Not Important to Safety	Class 2
Lubricant	Not Important to Safety	Class 2
Auxiliary Equipment		
Vacuum Drying System	Not Important to Safety	Class 2
Automatic Welding System	Not Important to Safety	Class 2
HSM Temperature Monitoring	Not Important to Safety	Class 2

Notes:
1. Graded Quality (per Standardized NUHOMS® SAR [3.3.2]
2. Part 71/72 Important to Safety
3. For use under the 10 CFR 50 license during fuel loading operations only.

Table 3-12

Summary of ISFSI Design Criteria

Maximum load capacity of Gantry Crane	130 tons (cantilever section)
Peak fuel cladding temperature	379°C Long Term 570°C Short Term
Criticality factor	$k_{eff} \leq 0.95$ considering optimum moderation and all applicable biases and uncertainties to a 95/95 confidence level
Maximum dose rates	≤ 2 mrem/hr dose rate at ISFSI fence ≤ 25 mrem/yr dose to real individual at or beyond the site boundary
Maximum accident exposure	<5 rem (whole body) and <50 rem (skin) at controlled area boundary
Ambient temperature(1)	-20°F to 117°F
Ambient humidity	0 to 100%
Tornado wind velocities (rotational and translational)	In accordance with RG 1.76 and NUREG 0800
Tornado pressure drop	2.0 psi for 1.5 seconds
Maximum winds	360 mph
Design earthquake peak acceleration	0.25g vertical and 0.17g horizontal with response spectra and damping values per RG 1.60 and 1.61
Explosion peak overpressure	1 psi
Flood	The Rancho Seco ISFSI is not subject to floods.
Storage structures	22 HSMs in a 2x11 array and 2 Casks

Note:

1. The design basis minimum temperature for Rancho Seco is 19 ^{o}F. All analysis has been done to –20 ^{o}F to be consistent with the standardized NUHOMS design.

Table 3-13
Maximum Fuel Loading Parameters

Parameter	Value
Number of Assemblies, FO/FC-DSCs	24
Number of Assemblies, FF-DSC	13
Enrichment, w/o U235	≤ 3.43%
Minimum Burnup	0
Design Basis Fuel	B&W 15x15
Maximum Number of Failed Rods (FF-DSC only)	15/assy

Table 3-14

Design Basis Fuel Parameters for Criticality Analysis

Parameter	Value
Fuel Pellet Outside Diameter	0.3686 in
Fuel Clad Thickness	0.0265 in
Fuel Clad Outside Diameter	0.43 in
Fuel Rod Pitch	0.568 in
Active Fuel Height	141.8 in
Enrichment, w/o U-235	3.43%
UO2 Density, %Theoretical Dens.	95.0%
Rod Array (NxN Rods)	15
Fueled Rod Locations	208

Table 3-15

KENO Model Atom Densities

Fuel Pellet			**Aluminum**		
Element	H-R ID No.	atom/b-cm	Element	H-R ID No.	atom/b-cm
Oxygen	8100	4.6540E-02	Aluminum	13100	6.0552E-02
U235	(run-unique)	8.0802E-04			
U238	(run-unique)	2.2462E-02			
Zircaloy-4			**Absorber Plate**		
Element	H-R ID No.	atom/b-cm	Element	H-R ID No.	atom/b-cm
Chromium	24100	7.5166E-05	Aluminum	13100	3.9268E-02
Iron	26100	1.4696E-04	Boron	5100	2.4879E-02
Nickel	28100	2.3299E-06	Carbon	6100	7.6705E-03
Zirconium	40100	4.2711E-02			
C-Steel			**Lead**		
Element	H-R ID No.	atom/b-cm	Element	H-R ID No.	atom/b-cm
Iron	26100	8.3801E-02	Lead	82100	3.2960E-02
Manganese	25100	8.6048E-04			
S_Steel			**NS-3**		
Element	H-R ID No.	atom/b-cm	Element	H-R ID No.	atom/b-cm
Chromium	24100	1.7274E-02	Aluminum	13100	7.0275E-03
Iron	26100	5.9042E-02	Calcium	20100	1.4835E-03
Manganese	25100	1.7210E-03	Carbon	6100	8.2505E-03
Nickel	28100	7.4481E-03	Hydrogen	1101	5.0996E-02
			Iron	26100	1.0628E-04
			Oxygen	8100	3.7793E-02
			Silicon	14100	1.2680E-03

Table 3-16
KENO Model Moderator Atom Densities

Density (g/cc)	Scaling Factor	Hydrogen (at/b-cm)	Oxygen (at/b-cm)
1.00000	1.00177	6.68544e-02	3.34272e-02
0.99823	1.00000	6.67361e-02	3.33680e-02
0.90000	0.90160	6.01690e-02	3.00845e-02
0.80000	0.80142	5.34835e-02	2.67418e-02
0.70000	0.70124	4.67981e-02	2.33990e-02
0.60000	0.60106	4.01126e-02	2.00563e-02
050000	0.50089	3.34272e-02	1.67136e-02
0.40000	0.40071	2.67418e-02	1.33709e-02
0.30000	0.30053	2.00563e-02	1.00282e-02
0.20000	0.20035	1.33709e-02	6.68544e-03
0.10000	0.10018	6.68544e-03	3.34272e-03
0.0500	0.05009	3.34272e-03	1.67136e-03
0.00000	0.00000	0.00000e-00	0.00000e-00

Table 3-17

Summary of KENO Parametric Studies

Study	DSC Moderator Density	Ex-Cask Moderator Density	Neutron Shield	Radial Boundary Condition
FONIF	Varies	1.0	Intact	Specular
FONXF	1.0	Varies	Intact	Specular
FOHIF	Varies	1.0	None	Specular
FOHXF	1.0	Varies	None	Specular
GSDEF*	1.0	0.70	None	Specular
FOCL	1.0	0.70	Varies	Specular
IFNCI	Varies	1.0	Intact	Water
IFNCX	1.0	Varies	Intact	Specular
FFDSI	Varies	1.0	Intact	Water
FFDSX	1.0	Varies	Intact	Specular

* This parametric study only applies to the FO Can.

Table 3-18

MP187/FO-DSC KENO Results (Guide Sleeve Deformation)

Model	k_{eff}	+/-	1 s	$k_{eff} + 2\sigma$
gsdef00.ko	0.93271	+/-	0.00152	0.93575
gsdef025.ko	0.93097	+/-	0.00149	0.93395
gsdef05.ko	0.93197	+/-	0.00143	0.93483
gsdef08.ko	0.93093	+/-	0.00157	0.93407
gsdef10.ko	0.93337	+/-	0.00154	0.93645
gsdef11.ko	0.93221	+/-	0.00157	0.93535
gsdef12.ko	0.93610	+/-	0.00152	0.93914
gsdef14.ko	0.93672	+/-	0.00142	0.93956
gsdef16.ko	0.93723	+/-	0.00162	0.94047
gsdef18.ko	0.93966	+/-	0.00157	0.94280

Table 3-19

MP187/FO-DSC KENO Results (Cask Layer Removal)

Model	k_{eff}	+/-	1 s	$k_{eff} + 2\sigma$
FOCLA.KO	0.93818	+/-	0.00158	0.94134
FOCLB.KO	0.93758	+/-	0.00155	0.94068
FOCLC.KO	0.93966	+/-	0.00157	0.94280
FOCLD.KO	0.94015	+/-	0.00148	0.94311
FOCLE.KO	0.93513	+/-	0.00157	0.93827
FOCLF.KO	0.93859	+/-	0.00155	0.94169

Table 3-20

MP187/FO-DSC KENO Results

Model	keff	+/-	1 σ	Model	keff	+/-	1 σ
FONIF00.KO	0.33984	+/-	0.00095	FOHIF00.KO	0.34009	+/-	0.00100
FONIF05.KO	0.37054	+/-	0.00084	FOHIF05.KO	0.37023	+/-	0.00085
FONIF10.KO	0.53723	+/-	0.00110	FOHIF10.KO	0.53972	+/-	0.00112
FONIF20.KO	0.60799	+/-	0.00110	FOHIF20.KO	0.61329	+/-	0.00120
FONIF30.KO	0.66953	+/-	0.00126	FOHIF30.KO	0.67393	+/-	0.00123
FONIF40.KO	0.71959	+/-	0.00136	FOHIF40.KO	0.72584	+/-	0.00140
FONIF50.KO	0.76679	+/-	0.00141	FOHIF50.KO	0.76799	+/-	0.00134
FONIF60.KO	0.80562	+/-	0.00145	FOHIF60.KO	0.81102	+/-	0.00146
FONIF70.KO	0.84433	+/-	0.00149	FOHIF70.KO	0.84750	+/-	0.00156
FONIF80.KO	0.87400	+/-	0.00156	FOHIF80.KO	0.88223	+/-	0.00144
FONIF90.KO	0.90667	+/-	0.00167	FOHIF90.KO	0.91096	+/-	0.00158
FONIF100.KO	0.93159	+/-	0.00158	FOHIF100.KO	0.93389	+/-	0.00151

Model	keff	+/-	1 σ	Model	keff	+/-	1 σ
FONXF00.KO	0.93436	+/-	0.00152	FOHXF00.KO	0.93548	+/-	0.00156
FONXF05.KO	0.93290	+/-	0.00158	FOHXF05.KO	0.93524	+/-	0.00153
FONXF10.KO	0.93030	+/-	0.00151	FOHXF10.KO	0.93899	+/-	0.00153
FONXF20.KO	0.93116	+/-	0.00149	FOHXF20.KO	0.93712	+/-	0.00158
FONXF30.KO	0.93397	+/-	0.00147	FOHXF30.KO	0.93950	+/-	0.00156
FONXF40.KO	0.93330	+/-	0.00159	FOHXF40.KO	0.93607	+/-	0.00152
FONXF50.KO	0.93269	+/-	0.00148	FOHXF50.KO	0.93604	+/-	0.00152
FONXF60.KO	0.93100	+/-	0.00149	FOHXF60.KO	0.93885	+/-	0.00147
FONXF70.KO	0.93331	+/-	0.00153	FOHXF70.KO	0.93966	+/-	0.00157
FONXF80.KO	0.93052	+/-	0.00161	FOHXF80.KO	0.93600	+/-	0.00156
FONXF90.KO	0.93074	+/-	0.00153	FOHXF90.KO	0.93734	+/-	0.00158
FONXF100.KO	0.93159	+/-	0.00158	FOHXF100.KO	0.93389	+/-	0.00151

Table 3-21

MP187/FF-DSC KENO Results

Model	keff	+/-	1 σ	Model	keff	+/-	1 σ
IFNCI000.KO	0.26349	+/-	0.00070	FFDSI000.KO	0.25584	+/-	0.00081
IFNCI005.KO	0.37248	+/-	0.00098	FFDSI005.KO	0.36074	+/-	0.00098
IFNCI010.KO	0.60350	+/-	0.00143	FFDSI010.KO	0.59256	+/-	0.00143
IFNCI020.KO	0.67650	+/-	0.00158	FFDSI020.KO	0.67131	+/-	0.00159
IFNCI030.KO	0.72586	+/-	0.00173	FFDSI030.KO	0.72868	+/-	0.00172
IFNCI040.KO	0.76875	+/-	0.00186	FFDSI040.KO	0.77097	+/-	0.00183
IFNCI050.KO	0.80214	+/-	0.00186	FFDSI050.KO	0.80398	+/-	0.00176
IFNCI060.KO	0.83485	+/-	0.00184	FFDSI060.KO	0.83614	+/-	0.00186
IFNCI070.KO	0.86550	+/-	0.00195	FFDSI070.KO	0.87030	+/-	0.00184
IFNCI080.KO	0.88686	+/-	0.00184	FFDSI080.KO	0.89273	+/-	0.00185
IFNCI090.KO	0.90846	+/-	0.00184	FFDSI090.KO	0.91861	+/-	0.00184
IFNCI100.KO	0.93493	+/-	0.00205	FFDSI100.KO	0.94382	+/-	0.00192

Model	keff	+/-	1 σ	Model	keff	+/-	1 σ
IFNCX000.KO	0.93065	+/-	0.00195	FFDSX000.KO	0.94015	+/-	0.00182
IFNCX005.KO	0.93402	+/-	0.00186	FFDSX005.KO	0.94498	+/-	0.00183
IFNCX010.KO	0.93828	+/-	0.00198	FFDSX010.KO	0.94164	+/-	0.00188
IFNCX020.KO	0.93636	+/-	0.00184	FFDSX020.KO	0.94164	+/-	0.00194
IFNCX030.KO	0.94004	+/-	0.00186	FFDSX030.KO	0.93913	+/-	0.00196
IFNCX040.KO	0.93472	+/-	0.00186	FFDSX040.KO	0.94024	+/-	0.00190
IFNCX050.KO	0.93430	+/-	0.00194	FFDSX050.KO	0.94159	+/-	0.00198
IFNCX060.KO	0.93829	+/-	0.00189	FFDSX060.KO	0.94471	+/-	0.00193
IFNCX070.KO	0.93336	+/-	0.00189	FFDSX070.KO	0.94150	+/-	0.00189
IFNCX080.KO	0.93152	+/-	0.00186	FFDSX080.KO	0.94598	+/-	0.00185
IFNCX090.KO	0.92930	+/-	0.00187	FFDSX090.KO	0.94417	+/-	0.00189
IFNCX100.KO	0.93405	+/-	0.00192	FFDSX100.KO	0.94015	+/-	0.00189

Single-ended Shear

Model	keff	+/-	1 σ	Model	keff	+/-	1 σ
FFSS010.KO	0.92971	+/-	0.00209	FFSS020.KO	0.93198	+/-	0.00195
FFSS030.KO	0.93866	+/-	0.00190	FFSS040.KO	0.93735	+/-	0.00194
FFSS048.KO	0.93920	+/-	0.00183				

Table 3-22

MP187/FF-DSC KENO Results (Cask Layer Removal)

File	Cask Layer	K_{eff}	+/-	1σ	$K_{eff} + 2\sigma$
FFCL80A.KO	Nominal	0.94248	+/-	0.00194	0.94636
FFCL80B.KO	N Shield Panel	0.94008	+/-	0.00180	0.94368
FFCL80C.KO	N Shield	0.94598	+/-	0.00185	0.94968
FFCL80D.KO	Cask Structural Shell	0.94336	+/-	0.00184	0.94704
FFCL80E.KO	Gamma Shield	0.93934	+/-	0.00189	0.94312
FFCL80F.KO	Cask Inner Shell	0.94492	+/-	0.00191	0.94874

Table 3-23

Benchmark Calculation Results

TNW Ref. #	Enrich-ment (w%)	Rod Pitch (mm)	Absorber Material	Absorber Thickness (mm)	Abs. to Cluster Distance (mm)	Reflector Material	Refl. to Cluster Distance (mm)	Critical Cluster Sep. (mm)	$k_{eff} \pm 1\sigma$
1	4.31%	25.40	N/A	N/A	N/A	Moderator	N/A	117.2	1.00462±0.00269
2	4.31%	25.40	N/A	N/A	N/A	Moderator	N/A	116.8	1.01742±0.00269
3	***4.31%***	***25.40***	***SS304L***	***4.85***	***2.45***	***Moderator***	***N/A***	***85.8***	***1.00582±0.00262***
4	***4.31%***	***25.40***	***SS304L***	***4.85***	***32.77***	***Moderator***	***N/A***	***96.5***	***1.00710±0.00275***
5	***4.31%***	***25.40***	***SS304L***	***3.02***	***4.28***	***Moderator***	***N/A***	***92.2***	***1.01261±0.00286***
6	***4.31%***	***25.40***	***SS304L***	***3.02***	***32.77***	***Moderator***	***N/A***	***97.6***	***1.01379±0.00235***
7	4.31%	25.40	SS304L1.1%	2.98	4.32	Moderator	N/A	61.0	1.01631±0.00259
8	4.31%	25.40	SS304L1.1%	2.98	32.77	Moderator	N/A	80.8	1.00913±0.00286
9	4.31%	25.40	SS304L1.6%	2.98	4.32	Moderator	N/A	57.6	1.01346±0.00284
10	4.31%	25.40	SS304L1.6%	2.98	32.77	Moderator	N/A	79.0	1.00796±0.00288
11	***4.31%***	***25.40***	***BoralA***	***7.13***	***32.77***	***Moderator***	***N/A***	***67.2***	***1.00822±0.00278***
12	4.31%	25.40	Copper	6.46	0.84	Moderator	N/A	81.5	1.00959±0.00273
13	4.31%	25.40	Copper	6.46	32.77	Moderator	N/A	94.2	1.00687±0.00282
14	4.31%	25.40	Copper	3.37	0.00	Moderator	N/A	84.8	1.00574±0.00255
15	4.31%	25.40	Copper	3.37	42.41	Moderator	N/A	96.4	1.00715±0.00252
16	4.31%	25.40	Cu/Cd	3.57	0.00	Moderator	N/A	66.6	1.01169±0.00283
17	4.31%	25.40	Cu/Cd	3.57	42.41	Moderator	N/A	83.5	1.01121±0.0028
18	4.31%	25.40	Cadmium	0.291	7.01	Moderator	N/A	59.3	1.01198±0.00258
19	4.31%	25.40	Cadmium	0.291	32.77	Moderator	N/A	74.2	1.00945±0.00284
20	4.31%	25.40	Cadmium	0.610	6.69	Moderator	N/A	59.6	1.01339±0.0028
21	4.31%	25.40	Cadmium	0.610	32.77	Moderator	N/A	74.2	1.01292±0.00269
22	4.31%	25.40	Cadmium	0.901	6.40	Moderator	N/A	58.7	1.01386±0.00277
23	4.31%	25.40	Cadmium	0.901	32.77	Moderator	N/A	73.8	1.01380±0.00271
24	4.31%	25.40	Cadmium	2.006	5.29	Moderator	N/A	56.8	1.01429±0.00248
25	4.31%	25.40	Cadmium	2.006	32.77	Moderator	N/A	72.8	1.00814±0.00264
26	4.31%	25.40	Aluminum	6.25	1.05	Moderator	N/A	107.2	1.00970±0.00279
27	4.31%	25.40	Aluminum	6.25	32.77	Moderator	N/A	107.7	1.01194±0.00278
28	4.31%	25.40	Zircaloy-4	6.52	0.78	Moderator	N/A	109.2	1.01439±0.00263
29	4.31%	25.40	Zircaloy-4	6.52	32.77	Moderator	N/A	108.6	1.00760±0.00269
30	2.35%	20.32	N/A	N/A	N/A	Steel	0	98.9	1.01780±0.00279
31	2.35%	20.32	N/A	N/A	N/A	Steel	6.6	104.4	1.00790±0.00263

Note: ***Bold face*** benchmarks are most applicable to the Rancho Seco ISFSI

Table 3-23

Benchmark Calculation Results (Continued)

TNW Ref. #	Enrich-ment (w%)	Rod Pitch (mm)	Absorber Material	Absorber Thickness (mm)	Abs. to Cluster Distance (mm)	Reflector Material	Refl. to Cluster Distance (mm)	Critical Cluster Sep. (mm)	$k_{eff} \pm 1\sigma$
32	2.35%	20.32	N/A	N/A	N/A	Steel	13.21	104.4	1.01002±0.00263
33	2.35%	20.32	N/A	N/A	N/A	Steel	26.16	96.0	1.00366±0.00281
34	2.35%	20.32	N/A	N/A	N/A	Steel	39.12	87.5	1.00960±0.00269
35	2.35%	20.32	N/A	N/A	N/A	Moderator	N/A	74.7	1.01039±0.00264
36	2.35%	16.84	N/A	N/A	N/A	Steel	0	85.7	1.01221±0.00285
37	2.35%	16.84	N/A	N/A	N/A	Steel	6.6	91.7	1.00653±0.00263
38	2.35%	16.84	N/A	N/A	N/A	Steel	13.21	91.0	1.00398±0.00304
39	2.35%	16.84	N/A	N/A	N/A	Steel	16.84	92.5	1.00716±0.00285
40	2.35%	16.84	N/A	N/A	N/A	Steel	23.44	88.7	1.00694±0.00262
41	2.35%	16.84	N/A	N/A	N/A	Steel	30.05	86.5	0.99909±0.00255
42	2.35%	16.84	N/A	N/A	N/A	Steel	39.12	81.3	1.00414±0.0025
43	2.35%	16.84	N/A	N/A	N/A	Steel	67.26	72.6	1.00344±0.00245
44	2.35%	16.84	N/A	N/A	N/A	Moderator	N/A	68.3	1.00699±0.00242
45	4.31%	25.40	N/A	N/A	N/A	Steel	0	117.6	1.01096±0.00309
46	4.31%	25.40	N/A	N/A	N/A	Steel	6.6	131.2	1.00367±0.00326
47	4.31%	25.40	N/A	N/A	N/A	Steel	13.21	129.9	1.01098±0.00268
48	4.31%	25.40	N/A	N/A	N/A	Steel	26.16	113.1	1.00763±0.00316
49	4.31%	25.40	N/A	N/A	N/A	Steel	54.05	86.7	1.00920±0.00309
50	4.31%	25.40	N/A	N/A	N/A	Moderator	N/A	71.0	1.01488±0.00262
51	4.31%	18.92	N/A	N/A	N/A	Steel	0	143.9	1.00208±0.003
52	4.31%	18.92	N/A	N/A	N/A	Steel	6.6	152.6	1.00385±0.00301
53	4.31%	18.92	N/A	N/A	N/A	Steel	13.21	153.9	1.00582±0.00322
54	4.31%	18.92	N/A	N/A	N/A	Steel	19.56	153.6	1.00894±0.00323
55	4.31%	18.92	N/A	N/A	N/A	Steel	26.16	149.7	1.00040±0.00298
56	4.31%	18.92	N/A	N/A	N/A	Steel	54.05	133.4	1.00556±0.00323
57	4.31%	18.92	N/A	N/A	N/A	Moderator	N/A	124.8	1.01910±0.00261
58	4.31%	18.92	N/A	N/A	N/A	Moderator	N/A	125.0	1.01506±0.00275
59	2.35%	16.84	N/A	N/A	N/A	Steel	13.21	91.0	1.02322±0.00258
60	***2.35%***	***16.84***	***SS304L***	***3.02***	***N/A***	***Steel***	***13.21***	***78.7***	***1.00976±0.0026***
61	2.35%	16.84	SS304L1.1%	2.98	N/A	Steel	13.21	43.9	1.01084±0.00267
62	***2.35%***	***16.84***	***BoralB***	***2.92***	***N/A***	***Steel***	***13.21***	***22.8***	***1.00711±0.0031***
63	2.35%	16.84	Boroflex	2.26	N/A	Steel	13.21	25.7	1.01203±0.00304

Table 3-23

Benchmark Calculation Results (Continued)

TNW Ref. #	Enrich-ment (w%)	Rod Pitch (mm)	Absorber Material	Absorber Thickness (mm)	Abs. to Cluster Distance (mm)	Reflector Material	Refl. to Cluster Distance (mm)	Critical Cluster Sep. (mm)	$k_{eff} \pm 1\sigma$
64	2.35%	16.84	Cadmium	0.61	N/A	Steel	13.21	34.5	1.00903±0.00259
65	2.35%	16.84	Copper	3.37	N/A	Steel	13.21	73.8	1.00367±0.00267
66	2.35%	16.84	Cu/Cd	3.57	N/A	Steel	13.21	50.2	1.00470±0.00273
67	4.31%	18.92	N/A	N/A	N/A	Steel	19.56	153.6	0.99884±0.0029
68	***4.31%***	***18.92***	***SS304L***	***3.02***	***N/A***	***Steel***	***19.56***	***132.7***	***1.00254±0.00298***
69	4.31%	18.92	SS304L1.1%	2.98	N/A	Steel	19.56	93.5	1.00232±0.00299
70	***4.31%***	***18.92***	***BoralB***	***2.92***	***N/A***	***Steel***	***19.56***	***78.2***	***1.00499±0.00302***
71	4.31%	18.92	Boroflex	2.26	N/A	Steel	19.56	78.9	0.99841±0.00289
72	4.31%	18.92	Cadmium	0.61	N/A	Steel	19.56	84.6	1.00984±0.00304
73	4.31%	18.92	Copper	3.37	N/A	Steel	19.56	129.9	1.00542±0.00296
74	4.31%	18.92	Cu/Cd	3.57	N/A	Steel	19.56	100.9	1.00094±0.00302
75	2.35%	16.84	N/A	N/A	N/A	Uranium	0	76.5	0.98897±0.00269
76	2.35%	16.84	N/A	N/A	N/A	Uranium	13.21	90.9	0.98111±0.00225
77	2.35%	16.84	N/A	N/A	N/A	Uranium	26.16	94.2	0.98716±0.00263
78	2.35%	16.84	N/A	N/A	N/A	Uranium	39.12	87.8	1.00210±0.00262
79	2.35%	16.84	N/A	N/A	N/A	Lead	0	96.5	1.00826±0.00254
80	2.35%	16.84	N/A	N/A	N/A	Lead	6.6	97.0	1.01217±0.00276
81	2.35%	16.84	N/A	N/A	N/A	Lead	32.75	80.9	1.00471±0.00261
82	2.35%	16.84	N/A	N/A	N/A	Moderator	N/A	61.8	1.00797±0.00259
83	2.35%	16.84	N/A	N/A	N/A	Moderator	N/A	68.1	1.00909±0.00266
84	4.31%	18.92	N/A	N/A	N/A	Uranium	0	148.5	0.97412±0.00294
85	4.31%	18.92	N/A	N/A	N/A	Uranium	6.6	162.3	0.98458±0.00283
86	4.31%	18.92	N/A	N/A	N/A	Uranium	13.21	177.9	0.98622±0.00333
87	4.31%	18.92	N/A	N/A	N/A	Uranium	19.56	187.6	0.99678±0.00262
88	4.31%	18.92	N/A	N/A	N/A	Uranium	26.16	188.9	0.99595±0.00294
89	4.31%	18.92	N/A	N/A	N/A	Uranium	32.75	183.0	0.99409±0.00284
90	4.31%	18.92	N/A	N/A	N/A	Uranium	54.05	159.2	0.99861±0.00275
91	4.31%	18.92	N/A	N/A	N/A	Moderator	N/A	118.8	1.01400±0.00252
92	4.31%	18.92	N/A	N/A	N/A	Lead	0	172.6	1.00956±0.00291
93	4.31%	18.92	N/A	N/A	N/A	Lead	6.6	177.0	1.01158±0.00269
94	4.31%	18.92	N/A	N/A	N/A	Lead	19.56	169.5	1.00399±0.00286
95	4.31%	18.92	N/A	N/A	N/A	Lead	50.01	138.7	1.00093±0.00286

Table 3-23

Benchmark Calculation Results (Continued)

TNW Ref. #	Enrich-ment (w%)	Rod Pitch (mm)	Absorber Material	Absorber Thickness (mm)	Abs. to Cluster Distance (mm)	Reflector Material	Refl. to Cluster Distance (mm)	Critical Cluster Sep. (mm)	$k_{eff} \pm 1\sigma$
96	2.35%	20.32	N/A	N/A	N/A	Moderator	N/A	119.2	1.01148±0.00268
97	2.35%	20.32	N/A	N/A	N/A	Moderator	N/A	84.1	1.00789±0.00238
98	2.35%	20.32	N/A	N/A	N/A	Moderator	N/A	84.2	1.00908±0.00238
99	2.35%	20.32	N/A	N/A	N/A	Moderator	N/A	100.5	1.01146±0.00265
100	2.35%	20.32	N/A	N/A	N/A	Moderator	N/A	63.9	1.01521±0.0025
101	2.35%	20.32	N/A	N/A	N/A	Moderator	N/A	80.1	0.99060±0.00262
102	2.35%	20.32	N/A	N/A	N/A	Moderator	N/A	44.6	1.00657±0.00229
103	***2.35%***	***20.32***	***SS304L***	***4.85***	***6.45***	***Moderator***	***N/A***	***68.8***	***1.00958±0.00239***
104	***2.35%***	***20.32***	***SS304L***	***4.85***	***27.32***	***Moderator***	***N/A***	***76.4***	***1.01405±0.00266***
105	***2.35%***	***20.32***	***SS304L***	***4.85***	***40.42***	***Moderator***	***N/A***	***75.1***	***1.00381±0.00271***
106	***2.35%***	***20.32***	***SS304L***	***3.02***	***6.45***	***Moderator***	***N/A***	***74.2***	***1.00641±0.00251***
107	***2.35%***	***20.32***	***SS304L***	***3.02***	***40.42***	***Moderator***	***N/A***	***77.6***	***1.01278±0.00253***
108	***2.35%***	***20.32***	***SS304L***	***3.02***	***6.45***	***Moderator***	***N/A***	***104.4***	***1.00873±0.00242***
109	***2.35%***	***20.32***	***SS304L***	***3.02***	***40.42***	***Moderator***	***N/A***	***114.7***	***1.00964±0.00238***
110	2.35%	20.32	SS304L1.1%	2.98	6.45	Moderator	N/A	75.6	1.00658±0.00244
111	2.35%	20.32	SS304L1.1%	2.98	40.42	Moderator	N/A	96.2	1.01159±0.00259
112	2.35%	20.32	SS304L1.6%	2.98	6.45	Moderator	N/A	73.6	1.00728±0.0026
113	2.35%	20.32	SS304L1.6%	2.98	40.42	Moderator	N/A	95.2	1.00881±0.00247
114	***2.35%***	***20.32***	***BoralA***	***7.13***	***6.45***	***Moderator***	***N/A***	***63.3***	***1.00871±0.00245***
115	***2.35%***	***20.32***	***BoralA***	***7.13***	***44.42***	***Moderator***	***N/A***	***90.3***	***1.01064±0.00237***
116	***2.35%***	***20.32***	***BoralA***	***7.13***	***6.45***	***Moderator***	***N/A***	***50.5***	***1.00950±0.00246***
117	2.35%	20.32	Copper	6.46	6.45	Moderator	N/A	66.2	1.01099±0.0025
118	2.35%	20.32	Copper	6.46	27.32	Moderator	N/A	77.2	1.00594±0.0026
119	2.35%	20.32	Copper	6.46	44.42	Moderator	N/A	75.1	1.00779±0.00246
120	2.35%	20.32	Copper	3.37	6.45	Moderator	N/A	68.8	1.00577±0.00228
121	2.35%	20.32	Copper	3.37	40.42	Moderator	N/A	70.0	1.00410±0.00227
122	2.35%	20.32	Cu/Cd	3.57	6.45	Moderator	N/A	51.5	1.00808±0.00244
123	2.35%	20.32	Cadmium	0.61	6.45	Moderator	N/A	67.4	1.00808±0.00244
124	2.35%	20.32	Cadmium	0.61	14.82	Moderator	N/A	76.0	1.00606±0.00253
125	2.35%	20.32	Cadmium	0.61	40.42	Moderator	N/A	93.7	1.00616±0.00249
126	2.35%	20.32	Cadmium	0.291	14.82	Moderator	N/A	77.8	1.01101±0.0023
127	2.35%	20.32	Cadmium	0.291	40.42	Moderator	N/A	94.0	1.01378±0.00281

Table 3-23

Benchmark Calculation Results

(Concluded)

TNW Ref. #	Enrich-ment (w%)	Rod Pitch (mm)	Absorber Material	Absorber Thickness (mm)	Abs. to Cluster Distance (mm)	Reflector Material	Refl. to Cluster Distance (mm)	Critical Cluster Sep. (mm)	$k_{eff} \pm 1\sigma$
128	2.35%	20.32	Cadmium	0.901	14.82	Moderator	N/A	75.4	1.00665±0.00241
129	2.35%	20.32	Cadmium	0.901	40.42	Moderator	N/A	93.9	1.01462±0.00257
130	2.35%	20.32	Aluminum	6.25	6.45	Moderator	N/A	86.7	1.01100±0.00245
131	2.35%	20.32	Aluminum	6.25	40.42	Moderator	N/A	87.8	1.00750±0.00255
132	2.35%	20.32	Aluminum	6.25	44.42	Moderator	N/A	88.3	1.00336±0.00252
133	2.35%	20.32	Zircaloy-4	6.52	6.45	Moderator	N/A	87.9	1.01140±0.00241
134	2.35%	20.32	Zircaloy-4	6.52	40.42	Moderator	N/A	87.8	1.00958±0.00244

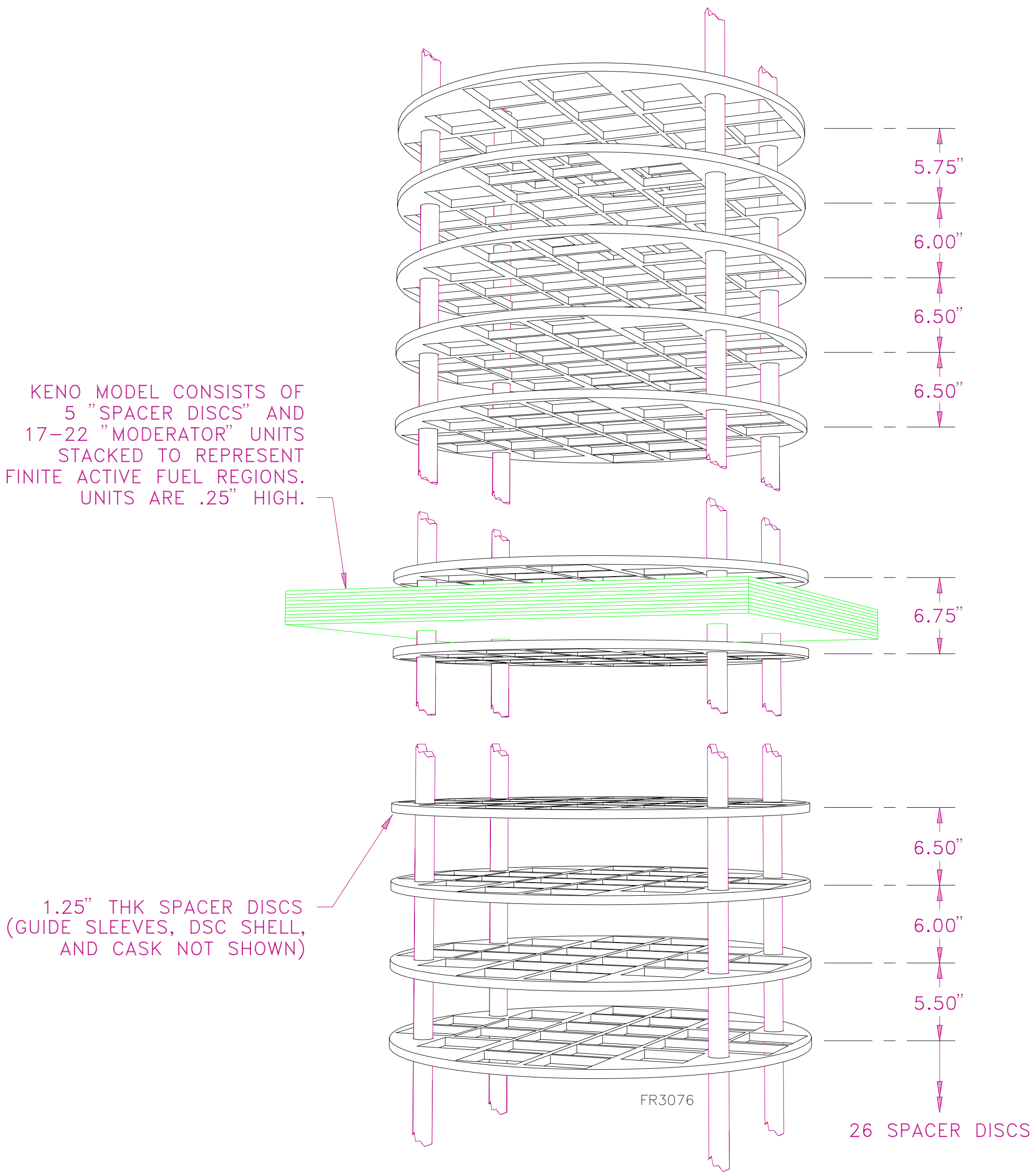


Figure 3-1

KENO Model and DSC Basket

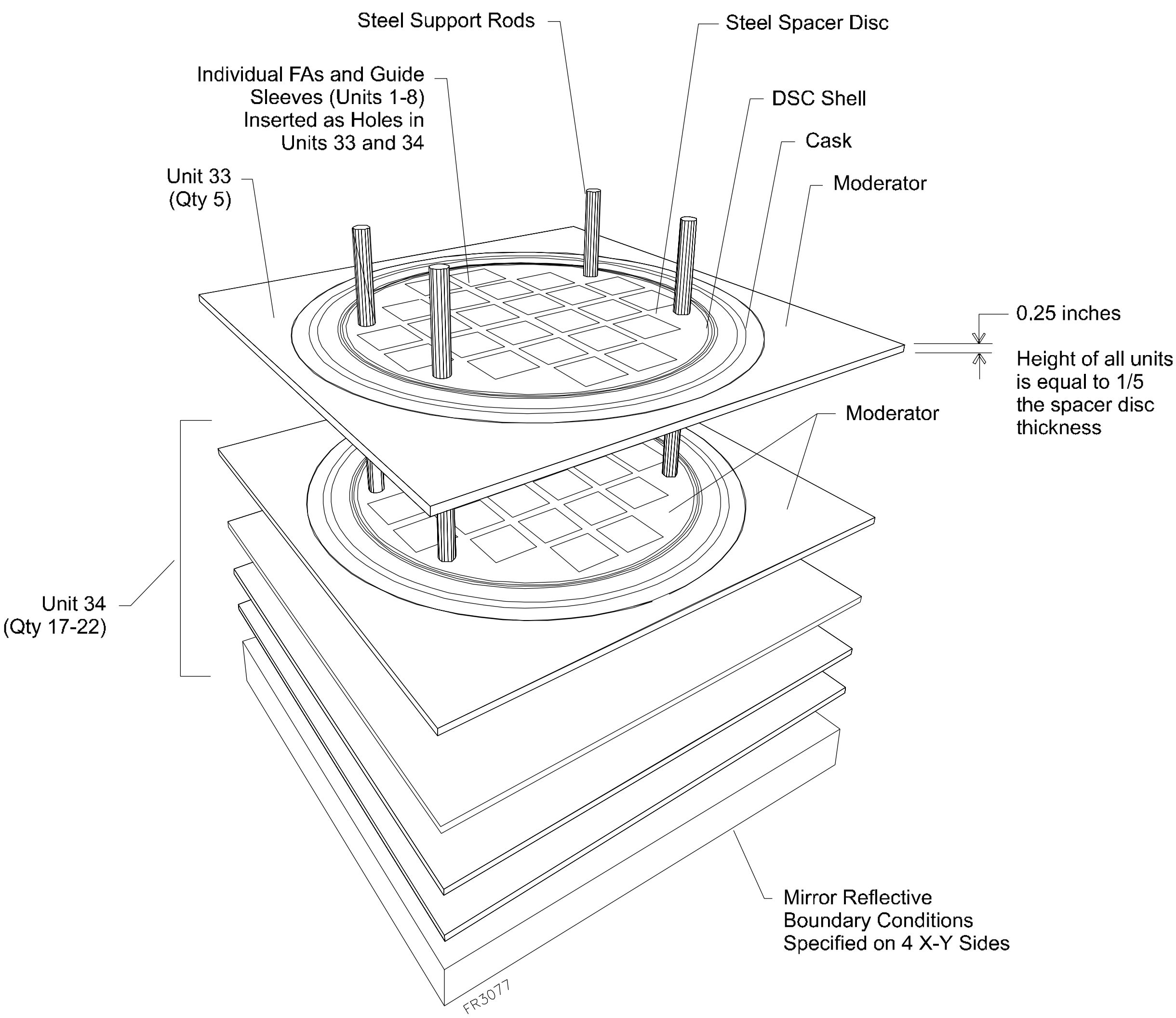


Figure 3-2

Exploded View of KENO Model

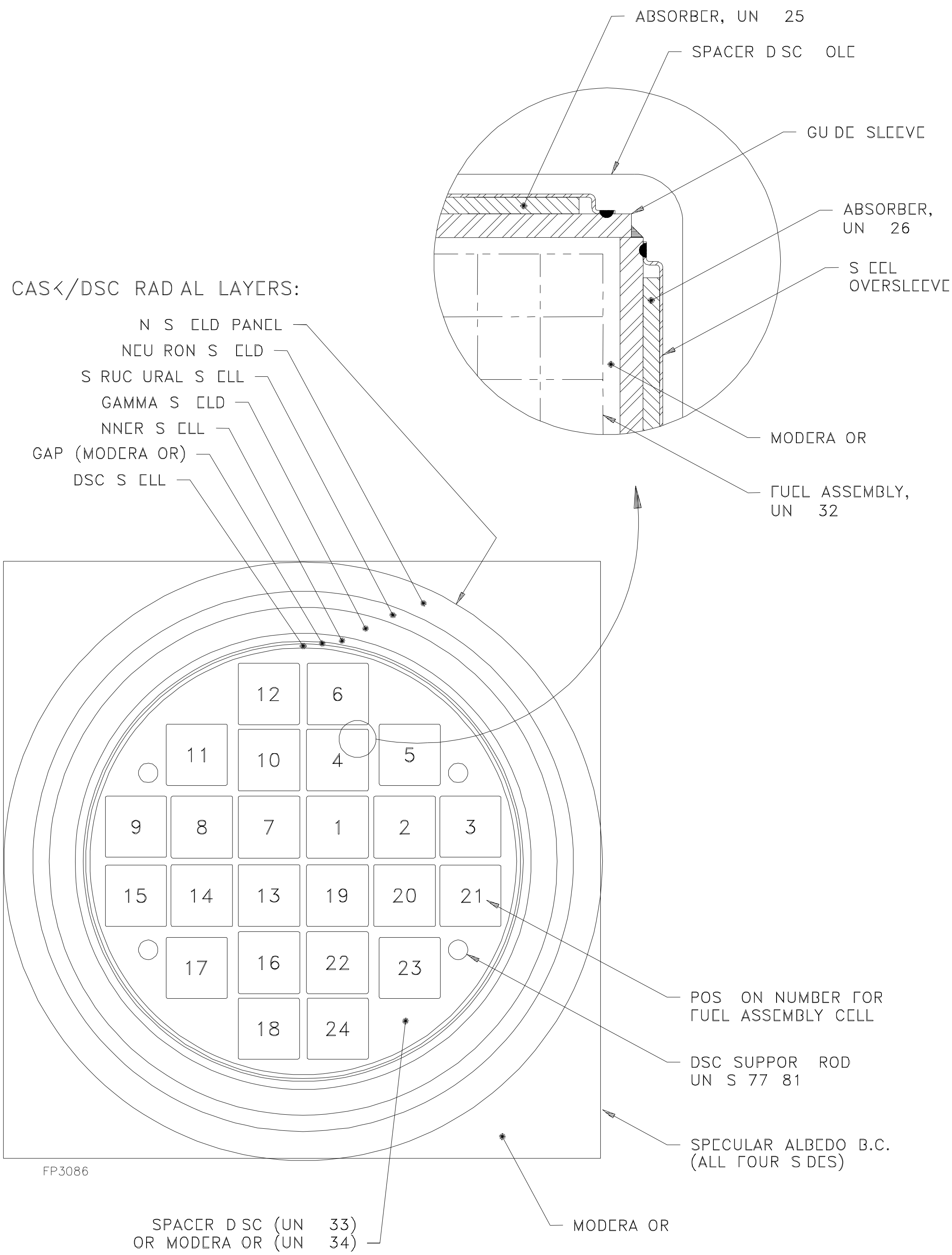


Figure 3-3
Structure of KENO Model UNITS 33 and 34

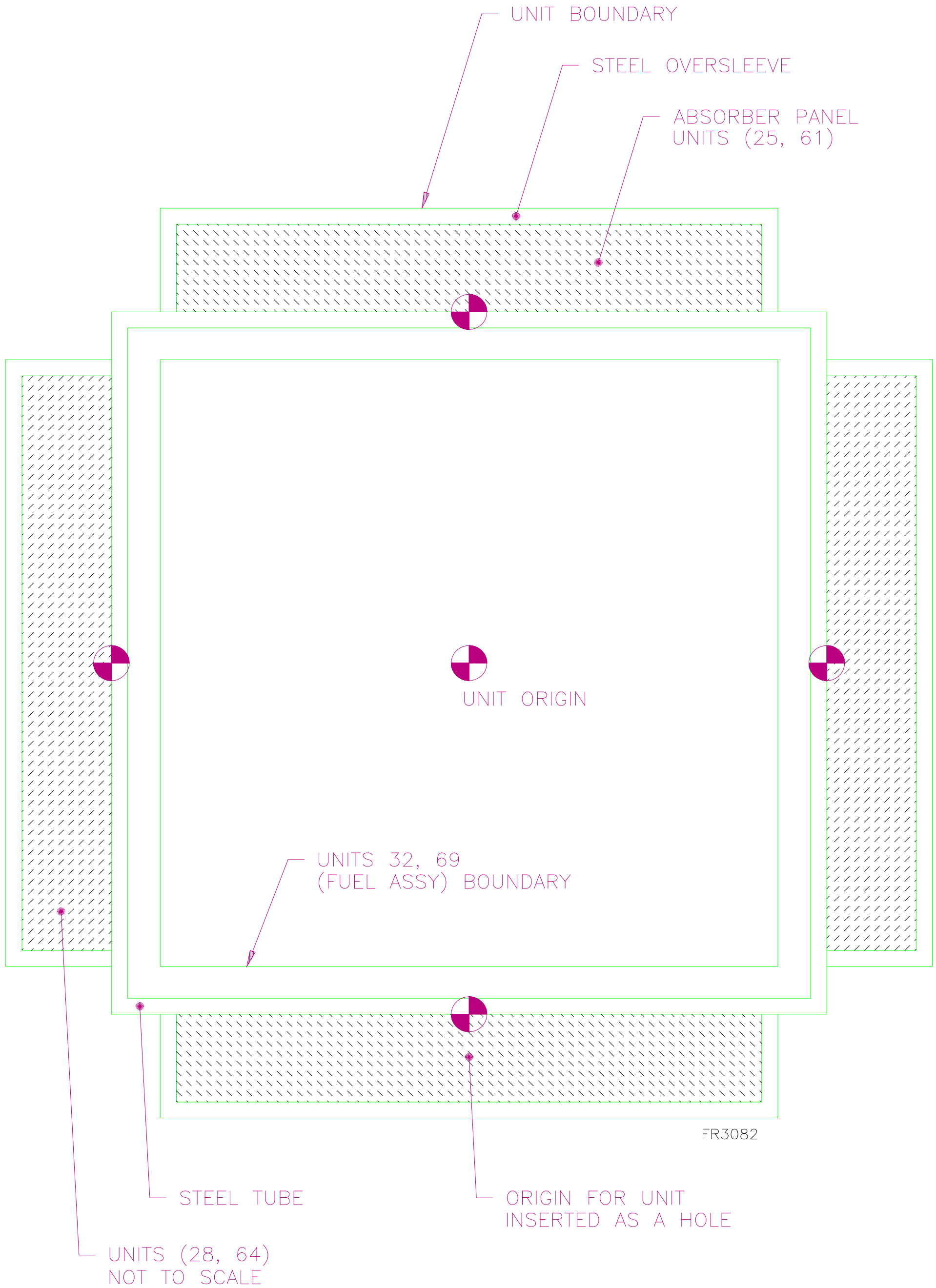


Figure 3-4

KENO Model UNITS 1-8

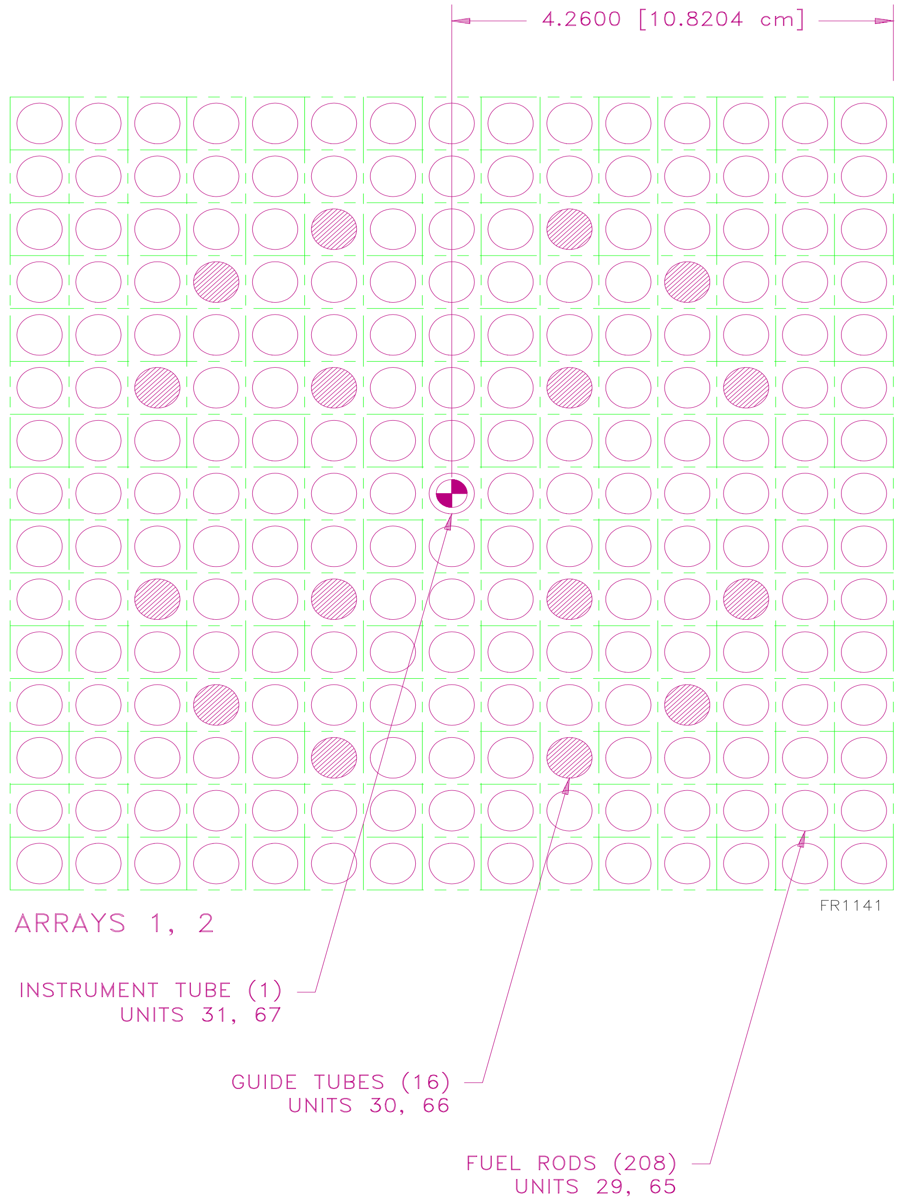


Figure 3-5

KENO Model of a Design Basis Fuel Assembly

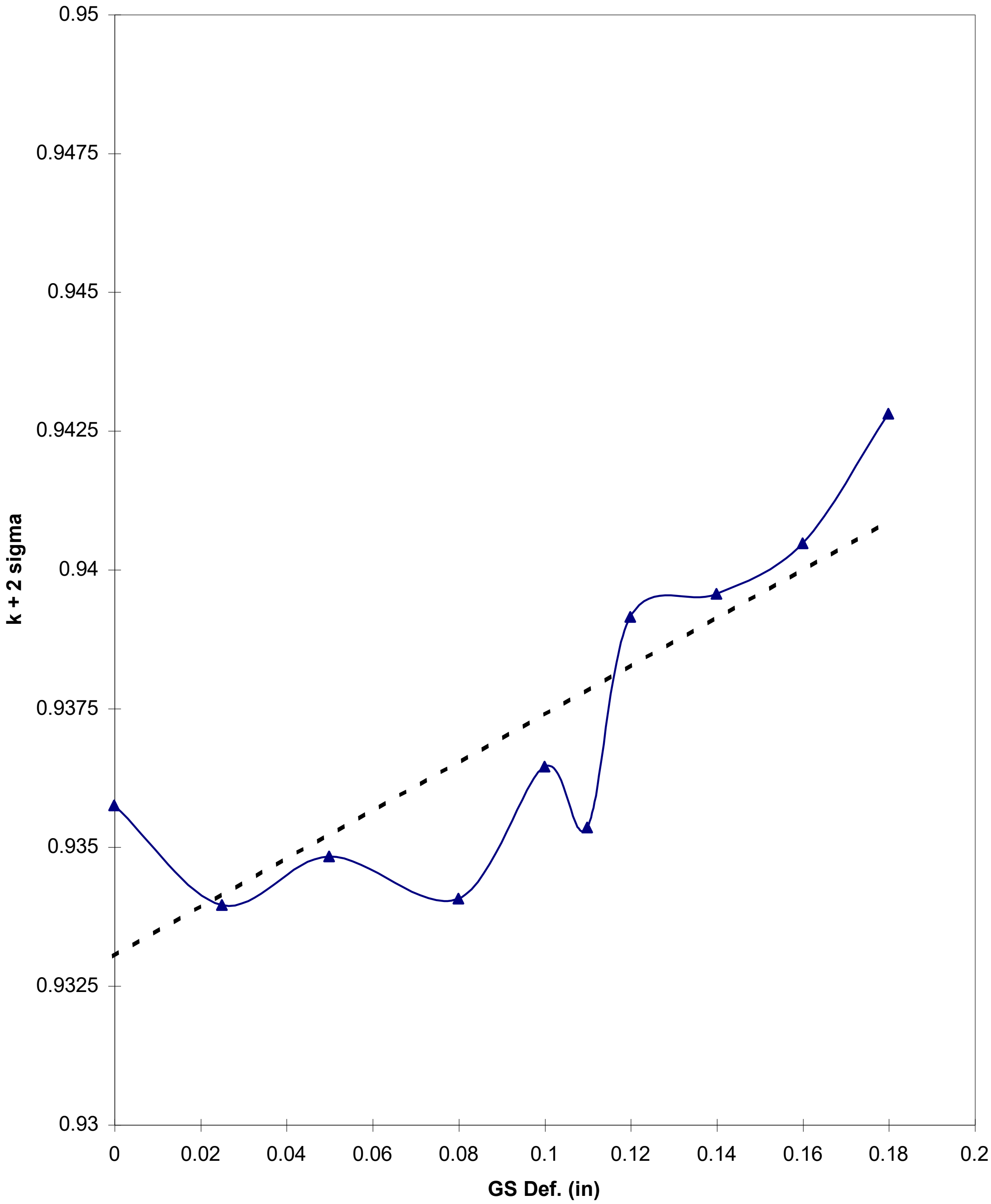


Figure 3-6

K vs. Guide Sleeve Deformation

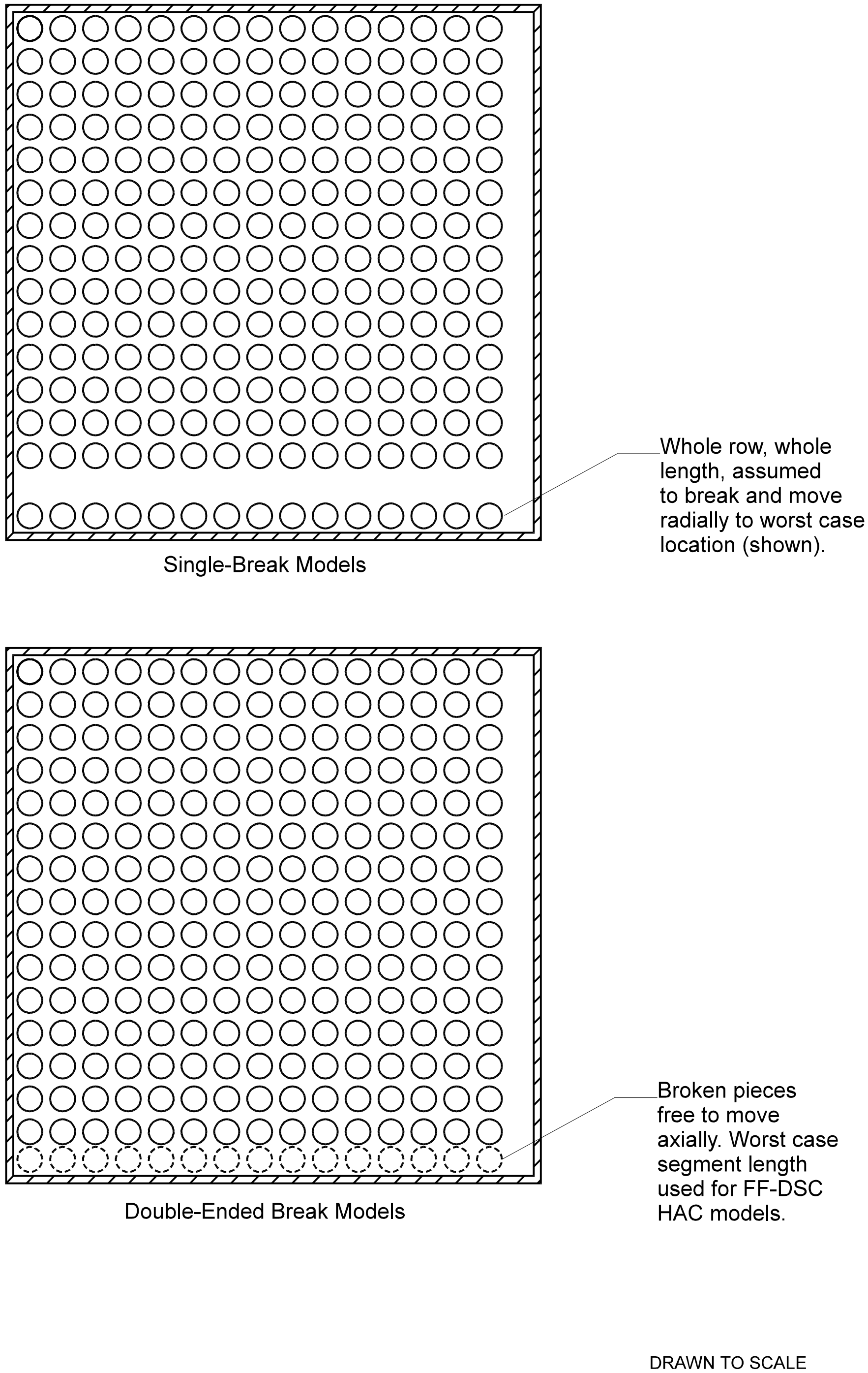


Figure 3-7

FF-DSC Broken Fuel Rod Models

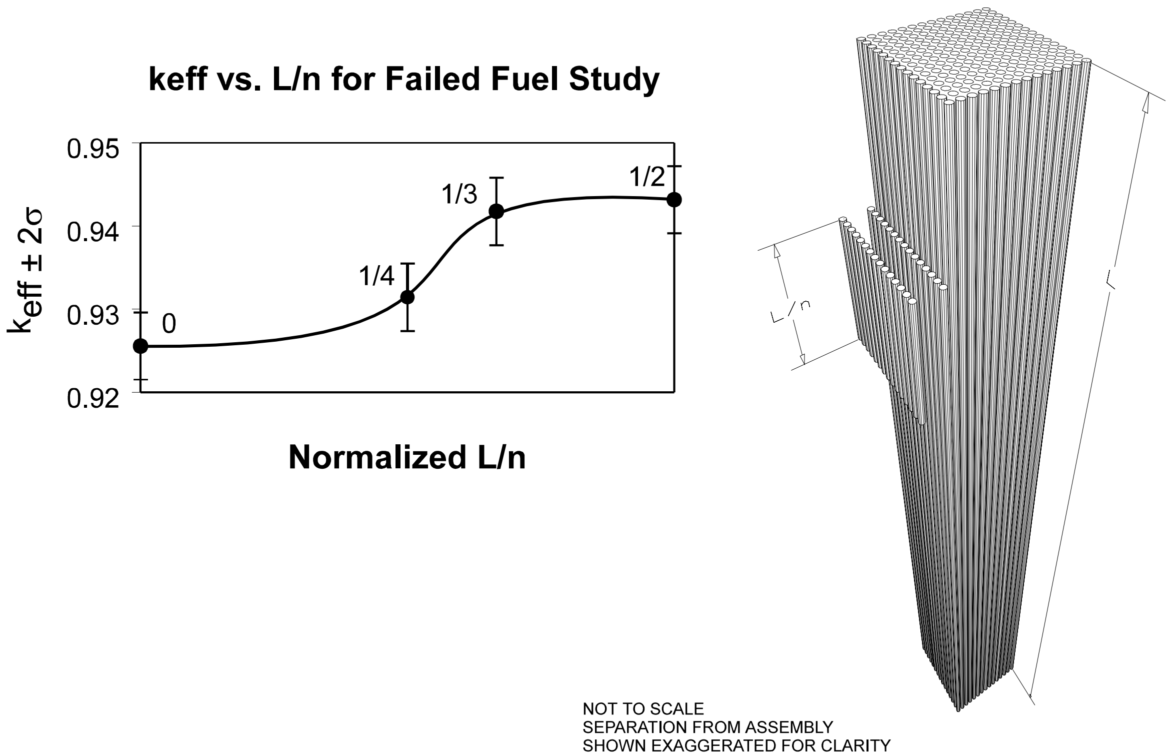
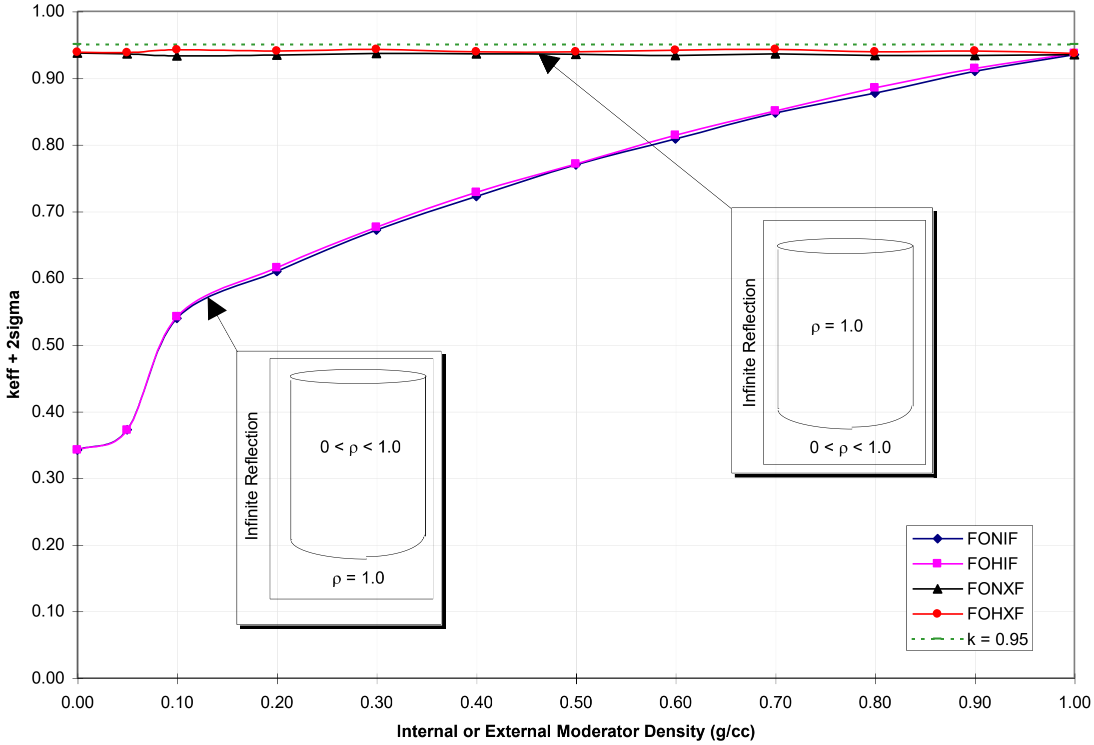


Figure 3-8

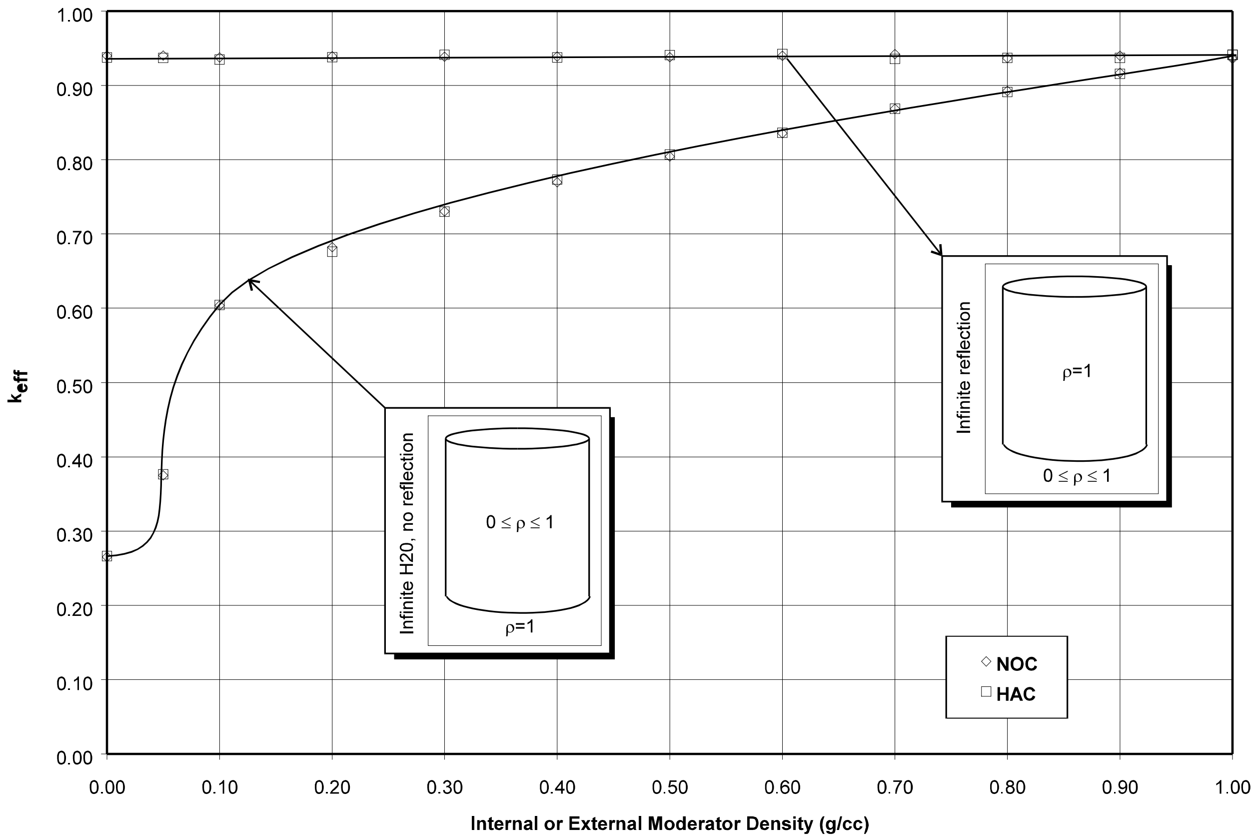
FF-DSC Double-Ended Rod Break Models



± 2σ error bars omitted for clarity

Figure 3-9

NUHOMS®-MP187 Cask/FO-DSC Criticality Results



± 2σ error bars omitted for clarity

Figure 3-10

NUHOMS®-MP187 Cask/FF-DSC Criticality Results

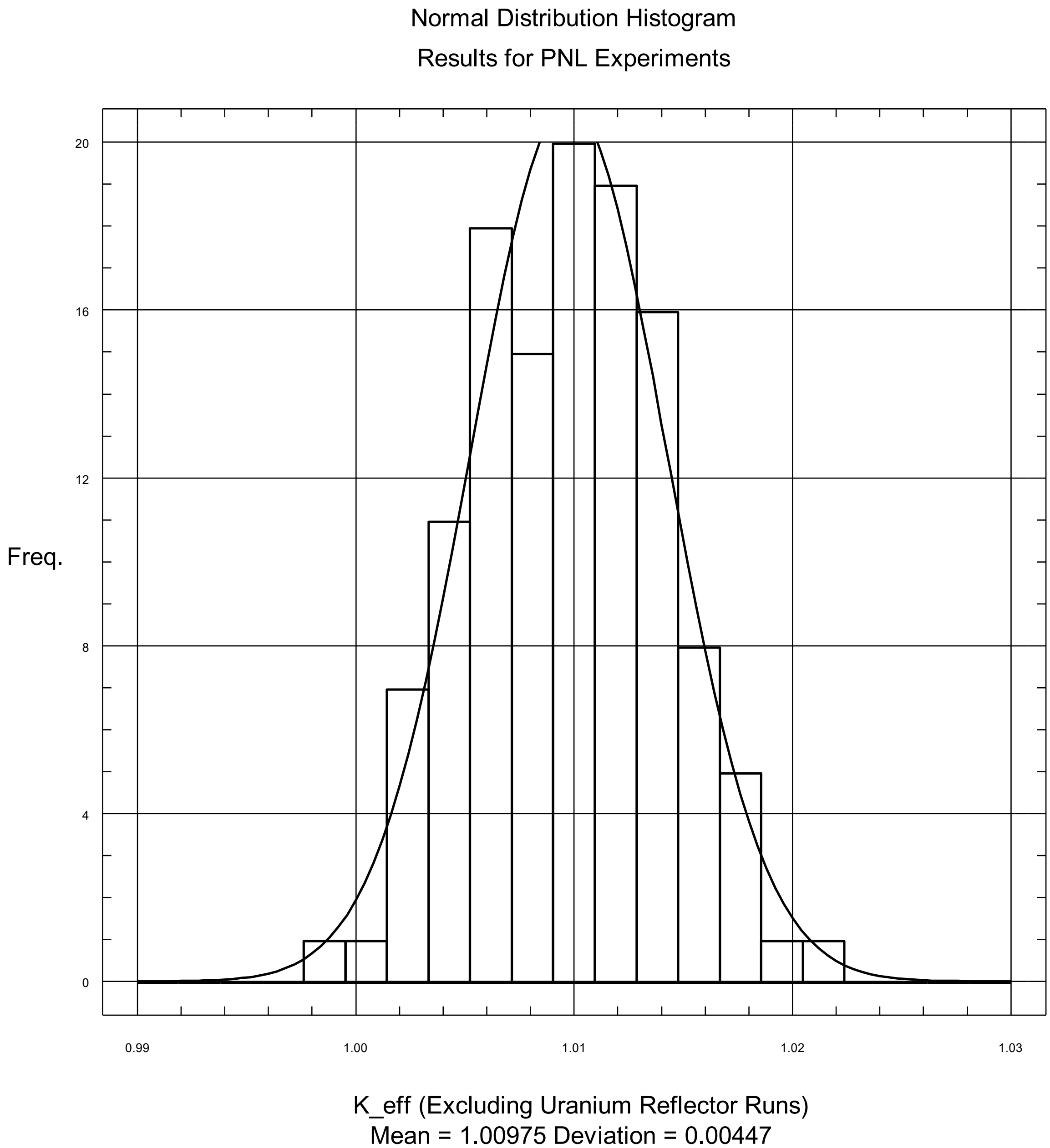


Figure 3-11

Critical Benchmark Results

4. INSTALLATION DESIGN

4.1 Summary Description

This chapter provides more detailed descriptions of the Rancho Seco HSMs, DSCs, and cask. This chapter also relates the design bases and use of industrial codes to the design criteria presented in Chapter 3.

4.1.1 Location and Layout of Installation

The ISFSI is designed in accordance with 10 CFR 72, Subpart F - General Design Criteria. Chapter 1 provides information regarding the location of the ISFSI. The concrete slab has been built approximately 600 feet west of the IOSB. The operational areas of the Rancho Seco ISFSI include the concrete slab and the immediate surrounding area. The ISFSI is located within the RSNGS Owner Controlled Area, as described in Chapter 2.

The ISFSI consists of two back to back rows of 11 HSMs and shield walls. The HSMs rest on a 2' thick concrete slab. The total ISFSI pad is approximately 225 X 170 feet, within the ISFSI fenced-in area. The slab is designed to accommodate 26 HSMs; each HSM is approximately 19 feet long, 15 feet high, and 9.7 feet wide. There is 6 inches between HSMs as positioned on the slab. The outside end walls of an HSM array are shielded by 2'-0" thick shield walls. The ISFSI is surrounded by chain link fences with entrance gates on the east side. A 10' x 10' pre-engineered building has been erected in one corner of the ISFSI to house lighting and security equipment. This building is designated as the Electrical Building. The ISFSI layout is shown in Figure 1-2.

4.1.2 Principal Features

The principal features of the Rancho Seco NUHOMS® ISFSI installation are described in Sections 1.2 and 1.3, and in Chapter 4 of the Standardized NUHOMS® SAR [4.4.1]. The location of the boundary and site layout are shown on Figure 1-1.

> See Appendix B for Standardized SAR, Section 1.2 (pages 1.2-1 to 1.2-14), and Section 1.3 (pages 1.3-1 to 1.3-23).

4.2 Storage Structures

The Rancho Seco ISFSI uses the HSMs the for storage of DSCs. All Rancho Seco DSCs have the same exterior dimensions, and are designed to be accepted by the HSMs and cask.

The Rancho Seco ISFSI utilizes three types of DSCs, summarized as follows (detailed descriptions are provided in the following sections):

1. Fuel Only DSC (FO-DSC) This type of DSC is designed to hold 24 fuel assemblies only (i.e., without control components). The cavity length of FO-DSCs are suitable for Rancho Seco fuel. There are two FO-DSCs at the Rancho Seco ISFSI.

2. Fuel with Control Components DSC (FC-DSC) This type of DSC is designed to hold 24 fuel assemblies with control components. The cavity length of the FC-DSC is larger than the FO-DSC while keeping the overall length the same. The top and bottom shield plugs for the FC-DSC are lead as compared to the FO-DSC which has carbon steel shield plugs. There are eighteen FC-DSCs at the Rancho Seco ISFSI.

3. "Failed Fuel" DSC (FF-DSC) The third type of DSC is designed to hold 13 damaged fuel assemblies inside removable cans. The fuel cans provide for containment of fuel pellets/shards. The basket assembly is designed such that the overall DSC dry loaded weight is approximately equal to the FO- and FC-DSCs. The internal cavity dimensions for the FF-DSC are the same as for the FO-DSC. There is one FF-DSC at the Rancho Seco ISFSI.

4.2.1 Design Bases and Safety Assurance

The intent of the ISFSI is to provide safe containment during dry storage of spent nuclear fuel. In accordance with 10 CFR 72.3, the only components at the Rancho Seco ISFSI important to safety are the DSCs, HSMs, and transfer cask. These components are self-contained, independent, passive systems and do not rely on any other systems or components for their operation. The cask and DSCs rely on other systems during fuel loading, cask handling, and transfer operations; but in storage, the DSC is self-contained, independent, and passive.

The Rancho Seco HSM design bases are similar to those described in the Chapters 3 and 4 of the Standardized NUHOMS® SAR [4.4.1]. The Rancho Seco cask design bases are similar to the NUHOMS® transfer cask design bases as described in Chapters 3 and 4 of the Standardized NUHOMS® SAR [4.4.1]. The SMUD Rancho Seco DSC's design bases are similar to those defined in Chapters 3 and 4 of the Standardized NUHOMS® SAR [4.4.1].

As described in Chapter 8, the design and operation of the Rancho Seco ISFSI ensures that a single failure will not result in the release of significant radioactive material.

The following sections discuss the conformance of the Rancho Seco ISFSI with applicable 10 CFR 72 design criteria.

4.2.2 Compliance with General Design Criteria

4.2.2.1 10 CFR 72.122 Overall Requirements

1. Quality standards Quality assurance requirements are addressed in Chapter 11.

2. Protection against environmental conditions and natural phenomena Extreme environmental conditions for the ISFSI are defined in Chapter 2. The design criteria require that the storage system be designed to withstand the design earthquake, high ambient temperature and humidity, and extreme winds.

   Lightning protection is provided by lightning rods installed on certain light poles at the ISFSI.

3. Protection against fire and explosion The design criteria require that the storage system be designed so that it can continue to perform its safety functions effectively under credible fire and explosion exposure conditions. As discussed in Section 3.3.6, no large fire or explosion within the Rancho Seco ISFSI is considered credible.

4. Sharing of structures, systems, and components The storage system and other ISFSI support systems will not be shared with any other facilities, and ISFSI activities will not impair any activities at RSNGS. The source of backup electrical power to the ISFSI is the emergency diesel generator associated with the microwave communications building.

5. Proximity of sites The design and operation of the ISFSI will result in minimal risk to the health and safety of the public. During the fuel transfer campaign, RSNGS will remain shutdown, as decommissioning activities continue.

   a. 6. Testing and maintenance of systems and components The design criteria require that the HSMs be designed to permit inspection, maintenance, and testing. Although the storage system requires minimum maintenance, the design of the ISFSI will allow for appropriate testing, inspection, and maintenance, if required.

7. Emergency capability Scenarios requiring emergency actions are neither considered credible, nor postulated to occur. Nevertheless, emergency facilities, as described in the RSNGS Emergency Plan [4.4.2], would be available, if needed. After the 10 CFR 50 license is terminated, the Rancho Seco ISFSI Emergency Plan will remain in effect to meet the requirements in 10 CFR 72.32.

8. Confinement barriers and systems The design of the storage system will ensure that spent fuel cladding is protected from degradation during storage and that stored fuel is maintained in a safe condition.

9. Instrumentation and control systems No control systems are needed for the storage system to perform its safety functions. The parameters that affect the long-term safe storage of spent nuclear fuel are structural integrity of confinement, shielding, passive cooling (heat rejection), and criticality control. To ensure adequate thermal performance of the ISFSI components, instrumentation is provided to monitor HSM concrete temperature.

   The Standardized NUHOMS® SAR [4.4.1] has demonstrated that the NUHOMS® ISFSI is safe under all credible normal, off-normal, or accident conditions. There are no accident scenarios which require instrumentation or control system monitoring to verify the safe operation of a NUHOMS® ISFSI.

10. Control room or control areas The Rancho Seco ISFSI is a passive installation, with no need for operator actions. No control room is needed for normal ISFSI operations; however, the instrumentation used to monitor HSM concrete temperature has a readout in the control room.

11. Utility services There are no utility or emergency systems required to perform safety functions at the ISFSI. Section 4.3 addresses auxiliary system requirements.

12. Retrievability By using a transportable storage system, the stored fuel can be transferred directly to a DOE facility after DOE acceptance of the fuel. The steps involved in placing a loaded vertical cask on the transfer trailer and transferring the cask from the trailer to a rail car and preparing it for transport are covered in the NUHOMS®-MP187 Multi-Purpose Cask Transportation Package Safety Analysis Report, Document No. NUH-05-151 submitted in accordance with 10 CFR 71.

4.2.2.2 10 CFR 72.124 - Criteria for Nuclear Criticality Safety

1. Design for criticality safety The design criteria require that the DSCs be designed to maintain subcriticality at all times, assuming a single active or credible passive failure.

2. Methods of criticality control The primary nuclear criticality safety design criterion is to provide design features that ensure that the fuel contained in the DSCs remains subcritical under normal, off-normal, and accident conditions. Primary control methods for the prevention of criticality are discussed in the NUHOMS®-MP187 Transportation SAR [4.4.7]. The Rancho Seco DSC designs also include the use of fixed, borated neutron absorbing panels to facilitate transportation. The methods used will be effective during normal, off-normal, and accident event conditions.

Additional design control methods include conservative analyses, specified error contingency criteria, and analysis verification.

3. Criticality monitoring Due to the criticality safety design of the DSCs, no criticality monitoring of the cask is required.

4.2.2.3 10 CFR 72.126 Criteria for Radiological Protection

1. Exposure control Operations at the Rancho Seco ISFSI will be conducted in accordance with ALARA procedures. Minimal maintenance operations are needed following DSC placement at the ISFSI. DSC loading, sealing, decontamination, and preparation will be performed in accordance with plant procedures. While fuel is stored at the ISFSI, access will be controlled by a double fence with locked gates.

2. Radiological alarm systems No radioactive releases are considered credible at the Rancho Seco ISFSI, and no alarm systems are needed.

3. Effluent and direct radiation monitoring Operation of the Rancho Seco ISFSI will not result in radioactive contamination of any plant effluents. No safety-related monitors are needed. Dosimeters will be used to monitor direct radiation around the ISFSI.

4. Effluent control No radioactive releases are considered credible at the Rancho Seco ISFSI.

4.2.2.4 10 CFR 72.128 Criteria for Spent Fuel, High-level Radioactive Waste, and other Radioactive Waste Handling and Storage

1. Spent fuel and radioactive waste storage and handling systems The design criteria require that the storage system provides sufficient shielding to lower surface doses to below prescribed levels, maintain containment, and maintain fuel in a safe condition under all normal and credible accident conditions. Any radioactive waste generated would be during cask decontamination prior to the cask leaving the Fuel Storage Building.

2. Waste treatment As stated in Section 1.3.3, Vacuum Drying System (VDS) exhaust and general cask decontamination waste are generated during DSC drying and sealing operations. During DSC drying and sealing operations, all discharges from the DSC cavity, whether gas or water, will be handled by the RSNGS radioactive waste system. Water from the DSC cavity may be routed back to the fuel pool, as appropriate. Both VDS exhaust and general cask decontamination waste are managed with established waste processing practices.

4.2.2.5 10 CFR 72.130 Criteria for Decommissioning

Operation of the Rancho Seco ISFSI will not result in contamination on the outside surface of the DSCs or any other ISFSI components above administrative limits. Decommissioning considerations for the cask are discussed in Volume III, Section 4.6.

4.2.3 Structural Specifications

Safe storage of the spent fuel assemblies depends only on the capability of the storage system to fulfill its design functions. The design criteria for the storage system ensures that its exposure to credible site hazards will not impair their safety function. Refer to Chapter 4 of the Standardized NUHOMS® SAR [4.4.1] for an itemized list of the Codes of Construction for the NUHOMS® ISFSI components. Appendix A provides a listing of ASME Code exceptions for the DSCs and the cask.

The HSMs are placed to ensure that the design criteria listed in the Standardized NUHOMS® SAR [4.4.1] are not exceeded, and that the safety of RSNGS is not impaired.

The slab was built in accordance with applicable commercial grade codes and standards and is approximately 2 feet-thick reinforced concrete under the HSMs. As described in the Standardized NUHOMS® SAR [4.4.1], the slab is non-safety related and provides a uniform level surface for storing the HSMs. The transportation route and area surrounding the slab was constructed to properly support the transporter used for handling the loaded multi-purpose cask. The compacted area around the slab allows for the movement and positioning of the cask handling equipment.

4.2.4 Installation Layout

Volume IV contains detailed drawings of the ISFSI cask and canisters.

4.2.5 Individual Unit Description

4.2.5.1 Horizontal Storage Module (HSM)

Chapter 4 of the Standardized NUHOMS® SAR [4.4.1] provides a description of the standardized NUHOMS® HSM.

4.2.5.2 Dry Shielded Canister

The Rancho Seco DSCs are high integrity stainless steel, welded pressure vessels that provide confinement of radioactive materials, encapsulate the fuel in a helium atmosphere; and together with a cask or HSM, provide biological shielding during DSC closure operations, transfer and long term storage. The Rancho Seco DSCs are shown in Figure 1-4 and detailed design drawings are contained in Volume IV.

To the extent practicable, the DSCs are designed and built to meet the requirements of ASME Section III. An N-stamp will not be applied to the DSC. The DSC shell is designed to the requirements of Section III, Subsection NB. The weld configuration for the outer top cover plate prevents radiography testing (RT), and will therefore be examined by multi-layered liquid penetrant testing (PT). Criticality control structures (basket materials) are generally designed per Subsection NG. See Appendix A for a list of code exceptions. The following subsections describe the three different types of Rancho Seco DSCs.

The Rancho Seco DSCs are being licensed for transfer, storage, and offsite transportation under both 10 CFR 71 and 10 CFR 72. The requirements of 10 CFR 71 have necessitated that the basket be designed to account for fuel assembly loading of the guide sleeves. The Rancho Seco basket design incorporates 26 spacer discs to address this concern.

The spacer discs are held in place by support rod sleeves which are held in place by compression by a nut installed on the support rod end during a pretensioning operation performed during the basket assembly for the Rancho Seco DSCs. The spacer disc material used in the Rancho Seco DSC is a high strength carbon steel and the Rancho Seco FO and FC DSCs also use neutron poisons for criticality control.

The Rancho Seco DSCs uses two shield plug configurations. The FC-DSC uses lead shield plugs to provide a longer cavity to accommodate assemblies with control components without increase in overall canister length. This DSC also incorporates welded angles at the top of the guide sleeves to limit displacement of the guide sleeves during cask/canister drop scenarios. The FO-DSC uses steel shield plugs since the assembly length without control components can be accommodated with a thicker shield plug. The shield plugs are analyzed to the criteria in ASME Section III, Subsection NB. As noted in the drawings, material fabrication and inspection of the shield plugs follow commercial standards.

Carbon steel DSC components are coated with electroless nickel for corrosion protection during staging prior to use. The NUHOMS MP-187 Transportation Cask Safety Analysis Report, Section 2.4.4 provides a discussion of chemical and galvanic reactions during the time the electroless nickel coated surfaces are in contact with the borated water in the spent fuel pool during DSC loading and unloading operations.

4.2.5.2.1 FO-DSC

The FO-DSC has solid steel shield plugs and the basket assembly consists of 24 guide sleeve assemblies with integral poison plates, 26 spacer discs and four support rods. During SFA dry storage, criticality control is maintained using the poison plate design of the FO-DSC basket assembly.

4.2.5.2.2 FC-DSC

The FC-DSC has lead/steel composite shield plugs which provide a 173" cavity length to accommodate control components. The FC-DSC envelope dimensions are identical to those of the FO-DSC. The FC-DSC also has an internal poisoned basket.

The FC-DSC and the FO-DSC basket assemblies are nearly identical. One notable exception is the length of the support rods above the top spacer disc. The FC-DSC support rods are 6" longer than those used in the FO-DSC to accommodate a longer cavity required for fuel assembly components. Another difference is the addition of angle iron extensions at the top of each guide sleeve.

The bottom end inner plate and the inner cover plate welds form the inner pressure boundary of the DSC. The outer bottom cover plate and the top outer cover plate welds form the outer pressure boundary of the DSC.

See Appendix B for Standardized SAR, Section 4.2.3, pages 4.2-3 to 4.2-10

4.2.5.2.3 FF-DSC

The FF-DSC is similar to the FC-DSC in most respects with the exception of the basket assembly. The FF-DSC shell assembly, bottom shield plug, top shield plug, grapple ring, drain and vent ports, and outer cover plate are similar to the FC-DSC shell assembly.

See Appendix B for Standardized SAR, Section 4.2.3, pages 4.2-3 to 4.2-10

The FF-DSC shell and top and bottom end assemblies enclose a basket assembly which serves as the structural support for failed fuel assemblies. The FF-DSC basket assembly consists of fifteen 2" thick carbon steel or austenitic Stainless Steel (Type XM-19) spacer discs, eight carbon steel support plates, and thirteen stainless steel fuel can bodies. The spacer discs maintain the cross-sectional spacing of the fuel assemblies and provide lateral support for the fuel assemblies and fuel cans. The spacer discs are held in place by the support plates which maintain longitudinal separation during the postulated cask drop accident. The fuel can bodies are intended to be removable and, therefore, are not permanently attached to the basket assembly or DSC shell.

The spacer discs have thirteen 10" square cut-outs. Additionally, eight 2" thick or 4" thick by 12" wide by 172.5" long axial support plates are fitted between cut-outs in the spacer discs. Sets of two 2" thick support plates or a single 4" thick support plate are welded between the spacer discs at the 45, 135, 225, and 315° azimuth positions.

The FF-DSC fuel can consists of a seam welded stainless steel angle plate body with welded bottom lid assembly, welded top flange assembly and removable top lid assembly. The fuel cans are not "poisoned", however, they provide for containment of fuel pellets/shards by means of the fixed bottom screen and removable top screen. The bottom lid and top lid stainless steel screens allow for dewatering of the fuel can. The bottom end includes provision for fuel support. Each can is designed to be removable from the basket.

The completed FF-DSC carbon steel basket assembly (including the spacer discs and support plates, only) is coated with a thin corrosion resistant layer of electroless nickel identical to that used for the FO- and FC-DSCs. Electroless nickel is applied to the carbon steel components of the DSC basket for corrosion protection during staging prior to use. This treatment is intended to provide steel surfaces which meet the requirements of ANSI/ASME N45.2.1, Cleanness Class C. Corrosion properties are not relevant after fuel loading in the spent fuel pool since the storage atmosphere is inert. The austenitic stainless steel basket assembly is not required to be coated with electroless nickel

The NUHOMS MP-187 Transportation Cask Safety Analysis Report, Section 2.4.4 provides a discussion of chemical and galvanic reactions during the time the electroless nickel coated surfaces are in contact with the borated water in the spent fuel pool during DSC loading and unloading operations.

4.2.5.3 The NUHOMS®-MP187 Cask

The NUHOMS®-MP187 multi-purpose cask consists of an inner pressure retaining cylindrical shell welded to a forged bottom assembly and forged top flange with a bolted top cover plate and an outer structural shell welded to the forged bottom assembly and the forged top flange ring. To the maximum extent practical, the cask is designed and built to meet the requirements of ASME Section III. An N-Stamp will not be applied to the cask. The inner pressure retaining containment portion of the cask is designed to the requirements of Section III, Subsection NB. The weld configuration between the cask inner shell and the top closure forging is in accordance with Subsection NB. NUREG-3019 classifies the remaining cask components as “Other Safety Related” and permits the design to meet the requirements of ASME Code Section VIII or Section NF, but are conservatively analyzed to the requirements of NB or NF. The cask neutron shield is fabricated and inspected to the requirements of the SAR drawings. The neutron shield does not follow classical component support design, and therefore can not follow typical NF fabrication. See Appendix A for a list of ASME code exceptions.

The cask is designed and analyzed in accordance with ASME Code requirements. The NRC has also licensed the NUHOMS®-MP187 cask per 10 CFR 71 for the transportation of Rancho Seco canisterized fuel (DSCs). Many of the structural details of the cask reflect these design requirements. The cask design is illustrated in Figure 1-5. Detailed cask design drawings are contained in Volume IV.

The cask is designed for on-site transfer of any of the three types of DSCs described in Section 4.2.5.2. The actual cask to be used for on-site transfers may be the NUHOMS®-MP187 cask as described herein, or another previously NRC reviewed and approved design such as the transfer cask designs documented in the standardized NUHOMS® SAR [4.4.1], the NUHOMS®-24P Topical Report [4.4.8], or the Oconee Nuclear Station Safety Analysis

Report [4.4.9]. The cask provides the principal biological shielding and heat rejection mechanism for the DSC and SFAs during handling in the fuel building, DSC closure operations, transport to the ISFSI, and transfer to the HSM. The NUHOMS®-MP187 cask also provides primary protection for the loaded DSC during off-normal and drop accident events postulated to occur during the transport operations.

The cask cross section is constructed from two primary concentric cylindrical shells to form an inner annulus. This inner annulus is filled with lead. The inner pressure retaining and outer structural shell are welded to a heavy forged ring assembly at the top (top flange) and a heavy bottom plate (as shown on the design drawings in Volume IV). An outer neutron shield annulus is formed by welding a jacket to top and bottom support rings and longitudinal support angles which in turn are welded onto the outer surface of the structural shell. The neutron shield annulus is approximately the length of the active fuel and is filled with a solid neutron absorbing material. The cask shells, bottom end closure plate, top flange, neutron shield jacket, top cover plate, and ram access cover plate are all fabricated from stainless steel.

The material selected for use as a neutron shield is a cementitious shop castable, fire resistant material with a high hydrogen content which is designed for use in shielding doors, hatches, plugs, and other nuclear applications. The solid neutron shield material used in the cask outer annular cavity, and temporary shield plugs, produces water vapor and a small quantity of non-condensible gases when heated above 212°F. The off-gassing produces an internal pressure which increases with temperature. As the temperature is reduced, the off-gas products are reabsorbed into the matrix, and the pressure returns to atmospheric. The maximum steady state temperature of the material is calculated conservatively for an extreme ambient day with a design basis decay heat load. This temperature, assumed to exist throughout the entire shield, results in an internal cavity pressure. This pressure is well within the design allowable value for the neutron shield cavity. The release of off-gas products does not affect the predicted neutron doses because the hydrogen content assumed in the shielding analysis is conservative.

The gaseous effluents from the neutron shield material in the neutron shield cavity of the cask at temperatures greater than 212°F will not be vented. The neutron shielding cavity of the cask is equipped with a rupture disk with a burst pressure of 75 psig. The worst case average temperature of neutron shield under all conditions of storage operation is less than 230°F. The maximum pressure increase caused by the gaseous effluents at 240°F is 21.1 psig which is less than the burst pressure of the rupture disk. Therefore, the gaseous effluents from the neutron shield will not be vented out of the neutron shielding cavity. Manufacturer's test data, Bisco Products Technical Report NS-3-029, "Moderate Temperature (250°F) Weight Loss of NS-3," shows that the gaseous effluents are mostly water vapor which will be reabsorbed into the neutron shield matrix at temperatures below 212°F. Therefore, the effectiveness of the neutron shield will not degrade. However, for conservatism, the shielding analysis is carried out assuming a 10% hydrogen loss from the neutron shield.

The cask is designed to provide adequate shielding to maintain the maximum radiation surface dose to less than 5 REM/hr combined gamma and neutron for a cask drop accident event assuming a complete loss of neutron shielding.

The top cover plate is bolted snug-tight to the top flange of the cask during transport from the plant's fuel building to the ISFSI. The top cover plate assembly consists of a thick stainless steel structural plate which has countersunk scalloped bolt holes, and is attached to the cask top flange with 36 two inch diameter socket head cap screws. Only 12 of the 36 screws are required for on-site transfer. O-ring seals need not be installed during on-site transfer, although either elastomeric or metallic seals may be used.

The cask bottom ram penetration cover plate is a water tight, pressure retaining closure used during fuel loading in the fuel pool, during DSC closure operations in the cask decontamination area, and during cask handling operations in the fuel building. The cask bottom end assembly is welded to the cask shell assembly and includes two o-ring seals for the ram grapple access penetration. The bolted ram access penetration cover plate assembly may be replaced by a two piece neutron shield plug assembly for transfer operations from the fuel building to the ISFSI. At the ISFSI site, the inner shield plug of the neutron shield plug assembly is removed to provide access for the ram and grapple to push the DSC into the HSM. The temporary shield is designed such that the contact dose rate is ALARA.

The cask inner shell is designed to be a pressure retaining vessel. The inner shell is designed to resist the internal pressure due to a postulated DSC rupture at the maximum expected operating temperature as described in Chapter 8. The pressure is contained inside the top cover plate and bottom cover plate (ram access penetration) through the use of metallic o-rings. The o-rings are designed to seal against helium back-fill gas which is used to inert the cask contents during transport and long term storage.

The cask neutron shield cavity is also fabricated as a pressure retaining vessel because it is desirable to have this cavity remain leak tight to prevent intrusion of contaminated spent fuel pool water. To this end, the neutron shield cavity is designed to withstand the expected hydrogen off-gassing from the solid neutron absorbing material. Also, the support members for the outer jacket of the solid neutron shield are angled at about 45° with respect to the cask structural shell to further enhance shielding and decay heat removal.

Two trunnion assemblies are provided in the upper region of the cask for lifting of the cask/DSC inside and outside of the fuel building, and for supporting the cask on the skid for transport to and from the ISFSI. An additional pair of trunnions in the lower region of the cask are used to position the cask on the support skid, serve as the rotation axis during down-ending of the cask, and provide support for the bottom end of the cask during transfer operations.

Alignment of the DSC with the cask is achieved by the use of permanent alignment marks on the DSC and cask top surfaces. These marks facilitate orienting the DSC to the required azimuthal tolerances for fuel loading using the fuel handling bridge in the Fuel Storage Building.

The yoke design used for cask handling is a non-redundant two point lifting device with a single pinned connection to the crane hook as described in Section 4.7.1.2. The yoke is designed in accordance with the requirements of ANSI N14.6 [4.4.5] and NUREG-0612 [4.4.6]. The yoke balances the cask weight between the two trunnions, and has sufficient margin for any minor eccentricities in the cask vertical center of gravity which may occur.

Neither the cask nor the trunnions are special lifting devices per ANSI N14.6. Both upper and lower trunnions are designed to the requirements of the ASME code. The upper trunnions have been evaluated and meet the design guidance, as to acceptable stress level, contained in ANSI N14.6.

Per the Standardized NUHOMS® SAR [4.4.1], the maximum stress intensity is 7.8 ksi. Per the ASME Code, the yield strength for the material (SA-240, type XM-19) is 55 ksi at 100°F and the ultimate strength of the material is 100 ksi at 100°F. Therefore, the guidance provided by ANSI N14.6 of stresses ≤ material yield strength with six times the load and stresses ≤ material ultimate (tensile) strength with ten times the load is met.

Per the Standardized NUHOMS® SAR [4.4.1], the upper trunnions will have a one-time pre-service load test equal to 150% of the maximum working load per ANSI N14.6. Therefore, no additional pre-service load testing is required. In addition, no periodic load testing is required.

The cask is designed to allow an inflatable seal to be inserted between the cask liner and the DSC. The seal is fabricated from reinforced elastomeric material rated for temperatures well above boiling. The seal is placed after the DSC is located in the cask and serves to isolate the clean water in the annulus from the contaminated water in the spent fuel pool. After installation, the seal is inflated to prevent contamination of the DSC exterior surfaces by waterborne particles.

The cask features include internal rails to facilitate DSC transfer. The rails are fabricated from a hardened non-galling wear resistant material coated with a lubricant.

The cask external features include a shear key way designed to resist off-site transportation loads as required by 10 CFR 71 for transportation of canisterized fuel. The shear key way consists of stainless steel bearing blocks and tie bars welded to a stainless steel pad plate which in turn is welded to the structural shell. The shear key way is not a required feature for on-site transfer of DSCs per 10 CFR 72. During DSC loading and prior to transfer operations, a temporary neutron/gamma shield plug is attached to the shear key. The temporary shear key plug is designed such that the contact dose rate is ALARA.

The structural materials and fabrication requirements for the cask are delineated in Volume IV. In general, these requirements are in accordance with the applicable portions of the

ASME Code, Section III, Division 1, Subsection NB for Class 1 Vessels to the extent possible. No N-stamp is required. All structural welds are volumetrically examined and/or tested by the dye penetrant method to the extent possible, as limited by joint configuration and location. These stringent design and fabrication requirements ensure the structural integrity of the cask and the performance of its intended safety function.

The cask is designated important to safety because it provides biological shielding and structural protection for the DSC from impact loads.

4.3 Auxiliary Systems

The Rancho Seco ISFSI is a self-contained, passive storage facility which requires no auxiliary systems. There are no utility or emergency systems required to perform any safety functions at the ISFSI; however, as discussed in Section 1.3, the following ancillary systems are present at the storage site: lighting, closed circuit television (CCTV) intrusion detection, lightning protection, and HSM temperature monitoring.

During fuel loading and cask/DSC transfer operations, the ISFSI requires use of some of the RSNGS auxiliary systems. During DSC drying and sealing operations the Vacuum Drying System (VDS), automatic welding system and the RSNGS waste processing system are utilized. Section 4.3 of the Standardized NUHOMS® SAR [4.4.1] describes these interface requirements of the ISFSI and RSNGS auxiliary systems.

See Appendix B for Standardized SAR, Section 4.3 (pages 4.3-1 to 4.3-3).

The following sections describe the Rancho Seco ISFSI specific auxiliary and utility system requirements.

4.3.1 Ventilation and Offgas Requirements

Spent fuel confined in the DSC is cooled by conduction and radiation within the DSC; and by conduction, convection and radiation from the DSC surface. Air inlets near the bottom of the HSM side walls and outlets near the HSM roof allow convective cooling by natural circulation. The driving force for this ventilation process is thermal buoyancy. The analysis of the HSM ventilation system is described in Section 8.1.3 of the Standardized NUHOMS® SAR. No auxiliary ventilation is used or required at the ISFSI. Fuel loading and DSC closure operations take place in the fuel storage building which uses the existing ventilation system.

There are no off-gas systems required for at the ISFSI. Any off-gas systems required during the DSC drying and backfilling operations use existing plant systems.

4.3.2 Electrical System Requirements

The ISFSI is a passive installation, and there are no operations to control (i.e., no motorized fans, dampers, louvers, or valves; or no electrically operated cranes or lifts). The only utility associated with the Rancho Seco ISFSI for dry SFA storage is non-safety related electrical power for lights, communications, HSM temperature monitoring, security equipment, and general utility. These functions are supportive in nature, and are not needed for effective storage system function.

Electric power is not required to support functions of the Rancho Seco ISFSI that are important to safety. The storage system does not require electric power to perform its function; therefore, loss of electricity will not jeopardize the safety of the facility.

During DSC drying and sealing operations power is required to operate the Vacuum Drying System and automatic welding machine. Power will be provided in the RSNGS spent fuel building to provide for these operations.

The lighting system will consist of light poles capable of illuminating the ISFSI pad and the perimeter fences.

The source of electric power for the ISFSI is from an existing 12 kV line feeding the IOSB through three single phase 12 kV - 480 V pole mounted transformers. The emergency diesel generator associated with the microwave communications building can provide backup electrical power to the ISFSI, if required.

4.3.3 Air Supply System

An air supply system may be used to force water from the DSC during closure operations.

4.3.4 Steam Supply and Distribution System

There are no steam systems used.

4.3.5 Water Supply System

Water is not required at the Rancho Seco ISFSI and none is provided. The storage system does not require a continuous water supply for cooling, makeup, cleaning, or any other reason.

Potable water is not required because the ISFSI is staffed on an infrequent basis by a small number of people during cask handling operations and inspections.

Cask decontamination takes place at the Fuel Storage Building prior to its transfer to the ISFSI.

Fire suppression water is not required because no large credible fire exists.

4.3.6 Sewage Treatment System

There are no sewage treatment systems required for the ISFSI.

4.3.7 Communication and Alarm Systems

The ISFSI is not staffed on a continuous basis. Any instrumentation provided will not be required for safe operation of the ISFSI, and therefore will not be safety-related.

4.3.8 Fire Protection System

As described in Section 3.3.6, no fires are considered credible at the Rancho Seco ISFSI. Therefore, the Rancho Seco ISFSI does not require a fixed fire protection system. However, the ISFSI electrical building includes a fire detection system to monitor the electrical and electronic components but not to satisfy or imply any regulatory requirement for fire protection.

4.3.9 Cold Chemical System

There are no cold chemical systems used at the ISFSI.

4.3.10 Air Sampling System

No air sampling systems are required for the ISFSI. Any airborne activity which may occur during fuel loading and DSC closure operations is monitored by the existing fuel storage building ventilation and radiological detection systems.

4.4 Decontamination System

4.4.1 Equipment Decontamination

No decontamination equipment is required at the ISFSI.

The principal decontamination activity performed in the fuel storage building is the removal of contamination from the outside surfaces of the cask, lifting yoke, and upper end of the DSC shell. Such contamination is due to immersion in the spent fuel pool. To prevent contamination of the DSC exterior surface and the cask cavity by pool water, the annulus between the DSC and cask is filled with clean demineralized water prior to insertion into the pool. The annulus is then sealed closed with an inflatable seal.

Upon withdrawal from the fuel pool, the exterior surfaces of the cask, lifting yoke, and upper end of the DSC are decontaminated prior to proceeding with transfer operations to the ISFSI. Decontamination operations are generally performed in the Fuel Storage Building.

As part of DSC closure operations, the seal is removed and the water in the cask/DSC annulus drained by means of the cask drain. The DSC exterior surface is checked for smearable contamination to a depth of about one foot below the top surface to verify that neither the exterior of the DSC nor the cask cavity has become contaminated. If no smearable contamination has penetrated to this depth, the DSC exterior is presumed to be clean throughout its length. If smearable contamination exceeds administrative limits, then the annulus is flushed with clean demineralized water until acceptable smearable contamination levels are obtained.

Decontaminating the casks after loading fuel is discussed in Section 9.6.2.2 of DSAR, Amendment 4 [4.4.3]. After the cask leaves the Fuel Storage Building, there are no credible mechanisms that could result in contamination of the outside surface of the DSCs, other ISFSI components, or individuals. Therefore, the Rancho Seco ISFSI does not require provisions for decontamination.

4.4.2 Personnel Decontamination

No personnel decontamination facilities are needed at the ISFSI.

Personnel decontamination will be conducted, if necessary, using existing plant equipment and procedures.

4.5 Repair and Maintenance

4.5.1 Repair

No repair operations are anticipated once the DSCs are placed into storage. Periodic maintenance is not required. Maintenance of a minor nature can be performed within the ISFSI area, without the need to move the DSCs.

4.5.2 Maintenance

Major maintenance operations are not required at the Rancho Seco ISFSI. Storage system design features minimize or eliminate the need for maintenance. The DSCs are made of corrosion-resistant stainless steel shells with electroless nickel plated corrosion resistant carbon steel or austenitic stainless steel (optional for FF-DSC) basket assemblies. Other equipment will be specified and selected to withstand the effects of the environment at the site.

Incidental mechanical operations involving storage system components include receiving new DSCs, HSMs, and the cask from the supplier, temporary storage (empty), and DSC and cask transfer to the Fuel Storage Building. During these operations the system components will be inspected and abnormalities evaluated for correction.

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 stainless steel DSC requires no corrosion protection for any foreseeable event. The carbon steel portions of the basket in the DSC are protected from corrosion by a thermally applied metallic coating. This coating protects the basket components for the duration between fabrication and fuel loading. After the DSC is sealed, dried, and backfilled with helium, the basket is maintained in an inert environment and is not subject to corrosion.

The DSC support structure in the HSM is coated carbon steel and requires no additional protection against the expected environment. The HSM heat shield is galvanized for corrosion protection and is painted on one side to enhance radioactive heat transfer.

A detailed discussion of chemical and galvanic reactions within the DSCs, including the potential for hydrogen generation, is provided in Section 2.4.4 of the MP187 transportation SAR [4.4.7].

4.7 Fuel Handling Operation Systems

With the exception of the description of the systems used to load failed fuel in a FF-DSC in a cask, all cask/DSC handling operation systems are similar to those described in the Standardized NUHOMS® SAR. Refer to Section 4.7 of the Standardized NUHOMS® SAR for a description of all other NUHOMS® standard fuel handling operation systems.

See Appendix B for Standardized SAR, Section 4.7 (pages 4.7-1 to 4.7-17).

The only interactions between the ISFSI and RSNGS are those regarding loading the DSCs in the Fuel Storage Building, and handling the cask using the Turbine Building Gantry Crane. Loading and handling of the DSCs will be performed in accordance with applicable RSNGS procedures and 10 CFR 50 license, and is discussed in Chapter 5. Radiation protection of individuals involved in handling spent fuel is addressed in Chapter 7.

Fuel handling and cask loading will be performed using systems and equipment already used for this or equivalent purposes in the Fuel Storage Building. Since cask loading does not present unique handling procedures, equipment contamination and the need for disposal of contaminated equipment is not expected.

Performance objectives during fuel loading are to transfer fuel assemblies from their storage location to the DSCs without damaging the fuel. The District will conduct all operations within and outside the Fuel Storage Building in a manner that does not jeopardize other ongoing activities, present a hazard to the stored fuel, or result in accidental releases of radioactive gases in excess of regulatory requirements.

4.7.1 Individual Unit Description

4.7.1.1 Function

The transfer system function is for moving the loaded cask from the Fuel Storage Building to the ISFSI, and placing a DSC into an HSM. This equipment includes a tractor (prime mover), lifting yoke, lifting yoke extensions[1], support skid, ram trunnion support frame, skid positioning system, transport trailer, hydraulic ram, and auxiliary equipment as described in Section 1.3.2. The components of the transfer system are described in Chapter 5.

4.7.1.1.1 Loading and Unloading

With the exception of the requirements for loading and unloading a FF-DSC, the loading and unloading operating system requirements for the cask and DSCs are similar to those described in Section 4.7 of the Standardized NUHOMS® SAR [4.4.1]. Loading and unloading a FF-DSC (versus a FO or FC-DSC) involves the requirement to place the fuel can

[1] A lifting extension may be a rigid extension or a sling.

body top lids following placement of the failed fuel assemblies into the fuel cans. All other loading and unloading operations for the FF-DSC are similar to those described in the Standardized NUHOMS® SAR.

See Appendix B for Standardized SAR, Section 4.7 (pages 4.7-1 to 4.7-17).

4.7.1.2 Lifting Yoke and Extension

The cask typically uses the standard NUHOMS® transfer system as described by the Standardized NUHOMS® SAR [4.4.1]. The main difference between the certified Standardized NUHOMS® SAR transfer equipment and the Rancho Seco cask transfer equipment is the design of the lifting yoke.

The lifting yoke and extensions provide the means for performing all cask handling operations within, and outside, the Fuel Storage Building. The lifting yoke and extension have a lifting capacity of 130 tons versus 100 tons for the Standardized NUHOMS® SAR yoke and extension. A lifting pin connects the gantry crane hook and the lifting yoke and extension.

The codes and standards used to design and fabricate the lifting yoke are presented in Section 4.7.4 of the Standardized NUHOMS® SAR [4.4.1].

See Appendix B for Standardized SAR, Section 4.7.4 (pages 4.7-10 to 4.7-11).

4.7.2 Design Bases and Safety Assurance

The Standardized NUHOMS® SAR [4.4.1] provides a list of the codes and standards to which the transfer system equipment is fabricated.

See Appendix B for Standardized SAR, Section 4.7.4 (pages 4.7-10 to 4.7-11).

4.7.3 Structural Specifications

The gantry crane and spent fuel handling machine are described in DSAR, Amendment 4 [4.4.3]. The codes and standards for the transfer equipment are described in the Standardized NUHOMS® SAR [4.4.1] Section 4.7.4 and Volume III.

See Appendix B for Standardized SAR, Section 4.7.4 (pages 4.7-10 to 4.7-11).

4.7.4 Installation Layout

The Rancho Seco fuel building is shown in DSAR, Amendment 4 [4.4.3]. The layout of the Rancho Seco ISFSI is discussed in Section 4.1. General layout criteria for the Rancho Seco ISFSI site (including HSM location, multi-purpose cask location, fence location, distance to site boundary, distance to personnel, work areas, etc.) which have radiological dose impact are addressed in Chapter 8.

The propane tank along the transfer route from the fuel building to the ISFSI has been removed. The caustic and acid tanks have been removed. The liquid nitrogen tank and bottles have been removed, and the hydrogen bottles have been disconnected, vented, depressurized, and abandoned.

4.7.5 Individual Unit Descriptions

4.7.5.1 Function

The transport system used to move the loaded DSCs from the Fuel Storage Building to the ISFSI includes the cask (refer to Volume III), Turbine Building Gantry Crane, and transport trailer. All fuel movement is conducted onsite, thus precluding any licensing activities related to 10 CFR 71.

4.7.5.2 Components

The transfer equipment components are described in Section 4.7 of the Standardized NUHOMS® SAR [4.4.1]. The Turbine Building gantry crane is described in DSAR, Amendment 4 [4.4.3]. The casks will be transported from the Fuel Building to the ISFSI using a transfer trailer designed specifically for the storage system.

See Appendix B for Standardized SAR, Section 4.7 (pages 4.7-1 to 4.7-17).

4.7.6 Design Basis and Safety Assurance

All fuel handling equipment is designed in accordance with the codes and standards as required by the RSNGS 10 CFR 50 license. No unique transfer operations are required for the cask loaded with a DSC. All fuel and cask handling operations will be conducted in accordance with approved procedures.

The transfer trailer and associated transfer equipment will have dimensions that will allow cask movement along the transfer path from the Fuel Building to the ISFSI. The transfer trailer is designed to handle a loaded cask.

4.8 References

4.1 "Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.

4.2 Rancho Seco Nuclear Generating Station Emergency Plan, Docket No. 312.

4.3 Rancho Seco Nuclear Generating Station Defueled Safety Analysis Report, Docket No. 50-312.

4.4 ASME Boiler And Pressure Vessel Code, Section III, Division 1, 1992 Edition with Addenda through 1993.

4.5 ANSI N14.6 - 1986 "Special Lifting Devices for Shipping Containers Weighing 10,000 lbs. or More."

4.6 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," August 1980.

4.7 "Safety Analysis Report for the NUHOMS®-MP187 Multi-Purpose Cask," NUH-05-151, Revision 7, Docket 71-9255, Transnuclear West, Inc., August 1998.

4.8 U.S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safeguards, "Safety Evaluation Report Related to the Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel NUHOMS®-24P Submitted by NUTECH Engineers Inc.," NUH-002, Revision 1A, April 1989.

4.9 U.S. Nuclear Regulatory Commission, "Safety Evaluation Report for a Design Change to the Transfer Cask for the Duke Power Company's Independent Spent Fuel Storage Installation," February 1990.

5. OPERATION SYSTEMS

This Chapter presents the tasks required to transfer spent fuel to the Rancho Seco ISFSI. The tasks include preparation of the DSC and cask for fuel loading, closure of the DSC, and preparation for transport to the ISFSI. Other operations include DSC transfer into the HSM, HSM monitoring operations, and DSC retrieval from the HSM which are discussed in Volume II. The NUHOMS® transfer equipment and the existing RSNGS systems and equipment will be used to accomplish these operations.

5.1 Operation Description

The following sections outline the activities for which operating procedures for fuel and DSC handling in the fuel building and on the trailer exist. Procedures are based on the steps described in the Standardized NUHOMS® SAR [5.5.1], that minimize the amount of time required to complete the subject operations, minimize personnel exposure, and assure that all operations required for DSC loading, closure, and transfer are performed safely. Rancho Seco ISFSI procedures are in accordance with the requirements of 10 CFR 50 and 10 CFR 72.24(h).

5.1.1 Narrative Description

The following steps describe the activities for which operating procedures for the NUHOMS® system will exist.

1. Preparation of the Cask and DSC
2. DSC Fuel Loading
3. DSC Drying and Backfilling
4. DSC Sealing Operations
5. Cask Downending and Transport to ISFSI
6. Transferring a DSC into an HSM at the ISFSI
7. Removing fuel from a loaded DSC

5.1.2 Process Flow Diagram

Process flow diagrams for DSC loading and for DSC sealing, draining, and drying operations are presented in Figure 5-1 and Figure 5-2, respectively. Process flow diagrams for placement of a DSC in storage are presented in Volume II, Section 5.1.2.

5.1.3 Identification of Subjects for Safety Analysis

5.1.3.1 Criticality Control

Criticality safety for the NUHOMS® system is assured through a combination of geometrical separation of the fuel assemblies in the FO, FC, and FF DSCs, and the neutron absorbing capability of the internal basket poison sheets for the FO and FC-DSCs. The criticality analysis for the Rancho Seco ISFSI is described in Section 3.3.4.

5.1.3.2 Chemical Safety

There are no hazardous chemicals used in the NUHOMS® system that require special precautions.

5.1.3.3 Operation Shutdown Modes

NUHOMS® is a totally passive system and has no operational shutdown modes.

5.1.3.4 Instrumentation

Table 5-1 shows the typical instruments that will be used to measure conditions or control the operations during the DSC loading, closure, and transfer operations. The instruments are readily available, standard industry equipment.

5.1.3.5 Maintenance Techniques

NUHOMS® is a totally passive system and therefore does not require maintenance. However, to insure that the ventilation airflow is not interrupted, the HSM is periodically inspected to ensure that no debris is in the airflow inlet or outlet openings.

5.2 Fuel Handling Systems

5.2.1 Spent Fuel Handling and Transfer

The Rancho Seco ISFSI is designed to use existing RSNGS systems for handling spent fuel and cask. This section describes the spent fuel handling systems that are unique to NUHOMS® and used during the DSC loading and closure operations.

5.2.1.1 Function Description

Figure 5-3 and Figure 5-4 illustrate the DSC loading and closure operations.

Transfer System

The transfer system is composed of the cask, lifting yoke, support skid, skid positioning system, transport trailer, hydraulic ram, and auxiliary equipment as described in Section 1.3.2. The components of the transfer system used for the operations listed in Section 5.1 are described below. The remaining transfer equipment is described in Chapter 5 of Volume II.

Cask

The cask is used to transfer a loaded DSC to and from the HSM. The cask provides biological shielding during the transfer, and loading operations. Descriptions of the cask's design criteria and features used for HSM loading are discussed in Volumes II and III. The cask is also licensed under 10 CFR 71 for offsite transportation of spent fuel.

Cask Support Skid

The purpose of the cask support skid mentioned in Section 4.7.1.1 is to transport the cask in a horizontal position to the ISFSI and to maintain cask alignment during loading and retrieval operations. The skid is mounted on bearing plates and secured to the transport trailer during transport. These bearings permit the skid to be moved in the longitudinal and transverse directions with respect to the trailer using the skid positioning system mentioned in Section 4.7.1.1, allowing the DSC to be precisely aligned with the DSC support structure inside the HSM. Section 3.1.2.1 establishes the criteria for design of the cask support skid.

Transport Trailer

The function of the transport trailer is two-fold:

1. To transport the loaded cask in the horizontal position to the ISFSI.

2. To approximately align the cask with the HSM opening.

The trailer mentioned in Sections 4.7.1.1 and 4.7.5 is a standard heavy haul trailer capable of handling a 130 ton net payload.

Jack Support System

As the cask or DSC weight is being transferred to or from the trailer, the transfer of the load may cause the trailer deck to move relative to the ground. To prevent this occurrence, jacks at four locations on the trailer are used. The design criteria for the jack support system are established in Section 3.1.2.1 of the Standardized NUHOMS® SAR [5.5.1].

5.2.1.2 Safety Features

During the fuel loading and DSC closure operations the loaded DSC is always seated inside the cask cavity. The safety features used in handling the cask in the fuel building are governed by the RSNGS 10 CFR 50 operating license.

5.2.2 Spent Fuel Storage

Descriptions of the operations used for the transfer and retrieval of the DSC from the HSM are presented in Volume II.

5.2.2.1 Safety Features

The features, systems, and special techniques which provide for safe loading and retrieval operations are described in Section 5.2.1.2 and in Volume II.

5.3 Other Operating Systems

5.3.1 Operating System

NUHOMS® is a passive storage system and requires no operating systems other than those systems used in transferring the DSC to and from the HSM.

5.3.2 Component/Equipment Spares

As discussed in Section 8.2, the Rancho Seco ISFSI is designed to withstand all postulated design basis events. Therefore, no storage component or equipment spares are required after fuel transfer operations are completed. Section 5.4 discusses spare parts for the cask and associated support equipment.

5.4 Operation Support System

The Rancho Seco ISFSI is a self-contained passive system and requires no effluent processing systems during normal storage conditions.

5.4.1 Instrumentation and Control Systems

The instrumentation and controls necessary during DSC loading, closure and transfer are described in Section 5.1.3.4. During DSC storage in HSMs, the HSM roof concrete temperature is monitored with a temperature monitoring system. These systems are described in Volume II for HSM storage.

5.4.2 System and Component Spares

Spare parts for the cask and associated support equipment will be maintained, as appropriate.

5.5 Control Room and/or Control Areas

There are no control room or control areas for the Rancho Seco ISFSI since there are no control systems. However, the instrumentation used to monitor HSM concrete temperature has a readout in a continuously manned location and with a local readout also available in the ISFSI Electrical Building.

5.6 Analytical Sampling

There is no analytical sampling required for the Rancho Seco ISFSI.

If removing fuel from a loaded DSC becomes necessary, a sample of the atmosphere within the DSC will be taken prior to the inspection or removal of fuel.

5.7 References

5.1 "Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.

Table 5-1

Instrumentation Used During NUHOMS® System Loading Operations

Instruments	Function
Gamma/Beta/Neutron Dose Rate Detectors	Measure doses at DSC top shield plug and cover plates
Hydrogen monitors	Monitor for hydrogen generation
Helium detector	Monitor for helium leakage
Pressure and Vacuum Gauges	Measure helium, air, and vacuum pressures inside DSC

Figure 5-1

DSC Loading Operations Flow Chart

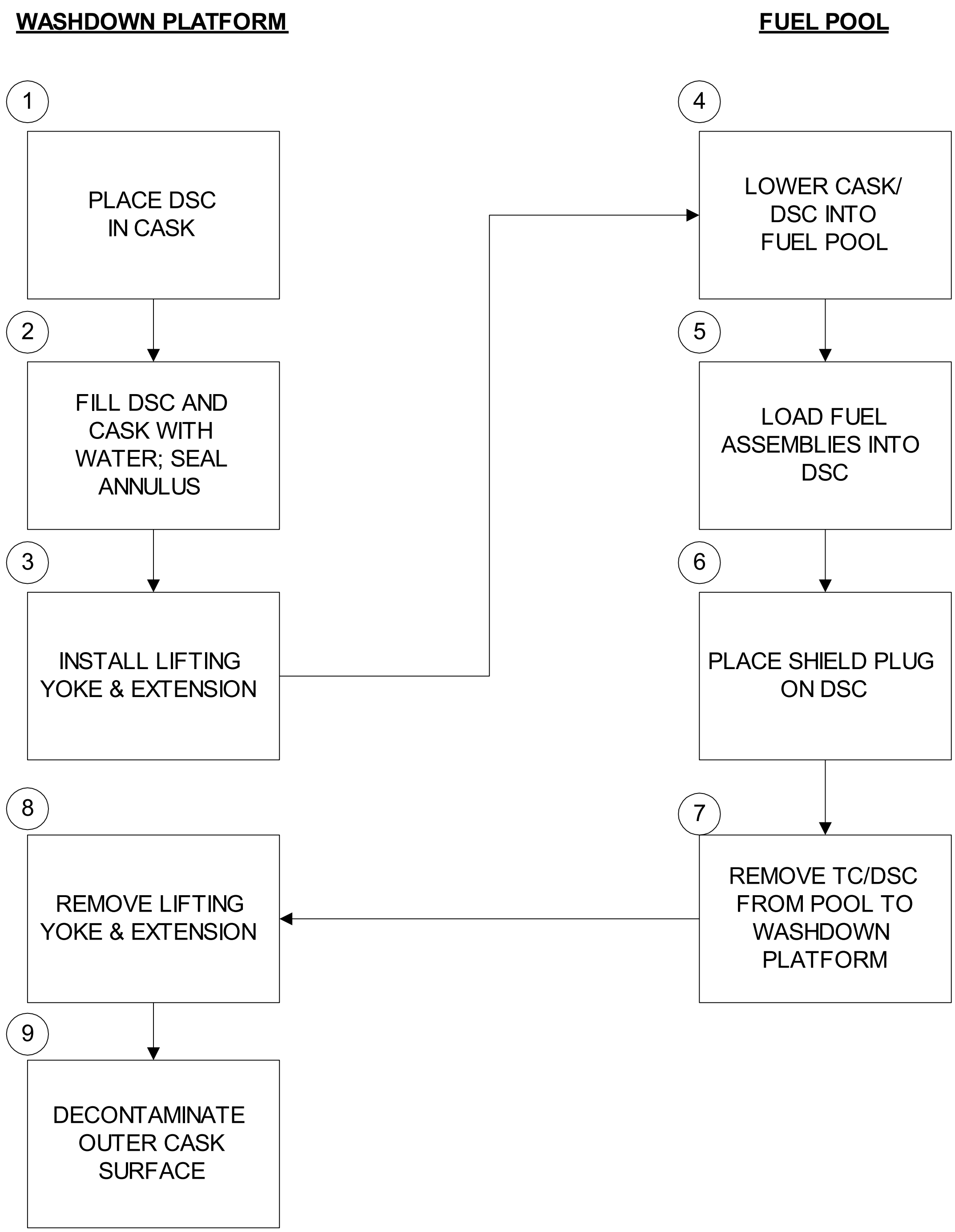


Figure 5-2
DSC Sealing, Draining and Drying Operations Flow Chart

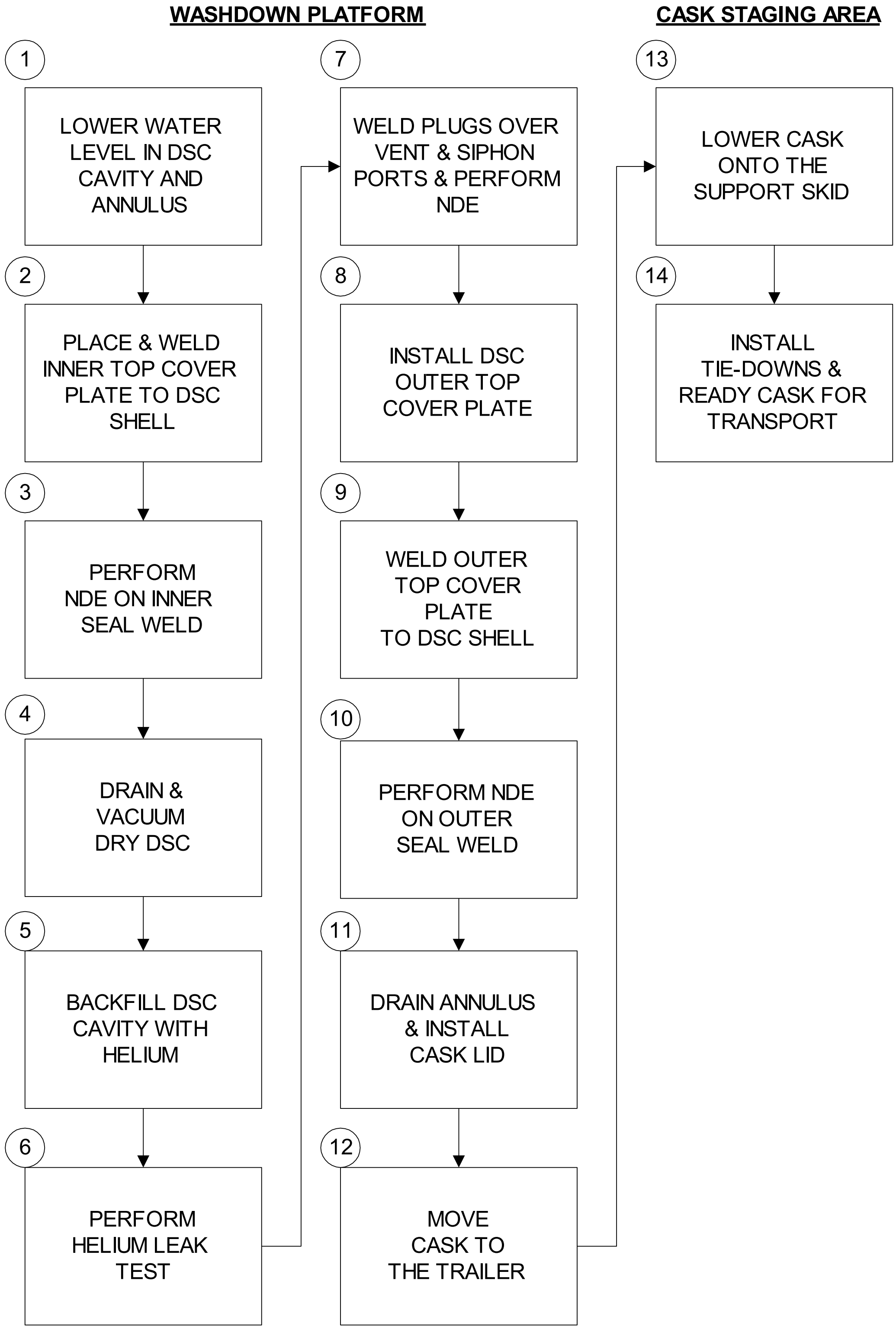


Figure 5-3
<u>Primary Operations for DSC Fuel Handling</u>

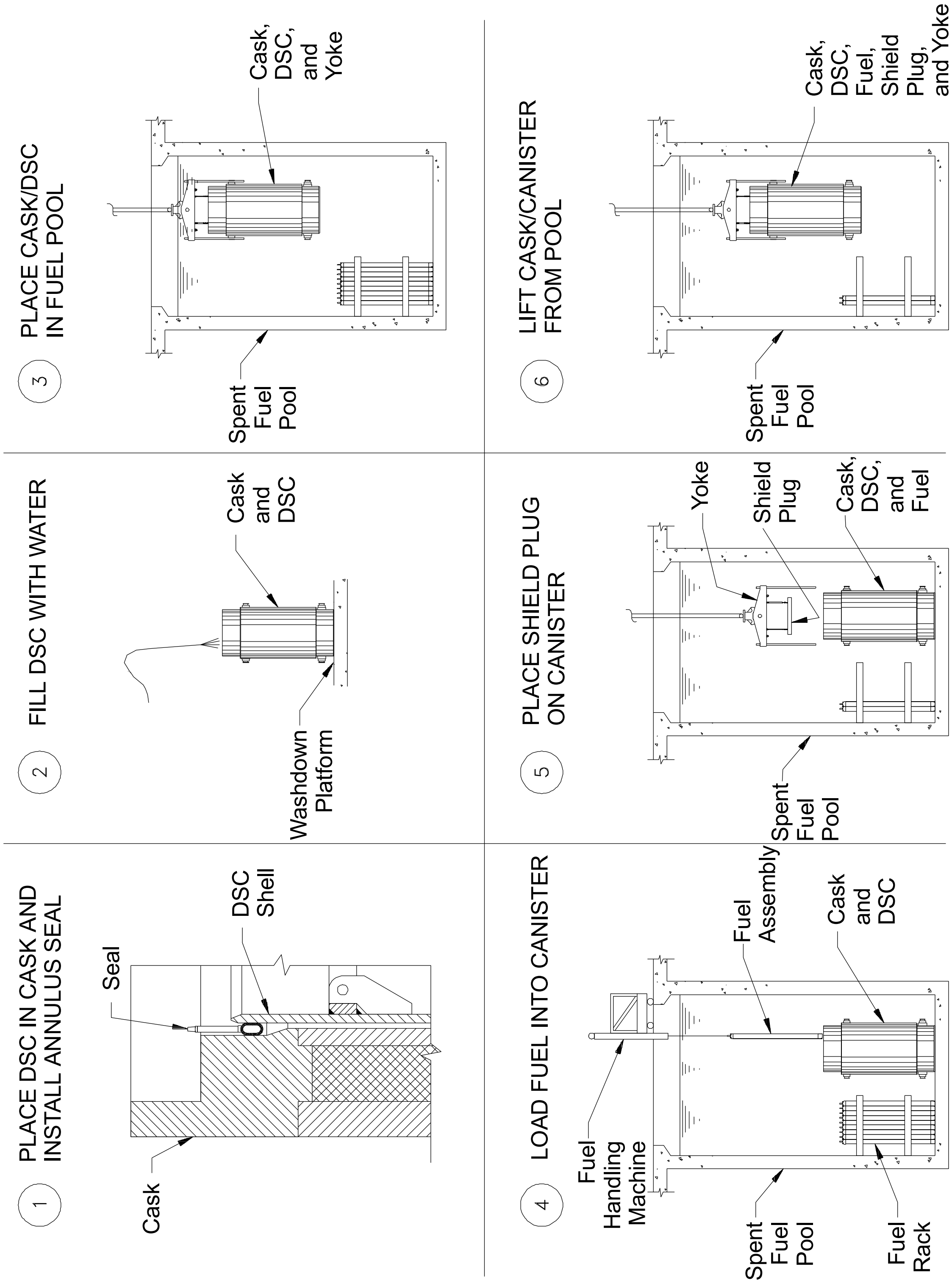
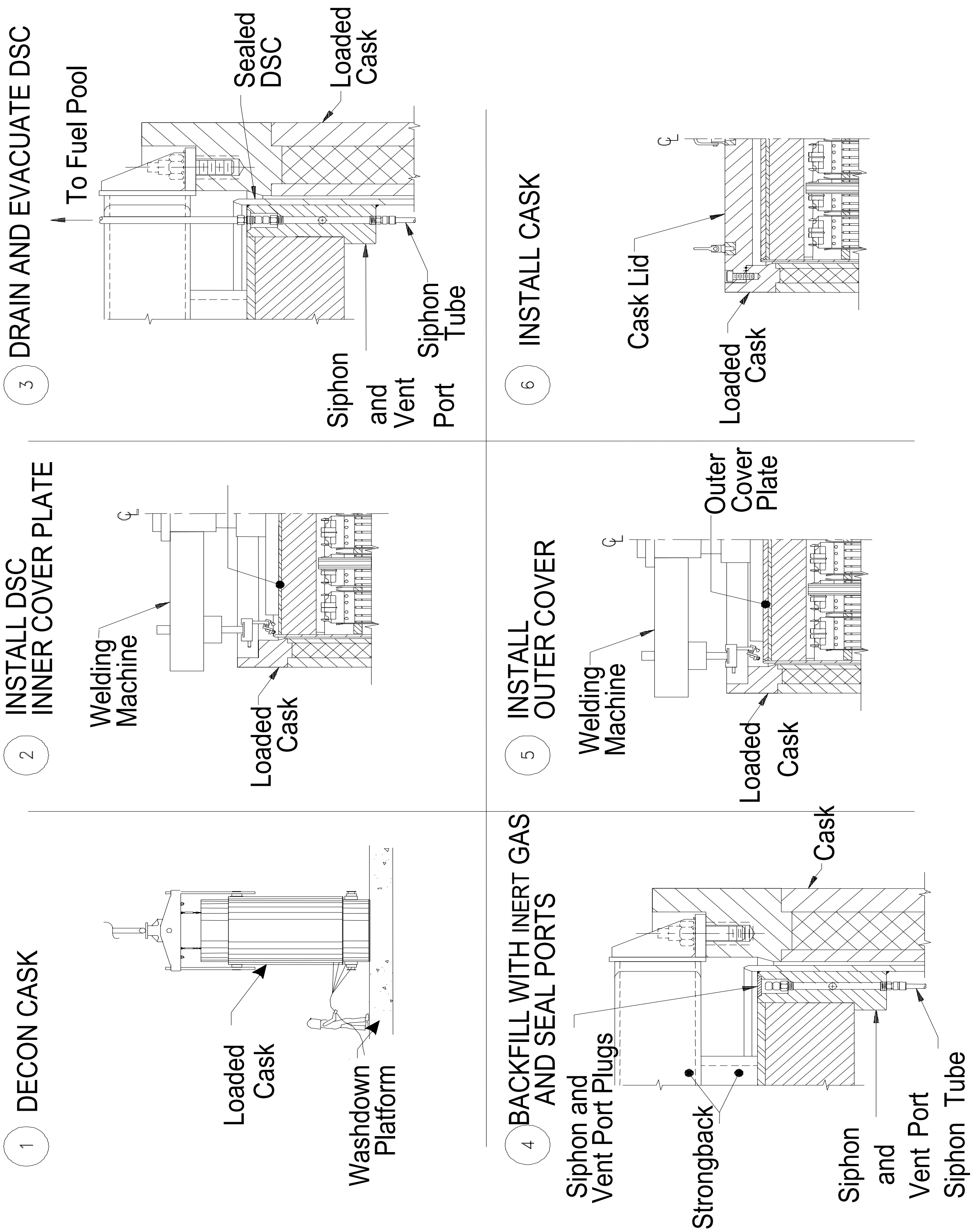


Figure 5-4
Primary Operations for DSC Closure



6. WASTE CONFINEMENT AND MANAGEMENT

6.1 Waste Sources

There are no radioactive wastes generated by the storage of spent fuel at the Rancho Seco ISFSI. The radioactive wastes generated in the fuel storage building during DSC loading, closure, or DSC re-flood during fuel removal operations are handled and processed using existing plant facilities and procedures.

6.2 Off-gas Treatment and Ventilation

There is no radioactive off-gas generated by the storage of spent fuel at the Rancho Seco ISFSI. Potentially contaminated air and helium purged from the DSC during evacuation, or DSC re-flood during fuel removal operations are redirected and processed using existing plant facilities and procedures.

6.3 Liquid Waste Treatment and Retention

There are no liquid wastes generated by the storage of spent fuel at the Rancho Seco ISFSI. The contaminated water purged from the DSC during closure operations may be drained back to the spent fuel pool with no additional processing. A small amount of liquid waste, estimated to be <15 cubic feet, results from decontamination of the transfer cask outer surface following removal from the spent fuel pool. Liquid waste will be processed using plant facilities and procedures.

6.4 Solid Wastes

There are no solid wastes generated by the storage of spent fuel at the Rancho Seco ISFSI. A small quantity of low level solid waste consisting of disposable Anti-C garments, tape, decon clothes, etc., are generated during DSC and HSM closure operations. Solid low level wastes are handled and processed using existing plant facilities and procedures.

6.5 Radiological Impact of Normal Operations - Summary

There are no gaseous, liquid effluents or solid wastes generated by the storage of spent fuel at the Rancho Seco ISFSI. The small volumes of waste generated during DSC loading and closure will have no significant impact on the ability of existing plant facilities to handle and process them.

6.6 References

6.1 "Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.

7. RADIATION PROTECTION

7.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)

7.1.1 Policy Considerations

The policy, programs, and organizational structure for maintaining occupational radiation exposures at the Rancho Seco ISFSI ALARA are the same as those for RSNGS. SMUD's ALARA policy details Rancho Seco management's commitment to maintaining individual and collective occupational radiation exposures ALARA. The ALARA Manual and associated implementing procedures describe the implementation of this policy. The ALARA policy is applicable to all employees at RSNGS and the Rancho Seco ISFSI. The ALARA Manual lists management responsibilities for administration, implementation, and oversight of the ALARA policy and lists the ALARA responsibilities of all plant personnel. The ALARA policy is consistent with 10 CFR 20, and the guidance in Regulatory Guides 8.8 [7.7.1] and 8.10 [7.7.2].

In addition, any changes to the ISFSI design or operating procedures will be evaluated under the provisions of 10 CFR 72.48 which includes an evaluation of the impact of the change on radiation exposures.

External dose to ionizing radiation will be controlled within NRC regulations and Rancho Seco guidelines. Procedures for work in radiological environments will include applicable provisions and requirements for work, commensurate with the radiological environment, in order to maintain exposures ALARA. SMUD conducts training to ensure that all individuals are adequately prepared to work responsibly in a radiological environment and implement the requirements of the ALARA Policy.

7.1.2 Design Considerations

The ISFSI is located approximately 600 feet west of the existing IOSB. SMUD chose this location based on the following ALARA considerations:

1. The ISFSI is located in an area with little ongoing activity such that the increased dose to RSNGS personnel is minimized.
2. The ISFSI is a facility that has limited occupancy and represents a low exposure potential for personnel.

The layout of the ISFSI is designed to maintain exposures ALARA since the HSMs have sufficient separation between them to allow for ease of surveillance operations.

The equipment design considerations are ALARA since the fuel will be stored dry, inside sealed, heavily shielded HSMs. The heavy shielding will minimize personnel exposures.

The DSCs will not be opened nor will fuel be removed from the DSCs while at the ISFSI. Storing fuel in DSCs eliminates the possibility of leakage of contaminated liquids. Gaseous releases are not considered credible. The exterior of the casks will be decontaminated to site administrative limits before transfer to the ISFSI. The required maintenance and surveillance of the HSMs will be minimal and therefore ALARA. This method of spent fuel storage is also considered ALARA because it minimizes direct radiation exposures and minimizes the potential for contamination incidents.

Regulatory Position 2 of Regulatory Guide 8.8 [7.7.1] provides guidance regarding facility and equipment design features. This guidance is being followed as described below:

1. Regulatory Position 2a regarding access control is met by use of a fence with a locked gate that surrounds the ISFSI and prevents unauthorized access once a loaded DSC is placed in the ISFSI.
2. Regulatory Position 2b regarding radiation shielding is met by the shielding provided by the cask and HSMs, which minimizes personnel exposures.
3. Regulatory Position 2c regarding process instrumentation and controls is met since there are no radioactive systems at the ISFSI. No process controls are required for the ISFSI.
4. Regulatory Position 2d regarding control of airborne contaminants is met because no gaseous releases are expected. No significant surface contamination is expected because the exterior of the casks will be decontaminated to meet the administrative limits before transfer to the ISFSI.
5. Regulatory Position 2e regarding crud control is not applicable to the ISFSI because there are no radioactive systems at the ISFSI that could transport crud.
6. Regulatory Position 2f regarding decontamination is met because the exterior of the cask is decontaminated before being released from the Fuel Storage Building.
7. Regulatory Position 2g regarding radiation monitoring is met because the DSCs are seal-welded. There is no need for airborne radioactivity monitoring since no airborne radioactivity is anticipated. Area radiation monitors will not be required because the ISFSI will not normally be occupied. Dosimetry will be installed to monitor direct radiation. Portable survey meters will normally be used. Personnel dosimetry will be used within the ISFSI, as required by Radiation Protection procedures.
8. Regulatory Position 2h regarding resin treatment systems is not applicable to the ISFSI because there will not be any radioactive systems containing resins.
9. Regulatory Position 2i regarding other miscellaneous ALARA items is not applicable because these items refer to radioactive systems not present at the Rancho Seco ISFSI.

ALARA goals and policy considerations are discussed in the ALARA Manual and in Section 7.1.1

All components of the Rancho Seco ISFSI take full advantage of the design and operational experience gained at similar installations. This includes NUHOMS® testing and storage programs at the H. B. Robinson, Oconee, Davis-Besse, and Calvert Cliffs plants and fuel shipment programs at numerous facilities. This experience has served to improve the efficiency of and reduce the occupational exposure received from each new NUHOMS® installation.

The design of the DSC and HSM comply with 10CFR72 ALARA requirements. Features of the NUHOMS® system design that are directed toward ensuring ALARA are:

A. Thick concrete walls and roof on the HSM to minimize the on-site and off-site dose contribution from the ISFSI.

B. A thick shield plug on each end of the DSC to reduce the dose to plant workers performing drying and sealing operations, and during transfer and storage of the DSC in the HSM.

C. Use of a heavy shielded transfer cask for DSC handling and transfer operations to ensure that the dose to plant and ISFSI workers is minimized.

D. Fuel loading procedures, which follow accepted practice and build on existing experience.

E. A recess in the HSM access opening to dock and secure the transfer cask during DSC transfer to reduce direct and scattered radiation exposure.

F. Double seal welds on each end of DSC to provide redundant containment of radioactive material.

G. Placement of demineralized water in the transfer cask/DSC annulus, then sealing the annulus to minimize contamination of the DSC exterior and the transfer cask interior surfaces during loading and unloading operations in the fuel pool.

H. Use of a heavy shielded door for the HSM to minimize direct and scattered radiation exposure.

I. Use of a passive system design for long term storage that requires minimal maintenance.

J. Use of proven procedures and experience to control contamination during canister handling and transfer operations.

K. Use of water in the DSC cavity during placement of the DSC inner seal weld to minimize direct and scattered radiation exposure.

L. Use of water in the cask/DSC annulus during DSC closure operations to reduce radiation streaming through the annulus.

M. Use of temporary shielding during DSC draining, drying, inerting and closure operations as necessary to further reduce the direct and scattered dose.

Further ALARA measures may be implemented, as necessary.

7.1.3 Operational Considerations

Consistent with SMUD's overall commitment to keep occupational radiation exposures ALARA, specific plans and procedures are followed by personnel to ensure that ALARA goals are achieved consistent with 10 CFR 20 and the intent of Section C.1 of Regulatory Guides 8.8 [7.7.1] and 8.10 [7.7.2]. Since the ISFSI is a passive system, no maintenance is expected on a normal basis. Maintenance operations on the cask, transfer equipment, and other auxiliary equipment is performed in a very low dose environment during periods when fuel movement is not occurring.

7.2 Radiation Sources

7.2.1 Characterization of Sources

The radioactive material to be stored in the Rancho Seco ISFSI consists of the RSNGS inventory of B&W 15x15 fuel assemblies, the associated control components, internal startup sources, and miscellaneous fuel structures. The ORIGEN2 computer code [7.7.4] was used to calculate [7.7.8] the worst case neutron and gamma-ray source terms for any assembly and control component in the Rancho Seco fuel pool, assuming that the ISFSI becomes operational after June 1996

The fuel assembly with the largest neutron source term is a 3.18 weight percent U-235 initial enrichment, 38,268 MWd/MTU burnup assembly cooled for 13 years. The fuel assembly with the largest gamma-ray source term is a 3.21 weight percent U-235 initial enrichment, 34,143 MWd/MTU burnup assembly cooled for 7 years. The control component with the largest gamma-ray source term is an axial power shaping rod assembly. These maximum neutron and gamma ray source terms were combined to form a composite design basis fuel assembly for use in all shielding calculations. The neutron and gamma-ray source strengths and spectra are given in Table 7-1 and Table 7-2, respectively.

The primary neutron source in the spent fuel assemblies is due to the spontaneous fission of Cm-244. The neutron spectrum shown in Table 7-1 is therefore the Cm-244 spontaneous fission spectrum [7.7.5].

In addition to the radioactive material associated with the RSNGS spent nuclear fuel, a sealed source of Sr-90 will be stored in the radioactive materials storage building located within the ISFSI controlled area. The source consists of 200µCi of SrO in ceramic.

7.2.2 Airborne Radioactive Material Sources

The release of airborne radioactive material is addressed for the following operations:

1. Fuel handling in the spent fuel pool
2. Drying and sealing of the DSC
3. DSC transfer and storage
4. Removing fuel from a loaded DSC

Potential airborne releases from irradiated fuel assemblies in the spent fuel pool are discussed in DSAR, Amendment 4.

DSC drying and sealing operations are performed using procedures which preclude airborne leakage. Once the DSC is dried and sealed, there are no design basis accidents that could result in a breach of the DSC and the airborne release of radioactivity. Design provisions to preclude the release of gaseous fission products as a result of accident conditions are discussed in Section 8.2.8 of the Standardized NUHOMS® SAR.

During transfer of the sealed DSC and subsequent storage in the HSM, the only postulated mechanism for the release of airborne radioactive material is the dispersion of non-fixed surface contamination on the DSC exterior. By filling the cask/DSC annulus with demineralized water, placing an inflatable seal over the annulus, and utilizing procedures which require examination of the annulus surfaces for smearable contamination, the contamination limits on the DSC can be kept minimal. There is no significant possibility of radionuclide release from the DSC exterior surface during transfer.

If it becomes necessary to unload fuel from a DSC, a sample of the atmosphere within the DSC will be taken prior to the inspection or removal of the fuel. Any radioactive gas will be directed through the gaseous effluent system in the Fuel Storage Building.

7.2.3 Sealed Sources

SMUD is committed to operating the Rancho Seco ISFSI in a manner that will ensure proper radiation protection to all employees, contractors, and the public. The radiation protection program is a robust, established program that implemented the requirements of the 10 CFR Part 50 license, which included the storage, control, and use requirements under 10 CFR Part 30.

The radiation protection program implemented during ISFSI operations is the same program that was implemented during RSNGS plant operations, decommissioning, and dry fuel storage transfer operations. Using the Sr-90 source as a check source to verify instrument response does not present any new challenges to the radiation protection program. The existing program is adequate to safely control all of the radiological aspects of passive dry fuel storage.

The Rancho Seco ALARA program is implemented in accordance with the requirements of 10 CFR Part 20 and additional NRC regulatory guidance. The ALARA policy states management's commitment to maintain exposures to workers and the public ALARA. This commitment is implemented by plant administrative procedures. The ALARA program is discussed further in ISFSI SAR, Section 7.1.1.

Location

The sealed source will be stored in a locked safe located inside the Fuel Transfer Equipment Storage Building. The Fuel Transfer Equipment Storage Building is also locked with access controls implemented in accordance with site procedures and monitored by site security.

Controls

Sealed sources are controlled in accordance with established site procedures that discuss the accounting and handling of radioactive sources. Sources are used, transported, and stored in such a way as to minimize personnel exposure.

Each source has a unique control number that is recorded in the source inventory. Each source is logged in and out of its storage location when required for use. The source will be handled only by an ANSI qualified Radiation Protection Technician (RPT).

Use

The Sr-90 source will be used as a check source to verify instrument response before using certain radiation detectors. The source will be kept in its storage location when not in use.

Personnel Qualifications

Rancho Seco staff meet the minimum education and experience standards specified in ANSI N18.1-1971 "Standard for Selection and Training of Personnel for Nuclear Power Plants." The Manager, Rancho Seco Assets meets the minimum qualifications for Radiation Protection Manager specified in Regulatory Guide 1.8, September 1975. All RPTs are ANSI qualified.

ISFSI SAR Section 9.3 discusses training programs for staff personnel. Retraining and replacement training and records of the qualifications, background, training, and retraining of each member of the organization are maintained in accordance with established programs.

Leakage Testing

The semi-annual leakage testing and the annual inventory are both required by the Rancho Seco Quality Manual (RSQM). All of the sealed sources are required to be tested to verify that there has been no significant loss of integrity of encapsulation during shipment and use . The source leakage acceptance criterion is that a tested source has less than 0.005 micro-curies of removable contamination during the leak test. The sealed source leakage testing and the accountability inventory are performed by ANSI-qualified RPTs.

7.3 Radiation Protection Design Features

7.3.1 Installation Design Features

The design considerations listed in Section 7.1.2 ensure that exposures to radiation are ALARA. All radiation sources are confined within DSCs that are stored in concrete HSMs. The arrangement of the Rancho Seco ISFSI is shown in Chapter 1. The shielding design features of the HSM and of NUHOMS® ISFSIs in general are discussed in Section 7.3.2 of the Standardized NUHOMS® SAR [7.7.3]. The shielding design features of the cask are discussed in Volume III, Section 7.3.2.

See Appendix B for Standardized SAR, Section 7.3.2 (pages 7.3-2 to 7.3-6).

7.3.2 Shielding

7.3.2.1 Radiation Shielding Design Features

Shielding design features of the HSMs and cask are discussed in Section 7.3.2.1 of the Standardized NUHOMS® SAR [7.7.3] and in Volume III, Section 7.3.2.1, respectively. Three types of DSCs are to be stored in the Rancho Seco ISFSI. The shielding design features of these DSCs are discussed below.

See Appendix B for Standardized SAR, Section 7.3.2.1 (pages 7.3-2 to 7.3-3).

The shielding design features of the FO-DSC are similar to those described in the Standardized NUHOMS® SAR [7.7.3], with the exception of the poisoned guide sleeves which have an insignificant effect on the shielding analysis. The shielding design features of the FC-DSC are similar to those described in Appendix H of the Standardized NUHOMS® SAR , again with the exception of the poisoned guide sleeves. The FF-DSC contains 13 fuel assemblies in a DSC shell identical to that of the FC-DSC. The neutron and gamma ray fluxes exterior to the FF-DSC are therefore bounded by those of the FC-DSC.

7.3.2.2 Shielding Analysis

The shielding analysis of the HSM is discussed in Volume II, Section 7.3.2. The shielding analysis of the casks is discussed in Volume III, Section 7.3.2.

7.3.3 Ventilation

The HSM is designed to provide natural-circulation cooling. The following features of the system design ensure that no credible site accident would result in a release of radioactive materials to the environment:

A. The use of a high integrity DSC with redundant seal welds at each end.

B. The passive nature of the system such as the HSM natural convection cooling system which ensures that fuel cladding integrity is maintained.

C. The operational limits and controls placed on DSC loading and closure and transfer operations.

7.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Area radiation and airborne radioactivity monitors are not needed at the Rancho Seco ISFSI. Monitoring devices are used to record dose rates along the ISFSI fence, however.

7.4 Estimated Onsite Collective Dose Assessment

7.4.1 Operational Dose Assessment

This section establishes the expected cumulative exposure received by operational personnel during the DSC loading, closure, and transfer activities associated with placing one DSC into dry storage in an HSM. Chapter 5 describes the ISFSI operational procedures, a number of which involve radiation exposure to personnel. The following discussion of the NUHOMS® operational doses is excerpted from Section 7.4.1 of the Standardized NUHOMS® SAR [7.7.3].

> *This SAR section establishes the anticipated cumulative dose exposure to site personnel during the fuel handling and transfer activities associated with utilizing one NUHOMS® HSM for storage of one DSC. Chapter 5 describes in detail the NUHOMS® operational procedures, a number of which involve potential 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 can be calculated by estimating the number of individuals performing each task and the amount of time associated with the operation. The resulting man-hour figures can then be multiplied by appropriate dose rates near the transfer cask surface, the exposed DSC top surface, or the HSM front wall. Dose rates can be obtained from the Section 7.3 results of dose rate versus distance from the cask side, DSC top end (with and without the top cover plate and cask lid in place) and HSM front wall.*
>
> *Every operational aspect of the NUHOMS® system, from canister loading through drying, sealing, transport, and transfer is designed to assure that exposure to occupational personnel is as low as reasonably achievable (ALARA). In addition, many engineered design features are incorporated into the NUHOMS® system which minimize occupational exposure to plant personnel during placement of fuel in dry storage as well as off-site dose to the nearest neighbor during long-term storage. The resulting dose at the ISFSI site boundary is to be within the limits specified by 10CFR72 and 40CFR190.*

The occupational dose received during fuel loading, closure, and transfer of the DSC to the HSM is estimated in Table 7-3 and Figure 7-1 [7.7.9]. The number of personnel and the amount of time required for each operation are based on operational experience at other NUHOMS® ISFSIs. The total occupational dose calculated for the placement of one DSC into an HSM is 2.5 man-rem. This result is consistent with previously calculated exposures. Actual experience with the NUHOMS® system as reported in the Standardized NUHOMS® SAR [7.7.3] shows that exposures less than one man-rem are likely.

7.4.2 Site Dose Assessment

Table 7-4, Figure 7-2, and Figure 7-3 provide dose rates at various locations both on and off the Rancho Seco site due to the Rancho Seco ISFSI [7.7.10]. This data was obtained using the methodology discussed below and includes direct and air-scattered neutrons and gamma rays. As shown in Figure 7-2, the 10 CFR 20.1301 unrestricted area dose limit of two mrem/hour is not exceeded at any location on the ISFSI fence.

The Rancho Seco ISFSI site dose assessment was performed using the Monte Carlo transport code MCNP [7.7.5]. The HSM array is modeled as a rectangular solid, with two casks modeled as spheres. Each of the 22 HSMs and two casks are assumed to contain a design basis FC-DSC.[1] This is an extremely conservative assumption as only 21 total DSCs containing spent fuel will be placed in storage of which 18 will contain control components. Additionally, about one-fourth of the Rancho Seco fuel assemblies have total sources less than half that of the design basis assembly.

The MCNP model of the ISFSI uses the HSM average dose rate on each surface of the array and the average dose rate on the surfaces of the casks as surface sources for the transport calculations. The HSM average surface dose rates reported in Volume II, Section 7.3 are used for the site dose assessment. The cask surface area-averaged dose rates are assumed to be 70 mrem/hour and 30 mrem/hour for gamma rays and neutrons, respectively. As discussed in Volume III, Section 7.3.2.2, these values bound the total (neutron plus gamma) predicted average dose rates on the cask surface. Source particles are generated on the HSM and cask surfaces with initial directions following a cosine distribution.

The ISFSI model consists of the ISFSI basemat, soil, and dry air at atmospheric pressure. No credit for self-shielding between the casks and the HSM array was taken. Computer model detectors are placed around the ISFSI fence and at the locations tabulated in Table 7-4. The flux at each modeled detector is converted to dose rate using tabulated flux-to-dose rate factors [7.7.7].

The ISFSI is surrounded by a large open area for operational and security purposes. Access to the ISFSI is restricted such that during storage, no access is allowed within the outer security fence except for security and operational activities. There are no work areas close to the ISFSI. The Fuel Transfer Equipment Storage Building, not routinely occupied, lies outside of the radiologically controlled area of the ISFSI. Dose to workers at RSNGS and other individuals in the unrestricted area due to exposure from the ISFSI is minimal and below regulatory limits.

The annual dose for HSM air inlet vent inspections is estimated to be 1.2 Rem. This value is derived by assuming that one inspector performs an inspection once every day, walking around the HSMs at a distance of 20 feet. The amount of time required for the inspection is assumed to be 10 minutes, and it is further assumed that the inspector is exposed to a dose rate of 20 mrem/hr.

[1] Two loaded casks are modeled in the dose calculation because it was initially postulated that two loaded casks could potentially be stored on the ISFSI pad.

7.5 Health Physics Program

7.5.1 Organization

The Radiation Protection organization is described in Section 11.10 of DSAR, Amendment 4. Qualified individuals will perform radiological surveillance, radioactive waste packaging and shipping, emergency planning, and environmental monitoring. The radiation protection functional responsibilities are:

1. Handling, receiving, storing, and shipping radioactive materials.
2. Monitoring personnel exposure to radioactivity.
3. Maintaining personnel exposure records, reporting exposure histories, and reporting abnormal exposure results.
4. Developing and implementing a program to calibrate equipment used in monitoring exposure and radiological conditions.
5. Implementing the ALARA program.
6. Solving programmatic issues related to operational health physics and radiation protection programs to assure employee and public radiation exposures are maintained ALARA.

As discussed in DSAR, Amendment 4, Section 11.10, all individuals assigned to the Rancho Seco site and all visitors are required to follow established administrative controls for protection against radiation and contamination. Delivery personnel and other visitors (non-badged) requiring access to the radiation controlled area are escorted and provided dosimetry, as required.

The qualifications and experience of Rancho Seco personnel are considered more than sufficient for the operation of the ISFSI because these individuals have gained considerable experience at RSNGS.

7.5.2 Equipment, Instrumentation, and Facilities

The radiation control equipment, instrumentation, and facilities for the Rancho Seco ISFSI will be those of RSNGS. Qualified technicians will conduct radiation surveys with portable instruments during activities at the ISFSI.

As indicated in Section 7.2.2, respiratory protection equipment will not be needed at the ISFSI. Similarly, protective clothing will also not be needed.

A variety of instruments are used to cover the entire spectrum of radiation measurements at RSNGS. These include instruments to detect and measure alpha, beta, gamma, and neutron radiation. Calibration sources, or other appropriate methods, are available to allow for instrument calibration, response checks, maintenance, and repair.

Portable radiation survey and monitoring instruments for routine use are the responsibility of the Radiation Protection Group. These instruments include:

1. Low and high-range beta-gamma survey meters
2. Neutron survey meters

3. Alpha survey instruments

Dosimetry procedures for the ISFSI will be the same as those used for RSNGS and will comply with appropriate regulatory guidance.

7.5.3 Procedures

The methods and procedures for conducting radiation surveys at the ISFSI will be those used at RSNGS. These procedures are maintained consistent with the requirements of 10 CFR 20 and 10 CFR 50, and are adhered to for all operations involving personnel radiation exposure.

Radiation Protection procedures and any required safety evaluations are reviewed and approved in accordance with plant administrative procedures.

The philosophies, policies, and objectives of Radiation Protection procedures are based on, and implement, Federal regulations and associated Regulatory Guides to maintain doses to workers and the public ALARA.

Administrative controls for radiation protection are subject to the same review and approval as those that govern other RSNGS procedures. These procedures include Radiation Work Permits (RWP), control of waste shipment and disposal, and access control. The RWP is an administrative tool used in the Radiation Protection Program at RSNGS and Rancho Seco ISFSI. The RWPs issued for work at the ISFSI will be used to inform workers of the radiological conditions in the area and the requirements for dosimetry and engineering controls. The RWP may be used to delineate job prerequisites, radiological safety practices, or additional requirements as needed. As an exposure tracking device, the RWP provides information necessary to ensure that exposures are kept ALARA. After the fuel has been placed at the ISFSI, the RWP and other administrative controls will be used to control access to the ISFSI.

Section 7.1 describes the radiation protection and ALARA procedures and planning that will be used for the ISFSI. Complete details are in the Rancho Seco Radiation Control Manual, ALARA Manual, and associated implementing procedures.

Access control will be accomplished by means of a fence with a locked gate surrounding the ISFSI. Control of the keys will be in accordance with appropriate administrative procedures.

7.6 Estimated Offsite Collective Dose Assessment

7.6.1 Effluent and Environmental Monitoring Program

No effluents are released from the ISFSI during operation. Effluents released during DSC loading are treated using existing RSNGS systems as described in Chapter 6. Since no effluents are released from the Rancho Seco ISFSI site, no effluent monitoring program is required. Direct radiation monitoring is discussed in Section 7.3.4.

A radiological release due to a cask drop in the Fuel Storage Building is discussed in DSAR, Amendment 4.

7.6.2 Analysis of Multiple Contribution

As shown in Table 7-4, the predicted annual dose equivalent at both the Rancho Seco site boundary and at the nearest residence (assuming 2080 hours per year at the boundary and 100% occupancy at the nearest residence) is well below the 10 CFR 72.104 and 40 CFR 190 limits of 25 mrem.

In accordance with NRC guidance, a normal operation confinement evaluation was performed assuming that fission products and actinides escape all 21 DSCs at the leak rate specified in Section 10.3.4. The calculated annual exposure, 2.3 mrem, is well below the 10CFR72.104 limit, even when added to the annual dose equivalent due to direct and air scattered radiation from Table 7-4. The doses to the thyroid and other critical organs are also below the 10CFR72.104 limits. It is therefore concluded that the radiation exposure due to the Rancho Seco ISFSI coupled with all other fuel cycle operations will not exceed the regulatory requirements of 10 CFR 72 and 40 CFR 190.

As discussed in Chapter 2, there are approximately 77 permanent residents within a two mile radius of the Rancho Seco site. The collective annual dose due to the ISFSI for this population is conservatively calculated by assuming that all of these persons are located at the closest residence to the ISFSI, 1000 meters away. The collective annual dose assuming 100% occupancy is then less than 15 person-mrem spread over 77 people. Considering the conservatisms in this calculation and the rapid attenuation of neutrons and gamma-rays with distance, the collective dose for the more distant population would be negligible.

The ISFSI restricted area fence will be approximately 350 feet from the edge of the ISFSI pad. The dose rate at this distance will be less than 0.1 mrem/hr. Assuming a conservative occupancy factor of 500 hr/yr, the annual dose to an individual member of the public would be 50 mrem/yr. This dose is below the regulatory limit of 100 mrem/yr in 10 CFR 20.1301.

7.6.3 Estimated Dose Equivalents

Since no airborne effluents are postulated to emanate from the ISFSI, the direct and air-scattered radiation exposure discussed in previous chapters comprises the total radiation exposure to the public. No estimation of effluent dose equivalents is necessary.

7.6.4 Liquid Release

No liquids are released from the Rancho Seco ISFSI.

7.7 References

7.1 "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable (ALARA)," Regulatory Guide 8.8.

7.2 "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," Regulatory Guide 8.10.

7.3 "Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.

7.4 "ORIGEN2-PC - Isotope Generation and Depletion Code - Matrix Exponential Method," CCC-371, Oak Ridge National Laboratory, RSIC Computer Code Collection, January 1987.

7.5 "MCNP4 - Monte Carlo Neutron and Photon Transport Code System," CCC-200A/B, Oak Ridge National Laboratory, RSIC Computer Code Collection, October 1991.

7.6 "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel," Pacific Nuclear Fuel Services, Revision 2A, NUH-002, 1991.

7.7 "American National Standard for Neutron and Gamma-Ray Flux-to-Dose Rate Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society, La Grange Park, Illinois, March 1977.

Reference Calculations

7.8 Radiological Source Term Calculation for Rancho Seco Fuel, VECTRA Calculation Number 2069.0500, Revision 1.

7.9 Rancho Seco NUHOMS® Occupational Exposure Calculation, VECTRA Calculation Number 2069.0503, Revision 2.

7.10 Rancho Seco NUHOMS® Site Dose Calculation, TN West Calculation Number 2069.0502, Revision 4.

Table 7-1

Design Basis Neutron Source Per Assembly and Energy Spectrum [7.7.8]

Group	Upper Energy (MeV)	Neutron Source[2] (n/s)
1	1.49e+01	3.469e+04
2	1.22e+01	1.970e+05
3	1.00e+01	7.686e+05
4	8.18e+00	3.039e+06
5	6.36e+00	7.163e+06
6	4.96e+00	9.697e+06
7	4.06e+00	2.058e+07
8	3.01e+00	1.653e+07
9	2.46e+00	3.878e+06
10	2.35e+00	2.109e+07
11	1.83e+00	3.627e+07
12	1.11e+00	3.084e+07
13	5.50e-01	1.956e+07
14	1.11e-01	2.236e+06
15	3.35e-03	1.127e+04
16	5.83e-04	8.191e+02
17	1.01e-04	5.387e+01
18	2.90e-05	7.782e+00
19	1.01e-05	1.678e+00
20	3.06e-06	2.615e-01
21	1.12e-06	5.764e-02
22	4.14e-07	1.665e-02
Total		1.719e+08

[2] The fixed neutron source spectrum is that of Cm-244.

Table 7-2

Design Basis Gamma-Ray Source per Assembly and Energy Spectrum [7.7.8]

Mean Energy (MeV)	In-Core (γ/s)	Top Nozzle (γ/s)	Bottom Nozzle (γ/s)	Gas-Plenum (γ/s)	Axial Power Shaping Rod(1) (γ/s)
0.010	9.432E+14	1.525E+11	2.485E+11	1.020E+11	1.554E+12
0.025	2.049E+14	2.602E+10	5.530E+10	6.206E+10	2.606E+11
0.038	2.441E+14	1.480E+10	2.734E+10	2.107E+10	1.474E+11
0.058	1.876E+14	1.667E+10	2.710E+10	1.096E+10	1.653E+11
0.085	1.130E+14	6.552E+09	1.066E+10	4.331E+09	6.503E+10
0.125	1.083E+14	2.517E+09	4.172E+09	1.933E+09	2.499E+10
0.225	9.269E+13	8.282E+08	2.519E+09	4.598E+09	8.242E+09
0.375	4.641E+13	2.321E+08	7.368E+09	2.432E+10	2.301E+09
0.575	1.668E+15	1.332E+07	9.009E+09	3.108E+10	1.320E+08
0.850	2.032E+14	4.027E+09	6.799E+09	1.866E+09	9.405E+10
1.250	1.400E+14	5.645E+12	9.141E+12	3.584E+12	5.591E+13
1.750	1.716E+12	3.780E-01	6.608E+00	2.096E+01	1.608E+01
2.250	1.714E+11	2.992E+07	4.845E+07	1.900E+07	2.964E+08
2.750	8.200E+09	9.258E+04	1.499E+05	5.879E+04	9.170E+05
3.500	1.049E+09	2.007E-14	1.475E-13	1.116E-13	8.348E-11
5.000	5.313E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00
7.000	6.125E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9.500	7.036E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Total	3.954E+15	5.869E+12	9.540E+12	3.848E+12	5.824E+13

Notes:

1. Both black and gray APSRAs were evaluated. The gray APSRA (shown) produces bounding dose rates for the in-core case. The black APSRA has a slightly greater top nozzle source and was used for the top nozzle calculations.

Table 7-3

Estimated Occupational Exposure for One HSM Load

Operation	Number of Personnel	Effective Time in Radiation Field (hours)	Total Personnel Dose (mrem)
Spent Fuel Pool Building			
Ready the DSC and Cask for Service(1)	2	0.0	0.0
Place the DSC into the Transfer Cask	3	1.0	6.0
Fill the Annulus and Install the Seal	2	2.0	8.0
Fill the DSC Cavity with Water	1	0.5	1.0
Place the Cask in the Fuel Pool	5	1.0	10.0
Verify and Load the Assemblies in the DSC	3	8.0	48.0
Place the Cask/DSC in the Decon Area	5	2.0	20.0
Cask Decontamination Area			
Cask Decontamination	7	1.0	90.3
Drain Water Above DSC Shield Plug	3	0.25	9.7
Decon the DSC Top Shield Plug(2)	N/A	N/A	N/A
Remove Water from the DSC Cavity	2	0.5	16.3(5)
Setup Welding Machine	2	1.5	72.8(5)
Weld the Inner Cover Plate and Perform NDE(3)	3	6.0	181.2(5)
Drain the DSC Cavity(3)	2	0.5	15.3(5)
Vacuum Dry and Helium Backfill(3)	2	0.5	15.3(5)

Table 7-3 (continued)

Estimated Occupational Exposure for One HSM Load

Operation	Number of Personnel	Effective Time in Radiation Field (hours)	Total Personnel Dose (mrem)
Helium Leak Test the Inner cover Plate	2	1.0	4.0
Seal Weld Vent and Siphon Ports	2	1.5	1095.0
Fitup the DSC Top Cover Plate	2	1.0	75.9
Weld the Top Cover and Perform NDE(3)	5	14.0	549.5
Drain the Annulus	2	0.25	20.7
Install the Cask Lid	2	1.0	56.0
North Laydown Area			
Ready the Skid and Trailer for Service(1)	2	0.0	0.0
Place the Cask onto the Skid	2	0.5	11.6
Install the Ram Trunnion Support and Cask Shielding	2	1.0	116.6
Secure the Cask to the Skid	2	1.0	69.4
ISFSI Site			
Ready the HSM and Ram for Service(1)	2	0.0	0.0
Transport the Cask to the ISFSI(4)	6	1.0	0.0
Position the Cask Close to the HSM(4)	3	1.0	0.0
Remove the Cask Lid	3	1.0	114.0

Table 7-3 (concluded)

Estimated Occupational Exposure for One HSM Load

Operation	Number of Personnel	Effective Time in Radiation Field (hours)	Total Personnel Dose (mrem)
Align and Dock the Cask with the HSM	2	0.25	51.5
Lift the Ram into Position and Align with the Cask	2	0.5	58.3
Transfer the DSC to the HSM(4)	3	0.5	0.0
Lift the Ram onto the Trailer and Undock the Cask	2	0.25	68.3
Install the HSM Access Door	2	0.5	43.9
TOTAL			2829

Notes:

1. This operation is performed away from any significant radiation field.

2. Not applicable. The DSC inner and outer top covers are installed above the top shield plug, which does not, therefore, require decontamination.

3. Monitoring operation - personnel may leave the radiation work area.

4. Workers are assumed to remain at a distance from the cask sufficient to expose them to a negligible dose.

5. Change in total personnel dose due to FC-DSC shield plug modifications.

Table 7-4

Rancho Seco ISFSI Area Dose Rates

Location	Neutron[1] (mrem/hr)	Gamma (mrem/hr)	Total (mrem/hr)
ISFSI Fence (max)	5.32×10^{-1}	1.35×10^{0}	1.89×10^{0}
Aeration Pond	1.01×10^{-2}	5.44×10^{-2}	6.45×10^{-2}
IOS Building	7.04×10^{-3}	3.50×10^{-2}	4.20×10^{-2}
Switch Yard	9.90×10^{-4}	5.00×10^{-3}	5.99×10^{-3}
West Site Boundary[3]	7.69×10^{-4}	3.54×10^{-3}	4.31×10^{-3}
Machine Shop	7.69×10^{-4}	3.54×10^{-3}	4.31×10^{-3}
Fab. Shop	6.35×10^{-4}	3.01×10^{-3}	3.65×10^{-3}
North Site Boundary	3.01×10^{-4}	1.34×10^{-3}	1.65×10^{-3}
T & R Building	1.47×10^{-4}	6.63×10^{-4}	8.10×10^{-4}
PAP Building	1.05×10^{-4}	4.12×10^{-4}	5.17×10^{-4}
Nearest Public Road	1.50×10^{-5}	5.31×10^{-5}	6.81×10^{-5}
Nearest Residence[2]	4.22×10^{-6}	1.42×10^{-5}	1.84×10^{-5}
Next Nearest Residence	1.76×10^{-7}	6.99×10^{-7}	8.75×10^{-7}
Reservoir/Park	1.66×10^{-8}	5.76×10^{-8}	7.43×10^{-8}

Notes:

1. Neutron doses are calculated theoretical values which are below the lower level of detectability.

2. The exposure received by an individual present at the nearest site boundary for 2080 hours in a year is 9.0 mrem. The annual exposure at the nearest residence, assuming 100% occupancy, is 0.16 mrem.

3. To provide additional assurance that the requirements of 10CFR72.104 will be satisfied, the exposure received by an individual at the nearest site boundary assuming 100% occupancy has also been estimated. As described in Reference 7.10, a correction factor of 0.484 was applied to the above dose rates to account for the additional decay time and the actual fuel irradiation parameters relative to those assumed in the design basis source term calculations. The resulting annual exposure assuming 100% occupancy at the site boundary is 18.3 mrem.

Figure 7-1

Occupational Exposure Contribution from Each DSC Loading Operation

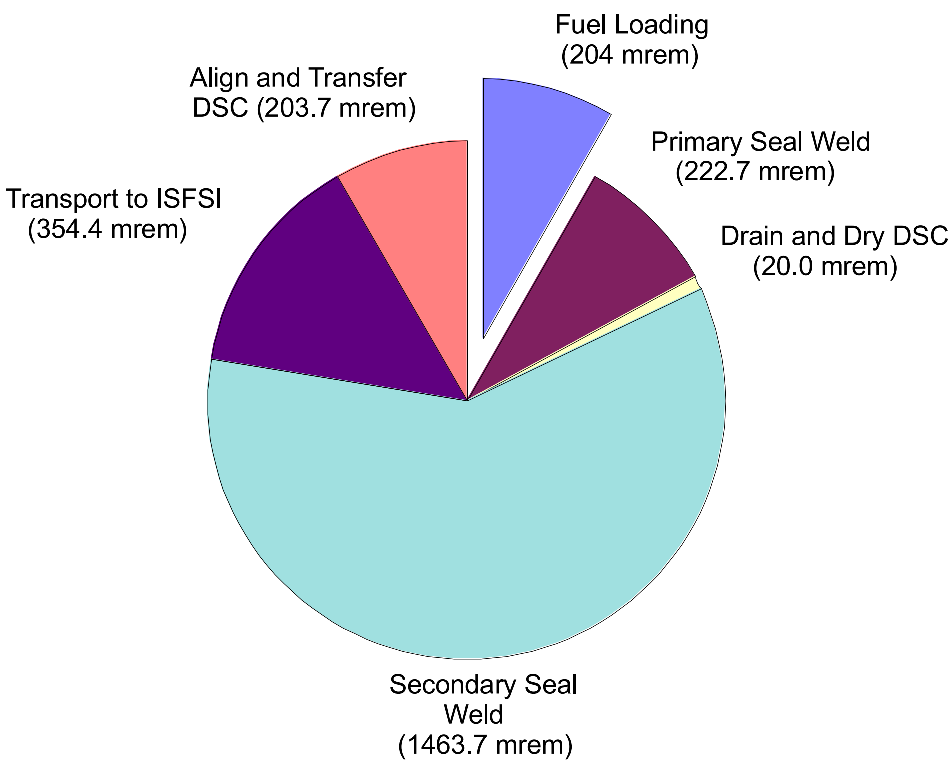
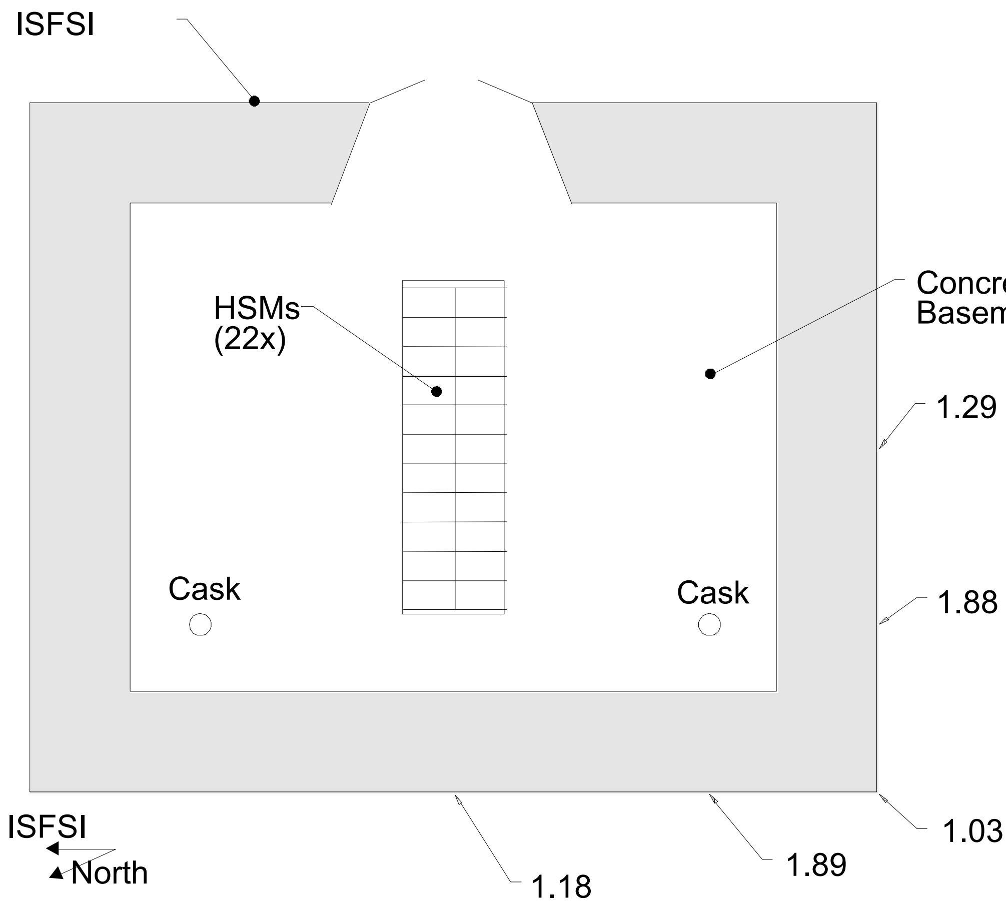


Figure 7-2

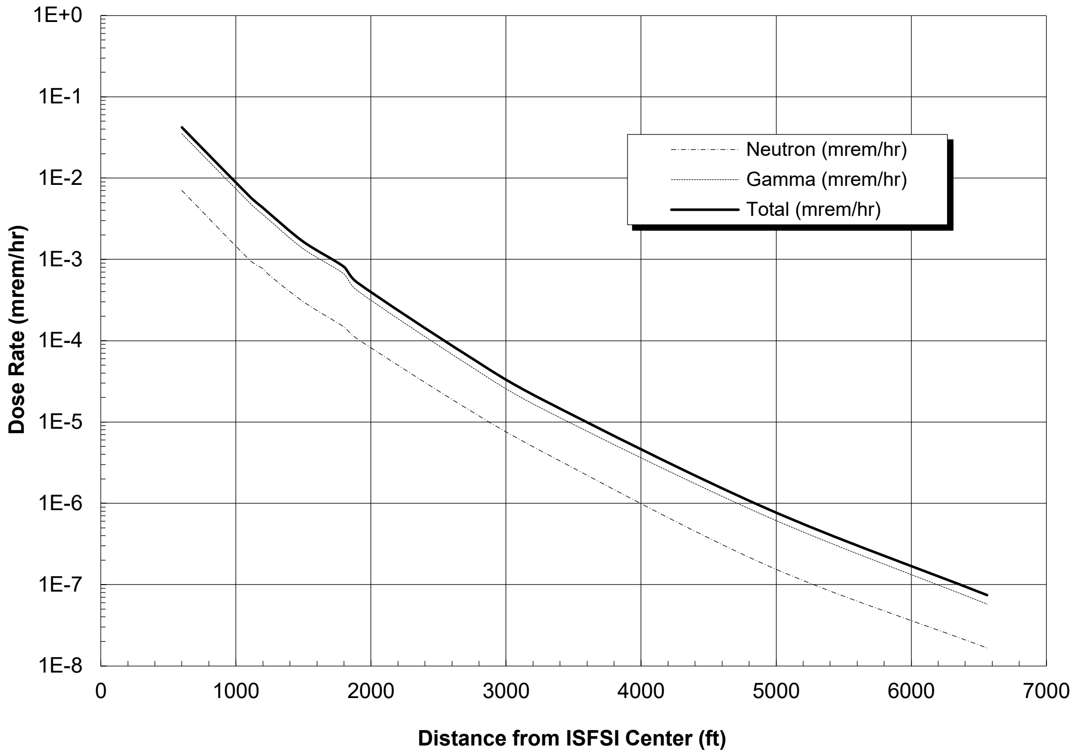
Rancho Seco ISFSI On-Site Dose Rates



Note: The presence of two storage casks was conservatively assumed for the off site dose calculations.

Figure 7-3

Dose Versus Distance from the Rancho Seco ISFSI



8. ANALYSIS OF DESIGN EVENTS

In previous chapters of this SAR, the features of the Rancho Seco ISFSI which are important to safety have been identified and discussed. The purpose of this chapter is to present the engineering analyses for normal and off-normal operating conditions, and to establish and qualify the system for a range of credible and hypothetical accidents.

In accordance with NRC Regulatory Guide 3.48 [8.8.1], the design events identified by ANSI/ANS 57.9-1984 [8.8.2] form the basis for the accident analyses performed for the NUHOMS® system. Four categories of design events are defined. Design event Types I and II cover normal and off-normal events and are addressed in Section 8.1. Design event Types III and IV cover a range of postulated accident events and are addressed in Section 8.2. The load combination evaluation of these events, presented in Section 8.3, provides a means of establishing that the ISFSI design satisfies the applicable operational and safety acceptance criteria as delineated herein.

8.1 Normal and Off-Normal Operations

The normal and off-normal ISFSI operations are addressed in this section. The loads due to normal and off-normal operations include:

1. Dead Weight Loads
2. Design Basis Internal Pressure Loads
3. Design Basis Thermal Loads
4. Operational Handling Loads
5. Design Basis Live Loads

Reference summaries are provided in Table 8-1 and Table 8-2. These loads are described in detail in the following paragraphs.

8.1.1.1 Dead Weight

A description of the dead weight conditions and the methodology used to evaluate the ISFSI system components for the various dead weight conditions are presented in the following paragraphs. The dead weight stresses in the cask, FO-DSC, FC-DSC, and FF-DSC components are summarized in Table 8-3 through Table 8-6, respectively.

1. NUHOMS®-MP187 Cask The effects of dead weight on the cask are evaluated for two cases, as discussed in the following paragraphs. The maximum dead load stresses in the cask components which are important to safety are summarized in Table 8-3.

a. Vertical Handling Dead Load The first case evaluated is for the cask hanging vertically by the two lifting trunnions, and loaded with its maximum payload weight. A bounding weight of 250,000 pounds is conservatively used for this condition.

The stresses in the cask outer shell in the region of the upper lifting trunnions are evaluated using a half symmetry 3-D finite element model. The model includes the cask outer shell and the upper and lower trunnions. The vertical handling dead weight load is applied to the upper trunnion at the location of the interface with the lifting hook. A linear elastic static analysis is performed using the ANSYS program [8.8.3] to determine the stresses in the cask outer shell for the vertical handling dead load. The maximum primary membrane, membrane plus bending and primary plus secondary stress intensities in the cask outer shell, occurring in the region of the upper trunnions, are 2.8 ksi, 9.1 ksi, and 14.8 ksi, respectively [8.8.9].

The cask inner shell stresses resulting from meridional and circumferential (hoop) tensile forces are determined for the cask dead weight and hydrostatic pressure loads using hand calculations. The resulting maximum primary membrane stress intensity in the cask inner shell is 0.4 ksi [8.8.9].

The stresses in the bottom end closure forging due to the vertical handling dead load are calculated for a simply supported annular plate subjected to a uniform pressure load using hand calculations. The weight of the heaviest DSC filled with water less the weight of the DSC cover plates is distributed as a uniform pressure load to the bottom end closure forging. The maximum bending stress in the bottom end closure forging due to the vertical handling dead weight loading is 1.0 ksi [8.8.9].

The ram access cover plate is loaded only by its own self weight in addition to the hydrostatic pressure from the water in the cask annulus for the vertical handling dead load. The bending stress in the ram access cover plate is calculated for a simply supported circular plate under uniform pressure loading using hand calculations. The maximum bending stress in the ram access cover plate due to the vertical dead weight loading is negligible.

b. Horizontal Transport Dead Load The second dead weight load case evaluated for the cask includes the loaded cask resting in a horizontal position on the support skid and transport trailer. In this orientation, the weight of the cask is shared between the lower support trunnions and the upper lifting trunnions resting in the pillow block supports of the support skid. A bounding weight of 239,700 pounds is conservatively used to evaluate the cask dead weight stresses for this condition.

The cask inner shell stresses for this condition are determined using hand calculations. The inner shell is conservatively assumed to support the entire weight of the cask lead shielding in addition to its self weight. The beam bending stresses in the cask inner shell are calculated assuming fixed ends. The maximum membrane plus bending stress intensity in the cask inner shell due to the horizontal transport dead load is 1.32 ksi [8.8.9].

The cask outer shell beam bending stresses resulting from the horizontal transport dead weight loads are calculated treating the cask as a beam supported at the upper and lower trunnion locations. The maximum membrane plus bending stress intensity in the cask outer shell due to beam bending behavior is 0.5 ksi [8.8.9]. The local stresses near the junctions between the cask outer shell and the trunnion sleeves are determined using a half symmetry 3-D finite element model. The model includes the cask outer shell and the upper and lower trunnions. The horizontal transport dead weight load is applied to the upper and lower trunnions at the location of the interface with the on-site skid pillow blocks. A linear elastic static analysis is performed using the ANSYS program [8.8.3] to determine the stresses in the cask outer shell for the horizontal transport dead load. The maximum primary membrane, membrane plus bending, and primary plus secondary stress intensities in the cask outer shell, occurring near the trunnions, are 1.0 ksi, 3.8 ksi, and 6.3 ksi, respectively [8.8.9].

The stresses in the bottom end closure forging and top corner forging are conservatively assumed equal to those calculated for the cask inner shell. The stresses in all other cask components are insignificant for the horizontal transport dead loads.

2. <u>FC-DSC</u> The FC-DSC shell assembly is very similar to that of the NUHOMS®-24P DSC shell analyzed in the NUHOMS Topical Report [8.8.4]. The stresses in the FC-DSC components due to dead weight loads, which are not unique to HSM storage, are evaluated for two cases: 1) DSC in the vertical orientation inside the cask, and 2) DSC in the horizontal orientation inside the cask. A description of the evaluations performed for the FC-DSC dead weight analyses are included in the following paragraphs. The stresses in the FC-DSC components which are important to safety are summarized in Table 8-5.

    a. <u>Vertical Dead Load</u> The dead weight stresses in the FC-DSC shell assembly and basket assembly components are calculated for the DSC in the vertical orientation. The vertical dead weight stresses result during both handling operations in the fuel building and during storage operations at the ISFSI. Only the most limiting conditions are considered in the FC-DSC vertical dead weight analysis.

There are two separate analyses performed for the FC-DSC. The first evaluation consists of a three dimensional finite element analysis of the shell assembly (including shell, cover plates, support ring, and grapple assembly). There are two models developed for the shell assembly analysis - a top end model and a bottom end model. For each model, a 90° representation (quarter symmetry) is used for the evaluation of axisymmetric loadings, such as the vertical deadweight, and a 180° representation (half symmetry) is used for the evaluation of non-axisymmetric loadings. The vertical dead load is determined by including the density of each component in the model and determining the stresses in the various components using a linear elastic analysis. The basket assembly and fuel assemblies bear on the bottom shield plug assembly and the weight of these components is not required for evaluation of the shell assembly. The welds between the top and bottom closure plates and the DSC shell are included in the finite element model. The cover plates are assumed to be pinned to the DSC shell so that the maximum bending stresses in the cover plates can be determined, and a separate evaluation considers the cover plates as fixed to the shell to determine the maximum bending stress in the shell. This bending stress is treated as a secondary stress. The shield plug assemblies are evaluated by classical closed form solutions.

The second evaluation consists of a three dimensional finite element analysis of the basket assembly (including support rods, support rod spacers, and spacer discs). There are two models developed for the basket assembly. The first model is a 90° representation (quarter symmetry) of the spacer discs and support rods and the second model is a 180° representation (half symmetry). The vertical dead load is determined by including the density of each component using a linear elastic analysis. The guide sleeves are evaluated separately by closed form solutions. The guide sleeves are assumed to support the weight of the neutron absorber panels in addition to their self weight, neglecting buoyancy effects encountered during handling operations in the fuel pool. The weight of the fuel assemblies bear on the bottom shield plug assembly and the weight of these components is not required for this evaluation.

The maximum primary membrane, membrane plus bending and primary plus secondary stress intensities in the FC-DSC shell and basket assemblies for the vertical dead load are shown in Table 8-5.

b. Horizontal Dead Load The stresses in the FC-DSC shell assembly and basket assembly components are determined for the DSC in the horizontal orientation inside the cask and HSM. The DSC is supported by the cask rails located at 30° on either side of the 180° azimuth. The horizontal dead load evaluation is performed using the finite element models described for the vertical dead load previously.

For the shell assembly, the horizontal model includes the cask and HSM rails so that local stresses due to this configuration can be determined. The stresses are determined by including the density of the material in the model. A separate model of the spacer disc, shell and cask is developed to determine the effects of the spacer disc on the shell for the horizontal loading. Linear elastic analysis is performed for the horizontal deadweight loading conditions.

The basket assembly evaluation uses the models described previously for the evaluation of the support rods. A separate model of the spacer disc is developed to determine the stresses in the spacer disc due to in-plane loadings (horizontal and thermal). The FC-DSC support rods support only their own self weight for the horizontal dead load. The maximum stress intensities in the support rods are calculated assuming a continuous beam along the length of the basket assembly using the model of the support rods and spacer discs.

Linear elastic analysis is performed for the horizontal deadweight loading condition. The FC-DSC spacer discs are loaded by the weight of the guide sleeve assemblies, 24 PWR fuel assemblies (with control components), support rods and their own self weight. The maximum load on any single FC-DSC spacer disc, based on the spacer disc tributary widths, is 2,060 pounds. The single spacer disc model with linear elastic analysis techniques is utilized for this evaluation.

3. <u>FO-DSC</u> The dead weight stresses in the FO-DSC shell assembly and basket assembly components are evaluated for the DSC in the vertical and horizontal orientations inside the cask. The FO-DSC is similar to the FC-DSC with the exception of the distance which the support rods extend beyond the top spacer disc and the addition of angle iron welded to the top four corners of each fuel sleeve. The FO-DSC components' horizontal and vertical dead weight stresses are the same as those calculated for the FC-DSC basket assembly components, including the guide sleeves [8.8.13].

   The stresses in the FO-DSC components which are important to safety are summarized in Table 8-4.

4. <u>FF-DSC</u> The dead weight stresses in the FF-DSC shell assembly and basket assembly components are evaluated for the DSC in the vertical and horizontal orientations inside the cask.

   The FF-DSC shell assembly configuration is essentially the same as that of the FC-DSC shell assembly. The differences between the two designs consist of the location of the drain and vent ports and the design of the DSC lifting lugs. The minor differences between the two DSC shell assembly designs have no effect upon their structural behavior. Since the weight of the loaded FF-DSC is less than that of the loaded FC-DSC, the dead weight stresses in the FF-DSC shell assembly components are bounded by those calculated for the FC-DSC shell assembly components. Consequently, the vertical and horizontal dead weight stresses for the FC-DSC shell assembly components for the cask handling and storage modes are conservatively used for the evaluation of the FF-DSC shell assembly.

The stresses in the FF-DSC support plates, spacer discs and fuel can bodies for the vertical dead weight condition are calculated by factoring the stress results from the 75g bottom end drop linear elastic analysis by 1/75 [8.8.12].

The horizontal dead weight stresses in the support plate and fuel can bodies are calculated by factoring the 75g side drop linear elastic analysis results by 1/75 [8.8.12].

The FF-DSC spacer disc horizontal transport dead weight stresses are calculated using a half symmetry finite element model. The spacer disc is modeled using quadrilateral solid elements with input thickness, using a plane stress option. The tributary mass of the fuel assemblies and fuel cans are distributed to the supporting spacer disc ligaments. Gap elements are used to model the interface between the spacer disc, DSC shell and cask inner shell. The spacer disc horizontal dead weight stresses are calculated for the 1 g load using the ANSYS program [8.8.3].

The vertical and horizontal dead weight stresses for the FF-DSC in the cask are summarized in Table 8-6.

8.1.1.2 Design Basis Internal Pressure

The range of FO-DSC, FC-DSC, and FF-DSC internal pressures for normal and off-normal operating conditions and postulated accident conditions and the associated helium gas temperatures are shown in Table 8-2. A bounding internal pressure of 10 psig is conservatively applied for the calculation of design basis internal pressure stresses for normal and off-normal operating conditions.

A description of the design basis internal pressure loads and the methodology used to evaluate the ISFSI components for the design basis internal pressure loads are presented in the following paragraphs. The design basis internal pressure stresses in the FO-DSC, FC-DSC, and FF-DSC components are summarized in Table 8-4 through Table 8-6, respectively.

1. FC-DSC The FC-DSC normal internal pressure load acts on the primary containment boundary of the FC-DSC, consisting of the DSC shell, inner top plate, inner bottom plate, and the associated welds.

2. FO-DSC The FO-DSC normal internal pressure load acts on the primary containment boundary of the FO-DSC, consisting of the DSC shell, inner top plate, inner bottom plate, and the associated welds. The Rancho Seco DSC shell assembly is analyzed for a normal internal pressure of 10 psig.

3. FF-DSC The FF-DSC shell assembly is identical to the FC-DSC shell assembly. The FC-DSC internal pressure analysis is performed for a 10 psig internal pressure load. Therefore, the stresses in the FF-DSC shell components due to the design basis internal pressure loading are bounded by the FC-DSC normal internal pressure stresses.

8.1.1.3 Design Basis Thermal Loads

The cask, FO-DSC, FC-DSC, and FF-DSC are subjected to the thermal gradient and thermal expansion loads associated with normal and off-normal operating conditions. The Rancho Seco NUHOMS® components are evaluated for a range of design basis ambient temperatures described in USAR, Appendix 2B.

The thermal analysis of the cask, FC-DSC, FO-DSC, and FF-DSC are presented in Volume III, Section 8.1.1. The temperature distributions derived from the range of normal and off-normal operating conditions encountered in the cask handling modes are considered in the structural analysis of the cask, FO-DSC, FC-DSC, and FF-DSC discussed in the following paragraphs. The temperature distributions for each component are used to determine the effects of thermal stresses and thermal cycling on the components.

The mechanical and thermophysical properties of materials used in the thermal and stress analyses of the ISFSI components are identical to those presented in Table 8.1-3 of the Standardized NUHOMS® SAR [8.8.5].

1. Cask The effects of thermal loads due to differential expansion between dissimilar material, through wall thermal gradients and circumferential thermal gradients in the cask are evaluated to determine the resulting stresses in the cask components which are important to safety during cask handling. The cask thermal stresses due to both differential thermal expansion between the cask inner shell, outer shell, and lead shielding and cask through-wall thermal gradients are determined using an axisymmetric finite element model of the cask. In addition, the effects of cask shell thermal loads due to circumferential thermal gradients are evaluated using a 3-D half symmetry finite element model of the cask shells.

   The controlling temperature distribution, resulting from the 0°F and 101°F ambient normal conditions and -20°F and 117°F ambient off-normal conditions are used for the cask thermal stress evaluations. The maximum stress determined from these analyses is 12.3 ksi [8.8.9], occurring in the top corner forging near the junction with the cask inner shell.

The maximum stresses in the cask inner and outer shells due to the circumferential thermal gradients are 2.0 ksi and 2.8 ksi, respectively [8.8.9]. The total thermal stress for each cask component is conservatively calculated as the absolute sum of the stresses due to the axisymmetric and circumferential thermal distributions, irrespective of location and ambient conditions.

The nominal gaps between the outer surface of the DSC and the cask at room temperature are 0.5 inches and 0.375 inches in the axial and radial directions, respectively. The gaps are provided to ensure that the DSC will not bind during HSM loading and retrieval operations due to combined tolerance buildup and differential thermal expansion between the DSC and cask. As a result, no thermal stresses are induced in the DSC shell or the cask due to thermal expansion of the DSC. This design feature also acts to minimize the effects of thermal cycling and fatigue on the cask and DSCs.

2. <u>FO-DSC and FC-DSC</u> The results of the FO-DSC and FC-DSC thermal analyses, presented in Volume III, Section 8.1.1, show that for the range of normal operating ambient temperature conditions and the maximum fuel decay heat load, the maximum DSC shell temperatures and gradients are significantly lower than those for which the Standardized NUHOMS®-24P DSC components are analyzed. Therefore, the thermal gradients for the NUHOMS®-24P DSC are conservatively used in the thermal stress evaluation of the FO-DSC and FC-DSC.

   The thermal stress analysis of the shell assembly is performed using the models described for the deadweight. The thermal stress analysis for the spacer disc is determined using the single spacer disc model developed for the in-plane loading conditions. Temperature conditions are imposed on the models and the resulting thermal stress conditions are determined. The highest stressed thermal condition is used for the various loading combinations that include thermal stresses.

   Additionally, the effects of differential thermal expansion are considered for the DSC. The differential expansion is evaluated for the longitudinal displacement of the support rod and shell cavity and for the radial displacement of the spacer disc and shell. These evaluations determine that the initial gaps between these components are sufficient to accommodate the thermal expansion of the various components without closing the gaps. Thermal stress results for the FO- and FC-DSC's are presented in Table 8-4 and Table 8-5, respectively.

3. <u>FF-DSC</u> The results of the FF-DSC thermal analysis, presented in Volume III, Section 8.1.1, show that for the range of normal operating ambient temperature conditions and the maximum fuel decay heat load, the maximum DSC shell

temperatures and gradients are significantly lower than those for which the Standardized NUHOMS®-24P DSC components are analyzed.

The effects of the thermal loads due to thermal gradients in the FF-DSC spacer discs resulting from cask storage and handling modes are evaluated using a half symmetry finite element model. The temperature distribution for the horizontal transfer condition with -20°F ambient air results in the maximum thermal gradients in the FF-DSC spacer discs. The FF-DSC spacer disc temperature distribution for this case is imposed on the FF-DSC spacer disc analytical model.

Additionally, the effects of differential thermal expansion are considered for the DSC. The differential expansion is evaluated for the longitudinal displacement of the support plates and fuel cans relative to the shell cavity and for the radial displacement of the spacer disc and shell. These evaluations determine that the initial gaps between these components are sufficient to accommodate the thermal expansion of the various components without closing the gaps. Thermal stress results for the FF-DSC are presented in Table 8-6.

8.1.1.4 Normal Operational Handling Loads

The stresses in the ISFSI components due to normal handling loads are determined for a variety of normal conditions. A description of the normal handling conditions and the methodology used to evaluate the ISFSI components for these conditions are presented in the following paragraphs. The maximum normal handling stresses in the cask, FO-DSC, FC-DSC, and FF-DSC components are summarized in Table 8-3 through Table 8-6, respectively.

1. Cask The major components of the cask affected by the normal handling loads are the structural shell including the top cover plate, bottom end closure forging, the upper and lower trunnions and the structural shell local to the trunnions. There are three normal operating cask handling cases which form the design basis for the cask. These normal handling conditions include critical lift conditions in the fuel building and at the ISFSI, transport handling conditions and HSM transfer handling conditions. A detailed description of the normal handling load conditions, analysis methodology and analysis results are included in the following paragraphs.

    a. Critical Lift Conditions The critical lift handling conditions include the cask lifting conditions inside the fuel building and at the ISFSI. These conditions include: 1) a vertical lift from the fuel pool, 2) positioning and downending of the cask onto the transfer skid and trailer, and 3) lifting the cask in the horizontal position at the ISFSI to transfer it from the on-site transfer trailer to the off-site transportation cradle fixture.

The controlling critical lift condition for the upper lifting trunnions is the vertical lift from the fuel pool. For this condition, the cask is hanging by the upper lifting trunnions, and being handled in an area of RSNGS which requires conformance with the requirements of 10 CFR 50 and ANSI N14.6 [8.8.6]. Accordingly, the upper trunnion allowable stresses are restricted to less than one sixth of the material yield strength and one tenth of the material ultimate strength for critical lifts per ANSI N14.6. The cask handling load is assumed to be shared equally between the two upper trunnions. The design load contains an additional load factor of 15% conservatively applied to the service load to account for the inertial effects of crane hoist motions in accordance with CMAA #70 [8.8.7] recommendations. The cask is designed so that the cask lifting yoke engages the outer most portion of the upper trunnion assembly. During the heaviest lift from the fuel pool, the cask/DSC is filled with water, the DSC top shield plug is in place, and the DSC and cask top cover plates are removed. For this condition the maximum ANSI N14.6 design load for the two upper trunnions due to a vertical lift is conservatively assumed to be equal to the maximum permissible gantry crane load of 260 kips, or 125 kips per trunnion, plus the 15% allowance, or 143.8 kips, acting vertically, with a moment arm measured from the center of the yoke lifting hook to the middle surface of the cask structural shell.

The stresses in the trunnions, trunnion sleeves, trunnion attachment bolts, and trunnion sleeve/cask outer shell welds are determined using hand calculations. The maximum stress intensity in the upper trunnion for this load case is 8.4 ksi [8.8.9] at the junction between the trunnion shoulder and the trunnion bolting flange. This compares with the ANSI N14.6 allowable stress of 9.2 ksi for the trunnion material. The maximum stress intensity in the trunnion sleeve/cask outer shell weld due to the vertical lifting load is 2.6 ksi [8.8.9]. The ANSI N14.6 allowable weld stress is 9.2 ksi. The maximum calculated stress in the upper trunnion attachment bolts is 21.2 ksi. The ANSI N14.6 allowable bolt stress is 25.0 ksi.

The maximum stress in the cask structural shell occurs at the junction with the upper trunnion sleeves. Stresses in the structural shell are calculated using a half symmetry finite element model. The model includes the cask outer shell and the upper and lower trunnions. The vertical lift handling load is applied to the upper trunnion at the location of the interface with the lifting hook. A linear elastic static analysis is performed using the ANSYS program [8.8.3] to determine the stresses in the cask outer shell for the vertical handling load. The maximum local membrane and membrane plus bending stress intensities in the cask structural shell for the vertical lift from the fuel pool are 2.8 ksi and 9.1 ksi, respectively [8.8.9]. The ANSI N14.6 allowable stress intensity value is 9.2 ksi. The maximum primary plus secondary stress intensity in the cask outer shell due to the vertical lift is 14.8 ksi [8.8.9] at the junction of the upper trunnion sleeve and the cask shell. The allowable primary plus secondary stress intensity, in accordance with the requirements of Subsection NC of the ASME code, is 99.9 ksi.

b. On-Site Transfer Handling Conditions The cask on-site transfer handling conditions, which occur during transport of the DSC from the plant's fuel building to the ISFSI, are identical to those discussed in Section 8.1.1.9(B) of the Standardized NUHOMS® SAR [8.8.5]. The normal on-site transfer handling loads consist of dead weight plus or minus a 1g acceleration load in the vertical, lateral, or longitudinal directions, or ±1/2g simultaneously in the vertical, lateral, and longitudinal directions. The allowable stresses for the on-site transfer load cases are governed by the ASME Code.

See Appendix B for Standardized SAR, Section 8.1.1.9(B) (pages 8.1-25 to 8.1-27).

The analysis methodology used to evaluate the cask components for the on-site transfer handling load is identical to that used for the handling loads for the vertical lift from the fuel pool. The maximum transport handling trunnion stress, due to the 134.8 kip dead weight plus 1g vertical handling load, occurs at the junction of the upper trunnion shoulder and bolting flange, and is 8.4 ksi. This compares with an ASME Code allowable primary membrane stress intensity of 33.3 ksi. The maximum membrane plus bending stress intensity in the cask outer shell, in the region near the lower trunnions, is 4.9 ksi. The ASME Code allowable primary membrane plus bending stress intensity is 30.0 ksi for the on-site transport handling conditions.

c. HSM Transfer Handling During transfer of a DSC from the cask to and from the HSM, the cask is restrained to the HSM to prevent any relative motion. The restraint device functions by firmly securing the cask lifting trunnions to embedded anchor points in the HSM front wall. The maximum load exerted on each cask lifting trunnion is approximately one half the maximum hydraulic ram load, or 40 kips. The stresses in the cask outer shell due to the HSM transfer handling load are determined by factoring the stresses calculated for the vertical lift at the fuel pool by the ratio of applied loads. The resulting maximum primary membrane, membrane plus bending and primary plus secondary stress intensities in the cask outer shell for the normal handling load (30k per trunnion) are 0.4 ksi, 1.1 ksi, and 1.9 ksi, respectively [8.9]. The stress intensities for the off-normal trunnion load are 1/3 greater than those for normal operations.

During transfer, the cask rail welds are loaded in shear by the friction of the sliding DSC. At an assumed shear of 80 kips, the stress on the rail welds is 5.8 ksi compared to an allowable value of 9.4 ksi.

2. FO-DSC, FC-DSC and FF-DSC The normal handling conditions for which the FO-DSC, FC-DSC, and FF-DSCs are analyzed include critical lift conditions in the fuel building and at the ISFSI, transport handling conditions, and HSM transfer handling conditions discussed above. The most significant normal handling load for the DSC shell components is the HSM transfer condition.

   The applied force from the hydraulic ram is applied to the DSC assembly at the grapple ring location. A uniform pressure load is applied to the center of the DSC bottom cover plate over an area approximately equal the contact area of the ram grapple assembly to evaluate the DSC insertion loading condition. A uniform pressure load is applied to the inner surface of the DSC grapple ring plate to evaluate the DSC retrieval loading condition.

   The DSC bottom cover plate and grapple ring stresses resulting from the DSC retrieval non-uniform load distribution of the ram grapple are analyzed using the 1/4 symmetry model. The load is applied to the grapple ring plate nodes corresponding to the contact area between the ram grapple arms and the grapple ring plate. The edges of the DSC bottom cover plate are conservatively modeled as pinned. The analysis results show that the load transferred to the DSC shell is uniformly distributed by the bottom cover plate.

   The transport loading provides the most critical conditions on the basket assemblies. This evaluation is performed using the basket models described for the vertical and horizontal deadweight loading conditions. This loading condition is evaluated by applying acceleration loads for the various transport loading conditions of 1g individually applied in the transverse, longitudinal and vertical directions, and 1/2g applied simultaneously in each direction. The controlling FO-DSC, FC-DSC, and FF-DSC normal handling stresses are tabulated in Table 8-4 through Table 8-6, respectively.

8.1.1.5 Off-Normal Handling Loads

The off-normal handling event postulates that the DSC binds or become jammed during HSM transfer operations due to misalignment between the cask and HSM. The systems involved in the jammed DSC event include the cask, DSC shell assembly, HSM, HSM support structure, on-site transfer skid and trailer, and the hydraulic ram. The postulated cause of a jammed DSC, methods provided to detect such an event and corrective actions required to return the system to normal conditions are described in Section 8.1.2.1 of the Standardized NUHOMS® SAR [8.8.5].

See Appendix B for Standardized SAR, Section 8.1.2.1 (pages 8.1-29 to 8.1-32).

As described in Sections 8.1.2.1(B) and (C) of the Standardized NUHOMS® SAR, a ram force of 80,000 pounds is postulated to develop due to axial sticking of the DSC on the cask rails and HSM DSC support rails and due to binding of the DSC. The design basis for the hydraulic ram was maintained at 80,000 pounds, and is conservative for Rancho Seco's slightly heavier DSCs.

The stresses in the cask due to the postulated jammed DSC loads are calculated using the methodology described in Section 8.1.2.1 of the Standardized NUHOMS® SAR. The cask component stresses due to the 80,000 pound off-normal handling load are equal to twice the stresses due to the 40,000 pounds normal HSM transfer handling load. The maximum stresses in the cask, FO-DSC, FC-DSC, and FF-DSC due to the jammed DSC loads are summarized in Table 8-3 through Table 8-6, respectively. The stresses in the DSC and cask due to the postulated jammed DSC loads are demonstrated to meet the ASME Code allowable stresses.

8.1.1.6 Design Basis Live Loads

As discussed in Section 3.2.4 of the Standardized NUHOMS®-24P SAR [8.8.5], a live load of 200 pounds per square foot is conservatively selected to envelope all postulated live loads acting on the cask, including the effects of snow and ice.

Snow and ice loads for the HSM are conservatively derived from ANSI A58.1-1982. The maximum 100 year roof snow load, specified for most areas of the continental United States for an unheated structure, of 5.27 kN/m^2 (110 psf) is assumed. For the purpose of this conservative generic evaluation, a total live load of 9.58 kN/m^2 (200 pounds per square foot) is used in the HSM analysis to envelope all postulated live loadings, including snow and ice. Snow and ice loads for the on-site transfer cask with a loaded DSC are negligible due to the smooth curved surface of the cask, the heat rejection of the SFAs, and the infrequent short term use of the cask.

8.1.1.7 DSC Fatigue Evaluation

Fatigue effects on the DSC are addressed using the criteria contained in NB-3222.4 of the ASME Code [8.8.8]. Fatigue effects need not be specifically evaluated provided the criteria contained in NB-3222.4(d) are met [8.8.16]. The fatigue evaluation demonstrates that the six criteria contained in NB-3222.4(d) are satisfied for all components of the DSC. Therefore, fatigue effects need not be specifically evaluated for the DSC.

8.1.1.8 Cask Fatigue Evaluation

Fatigue effects on the cask are addressed using the criteria contained in NB-3222.4 of the ASME Code [8.8.8]. Fatigue effects need not be specifically evaluated provided the criteria contained in NB-3222.4(d) are met [8.8.9].

The fatigue evaluation [8.8.9] demonstrates that the six criteria contained in NB-3222.4(d) are satisfied for all components of the cask.

8.1.1.9 Thermal Cycling of the Cask

The largest mean daily change of temperature at the Rancho Seco site is conservatively assumed equal to the largest mean daily change of temperature in the United States of 47°F, occurring in Reno, Nevada. Because of the large thermal mass of the cask, a period of approximately 1 day is needed to obtain steady state temperatures and a steady state thermal gradient. For conservatism, it is assumed that the 47°F maximum daily change could produce a steady state gradient every day for 50 years, for a total of 18,250 thermal cycles. From the S-N curve, a temperature fluctuation of up to 49°F, corresponding to 10^6 cycles is acceptable.

Fatigue effects, including that from thermal cycling, were originally analyzed for a design life of 50 years. A new analysis was performed to address the six criteria from ASME B&PV Code, Section III, [8.8]1, Subsection NB-3222.4 affecting fatigue for the DSCs and TC at Rancho Seco. The criteria are: 1) Atmosphere-to- Service Pressure Cycle; 2) Normal Service Pressure Fluctuation; 3) Temperature Difference - Startup and Shutdown; 4) Temperature Difference - Normal Service; 5) Temperature Difference – Dissimilar Materials; and 6) Mechanical Loads. The results of the analysis indicate that the limits of peak stress intensities, as governed by fatigue, have been satisfied by compliance with these six criteria and that no additional analysis is required for the PEO.

8.1.2 Horizontal Storage Module

The normal and off-normal events and the postulated accident events which are unique to the HSM storage mode of operation are addressed in Volume II, Chapter 8.

8.2 Accident Analyses for the ISFSI

Section 8.2 analyzes postulated ISFSI accidents to demonstrate that adequate safety margin exists and that radiological consequences are within regulatory limits. The postulated accidents addressed in this SAR section include:

1. Accidental cask drop
2. DSC leakage
3. Accident pressurization
4. Earthquake
5. Fire

For each postulated accident, the SAR discusses the postulated cause of the event, detection of the event, analysis of the effects and consequences of the event, and appropriate corrective actions. In addition, Section 3.2 discusses the safety criteria for natural phenomena events, including:

1. Tornado wind loadings and tornado generated missiles
2. Flood
3. Seismic design

The results of the analyses discussed above show that adequate safety margin exists for all postulated accidents and natural phenomena events. The only event with radiological consequences is the postulated DSC leakage event. While this is a non-credible event, it provides the bounding case for radiological consequences, and demonstrates that the radiation dose from an accident or natural phenomena event does not exceed the limits in 10 CFR 72.106(b). A summary of affected components for each load type is presented in Table 8-7.

8.2.1 Accidental Cask Drop

For cask-handling activities conducted under the 10 CFR 72 license, the postulated accidental cask drop event is discussed in Section 8.2.5 of the Standardized NUHOMS® SAR [8.8.5]. The results of a cask drop event for activities conducted under the 10 CFR 50 license are discussed in DSAR, Amendment 4.

See Appendix B for Standardized SAR, Section 8.2.5 (pages 8.2-26 to 8.2-42).

8.2.1.1 Postulated Cause of Event

The postulated causes of the accident drop event, accident scenarios and the load definitions are discussed in Section 8.2.5.1 of the Standardized NUHOMS® SAR [8.8.5].

See Appendix B for Standardized SAR, Section 8.2.5.1 (pages 8.2-26 to 8.2-29).

8.2.1.2 Detection of Event

No additional means or methods are required to be provided for the detection of the accidental cask drop event.

8.2.1.3. Analysis of Effects and Consequences

The analyses of the ISFSI components for the postulated accidental cask horizontal side drop, vertical end drop, and corner drop conditions are discussed in the following paragraphs.

1. Horizontal Side Drop The cask and DSC assembly components are analyzed for a 75g equivalent static side drop load. The stability of the system components as well as the stresses are considered in the evaluation.

   a. Cask Horizontal Side Drop The stresses in the cask inner and outer shells due to the postulated 75g side drop are analyzed using a half symmetry finite element model. The model includes the cask inner shell, outer shell, lead shielding material, neutron shield, and the top and bottom end plates. The cask inner and outer shells are modeled using 3-D quadrilateral shell elements. The cask lead shielding and neutron shield are modeled using 3-D brick elements. It is conservatively assumed that no shear transfer exists between the cask shells and the shielding materials. The load due to the weight of the DSC is applied as a uniform pressure load acting over the length of the shell and a width of a single rail. The depth of penetration of the cask into the concrete target, calculated using the modified National Defense Research Committee (NDRC) formula, is 1.65 inches [8.8.9], corresponding to a half angle of contact of 17°. Displacement constraints, representing the contact with the target, are applied to the nodes on the outer surface of the neutron shield along the entire length and the bottom 15° circumferential segment. A 75g horizontal side drop acceleration load is applied to the cask. A linear elastic static analysis is performed using the ANSYS program [8.8.3]. The maximum primary membrane and (local membrane+bending) stress intensities in the cask inner shell due to the 75g side drop are 42.3 ksi and 67.3 ksi, respectively [8.8.9]. The maximum primary membrane and (local membrane+bending) stress intensities in the cask outer shell due to the 75g side drop are 37.7 ksi and 67.1 ksi, respectively [8.8.9].

b. FC-DSC Horizontal Side Drop The spacer discs are evaluated for 0°, 45° and 18.5° (on a single cask rail) 75g horizontal side drops. Finite element models are developed for one half of a typical DSC spacer disc, and an entire spacer disc, in order to analyze the spacer disc for the postulated horizontal side drop conditions. The DSC shell and fixed nodes representing the inner liner of the transfer cask are included in the model. Gap elements are used between the spacer disc, DSC shell and cask liner to accurately capture the interaction of the components. The mass of the PWR fuel assemblies and guide sleeves are applied to the spacer disc ligaments as uniform pressure loads. The transfer cask liner nodes include the cask rails and are offset the thickness of the cask rails for the side drop analysis.

An elastic-plastic analysis of the spacer disc for the side drop conditions is performed to account for local yielding of the spacer disc during the side drop events. Classical bilinear plasticity with a 5% tangent modulus is used to represent the elastic-plastic material properties of the carbon steel spacer disc and stainless steel DSC shell.

Spacer disc buckling is considered for three side drop orientations. The minimum factor of safety against buckling for the 0° side drop is 1.87. The minimum factor of safety against elastic buckling required by the ASME Code [8.8.8] is 1.5. Therefore, the FC-DSC spacer disc meets the elastic buckling acceptance criteria. The analysis basis is similar to that for the Standardized NUHOMS®-24P DSC as quoted below.

> *In addition, an ANSYS bifurcation buckling analysis of the entire spacer disk is performed to evaluate the global buckling behavior and stability of the spacer disk. The spacer disk model shown in Figure 8.2-6 is used to perform this analysis. The spacer disk analytical model permits out-of-plane deformations, and is assumed to be supported both in-plane at the perimeter of the spacer disc that is in contact with the DSC shell, and out-of-plane at the four support rod locations. This analysis showed that out-of-plane buckling is the controlling buckling mode for the spacer disk. A factor of safety of 1.80 against collapse of the spacer disk is calculated for the postulated 75g horizontal drop.*

The stresses in the FC-DSC guide sleeve assembly due to the postulated 75g side drop are calculated using a half symmetry finite element model. The guide sleeve is modeled using 3-D quadrilateral elements. The guide sleeve inner tube is assumed to support its own self weight in addition to the weight of the neutron absorber panels and the poison support sleeves for a side drop. The material density of the guide sleeve inner tube is adjusted to include the weight of the neutron absorber panels and oversleeves, conservatively assuming the neutron absorber panels and oversleeves provide no structural support. A linear elastic static analysis is performed using the ANSYS program [8.8.3] to determine the stresses in the guide sleeve inner tube.The FC-DSC support rods are analyzed for the postulated 75g side drop loading using hand calculations. The FC-DSC component stresses due to the postulated 75g side drop are as per Table 8-10.

c. FO-DSC Horizontal Side Drop The stresses in the FO-DSC shell assembly components due to the postulated 75g side drop event are analyzed in a fashion similar to those for the FC-DSC shell and basket assembly components.

   The stresses in the FO-DSC and FC-DSC basket assembly components, including the guide sleeve assemblies, due to the postulated 75g side drop event are calculated for the enveloping loads.The FO-DSC spacer disc buckling load is the same as the FC-DSC spacer disc buckling load discussed in Section (b) above.

d. FF-DSC Horizontal Side Drop The FF-DSC shell assembly is the same as the FC-DSC shell assembly. The DSC shell assembly is loaded by its own self-weight, the weight of the basket assembly and the weight of the spent fuel for the postulated side drop. The total weight of the loaded FF-DSC is less than that of the loaded FC-DSC. Therefore, the FF-DSC shell assembly side drop loads and stresses are bounded by those calculated for the FC-DSC shell assembly in (b) above.

   The FF-DSC spacer discs are evaluated for 0°, 20°, and 45° (on a single cask rail) 75g horizontal side drops. The 0° side drop analyses is performed using a half symmetry finite element model. The 20° and 45° side drop analyses are performed using a full spacer disc finite element model. The spacer disc is modeled using quadrilateral solid elements with input thickness, assuming plane stress. Gap elements are used to model the interface between the spacer disc, DSC shell, and cask inner shell due to the side drop loading. The tributary mass of the fuel assemblies and fuel cans are modeled on the spacer disc ligament using generalized mass elements. Elastic analyses of the spacer disc for the 0°, 20° and 45° side drop conditions are performed using the ANSYS program [8.8.3].

   The FF-DSC spacer disc elastic stability is analyzed using a half symmetric model for 0° and a full spacer disc finite element model for 20° and 45° 75g side drops. The model consist of 3-D quadrilateral shell elements having three rotational degrees of freedom and three translational degrees of freedom at each node. The spacer disc is conservatively modeled with a 1/2 inch linear offset from the top to the bottom. The spacer disc is supported in-plane at the node along the outer edge in the regions supported by the DSC shell, as shown by the spacer disc 75g side drop analysis results. In addition, out-of-plane displacement constraints are applied to the nodes at the center of the support plates. The results of the FF-DSC spacer disc elastic buckling analyses show the minimum factor of safety against elastic buckling to be 4.0 for the 0° side drop orientation [8.8.12]. This is much larger than the 1.5 minimum factor of safety required by the ASME Code [8.8.8].

The fuel can bodies are analyzed for the 75g side drop loading using hand calculations. The fuel can body is treated as a beam with pinned support conditions at the spacer disc locations. The beam load is assumed to be a uniform line load equal to the combined weight of the fuel can body and the PWR fuel assembly. Stress results for the FF-DSC are provided in Table 8-11.

A structural evaluation has been performed to allow the use of SA-240, Type XM-19 austenitic stainless steel as an optional FF-DSC basket material. The allowable stresses for SA-240, Type XM-19 are greater than those of the original carbon steel material for all stress categories and at all temperatures. The elastic modulus of SA-240, Type XM-19 is up to 5% less than that of carbon steel, which slightly lowers the minimum factor of safety against elastic buckling. Although spacer disk elastic stability minimum factor of safety will be slightly lower, the current margin is more than 270% greater than required. The yield stress of SA-240, Type XM-19 is slightly less than that of the carbon steel materials. The effect of a lower yield stress only impacts the analysis of the support plates, which are shown to retain positive margins of safety in the bounding end drop case, discussed below. All aspects of the structural analysis, including stress analysis and elastic stability, are shown to retain positive large margins of safety, thus permitting the use of SA-240, Type XM-19 stainless steel as a FF-DSC basket material.

2. Vertical End Drop The cask and DSC assembly components are analyzed for a 75g equivalent static vertical end drop load. The stability of the system components as well as the stresses are considered in the evaluation.

   a. Cask Vertical End Drop The stresses in the cask components resulting from the postulated 75g top and bottom end vertical end drops are evaluated using an axisymmetric finite element model. The model includes the cask inner shell, outer shell, ram access cover plate, bottom end closure forging, top corner forging, top cover plate, lead shielding, neutron shielding, neutron shield jacket, and neutron shield support rings. The load from the DSC is modeled as a uniform pressure load acting on the supporting cask end plate. A 75g acceleration load is applied in the direction of the drop. Linear elastic static analyses are performed for the postulated top and bottom end drop loading using the ANSYS program [8.8.3]. The results of the cask vertical end drop analysis show that the maximum primary membrane, membrane plus bending and primary plus secondary stresses occur in the cask outer shell for the bottom end drop and are 9.1 ksi, 11.0 ksi and 14.1 ksi, respectively [8.8.9]. The cask vertical end drop stresses are lower than those calculated for the cask horizontal side drop and corner drop conditions.

b. <u>FC-DSC Vertical End Drop</u> The FC-DSC shell assembly and basket assembly components are analyzed for the postulated 75g top and bottom end drop events using a combination of hand calculations and finite elements computer models.

The stresses in the FC-DSC spacer discs and support rods due to the postulated 75g end drop loading are evaluated using a quarter symmetry finite element model of the basket assembly. The model includes a torsional spring element used to model support rod bending stiffness. The spacer disc is restrained from translating out of plane at nodes nearest the locations of the support rod centerlines. The spacer disc is loaded only by its own self-weight for the end drop condition. Linear elastic static analysis are performed for the end drop conditions using the ANSYS program [8.8.3]. The bounding results of the FC-DSC top and bottom end vertical drop analyses are summarized in Table 8-10 [8.8.13].

The support rod has a pretension of 100k and the spacer disc and rod sleeves are not welded together. Therefore, the support rod will not be under compression and no buckling will occur [8.8.13]. The guide sleeve end drop evaluation is performed by hand calculations for the 75g end drop loading. The results of this analysis are summarized in Table 8-10.

The guide sleeve buckling mode consists of panel buckling rather than beam buckling. The minimum factor of safety against elastic buckling is 2.61 [8.8.11]. Therefore, elastic buckling will not occur for the postulated 75g end drop loading and the minimum factor of safety of 1.50 against elastic buckling required by the ASME Code is satisfied.

The FC-DSC shell is loaded by its own self weight in addition to the weight of the top shield plug, inner top plate and outer top plate for a bottom end drop or the weight of the inner bottom plate, bottom shield plug and bottom cover plate for a top end drop. An enveloping load for FO-, FC-, and FF-DSCs has been used to calculate stresses in the shell for the 75g end drop. These stresses are shown in Table 8-10.

The controlling end drop condition for the inner bottom plate is the 75g top end drop for which the inner bottom plate supports its own self weight in addition to the weight of the 3.5 inch thick bottom end lead shield plug. The finite element described in Section 8.1.1.1 is used to analyzed the inner bottom plate stresses for the 75g top end drop loading. The 3/4" thick inner bottom plate is modeled with pinned edges representing the restraint provided by the double sided full penetration weld between the inner bottom plate and the DSC shell. No credit is taken for shear transfer between the lead shielding and the inner bottom cover plate. A 75g top end drop acceleration load is applied to the model.

The stresses in the inner bottom plate/DSC shell weld due to the 75g top end drop are determined using hand calculations for the maximum weld reaction forces from the inner bottom plate top end drop analysis. The bending stress in the bottom cover plate and shield plug assembly due to the 75g top end drop loading is determined using hand calculations. The bottom cover plate is conservatively analyzed as a simply supported circular plate under uniform pressure loading due to its own self weight.

The bending stresses in the inner and outer top cover plates due to the 75g bottom end drop loading are determined using hand calculations. The inner and outer top cover plates are conservatively analyzed as a simply supported circular plates under uniform pressure loading due to their own self weight. Stress results are provided in Table 8-10.

c. <u>FO-DSC Vertical End Drop</u> The FO-DSC shell assembly is identical to the FC- and FF-DSC shell assemblies except for the use of steel shield plugs. The DSC shell assembly is loaded by its own self weight in addition to the weight of the basket assembly and fuel. An enveloping load for all DSCs has been used to compute stresses in the FO shell for the 75g end drop. The resulting stresses are shown in Table 8-9.

   The FO-DSC basket assembly is identical to the FC-DSC basket assembly with the exception of the length of the support rods extending beyond the top end spacer disc and the addition of angle iron at the top corners of the fuel sleeves. The FO-DSC support rods extend 5.12 inches beyond the top spacer disc, while the FC-DSC support rods extend 11.12 inches beyond the top spacer disc. The difference between the FO-DSC and FC-DSC basket assembly designs will have no significant effect on the basket component stresses. The shorter length of the FO-DSC support rods makes the FO-DSC basket assembly more stable than the FC-DSC basket assembly for a top end drop. Therefore, the FO-DSC basket assembly stress analysis and stability analysis results for the top and bottom end drops are bounded by those of the FC-DSC discussed in (b) above. The FC-DSC basket assembly component stresses and elastic buckling loads are conservatively used for the evaluation of the FO-DSC basket assembly components.

d. <u>FF-DSC Vertical End Drop</u> The FF-DSC shell assembly is the same as the FC-DSC shell assembly. Therefore, the stresses in the FF-DSC shell assembly components due to the postulated 75g vertical end drop loading are equal to those calculated for the FC-DSC shell assembly components in (b) above, and are shown in Table 8-11.

The stresses in the FF-DSC spacer discs and support plates due to the postulated 75g end drop loading are evaluated using a quarter symmetry finite element model of the basket assembly. The spacer discs and support plates are modeled using 3-D quadrilateral shell elements and 3-D beam elements, respectively. Linear elastic static analyses are performed for the end drop conditions using the ANSYS program [8.8.3]. The bounding results of the FF-DSC top and bottom end vertical drop analyses are summarized in Table 8-11 [8.8.12].

The maximum support plate/spacer disc weld stress due to the 75g end drop is calculated using the maximum weld loads from the top and bottom end drop analyses. The maximum support plate/spacer disc weld stress occurs at the bottom end spacer disc for the 75g bottom end drop loading.

The FF-DSC support plate stability analysis has been performed using hand calculations. The minimum factor of safety against elastic buckling for this condition is 11.1 [8.8.12]. The minimum factor of safety required by the ASME Code [8.8.8] for elastic buckling is 1.5. Therefore, the FF-DSC support plates meet the ASME Code acceptance criteria.

The stresses in the FF-DSC fuel can bodies due to the 75g end drop load are determined using hand calculations. The elastic stability of the fuel cans is determined in accordance with F-1334.3 of the ASME Code [8.8.8]. The unbraced length of the fuel can is conservatively assumed as its total length (172.5 inches). An effective length factor of 2.1 is conservatively assumed. The fuel can body is analyzed for the maximum straightness tolerance of 0.12 inches. The minimum factor of safety against elastic buckling is 2.9 [8.8.12]. The minimum factor of safety required by the ASME Code [8.8.8] for elastic buckling is 1.5. Therefore, the FF-DSC fuel can bodies meet the ASME Code acceptance criteria.

A structural evaluation has been performed to allow the use of SA-240, Type XM-19 austenitic stainless steel as an optional FF-DSC basket material. The allowable stresses for SA-240, Type XM-19 are greater than those of the original carbon steel material for all stress categories and at all temperatures. The elastic modulus of SA-240, Type XM-19 is up to 5% less than that of carbon steel, which slightly lowers the minimum factor of safety against elastic buckling. The effect of the elastic modulus is discussed in Section 8.2.1.3(1)(d). The yield stress of SA-240, Type XM-19 is slightly less than that of the carbon steel materials. The effect of a lower yield stress only impacts the analysis of the support plates. The maximum stress ratio for the carbon steel support plates, which must be below 1.0 is 0.70. The same analysis methods are used to evaluate the support plates using the SA-240, Type XM-19 material properties. The resulting maximum stress ratio for the supp0rt plate is 0.88. Additionally, the buckling load limit for the support plates is a function of yield. The buckling load limit for SA-240, Type XM-19 remains greater than the applied axial load. All aspects of the structural analysis, including stress analysis and elastic stability, are shown to retain positive large margins of safety, thus permitting the use of SA-240, Type XM-19 stainless steel as a FF-DSC basket material.

3. Corner Drop The cask and DSC assembly components are analyzed for a 25g equivalent static corner drop load as described in the following paragraphs.

   a. Cask Corner Drop The cask is analyzed for the postulated 25g corner drop on the top and bottom corners using a finite element model. The cask finite element model consists of harmonic elements which have axisymmetric geometry but permit non-axisymmetric loading. Fourier series are used to define the corner drop non-axisymmetric loading due to the impact forces and the loading from the DSC. The results of the cask top and bottom corner drop analyses shows the maximum stresses occur at the 0° azimuth (bottom side) for all cask components. The maximum primary membrane and (local membrane + bending) stress intensities in the cask inner shell, resulting from the top end corner drop, are 42.3 ksi and 47.7 ksi, respectively [8.8.9]. The maximum primary membrane and (local membrane + bending) stress intensities in the cask outer shell are 36.8 ksi and 58.4 ksi, respectively [8.8.9]. The bounding cask top and bottom end corner drop stresses are summarized in Table 8-8.

   b. FO-DSC, FC-DSC and FF-DSC Corner Drop The stresses in the DSC shell and basket assembly components resulting from the postulated 25g corner drop are bounded by the 75g horizontal side drop and 75g vertical end drop results. Therefore, no analysis of the DSC shell and basket assembly components is necessary.

8.2.1.4 Corrective Actions

For drop heights of less than fifteen inches the cask will be loaded back onto the transfer skid/ trailer and moved to the HSM. The DSC will then be transferred to the HSM in the normal manner described previously. For drop heights greater than fifteen inches the transfer cask and contents will be returned to the Fuel Storage Building. There the DSC will be inspected for damage, and the DSC opened and the fuel removed for inspection, as necessary. Removal of the cask top cover plate may require cutting of the bolts in the event of a corner drop onto the top end. This operation will take place in the Fuel Storage Building after recovery of the cask. Removal of the DSC cover plates and shield plug assembly are described in Section 5.0.

Following recovery of the cask and unloading of the DSC, the cask will be inspected, repaired and tested as appropriate prior to reuse.

For drop heights approaching the design basis conditions, it may be necessary to develop a special sling/lifting apparatus to move the transfer cask from the drop site to the fuel pool. This may require several weeks of planning to ensure all steps are correctly organized. During this time, additional blankets can be added to the transfer cask to minimize on-site exposure to site operations personnel. The cask will be roped off to ensure the safety of the site personnel.

8.2.2 DSC Leakage

The DSC shell is designed as a pressure retaining containment boundary to prevent leakage of contaminated materials, as discussed in the Standardized NUHOMS® SAR [8.8.5]. The analyses of normal, off-normal, and accident conditions have shown that no credible conditions can breach the DSC shell or fail the double seal welds at each end of the DSC. However, as specified in Interim Staff Guidance No. 5 (ISG-5) Revision 1, an accident release is postulated assuming all of the fuel rods in a single DSC are breached and leak to the environment at the rate specified in Section 10.3.4.

This event postulates a month long release of 30% of all gasses, 0.02% of all volatiles, 0.003% of all fuel fines, and 100% of all CRUD contained in the DSC. Fission products that represent more than 0.1% of the design basis activity and actinides that represent 0.01% of the design basis activity are included in the evaluation. All other components of the ISFSI remain intact.

8.2.2.1 Postulated Cause of Event

There is no postulated cause of this event. Simultaneous rupture of all rods is assumed to occur and the DSC is postulated to leak at a rate of 10^{-5} std-cc/sec in order to demonstrate compliance with 10 CFR 72.106.

8.2.2.2 Detection of Event

Although unexpected, detection of this event would occur upon cask inspection following a drop event.

8.2.2.3 Analysis of Effects and Consequences

There are no structural or thermal consequences resulting from the DSC leakage accident described above. The radiological consequences of this accident are discussed below.

The postulated accident assumes that one DSC is leaking and that all the spent fuel rod cladding fails simultaneously such that 30% of all gasses, 0.02% of all volatiles, 0.003% of all fuel fines, and 100% of all CRUD contained in the DSC are available for release. Fission products that represent more than 0.1% of the design basis activity and actinides that represent 0.01% of the design basis activity are included in the evaluation. The whole body dose at the Rancho Seco ISFSI controlled area boundary is 0.195 rem [8.13] well within the 10 CFR 72.106 limit of 5 rem. All other organ doses are well below the remaining 10 CFR 72.106 limits. [8.13]

8.2.2.4 Corrective Actions

There are no corrective actions since DSC leakage combined with a simultaneous breach of all fuel rods is not a credible event. The purpose of this evaluation is to satisfy the criteria of ISG-5 to demonstrate the overall safety of the system.

8.2.3 Accident Pressurization

The cask, FO-DSC, FC-DSC, and FF-DSC are evaluated for postulated accident pressurization internal pressure loads. The accident pressurization event is postulated to occur during any handling or storage mode of operation. Accident pressure loads apply only to the cask for a hypothetical storage condition.

8.2.3.1 Postulated Cause of Event

During accident conditions, 100% of the fill gas and 30% of the fission gases are assumed to be released inside the DSC cavity due to the rupture of 100% of the fuel rods from all the 24 fuel assemblies. Also, control components were considered including black and gray axial power shaping rod assemblies, burnable poison rod assemblies, control rod assemblies, and orifice rod assemblies. 100% rupture of the control components, 100% release of the fill gas, and 30% release of generated gases into the FF/FC DSCs are assumed.

The pressures corresponding to the postulated accident conditions and the associated helium gas temperature without control components are shown in Table 8-2a. The pressures corresponding to the postulated accident conditions and with control components are shown in Table 8-2b. Note that the 125°F ambient temperature, blocked vents, case from the Standardized NUHOMS® SAR [8.8.5] is conservatively used to estimate the HSM concrete temperatures in the calculation of maximum DSC accident pressures.

8.2.3.2 Detection of Event

No additional means or methods are required to be provided for the detection of the accident pressurization event.

8.2.3.3 Analysis of Effects and Consequences

The accident pressurization analysis methodology is identical to the methodology used to evaluate the DSC for the normal internal pressure loads in Section 8.1.1.2. The DSCs are evaluated for a maximum accident internal pressure of 50 psig. The inner top cover plate is evaluated for an accidental pressure of 41 psig for Service Level C. The outer top cover plate is evaluated for an accidental pressure of 50 psig for Service Level D. The FO-DSC, FC-DSC, and FF-DSC accident internal pressure stress analysis results are summarized in Table 8-9 through Table 8-11.

8.2.3.4 Corrective Actions

No corrective actions are required in the event of an accidental DSC pressurization. The analysis of the DSCs for the accident pressure load show that no significant deformations occur in the DSC which could prevent retrieval from the HSM or cask or inhibit normal transport or on-site transfer operations. In addition, the DSC pressure boundaries are analyzed to withstand the accident internal pressure to prevent release of any radioactive materials to the environment.

8.2.4 Earthquake

As discussed in Section 3.2.3, the cask and DSC are analyzed for the enveloping design basis earthquake for the ISFSI conditions.

8.2.4.1 Postulated Cause of Event

As discussed in Section 3.2.3, enveloping design basis seismic forces are assumed to act on the system components. For this conservative evaluation, the design response spectra of NRC Regulatory Guide 1.60 (8.35) were used for the seismic analysis of the system components.

8.2.4.2 Detection of Event

No additional means or methods are required to be provided to the detection of the design basis seismic event.

8.2.4.3 Analysis of Effects and Consequences

The seismic input criteria and analysis methodology are described in Section 3.2.3. Stability and stress analyses are performed for the cask in the on-site transfer mode for the postulated design basis earthquake loads. The bounding seismic stress results for the cask, FO-DSC, FC-DSC and FF-DSC are reported in Table 8-8 through Table 8-11, respectively.

Evaluations of the effects and consequences of the earthquake event for HSMstorage is addressed in Volume II.

1. Cask Overturning Analysis During on-site transfer operations, the cask is secured to the on-site transfer skid and trailer in the horizontal position. The cask in this orientation is much more stable than for the postulated cask storage mode. The results of the postulated cask storage seismic stability analysis, contained in Volume III, Section 8.3.2.3(1), show a minimum margin of safety against overturning of 38%. Therefore, overturning due to the design basis seismic event will not occur.

2. Seismic Stress Analysis The stresses in the cask, FC-DSC, FO-DSC and FF-DSC due to the design basis seismic event are calculated using the methodology discussed in Section 3.2.3.2. Equivalent static loads are calculated based on the

dominant natural frequencies of the ISFSI components. The equivalent static vertical and horizontal seismic accelerations for the cask, conservatively including a 1.5 factor for possible multi-mode excitation, are 0.65g and 0.95g, respectively. Similarly, the equivalent static vertical and horizontal seismic accelerations for the DSC inside the HSM are 0.17g and 0.37g, respectively.

The stresses in the cask outer shell near the trunnions are calculated for the cask in the horizontal position fixed to the on-site transfer trailer and skid using finite element model of the cask outer shell and upper and lower trunnions. The trunnion reaction loads due to the 0.65g vertical and 0.95g horizontal equivalent static loads are applied to the model at the trunnion support locations. The maximum local membrane stress intensity in the cask outer shell due to the seismic loading is 3.4 ksi [8.8.9].

The stresses in all other cask components due to the 0.65g vertical and 0.95g horizontal seismic accelerations are calculated by factoring the appropriate horizontal and vertical dead load analysis results. The total stress intensities in the cask components are calculated as the SRSS of the stresses due to the vertical, lateral and axial seismic loads. The resulting cask seismic stress intensities are reported in Table 8-8 [8.8.9].

The seismic stresses in the DSC components are calculated for the cask and DSC in the horizontal orientation, with the DSC resting on the cask rails, using the same methodology used in the Standardized NUHOMS® SAR [8.8.5]. The DSC seismic stresses for the cask handling mode are calculated for a horizontal acceleration of 3g and a vertical acceleration of 1g. The horizontal and vertical seismic stresses are conservatively added absolutely. The spacer discs and support rods are conservatively evaluated for 1.5g horizontal and 1.0g vertical accelerations. The maximum primary membrane and membrane plus bending stress intensities in the FO-DSC and FC-DSC spacer discs are 9.2 ksi and 13.2 ksi, respectively. The maximum primary membrane and membrane plus bending stress intensities in the FF-DSC spacer disc are 1.7 ksi and 6.0 ksi, respectively.

8.2.4.4 Corrective Actions

No corrective actions are required in the event of an earthquake.

8.2.5 Fire

The ISFSI system is analyzed for a postulated fire accident which takes place either during transfer when the DSC is in the cask, loading, or storage of the DSC in the HSM.

8.2.5.1 Postulated Cause of Event

The tow vehicle which is used to transfer the TC/DSC to the ISFSI site utilizes flammable diesel fuel. In addition, the skid positioning system and hydraulic ram uses hydraulic oil which could potentially spill and ignite.

8.2.5.2 Detection of Event

No additional means or methods are required to be provided for the detection of the fire event.

8.2.5.3 Analysis of Effects and Consequences

A worst case bounding fire is postulated and analyzed using the HEATING7 code[8.21]

The maximum expected flammable fuel either during the transfer operation or inside the ISFSI is 300 gallons. A worst case fire is therefore postulated for the transfer condition in which the 300 gallons forms a pool directly beneath the transfer cask. The diameter of the pool is assumed to be 201.5 in, or the nominal length of the transfer cask. The thickness of such a pool would be 2.17 in. The assumption of a 2.17 inch thick pool is very conservative as the pool would tend to spread out very quickly and only a small thickness of fuel would result due to capillary pressures. This pool is assumed to burn at a minimum rate of 0.15 in/min [8.18]. The 2.17 inch thick pool would burn for 14.5 minutes. 15 minutes is conservatively used in the calculation. The fire is assumed to engulf the entire TC/DSC structure. The fire parameters from 10CFR71.73 [8.19] are used.

Forced convection from the fire to the cask is described by using the temperature dependent convection coefficients derived for the transportation fire accident considered for the MP187 package [8.20, Table 3.2-3]. The fire is assumed to be initiated during the maximum off-normal ambient conditions, 117°F, with maximum solar insolation. The thermal properties of the NS-3 annular region of the transfer cask, which has aluminum stiffeners, is assumed to be conservatively replaced by air following the fire accident, since the material temperature limit of the NS-3 will be exceeded during the fire. The HEATING7 model which is described in Section 8.1.1.1 of Volume III of this SAR is used as the basis for the transient fire analysis. The boundary conditions are changed according to the descriptions of the fire parameters presented here. The transient is run for approximately 25 days after the fire to obtain the maximum temperatures for all the components. The results show that the component temperatures remain well below their material temperature limits for all materials except NS-3 and aluminum stiffeners in the NS-3 annular region. The integrity of the DSC shell and cover plates, the TC structural steel and all the seals and o-rings are therefore ensured during and after the fire. The neutron shield rupture disks have been designed for the 10CFR71 postulated fire as described in the NUHOMS®-MP187 transportation SAR and will adequately relieve any pressure buildup in the neutron shield.

Assuming that the entire neutron shield is lost during the fire, dose rates on the surface of the cask as high as about 1.2 rem per hour would be expected. This estimate is based on the accident dose rate calculations for the cask shown in Figure 5.4-9 of the NUHOMS®-MP187 transportation SAR. These calculations assumed that the neutron shield was completely removed from the cask during a fire accident. The dose rate peak shown in Figure 5.4-9 of the transportation SAR has been neglected in this estimate because it is due to lead slump from a drop accident and is not related to a loss of neutron shield. During recovery, workers should use temporary shielding as necessary to keep exposures ALARA.

The effects of the 15 minute cask fire on the DSC basket are bounded by the HSM blocked vent case already considered. For the blocked vent accident case, the maximum DSC shell temperature is higher and subsequently the internal components are also higher. Therefore, the DSC basket component temperatures remain well below their material limits and the integrity of the fuel cladding, guide sleeve, spacer disc, and neutron absorbing material are ensured. Also, the accident pressure resulting from the fire event is bounded by the HSM blocked vent accident pressure.

Direct engulfment of the HSM or DSC during storage in the HSM is not credible due to the following:

1. The HSM vent is approximately 1 foot above the ISFSI slab and 3 feet behind the front of the HSM. Therefore, fuel can not accumulate above the level of the vents.

2. The ISFSI apron design requires a 1% slope for drainage purposes. Therefore, any fuel spill inside the ISFSI will drain away from the HSMs to the edge of the ISFSI foundation.

Therefore, direct engulfment of the HSM is not credible. Any fire within the ISFSI boundary while the DSC is in the HSM would be bounded by the fire during transfer when the DSC is in the TC already considered. The HSM concrete acts as a significant insulating fire wall to protect the DSC from the high temperatures of the fire. The thermal resistance of the HSM is much greater than the resistance of the cask. Therefore, the fire case of the DSC in the HSM is bounded by the fire case of the DSC in the transfer cask.

8.2.5.4 Corrective Action

The NS-3 material may be damaged during the fire event. Therefore, for ALARA purposes, it is recommended that the transfer cask be unloaded and the effect of the fire on the transfer cask be evaluated prior to its reuse.

8.3 Load Combination Evaluation

The bounding load combination results for all storage and handling modes of operation are summarized in this section. The analysis of the ISFSI load conditions and HSM storage conditions, addressed in Chapter 8 of Volumes I, II and III, respectively, are combined in accordance with the load combinations specified in Section 3.2. Detailed load combination evaluations for each of the ISFSI components are included in the calculation packages contained in Volume IV.

8.3.1 DSC Load Combination Evaluation

As described in Section 3.2.5, the stress intensities in the FO-DSC, FC-DSC and FF-DSC at various critical locations for the appropriate normal operating conditions are combined with the stress intensities resulting from the postulated off-normal and accident conditions. It is assumed that only one postulated accident event occurs at any one time. The results of the structural analyses show that the ISFSI components which are important to safety meet the applicable structural and mechanical safety criteria specified in Section 3.2.

1. FO-DSC and FC-DSC The stress intensities in the FO-DSC for the individual load conditions are combined in accordance with the load combinations presented in Section 3.2.5.2. The maximum stress intensities in each of the FO-DSC and FC-DSC components which are important to safety for the enveloping normal/off-normal and accident load combinations, and the corresponding ASME Code allowable stresses and maximum stress ratios, are shown in Table 8-15 through Table 8-17 and Table 8-18 through Table 8-20, respectively. The results of the FO-DSC and FC-DSC load combination evaluation show a maximum stress ratio of 0.99 in the spacer disc for load combination D2. The controlling load combination D2 stress is primarily due to the accidental cask drop load condition. Therefore, the stresses in the FO-DSC and FC-DSCs meet the ASME Code allowable stresses for all load combinations.

2. FF-DSC The stress intensities in the FF-DSC for the individual load conditions are combined in accordance with the load combinations presented in Section 3.2.5.2. The maximum stress intensities in each of the FF-DSC components which are important to safety for the enveloping normal/off-normal and accident load combinations, and the corresponding ASME Code allowable stresses and maximum stress ratios, are shown in Table 8-21 through Table 8-23. The results of the FF-DSC load combination evaluation show a maximum stress ratio of 0.94 in the DSC shell for load combination B2. The controlling load combination B2 secondary stress intensity is primarily due in combination to thermal loads and handling loads for the postulated jammed canister during retrieval from the HSM. Therefore, the stresses in the FF-DSC meet the ASME Code allowable stresses for all load combinations.

8.3.2 Cask Load Combination Evaluation

As described in Section 3.2 of volumes I, II and III, the stress intensities in the casks at various critical locations for the appropriate normal operating conditions are combined with the stress intensities resulting from the postulated off-normal and accident conditions. It is assumed that only one postulated accident event occurs at any one time. The stress intensities in the cask for the individual load conditions are combined irrespective of location in accordance with the load combinations presented in Section 3.2.5.3. The maximum stress intensities in each of the cask components which are important to safety for the enveloping normal/off-normal and accident load combinations, and the corresponding ASME Code allowable stresses and maximum stress ratios, are shown in Table 8-12 through Table 8-14. The results of the cask load combination evaluation show a maximum stress ratio of 0.98 in the cask inner and outer shell for load combination D1/D3. Therefore, the stresses in the cask meet the ASME Code allowable stresses for all load combinations.

8.3.3 Summary of Design Requirements Met

As discussed in Chapter 3 the analytical results are compared to the principal design criteria to determine the acceptability of the ISFSI components. The results of the structural analyses presented in Sections 8.3.1 and 8.3.2 show that the ISFSI components which are important to safety meet the applicable structural and mechanical safety criteria specified in Section 3.2. Therefore, the NUHOMS®-MP187 cask serves its intended purpose of protecting the DSC from all normal and off-normal condition loads as well as a range of credible postulated accident loads. In addition, the cask provides passive radiation shielding and heat removal capabilities.

8.4 Site Characteristics Affecting Safety Analysis

All site characteristics affecting the safety analysis of the ISFSI are noted throughout this SAR where they apply.

8.5 References

8.1 U.S. Nuclear Regulatory Commission (U.S. NRC), "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)," Regulatory Guide 3.48 (August 1989).

8.2 American National Standard, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," ANSI/ANS 57.9-1984, American Nuclear Society, La Grange Park, Illinois (1984).

8.3 Swanson Analysis Systems, Inc., ANSYS Engineering Analysis System User's Manual, Version 4.4, Swanson Analysis Systems, Inc., Pittsburgh, PA.

8.4 "Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-002, Revision 1A, Pacific Nuclear Fuel Services.

8.5 "Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.

8.6 American National Standard for Radioactive Materials, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," ANSI N14.6-1986, American National Standards Institute, New York, N.Y. (1987).

8.7 Crane Manufacturers Association of America, Inc., "Specifications for Electric Overhead Traveling Cranes," CMAA Specification #70, 1988.

8.8 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1992 Edition with 1993 Addenda.

8.9 MP187 Cask 10CFR72 Structural Analysis, TN West Calculation Number 2069.0203, Revision 2.

8.10 NUHOMS®-MP187 FC-DSC 10CFR72 Structural Analysis, TN West Calculation Number 2069.0201, Revision 3.

8.11 NUHOMS®-MP187 Guide Sleeve Structural Analysis, TN West Calculation Number 2069.0204, Revision 5.

8.12 NUHOMS®-MP187 FF-DSC 10CFR72 Structural Analysis, TN West Calculation Number 2069.0205, Revision 2.

8.13 Rancho Seco NUHOMS® DSC Confinement Evaluation, TN West Calculation Number 2069.0507, Revision 0.

8.14 Rancho Seco NUHOMS® FO/FC Dry Shielded Canister Basket Assembly Structural Analysis, TN West Calculation Number 2069.0217, Revision 0.

8.15 Rancho Seco NUHOMS® Long Cavity (FC) 24P DSC Shell Assembly Structural Analysis, TN West Calculation Number 2069.0219, Revision 0.

8.16 Rancho Seco NUHOMS®-Short Cavity (FO) 24P Dry Shielded Canisters Shell Assembly Structural Analysis, TN West Calculation Number 2069.0216, Revision 0.

8.17 Evaluation of FO, FC and FF-DSC Lifting Lugs, TN West Calculation Number NUH005.0262, Revision 0.

8.18 SAND85-0196, TTC-0659, Gregory et. al., "Thermal Measurement in a Series of Large Pool Fires," Sandia National Laboratories, 1987.

8.19 "Packaging and Transportation of Radioactive Material," Title 10 Code of Federal Regulations, Part 71 (10CFR71), USNRC, January 1996.

8.20 Safety Analysis Report for the NUHOMS®-MP187 Multi-Purpose Cask, Revision 9, NRC Docket Number 71-9255.

8.21 "HEATING7, Multidimensional, Finite-Difference Heat Conduction Analysis", PSR-199, Oak Ridge National Laboratory, March 1993

Table 8-1

NUHOMS® ISFSI Normal and Off-Normal Operating Loading Summary

Load Type	Reference Section	Affected Component		
		Cask	DSC Shell Assembly	DSC Internals
Dead Weight	8.1.1.1	X	X	X
Internal Pressure	8.1.1.2		X	
Normal Thermal	8.1.1.3	X	X	X
Normal Handling	8.1.1.4	X	X	X
Off-Normal Handling	8.1.1.5	X	X	X
Live Loads	8.1.1.6	X		
Fatigue	8.1.1.7&8	X	X	
Thermal Cycling	8.1.1.9	X		

Table 8-2a

FO/FC/FF DSC Cavity Normal, Off-Normal, and Accident Pressures Without Control Components[1,2]

Ambient Air Temperature (°F)	Average Helium Temperature (°F)	Normal Pressure[1] (psia)	Off-Normal Pressure[2] (psia)	Design Basis Accident Pressure (psia)
70	524	18.6	N/A	59.1
101[4]	541	18.9	N/A	60.1
117[3]	542	N/A	22.8	60.1
Blocked HSM Vents (117)[3]	611	N/A	N/A	64.2

Notes:

1. Maximum normal operating pressure with 1% of fuel rods ruptured, 100% release of fuel rod fill gas, and 30% of fission gases assumed to be released.

2. Maximum off-normal operating pressure with 10% fuel rods ruptured, 100% release of fuel rod fill gas, and 30% of fission gases assumed to be released.

3. Enveloping accident pressures with 100% fuel rods ruptured, 100% release of fuel rod fill gas, and 30% of fission gases assumed to be released.

4. The 70°F, 101°F, and 117°F cases correspond to the DSC in a horizontal position in the cask during transfer conditions. This case bounds the DSC in HSM storage.

Table 8-2b
Pressures in the FC/FF DSC with Control Components[1,2]

Ambient Air Temperature (°F)	Average Helium Temperature (°F)	Normal Pressure[1] (psia)	Off-Normal Pressure[2] (psia)	Design Basis Accident Pressure (psia)
101[4]	541	19.1	N/A	<64.3
117[3]	542	N/A	22.9	<64.3
Blocked HSM Vents (117)[3]	588	N/A	N/A	64.3

Notes:

1. Maximum normal operating pressure with 1% of fuel and control component rods ruptured, 100% release of fuel and control component rod fill gas, and 30% release fraction of fission gases and gases generated by the control components.

2. Maximum off-normal operating pressure with 10% fuel and control component rods ruptured, 100% release of fuel and control component rod fill gas, and 30% release fraction of fission gases and gases generated by the control components.

3. Enveloping accident pressures with 100% fuel and control component rods ruptured, 100% release of fuel and control component rod fill gas, and 30% release fraction of fission gases and gases generated by the control components.

4. The 101°F and 117°F cases correspond to the DSC in a horizontal position in the cask during transfer conditions. This case bounds the DSC in HSM storage.

Table 8-3

Cask ISFSI Normal and Off-Normal Operating Condition Stresses

Cask Component	Stress Type	Stress (ksi)[1]				
		Dead Weight				
		Vertical Handling	Horizontal Transport	Thermal	Normal Handling	Off-Normal Handling
Inner Shell	Primary Membrane	0.4	1.3	N/A	2.6	0.1
	Membrane + Bending	0.4	1.3	N/A	2.6	0.1
	Primary + Secondary	0.5	1.3	14.3	2.6	0.2
Outer Shell	Primary Membrane	2.4	1.0	N/A	3.7	0.8
	Membrane + Bending	7.9	3.8	N/A	13.9	2.6
	Primary + Secondary	12.9	6.3	13.9	17.7	4.2
Top Cover Plate	Primary Membrane	N/A[2]	0.0	N/A	0.0	N/A[3]
	Membrane + Bending	N/A[2]	0.0	N/A	1.0	N/A[3]
	Primary + Secondary	N/A[2]	0.0	7.3	1.0	N/A[3]
Top Corner Forging	Primary Membrane	0.4	1.3	N/A	2.6	0.0
	Membrane + Bending	0.4	1.3	N/A	2.6	0.0
	Primary + Secondary	0.5	1.3	12.3	2.6	0.0
Bottom End Closure Forging	Primary Membrane	0.4	1.3	N/A	2.6	0.0
	Membrane + Bending	1.5	1.3	N/A	2.6	0.0
	Primary + Secondary	1.6	1.3	10.1	2.6	0.0
Ram Access Cover Plate	Primary Membrane	0.0	0.0	N/A	0.0	N/A[3]
	Membrane + Bending	0.1	0.0	N/A	0.1	N/A[3]
	Primary + Secondary	0.1	0.0	2.3	0.1	N/A[3]

Notes:

1. Values shown are maximum irrespective of location.
2. The cask top cover plate is not in place during the vertical lift from the fuel pool.
3. The cask top cover plate and ram access cover plate are not in place during HSM transfer handling operations.

Table 8-4

FO-DSC ISFSI Normal and Off-Normal Operating Condition Stresses

FO-DSC Component	Stress Type	Stress (ksi)[1] Dead Weight Vertical	Dead Weight Horizontal	Internal Pressure	Thermal	Normal Handling[3]	Off-Normal Handling
Shell	Primary Membrane	9.5[2]	2.6	1.6	N/A	12.1	16.1
	Membrane + Bending	26.3[2]	5.2	3.4	N/A	17.2	23.0
	Primary + Secondary	N/A[2]	4.9	9.7	32.2	19.7	N/A
Outer Top Cover Plate	Primary Membrane	0.02	1.2	2.1	N/A	0.0	0.0
	Membrane + Bending	0.04	1.85	7.1	N/A	0.0	0.0
	Primary + Secondary	0.03	1.27	6.3	23.9	0.0	0.0
Inner Top Cover Plate	Primary Membrane	0.00	0.76	0.9	N/A	0.0	0.0
	Membrane + Bending	0.03	2.2	4.5	N/A	0.0	0.0
	Primary + Secondary	0.03	2.1	3.4	24.9	0.0	0.0
Outer Bottom Cover Plate	Primary Membrane	0.0	0.71	0.4	N/A	12.7	16.9
	Membrane + Bending	0.0	1.21	0.7	N/A	23.4	31.1
	Primary + Secondary	0.0	1.13	0.5	30.3	22.2	N/A
Inner Bottom Cover Plate	Primary Membrane	0.0	0.71	0.3	N/A	11.5	15.7
	Membrane + Bending	0.0	0.83	0.8	N/A	18.8	25.5
	Primary + Secondary	0.0	0.82	0.8	28.0	18.8	N/A
Spacer Disc	Primary Membrane	0.0	2.3	N/A	N/A	N/A	N/A
	Membrane + Bending	11.6	3.3	N/A	N/A	N/A	N/A
	Primary + Secondary	14.2	3.3	N/A	42.6	N/A	N/A
Support Rods	Primary Membrane	31.8	0.0	N/A	N/A	N/A	N/A
	Membrane + Bending	N/A	0.3	N/A	N/A	N/A	N/A
	Primary + Secondary	N/A	0.3	N/A	15.4	N/A	N/A
Guide Sleeves	Primary Membrane	0.1	0.1	N/A	N/A	N/A	N/A
	Membrane + Bending	0.1	0.9	N/A	N/A	N/A	N/A
	Primary + Secondary	N/A	0.9	N/A	0.0	N/A	N/A
Support Ring	Primary Membrane	12.8[2]	0.2	0.5	N/A	N/A	N/A
	Membrane + Bending	28.9[2]	0.2	0.5	N/A	N/A	N/A
	Primary + Secondary	0.2	0.2	0.5	5.2	N/A	N/A

Notes:

1. Values shown are maximum irrespective of location.
2. Local stresses in the vicinity of lifting lugs, when an empty DSC is lifted through various configurations of cross-bar. The DSC top cover plate and inner top cover plate are not in place for the vertical handling dead load condition in the fuel building.
3. Normal handling load for transfer of DSC into HSM.

Table 8-5

FC-DSC ISFSI Normal and Off-Normal Condition Stresses

FC-DSC Component	Stress Type	Stress (ksi)[1] Dead Weight Vertical	Dead Weight Horizontal	Internal Pressure	Thermal	Normal Handling	Off-Normal Handling
Shell	Primary Membrane	9.5[2]	2.6	1.6	N/A	12.1	16.1
	Membrane + Bending	26.3[2]	5.2	3.4	N/A	17.2	23.0
	Primary + Secondary	N/A	4.9	9.7	32.2	19.7	N/A
Outer Top Cover Plate	Primary Membrane	0.0	1.2	2.1	N/A	0.0	0.0
	Membrane + Bending	0.3	1.85	7.1	N/A	0.0	0.0
	Primary + Secondary	0.3	1.27	6.3	23.9	0.0	0.0
Inner Top Cover Plate	Primary Membrane	0.0	0.76	0.9	N/A	0.0	0.0
	Membrane + Bending	0.2	2.2	4.5	N/A	0.0	0.0
	Primary + Secondary	0.2	2.1	3.4	24.9	0.0	0.0
Outer Bottom Cover Plate	Primary Membrane	0.02	0.53	0.07	N/A	0.13	0.18
	Membrane + Bending	0.15	0.54	0.68	N/A	1.22	1.63
	Primary + Secondary	0.15	N/A	0.68	17.8	1.22	1.63
Inner Bottom Cover Plate	Primary Membrane	0.01	0.53	0.40	N/A	0.02	N/A
	Membrane + Bending	0.38	0.54	15.0	N/A	0.75	N/A
	Primary + Secondary	0.38	0.54	15.0	28.0	0.75	N/A
Spacer Disc	Primary Membrane	0.0	2.3	N/A	N/A	N/A	N/A
	Membrane + Bending	11.6	3.3	N/A	N/A	N/A	N/A
	Primary + Secondary	14.2	3.3	N/A	42.6	N/A	N/A
Support Rods	Primary Membrane	31.8	0.0	N/A	N/A	N/A	N/A
	Membrane + Bending	N/A	0.3	N/A	N/A	N/A	N/A
	Primary + Secondary	N/A	0.3	N/A	15.4	N/A	N/A
Guide Sleeves	Primary Membrane	0.1	0.1	N/A	N/A	N/A	N/A
	Membrane + Bending	0.1	0.9	N/A	N/A	N/A	N/A
	Primary + Secondary	N/A	0.9	N/A	0.0	N/A	N/A
Support Ring	Primary Membrane	12.8[2]	0.2	0.5	N/A	N/A	N/A
	Membrane + Bending	28.9[2]	0.2	0.5	N/A	N/A	N/A
	Primary + Secondary	N/A	0.2	0.5	5.2	N/A	N/A

Notes:

1. Values shown are maximum irrespective of location.
2. Local stresses in the vicinity of lifting lugs, when an empty DSC is lifted through various configurations of cross-bar. The DSC outer and inner top cover plates are not in place for the vertical handling dead load condition in the fuel building.
3. Normal handling load for transfer of DSC into HSM.

Table 8-6

FF-DSC ISFSI Normal and Off-Normal Condition Stresses

FF-DSC Component	Stress Type	Stress (ksi)[1]					
		Dead Weight					
		Vertical Handling	Horizontal Transport	Internal Pressure	Thermal	Normal Handling	Off-Normal Handling
Shell	Primary Membrane	9.5[2]	2.6	1.6	N/A	12.1	16.1
	Membrane + Bending	26.3[2]	5.2	3.4	N/A	17.2	23.0
	Primary + Secondary	N/A	4.9	9.7	32.2	19.7	N/A
Outer Top Cover Plate	Primary Membrane	0.0	1.2	2.1	N/A	0.0	0.0
	Membrane + Bending	0.3	1.85	7.1	N/A	0.0	0.0
	Primary + Secondary	0.3	1.27	6.3	23.9	0.0	0.0
Inner Top Cover Plate	Primary Membrane	0.0	0.76	0.9	N/A	0.0	0.0
	Membrane + Bending	0.2	2.2	4.5	N/A	0.0	0.0
	Primary + Secondary	0.2	2.1	3.4	24.9	0.0	0.0
Outer Bottom Cover Plate	Primary Membrane	0.02	0.53	0.07	N/A	.13	0.18
	Membrane + Bending	0.15	0.54	0.68	N/A	1.22	1.63
	Primary + Secondary	0.15	N/A	0.68	17.8	1.22	1.63
Inner Bottom Cover Plate	Primary Membrane	0.01	0.53	0.40	N/A	0.02	N/A
	Membrane + Bending	0.38	0.54	15.0	N/A	0.75	N/A
	Primary + Secondary	0.38	0.54	15.0	28.0	0.75	N/A
Spacer Disc	Primary Membrane	0.0	1.0	N/A	N/A	2.5	N/A
	Membrane + Bending	0.4	1.0	N/A	N/A	2.7	N/A
	Primary + Secondary	N/A	N/A	N/A	27.2[4]	2.7	N/A
Support Plates	Primary Membrane	0.1	0.0	N/A	N/A	0.1	N/A
	Membrane + Bending	0.4	0.1	N/A	N/A	0.5	N/A
	Primary + Secondary	N/A	N/A	N/A	0.0[4]	0.5	N/A
Fuel Body Can	Primary Membrane	0.0	0.1	N/A	N/A	0.3	N/A
	Membrane + Bending	0.1	0.1	N/A	N/A	0.3	N/A
	Primary + Secondary	N/A	N/A	N/A	0.0	0.3	N/A

Notes:

1. Values shown are maximum irrespective of location.
2. Local stresses in the vicinity of lifting lugs, when an empty DSC is lifted through various configurations of cross-bar. The DSC outer and inner top cover plates are not in place for the vertical handling dead load condition in the fuel building.
3. Normal handling load for transfer of DSC into HSM.
4. Thermal results shown are based on the analysis using SA-537, Class 2 carbon steel.

Table 8-7

Postulated ISFSI Accident Loading Summary

Load Type	Section Reference	Affected Component		
		Cask	DSC Shell Assembly	DSC Internals
Cask Drop	0	X		
DSC Leakage	0	X		
Accident Pressurization	0	X	X	X
Seismic	0	X	X	

Table 8-8

Cask ISFSI Accident Condition Stresses

Cask Component	Stress Type	Stress (ksi)[1] Accidental Cask Drop End	Side	Corner	Accident Pressure	Earthquake
Inner Shell	Primary Membrane	6.5	24.9	42.3	0.5	1.5
	Membrane + Bending	7.1	67.3	47.7	0.5	1.5
Outer Shell	Primary Membrane	9.1	24.5	37.7	0.5	3.4
	Membrane + Bending	11.0	67.1	58.4	0.5	3.4
Top Cover Plate	Primary Membrane	6.6	(3)	38.1	0.5	0.0
	Membrane + Bending	6.6	(3)	47.2	2.0	0.1
Top Corner Forging	Primary Membrane	7.6	(3)	40.2	0.5	1.5
	Membrane + Bending	7.6	(3)	47.7	0.5	1.5
Bottom End Closure Forging	Primary Membrane	7.8	(3)	35.3	0.5	1.5
	Membrane + Bending	7.8	(3)	40.0	1.5	2.1
Ram Access Cover Plate	Primary Membrane	2.1	(3)	11.1	0.0	0.0
	Membrane + Bending	2.1	(3)	11.1	0.5	0.1

Notes:

1. Values shown are maximum irrespective of location.
2. The cask top cover plate and ram access cover plate are removed prior to HSM transfer operations.
3. Side drop stress intensities for these cask components are bounded by the corner drop stress intensities, therefore are not reported here.

Table 8-9

FO-DSC ISFSI Accident Condition Stresses

FO-DSC Component	Stress Type	Stress (ksi)[1] Accidental Cask Drop End	Stress (ksi)[1] Accidental Cask Drop Side	Stress (ksi)[1] Accident Pressure
Shell	Primary Membrane	9.8	28.9	5.2
	Membrane + Bending	24.5	46.2	15.0
Outer Top Cover Plate	Primary Membrane	1.4	36.5	7.3
	Membrane + Bending	2.2	51.9	41.2
Inner Top Cover Plate	Primary Membrane	1.4	33.7	6.6
	Membrane + Bending	1.9	50.8	18.5
Outer Bottom Cover Plate	Primary Membrane	1.4	23.9	6.3
	Membrane + Bending	2.3	34.7	25.1
Inner Bottom Cover Plate	Primary Membrane	7.6	39.4	1.7
	Membrane + Bending	25.7	40.5	3.7
Spacer Disc	Primary Membrane	0.0	50.4	N/A
	Membrane + Bending	76.5	78.3	N/A
Support Rods	Primary Membrane	0.0	2.0	N/A
	Membrane + Bending	0.0	21.5	N/A
Guide Sleeves	Primary Membrane	4.5	16.0	N/A
	Membrane + Bending	4.5	29.1	N/A
Support Ring	Primary Membrane	8.1	21.8	2.5
	Membrane + Bending	14.0	21.9	2.7

Notes:

1. Values shown are maximum irrespective of location.

Table 8-10

FC-DSC ISFSI Accident Condition Stresses

FC-DSC Component	Stress Type	Stress (ksi)(1)		
		Accidental Cask Drop		Accident Pressure
		End	Side	
Shell	Primary Membrane	9.8	28.9	5.2
	Membrane + Bending	24.5	46.2	15.0
Outer Top Cover Plate	Primary Membrane	1.4	36.5	7.2
	Membrane + Bending	19.5	51.9	41.2
Inner Top Cover Plate	Primary Membrane	1.4	33.7	6.6
	Membrane + Bending	18.0	50.8	18.5
Outer Bottom Cover Plate	Primary Membrane	1.18	39.4	0.37
	Membrane + Bending	10.9	40.5	3.39
Inner Bottom Cover Plate	Primary Membrane	0.81	39.4	N/A
	Membrane + Bending	27.3	40.5	N/A
Spacer Disc	Primary Membrane	0.0	50.4	N/A
	Membrane + Bending	76.5	78.3	N/A
Support Rods	Primary Membrane	0.0	2.0	N/A
	Membrane + Bending	0.0	21.5	N/A
Guide Sleeves	Primary Membrane	4.5	16.0	N/A
	Membrane + Bending	4.5	29.1	N/A
Support Ring	Primary Membrane	8.1	21.8	2.5
	Membrane + Bending	14.0	21.9	2.7

Notes:

1. Values shown are maximum irrespective of location.

Table 8-11

FF-DSC ISFSI Accident Condition Stresses

FF-DSC Component	Stress Type	Stress (ksi)[1]		
		Accidental Cask Drop		Accident Pressure
		End	Side	
Shell	Primary Membrane	9.8	28.9	5.2
	Membrane + Bending	24.5	46.2	15.0
Outer Top Cover Plate	Primary Membrane	1.4	36.5	7.2
	Membrane + Bending	19.5	51.9	41.2
Inner Top Cover Plate	Primary Membrane	1.4	33.7	6.6
	Membrane + Bending	18.0	50.8	18.5
Outer Bottom Cover Plate	Primary Membrane	1.18	39.4	0.37
	Membrane + Bending	10.9	40.5	3.39
Inner Bottom Cover Plate	Primary Membrane	0.81	39.4	N/A
	Membrane + Bending	27.3	40.5	N/A
Spacer Disc	Primary Membrane	0.0	29.9	N/A
	Membrane + Bending	26.8	68.4	N/A
Support Plates	Primary Membrane	10.0	1.0	N/A
	Membrane + Bending	38.6	3.9	N/A
Fuel Can	Primary Membrane	4.2	4.8	N/A
	Membrane + Bending	4.2	4.8	N/A

Notes:

1. Values shown are maximum irrespective of location.

Table 8-12

Cask Enveloping Load Combination Results for Normal and Off-Normal Loads (ASME Service Levels A and B)

Cask Component	Stress Type	Controlling Load Combination[1]	Stress (ksi) Calculated	Stress (ksi) Allowable	Stress Ratio
Inner Shell	Primary Membrane	A[2]	4.0	20.0	0.2
	Membrane + Bending	A[2]	4.0	30.0	0.13
	Primary + Secondary	A[2]	18.4	60.0	0.31
Outer Shell	Primary Membrane	A[2]	6.2	20.0	0.31
	Membrane + Bending	A[2]	21.9	30.0	0.73
	Primary + Secondary	A[2]	44.8	60.0	0.75
Top Cover Plate	Primary Membrane	A[2]	0.1	20.0	0.01
	Membrane + Bending	A[2]	1.5	30.0	0.05
	Primary + Secondary	A[2]	8.8	60.0	0.15
Top Corner Forging	Primary Membrane	A[2]	4.0	20.0	0.20
	Membrane + Bending	A[2]	4.0	30.0	0.13
	Primary + Secondary	A[2]	16.4	60.0	0.27
Bottom End Closure Forging	Primary Membrane	A[2]	4.0	20.0	0.20
	Membrane + Bending	A[2]	4.4	30.0	0.15
	Primary + Secondary	A[2]	14.6	60.0	0.24
Ram Access Cover Plate	Primary Membrane	A[2]	0.0	31.4	0.00
	Membrane + Bending	A[2]	0.3	47.1	0.01
	Primary + Secondary	A[2]	2.6	94.2	0.03

Notes:

1. The cask load combinations are defined in Table 3-10.
2. The enveloping Service Level A cask hypothetical storage load combination results are conservatively calculated as the sum of the bounding maximum stresses for all dead weight and normal handling conditions considered and the maximum normal/off-normal thermal and internal pressure.

Table 8-13

Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level C)

Cask Component	Stress Type	Controlling Load Combination[1]	Stress (ksi) Calculated	Stress (ksi) Allowable	Stress Ratio
Inner Shell	Primary Membrane	C1/C2[2]	5.9	24.0	0.25
	Membrane + Bending	C4[3]	16.2	36.0	0.45
Outer Shell	Primary Membrane	C1/C2[2]	10.0	24.0	0.42
	Membrane + Bending	C4[3]	29.2	36.0	0.81
Top Cover Plate	Primary Membrane	C4[3]	0.5	24.0	0.02
	Membrane + Bending	C4[3]	4.9	36.0	0.14
Top Corner Forging	Primary Membrane	C1/C2[2]	5.9	25.0	0.24
	Membrane + Bending	C1/C2[2]	5.9	37.5	0.16
Bottom End Closure Forging	Primary Membrane	C1/C2[2]	5.9	25.0	0.24
	Membrane + Bending	C1/C2[2]	7.7	37.5	0.21
Ram Access Cover Plate	Primary Membrane	C3	0.1	43.4	0.00
	Membrane + Bending	C4[3]	3.5	65.1	0.05

Notes:

1. The cask load combinations are defined in Table 3-10.
2. Load combination C1/C2 includes the bounding dead weight, thermal and normal handling stresses and therefore bounds load combinations C1 and C2.
3. C4, the governing load combination, includes flood loads on the cask during a hypothetical storage mode. This bounds the other load combinations, including the transfer mode.

Table 8-14

Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level D)

Cask Component	Stress Type	Controlling Load Combination[1]	Stress (ksi)		Stress Ratio
			Calculated	Allowable	
Inner Shell	Primary Membrane	D1	42.3	46.2	0.92
	Membrane + Bending	D1	67.3	68.5	0.98
Outer Shell	Primary Membrane	D1	37.7	46.2	0.82
	Membrane + Bending	D1	67.1	68.5	0.98
Top Cover Plate	Primary Membrane	D1	38.1	46.2	0.82
	Membrane + Bending	D1	47.2	66.0	0.72
Top Corner Forging	Primary Membrane	D1	40.2	48.0	0.84
	Membrane + Bending	D1	47.7	72.0	0.66
Bottom End Closure Forging	Primary Membrane	D1	35.3	48.0	0.74
	Membrane + Bending	D1	40.0	72.0	0.56
Ram Access Cover Plate	Primary Membrane	D1	11.1	66.0	0.17
	Membrane + Bending	D1	11.1	94.3	0.12

Notes:

1. The cask load combinations are defined in Table 3-10.

Table 8-15

FO-DSC Enveloping Load Combination Results for Normal and Off-Normal Loads (ASME Service Levels A and B)

FO-DSC Component	Stress Type	Controlling Load Combination	Stress (ksi)		Stress Ratio
			Calculated	Allowable	
Shell	Primary Membrane	B2	15.13	18.7	0.81
	Membrane + Bending	B2	23.38	26.3	0.89
	Primary + Secondary	B2	45.17	54.3	0/83
Outer Top Cover Plate	Primary Membrane	A3	4.45	18.1	0.25
	Membrane + Bending	A3	10.75	26.3	0.41
	Primary + Secondary	A3	27.03	54.3	0.50
Inner Top Cover Plate	Primary Membrane	A3	3.53	18.1	0.20
	Membrane + Bending	A3	9.24	26.3	0.35
	Primary + Secondary	A3	31.55	54.3	0.58
Outer Bottom Cover Plate	Primary Membrane	B2	13.77	18.1	0.74
	Membrane + Bending	B2	25.30	26.3	0.90
	Primary + Secondary	B2	35.56	54.3	0.65
Inner Bottom Cover Plate	Primary Membrane	B2	16.61	19.3	0.86
	Membrane + Bending	B2	27.06	29.0	0.93
	Primary + Secondary	B2	35.23	54.3	0.65
Spacer Disc	Primary Membrane	A3	9.20	26.4	0.35
	Membrane + Bending	A3	18.20	39.6	0.46
	Primary + Secondary	A3	63.40	79.2	0.80
Support Rods	Primary Membrane	A3	31.80	43.8	0.73
	Membrane + Bending	A3	33.00	65.7	0.50
	Primary + Secondary	A3	48.40	131.4	0.37
Guide Sleeves	Primary Membrane	A3	0.24	16.0	0.02
	Membrane + Bending	A3	2.73	24.0	0.11
	Primary + Secondary	N/A	N/A	N/A	N/A

Table 8-16

FO-DSC Enveloping Load Combination Results for Accident Loads

(ASME Service Level C)

FO-DSC Component	Stress Type	Controlling Load Combination	Stress (ksi) Calculated	Stress (ksi) Allowable	Stress Ratio
Shell	Primary Membrane	C4	20.81	22.4	0.93
	Membrane + Bending	C4	32	33.7	0.95
Outer Top Cover Plate	Primary Membrane	C1	7.97	21.7	0.37
	Membrane + Bending	C5	30.41	32.6	0.93
Inner Top Cover Plate	Primary Membrane	C1	7.85	21.7	0.36
	Membrane + Bending	C5	18.12	32.6	0.56
Outer Bottom Cover Plate	Primary Membrane	C4	19.25	22.4	0.86
	Membrane + Bending	C4	34.23	35	0.98
Inner Bottom Cover Plate	Primary Membrane	C4	12.59	22.4	0.56
	Membrane + Bending	C4	16.63	33.7	0.49
Spacer Disc	Primary Membrane	C1	11.5	39.6	0.29
	Membrane + Bending	C1	16.5	59.4	0.28
Support Rods	Primary Membrane	C1	31.8	65.7	0.48
	Membrane + Bending	C1	32.7	98.6	0.33
Guide Sleeves	Primary Membrane	C1	0.22	24.0	0.01
	Membrane + Bending	C1	2.53	36.0	0.07

Table 8-17

FO-DSC Enveloping Load Combination Results for Accident Loads

(ASME Service Level D)

FO-DSC Component	Stress Type	Controlling Load Combination	Stress (ksi) Calculated	Stress (ksi) Allowable	Stress Ratio
Shell	Primary Membrane	D2	28.92	44.5	0.65
	Membrane + Bending	D2	46.17	57.2	0.81
Outer Top Cover Plate	Primary Membrane	D2	36.51	44.5	0.82
	Membrane + Bending	D2	51.90	57.2	0.91
Inner Top Cover Plate	Primary Membrane	D2	33.70	44.5	0.76
	Membrane + Bending	D2	51.83	57.2	0.91
Outer Bottom Cover Plate	Primary Membrane	D2	23.86	44.5	0.54
	Membrane + Bending	D3	55.96	64.4	0.87
Inner Bottom Cover Plate	Primary Membrane	D2	39.40	44.5	0.89
	Membrane + Bending	D2	40.50	57.2	0.71
Spacer Disc	Primary Membrane	D2	50.40	55.5	0.91
	Membrane + Bending	D2	78.30	79.3	0.99
Support Rods	Primary Membrane	D2	33.80	105.1	0.32
	Membrane + Bending	D2	53.30	131.4	0.41
Guide Sleeves	Primary Membrane	D2	16.00	44.5	0.36
	Membrane + Bending	D2	29.10	57.2	0.51

Table 8-18

FC-DSC Enveloping Load Combination Results for Normal and Off-Normal Loads

(ASME Service Levels A and B)

FC-DSC Component	Stress Type	Controlling Load Combination	Stress (ksi) Calculated	Stress (ksi) Allowable	Stress Ratio
Shell	Primary Membrane	B2	15.13	18.7	0.81
	Membrane + Bending	B2	22.38	26.3	0.89
	Primary + Secondary	B2	45.17	54.3	0.83
Outer Top Cover Plate	Primary Membrane	A3	4.45	18.1	0.25
	Membrane + Bending	A3	10.75	26.3	0.41
	Primary + Secondary	A3	27.03	54.3	0.50
Inner Top Cover Plate	Primary Membrane	A3	3.65	18.7	0.21
	Membrane + Bending	A2	9.39	28.1	0.36
	Primary + Secondary	A3	31.7	54.3	0.58
Outer Bottom Cover Plate	Primary Membrane	B2	0.25	18.7	0.01
	Membrane + Bending	B2	2.30	28.1	0.09
	Primary + Secondary	B2	20.10	54.3	0.37
Inner Bottom Cover Plate	Primary Membrane	A3	0.47	18.1	0.03
	Membrane + Bending	A3	15.7	27.2	0.58
	Primary + Secondary	A3	43.7	54.3	0.80
Spacer Disc	Primary Membrane	A3	9.20	26.4	0.35
	Membrane + Bending	A3	18.20	39.6	0.46
	Primary + Secondary	A3	63.40	79.2	0.80
Support Rods	Primary Membrane	A3	31.80	43.8	0.73
	Membrane + Bending	A3	33.00	65.7	0.50
	Primary + Secondary	A3	48.40	131.4	0.37
Guide Sleeves	Primary Membrane	A1	0.24	16.0	0.02
	Membrane + Bending	A1	2.73	24.0	0.11
	Primary + Secondary	N/A	N/A	N/A	N/A

Table 8-19

FC-DSC Enveloping Load Combination Results for Accident Loads

(ASME Service Level C)

FC-DSC Component	Stress Type	Controlling Load Combination	Stress (ksi)		Stress Ratio
			Calculated	Allowable	
Shell	Primary Membrane	C4	20.81	22.4	0.93
	Membrane + Bending	C4	32	33.7	0.95
Outer Top Cover Plate	Primary Membrane	C1	7.97	21.7	0.37
	Membrane + Bending	C5	30.41	32.6	0.93
Inner Top Cover Plate	Primary Membrane	C1	7.85	21.7	0.36
	Membrane + Bending	C5	18.12	32.6	0.56
Outer Bottom Cover Plate	Primary Membrane	C1	0.33	22.4	0.02
	Membrane + Bending	C1	3.05	33.7	0.09
Inner Bottom Cover Plate	Primary Membrane	C1	0.01	22.4	0.00
	Membrane + Bending	C1	0.30	33.7	0.01
Spacer Disc	Primary Membrane	C1	11.5	39.6	0.29
	Membrane + Bending	C1	16.5	59.4	0.28
Support Rods	Primary Membrane	C1	31.8	65.7	0.48
	Membrane + Bending	C1	32.7	98.6	0.33
Guide Sleeves	Primary Membrane	C1	0.22	24.0	0.01
	Membrane + Bending	C1	2.53	36.0	0.07

Table 8-20

FC-DSC Enveloping Load Combination Results for Accident Loads

(ASME Service Level D)

FC-DSC Component	Stress Type	Controlling Load Combination	Stress (ksi)		Stress Ratio
			Calculated	Allowable	
Shell	Primary Membrane	D2	28.92	44.5	0.65
	Membrane + Bending	D2	46.17	57.4	0.81
Outer Top Cover Plate	Primary Membrane	D2	36.51	44.5	0.82
	Membrane + Bending	D2	51.90	57.2	0.91
Inner Top Cover Plate	Primary Membrane	D2	33.70	44.5	0.76
	Membrane + Bending	D2	51.83	57.2	0.91
Outer Bottom Cover Plate	Primary Membrane	D2	39.4	44.5	0.89
	Membrane + Bending	D2	40.5	57.2	0.71
Inner Bottom Cover Plate	Primary Membrane	D2	39.4	44.5	0.89
	Membrane + Bending	D2	40.5	57.2	0.71
Spacer Disc	Primary Membrane	D2	50.40	55.5	0.91
	Membrane + Bending	D2	78.30	79.3	0.99
Support Rods	Primary Membrane	D2	33.80	105.1	0.32
	Membrane + Bending	D2	53.30	131.4	0.41
Guide Sleeves	Primary Membrane	D2	16.00	44.5	0.36
	Membrane + Bending	D2	29.10	57.2	0.51

Table 8-21
FF-DSC Enveloping Load Combination Results for Normal and Off-Normal Loads (ASME Service Levels A and B)

FF DSC Component	Stress Type	Controlling Load Combination[1]	Stress Intensity (ksi)		Stress Ratio
			Calculated	Allowable	
Shell	Primary Membrane	B2	15.13	18.7	0.81
	Membrane + Bending	B2	23.38	26.3	0.89
	Primary + Secondary	B2	45.17	54.3	0.83
Outer Top Cover Plate	Primary Membrane	A3	4.45	18.1	0.25
	Membrane + Bending	A3	10.75	26.3	0.41
	Primary + Secondary	A3	27.03	54.3	0.50
Inner Top Cover Plate	Primary Membrane	A3	3.65	17.5	0.21
	Membrane + Bending	A3	9.39	26.3	0.36
	Primary + Secondary	A3	31.7	54.3	0.58
Outer Bottom Cover Plate	Primary Membrane	B2	0.25	18.7	0.01
	Membrane + Bending	B2	2.30	28.1	0.09
	Primary + Secondary	B2	20.10	54.3	0.37
Inner Bottom Cover Plate	Primary Membrane	A3	0.47	18.1	0.03
	Membrane + Bending	A3	15.7	27.2	0.58
	Primary + Secondary	A3	43.7	54.3	0.80
Spacer Disc[2]	Primary Membrane	A3	3.5	26.4	0.13
	Membrane + Bending	A3	3.7	39.6	0.09
	Primary + Secondary	A3	30.9	79.2	0.39
Support Plates[2]	Primary Membrane	A3	0.2	26.5	0.01
	Membrane + Bending	A3	0.9	39.8	0.02
	Primary + Secondary	A3	0.9	79.5	0.01
Fuel Can	Primary Membrane	A3	0.4	17.1	0.02
	Membrane + Bending	A3	0.4	25.7	0.02
	Primary + Secondary	A3	0.4	51.3	0.01

Note: 1. The DSC load combinations are defined in Table 3-9.

2. Thermal results shown are based on the analysis using SSA-537, Class 2 carbon steel. Additional evaluations show that the stress ratios for all load combinations are acceptable using SA-240, Type XM-19 stainless steel.

Table 8-22

FF-DSC Enveloping Load Combination Results for Accident Loads (ASME Service Level C)

FF DSC Component	Stress Type	Controlling Load Combination[1]	Stress Intensity (ksi) Calculated	Stress Intensity (ksi) Allowable	Stress Ratio
Shell	Primary Membrane	C4	20.81	22.4	0.93
	Membrane + Bending	C4	32	33.7	0.95
Outer Top Cover Plate	Primary Membrane	C1	7.97	21.7	0.37
	Membrane + Bending	C5	30.41	32.6	0.93
Inner Top Cover Plate	Primary Membrane	C1	7.85	21.7	0.36
	Membrane + Bending	C5	18.12	32.6	0.56
Outer Bottom Cover Plate	Primary Membrane	C1	0.33	22.4	0.02
	Membrane + Bending	C1	3.05	33.7	0.09
Inner Bottom Cover Plate	Primary Membrane	C1	0.01	22.4	0.00
	Membrane + Bending	C1	0.30	33.7	0.01
Spacer Disc[2]	Primary Membrane	C1	5.0	39.6	0.13
	Membrane + Bending	C1	5.0	59.4	0.08
Support Plates[2]	Primary Membrane	C1	0.1	39.8	0.00
	Membrane + Bending	C1	0.8	59.6	0.01
Fuel Can	Primary Membrane	C1	0.5	25.7	0.02
	Membrane + Bending	C1	0.5	38.5	0.01

Notes:

1. The DSC load combinations are defined in Table 3-9.

2. Thermal results shown are based on the analysis using SSA-537, Class 2 carbon steel. Additional evaluations show that the stress ratios for all load combinations are acceptable using SA-240, Type XM-19 stainless steel.

Table 8-23

FF-DSC Enveloping Load Combination Results for Accident Loads

(ASME Service Level D)

FF DSC Component	Stress Type	Controlling Load Combination[1]	Stress Intensity (ksi) Calculated	Stress Intensity (ksi) Allowable	Stress Ratio
Shell	Primary Membrane	D2	28.92	44.5	0.65
	Membrane + Bending	D2	46.17	57.2	0.81
Outer Top	Primary Membrane	D2	36.51	44.5	0.82
Cover Plate	Membrane + Bending	D2	51.90	57.2	0.91
Inner Top	Primary Membrane	D2	33.70	44.5	0.76
Cover Plate	Membrane + Bending	D2	51.83	57.2	0.91
Outer Bottom	Primary Membrane	D2	39.4	44.5	0.89
Cover Plate	Membrane + Bending	D2	40.5	57.2	0.71
Inner Bottom	Primary Membrane	D2	39.4	44.5	0.89
Cover Plate	Membrane + Bending	D2	40.5	57.2	0.71
Spacer	Primary Membrane	D2	29.9	55.5	0.54
Disc[2]	Membrane + Bending	D2	68.4	79.3	0.86
Support	Primary Membrane	D2	10.0	55.7	0.18
Rods[2]	Membrane + Bending	D2	38.6	79.5	0.49
Guide	Primary Membrane	D2	4.8	41.0	0.12
Sleeves	Membrane + Bending	D2	4.8	61.0	0.08

Notes:

1. The DSC load combinations are defined in Table 3-9.

2. Thermal results shown are based on the analysis using SSA-537, Class 2 carbon steel. Additional evaluations show that the stress ratios for all load combinations are acceptable using SA-240, Type XM-19 stainless steel.

Table 8-24

Maximum Pressure Differential Across DSC Shell(2,3)

Case	Ambient Air Temperature (°F)	DSC Cavity Average Helium Temperature (°F)	Maximum DSC Cavity Helium Pressure (psia)	Cask Annulus Average Temperature (°F)(2)	Maximum Cask Annulus Pressure(2) (psia)	Maximum Pressure Differential Across DSC Shell(1) (psi)
1	70 Vert in Cask	464	17.2	274	22.2	7.5
2	101 Vert in Cask	482	17.5	300	23.0	8.0
3	117 Vert in Cask	491	17.7	316	23.5	8.4

1. The direction of the pressure differential is inward toward the DSC cavity.
2. The postulated vertical DSC in storage is the worst case for differential pressure across DSC shell.
2. These differential pressures apply to the case of a non-leaking DSC postulated to be in storage in the MP187 cask (conservative worst case.

Table 8-25

Cask Cavity Pressure Assuming DSC Leakage After Placement in Storage

Deleted

9. CONDUCT OF OPERATIONS

This chapter describes the organization and general plans for operating the Rancho Seco ISFSI. The organization section includes a brief description of the responsibilities of managers, supervisors, and other key personnel. The training program for the plant staff is described, along with a more general discussion of replacement and retraining plans. Standards and procedures that govern daily operations and the records developed as a result of these operations are also discussed, as are the controls used to promote safety and ensure compliance with the license and the regulations under which the facility operates.

Initially, the managerial and administrative controls for the conduct of operations at the Rancho Seco ISFSI will be built upon the existing organization under the 10 CFR 50 license. The Superintendent, Rancho Seco Assets is currently responsible for oversight of the Rancho Seco facility and for ensuring the safe storage of the spent nuclear fuel and irradiated core components. This individual will continue to be responsible for safe storage of the fuel, and will be responsible for the safe management of the Rancho Seco ISFSI.

The administrative and procedural controls applicable to the 10 CFR 50 license have been expanded to include the requirements of the 10 CFR 72 license. Programs such as radiation protection, environmental monitoring, emergency preparedness, quality assurance, and training will be adapted as necessary to ensure the safe management of the ISFSI. SMUD has submitted and the NRC has approved the ISFSI security program which addresses the specific requirements for ISFSI security.

Upon termination of the 10 CFR 50 license, those license requirements will be removed from the procedures. Appropriate 10 CFR 72.48 reviews will be conducted to ensure continued compliance with ISFSI license requirements. This process will result in stand-alone ISFSI programs that implement the 10 CFR 72 license. SMUD will maintain the appropriate administrative and managerial controls at the Rancho Seco ISFSI until the DOE takes title to the fuel.

9.1 Organizational Structure

9.1.1 Corporate Organization

SMUD's organization and its relationship to the nuclear organization is presented in the Rancho Seco Defueled Safety Analysis Report (DSAR) [9.7.1]. Both Rancho Seco licensed facilities (ISFSI and Interim Onsite Storage Building) are managed under the same organization.

9.1.1.1 Corporate Functions, Responsibilities, and Authorities

SMUD's Board of Directors is the policy-making body which has ultimate responsibility for the Rancho Seco ISFSI license. The Chief Executive Officer & General Manager (GM) reports directly to the Board of Directors. The GM, through the Chief Generation and Grid Assets Officer, and Director, Power Generation has corporate responsibility for overall safety and management of the facility and shall take any measures needed to ensure acceptable performance of the staff in managing, maintaining, and providing technical support to the facility to ensure nuclear safety.

9.1.1.2 In-House Organization

The facility organization is described in the DSAR[9.7.1].

9.1.1.3 Interrelationship with Contractors and Suppliers

The prime contractor for design and analysis of the Rancho Seco ISFSI dry shielded canisters, horizontal storage modules, auxiliary and transfer equipment and casks is Transnuclear West, Inc. of Fremont, California. The prime contractor for the design of the Rancho Seco ISFSI civil facilities, including the storage pad, fencing and lighting system, etc. was Impell Corporation of San Ramon, California. Construction of the Rancho Seco ISFSI was the responsibility of BRCO Constructors, Inc. of Loomis, California. The Rancho Seco ISFSI is owned and operated by SMUD.

9.1.1.4 Technical Staff

The Corporate technical staff supporting the Rancho Seco ISFSI is described in the DSAR [9.7.1].

9.1.2 Operating Organization, Management and Administrative Control System

9.1.2.1 Onsite Organization

The RSNGS organization is responsible for management of the Rancho Seco ISFSI. This organization is described in DSAR [9.7.1].

9.1.2.2 Personnel Functions, Responsibilities and Authorities

The responsibilities and authority of major RSNGS positions or departments are summarized below. RSNGS personnel are selected and trained for their assigned duties, with particular emphasis on the supervisory and technical staffs to assure safe and efficient management of the Rancho Seco facilities.

Chief Generation and Grid Assets Officer

The Chief Generation and Grid Assets Officer is responsible for the overall Rancho Seco facility and the Rancho Seco organization. This includes ensuring the safe storage of irradiated core components, ensuring effective day-to-day management, and maximizing the effectiveness of nuclear policies and procedures.

Director, Power Generation

The Director, Power Generation is responsible for ensuring effective management of the licensed facilities, and ensuring the safe storage of irradiated core components.

Manager, Rancho Seco Assets

The Manager, Rancho Seco Assets (MRSA) is the lead SMUD representative at the Rancho Seco site and is responsible for all facets of day-to-day management of the licensed facilities.

The MRSA is responsible for site security during routine, emergency, and contingency operations. The MRSA is also responsible for the implementation and maintenance of the Physical Protection Plan.

The MRSA meets all qualifications for and is the Radiation Protection Manager and implements the Radiation Protection program. The MRSA is responsible for health physics surveillance, personnel monitoring and record keeping, radwaste management, emergency preparedness and environmental monitoring.

The MRSA utilizes available SMUD and contract personnel to resolve engineering, design, and other technical issues required to support the 10 CFR 72 ISFSI licensing process in accordance with applicable regulations as well as similar issues conducted under the 10 CFR 50 license.

The MRSA is responsible for ensuring that management of the Rancho Seco ISFSI is conducted in accordance with Technical Specifications, federal and state regulations, Physical Protection Plan, and plant procedures and has the primary responsibility for cask and canister handling operations.

Staff under the direction of the MRSA is engaged in a continual retraining program, as described in Section 9.3, to ensure that ISFSI operations are conducted in a safe and efficient manner.

Personnel under the direction of the MRSA as designated by site procedures check, analyze, and log system parameters, and initiate corrective actions when abnormal conditions exist. These personnel perform initial fire response and notifications in accordance with the fire protection program.

Individuals on shift are trained and qualified to implement appropriate radiation protection procedures.

Supporting Organizations outside Generation and Grid Assets

Audit & Quality Services is responsible for ensuring that the quality assurance program is implemented in accordance with regulatory requirements. The Audit & Quality Services organization has the authority to take any issue regarding the quality of program management at Rancho Seco to the General Manager and the Chief Generation and Grid Assets Officer.

Emergency Preparedness is responsible for maintaining and administering the Emergency Plan under the direction of the Manager, Rancho Seco Assets. The Emergency Preparedness staff trains all personnel implementing the Emergency Plan as well as directing drills and other activities necessary to maintain regulatory compliance.

Security is responsible for providing personnel as required to implement the Physical Protection Plan. Security is also responsible for staffing the security functions as required during routine, emergency and contingency conditions at the facility. Any or all of the Security function may be staffed by contracted personnel in the future: all contracted work will be under the direction of the Manager, Rancho Seco Assets.

9.1.3 Personnel Qualification Requirements

Each member of the Rancho Seco staff meets or exceeds the minimum qualifications of ANSI N18.1-1971 for comparable positions, except the Radiation Protection Manager position which meets or exceeds the qualifications of Regulatory Guide 1.8, September 1975.

Facility personnel are selected and trained for their assigned duties to ensure safe and efficient Rancho Seco ISFSI management.

Training, retraining, and replacement training programs for the maintenance staff and security force are maintained and conducted in accordance with approved procedures.

9.1.4 Liaison with Other Organizations

Interface with DOE, Transnuclear West, and other outside organizations is performed in accordance with contractual agreements.

9.2 Pre-Operational Testing and Operation

Before the operation of the Rancho Seco ISFSI, the electrical system, communications system, and transportable storage system will be tested to ensure their proper functioning.

The electrical system will be tested to ensure that power is available for lighting, security systems, and general service receptacles. The communications system will be tested to ensure that all ISFSI telephones are properly connected into the station phone system.

To the extent practicable, functional tests of the in-plant operations, transfer operations, and HSM loading and retrieval will be performed to verify that the storage system components (e.g., DSC, cask, transfer trailer, etc.) can be operated safely and effectively. Pre-operational testing may be performed using the actual cask and canister or a training cask and canister with test weights, as appropriate. The training cask and canister were designed and fabricated to approximate the size, weight, and behavior of an MP187 cask and canister.

9.2.1 Administrative Procedures for Conducting Test Program

The system for preparing, reviewing, approving, and implementing testing procedures, and instructions for the Rancho Seco ISFSI will be the same as those used for RSNGS. Any changes to, or deviations from, these procedures and instructions will be reviewed and approved in accordance with Technical Specification requirements of the 10 CFR 50 and 10 CFR 72 licenses.

9.2.2 Test Program Description

The objectives of the pre-operational testing program are to ensure that the storage system performs its intended safety functions and meets the operating controls and limits proposed in Chapter 10.

9.2.2.1 Physical Facilities and Operations

9.2.2.1.1 DSC and Associated Equipment

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

The part-length mock-up will be used for checkout of the automated welding and cutting equipment including actual welding and removal of the top cover plates. Emphasis will be placed on acceptability of the weld, as well as compliance with approved ALARA practices.

9.2.2.1.2 Cask and Handling Equipment

Functional testing will be performed with the cask and lifting yoke. These tests will ensure that the cask can be safely transported from the trailer loading area to the cask washdown area. From there, it will be placed into the spent fuel pool to verify clearances and travel path.

9.2.2.1.3 Off-Normal Testing of the DSC and Cask

In the unlikely event that a problem arises during actual loading of the spent fuel assemblies (SFAs) into the DSC, seal welding of the DSC, or emplacement of a loaded DSC into an

HSM, no immediate action would be required since the fuel assemblies would be in a safe condition. The pre-operational testing program will confirm that the SFAs can be safely removed from the DSC by demonstrating that the DSC lids can be removed.

9.2.2.1.4 Transfer Trailer and HSM

The cask will be placed on the transfer trailer, which will then be transported to the Rancho Seco ISFSI and aligned with an HSM. Compatibility of the transfer trailer with the transfer cask, negotiation of the travel path to the Rancho Seco ISFSI, and maneuverability within the confines of the Rancho Seco ISFSI will be verified.

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

9.2.2.1.5 Off-Normal Testing of the Transfer Trailer and HSM

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

The most severe condition would occur if a failure of the hydraulic ram, after partial insertion of a DSC into an HSM, were to prevent complete emplacement of the DSC. (Radiological shielding and decay heat removal are not compromised by this condition, but the transfer trailer may not be moved away until the DSC is completely within the confines of either the cask or the HSM.) Pre-operational testing will verify that reversal of DSC movement can be completed by the operator of the hydraulic ram.

9.2.3 Test Discussion

1. The purpose of the pre-operational tests is to ensure that a DSC can be properly and safely placed in the spent fuel pool, loaded with SFAs, inerted and sealed, transported to the Rancho Seco ISFSI, emplaced in the HSM, and removed from the HSM. Proper operation of the DSC, cask, and transfer trailer, as well as the associated handling equipment (e.g., lifting yoke, welding equipment, vacuum drying equipment), will provide such assurance.
2. Pre-operational test requirements will be specific. Detailed procedures will be developed and implemented by RSNGS personnel who will be responsible for ensuring that the test requirements are satisfied.
3. The expected results of the pre-operational tests are the successful completion of the following:
    - Loading a DSC into the transfer cask,
    - seal-welding and removal of the lids of the mock-up DSC,
    - placing a DSC and cask into and out of the spent fuel pool,

- transporting the cask loaded with a DSC and test weights to the Rancho Seco ISFSI, and
- DSC emplacement in an HSM and removal from an HSM,

The tests will be deemed successful if the expected results are achieved safely and without damage to any of the components or associated equipment.

4. Should any equipment or components require modification in order to achieve the expected results, it will be retested to affirm that the modification is sufficient. If any pre-operational test procedures are changed in order to achieve the expected results, the changes will be incorporated into the appropriate operating procedures.

9.3 Training Program

9.3.1 Program Description

The objective of SMUD's training program for the Rancho Seco ISFSI is to ensure a qualified work force for safe and efficient ISFSI management. The RSNGS training program will be used to provide this training and indoctrination and will be revised, as appropriate, to include information pertinent to the Rancho Seco ISFSI. All individuals working in the fuel storage area will receive radiation and safety training and those performing cask and fuel handling operations will be provided additional training, as required.

The training programs, in concert with other management systems, ensure that qualified individuals will be available to perform planned and unplanned tasks while protecting the health and safety of plant personnel and the public. SMUD will maintain additional training to support the emergency plan, physical security plan, quality assurance plan, and administrative and safety requirements, as required.

9.3.1.1 Scope of Training

The scope of training given to the Rancho Seco ISFSI staff will provide individuals with the necessary knowledge and skills to perform their job functions. SMUD will provide specialized training applicable to specific activities, tasks, and conditions, as needed. Contractors will be given safety, radiological, security, and site-specific training commensurate with their required duties.

Regarding the training requirements for access to the Rancho Seco ISFSI Controlled Area, individuals will be assigned to one of two categories: visitors or staff.

Visitors

Individuals who require access to Rancho Seco licensed facilities infrequently (e.g., tour groups, vendors, visiting managers) will be escorted by an approved, qualified employee while at Rancho Seco and will receive training in accordance with 10 CFR 20.

Staff

Individuals who will routinely work within Rancho Seco licensed facilities must have satisfactorily completed CAT I General Employee Training (GET), prior to working in these areas. Individuals who will routinely work in radiologically controlled areas must have satisfactorily completed CAT I and Controlled Area Radiation Protection training (CAT II).

Additionally, CAT I and CAT II training must be satisfactorily refreshed annually by taking Site Access Refresher Training and Controlled Area Refresher Training to maintain access to work areas.

In addition to the existing training program at RSNGS, SMUD will develop a training program for individuals involved in ISFSI management. The training program will be developed in accordance with administrative procedures, and will establish the requirements for the training and proficiency testing of individuals involved in ongoing management of the ISFSI. ISFSI training will include:

1. ISFSI facility design (overview)

2. License conditions and technical specifications (overview)
3. Off-normal event procedures

Training methods may include classroom instruction, on-the-job training, group briefings, or reading assignments.

Table 3-11 identifies the major components at the ISFSI that are important to safety. In the current long term storage condition, operation of equipment important to safety is not anticipated as a routine occurrence. Prior to initiation of operations involving use of equipment important to safety, all necessary training will be completed. Individuals who operate equipment that has been designated as important to safety will either be trained, or under the direct visual supervision of someone who is trained. Supervisory personnel who personally direct the operation of equipment that is important to safety will also be trained in such operations.

SMUD will select individuals for ISFSI operations to provide reasonable assurance that their physical condition and general health will not be such as might cause operational errors that could endanger in-plant personnel or public health and safety. The process for selecting individuals for ISFSI operations will give consideration for any condition that might cause impaired judgement or motor coordination. The following sections address the training requirements for individuals operating equipment important to safety.

9.3.1.1.1 Initial Training

The responsibility for each discipline training program is assigned to the MRSA. Classroom and laboratory training are provided when appropriate or necessary. On the Job Training (OJT) is provided within most disciplines. OJT consists of, but is not limited to, task training and evaluation, procedure training, and specific discipline-related training requirements.

9.3.1.1.2 Continuing Training

Training programs are designed to meet the specific needs of the participating disciplines and may include facility change review, procedure change review, administrative training commitments, OJT training review, and material from the initial training program.

9.3.1.2 Radiation Protection Technician Training

Both SMUD and contract Radiation Protection Technicians will be ANSI qualified. SMUD will provide initial training to all Radiation Protection Technicians to ensure they are qualified to perform assigned tasks. The Radiation Protection Technician shall participate in continuing training as needed.

9.3.1.3 Dry Fuel Storage Equipment Operator Training

Certified Dry Fuel Storage Equipment Operators will be responsible for fuel loading and cask/DSC handling and transfer operations. These individuals will be certified by the MRSA and meet the requirements of the Dry Fuel Storage Equipment Operator Training and Certification Program. This program meets the

requirements of 10 CFR 72, Subpart I. The Certified Dry Fuel Storage Equipment Operators shall participate in initial and proficiency training programs.

9.3.1.4 Maintenance Training

Each individual will be given instructions regarding the hazards and safety precautions applicable to the type of work to be performed, general work place hazards, and the procedures for protecting themselves from injury. Only qualified individuals will operate equipment, machinery, and cranes.

Also, maintenance personnel are trained in the operation of fork lifts and cranes, and they should have a working knowledge of the facility drawing system and the vendor manual system.

9.3.1.5 Trainer Qualifications

SMUD will select trainers to ensure they possess the knowledge, experience, and abilities to provide the required training. Instructors who teach the Certified ISFSI Operator Training Program shall be certified on the equipment being taught and shall participate in Proficiency Training.

9.3.2 Administration and Records

Training courses are prepared by individuals qualified in the particular topical or functional area.

SMUD will maintain training records and documents in accordance with appropriate regulatory requirements.

9.4 Normal Operations

9.4.1 Procedures

SMUD will prepare, review, and approve written procedures for all normal operations, maintenance, and testing at the Rancho Seco ISFSI prior to its operation. These procedures will be reviewed and approved as specified in the RSNGS Technical Specifications.

9.4.1.1 Administrative Procedures

Administrative procedures provide rules and instructions to all Rancho Seco ISFSI personnel regarding operating philosophy and management policies. These procedures include instructions pertaining to personnel conduct and control, including consideration of job related factors that influence the effectiveness of operating and maintenance personnel (e.g., work hours, entering and exiting the Rancho Seco ISFSI, organization, responsibilities, etc).

SMUD will establish procedures to ensure that the operation and maintenance of the Rancho Seco ISFSI is performed in accordance with the QA program described in Chapter 11.

9.4.1.2 Radiation Protection Procedures

Radiation Protection procedures are used to implement the radiation control program. The radiation control program involves the acquisition of data and use of equipment to perform radiation surveys, measurements, and evaluations to assess and control radiation hazards associated with the operation of the Rancho Seco ISFSI. SMUD has implemented procedures for:

1. Monitoring exposures to employees.
2. Using accepted radiation control techniques.
3. Performing radiation surveys of work areas.
4. Performing radiation monitoring of maintenance activities.
5. Maintaining records regarding measures taken to maintain radiation exposures to employees ALARA, and within administrative limits.

Entrance to the Rancho Seco ISFSI will be controlled by administrative procedures. SMUD will revise procedures, as necessary, to ensure the safety of individuals performing surveillance and maintenance at the Rancho Seco ISFSI.

9.4.1.3 Maintenance Procedures

SMUD will establish maintenance procedures for performing preventative and corrective maintenance on Rancho Seco ISFSI equipment. SMUD may perform preventative maintenance on a periodic basis to preclude the degradation of Rancho Seco ISFSI systems, equipment, and components. Unexpected system, equipment, or component malfunction will be evaluated to determine any corrective action needed.

9.4.1.4 Operating Procedures

The operating procedures will provide instructions for handling, loading, sealing, transporting, and storing the Rancho Seco ISFSI cask. Procedures will also be developed for removing fuel from a loaded DSC.

9.4.1.5 Test Procedures

Periodic test procedures will ensure that Rancho Seco ISFSI systems, equipment, and components are observed on an as needed basis to verify operability.

9.4.1.6 Pre-operational Test Procedures

As stated in Section 9.2, SMUD will establish pre-operational test procedures to ensure that Rancho Seco ISFSI systems and components will satisfactorily perform their required functions. These test procedures will further ensure that the Rancho Seco ISFSI has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public.

9.4.2 Records

The Rancho Seco ISFSI records will be maintained in accordance with the requirements of 10 CFR 72. Procedures will be established for retention of records during the construction phase, fuel loading phase and storage phases of the project.

For Special Nuclear Material (SNM) accountability, surveillance procedures have been developed for record-keeping relative to storage of fuel at the Rancho Seco ISFSI. The requirements of 10 CFR 72.72, 10 CFR 72.74, 10 CFR 72.76, and 10 CFR 72.78 are met by these procedures.

Each DSC will be labeled with a unique alpha-numeric designator. The alpha-numeric designator will be stamped into the DSC grapple ring using a low-stress die stamp. Also, each fuel assembly location within the DSC will be uniquely identified, as described in Rancho Seco ISFSI procedures. Each HSM will also have a unique designation.

HSM and DSC identification numbers along with individual assembly locations within a DSC, will be maintained in the SNM database. In this way, Rancho Seco ISFSI SNM accountability requirements will be met. Periodic physical inventory requirements will be met by verifying that HSMs have not been tampered with since the previous inventory.

9.5 Emergency Planning

The Rancho Seco Emergency Plan describes the organization, assessment actions, conditions for activation of the emergency organization, notification procedures, emergency facilities and equipment, training, provisions for maintaining emergency preparedness, and recovery criteria used at the Rancho Seco licensed facilities. This emergency plan is also used for emergencies that may arise at the Rancho Seco ISFSI. After the 10 CFR 50 license is terminated, the Rancho Seco emergency plan will remain in effect for the ISFSI to meet the requirements in 10 CFR 72.32.

SMUD will modify appropriate portions of the Emergency Plan and applicable implementing procedures to reflect the actions to be taken during the following events described in Chapter 8 for the Rancho Seco ISFSI:

1. Cask drop greater than 15 inches through air.

9.6 Decommissioning Plan

Due to the zero-leakage design of the NUHOMS DSCs, SMUD expects no residual contamination on the ISFSI concrete base pad. Nor does SMUD expect that the concrete pad will become activated. The loaded DSCs, HSMs, and the MP-187 cask are the only components at the Rancho Seco ISFSI that may need to be removed to complete radiological decommissioning. No other decommissioning activities are envisioned because of the absence of contaminated sources. The base pad, security fence, lighting, and peripheral utility structures will in effect be decommissioned when the last DSC, HSM, and MP-187 cask are removed. The non-contaminated components will be removed during ISFSI site restoration.

Five years prior to DOE taking title to the last of Rancho Seco's spent fuel, SMUD will conduct a decommissioning cost study that evaluates the various decommissioning options available. Based on the final decommissioning option chosen, SMUD will evaluate the need for decommissioning funding, if any. Funding for ISFSI decommissioning will be in accordance with the requirements in 10 CFR 72.30(c).

Record keeping in support of ISFSI decommissioning will be comprised of radiological records, fuel records, DSC records, and records of the facility including engineering drawings, plans, specifications, cost studies, etc. that affect decommissioning safety. The requirements for the retention of records are discussed in Section 10.2 of the ISFSI Safety Analysis Report (SAR).

The only component of the Rancho Seco ISFSI that may become contaminated is the MP-187 cask. The HSMs may become slightly radioactive due to neutron activation. After DOE has removed the last of the fuel from the ISFSI, SMUD will attempt to find another owner(s) for the HSMs and cask. If SMUD cannot find a second owner(s), the HSMs and cask will remain at the ISFSI to decay until they can be released for unrestricted use.

During the decay period, SMUD will maintain the HSMs and MP-187 cask intact and in a sound condition. Plant staff will perform minimal inspection and maintenance, and access to the ISFSI will be secured to provide controlled access. Anticipated activities include preventive and corrective maintenance on required security systems, area lighting, and general area maintenance and routine radiological inspections of the area. The Security Program will be designed to prevent inadvertent public access, and to provide a reasonable level of industrial security. At the end of the dormancy period, SMUD will dispose of the HSMs and MP-187 cask, as appropriate.

The Rancho Seco ISFSI Decommissioning Plan discusses the most likely scenarios for the disposition of spent fuel, and disposal of the ISFSI fuel storage system and support equipment. These scenarios may require amendment as the Federal Waste Management System matures.

DSCs

DOE has agreed to consider accepting Rancho Seco's fuel, as canistered in the NUHOMS storage system. DOE will accept the spent fuel at the ISFSI fence railroad gate, and will use an MP-187 cask to ship the fuel. DOE will not hold SMUD accountable if they later decide to repackage Rancho Seco's spent fuel for permanent disposal.

HSMs

The HSMs may become slightly radioactive due to neutron activation. District calculation Z-XXX-N0057, Revision 1, estimates the amount of activation that may occur within the NUHOMS HSMs. Components evaluated in the calculation include the concrete, heat shield, and canister support structure. Calculation Z-XXX-N0057 was provided as Appendix A to the Decommissioning Plan.

After DOE has removed the DSCs from the Rancho Seco ISFSI, the associated HSMs can be made available to DOE, or others, who will provide handling and transportation costs. Some support equipment may be reusable. If the HSMs cannot be sold, they will remain at the Rancho Seco ISFSI site until any activated material has decay to below releasable levels (approximately 1-2 years after removal of the loaded DSCs). The internal metal structures will be removed and recycled. The concrete will be demolished and buried.

MP-187 Cask

The MP-187 cask may also become slightly activated, and may have some internal and/or external contamination. After DOE has accepted the fuel, the cask may be made available to DOE, who will ultimately be responsible for cask decommissioning. If DOE has no use for the cask, it may be made available to others, who will provide handling and transportation costs, etc. If the MP-187 cask cannot be sold, it will remain in storage at the Rancho Seco site until it is free releasable, and can be disposed of. ISFSI concrete pad and remaining support equipment will not be activated or contaminated. These components will be demolished and disposed of. Based on the above, the DSCs, HSMs, and MP-187 cask can be disposed of without the need for low level waste disposal. Therefore, the only funding required for ISFSI decommissioning is that already provided for in the site restoration phase of decommissioning Rancho Seco. SMUD's Board of Directors has agreed, by resolution, to fund Rancho Seco site restoration and ISFSI decommissioning, and will begin funding after decommissioning. The funding program will be in accordance with 10 CFR 50.75(e) and/or 10 CFR 72.30(c), as appropriate.

9.7 References

9.7.1 Rancho Seco Nuclear Generating Station Defueled Safety Analysis Report (DSAR), Docket No. 50-312.

9.7.2 Rancho Seco Independent Spent Fuel Storage Installation Materials License SNM-2510 Amendment 4, November 2017 (Docket 72-11).

9.7.3 U.S. Nuclear Regulatory Commission, NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," June 2016.

9.7.4 ASME B&PV Code, Section III, Division 1, 1992 Edition, with Addenda through 1993.

9.7.5 Nuclear Energy Institute, "Format, Content and Implementation Guidance for Dry Storage Operations-Based Aging Management," NEI 14-03 Revision 2, 2016.

9.7.6 Enclosure 4 to Letter from Dan Tallman to NRC Document Control Desk, "RESPONSE TO REQUEST FOR CLARIFICATION OF RESPONSE TO ADDITIONAL INFORMATION FOR THE TECHNICAL REVIEW OF THE APPLICATION FOR RENEWAL OF THE RANCHO SECO INDEPENDENT SPENT FUEL STORAGE INSTALLATION LICENSE NO. SNM-2510 (CAC/EPID NOS. 001028/L-2018- RNW-0005; 000993/L-2018-LNE-0004)," dated July 12, 2019 (ADAMS ML19204A239)

9.7.7 U.S. Nuclear Regulatory Commission, "Safety Evaluation Report for the Rancho Seco Independent Spent Fuel Storage Installation", June 30, 2000. (ADAMS ML003729758)

9.8 Aging Management

Aging management activities (AMAs) have been undertaken in order to identify the structures, systems, and components (SSCs) and associated subcomponents of the Rancho Seco ISFSI that are within the scope of the Rancho Seco Independent Spent Fuel Storage Installation Materials License SNM-2510 Amendment 4, November 2017, [9.7.2] renewal. The methodology used to perform this scoping evaluation is based on the guidance contained in NUREG-1927 Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel, [9.7.3]. The scoping evaluation identifies the ISFSI SSCs that are within the scope of renewal, and therefore require an AMR. The AMR identifies the materials and environment for the SSCs and associated subcomponents determined to be within the renewal scope. These SSCs within the scope of renewal are further subjected to evaluation for potential degradation due to aging effects. After potential aging effects are identified, it is determined for each in-scope SSC whether they can be addressed by a TLAA, or if they will require an AMP. Additional details of the AMR can be found in Chapter 3. Listed in the subsections below are a description of the scoping evaluation methodology and the results of the scoping evaluation, the results of the AMR, TLAAs and the AMPs selected to manage the effects of aging over the PEO.

9.8.1 Scoping Evaluation Methodology

The scoping evaluation is performed based on the two-step process described in Section 2.4.2 of NUREG-1927, [9.7.3]. SSCs are considered to be within the scope of renewal if they satisfy either of the following two criteria:

Criterion 1: The SSC is classified as important-to-safety (ITS) as it is relied on to perform one of the following functions:

A. Maintain the conditions required by the regulations, specific license or certificate of compliance (CoC) 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.

These SSCs ensure that important safety functions are met for (1) criticality, (2) shielding, (3) confinement, (4) heat transfer, (5) structural integrity, and (6) spent fuel canister retrievability.

Criterion 2: The SSC is classified as not important-to-safety (NITS), but according to the licensing basis, its failure could prevent fulfillment of a function that is ITS, or its failure as a support SSC could prevent fulfillment of a function that is ITS.

The second step of the scoping evaluation includes a more detailed review of the SSCs that are determined to be within the scope of the renewal to identify and describe the subcomponents and subcomponent parts that support the intended function or functions of the SSCs. The intended functions of the SSC subcomponents (and the corresponding abbreviations used to denote this function) include:

- Providing criticality control of the spent fuel (CC),
- Providing heat transfer (HT),
- Directly or indirectly maintaining a pressure boundary (PB),
- Providing radiation shielding (SH),
- Providing structural support, functional support, or both, to SSCs that are ITS (SS) and
- Providing retrieval of spent fuel canister from the storage cask (RE).

The scoping of the spent fuel assemblies (SFAs) authorized for storage at the Rancho Seco ISFSI has been addressed as specified in Section 2.4.2.1 of NUREG-1927, [9.7.3].

The HSM, DSC, and transfer cask (TC) drawings list the associated subcomponents of the major SSCs, their quality categories and their materials of construction. The drawings and other licensing basis documents were reviewed to determine the SSCs and associated subcomponents that meet either Scoping Criterion 1 or 2. Based on this review, those SSCs and subcomponents that perform or support any of the identified intended functions are determined to require an AMR. Those SSCs and associated subcomponents that do not perform or support one or more of these intended safety functions are excluded from further evaluation.

9.8.2 Results of Scoping Evaluation

Table 9.8-1 summarizes the results of the scoping evaluation, listing the SSCs that are identified to be within the scope of renewal and the criterion upon which this determination is based. The SSCs determined to be within the scope of Rancho Seco ISFSI license renewal are the DSC (containing SFAs and GTCC waste), HSM, TC and ISFSI basemat. Detailed scoping evaluation results for each SSC and associated subcomponents have been generated. These results are included in Tables 9.8-3 to 9.8-8.

At the Rancho Seco ISFSI, all fuel is currently in dry storage and no more fuel or new canisters are to be loaded. Hence, the TC and the auxiliary equipment will not be used for any loading operations. The only remaining TC safety function during the PEO is retrieval of the DSCs from the HSMs for inspection or loading directly into a transport package for offsite shipment of the DSC.

The SFAs, which are stored in an inert and sealed environment and are supported inside the FO/FC/FF DSC basket assembly, are also determined to be within the scope of renewal. The in-scope subcomponents of the SFAs and their intended functions are identified in Table 9.8-2.

Note: The external design characteristics of the greater than class C (GTCC) waste containing DSC are identical to the fuel-containing DSCs (FO/FC/FF DSCs). Hence, the term “DSC” has been used to address both DSC types in this application, unless a distinction is necessary to address a specific characteristic of the GTCC DSC.

The storage pad (also referred to as the basemat) is a NITS structure. Since failure of the basemat may affect retrievability of the DSC, the basemat is included within the scope of renewal.

Structures, systems, and components that are not in the scope of renewal include the fuel transfer and auxiliary equipment. These components are classified as NITS and do not meet scoping Criterion 2. Also not within the scope are those NITS subcomponents of the DSC, HSM and TC that do not meet Criterion 2 because their failure does not prevent fulfillment of an ITS function.

The approach or apron slab is a NITS reinforced concrete structure, designed and constructed to Rancho Seco specific conditions. The slab provides access to the HSM but does not support the HSM. It does not provide a safety function, and its failure would not prevent fulfillment of a safety function of the HSM loaded with a DSC

9.8.3 Aging Management Review

The purpose of the AMR is to assess the proposed aging management activities of SSCs determined to be within the scope of renewal for the Rancho Seco ISFSI. The AMR addresses aging mechanisms and effects that could adversely affect the ability of the SSCs (and associated subcomponents) from performing their intended functions during the PEO.

The AMR process involves the following major steps:

- Identification of materials and environments;
- Identification of aging effects and mechanisms requiring management;
- Determination of the activities required to manage the effects and mechanisms of aging. This involves the identification of TLAAs or AMPs for managing the effects of aging;
- Evaluation of spent fuel (canister) retrievability during the period of extended operation.

For each SSC, the material of construction and the environment to which each SSC is exposed are determined. The component environments are determined based on the location of the component within the storage system. Once the component material/environment combinations are determined, potential aging effects requiring management are determined. Engineering literature, related research and industry information, and existing operating experience (OE) are reviewed to identify expected aging degradation mechanisms for different materials and environments. After the expected aging effects are identified, it is determined whether the effects can be addressed by analysis (TLAA), or will require an AMP.

The environments to which SSCs and associated subcomponents are exposed play a critical role in the determination of potential aging effects and mechanisms. The environments to which the Rancho Seco SSCs are exposed are affected by the characteristics of the ISFSI site environment, as well as by the component location within the storage system. There are five basic environments that apply for the Rancho Seco ISFSI SSCs.

- Inert Gas – Inert environment inside the DSC cavity. The spent fuel assemblies (SFAs), the DSC internal basket assembly, and the inside surfaces of the DSC shell and interior shell assembly subcomponents (e.g., inner top and bottom cover plates and top shield plugs) are exposed to the inert gas (helium) environment inside the shell assembly cavity. These components are exposed to significant neutron and gamma radiation. Refer to DSC cavity helium temperatures and pressures presented in Volume I, Tables 8-2a and 8-2b, for normal and off-normal conditions. Both temperature and pressure are expected to continuously decrease over the period of extended operation.
- Sheltered – Protected environment, such as HSM interior or TC stored indoors in an uncontrolled environment. A sheltered environment is a protected environment with no direct exposure to sun, wind, or precipitation. A sheltered environment may contain moisture and salts or other contaminants from the external ambient air. The temperature inside the HSM depends on the ambient air temperature and the heat load of the loaded canister. Components exposed to the HSM sheltered environment (interior side of the HSM walls, HSM steel, and DSC external shell assembly components) are exposed to neutron and gamma radiation to a lesser extent than those of the interior cavity of the DSC. Temperatures and radiation sources are expected to decrease over the PEO.
- Embedded or Encased – This environment applies for materials that are embedded or encased (sealed) inside another material. These include rebar and anchorage embedded in the HSM concrete, DSC bottom shield plug encased between the inner and outer cover plates, NS-3 solid neutron shielding material encased in the TC lids and between the TC neutron shield jacket and the structural shell, and the lead encased between the inner liner and the structural shell of the TC. Embedded or encased environments are exposed to radiation. The radiation source is expected to decrease over the period of extended operation.
- External (Yard and Outdoor) – During storage, the exterior surfaces of the HSM are exposed to all weather conditions, including insolation, wind, rain, snow, and plant-specific ambient air conditions, including moist atmospheric air, ambient temperatures, and humidity.
- Underground – At the Rancho Seco ISFSI, the HSMs are installed on a reinforced concrete basemat, which is constructed on compacted, engineered fill. The surfaces of the ISFSI basemat located below grade are exposed to soil. As discussed in Section 2.4.6, the groundwater at the Rancho Seco site is approximately 150 feet below grade and therefore, there is no source of soil moisture.

After the component material/environment combinations are determined, potential aging management effects are determined. Aging effects are the manifestation of aging mechanisms. The AMR process identifies both the aging effects and the associated aging mechanisms that cause them. Each subcomponent that was subjected to AMR was evaluated to determine if the potential aging effects and mechanisms were credible considering the various material/environment combinations.

Supplemental Evaluations

For the following aging mechanisms, supplemental evaluations were performed to show that the aging mechanisms did not require aging management:

A. Irradiation Embrittlement

This supplemental evaluation assessed the effect of neutron and gamma radiation on the DSC and HSM structural materials. Although the license renewal period is for 40 years, this evaluation takes into account the initial 20 years of storage period and an additional 80 years of storage period, for a total of 100 years. The maximum fluence is calculated as **[** Note[1] **]** for the DSC fuel compartment and the total gamma exposure is calculated as 3.58 E+09 rad. The neutron fluence and gamma exposure are below the threshold levels of concern for the DSC and HSM materials. The results indicate that there is no credible mechanical degradation occurring in compressive strength and tensile strength of the DSC shell, shield plug, and HSM components due to neutron fluence and gamma exposure levels. Therefore, irradiation embrittlement is not an aging effect requiring management for the DSC and HSM components.

B. Combustible Gas Generation

This supplemental evaluation documents an analysis performed to determine the quantity of combustible gases generated as a result of irradiation of the NS-3 neutron shield material for the MP187 TC during its function as a transfer cask. The combustible gas generation in the neutron shield material of the transfer cask was calculated based on the design basis radiological fuel loaded in a DSC inside the transfer cask. Considering that the total hydrogen mass originally present in the neutron shield material is approximately 209 kg, the fraction of hydrogen liberated from the neutron shield material over the service period of the cask is **[**Note[1]**]** by weight, which is statistically insignificant. Therefore, generation of combustible gases is not an aging mechanism requiring management for the NS-3 neutron shield material.

9.8.3.1 Results of Aging Management Review – DSC

The DSC performs the following intended functions:

CC Provides criticality control of spent fuel

HT Provides heat transfer

PB Directly or indirectly maintains a pressure boundary (confinement)

RE Provides retrievability of SFAs

SH Provides radiation shielding

SS Provides structural support (structural integrity)

The GTCC DSC is designed to store greater than Class C waste and, thus, criticality is not an intended function for the GTCC DSC.

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[1] The values stated on page C-7 of Reference [9.7.6] are incorporated by reference into this IFSAR

The Rancho Seco DSCs consist of two main subcomponents: the shell assembly and the internal basket assembly. The materials of construction for the DSC subcomponents that are subject to further AMR include stainless steel, carbon steel, and Boral. The following aging effects and mechanisms are applicable for DSC steel components:

Loss of Material

- Loss of material due to crevice corrosion – stainless steel
- Loss of material due to pitting corrosion – stainless steel
- Loss of material due to galvanic corrosion – Nitronic® 60 rail plate and graphite lubricant / stainless steel

Cracking

- Cracking due to stress corrosion cracking – stainless steel
- Cracking due thermal fatigue - stainless steel DSC pressure boundary

Loss of Criticality Control

- Loss of criticality control due to boron depletion - Boral

Tables 9.8-3 through 9.8-6 summarize the results of the AMR for the FO, FC, FF and GTCC DSCs, respectively. These tables include for each in-scope subcomponent the material, intended function, the environment, the associated aging effects requiring management, and the resulting aging management activity.

9.8.3.2 Results of Aging Management Review – HSM

The evaluation boundary for the HSM includes the entire HSM concrete structure and the steel support structure for the DSCs, which perform the following intended functions:

HT Provides heat transfer

RE Provides retrievability of DSC

SH Provides radiation shielding

SS Provides structural support (structural integrity)

The materials of construction for the subcomponents of the HSM are presented in the AMR tables listed in this section, and consist of reinforced concrete, carbon steel, Nitronic® 60 stainless steel, plain concrete and stainless steel.

The following aging effects and mechanisms for HSM concrete components are applicable:

Loss of Material

- Aggressive chemical attack
- Corrosion of embedded steel

Cracking

- Corrosion of embedded steel
- Reactions with aggregates

Change in Material Properties

- Leaching of Ca(OH)2
- Aggressive chemical attack

The following aging effects/mechanisms for HSM steel and other metal components are applicable:

Loss of Material

- Loss of material due to general corrosion – carbon steel
- Loss of material due to galvanic corrosion – Nitronic® 60 stainless steel DSC support rail plates
- Loss of material due to crevice corrosion – carbon steel and stainless steel
- Loss of material due to pitting corrosion – carbon steel and stainless steel

Cracking

- Stress Corrosion Cracking - welds attaching Nitronic® 60 stainless steel rail plate.

Table 9.8-7 summarizes the results of the AMR for the RS HSMs.

9.8.3.3 Results of Aging Management Review – Concrete Basemat

The basemat is a NITS reinforced concrete structure designed to support the HSMs and constructed to plant-specific site conditions. Failure of the ISFSI basemat may affect retrievability of the DSC. Therefore, the basemat is within scope of renewal.

The basemat is evaluated in support of the following intended function:

- Provides support for DSC retrievability.

The material of construction for the basemat is reinforced concrete.

The following aging effects and mechanisms are applicable for the concrete basemat:

Loss of Material

- Delayed Ettringite Formation (DEF)
- Aggressive chemical attack
- Corrosion of embedded steel

Cracking

- Reactions with aggregates
- Settlement
- Corrosion of embedded steel

Change in Material Properties
- Leaching of Ca(OH)2
- Aggressive chemical attack

The following aging effect and mechanisms are applicable to the carbon steel rebar:

Loss of Material
- General corrosion
- Pitting corrosion
- Crevice corrosion

The portion of the AMR associated with the concrete in Table 9.8-7 is also applicable to the concrete basemat exposed to the sheltered, external and underground environments.

9.8.3.4 Results of Aging Management Review – Transfer Cask

This section summarizes the results of the AMR for the in-scope subcomponents of the NUHOMS® MP187 Transfer Cask (MP187 TC). The evaluation boundary for the TC includes the TC subcomponents, which perform the following intended functions:

HT Provides heat transfer

RE Provides retrievability of DSC

SH Provides radiation shielding

SS Provides structural support (impact resistance, lifting, etc.)

The materials of construction for the subcomponents of the TC are presented in the AMR tables listed in this section, and consist of stainless steel, carbon steel, Elastomer O-rings, NS-3 neutron shielding, lead gamma shielding, and Nitronic® 60 canister rails.

Note: All of the Rancho Seco SFAs and GTCC waste have been loaded and placed into the HSMs for interim storage. The MP187 TC is currently stored in a sheltered environment and the only remaining MP187 TC function is to retrieve the DSCs for inspection or offsite shipment.

The MP187 TC is a cylindrical vessel with a welded bottom assembly and a bolted top cover plate. The TC is constructed from three concentric cylindrical shells to form an inner and outer annulus. The TC is normally stored in a sheltered environment between uses and during staging activities prior to each use. The total time exposure of the TC to the spent fuel pool environment (during prior loading operations) and to the external environment during transfer and retrieval operations) represents a negligible fraction of its total life span, including PEO. Hence, the only environment considered for the TC AMR is the sheltered environment.

The following aging effects and mechanisms for steel components are applicable:

Loss of Material

- Loss of material due to general corrosion – carbon steel
- Loss of material due to crevice corrosion – carbon steel and stainless steel
- Loss of material due to pitting corrosion – carbon steel and stainless steel
- Loss of material due to galvanic corrosion - stainless steel rail, inner shell, and bottom end closure.

Cracking

- Cracking due to thermal fatigue - carbon steel and stainless steel

Table 9.8-8 summarizes the results of the AMR for the TC.

9.8.3.5 Results of Aging Management Review – Spent Fuel Assemblies

The SFA principal function during dry storage is to maintain proper geometry and position of radioactive material through confinement. Although fuel cladding provides a confinement barrier, no credit is taken in the safety analysis for the fuel cladding as a confinement boundary. The evaluation boundary for the SFA includes the fuel cladding and end plugs, guide tubes, grid assemblies, upper nozzle, and bottom nozzle, which perform the following intended functions:

CC Provides criticality control

HT Provides heat transfer

PB Directly or indirectly maintains a pressure boundary (confinement)

SS Provides structural support (structural integrity)

The materials of construction of the SFA hardware consist of zirconium-based alloys, stainless steel and nickel-based alloys.

The AMR for the SFAs focuses primarily on the fuel rod cladding as it is considered the limiting component of the fuel assembly hardware because it serves as a barrier to fission products, provides defense-in-depth, and maintenance of its structural integrity ensures its retrievability from the DSC.

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

External

For SFAs, external environment refers to the internal DSC atmosphere. The storage atmosphere is predominantly helium with trace amounts of water vapor and air.

Internal

For SFAs, internal environment refers to the fuel rod interior. The fuel rod internal environment is assumed to be a combination of the original helium fill gas (during manufacturing) and fission products produced during reactor operation.
The materials inside the DSC, including the SFAs, cannot practically be inspected in-situ due to radiation levels and accessibility (i.e., DSC is seal welded). In preparation for dry storage, the DSC internals are vacuum dried and backfilled with helium to establish an inert gas environment in the DSC cavity. The DSC is leak tested to ensure that the inert gas environment is maintained so that the SFAs will not become subject to age-related degradation mechanisms during the storage period. A demonstration project provides the basis for the assertion that the SFAs will not degrade to unacceptable levels during the PEO. Therefore, no aging management program or activities are credited during the PEO for the Rancho Seco low burnup SFAs and associated subcomponents.

9.8.4 Summary of Time-Limited Aging Analyses

Time-limited aging analyses are prepared to assess SSCs that have a time-dependent operating life to demonstrate that the existing licensing basis remains valid and that the intended functions of the SSCs in scope of renewal are maintained during the PEO. Time dependency may entail fatigue life (cycles), change in a mechanical property, such as fracture toughness or strength of materials due to irradiation, or time-limited operation of a subcomponent.

9.8.4.1 DSC Time-Limited Aging Analyses

The following are summary descriptions of the TLAAs that were identified and prepared based on the AMR of the DSC.

A. Fatigue Evaluation of the DSCs

This TLAA documents the evaluation of the DSC pressure boundary subcomponents for pressure and temperature fluctuations in accordance with the provisions of NB 3222.4(d) of the ASME B&PV Code, Section III, Division 1, 1992 Edition, with Addenda through 1993, [9.7.4]. As provided by NB 3222.4(d) of the ASME B&PV Code, fatigue effects need not be specifically evaluated provided the six criteria in NB 3222.4(d) are met. This evaluation is performed considering a 60-year service life using maximum bounding initial DSC pressures and temperatures (at the beginning of storage). This evaluation shows that the six criteria of NB 3222.4(d) are met.

B. Boron Depletion

This TLAA performs an analysis to determine the amount of boron depletion in the FO and FC DSC poison plates during the PEO. Although the license renewal period is for 40 years, this evaluation takes into account the initial 20 years of storage period and an additional 80 years of storage period, for a total of 100 years.

Over a period of 100 years, the evaluation considers a bounding neutron irradiation rate, and indicates that the amount of B-10 depleted is negligible.

9.8.4.2 HSM Time-Limited Aging Analyses

No TLAAs are used to manage any expected aging effects of HSM components.

9.8.4.3 Transfer Cask Time-Limited Aging Analyses

The following are summary descriptions of the TLAAs that were identified and prepared based on the AMR of the TC.

Fatigue Evaluation of the TC

This TLAA documents the thermal fatigue analysis of the TC for pressure and temperature fluctuations in accordance with the provisions of NB 3222.4(d) of the ASME BP&V Code. As provided by NB 3222.4(d), fatigue effects need not be specifically evaluated provided the six criteria in NB 3222.4(d) are met. This evaluation is performed considering a 60-year service life using maximum bounding initial TC pressures and temperatures (at the beginning of storage). This evaluation shows that all the six criteria of NB 3222.4(d) are met.

9.8.4.4 ISFSI Basemat Time-Limited Aging Analyses

No TLAAs are used to manage any expected aging effects of the basemat components.

9.8.5 Summary of Aging Management Programs

Aging management programs are developed for managing the effects of aging. As appropriate, an AMP was created to summarize the activities or procedures implemented to monitor and manage the aging effects.

9.8.5.1 DSC Aging Management Program

The DSC External Surfaces AMP is employed to manage the aging effects and mechanisms for the DSC. The scope of the DSC AMP, parameters to be monitored, the criteria for selecting a DSC for the AMP, the detection of aging effects and the acceptance criteria are described in Table 9.8-9. Rancho Seco will implement the first (baseline) DSC inspections under this AMP on the selected DSCs in accordance with the program schedule as described in the table.

9.8.5.2 HSM Aging Management Program

The following program is employed to manage the aging effects and mechanisms for the HSM concrete and steel components:

HSM Aging Management Program for External and Internal Surfaces (applicable to HSM, DSC support structures)

The scope of the HSM AMP, parameters to be monitored, the criteria for selecting an HSM for the AMP, the detection of aging effects and the acceptance criteria are described in Table 9.8-10. Rancho Seco will implement the first (baseline) HSM inspections under this AMP on the selected HSM in accordance with the program schedule as described in the table.

9.8.5.3 TC Aging Management Program

The TC AMP is employed to manage the aging effects and mechanisms for the TC components. The scope of the TC AMP, parameters to be monitored, the detection of aging effects and the acceptance criteria are described in Table 9.8-11. Rancho Seco will perform the inspections and monitoring activities described in this AMP prior to use to identify areas of degradation. Evaluation of this information during preparations for DSC retrieval and transfer provides adequate predictability and allows time for corrective action, if required, in order for the TC to perform its intended functions.

9.8.5.4 ISFSI Basemat Aging Management Program

The Rancho Seco ISFSI Basemat AMP is employed to manage the aging effects and mechanisms for the basemat concrete. The scope of the ISFSI basemat AMP, the detection of aging effects, and the acceptance criteria are described in Table 9.8-12. Rancho Seco will implement the first (baseline) basemat inspections under this AMP in accordance with the program schedule as described in the table.

9.8.6 Aging Management Tollgates

The AMPs listed in Section 9.8.5 are subject to modification under 10 CFR 72.48 as new OE accumulates. For these AMPs, Rancho Seco will implement a program that is consistent with the guidelines of a generic AMP tollgate process described below.

The following definitions are reproduced from NEI 14-03, Format, Content and Implementation Guidance for Dry Storage Operations-Based Aging Management, [9.7.5]:

Tollgate: A requirement included in a renewed ISFSI license and associated UFSAR for the licensee to perform and document an assessment of the aggregate impact of aging-related dry cask storage (DCS) OE, research, monitoring, and inspections at specific points in time during the renewed operating period.

Tollgate Assessment: A written evaluation, performed by licensees at each tollgate, of the aggregate impact of aging-related DCS OE, research, monitoring, and inspections on the intended functions of in-scope DCS SSCs. Tollgate assessments are intended to include non-nuclear and international operating information on a best-effort basis. Corrective or mitigative actions arising from tollgate assessments are managed through the corrective action programs of the licensee, the certificate holder, or both.

Corrective actions may include

- Modification of TLAAs
- Adjustment of the scope, frequency, or both of AMPs
- Repair or replacement of SSCs

Rancho Seco will assess new information relevant to aging management, as it becomes available, in accordance with normal corrective action and OE programs. Tollgates are an opportunity to seek out other information that may be available and perform an aggregate assessment. Tollgate assessments are not stopping points. No action other than performing an assessment is required to continue NUHOMS® dry storage system operation.

SMUD will perform tollgate assessments during the PEO of the Rancho Seco ISFSI, spanning the period from June 30, 2020 through June 30, 2060 or the date the last licensed material is removed from the Rancho Seco ISFSI, whichever occurs sooner. Tollgate assessments for the Rancho Seco ISFSI will be performed as shown in Table 9.8-13.

Table 9.8-1
Scoping Evaluation of Rancho Seco ISFSI SSCs

SSC	Criterion 1	Criterion 2	In-Scope
Dry Shielded Canister (DSC)(1)	Yes	N/A	Yes
HSM(2)	Yes	N/A	Yes
Transfer Cask (TC)(3)	Yes	N/A	Yes
Transfer Cask Lifting Yoke and extensions (4)	No	No	No
Spent Fuel Assemblies(5)	Yes	N/A	Yes
ISFSI Basemat(6)	No	Yes	Yes
ISFSI Approach Slab	No	No	No
Other Transfer Equipment(7)	No	No	No
Auxiliary Equipment(8)	No	No	No
Miscellaneous Equipment(9)	No	No	No
GTCC Waste	No	No	No

Notes:

(1) The DSC includes (but is not limited to) the DSC shell confinement boundary assembly and the internal basket assembly, siphon and vent block, support ring, lifting lugs. There are three types of DSCs and one GTCC canister licensed for Rancho Seco ISFSI: NUHOMS® FO-DSC, NUHOMS® FC-DSC, NUHOMS® FF-DSC and NUHOMS® GTCC DSC.

(2) The HSM includes (but is not limited to) the HSM reinforced concrete walls, roof, and end/rear shield walls; DSC steel structure support assembly; HSM accessories (DSC seismic retainer, heat shield panels, shielded door assemblies and door supports); associated attachment/installation hardware (tie rods, bolts, nuts, washers, embedment assemblies, mechanical splices); ventilation inlet vent openings and bird screens, ventilation outlet vent openings and bird screens, and outlet vent reinforced concrete covers.

(3) Transfer Cask includes (but is not limited to) the MP187 cask structural shell assembly, cask inner liner, upper and lower trunnion assemblies, lead gamma shielding, neutron shield plug, solid neutron shielding, top cover assembly, ram access penetration, bottom cover assembly.

(4) The TC lifting yoke and extensions were used for handling of the TC within the fuel/reactor building and were designed and procured as “safety-related” components by the licensee under 10 CFR Part 50. Per FSAR Section 3.2, they are not important-to-safety for storage purposes. They will not be used again since all spent fuel has been transferred into DSCs and the site has been decommissioned and the Part 50 license terminated by the NRC. Therefore, they are out of scope.

(5) Spent Fuel Assemblies - SFA cladding and assembly hardware listed in Table 2-9 are included in-scope.

(6) ISFSI Basemat - See discussion in Section 2.3.1.

(7) Other transfer equipment includes a hydraulic ram system, a prime mover for towing, a transfer trailer, a ram support assembly, a cask support skid, auxiliary equipment mounted on the skid, and a skid positioning system.

(8) Auxiliary equipment to facilitate canister loading, draining, drying, inerting, and sealing operations includes (but is not limited to) the following five systems: a vacuum drying system, an automatic welding system, the waste processing system, the security system, and the temperature monitoring system.

(9) Miscellaneous equipment includes (but is not limited to) ISFSI security fence and gate(s), lighting, lightning protection, communications, monitoring, and alarm systems.

N/A: Not Applicable.

Table 9.8-2
Scoping Evaluation Results for SFAs

Subcomponent	Intended Function					
	Confinement	**Shielding**	**Criticality**	**Structural**	**Thermal**	**Retrievability**
Fuel Pellets	Not in-scope: fuel pellets are not credited in any safety analyses and do not affect any safety functions					
Fuel Cladding (2)	Yes(1)	No	Yes	Yes	Yes	Note 3
Spacer Grid Assemblies	No	No	Yes	Yes	No	Note 3
Upper End Fitting/Nozzle (and related subcomponents	No	No	No	Yes	No	Note 3
Lower End Fitting/Nozzle (and related subcomponents)	No	No	No	Yes	No	Note 3
Guide Tubes	No	No	Yes	Yes	No	Note 3
Hold Down Spring and Upper End Plugs	Not in-scope: control components are not credited in any safety analyses and do not affect any safety functions					
Control Components	Not in-scope: hold down springs and upper end plugs are not credited in any safety analyses and do not affect any safety functions					

Notes:

(1) Though fuel cladding is the first barrier for confinement of radioactive materials, no credit for confinement of radioactive material is taken for the fuel cladding in the SMUD ISFSI design and licensing basis. The DSC pressure boundary is the only credited confinement boundary.

(2) Zircaloy-4 for Rancho Seco.

(3) The licensing basis for retrievability with respect to the Rancho-Seco ISFSI is defined on a canister basis, consistent with ISG-2 Revision 0, as described in ISFSI FSAR Section 4.2.2.1.12. The NRC has accepted this approach as documented in SER Section 4.3.7 [9.7.7].

Table 9.8-3
Rancho Seco FO-DSC Intended Functions and AMR Results
(2 pages)

Subcomponent(2)	Subcomponent Parts(2)	Material(2)	Drawings/Part #(2)	Intended Function(1)(2)	Environment	Aging Effects Requiring Management	Aging Management Activity
Main Assembly	Cylindrical Shell	Note 3	Note 3	PB, SH, SS, HT, RE	Inert Gas / Sheltered	Loss of Material, Cracking	AMP, TLAA
Main Assembly	Outer Bottom Cover	Note 3	Note 3	SH, SS, RE	Sheltered / Embedded	Loss of Material, Cracking	AMP
Main Assembly	Grapple Ring	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Grapple Ring Support	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Inner Bottom Cover	Note 3	Note 3	PB, SH, SS	Inert Gas / Embedded	Cracking	TLAA
Main Assembly	Spacer Discs (Type “A” “B” and Type “C”)	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Guide sleeve	Note 3	Note 3	CC, SS, HT	Inert Gas	None Identified	None Required
Main Assembly	Oversleeve	Note 3	Note 3	CC, SS, HT	Inert Gas	None Identified	None Required
Main Assembly	Neutron Absorber Sheet	Note 3	Note 3	CC, HT	Inert Gas	Loss of Criticality Control	TLAA
Main Assembly	Support Rod	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Shear Key	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Extension Plate	Note 3	Note 3	CC, SS, HT	Inert Gas	None Identified	None Required
Main Assembly	Key	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Siphon & Vent Block	Note 3	Note 3	PB, SH, SS	Inert Gas	Cracking	TLAA
Main Assembly	Lifting Lug	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Support Ring	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Top Shield Plug	Note 3	Note 3	SH, SS	Inert Gas	None Identified	None Required
Main Assembly	Bottom Shield Plug	Note 3	Note 3	SH, SS	Embedded	None Identified	None Required
Main Assembly	Inner Top Cover Plate	Note 3	Note 3	PB, SH, SS	Embedded / Inert Gas	Cracking	TLAA
Main Assembly	Outer Top Cover Plate	Note 3	Note 3	PB, SH, SS, RE	Embedded / Sheltered	Loss of Material, Cracking	AMP, TLAA
Main Assembly	Siphon & Vent Port Cover Plate	Note 3	Note 3	PB, SH, SS	Inert Gas / Embedded	Cracking	TLAA

Table 9.8-3
Rancho Seco FO-DSC Intended Functions and AMR Results
(2 pages)

Subcomponent[(2)]	Subcomponent Parts[(2)]	Material[(2)]	Drawings/Part #[(2)]	Intended Function[(1)(2)]	Environment	Aging Effects Requiring Management	Aging Management Activity
Main Assembly	Top and Bottom End Spacer Sleeve	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Spacer Sleeves (Type 1, 2, 3, 4, 5, 6)	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Stop Plate	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Plate 0.085 Thk	Note 3	Note 3	SS	Inert Gas	None Identified	None Required

Notes:

(1) Abbreviations for Intended Function Column:

PB - Directly or indirectly maintains a pressure boundary (confinement)
SH - Provides radiation shielding
CC - Provides criticality control of spent fuel
SS - Provides structural support
HT - Provides heat transfer
RE - Retrievability

(2) Only in-scope subcomponents from Chapter 2, Table 2-6 of Reference [9.7.6] are listed in this table.

(3) The values stated on page 3-80 through 3-81 of Reference [9.7.6] are incorporated by reference into this IFSAR.

Table 9.8-4
Rancho Seco FC-DSC Intended Functions and AMR Results
(2 pages)

Subcomponent[(2)]	Subcomponent Parts[(2)]	Material[(2)]	Drawings/Part #[(2)]	Intended Function[(1) (2)]	Environment	Aging Effects Requiring Management	Aging Management Activity
Main Assembly	Cylindrical Shell	Note 3	Note 3	PB, SH, SS, HT, RE	Inert Gas / Sheltered	Loss of Material, Cracking	AMP, TLAA
Main Assembly	Outer Bottom Cover	Note 3	Note 3	SH, SS, RE	Sheltered / Embedded	Loss of Material, Cracking	AMP
Main Assembly	Lead Shielding	Note 3	Note 3	SH	Embedded / Encased	None Identified	None Required
Main Assembly	Grapple Ring	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Grapple Ring Support	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Inner Bottom Cover	Note 3	Note 3	PB, SH, SS	Inert Gas / Embedded	Cracking	TLAA
Main Assembly	Bottom Plug Post	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Spacer Discs (Type “A” “B” and Type “C”)	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Guide sleeve	Note 3	Note 3	CC, SS, HT	Inert Gas	None Identified	None Required
Main Assembly	Oversleeve	Note 3	Note 3	CC, SS, HT	Inert Gas	None Identified	None Required
Main Assembly	Neutron Absorber Sheet	Note 3	Note 3	CC, HT	Inert Gas	Loss of Criticality Control	TLAA
Main Assembly	Support Rod	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Shear Key	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Extension Plate	Note 3	Note 3	CC, SS, HT	Inert Gas	None Identified	None Required
Main Assembly	Key	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Siphon & Vent Block	Note 3	Note 3	PB, SH, SS	Inert Gas	Cracking	TLAA
Main Assembly	Lifting Lug	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Support Ring	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Inner Top Cover Plate	Note 3	Note 3	PB, SH, SS	Embedded / Inert Gas	Cracking	TLAA
Main Assembly	Outer Top Cover Plate	Note 3	Note 3	PB, SH, SS, RE	Embedded / Sheltered	Loss of Material, Cracking	AMP, TLAA
Main Assembly	Siphon & Vent Port Cover Plate	Note 3	Note 3	PB, SH, SS	Inert Gas / Embedded	Cracking	TLAA

Table 9.8-4
Rancho Seco FC-DSC Intended Functions and AMR Results
(2 pages)

Subcomponent[2]	Subcomponent Parts[2]	Material[2]	Drawings/Part #[2]	Intended Function[1] [2]	Environment	Aging Effects Requiring Management	Aging Management Activity
Main Assembly	Top Shield Plug Casing	Note 3	Note 3	SH, SS	Inert Gas / Embedded	None Identified	None Required
Main Assembly	Top Shield Plug Post	Note 3	Note 3	SS	Embedded	None Identified	None Required
Main Assembly	Plate Stiffening	Note 3	Note 3	SS	Embedded	None Identified	None Required
Main Assembly	Top and Bottom End Spacer Sleeve	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Spacer Sleeves (Type 1, 2, 3, 4, 5, 6)	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Angle, 1-1/4 x 1-1/4 x ¼	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Plate 1.25x1.25 x ¼	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Stop Plate	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Plate 0.085 Thk	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Bottom Plug Top and Side Casing	Note 3	Note 3	SH, SS	Embedded	None Identified	None Required
Main Assembly	Plate Stiffening	Note 3	Note 3	SS	Embedded	None Identified	None Required

Notes:

(1) Abbreviations for Intended Function Column:

PB - Directly or indirectly maintains a pressure boundary (confinement)
SH - Provides radiation shielding
CC - Provides criticality control of spent fuel
SS - Provides structural support
HT - Provides heat transfer
RE - Retrievability

(2) Only in-scope subcomponents from Chapter 2, Table 2-6 of Reference [9.7.6] are listed in this table.

(3) The values stated on pages 3-82 through 3-84 of Reference [9.7.6] are incorporated by reference into this IFSAR.

Table 9.8-5
Rancho Seco FF-DSC Intended Functions and AMR Results
(2 pages)

Subcomponent(2)	Subcomponent Parts(2)	Material(2)	Drawings/Part #(2)	Intended Function(1) (2)	Environment	Aging Effects Requiring Management	Aging Management Activity
Main Assembly	Cylindrical Shell	Note 3	Note 3	PB, SH, SS, HT, RE	Inert Gas / Sheltered	Loss of Material, Cracking	AMP, TLAA
Main Assembly	Outer Bottom Cover	Note 3	Note 3	SH, SS, RE	Sheltered / Embedded	Loss of Material, Cracking	AMP
Main Assembly	Key	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Grapple Ring	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Grapple Ring Support	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Inner Bottom Cover	Note 3	Note 3	PB, SH, SS	Inert Gas / Embedded	Cracking	TLAA
Main Assembly	Spacer Discs	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Inner and Outer Support Plate	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Lead Shielding	Note 3	Note 3	SH	Embedded / Encased	None Identified	None Required
Main Assembly	Bottom Plug Post	Note 3	Note 3	SS	Embedded	None Identified	None Required
Main Assembly	Siphon & Vent Block	Note 3	Note 3	PB, SH, SS	Inert Gas	Cracking	TLAA
Main Assembly	Lifting Lug	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Support Ring	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Inner Top Cover Plate	Note 3	Note 3	PB, SH, SS	Embedded / Inert Gas	Cracking	TLAA
Main Assembly	Outer Top Cover Plate	Note 3	Note 3	PB, SH, SS, RE	Embedded / Sheltered	Loss of Material, Cracking	AMP, TLAA
Main Assembly	Siphon & Vent Port Cover Plate	Note 3	Note 3	PB, SH, SS	Inert Gas / Embedded	Cracking	TLAA
Main Assembly	Top Shield Plug Casing	Note 3	Note 3	SH, SS	Inert Gas Embedded	None Identified	None Required
Main Assembly	Top Shield Plug Post	Note 3	Note 3	SS	Embedded	None Identified	None Required
Main Assembly	Liner	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Flange Plate	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Shear Key	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP

Table 9.8-5
Rancho Seco FF-DSC Intended Functions and AMR Results
(2 pages)

Subcomponent(2)	Subcomponent Parts(2)	Material(2)	Drawings/Part #(2)	Intended Function(1) (2)	Environment	Aging Effects Requiring Management	Aging Management Activity
Main Assembly	Top Lid Cover Plate	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Bottom Lid Adapter Plate	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Top Lid Lifting Pintle	Note 3	Note 3	SS	Inert Gas	None Identified	None Required
Main Assembly	Mesh, 6x6	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Washer Plate	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Spacer Bar	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Cover Plate	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Side Lid Plate	Note 3	Note 3	CC, SS	Inert Gas	None Identified	None Required
Main Assembly	Bottom Plug Top and Side Casing	Note 3	Note 3	SH, SS	Embedded	None Identified	None Required
Main Assembly	Plate Stiffening	Note 3	Note 3	SS	Embedded	None Identified	None Required
Main Assembly	Plate Stiffening	Note 3	Note 3	SS	Embedded	None Identified	None Required

Notes:

(1) Abbreviations for Intended Function Column:

PB - Directly or indirectly maintains a pressure boundary (confinement)
SH - Provides radiation shielding
CC - Provides criticality control of spent fuel
SS - Provides structural support
HT - Provides heat transfer
RE - Retrievability

(2) Only in-scope subcomponents from Chapter 2, Table 2-6 of Reference [9.7.6] are listed in this table.

(3) The values stated on pages 3-85 through 3-87 of Reference [9.7.6] are incorporated by reference into this IFSAR.

Table 9.8-6
Rancho Seco GTCC DSC Intended Functions and AMR Results

Subcomponent[(2)]	Subcomponent Parts[(2)]	Material[(2)]	Drawings/Part #[(2)]	Intended Function[(1)(2)]	Environment	Aging Effects Requiring Management	Aging Management Activity
Main Assembly	Cylindrical Shell	Note 3	Note 3	PB, SH, SS, RE	Inert Gas / Sheltered	Loss of Material, Cracking	AMP, TLAA
Main Assembly	Bottom Shield Plug	Note 3	Note 3	PB, SH, SS, RE	Embedded / Inert Gas / Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Grapple Ring	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Grapple Ring Support	Note 3	Note 3	SS, RE	Sheltered	Loss of Material, Cracking	AMP
Main Assembly	Top Shield Plug	Note 3	Note 3	SH	Embedded (Inert Gas	None Identified	None Required
Main Assembly	Outer Top Cover Plate	Note 3	Note 3	PB, SH, SS, RE	Sheltered / Embedded	Loss of Material, Cracking	AMP, TLAA
Main Assembly	Outer Bottom Cover Plate	Note 3	Note 3	SH, SS, RE	Sheltered / Embedded	Loss of Material, Cracking	AMP
Main Assembly	Siphon & Vent Port Cover Plate	Note 3	Note 3	PB, SH, SS	Inert Gas / Embedded	Cracking	TLAA
Basket	Bottom Plate	Note 3	Note 3	SH	Inert Gas	None Identified	None Required
Basket	Cylindrical Shell	Note 3	Note 3	SH	Inert Gas	None Identified	None Required

Notes:

(1) Abbreviations for Intended Function Column:

PB - Directly or indirectly maintains a pressure boundary (confinement)
SH - Provides radiation shielding
CC - Provides criticality control of spent fuel
SS - Provides structural support
HT - Provides heat transfer
RE - Retrievability

(2) Only in-scope subcomponents from Chapter 2, Table 2-6 of Reference [9.7.6] are listed in this table.

(3) The values stated on page 3-88 of Reference [9.7.6] are incorporated by reference into this IFSAR.

Table 9.8-7
Rancho Seco HSM Intended Functions and AMR Results
(3 pages)

Subcomponent(2)	Subcomponent Parts(2)	Material(2)	Drawings/Part #(2)	Intended Function (1) (2)	Environment	Aging Effects Requiring Management	Aging Management Activity
Base Unit Assembly	HSM Base Walls & Floor Slab	Note 3	Note 3	SH, SS, HT, RE	External / Sheltered	Loss of Material, Cracking, Change in Material Properties	AMP
Roof Slab Assembly	HSM Roof Slab	Note 3	Note 3	SH, SS, HT	External / Sheltered	Loss of Material, Cracking, Change in Material Properties	AMP
End/Rear Shield Walls	End and Rear Shield Walls	Note 3	Note 3	SH, SS, HT	External / Sheltered	Loss of Material, Cracking, Change in Material Properties	AMP
DSC Support Structure Assembly	Support Rail Beams and Cross Beams	Note 3	Note 3	SS, RE	Sheltered	Loss of Material	AMP
DSC Support Structure Assembly	Support Rail Plate	Note 3	Note 3	RE	Sheltered	Loss of Material / Cracking	AMP
DSC Support Structure Assembly	Support Structure Steel (Rail Extension Plate, DSC Stop Plates, Stiffener Plates, Gussets, Mounting Plates, Base Plates, Support Plate, Wall Attachment Channel and Angles	Note 3	Note 3	SS, RE	Sheltered	Loss of Material	AMP
DSC Support Structure Assembly	Tube Steel Leg Column	Note 3	Note 3	SS	Sheltered	Loss of Material	AMP
DSC Support Structure Assembly	Bolts	Note 3	Note 3	SS	Sheltered	Loss of Material	AMP
DSC Support Structure Assembly	Nuts	Note 3	Note 3	SS	Sheltered	Loss of Material	AMP
DSC Support Structure Assembly	Wall Attachment Hardware (Heavy Hex Bolt/Hardened Washer)	Note 3	Note 3	SS	Sheltered	Loss of Material	AMP
HSM Shielded Door Assembly	Steel Plates (Various)	Note 3	Note 3	SH, SS, RE	External	Loss of Material	AMP
HSM Shielded Door Assembly	Encased Concrete Core	Note 3	Note 3	SH, RE	Embedded / Encased	None Identified	None Required
Canister Axial Retainer Assembly	Axial Retainer Rod/Mounting Plate /Bolts /Hardened Washer	Note 3	Note 3	SS	External / Sheltered	Loss of Material	AMP

Table 9.8-7
Rancho Seco HSM Intended Functions and AMR Results
(3 pages)

Subcomponent(2)	Subcomponent Parts(2)	Material(2)	Drawings/Part #(2)	Intended Function (1) (2)	Environment	Aging Effects Requiring Management	Aging Management Activity
Cask Docking Ring Assembly	Rings /Plates /Nelson Studs / Door Clamps /Hex Bolts	Note 3	Note 3	SS, RE	Embedded / Encased / External	Loss of Material	AMP
Heat Shield Assemblies	Roof and Side Wall Mounted Heat Shields	Note 3	Note 3	HT	Sheltered	Loss of Material	AMP
Heat Shield Assemblies	ZEE Brackets (for the Roof Mounted Heat Shields)	Note 3	Note 3	SS	Sheltered	Loss of Material	AMP
Heat Shield Assemblies	Heat Shield Attachment Hardware (Rods, Nuts)	Note 3	Note 3	SS	Sheltered	Loss of Material	AMP
Heat Shield Assemblies	Heat Shield Embedment Assemblies (Bolts/Sleeve Nuts)	Note 3	Note 3	SS	Embedded / Encased / Sheltered	Loss of Material	AMP
Cask Restraint Embedment Assembly	Rods/Sleeve Nuts/Hexagonal Nuts	Note 3	Note 3	SS, RE	Embedded / Encased / Sheltered	Loss of Material	AMP
Wall & Floor Mounted Canister Support Structure Embedment Assembly	Bolt/Sleeve Nut	Note 3	Note 3	SS	Embedded / Encased / Sheltered	Loss of Material	AMP
Roof Attachment Assembly	Roof Mounted/Wall Mounted Attachment Assemblies (Sleeve Nut)	Note 3	Note 3	SS	Embedded / Encased / Sheltered	Loss of Material	AMP
End and Rear Shield Wall Attachment Hardware	Embedment Bolts/Sleeve Nuts	Note 3	Note 3	SS	Embedded / Encased / Sheltered	Loss of Material	AMP
End Shield Wall Attachment Hardware	Embedment Bolts/Sleeve Nuts	Note 3	Note 3	SS	Embedded / Encased / Sheltered	Loss of Material	AMP
End and Rear Shield Wall Attachment Hardware	Cast-In Place Bolts/Nuts	Note 3	Note 3	SS	Embedded / Encased / Sheltered	Loss of Material	AMP
End & Rear Shield Wall Attachment Hardware	Tie Plate	Note 3	Note 3	SS	External	Loss of Material	AMP
HSM-to-HSM Spacer Channels	Spacer Channels	Note 3	Note 3	SS	External	Loss of Material	AMP
End and Rear Shield Wall Support Bolt Assembly	Shield Wall Support Bolt Assembly (Bolts, and Nuts)	Note 3	Note 3	SS	External	Loss of Material	AMP

Notes:

(1) Abbreviations for Intended Function Column:

PB - Directly or indirectly maintains a pressure boundary (confinement)
SH - Provides radiation shielding
CC - Provides criticality control of spent fuel
SS - Provides structural support
HT - Provides heat transfer
RE - Retrievability

(2) Only in-scope subcomponents from Chapter 2, Table 2-7 of Reference [9.7.6] are listed in this table.

(3) The values stated on pages 3-117 through 3-119 of Reference [9.7.6] are incorporated by reference into this IFSAR.

Table 9.8-8
MP187 TC Intended Functions and AMR Results
(4 pages)

Subcomponent(2)	Subcomponent Parts(2)	Material(2)	Drawings/Part #(2)	Intended Function(1) (2)	Storage Environment (3)	Aging Effects Requiring Management (4)	Aging Management Activity (4)
Main Assembly	Inner Shell	Note 5	Note 5	SH, SS, HT, RE	Sheltered / Encased	Loss of Material	AMP
Main Assembly	Bottom End Closure	Note 5	Note 5	SH, SS	Sheltered	Loss of Material	AMP
Main Assembly	Bottom Structural Shell	Note 5	Note 5	SH, SS, HT, RE	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Top Structural Shell	Note 5	Note 5	SH, SS, HT, RE	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Top Flange	Note 5	Note 5	SH, SS	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Gamma Shielding	Note 5	Note 5	SH, HT	Encased	None Identified	None Required
Main Assembly	Upper Trunnion Plug Cover & Side Plate	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Upper Trunnion Sleeve	Note 5	Note 5	SH, SS, RE	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Lower Trunnion Sleeve	Note 5	Note 5	SH, SS	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Pad Plate	Note 5	Note 5	SS	Encased	None Identified	None Required
Main Assembly	Bearing Block	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Tie Bar	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP
Main Assembly	NSP Top & Bottom Support Ring	Note 5	Note 5	SH, SS, HT	Encased / Sheltered	Loss of Material	AMP
Main Assembly	NSP Support Angle, Outer	Note 5	Note 5	SH, SS, HT	Encased	None Identified	None Required
Main Assembly	Rupture Plug	Note 5	Note 5	SH	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Plugs	Note 5	Note 5	SH	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Neutron Shield Shell	Note 5	Note 5	SH, SS, HT	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Upper Trunnion Plug Bottom Plate	Note 5	Note 5	SH, SS	Encased / Sheltered	Loss of Material,	AMP
Main Assembly	Rails	Note 5	Note 5	SS, RE	Sheltered	Loss of Material	AMP
Main Assembly / On-Site Transfer	Castable Neutron Shielding Material	Note 5	Note 5	SH	Encased	None Identified	None Required
Main Assembly	Ram Closure Plate	Note 5	Note 5	SH, SS, RE	Sheltered	Loss of Material	AMP
Main Assembly	Top Closure Plate	Note 5	Note 5	SH, SS, RE	Sheltered	Loss of Material	AMP
Main Assembly	Screw, Cap Hd. Soc.	Note 5	Note 5	SS, RE	Sheltered	Loss of Material	AMP
Main Assembly	Screw, Cap Hd. Soc.	Note 5	Note 5	SS	Sheltered	Loss of Material	AMP

Table 9.8-8
MP187 TC Intended Functions and AMR Results
(4 pages)

Subcomponent(2)	Subcomponent Parts(2)	Material(2)	Drawings/Part #(2)	Intended Function(1) (2)	Storage Environment (3)	Aging Effects Requiring Management (4)	Aging Management Activity (4)
Main Assembly	Filler Plate	Note 5	Note 5	SH, SS	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Hardened Washer (3" & 1.5” OD)	Note 5	Note 5	SS	Sheltered	Loss of Material	AMP
Main Assembly	Test Port Screw	Note 5	Note 5	SH	Sheltered	Loss of Material	AMP
Main Assembly	Vent/Drain Port Screw	Note 5	Note 5	SH	Sheltered	Loss of Material	AMP
Main Assembly	Threaded Insert and Port Plugs	Note 5	Note 5	SH	Sheltered	Loss of Material	AMP
Main Assembly	Lower Trunnion Plug Cover Plate	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Lower Trunnion Plug Shield Block	Note 5	Note 5	SH	Encased / Sheltered	Loss of Material	AMP
Main Assembly	Screw, Flat Hd. Cap	Note 5	Note 5	SS	Sheltered	Loss of Material	AMP
Main Assembly	NSP Support Angle, Inner	Note 5	Note 5	SH, SS, HT	Embedded	None Identified	None Required
Main Assembly	Screw Thread Insert (1” and 2”)	Note 5	Note 5	SS, RE	Encased / Sheltered	Loss of Material	AMP
Main Assembly	10 Gage Sheet	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP
On-Site Transfer	Outer Plug Cover Plate	Note 5	Note 5	SH	Encased / Sheltered	Loss of Material	AMP
On-Site Transfer	Inner Plug Cover Plate	Note 5	Note 5	SH	Encased / Sheltered	Loss of Material	AMP
On-Site Transfer	Inner Plug Inside Sleeve	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material(6)	AMP
On-Site Transfer	Bolt, 1-8UNC-2A	Note 5	Note 5	SS	Sheltered	Loss of Material	AMP
On-Site Transfer	Outer Plug Support Bracket	Note 5	Note 5	SS	Sheltered	Loss of Material	AMP
On-Site Transfer	Key Plug Cover Plate	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP
On-Site Transfer	Flat Hd Socket Cap Screw	Note 5	Note 5	SS	Sheltered	Loss of Material	AMP
On-Site Transfer	Socket Hd Cap Screw	Note 5	Note 5	SS	Sheltered	Loss of Material	AMP
On-Site Transfer	Lower Trunnion	Note 5	Note 5	SH, SS	Sheltered	Loss of Material	AMP
On-Site Transfer	Upper Trunnion	Note 5	Note 5	SH, SS, RE	Encased / Sheltered	Loss of Material	AMP
On-Site Transfer	Trunnion Back	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP
On-Site Transfer	Key Plug Side Plate	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP

Table 9.8-8
MP187 TC Intended Functions and AMR Results
(4 pages)

Subcomponent(2)	Subcomponent Parts(2)	Material(2)	Drawings/Part #(2)	Intended Function(1) (2)	Storage Environment (3)	Aging Effects Requiring Management (4)	Aging Management Activity (4)
On-Site Transfer	Key Plug Bottom Plate	Note 5	Note 5	SS	Encased / Sheltered	Loss of Material	AMP

Notes:

(1) Abbreviations for Intended Function Column:

PB Directly or indirectly maintains a pressure boundary (confinement)
SH Provides radiation shielding
CC Provides criticality control of spent fuel
SS Provides structural support
HT Provides heat transfer
RE Retrievability

(2) Only in-scope subcomponents from Chapter 2, Table 2-8 of Reference [9.7.6] are listed in this table.

(3) The TC operations are intermittent and following the completion of each fuel loading campaign all exposed cask surfaces are thoroughly cleaned to remove potential contamination. Therefore, the Aging Management Results are based on the Encased / Sheltered environments.

(4) Cracking due to thermal fatigue of the transfer cask subcomponents has been addressed by performing a TLAA that bounds all subcomponents.

(5) The values stated on pages 3-132 through 3-135 of Reference (9.7.6] are incorporated by reference into this IFSAR.

Table 9.8-9
DSC External Surfaces Aging Management Program
(2 pages)

Subcomponents	Number Inspected	Environment	Inspection Type	Frequency	Trending	Acceptance Criteria	Corrective Action
DSC Shell External Surfaces (normally non-accessible areas) • portions of the outer top cover plate, closure welds and HAZ; • portions of the DSC shell bottom surface • DSC surfaces, welds and HAZ, crevice locations near DSC support rails, inspected for discontinuities and imperfections; localized corrosion (e.g. general, pitting and crevice corrosion); cracking and stains caused by leaking rainwater; appearance and location of atmospheric deposits on DSC surfaces are recorded; no additional action for rainwater stains or discoloration unless corrosion is exhibited • outer bottom cover plate, grapple ring assembly, shear key, closure welds and HAZ	1	Sheltered	Remote Visual	Baseline no later than 2 years after PEO begins; 10 +/- 2 yrs thereafter. If "major" corrosion is identified, Increase frequency to 5 yrs +/- 1 yr	Data taken from the inspections are to be monitored by comparison to past site data taken as well as comparison to industry OE.	<u>Visual Exams</u>: The presence of a major corrosion indication anywhere on the DSC, or a minor corrosion indication within 2" of a weld, will receive a supplemental surface or volumetric examination. A minor corrosion indication more than 2" from a weld will receive a supplemental VT-1 exam. <u>Augmented Exams</u>: absence of flaws, or flaw is a round indication, or does not have corrosion products present, or does not have crack-like morphology. <u>Flaw Evaluation</u>: Determine when 75% through-wall is reached.	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Identification of major corrosion requires an expansion of the sample. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size.

Table 9.8-9
DSC External Surfaces Aging Management Program
(2 pages)

Subcomponents	Number Inspected	Environment	Inspection Type	Frequency	Trending	Acceptance Criteria	Corrective Action
DSC Shell (inaccessible areas) • upper surface of DSC shell where atmospheric particulate may settle • majority of the outer top cover plate, welds and HAZ • DSC shell crevice locations where shell rests on support rails	As required by inspection findings	Sheltered	In accordance with Corrective Actions AMP Section B.3.5(7)	In accordance with Corrective Actions AMP Section B.3.5(7).	In accordance with Corrective Actions AMP Section B.3.5(7).	Via the SMUD corrective action program to ensure the aging effect is adequately managed and that the intended function is maintained during the PEO.	Further evaluation and disposition per SMUD corrective action program (See AMP Section B.3.5(7), including more frequent inspections.

Table 9.8-10
HSM Aging Management Program for External and Internal Surfaces
(4 pages)

Subcomponents	Number Inspected	Environment	Inspection Type	Frequency	Trending	Acceptance Criteria	Corrective Action
HSM Concrete - External • front, back, and side walls & rebar • roof exterior • HSM access door	All	External/ Embedded	Direct Visual (accessible areas)	Baseline no later than 2 years after PEO begins; 10 +/- 2 yrs thereafter. Increase frequency to 5 yrs +/- 1 yr if acceptance criteria exceeded.	Data taken for these inspections are to be monitored by comparison to past site data taken as well as comparison to industry OE, including data gathered by the AMID, as discussed in NEI 14-03	Cracking, spalling, scaling, loss of material and other anomalies do not exceed ACI-349.3R. The following anomalies are considered acceptable: • absence of leaching and chemical attack • absence of signs of corrosion of steel reinforcement • absence of Drummy areas (poorly consolidated concrete, air voids with paste deficiencies per ACI-201.1R) • popouts and voids less than 2" in dia or equal surface area • scaling less than 1-1/8" in depth • spalling less than 3/4" in depth and less than 8" for any dimension • absence of corrosion staining of undefined source on concrete • passive cracks less than 0.04" in maximum width	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size.

Table 9.8-10
HSM Aging Management Program for External and Internal Surfaces
(4 pages)

Subcomponents	Number Inspected	Environment	Inspection Type	Frequency	Trending	Acceptance Criteria	Corrective Action
HSM Concrete - Internal • visible portions of front, back, and side walls • portions of the HSM concrete floor (base)	1	Sheltered/ Embedded	Remote Visual (normally non-accessible)	Baseline no later than 2 years after PEO begins; 10 +/- 2 yrs thereafter. Increase frequency to 5 yrs +/- 1 yr if acceptance criteria exceeded.	Data taken for these inspections are to be monitored by comparison to past site data taken as well as comparison to industry OE, including data gathered by the AMID, as discussed in NEI 14-03.	Cracking, spalling, scaling, loss of material and other anomalies do not exceed ACI-349.3R. The following anomalies are considered acceptable: • absence of leaching and chemical attack • absence of signs of corrosion of steel reinforcement • absence of Drummy areas (poorly consolidated concrete, air voids with paste deficiencies per ACI-201.1R) • popouts and voids less than 2" in dia or equal surface area • scaling less than 1-1/8" in depth • spalling less than 3/4" in depth and less than 8" for any dimension • absence of corrosion staining of undefined source on concrete • passive cracks less than 0.04" in maximum width	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size.

Table 9.8-10
HSM Aging Management Program for External and Internal Surfaces
(4 pages)

Subcomponents	Number Inspected	Environment	Inspection Type	Frequency	Trending	Acceptance Criteria	Corrective Action
HSM Concrete & Steel Internal (inaccessible areas) • internal surface of the HSM roof due to the upper heat shield • heat shields at internal surface of the roof and side walls	As required by inspection findings	Sheltered/ Embedded	In accordance with Corrective Actions AMP Section B.4.5(7)	In accordance with Corrective Actions AMP Section B.4.5(7).	In accordance with Corrective Actions AMP Section 5 B.4.5(7).	Via the SMUD corrective action program to ensure the aging effect is adequately managed and that the HSMs intended function is maintained during the PEO.	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size.
HSM Steel - External • HSM access door • attachment hardware	All	External/Embedded	Direct Visual (accessible areas)	Baseline no later than 2 years after PEO begins; 10 +/- 2 yrs thereafter. Increase frequency to 5 yrs +/- 1 yr if acceptance criteria exceeded.	Data taken for these inspections are to be monitored by comparison to past site data taken as well as comparison to industry OE, including data gathered by the AMID, as discussed in NEI 14-03.	VT-3: ASME Section XI, Subarticle IWF-3400 and any indications of the following are evaluated: • corrosion and material loss • crevice, pitting, and galvanic corrosion • corrosion stains on adjacent components and structures • surface cracks • stains caused by leaking rainwater if evidence of corrosion exhibited • loose bolts and nuts and cracked bolts are not acceptable unless approved by the Engineering evaluation.	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size.

Table 9.8-10
HSM Aging Management Program for External and Internal Surfaces
(4 pages)

Subcomponents	Number Inspected	Environment	Inspection Type	Frequency	Trending	Acceptance Criteria	Corrective Action
HSM Steel - Internal • DSC support structure including Nitronic® 60 rail plates and welds • attachment hardware	1	Sheltered/ Embedded	Remote Visual (normally non-accessible)	Baseline no later than 2 years after PEO begins; 10 +/- 2 yrs thereafter. Increase frequency to 5 yrs +/- 1 yr if acceptance criteria exceeded.	Data taken for these inspections are to be monitored by comparison to past site data taken as well as comparison to industry OE, including data gathered by the AMID, as discussed in NEI 14-03.	VT-3: ASME Section XI, Subarticle IWF-3400 and any indications of the following are evaluated: • corrosion and material loss • crevice, pitting, and galvanic corrosion • corrosion stains on adjacent components and structures • surface cracks • Stains caused by leaking rainwater if evidence of corrosion exhibited • loose bolts and nuts and cracked bolts are not acceptable unless approved by the engineering evaluation.	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size.

Table 9.8-11
TC Aging Management Program

Subcomponents	Number Inspected	Environment	Inspection Type	Frequency	Trending	Acceptance Criteria	Corrective Action
MP187 External Surfaces • Cask lid surfaces • bearing surfaces of upper and lower trunnions • attachment fasteners • exterior cask surfaces	1	Sheltered	Direct Visual of external surfaces accessible areas	Within one year prior to use	Data taken from the inspections are to be monitored by comparison to past site data taken as well as comparison to industry OE.	• No indications of corrosion or wear on TC external surfaces	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size.
MP187 Internal Surfaces • Cask cavity inner liner • Nitronic 60 rails	1	Sheltered	Direct Visual, Remote Visual or both of accessible areas	Within one year prior to use	Data taken from the inspections are to be monitored by comparison to past site data taken as well as comparison to industry OE.	• No indications of corrosion on TC internal surfaces • No wear of inner liner thickness	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size.

Table 9.8-12
ISFSI Basemat Aging Management Program

Subcomponents	Number Inspected	Environment	Inspection Type	Frequency	Trending	Acceptance Criteria	Corrective Action
Basemat Concrete	1	External/ Embedded/ Sheltered/ Underground	Direct Visual of the accessible above-grade surfaces.	Baseline no later than 2 years after PEO begins; 10 +/- 2 yrs thereafter. If acceptance criteria is exceeded, increase frequency to 5 yrs +/- 1 yr.	Data taken from the inspections are to be monitored by comparison to past site data taken as well as comparison to industry OE.	Cracking, spalling, scaling, loss of material and other anomalies do not exceed ACI-349.3R and ACI-201.1R. The following anomalies are considered acceptable: • absence of leaching and chemical attack • absence of signs of corrosion of steel reinforcement • absence of Drummy areas (poorly consolidated concrete, air voids with paste deficiencies per ACI-201.1R) • popouts and voids less than 2" in dia or equal surface area • scaling less than 1-1/8" in depth • spalling less than 3/4" in depth and less than 8" for any dimension • absence of corrosion staining of undefined source on concrete • passive cracks less than 0.04" in maximum width • passive settlement or deflection within original design limits	Conditions adverse to quality are evaluated in accordance with SMUD's Corrective Action Program. Evaluations should use the same methodology used in licensing and design basis as much as practical. Extent of condition may trigger additional inspections, increased inspection frequency or expanded inspection sample size. Repair, restoration meets ACI-224.4R and ASME Sect. XI, IWA-4000.

Table 9.8-13
Rancho Seco ISFSI Tollgates

TOLL GATE	DUE DATE	ASSESSMENT
1	6/30/2025	Evaluate information from the following sources and perform a written assessment of the aggregate impact of the information, including but not limited to applicable and relevant trends, corrective actions required, and the effectiveness of the AMPs with which they are associated: – Results, if any, of research and development programs focused specifically on aging-related degradation mechanisms identified as potentially affecting DSS ISFSIs; – Relevant domestic and international OE including research results on aging effects/mechanisms (including non-nuclear on an opportunistic basis); – Relevant results of domestic and international ISFSI and DSS performance monitoring; – Relevant results of domestic and international ISFSI and DSS inspections Topics of particular interest for the Rancho Seco ISFSI tollgate assessment should include the following: – Reinforced concrete degradation in general, and degradation of NUHOMS® HSMs in particular – Deterioration of carbon steel and coatings
2	6/30/2030	Evaluate additional information gained from the sources listed in Tollgate 1 along with any new relevant sources and perform a written assessment of the aggregate impact of the information. This evaluation should be informed by the results of Tollgate 1. The aging effects and mechanisms evaluated at this Tollgate, and the time at which it is conducted, may be adjusted based on the results of the Tollgate 1 assessment.
3 and later	No more than five years after completion of the previous tollgate assessment	Same as Tollgate 1, as informed by the results of Tollgates 1 and 2

10. OPERATING CONTROLS AND LIMITS

10.1 Proposed Operating Controls and Limits

The Rancho Seco ISFSI storage system is totally passive and requires minimal operating controls during canister loading, closure, and transfer operations. The Rancho Seco ISFSI employs a proven technology, stringent codes of construction, and comprehensive quality assurance measures. As a result, it has substantial design and safety margins. The areas where controls and limits are necessary to ensure safe operation of the Rancho Seco ISFSI are shown in Table 10-1.

Operating controls and limits proposed for the HSMs are discussed in Chapter 10 of Volume II. The items to be controlled are selected based on the design criteria and safety analyses for normal, off-normal, and accident conditions documented in Chapters 3, 7 and 8 of Volumes I, II, and III.

10.2 Development of Operating Controls and Limits

This section provides an overview of and the general bases for the operating controls and limits specified for the Rancho Seco ISFSI to ensure the protection of the public's health and safety. The ISFSI Technical Specifications and/or SAR Section 10.3 provide a full description and discussion of these operating limits.

Operating controls and limits unique to HSM storage are developed in Section 10.2 of Volume II.

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

This category of operating controls and limits applies to operating variables that are observable and measurable. A temperature monitoring system will be used to monitor the temperature of each HSM. Other limits and controls which are applied to the system during DSC loading, closure and transfer to the ISFSI are the fuel selection criteria, DSC surface contamination limits, DSC vacuum and helium backfill pressures, DSC closure weld examination requirements, cask/DSC lifting heights, and dose rates.

The functional limits for the fuel to be stored in the Rancho Seco ISFSI are provided in Chapter 2 of the Technical Specifications.

10.2.2 Limiting Conditions for Operation

10.2.2.1 Equipment

No limiting conditions regarding minimum available equipment or operating characteristics which are important to safety apply to the Rancho Seco ISFSI. The components of storage, the DSC, the HSM, and the cask have been analyzed for all credible equipment failure modes and extreme environmental conditions. No postulated event results in damage to fuel, release of radioactivity, or danger to the public health and safety. All operational equipment is to be maintained, tested, and operated according to the implementing procedures developed for the ISFSI. The failure or unavailability of any operational component can delay the transfer of the DSC to the HSM, but would not result in an unsafe condition.

10.2.2.2 Technical Conditions and Characteristics

The following technical conditions and characteristics are required for the Rancho Seco ISFSI and DSCs:

1. Spent Fuel Specifications (Technical Specifications Section 2.1.1)
2. DSC Vacuum Pressure During Drying (Technical Specifications Section 3.1.1)
3. DSC Helium Backfill Pressure (Technical Specifications Section 3.1.3)

4. DSC Helium Leak Rate of Inner Seal Weld (Technical Specifications Section 3.1.2)

5. DSC Dye Penetrant Test of Closure Welds (SAR, Volume I, Section 10.3.5)

6. DSC Surface Contamination (Technical Specifications Section 5.5.4)

7. ISFSI Security Area Dose Rates (SAR, Volume I, Section 10.3.7)

8. DSC Inspection Following Cask Drop (Technical Specifications Section 5.6.2)

9. Post Fire Recovery Plan (SAR, Volume I, Section 10.3.9)

10. Cask/DSC Lifting Heights (Technical Specifications Section 5.6.1)

11. DSC Top End Dose Rates (SAR, Volume I, Section 10.3.11)

12. HSM Dose Rates (SAR, Volume I, Section 10.3.12)

13. Transfer Cask Dose Rates (SAR, Volume I, Section 10.3.13)

14. DSC Re-flood flow Rate (SAR, Volume I, Section 10.3.14)

15. Heat-up duration of a Loaded DSC filled with Water (SAR, Volume I, Section 10.3.15)

16. HSM Thermal Monitoring (Technical Specifications Section 5.5.3)

A description of the bases for selecting the above conditions and characteristics are detailed in the bases section for the Technical Specifications or in the operating limits in Section 10.3, as appropriate. Technical conditions and characteristics for the HSMs are discussed in Section 10.2.2.2 of Volumes II.

The overall technical and operational considerations are to:

1. Assure proper internal DSC atmosphere to promote heat transfer, minimize uranium dioxide oxidation, and prevent an uncontrolled release of radioactive material.

2. Assure that dose rates in areas where operators must work are as-low-as-reasonably-achievable and that all relevant dose limits are met.

3. Assure that the fuel cladding is maintained at a temperature sufficiently low to preclude cladding degradation during normal storage conditions.

Through the analyses and evaluations provided in Chapters 7 and 8, the Technical Specifications and this SAR demonstrate that the above technical conditions and characteristics are adequate and that no significant public or occupational health and safety hazards exist.

10.2.3 Verification Requirements

Analysis has shown that the Rancho Seco ISFSI can fulfill its safety functions during all normal and off-normal operating conditions and during all accident conditions as described in Chapter 8. No verification of the DSC is required during long-term storage. HSM verification requirements are discussed in Section 10.2.3 of Volume II.

10.2.4 Design Features

The following storage system design features are important to the safe operation of the Rancho Seco ISFSI and require design controls and limits:

1. Material Mechanical Properties for Structural Integrity Containment, and Shielding

2. Material Composition and Dimensional Control for Subcriticality

3. Decay Heat Removal

Component dimensions are not specified here since the combination of materials, dose rates, criticality safety, and component fit-up define the operable limits for dimensions (i.e., thickness of shielding materials, thickness of concrete, DSC plate thicknesses, etc.). The values for these design parameters are specified on the Volume IV drawings. Changes to any of these design features will be implemented only after conducting a safety evaluation in accordance with 10 CFR 72.48.

The combination of the above controls and limits and those discussed in the previous subsections of Section 10.2 define requirements for the Rancho Seco ISFSI components that provide radiological protection and structural integrity during normal storage and postulated accident conditions.

10.2.5 Administrative Controls

Use of SMUD's existing organizational and administrative systems and procedures, record keeping, review, audit, and reporting requirements coupled with the requirements of the Technical Specifications and this SAR ensure that the operations involved in the storage of spent fuel in the Rancho Seco ISFSI are performed in a safe manner. This includes both the selection of assemblies qualified for ISFSI storage and the verification of assembly identification numbers prior to and after placement into individual storage canisters.

10.2.5.1 Qualification of Spent Fuel

Fuel assembly qualification is based on the requirements for criticality safety, decay heat removal, radiological protection, and structural integrity. The analyses presented in Chapters 7 and 8 of Volumes I, II, and III of this SAR document the qualification of the complete Rancho Seco inventory of spent fuel assemblies for storage in any of the three Rancho Seco DSC designs. Additionally, these analyses document the qualification of the complete Rancho Seco inventory of control components for storage in the Rancho Seco FC DSC design. The analyses of the ISFSI decay heat removal and radiological protection are valid for DSC loading dates after June 1996.

During RSNGS operation, primary coolant chemistry was monitored to provide an indication of leaking fuel cladding. In addition, the spent fuel pool water chemistry has been carefully maintained in accordance with the 10 CFR 50 Technical Specification requirements to minimize any long term degradation of the fuel, and to ensure long term safe storage in the pool. All fuel movements are performed by trained and qualified individuals using written approved procedures which have been proven over the operating history of RSNGS to ensure safe handling of the fuel. Fuel assembly handling will be done using the permanent installed fuel handling equipment which includes interlocks and controls designed to preclude fuel damage during handling.

As discussed in EPRI Report NP-4804, the predominant fuel damage in PWR reactors is caused by debris fretting from the primary coolant system. Further, there is no historical basis for assuming that more than 15 rods per assembly would be found to be defective. Typical PWR fuel leakage involves less than 4 fuel rods per assembly.

To identify fuel assemblies with visible cladding damage, underwater cameras were used to visually inspect the accessible areas of each fuel assembly. Telephoto lenses were used, as required, to obtain the necessary magnification for detailed examination. Fuel assemblies with cladding damage should be clearly visible with this type of examination. The inspections were video taped.

Based on the visual inspection of the accessible areas of each spent fuel assembly, 10 fuel assemblies were determined to have some cladding damage, and no assemblies are believed to have cladding damage in more than 15 fuel rods. The visual inspections, along with the known history of plant operations and long term fuel storage, provides a high level of confidence the fuel will meet the criteria for storage in the appropriate DSC.

The inspection records document fuel assemblies with visible cladding damage. Rancho Seco will develop the fuel loading schedule to ensure that damaged fuel assemblies are not loaded in either the FO or FC DSCs. Up to 13 assemblies with visible cladding damage in 15 or fewer fuel pins are qualified for storage in the FF DSC. If the structural integrity criterion is met, then approval for dry storage for a given assembly is made. This qualification will be documented and subsequently referenced through Rancho Seco ISFSI operating procedures prior to loading fuel into the DSC.

10.2.5.2 Spent Fuel Identification

The following controls will ensure that each fuel assembly is loaded into a known cell location within a DSC:

1. A loading schedule will be independently verified and approved.

2. A fuel movement schedule will be based upon the written loading plan. All fuel movements from any rack location will be performed under controls that will ensure strict, verbatim compliance with the fuel movement schedule.

3. Prior to placement of the shield plug, all fuel assemblies will be video taped and independently verified, by ID number, to match the fuel movement schedule.

4. A third independent verification will be performed by a senior manager. This third verification verifies that fuel in the DSCs is placed in accordance with the original cask loading plan.

10.3 Operating Control and Limit Specifications

The operating controls and limits applicable to the Rancho Seco ISFSI as documented in this SAR to be implemented by SMUD are delineated in the sections which follow. Operating controls and limits applicable to the Rancho Seco ISFSI HSMs are provided in Section 10.3 of Volume II.

10.3.1 Spent Fuel Specifications

10.3.1.1 FO and FC-DSC Fuel Specifications

See Technical Specifications Section 2.1.1

10.3.1.2 FF-DSC Fuel Specifications

See Technical Specifications Section 2.1.1

10.3.2 DSC Vacuum Pressure During Drying

See Technical Specifications Section 3.1.1

10.3.3 DSC Helium Backfill Pressure

See Technical Specifications Section 3.1.3

10.3.4 DSC Helium Leakage Rate of Inner Seal Weld

See Technical Specifications Section 3.1.2

10.3.5 DSC Dye Penetrant Test of Closure Welds

Operating Limit: The acceptance standards for liquid penetrant examination contained in the ASME Boiler and Pressure Vessel Code Section III, Division l, Subsection NB-5350 (1992) (1993 Addenda) Liquid Penetrant Acceptance Standards shall apply.

Applicability: This operating limit is applicable to the inner top cover plate, vent and siphon port covers, and outer top cover plate closure welds of all DSCs.

Objective: To ensure that the DSC is adequately sealed in a redundant manner and to assure that all radioactive materials are confined for all design conditions.

Action: If the liquid penetrant test indicates that the weld is unacceptable:

a. The weld shall be repaired in accordance with approved procedures.

b. The weld shall be re-examined in accordance with this operating limit.

Verification: Verify that the acceptance standards for liquid penetrant examination contained in the ASME Boiler and Pressure Vessel Code Section III, Division l, Subsection NB-5350 (1992) (1993 Addenda) Liquid Penetrant Acceptance Standards, are applied to DSC closure welds.

Bases: Article NB-5000 Examination

ASME Boiler and Pressure Vessel Code

Section III - Division I

Subsection NB (1992) (1993 Addenda)

10.3.6 DSC Surface Contamination

See Technical Specifications Section 5.5.4

10.3.7 ISFSI Security Area Dose Rate

Operating Limit: The dose rate at any point on the Rancho Seco ISFSI outer security fence area boundary shall be less than two mrem per hour.

Applicability: This operating limit is applicable to the entire Rancho Seco ISFSI outer security fence area boundary.

Objective: The objective of this operating limit is to guarantee compliance with the 10 CFR 20.1301 unrestricted area dose limit, the 10 CFR 72.106 controlled area dose limit, and to maintain offsite exposures as-low-as-reasonably achievable.

Action: If the dose rates are exceeded, evaluate and correct the problem using the RSNGS corrective action program.

Verification: The Rancho Seco ISFSI outer security fence area boundary shall be checked to verify that this operating limit has been met after each DSC is placed in storage.

Bases: The dose rate stated in this operating limit is selected to maintain exposures on-site and offsite in accordance with 10 CFR 20 and 10 CFR 72. Based on the results presented in Chapter 7 of Volume I, the security area boundary dose rate exhibits the smallest margin of safety below the applicable regulatory limit. Compliance with the above operating limit will therefore guarantee compliance with the remaining dose limits discussed in Chapters 3 and 7.

10.3.8 Cask and DSC Inspection Following Accidental Cask Drop

See Technical Specifications Section 5.6.2

10.3.9 Post Fire Recovery Plan

Operating Limit: If a fire occurs in the ISFSI, a post fire recovery plan will be formulated.

Applicability: This operating limit applies to all components located within the ISFSI.

Objective: To assure the effects of any fire within the ISFSI will have no impact on the public health and safety.

Action: The scope of the post fire recovery plan will vary dependent on the size and intensity of the fire, but as a minimum it will include:

a. Conduct radiological surveys of the potentially affected areas.

b. Visual inspection of affected cask exterior surface areas, and an interior inspection of the cask if damage is suspected..

c. Visual inspection of potentially affected DSCs.

d. Event analysis and implementation of corrective actions as required.

Verification: No Verification is required.

Bases: A post fire recovery plan will ensure that any effects of a fire are adequately analyzed to determine and correct any possible safety consequences.

10.3.10 DSC Lifting Heights

See Technical Specifications Section 5.6.1

10.3.11 DSC Top End Dose Rates

Operating Limit: Dose rates at the following locations shall be limited to levels which are less than or equal to:

a. 200 mrem/hr at top shield plug surface at centerline with water in cavity.

b. 400 mrem/hr at top cover plate surface at centerline without water in cavity.

Applicability: This operating limit applies to all DSCs.

Objective: The dose rate is limited to this value to ensure that the DSC has not been inadvertently loaded with fuel not meeting the operating limits in Technical Specification Section 2.1.1 and to maintain dose rates as low as reasonably achievable during DSC closure operations.

Action: If specified dose rates are exceeded, evaluate and correct the problem using the RSNGS corrective action program.

Verification: Dose rates specified in 10.3.11.a shall be measured before installing the inner top cover plate. Dose rates specified in 10.3.11.b shall be measured before welding the outer top cover plate to the DSC shell.

Basis: The basis for this limit is the shielding analysis presented in Chapter 7 of the Standardized SAR.

10.3.12 HSM Dose Rates

Operating Limit: Dose rates at the following locations shall be limited to levels which are less than or equal to:

a. 400 mrem/hr at 3 feet from the HSM surface.

b. 100 mrem/hr outside of HSM door on center line of DSC.

c. 20 mrem/hr at end shield wall exterior.

Applicability: This operating limit is applicable to all HSMs which contain a loaded DSC.

Objective: The dose rate is limited to this value to ensure that the cask (DSC) has not been inadvertently loaded with fuel not meeting the operating limits in Technical Specification Section 2.1.1 and to maintain dose rates as low as is reasonably achievable (ALARA) at locations on the HSMs where Verification is performed, and to reduce offsite exposures during storage.

Action: If specified dose rates are exceeded, evaluate and correct the problem using the RSNGS corrective action program.

Verification: The HSM and ISFSI shall be checked to verify that this operating limit has been met after each DSC is placed into storage and the HSM door is closed.

Basis: The basis for this limit is the shielding analysis presented in Chapter 7 of the Standardized SAR. The specified dose rates provide as low as reasonably achievable on-site and offsite doses in accordance with 10 CFR 20 and 10 CFR 72.104(a).

10.3.13 Transfer Cask Dose Rates

Operating Limit: Dose rates from the transfer cask shall be limited to levels which are less than or equal to:

a. 200 mrem/hr at 3 feet with water in the DSC cavity.

b. 500 mrem/hr at 3 feet without water in the DSC cavity.

Applicability: This operating limit is applicable to the transfer cask containing a loaded DSC.

Objective: The dose rate is limited to this value to ensure that the DSC has not been inadvertently loaded with fuel not meeting the operating limits in Technical Specification Section 2.1.1 and to maintain dose rates as low as is reasonably achievable during DSC transfer operations.

Action: If specified dose rates are exceeded, evaluate and correct the problem using the RSNGS corrective action program.

Verification: The dose rates shall be measured as soon as possible after the transfer cask is removed from the spent fuel pool.

Basis: The basis for this limit is the shielding analysis presented in Chapter 7 of the Standardized SAR.

10.3.14 DSC Re-flood Flow Rate

Operating Limit: The DSC re-flood flow rate shall be controlled to ensure that the DSC internal pressure is maintained ≤16 psig during the initial steaming and solid water cooling modes.

Applicability: This operating limit is applicable to any loaded DSC required to be re-flooded.

Objective: To prevent DSC over-pressurization and unacceptable pressure spikes while maintaining a reasonable cool down rate.

Action: If the DSC internal pressure exceeds the specified rate, adjust the flow rate to bring the DSC pressure to within the specified limit or terminate re-flood operations until an acceptable DSC internal pressure can be established.

Verification: Verify the DSC internal pressure is maintained within the specified limit.

Basis: Controlling the re-flood rate ensures that the DSC pressure stays below 16 psig which provides a 20% margin below the DSC design pressure of 20 psig.

10.3.15 Heat-up Duration of a Loaded DSC Filled with Water

Operating Limit: After a loaded DSC filled with water is removed from the spent fuel pool, the initial draining operations shall be completed within the duration specified below:

For Spent Fuel Pool Water Temperature ≤ 110 °F

Decay Heat (Kw)	≤13.5 Kw	≤9.3Kw	≤8.4 Kw
Duration (hrs.)	46	67	74

For Spent Fuel Pool Water Temperature ≤ 80 °F

Decay Heat (Kw)	≤13.5 Kw	≤9.3Kw	≤8.4 Kw
Duration (hrs.)	59	86	96

Applicability: This operating limit is applicable to any loaded DSC removed from the spent fuel pool.

Objective: To ensure that the water in the loaded DSC does not exceed 212 °F.

Action: If the initial draining operations are not completed within the specified time limit, take appropriate actions to ensure that water in the DSC does not exceed 212 °F.

Verification: Verify that the loaded DSC is drained within the specified time limit.

Basis: Based on conservative calculations, staying below the specified duration times ensures that water in a loaded DSC will not exceed 212 °F.

10.4 References

10.1 "Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.

Table 10-1

Area Where Controls and Limits Are Specified

Areas for Operating Controls and Limits	Conditions or Other Items To Be Controlled
Fuel characteristics	Physical condition
Cask	Inspection following drop accident
Dry Shielded Canister	Dye penetrant test of closure welds. Vacuum pressure during drying. Helium backfill pressure and leakage rate. Surface contamination.
ISFSI Site	Security area dose rate
Administrative Controls	Fuel loading verification including fuel assembly location
Training	Operations, maintenance and Verification

11. QUALITY ASSURANCE

11.1 Sacramento Municipal Utility District Quality Assurance Program

10 CFR 72.140 requires that licensees establish, maintain, and execute a quality assurance (QA) program satisfying each of the applicable criteria in 10 CFR 72, Subpart G. 10 CFR 72.140(d) states that an NRC-approved QA program that satisfies the criteria of 10 CFR 50, Appendix B, and that is established, maintained, and executed with regard to an ISFSI is acceptable for satisfying the QA program requirements.

SMUD has established and implemented a QA program based on the criteria in 10 CFR 50, Appendix B for the RSNGS. This program will be implemented for the structures, systems, and components of the Rancho Seco ISFSI that are important to safety.

The Plant Manager is responsible for the safe and reliable decommissioning of Rancho Seco. The Plant Manager has the responsibility and authority to implement the Rancho Seco Quality Assurance Program and ensure optimum quality performance of Rancho Seco.

The governing document for this program is the Rancho Seco Quality Manual (RSQM) [11.11.1] which has been reviewed and approved by the NRC. The program is implemented through the RSQM and appropriate administrative procedures. The objective of the QA program for operating nuclear power stations is to comply with the criteria as expressed in 10 CFR 50, Appendix B, and with the QA program requirements for nuclear power plants as referenced in the Regulatory Guides and ANSI standards. The Rancho Seco RSQM will be applied to those activities associated with the Rancho Seco ISFSI that are important to safety.

11.2 Quality Assurance Program – Contractors

11.2.1 Architect-Engineer

SMUD has the responsibility to ensure that the design and engineering of the Rancho Seco ISFSI is performed in accordance with the applicable requirements and design bases. Contractors hired to perform design or engineering of the ISFSI will perform their work in accordance with District approved quality requirements.

11.2.2 Storage System Supplier

SMUD has the responsibility to ensure that components are manufactured in accordance with applicable requirements and design bases. The Transfer Cask and Yoke, Dry Shielded Canister, and Horizontal Storage Modules are designed and manufactured under a District approved Quality Assurance Program.

11.3 References

11.1 Rancho Seco Quality Manual.

11.2 "Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 4A, VECTRA Technologies, Inc., June 1996.