<u>ATTACHMENT (11)</u>

PACIFIC NUCLEAR DOCUMENT NUH-002 –

NON-PROPRIETARY VERSION

NUH-002
Revision 2A
NUH002.0103

TOPICAL REPORT
FOR THE
NUTECH HORIZONTAL MODULAR
STORAGE SYSTEM FOR
IRRADIATED NUCLEAR FUEL
NUHOMS®-24P

VOLUME I

NONPROPRIETARY

Submitted to the
UNITED STATES NUCLEAR REGULATORY COMMISSION

By
Pacific Nuclear Fuel Services, Inc.
San Jose, California

April 1991

REVISION CONTROL SHEET

TITLE: Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P

DOCUMENT FILE NUMBER: NUH-002 NUH002.0103

NAME / TITLE	INITIALS
W. J. McConaghy/Project Manager	WJM
R. A. Lehnert/Project Engineer	RAL
J. R. Barker/Specialist	JRB
Dr. J. R. Bondre/Senior Consultant	JRB
Dr. N. G. Cofie/Principal Consultant	NGC
N. Eng/Principal Consultant	NEng
B. W. Ferguson/Engineer	BWF
J. H. Kessler/Consultant II	BDT for JHK
R. T. Mahini/Consultant I	RTM.
I. D. McInnes/Staff Engineer	IDM
J. D. Rollins/Supervising Engineer	RAL for JDR
K. C. Smith/Consultant II	JRB for KCS
B. D. Thomas/Engineer	BDT
J. M. Rosa/Staff Engineer	BDT for JMR WMR 7/13/89

QEP 3-3.1.1
REV 1

REVISION CONTROL SHEET
(CONTINUATION)

TITLE: Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P

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AFFECTED PAGE(S)	DOC REV	PREPARED BY / DATE	ACCURACY CHECK BY / DATE	CRITERIA CHECK BY / DATE	REMARKS
i	1	DMM/8-9-88	RAL 8-9-88	RAL 8-9-88	Revision 1 provides responses to NRC questions for Project M-49 dated May 24, 1988. All pages in this document are identical to original issue unless specifically noted on these sheets and shown by revision bars. The revision control sheets for the initial issue are contained in File No. DUK003.0103 with report NUH-001, Rev. 2.
iii	1				
v thru xxx	1				
0.0	1	DMM/8-9-88	RAL 8-9-88		
1.1-1	1	BJT for JMR 8/9/88	DMM/8-9-88		
1.1-3 thru 1.1-4	1				
1.2-1 thru 1.2-11	1				
1.3-1 thru 1.3-10	1				
1.3-12 thru 1.3-19	1				
1.4-1	1				
1.5-1	1				
1.6-1	1	BJT for JMR 8/9/88	DMM/8-9-88		
2.1-1	1	DMM/8-9-88	BJT for JMR 8/9/88		
3.1-1 thru 3.1-9	1	JRB/8-9-88			
3.1-11 thru 3.1-12	1	JRB/8-9-88			
3.2-1 thru 3.2-7	1	DMM/8/9/88	JRB for NGC 8-9-88		
3.2-9 thru 3.2-14	1				
3.2-16	1				
3.2-19 thru 3.2-20	1	DMM/8-9-88	JRB for NGC 8-9-88	RAL 8-9-88	

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3.2-23 thru 3.2-24	1	IDum 8/9/88	RAL 8-9-88	RAL 8-9-88	
3.3-3 thru 3.3-7	1	BDT 8/9/88			
3.3-9 thru 3.3-15	1				
3.3-17 thru 3.3-18	1				
3.3-20	1				
3.3-23 thru 3.3-25	1				
3.3-27 thru 3.3-31	1				
3.3-35 thru 3.3-36	1	BDT 8/9/88 BDT for JMR 8/9/88			
3.5-1	1		RAL 8-9-88		
3.6-1 thru 3.6-4	1	JRB/8-9-88	BDT for JMR 8/9/88		
4.2-1 thru 4.2-21	1	BDT for JMR 8/9/88	IDum/8-9-88		
4.3-1 thru 4.3-3	1				
4.5-1	1				
4.6-1	1				
4.7-1 thru 4.7-5	1				
5.0-1	1		IDum 8/9/88	RAL 8-9-88	

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5.1-1 thru 5.1-14	1	EDT for JMR 8/9/88	RAL 8-9-88	RAL 8-9-88	
5.1-18 thru 5.1-23	1				
5.2-1 thru 5.2-4	1		RAL 8-9-88		
7.2-1	1	JRB/8-9-88	EDT 8/9/88		
7.3-2 thru 7.3-5	1				
7.3-8 thru 7.3-9	1				
7.4-1	1				
7.4-6 thru 7.4-7	1				
7.6-1	1	JRB/8-9-88			
8.1-1 thru 8.1-6	1	IDMM 8-9-88	JRB for MGC 8-9-88		
8.1-10 thru 8.1-11	1				
8.1-13 thru 8.1-28	1				
8.1-30 thru 8.1-31	1				
8.1-33	1	IDMM 8-9-88	JRB for MGC 8-9-88		
8.1-36 thru 8.1-41	1	JRB/8-9-88	IDMM 8-9-88		
8.1-44 thru 8.1-50	1	JRB/8-9-88	IDMM 8-9-88	RAL 8-9-88	

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8.1-54	1	IDMM 8.9.88	JRB for TGC 8.9.88	RAL 8-9-88	
8.1-56 thru 8.1-59	1	IDMM 8.9.88	JRB for TGC 8.9.88		
8.1-61 thru 8.1-62	1	JRB/8-9-88	BDT 8/9/88		
8.1-68	1	IDMM 8.9.88	JRB for TGC 8.9.88		
8.1-70 thru 8.1-75	1	IDMM 8.9.88	JRB for TGC 8.9.88		
8.1-81	1	JRB/8-9-88	BDT 8/9/88		
8.1-87 thru 8.1-88	1	JRB/8-9-88	BDT 8/9/88		
8.1-97 thru 8.1-98	1	JRB for TGC 8.9.88	IDMM 8.9.88		
8.1-104 thru 8.1-105	1	JRB for TGC 8.9.88	IDMM 8.9.88		
8.1-115	1	JRB/8-9-88	BDT 8/9/88		
8.2-1 thru 8.2-8	1	IDMM 8.9.88	BDT 8/9/88		
8.2-11 thru 8.2-12	1		JRB for TGC 8.9.88		
8.2-14 thru 8.2-26	1	IDMM 8.9.88			
8.2-30	1	JRB 8.9.88			
8.2-32	1	IDMM 8.9.88			
8.2-39	1				
8.2-42 thru 8.2-53	1	IDMM 8.9.88	JRB for TGC 8.9.88	RAL 8-9-88	

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AFFECTED PAGE(S)	DOC REV	PREPARED BY / DATE	ACCURACY CHECK BY / DATE	CRITERIA CHECK BY / DATE	REMARKS
8.2-57 thru 8.2-65	1	JRB 8.9.88	JRB for MGL 8.9.88	RAL 8-9-88	
8.2-67	1	JRB 8.9.88	JRB for MGL 8.9.88		
8.4-1	1	IDMM 8.9.88	JRB 8.9.88		
8.4-5	1	IDMM 8.9.88	JRB 8.9.88		
9.2-2	1	BDT for JHR 8/9/88	IDMM 8.9.88		
10.1-2	1		BDT 8/9/88		
10.2-3	1				
10.3-5 thru 10.3-6	1				
10.3-15	1				
10.3-20	1				
10.3-22 thru 10.3-23	1			RAL 8-9-88	
A.3 thru A.4	1	BDT 8/9/88	JRB 8-9-88	JRB 8-9-88	
A.6	1		JRB 8-9-88	JRB 8-9-88	
B.1-1	1	RJM 8/9/88	BDT for JHR 8/9/88		
B.1-6	1				
B.2-2	1				
B.3-1	1	RJM 8/9/88		JRB 8-9-88	
C.1-2 thru C.1-16	1	IDMM 8.9.88	JRB for MGL 8.9.88	RAL 8-9-88	
C.2-3	1	JRB for MGL 8.9.88	IDMM 8.9.88		
C.3-2 thru C.3-3	1	JRB for MGL 8.9.88	IDMM 8.9.88		
D.1-1 thru D.1-4	1	IDMM 8.9.88	JRB for MGL 8.9.88	RAL 8-9-88	
E.1	1	(Per dwgs.)	(Per dwgs.)	(Per dwgs.)	

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AFFECTED PAGE(S)	DOC REV	PREPARED BY / DATE	ACCURACY CHECK BY / DATE	CRITERIA CHECK BY / DATE	REMARKS
i	1A	JWR 7/13/89	RAL 7/13/89	RAL 7/13/89	Changes reflect NRC approval status and registered trademark status for NUHOMS
iii	1A		RAL 7/13/89	RAL 7/13/89	
v	1A	JWR 7/13/89			
xxxiii-xxxvi	1A	NA	NA	NA	NRC's SER and cover letter and list of NUTECH supporting submittals
SER-i	1A	NA	NA	NA	
All remaining pages of Rev. 1 document	1A	JWR 7/13/89	RAL 7/13/89	RAL 7/13/89	Only the revision number has been changed to reflect NRC approval status
i	2	RAL/3-14-90	JWR 3/14/90	JWR 3/14/90	Revised to reflect organization change
11.1-1	2	RAL/3-14-90	JWR 3/14/90	JWR 3/14/90	
PS-i	2	RAL/3-27-90	JWR 3/27/90	JWR 3/27/90	Proprietary Supplement Cover Page
i	2A	RAL/7-29-91	JWR 7/29/91	RWL 7/29/91	Revised to reflect NRC approval status
11.1-1	2A	RAL/7-29-91			
PS-i	2A	RAL/7-29-91	JWR 7/29/91	JWR 7/29/91	

QEP 3-3.1.2
REV 1

PROPRIETARY INFORMATION NOTICE

This document contains confidential information of NUTECH Engineers, Inc. (hereinafter referred to as NUTECH) and its subcontractors. It is submitted to the Nuclear Regulatory Commission for review under 10CFR72. No other use, direct or indirect, and no dissemination or distribution (including publication), direct or indirect, other than to those employees of the Nuclear Regulatory Commission who require the document or any information contained herein for evaluation purposes is authorized.

Executive Summary

This Topical Report (No. NUH-002, NRC Project No. M-49) provides a generic safety analysis report for the NUTECH Horizontal Modular Storage (NUHOMS®)[1] system for twenty-four PWR fuel assemblies (NUHOMS-24P). This system provides for the safe, dry storage of spent fuel assemblies in an Independent Spent Fuel Storage Installation and is in full compliance with the requirements of 10CFR72 and ANSI 57.9. This Topical Report (No. NUH-002) was approved by the United States Nuclear Regulatory Commission on April 21, 1989 and is now designated NUH-002, Revision 1A. The related NUHOMS Topical Report (No. NUH-001, Revision 1A, NRC Project No. M-39) was approved by the United States Nuclear Regulatory Commission on March 28, 1986 for storage of seven spent PWR fuel assemblies per module (NUHOMS-07P). Accordingly, the NUHOMS-07P and the NUHOMS-24P Topical Reports are acceptable for reference in site specific license applications and it is NUTECH's understanding that the Nuclear Regulatory Commission staff does not intend to repeat the review of the related features important to safety.

The principal features of the NUHOMS-24P system which differ from those previously approved NUHOMS-07P system are:

1. An increase in the dry shielded canister and horizontal storage module capacity to 24 PWR fuel assemblies (NUHOMS-24P).

2. The utilization of credit-for-burnup assumptions in the design of the dry shielded canister and in the associated criticality analyses.

3. The addition of design details and safety analyses for an on-site transfer cask used to safely transport the fuel canisters from the fuel building to the horizontal storage module.

The NUHOMS-24P system provides long-term interim storage for spent fuel assemblies which have been out of the reactor for the equivalent of ten years or longer in accordance with the criteria set forth in this Topical Report. The fuel assemblies are confined in a helium atmosphere by a stainless steel canister. The canister is protected and shielded by a massive reinforced concrete module. Decay heat is removed by thermal radiation,

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(1) NUHOMS® is a registered trademark of NUTECH, Inc.

conduction and convection from the canister to an air plenum inside the concrete module. Air flows through this internal plenum by natural draft convection.

The canistered spent fuel assemblies are transferred from the reactor fuel pool to the concrete module in a transfer cask. The cask is aligned with the storage module and the canister is inserted into the module by means of a hydraulic ram. The NUHOMS System is a totally passive installation that is designed to provide shielding and safe confinement of spent fuel for a range of postulated accident conditions and natural phenomena.

This Topical Report describes the design features and provides the safety analysis for the storage of PWR fuel in the NUHOMS-24P system. Spent BWR fuel may also be stored using the NUHOMS system. The associated safety analysis will be provided in a future amendment to this Topical Report or in site specific license applications.

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LIST OF ABBREVIATIONS

ACI	American Concrete Institute
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
BWR	Boiling Water Reactor
10CFR	Code of Federal Regulations, Title 10
CE	Combustion Engineering
DBT	Design Basis Tornado
DSC	Dry Shielded Canister
He	Helium
HSM	Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
NDRC	National Defense Research Committee
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NUHOMS	NUTECH Horizontal Modular Storage
NUREG	Nuclear Regulatory Guide
OBE	Operating Basis Earthquake
OSHA	Occupational Health and Safety Administration
PI	Project Instruction
PWR	Pressurized Water Reactor
QEP	Quality Engineering Procedure
R.G.	NRC Regulatory Guide
SAR	Safety Analysis Report
SFA	Spent Fuel Assembly
SSE	Safe Shutdown Earthquake
TR	Topical Report

LIST OF ABBREVIATIONS
(Continued)

U.S.	United States
W	Westinghouse
atm	atmosphere
bar	bar
cm	centimerter
°C	degrees Centigrade
°F	degrees Fahrenheit
fps	feet per second
ft/s	feet per second
ft	foot
ft-lb	foot pounds
He	helium
in	inch
kg	kilogram
kw	kilowatt
k-in	kip inch
ksi	kips per square inch
MWD/MTU	megawatt days per metric ton uranium
MWe	megawatts electric
MWt	megawatts thermal
Hg	Mercury
m	meter
μ Ci/cm^2	microcuries per square centimeter
MeV	Megaelectron volt
mph	miles per hour
mm	millimeter
mrem/hr	millirem per hour
mR/hr	milliroentgen per hour
n	neutron
k_{eff}	neutron multiplication factor, effective
kN	kilonewton
N	Newton

LIST OF ABBREVIATIONS
(Concluded)

lb	pound
lbf	pounds-force
psf	pounds per square foot
psi	pounds per square inch
psia	pounds per square inch, atmospheric
psig	pounds per square inch, gauge
sec	second
sq. mi.	square mile
kips	thousand pounds
ton	ton

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

RECEIVED
APR 26 1989
NUTECH

APR 21 1989

Project M-49

NUTECH Engineers, Inc.
ATTN: Mr. William J. McConaghy, P.E.
Vice President
Waste Business Group
145 Martinvale Lane
San Jose, California 95119

Dear Mr. McConaghy:

SUBJECT: ACCEPTANCE AS A REFERENCE OF "TOPICAL REPORT FOR THE NUTECH HORIZONTAL MODULAR STORAGE SYSTEM FOR IRRADIATED NUCLEAR FUEL, NUHOMS®-24P" (NUH-002), REVISION I

The Nuclear Regulatory Commission (NRC) staff has completed its review of NUTECH Engineers, Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS®-24P (NUH-002), Revision 1. Based on this review, NRC staff has concluded that the NUTECH Horizontal Modular System (NUHOMS®) design meets the requirements of 10 CFR Part 72 as defined in this letter and subject to appropriate specifications expressed in its enclosure, the NRC staff's safety evaluation report (SER), for the safe receipt, handling, and storage of spent fuel at an independent spent fuel storage installation (ISFSI) to be located at a nuclear power plant site. This acceptability is limited to conditions and the spent fuel detailed in the TR (i.e., Revision 1), augmented by information submitted after the filing of Revision 1 and in this letter with its enclosure.

By letter dated February 26, 1988, NUTECH Engineers, Inc., submitted for review a topical report entitled, "Topical Report Amendment 2 for the NUTECH Modular Storage System for Irradiated Nuclear Fuel, NUHOMS®-24P," (NUH-001), dated February 1988 (docketed under Project No. M-49). In response to NRC staff comments, a revision to the original NUTECH report was subsequently submitted and docketed. This was Revision 1 entitled, "Topical Report for the NUTECH Horizontal Modular Storage System For Irradiated Nuclear Fuel NUHOMS®-24P," (NUH-002), Revision 1, dated July 1988.

In the SER, the staff's review examined how the submitted NUHOMS® concrete horizontal storage module (HSM) and steel dry shielded canister (DSC) design for an ISFSI meets specific requirements of 10 CFR Part 72 with respect to design, operation, and decommissioning. The staff's review addresses normal and off-normal operating conditions and accidents. Radiological, shielding, criticality, structural, and thermal aspects of the storage module design and the vendor's Quality Assurance Program have been reviewed for compliance with applicable requirements of Subparts E, F, and G of 10 CFR Part 72.

Requirements for physical protection in 10 CFR Part 73 and for offsite transport of radioactive materials in 10 CFR Part 71 were not within the scope of the TR and were not addressed in the staff's review.

Operating limits established for the module and its spent fuel content have been reviewed, and limitations and operating conditions applicable to fuel loading, onsite transfer, insertion of a canister into a module, storage operations, and surveillance are detailed in Chapter 12 of the SER (see enclosure). These specify the limitations under which the TR, with its described design and spent fuel, is accepted as a reference in a Safety Analysis Report in a 10 CFR Part 72 site-specific spent fuel storage license application. However, these are not complete; other appropriate technical specifications and limitations will apply, depending on siting or other conditions associated with a specific license application.

As a result of its evaluation, the NRC staff finds that the NUTECH Engineers Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS®-24P" (NUH-002), Revision 1, as augmented by additional information received and docketed after submittal of Revision 1, is acceptable as a reference, under the limitations delineated in the TR, as modified and expanded in the SER (enclosure), with the following exception:

> Chapter 10, Operating Controls and Limits, of the TR is not to be cited as a reference. A site-specific license application should explicitly list its proposed technical specifications. This does not preclude a license applicant's use of Chapter 10 of the TR as guidance along with Chapter 12 of the NRC staff's SER (enclosure).

It is requested that NUTECH Engineers, Inc., publish an approved version of this report, with proprietary information in a separate binder, as per Item 3, "Proprietary Information," of the Introduction of Regulatory Guide 3.48, within three (3) months of the receipt of this letter and submit 25 copies for docketing.

This revision is also to incorporate this letter with its enclosure, the SER, following the title page and a listing identifying with submittal dates, supporting supplemental information submitted after the TR, i.e., Revision 1, and docketed under Project M-49. The report identification of the approved report is to have an "A" suffix.

The NRC staff does not intend to repeat the review of the features important to safety described in the TR and found acceptable when it appears as a reference in a license application except to assure that the material presented is applicable to the application involved. The NRC staff's acceptance applies only to the features described in the TR, as augmented by the supplemental information submitted subsequent to the filing of the TR (i.e., Revision 1).

Should NRC criteria or regulations change, such that our conclusions as to the acceptability of the report are invalidated, NUTECH Engineers Inc., and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

John P. Roberts

John P. Roberts, Section Leader
Irradiated Fuel Section
Fuel Cycle Safety Branch
Division of Industrial and
Medical Nuclear Safety

Enclosure:
Safety Evaluation Report

List of Submittal Dates for Supporting Information

NUTECH Letter No.	Date	Subject
DUK-03-526	April 10, 1989	Off-site Dose Back-up Calculation
WJM-89-045	March 13, 1989	Non-proprietary Responses to Criticality Questions
WJM-89-028	February 17, 1989	Proprietary Response to Boron Credit Question
WJM-89-037	February 16, 1989	Back-up Calculations for Cask Cover Plate Bolts
WJM-89-034	February 16, 1989	Additional Criticality Computer Output
WJM-89-008	January 9, 1989	Proprietary Responses to Criticality Questions
WJM-88-258	December 7, 1988	Additional Cask Drop Computer Output
WJM-88-255	December 1, 1988	Supplementary Response to Cask Drop Load Definition Question
WJM-88-226	November 9, 1988	Responses to Remaining Follow-up Questions and Cask Drop Computer Output
WJM-88-189	September 26, 1988	Additional Responses to Selected Follow-up Questions
WJM-88-182	September 13, 1988	Responses to Selected Follow-up Questions
WJM-88-166	August 17, 1988	Additional Back-up Calculations and Computer Output
WJM-88-164	August 10, 1988	Revised Topical Report to Incorporate Questions Responses
WJM-88-163	August 9, 1988	Additional Back-up Calculations and Computer Output
WJM-88-153	August 1, 1988	Structural, Thermal, Shielding and Criticality Back-up Calculations and Computer Output
WJM-88-127	July 7, 1988	Responses to Initial Topical Report Questions
WJM-88-047	March 18, 1988	Revised Report Number for Topical Report
WJM-88-033	February 26, 1988	Submittal NUHOMS-24P Topical Report

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.

U.S. Nuclear Regulatory
Commission

Office of Nuclear Material Safety
and Safeguards

April 1989

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1.0 GENERAL DESCRIPTION

1.1 INTRODUCTION

1.1.1 Objective

This is a Safety Evaluation Report (SER), which documents the Nuclear Regulatory Commission (NRC) staff analysis and recommendations on the suitability and acceptability of the NUTECH "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS*-24P" (Reference 1), an independent spent fuel storage installation system (ISFSI), hereafter referred to as the Topical Report (TR). The TR uses the format of NRC Regulatory Guide 3.48 (Reference 2).

1.1.2 Scope

The review of the TR is oriented toward determining and justifying the extent that the TR can be used as a reference to satisfy the requirements of 10 CFR 72 (Reference 3) for an ISFSI license application. This use of the TR would be by inclusion by clear and specific reference or by repetition in the license application documents. The application contents and licensing documents, which would typically make maximum use of the TR, are the safety analysis report (SAR) (10 CFR 72.24) and the technical specifications (10 CFR 72.26), described in 10 CFR 72.44(c)).

The review also addresses the suitability of material contained in the TR to be incorporated by reference in the other documents required to be submitted with the license application, specifically: the decommissioning plan, emergency plan, environmental report, and quality assurance program. The review does not address potential reference in conjunction with the following licensing documents: physical security plan, design for physical protection, safeguards contingency plan, or personnel training program.

The review includes considerations of the appropriate parts of 10 CFR 20 for radiation protection during onsite handling, movement, and storage of spent fuel.

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* NUHOMS is a registered trademark of NUTECH Engineers, Inc.

The recommendations for approval of the NUTECH ISFSI system are limited to the level to which the system is defined. The drawings and descriptions in the TR do not constitute final construction drawings and specifications. However, except as otherwise indicated in the recommendations, the level of design and supporting rationale and analyses presented are adequate to permit the development of such designs and specifications following standard codes and practice, and sufficiently bound the final design as to not require further NRC detailed review.

This SER includes descriptions of the different functional elements of NUTECH ISFSI system; general design criteria; and evaluations of the designs, proposed operating procedures, proposed acceptance tests and maintenance program, radiological protection, decommissioning discussion, proposed operating controls and limits, and proposed quality assurance. In general, the SER has been prepared for use together with Reference 1. Figures, tables, and text of the TR are not repeated in the SER but are referenced, except where such repetition is considered essential for clarity of the SER.

The descriptions of the NUTECH ISFSI system included in Section 1.2 are for general orientation of the reviewer. The descriptions are believed to be accurate representations, but they did not form the basis for the detailed evaluations. The evaluation and recommendations are based directly on the contents of the TR (Reference 1).

1.1.3 Context

A topical report for an ISFSI constitutes a potential reference which may be cited in subsequent license applications to the NRC for permission to construct, own, use, and operate ISFSI at specific sites, or may be cited in subsequently submitted other topical reports. NRC action on a topical report may be approval, disapproval, or approval with limitations or other qualifications.

The principal use of an approved topical report in a license application is by inclusion (by repetition or specific reference) in the accompanying safety analysis report (SAR) and proposed technical specifications. Requirements for SAR are stated in 10 CFR 72.24.

Requirements for technical specifications are stated in 10 CFR 72.28 and 10 CFR 72.44. Incorporation of TR contents by reference is under the provisions of 10 CFR 72.18, which requires that those references are clear and specific. Designs and descriptions in the topical report, as it is approved, may be incorporated fully or partially. Changes and omitted material must however be fully addressed in the license application.

A topical report cannot constitute (by reference) all of an SAR or technical specification. Actual site conditions, procedures of the individual company making the application, and elements of a site's existing final safety analysis report (FSAR) also impact or must be addressed in the SAR. A topical report may provide a reference for all of the technical specification if all of the requirements of 10 CFR 72.26 and 10 CFR 72.44(c) are met and are specifically referenced in the technical specifications submitted with the license applications.

The format and content of an SAR generally follow the structure suggested by Regulatory Guide 3.48 where it is applicable. The NUTECH TR follows the same format. There is no specific requirement for the contents of a TR; it is the SAR that must be complete. TRs are reviewed to determine adequacy in meeting requirements for an SAR. TR elements become part of an SAR by reference or repetition. The TR is also reviewed for validity that the design of the included ISFSI systems meets the requirements for such systems. The requirements are principally as presented in 10 CFR 72, and as further implemented by Regulatory Guides 3.48 and 3.60 (Reference 4).

1.2 GENERAL DESCRIPTION OF NUHOMS-24P SYSTEM

The NUHOMS system is an ISFSI system that provides for horizontal, dry storage of irradiated nuclear fuel assemblies. The fuel assemblies are contained in a dry shielded canister (DSC) made of stainless steel and lead, which is transported within a heavily shielded transfer cask (TC), and which is placed inside a reinforced concrete horizontal storage module (HSM) for long term storage.

In addition to the DSC, TC, and HSM, the NUHOMS system also requires:

1. Handling and transfer equipment to load the DSC with fuel, to seal the DSC, to move the loaded DSC inside the TC from the fuel pool building to the HSM (elsewhere on the site), and to insert the DSC into the HSM; and

2. An infrastructure of procedures, interfaces with the host plant, personnel qualifications, organization, training, quality assurance, and support services. Figures 1.1 and 1.2 show schematically the major physical components and operations of the NUHOMS system.

The TR presents for review and approval a design in which the DSC holds 24 irradiated pressurized water reactor (PWR) fuel assemblies and in which the HSMs are arranged in back-to-back arrays. There may be any number of arrays; however, the overall exposure levels are dependent on the actual number of arrays, and must therefore be checked in any license application.

The designs of the HSM, DSC, TC, handling and transfer equipment, and nuclear fuel assemblies to be stored are described in more detail in the following subsections.

1.2.1 Horizontal Storage Module

HSMs are constructed in arrays of reinforced concrete and structural steel. An HSM within a back-to-back side-by-side array is 6.096 m (20′) deep, 4.572 m (15′) high (plus 0.914m (3′) high air outlet shielding blocks), and has the DSC stored 2.642m (8′-8") on centers. A 3x2 HSM array would be 12.192 m (40′) deep and 9.144m (30′) across. The concrete walls and roof are intended to be of sufficient thickness to attenuate radiation so that the average contact dose rate on the outside surface of the HSM is less than 20 mrem/hour.

The TR reference design is based on an installation of six modules arranged in a 2x3 array on a load-bearing foundation. Each HSM can hold one DSC. The modules are arranged back-to-back so that loading of each module is accomplished through an opening in the front. The center of the opening is approximately 2.591m (8.5 feet) above the surface of the foundation.

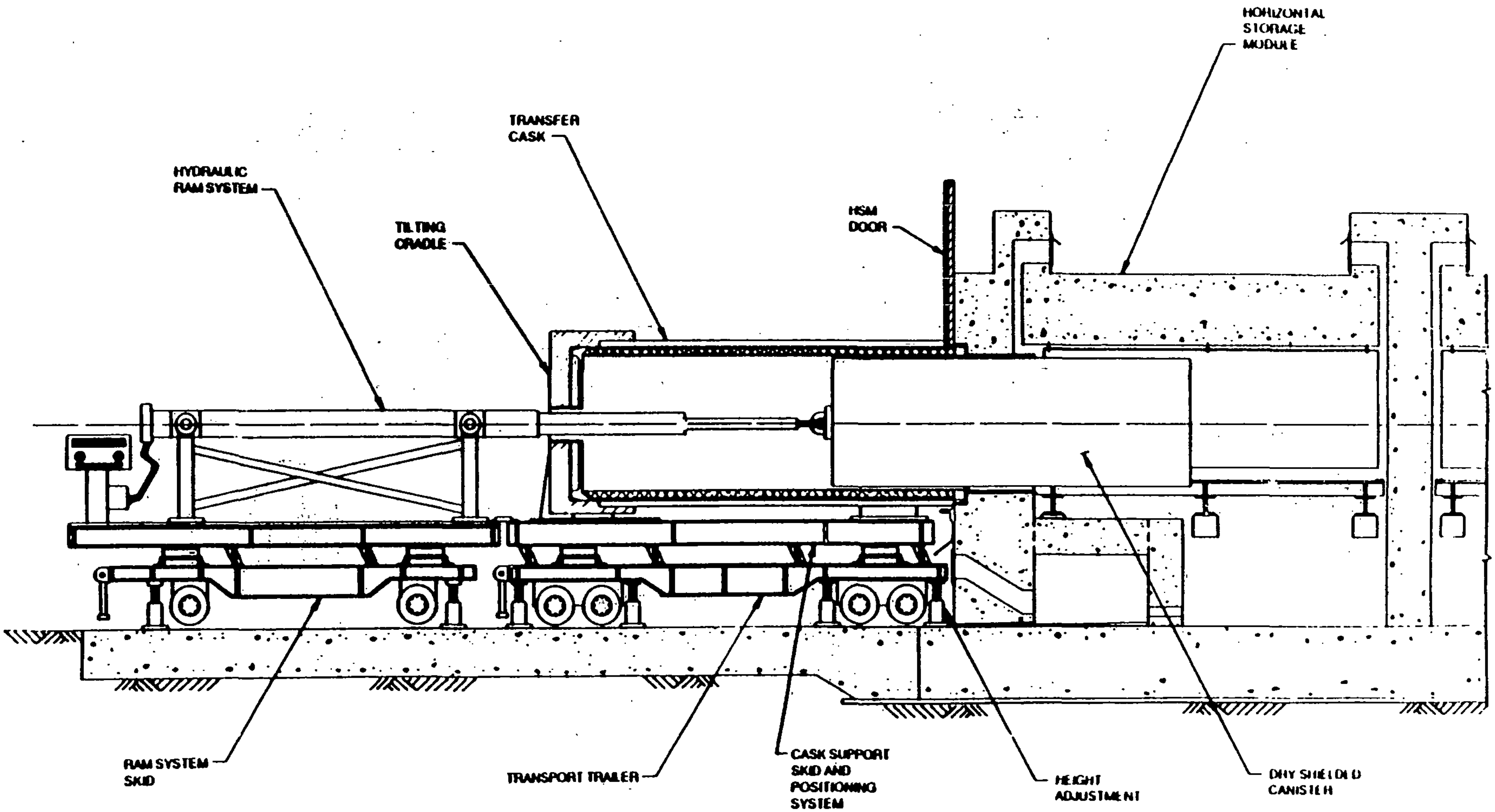

Figure 1.1

PRIMARY COMPONENTS OF THE NUHOMS SYSTEM

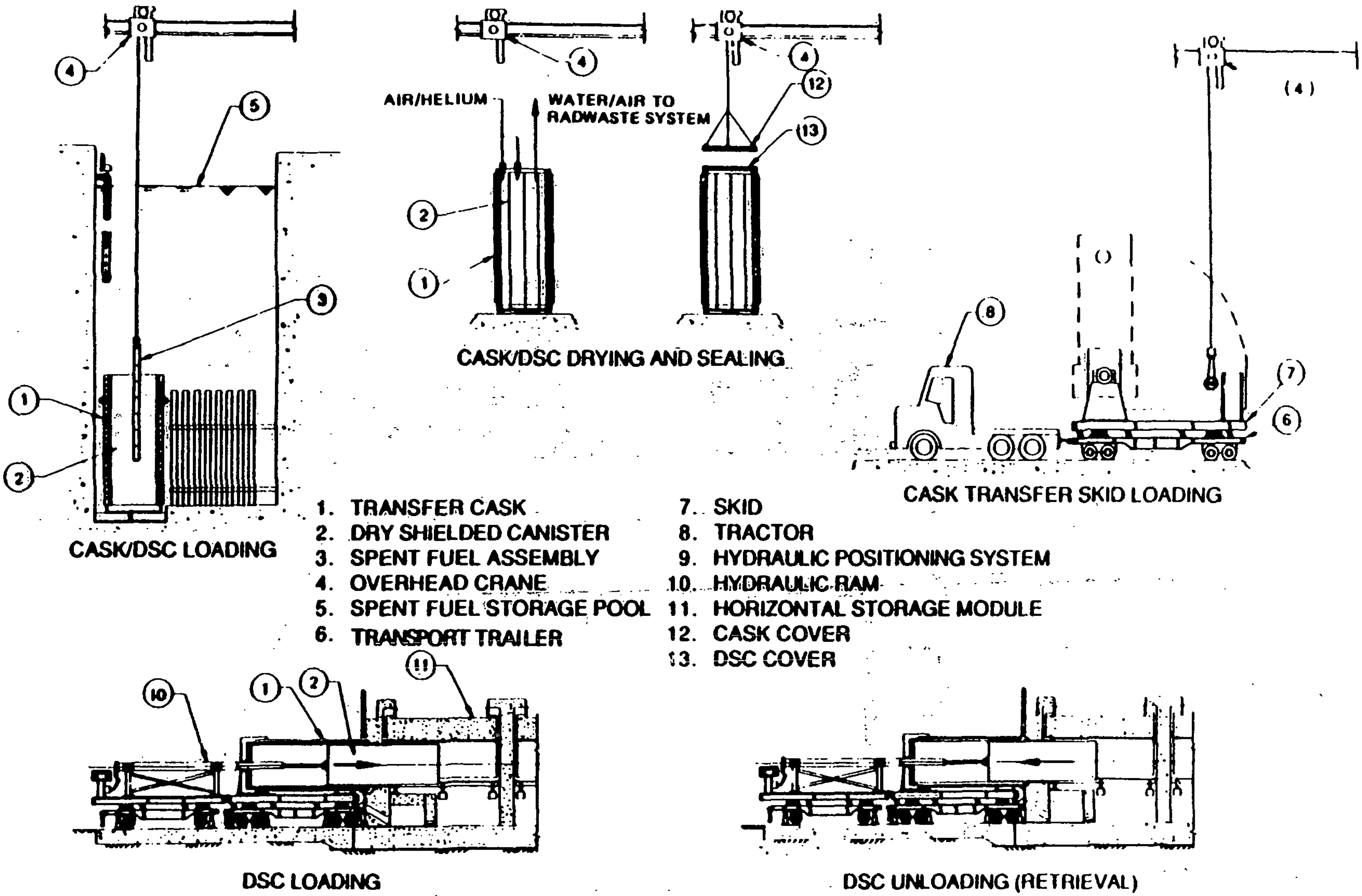

Figure 1.2

PRIMARY OPERATIONS FOR THE NUHOMS SYSTEM

After the HSM has been loaded with a DSC, a steel door is lowered down over the front opening and tack welded in place.

There are two steel rails inside the HSM running front-to-back which support the DSC while it is in storage. Each HSM has an air inlet on the front below the DSC opening and two air outlets on the roof to permit natural convective air cooling of the DSC while it is in storage. The inlet and outlets are shielded to reduce radiation doses at the exterior of the HSM.

1.2.2 Dry Shielded Canister

The DSC consists of a stainless steel (ASME SA-240, Type 304) cylindrical body, two shielded end plugs, and an internal basket to hold and support twenty-four irradiated pressurized water reactor (PWR) fuel assemblies.

The DSC body is a 15.9 mm (0.625") thick stainless steel cylinder. It has an outside diameter 1.708 m (67.25"). Its length is 4.724 m (186.00"). When welded shut, the DSC may be evacuated through valves, backfilled with helium at 1.2 bar. The valves may then be fully sealed by welding.

The internal basket is composed of twenty-four separately formed square cells. Structural support of the cells inside the DSC is provided by circular stainless-steel spacer disks. Longitudinal support of the disks is provided by four support rods that run the length of the canister from one end shield to the other. The cells and supporting assembly are fabricated of Type 304 stainless steel.

Each end of the DSC is equipped with a shielded end-plug so that when the canister is inside the transfer cask or the HSM, the radiation dose at the ends is limited. The top end shield is 184 mm (7.25") total thickness of stainless steel and lead. The bottom end shield is 147.0 mm (6.0") total thickness of stainless steel and lead.

The DSC has redundant seal welds for the top and bottom end plugs. The bottom is shop-welded during fabrication. The top cover plates are welded at the site after fuel is loaded in the DSC. The valve connections (drain and air purge lines) are also sealed at the site.

The DSC has a grapple attachment integrated with its bottom end to provide for insertion and withdrawal at the HSM by use of a trailer mounted hydraulic ram (ram design is left to site specific license application). The ram is inserted through the bottom port of the TC, is connected to the DSC, and inserts the DSC by pushing it out of the TC into the HSM or withdraws it by pulling it out of the HSM into the TC. The DSC slides on the HSM rails and internal TC surface in these operations.

1.2.3 Transfer Cask

The transfer cask consists of a structural steel and lead shell with a neutron shield water jacket and overflow tank, an integral steel bottom end incorporating a solid neutron shield, a bolted-on vented steel top cover incorporating a solid neutron shield, and a smaller diameter bolted-on steel bottom cover over the ram access port incorporating a solid neutron shield. The TC is equipped with a drain plug for draining the cask and provisions for filling and venting the neutron shield water jacket.

The TC has an outer diameter of 2.165 m (85.27") (exclusive of the overflow tank), an inner diameter of 1.727 m (68"), an inner clear length of 4.750 m (187"), and an overall outer length of 5.009 m (197.2").

The TC is intended to be hoisted by trunnions on its sides. The DSC is to be loaded with fuel assemblies in a vertical orientation within the TC. For transport the TC is placed in a specially designed carrying assembly and rotated to the horizontal position (as shown in Figure 1.2).

1.2.4 Handling and Transfer Equipment

In order to support the operation of the NUHOMS system, several additional components are needed for the handling of both the fuel and the DSC and for the transfer of the loaded and sealed DSC to the HSM. Designs or selection of these items are left to the site-specific license application. They include the following major components:

1. Lifting assemblies or crane adaptor assemblies for the DSC, TC, DSC cover, and TC cover.

2. Welding machine suited to remote welding of the DSC cover.

3. DSC evacuation and helium backfill systems.

4. Transfer vehicle capable of moving the loaded cask across the site.

5. Jack support system for the transfer vehicle to be used to restrict relative motion between the ground (loading apron) and the trailer.

6. Cask positioning skid to adjust the cask position at the HSM to allow proper alignment before the DSC is transferred to the HSM.

7. Cask restraint system to prevent relative motion between the cask/skid and the HSM during inserting or withdrawing operation.

8. Optical alignment system to align the loaded cask with the HSM opening.

9. Ram and grapple apparatus to push the DSC from the TC into the HSM and to withdraw the DSC from the HSM into the TC.

10. Components to reverse the process in order to retrieve fuel assemblies from the DSC.

The staff has reviewed these components primarily from the point of view of feasibility. That is, these components have been reviewed only to determine if the staff believes that all operations required to support the NUHOMS system can be performed by current technology, that such equipment exists or can be fabricated, and that such a system could perform its required functions. Review and approval of all NUHOMS-24P ISFSI system physical components is left to the site specific license application, with the specific exceptions of the DSC, TC, and HSM.

1.2.5 Stored Materials

Each HSM holds one DSC and each DSC holds twenty-four irradiated PWR fuel assemblies. The proposed system is designed to permit storage of any PWR fuel with the following criticality and radiological characteristics:

1. Initial uranium content: 472 kg/assembly or less.

2. Initial enrichment: 4.0% (^{235}U equivalent) or less.

3. Fuel rod cladding of zircaloy.

4. No known or suspected cladding damage.

5. Irradiated fuel initial enrichment less than or equal to 1.45 weight percent ^{235}U unirradiated fuel.

6. Post irradiation cooling time such that:

   a) Decay Heat Power per Assembly <0.66 kW,
   b) Total Gamma Ray Source per DSC <3.85 x 10^{16} MeV/sec, (1.11 x 10^{17} gammas/sec)
   c) Total Neutron Source per DSC <3.715 x 10^{9}.

7. Initial fuel rod fill gas pressure of less than 480 psig.

A fuel assembly not meeting the specified conditions must be analyzed specifically before it can be stored in the proposed NUHOMS design.

1.3 IDENTIFICATION OF AGENTS AND SUBCONTRACTORS

No subcontractors for design are identified in subsection 1.4 of the TR, however Duke Power Company, Inc. is identified as responsible for the design of the HSM and performance of the criticality analysis.

1.4 GENERIC HORIZONTAL STORAGE MODULE ARRAYS

The TR is based on a 2x3 array of front-loaded HSMs. Although the TR states that other arrays are possible, none were presented for review and approval. Review of the design indicates that the following other HSM arrays are adequately included by the design of the 2x3 array: 2x1 and 2x2, using the exterior wall designs of the 2x3 array. Shielding calculations provided only cover the situation where HSM are installed back-to-back.

2.0 PRINCIPAL DESIGN CRITERIA

2.1 SUMMARY AND CONCLUSIONS

This section of the SER presents a review of the design criteria developed and presented in the TR to determine the suitability of the NUHOMS-24P design criteria. Sections 3 through 10 evaluate the use and satisfaction of the criteria in the designed system components. Subpart F of 10 CFR Part 72 sets forth design criteria for the design, fabrication, construction, testing and performance of structures, systems and components important to safety in an ISFSI. This section presents a discussion of the applicability of these criteria to the NUHOMS system and the degree to which the NUTECH, Inc. TR is in compliance with these criteria. Section headings in this section are the same as applicable subsections of Subpart F of Part 72.

Section 3 of the TR identifies sources for design criteria. These sources, and their acceptability are summarized in Table 2.1. The NRC staff concurs in the selection of sources in the TR, with the following exception:

ACI 349-85 was used in lieu of ACI 349-80 (Reference 5), which is cited in paragraph 6.17.2 of ANSI/ANS-57.9-1984 (Reference 6). This standard is endorsed with modifications in Regulatory Guide 3.60 by reference. There is no impact on the designs in the TR however and therefore NUTECH's use of ACI 349-85 in this TR for design of the HSM is considered acceptable. The NRC staff also notes that as Regulatory Guide 3.60 is a "guide," the use of a substitute determined to be acceptable to the NRC is satisfactory.

Section 3 of the TR also establishes design criteria used subsequently for design procedures and designs discussed in Sections 4 and 8 of the TR. These design criteria, as presented in Section 3 of the TR, are considered acceptable with the following exceptions:

1. There are two discrepancies in the methodology used by NUTECH for combining loads for the HSM. The factors to be applied to the dead load and the live load are not consistent with ANSI 57.9-84; however, the safety of the HSM is not compromised because of the design margin.

TABLE 2.1 DESIGN CRITERIA SOURCES CITED IN THE TR

(Sources are more fully described at TR Section 3.6)
(Similar citations within TR Section 3 are not repeated)

TR Ref	Source	Use	NRC Staff Comments
3.1.1.2, Tbl 3.1-3	NUREG/CR-2397 (ORNL-CSD-90)	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.2, Tbl 3.1-3	ORNL/TM-7431	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.2, Tbl 3.1-3	ANSI/ANS-5.1-1979	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.2, Tbl 3.1-3	A.D. Little, Inc., "Tech Spt for Rad Stds for Hi-Lvl Rad Waste Mgt"	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.3, Tbl 3.1-4	NUREG/CR-2397 (ORNL/TM-7431)	Development of radiological criteria using ORIGEN calculations	Acceptable
3.1.2.2	ANSI 57.9-1984	Design std for cask handling crane	Acceptable
3.1.2.2	Reg Guide 1.60	Seismic Design Response Spectra	Acceptable
3.1.2.2	Reg Guide 1.61	Seismic Design Damping Values	Acceptable
3.1.2.2	ANSI/ANS 57.9-1984	Operational Handling Loads	Acceptable
3.1.2.2	ANSI/ANS 57.9-1984	Accident Drop Loads	Acceptable
3.1.2.2	ANSI/ANS 57.9-1984	Thermal and Dead Loads	Acceptable
3.1.2.2	Reg Guide 1.76	Tornado Wind Loads	Acceptable
3.2.1	NUREG 0800	Tornado Missiles	Acceptable
3.2.1.2	ANSI A58.1-1982	Tornado Wind MPH to Pressure Conversion	Acceptable
3.2.4	ANSI A58.1-1982	Snow and Ice Loads	Acceptable
3.2.5.1	ACI 349-85	Reinforced Concrete Design	Acceptable, however ACI 349-80 is currently approved by NRC (per Reg Guide 3.60)
3.2.5.1	ANSI/ANS 57.9-1984	Load Combinations for HSM Design	Acceptable
3.2.5.2	ASME B&PV Code (1983) Sect III, Div 1, Subsec NB for Class 1 comp.	DSC allowable stresses	Acceptable
3.2.5.3	ASME B&PV Code (1983) Sect III, Subsec NC for Class 2 comp.	TC allowable stresses	Acceptable
3.2.5.3	ANSI N14.6-1986	Allowable stresses for lifting trunnions in fuel bldg.	Acceptable
Tbl 3.2-1	AISC Code for Struct Steel	DSC Support Assy Design	Acceptable for design stresses, not for load combs.

TABLE 2.1 DESIGN CRITERIA SOURCES CITED IN THE TR (cont'd)

TR Ref	Source	Use	NRC Staff Comments
Tbl 3.2-1	ASME B&PV Code (1983) Sect III, Subsec NC for Class 2 comp.	Allowable stresses for lifting and support trunnions on site transfer	Acceptable
3.3.2.1	ASME B&PV Code (1983) Sect III, Div 1, NB	DSC pressure boundary weld inspection	Acceptable
3.3.4.2	STUDSVIK/NR-81/3	CASMO-2 Fuel Assy Burnup Prog.	Acceptable
3.3.4.2	ORNL, "SCALE-3:_"	Shielding Anal Seq. No. 2	Acceptable
3.3.4.2	ORNL, "SCALE-3:_"	Criticality Safety Anal Seq. No. 2	Acceptable
	SAND 87-0151	Major neutron absorbers	Acceptable
3.3.4.3	ANSI/ANS 57.2-1983	Criticality criteria	Acceptable
	ANSI/ANS 8.17-1984	Fuel burnup credit	Acceptable
3.3.4.4	ANSI/ANS 8.17-1984	Double contingency principle	Acceptable
3.3.7.1	PNL 6189	Temp limits for dry stored fuel	Acceptable

2. The design criteria for the DSC support assembly does not include the dead load of the DSC for the off-normal case; however, the actual analysis does include the DSC dead load.

3. The derivation of the allowable shear stress for the DSC support assembly as used by NUTECH would result in exceeding the code specified in ANSI 57.9-84 section 6.17.3.2.1 for steel design. Because NUTECH selected an overly conservative temperature in conjunction with the seismic event, the NRC judged that the material allowable was also conservative. There will not be a safety problem if a lower temperature is used in the derivation.

4. NUTECH proposed a 10% value of critical damping for the DSC and TC for the accident drop case. This value is higher than recommended by Regulatory Guide 1.61 (Reference 7). The staff evaluated this deviation and determined that 7% is a conservative estimate for the damping coefficient and also determined that no safety problems will occur for the drop if 7% damping is used.

2.2 FUEL TO BE STORED

The NUHOMS-24P system is designed for dry, horizontal storage of irradiated PWR fuel from nuclear power stations. Acceptable fuel characteristics are presented in subsection 1.2.5 of this report and elaborated in Table 3.2-1 of the TR. The principal design parameters of the fuel to be stored are intended to accommodate standard PWR fuel designs manufactured by Babcock and Wilcox, Combustion Engineering, Westinghouse and Advanced Nuclear Fuels.

The physical parameters of the DSC design are based on a hybrid set of design parameters which will accommodate standard fuel assembly arrays of (1) 15x15/208 and 17x17/264 designed by Babcock and Wilcox, and (2) 14x14/176, 15x15/216, and 16x16/236 designed by Combustion Engineering. The fuel assemblies 14x14/179, 15x15/204, and 17x17/264 designed by Westinghouse were listed in Table 3.1-2 of the TR for general reference only and are not bounded by the design case for the TR. The design case is B&W 15x15/208.

The design basis for nuclear criticality safety is based on standard Babcock and Wilcox 15 x 15 fuel assemblies with an initial enrichment of 4 weight percent ^{235}U. The design basis for radiation protection is based on 4.0 weight percent ^{235}U B&W 15 x 15 fuel irradiated to 40,000 MWd/MTHM at a specific power of 37.5 MW/MTHM with a post irradiation cooling time of ten years before being stored in the NUHOMS-24P system.

The fuel cladding temperature limits used by the applicant are based on the work of I.S. Levy, et al. (Reference 36). In developing limits, the applicant relied upon the following restrictions: (1) burnup $\leq$40,000 MWD/MTU, (2) rod fill pressure up to 480 psig, and (3) cooling times of ten years or more. These restrictions must be satisfied by the stored fuel. The last restriction limits the assembly power to 0.66 kW, and to 0.66 kW at ten years cooling time.

The results of this safety review are that the use of these design parameters for the fuels meet the requirements of 10 CFR Part 72 as applied to the DSC design, criticality design, and shielding design.

2.3 QUALITY STANDARDS

Quality standards for structures, systems and components important to safety are required by Sections 72.122(a) and 72.140 of 10 CFR 72. Sections 3.4 and 11 (which incorporates Section 11 of Reference 4 by reference) of the TR identify components of the NUHOMS-24P system that are classified as important to safety. A quality standard provides numerical criteria and/or acceptable methods for the design, fabrication, testing, and performance of the structures, systems and components important to safety. These standards should be selected or developed to provide sufficient confidence in the capability of the structure, system, or component to perform the required safety function. Since quality standards are generally embodied in widely accepted codes and standards dealing with design procedures, materials, fabrication techniques, inspection methods, etc., judgments regarding the adequacy of the standards cited in the TR are presented in the sections of this report where the standards are applicable.

2.4 PROTECTION AGAINST ENVIRONMENTAL CONDITIONS AND NATURAL PHENOMENA

Section 72.122(b) of 10 CFR 72 requires the licensee to provide protection against environmental conditions and natural phenomena. Section 3.2 of the TR describes the structural and mechanical criteria for tornado and wind loadings, tornado missile protection, flood protection, seismic design, snow, ice and dead loads, pressure and thermal loads resulting from normal operating conditions and accident conditions, normal and accident handling loads, accidental drop loads, and combined loads.

This section discusses the adequacy of the selected criteria for protecting the following components against environmental conditions and natural phenomena: (1) the reinforced concrete HSM and the HSM passive ventilation systems, (2) the DSC support assembly, (3) the DSC, including the internal basket components and the shielded end plugs, and (4) the on-site transfer cask, including the shield materials, structure and upper and lower trunnions. The above mentioned structures and component are important to safety because they contribute to the safe confinement of the radioactive spent fuel assemblies. The technical bases for determining the adequacy of these criteria are specified by the regulatory requirements to consider the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take account of limitations of data. Since the NUHOMS system was not designed for a specific site, the regulatory requirement is interpreted to mean that the NUHOMS system should be reviewed against the environmental conditions and natural phenomena provided for either by the limits specified in the TR or against the most severe of the natural phenomena that may occur within the boundaries of the United States. Table 2.2 summarizes the design criteria used in the TR for design or evaluation for normal operating conditions. Table 2.3 summarizes the criteria for off-normal operating conditions and Table 2.4 summarizes the criteria for the accident conditions.

As can be seen in Tables 2.2, 2.3, and 2.4, some of the design criteria for safety related components are not explicitly defined by codes or regulations. In some cases NUTECH has applied engineering judgment to determine a performance envelope or design criteria for the system based on the intent of 10 CFR 72.122. The SER review is oriented on satisfaction of Regulatory Guides 3.48 and 3.60 primarily, with recognition that these are

TABLE 2.2 SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Req. Guides Cited by NUTECH	NRC Staff Comments/Suitability
Horizontal Storage Module	Dead load	8.1.1.5	Dead weight including loaded DSC	ANSI 57.9-1984 ACI 349-85 and ACI 349R-85	Verified by SER
	Load combination	Tbl 3.2-5	Load combination methodology	ANSI 57.9-6.17.1.1	Acceptable if dead weight increased by 5% over estimated value. Acceptable if live load varied between 0% and 100% of estimated load to achieve most adverse conditions.
	Design Basis operating temp	8.1.1.5	DSC with spent fuel rejecting 15.8 kW decay heat. Ambient air temperature range $0^{o}F$ to $100^{o}F$	ANSI 57.9-1984	Verified by SER
	Normal handling loads	8.1.1.4	Hydraulic ram load: 20,000 lb. (25% loaded DSC weight)	ANSI 57.9-1984	Verified by SER
	Snow and Ice Loads	3.2.4	Maximum load: 110 psf (included in live load)	ANSI 57.9-1984	Verified by SER
	Live Loads	8.1.1.5	Design load: 200 psf	ANSI 57.9-1984	Verified by SER
Dry Shielded Canister	Dead Loads	8.1.1.2	Weight of loaded DSC: 72,000 lb. nominal, 80,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
	Design Basis Internal Pressure Load	8.1.1.1	DSC internal pressure $\leq$9.7 psig	ANSI 57.9-1984	Verified by SER
Structural Design		Tbl 3.2-6	Service Level A and B	Also see ASME B&PV Code Section III, Div 1, NB, Class 1, Service Level A,B	Verified by SER

TABLE 2.2 SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Req. Guides Cited by NUTECH	NRC Staff Comments/Suitability
DSC	Design Basis Operating Temp Loads	8.1.1.2 8.1.2.2 Tbl. 3.2-6	DSC decay heat 15.8 kW. Ambient air temperature $0^{o}F$ to $100^{o}F$	ANSI 57.9-1984	Verified by SER
	Operational Handling Loads	8.1.1.2 Tbl. 3.2-6	Hydraulic ram load: 20,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
DSC Support Assembly	Dead Loads	8.1.1.4 Tbl. 3.2-7	Loaded DSC + self weight: 85,000 lb.	ANSI 57.9-1984 AISC Code	Verified by SER
	Operational Handling Load	8.1.1.4 Tbl. 3.2-7	DSC reaction load with hydraulic ram load: 20,000 lb.	ANSI 57.9-1984	Verified by SER
Transfer cask (on-site) Structure: Shell, rings, ends, etc.	Normal operating condition	Tbl. 3.2-8	Service Level A and B	ASME B&PV Code, Section III, Div. 1, NC, Class 2	Verified by SER
	Dead Loads	8.1.1.9	a) Vertical orientation, self weight + loaded DSC + water in cavity: 200,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
Structure: Shell, rings, ends, etc.			b) Horizontal orientation, self weight + loaded DSC on transfer skid: 193,000 lb. nominal, 200,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
	Snow and Ice Loads	3.2.4	External surface temperature of cask will preclude buildup of snow and ice loads wr use: 0 psf	10 CFR 72.122(b)	Verified by SER

TABLE 2.2 SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Req. Guides Cited by NUTECH	NRC Staff Comments/Suitability
Shell, rings, ends	Design Basis Operating Temp. Loads	8.1.1.9, 8.1.2.2	Loaded DSC rejecting 15.8 kW decay heat. Ambient air temperature range 0^{o}F to 100^{o}F.	ANSI 57.9-1984	Verified by SER
TC Upper Trunnions	Operational Handling Loads	8.1.1.9	a) Upper lifting trunnions while in fuel building:		
			i) stress must be less than yield stress for 6 times critical load of 115,000 lb./trunnion, nominal	ANSI N14.6-1978	Verified by SER
			ii) stress must be less than ultimate stress for 10 times critical load	ANSI N14.6-1978	Verified by SER Also see: NUREG-0612 and NOG-1-1983 and WRC-297
Upper Trunnions	Op. Handling	Append. C	b) Upper lifting trunnions for on-site transfer: 118,000 lb./trunnion, 94,000 lb./shear, 29,500 lb./trunnion axial.	ASME B&PV Code Section III, NC, Class 2	Verified by SER
Lower Trunnions	Op. Handling	8.1.1.9	c) Lower support trunnions: weight of loaded cask during down loading and transit to HSM	ASME B&PV, Section III, NC Class 2	Verified by SER Also see WRC-297
Shell	Op. Handling	8.1.1.9	d) Hydraulic ram load due to friction of extracting loaded DSC: 20,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
Bolts	Normal operation	8.1.1.9 Tbl. 3.2-9	Service Levels A, B and C Ave. Stress less than 2 Sm Max. Stress less than 3 Sm	ASME B&PV Section III, NC, Class 2	Verified by SER

TABLE 2.3 SUMMARY OF DESIGN CRITERIA FOR OFF-NORMAL OPERATING CONDITIONS

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
HSM	Off-normal Temperature	8.1.1.5	-40^{o} to 125^{o}F ambient temperature	ANSI 57.9-1984	Verified by SER
	Jammed Condition Handling	8.1.1.4	Hydraulic ram load equal to 100% of DSC: 80,000 lb. nominal	ANSI 57.9-1984	Verified by SER
	Load combination	Tbl. 3.2-5	Load combination methodology	ANSI 57.9-1984 - 6.17.1.1	Acceptable if dead weight increased by 5% over estimated value. Acceptable if live load varied between 0% and 100% to achieve the most adverse conditions
DSC	Off-normal Temperature	8.1.2.2	-40 to 125^{o}F ambient temperature	ANSI 57.9-1984	Verified by SER
	Off-normal Pressure	8.1.1.2	DSC internal pressure less than 9.7 psig	ANSI 57.9-1984	Verified by SER
	Jammed Condition Handling	8.1.2.1	Hydraulic ram load equal to 80,000 lb. nominal	ANSI 57.9-1984	Verified by SER
Structural Design	Off-normal Conditions	Tbl. 3.2-6	Service Level C	ASME B&PV Section III, Div 1, NB, Class 1	
DSC Support	Jammed Condition Handling	8.1.1.4	Hydraulic ram load: 80,000 lb. nominal	ANSI 57.9-1984	Verified by SER

TABLE 2.3 SUMMARY OF DESIGN CRITERIA FOR OFF-NORMAL OPERATING CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Req. Guides Cited by NUTECH	NRC Staff Comments/Suitability
DSC Support	Load combination	Tbl. 8.2-11	Load combination methodology		Neglects deadload of DSC, which must be present. (Note SER verified that actual design did include DSC)
	Combined stresses	Tbl. 8.2-11	Calculation of allowable stresses	ANSI 57.9-1984, 6.17.3.2.1	Shear stress limit in TR Tbl. 8.2-11 is higher than allowed by code.
TC	Off-normal Temperature	8.1.1.8 8.1.2.2	-40 to 125°F ambient temperature	ANSI 57.9-1984	Verified by SER
	Jammed Condition Handling	8.1.2.1	Hydraulic ram load: 80,000 lb. nominal	ANSI 57.9-1984	Verified by SER
Structural Design	Off-normal Conditions	Tbl. 3.2-8	Service Level C	ASME B&PV Section III, Div 1, NC, Class 2	Acceptable
Bolts	Off-normal Conditions	Tbl. 3.2-9	Service Level C Ave. stress less than 2 Sm Max. stress less than 3 Sm	ASME B&PV Section III, Div. 1, NC, Class 2	Acceptable

TABLE 2.4 SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Req. Guides Cited by NUTECH	NRC Staff Comments/Suitability
HSM	Design Basis Tornado	3.2.1	Max. velocity 360 mph Max. wind pressure 304 psf	NRC Reg. Guide 1.76 ANSI 58.1 1982	Adequate
	Load Combination	Tbl. 3.2-5	Load combination methodology	ANSI 57.9-84 6.17.1.1	Acceptable if dead weight increased by 5% over estimated value. Acceptable if live load varied between 0% and 100% to achieve most adverse conditions.
	DBT Missiles	3.2.1	Max. velocity 126 mph Types: Automobile 3967 lb. 8 in. diam shell 276 lb. 1 in. solid sphere	NUREG-0800 Section 3.5.1.4	Verified by SER
	Flood	3.2.2	Max. water height: 50 ft. Max. velocity: 15 ft/sec	10 CFR 72.122	Adequate for limit design. Licensee to determine site design parameters and check against ACI 349-80 equation 2.4.7.
	Seismic	3.2.3	Horizontal ground acceleration 0.25 g (both directions) Vertical ground acceleration 0.17 g 7% critical damping	NRC Reg Guides 1.60 and 1.61	Adequate
	Accident Condition Temperature	8.2.7.2	HSM vents (inlet/outlet) blocked for 48 hrs or less. HSM inside surface temp: 395°F	ANSI 57.9-1984	Verified by SER
	Fire and Explosions	3.3.6	Site specific	10 CFR 72.122(c)	Not designed for by NUTECH. Must evaluate on site specific basis.

TABLE 2.4 SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
DSC	Accident Drop	8.2.5	Equivalent static deceleration: 75 g vertical end drop 75 g horizontal side drop 25 g corner drop with slapdown (corresponds to an 80 inch drop height)	10 CFR 72.122(b)	Verified by SER Also see i) EPRI report NP-4830 ii) LLNL report UCID-21246
			Structural damping during drop: 10%	Reg Guide 1.61	10% damping value exceeds R.G. 1.61 guidance. A 7% value has been evaluated by the staff and has been accepted. Verified by SER.
	Flood	3.2.2	Maximum water height 50 ft.	10 CFR 72.122	Adequate for limit design
	Seismic	3.2.3	Horizontal acceleration 1.0 g Vertical acceleration 0.68 g 3% critical damping	Reg. Guides 1.60 and 1.61	Verified by SER
	Accident Internal Pressure (Loss of cask neutron shield)	8.2.9	DSC internal pressure: 49.1 psig based on 100% fuel clad rupture and fill gas release, 30% fission gas release, and ambient air temperature = 125°F.	10 CFR 72.122(b)	Verified by SER
	Accident Internal Pressure (HSM vents blocked)	8.2.7	DSC internal pressure: 46.7 psig based on 100% fuel clad rupture and fill gas release, and ambient air temperature = 125°F. DSC shell temperature 455°F	10 CFR 72.122(b)	Verified by SER
	Accident Conditions	Tbl. 3.2-6	Service Level D	ASME B&PV Section III Div. 1, NB, Class 1	Acceptable with operational controls. See para. 10.3.2.9 of TR.

TABLE 2.4 SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
DSC Support Assembly	Seismic	3.2.3	DSC reeaction loads: horizontal acceleration 0.4g vertical acceleration 0.27 g 7% critical damping	Reg. Guides 1.60 and 1.61	Verified by SER
	Load combination	Tbl. 8.2-11	Load combination methodology	ANSI 57.9-84 6.17.3.2.1	Shear stress limit in TR Tbl. 8.2-11 is higher than allowed by code.
TC	Design Basis Tornado	3.2.1	Max. wind velocity 360 mph Max. wind pressure 397 psf	Reg. Guide 1.76 and ANSI 58.1-1982	Verified by SER
	DBT Missiles	3.2.1	Automobile 3967 lb. 8 in. diameter shell 276 lb.	NUREG-0800 Section 3.5.1.4	Verified by SER
	Flood	3.2.2	Cask use to be restricted by administrative controls.	10 CFR 72.122	Adequate, must verify license application invokes controls.
	Seismic	3.2.3	Horizontal ground acceleration 0.25 g (both directions) Vertical acceleration 0.17 g 3% critical damping	Reg. Guides 1.60 and 1.61	Verified by SER
	Accident Drop	8.2.5	Equivalent static deceleration 75 g vertical end drop 75 g horizontal side drop 25 g corner drop with slapdown (corresponds to an 80 inch drop height)	10 CFR 72.122(b)	Verified by SER Also see i) EPRI report NP-4830 ii) LLNL report UC.D-1246
			Structural damping during drop 10%	Reg. Guide 1.61	10% damping exceeds R.G. 1.611 guidance; however, 7% has been evaluated by the staff and accepted. Verified by SER

TABLE 2.4 SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
TC bolts	Accident Drop	Tbl. 3.2-9	Service Level D	ASME B&PV Section III, NC, Class 2	Verified by SER
TC Structural Design	Accident	Tbl. 3.2-8	Service Level D	ASME B&PV Section III, NC, Class 2	Verified by SER Acceptable with operational controls. (See para. 10.3.2.9 of TR)
	Fire and Explosions	3.3.6	Site specific	10 CFR 72.122(c)	Not designed for by NUTECH. Must evaluate on site specific basis.
	Internal Pressure	-	Not applicable because DSC provides pressure boundary	10 CFR 72.122(b)	Verified by SER

"guides." Where deviations from these guides occur, or for areas not specifically covered, the NRC staff has reviewed NUTECH's selection or derivation of criteria using the principles of the guides, accepted codes, and engineering practices as standards.

There are cases where the minimum criteria used by NUTECH are considered unacceptable (see Tables 2.2, 2.3, and 2.4), being contrary to the Regulatory Guides or accepted codes and standards. This is not prima-facie evidence that there is a safety problem, however, since the actual design may have exceeded the minimum criteria to the extent of satisfying acceptable higher criteria. In addition, different codes use different design bases, combinations of factored loads, and derivation of allowable stresses. Thus apparent problems may not exist on more detailed examinations of the actual design (SER Section 3).

Some load sources and load combinations that would normally be included in a structural analysis have been omitted by NUTECH. The basis is typically that another, more severe loading case envelopes the condition. For example, the design basis tornado (DBT) wind loadings are typically higher than non-DBT wind loadings. These rationale have been reviewed by the NRC staff in conjunction with review of the criteria, and except where noted to the contrary, have been determined to be acceptable.

NUTECH defined the following normal operating events: dead weight loads, design basis internal pressure loads, design basis thermal loads, operational handling loads, and design basis live loads. The criteria associated with these loads are presented in Table 2.2. The staff has reviewed these criteria and with the following exceptions, considers them to be acceptable.

1. Failure to comment that dead load should be, or was, increased by 5% over the estimated value, as stated in ANSI/ANS 57.9-84 section 6.17.1.1, and applicable to both concrete and steel design. [NOTE: Analytical results suggest that design margins exceeded this value and thus it does not result in a safety concern.]

2. Failure to comment that the live load should be, or was, varied between 0% and 100% of estimated load to simulate the most adverse

conditions for the structure. [NOTE: The discussions of actual analyses that indicate that the worst case loading conditions were assumed and/or the design margin of the actual design cause this not to be a safety concern.]

Off-normal events that can be expected to occur on a moderate frequency were postulated by NUTECH. They included: jammed DSC during transfer, off-normal temperatures (-40^{o}F to 125^{o}F), and off-normal pressurization of the DSC. The criteria associated with these conditions are shown in Table 2.3. The staff has reviewed these criteria and considers them to be acceptable with the following exceptions.

1. The design criteria (Table 3.2-1 of the TR) for the DSC support assembly off-normal case include the dead load of the support assembly (about 5000 lbs.) and handling loads due to the jammed DSC, but not the dead load of the DSC itself. In the actual analysis (TR section 8.1.2.1.B) a vertical load corresponding to the DSC dead weight was used. As a result the omission of the dead load from the table presenting the criteria is not a safety problem.

2. An increase in allowable shear stress of 50% was used for the accident condition criteria for the DSC support assembly. This exceeds the absolute 40% maximum increase allowed by ANSI/ANS 57.9-84 section 6.17.3.2.1 for steel design. The allowable shear stress used was determined by factoring a tensile yield stress based on an elevated temperature. This temperature (600^{o}F) would not be approached except under an accident condition of blockage of air inlets and outlets (TR section 8.2.7.2). As the critical stress level is produced by seismic forces the allowable is, in effect, based on the simultaneous occurrence of two "accidents" (a 48-hour vent blockage in 125^{o}F ambient air, and an earthquake). NUTECH used an extreme off-normal (less than "accident") temperature for determining the tensile yield stress. From that, they determined the allowable shear stress in a non-thermal accident (i.e., $0.4xF_{y}x1.4$) which provides an allowable shear stress higher than that used. As a result of the overly conservative derivation, the allowable shear stress is not considered to be a safety problem.

As stated by 10 CFR 72.122, those structures of an ISFSI that are important to safety must be able to withstand the effects of accident conditions resulting from extreme environmental conditions, natural phenomena and postulated accidents. The extreme environmental and natural phenomena conditions include: 1) tornado winds and tornado missiles, 2) flood, 3) earthquakes, and 4) lightning. The accident conditions include: 1) loss of HSM air outlet shielding blocks, 2) blockage of the HSM air inlets and outlets, 3) accidental internal pressure in the DSC, 4) postulated DSC leakage, and 5) a postulated drop of the DSC (a drop distance of 80 inches while in the transfer cask) resulting in a 75g deceleration if dropped in the vertical or the horizontal orientations, and a 25g deceleration if dropped on the corner with subsequent slapdown. The bases associated with each of these load conditions are discussed below and are summarized in Table 2.4. The staff has reviewed these criteria and considers them to be acceptable as defined in Table 2.4 with the exceptions discussed below.

Three of the exceptions noted in Section 2.1 have been discussed. Two relate to the methodology for combining loads for dead weight and live load for the HSM. A third relates to the derivation of the shear stress limit for the DSC support structure. See previous discussion. The fourth exception (noted in Section 2.1) to the criteria used for the designs of the DSC and its internal basket assembly is the value proposed for damping of the DSC during the drop accident. The selection of 10% as the value for critical damping for the accident drop case deviates from guidance provided by Regulatory Guide 1.61 (Reference 7). The Regulatory Guide suggests that a damping value of 4% for welded steel structures and 7% for bolted steel structures be used for calculating loads in seismically loaded structures in nuclear power plants. The DSC is a completely welded structure and the cask is welded except for the top lid which is bolted. Thus a conservative damping value based on Regulatory Guide 1.61 would be 4%. The NRC evaluated this deviation from the Regulatory Guide and determined that 7% is acceptable based on several sources in the open literature as well as several additional technical considerations. References 8 and 9 indicate that welded steel structures stressed to levels at or just below the yield point of the material have critical damping values of 5%, and if the yield point of the material is exceeded, as NUTECH predicts in the event of a drop, then 7-10% damping can be expected. A more recent study (Reference 10) also shows a strong correlation of increased damping as the stress levels increase from linear elastic to stress levels in the plastic region for structures with inherently ductile materials.

Damping associated with impact is somewhat different than damping associated with seismic events. Components in nuclear power plants subjected to a seismic event may be suspended so that they are free to vibrate at their natural frequencies. Portions of a structure (such as a cask) which are located on the direct load path during a drop impact event are likely to be critically damped, i.e., 100% (Reference 11). At the point of contact, freedom of movement of the dropped object is reduced because of the high compressive forces between the object and the target. At locations other than the direct load path, vibration of the object will not be critically damped. At the impact areas for both the DSC and the TC the local stress levels exceed yield stresses, but remain below the allowable stress level for Service Level D (see Reference 12, ASME Boiler and Pressure Vessel Code, Section III). Thus, at the contact area, stress levels will exceed yield and a higher damping level will exist.

Based on the above references and observations, the NRC staff accepts 7% as a conservative damping coefficient for the drop accident case for both the DSC and the TC. The fact that NUTECH used 10% does not pose a safety concern because the NRC staff calculated the DSC acceleration levels associated with the dynamic load factors for 7% damping and found that the acceleration levels used by NUTECH were equal to or greater than those accelerations determined by the NRC staff.

The HSM and the TC are designed to withstand DBT and tornado generated missiles. The transfer cask resistance to DBT and the potential safety hazard of a tornado generated missile was evaluated. The DSC was not designed or evaluated for DBT or tornado generated missiles as it would be continually housed in either the TC or the HSM when outside of the fuel pool building, and both the TC and HSM were shown to provide satisfactory shielding of the DSC. Safety of the fuel rods and the containers while in the fuel pool building is outside the scope of the TR.

The design and/or safety evaluation criteria used for DBT and tornado generated missiles as described in the TR are considered acceptable. [NOTE: Adverse effects might result from overturning of the TC with contained DSC while on its transporter, which would be bracketed by the separately analyzed drop scenario. Another accident scenario analyzed was the possible puncture of the neutron shield water reservoir around the TC. The puncture and drainage of the TC neutron shield under detectable conditions (such as following a tornado) are not considered significant safety problems.]

The DSC and HSM design criteria include flood parameters of 15 fps velocity and 50-foot flood height. These flood conditions are assumed to exist during the normal situation of the DSC in storage in the HSM. The TR indicates that plant procedures are expected to be sufficient to avoid need to design or assess the case where the DSC is within the transfer cask during a flood. These are considered to be satisfactory assumptions for the TR. A site-specific license application, therefore, will be required to either validate these assumptions or provide further analysis if more severe flood parameters may occur.

A horizontal acceleration of 0.25g was established as a basis for seismic design in Section 3.2.3 of the TR. This selected acceleration is acceptable to the staff as a representative value for use in the TR. This acceptance recognizes that a site-specific evaluation will be required to establish geological and seismological requirements for each site-specific ISFSI application, as required by 10 CFR 72.102.

The vertical acceleration of 0.17g established in Section 3.2.3 of the TR is acceptable to the staff since this value is consistent with the Regulatory Guide 1.60 requirement that the vertical acceleration be 2/3 of the horizontal acceleration.

2.5 PROTECTION AGAINST FIRE AND EXPLOSION

The NUTECH TR does not specifically address protection of the NUHOMS system from potential fires or explosions. Instead, it relegates such analyses to a site-specific situation.

There are no flammable or explosive materials used in the construction and operation of the DSC or the HSM. Nevertheless, site-specific conditions can exist with the potential for fire and explosions in or around the HSM and DSC. Therefore, any application of the NUHOMS system to a specific site must analyze the consequences of fires and explosions and provide for protective and mitigative measures, as deemed necessary. NUTECH stated that the DSC has been calculated to withstand an external pressure of 21.7 psi. This external pressure is that which would result from immersion in fifty feet of water, a postulated accident considered in the flood analysis, Section 8.2.4 of the TR.

2.6 CONFINEMENT BARRIERS AND SYSTEMS

Subpart F of 10 CFR 72 provides the general design criteria and within that subpart, 72.122(h) deals with confinement barriers and systems. For the NUHOMS-24P system, 72.122(h)(1) is relevant to the dry storage of spent fuel as follows: "The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate" (Reference 3).

The TR takes the position that the inert helium atmosphere in the DSC will not leak out and that the fuel cladding temperature will be held below levels at which damage could occur. The staff accepts that the helium atmosphere will be maintained during storage. The staff then analyzed the impact of long-term storage on the behavior of spent fuel, using a diffusion controlled cavity growth (DCCG) mechanism as the basis for this calculation since it appears that this damage mechanism is the only one applicable to these storage conditions. Under the influence of stress and temperature, this damage mechanism progresses by the nucleation and growth of cavities along grain boundaries.

The staff also evaluated the impact of the cask dry-out procedure and off-normal operation on the behavior of the spent fuel. This evaluation was based on the concerns for the potential oxidation and creep of the fuel, respectively. All of these concerns and the evaluation results are discussed in Section 5, Confinement Barriers and Systems.

The analyses are predicated on the knowledge and control of the character of the spent fuel loaded into the DSC, particularly the quantity, specific power, and age of the fuel assemblies, and the heat dissipation properties of the system. The thermal evaluation is addressed in Section 4.

2.7 INSTRUMENTATION AND CONTROL SYSTEMS

Instrumentation and control systems are addressed in Part 72.126 of 10 CFR which requires the provision of (1) protection systems for radiation exposure control; (2) radiological alarm systems; (3) systems for monitoring effluents and direct radiation; and (4) systems to control the release of radioactive materials in effluents.

The TR takes the position that because of the passive nature of the NUHOMS-24P system, no instrumentation is necessary. Since the DSC was conservatively designed to perform its containment function during all worst-case conditions, as can be shown by analysis, there is no need to monitor the internal cavity of the DSC for temperature or pressure during normal operations. The staff concurs with the position that instrumentation and control systems are not required for the NUHOMS-24P.

2.8 CRITERIA FOR NUCLEAR CRITICALITY SAFETY

The requirement stated in 10 CFR 72.124 is that spent fuel handling, transfer and storage systems be designed to be maintained in a subcritical configuration and to ensure that before a nuclear criticality accident is possible, at least two unlikely independent, concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design safety margins must reflect design uncertainties including uncertainties in handling, transfer, and storage conditions, data and methods used in calculations, and adverse accident environments. Section 72.124 also requires that the design be based on either favorable geometry or permanently fixed neutron absorbing materials. A criticality monitoring system is required in each area where special nuclear fuel is handled, used, or stored, except where the material is packaged in its stored configuration. Section 3.3.4 of the NUHOMS TR addresses nuclear criticality safety, and Section 7 of this report reviews the criticality analysis.

The acceptance criteria for nuclear criticality safety established in the present review was a 95% probability/95% confidence effective multiplication factor of 0.95 for storage and 0.98 for loading. The initial enrichment of unirradiated fuel assemblies was assumed in both cases. The maximum effective reactivity factor includes method and cross section biases, uncertainties in design parameters, and assumes optimal moderation conditions. Optimal moderation might occur if moderator boiling temperatures were reached; however, unirradiated fuel provides no moderator heating. The circumstances can be conceived for optimal moderation for irradiated fuel; however, the reactivity of the fuel would be significantly reduced.

2.9 CRITERIA FOR RADIOLOGICAL PROTECTION

Parts 72.24, 72.104(a), and 72.106(b) of 10 CFR 72 require the licensee to provide the means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR 20, for limiting the annual dose equivalent to any individual beyond the controlled area, and for meeting the objective of maintaining exposures as low as reasonably achievable (ALARA). Part 72.126 of 10 CFR requires the provision of (1) protection systems for radiation exposure control; (2) radiological alarm systems; (3) systems for monitoring effluents and direct radiation; and (4) systems to control the release of radioactive materials in effluents.

Part 20.101(a) of 10 CFR 20 states that any individual in a restricted area shall not receive in any period of one calendar quarter from radioactive material and other sources of radiation a total occupational dose in excess of 1.25 rems to the whole body. Part 20.101(b) states that, under certain conditions, the quarterly dose limit to the whole body is 3 rems in any calendar quarter. Guidance for ALARA considerations is also provided in NRC Regulatory Guides 8.8 and 8.10 (References 13 and 14).

The TR establishes shielding criteria for the NUHOMS module of an average external surface dose of less than 20 mrem/hr. In addition, criteria were established of 200 mrem/hr for the transfer cask side surfaces and 100 mrem/hr on the DSC top lead plug. The shielding capability of the system relies primarily upon the bulk concrete shielding of the NUHOMS module and the DSC top lead plug.

The radiological protection design features of the NUHOMS-24P are described in Sections 3 and 7 of the TR. These features consist of (1) radiation shielding provided by the transfer cask, DSC, and HSM; (2) radioactive material confinement within the DSC, specifically the integrity of the double seal welds; (3) prevention of external surface contamination; and (4) site access control. Access to the site of the NUHOMS-24P array, although not specifically addressed in the TR, would be restricted by a periphery fence to comply with 10 CFR 72.106 controlled area requirements. The details of the access control features are site-specific, and would be described in the applicant's site license application.

Radiation protection for on-site personnel is considered acceptable if it can be shown that the non-site-specific considerations (1) will maintain occupational radiation exposures at levels which are as low as reasonably achievable (ALARA), (2) are in compliance with appropriate guidance and/or regulations, and (3) will assure that the dose from associated activities to any individual does not exceed the limits of 10 CFR 20.

Off-site radiological protection features of the NUHOMS-24P system are deemed acceptable if it can be shown that design and operational considerations which are not site-specific result in off-site dose consequences which are (1) in compliance with 10 CFR 72.104(a) for normal operations and anticipated occurrences, (2) in compliance with 10 CFR 72.106(b) for design basis accidents, and (3) are as low as reasonably achievable.

Based on analyses presented in the TR, the staff concludes that the NUHOMS-24P system, if properly sited, meets the requirements for on-site and off-site radiological protection, including the incorporation of ALARA principles. Radiological alarm systems and systems for monitoring effluents are not required in the NUHOMS design because of the integrity of the double seal weld on the DSC.

2.10 CRITERIA FOR SPENT FUEL AND RADIOACTIVE WASTE STORAGE AND HANDLING

Pursuant to 10 CFR 72.128(a), the licensee is required to design the spent fuel and radioactive waste storage systems to ensure adequate safety under normal and accident conditions. These systems must be designed with

(1) a capability to test and monitor components important to safety; (2) suitable shielding for radiation protection under normal and accident conditions; (3) confinement structures and systems; (4) a heat-removal capability having testability and reliability consistent with its importance to safety; and (5) means to minimize the quantity of radioactive waste generated. Part 72.128(b) further requires that radioactive waste treatment facilities be provided for the packing of site-generated low-level wastes in a form suitable for storage on-site awaiting transfer to disposal sites.

Criteria covering items (1) through (4) above have been addressed in this SER in the preceding subsections of this Section. The TR does not specifically address the issue of minimization of radioactive waste generation. Solid wastes will likely be limited to small amounts of sampling or decontamination materials such as rags or swabs, while liquid wastes will consist mainly of small amounts of liquid resulting from decontamination activities. Contaminated water from the spent fuel pool and potentially contaminated air and helium from the DSC, which are generated during cask loading operations, will be treated using plant-specific systems and procedures. No radioactive wastes requiring treatment are generated during the storage period under either normal operating or accident conditions.

The staff agrees that the design of the NUHOMS-24P provides for minimal generation of radioactive wastes, and that any wastes that are generated would be easily accommodated by existing plant-specific treatment or storage facilities.

2.11 CRITERIA FOR DECOMMISSIONING

Part 72.130 of 10 CFR provides criteria for decommissioning. It requires that considerations for decommissioning be included in the design of an ISFSI, and that provisions be incorporated to (1) decontaminate structures and equipment; (2) minimize the quantity of waste and contaminated equipment; and (3) facilitate removal of radioactive waste and contaminated materials at the time of decommissioning.

Part 72.30 of 10 CFR defines the need for a decommissioning plan, which includes financing. Such a plan, however, is only appropriate to a site-specific situation, and 10 CFR 72.30 is therefore considered not applicable to this review.

The NUHOMS-24P TR claims that the DSC is designed to interface with a transportation system capable of transporting intact canistered assemblies to either a monitored retrievable storage (MRS) facility or a geologic repository. The TR does not identify any transportation system that could accept the DSC, hence the staff cannot comment about this claim. However, if the fuel must be removed from the DSC, the internal surface of the DSC will be contaminated and may be slightly activated. After the interior is cleaned to remove loose contamination, the DSC can be disposed of as low-level waste, or possibly even as scrap.

The current design of the NUHOMS system is based on the intended eventual disposal of each DSC following fuel removal. However, it is also possible that the DSC shell/basket assembly could be reused. Such an alternative would be dependent on economic and regulatory conditions at the time of fuel removal.

To facilitate decommissioning of the HSM, the design should be such that:

1. There is no credible chain of events that would result in widespread contamination outside of the DSC; and

2. Contamination of the external surfaces of the DSC must be maintained below applicable surface contamination limits. The TR uses the following surface removable contamination limits as a guide:

   Beta-gamma emitters: 10^{-4} uCi/cm^2
   Alpha emitters: 10^{-5} uCi/cm^2

A detailed decommissioning plan, which would consider such factors as decommissioning options available, likely further use of the site, environmental impact, and available waste transportation and disposal capabilities, would be developed on a site-specific basis.

The staff concludes that adequate attention has been paid to decommissioning in the design of the NUHOMS-24P.

3.0 STRUCTURAL EVALUATION

3.1 SUMMARY AND CONCLUSIONS

This section presents the results of the NRC staff evaluations of the structural analyses and designs included in the TR. The evaluations are made against the accepted criteria, as evaluated in Section 2 of this SER. Where the criteria were not acceptable the staff examined the design results and compared those against acceptable codes, standards, or engineering practice, as appropriate. Thus, the actual design might be acceptable even if the stated design criteria are not acceptable.

The system descriptions presented in the TR are reviewed at two levels to determine: 1) whether the designs and descriptions are in themselves acceptable, and 2) the extent that the system description and analyses satisfy the requirements for a potential site-specific license application, and might thus be incorporated by reference or repetition in the application documentation.

This review includes an evaluation of all structural design criteria, analysis methodologies, material specifications, allowable stress levels and structural analyses. The staff has reviewed the structural design of the NUHOMS system proposed by NUTECH and confirms that the design is in compliance with 10 CFR 72.122 with the exceptions outlined below.

The staff has reviewed the structural analysis methodologies used in evaluating the structures and found them to be acceptable with the following exceptions:

1. Some discrepancies between the TR statements and the actual loads used in the HSM dead load analysis (see Table 8.1-9 and Section 8.1.1.5A in the TR) exist. The staff evaluated these discrepancies and concluded that the results of the analysis are satisfactory.

2. The method of accounting for concrete creep and shrinkage for the HSM was not documented in the TR; however, the staff reviewed the method used and determined that the methodology is appropriate.

3. The methodology used by NUTECH to calculate local DSC shell bending was not considered conservative by the NRC staff. A different model was selected by the staff and evaluated. The staff concludes that even with the more conservative method, the DSC design is adequate.

4. The type of finite element used by NUTECH to model the DSC and TC is a two-dimensional isoparametric element that only calculates membrane stress unless two elements are used together to model the shell thickness. Since NUTECH only used one element to model the thickness no bending stresses were calculated. However, the ASME Code requires that bending stresses as well as membrane stresses be evaluated for Class 1 components. The staff evaluated the results using alternative methods and concurs that the resulting design is satisfactory despite the flaw in the methodology.

The staff has reviewed the material specifications and allowable stress levels used in evaluating the system and confirmed that these data are in compliance with 10 CFR 72.122 with the following exception:

The material allowable stresses were evaluated for 400°F by NUTECH for the DSC (see Table 8.2-9c of the TR). The staff notes that the maximum temperature experienced by the DSC is 513°F for the Service Levels C and D. Even though the Code does not require an evaluation of thermal stresses for Service Levels C and D, the appropriate material allowable stresses must be used. The staff made this adjustment to lower allowables and concludes that the design for the DSC is satisfactory.

The staff has reviewed the structural analyses and designs presented in the TR for satisfaction of the requirements of 10 CFR 72.122 and finds that they are acceptable with the following exceptions:

1. The thicknesses of the top and bottom cover plates as modeled by the computer analyses <u>do not</u> agree with the thicknesses of the plates in the design drawings. The staff used a ratio of the squares of the thicknesses to multiply the computer listings by in order to estimate the stresses. All stress levels were found satisfactory after this adjustment.

2. Although the TR did not provide any drawings or analysis of the DSC seismic restraint, NUTECH did provide responses to the staff's request for

additional information on this restraint. The staff performed an independent check to confirm that adequate shear area is provided by the design. NUTECH should include details of this design in the revision of their TR.

The exceptions noted by the staff in the areas of methodology, material specifications and allowable stresses, as well as the analysis for all systems important to safety, do not result in safety concerns. They are noted in this SER as a matter of record.

The staff reviewed the structural analyses and designs presented in the TR to determine the extent that the TR would satisfy requirements for a site-specific license application SAR as expressed in 10 CFR 72.24. The staff finds that the description and design of the HSM and the integral DSC support assembly, the DSC, and the TC are satisfactory for incorporation in a SAR by reference for the following technical information requirements, to the extent concerned with the structural design, with the exception of site-specific considerations (such as HSM foundation and validation of maximum accident condition parameters):

1. Description and discussion, per 10 CFR 72.24(b).

2. Design, per 10 CFR 72.24(c).

3. Impact on public health and safety, per 10 CFR 72.24(d), but only to the extent of protecting against accidental rupture and/or exposure of nuclear reactor fuel, and not as relates to radiological or other hazards (see other corresponding SER sections).

4. Selection of license conditions and specifications as relate to the structural design, per 10 CFR 72.24(g).

5. Proof of functional adequacy or reliability, per 10 CFR 72.24(i) (requirements related to design or materials).

Section 4.2.1 of the TR lists codes and standards applicable to the fabrication and construction of the components, equipment, and structures identified through the TR. The staff has reviewed these and considers them acceptable with the following exceptions:

1. The reference to the AISC Code Eighth Edition should be to the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," effective November 1, 1978, which is contained in the AISC Steel Construction Manual, Eighth Edition (Reference 15).

2. The ACI 318-83 (Reference 16) code is listed as the code of construction for the HSM in Section 4.2.1 of the TR, and is specified in Appendix E, drawing DUK-03-1000, sheet 1 for construction of the reinforced concrete. This code is not endorsed by Reg. Guide 3.60 or ANSI/ANS 57.9-1984, which cites the ACI 349-80 (Reference 5) code as suitable for concrete design. However, the NRC staff performed an extensive comparison of ACI 318-83 with ACI 349-80 and concluded that <u>for the purposes of construction only</u> the impact on the HSM would be negligible. ACI 349-80 has more stringent quality assurance requirements than ACI 318-83, but the assessment of the staff is that there will be no safety consequences as a result of using ACI 318-83 as the code of construction.

3. Reference to non-specific codes of construction for transfer equipment are not considered appropriate since the designs for such equipment are not included in the TR. The listing of codes for other than the HSM (with DSC support assembly), DSC, or TC does not provide a suitable reference. The appropriate codes should be identified in the site-specific SAR.

3.2 DESCRIPTION OF REVIEW

This section evaluates the structural capability of the HSM, DSC, and transfer cask to withstand loads due to normal operating conditions, off-normal conditions, accident conditions, environmental conditions, and natural phenomena. The review addresses the assumed loads, the material properties, and the allowable stress limits. The review provides an evaluation of the structural analysis in the TR for each of the components and systems important to safety.

The SER review only addresses forces and conditions external to the fuel pool building. Determination of the adequacy of the design for normal, off-normal, and accident conditions within the fuel pool building at a specific site will be addressed as part of the 10 CFR 50 safety review.

3.2.1 Applicable Parts of 10 CFR 72

The parts of 10 CFR 72 that are applicable to the structural evaluation are as follows: 72.122(a), which deals with quality standards; 72.122(b), which requires that structures important to safety be protected against environmental conditions and natural phenomena, as well as appropriate combinations of effects including accident conditions; 72.122(c), which requires protection against fires and explosions; 72.122(f), which requires design to permit inspection, maintenance, and testing; and 72.122(h), which requires protection of the fuel cladding against degradation and gross rupture.

The structural descriptions, analyses, and designs in the TR were reviewed for potentially meeting the requirements of 10 CFR 72.24 for a site-specific license application. Although there is no stated scope for an ISFSI TR, the utility of the TR is in providing already approved elements of site-specific license application documentation, in which it may be incorporated by specific reference (per 10 CFR 72.18) or repetition. The requirements of 10 CFR 72.24, which might be met by the structural descriptions, analyses, and designs of a TR, are section 72.24: "(b) description and discussion of structures;" "(c) design, including criteria, bases, special information, and codes and standards;" "(d) analysis and evaluation of the design and performance;" "(g) subjects for license conditions and technical specifications (per 10 CFR 72.44);" and "(i) need for demonstration of functional adequacy or reliability."

3.2.2 Review Procedure

The TR was reviewed to determine compliance with the applicable parts of 10 CFR 72 outlined above. The systems comprising the NUHOMS ISFSI including the HSM, DSC support assembly, DSC, and transfer cask were considered first as systems and secondly as individual parts making up complete systems. Normal operating conditions, off-normal operating conditions, extreme natural phenomena, and accident conditions and resulting loading combinations that were analyzed by NUTECH were reviewed for completeness by the staff.

3.2.2.1 Design Descriptions

A brief description of the NUHOMS ISFSI was given in the first section of this report. A more detailed description of the design in this section highlights aspects of the design that are important to the structural evaluation.

The safe storage of irradiated fuel is provided by the DSC and HSM. The DSC provides confinement of radioactive material. The HSM and DSC provide shielding for biological protection. The DSC and TC provide shielding during handling and transportation of the fuel. Both the DSC-TC and DSC-HSM combinations must provide for adequate steady-state heat transfer.

The HSM is a reinforced concrete structure that provides projectile impact and weather protection for the DSC and serves as the primary biological shield for the irradiated fuel during storage.

The HSM is designed to be constructed of 5000 psi (minimum specified compressive strength) normal weight (145 pcf minimum density) concrete with Type II Portland cement meeting the requirements of ASTM C150. The aggregate is to meet the specifications of the ASTM C33. The reinforcing steel is #9 bars ASTM A615 Grade 60 spaced 6" on centers each way each face, top, sides, front, back, and foundation.

The HSM wall thicknesses were designed to meet shielding requirements and was checked against structural criteria. The walls protect the DSC against tornado generated missiles, which effectively bounds reasonable impact accidents, as well as other environmental conditions, natural phenomena, and accidents.

The structural properties of the concrete when subjected to the elevated surface temperature for the long term are discussed in Section 3.3.2 of this report.

The HSM is designed to accommodate the transfer of the DSC from and back into the TC. This is provided by an oversize inset steel collar forming the opening of the HSM storage vault. The collar includes an access opening sleeve into which the top of the horizontal TC may be slid. The adjacent external face of the HSM includes connection points to provide a tensile reaction against the piston mechanism used to drive the DSC out of TC and into the HSM.

The DSC end plugs and the canister shell provide confinement and radiation shielding. The bottom end sandwiches lead between an outer plate and an inner plate of Type 304 stainless steel. The bottom end plug also includes a grapple attachment assembly for insertion and removal from the HSM. The top plug is formed by two covers, separately welded to the DSC shell. The inner cover and the outer cover are manufactured from Type 304 stainless steel with lead placed between these cover plates. The DSC ends serve as pressure boundaries. The welds are multiple pass and are to be tested either ultrasonically or by multilevel dye penetrant examination. In addition, a helium leak test will be performed after welding the inner top cover in place.

The DSC shell will be assembled using longitudinal and circumferential full penetration butt welds. These welds are to be fully radiographed and inspected according to the requirements of the ASME B&PV Code Section III, Division 1, Subsection NB (Reference 12). The material is 0.625 inch thick 304 stainless steel.

The canister encloses a basket assembly, which can house twenty-four irradiated fuel assemblies. The basket consists of eight spacer discs of Type 304 stainless steel that are fixed axially by four 3.0 inch diameter 304 stainless steel rods running the length of the canister. There are twenty-four square section guide sleeves of Type 304 stainless steel that house the spent assemblies. The primary structural function of the spacer discs and axial support rods is to maintain dimensional stability for the guide sleeves that house the spent fuel in the event of a vertical or horizontal drop. The axial location of the spacer discs corresponds to the grid spacing of the specific fuel to be stored.

The DSC rests on a fabricated support rail assembly that rests on brackets attached by anchor bolts cast in the interior walls of the HSM. The support rails are also welded to the cast-in-place sleeve forming the HSM front opening. Thermal expansion of the support rails and crossbeams is allowed by using slotted bolt holes. Corrosion of the structural carbon steel will be retarded by either zinc paint, galvanizing, or hard plating.

During loading of the DSC into the HSM, frictional loads between the DSC and the support rails in the HSM will be reduced by the use of a dry film lubricant applied to both sliding surfaces. The particular product selected by NUTECH is Everlube 823, which was designed for radiation service.

The DSC is prevented from sliding longitudinally along the rail during a seismic event by seismic restraints. Permanent restraints are welded to the rails at their inside ends and a removable restraint is attached to the access sleeve at the HSM front opening after placement of the DSC.

A specially designed TC is a major component of the NUHOMS-24P system. The DSC is placed in the TC prior to loading spent fuel rods into it and remains in the TC until it is pushed from the TC into the HSM. The DSC is loaded, sealed, drained, and the TC is drained prior to departure from the fuel rod storage pool enclosure. The DSC will be pulled from the HSM back into the TC for removal and will remain in the TC at least until the fuel rods are removed.

The TC provides radiation shielding and a protective enclosure for the DSC. During transportation from the fuel pool building to the HSM site, the TC provides DBT projectile impact protection. The TC does not provide a pressure boundary in addition to the DSC liner.

The TC design consists of: (1) a bolted-on vented top cover plate, (2) a lower water-tight bolted-on cover sized to permit the grapple of the hydraulic ram to enter and act on the DSC, (3) an annulus between the TC and DSC that can be filled with fluid and drained, and (4) a neutron shielding fluid-filled jacket with external expansion tank.

The TC cylindrical wall section is comprised of a 1/2 inch thick type 304 stainless steel inner shell, 3.5 inches of cast in place lead, a 1.5 inch thick ASME SA-516, Grade 70 structural steel shell, a 3.0 inch radial width fluid chamber, and a .13 inch thick type 304 stainless steel outer neutron shield (tank) shell. The neutron shield is to be filled with a water-antifreeze mixture.

The lower end of the TC forms a radiation shield, provides for fluid retention, and has a bolted-on cover over the access port. The end is formed of type 304 stainless steel 2 inch thick inner and a 3/16 inch thick outer cover plates and approximately 2 3/4 inches of solid neutron shield (Borosilicone (reg) No. 237) in between. The upper TC cover is formed of ASTM A516, Grade 70 steel 3 inch thick inner and 1/4 inch outer cover plates and an intermediate 2 inch thick solid neutron shield (also Borosilicone).

The TC is assembled by welding the concentric cylindrical walls and the lower end to heavy forged ring assemblies made of ASTM SA-182, Type F304N steel at the top and bottom of the cask. Lead is poured to fill the annulus formed by the inner two shells. Hydraulic fittings, tubing, and an external expansion tank permit use of the annulus formed by the middle and outer shells as a water-filled neutron shield.

The TC has two pairs of trunnions. The upper pair is used to lift the cask vertically and to support it while in a horizontal orientation. The lower pair is used for support on the transfer trailer and serves as pivots for rotating between vertical and horizontal orientation (and vice versa) when the TC is on the transfer trailer.

The upper end of the TC fits within a receiving collar at the opening of the HSM to provide continuous radiation shielding during DSC transfer into and out of the HSM from/into the TC.

The TC does not constitute a radioactive material confinement boundary, although it is essential for radiation shielding. As a result, the TC is not required to meet as stringent design criteria as the DSC. The TC is designed to meet the requirements of subsection NC, for Class 2 components, of Reference 17.

3.2.2.2 Acceptance Criteria

The structural integrity of the NUHOMS HSM, DSC, DSC support assembly, and TC are judged adequate if it can be demonstrated that the stresses induced by the loads noted below in Section 3.3.1 are lower than the allowable stress limits for the components important to safety and that all other material properties are consistent with applicable code requirements. The allowable stress limits are documented in the TR (Reference 1, Section 3.2.5, Tables 3.2-4, 3.2-5, 3.2-6, 3.2-7, 3.2-8, and 3.2-9).

3.2.2.3 Review Method

The method of review used to assure that the TR was in compliance with 10 CFR 72 involved several steps. The TR was reviewed first for completeness, to ensure that all areas specified by 10 CFR 72 Subpart F were addressed, and that the standard format, content, and design guidance specified by Regulatory Guides 3.48 (Reference 2) and 3.60 (Reference 4) were followed to the extent applicable for a non site-specific TR to be used in conjunction with subsequent site-specific license applications. Sources cited by the TR were reviewed to determine applicability to the design of the NUHOMS system. Section 3 of the TR, which sets out the design criteria, was examined critically for appropriateness. Assumptions stated in the TR were assessed with respect to those suggested by the professional societies which guide design practice for pressure vessels and reinforced concrete structures for nuclear safety related items. The societies and their respective codes are: the American Society of Mechanical Engineers (ASME Boiler and Pressure Vessel Code for Nuclear Power Plant Components, Section III, Division 1, Subsection NB, Class 1 Components, 1983) for the DSC and the American Concrete Institute (ACI 349-80, ACI 349R-80 and 1984 Supplement to Code Requirements for Nuclear Safety Related Concrete Structures) for the HSM. The design of the DSC support system was compared to requirements of the Manual of Steel Construction published by the American Institute of Steel Construction. The design of the TC was reviewed against the requirements of the ASME B&PV Code (Section III, Subsection NC, Class 2 Components, 1983). The design of the lifting trunnions of the TC was reviewed against the requirements of ANSI N14.6-1986 (Reference 18).

Secondly, Section 8 of the TR, which covers the analysis of the design events, was reviewed in detail. This included verifying all calculations that could be executed without resorting to computer models. The finite element computer models performed by NUTECH were verified for accuracy by examining the input and output printouts for all ANSYS and STRUDL (References 19 and 20) computer runs that were referenced in the TR, and NUTECH post-processor codes. All results that were included in the TR (Tables 8.1-7, 8.1-7a, 8.1-8, 8.1-9, 8.1-9a, 8.1-10, 8.1-10a, 8.1-10b, 8.2-3, 8.2-7, 8.2-7a, 8.2-9, 8.2-9a, 8.2-9b, 8.2-10, 8.2-11, 8.2-12, 8.2-13, 8.2-14, and 8.2-15 (Reference 1)) were either verified by hand calculations or by examining the computer printouts. No independent computer analysis was performed.

3.2.2.4 Key Assumptions

Assumptions made by staff reviewers in verifying NUTECH's models are discussed on a case-by-case basis in the following sections.

3.3 DISCUSSION OF RESULTS

The following evaluation covers loads, materials, stress intensity limits, and structural analyses results.

3.3.1 Loads

The loads specified in the TR for use in designing the NUHOMS system are described in this section together with comments by the staff regarding their acceptability. Loads are described for normal operating, off-normal operating and accident conditions. The staff evaluation of the design criteria sources was summarized earlier in Table 2.1.

3.3.1.1 Normal Operating Conditions

Section 8.1.1 of the TR defines the normal operating conditions of the NUHOMS system. The normal operating loads of the NUHOMS system are dead weight loads, design basis internal pressure loads, design basis thermal loads, operational handling loads, and design basis live loads. The staff evaluation of criteria associated with these loads is summarized in Table

2.2 for the system components, and includes sources and the results of the staff review.

3.3.1.2 Off-Normal Conditions

Section 8.1.2 of the TR defines the off-normal events. These are events that are expected to occur on a moderate frequency. The events included are: a jammed DSC during HSM loading or unloading, and extreme ambient temperatures ($-40^{o}F$ and $125^{o}F$). The staff evaluation of criteria associated with each of these loads is summarized in Table 2.3, and includes sources and the results of the staff review.

3.3.1.3 Accident Conditions

Section 8.2 of the TR defines the accident conditions resulting from extreme environmental conditions, natural phenomena conditions, and postulated accidents, which include the following conditions:

1. Loss of HSM air outlet shielding blocks.
2. Tornado winds and tornado generated missiles.
3. Design basis earthquake.
4. Design basis flood.
5. Accidental TC drop with loss of neutron shield.
6. Lightning.
7. Debris blockage of HSM ventilation air inlets and outlets.
8. Postulated DSC leakage.
9. Pressurization due to fuel cladding failure within the DSC.

The staff evaluation of design criteria associated with the accident loads is summarized in Table 2.4, and includes sources and the results of the staff review.

3.3.2 Materials

The mechanical properties of all materials used in the fabrication of components important to safety are listed in Section 8.1.1, Table 8.1-2 of the TR. The source identified in these tables for properties of steel is the ASME Boiler and Pressure Vessel Code, Section III-1 Appendices. This source is an acceptable standard and is in compliance with the quality requirements of 10 CFR 72 Part F.

The source identified in TR Table 8.1-2 for the mechanical properties of concrete and reinforcing steel is the <u>Handbook of Concrete Engineering</u> (Reference 21), a document that is not considered to constitute a standard meeting the quality requirements of 10 CFR 72.122. NUTECH supplements these data by a review of concrete behavior under sustained elevated temperatures that is presented in Appendix D to the TR. The Appendix D data are substantiated by a number of references, most of which are publications of the American Concrete Institute and the Portland Cement Association (PCA). Both of these organizations publish recognized standards consistent with the quality requirements of 10 CFR 72.122(a).

The temperature of some HSM concrete may exceed 150^{o}F under "normal conditions" (sustained ambient temperature up to 100^{o}F), over 200^{o}F under off-normal conditions (ambient temperature to 125^{o}F for 48 hours), and up to 395^{o}F for accident conditions (ambient temperature of 125^{o}F with all vents plugged for 48 hours, see TR Table 8.1-12). The analytical procedures by which these projected temperatures were calculated were reviewed, with appropriate amplification provided by direct informal contact, and were determined to be acceptable.

The ACI Code, ACI 349-85, used for the TR design criteria provides for limits on concrete temperatures as follows: general limit for concrete in structures of 150^{o}F, 200^{o}F for local areas for long terms periods, 350^{o}F for concrete for short term periods, and 650^{o}F for local areas (if due to steam or water jets) in an accident or other short time period. Section A.4.3 of the ACI Code indicates that higher temperatures may be allowed if tests are provided to evaluate the reduction in strength and this reduction is applied to the design allowables, <u>and</u> that evidence be provided that verifies that the increased temperatures do not cause deterioration of the concrete either with or without load.

The TR includes a review of concrete behavior under sustained elevated temperatures (TR Appendix D), and a strength reduction is applied (TR Table 8.1-2, $f_c' = 4.5$ ksi for 5 ksi concrete) and used in the comparison of calculated versus allowable moments (TR Table 8.2-10).

The NRC staff reviewed the submitted discussion of concrete at high temperatures, reduction of allowable stresses (based on ultimate compressive strength), extent of concrete affected, and the ACI Code provisions, and concludes that the HSM design is acceptable with regard to the projected temperatures.

The review also addressed the issue of need for evidence that the continued long term temperatures will not result in degradation of the concrete. It is noted that the HSM for the NUHOMS-7P design is projected to have higher operating temperatures than the HSM which is addressed by this SER and that a condition of the NRC approval of Reference 22 is that results of field tests on the concrete be submitted after that system is placed in use.

The NRC staff considers that because of the following results, there is no requirement for subsequent or further submission of test data on possible concrete degradation under elevated temperatures for the NUHOMS-24P system HSM: 1) the relatively small concrete areas affected by elevated (over $150^{o}F$) temperatures, 2) the small magnitude of the elevated temperature, and 3) the empirical data on concrete behavior on-hand. This is subject to further NRC review if data or analyses made available to the NRC after final action on this TR suggest that a problem could exist.

The sources identified in TR Table 8.1-2 for the structural properties of lead are not recognized standards consistent with the quality requirements of 10 CFR 72.122(a). However, the material strength properties for lead shown in the TR were used in a conservative way that would not invalidate the analysis. No bending stiffness is assumed to be imparted by lead shielding plug because coupled nodes are used at the interface of lead and steel. This coupling of nodes permits only tension or compression and no shear forces to be transmitted through the lead. Thus the staff concludes that the way the data were used meets the intent of the quality requirements of 10 CFR 72.122(a) for material properties.

3.3.3 Stress Intensity Limits

The mechanical properties of the structural materials used in the design of the NUHOMS system are listed in Table 8.1-2 of the TR. These properties, including allowable stress intensities, are listed as a function of temperature for a variety of materials as described below.

The ASME Boiler and Pressure Vessel Code, Section III-1, was identified in Table 8.1-2 of the TR as the source for stress allowables for the two Type 304 stainless steels, the SA-516 carbon steel, the SA-564 and SA-533 alloy steels used for the DSC and TC shell and trunnions, Grades A36 and A325 steel and SA-193 alloy steel used for bolts. As discussed in Section 3.3.2 of this SER, these stress allowables are acceptable to the staff for use in the design of the NUHOMS system.

The Handbook for Concrete Engineering is identified in Table 8.1-2 of the TR as the source for stress allowables for concrete and reinforcing steel. As discussed in Section 3.3.2 of this SER, the staff does not concur in the use of this handbook as the authoritative source for concrete and reinforcing steel allowable stresses. However, the staff has reviewed the pertinent ACI and PCA data and concurs in the stress allowable values as presented in Table 8.1-2 of the TR for these materials.

3.3.4 Structural Analysis

3.3.4.1 HSM

A linear elastic structural analysis of a one foot section of a ten bay HSM was performed, using the STRUDL finite element computer program to determine the worst internal forces due to normal, off-normal, environmental and accident loading conditions. The combinations of the resultant forces were performed based on the requirements of ANSI-57.9-1984.

The staff reviewed and accepts the finite element modeling techniques of the HSM reinforced concrete structure. The following presents an overview of the evaluation.

3.3.4.1.1 Normal Operating Conditions

The HSM concrete structure was analyzed for the effects of dead loads (including effects of creep and shrinkage), live loads, and design basis temperature loads. In addition, the HSM door was analyzed for dead loads and normal handling loads. The HSM door supports are not designed to withstand dropping the door during closing or opening. The NRC staff reviewed this situation and determined that failure of the supports and possible drop of the door to the ground level would not constitute a safety problem. The staff did determine that the door and support design and method of locking (welding) are acceptable.

The HSM concrete structure analysis results are presented in TR Section 8.1.1.5 and Table 8.1-10 for all of the load combinations considered. These results are presented in the form of maximum moments and shears, which are compared with the ultimate moment and shear capacities of the respective structural section. The maximum moments and shears are developed for normal conditions in load combinations 1, 2, and 4 of TR Table 8.2-10 (included in this SER as Table 3.1). It is seen from this table that these maximum moments and shears are significantly lower than the associated capacities of the module.

<u>HSM dead and live load analyses</u>

The dead weight of HSM plus the weight of DSC and the DSC support assembly were considered. The actual concentrated loads (reactions) due to the dead weight of the DSC and the DSC support assembly were calculated and reported in Table 8.1-9 of the TR. However, these loads were not used in the finite element analysis as stated in TR Section 8.1.1.5.A. Instead, one-sixth of the total weight of the DSC was applied at the embedded support connection. Even though there is almost a factor of two difference in the actual load and the one-sixth estimated loads, the staff accepted this condition since only the properties of a one foot section of the HSM were used. The staff used the actual load to perform hand calculations to verify suitability of the design. The vendor performed a series of computer runs based on alternating loaded and empty HSMs to determine the worst set of internal HSM forces. The dead weight of the HSM was not considered as was stated in TR Section 8.1.1.5.A. The weight of the reinforced concrete was distributed throughout the members of the finite element model.

A live load of 200 psf was applied to the HSM roof to envelope all live loads. The resulting calculated maximum dead and live loads are tabulated in Table 8.1-10 of the TR. The staff reviewed the tabulated results as well

Table 3.1 SUMMARY REVIEW OF HSM STRUCTURAL DESIGN

HSM ENVELOPING LOAD COMBINATION RESULTS
(Table 8.2-10, Reference 1)

Load[1] Combination	Loading Combination Description	Maximum Loading V_{max} (k)	Maximum Loading M_{max} (k.in.)	Capacities[7] V_u (k)	Capacities[7] M_u (k.in.)	NRC Staff Comments
1 Norm	1.4D + 1.7L	4.8	233	43.8	3570	Acceptable
2 Norm	1.4D + 1.7L + 1.7H	4.8	233	43.8	3570	Acceptable
3 Accid	0.75(1.4D + 1.7L + 1.7H + 1.7T + 1.7W)	34.5	867	43.8	3570	Acceptable
4 Norm	0.75(1.4D + 1.7L + 1.7H + 1.7T)	34.5	867	43.8	3570	Acceptable
5 Accid	D + L + H + T + E	42.8	1220	43.8	3570	Acceptable
6 Accid	D + L + H + T + F	40.9	1100	43.8	3570	Acceptable
7 Off-normal & Accid	D + L + H + T_a	79.4	2800	104.[8]	3570	Acceptable

D = Dead Weight, E = Earthquake Load, F = Flood Induced Loads, H = Lateral Soil Pressure Load, L = Live Load, T = Normal Condition Thermal Load, T_a = Off-normal or Accident Condition Thermal Load, W = Tornado Wind and Missile Loads — Acceptable

Notes:

1. Load combinations are based on ANSI-57.9 as shown in Table 3.2-5 of the TR. — Acceptable
2. Maximum loads shown are irrespective of locations. — Acceptable
3. Thermal accident load (T_a) is based on 125°F ambient with air inlets and outlets blocked. — Acceptable
4. V_{max}, V_u, M_{max}, and M_u calculated per 12" section of HSM. — Acceptable
5. Results of load combinations 3 through 7 are based on cracked section. Others based on uncracked sections. — Acceptable
6. Material properties taken at 400°F for all load combinations. — Acceptable
7. V_u values based on allowable shear for deep flexural members, ACI 349-85, Section 11.8. [Note: ACI 349-80 identical.] — Acceptable
8. The shear capacity V_c is calculated using Equation 11-29 of ACI 349-85. [Note: ACI 349-80 identical.] — Acceptable

as the computer output, and concurs with the results as summarized in Table 3.1 of the SER.

Concrete creep and shrinkage analysis

The strains due to creep and shrinkage were calculated and then the total axial change in length were calculated for the HSM members. From a knowledge of the axial change in length, it was possible to calculate the axial forces. The TR does not document the method used for determining the column in Table 8.1-10 of the TR for "creep effects." The staff discussed this problem with the vendor's contractor and reviewed the computer inputs of the dead weight load case and found that the member forces due to creep and shrinkage were combined with the HSM dead weight load case where these effects increased the calculated forces. The staff accepted the approach and the load case.

HSM thermal analysis

Analyses were performed for ambient operating temperatures of 0^oF, 70^oF and 100^oF. The results of the heat transfer analysis of the 100^oF ambient with solar heat flux case indicated a maximum local temperature of 179^oF as shown in Figure 8.1-3a of the TR. The results are tabulated in TR Table 8.1-10. This localized temperature is within the allowable limits of ACI 349-85, Section A.4.1.

From a knowledge of the thermal gradient from the heat transfer analysis, the internal axial forces and bending moments were calculated and applied to the members of the HSM model to determine resultant forces and moments. The results are based on the uncracked section properties of concrete. The cracked section moment of inertia of the HSM members were calculated and then the ratio of cracked to uncracked moment of inertias were multiplied by the moment values taken from the computer output to determine the actual moment values of the cracked sections for the thermal load case. The approach was found acceptable. The staff reviewed the calculations, checked tabulated results in TR Table 8.1-10 against the computer output and found them acceptable.

The staff concludes that the structural design of the HSM for normal operating conditions is acceptable.

3.3.4.1.2 Off-Normal Conditions

The effect of increased temperatures due to high ambient temperatures (125^{o}F) was the only off-normal event considered in the TR to have an effect on the HSM. A thermal analysis was performed for this event as described in Section 4.3.2.2 of this SER. The results from this thermal analysis were used to perform a structural analysis on the HSM, as reported in Section 8.1.2.2 of the TR. Load combination 7 on Table 3.1 is applicable to this condition; however, the effects of elevated temperature due to the accident condition of vent blockage envelop this off-normal condition.

It is stated in Section 8.1.2.2 of the TR that this structural analysis considered the effect of a cracked cross-section when performing stiffness calculations. The staff agrees with the use of this cracked section analysis procedure since it is permitted as a special case in the ACI 349-80 Code. The staff has reviewed the procedure used to perform this cracked section analysis together with a review of the special conditions placed on its use by the ACI 349-80 code. The staff concludes that the results from this stress analysis are acceptable.

The only other off-normal event considered in the TR is the jamming of a canister against either the DSC support or HSM components during loading or unloading. Although the effect of this off-normal event on the DSC and the DSC support assembly within the HSM is considered in the TR, there is no actual analysis of this event for the HSM reinforced concrete structure. The staff has performed the analysis for the effect of this off-normal event on the HSM and concludes that the effect of this loading is negligible.

3.3.4.1.3 Accident Conditions

<u>HSM accident load analysis</u>

The postulated accident conditions for design specified by ANSI/ANS 57.9-1984 and other credible accidents which could affect the safety of the NUHOMS system were considered. The postulated accident conditions addressed in the TR include:

1. Loss of HSM air outlet shielding blocks.
2. Tornado winds and tornado generated missiles.
3. Design basis earthquake.
4. Flood.
5. Lightning.
6. Debris blockage of the HSM ventilating air inlets and outlets.

Loss of HSM air outlet shielding blocks

Air outlet shielding blocks are the only components of the NUHOMS system that are not designed to withstand tornado generated missiles. The vendor argues that there are no structural or thermal consequences of the loss of the shield blocks from the HSM. Increases to off-site radiological dose are discussed in Section 10 of this SER. The staff reviewed the drawings of the shielding blocks as shown in the TR and concurs that minimal structural damage to the HSM will result if the shield blocks are lost due to tornado generated missiles. There may be some difficulty in replacing the shield blocks if the method of attachment is via bolts and the bolts are damaged by the tornado. There may be much more difficulty encountered in replacing the shield blocks if they are cast in place. As the drawings supplied with the TR do not indicate the type of attachment, the question will need to be addressed in a site-specific application.

Tornado winds and tornado missiles

The analyses performed to evaluate the effect of tornado winds and tornado missiles is presented in Section 8.2.2 of the TR. The tornado wind analysis includes an evaluation of the possible overturning of an unanchored module and the computation of wind induced maximum moments and shears in an anchored module. Design basis for this postulated accident analysis was taken from NRC Regulatory Guide 1.76 and NUREG-0800 3.5.3 and 3.5.1.4 (References 23 and 24). The bending moments and shear forces at critical locations in the HSM member were calculated by performing a linear elastic finite element analysis. The resulting moments and shear forces are tabulated in TR Table 8.2-3.

The analysis performed to evaluate potential sliding and/or overturning of an unanchored module showed that a single unanchored module (or multiple modules) would not overturn or slide when subjected to the tornado wind event. Tie-downs or anchorage between the HSM and its foundation would still be used to reduce the potential risk of sliding. The NRC staff concurs with the analyses that anchors (e.g., monolithic construction, reinforcing steel) should be used.

The computation of wind induced maximum moments and shears was performed using selected critical sections and finite element analysis. The maximum moments and shears resulting from this analysis are presented in TR Table 8.2-3 and are included in the combination of loads analysis (combination 3 on Table 3.1). The NRC staff considers the analyses and results to be acceptable.

Analyses are included in TR Section 8.2.2.2 to evaluate the effect of a penetrating missile and a massive missile impact on the HSM. These analyses consider the impact of a 276 pound, 8-inch diameter blunt-nosed steel object on the HSM walls/roof and on the 3-inch thick steel door at the front of the HSM, and evaluate the impact of a 3,976 pound automobile on the side wall of the HSM. The outer walls and roof of the HSM are not less than 36" thick reinforced concrete. The staff performed hand calculations to determine equivalent static impact force and then the maximum bending moments. The staff reviewed and accepts the analyses of each postulated case and the results.

Design basis earthquake

Analyses performed to evaluate the effect of earthquakes on the HSM are presented in Section 8.2.3 of the TR. These analyses include an evaluation of the possible overturning and sliding of an unanchored single module and the computation of seismically induced maximum moments and shears in an anchored module.

A horizontal acceleration of 0.25 g and a vertical acceleration of 0.17 g were used as bases for seismic design. To evaluate the seismic response of the HSM, an equivalent static seismic analysis was performed. To determine the HSM fundamental frequency, the STRUDL-DYNAL finite element model of single section of HSM was developed. Based on the computer output, the lowest fundamental frequency was 25 Hz.

The corresponding vertical and horizontal accelerations for 25 Hz were multiplied by a factor of 1.5 for the members of the HSM finite element model. The values of the acceleration used for the analysis were slightly higher than the actual calculated values for the HSM members. Even though the calculated values of the DSC support structure acceleration are higher than the acceleration values for the HSM structure, the values for the HSM structure were used to determine the seismic forces for the DSC support structure. The staff reviewed this discrepancy and accepts the resulting shear forces and moments tabulated in TR Table 8.2-3 since the combination is summed absolutely.

The maximum moments and shears resulting from the seismic analysis of the HSM are shown in TR Table 8.2-6. They were used in the combination of loads design validation, shown in Table 3.1 of this SER (combination 5). The analyses and results are considered to be acceptable to the NRC staff.

The analyses performed to evaluate potential sliding and overturning of an unanchored module showed that a single unanchored module would not either overturn or slide when subject to the design earthquake. The staff has reviewed these analyses and concurs with their results.

<u>HSM flooding analysis</u>

The analysis performed to evaluate the effect of flood on the HSM is presented in Section 8.2.4 of the TR. The analysis assumed a 50 foot static head and 15 fps maximum flow velocity. This analysis demonstrates that a single, unanchored, submerged module (or multiple modules) would not slide or overturn under the design conditions. Maximum shears and moments due to flood forces were calculated and used in the combination of loads expression (Table 3.1, combination 6). The NRC staff has reviewed and concurs with the analyses and results. However, each site-specific application must validate that its potential flood parameters are within those assumed in the TR or a separate analysis would be required.

Lightning

The lightning protection will be provided at the site. The TR also states that resulting current surge from the lightning will not affect the normal operating condition of the HSM. The staff agrees and accepts the statements in Section 8.2.6 of the TR.

Blockage of HSM ventilation air inlets and outlets

The analysis performed to evaluate the effect of air inlet and outlet blockage is presented in Section 8.2.7 of the TR. Section 8.2.7.1 states that the structural consequences due to the weight of debris blocking the air inlets and outlets are bounded by the structural consequences of tornado and earthquake accidents. The staff agrees with this statement.

An analysis was performed to determine the effect of high temperatures caused by the blockage of both air inlets and outlets on the structural behavior of the HSM. The results from this analysis were used in the combination of loads (Table 3.1, combination 7).

The complete blockage of the HSM ventilation air inlets and outlets was considered as an accident condition. The thermal effects of this accident result from the increased temperatures of the DSC and the HSM at extreme ambient condition of 125°F. NUTECH evaluated this blockage for 48 hours, at which time it was assumed that corrective action would be completed and natural circulation air flow would be restored to the HSM.

At the end of the 48 hours of blockage, the maximum HSM inside surface temperature was calculated to be 395°F. The staff reviews conclude that this high temperature is an acceptable temporary localized condition for the HSM based on the limitation from ACI 349-85-A.4.2. NUTECH's contractor calculated the linear thermal gradients and then calculated the fixed moments and forces. These were input to the STRUDL finite element model to determine the internal forces in the HSM members. Then the internal forces and moments were modified to account for the concrete cracked section properties.

The resultant calculated thermal moment and shear forces are tabulated in Table 8.2-3 of the TR. The staff reviewed the calculations, checked the tabulated results against results from the computer output, and find them acceptable.

HSM load combination

The maximum bending moments and shear forces due to normal and off-normal loads are listed in TR Table 8.1-10. Similarly, the results due to the accident loads are listed in TR Table 8.2-3. The combination of the resultant bending moments and the shear forces was performed based on the requirements of ANSI-57.9-1984 and the results are tabulated in TR Table 8.2-10.

The combinations were checked against the calculated ultimate capacities of the concrete at $400^{0}F$ based on the formulas from ACI 349-85. The staff agrees and accepts the load combination results.

3.3.4.2 DSC and Internals

3.3.4.2.1 DSC Normal Operating Conditions

The dry shielded canister was analyzed for: (1) dead weight loads, (2) design basis internal pressure, (3) design basis operating temperature loads, and (4) normal operation handling loads. The canister internal parts were analyzed for: (1) dead weight loads, and (2) design basis operating temperature loads. Table 3.2 summarizes all the stress analysis results for normal operating conditions. The summary table shows stresses for each DSC component for each load condition analyzed by NUTECH and the corresponding stress as verified by the NRC staff. Each stress value was compared to the allowable stress intensity for the particular material at the stated temperature as defined by the ASME Code for Service Levels A and B conditions. All calculated stresses are below allowable levels.

Table 3.2

DSC STRESS ANALYSIS RESULTS
FOR NORMAL LOADS
Service Level A

DSC Component	Stress Type	Stress (ksi) Dead Weight NUTECH	Dead Weight NRC	10.7 psig Int. Pressure NUTECH	10.7 psig Int. Pressure NRC	100°F Thermal NUTECH	100°F Thermal NRC	Normal Handling NUTECH/NRC	Allowable* Level A & B
DSC Shell	Pri Memb	0.1	0.2	0.5	0.6	N/A	N/A	0.2	18.7
	Memb + Bend	3.7	13.4	0.5	0.6	N/A	N/A	1.8	28.0
	Pri + Second	3.7	13.4	6.2	6.6	17.5	17.8	-	56.1
Inner Top Cover Plate	Pri Memb	0.1	0.7	0	0	N/A	N/A	0.1	18.7
	Memb + Bend	0.5	0.5	4.6	4.6	N/A	N/A	0.3	28.0
	Pri + Second	0.2	0.2	3.4	7.2	0.3	0.3	N/A	56.1
Outer Top Cover Plate	Pri Memb	0.1	0.2	N/A	N/A	N/A	N/A	0.1	18.7
	Memb + Bend	0.4	0.4	4.6	4.6	N/A	N/A	0.3	28.0
	Pri + Second	0.2	0.2	3.4	7.2	1.0	1.0	N/A	56.1
Bottom Cover Plate	Pri Memb	0.1	1.2	0	0	N/A	N/A	0.7	18.7
	Memb + Bend	0.3	0.3	1.	1.	N/A	N/A	1.6	28.0
	Pri + Second	0.3	0.3	0.5	1.5	1.7	4.	0.8-2.4	56.1
Spacer Disc	Pri Memb	0.5	0.5	0	0	N/A	N/A	0	18.7
	Memb + Bend	0.3	0.6	N/A	N/A	46.5	46.5	N/A	56.1

*Allowable stress for Service Levels A and B

Primary Membrane	Sm	= 18.7 ksi
Primary Memb + Bend	1.5 Sm	= 28.0
Primary + Secondary	3.0 Sm	= 56.1
Shell, Disc and end plates for 400°F	SA 204 Type 304	

Dead weight loads for DSC

The dead load analysis for the DSC is presented in Section 8.1.1.2.A of the TR. Both beam bending and shell bending were considered. For the beam bending, a two-span continuously loaded beam, simply supported at three locations corresponding with the DSC support structure, was assumed. The maximum membrane plus bending stress for this condition is 200 pounds per square inch or 0.2 ksi, which is below the 18.7 ksi for ASME allowable stress for Service Level A. The canister was also modeled for local shell bending by considering that the total dead weight was supported uniformly by the two continuous T-section support rails. The NRC staff checked the reference cited in the TR and concludes that it is not a conservative model because shape of the elastic deformation in Bednar (Reference 25) is not consistent with the actual deformed shape caused by two support rails. The NRC staff used a more conservative approach from Roark (Reference 26). The results shown in Table 3.2 are below the ASME Code allowable stress.

Design basis internal pressure

Table 8.1-4 of the TR shows eight cases for operating and accident pressures. NUTECH used the ANSYS (Reference 19) finite element code to model the internal pressure load for the top and bottom portions of the DSC. NUTECH used 1 psig for the internal pressure and then multiplied the stress results by a factor corresponding to the particular load case per TR Table 8.1-4.

Figures 8.1-5 and 8.1-6 of the TR show how NUTECH used symmetry to model the top and bottom portions of the DSC. It is seen that a single element was used to model the thickness of the steel shell, as well as the inner and outer top cover plates and the lower cover plate. The ANSYS user's manual describes the particular element type that NUTECH used as a "two-dimensional isoparametric element," which has two degrees of freedom at each node. It was used by NUTECH as an axisymetric ring element. In this configuration, the computer code only calculates membrane stresses. It is possible to calculate bending stresses with this element, provided two or more elements are used through the thickness of the shell. NUTECH did not model the DSC by using two elements in the thickness. Therefore, none of the bending stresses shown in the TR summary tables are, strictly speaking, bending stresses. Membrane and shear stresses are calculated at the centroid of the element and referred to the edge face and/or to the node as an output option. NUTECH used these output options to "estimate" bending stresses. Section III of the ASME Code requires that bending as well as membrane stresses be evaluated for Class 1 components.

For the internal pressure case, the DSC has local bending stresses at the gross structural discontinuity between the thick cover plates and the shell and also in the middle of the flat end plates. The NRC staff calculated the shell bending stresses at the shell/end plate discontinuity. The method used is given in Roark (Reference 26, p. 465). The result was 1.1 ksi for bending stress. NUTECH reported 6.2 ksi by using the stresses referred from the centroid of the shell element to the inside face. From this procedure, the NRC staff concludes that although the ANSYS program does not calculate bending stresses, the stress used by NUTECH is higher than the bending stress calculated by the NRC staff. In both cases, the stress is substantially below the Code allowable for primary plus secondary stresses for Levels A and B (56.1 ksi).

Similar checks were made by the NRC staff for bending stresses in the DSC inner and outer top cover plates and the bottom cover plate. In all cases the calculated stresses are below the allowable level.

As a final observation regarding NUTECH's computer modeling of the top and bottom portions of the DSC, the NRC staff noted that the thicknesses of the plates as modeled in the computer analyses <u>do not</u> agree with the thicknesses of the plates in the design drawings. In order to predict approximately correct stresses for the plates in question, the NRC staff multiplied the stresses as listed in the computer output by the ratio of the squares of the thicknesses involved. These stresses are shown in the summary Table 3.2 of this SER in the columns headed by "NRC."

<u>Design basis operating temperature</u>

NUTECH has provided for axial thermal expansion of the basket assembly and the inner surfaces of the top and bottom end plates; thus no thermal stresses are induced due to restriction of expansion of internal parts. Similarly, they have sized the spacer disc smaller than the inside diameter of the DSC shell to preclude induced thermal stresses. However, NUTECH did perform five different finite element analyses to determine thermal stresses for differential expansion of the shell, the spacer disc, and the shell/end-cover interface. These analyses were performed at ambient conditions of 100^{o}F, except for one case where the shell was analyzed at 125^{o}F ambient temperature.

The thermal stresses are always defined as "secondary stresses" by the ASME Code. This means that higher allowable stresses are permitted <u>and</u> only Service Level A (for normal operations) and Service Level B (for off-normal operations) need be considered.

For normal operations at an ambient temperature of 100^{o}F, the maximum primary plus secondary stress for all thermal cases considered is 46.5 ksi for the spacer disk. The allowable stress is 56.1 ksi. The staff has reviewed all the documentation provided with the TR and concurs that thermal stresses for the DSC for normal operations meet ASME Code requirements. They are shown in Table 3.2 of the SER.

<u>Operational handling loads for DSC</u>

The only normal operational handling load considered by NUTECH was due to the axial force of 20,000 pounds due to the hydraulic ram acting against the DSC bottom assembly. The resulting stresses are much lower than allowable stress as shown in Table 3.2.

<u>DSC internal basket analyses</u>

Section 8.1.1.3 of the TR discusses the stress analysis considerations of the basket components, i.e., the spacer disc, the 24 guide sleeves and the four 3-inch diameter support rods. The spacer disc was analyzed using a finite element program for the 75 g drop case. For the normal dead weight, the stress levels were divided by 75. The results show stress values lower than the Code allowables.

Because the axial location of the spacer discs coincides with the grid spacers of the fuel assemblies, the weight of the fuel assemblies is transmitted directly to the spacer disc. Thus the guide sleeves and support rods only have to resist their self weight, which is trivial for the spacer disc spacing of 21 inches.

3.3.4.2.2 DSC Off-Normal Events

Three off-normal events were evaluated by NUTECH for the DSC. They were off-normal pressure, jammed DSC during transfer and off-normal temperature. The off-normal temperature of 125°F ambient and the jammed DSC bound the range of loads.

Jammed DSC during transfer

The basis for the postulated off-normal event involving jamming of the DSC during transfer into the HSM is axial misalignment of the DSC. Should this occur, the hydraulic ram could exert an axial force equal to the static weight of the DSC of 80,000 pounds, before a relief valve would prevent further load. The bending stress in the bottom cover plate of the DSC is smaller than the allowable. Also, the bending stress in the DSC shell is well below the allowable stress. These results are shown in Table 3.3 of this report.

Binding of DSC during transfer

A variation of the jammed case involves an angular misalignment of the DSC with respect to the HSM. This condition also results in stresses lower than the allowables.

DSC off-normal thermal/pressure analysis

The off-normal temperature range was taken as -40°F to 125°F for the DSC inside the HSM. The off-normal thermal analysis is the basis for higher thermal stresses for the spacer disc, and the cause of higher internal pressures causing higher shell and end plate stresses. The table in the TR which reports these stresses (Table 8.1-7a) does not, in fact, show higher thermal or pressure stresses for any component except the DSC shell for the thermal event. Pressure stresses are shown to be constant and thermal stresses for the spacer disc are also shown constant. (Compare TR Tables 8.1-7 and 8.1-7a.)

Table 3.3

DSC STRESS ANALYSIS RESULTS
FOR OFF-NORMAL LOADS
Service Level B

DSC Component	Stress Type	Stress (ksi) Internal Pressure 10.7 psig NUTECH	NRC	Thermal 125°F NUTECH	NRC	Off-Norm. Hand NUTECH	NRC	Allowable*
DSC Shell	Pri Memb	0.5	0.6	N/A	N/A	1.2	1.2	18.7
	Memb + Bend	0.5	0.6	N/A	N/A	7.0	7.0	28.
	Pri + Second	6.8	6.6	20.9	21.6	-	-	56.1
Inner Top Cover Plate	Pri Memb	0	0	N/A	N/A	0	0	18.7
	Memb + Bend	4.6	4.6	N/A	N/A	0	0	28.0
	Pri + Second	3.4	7.2	0.3	1.3	0	0	56.1
Outer Top Cover Plate	Pri Memb	0	0	N/A	N/A	0	0	18.7
	Memb + Bend	4.6	4.6	N/A	N/A	0	0	28.0
	Pri + Second	3.4	7.2	1.0	1.8	0	0	56.1
Bottom Cover Plate	Pri Memb	0	0	N/A	N/A	0	0	18.7
	Memb + Bend	1.0	1.	N/A	N/A	6.5	6.5	28.0
	Pri + Second	0.5	1.5	1.7	4.4	3.1	9.5	56.1
Spacer Disc	Pri Memb	0	0	N/A	N/A	0	0	18.7
	Memb + Bend	N/A	N/A	46.5	50.7	0	0	56.1

*Allowable stress is taken for Service Level B for SA 204 Type 304 material at 400°F.

The NRC staff evaluated the TR and concluded that NUTECH did not perform a finite element analysis for the spacer disc for the higher temperature. In order to estimate the higher thermal stress, the staff multiplied the thermal stress for the normal case by 1.09, a factor obtained by comparing DSC outer surface temperatures for 100°F and 125°F ambient conditions (see Table 8.1-12 of TR). The assumption made by the staff is that the thermal stresses are linearly proportional to the temperature. The resulting higher estimated thermal stress is 50.7 ksi as shown in Table 3.3 of the SER. This level is still lower than the allowable of 56.1 ksi.

For the pressure stresses, the staff concluded that NUTECH did not make a separate evaluation for the higher pressure due to the off-normal case. Although the NUTECH documentation is not accurate, the staff can accept the results in TR Table 8.1-7a for the pressure stress since they are well below allowables. This conclusion is based on the very small difference in internal pressure of the helium for the off-normal case. The partial pressure is 5.9 vs. 6.1 psig for 100°F and 125°F, respectively.

DSC load combinations for normal and off-normal conditions

Table 3.2-5a of the TR outlines the different load combinations considered for normal and off-normal conditions. These conditions correspond to Service Levels A and B of the ASME Code. Altogether there were four combinations for both service levels; however, due to the fact that NUTECH did not present data in their TR for the off-normal thermal case and off-normal pressure case, the NRC staff combined load combinations A3 and A4, as well as B2 and B3 for the purposes of presenting the results shown in Table 3.4 of this SER. The staff summarized the combinations as described and finds that all stresses are below the allowables for Service Levels A and B.

3.3.4.2.3 DSC Accident Conditions

Section 8.2 of the TR defines the accident conditions associated with the NUHOMS system. The accident conditions which were examined for the DSC are: (1) earthquake, (2) flood, (3) accident pressure, (4) accident thermal, and (5) accidental drop of the transfer cask with the DSC inside. Of these accidents, the drop case is by far the most severe. NUTECH

Table 3.4

DSC LOAD COMBINATIONS FOR
NORMAL AND OFF-NORMAL OPERATING CONDITIONS

DSC Component	Stress Type	Stress (ksi) Case A1 NUTECH	Case A1 NRC	Case A2 NUTECH	Case A2 NRC	Case[1] A3/A4 NUTECH	Case[1] A3/A4 NRC	Case B1 NUTECH	Case B1 NRC	Case[2] B2/B3 NUTECH	Case[2] B2/B3 NRC	Case B4 NUTECH	Case B4 NRC	Allowable Level*
DSC Shell	Pri Memb	0.1	0.2	0.6	0.8	0.8	1.0	1.8	2.0	1.8	2.0	0.6	0.8	18.7
	Memb + Bend	3.7	13.4	4.2	14.0	6.0	15.8	11.2	21.0	11.2	21.0	4.2	14.0	28.0
	Pri + Second	3.7	13.4	27.4	37.8	27.4	37.8	27.4	37.8	30.8	41.2	30.8	41.6	56.1
Inner Top Cover Plate	Pri Memb	0.1	0.7	0.1	0.7	0.2	0.8	0.1	0.7	0.1	0.7	0.1	0.7	18.7
	Memb + Bend	0.5	0.5	5.1	5.1	5.4	5.4	5.1	5.1	5.1	5.1	5.1	5.1	28.0
	Pri + Second	0.2	0.2	3.9	7.7	3.9	7.7	3.9	7.7	3.9	8.7	3.9	8.7	56.1
Outer Top Cover Plate	Pri Memb	0.1	0.2	0.1	0.2	0.2	0.3	0.1	0.2	0.1	0.2	0.1	0.2	18.7
	Memb + Bend	0.4	0.4	5.0	5.0	5.3	5.3	5.0	5.0	5.0	5.0	5.0	5.0	28.0
	Pri + Second	0.2	0.2	4.6	8.4	4.6	8.4	4.6	8.4	4.6	9.2	4.6	9.2	56.1
Bottom Cover Plate	Pri Memb	0.1	1.2	0.1	1.2	.8	1.9	0.1	1.2	0.1	1.2	0.1	1.2	18.7
	Memb + Bend	0.3	0.3	1.3	1.3	2.9	2.9	7.8	7.8	7.8	7.8	1.3	1.3	28.0
	Pri + Second	0.3	0.3	2.5	5.8	3.3	8.2	5.6	15.3	5.6	15.7	2.5	6.2	56.1
Spacer Disc	Pri Memb	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	18.7
	Memb + Bend	0.3	0.6	46.8	47.1	46.8	47.1	46.8	47.1	46.8	51.3	46.8	51.3	56.1

*Allowable stress is taken for Service Levels A and B for SA 204 Type 304 Material at 400°F.

1. Load cases A3 and A4 were combined into one case because the stresses for the normal and off-normal pressure cases were not supplied by NUTECH.
2. Load cases B2 and B3 were combined into one case because the stresses for the thermal case with the DSC inside the cask or inside the HSM at $T_{ambient}$ = 125°F were not supplied by NUTECH.

classified the thermal accidents and the drop accidents as Service Level D conditions and the remaining accidents as Service Level C conditions. The NRC staff concurs with this classification.

A consequence of classifying the thermal accidents as Service Level C or D is that the ASME Code does not require any stress analysis because of the ASME definition of thermal stresses as "secondary" stresses or "self-relieving" stresses. The only way in which NUTECH was required to give any consideration of the accident thermal cases was in a reduction of material properties at the higher temperature.

Following is a discussion of the results of the accident review.

DSC seismic condition

NUTECH considered the response of the DSC to a seismic event when it is resting on the two support rails. They first performed a rigid-body stability analysis to show no possibility of roll-out. For this purpose they used a factor of 1.5 times .25 g and .17 g for the horizontal and vertical accelerations. The 1.5 factor accounted for the elevation of the DSC in the HSM. No roll-out was possible.

Next NUTECH calculated the natural frequency of the shell ovaling mode and the beam bending mode of vibration. Since the frequency for the ovaling mode was 13.8 Hertz, NUTECH applied an amplification factor of 2.5. (See Regulatory Guide 1.60, Reference 27). The resulting spectral accelerations were 1.0 g and 0.68 g for horizontal and vertical directions, respectively. To account for possible multi-mode excitation, NUTECH used a "safety factor" of 1.5. The total equivalent static load factor used to simulate the seismic event was 1.0 g for the vertical direction and 1.5 g for the horizontal direction.

The stress intensities for the 1 g vertical case were calculated by factoring the dead load analysis results by 1.0. The stress intensities for the 1.5 g horizontal case were calculated by assuming that the DSC is supported by a single T-section rail inside the HSM. Lateral bending stresses were summed absolutely with vertical bending stresses to obtain a combined stress of 21 ksi. The NRC used a more conservative model for lateral bending (Reference 26, Table 5, case 1) and obtained 27.7 ksi. Both values are below the 33.7 ksi allowable for Service Level C.

DSC flood condition

The flood conditions postulated by NUTECH consisted of a 50 foot static head of water and a 15 foot per second flow velocity. It will be necessary for each license applicant to demonstrate that these conditions bound the flood conditions for each individual site.

The static head resulted in a 21.7 psi external pressure which caused 1.2 ksi stresses in the DSC shell and 19.4 ksi and 9.9 ksi stresses in the outer top cover plate and bottom cover plate, respectively. These stress levels are below the 33.7 ksi allowable levels for Service Level C. The NRC calculations as well as the NUTECH calculations are reported in Table 3.5 of this SER.

DSC accident pressure

The bounding DSC internal accident pressure is 49.1 psig according to Table 8.1-4 of the TR. This internal pressure could occur if the transfer cask neutron shield were lost during transfer operations on a day when the ambient air temperature is 125°F. Further assumptions were that all cladding failed and that 100% of the fill gas and 30% of the fission gas were released inside the DSC. Under these unlikely conditions, the internal pressure could reach 49.1 psig. Table 3.5 of this SER shows the stress results of this case. All stress intensities are lower than the allowables.

It should be noted that NUTECH used 400°F as the appropriate temperature to select the allowable stresses for the materials in the DSC (TR Table 8.2-9c). However, Table 8.1-13 of the TR indicates that the DSC shell reaches a maximum temperature of 513°F for this accident; therefore, the NRC staff used lower material allowable stresses for this case and all accident load combinations that have this load case as a part of the load combination.

Table 3.5

DSC STRESS ANALYSIS RESULTS
FOR ACCIDENT CONDITIONS
Service Level C

DSC Component	Stress Type	Stress (ksi) Seismic NUTECH	Seismic NRC	Flood 50' NUTECH	Flood 50' NRC	Accident Pressure 49.1 NUTECH	Accident Pressure 49.1 NRC	Thermal 125°F[1] NUTECH	Thermal 125°F[1] NRC	Accident Handling NUTECH	Accident Handling NRC	Allowable*
DSC Shell	Pri Memb	-	-	1.2	1.2	2.6	2.6	N/A	N/A	1.2	1.2	22.4
	Memb + Bend	21.0	27.7	-	-	6.5	2.6	N/A	N/A	7.0	7.0	33.7
	Pri + Second					31.2	30.6	N/A	N/A			N/A
Inner Top Cover Plate	Pri Memb	-	-		-	0	0	N/A	N/A	0	0	22.4
	Memb + Bend	-	-		-	23.2	23.1	N/A	N/A	0	0	33.7
	Pri + Second	-	-		-			N/A	N/A	0	0	N/A
Outer Top Cover Plate	Pri Memb	-	-	-	-	0	0	N/A	N/A	0	0	22.4
	Memb + Bend	-	-	13.5	19.4	23.2	23.1	N/A	N/A	0	0	33.7
	Pri + Second	-	-	-	-			N/A	N/A	0	0	N/A
Bottom Cover Plate	Pri Memb	-	-	-		0	0	N/A	N/A	0	0	22.4
	Memb + Bend	-	-	7.6	9.9	4.9	4.9	N/A	N/A	6.5	6.5	33.7
	Pri + Second	-	-	-				N/A	N/A		9.5	N/A
Spacer Disc	Pri Memb	0	0.5	-	-	0	0	N/A	N/A	0	0	22.4
	Memb + Bend	0	0.6	-	-	0	0	N/A	N/A	0	0	N/A

*Allowable stress for Service Level C

P_m larger of 1.25 m or Sy = 22.4
$P_L = P_L+P_B$ = larger of 1.8 Sm or 1.5 Sy = 33.7

1. No secondary stress needs to be evaluated according to ASME Code for Service Level C. This includes thermal as well as secondary bending stresses for pressure cases.

DSC thermal accident cases

NUTECH indicates in TR Table 3.2-5a that thermal accident cases were considered as separate load cases and as a part of the load combinations. The NRC staff has noted in this SER that the ASME Code does not require any stress evaluation for thermally induced stresses for Service Levels C and D. Since NUTECH categorized the two thermal accident cases as Service Levels C and D, they were not obliged to evaluate them. However, the material properties for load conditions, such as the case of the pressure stress at the higher DSC temperature, as described in the preceding paragraph, should have been taken at the higher temperature. The NRC staff did this in all tables in this SER. The results are satisfactory.

DSC load combinations for Service Level C accident conditions

Table 3.6 shows the results of seven load combinations. Load combinations, as defined in Table 3.2-5a of the TR, are a bit misleading because case C4 and C5 are actually the same, as well as cases C6 and C7. The only difference in both of these sets of cases is thermal stresses, which NUTECH did not evaluate. As may be seen from Table 3.6, all stresses are below allowable levels.

Discussion of cask drop

Because the cask drop accidents postulated by NUTECH cause the highest stresses in both the DSC and the transfer cask, it is appropriate to discuss the basis for selecting some of the parameters and assumptions for this case. It should be pointed out that the drop situations that were postulated by NUTECH all involve dropping the TC with the DSC inside at a maximum height of 80 inches. The NRC staff considers these assumptions reasonable, because the DSC will always be in the TC or inside the HSM whenever it is outside of the building which houses the spent fuel pool. The requirements of 10 CFR 72 must be met whenever the irradiated fuel is outside of the spent fuel pool building. Inside the building, 10 CFR 50 governs. The centerline of the HSM is located at 102 inches above the base pad and therefore the maximum drop height would be about 60 inches for the DSC, should it fall off of the transport trailer during loading or during

Table 3.6

DSC LOAD COMBINATIONS FOR ACCIDENT
Service Level C Cases[1]

DSC Component	Stress Type	Case[2] C1 NUTECH	Case[2] C1 NRC	Case C2 NUTECH	Case C2 NRC	Case C3 NUTECH	Case C3 NRC	Case[3] C4/C5 NUTECH	Case[3] C4/C5 NRC	Case[4] C6/C7 NUTECH	Case[4] C6/C7 NRC	Allowable
		Stress (ksi)										
DSC	Pri. Memb	2.7	2.0	1.8	1.4	2.9	3.0	2.7	2.8	3.9	4.0	22.4
	Memb + Bend	31.2	30.3	3.7	13.4	12.0	17.8	10.2	16.0	17.2	21.9	33.7
Inner Top Cover Plate	Pri. Memb	0.1	0.7	0.1	0.7	0.2	0.8	0.1	0.7	0.1	0.7	22.4
	Memb + Bend	23.7	23.6	5.1	5.1	24.0	23.9	23.7	23.6	23.7	23.6	33.7
Outer Top Cover Plate	Pri. Memb	0.1	0.2	0.1	0.2	0.2	0.3	0.1	0.2	0.1	0.2	22.4
	Memb + Bend	23.6	23.5	18.1	19.8	23.9	23.8	23.6	23.5	23.6	23.5	33.7
Bottom Cover Plate	Pri. Memb	0.1	1.2	0.1	1.2	0.8	1.9	0.1	1.2	0.1	1.2	22.4
	Memb + Bend	5.2	5.2	8.9	11.2	6.8	6.8	5.2	5.2	11.7	11.7	33.7
Spacer Disc	Pri. Memb	0.5	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	22.4
	Memb + Bend	0.3	1.2	0.3	0.6	0.3	0.6	0.3	0.6	0.3	0.6	33.7

1. Secondary stresses are not required for Service Level C.
2. Seismic stresses are considered "mechanical loads" and must be combined with DW and accident pressure for C1.
3. Thermal stresses are secondary and need not be evaluated for Service Level C. Therefore, both C4 and C5 are identical cases.
4. Because thermal stresses need not be evaluated for Service Level C, cases C6 and C7 are identical.

transport between the spent fuel pool building and the ISFSI site. Thus, 80 inches is conservative.

Five different drop orientations were considered: (1) a horizontal drop, (2 and 3) a vertical end drop onto the top or bottom of the TC, and (4 and 5) a corner drop at an angle of 30^{0} onto the top or bottom corner of the TC. The drop height was 80 inches for all orientations.

The magnitude of the deceleration for each case was defined in Section 3 of the TR as 75 g for either vertical or horizontal drop orientations and 25 g for the corner drop. NUTECH based these values on an EPRI report (Reference 28), which described a method of predicting maximum decelerations of casks as a function of drop height, target hardness (i.e., hardness of concrete pad), and cask orientation.

Because Reference 28 does not document the deceleration time history, it was necessary for the NRC staff to establish what the representative time histories and damping coefficients for the three orientations would be, in order to predict appropriate dynamic load factors (DLF). NUTECH provided additional material which included references to drop test data for a 90 ton rail cask (Reference 29). The time histories from this reference were used to determine the DLFs for the different drop orientations. As discussed in Section 2 of this SER, the DLFs are also dependent on structural damping. The staff determined that a damping value of 7% is conservative. This was based on sources in the open literature as well as the information provided by NUTECH. The NRC staff concluded that the DLFs for the vertical, horizontal, and corner drops are 1.50, 1.75, and 1.25, respectively. These factors, when multiplied times the unfactored decelerations levels, produced values of 73.5 g, 66.5 g and 25.0 g for the three drop orientations. These values compare favorably with the deceleration values of 75 g, 75 g, and 25 g selected by NUTECH in their design criteria. Based on the above review, the NRC staff finds that the deceleration levels used by NUTECH are appropriate for the drop cases considered.

The deceleration levels specified by NUTECH provide a margin of safety for ensuring the fuel integrity against the effects of impact according to Reference 30. The reference indicates that, for the type of fuel which the NUHOMS-24P system was designed around, there is ample safety margin to meet the requirements of 10 CFR 72.122(h). The B&W 15 x 15 fuel assemblies should not fail if the dynamic impact loads are below 147 g for end drops and 101 g for horizontal drops. As can be seen from the preceding paragraphs, the NUTECH loadings are substantially below these levels.

In all cases, NUTECH used the ANSYS finite element code to model the drop cases for the DSC and TC cask components. For the vertical drop, an axisymetric load and an axisymetric geometry were modeled, using an equivalent 75 g static load. For the horizontal and corner drop cases, NUTECH modeled an axisymetric structure with non-axisymetric loading. The asymmetrical loading was approximated with a Fourier series technique in conjunction with an ANSYS element type designed to facilitate the use of the Fourier (harmonic) series.

The distribution of impact force for horizontal and corner drop cases was approximated by cosine functions, which in turn were approximated by the Fourier series. NUTECH calculated the depth of concrete penetration by the dropped cask using the modified Petry formula (Reference 31), which predicted a smaller penetration depth than Reference 29. Had NUTECH used the deeper crush depth as predicted by Reference 29, the impact force would have been distributed over a larger area of the TC than NUTECH actually used to develop the Fourier series coefficients. Hence the calculations provided in the appendices of the TR are based on conservative assumptions. The computer analyses use these assumptions.

The finite element analysis calculations which NUTECH made modeled the DSC inside of the cask for the end and corner drops. See Figures 8.2-6 and 8.2-7 of the TR. Note that the DSC shell and upper and lower cover plates as well as the cask top cover plate and cask bottom cover plate were all modeled using one element through the thickness. Consequently, as described in Section 3.3.4.2.1 of this SER, the computer code only calculated membrane stresses and did not calculate any bending stresses except in the structural shell of the TC and at the outside diameter of the top and bottom cover plates of the TC. NUTECH "estimated" bending stresses by referring the membrane stresses calculated at the centroid of the element to an outer face and/or node of the element.

The results of these analyses are reported in Table 8.2-7 of the TR. The NRC staff has summarized the results and included the findings of the staff review in Table 3.7. Discrepancies between the TR results and staff results can be attributed to two principal causes. Some results reported in the TR have been superseded by additional calculations that NUTECH performed following submittal of the Revision 1 of the TR and consequently are not shown in the TR. Another source of discrepancy is due to the difference in thickness of the DSC cover plates as calculated and as specified on the drawings. The staff increased the stresses listed in the computer output listing by a ratio of the squares of the end plate thicknesses. The results in Table 3.7 show that the stresses for all components for all drop orientations are lower than the ASME Code allowable stresses for Service Level D. The NRC staff evaluated the material properties for the worst case temperature reported by NUTECH, i.e., load case D1. Consequently, the allowable stresses are slightly lower than NUTECH used. In all cases, the calculated stresses are lower than the allowable stresses.

Two analyses were carried out to verify the design adequacy of the spacer disc. A finite element analysis of one half of a spacer disc, symmetrically loaded with 75 times the vertical static load, was performed. Also, a stress and buckling stability analysis of the entire disc was performed using ANSYS. This analysis assumed the disc was supported in-plane around the circumference of the disc and out-of-plane at the four support rod locations. Again, the load consisted of 75 times the dead weight. As Table 3.7 of this SER shows, the spacer disc satisfies the ASME Code allowable stresses for both vertical and horizontal drop orientations. The NRC staff verified that the eigenvalue buckling solution is 1.8 times the load for horizontal load cases. No buckling analysis was performed for the vertical drop case.

The guide sleeves were checked for bending plus membrane stress when loaded horizontally by 75 times self weight and simply supported between spacer discs. The resulting stress intensity was only 2 ksi, far below the allowable of 63.5 ksi.

The four support rods running the length of the basket were also checked for stress as well as critical buckling during a vertical drop. For the drop accident, NUTECH postulated the load for each rod to be one quarter

Table 3.7

DSC DROP ACCIDENT LOADS
Service Level D

DSC Component	Stress Type	Stress (ksi) Vertical NUTECH	Vertical NRC	Horizontal NUTECH	Horizontal NRC	Corner NUTECH	Corner NRC	Allowabl
DSC Shell	Pri. Memb.	6.2	33.2	9.2	17.6	12.7	18.2	43.4
	Memb + Bend	19.3	29.2	12.4	24.7	28.6	28.8	63.5
Inner Top Cover	Pri. Memb.	0	0	14.	16.9	17.6	17.6	43.4
	Memb + Bend	33.8	34.0	15.9	18.7	8.2	12.0	63.5
Outer Top Cover	Pri. Memb.	0	0	9.5	9.5	10.2	10.2	43.4
	Memb + Bend	20.1	29.0	14.6	21.0	6.2	14.7	63.5
Bottom Cover	Pri. Memb.	0	0	9.5	9.5	34.7	39.4	43.4
	Memb + Bend	19.1	30.5	14.6	21.0	23.2	43.8	63.5
Spacer Disc	Pri. Memb.	1.	24.4	36.4	37.3	-	-	43.4
	Memb + Bend	22.4	27.9	26.1	47.6	-	-	63.5
Support Rods	Primary	32.6	32.6			32.6	32.6	49.9
Top End Struct. Weld	Primary (shear)	5.	5.	9.5	9.5	9.5	9.5	43.4
Bottom End Struct. Weld	Primary (shear)	4.1	4.1	9.5	9.5	9.5	9.5	43.4

* Allowables taken at worst case temperature, i.e., for Case D1, T=513^{o}F shell temperature.

of: the dead weight of eight spacer discs, the weight of the guide sleeves, and the self weight of one rod. The primary axial stress was only 32.6 ksi compared with an allowable of 49.9 ksi. Also, the critical buckling load was found to be 180 ksi, well above the actual load. Based on the above evaluation, the NRC staff concurs that the support rod design is satisfactory.

DSC load combinations

Table 8.2-9b of the TR summarized the enveloping load combination stress results for the DSC accident conditions. Table 3.2-5a of the TR defined the load cases for each load combination. The stress intensities in the DSC at various critical locations were evaluated by combining the dead load, accident pressure load, and the worst drop orientation load. Table 3.8 of this SER uses material allowables for Service Level D for the worst thermal condition reported in the TR. These allowables are somewhat lower than the TR used; however, it may be seen that even with these lower allowable stresses, the DSC components meet the ASME Code requirements.

It should be noted that NUTECH elected to use Service Level D for accident case allowable stresses. While the NRC staff concurs with this decision, it must be coupled with the NUTECH operating controls and limits as proposed in Section 10 of the TR. Following a cask drop of fifteen inches or greater, the DSC must be retrieved, and the DSC and the internals must be inspected for damage. The NRC staff sets this operational control because it is in keeping with the high allowable stress of the Service Level D, i.e., permanent deformations of the DSC confinement boundary and the DSC internals are permitted under Service Level D conditions.

DSC fatigue evaluation

Section NB-3222.4a of Section III of the ASME Code (Reference 12) requires that components be qualified for cyclic operation under Service Level A limits unless the specified service loadings of the components meet all six conditions defined by NB-3222.4d. Although it is superficially clear that the DSC is inherently not subjected to high cycles of pressure, temperature, temperature difference, or mechanical loads, NUTECH evaluated each of the six conditions defined by the ASME Code. The NRC staff

Table 3.8

DSC ENVELOPING LOAD COMBINATION
RESULTS FOR ACCIDENT LOADS
Service Level D

DSC Component	Stress Type	Controlling Load Combination	Stress (ksi) Calculated NUTECH	NRC	Allowable*
DSC Shell	Pri. Memb. Memb + Bend	D2	11.9 25.9	35.0 49.8	43.4 63.5
Inner Top Cover	Pri. Memb. Memb + Bend	D2	14.0 57.5	18.3 57.6	43.4 63.5
Outer Top Cover	Pri. Memb. Memb + Bend	D2	9.5 43.5	10.4 52.5	43.4 63.5
Bottom Plate	Pri. Memb. Memb + Bend	D2	9.5 28.4	40.6 49.0	43.4 63.5
Spacer Disc	Pri. Memb. Memb + Bend	D2	36.4 26.4	37.8 48.2	43.4 63.5
Guide Sleeve	Memb + Bend	D2	2.	2.0	63.5
Support Rods	Pri. Memb.	D2	32.6	32.6	49.9
Top End Structural Weld	Primary (shear)	D2	15.5	15.5	43.4
Bottom End Struc. Weld	Primary (shear)	D2	12.0	12.0	43.4

* Allowables taken at worst case temperature, i.e., Case D1, T=513°F shell temperature.

evaluated NUTECH's analysis and concurs with the finding that the service loading of the DSC meets all conditions, and therefore does not require a separate analysis for cyclic service.

3.3.4.3 DSC Support Assembly Analysis

A linear elastic structural analysis was performed using the STRUDL finite element computer program to determine the deflections, forces and stresses under normal, off-normal and accident loading conditions. Three load combinations were performed to determine the worst resultant stresses and the end forces. The boundary conditions used in the finite element mathematical model do not reflect the boundary condition shown on the drawings in the TR. The drawings indicate that the ends of the T-section guide rails are welded to the access opening sleeve, whereas analytically they were modeled as free ends instead of fixed ends. The staff discussed this discrepancy with the vendor's contractor and concluded that this is a conservative approach for the loading condition listed above. Therefore, the staff accepts the finite element mathematical modeling technique for the analysis of the support assembly.

3.3.4.3.1 DSC Support Assembly Normal Operating Condition

The normal operating condition loads consist of the dead weight of the support assembly, the dead weight of the DSC, the DSC operational handling loads and temperature loads.

<u>DSC support assembly dead weight analysis</u>

The staff checked and concurs with the results due to the dead weight of the support assembly and the weight of the DSC from the STRUDL computer output. The worst stresses are listed in Table 8.1-8 of the TR, forces at the end connections are listed in Table 8.1-9, and the maximum vertical deflection is listed in TR Table 8.1-9a. The tabulated values of deflection and stresses meet the AISC allowable limits for normal conditions.

DSC support assembly operational handling analysis

The normal operating handling load considered was a 20,000 pound load applied axially to the rails. This models the normal condition of loading the DSC into the HSM with a coefficient of friction of 0.25. The staff checked the results from the STRUDL computer output. The worst stresses are listed in Table 8.1-8 of the TR. Forces at the end connections are listed in Table 8.1-9, the maximum vertical deflection was listed in TR Table 8.1-9a. The tabulated values of deflection and stresses meet the AISC allowable limits for normal conditions.

DSC support assembly thermal analysis

After reviewing the drawings in the TR and discussing the analytical model with the vendor's contractor, the staff agrees that no thermal stresses will be induced into the support assembly system. To permit free thermal expansion, slotted bolt holes are used at the connections. The bolts will be installed "snug tight" in accordance with AISC requirements with lock nuts added to ensure that the bolts remain in place; then the friction in the bolted assembly can be overcome by the thermal expansion of the members during normal heatup conditions.

3.3.4.3.2 DSC Support Assembly Off-Normal Event

Section 8.2 of the TR discusses off-normal events as they relate to the support assembly. The off-normal event conservatively considered was a jammed condition where the hydraulic ram exerted a force equal to the weight of DSC and was applied axially to the rails of the support assembly. Results were combined with the results due to the dead weight of the support assembly. The combination from the STRUDL output was checked by the staff. The worst stresses are listed in Table 8.2-11 of the TR and the maximum end loads are listed in Table 8.2-12. The tabulated stress values meet the AISC normal allowable limits. The staff checked the tabulated results against the computer output and concurs with the results as shown in TR Table 8.2-11. However, the qualification of the weld joints between the rails and the embedded access opening sleeve was not documented in the TR. The staff performed independent calculations for these weld joints and found them to be acceptable.

3.3.4.3.3 DSC Support Assembly Accident Analysis

The only loading that the DSC support assembly experiences during an accident analysis is the loading combination associated with a seismic event. Hand calculations were performed to determine the lowest frequency of the support structure. The staff review concurs with the frequency as calculated by NUTECH's contractor. The corresponding values of accelerations at 18.3 Hz are 0.40 g in the vertical and 0.60 g in the horizontal directions; however, 0.48 g acceleration was used in both horizontal orthogonal directions for the finite element computer analysis. The staff reviewed the load combinations from the computer input list, and found none of them reflect the accident load combination as shown in Table 8.2-11 of the TR. The tabulated stresses were taken from load combination 21 of the computer runs which is the combination of the load cases 1, 3, 7, 27, 28. They are: the dead weight of the support assembly, the normal axial handling load, and the transverse seismic (vertical Y) added to the X and directions for seismic (horizontal). The staff has not confirmed that this produces the worst load combination. However, conservatisms built into the model, such as boundary conditions of the structure, inclusion of the normal axial handling load, simultaneous application of the seismic forces in all three orthogonal directions and summation of the results absolutely should lead to an acceptable combination. The stresses are tabulated in TR Table 8.2-11. The calculated stresses of this accident load combination are lower than the accident allowable stress of 1.5 times the normal allowable stresses at 600^0 as shown in TR Table 8.2-11. They are also lower than the accident allowable stress of 1.33 times for normal allowable stresses as prescribed by AISC steel construction manual.

The staff compared the tabulated stress and end load combinations from TR Tables 8.2-11 and 8.2-12 against the computer output and finds them acceptable.

Qualification of the embedded support connection to the HSM for the DSC support assembly

The licensee submitted additional detail drawings of the embedded support connections, but did not submit any qualification calculations. The staff evaluated the drawings and performed some hand calculations to determine that under the reaction forces at the boundary connections of the DSC support assembly, the embedded support connections are acceptable.

DSC support assembly load combinations

Three load combinations were considered. Load combination one consists of the DSC plus the support assembly dead weight, plus the DSC handling loads for a typical normal operating load case. Load combination two includes the dead weight of the support structure plus DSC handling loads in the jammed condition, representing an off-normal loading. The third load combination includes the total dead weight plus design basis seismic loads for an accident event. These results were compared to AISC code allowables and they are within the allowable limits.

DSC seismic restraint analysis

The DSC seismic restraint is located inside the HSM access opening. The restraint and its attachment were designed for a lateral force equal to the mass of the DSC times the horizontal acceleration times an impact factor of 1.5. The staff performed an independent review including hand calculations and concludes that the DSC seismic restraint is acceptable under the loading condition described above.

3.3.4.4 Transfer Cask

3.3.4.4.1 TC Normal Operating Conditions

The transfer cask was evaluated for the three normal operating conditions of: (1) dead weight load, (2) thermal loads, and (3) normal operation handling loads. Table 3.9 summarizes all the stress analysis results for the normal operating conditions. The summary table shows stresses for each of the TC components for each of the three loads as

Table 3.9

TRANSFER CASK STRESS ANALYSIS RESULTS FOR NORMAL LOADS
Service Levels A and B Allowables

Cask Component	Stress Type	Stress (ksi) Dead Weight		Thermal**		Normal Handling		*Allowable
		NUTECH	NRC	NUTECH	NRC	NUTECH	NRC	
Cask Shell	Pri Memb	0.7	0.7	NA	NA	0.5	-	21.7
	Memb + Bend	0.8	0.5	NA	NA	30.3	-	32.6
	Pri + Sec.	0.4		20.3	12.2	35.6	48.4	65.1
Top Cover Plate	Pri Memb	0.2	-	NA	NA	-	-	21.7
	Memb + Bend	0.6	-	NA	NA	6.3	-	32.6
	Pri + Sec.	0.5	-	7.4	9.2	-	-	65.1
Bottom Cover Plate	Pri Memb	0.2	-	NA	NA	-	-	18.7
	Memb + Bend	1.3	-	NA	NA	14.2	-	28.
	Pri + Sec.	1.4	-	5.3	-	-	-	56.1
Top Ring	Pri Memb	.2	-	NA	NA	-	-	20.3
	Memb + Bend	.1	-	NA	NA	-	-	30.5
	Pri + Sec.	.5	-	4.5	-	-	-	60.9
Bottom Ring	Pri Memb	.4	-	NA	NA	-	-	20.3
	Memb + Bend	.3	-	NA	NA	-	-	30.5
	Pri + Sec.	.6	-	14.9	-	-	-	60.9

* Allowables taken at 400^{o}F
** Thermal stresses are considered secondary stresses only

analyzed by NUTECH. The NRC staff verified selected components and has recorded them adjacent to the NUTECH stress levels. The ASME Code allowable stresses for the various materials were taken at 400°F for Service Levels A and B. All calculated stresses are below allowable levels.

Dead weight loads for the TC

The dead weight loads were evaluated for the TC in a vertical orientation, suspended from the lifting trunnions, as well as a horizontal orientation supported by the pillow blocks of the TC support skid. See Figure 1.3-4 of the TR for a sketch of the skid. All stress levels are one to two orders of magnitude lower than allowables.

Thermal loads for the TC

Section 8.1.1.9.C of the TR describes the thermal analysis performed by NUTECH to verify that the thermal stresses in the TC are below allowable stresses for Service Levels A and B. These service levels are the only ones that NUTECH was required to evaluate according to the ASME Code for Class 2 components (Reference 17). Table 3.2-5b of the TR defines the temperature at which specific load cases were evaluated, i.e., an ambient temperature of 100°F for normal conditions, and an ambient temperature of 125°F for off-normal conditions. The NRC staff reviewed the computer analyses for the thermal case and confirmed that only one run was made for the 125°F case, although it was not possible to confirm that the temperature distribution for the model was correct. No information was provided for a temperature distribution. Also the NRC staff noted that the thermal stresses reported by NUTECH in TR Tables 8.1-10a and 8.1-10b show identical thermal stresses for normal and off-normal. The staff checked the computer output and recorded the stresses as shown in the summary table of this SER. In all cases the staff confirmed that the thermal stresses are below allowables.

Operational handling loads for TC

As described in the dead weight load section above, there are two normal operating handling cases for the TC: vertically supported by the crane, and horizontally supported by the skid. The former is governed by ANSI N14.6 rules (Reference 18) and the latter is governed by the ASME Code.

The ANSI code is concerned with critical loads and consequently only addresses the lifting trunnion design and the TC shell in the vicinity of the lifting trunnion. Table 3.9 of this SER summarizes the results of stress analysis for the TC shell and top and bottom cover plates. All results for the normal handling case are satisfactory for Service Level A.

TC Trunnion loads and stresses

The relevant design criteria for lifting a "critical load," i.e., the spent fuel loaded in the DSC inside the TC while in the fuel building are covered by ANSI N14.6, 1987 (Reference 18) and NUREG-0612 (Reference 32). Critical loads, as defined by N14.6, are defined as loads "whose uncontrolled movement or release could adversely affect any safety-related system or could result in potential off-site exposures comparable to the guideline exposures outlined in 10 CFR Part 100." In the case of the transfer cask, the cask lifting and tilting trunnions shall be considered as a special lifting device for the DSC. Because its design does not provide a dual-load path, the design criteria requires that load bearing members shall be designed with a safety factor of two times the normal stress design factor for handling the critical load. Thus the load bearing members must be sized so that yield stresses are no more than one-sixth minimum tensile yield strength of the material or no more than one-tenth the minimum ultimate tensile strength of the material. An additional allowance for crane hoist motion loads is recommended by NUREG-0612. Although Reference 32 does not quantify the magnitude of this dynamic load, ANSI NOG-1-1983 (Reference 33) does specify 15%, which was used by NUTECH. Therefore the NUTECH assumption is appropriate.

Table 3.10 summarizes the results for the lifting trunnion assemblies, weld regions and cask shell. This table presents summary results for the lifting and supporting trunnions that are designed in accordance with: (1) ANSI N14.6 for critical lift loads, and (2) ASME for horizontal Table support loads. The local stresses in the TC at the intersection of the trunnion sleeve and the shell stiffener insert (see Figure C.1-1 of the TR) are calculated by using the method of the Welding Research Council, WRC-297 (Reference 34). The local stresses at the intersection of the shell and the shell stiffener insert are also evaluated by the same method. Summary Table 3.10 shows that all stresses are less than the allowables for both the ANSI-

Table 3.10

SUMMARY OF STRESS ANALYSIS FOR
LIFTING TRUNNION ASSEMBLIES, WELD REGIONS
AND CASK SHELL FOR LIFTING CASES

Component Location	Stress (ksi): Critical Handling Loads per ANSI N14.6 — Stress Intensity			Critical Handling Loads per ANSI N14.6 — Allowable			On-site Transfer per ASME III Class 2 — Stress intensity			On-site Transfer per ASME III Class 2 — Allowable*		
Trunnion Lift Pin	5.9			13.5			NA			NA		
Trunnion Rest Pin	6.3			13.5			3.6			43.8		
1.5" Sleeve @ Insert	5.7			9.0			4.1			32.6		
Weld @ Rest Pin/ Sleeve	Plane 1	Plane 2		Plane 1	Plane 2		Plane 1	Plane 2		Plane 1	Plane 2	
	5.0	7.0		9.0	8.0		5.1	6.8		45	45	
Weld @ Sleeve/ Insert	Pl 1	Pl 2	Pl 3	Pl 1	Pl 2	Pl 3	Pl 1	Pl 2	Pl 3	Pl 1	Pl 2	Pl 3
	5.	3.8	4.	7.	9.0	5.6	5.	3.8	4.	32.6	45.	32.6
Cask Stiffener Plate ASME	Vert	Tilt	Horiz	All Cases			DL± Vert	DL± Lat	DL± Comb	All cases		
	22.6	22.8	19.1	32.6			22.3	41.0	31.3	65.1		
Cask Shell @ Stiff. Plt. ASME	Critical Lift						On-Site Trans.					
	17.4			32.6			48.4			65.1		

*Service Level A

and the ASME-governed load conditions. All stresses shown are the result of the NRC staff calculations.

Table 3.11 shows the results for the tilting trunnion assemblies. Comparisons between the NUTECH-derived and NRC staff-derived stresses show that all the stresses are lower than the allowable except for the tilting trunnion shell to sleeve intersection. The discrepancy arises due to the NRC staff using material allowables evaluated at 400°F. Table 8.2-13 of the TR also shows material allowables evaluated at 400°F. The staff considers that this temperature may be overly conservative, if Table 8.1-14 of the TR is consulted. There, the maximum exterior cask temperature noted by NUTECH was 248°F. If the NRC staff considers the maximum exterior temperature of the TC to be 300°F (still conservative), then the material allowable would be 67.5 ksi. With this adjustment in allowable stress, all calculated stresses are below the allowables.

3.3.4.4.2 TC Off-Normal Operating Conditions

The only off-normal operating condition considered by NUTECH was for an ambient temperature of 125°F. Since NUTECH reported the same stresses in Tables 8.1-10a and 8.1-10b of the TR, and actually only evaluated thermal stresses for 125°F, the results are the same. Table 3.9 of this SER shows these results. They are all satisfactory.

TC load combinations for normal and off-normal conditions

Table 3.2-5b of the TR defines the different load combinations for normal and off-normal events. These conditions correspond to Service Levels A and B of the ASME Code. Altogether there are five Level A conditions and two Level B conditions; however, NUTECH does not present data for all the cases, so Table 3.12 of this SER has combined the conditions as follows. NUTECH only evaluated the thermal case for an ambient temperature of 125°F. Consequently there is no difference between their load cases A4 and B1, and similarly for A5 and B2. In all cases the allowable stresses were evaluated for a material temperature of 400°F, a conservative value. As shown in Table 3.12, all the stresses are lower than the allowables.

Table 3.11

SUMMARY OF STRESS ANALYSIS FOR TILTING
TRUNNION ASSEMBLIES, WELD REGIONS AND CASK SHELL

For On-Site Transportation Cases
Per ASME III Class 2

Component, Location	Stress Intensity (ksi)		Allowable (ksi)*
	NUTECH	NRC	
Trunnion/Sleeve Intersection	5.6	9.	18.7
Sleeve/Shell Intersection	9.3	8.4	21.7
Trunnion/Sleeve Weld	12.6	12.4	18.7
Sleeve/Shell Weld	9.5	7.2	21.7
Shell/Sleeve Intersection	67	65.4	65.1 67.5**
Shell membrane stress	5.2	5.2	28.4

* Allowable stresses taken at 400°F
** Allowable stresses taken at 300°F

Table 3.12

TRANSFER CASK LOAD COMBINATIONS
FOR NORMAL OPERATING CONDITIONS
Service Levels A and B

Cask Component	Stress Type	Load Combination	Calculated Stress (ksi) NUTECH	NRC	Allowable Stress (ksi)
Cask Shell	Pri Memb	A4/B1	1.2	1.2	21.7
	Memb + Bend	A4/B1	31.1	30.8	32.6
	Pri + Sec.	A4/B1	56.3	61.0	65.1
Top Cover Plate	Pri Memb	A2/B5	0.2	0.2	21.7
	Memb + Bend	A4/B1	6.9	6.9	32.6
	Pri + Sec.	A4/B1	7.9	9.7	65.1
Bottom Cover Plate	Pri Memb	A1	0.2	0.2	18.7
	Memb + Bend	A4/B1	15.5	15.5	28.
	Pri + Sec.	A4/B1	6.7	6.7	56.1
Top Ring	Pri Memb	A1	0.2	0.2	20.3
	Memb + Bend	A3	0.1	0.1	30.5
	Pri + Sec.	A1	5.0	5.0	60.9
Bottom Ring	Pri Memb	A3	0.4	0.4	20.3
	Memb + Bend	A3	0.3	0.3	30.5
	Pri + Sec.	A3	15.5	15.5	60.9

No load combinations for either case B1 or B2 were presented by NUTECH. The TR distinguished case B1 from A4 and case B2 from A5 by indicating a higher ambient temperature (125^{o}F). However, NUTECH only calculated stresses associated with the 125^{o}F temperature.

3.3.4.4.3 TC Accident Conditions

Section 8.2 of the TR defines the accident conditions that affect the transfer cask. These conditions are: (1) earthquake, and (2) accidental drop of the TC with the DSC inside. NUTECH also considered a third case, as defined on page 8.2-6 and 8.2-7 of the TR; however, this case was not incorporated in TR Table 3.2-5b, and the results were never incorporated into the enveloping load combination Table 8.2-14 in the TR.

The unincorporated case is for design basis winds. NUTECH postulated a pressure of 595 pounds per square foot (psf) pressure acting on the surface of the TC when supported by the transport trailer. This was based on a maximum wind pressure of 397 psf. NUTECH showed that if the height to the top of the cask is 146 inches, and the track of the transport vehicle is 132 inches, there is a safety factor of 1.5 against overturning. Shell stresses were also evaluated and found to be 3.8 ksi, well below the 26 ksi allowable for Service Level C. The NRC staff concurs with the results for the DBT winds, provided the site-specific equipment, i.e., the trailer and the skid, correspond dimensionally with the example in the TR.

TC seismic condition

NUTECH evaluated the effects of a seismic event on a loaded DSC inside the TC for two conditions. The first case postulated was for the TC in a vertical orientation in the decontamination area during closure of the DSC. For this case NUTECH showed that the loaded TC would not overturn during an earthquake, provided the loaded TC weighed 190 kips and experienced a horizontal acceleration of 0.4g. Since the seismic criteria calls for 0.25 g at ground level, even when both orthogonal directions are summed by the SRSS and the resultant 0.35 g is used to calculate the stability, the NRC staff calculated a safety factor of 1.18 against overturning.

The second case postulated by NUTECH was for a seismic event occurring during the normal transport of the TC loaded on the trailer. NUTECH stated that this case is enveloped by the handling case of $\pm 0.5g$ acting in the vertical, axial and transverse directions simultaneously. On page 8.2-21 of the TR, the statement is made that the calculated stress intensities for normal transport case are 17.9 ksi for the cask structural shell and 2.0 ksi for the trunnions, and furthermore that these were "conservatively used as the maximum seismic stresses in the load combination results" in Tables 8.2-13 and 8.2-14 of the TR. These tables do not reflect this statement, since Table 8.2-14 shows a shell stress of 31.1 ksi for load combination C1 which is dead weight loads, transportation handling loads, and seismic. If NUTECH had used 17.9 ksi for seismic as well as handling, they would have recorded at least 35.8 ksi. The staff evaluated this load condition and arrived at 31.2 ksi. This stress is lower than the allowable, as are the other stresses for the seismic case as shown in Table 3.13 of this SER.

<u>TC load combinations for Service Level C accident conditions</u>

Table 3.13 of this SER shows the results of two load combinations, as defined in Table 3.2-5b of the TR. The only difference between cases C1 and C2 is in the calculation of handling loads, i.e., during actual transport with ± 0.5 g acting in all three directions and during the transfer of the DSC into or out of the HSM. NUTECH does not present any results for the latter case, but it is clear that the higher loading case occurs during actual over-the-road transport between the spent fuel pool and the HSM pad. The NRC staff also included the stress intensities resulting from the DBT winds in this single load combination. As may be seen from Table 3.13, all the stresses are below the allowable levels for Service Level C conditions.

<u>Cask drop accident</u>

Section 3.3.4.2.3 (DSC Accident Conditions) of this SER presents a detailed discussion of the cask drop accidents postulated by NUTECH. This discussion includes the basis for the selection of the parameters and the assumptions used for the ANSYS finite element models. Because the previous discussion covered the DSC as well as the TC, it will not be repeated here.

Table 3.14 summarizes the results of all five drop orientations postulated by NUTECH. All the structural components of the TC are reported including: the cask shell, the top and bottom rings, and the top and bottom cover plates. The cask liner, bolts for the top cover plate, and the bottom sheet steel plates are also included for completeness. The temperature chosen by NUTECH to evaluate the material properties is 400^{o}F. Because the outside surface of the TC does not exceed 248^{o}F, this material temperature

Table 3.13

TRANSFER CASK STRESS ANALYSIS RESULTS
FOR ACCIDENT LOADS
Service Level C** Allowables

Cask Component	Stress Type	Stress (ksi) Handling	Seismic	DBT Wind	Load Comb C1***	Allowables*
Cask Shell	Pri Memb	0.5	.5	3.8	5.5	26.
	Memb + Bend	30.3	0.4		31.2	39.
Top Cover Plate	Pri Memb	-	.2	0.5	.7	26.
	Memb + Bend	6.3	.3		7.2	39.
Bottom Cover Plate	Pri Memb	-	.2	0.5	.9	22.4
	Memb + Bend	14.2	.3		15.8	33.7
Top Ring	Pri Memb	-	.2	-	.4	24.3
	Memb + Bend	-	-		-	36.5
Bottom Ring	Pri Memb	-	.2	-	.6	24.3
	Memb + Bend	-	-		-	36.5

* Allowables taken at 400°F

** No secondary stresses need to be evaluated according to the ASME Code for Service Level C.

*** The C1 load combination includes deadweight, seismic, handling loads, and DBT wind loads.

Table 3.14

TRANSFER CASK DROP ACCIDENT LOADS
Service Level D Allowables

Cask Component	Stress Type	Stress (ksi) Vertical Top Drop NUTECH	NRC	Vertical Bottom Drop NUTECH	NRC	Horizontal Drop With DW NUTECH	NRC	Corner Top NUTECH	NRC	Corner Bottom NUTECH	NRC	Allowables*
Cask	Pri. Memb	9.6	30.1	8.7	8.7	3.8	22.7	3.2	7.6	4.6	8.5	49.
Shell	Memb + Bend	10.2	33.6	-	-	15.5	21.9	6.6	7.5	13.9	11.3	70.
Cask	Pri. Memb	19.3	12.3	12.9	12.9	9.3	12.7	4.2	7.4	8.8	18.2	44.9
Liner	Memb + Bend			-	11.4	-	17.4	7.4	5.8	25.7	28.9	64.9
Top	Pri. Memb	25.2	24.2	-	-	12.2	17.3	2.1	7.5	-	-	48.7
Ring	Memb + Bend	-	46.4	-	-	-	22.6	2.9	12.6	-	-	73.1
Top	Pri. Memb	24.2	20.3	-	-	5.8	7.7	2.7	11.7	-	-	49.
3" Cover	Memb + Bend	-	22.5	3.7	3.7	-	8.0	14.1	14.1	-	-	70.
Bottom	Pri. Memb	-	-	22.9	5.8	5.8	6.4	-	-	-	33.1	44.9
2" Cover	Memb + Bend	14.4	14.4	10.2	10.2	-	11.6	-	-	33.1	28.6	64.9
Bottom	Pri. Memb	-	-	14.0	26.7	12.2	12.2	-	-	9.7	10.7	48.7
Ring	Memb + Bend	-	-	-	-	-	25.9	-	-	4.6	33.9	73.1
Bottom	Pri. Memb	-		11.1	11.1	5.8	6.4	-	-			44.9
1/4" PL	Memb + Bend	11.1	11.1	-	-	-	11.6	-	-	14.5	14.5	64.4
Bolts for Top Cover	Ave. Tension	-	-	-	-	-	-	27.1	29.7	-	-	77.0

* Allowables taken at $400^{o}F$

is conservative. None of the stresses reported in the SER summary Table 3.14 exceed the allowables for Service Level D Conditions.

It is interesting to note that the ANSYS models predict that the stresses will exceed the yield stress for all major structural components except the top cover plate. Thus the previous discussion concerning the selection of a 7% critical damping value is partially justifiable, by virtue of stress levels in excess of the yield stress. (See Sections 2.4 and 3.3.4.2.3 of the SER). As discussed in the structure analysis of the DSC, any drop height higher than fifteen inches shall require the retrieval and inspection of the DSC and its internals, in keeping with the guidelines of the ASME Code when using Service Level D allowables.

In docketed responses to NRC staff's questions, NUTECH presented results of a fourth accident condition, namely design basis tornado (DBT) generated missiles. The two missiles considered are those suggested in NUREG-0800 (Reference 24), a 3967 pound automobile, and a 276 pound eight inch diameter shell. TC stability, penetration resistance, and shell and end plate stresses were calculated and shown to be below the allowable stresses for Service Level D stresses. The results are given in Table 3.15.

<u>TC load combinations for Service Level D accident conditions</u>

Table 8.2-9b of the TR summarizes the enveloping load combination stress results for the TC drop accident. Table 3.2-5b of the TR defines the load cases for each load combination. In Revision 1 of the TR, only three cases were postulated to envelop all Service Level D conditions. They were: (1) vertical drop, (2) corner drop, and (3) horizontal drop. In each drop case the dead weight loads were combined with the drop loads. Table 3.15 of this SER shows the results and the material allowables at 400°F for the various materials specified in the drawings. These allowables are in most cases somewhat higher than given in the TR, but they represent the values for the specified materials. In docketed responses to staff questions, NUTECH summarized the results of DBT winds and DBT-generated missiles. The results of these three additional accident cases are also shown in Table 3.15. The results of these cases need to be incorporated in Revision 2 of the reference TR. In all cases the actual stress intensities are lower than

Table 3.15

TRANSFER CASK LOAD COMBINATIONS
FOR ACCIDENT CONDITIONS
Service Level D

Cask Component	Stress Type	Stress (ksi) Case D1 (Vert) NUTECH	Case D1 (Vert) NRC	Case D2 (Corner) NUTECH	Case D2 (Corner) NRC	Case D3 (Horiz) NUTECH	Case D3 (Horiz) NRC	DBT Wind* NUTECH	Massive Missile* NUTECH	Pen. Resist. Missile* NUTECH	Allowable** ksi
Cask	Pri. Memb	10.3	30.8	5.3	9.2	4.5	23.4	0.9	6.4	4.9	49.
Shell	Memb + Bend	11.0	34.1	14.7	11.8	16.3	22.4	2.9	20.5	30.3	70.
Top	Pri. Memb	25.4	24.4	2.3	7.5	12.4	17.3	NA	NA	NA	48.7
Ring	Memb + Bend	25.4	46.5	3.0	12.6	.1	22.6	NA	NA	NA	73.1
Top	Pri. Memb	24.4	20.3	2.9	11.7	6.0	7.7	0.	0.	0.	49.
Cover	Memb + Bend	24.4	22.5	14.7	14.7	.6	8.0	0.4	19.7	13.2	70.
Bottom	Pri. Memb	14.4	26.7	.4	-	12.6	12.2	NA	NA	NA	48.7
Ring	Memb + Bend	.3	26.7	.3	-	.3	25.9	NA	NA	NA	73.1
Bottom	Pri. Memb	23.1	5.8	.2	33.1	6.0	6.4	0.	0.	0.	44.9
Cover	Memb + Bend	15.7	10.2	34.4	28.6	1.3	11.6	0.3	17.5	22.2	64.4

* Data obtained from responses to NRC staff questions. This information needs to be incorporated in a Revision 2 to Reference 1.

** Service Level D Allowables

the allowables. Thus the TC meets the ASME Code for Service Level D conditions.

TC fatigue evaluation

Section C.4.2 of the TR presents an evaluation of the loading cycles of the TC to show that the six criteria associated with NC-3219.2 of the ASME Code are met. The NRC staff evaluated Section C.4.2 and concurs with NUTECH that all six ASME criteria are met; however, the margin is very small for the sixth criteria for mechanical loads. Using the WRC Bulletin No. 297 (Reference 34), the NRC staff calculated local stresses in the cask shell to be 48.4 ksi. These local stresses are due to normal mechanical handling loads. For the 5000 stress cycles selected by NUTECH, the allowable stress, Sa, is only 50. ksi. Thus even a small deviation in service cycles, material specification, or load could result in a situation where a licensee would be required to evaluate the cyclic operation according to Section NC-3219.2 of the ASME Code.

3.3.4.5 HSM Loading and Unloading

The actions and equipment associated with loading the DSC into the HSM are addressed in Section 5.1.1.6 of the TR. Unloading is addressed in Section 5.1.1.8 of the TR. An off-normal situation of a jammed DSC occurring during loading or unloading is addressed in Section 8.1.2.1 of the TR.

The TR states that approval of the procedure descriptions in Section 5 is not sought, that the included descriptions are for information and illustrate the feasibility and suitability of the prepared system. The TR states that actual proposed procedures would necessarily be the subject of a site-specific application. Equipment identified as required for the loading and unloading operations are: a trailer to hold and position the transfer cask which includes a skid positioning system and jacks for vertical position adjustment; a "porta-crane" to remove and/or replace the HSM door, the TC top cover plate, and the TC bottom ram access port cover; cover plate lifting cables; a cask restraint system to secure the TC to the HSM; an optical alignment system to align the TC with the HSM; a hydraulic ram system to push the DSC; and the trailer prime mover. In addition, tools and equipment would be required for removing and securing cover plate bolts/nuts and welding the HSM door in place (or cutting the welds for unloading). The TR does not include defined designs for any of the loading and unloading equipment.

The NRC staff reviewed the TR and concurs with the descriptive material of DSC loading into the HSM and unloading procedures. However, the staff considers that the design of the following have safety implications and therefore, since adequately defined designs are not included in the TR, such designs must be included in site-specific applications to use the NUTECH NUHOMS-24P system:

1. TC transfer skid and trailer, due to the potential for overturning and exceeding the limits on cask drop used in the accident condition analyses; the need to provide a stable base during DSC transfer operations; and the interfaces with the HSM, TC, ram system, and the cask restraint system.

2. Hydraulic ram system, due to the need to prevent excessive force on the DSC, provide a stable and linear motion, and interfaces with the TC, cask restraint system and/or trailer/skid and/or HSM.

3. Cask restraint system, due to the need to provide a secure mating of the TC with the HSM during DSC transfer and interfaces with the TC, HSM, trailer/skid and/or hydraulic ram system.

The staff reviewed the identification of normal, off-normal, and accident situations involving the TC to HSM and HSM to TC DSC loading and unloading operations and equipment and considers that they are adequate with regard to the DSC, TC, and HSM designs submitted in the TR. Additional off-normal and accident conditions may be appropriate to the equipment whose designs were not included (as noted above). These would include: determination of actual potential forces on the TC, DSC, and fuel rods if the actual trailer/skid design may permit a greater equivalent drop than used in the TR analyses; determination of the actual cask restraint system could produce overstresses on the attachment points on the HSM, TC, and any other connections; determination of the actual forces which might be exerted on the DSC by the hydraulic ram, as in a jammed condition or at either end of its travel; examination of the potential for failure of the ram to disengage from the DSC; and examination of the possibilities for TC movement relative to the HSM during DSC transfer.

3.3.4.6 Fuel Assemblies and Rods

10 CFR 72.130 briefly discusses the criteria for decommissioning of the ISFSI. Implicit in either decommissioning or in inspection for possible damage following a drop accident or a DSC containment leak is the ability of operators to remove the fuel assemblies from the DSC. 10 CFR 72.126 discusses the criteria for radiological protection including exposure control in Subpart (a) and effluent and direct radiation monitoring in Subpart (c) that must be followed during these operations. Normal and accident conditions are discussed below.

3.3.4.6.1 Normal Operating Conditions

Decommissioning, after completion of the storage period under normal conditions, is the only time when it would become necessary to remove the fuel assemblies from the DSC. The only possible problem that could be postulated as a result of the long-term storage in the horizontal condition is the sagging of the fuel rods due to creep, such that the fuel assemblies could not be removed from the DSC basket assembly.

An analysis of the potential creep and sag of the fuel rods was conducted. The fuel temperature decay was assumed to follow the ORIGEN-2 prediction for 10-year old fuel within the NUHOMS facility. The creep equation of M. Peehs et al. (Reference 35) was first used to determine whether creep of the fuel rods due to internal pressure could occur. The creep of the fuel rods for the total storage period was found to be less than 1%. This permitted the sag of fuel rods between grids to be calculated, since creep could be discounted. The sag was calculated using the standard beam equations for a tubular cross-section linearly loaded. The maximum sag was found to be 0.020 inch, which should not impede the removal of the fuel assemblies from the DSC.

3.3.4.6.2 Accident Conditions

Section 8.2.5.4 of the TR discusses recovery from a drop accident. Section 10.3.2.9 of the TR discusses fuel assembly retrieval and inspection following a cask accident. "Recovery" implies the removal of the spent fuel assemblies (SFA) from the DSC, i.e., it must be possible to easily extract the SFAs from the guide sleeves of the fuel basket. This is required by 10 CFR 72.122(l) and 72.126(a)(5). Both sections of the TR specify that for a cask drop of less than 15 inches, no inspection is required. However, for drop heights of 15 inches or greater, the transfer cask must be returned to the plant's fuel building where the DSC will be cut open and inspected for damage.

As noted in Section 3.3.4.6, radiological protection of the workers must be provided to assure that an aerosol of oxidized fuel particulate is not inhaled during inspection and removal operations. Fuel particulate can form if the spent fuel oxidizes at elevated temperature, due to air ingress into the DSC and the availability of failed cladding that could expose fuel to that air. The TR states that this work will be performed under the site's standard health physics guidelines for handling potentially contaminated equipment. These procedures <u>may</u> require personnel to work using respirators or supplied air. However, the staff finds that these precautions <u>must</u> be taken when the DSC is opened to protect the health of the operations personnel.

NUTECH used the finite element code, ANSYS, to predict the maximum elastic deflection of the spacer disc ligaments for the 75 g horizontal drop. They also postulated a three hinge collapse mechanism for plastic deformation. These deformations were 0.050 and 0.022 inches respectively. By summing these two deformations, an estimate of potential interference or binding between the guide sleeve and the SFA was predicted to be 0.072 inches, which is much less than the clearance available. Therefore, there should be no possibility of binding for the worst assumed case.

The NRC staff also compared NUTECH's maximum deceleration level (75 g) with a minimum predicted deceleration level required to yield B&W 15x15 fuel assemblies. Reference 27 predicts that 101 g deceleration is necessary to cause yielding of the fuel rods for a horizontal drop. The same reference also considered what vertical deceleration would be necessary to cause axial buckling. The level was 147 g, considerably higher than the 75 g level used by NUTECH in their design criteria. Thus there is considerable margin both with regard to the minimum deceleration levels required to cause yielding or buckling of the fuel rods as well as the clearance available between the fuel rods and the guide sleeve. The NRC staff concurs with NUTECH's statement that the SFAs could be extracted from the fuel basket following an accidental drop involving 75 g or less deceleration.

4.0 THERMAL EVALUATION

4.1 SUMMARY AND CONCLUSIONS

The staff has reviewed the thermal features of the NUHOMS-24P design and finds that they conform to appropriate sections of 10 CFR 72 and are acceptable.

4.2 DESCRIPTION OF REVIEW

4.2.1 Applicable Parts of 10 CFR 72

The thermal analysis was reviewed for conformance to 10 CFR 72 Subpart F. For normal and accident conditions 10 CFR 72.122(h) requires that the fuel cladding be protected against degradation and gross rupture. Sections 10 CFR 72.122(b,c) require that the system design provide protection against environmental conditions, natural phenomena and fires.

4.2.2 Review Procedure

4.2.2.1 Design Description

The NUHOMS system provides for the horizontal storage of irradiated fuel in a dry, shielded canister (DSC), which is placed in a concrete horizontal storage module (HSM). Decay heat is removed from the fuel by conduction and radiation within the DSC and by convection and radiation from the surface of the DSC. Natural circulation flow of air through the HSM and conduction of heat through concrete provide the mechanisms of heat removal from the HSM.

Spent fuel assemblies are loaded into the DSC while it is inside a transfer cask in the fuel pool at the reactor site. The transfer cask containing the loaded DSC is removed from the pool, dried, purged, backfilled with helium and sealed. The DSC is then placed in a transfer cask and moved to the HSM. The DSC is pushed into the HSM by a horizontal hydraulic ram.

The DSC is constructed from stainless steel with an outside diameter of 67.25 inches, a wall thickness of 0.625 inches and a length of 186 inches. Within the DSC, there is a stainless steel basket consisting of twenty-four square cells. An intact PWR spent fuel assembly is loaded into each cell for a total of twenty four assemblies per DSC. Spacer disks are used for structural support. The DSC has double seal welds at each end and rests on two steel rails when placed in the HSM.

The HSM is constructed from reinforced concrete, carbon steel and stainless steel. Passageways for air flow through the HSM are designed to minimize the escape of radiation from the HSM but at the same time to permit adequate cooling air flow. Decay heat from the spent fuel assemblies within the DSC is removed from the DSC by natural draft convection and radiation. Air enters at the bottom of the HSM, flows around the canister and exits through the flow channels in the top shield slab. Heat is also radiated from the DSC to the inner surface of the HSM walls where again, natural convection air flow removes the heat. Some heat is also removed by conduction through the concrete.

The NUHOMS system utilizes a transfer cask (TC), transporter, skid and horizontal hydraulic ram. The transporter, skid and horizontal hydraulic ram are not affected by the thermal analysis. During transport and vacuum drying of the fuel in the DSC, heat is removed by conduction through the TC.

4.2.2.2 Acceptance Criteria

Temperature limits for dry storage were developed by I.S. Levy, et al, in Reference 36. The NRC staff has reviewed and accepted the temperature limits developed in Reference 36. These limits are in the form of a family of generic limit curves of recommended maximum allowable initial cladding temperature as a function of cladding hoop stress. Fuel cooling time at the beginning of dry storage is a parameter. Based on the results presented in Reference 36, NUTECH derived a long term fuel cladding temperature storage limit of $340^{o}C$. This limit was derived by applying the methodology of Reference 36 for a range of rod fill pressures (up to 480 psig), burnups, (up to 40,000 MWD/MTU) and ten years or less cooling time. Since it is possible to exceed the fuel performance limits of Reference 36 (while meeting the 340^{o} criteria) for higher burnup fuel, higher initial rod fill pressure fuel and/or fuel with cooling time greater than ten years, additional restrictions are required. To be stored in NUHOMS-24P dry storage a fuel assembly must have the following characteristics:

1. Maximum burnup less than 40,000 MWD/MTU
2. Maximum initial fill gas pressure less than 480 psig
3. Generated less than 660 watts of decay heat at ten years cooling time.

This limit is more conservative than the 380^{o}C maximum temperature used for the NUHOMS-07P (Reference 22) design, and is acceptable for normal operating conditions. Meeting these acceptance criteria assures that the requirements of 10 CFR 72.122(h) are satisfied.

Reference 37 establishes that no rods have failed in inert gas exposures up to 570^{o}C, and rods forced to failure required temperatures from 765 to 800^{o}C to produce ruptures. An accident temperature limit of 570^{o} is the acceptance criteria for accidents based on the above evaluation.

The thermal analysis review addresses the correctness of the reported concrete temperatures, and also the thermal input for stress analysis. Acceptability of the concrete temperatures relative to ACI-349-80 is addressed in Section 3 of this SER.

4.2.2.3 Review Method

The TR thermal analysis was reviewed for completeness, applicability of the methods used, adequacy of the key assumptions and correct application of the methods. The NUTECH thermal analysis was performed primarily with the HEATING-6 (Reference 38) computer program. HEATING-6 is a part of the Oak Ridge National Laboratory SCALE package and is an industry standard code for thermal analysis. Representative input and output was reviewed to establish that the code use was appropriate and that the results were reasonable. Independent calculations were performed to check other portions of the analysis that did not use the HEATING-6 code. This includes the natural convection cooling calculation which determines the magnitude of the air flow through the HSM. Since the heat flux through the DSC surface is significantly increased for the NUHOMS-24P design compared to the NUHOMS-07P design, the ability to remove heat by air cooling is particularly important. An independent determination of the form losses and friction pressure drop, together with a balancing of the buoyancy and flow loss, confirmed the adequacy of the NUTECH analysis.

4.2.2.4 Key Design Information and Assumptions

The key assumptions made in the NUTECH thermal analysis are listed below.

1. The total heat generation rate for each fuel assembly is less than or equal to 660 Watts. This value is based on ORIGEN calculations and data published in the literature. All heat is assumed to be generated in the fuel region.

2. Each dry storage canister contains 24 intact PWR assemblies.

3. A factor of 1.08 to account for axial power peaking in the fuel during operation was assumed for thermal analysis inside of the DSC.

4.3 DISCUSSION OF RESULTS

The following discussion covers the analytical methods used by NUTECH for normal, off-normal, and accident conditions for the HSM, DSC, and TC, which were evaluated by the NRC staff. It also covers independent analyses that were performed by the staff.

4.3.1 Analytical Methods Used by NUTECH

The TR thermal analysis was done for the horizontal storage module, the dry shielded canister in the horizontal storage module and the dry shielded canister in the transfer cask. The HEATING-6 computer program was used to perform the major portion of the thermal analysis. HEATING-6 solves steady state and/or transient heat conduction problems in one, two or three dimensional Cartesian or cylindrical coordinates.

Air temperatures within the HSM were first established by a natural circulation cooling analysis. Steady-state circulation flow will occur when the buoyancy forces are balanced by friction and form loss forces. Flow areas and loss factors were designed to allow sufficient flow to maintain the desired temperature difference between the inlet and outlet air temperature. An independent analysis, including determination of friction and form losses, was performed by the staff to confirm the NUTECH results.

Thermal analysis of the HSM is performed to obtain the temperature of the outside surface of the DSC and the temperature distribution of the concrete module, given a heat flux across the canister surface corresponding to the spent fuel heat generation rate. Once this temperature is established, detailed analysis of the temperature distribution within the canister is done. A thermal analysis of the canister within a hypothetical TC is done to determine the peak fuel clad temperatures during normal and off-normal situations. The vacuum drying operation and loss of liquid shielding accident are also analyzed.

A two dimensional Cartesian model is used to represent the HSM for HEATING-6 analysis. The HSM is assumed to be infinitely long with the axial average heat flux determined over the DSC length. Only one-half of the module is modeled in HEATING-6, since symmetry exists about the vertical centerline. Both a single free standing unit and a 2 x 10 array of HSMs were considered. The model includes the 3 feet thick concrete ceiling, concrete side walls and the floor. The external surfaces of the side walls are assumed to be adiabatic for interior walls centered in a group of modules, or to be exposed to ambient conditions for exterior walls or for modules with no DSC stored in adjacent locations. The floor was taken as seven feet of concrete with a constant temperature at the bottom.

The DSC located within the module is modeled as a cylindrical shell represented as a series of 40 small rectangular slabs. The total surface area of these slabs is equal to the surface area of the canister. Heat transfer by convection and radiation is considered in the air gap between the canister and the interior surface of the module. Convection heat transfer at the outer surface of the module ceiling is included, as is solar heat loading on the outer surface of the module ceiling. The heat source consists of 24 PWR assemblies, each with an assumed heat generation rate of 660 Watts.

Temperatures within the DSC are determined using a second HEATING-6 model. A two dimensional Cartesian model is used to represent the DSC and the internal helium, stainless steel sleeves and fuel regions. The surface heat flux is based on the 144 inch active fuel length and a 1.08 axial peaking factor. All of the heat is conservatively assumed to be generated in the fuel regions. The regions representing the DSC wall are at fixed temperatures determined from the HSM HEATING-6 analysis. An effective thermal conductivity was determined for the fuel regions based on experimental results at E-MAD (Reference 39). These results were shown to be in agreement with the Wooten-Epstein correlation which has been previously used for TC thermal analysis.

Temperature profiles for the DSC within the transfer cask were determined from the steady-state heat conduction solution for a composite cylinder with combined radiation and convection heat transfer at the outer surface of the TC. Radiation, conduction, and convection were modeled in the air gap between the DSC and the TC.

4.3.2 HSM and Internals

4.3.2.1 Normal Operating Conditions

A total of three cases were considered for normal operating conditions based on the temperature of the air at the inlet of the module. These are: (1) entering air at $0^{o}F$ representing "minimum normal conditions," (2) entering air at $70^{o}F$ representing "normal conditions," and (3) entering air at $100^{o}F$ representing "maximum normal conditions." The method of calculating concrete temperatures is conservative and acceptable. Satisfaction of the limiting condition for operation of a $60^{o}F$ maximum air temperature rise on exit from the HSM gives a reasonable degree of assurance that adequate cooling is achieved.

Temperature gradients through the walls and roof were determined from the HEATING-6 results. These are acceptable temperature gradients for use in the reinforced concrete structural analysis.

4.3.2.2 Off-Normal Conditions

The off-normal conditions considered were an inlet temperature of -40°F representing extreme winter minimum and 125°F representing extreme summer maximum. The concrete temperature on the inside surface of the HSM reaches a maximum of 215°F for the extreme condition of 125°F ambient temperature. The results are acceptable for use in structural and concrete integrity evaluations.

4.3.2.3 Accident Conditions

The total blockage of all air inlets and exits was analyzed as the accident case. Adiabatic heatup of the various components was assumed, with the HSM providing the slowest heatup rate. Adiabatic heating starting at the 125°F inlet temperature condition is the limiting case for maximum concrete and fuel clad temperatures. The resulting concrete temperatures are reasonable and acceptable for use in the thermal loads analysis. Since it is assumed that the blockage will be cleared within 48 hours, heatup was calculated over this period.

4.3.3 DSC and Internals

4.3.3.1 Normal Operating Conditions

The normal operating conditions at 70°F and 100°F ambient air inlet temperature were analyzed for the DSC and internals. HEATING-6 input and output for the 70°F case and the corresponding HSM run were reviewed. No errors were detected. Trends and magnitude of the resulting temperature distributions are reasonable. Maximum fuel cladding temperatures were calculated and found to be less than the 340°C limit for the 70°F ambient temperature case. Maximum fuel cladding temperature was 349°C for the 100°F "maximum normal condition."

4.3.3.2 Off-Normal Conditions

The off-normal condition considered was the 125°F ambient inlet air temperature. HEATING-6 calculations were performed which yielded a maximum fuel clad temperature of 353°C compared to the acceptance criterion of

570^{o}C. Results from this case were conservatively used to determine thermal loadings for purposes of the staff review.

4.3.3.3 Accident Conditions

Temperature distribution within the DSC was determined for the case of all air inlets and exits blocked for a 48-hour period. A steady-state temperature distribution was assumed within the DSC, since its heatup rate is faster than that of the HSM. The resulting temperature distribution is acceptable for use in determining thermal loads. Maximum fuel cladding temperature was calculated to be 403^{o}C, which is below the 570^{o}C accident limit.

4.3.4 Transfer Cask and Fuel

During loading, evacuation and transport to the HSM, the DSC is located within the TC. In this case, the inside surface temperature of the transfer cask was determined by calculating the steady-state temperature distribution through the cask which was modeled as a series of cylindrical annular regions. The surface temperature of the DSC was then determined from the conduction, convection and radiation heat losses from the canister to the cask. Two cases were considered: the top half of the DSC which is not in contact with the TC, and the bottom half which is assumed to contact the TC over its entire surface. This models the situation where the DSC and TC are in the horizontal position for transport.

Two normal, two off-normal and one accident conditions were analyzed. Normal minimum and maximum ambient air temperatures of 0^{o}F and 100^{o}F were analyzed, along with -40^{o}F minimum and +125^{o}F maximum ambient air temperature off-normal cases. The accident condition analyzed was the loss of liquid neutron shield. The vacuum drying operation with the evacuated DSC cavity was also analyzed since this is a limiting condition. Maximum fuel clad temperature was 421^{o}C for the loss of liquid neutron shield accident, which is within the acceptance criteria of 570^{o}C. For the evacuated DSC, the maximum fuel cladding temperature was 410^{o}C. When the DSC is not evacuated, the maximum temperature will be significantly lower due to the higher effective thermal conductivity within the DSC. Since the evacuated condition is short term, the acceptance criterion of 570^{o}C is satisfied.

5.0 CONFINEMENT BARRIERS AND SYSTEMS

5.1 SUMMARY AND CONCLUSIONS

The staff has reviewed the features of the NUHOMS-24P design which provide confinement of radioactive material and, specifically, protection of the fuel rod cladding. The review was directed at two aspects of the design: (1) the mechanical integrity of the DSC and (2) the long term behavior of cladding in an inert helium atmosphere.

As a result of this review, the staff concludes that the NUHOMS design conforms to applicable parts of 10 CFR 72.122(h). Confinement is assured by a radiographic inspection of the longitudinal full penetration weld and the bottom circumferential weld, radiographic inspection of the two welds for the bottom plug, and helium leak testing and dye penetrant testing of the welds for the top lead plug and top plate, respectively. The acceptance leak rate for helium leak testing is less than or equal to 10^{-4} atm - cc/sec. The less rigorous dye test procedure used for the top end plate can be considered acceptable due to the helium leak testing of the inner weld, and due to the fact that two seals are used instead of one, as for the longitudinal weld. Radiographic inspection of the top plug welds is not feasible due to the fact that irradiated fuel will already be installed before the tests can be made.

The staff considered three potential mechanisms for the deterioration of the integrity of fuel rods. The first was potential failure of the cladding by the diffusion controlled cavity growth mechanism. The staff determined that the area of decohesion was less than 4 percent, not high enough to cause any concern. The second mechanism examined was creep or sag of the fuel cladding. It was found to be 0.020 inches, much less than the clearance available for removal of the rods. The third mechanism examined was oxidation of the fuel during the dry-out period. Cladding strain was determined to be much less than 1% for postulated fuel oxidation of defective fuel rods. The staff concludes that the NUHOMS design has provided sufficient means to assure that the fuel cladding is protected against degradation.

5.2 DESCRIPTION OF REVIEW

5.2.1 Applicable Parts of 10 CFR 72

Paragraph (1) of Section 72.122(h) is pertinent to storage of spent fuel in NUHOMS. It requires that "spent fuel cladding must be protected during storage against degradation that leads to gross ruptures" and "that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage." Paragraphs (2) and (3) of that section relate to underwater storage of fuel and to the off-gas and ventilation systems, respectively, and are not applicable to this review. Paragraphs (4) and (5) deal with monitoring and handling and retrievability operations, respectively, and are addressed elsewhere in this document.

5.2.2 Review Procedure

5.2.2.1 Design Description

The NUHOMS design provides protection of the fuel cladding by storing fuel assemblies in an inert atmosphere of helium. The helium atmosphere is first established after the fuel is loaded into the DSC. The loaded DSC is welded closed, and the weld tested with the dye penetrant method, drained of water by pressurizing the cavity with helium, and evacuated. A vacuum of 3 Torr is drawn on the DSC cavity for not less than 30 minutes. This stable vacuum pressure of 3 Torr will result in an inventory of oxidizing gases in the cavity of less than 0.25 volume %. Then the DSC is back-filled with helium to an unspecified pressure for purposes of helium leak-testing of the primary weld.

After the end weld is checked for leaks, the DSC is again evacuated and backfilled with helium at 2.5 ± 2.5 psig. The evacuation lines are sealed and the top end cover is welded to the DSC. The field welds and the shop welds on the bottom and along the longitudinal seam are expected to maintain the internal helium atmosphere intact for the full time of storage of the DSC in the HSM. No device (e.g., gauge) is made part of the system for verifying the maintenance of the helium atmosphere.

5.2.2.2 Acceptance Criteria

The confinement barriers and systems design will be considered acceptable if the TR shows that: (1) there is a high likelihood that the DSC internal helium atmosphere will remain intact; (2) there is no long term cladding degradation mechanism in a helium atmosphere which could cause significant degradation or gross ruptures; and (3) there is insufficient time for cladding or fuel degradation during cask dry-out or off-normal behavior that could pose operational problems with respect to the removal of fuel from storage.

5.2.2.3 Review Method

The NRC staff review of the TR was directed at two aspects of the design: (1) the mechanical integrity of the DSC; and (2) the long term behavior of the cladding in an inert environment. The review was also directed at the impact of cask dry-out and off-normal behavior on fuel removal.

The staff reviewed DSC integrity from the point of view of weld quality and inspections, adequacy of leak check methods on welds, other leakage paths, and long term helium migration. Reviewers also checked the calculated stresses in the DSC under normal, off-normal and accident conditions in order to verify that they are in the acceptable range. Cyclic fatigue of the DSC was also reviewed.

The staff evaluated cladding degradation by reviewing the pertinent technical literature in order to identify known and postulated mechanisms of gross failure of fuel in inert atmosphere. Based on the literature search, calculations were performed of postulated failures by the mechanism of diffusion controlled cavity growth using a conservative set of assumptions. This was the only failure mechanism considered likely under the NUHOMS storage conditions. The staff also evaluated the possible long term creep and sag of the spent fuel under these storage conditions since creep or sag could impact on removal of the fuel from storage.

5.2.2.4 Key Assumptions

The assumptions made in review of the TR regarding confinement systems are listed below.

1. The diffusion rate of helium through the DSC is no greater than 10^{-8} g-moles/year at nominal design conditions and as much as 10^{-5} g-moles/year at accident conditions, as stated by the applicant.

2. The values used for various properties of the zircaloy cladding in the analysis of diffusion controlled cavity growth (DCCG) and the DCCG mathematical model lead to a very conservative estimate of degradation.

3. The fuel cladding is protected at steady state temperatures of up to 340°C and on short term transients up to 570°C if in an inert atmosphere.

5.3 DISCUSSION OF RESULTS

The following evaluation covers DSC integrity, potential for long term fuel rod failure, potential cladding creep or sag, and potential oxidation of fuel during cask dry-out or off-normal behavior.

5.3.1 DSC Integrity

In the review of the structural analysis of the DSC, the staff found the design acceptable.

The commitment to design and fabricate the DSC's bottom circumferential and longitudinal welds to the ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsections NB and NF for Class 1 components provides assurance of leak-tightness at these locations.

The top end plate welds are made in the field. The TR states that end plate welds are to be ultrasonically tested, or tested by dye penetrant method in accordance with the ASME Code as stated above. The dye penetrant method of testing reveals information about the weld surface only, hence a weld tested by this method does not yield as much information as the radiographic method. However, the top end welds cannot be radiographed because irradiated fuel will be in place before these tests can be performed. The staff finds this procedure acceptable because the primary welds are first leak-tested by a helium detector and because the two top end plates represent a double seal.

Since the DSC contains no penetrations for sampling or gauges, there are no diffusion or leakage paths for helium other than the welds and the primary metal. Presuming the weld integrity to be equivalent to that of the parent metal, the staff also concludes that diffusion is not a potential mechanism to permit escape of helium and ingress of oxygen.

The staff concludes that DSC design and fabrication methods will result in a high likelihood that the internal DSC helium atmosphere will remain intact over its storage lifetime.

5.3.2 Potential for Long Term Fuel Rod Failure

Calculations were performed on the potential failure of the cladding by the diffusion controlled cavity growth mechanism, which is the only mechanism postulated to occur under the NUHOMS storage conditions. The method used is described in Appendix A of the SER for the initial NUHOMS TR (Reference 42). Following the earlier assumptions, a constant ambient temperature of 70^{o}F was used in the analysis. The temperature dependence of grain boundary decohesion is established using the temperature decay curve provided in the current TR in Figure 8.1-28. Since the data terminates at about 10 years from the beginning of storage, it was conservatively assumed that the temperature would remain constant thereafter, that is, for the remaining ten years.

Since the values of all parameters in the equation given in Appendix A of the initial SER, except for the exponential term involving the temperature decay, were identical, the calculation reduces to a comparison of the integrals of the two exponential terms. The difference was found to be insignificant. Therefore, the area of decohesion at the end of the twenty-year storage life is the same as that found previously, less than 4 percent. Hence, the requirements of 10 CFR 72 Section 72.122(h) are met.

5.3.3 Potential Cladding Creep or Sag

Cladding creep or sag could impact on the removal of fuel from storage. The potential for cladding creep was analyzed first, using the creep equations of Peehs et al. (Reference 35). The temperature profile was conservatively broken down into five or ten year constant temperature periods to estimate the cladding creep. For stresses of 80 or 100 MPa, the creep was found to be less than 1 percent. The sag of the cladding was then calculated using a standard linearly loaded beam formula. If no credit for inertia is taken for the fuel itself, the maximum sag was found to be 0.015 inches. If the fuel also resists bending, then the maximum sag was found to be 0.006 inches. For this analysis, the inter-grid distance was assumed to be 24 inches. For an inter-grid spacing of 26 inches, the maximum sag was found to be 0.020 inches. Since the space available between the fuel rods and the DSC basket is much greater than the 0.020 inches, it is not anticipated that sag would impede the removal of the fuel assemblies.

5.3.4 Potential Oxidation of Fuel During Cask Dry-Out or Off-Normal Behavior

The NUTECH thermal analysis, with which the staff concurs, indicates that cask dry-out or off-normal behavior could involve a temperature excursion of up to 375°C in 48 hours. It was conservatively assumed that air was present for the entire time period. The temperature profile given in Figure 8.2-12 of the TR was divided into eight six-hour periods. For each time segment, the oxidation rate was determined. Oxidation front velocity data were taken from Einziger and Cook (Reference 40) and Kohli et al. (Reference 41). The maximum length of fuel oxidized was found to be 2.1 inches for fuel rods containing defects. The cladding strain was estimated to be much less than 1 percent, so that defect extension or fuel powdering is not anticipated. However, as noted previously, radiological precautions must be taken to protect personnel during operations in which fuel could be exposed.

6.0 SHIELDING EVALUATION

6.1 SUMMARY AND CONCLUSIONS

The methods used for designing the NUHOMS-24P shielding, and the resultant shielding design, are similar to the design for the NUHOMS-07P system containing seven irradiated fuel assemblies which has been reviewed in Reference 42. The neutron and gamma ray design basis source strengths are slightly higher than the smaller capacity design; however, the basic shielding design of the NUHOMS system readily accommodates the slightly higher source strengths. The results of experimental measurements from a real cask similar to the NUHOMS-24P design also provides a benchmark for the shielding design methods.

The NUHOMS shielding design conforms to the ALARA requirements of 10 CFR 72 and to acceptable shielding methods and practices. The staff concludes, based on the TR analysis, that the shielding is designed to ensure that the surface dose rates satisfy the criteria established in the TR subject to the following conditions:

1. No more than twenty-four (24) fuel assemblies meeting the specifications discussed in Chapter 12 of this report are contained in the DSC, or

2. The maximum neutron source strength per DSC is $<3.715\times10^{9}$ neutrons/sec, and the maximum gamma ray source strength per DSC is $<3.85\times10^{16}$ MeV/sec (1.11×10^{17} gammas/sec).

6.2 DESCRIPTION OF REVIEW

6.2.1 Applicable Parts of 10 CFR 72

The applicable part of 10 CFR 72 regarding the shielding evaluation of the NUHOMS-24P is the requirement of 10 CFR Part 72.3 related to ensuring that occupational exposures to radiation are as low as reasonably achievable, and 10 CFR Part 72.126 relating to criteria for radiological protection.

6.2.2 Review Procedure

6.2.2.1 Design Description

The principal design criterion for the NUHOMS-24P module is to limit the average external surface contact dose (gamma ray and neutron) to site workers to less than 20 mrem/hr. The design criteria during handling and transfer operations is to limit contact dose to less than 200 mrem/hr.

6.2.2.2 Acceptance Criteria

The shielding design is acceptable if the shielding evaluation results provide reasonable assurance that the design criteria indicated above are satisfied in the NUHOMS-24P system design.

6.2.2.3 Review Method

The TR shielding analysis was reviewed. Independent or confirmatory calculations were not performed. Rather, an assessment of the appropriateness of the shielding methods was made. Checks of the results for consistency with the similar NUHOMS-07P system were made as well as checks for self-consistency of the results.

6.2.2.4 Key Assumptions and Computer Codes

The major assumption in the shielding design is the source strength specification for the fuel to be stored, which is described in Section 2.2 of this report.

Two computer codes were used in the shielding analysis reported in the TR. ANISN, a one-dimensional discrete ordinates code, was used to estimate the neutron and gamma-ray dose rates at the outer HSM wall, the DSC top and bottom cover plate surfaces, and the TC outer surfaces. The ANISN calculations used the CASK cross section library, which includes 22 neutron energy groups and 18 gamma ray energy groups. QAD-CGGP, a three dimensional point kernel shielding code, was used for the gamma ray shielding analysis of the HSM door, the DSC and cask end sections, the DSC-cask annular gap and the HSM air vent penetrations.

6.3 DISCUSSION OF RESULTS

The following evaluation covers source specifications, HSM dose rates, and cask-DSC dose rates.

Source Specifications

The neutron and gamma radiation sources include the design basis irradiated fuel and activated portions of the fuel assemblies. The shielding analysis includes both primary neutrons and gamma-rays from these sources as well as an approximation of the secondary gamma rays from interactions of neutrons with the DSC and shielding materials. A more rigorous estimate of secondary gamma rays is included in the ANISN calculations, while in the QAD-CGGP calculations, secondary gamma rays have been approximated by increasing the primary gamma ray source strength. This approximation was justified by comparison with experimental data to confirm that the calculational results are giving conservative results.

The shielding is designed for a neutron source strength of 3.715×10^9 neutrons per second and a gamma ray source strength of 3.85×10^{16} MeV per second. Any combination of fuel irradiation time, burnup, specific power, enrichment, post irradiation time, and selection of assemblies to be loaded into a DSC is acceptable for storage in the HSM if the neutron and gamma ray source strengths do not exceed these criteria. The design basis is derived from a burnup analysis of 4 weight percent ^{235}U initial enrichment PWR fuel irradiated to an average fuel burnup of 40,000 MWd/MTHM at a specific power of 37.5 MW/MTHM, and a post irradiation time of ten (10) years. Irradiated fuel assemblies that meet these criteria are bounded by the neutron and gamma ray sources used in the shielding analysis.

The neutron and gamma ray source energy spectrum used for the shielding analysis were derived from an ORIGEN burnup calculation and are reported in TR Tables 7.2-1 and 7.2-2, respectively.

HSM Dose Rates

A dose rate of 7 mrem/hr for the HSM top and lateral surfaces was calculated using the ANISN discrete ordinates transport code and a cylindrical model of the DSC/HSM. This contact surface dose rate is less than the design criteria of 20 mrem/hr. Dose rates of 45 and 24 mrem/hr were calculated at the HSM door and 2 meters from the door, respectively, using a combination of results from an ANISN slab calculation modeling the ends of the DSC and a QAD calculation of the shielding effectiveness of the HSM door. These dose rates are less than the design criteria contact dose rates of 200 mrem/hr for workers performing operations.

Dose rates at the air ventilation inlets and outlets were calculated using the QAD-CGGP code and a manual albedo method to account for radiation streaming through the air ducts. Dose rates of 96 mrem/hr and 63 mrem/hr were calculated at the center of the air inlet and outlet ducts, respectively. A dose rate of 3600 mrem/hr at the air outlet duct was calculated for the accident condition of the shielding cap being removed.

Cask-DSC Dose Rates

Maximum cask surface dose rates were calculated to be 168 mrem/hr on the radial surfaces of the cask using a cylindrical model of the cask-DSC in the ANISN code. A maximum dose rate of 191 mrem/hr was calculated in the cask-DSC annulus. These dose rates are less than or equal to the design criteria dose rates of 200 mrem/hr.

The NRC staff concludes that, based on the material supplied in the TR, the NUHOMS-24P design meets the design criteria as stated in the TR.

7.0 CRITICALITY EVALUATION/BURNUP

7.1 SUMMARY AND CONCLUSIONS

Criticality safety cannot be assured for the NUHOMS-24P system under the conditions that have previously been considered for licensing of independent spent fuel storage installations using a dry storage concept, i.e., criticality safety is assured assuming loading of system with unirradiated fuel of maximum initial enrichment with optimal interstitial water density. Additional measures have been considered to provide assurance of nuclear criticality safety for the NUHOMS-24P system, including:

1. evaluation of fissile isotope concentrations and stable fission product absorbers in irradiated fuel (i.e., burnup credit)

2. modification of operational procedures to ensure fuel loading in a moderator solution with sufficient soluble poisons to assure nuclear criticality safety

Since consideration of the generic issue of the use of allowance for burnup credit in the safety evaluation of Independent Spent Fuel Storage Installations has not yet been completed by the Nuclear Regulatory Commission, allowance for burnup credit has not been accepted by NRC staff as a basis for the safety evaluation of the NUHOMS-24P system. However, nuclear criticality safety can be assured in the NUHOMS-24P design if the DSC is filled with borated water ($\geq$ 1810 ppm boron) during loading and unloading operations and if the irradiated fuel assemblies are loaded with the DSC submerged in a borated-water PWR spent fuel pool. The maximum effective reactivity under these conditions with optimal moderator density and 4% unirradiated fuel has been determined to be < 0.98.

Since unirradiated fuel will not be loaded into the DSC, there will be a reactivity safety margin realized, although it has not been quantitatively evaluated in this safety evaluation because of the outstanding research issues. Thus, it is concluded, based on the analysis presented in the TR and response to questions, that the NUHOMS-24P system is designed to provide assurance of nuclear criticality safety. The NUHOMS-24P system is determined to be in compliance with 10 CFR 72.124 as long as the following conditions are met:

1. Irradiated fuel initial enrichment equivalent[a] $\leq$ 1.45 weight percent ^{235}U;

2. DSC filled with borated water ($\geq$ 1810 ppm boron) and submerged in borated-water PWR spent fuel pool during loading and unloading operations;

3. The irradiated fuel assemblies are not more reactive than the design basis 15x15 rod fuel assemblies;

4. Borated water is drained from the DSC within 50 hours of being removed from the spent fuel pool;

7.2 DESCRIPTION OF REVIEW

7.2.1 Applicable Parts of 10 CFR 72

The applicable part of 10 CFR 72 regarding nuclear criticality safety is the requirement of 10 CFR 72.124.

a The initial enrichment equivalent of an irradiated fuel assembly is the ^{235}U enrichment of unirradiated fuel assemblies which would exhibit the same reactivity as the irradiated fuel assembly.

7.2.2 Review Procedure

7.2.2.1 Design Description

The NUHOMS-24P DSC is designed to provide nuclear criticality safety during wet loading and unloading operations. After fuel loading and DSC drying, the irradiated fuel assemblies are not moderated and therefore criticality safety is assured for subsequent operations and configurations.

The moderator density conditions are an important factor for criticality safety during fuel loading into the DSC and if removal of fuel assemblies from the DSC is necessary for any reason. The DSC is initially filled with borated water ($\geq$ 1810 ppm boron) prior to placement in the spent fuel pool, which is also borated. The loaded DSC is removed from the fuel pool and the DSC cavity is subsequently dried and backfilled with helium as part of the DSC closure operation. Flow paths are provided in the DSC design to ensure that the DSC draining or refilling process is a controlled and determinate process.

During transfer and storage of the canistered spent fuel, ingress of water into the helium-filled DSC is precluded by its welded seals and its presence in the TC and HSM, respectively. To ensure criticality safety, the NRC staff also limits its approval of the use of this design to storage on flood-free sites, eliminating water intrusion into the DSC as a credible event.

Further, for loading or unloading of the DSC, any licensee shall have in place fuel selection and verification, fuel identification and verification, borated water measurement and verification, and fuel handling procedures.

7.2.2.2 Acceptance Criteria

The requirement of 10 CFR 72.73 is determined to be satisfied if the 95% probability/95% confidence (95/95) effective multiplication factor for the NUHOMS-24P design is demonstrated to be less than 0.95, and the 95/95 effective multiplication factor for unirradiated fuel, which is not to be stored, is demonstrated to be less than 0.98.

7.2.2.3 Review Method

The criticality analysis presented in the TR and supplementary response to questions was reviewed. Independent and confirmatory calculations were also performed to verify important sensitivities in the criticality analysis.

7.2.2.4 Key Factors/Assumptions

Key factors and assumptions in the criticality safety analysis were:

factors

1. The maximum initial fuel enrichment is 4.0 wt. % ^{235}U; note, however, that unirradiated fuel will not be loaded in the DSC.

2. The DSC is filled with borated water ($\geq$ 1810 ppm boron) and submerged in a borated-water spent fuel pool ($\geq$ 1810 ppm boron) during loading and unloading operation,

3. Only irradiated fuel assemblies with an initial enrichment equivalent $\leq$ 1.45 wt. % ^{235}U will be loaded in the DSC,

4. DSC draining is accomplished within 50 hours of removal from the spent fuel pool.

assumptions

1. Fuel assemblies to be stored are no more reactive than the design basis 15x15 rod array,[b]

2. The DSC can not be filled with unborated water or borated water with less than 1810 ppm boron,

b The 15x15 rod array was determined to be the most reactive of several fuel assemblies evaluated in the TR.

3. No accident can occur which could alter the mechanical configuration of the stored array of irradiated fuel assemblies.

7.3 DISCUSSION OF RESULTS

7.3.1 Analytical Methods

The criticality safety analysis presented in the TR was performed using the Criticality Safety Analysis Sequence No. 2 (CSAS2) included in the SCALE-3[c] package of codes. The CSAS2 sequence and the 123GROUPGMTH master cross-section library included in the SCALE-3 system were used in calculating the effective neutron multiplication factor, k_{eff}, for the design basis configuration and the several evaluations of sensitivities to design parameters. The CSAS2 analysis sequence used two cross-section processing codes (NITAWL and BONAMI), and a three dimensional Monte-Carlo code (KENO-V) for calculating the multiplication factor for the DSC fuel assembly arrays.

The NRC staff performed independent calculations for this safety evaluation report using the NITAWL and KENO-V codes from the same SCALE-3 safety evaluation system. The geometry modeling of the fuel assemblies and DSC internals were independently developed for these calculations.

7.3.2 Design Basis Calculations

Misloading of unirradiated fuel was determined to be the worst case nuclear criticality condition that can reasonably be conceived for the operation of the NUHOMS-24P system. Analyses were performed to confirm an adequate criticality safety margin for this worst case configuration.

The criticality safety analysis presented in the TR shows that the loading of 24 unqualified unirradiated fuel assemblies with an enrichment of 4 wt. % ^{235}U would result in a (95/95) k_{eff} of 0.887. If the draining of

c SCALE-3: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, Oak Ridge National Laboratory, Revision 3, December 1984.

the DSC were not accomplished prior to the onset of boiling and if optimal moderator density conditions were realized, the worst case (95/95) k_{eff} is determined to be 0.979. The staff's evaluation confirms these results.

Note that in this instance, attainment of a condition of optimal moderation for unirradiated fuel is a non-mechanistic assumption. Unirradiated fuel does not constitute a significant heat source to reduce water density to an optimum, as might be postulated for irradiated fuel.

For design basis irradiated spent fuel with a 0.66/kW/assembly decay heat rate, the heatup rate for the moderator in the DSC cavity is no greater than 2^{o}F/hr following removal from the spent fuel pool. Thus for a nominal fuel pool temperature of 100^{o}F, the boiling point of the DSC moderator would not be reached for 50 hours. It is also recognized that the reactivity of irradiated fuel assemblies is less than unirradiated fuel assemblies, although the reactivity of irradiated fuel assemblies is not evaluated in this report.

In the event that the DSC must be unloaded following the second backfilling with helium gas, the irradiated fuel assemblies may have reached a temperature in excess of 600^{o}F and the DSC may have reached a temperature in excess of 200^{o}F (Reference 1). Before safe unloading can be accomplished, the DSC must be reflooded with borated water. The staff has considered this case.

Based on data from references in footnotes d, e, f, and g, the solubility of boric acid in water at 100^{o}F is approximately 13,000 ppm boron. This is well in excess of the 1810 ppm solution being injected into the cask. The solubility of boric acid in water increases with increasing temperature. Therefore, the raised solution temperature due to heat transfer from the fuel would only further increase the solubility of boric acid in water. A bounding

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d WCAP-1570, January, 1961, D.E. Byrnes and W.E. Foster.

e Boric Acid Properties Data, U.S. Borax Research Corporation, Anaheim, California.

f Supplement to Mellor's Comprehensive Treatise on Inorganic and Theoretical Chemistry, Volume V, Boron, Part A: Boron-Oxygen Compounds.

g Boron, Metallo-Boron Compounds and Boranes, edited by Roy M. Adams, Interscience Publishers.

conservative heat transfer calculation indicated that only about 0.3% of the water inventory could be removed by steaming in the first hour after the water is introduced to the fuel array. This represents an insignificant increase in the boric acid concentration. In conclusion, there is a very large margin between the boric acid concentration and the solubility limit of boric acid in the temperature range of interest and a reduction of the boric acid concentration is not possible for this scenario.

Further, as discussed above, any licensee shall have in place site-specific procedures for fuel selection and verification; fuel identification and verification; borated water measurement and verification; and fuel handling. The staff concludes that such procedures are necessary to assure that spent fuel is "...handled, stored and transported in a manner providing a sufficient factor of safety to require at least two unlikely independent and concurrent changes in conditions before a criticality accident is possible."[h] The staff's intent here is to identify those site-specific procedures that shall be implemented to assure the staff's factors and assumptions set forth in Section 7.2.2.4 for this criticality analysis are validated.

On the basis of the analysis presented in the TR, the supplementary analysis presented in response to questions, and the operating controls and limits, it is concluded that the NUHOMS-24P system is designed to be maintained in a subcritical configuration and to prevent a nuclear criticality accident in compliance with 10 CFR 72.124.

h ANSI/ANS 8.17-1984 "Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors," p. 2. Endorsed with modification by Regulatory Guide 3.58.

8.0 OPERATING PROCEDURES

8.1 SUMMARY AND CONCLUSIONS

The staff reviewed proposed operations described in Sections 5 and 9* of the TR. Portions of Sections 1 (1.3.1), 3 (3.1.2) and 4 (4.2.3, 4.5, and 4.7) of the TR contain summaries of operating procedures and were also reviewed.

Operations described in the TR are intended to serve as an example only and are not submitted for approval. Therefore, this review is limited to evaluating the feasibility of accomplishing the various activities. Approval of operations by the staff must await submittal of a site-specific application.

The staff concludes from its review that the operating sequence and steps proposed in the TR are feasible. If a site-specific applicant develops his own detailed operating procedures from the TR descriptions, there is no reason to believe they could not be made to meet the NRC's regulatory requirements. However, since NUHOMS is a new system that has not been built and tested, approval of site-specific procedures will be contingent upon successful demonstration of most "first-of-a-kind" features.

8.2 DESCRIPTION OF REVIEW

8.2.1 Applicable Regulations

The regulations used in the review of the TR included appropriate parts of 10 CFR 20 under the heading of "PERMISSIBLE DOSES, LEVELS, AND CONCENTRATIONS," and those paragraphs of Subparts E and F of 10 CFR 72 related to potential operational accidents (e.g., cask drop), off-normal events, and radiological doses.

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* Section 9.6, "Decommissioning Plan," of the TR is reviewed in Section 11 of this SER.

8.2.2 Review Procedure

8.2.2.1 Design Description

Section 5 of the TR presents a generic description of the handling, transfer and storage operations for NUHOMS. The operations considered unique to this system include:

1. Water filling of DSC and sealing of Cask/DSC annulus prior to lowering cask into spent fuel pool

2. DSC top lead cover welding of inner seal and weld inspection

3. DSC evacuation and helium backfill

4. DSC top cover plate welding and weld inspection

5. Draining of cask/DSC annulus and placing TC top cover

6. Transfer of fuel across the site on a specially designed vehicle in the TC which is built specifically for this use and is able to unload the DSC at the HSM

7. Positioning and aligning the TC with respect to the HSM opening while it sits on the transfer vehicle

8. Pushing the DSC into the HSM from the TC cavity

9. Reversing the order of loading the DSC into the HSM in order to be able either (1) to retrieve the spent fuel from the DSC on-site or (2) to ship the loaded DSC off-site.

8.2.2.2 Review Criteria

Since all operations are generic and no approval is sought, acceptance criteria are not applicable to this review. The review criteria for suitability of the operating procedures are based on: (1) the identification of appropriate steps for the protection of operating personnel, the public and the equipment, and (2) the feasibility of performing the operations.

8.2.2.3 Review Method

The sequence of operations and the step-by-step procedures proposed in the TR for the handling, transfer and storage of spent fuel were reviewed to determine if any portion of the proposed system might not function as planned. The reviewers used engineering judgment and past experience in a review of all proposed steps to reach a determination of feasibility. For those situations in which accidents might occur, a judgment was made of whether the results reported in the TR were reasonable or, lacking results, whether mitigating measures were available that could be implemented on a site-specific basis.

In the review of NUHOMS operations, special attention was given to the following issues.

1. Are inspection procedures and records normally available to determine the characteristics and the mechanical and structural integrity of fuel assemblies prior to loading them into a DSC?

2. Is the DSC able to withstand some reasonable combinations of loads, including various drops at normal TC transfer and placement heights, when used with the TC and transfer vehicles, while still maintaining its mechanical integrity, including retrievability of the fuel?

3. Are the dose rates, distances, and worker residence times during the DSC top welding operations reasonable and do they result in acceptably low exposures?

4. Are the dose rates, distances, and worker residence times for loading the DSC into the HSM reasonable, and do they result in acceptably low exposures?

5. Are the dose rates, distance of personnel from the DSC in the HSM, and personnel residence time during normal operation, off-normal events and

accidents reasonable and do they result in dose rates below levels specified by regulations?

6. Are the alignment dimensional tolerances between the HSM and the TC achievable and can the DSC be easily retrieved from the HSM after 20 years of storage?

7. Are the dose rates, distances, and worker residence times for the removal of fuel from the DSC reasonable and do they result in acceptably low exposures?

8.2.2.4 Key Assumption

It is assumed that approval of operating procedures will be given only on a site-specific basis.

8.3 DISCUSSION OF RESULTS

Although NUTECH is not seeking approval of the generic operations outlined in Section 5 of its TR, the NRC staff has evaluated Section 5, as well as the above issues. Based on engineering judgment and past experience with nuclear plant equipment and general level of personnel capability, the staff believes that the appropriate steps have been identified for the protection of operating personnel, the public and the equipment, and that the proposed operations are feasible.

The radiological guidelines for handling potentially contaminated equipment that apply to the removal of fuel from the DSC must include the requirement that personnel use respirators or supplied air to protect the health of the operations personnel.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

9.1 SUMMARY AND CONCLUSIONS

The staff has reviewed the proposed acceptance tests and maintenance programs for the storage of spent fuel in NUHOMS. Most of these activities are site-specific or are included as a part of the Codes of Design and Construction, which the vendor has committed to via its quality assurance program. The TR does specify the following generic requirements that must be met by the system:

1. The dose rates at the end of the DSC shield plug and at the surface of the HSM after the DSC is first inserted are restricted to specified values consistent with ALARA principles.

2. The maximum rise in air temperature from the HSM inlet to the HSM outlet after the DSC is initially loaded into the HSM is limited to a predetermined figure of 60°F. At this value the maximum fuel cladding temperature is predicted to remain below 340°C.

3. Daily inspections (surveillance) of the HSM air inlets and outlets is required to ensure that airflow is not interrupted. An annual inspection of the HSM internals is also recommended to identify potential airflow blockage and material degradation. The results of such inspections may require corrective action, which could be classified in the category of maintenance.

The staff finds that these generic activities, when augmented by a complete set of site-specific acceptance tests and maintenance programs, will provide for safe operation of the TC, the DSC, and the HSM with one exception. The staff requires the same level of pre-operational testing for the 24P design as was required for the 07P design. See Table 9.2-1 of Reference 22. Table 9.2-1 of Reference 22 outlines three sets of pre-operational tests for the NUHOMS-07P system. Two of the three are incorporated in Table 9.2-1 of the TR. The set missing involves the heat transfer, temperatures and air flow in various configurations for a DSC inserted in the HSM. The NRC staff requires the licensee to perform this set of tests for a DSC loaded with a heat source other than radioactive material, prior to spent fuel loading.

9.2 DESCRIPTION OF REVIEW

The review was performed by grouping the proposed test and maintenance activities into the following phases:

1. Design, procurement and fabrication of components.

2. Site commissioning: construction and installation of the system leading to start-up, including pre-operational testing.

3. Operational.

The tests and maintenance activities proposed in the TR for each phase were evaluated for completeness. Those activities that will be the subject of a site-specific application were not reviewed in detail. Those proposed generically were reviewed to determine whether they provide for safe operation of the three components which are important to safety.

9.3 DISCUSSION OF RESULTS

The material relating to acceptance tests and maintenance programs is very sparse and widely scattered throughout the TR. Table 3.3-4 of the TR identifies three major components as being important to safety: (1) the transfer cask, (2) the dry shielded canister, and (3) the horizontal storage module. These three components have codes of design and codes of construction associated with them. The vendor has formally committed to these codes of design and construction as an integral part of the NUHOMS 24P system.

9.3.1 Acceptance Tests

Although acceptance tests during spent fuel handling, transfer and storage are for the most part considered to be site-specific, subsection 10.3.2 of the TR does establish limiting conditions on certain critical parameters prior to the time that passive storage begins. Dose rates at the end of the DSC shield plug and at the surface of the HSM after the DSC is first inserted are restricted to specified values consistent with ALARA principles. The maximum rise in air temperature from the HSM inlet to the HSM outlet after the DSC is initially loaded into the HSM is also limited to a predetermined figure of 60°F. At this value, the maximum fuel cladding temperature is predicted to remain below 340°C.

Acceptance tests are not presented as such in the TR. They are primarily implied as a part of the codes of design and construction. Examples of these codes are given below:

1. Code of Design for DSC: ASME Code Section III, Division 1, Subsection NB.

2. Code of Design for TC: ASME Code Section III, Division 1, Subsection NC.

3. Code of Design for TC lifting trunnions: ANSI N14.6.

4. Code of Design for HSM: ACI 349-85.

5. Code of Design for DSC Supports: AISC Code 8th Edition.

6. Code of Construction for DSC: same as design code.

7. Code of Construction for TC: same as design code.

8. Code of Construction for TC trunnions: ASME Code Section III, Division 1, Subsection NC.

9. Code of Construction for HSM: ACI 318-83.

10. Code of Construction for DSC supports: same as design code.

Section 10 of the TR has various operational controls and limits for the performance of the system prior to service as well as immediately following emplacement of a DSC into an HSM. Examples of these controls and limits are:

1. Fuel characteristics
2. Dose rates for the on-site TC
3. Weld inspection standards for the DSC
4. Vacuum pressure required for drying the DSC following seal welding
5. Helium leak testing of the DSC and content of helium following backfilling
6. Dose rates for HSM
7. Surveillance of the HSM air inlets and outlets.

Section 9.2 of the TR refers to a pre-operational testing program that was specified for the NUHOMS-07P design. Although this program is relevant to the 24P design, the TR does not present any data or results. Thus the claim made in the TR that pre-operational testing of the 07P design "will provide sufficient data to demonstrate that the analytical methods described in this report provide conservative thermal and radiological results," is premature. The NRC staff therefore requires the same level of pre-operational testing for the 24P design as was required for the 07P design. (See Section 9.2 and Table 9.2-1 of Reference 22.) This is necessary to confirm the validity of the analytical methods with regard to the thermal hydraulic calculations.

The calculated maximum fuel cladding temperature, assuming 70°F ambient conditions, is 339°C, just 1°C below the limit of 340°C. The staff therefore requires that acceptance testing be performed in the same manner as for the 07P design, to confirm the validity of this design. Such testing must confirm that the air temperature rise from inlet to outlet is less than 60°F for a fully loaded (15.8 kW) DSC.

In general, the generic activities, when augmented by a complete set of site-specific acceptance tests, will provide for the safe operation of the DSC, TC and HSM. However, the validity of the analytical thermal hydraulic model must be confirmed by acceptance testing. The staff accepts the TR sections that refer to acceptance tests (Chapters 3, 4, 5, 9 and 10) except for Section 9.2.

9.3.2 Maintenance Program

Maintenance of NUHOMS is addressed in the TR in Sections 4.5 and 5.1.3. Section 4.5 of the TR covers the routine and annual inspection considered necessary for the TC. It includes: (1) visual inspection of such items as the cask exterior for cracks, damaged bearing surfaces, leakage from the neutron shield fittings, (2) visual inspection of all threaded parts for wear and burrs, (3) check of quick-connect fittings, and (4) visual inspection of interior surfaces of the cask. It also includes an annual inspection of the neutron shield pressure relief system and the cask lifting points and cask lifting yoke.

Sections 5.1.3 and 10.3.3.1 of the TR discuss the surveillance requirements of the HSM air inlets and outlets. Basically, however, the TR maintains that:

1. Maintenance of the system in order to assure continuous operation is not required since the system is totally passive once the spent fuel is in long term storage. However, daily inspection (surveillance) of the HSM air inlets and outlets is required to ensure that airflow is not interrupted. An inspection of the HSM internals at intervals of five and ten years after initial storage is also required for at least one HSM per installation to identify potential material degradation. The detailed procedures to be used during such inspections, which must address criteria for determining the effect of degradation, are site-specific. The results of such inspections may require corrective action, which could be classified in the category of maintenance.

2. Maintenance of the fuel handling and transfer equipment is site-specific. The major components involved are the transfer cask, transfer trailer and skid, cask restraint system, and hydraulic ram. (Note: the devices used for lifting heavy loads while the DSC is in the reactor or spent fuel pool building are assumed to be covered under a 10 CFR Part 50 license).

The NRC reviewers found the information on maintenance of equipment and procedures supplied in the TR to be adequate.

In summary, the TR treats acceptance testing and maintenance in the following ways:

1. Pre-operational acceptance testing of the system is site-specific.

2. Acceptance testing of components that are important-to-safety (the DSC, DSC internals and HSM) is subject to industry codes and standards, NUTECH's quality assurance program (as applicable), a site-specific applicant's quality assurance program (as applicable) and various procurement specifications, the last two items being site-specific.

3. Generic limiting conditions for operation are applied in the TR which, if not met, require corrective action.

4. Surveillance of the HSM exterior during the passive storage phase is required, which may result in maintenance activities if the NUHOMS performance is jeopardized. As noted above, detailed surveillance procedures are site-specific.

5. Maintenance of equipment used in handling and transfer of spent fuel is a site-specific requirement.

With one exception, the staff finds that this treatment is acceptable and that the generic activities, when augmented by a complete set of site-specific items, will provide for safe operation of the TC, DSC and the HSM when applied to a site-specific situation. Special attention needs to be given to establishing criteria which define when corrective actions are required. The single exception is that the pre-operational test requirements of the 24P design need to be modified to reflect the level of testing required for the 07P design.

10.0 RADIOLOGICAL PROTECTION

10.1 ON-SITE RADIOLOGICAL PROTECTION

10.1.1 Summary and Conclusions

The shielding, confinement, and handling design features of the NUHOMS-24P conform to the on-site radiological protection requirements of 10 CFR 20, and are considered acceptable for the set of conditions assumed in this review. The NUHOMS-24P design and operational procedures are also consistent with the objective of maintaining occupational exposures as low as reasonably achievable (ALARA). Detailed discussions of access control, surveillance, and other operational aspects affecting on-site exposure are deferred to the site-specific license application.

10.1.2 Description of Review

10.1.2.1 Applicable Parts of 10 CFR 72

Part 72.24 of 10 CFR requires the licensee to provide the means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20, and for meeting the objective of maintaining exposures as low as reasonably achievable.

Part 72.126(a) of 10 CFR requires that radiation protection systems shall be provided for all areas and operations where on-site personnel may be exposed to radiation or airborne radioactive materials.

Part 20.101(a) of 10 CFR 20 states that any individual in a restricted area shall not receive in any period of one calendar quarter from radioactive material and other sources of radiation a total occupational dose in excess of 1.25 rems to the whole body. Part 20.101(b) states that, under certain conditions, the quarterly dose limit to the whole body is 3 rems in any calendar quarter.

Guidance for ALARA considerations is also provided in NRC Regulatory Guides 8.8 and 8.10 (References 13 and 14, respectively).

10.1.2.2 Review Procedure

10.1.2.2.1 Design Description

The main radiation protection features of the NUHOMS-24P design include (1) radiation shielding; (2) radioactive material confinement; (3) prevention of external surface contamination; and (4) site access control. Access to the site of the NUHOMS-24P array, although not specifically addressed in the TR, would be restricted by a periphery fence to comply with 10 CFR 72.106(b) restricted area requirements. The details of the access control features are site-specific, and would be described in the applicant's site license application.

The shielding features of the NUHOMS-24P are discussed in Section 7.3.2 and Appendix A of the TR. Shielding includes many features designed to reduce direct and scattered radiation exposure, including:

1. Thick concrete walls and roof on the HSM which limit the contact dose rate to site workers to below an average of 20 mrem/hr

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

3. Use of a shielded transfer cask for DSC handling and transfer operations which limits the contact dose rate to 200 mrem/hr or less

4. Placing external shielding blocks over the HSM air outlets

5. Use of an internal shielding slab and wall around the HSM air inlet opening

6. Filling of the DSC cavity with borated water and the DSC-transfer cask annulus with demineralized water

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

The confinement features of the NUHOMS-24P control the release of gaseous or particulate radionuclides and are described in Section 3.3.2 of the TR. These features include:

1. The cladding of the stored fuel assemblies

2. The DSC containment pressure boundary

3. The inner and outer seal welds of the DSC

4. The DSC shielded end plugs and cover plates.

The DSC has been designed as a weld-sealed confinement pressure vessel with no mechanical or electrical penetrations. All the DSC pressure boundary welds are inspected according to the appropriate articles of the ASME Boiler and Pressure Vessel Code to ensure that the weld metal is as sound as the parent metal.

10.1.2.2.2 Acceptance Criteria

Radiation protection for on-site personnel is considered acceptable if it can be shown that the non-site-specific considerations will: (1) maintain occupational radiation exposures at levels which are as low as reasonably achievable, (2) be in compliance with appropriate guidance and/or regulations, and (3) assure that the dose from associated activities to any individual does not exceed the limits of 10 CFR 20.

10.1.2.2.3 Review Method

The calculational methods used in the estimation of on-site doses are described in detail in the TR. These methods focused on the use of the ANISN, QAD-CGGP, SKYSHINE-II, and MICROSHIELD radiation transport codes, as well as manual albedo calculations, to calculate exposure rates around the DSC in a TC and an HSM. Independent or confirmatory calculations of these exposure rate calculations were not made. Rather, the calculational methods and results presented in the TR were reviewed for completeness, correctness, and internal consistency. The dose rate results were then used with estimated distance and occupancy rate data to assess the individual and collective on-site doses.

10.1.2.2.4 Key Assumptions

Radiation doses to on-site workers are not calculated in the TR. Rather, a summary of the operational procedures which lead to occupational exposures is presented, as are the number of personnel required, the estimated time for completion of each operation, and the average source-to-subject distance. Dose rate estimates for the specific areas to be occupied during these operations are not presented directly, but can be estimated from the exposure rate data, which are presented. The TR notes that the operations and labor estimates are provided only as an example, since a collective dose calculation of this type is required for a site-specific license application. The TR also states that the dose rates for the NUHOMS-24P system are similar to those of the NUHOMS-07P system, although the number of specific operations and the time required for their completion, as listed in Table 7.4-1 of the respective TRs, differ significantly.

The following method was used in this review to estimate the on-site dose. It was assumed that the working area dose rates around the surfaces of the TC or HSM are similar to those presented in Table 7.4-1 of the NUHOMS-07P TR, and that the labor-hour requirements for specific operations are similar to those listed in Table 7.4-1 of the NUHOMS-24P TR. For operations which were not listed in the NUHOMS-07P TR, the dose rates which are used were assessed in a recent site-specific application.

Both collective and maximally exposed individual dose results are presented for the phase of ISFSI operations from loading of the DSC to insertion into the HSM. The assessment does not include the dose to on-site workers not directly involved in ISFSI operations, which is highly dependent on site-specific factors.

10.1.3 Discussion of Results

The following evaluation covers ALARA considerations, radiation protection design features, and on-site dose assessments.

10.1.3.1 ALARA Considerations

The design of the NUHOMS-24P exhibits several features that are specifically directed toward ensuring that occupational doses are in accordance with the ALARA guidance given in Regulatory Guide 8.8, in addition to satisfying the requirements of 10 CFR 20. In addition to the radiation protection design features discussed below, specific considerations include administrative programs such as access control, the application of maximum acceptable dose rates related to access requirements, and provisions for shielding based on demonstrably conservative assumptions. Other considerations are identified in Section 7.1 of the TR.

10.1.3.2 Radiation Protection Design Features of the NUHOMS-24P

There are several radiation protection design features of the NUHOMS-24P described in Sections 7.1.2 and 7.3 of the TR. The principal radiation protection design features include provisions for shielding, confinement, and contamination control.

Shielding includes many features designed to reduce direct and scattered radiation exposure. The specific features were listed in Section 10.1.2.2.1.

The DSC has been designed as a weld-sealed confinement pressure vessel with no mechanical or electrical penetrations. All the DSC pressure boundary welds are inspected according to the appropriate articles of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB. These criteria ensure that the weld metal is as sound as the parent metal.

Contamination of the DSC exterior and transfer cask interior surfaces is controlled by placing demineralized water in the TC during loading operations, then sealing the DSC/cask annulus with a donut-shaped inflatable rubber tube.

10.1.3.3 On-Site Dose Assessment

Radiation doses to on-site workers were not calculated in the TR. Rather, a summary of the operational procedures which lead to occupational exposures is presented, as are the number of personnel required, the estimated time for completion, and the average source-to-subject distance. Dose rate estimates for the specific areas to be occupied during these operations are not presented. The TR notes that the operations and labor estimates are provided only as an example, since a collective dose calculation of this type is required for a site-specific license application. The TR also states that the dose rates for the NUHOMS-24P system are similar to those of the NUHOMS-07P system.

In the NUHOMS-07P TR, the estimated collective dose for the loading, transfer, and insertion of one DSC was about 0.26 person-rem, while the maximum individual dose for these operations was about 125 mrem. Using the assumptions stated in Section 10.1.2.2.4, the collective dose associated with the loading, transfer, and insertion of one DSC would be about 1.4 person-rem, and the maximum individual dose would be about 610 mrem. This dose rate could require the use of multiple worker crews, depending on the number of transfers in a given year.

A detailed assessment of operator doses and the possible provision of management or administrative controls to meet ALARA criteria is deferred to a site-specific license application.

Other workers at the nuclear power plant site will also be exposed to direct and air-scattered (skyshine) radiation during the transfer and storage phases of ISFSI operation. Examples of activities involving such exposure are surveillance of the HSMs, and site operations which are not associated with spent fuel storage but which are performed in the general vicinity of the storage area. Major factors influencing the magnitude of the exposures are the occupancy times and spatial distribution of workers, and the intensity of the radiation field. An assessment of the expected on-site doses incurred by site personnel not directly involved in ISFSI operations is deferred to site-specific applications.

10.2 OFF-SITE RADIOLOGICAL PROTECTION

10.2.1 Summary and Conclusions

The shielding and confinement design features of the NUHOMS-24P conform to the off-site radiological protection requirements of 10 CFR 72 and are considered acceptable for the set of conditions assumed in this review. The use of high-integrity double-seal welds on the DSC ensures that during normal operation, there are no effluent streams from the NUHOMS-24P. Off-site dose is, therefore, due strictly to direct and scattered radiation, the intensity of which is a function of distance. Site-specific factors such as the number of HSMs in the storage array, the distance and direction of the nearest boundary of the controlled zone, the contribution of reactor plant effluents to the off-site dose, and resultant collective off-site dose must be considered in the compliance evaluation for a proposed NUHOMS-24P at a specific site.

10.2.2 Description of Review

10.2.2.1 Applicable Regulations

Sections 72.24(l) and (m) of 10 CFR require, in part, that a safety assessment be performed on the potential dose or dose commitment to an individual located outside the controlled area as a result of radioactivity releases caused by accidents or natural phenomena events.

Section 72.104(a) of 10 CFR requires that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual located beyond the controlled area shall not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to: (1) planned discharges of radioactive materials (except for radon and its daughter products) to the general environment, (2) direct radiation from NUHOMS-24P operations, and (3) any other radiation from uranium fuel cycle operations within the region.

Section 72.106(b) requires that any individual located on or near the closest boundary of the controlled area (at least 100 m) shall not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident.

10.2.2.2 Review Procedure

The two principal design features which limit off-site exposures during normal operations are the confinement features of the double-seal welded DSC, and the radiation shielding of the DSC and the HSM. During transfer operations, shielding in the radial direction is provided by the transfer cask. The confinement features of the DSC control the release of gaseous or particulate radionuclides and are described in Section 3.3.2 of the TR. The radiation shielding design features limit the direct radiation exposure rate and are described and analyzed in Section 7.3.2 and Appendix A of the TR. Additionally, Section 7.4 provides a dose-versus-distance curve from the shield analysis results.

Off-normal events and postulated accidents that could result in the loss of shielding or the release of radionuclides are analyzed in Sections 8.1 and 8.2 of the TR. In particular, an accident resulting in the loss of both air outlet shielding blocks is analyzed in Section 8.2.1, while an instantaneous release of 30 percent of fission gas inventory is assessed in Section 8.2.8. Other accidents are assessed in Section 8.2 (e.g., floods, tornadoes, earthquakes, accidental cask drop, blockage of air inlets and outlets, etc.), but the TR concludes that none of these other accidents represent credible sources of off-site dose consequences. The NRC staff accepts this conclusion.

This evaluation focuses on the off-site doses resulting from normal operations and from the two postulated accident events which can have off-site dose consequences. These doses are assessed for compliance with 10 CFR 72. The minimum distance selected for the evaluation of compliance with this section is 200 m, which is a reasonable approximation of the minimum distance to the nearest residence beyond the 100 m controlled area required by 10 CFR 72.106.

10.2.2.3 Acceptance Criteria

Off-site radiological protection features of the NUHOMS-24P system are deemed acceptable if it can be shown that design and operational considerations which are not site-specific result in off-site dose consequences which are in compliance with the applicable sections of 10 CFR 72, and that these doses to off-site individuals are as low as reasonably achievable.

10.2.2.4 Review Method

The review for off-site radiological protection mainly involved a detailed evaluation of the methods applied and the results obtained in the applicable TR sections, supplemented by additional information provided by NUTECH on these methods and results. For the case of off-site doses from direct and scattered (or "skyshine") radiation, an evaluation was performed on the application of the ANISN, SKYSHINE-II, and MICROSHIELD computer codes, which were used to calculate gamma-ray and neutron dose equivalent rates at various locations in and around the HSM, and to generate a dose-versus-distance curve. The dose rates predicted by this curve for an off-site distance of 200 m was used to assess the general level of compliance with the minimum 100 m criterion of 10 CFR 72.104(a).

The accident analyses provided in Section 8.2 of the TR were evaluated for technical soundness, and the results of the DSC leakage event, which provides the highest off-site dose, were verified by independent calculation. The dose consequences were assessed at 200 m and 300 m. The former distance is used as a reasonable estimate for the distance to the nearest resident. The 300-m distance is used in the TR and is used here for comparison purposes.

10.2.2.5 Key Assumptions

The assessment of off-site dose from normal operations assumes the following:

1. The recipient of the dose resides at a distance of 200 m or 300 m from a two-by-ten array of NUHOMS-24P modules, which are filled with design basis spent fuel.

2. An occupancy factor of unity is assumed, and no credit is taken for attenuation in building materials.

3. The dose rate as a function of distance from a filled NUHOMS-24P is as illustrated in Figure 7.4-1 of the TR.

The consequences of the loss of shielding blocks event assume the following:

1. The air outlets on a single HSM or all air outlets on a 2x10 array of HSMs lose their shielding blocks and remain unshielded for a period of 7 days.

2. The resultant dose rate at the surface of the air outlet is 3600 mrem/hr, and decreases with distance according to the results presented in Table 8.2-2 of the TR.

3. The recipient of the dose is present for the entire duration of the recovery at a distance of 200 m or 300 m.

The consequence assessment of the DSC leakage event assumes the following:

1. The fraction of the noble gas (assumed to consist entirely of Kr-85) inventory which is released is either 0.1, as recommended by NUREG-0575 (Reference 43), or 0.3, as recommended by Regulatory Guide 1.25 (Reference 44), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in

the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

2. The release is short-term (i.e., assumed to last from 0 to 8 hours).

3. Short-term atmospheric dispersion factors were obtained from Regulatory Guide 1.4 (Reference 45), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors", and assumes Class F stability, 1 m/sec wind speed, and ground-level release.

4. External dose conversion factors were obtained from Regulatory Guide 1.109 (Reference 46), "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I". The inhalation dose conversion factor for I-129 was taken from NUREG-0172 (Reference 47), "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake."

5. The distance from the release point to the receptor is 200 or 300 m.

10.2.3 Discussion of Results

The evaluation covers both normal operating and accident conditions.

10.2.3.1 Normal Operating Conditions

The dose to an off-site individual residing at a distance of 200 m from a filled NUHOMS-24P array is conservatively estimated as 110 mrem/yr. At 300 m, the dose is estimated as 32 mrem/yr. Since the assessment methodology conservatively assumed peak irradiated fuel, minimum post-irradiation time, full-time occupancy in the direction of maximum off-site dose, and no attenuation by building materials, it is likely that off-site doses to a "real" individual would be significantly lower, perhaps by a factor of five or more. Although site-specific factors (e.g., distance and direction of the nearest off-site residence, fuel conditions, contribution of off-site dose from reactor plant effluents, etc.) must be carefully considered, it is likely that normal operation of an NUHOMS-24P would comply with the requirements of 10 CFR 72.

10.2.3.2 Accident Conditions

The TR evaluated the dose to an off-site individual at several distances as a result of a loss of air outlet shielding block accident. Based on the TR evaluation, the dose to an individual at a distance of 200 m or 300 m is computed as approximately 1.3 mrem or 0.6 mrem, respectively, for a single HSM, or 19 mrem and 9 mrem for a 2x10 array of affected HSMs. These doses are well below the limits prescribed by 10 CFR 72.106 (b).

The following accident dose consequence results have been calculated for an offsite individual at distances of 200 m or 300 m. This assessment uses the method of Regulatory Guide 1.25 (i.e., 30% of fission gas inventory released), dispersion factors from Regulatory Guide 1.4, and dose factors from Regulatory Guide 1.109. For comparison, doses for a 10% fission gas release are also presented. These results are as follows:

	Dose Equivalent (rem)			
	10% Fission Gas Release		30% Fission Gas Release	
Organ	200 m	300 m	200 m	300 m
Whole Body	0.21	0.094	0.62	0.28
Skin	3.1	1.4	9.3	4.2
Thyroid	1.2	0.56	3.7	1.7

With the exception of skin dose at 200 m from 30% fission gas release, these doses are all within the 5 rem limit for whole body or any organ prescribed by 10 CFR 72.106(b). The DSC leakage event should be further assessed for site-specific applications. It should also be noted that, as indicated in the TR, no credible conditions have been identified which could breach the canister body or fail the double seal welds at each end of the DSC. Thus, these dose results are only presented to bound the consequences that could conceivably result, and to evaluate compliance with the 10 CFR 72.106 standard.

11.0 DECOMMISSIONING

11.1 SUMMARY AND CONCLUSIONS

The applicant has summarized decommissioning considerations for the NUHOMS-24P. The TR takes the position that the basic design of the NUHOMS-24P recognizes the need to decommission at the end of its useful life, and that a decommissioning plan would be developed based on site-specific factors. The TR also states that the DSC is designed to interface with a transportation system planned to transport canistered intact fuel assemblies (i.e., filled DSCs) to either a monitored retrieval storage facility (MRS) or a geologic repository. Once the DSCs have been removed, only small amounts of residual contamination are expected to remain in the HSM passages, thereby facilitating easy decommissioning. This position is based mainly on the fact that external contamination of the DSC is limited by its confinement features and through the contamination control procedures used during DSC fuel loading.

The staff finds that the proposed design and procedures are in conformance with the intent of 10 CFR 72.130, but withholds formal approval pending review of a site-specific case.

11.2 DESCRIPTION OF REVIEW

11.2.1 Applicable Parts of 10 CFR 72

Part 72.130 of 10 CFR provides criteria for decommissioning. It requires that considerations for decommissioning be included in the design of an ISFSI, and that provisions (1) facilitate the decontamination of structures and equipment, (2) minimize the quantity of radioactive wastes and contaminated equipment, and (3) facilitate removal of radioactive wastes and contaminated materials at the time of decommissioning.

Part 72.30 of 10 CFR defines the need for a decommissioning plan, which includes financing. Such a plan, however, is only appropriate to a site-specific situation, and 10 CFR 72.30 is therefore considered not applicable to this review.

11.2.2 Review Procedure

11.2.2.1 Design Description

The three primary components reviewed against 10 CFR 72.130 are the TC, the DSC and the HSM. Contamination levels on the external surface of the DSC are minimized by the use of uncontaminated water in the cask/DSC annulus during fuel pool loading operations. This prevents contaminated fuel pool water from contacting the DSC exterior.

Based on the proposed procedures described in Section 3.3.2.1, 4.4.1, and 5.1.1 of the TR, the contamination levels of the DSC will be determined by taking surface swipes of the upper one foot of the DSC exterior while the DSC is in the transfer cask prior to making the first closure weld. This swipe will be used as a representative sample of the DSC body. If the specified limits are exceeded, the annular space between the DSC and TC will be flushed with demineralized water until the contamination levels are within these limits. By minimizing DSC contamination, the potential for the internal surfaces of the HSM are kept to a minimum.

The current design of the NUHOMS system is based on the intended eventual disposal of each DSC following fuel removal. However, it is also possible that the DSC shell/basket assembly could be reused. Such an alternative would be dependent on economic and regulatory conditions at the time of fuel removal.

A detailed decommissioning plan, which would consider such factors as decommissioning options available, likely further use of the site, environmental impact, and available waste transportation and disposal capabilities, would be developed on a site-specific basis.

11.2.2.2 Acceptance Criteria

Although 10 CFR 72.130 does not provide specific criteria for acceptance, the licensee is required to design the ISFSI for decommissioning. Therefore, the NUHOMS-24P design has been reviewed against good nuclear engineering practices which include (1) means to control the spread of contamination, and (2) a design that facilitates decontamination.

11.2.2.3 Review Method

Decommissioning considerations are addressed in a general manner in Section 3.5 of the TR. Other applicable descriptions in the TR include (1) Sections 3.3.2 and 4.2.3.1, which describe the confinement features of the DSC, (2) Sections 4.4.1 and 5.1.1, which describe methods used to limit external contamination of the DSC, and (3) Section 3.3.7.1, which provides guidelines for external surface contamination limits. These sections were reviewed to assess the adequacy of the proposed design in meeting the acceptance criteria.

11.2.2.4 Key Assumptions

It has been assumed for the purpose of this review that:

1. There is no credible chain of events that would cause the DSC confinement to fail, resulting in contamination of the HSM passages

2. Contamination of the external surfaces of the DSC and the internal surfaces of the HSM can be maintained below applicable surface contamination limits. The TR uses the following surface removable contamination limits as a guide:

   Beta-gamma emitters: 10^{-4} uCi/cm^2
   Alpha emitters: 10^{-5} uCi/cm^2

11.2.3 Discussion of Results

The material presented in Section 3.5 of the TR addresses decommissioning of the HSM. This section claims that the DSC is designed to interface with a transportation system capable of transporting intact canistered assemblies to either a monitored retrievable storage (MRS) facility or a geologic repository. However, no evidence to support this statement was provided in the TR. The NRC staff has not evaluated this aspect of the NUHOMS-24P system. For purposes of decommissioning, the NRC staff has assumed that the spent fuel must be removed from the DSC by a cutting operation. For personnel protection, see Section 8.3 of this SER.

When the fuel must be removed from the DSC, the internal surface of the DSC will be contaminated and may be slightly activated. After the interior is cleaned to remove loose contamination, the DSC can be disposed of as low-level waste, or possibly even as scrap. Decommissioning of the transfer cask is considered a site-specific issue.

The current design of the NUHOMS system is based on the intended eventual disposal of each DSC following fuel removal. However, it is also possible that the DSC shell/basket assembly could be reused. Such an alternative would be dependent on economic and regulatory conditions at the time of fuel removal. A detailed decommissioning plan, which would consider such factors as decommissioning options available, likely further use of the site, environmental impact, and available waste transportation and disposal capabilities, would be developed on a site-specific basis.

The primary reason for requiring a clean exterior surface of the DSC is to reduce the total amount of activity available as a source of potential contamination for the HSM interior. The DSC surface contamination limits can be converted to a maximum total activity of roughly 15 microcuries of beta-gamma emitters, and 1.5 microcuries of alpha emitters. If the DSC exterior is initially below the contamination guidelines, contamination of the HSM interior will be much lower than these values.

The applicant has also claimed that failure of the DSC and release of radionuclides is not feasible under normal, off-normal, or accident conditions. Therefore, the contamination levels of the HSM are limited to levels which are much less than the initial DSC surface levels. This would probably allow the demolition and disposal of the HSM by conventional methods.

The staff concludes that adequate attention has been paid to decommissioning in the design of the NUHOMS-24P, considering the current state of knowledge. It will be necessary to review each site-specific application before determining whether demolition and removal of the HSM can be performed by conventional methods. The staff also notes that decommissioning of the DSCs, TC, and other equipment, as well as preparation of a decommissioning plan, are matters properly addressed in a site-specific application.

12.0 OPERATING CONTROLS AND LIMITS

12.1 SUMMARY AND CONCLUSIONS

Although operating controls and limits are normally reviewed as part of an application for a site-specific license, the staff has reviewed the set of generic operating controls and limits found in Chapter 10 of the TR. These controls and limits are summarized and expanded upon by the NRC staff in Table 12-1 of this SER. Operating controls and limits as stated in Table 12-1 of this SER are found acceptable.

12.2 DESCRIPTION OF REVIEW

12.2.1 Applicable Parts of 10 CFR 72

10 CFR 72.44 defines the requirements for operating limits and controls. That section only applies to specific licenses, not to reviews and approvals of topical reports. However, to the extent that operating controls and limits in a topical report are referenced in an application for a license, they require approval by the NRC.

12.2.2 Review Procedure

The staff has reviewed Sections 3, 7, 8, and 10 of the TR with special attention given to those parts which form the basis for a set of generic operating controls and limits. The criteria for and results of the safety analyses provided in the first three above-mentioned sections were used to review the limiting conditions proposed in Section 10 of the TR.

12.3 DISCUSSION OF RESULTS

Section 10.3 of the TR presents one fuel specification, nine limiting conditions for operation, one surveillance requirement, and two limiting conditions for transfer cask operation. The TR identifies these controls and limits as being generic and necessary for safe operation. The NRC staff has reviewed this set of operating controls and limits and added some additional conditions based on the overall evaluation of the TR. The

Table 12-1 Summary of Specifications from Section 10 of NUHOMS TR

Topic	Specification		TR Reference
Fuel Specifications (nominal)	Type	PWR Fuel	10.3.1.1 (3.1.1.1)
	Fuel Cladding	Zircaloy-clad fuel with no known or suspected cladding damage	10.3.1.1
	Burnup	$\leq$ 40,000 MWd/MT	10.3.1.1 (3.1.1.1)
	Post Irradiation Time	$\geq$ 10 years	10.3.1.1 (3.1.1.1)
	Initial (Beginning of Life) Enrichment	$\leq$ 4.0 wt % U-235	10.3.1.1 (3.1.1.1)
	Initial Enrichment equivalent of stored assemblies	$\leq$ 1.45 wt % U-235	3.3.4.1
	Weight Per Distance Between Any Adjacent Spacers, Per Assembly	$\leq$ 109.00 kg	Table 3.1-2
	Distance Between Spacers	$\leq$ 0.574 m	Table 3.1-2
	Maximum initial rod fill gas pressure	$\leq$ 480 psig (at 32° F and 1 atm)	8.2.9.1
	Any fuel not specifically filling the above requirements for burnup and post irradiation time may still be stored in the NUHOMS system, if all the following requirements are met:		
	Decay Power Per Assembly	$\leq$ 0.66 kw at 10 years post irradiation time	
	Neutron Source Per DSC	$\leq 3.715 \times 10^9$ n/sec/DSC, with spectrum bounded by Table 3.1-4	Table 3.1-1
	Gamma Source Per DSC	$\leq 3.85 \times 10^{16}$ MeV/sec/DSC with spectrum bounded by that shown in Table 3.1-4	Table 3.1-1

Table 12-1 Summary of Specifications from Section 10 of NUHOMS TR (cont'd)

Topic	Specification		TR Reference
DSC Vacuum Pressure During Drying	Vacuum Pressure Time at Pressure:	3 torr Not less than 30 minutes	10.3.2.1
DSC Helium Backfill Pressure	Helium backfill pressure (stable for 30 minutes filling)	2.5 psig ± 2.5 psig	10.3.2.2
DSC Moderator During Loading and Unloading	Boron concentration	≥ 1810 ppm	NRC 10-11 Feb. 1989, page 4
Time Limit to Complete DSC Draining After Removal From Spent Fuel Pool	Time	≤ 50 hours	NRC 10-11 Feb. 1989, page 3
DSC Helium Leakage Rate Test of Primary Weld	Leakage rate of primary weld 10^{-4} atm - cc/sec		10.3.2.3
DSC Dye Penetrant Test of Secondary Weld	Acceptance standards for liquid penetrant examination ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsection NB-5350 (1983) Liquid Penetrant Acceptance Standards		10.3.2.4
Dose Rate at End of DSC Lead Shield Plug	Dose Rates at the following locations:		10.3.2.5
	Center of Lead Shield Plug with water in cavity of DSC	100 mrem/hr	
	Center of DSC Top Cover Plate with Temporary Shield in place	200 mrem/hr	
Location of HSM	The HSM shall be located on a flood-free site.		SER Section 7.3.2
Surface Dose Rates on the HSM While the DSC is in Storage	Surface dose rates at the following locations:		10.3.2.6
	1) Outside of HSM door on centerline of DSC	100 mrem/hr	
	2) Center of air inlets	200 mrem/hr	
	3) Center of air outlet shielding caps	125 mrem/hr	
	4) Exterior side walls	20 mrem/hr	

Table 12-1 Summary of Specifications from Section 10 of NUHOMS TR (cont'd)

Topic	Specification		TR Reference
Maximum Air Temperature Rise from HSM Inlet to Outlet	Maximum air temperature 60°F measured 24 hours after DSC emplacement.		10.3.2.7
Alignment of Cask and HSM for DSC Transfer Operation	The cask must be aligned with respect to the HSM so that the longitudinal centerline of the DSC in the cask is within ± 1/8" of its true position when the DSC rests on the HSM.		10.3.2.8
DSC Retrieval and Inspection	The DSC must be retrieved and inspected subsequent to any cask drop of 15 inches or greater.		3.3.4.2.3 of SER
Surveillance of the HSM Air Inlets and Outlets	Normal visual inspection	Every 24 hours	10.3.3.1
Maximum Surface Dose Rate on Transfer Cask	Transfer cask lid	≤ 250 mrem/hr	10.3.4.1
	Body of TC with neutron shield filled with water	≤ 250 mrem/hr	
	Bottom of transfer cask with bottom cover plate installed	≤ 250 mrem/hr	
Transfer Route Selection	The surface within an eight foot proximity of the transfer trailer roadway shall be at the same elevation to ensure that the potential drop height of 80 inches is not exceeded.		10.3.4.2

complete list of operating controls and limits is given in Table 12-1 of this SER.

The requirements which were provided comprise a set of controls and limits for use with the proposed design. They will have to be augmented by additional specifications or revised to accommodate site-specific issues, but they do serve as a basis for review as a minimum set of requirements.

12.3.1 Fuel Specification

The fuel specification of Section 10.3.1.1 of the TR restricts the type of fuel acceptable for storage in the proposed design to ensure that peak fuel rod temperatures, radiation source terms, neutron multiplication factor, and stress on the DSC and its internals are below specified design limits.

12.3.2 Limiting Conditions for Operation

The nine limiting conditions for operation (LCO) are acceptable as proposed (see Section 10.3.2 of the TR).

12.3.3 Surveillance Requirements

The surveillance requirements are acceptable.

12.3.4 Limiting Conditions for Operation for TC Containing Fuel

The two limiting conditions for operation are acceptable as proposed in Section 10.3.4 of the TR.

13.0 QUALITY ASSURANCE

13.1 DISCUSSION

NUTECH's quality assurance (QA) program is addressed in the NUHOMS-24P topical report by incorporation of Section 11 of Reference 22. The QA program describes how NUTECH ensures the quality of the ISFSI described in the topical report. NUTECH is expected to be responsible for final design, specifications, procurement, fabrication, assembly, delivery, and preoperational testing associated with the dry storage canister and transfer cask. NUTECH may also have responsibility as consultant, supplier, installer, and/or on-site engineer for the horizontal storage module. The staff reviewed the QA program description against the acceptance criteria in Reference 48, the "Standard Review Plan for Quality Assurance Programs for an Independent Spent Fuel Storage Installation (ISFSI)."

The staff found that NUTECH's QA program described in the topical report adequately addresses the QA functions appropriate for these responsibilities, that the commitments meet the requirements of Subpart G of 10 CFR Part 72, and that the QA program is acceptable. The topical report can be referenced without further QA review in a license application to receive and store spent fuel under 10 CFR Part 72, provided the applicant applies its NRC-approved QA program that meets the requirements of Appendix B to 10 CFR Part 50 to the design, construction, and use of the spent fuel storage installation.

13.2 CONCLUSION

The staff concludes that the QA program described in the NUHOMS-24P topical report is acceptable as an appropriate reference for partial fulfillment of QA program requirements for ISFSI license applications.

14.0 SUMMARY EVALUATION

14.1 SUITABILITY AS REFERENCE

The NRC staff finds the TR to be suitable as a reference for ISFSI license applications except as specifically noted in Table 14.1. This evaluation is based on the extent of compliance with 10 CFR 72 using guidance as provided in NRC Regulatory Guides 3.48 and 3.60.

The NRC staff approves use of the TR as reference or for direct incorporation in license application documents with the exceptions noted in Table 14.1. Any modifications beyond those specified in Table 14.1 in material taken from or referenced in the TR, or any use of material in the TR by incorporation or reference which is outside the limitations, assumptions, conditions, or other context as stated in the TR or this SER requires full explanation, calculations, and/or descriptions, and is subject to further review by the NRC.

14.2 SATISFACTION OF LICENSE APPLICATION REQUIREMENTS

The NRC staff finds that the TR can partially fulfill ISFSI license application documentation requirements stated in 10 CFR 72 when used by clear and specific reference.

TABLE 14.1 EXCEPTIONS TO USE OF TR AS REFERENCE IN ISFSI LICENSE APPLICATION DOCUMENTATION

License Application Documentation	10 CFR Reference	Subject	Limitations on Use of TR Sections as References
License Application	72.22	General & Financial Info.	Not a reference
Safety Analysis Report	72.24(a)	Site Description & Safety Assessment	Background only
	72.24(b)	Description & Discussion of Structures	
		- Design	For DSC, TC, HSM (less Found. Anal) only
		- Operating Characteristics	For DSC, TC, HSM only
		- Unusual/Novel Design Features	Not a reference
		- Principal Safety Considerations	DSC, TC, HSM (less Found. Anal) only
	72.24(c)	Design Sufficient to Support 72.31 findings	For DSC, TC, HSM (less Found. Anal) only, except as indicated below.
	72.122(b)(1)	Design for compatibility w/site characteristics & environment	Not complete. SAR to validate that satis. for site & environment.
	72.122(b)(2)	Design for Natural Phenomena for site.	Not complete. SAR to validate that site phenom. within design envelope.
	72.122(b)(3)	Capability to determine intensity of Natural Phenomena	Not a reference
	72.122(c)	Protection vs Fires & Explosion	Background only
	72.122(d)	Sharing of Structures, Systems & Components	Background only
	72.122(e)	Proximity of Sites	Background only
	/ 72.122(f) \ \ 72.44(c)(3) /	Testing & Installation of Systems & Components	Background only. Satisfaction of surveillance requirements left to SAR.
	72.122(g)	Emergency Capability	Background only
	72.122(h)(3)	Ventilation and off-gas systems	Background only
	/ 72.122(i) \ \ 72.44(c)(3) /	Instrumentation & Control systems	Background only. Satisfaction of surveillance requirements left to SAR.
	72.122(j)	Control room or area	Not a reference
	72.122(k)	Utility services	Background only
	72.126(a)	Exposure control	Background only
	72.126(b)	Rad alarm systems	Background only
	72.126(c)	Effluent & Direct Rad Monitoring	Background only
	72.126(d)	Effluent Control	Background only
	72.128(b)	Waste Treatment	Background only
	72.130	Criteria for Decommissioning	Background only
	72.24(c)(3)	Applicable Codes and Standards	Not a reference

TABLE 14.1 EXCEPTIONS TO USE OF TR AS REFERENCE IN ISFSI LICENSE APPLICATION DOCUMENTATION (cont'd)

License Application Documentation	10 CFR Reference	Subject	Limitations on Use of TR Sections as References
SAR (cont'd)	72.24(d)	Structures, Systems & Components Important to Safety	For DSC, TC, HSM (less Found. Anal)
	72.24(e)	Satisfaction of 10 CFR 20 occupational radiation exposure limits	Background only
	72.24(f)	Design & Operating mode features to maintain low waste volume	Background only
	72.24(g)	Probable License Conditions and Tech Specs	Design background only
	72.24(h)	Plan for Conduct of Operations	Not a reference
	72.24(i)	Resolution of Safety Questions	Background only
	72.24(j)	Applicant's Tech Qualifications	Not a reference
	72.24(k)	Description of Emergency Plans	Not a reference
	72.24(l)	Equipment to Maintain Control over Gas and Liquid Effluent	Not a reference
	72.24(m)	Individual dose/dose commitment outside controlled area	Not complete. SAR to address actual site conditions.
	72.24(n)	Description of QA program	Not complete. SAR to address or integrate w/applicant QA program.
	72.24(o)	Description of Detailed Security Measures for Physical Protection	Not a reference
	72.24(p)	Descr of Program for Preoperational testing & initial operations	Not complete. SAR to address or integrate w/applicant's program.
	72.24(q)	Description of Decommissioning Plan	Background only
Decommissioning Plan	72.30	Decommissioning Plan	Background only
Emergency Plan	72.32	Emergency Plan	Background only
Environmental Report	72.34	Environmental Report	Background only
Quality Assurance Program	72. Subpart G	Quality Assurance Program	Not complete. SAR to address or integrate w/applicant QA program.
Physical Security Plan	72.180	Physical Security Plan	Not a reference
Design for Physical Protection	72.182	Design for Physical Protection	Not a reference
Safeguards Contingency Plan	72.184	Safeguards Contingency Plan	Not a reference

15.0 REFERENCES

1. NUTECH Engineers, Inc., "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P," (NUH-002), Revision 1, July 1988.

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The Topical Report
for the
NUTECH Horizontal Modular Storage System
for Irradiated Nuclear Fuel
NUHOMS-24P

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

This report describes the design and the generic licensing basis for the NUTECH Horizontal Modular Storage (NUHOMS) system for twenty-four PWR fuel assemblies. The NUHOMS system provides for the horizontal storage of spent fuel in a dry shielded canister (DSC) which is placed in a concrete horizontal storage module (HSM). The NUHOMS system is designed to be installed at any reactor site or any new site where an independent spent fuel storage installation (ISFSI) is required. The NUHOMS system described in this report is for dry storage of pressurized water reactor (PWR) fuel. A later amendment to this report or site specific license applications will address dry storage of boiling water reactor (BWR) fuel.

The NUHOMS Topical Report (NUH-001, Revision 1A, NRC Project No. M-39) was approved by the United States Nuclear Regulatory Commission (NRC) on March 28, 1986 for storage of seven spent PWR fuel assemblies per HSM (NUHOMS-07P). The NUHOMS-07P system was designed to be compatible with the General Electric IF-300 shipping cask. The DSC internal basket consisted of seven square cells composed of stainless steel clad Boral . The Boral provides criticality control in the DSC during wet loading operations.

The NUHOMS Topical Report was revised (NUH-002, Revision 0, NRC Docket No. M-49) to provide the generic design criteria and safety analysis for the larger 24 spent PWR fuel assembly design (NUHOMS-24P) and its associated on-site transfer cask (10CFR72). Unlike the NUHOMS-07P design, no borated neutron absorbing material is used in the design of the NUHOMS-24P DSC. The NUHOMS-24P internal basket consists of 24 stainless steel guide sleeves.

The remainder of this report section provides a general overview of the NUHOMS system and describes the format and contents of this topical report.

1.1 Introduction

Due to the unavailability of nuclear fuel reprocessing in the United States (U.S.), long-term storage of spent fuel assemblies (SFA) has become necessary. To date, storage systems have, to a large extent, relied on the plant's fuel pools. However, as existing pools have begun to approach full capacity (with high-density storage racks), new out-of-pool dry storage designs have emerged. The NUHOMS system is one of these new dry storage designs. Figure 1.1-1 shows the primary components of the NUHOMS system.

The SFAs are loaded into the DSC (which is resting inside the transfer cask) in the fuel pool at the reactor site. The transfer cask containing the loaded DSC is removed from the pool and placed in the cask lay-down area where sealing, draining, and drying operations are performed. The DSC cavity is then backfilled with helium. Multi-pass, double seal welds at each end of the DSC and multi-pass circumferential and longitudinal welds are volumetrically examined to assure that no leakage of helium will occur. The cask is then placed on a transport trailer in the plant's fuel building and towed to the on-site ISFSI location. At the ISFSI location, the loaded transfer cask is aligned with the HSM and the DSC is pushed out of the transfer cask into the HSM by a horizontal hydraulic ram. Once inside the HSM, the DSC is in safe, dry storage.

The various components of the NUHOMS system are further described in Sections 1.2 and 1.3. The design and the conservative generic analyses of the system components is described in detail in the remainder of this topical report. The principal design features of a NUHOMS ISFSI are:

1. Horizontal transfer of the DSC into and out of the HSM

   This removes the need for a critical heavy lift of the SFAs at the storage location (i.e. away from the plant's safety-related systems), optimizes the amount of material required for biological shielding, and results in a passive, low profile, impact-resistant storage structure.

2. Transport of the DSC from the fuel building to the HSM in an on-site transfer cask

   This provides radiation shielding and structural protection for the DSC during the transfer operation while providing passive heat removal.

3. Shielded End Plug Assemblies on the DSC

   This enables the direct handling and monitoring of the DSCs at the top and bottom end positions when the DSC is inside the cask or the HSM.

4. Phased Construction of HSMs

   This facilitates site licensing and modular construction of the HSM arrays, thus economizing and distributing the cost for fuel storage over the time span when storage is required.

5. Natural Circulation Air Cooling

   This keeps the maximum fuel rod cladding temperature below acceptable limits to preclude damage during long term storage.

6. Acceptance of Equivalent PWR Spent Fuel

   The NUHOMS-24P system accepts spent fuel assemblies with equivalent decay heat and radiological source term values enveloped by those corresponding to PWR fuel with a cooling time of ten years, an initial uranium content of 472 Kg/assembly, an initial enrichment of 4.0%, and a fuel burnup of 40,000 MWD/MTU as outlined in Section 3 of this Topical Report.

7. Helium Storage Atmosphere

   This provides effective heat transfer and prevents uranium dioxide (UO_2) oxidation of the SFAs. The double seal welded DSC pressure boundary assures that the helium atmosphere will be maintained.

This topical report was written for the U.S. Nuclear Regulatory Commission (NRC) for review under Title 10 of the Code of Federal Regulations, Part 72 (10CFR72) (1.1).* The purpose of this topical report is to provide directly referenceable information to any site license applicant submitting a Safety Analysis Report (SAR) for review and licensing under 10CFR72. To facilitate this direct referencing, the format, the numbering system, the section headings, and the content have followed NRC Regulatory Guide 3.48 (1.2).

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* Numbers in parentheses indicate references which are listed at the end of each section.

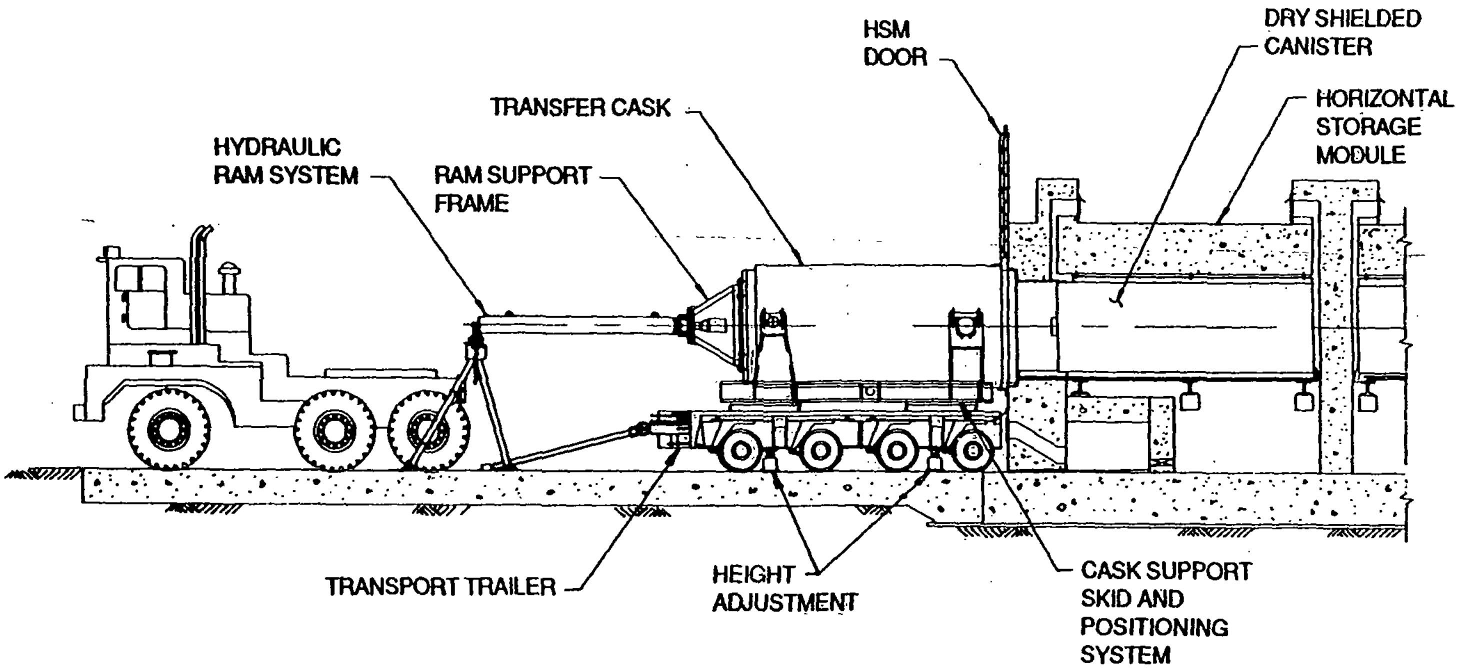


Figure 1.1-1

PRIMARY COMPONENTS OF THE NUHOMS SYSTEM

1.2 General Description of Installation

1.2.1 Arrangement of Major Structures and Equipment

The NUHOMS system provides for the horizontal, dry storage of canisterized SFAs in a concrete HSM. The primary system components are a reinforced concrete HSM and a DSC pressure vessel with an internal basket assembly which holds the SFAs.

In addition to these primary components, the NUHOMS system also utilizes transfer equipment to move the DSCs from the plant's fuel building where they are loaded with SFAs and readied for storage, to the HSMs where they are stored. This transfer system consists of a transfer cask, a lifting yoke, a hydraulic ram system, a prime mover for towing, a transport trailer and a cask positioning skid. This transfer system requires interfacing with the existing plant fuel pool, the crane, the site layout (i.e. roads and topography) and other site specific design and procedural requirements. This topical report will primarily address the design and analysis of the primary system components which, in accordance with 10CFR72, are important to safety including the DSC, the HSM, and the transfer cask. Sufficient information for the remaining transfer system equipment is also included to demonstrate that means for safe operation of the system are provided.

Each NUHOMS system model type is designated by NUHOMS-XXY. The two digits (XX) refer to the number of fuel assemblies stored in the DSC, and the character (Y) is a P for PWR, or B for BWR, to designate the type of fuel stored. The number of HSMs to be constructed at any one time is site specific and will be addressed in the individual site license applications. Future requirements for NUHOMS systems for different sizes of DSCs, HSMs, and transfer cask to accommodate different fuel types will be addressed in amendments to this topical report or in site specific license applications.

Although this topical report addresses only the standard NUHOMS-24P system, the basic design of the DSC, the HSM, and the transfer cask can be sized, depending on plant specific capabilities, to accommodate internal baskets which hold a single fuel assembly or up to 32 PWR fuel assemblies. Table 1.2-1 shows approximate sizes for various DSCs. In addition to the various sizes of DSCs and HSMs, the NUHOMS system can be designed for stand-alone HSMs (containing one DSC) or for interconnected HSMs which are constructed in arrays to accommodate a year or more of spent fuel discharged from a plant (50-70 PWR assemblies or 100-150 BWR assemblies). These two variations are shown in Figure 1.2-1. Any number of HSMs can be interconnected to form HSM units. The reinforced concrete HSMs can be constructed on-site to form an interconnected storage unit (poured in place), or they can be formed off-site (pre-cast) and then erected and tied together at the on-site storage installation. These NUHOMS system variations will be addressed in future topical report amendments and/or plant specific license applications submitted to the NRC for review and approval.

The specific size of the DSC and HSM depends on the interfaces with the existing plant equipment and siting requirements, future DOE repository and transportation requirements, and other economic and design considerations. These future considerations and individual utility choices are not fully addressed in this generic report and will be the subject of subsequent site license application submittals by individual licensees. This topical report deals specifically with the NUHOMS DSC, HSM, and on-site transfer cask sized to store 24 PWR fuel assemblies. It is unlikely that utilities would store less than the number of fuel assemblies corresponding to one year's core discharge per year. This topical report, however, addresses both a single HSM and interconnected HSM arrays which are grouped together to form units. The specific size of each HSM unit will vary depending on site specific requirements, but will typically be sized to store at least one year's discharge of fuel. This TR provides the design description and analyses for HSM units ranging in size from a single stand alone HSM up to a 2x10 array of 20 back-to-back HSMs. These two extremes bound all loading conditions for all intermediate sized HSM arrays. HSM units larger than 2x10 will be constructed of 2x10 or smaller arrays with provisions for thermal expansion between the HSM units.

1.2.2 Principal Design Criteria

The principal design criteria and parameters upon which this topical report is based are shown in Table 1.2-2.

Structural Features: The HSM is a low profile, reinforced concrete structure designed to withstand all normal operating loads as well as the abnormal loads created by earthquakes, tornadoes, flooding, and other natural phenomena. The HSM is also designed to withstand accident loadings postulated to occur during system operation.

The structural features of the DSC design depend, to a large extent, on the postulated design basis cask drop accident (described in Section 8.2.5). The DSC shell, the double containment welds on each end, and the DSC internals are designed to ensure that the intended safety functions of the system are not impaired following a postulated drop accident. The limits established for equivalent decellerations due to a postulated drop accident are necessarily site specific. They depend on various parameters such as the transfer cask design features provided for handling operations, the type of handling equipment used, the cask on-site transfer route, the maximum feasible drop height and orientation, and the postulated drop site surface conditions. Based on the parametric studies described in Section 8.2.5 of this TR, the transfer cask and DSC are designed to accommodate equivalent drop deceleration magnitudes of 75g in either the horizontal or vertical orientation, and 25g for the oblique (corner) orientation.

Decay Heat Removal: The decay heat of the SFAs during storage in the HSM is removed from the DSC by natural circulation convection and by conduction through the HSM walls and roof. Air enters the lower ventilation plenum of the HSM, circulates and rises around the DSC and exits through shielded openings in the HSM roof slab. The cross-sectional areas of the air inlet and outlet openings, and the interior flow channels and flow paths are designed to optimize ventilation air flow in the HSM for decay heat removal even for worst case extreme summer ambient conditions.

External Atmosphere Criteria: Given the corrosion resistant properties and coatings used in the design and construction of the NUHOMS system components, and the hot and dry environment which exists within the HSM, no limits on the range of acceptable external atmospheric conditions are required. All components are either stainless steel, are coated with inorganic zinc-based coatings, or are galvanized. Hence, all metallic materials are protected against corrosion. The interior of the HSM is a concrete surface and is void of any substance which would be conducive to the growth of any organic or vegetative matter. The design of the HSM also provides for drainage of ambient moisture which further eliminates any need for external atmospheric limitation.

The ambient temperatures selected for the design of the NUHOMS system range from -40°F to 125°F, with a lifetime average ambient temperature of 70°F. The extreme ambient temperatures of -40°F and 125°F are expected to last for a short period of time. The minimum and maximum average ambient temperatures of 0°F and 100°F are expected to last for longer periods of time.

1.2.3 Operating and Fuel Handling Systems

Some handling equipment and operating systems for the NUHOMS system are site-specific as previously discussed. However, the NUHOMS-24P on-site transfer cask is designed to satisfy a range of site-specific conditions and requirements. The general operations for a typical NUHOMS system installation are outlined in Table 1.2-3. A more detailed procedure for this sequence of operations is provided in TR Section 5.1. The primary design parameters of interest for site specific applications are listed in Table 1.2-4. The majority of the fuel handling operations involving the DSC and transfer cask (i.e. fuel loading, draining and drying, transport trailer loading etc.) are standard procedures used at power plants for SFA shipment. The remaining operations (cask-HSM alignment and DSC transfer) are unique to the NUHOMS system.

1.2.4 Safety Features

The principal safety features of a NUHOMS ISFSI include the high integrity containment for spent fuel materials and the axial shielding provided by the DSC pressure vessel, and the extensive biological shielding and protection against extreme natural phenomena provided by the massive reinforced concrete HSM. The shielding materials incorporated into the DSC and HSM designs reduce the gamma and neutron flux emanating from the SFAs so that the average contact dose rate on the outside surface of the HSM is less than 20 mrem/hour. The radiological safety features of the NUHOMS system include:

	Feature	Purpose
1.	Filling the DSC and cask with water prior to lowering these components into the fuel pool.	Minimizes contamination of DSC exterior surface during loading of SFAs in the fuel pool. Also lowers dose during subsequent DSC closure operations.
2.	A partial height shield wall and slab inside the HSM to form a ventilation plenum.	Minimizes the scatter dose through the ventilation air inlet opening.
3.	External shield blocks over HSM ventilation air outlet openings.	Minimizes the scatter dose and skyshine through the HSM ventilation air outlet openings.
4.	A heavy steel door with additional neutron absorbing material covering the HSM access opening.	Minimizes the direct and scatter dose through the HSM access opening.

	Feature	Purpose
5.	Top and bottom shield plugs on DSC.	Reduces the dose during DSC draining, drying, back-filling and seal welding operations. Also reduces the axial dose during storage in the HSM.
6.	High integrity pressure boundary with redundant containment closure welds on both ends of the DSC.	Prevents leakage of inert atmosphere, radioactive gases, or particulates should a fuel rod failure occur.

These radiological safety features allow the NUHOMS system to be deployed with a minimum of additional, site specific radiation protection and monitoring measures.

1.2.5 Radioactive Waste and Auxiliary Systems

Because of the passive nature of the NUHOMS system, there are no radioactive waste or auxiliary systems required during normal storage operations. There are, however, some waste and auxiliary systems required during the loading, draining, drying, back-filling, sealing and transfer operations for the DSC. The plant's radwaste systems are utilized to process water and off-gas which are purged from the DSC and cask during draining and drying. Auxiliary handling systems (such as the fuel building crane, cask lifting yoke, hydraulic ram system, cask support skid positioning system, the alignment equipment, a small porta-crane, etc.) are also utilized during the DSC transfer operation. Additional plant support systems such as compressed air, a helium source, demineralized water, and AC power are also utilized. The waste and auxiliary systems are further described in Sections 3, 4, and 5.

1.2.6 Principal Characteristics of the Site

The principal characteristics of a NUHOMS ISFSI site are necessarily site specific and, hence, are not included in this topical report.

Table 1.2-1

SAMPLE DSC AND HSM SIZES (1) (5)

No. of PWR [BWR] Assemblies	Typical Transfer Cask: Cavity Size (2)	Typical Transfer Cask: Gross Weight	DSC Size (3)	HSM Size (4)
1[2]	0.34 x 4.57 (13.5" x 180")	22,700 (50,000)	0.33 x 4.55 (13" x 179")	5.79 x 3.05 x 1.09 (19′ x 10′ x 3.6′)
7[14]	0.95 x 4.57 (37.5" x 180")	70,000 (154,000)	0.94 x 4.55 (37" x 179")	5.79 x 3.65 x 1.70 (19′ x 12′ x 5.6′)
10[20]	1.16 x 4.57 (45.5" x 180")	75,000 (165,000)	1.14 x 4.55 (45" x 179")	6.10 x 3.90 x 1.91 (20′ x 12.75′ x 6.25′)
12[24]	1.23 x 4.57 (48.5" x 180")	80,000 (175,000)	1.22 x 4.55 (48" x 179")	6.10 x 3.96 x 2.06 (20′ x 13′ x 6.75′)
21[42]	1.52 x 4.57 (60.0" x 180")	84,000 (185,000)	1.51 x 4.55 (59.5" x 179")	6.10 x 4.27 x 2.44 (20′ x 14′ x 8′)
24[52]	1.73 x 4.74 (68.0" x 186.8")	86,300 (190,000)	1.71 x 4.72 (67.3" x 186")	6.10 x 4.57 x 2.64 (20′ x 15′ x 8.67′)
32[64]	1.97 x 4.57 (77.5" x 180")	101,000 (225,000)	1.95 x 4.55 (76.7" x 179")	6.10 x 4.8 x 2.9 (20′ x 15.75′ x 9.5′)

Notes:
1. Unless otherwise noted, length units are meters and weight units are kilograms (inches or feet and pounds are used in the parenthesis below).
2. Cask cavity size is internal diameter x length.
3. DSC size is outside diameter x length.
4. HSM size is length x height x width.
5. Dimensions and weights are typical for the number of fuel assemblies to be stored. This information is provided only for the purpose of illustrating the flexibility of the NUHOMS system. Table 1.2-2 provides specific information for the standard NUHOMS-24P design presented in this TR.

Table 1.2-2

DESIGN PARAMETERS FOR THE NUHOMS-24P SYSTEM

Category	Criteria or Parameter	Value
Fuel Assembly Criteria	Initial Uranium Content	472 kg/assembly
	Initial Enrichment	4.0% (^{235}U Equivalent)
	Fuel Burnup	40,000 MWD/MTU
	Gamma Radiation Source	4.62E15 photons/ sec/assembly
	Neutron Radiation Source	1.548E8 neutron/sec./ assembly
	Decay Heat Power	0.66 kw/assembly
Dry Shielded Canister	Capacity per DSC	24 PWR Fuel Assemblies
	Size:	
	Overall Length	4.72m (186.0 in.)
	Outside Diameter	1.71m (67.25 in.)
	Shell Thickness	16mm (0.625 in.)
	Heat Rejection	15.8 kw
	Internal Atmosphere	Helium
	Maximum Design Pressure	Conservatively Based on 100% Release of Fill Gas and 30% Release of Fission Gas
	Equivalent Cask Drop Deceleration	75g Vertical and Horizontal 25g Oblique/Corner
	Material of Construction	Stainless Steel with Lead Shielded End Plugs
	Service Life	50 Years(1)

(1) Expected life is much longer (hundreds of years), however, for the purpose of this generic topical report, the service life is taken as 50 years.

Table 1.2-2

DESIGN PARAMETERS FOR THE NUHOMS-24P SYSTEM
(Concluded)

Category	Criteria or Parameter	Value
Transfer Cask	Payload Capacity	90,000 lbs.
	Gross Weight	190,000 lbs.
	Maximum Contact Dose Rate	200 mrem/hr.
	Equivalent Cask Drop Deceleration	75g Vertical and Horizontal, 25g Oblique/Corner
	Materials of Construction	Steel, Lead, and Neutron Absorbing Material
	Service Life	50 Years
Horizontal Storage Module	Capacity	One DSC per HSM
	Array Size	Single Module to 2x10 Module Array
	HSM Size:(2)	
	Length	6.1m (20 ft.)
	Height	4.6m (15 ft.)
	Width	2.6m (8.7 ft.)
	Average Contact Dose Rate	<20 mrem/hr.
	Maximum Contact Dose Rate	<100 mrem/hr
	Heat Removal	Natural Circulation
	Material of Construction	Reinforced Concrete and Structural Steel
	Service Life	50 years

(2) HSM length and width are for an interior HSM in an HSM array.

Table 1.2-3

NUHOMS SYSTEM OPERATIONS OVERVIEW(*)

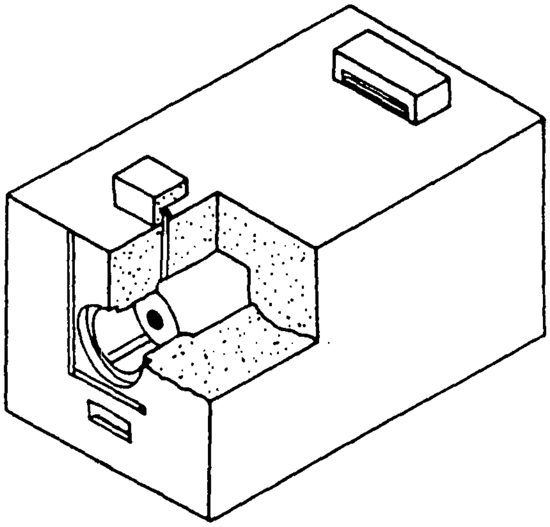
1. Clean and load the DSC into the transfer cask
2. Fill the DSC and cask with water and seal the cask/DSC annulus
3. Place the transfer cask containing the DSC in the fuel pool
4. Load the fuel into the DSC
5. Place the top shielded end plug on the DSC
6. Lift the cask containing the loaded DSC out of the fuel pool and place it in the decon area
7. Lower the water level in the DSC cavity and the DSC/cask annulus
8. Seal weld the top shield plug to the DSC shell and perform NDE
9. Drain the water from the DSC
10. Evacuate and dry the DSC
11. Backfill the DSC with helium
12. Perform helium leak test on the seal weld
13. Seal weld plugs in the siphon and vent ports of the DSC
14. Perform nondestructive examination on the port cover seal welds
15. Install and weld the top cover plate on the DSC
16. Perform nondestructive examination on seal weld
17. Drain the water from the transfer cask/DSC annulus
18. Place and secure the transfer cask top cover plate
19. Lift the transfer cask onto the transport trailer and lower it to a horizontal position
20. Tow the transport trailer to the HSM
21. Remove the HSM door
22. Position the transfer cask with the HSM access opening
23. Remove transfer cask top cover plate
24. Align and secure the transfer cask to the HSM
25. Remove the transfer cask center bottom shield plug
26. Insert the hydraulic ram
27. Push the DSC into the HSM and then retract the ram
28. Disengage the transfer cask from the HSM
29. Place and secure the HSM door

* See Section 5.1 for more detailed system operation description.

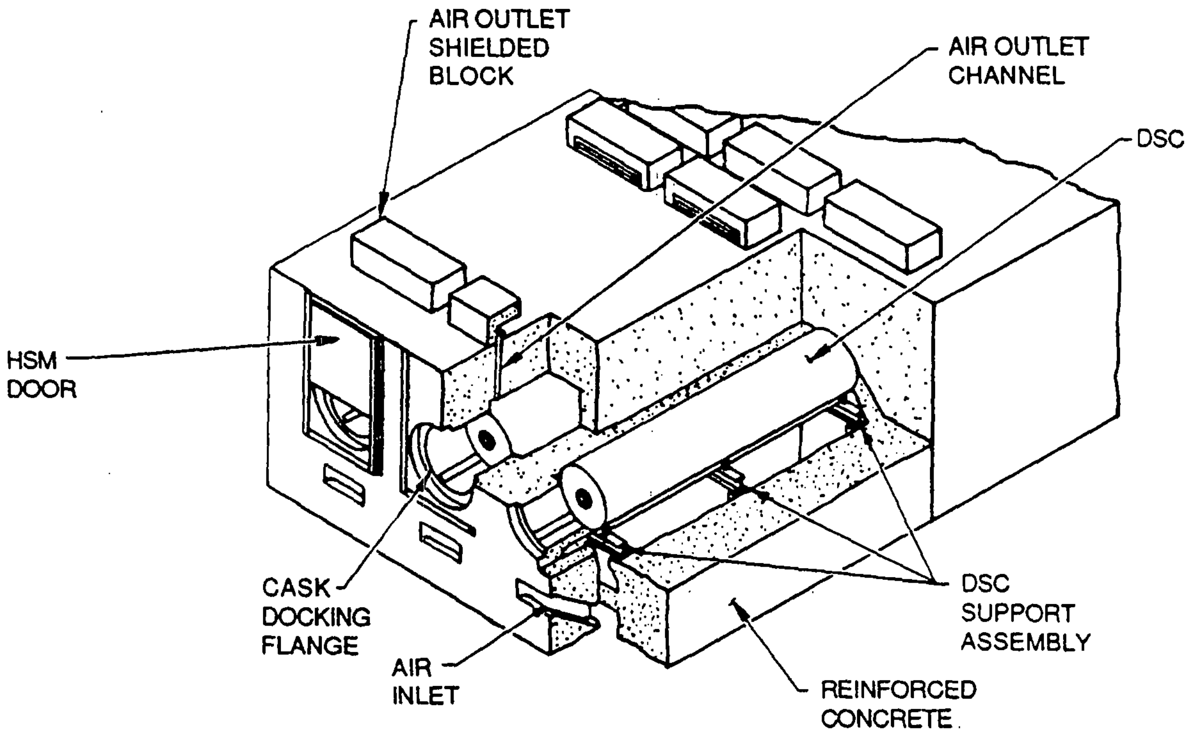
Table 1.2-4

PRIMARY DESIGN PARAMETERS FOR
OPERATING THE NUHOMS-24P SYSTEM

System	Parameters	Value
Transfer Cask	Nominal Cavity Diameter Nominal Cavity Length Empty Weight Payload Capacity Heat Rejection Capacity Contact Dose Rate Materials of Construction	1.73m (68 in.) 4.74m (186.85 in.) 45,400 kg (100,000 lb.) 40,900 kg (90,000 lb.) 15.8 kw ≤200 mrem/hr. Stainless & Carbon Steel, Lead Gamma Shielding, Liquid/Solid Neutron Shielding
Cask Handling	Lifting Yoke & Plant Crane. Oriented Vertical to Horizontal Using Crane	100 Ton Lifting Capacity
Cask Top Cover Plate	Remove & Replace with Cask in Horizontal Position	Porta-crane
Transport Trailer and Cask Support Skid	Towed by Tractor or Truck	Compatible with Trailer Design & Site Road Surfaces and Gradients
	Capacity (Trailer) (Skid)	109,000 kg (120 tons) 100,000 kg (110 tons)
	Positioning Capability	Capable of Positioning Transfer Cask Vertically, Axially and Laterally to Align with HSM Access Opening
HSM Door	Remove and Replace	Porta-Crane



SINGLE STAND-ALONE HSM
(FEATURES ARE SAME AS THOSE IDENTIFIED BELOW)



ARRAY OF TWO TO TWENTY HSMs

Figure 1.2-1

STAND-ALONE AND INTERCONNECTED HSMs

1.3 General Systems Descriptions

The primary components, structures and equipment which make up the NUHOMS system are listed in Table 1.3-1. The following subsections briefly describe the design features and operation of these NUHOMS system elements.

1.3.1 Systems Descriptions

1.3.1.1 Dry Shielded Canister The principal design features of the NUHOMS-24P DSC are listed in Table 1.3-1 and shown in Figure 1.3-1. Table 1.2-2 lists the capacity, dimensions and design parameters for the NUHOMS-24P DSC. The stainless steel cylindrical shell, and the top and bottom cover plate assemblies form the pressure retaining containment boundary for the spent fuel.

The component parts of the internal basket assembly of the DSC are listed in Table 1.3-1 and shown in Figure 1.3-1. The basket assembly contains a stainless steel guide sleeve for each fuel assembly. As discussed in Section 3.3.4, the criticality analysis performed for the NUHOMS-24P system accounts for fuel burnup and demonstrates that borated neutron absorbing material is not required in the basket assembly of the NUHOMS-24P DSC for criticality control.

Structural support for the guide sleeves in the lateral direction is provided by circular spacer disk plates. The DSC basket is designed so that there is one spacer disk at each fuel assembly grid spacer location. Longitudinal support for the DSC basket is provided by four support rods which extend over the full length of the DSC cavity and bear on the top and bottom cover plate assemblies.

The DSC is equipped with two shielded end plugs so that the radiation dose at the ends is minimized for drying, sealing, and handling operations. The shielded end plugs are constructed of a stainless steel casing with a poured lead core.

The DSC has double, redundant seal welds which join the shell and the top and bottom end plug and cover plate assemblies to form the containment boundary. The bottom end assembly containment boundary welds are made during fabrication of the DSC. The top end assembly containment boundary welds are made after fuel loading. Also, all top plug penetrations (siphon and vent ports) are redundantly sealed after fuel loading operations are complete. This assures that no single failure of the DSC top or bottom end assemblies will breach the DSC pressure boundary. Furthermore, there are no credible accidents which could breach the the primary containment pressure boundary of the DSC.

Subcriticality during wet loading, drying, sealing, transfer, and storage operations is maintained through the geometric separation of the fuel assemblies by the DSC basket assembly and the neutron absorbing capability of the DSC materials of construction (i.e., stainless steel).

1.3.1.2 Horizontal Storage Module An isometric view of a single HSM and an array of HSMs is shown in Figure 1.2-1. Each HSM provides a self-contained modular storage facility for spent fuel contained in a DSC as illustrated in Figure 1.3-1a. The HSM is constructed from reinforced concrete, structural steel sections and steel plate. The thick concrete roof and walls of the HSM provide substantial neutron and gamma shielding. Average contact doses for the HSM are less than 20 mrem/hour and are less than 100 mrem/hour at vent penetration and access door locations.

The nominal thickness of the HSM roof and exterior walls of an HSM array for biological shielding is three feet. Sufficient shielding between HSMs in an HSM array, in order to prevent scatter in adjacent HSMs during loading and retrieval, is provided by two-foot thick concrete walls.

The HSM provides a means of removal of spent fuel decay heat by a combination of radiation, conduction and convection. Ambient air enters the HSM through a ventilation inlet opening in the lower front wall of the HSM and passes through a shielded plenum in the interior of the HSM. Ventilation air exits the plenum at three locations and circulates around the DSC and the heat shield and then exits the HSM through two outlet openings in the HSM roof slab. Decay heat is rejected from the DSC to the HSM air space by convection and then is removed from the HSM by a natural circulation air flow. Heat is also radiated from the DSC surface to the heat shield and HSM walls where again the natural convection air flow and conduction through the walls removes the heat. Figure 1.3-2 shows the ventilation flow paths for the DSC and the HSM. The passive cooling system for the HSM was designed to assure that peak cladding temperatures during long term storage are less than 340°C (644°F).* As discussed in Section 3.3.7, current research has shown that maximum initial cladding temperatures for dry storage should be within the range of approximately 340°C to 380°C for spent fuel with a post discharge cooling time

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* SI units are used in the first three sections of this report. Where the general design features of the system are discussed, commonly used units are included in parentheses. For Sections 4 through 11, the units commonly used in the U.S. for the various design and analysis work described are used. SI units are provided in parentheses where meaningful.

of greater than five years (1.3). Research in the industry has also shown that cladding temperatures of up to 570°C (1058°F) may occur during short term operational and postulated accident conditions, without affecting the fuel cladding integrity (1.4).

The NUHOMS system HSMs provide an independent, passive system with substantial structural capacity for the dry storage of spent fuel assemblies. To this end, the HSMs are designed to ensure that normal transfer operations and postulated accidents or natural phenomena do not impair the DSC or pose a hazard to plant personnel.

The HSMs are constructed on a load bearing foundation which consists of a reinforced concrete basemat and compacted engineered fill. The HSMs are located in the plant's owner controlled area within a fenced, secured location with controlled access. The necessary civil work required will be performed on a site specific basis to prepare the storage site for the HSM foundation and access area, as well as adequately grading the site for drainage.

1.3.1.3 Transfer Cask

The transfer cask used in the NUHOMS system provides shielding and protection from potential hazards during the DSC closure operations and transfer to the HSM. Any suitable cask which meets the size, weight, shielding and structural requirements for the DSC and 10CFR72 can be used in the NUHOMS system. The NUHOMS-24P transfer cask described in this report meets these requirements.

The transfer cask for the NUHOMS-24P system, has a 4.75m (186.85 inches) long inner cavity, a 1.73m (68 inches) inside diameter and a maximum payload capacity of 40,900 kg (90,000 pounds). The NUHOMS-24P cask is designed to meet the requirements of 10CFR72 for on-site transfer of the DSC from the plant's fuel pool to the HSM. As shown in Figure 1.3-2a, the NUHOMS-24P transfer cask is constructed with two concentric cylindrical steel shells with a bolted top head cover plate and a welded bottom end assembly. The annulus formed by these two shells is filled with poured common lead to provide gamma shielding. The transfer cask also includes an outer steel jacket which is filled with a water-based solution for neutron shielding.

The top and bottom end assemblies also incorporate a solid neutron shield material. The transfer cask design provides sufficient shielding to ensure that the contact dose rate does not exceed 200 mrem/hour. Two lifting trunnions are provided for handling the transfer cask in the plant's fuel building using a lifting yoke and crane. Lower support trunnions are provided on the cask for rotating the transfer cask from/to the vertical and horizontal positions on the support skid/transport trailer. A cover plate is provided to seal the bottom hydraulic ram access penetration of the cask during fuel loading and transport to or from the HSM.

1.3.1.4 Transport Trailer The NUHOMS-24P transport trailer consists of a heavy industrial trailer with a capacity of 120 tons. The trailer carries the cask skid and the loaded transfer cask to and from the plant's fuel building and the HSMs of the ISFSI. The trailer is designed to ride as low to the ground as possible to minimize the HSM height and the transfer cask height during transport and DSC transfer operations. Figure 1.3-3 shows a heavy industrial trailer acceptable for use with the NUHOMS system. The trailer is equipped with four hydraulic leveling jacks to provide vertical travel for alignment of the cask and HSM. The trailer is typically towed by a conventional truck tractor or other suitable prime mover. The nominal trailer bed height during transport to the HSM site is 41 inches. At the HSM site, the trailer bed is raised to approximately 42 inches using the hydraulic jacks.

1.3.1.5 Cask Support Skid The NUHOMS system cask support skid is similar in design and operation to other cask transportation skids used for shipment of fuel. The major differences are:

1. There is no ancillary equipment mounted on the skid.

2. The skid is mounted on a surface with sliding support bearings and hydraulic positioners to provide alignment of the cask with the HSM. Brackets with locking bolts are provided to prevent movement during trailer towing.

3. The cask support skid is mounted on a low profile heavy industrial trailer.

The above noted cask support skid features are noted on Figure 1.3-4. The plant's fuel building crane is used to lower, the cask onto the support skid which is secured to the transport trailer. Specific details of this operation and the fuel building arrangement are plant specific and are covered by the provisions of 10CFR50. The cask support skid, when mounted on the transport trailer, is approximately 5'-0" high x 8'-6" wide x 16'-0" long. During transfer operations the bottom of the transfer cask is approximately 59" above the ground surface when secured to the support skid/transport trailer.

1.3.1.6 Horizontal Hydraulic Ram The horizontal hydraulic ram is a hydraulic cylinder with a capacity and a reach sufficient for DSC loading and unloading to and from the HSM. The NUHOMS-24P system has a ram with a capacity of 360 KN (80,000 lbf) and a reach of 6.55m (21.5 feet). Figure 1.3-5 shows the NUHOMS hydraulic ram system. The design of the ram support system provides a direct load path for the hydraulic ram reaction forces during DSC transfer. The design uses a tripod for vertical support and alignment at the rear of the ram, and a frame structure, bolted to the cask base, as a front support. This design provides positive alignment of the major components during DSC transfer. During DSC transfer the ram reaction forces will be transferred through the frame system into the transfer cask, and from the cask to the HSM, through the cask restraining system shown in Figure 4.2-6.

<u>1.3.1.7 System Operation</u> The primary operations (in sequence of occurrence) for the NUHOMS system are:

1. Cask Preparation
2. DSC Preparation
3. Placement of DSC in Cask
4. Fill with Water and Seal DSC/Cask Annulus
5. Cask Lifting and Placement in Pool
6. DSC Spent Fuel Loading
7. DSC Top Lead Plug Placement
8. Cask Lifting from Pool
9. Inner DSC Seal Weld Application
10. DSC Drying and Helium Backfilling
11. Outer DSC Seal Weld Application
12. Cask/DSC Annulus Draining and Top Cover Plate Placement
13. Placement of Cask on Transport Skid/Trailer
14. Transport of Loaded Cask to HSM
15. Cask/HSM Preparation
16. Transfer of DSC into HSM
17. Storage
18. Retrieval

The operations are shown schematically in Figure 1.3-6 and are described in the following paragraphs. These descriptions are intended to be generic. Plant specific requirements may affect these operations and will be addressed in 10CFR72 site license applications.

<u>Cask Preparation:</u> Cask preparation includes exterior washdown and interior decontamination. These operations are done on the decontamination pad/pit outside the fuel pool area. The operations are standard cask operations and have been performed by numerous utility personnel and others in the nuclear industry. Detailed procedures already exist for these operations and will be described in specific site licenses.

DSC Preparation: The internals and externals of the DSC are thoroughly washed. This ensures that the newly fabricated DSC will meet existing plant-specific criteria for placement in the spent fuel pool.

Placement of DSC in Cask: The empty DSC is inserted into the cask. Proper alignment is assured by visual inspection of the alignment match marks on the DSC and cask.

Fill with Water and Seal Cask/DSC Annulus: The cask and DSC inside the cask are filled with water. This prevents an in-rush of pool water when they are placed in the pool. The cask/DSC annulus is then sealed. This will also reduce (if not prevent) contamination of the DSC outer surface by the pool water.

Cask Lifting and Placement in Pool: The water-filled cask, with the DSC inside, is then lifted into the fuel pool and positioned in the cask pit area.

DSC Spent Fuel Loading Spent fuel assemblies are placed into the DSC basket. This operation is identical to that presently used at plants for cask loading.

DSC Top Lead Plug Placement This operation consists of placing the DSC top shielded end plug assembly onto the DSC using the plant's crane.

Lifting Cask from Pool: The loaded cask is lifted out of the pool and placed (in the vertical position) on the drying pad in the decon pit. This operation is identical to that presently used for cask lifting operations.

Inner DSC Sealing: Using a pump, the water level in the DSC is lowered below the inside surface of the upper DSC shielded end plug. A fillet weld is made between the edge of the top shielded end plug and the DSC shell. This weld provides the primary seal for the DSC.

DSC Drying and Backfilling: A pressure line is connected to the DSC vent port and pressurized with helium. The remaining liquid water in the DSC cavity is forced out the siphon tube and routed back to the pool or to the plant's liquid radwaste processing system, as appropriate. The DSC is then evacuated to remove the residual liquid water and water vapor in the DSC cavity. The vacuum drying process is based on operational experience with similar spent fuel storage containers.

The cavity is evacuated in steps, with hold points at each pressure plateau to prevent the formation of ice in the DSC or in the vacuum lines. Hold points also allow observation of the system behavior indicating the presence or absence of liquid water in the DSC. The system is evacuated by steps to a pressure of less than three torr. When the system pressure has stabilized below three torr, the DSC is backfilled with one atmosphere of helium and re-evacuated. The second backfill and evacuation is a rinsing procedure which removes essentially all the remaining water vapor in the cavity.

After the second evacuation, the DSC is again backfilled with helium, this time to a pressure of 1.5 atmospheres, and a helium leak check of the shield plug seal weld performed. The helium pressure is then reduced to 1.2 atmospheres, the helium lines removed, and the two DSC penetrations weld sealed.

<u>Outer DSC Sealing</u> After helium backfilling, the DSC top cover plate is positioned and a seal weld is made between the cover plate and the DSC shell. Together with the inner seal weld, this weld provides a redundant seal at the upper end of the DSC. The lower end has redundant seal welds which were provided and tested during fabrication.

<u>Cask/DSC Annulus Draining and Top Cover Plate Placement:</u> The cask is drained, removing all the demineralized water from the cask/DSC annulus. After draining, clean demineralized water is flushed through the cask/DSC annulus to remove any contamination left on the DSC exterior. A swipe is then taken over the DSC exterior at the DSC upper surface and one foot below the DSC head. The transfer cask top cover plate is then put in place using the plant's crane. The cask lid is then bolted closed and secured for subsequent transfer operations.

<u>Placement of Cask on Transport Trailer Skid:</u> The cask is then lifted onto the cask support skid. The plant's crane is used to reorient the cask from a vertical to a horizontal position. The cask is then secured to the skid and readied for the subsequent transport operations.

<u>Transport of Loaded Cask to HSM:</u> Once loaded and secured, the transport trailer is towed to the HSM along a predetermined route on a prepared road surface. Upon entering the HSM secured area, the cask is generally positioned and aligned with the particular HSM in which a DSC is to be transferred.

<u>Cask/HSM Preparation</u> At the HSM storage area, the cask top cover plate and is removed. The transfer trailer is backed into position and the HSM door is raised and removed. An optical alignment system and the skid hydraulic positioning system are used for the final alignment and docking of the cask with the HSM.

Loading DSC into HSM: After final alignment of the cask, HSM, and hydraulic ram, and removal of the ram access penetration cover plate on the cask, the DSC is pushed into the HSM by the hydraulic ram (located at the rear of the cask).

Storage After the DSC is inside the HSM, the hydraulic ram is released from the DSC and withdrawn through the cask. The transfer trailer is pulled away, the DSC seismic restraint installed, and the HSM shield door closed and secured in place. The DSC is now in safe storage within the HSM.

Retrieval For retrieval, the cask is positioned as previously described and the DSC is transferred from the HSM to the cask. The hydraulic ram is used to pull the DSC into the cask. All transfer operations are performed in the same manner as previously described. Once back in the cask, the DSC with its SFAs is ready for return to the plant fuel pool and/or transfer to a shipping cask for shipment to a repository or another storage location.

1.3.2 Storage Mode and Arrangement of Storage Structures

The DSC, containing the SFAs, is transferred to, and stored in, the HSM in the horizontal position. Multiple HSMs are interconnected to form arrays which are grouped together to form units whose size is determined to meet plant-specific needs. Units of HSMs are arranged at the storage installation site on a concrete pad(s) with the entire area enclosed by a security fence. HSM units can be arranged directly adjacent to each other since the decay heat for each HSM is primarily removed by internal natural circulation flow and not by conduction through the HSM walls. Figure 1.3-7 shows a typical layout for an at-reactor fuel storage installation for approximately 20 years capacity using the NUHOMS-24P system. The parameters of interest in planning the installation layout are the configuration of the HSM unit and an area in front of each HSM to provide adequate space for backing and aligning the transport trailer.

Table 1.3-1

PRIMARY COMPONENTS, STRUCTURES AND EQUIPMENT
FOR THE NUHOMS-24P SYSTEM

Dry Shielded Canister

- DSC Basket
  - Guide Sleeves (24)
  - Spacer Disks (8)
  - Support Rods (4)
- DSC Shell
- Shielded End Plugs (Top and Bottom)
- Cover Plates (Top and Bottom)
- Drain and Fill Port
- Grapple Ring

Horizontal Storage Module

- Reinforced Concrete Walls, Roof, and Basemat
- DSC Structural Steel Support Assembly
- DSC Seismic Retainer
- Cask Docking Flange and Tie-Down Restraints
- Heat Shield
- Shielded Access Door
- Ventilation Air Openings (One Inlet, Two Outlets)
- Shielded Ventilation Air Inlet Plenum
- Ventilation Air Outlet Shielding Blocks

Transfer Cask

- Cask Structural Shell Assembly
- Bolted Cover Plate Assembly

Table 1.3-1

PRIMARY COMPONENTS, STRUCTURES AND EQUIPMENT

FOR THE NUHOMS-24P SYSTEM

(Concluded)

Transfer Cask (Continued)

- Upper Lifting Trunnions
- Lower Support Trunnions
- Lead Gamma Shielding
- Neutron Shield Assembly
- Ram Access Penetration Cover Plate and Temporary Shield Plug

Transport Trailer

- Heavy Industrial-Grade Trailer
- Cask Support Skid
- Skid Positioning and Alignment System

Hydraulic Ram

- Hydraulic Cylinder and Supports
- Hydraulic System
- Grapple Assembly

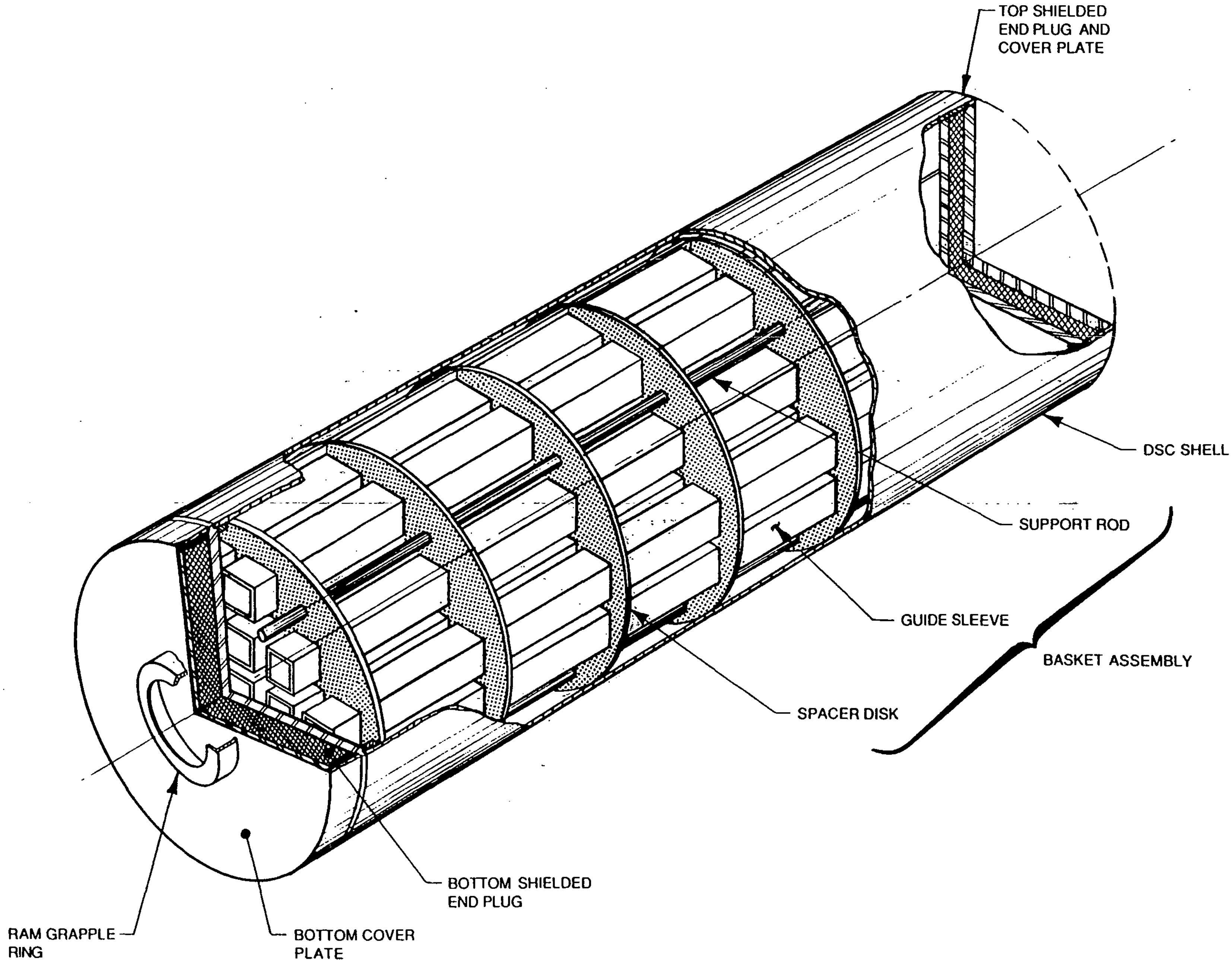


Figure 1.3-1

NUHOMS-24P DRY SHIELDED CANISTER ASSEMBLY COMPONENTS

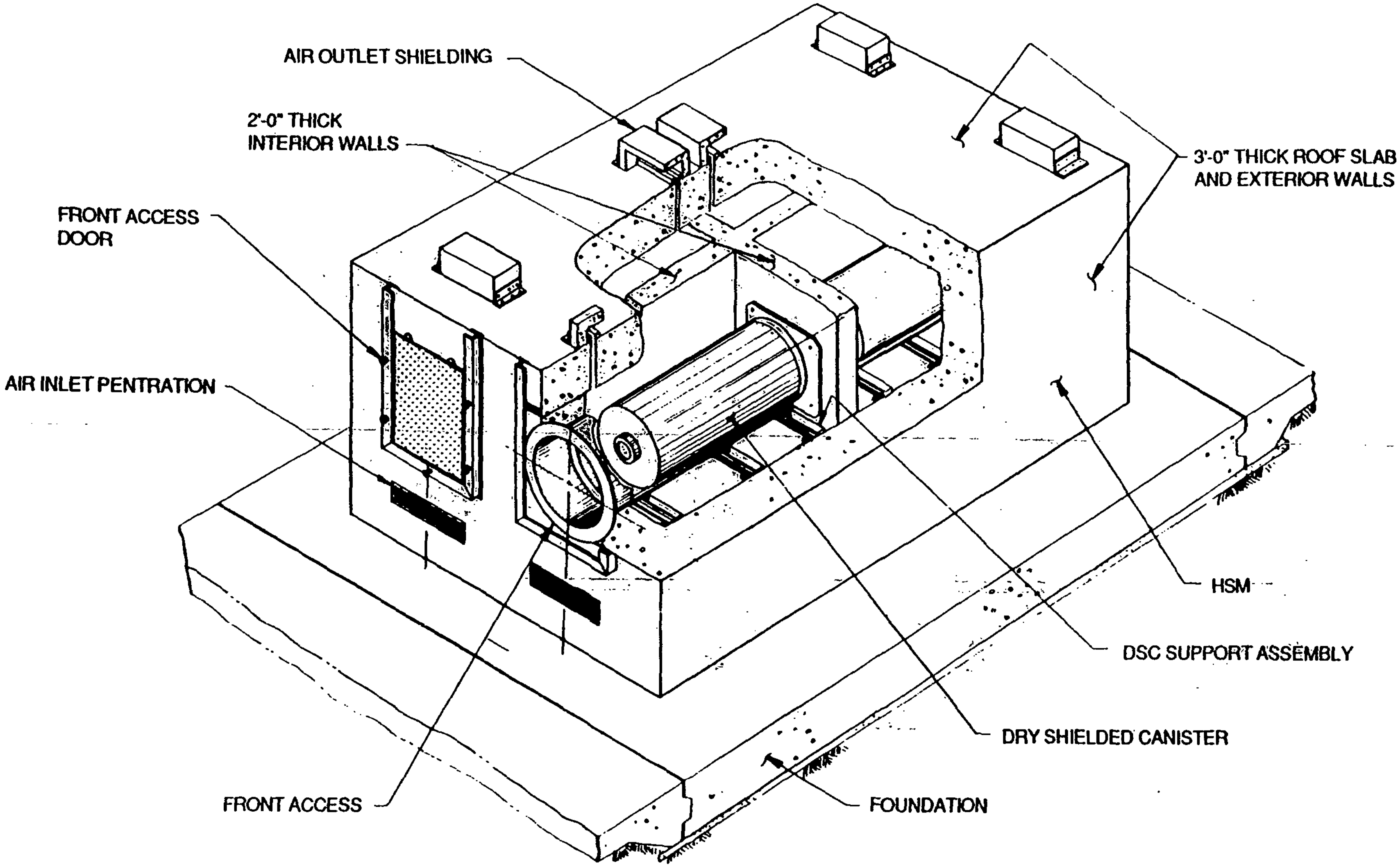


Figure 1.3-1a

NUHOMS-24P HORIZONTAL STORAGE MODULE COMPONENTS

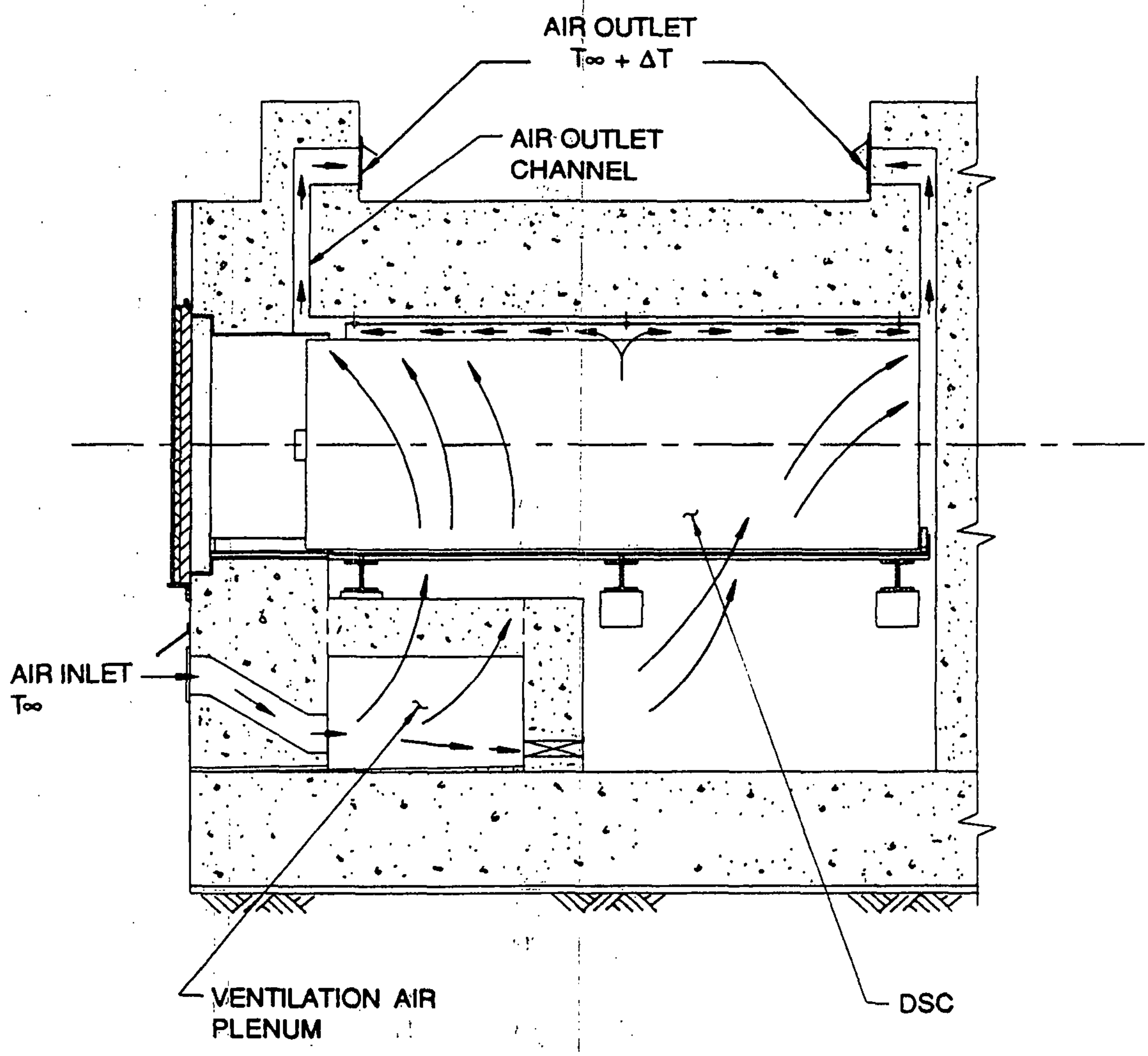


Figure 1.3-2

<u>HSM AIR FLOW DIAGRAM</u>

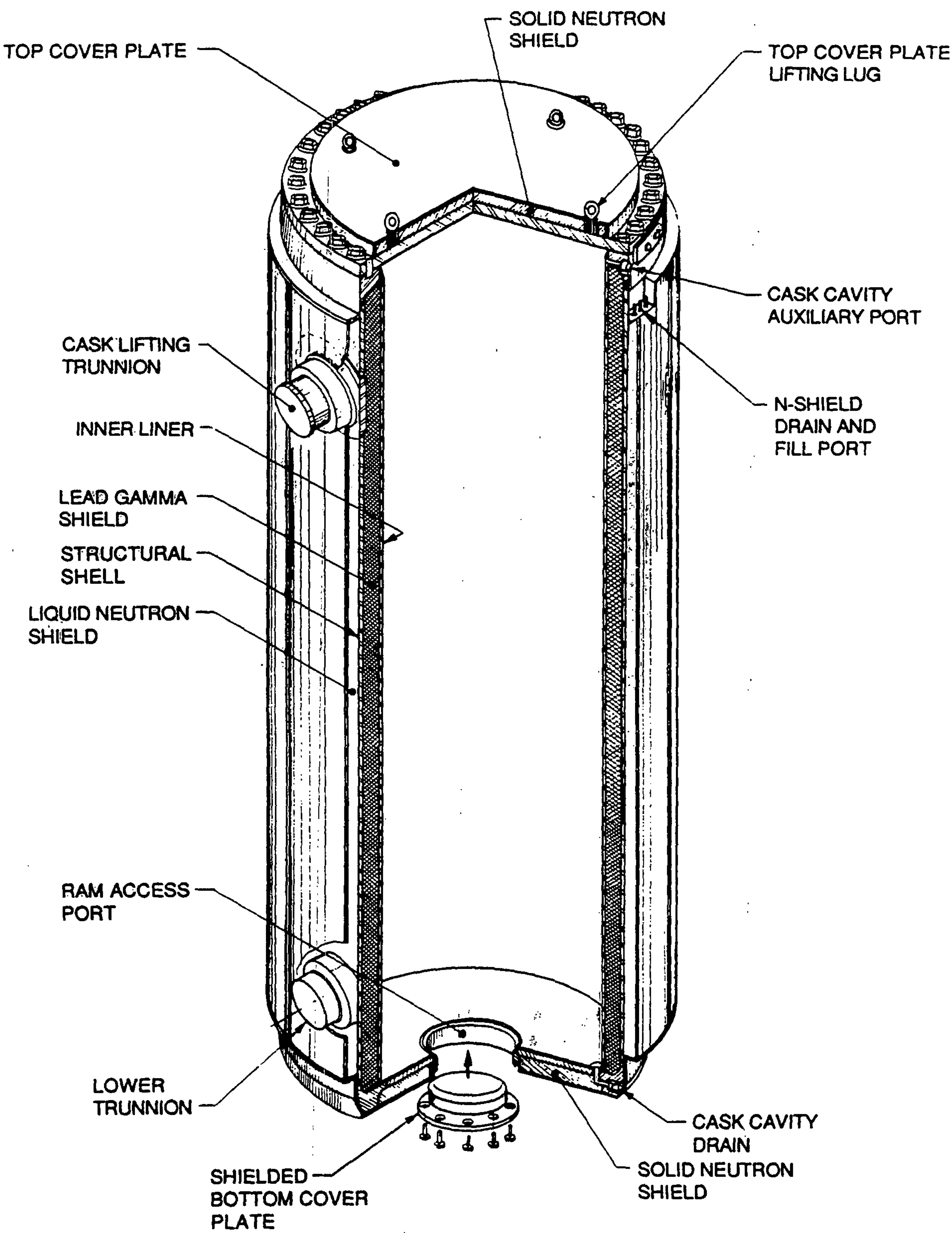


Figure 1.3-2a

NUHOMS ON-SITE TRANSFER CASK

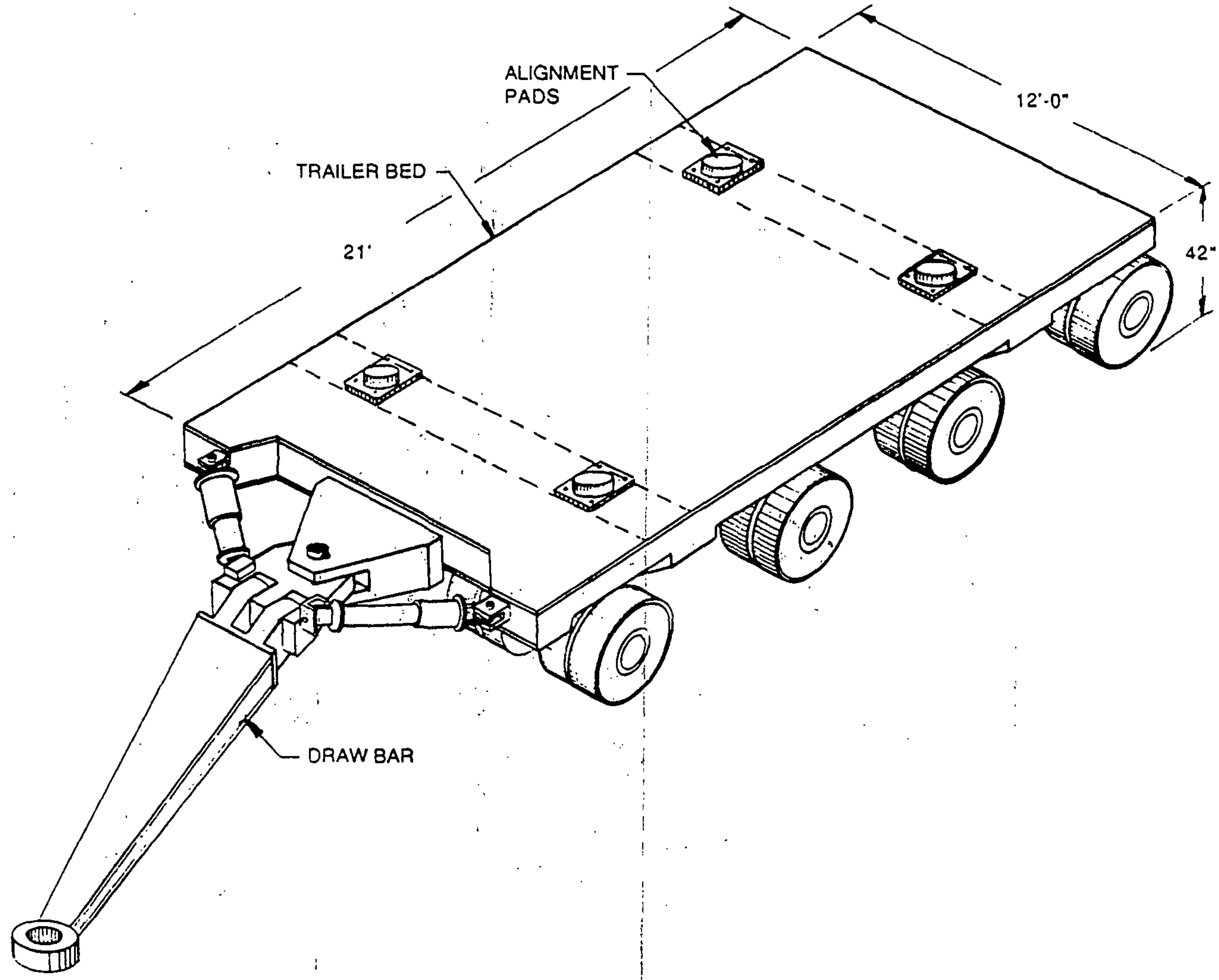


Figure 1.3-3

TYPICAL TRANSPORT TRAILER FOR NUHOMS

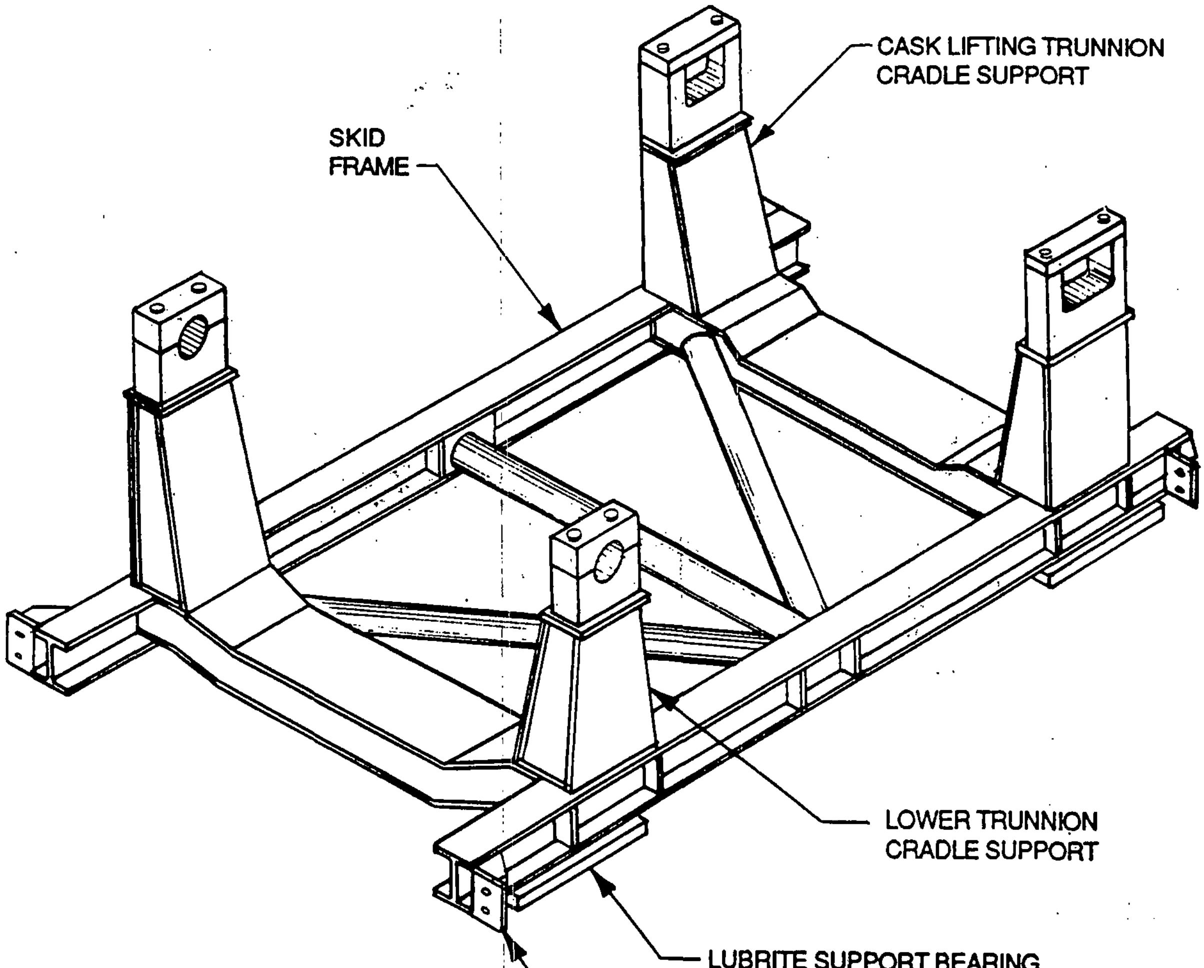


Figure 1.3-4

TYPICAL CASK SUPPORT SKID FOR THE NUHOMS SYSTEM

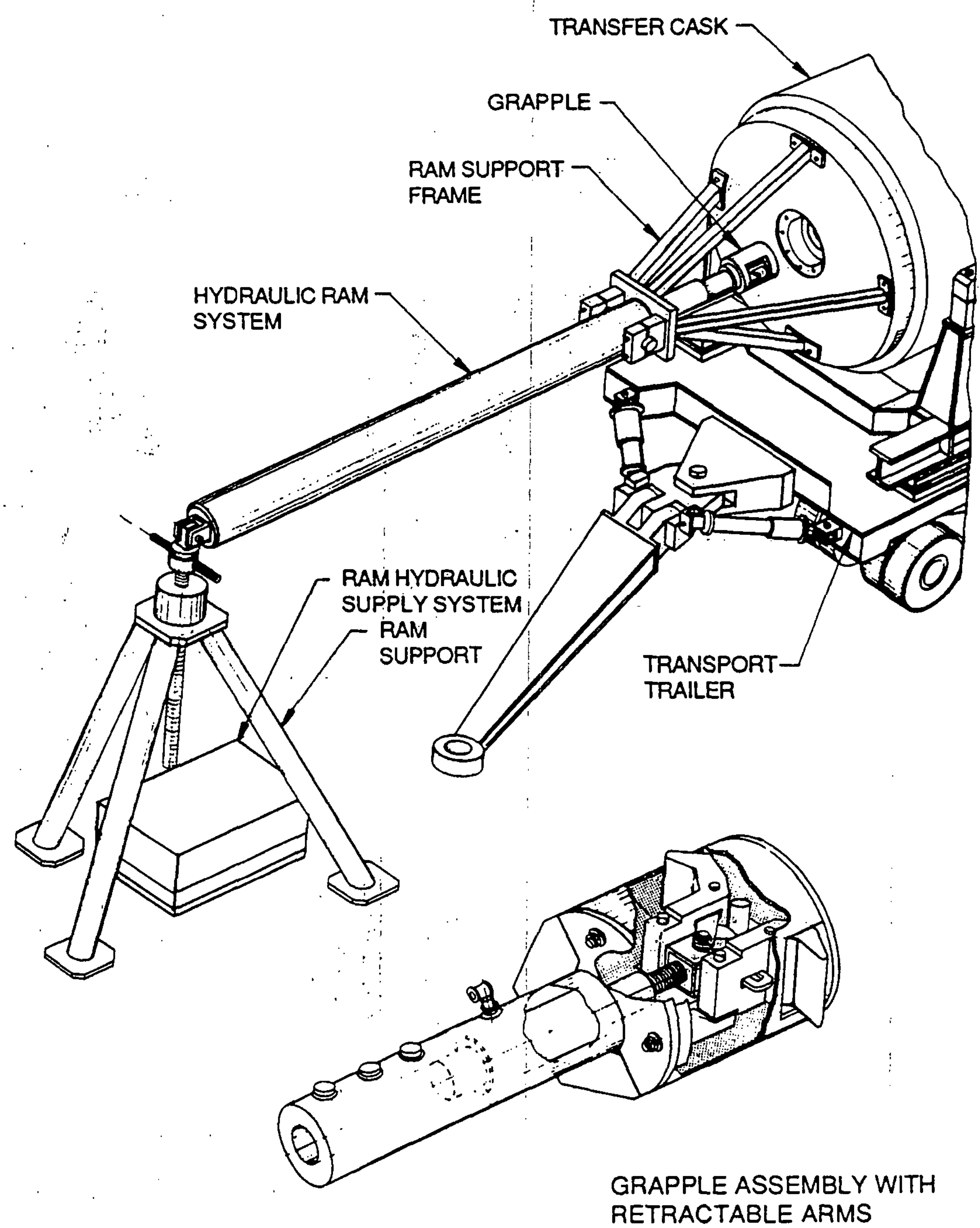


Figure 1.3-5

TYPICAL HYDRAULIC RAM SYSTEM FOR NUHOMS

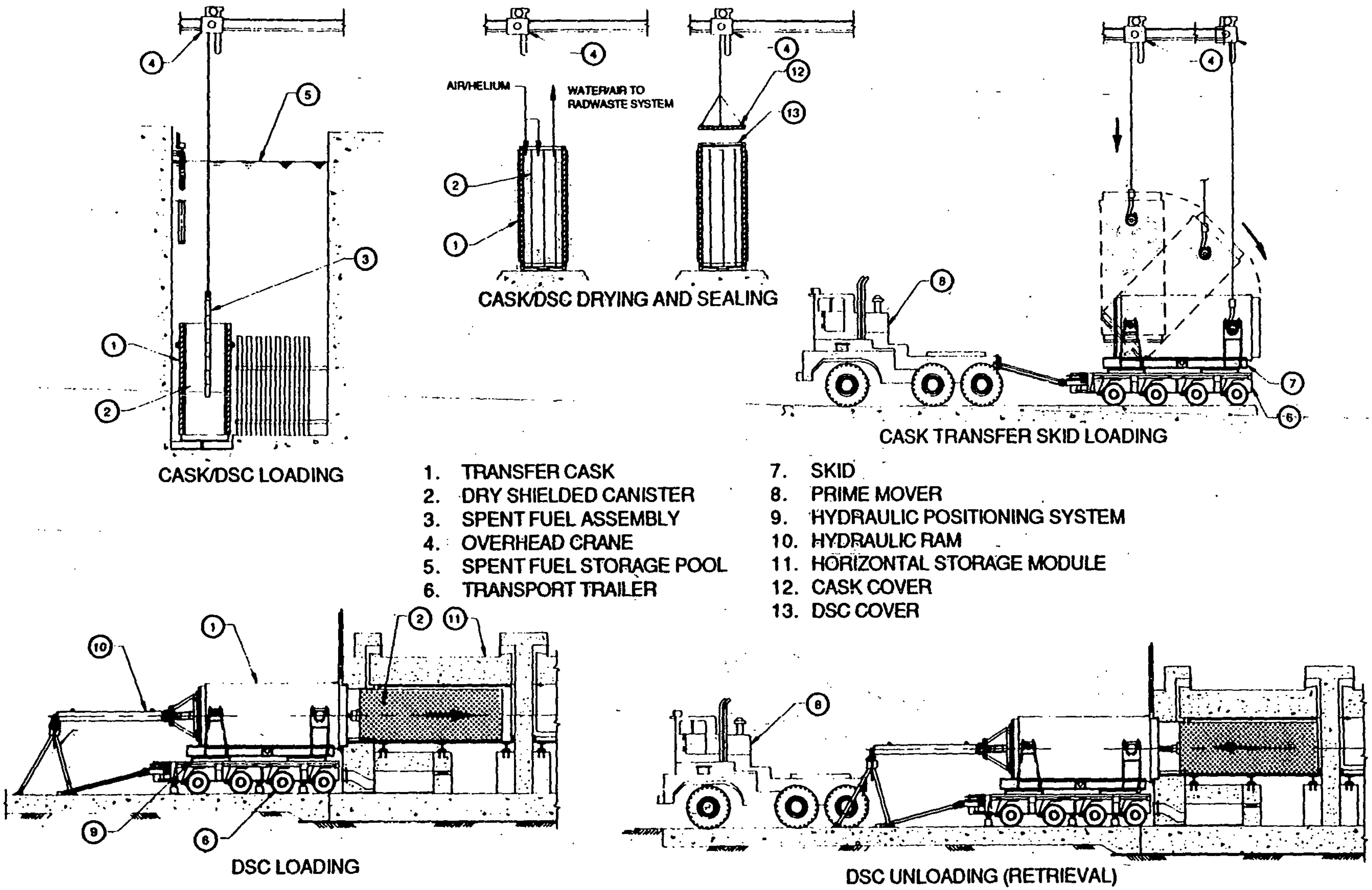
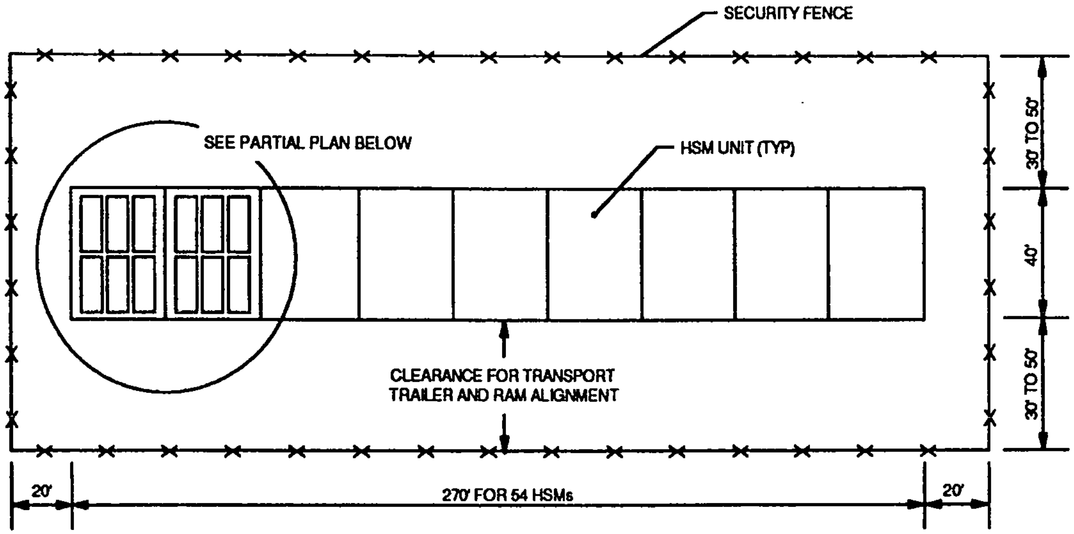


Figure 1.3-6

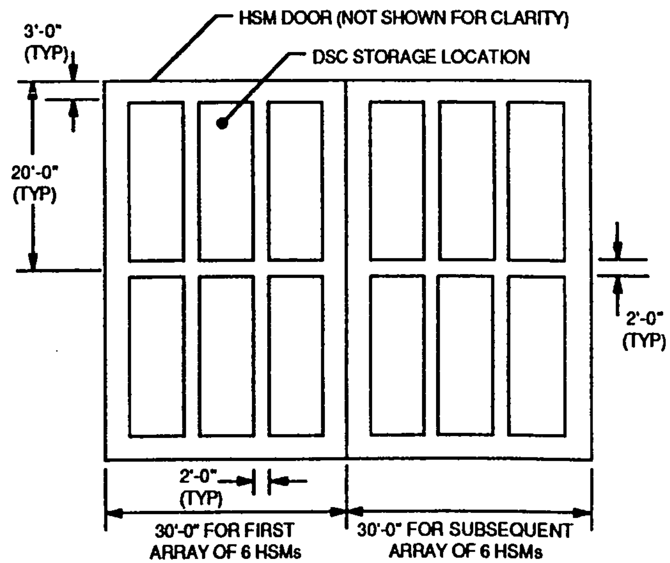
PRIMARY OPERATIONS FOR THE NUHOMS SYSTEM



ISFSI HSM SITE PLAN

Note:

HSM units with an array size of 2 x 3 are shown here for the purpose of illustration. HSM units may have variable array sizes up to 2 x 10.



ISFSI HSM PARTIAL PLAN

Figure 1.3-7

TYPICAL NUHOMS-24P ISFSI SITE LAYOUT FOR 20 YEARS STORAGE

1.4 Identification of Agent and Contractors

The prime contractor for design and procurement of the NUHOMS-24P system components is NUTECH Inc. of San Jose, California. NUTECH will subcontract the fabrication and on-site construction to qualified firms on a project specific basis.

The generic design activities for the NUHOMS-24P Topical Report (NUH-002, NRC Project No. M-49) were performed by NUTECH and by Duke Power Company, Inc. NUTECH was responsible for the design of the DSC, the on-site transfer cask, and the associated transfer equipment. Duke Power Company, Inc. was responsible for the design of the HSM and the criticality analysis.

1.5 Material Incorporated by Reference

The NUHOMS Topical Report, inclusive of Revision 1A for the NUHOMS-07P system, does not make direct reference to any other previously licensed documents. The NUHOMS-07P information is deleted from this NUHOMS-24P Topical Report amendment for clarity. Information has been added to document the generic design criteria and safety analysis for the NUHOMS-24P system. Editorial changes have also been made to add clarity where necessary. All revised text in this NUHOMS-24P Topical Report revision has been clearly identified with a vertical line in the right hand margin of each page. Where material has not been revised (i.e., no vertical line appears in the right hand margin of the page), it is identical to the previous revision of the NUHOMS-24P Topical Report. The Quality Assurance Program description contained in Section 11 of the NUHOMS-07P Topical Report (NUH-001, Revision 1A) is incorporated by reference in Section 11 of this NUHOMS-24P Topical Report.

1.6 REFERENCES

1.1 U.S. Government, "Licensing Requirements for the Storage of Spent Fuel In An Independent Spent Fuel Storage Installation," Title 10 Code of Federal Regulations, Part 72, Office of the Federal Register, Washington, DC (1984).

1.2 U.S. Nuclear Regulatory Commission, "Standard Format and Content for the Safety Analysis Report For An Independent Spent Fuel Storage Installation (Dry Storage)," Regulatory Guide 3.48, U.S. NRC, (1981).

1.3 Levy, I. S., et al. 1987. Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas, May, 1987, PNL-6189 (UC-85), Pacific Northwest Laboratory, Richland, Washington.

1.4 Johnson, A. B. Jr., Gilbert, E. R., Technical Basis for Storage of Zircaloy-Clad Fuel in Inert Gases, PNL-4835, September, 1983, Pacific Northwest Laboratory, Richland, Washington.

1.5 Deleted.

1.6 Deleted.

1.7 Deleted.

1.8 Deleted.

1.9 Deleted.

2.0 SITE CHARACTERISTICS

Information on plant specific sites for a NUHOMS system ISFSI will be covered in site specific license applications. This shall include a geologic investigation for the proposed HSM site.

3.0 PRINCIPAL DESIGN CRITERIA

3.1 Purpose of Installation

The NUHOMS system provides an ISFSI for horizontal, dry storage (in a helium atmosphere) of SFAs in a high integrity stainless steel DSC which is placed inside a massive reinforced concrete HSM. The function of the DSCs and HSMs is to provide for the safe, controlled, long-term storage of SFAs.

It is feasible to adapt the NUHOMS system to store any of the various types of commercial fuel assemblies which presently reside in spent fuel pools. This topical report amendment addresses PWR spent fuel. BWR spent fuel will be addressed as a future TR amendment or in site specific license applications. The following subsection provides a description of the spent fuel assemblies to be stored.

The design storage capacity of a single NUHOMS-24P DSC and HSM is 24 PWR fuel assemblies. Multiple HSMs can be grouped together to form units which provide the needed storage capacity consistent with site space limitations and reactor fuel discharge rates.

3.1.1 Material To Be Stored

Table 3.1-1 lists the principal design parameters for the B&W 15x15 PWR spent fuel selected as the design basis for the design of the NUHOMS-24P system documented in this TR. Therefore, this TR covers fuels which are enveloped by the B&W 15x15 fuel and meet the fuel acceptance criteria specified herein. The following acceptance criteria is established for fuel other than the TR design basis B&W 15x15 fuel.

1. For shielding, the gamma and neutron source strengths and resulting contact doses must be less than or equal to those of this TR.

2. For thermal, the total decay heat power per DSC and the fuel loading temperature limits must be less than or equal to those of the TR.

3. For criticality, fuel/plant specific analysis which demonstrates the applicability of the TR burnup curve are to be performed using methods generically approved for this TR.

4. For structural, the fuel weight per spacer disc and the total weight of the DSC and transfer cask must be less than or equal to those of the TR. Also, the postulated cask

drop height and drop surface conditions must meet the limits specified in this TR.

Plants with fuel that does not meet this criteria will require more detailed plant specific analysis which will form the basis for site license applications. The parameters for candidate fuel assemblies are described further in the subsections which follow.

<u>3.1.1.1 Physical Characteristics</u> The NUHOMS-24P system can be adapted to store the types of PWR fuel assemblies shown in Table 3.1-2. The key physical parameters of interest are the weight, length, cross-sectional dimensions, and the axial distance between fuel assembly grid spacers. The values of these parameters form the basis for the mechanical and structural design of the DSC and its internals. The DSC and transfer cask designs for the NUHOMS-24P system presented in this TR are based on the B&W 15x15 fuel assembly parameters listed in Table 3.1-1. DSC internal basket assembly designs for the other fuel types listed in Table 3.1-2, will be similar with dimensional variations to suit the key parameters for a specific fuel assembly design.

<u>3.1.1.2 Thermal Characteristics</u> The key parameters utilized to determine the heat removal requirements for the NUHOMS system design is the SFA decay heat power. The total decay heat power per spent fuel assembly is dependent on the average burnup per assembly and the cooling time as indicated in Table 3.1-3. To a lesser extent, total decay heat power is dependent on the initial ^{235}U enrichment, specific power (MW/MTHM) and neutron flux energy spectrum. The total heat rejected to the DSC and HSM is conservatively taken to be less than or equal to 0.66 kilowatt per fuel assembly (15.84 kw/DSC) for the NUHOMS-24P design (3.1) as listed in Table 3.1-1.

For thermal characteristics, fuel assembly burnup and cooling time can be used to determine the acceptability of a candidate SFA for dry storage using the NUHOMS-24P system. Figure 3.1-1 shows a plot of decay heat versus time for various burnups. This figure was derived from ORIGEN calculations performed for a range of key fuel assembly parameters (Cases 1, 3, 6, and 8 from Appendix C of Reference 3.2). These results were verified by comparisons with other sources and calculational techniques (3.3, 3.4 and 3.5). Figure 3.1-1 is based on conservative estimates of the total decay heat for assemblies with specific powers of 37.5 MW/MTHM or less. As such, if the burnup and cooling time for an assembly are known, Figure 3.1-1 can be used to determine its acceptability for dry storage using the NUHOMS-24P system. Other methods, such as specific ORIGEN calculations for a candidate fuel assembly to determine calorimetry or burnup test measurements, are also acceptable for determining the acceptability of the SFA. These determinations may be included in site license applications.

3.1.1.3 Radiological Characteristics The principal design parameters for acceptable radiological characteristics and shielding design for the NUHOMS system are that the gamma and neutron source term values for the candidate spent fuel be equal to or less than those shown in Table 3.1-4. These limits, as with the thermal limits, may be met with a variety of initial enrichments, fuel burnup, and cooling time combinations. Further, since the radiological design criteria are based on the total DSC contents and not on individual assemblies, the hottest (radioactively) assembly can differ significantly from the average assembly as long as the source term values listed in Table 3.1-4 are not exceeded.

Three alternative measures can be taken by site license applicants to qualify candidate fuel assemblies for storage in the NUHOMS-24P system as follows:

1. The limits for three fuel management parameters including initial enrichment, burnup, and cooling time as specified in Table 3.1-1 must be met. Using these parameters as acceptance criteria, existing fuel records and plant specific procedures form the basis for controlling the selection and placement of candidate fuel assemblies.

2. The limits for the neutron and gamma source terms specified in Table 3.1-4 can be used directly. An engineering analysis can be performed by the license applicant to develop the source term for the candidate fuel (i.e., ORIGEN calculations).

3. In the event that calculations are performed leading to neutron and gamma source terms which are not bounded by those specified in Table 3.1-4, detailed dose calculations may be performed to demonstrate that contact doses for the cask and HSM are below the acceptance limits specified in this TR.

The criteria shown in Table 3.1-4 are based on ORIGEN calculations for PWR fuel with a 4% initial ^{235}U enrichment and a burnup of 40,000 MWd/MTHM at a specific power of 37.5 MW/MTHM (3.1, 3.2, and 3.44).

SFAs with an equivalent maximum initial ^{235}U enrichment of 4%, a burnup of less than 40,000 MWd/MTHM, a specific power of less than or equal to 37.5 MW/MTHM, and a post-discharge cooling time greater than or equal to ten years will have gamma and neutron source strengths less than, or equal to, the values used for design of the NUHOMS-24P system. Hence, these assemblies will be acceptable for storage in the NUHOMS-24P system presented in this topical report. Assemblies with a combination of lower initial enrichment, higher burnup, or shorter cooling times may also be acceptable and may be addressed in plant specific license applications as discussed above.

3.1.2 General Operating Functions

<u>3.1.2.1 Functional Overview of the Installation</u> For the NUHOMS system, SFAs are loaded into the DSC as discussed in Section 1.3. During loading, the DSC is resting in the cavity of the transfer cask, in the fuel pool cask laydown area. After removal from the pool, the DSC is dried and backfilled with helium. After drying, the DSC (still inside the transfer cask) is moved to the cask skid/trailer and transported to the HSM. The DSC is pushed from the transfer cask into the HSM by a hydraulic ram.

Once inside the HSM, the DSC and its payload of SFAs is in passive dry storage. Safe storage in the HSM is assured by a natural convection heat removal system, and massive concrete walls and slabs which act as biological radiation shielding. The storage operation of the HSMs and DSCs is totally passive. No active systems are required.

<u>3.1.2.2 Handling and Transfer Equipment</u> The handling equipment designs required to implement the NUHOMS system will be site specific. This equipment includes a cask handling crane (3.36) at the reactor fuel pool, a cask lifting yoke, a transfer cask, a cask support skid and positioning system, a low profile heavy-haul transport vehicle and a hydraulic ram system. This equipment will be designed and tested to the applicable governmental and industrial standards and will be maintained and operated according to the manufacturer's specifications. Performance criteria for this equipment, excluding the fuel building cask handling crane, is given in the following sections. The criteria are summarized in Table 3.1-5.

<u>Cask Criteria:</u> Virtually any large cask licensed for shipment of spent fuel under the rules of 10CFR71 (3.22) could be used as a NUHOMS system on-site transfer cask by using an appropriate cask liner, or by making minor modifications to the HSM and DSC design. Transfer casks licensed under 10CFR72, are also acceptable for on-site transfer of fissile materials, and could be utilized in a NUHOMS system if all components of that system

remain within the plant's site boundary. The on-site transfer cask developed for the NUHOMS-24P DSC is the basis of the operations described in Sections 4 and 5, and the analysis descriptions in Section 8 of this Topical Report.

The transfer cask used for the NUHOMS system has certain basic requirements. The DSC must be transferred from the plant's fuel pool to the HSM inside the transfer cask. The cask should provide neutron and gamma shielding adequate for biological protection at the outer surface of the cask. The cask should be capable of rotation, from the vertical to the horizontal position in an appropriate cradle or skid. The cask should have a lid or top cover plate that is removable in the horizontal position, or that can be fitted with an attachment allowing removal when the cask is oriented horizontally. The cask must be capable of rejecting the design basis decay heat load to the atmosphere assuming the most severe ambient conditions postulated to occur during normal, off-normal and accident conditions. For the NUHOMS-24P DSC, the cask should have a clear cylindrical cavity of 1.73m (68 in.) diameter and 4.75m (186.85 inches) in length and a cavity capacity of 41,000 Kg (90,000 pounds). The cask and the associated lifting yoke shall be designed and operated such that the consequences of a postulated drop satisfy the plant's current 10CFR50 licensing basis.

The NUHOMS-24P transfer cask is designed to meet the requirements of 10CFR72 for normal, off-normal and accident conditions. The NUHOMS-24P transfer cask is designed for the following conditions:

1.	Seismic	Reg. Guide 1.60 and 1.61
2.	Operational Handling Loads	ANSI/ANS-57.9-1984 (3.36)
3.	Accident Drop Loads	ANSI/ANS-57.9-1984
4.	Thermal and Dead Loads	ANSI/ANS-57.9-1984
5.	Tornado Wind Loads	Reg. Guide 1.76 (3.7)

Extreme environmental conditions due to tornado generated missiles and floods are not considered to be credible because of the infrequent use (normally four to five times a year) and short period of time (normally about a day) for which the transfer cask is utilized for DSC transfer operations. The transfer cask has been designed for tornado wind loads, however, in accordance with 10CFR72.72. Since the DSC and the double closure welds on the DSC form the pressure containment boundary for the spent fuel materials, the transfer cask need not be designed for internal pressure. Seals are, however, provided for the bottom ram access penetration closure plate to prevent the ingress of water during DSC loading. The designs of other 10CFR72 casks and transport systems proposed for use at a storage site utilizing the NUHOMS system can be described in site specific license applications.

Cask Support Skid and Positioning System Criteria: The cask support skid and positioning system utilized in the NUHOMS system has a unique criterion in that it must be capable of transverse movement relative to the transport trailer. The transport trailer is a low profile, heavy-haul, truck-towed trailer. The cask support skid shall be capable of allowing rotation of the cask from vertical to horizontal. The skid shall not extend past the top end of the cask in order to allow the HSM cask docking collar to seat with the cask and prevent radiation streaming during DSC transfer.

Transverse movement is necessary to align the cask with the HSM so that the DSC transfer may be accomplished smoothly without sticking or binding. To illustrate this criterion, one design uses Lubrite spherical bearing plates supporting the skid and cask at the four corners of the skid. These bearing plates slide on wide flange beams supported by a trailer. Air pads, such as the Rolair system or Hilman multi-ton rollers may also be used. The amount of transverse motion required is a few inches. The motion system shall have positive locks which prevent any possibility of movement or load shifting during transport or DSC transfer.

Trailer Criteria: The heavy-haul vehicle used to transport the cask, skid and DSC from the spent fuel pool to the HSM is a low profile, truck-towed, multi-axle trailer. Such a trailer is presented here to illustrate how the NUHOMS system may be implemented and is not meant to preclude use of a railcar or other suitable heavy-haul vehicle. The principal criteria for the transport vehicle are the capacity to bear the weight of the cask and the support skid, plus the additional inertia forces associated with transport operations; to minimize the height and limit the orientation of postulated cask drop accidents, and the HSM height; and have the ability to adjust the height of the cask in order to achieve precise alignment with the HSM. This latter requirement may also be accomplished with the support skid positioning system.

Once the height adjustment is complete, the cask support skid is brought into solid contact with a firm ground surface. That is, any springing, such as may result from tires, coil springs or other flexible members, should be eliminated from the cask support system in order to prevent cask movement while the DSC is being transferred into the HSM.

Hydraulic Ram System Criteria: The hydraulic ram system consists of a hydraulic ram, mounted on a firm base, with a grapple at the outer end used to apply a push or pull force to transfer the DSC to and from the cask and HSM. The hydraulic ram should be capable of exerting sufficient force during the entire insertion and retrieval strokes to effect the transfer. The ram should have the capacity to move the DSC assuming a coefficient of friction of one. The ram will be limited, during normal operation, to exert a force no greater than 25% of the weight of the fully loaded DSC. The stroke of the ram should be sufficient to complete the transfer. The nominal required stroke for the NUHOMS-24P system is 6.55m (21.5 feet). The piston when fully extended should be able to withstand a compressive load equal to the weight of the loaded DSC. During DSC transfer operations, the ram will be firmly attached to the cask to transfer the reaction load during operations.

3.1.2.3 Waste Processing, Packaging and Storage Areas The only waste produced during the NUHOMS storage operations is potentially contaminated water drained from the DSC and transfer cask after fuel loading, and the water used to decontaminate the outer surface of the transfer cask after removal from the spent fuel pool. This water will be removed from the DSC and cask in the plant's fuel building and routed back to the fuel pool, or the plant's radwaste treatment system.

Likewise, the air and helium evacuated from the DSC during the drying operations will be collected and checked for radioactive releases. If clean, the gas can be released. If contaminated, the gas should be routed to the plant's off-gas processing system or filtered and released.

A limited amount of dry active waste is generated from temporary protective clothing and material used during fuel loading, DSC drying, and sealing operations. The details of these operations are site specific.

The only other waste generated by the NUHOMS system will be during decommissioning. This topic is addressed in Section 9.6.

Table 3.1-1

PRINCIPAL DESIGN PARAMETERS FOR MATERIAL
TO BE STORED IN NUHOMS-24P DSC

Parameter	Value
PHYSICAL PARAMETERS	
Assembly Length (w/o control components)	4.232 m (166.68 in.)
Assembly length (w/control components and allowance for irradiation growth)	4.394 m (173.00 in.)
Nominal Cross-Sectional Envelope	0.2168 m (8.536 in.)
Maximum Assembly Weight (w/control components)	763 Kg (1,682 lb.)
Nominal Center-to-Center Distance Between Grid Spacers	0.574 m (22.60 in.)
No. of Assemblies per DSC	24
THERMAL CHARACTERISTICS (1)	
Decay Heat Power per DSC	≤15.84 kw
RADIOLOGICAL CHARACTERISTICS (1)	
Maximum Initial Uranium Content	472 Kg/assembly
Total Fissile Content (Equivalent ^{235}U percentage)	≤4.0%
Total Gamma Source per DSC	≤3.85E16 Mev/sec (1.11E17 photons/sec)
Total Neutron Source per DSC	≤3.715E9 neutrons/sec

Note:

1. The thermal and radiological characteristics of the design basis fuel for the NUHOMS-24P system are based on enveloping source term values which result from the storage of PWR fuel burned to 40,000 MWd/MTHM at 37.5 MW/MTHM and cooled for ten years. There are a number of combinations of initial enrichment, burnup, specific power, and cooling times which will result in acceptable radiological and thermal source term values. Acceptance of other fuel source term values can be demonstrated in site specific applications by performing ORIGEN (3.44) or other suitable calculations as described in Sections 3.1.1.2 and 3.1.1.3.

Table 3.1-2

GENERAL PWR FUEL ASSEMBLY DATA[1]

Vendor	Array Size/No. of Fuel Rods	Assembly/ Active Length (m)	Nominal Envelope (m) (in.)	Weight (kg) (Assembly/ Heavy Metal)	Nominal Distance Between Grid Spacers (m) (in.)
Babcock & Wilcox (B&W)	15 x 15/208	4.207/3.889	0.2168 (8.536)	688/464	0.5365 (21.12)
B&W	17 x 17/264	4.207/3.886	0.2168 (8.536)	681/454	0.5588 (22.00)
Combustion Eng. (CE)	14 x 14/176	4.636/3.810	0.2096 (8.25)	558/381	0.4272 (16.82)
CE	15 x 15/216	3.788/3.353	0.2117 (8.36)	614/405	0.3937 (15.50)
CE	16 x 16/236	4.529/3.810	0.2906 (8.25)	660/426	0.3763 (14.81)
Westinghouse (W)	14 x 14/179	4.084/3.658	0.1972 (7.763)	573/405	0.6652 (26.19)
W	15 x 15/204	4.097/3.658	0.2140 (8.426)	650/459	0.6652 (26.19)
W	17 x 17/264	4.097/3.658	0.2140 (8.426)	665/461	0.6205 (24.43)

Advanced Nuclear Fuels (ANF) (ANF manufactures all fuel types and their designs closely match the vendors listed above.)

Note:

1. This information is provided for general reference only. Fuels represent the maximum physical size and/or weight for each design.

Table 3.1-3[1]

TYPICAL VALUES OF BURNUP AND COOLING TIMES TO YIELD 0.66 KW/ASSEMBLY

Burnup (MWd/MTHM)	Cooling Time (Years)
18,000	5
33,000	9
46,000	24

Note:

1. Table 3.1-3 is based on information in references 3.2, 3.3, 3.4 and 3.5.

Table 3.1-4

RADIOLOGICAL CRITERIA FOR STORAGE OF MATERIAL IN THE NUHOMS-24P SYSTEM

Gamma Source Spectrum

E (Mean) [MeV]	Source Strength [Gamma/Sec. Assembly]	Source Strength [MeV/Sec. Assembly]
1×10^{-2}	1.10×10^{15}	1.10×10^{13}
3×10^{-2}	4.86×10^{14}	1.46×10^{13}
5.5×10^{-2}	2.73×10^{14}	1.50×10^{13}
8.5×10^{-2}	2.51×10^{14}	2.13×10^{13}
1.2×10^{-1}	1.52×10^{14}	1.82×10^{13}
1.7×10^{-1}	8.02×10^{13}	1.36×10^{13}
3.0×10^{-1}	9.06×10^{13}	2.72×10^{13}
6.5×10^{-1}	2.05×10^{15}	1.33×10^{15}
1.13	1.25×10^{14}	1.42×10^{14}
1.57	8.87×10^{12}	1.39×10^{13}
2.00	1.62×10^{10}	3.23×10^{11}
2.40	2.51×10^{10}	6.04×10^{10}
2.80	4.65×10^{9}	1.18×10^{9}
3.25	7.17×10^{8}	2.33×10^{9}
3.75	6.28×10^{6}	2.35×10^{7}
4.25	3.45×10^{6}	1.47×10^{7}
4.75	2.00×10^{6}	9.49×10^{6}
5.50	1.81×10^{6}	9.96×10^{6}
	4.62×10^{15}	1.60×10^{15}

Neutron Source Spectrum[1]

Energy Range [MeV]	Source Strength [Neutrons/Sec. Assembly]
6.43 --> 20	3.119×10^{6}
3.00 --> 6.43	3.561×10^{6}
1.85 --> 3.00	3.968×10^{7}
1.40 --> 1.85	2.223×10^{7}
9.00×10^{-1} --> 1.40	2.999×10^{7}
4.00×10^{-1} --> 9.00×10^{-1}	3.266×10^{7}
1.00×10^{-1} --> 4.00×10^{-1}	6.393×10^{6}
1.00×10^{-11} --> 1.00×10^{-1}	0
	1.548×10^{8}

Note:

1. See Table 7.2-1 for enveloping neutron source spectrum used for TR shielding calculations.

Table 3.1-5

NUHOMS TRANSFER EQUIPMENT CRITERIA

Component	Requirement	Criteria
Cask Interface	Overall Dimensions	2.17m dia. x 5.03m Length (85.5 in. dia. x 198 in. Length)
	Gross Weight	86,000 kg (190,000 lb.)
	Heat Rejection	15.8 kw total
	Orientation	Vertical to Horizontal
	Contact Dose	≤ 200 mrem/hr.
	Support Points	Upper Lifting Trunnions and Lower Support Trunnions
Cask Support Skid	Weight Capacity	Cask + DSC (100 tons)
	Cask Positioning	Horizontal Translation and Rotation About Vertical Axis
	Cask Orientation	Allow Vertical to Horizontal Rotation
	Length	Must not Extend Past Top Flange of Cask. Overall Length 16 ft.
	Support Points	Upper and Lower Trunnion Pillow Blocks
Transfer Trailer	Weight Capacity	Skid + Cask + DSC (120 tons)
	Length	6.40m (21 ft.)
	Cask Positioning	Vertical Translation at Each Corner
	Rigidity	Cask must be Solidly Supported During DSC Transfer Operation
Hydraulic Ram	Capacity	355,900 N (80,000 lbf) Push and Pull
	Load Limit	Maximum Force Must be Limitable
	Stroke	6.55m (21.5 feet)
	Base Mounting	Immobile During DSC Transfer

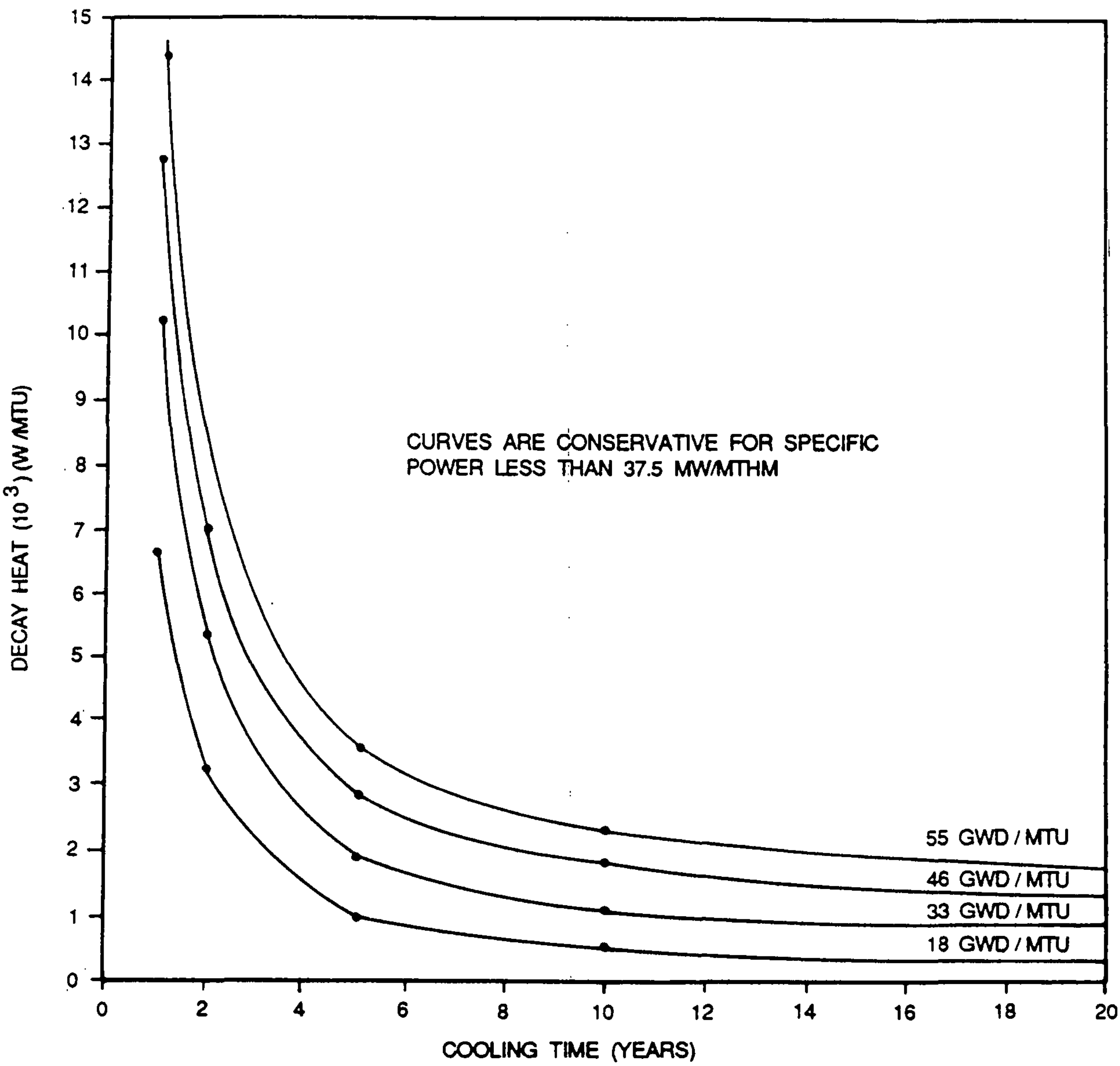


Figure 3.1-1

DECAY HEAT VS. COOLING TIME FOR VARIOUS BURNUPS

3.2 Structural and Mechanical Safety Criteria

The NUHOMS reinforced concrete HSM and its DSC support structure, the DSC and its internal basket assembly, and the transfer cask are the components that are important to safety. Consequently, they are designed and analyzed to perform their intended functions under the extreme environmental and natural phenomena specified in 10CFR72.72 (3.6) and ANSI-57.9 (3.36). Since NUHOMS is an independent, passive system, no other components or systems contribute to its safe operation.

Table 3.2-1 summarizes the design criteria for the equipment important to safety. This table also summarizes the applicable codes and standards utilized for design. The extreme environmental and natural phenomena design criteria discussed below comply with the requirements of 10CFR72.72 and ANSI-57.9. A description of the structural and mechanical safety criteria for the other design loadings listed in Table 3.2-1, such as thermal loads and cask drop loads, are provided in Section 8 of this report.

3.2.1 Tornado and Wind Loadings

A NUHOMS ISFSI is designed to be located anywhere within the contiguous United States. Consequently, the most severe tornado and wind loadings specified by NRC Regulatory Guide 1.76 (3.7) and NUREG-0800, Section 3.5.1.4 (3.8) were selected. Although 10CFR72.72 (b)(2) does not require an ISFSI to be protected against tornado missiles, the NUHOMS reinforced concrete HSMs are designed to safely withstand such missiles. Extreme wind effects are much less severe than tornado wind and missile loads or seismic effects and, therefore, are not evaluated in detail for the HSM.

Since the NUHOMS-24P transfer cask is used infrequently and for short durations, the possibility of a tornado occurring which would damage the cask/DSC in transit to the HSM is a low probability event. Nevertheless, the NUHOMS-24P transfer cask is designed for the effects of tornados, excluding design for tornado missiles in accordance with 10CFR72.72. This includes design for the effects of worst case tornado winds.

3.2.1.1 Applicable Design Parameters The design basis tornado (DBT) intensities used for the NUHOMS-24P transfer cask and HSM design were obtained from NRC Regulatory Guide 1.76. Region I intensities were utilized since they result in the most severe loading parameters. For this region, the maximum wind speed is 360 miles per hour, the rotational speed is 290 miles per hour, the maximum translational speed is 70 miles per hour, the radius of the maximum rotational speed is 150 feet, the pressure drop across the tornado is 3.0 psi, and the rate of pressure drop is 2.0 psi per second. The maximum transit time based on the five miles per hour minimum translational speed specified for Region I was not used since an infinite transit time is conservatively assumed.

<u>3.2.1.2 Determination of Forces on Structures</u> The effects of a DBT were evaluated for the NUHOMS-24P transfer cask and HSM. Tornado loads were generated for two separate loading phenomena: First, pressure forces created by drag as air impinges and flows past the transfer cask or HSM; and second, impact, penetration, and spalling forces created by tornado-generated missiles impinging on the HSM. The atmospheric pressure change induced forces are considered. In the following paragraphs, the determination of these forces is described.

The determination of the DBT velocity pressure was based on the following equation as specified in ANSI 58.1-1982 (3.9).

$$q = 0.00256\ K_z (IV)^2 \qquad (3.2\text{-}1)$$

Table 5 of ANSI A58.1 (3.9) defines the Importance Factor (I) to be 1.07 and the velocity pressure exposure coefficient (K_z) to be 0.8 applied to the full HSM height of 15 feet. Since the generic design basis HSM dimensions were relatively small compared to the 150 ft rotational radius of the DBT, the velocity value of combined rotational and translational wind velocity of 360 miles per hour was conservatively used in equation 3.2-1 as follows:

$$q = 0.00256 \times 0.8 \times [1.07 \times 360]^2 = 304 \text{ psf} \qquad (3.2\text{-}2)$$

The calculated DBT velocity pressure was converted to a design wind pressure by multiplying this value by the appropriate pressure coefficients specified in Figure 2 and gust response of ANSI A58.1-1982. The magnitude and direction of the design pressures for various HSM components and the corresponding pressure coefficients are tabulated in Table 3.2-2. The effects of overturning and sliding of the HSM under these design pressures were also evaluated and are reported with the stress analysis results in Section 8.2 of this report.

The transfer cask was also evaluated for a 397 psf DBT velocity pressure since this load magnitude envelops that for a closed cylindrical structure such as the cask. The transfer cask stress analysis for tornado wind loads is contained in Section 8.2.

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

For the overall effects of a DBT missile impact, overturning, and sliding on the HSM, the force due to the deformable massive missile impact was applied to the structure at the most adverse location. The force was evenly distributed over the impact area. The magnitude of the impact force was determined in accordance with the procedure established by Williamson and Alvy (3.10), as recommended by NUREG-0800, Section 3.5.3. (3.25).

For the local damage analysis of the HSM for DBT missiles, the rigid penetration-resistant missile was used for the evaluation of concrete penetration, spalling, scabbing and perforation thickness. The modified National Defense Research Committee (NDRC) empirical formula was used for this evaluation as recommended in NUREG-0800, Section 3.5.3. The results of these evaluations are reported in Section 8.2 of this report.

<u>3.2.1.3 Ability of Structures to Perform</u> The HSM protects the DSC from adverse environmental effects and is the principal NUHOMS 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 potential loss of the air outlet shielding blocks due to tornado missiles is also addressed. The transfer cask protects the DSC during transit to the HSM from adverse environmental effects such as tornado winds. The analyses of the HSM and transfer cask for tornado effects is contained in Sections 8.2.1 and 8.2.2

Since the HSMs are constructed outdoors in generally open areas, 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 Section 8.2.7.

3.2.2 Water Level (Flood) Design

The DSC and HSM components of the NUHOMS system were subjected to an enveloping design basis flood, postulated to result from natural phenomena such as a tsunami, and seiches, as specified by 10CFR72.72(b). For the purpose of this bounding generic evaluation, a 50 foot flood height and water velocity of 15 fps was used. The NUHOMS-24P HSM was evaluated for the effects of a water current of 15 fps impinging upon the side of a submerged HSM. The DSC was subjected to an external pressure equivalent to a fifty foot head of water. These evaluations are presented in Section 8.2.4. Due to its short term infrequent use, the NUHOMS-24P transfer cask is not explicitly evaluated for flood effects. Plant procedures will ensure that the transfer cask is not used for DSC transfer during flood conditions.

The calculated effects of the enveloping design basis flood have been included in the load combinations and reported stresses presented in Section 8.2.10. The plant specific design basis flood (if the possibility for flooding exists at a particular plant site) will be identified in site license applications and shown to be enveloped by the flooding conditions used for this generic evaluation of the NUHOMS-24P system DSC and HSM.

3.2.3 Seismic Design Criteria

The design basis response spectra of NRC Regulatory Guide (R.G.) 1.60 (3.11) was selected for the NUHOMS design earthquake as defined in 10CFR72.66(a)(6)(i). 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 NRC Regulatory Guide 1.61, Table 1 (3.12) was assumed to be applicable. Hence, a damping value of three percent of critical damping for the design bases safe shutdown earthquake was used. Similarly, from the same R.G. table, a damping value of seven percent of critical damping was 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 NUHOMS was 0.25g. The maximum vertical acceleration component selected was two-thirds of the horizontal component which is 0.17g. These ground acceleration values comply with the recommendations of 10CFR72.66 (a)(6) (ii) for sites underlaid by rock east of the Rocky Mountain front, except in the areas of known seismic activity. For ISFSI sites west of the Rocky Mountain front and areas of known potential seismic activity, seismicity should be evaluated using the techniques outlined in 10CFR100 Appendix A. These will be evaluated on a site specific basis and included in the site license application.

In order to establish the amplification factor associated with the generic design basis response spectra, various frequency analyses were performed for the different NUHOMS-24P components and structures. The results of these analyses indicated that the dominant lateral frequency for the reinforced concrete HSM was 25 Hertz. The dominant frequency of the DSC shell was calculated to be 13.3 Hertz. The corresponding horizontal seismic acceleration used for design of the HSM was 0.32g. Conservatively assuming that the dominant HSM vertical frequency is also 25 Hz. produces a vertical seismic design acceleration of 0.22g. The resulting seismic design accelerations used for the DSC are 1.0g horizontally and 0.68g vertically. The seismic analyses of the HSM and DSC are discussed further in Section 8.2.3.

3.2.4 Snow and Ice Loads

Snow and ice loads for the HSM were 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 110 psf is assumed. For the purpose of this conservative generic evaluation, a total live load of 200 pounds per square foot was used in the HSM analysis to envelope all postulated live loadings, including snow and ice. Snow and ice loads for the NUHOMS-24P transfer cask with a loaded DSC are negligible due to the 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 NUHOMS reinforced concrete HSM was designed to meet the requirements of ACI 349-85 (3.13). The ultimate strength method of analysis was utilized with the appropriate strength reduction factors as described in Table 3.2-4. The load combinations specified in Section 6.17.3.1 of ANSI 57.9-1984 were used for combining normal operating, off-normal, and accident loads for the HSM. All seven load combinations specified were considered and the governing combinations were selected for detailed design and analysis. The resulting HSM load combinations and the appropriate load factors are presented in Table 3.2-5. The effects of duty cycle on the HSM were considered and found to have negligible effect on the design. The HSM load combination results are presented in Section 8.2.10.

3.2.5.2 Dry Shielded Canister The DSC was designed by analysis to meet the stress intensity allowables of the ASME Boiler and Pressure Vessel Code (1983) (3.14) Section III, Division 1, Subsection NB for Class 1 components. The DSC was conservatively designed by utilizing linear elastic analysis methods. The load combinations considered for the DSC normal, off-normal and postulated accident loadings are shown in Table 3.2-5a. ASME Code Service Levels A and B allowables were conservatively used for normal and off-normal operating conditions. Service Levels C and D allowables were used for accident conditions such as a postulated drop accident. Using this acceptance criteria ensures that in the event of a design basis drop accident, the DSC containment pressure boundary will not be breached. As indicated by the results of the analysis of Section 8.2.5, the amount of deformation sustained by the spacer disks will not hinder the fuel retrieval operation. Maximum shear stress theory was used to calculate principal stresses. Normal operational stresses were combined with the appropriate off-normal and accident stresses. It was assumed that only one postulated accident condition occurs at any one time. The accident analyses are documented in Section 8.2 of this report. The structural design criteria for the DSC and DSC support assembly are summarized in Tables 3.2-6 and 3.2-7, respectively. The effects of fatigue on the DSC due to thermal and pressure cycling are addressed in Section 8.2.10.

3.2.5.3 NUHOMS-24P Transfer Cask The NUHOMS-24P transfer cask was designed by analysis to meet the stress allowables of the ASME Code (3.14) Subsection NC for Class 2 components. The cask was conservatively designed by utilizing linear elastic analysis methods. The load combinations considered for the transfer cask normal, off-normal, and postulated accident loadings are shown in Table 3.2-5b. Service Levels A and B allowables were 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 were conservatively developed to meet the requirements of ANSI N14.6-1986 (3.37) for critical loads. Maximum shear stress theory was used to calculate principal stresses in the cask structural shell. The appropriate dead load and thermal stresses were combined with the calculated drop accident scenario stresses to determine the worst case design stresses. The transfer cask structural design criteria are summarized in Tables 3.2-8 and 3.2-9. The transfer cask accident analyses are presented in Section 8.2. The effects of fatigue on the transfer cask due to thermal cycling are addressed in Section 8.2.10.

Table 3.2-1

SUMMARY OF NUHOMS-24P SYSTEM DESIGN LOADINGS

Component	Design Load Type	Report Section Reference	Design Parameters	Applicable Codes
Horizontal Storage Module:	-	-	-	ACI 349-85 and ACI 349R-85
	Design Basis Tornado	3.2.1	Max. wind pressure: 397 psf Max. speed: 360	NRC Reg. Guide 1.76 and ANSI A58.1 1982
	DBT Missile	3.2.1	Max. speed: 126 Types: Automobile 3967 lb., 8 in. diam. shell 276 lb., 1 in. solid sphere	NUREG-0800, Section 3.5.1.4
	Flood	3.2.2	Maximum water height: 50 feet Maximum velocity: 15 ft./sec.	10CFR72.72
	Seismic	3.2.3	Hor. ground acceleration: 0.25g (both directions) Vert. ground acceleration:0.17g	NRC Reg. Guides 1.60 & 1.61
	Snow and Ice	3.2.4	Maximum load: 110 psf (included in live loads)	ANSI A58.1-1982
	Dead Loads	8.1.1.5	Dead weight including loaded DSC (concrete density of 150 pcf)	ANSI 57.9-1984

Table 3.2-1

SUMMARY OF NUHOMS-24P SYSTEM DESIGN LOADINGS
(Continued)

Component	Design Load Type	Report Section Reference	Design Parameters	Applicable Codes
	Normal and Off-normal Operating Temperatures	8.1.1.5	DSC with spent fuel rejecting 15.8 kw of decay heat. Ambient air temperature range of $-40^{o}F$ to $125^{o}F$	ANSI 57.9-1984
	Accident Condition Temperatures	8.2.7.2	Same as off-normal conditions with HSM vents blocked for 48 hours or less	ANSI 57.9-1984
	Normal Handling Loads	8.1.1.1	Hydraulic ram load equal to 25% of loaded DSC weight: 20,000 lb. enveloping	ANSI 57.9-1984
	Off-normal Handling Loads	8.1.1.4	Hydraulic ram Load equal to 100% of loaded DSC weight: 80,000 lb. enveloping	ANSI 57.9-1984
	Live Loads	8.1.1.5	Design load: 200 psf (includes snow and ice loads)	ANSI 57.9-1984
	Fire and Explosions	3.3.6	Considered on a site specific basis	10CFR72.72(c)

Table 3.2-1

SUMMARY OF NUHOMS-24P SYSTEM DESIGN LOADINGS
(Continued)

Component	Design Load Type	Report Section Reference	Design Parameters	Applicable Codes
Dry Shielded Canister	-	-	-	ASME Code, Section III, Subsection NB, Class 1 Component
	Flood	3.2.2	Maximum water height: 50 ft.	10CFR72.72
	Seismic	3.2.2	Horizontal acceleration: 1.0g Vertical acceleration: 0.68g	NRC Reg. Guides 1.60 & 1.61
	Dead Loads	8.1.1.2	Weight of loaded DSC: 72,000 lb. nominal, 80,000 lb. enveloping	ANSI 57.9-1984
	Normal and Off-Normal Pressure	8.1.1.2	DSC internal pressure of $\leq$ 9.7 psig	ANSI 57.9-1984
	Normal and Off-normal Operating Temp.	8.1.1.2, 8.1.2.2	DSC with spent fuel rejecting 15.8 kw decay heat. Ambient air temperature -40^{o}F to 125^{o}F	ANSI 57.9-1984
	Normal Handling Loads	8.1.1.2	Hydraulic ram load equal to 25% of loaded DSC weight: 20,000 lb. enveloping	ANSI 57.9-1984

Table 3.2-1

SUMMARY OF NUHOMS-24P SYSTEM DESIGN LOADINGS
(Continued)

Component	Design Load Type	Report Section Reference	Design Parameters	Applicable Codes
	Off-normal Handling Loads	8.1.2.1	Hydraulic Ram load equal to 100% of loaded DSC weight: 80,000 lb. enveloping	ANSI-57.9-1984
	Accident Drop	8.2.5	Equivalent static deceleration of 75g for vertical end drop and horizontal side drops, and 25g corner drop with slapdown	10CFR72.72(b)
	Accident Internal Pressure	8.2.7 8.2.9	DSC internal pressure of 49.1 psig based on 100% fuel cladding rupture and fill gas release, 30% fission gas release, and ambient air temperature of 125^{o}F	10CFR72.72(b)
Dry Shielded Canister Support Assembly:	-	-	-	AISC Code for Structural Steel
	Dead Weight	8.1.1.4	Loaded DSC plus self weight	ANSI-57.9-1984
	Seismic	3.2.3	DSC reaction loads with horizontal acceleration of 0.40g and vertical acceleration of 0.27g	NRC Reg. Guides 1.60 & 1.61

Table 3.2-1

SUMMARY OF NUHOMS-24P SYSTEM DESIGN LOADINGS
(Continued)

Component	Design Load Type	Report Section Reference	Design Parameters	Applicable Codes
	Normal Handling Loads	8.1.1.4	DSC reaction loads with hydraulic ram load equal to 25% of loaded DSC weight: 20,000 lb. enveloping	ANSI-57.9-1984
	Off-normal Handling Loads	8.1.1.4	DSC reaction loads with hydraulic ram load equal to 100% of loaded weight: 80,000 lb. enveloping	ANSI-57.9-1984
On-site Transfer Cask:	-	-	-	ASME Code Section III, Subsection NC, Class 2 Component
	Design Basis Tornado Wind	3.2.1	Max. wind pressure: 397 psf Max. wind speed: 360 mph	NRC Reg. Guide 1.76 and ANSI 58.1-1982
	Flood	3.2.2	Not included in design due to infrequent short duration; use of cask restricted by administrative controls	10CFR72.72(b)

Table 3.2-1

SUMMARY OF NUHOMS-24P SYSTEM DESIGN LOADINGS
(Continued)

Component	Design Load Type	Report Section Reference	Design Parameters	Applicable Codes
	Seismic	3.2.3	Hor. ground accel.: 0.25g Vert. ground accel.: 0.17g	NRC Reg. Guides 1.60 & 1.61
	Snow and Ice	3.2.4	External surface temp. and circular section will preclude build-up of snow and ice loads when cask is in use	10CFR72.72(b)
	Dead Weight	8.1.1.8	a. Vertical orientation, self weight with loaded DSC and water in cavity: 200,000 lbs. enveloping	ANSI 57.9-1984
			b. Horizontal orientation self weight with loaded DSC on transfer skid: 193,000 lbs. nominal, 200,000 lbs. enveloping	ANSI 57.9-1984
	Normal and Off-normal Operating Temperatures	8.1.1.8, 8.1.2.2	Loaded DSC rejecting 15.8 kw decay heat. Ambient air temperature range: $-40^{\circ}F$ to $125^{\circ}F$	ANSI 57.9-1984

Table 3.2-1

SUMMARY OF NUHOMS-24P SYSTEM DESIGN LOADINGS
(Continued)

Component	Design Load Type	Report Section Reference	Design Parameters	Applicable Codes
	Normal Handling Loads	8.1.1.8	a. Upper lifting trunnions - in fuel building: $\leq$Stresses yield with 6 x load $\leq$and ultimate with 10 x load	ANSI N14.6-1978
			b. Upper lifting trunnions - on-site transfer	ASME Section III
			c. Lower support trunnions: Weight of loaded cask during down loading and proportional weight of loaded cask during transit to HSM	ASME Section III
			d. Hydraulic ram load/friction of moving DSC equal to 25% of DSC loaded weight: 20,000 lb. enveloping	ANSI 57.9-1984
	Off-normal Handling Loads	8.1.2.1	Hydraulic ram load/jammed DSC equal to 100% of DSC loaded weight: 80,000 lb. enveloping	ANSI 57.9-1984

Table 3.2-1

SUMMARY OF NUHOMS-24P SYSTEM DESIGN LOADINGS
(Concluded)

Component	Design Load Type	Report Section Reference	Design Parameters	Applicable Codes
	Accident Drop Loads	8.2.5	Equivalent static deceleration of 75g for vertical end drops and horizontal side drops, and 25g for corner drop and slapdown	10CFR72.72(b)
	Fire and Explosions	3.3.6	Considered on a site specific basis	10CFR72.72(c)
	Internal Pressure	--	N/A - DSC provides pressure boundary	10CFR72.72(h)

Table 3.2-2

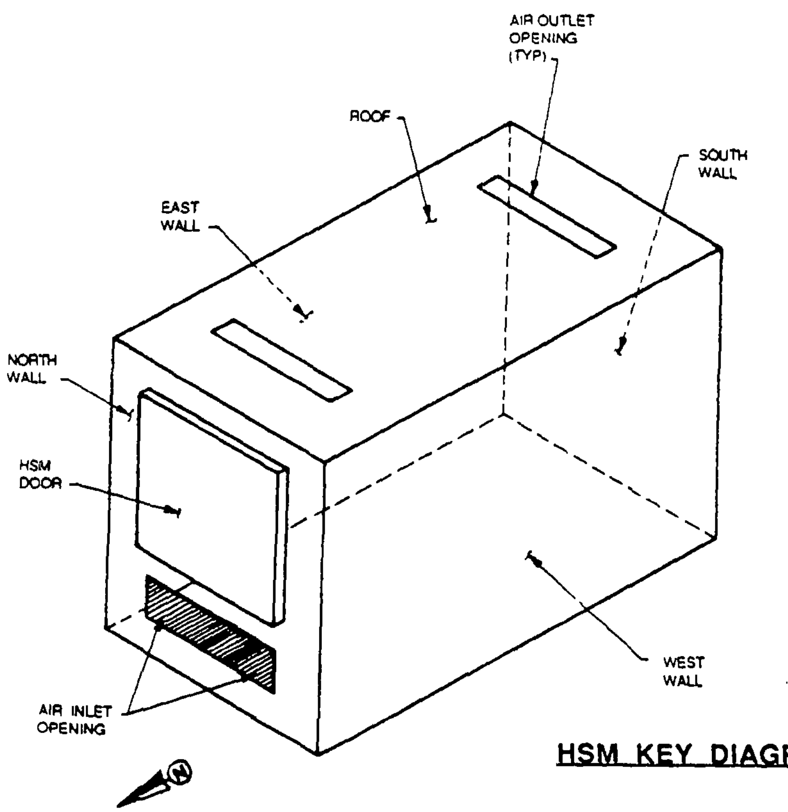
DESIGN PRESSURES FOR TORNADO WIND LOADING

Wall Orientation	Velocity Pressure (psf)	Max/Min Pressure Coefficient	Max/Min Design Pressure (psf)
North	304	+1.31	+397
East	304	-1.17	-357
South	304	-0.64	-196
West	304	-1.17	-357
Roof	304	-1.17	-357

Notes:

1. Wind direction assumed to be from North. Wind loads for other directions may be found by rotating table values to desired wind direction.

2. Gust factors G_h and G_z are conservatively assumed to be 1.32.



HSM KEY DIAGRAM

TABLE 3.2-3 HAS BEEN DELETED

Table 3.2-4

HSM ULTIMATE STRENGTH REDUCTION FACTORS

Type of Stress	Reduction Factor
Flexure	0.9
Axial Tension	0.9
Axial Compression	0.7
Shear	0.85
Torsion	0.85
Bearing	0.7

Table 3.2-5

HSM LOAD COMBINATION METHODOLOGY

Case No.	Load Combination	Loading Notation
1	1.4D + 1.7L	D = Dead Weight
2	1.4D + 1.7L + 1.7H	E = Earthquake Load
3	0.75(1.4D + 1.7L + 1.7H + 1.7T + 1.7W)	F = Flood Induced Loads
4	0.75(1.4D + 1.7L + 1.7H + 1.7T)	H = Lateral Soil Pressure Load
		L = Live Load
5	D + L + H + T + E	T = Normal Condition Thermal Load
6	D + L + H + T + F	
7	D + L + H + T_a	T_a = Off-normal or Accident Condition Thermal Load
		W = Wind Load

Notes:

1. The HSM load combinations are in accordance with ANSI-57.9. In Case 6 flood loads (F) are substituted for drop loads (A) which are not applicable to the HSM.

2. The effects of creep and shrinkage are included in the dead weight load for Cases 3 through 8.

3. Wind loads are conservatively taken as Design Basis Tornado (DBT) loads. These include wind pressure, differential pressure, and missile loads. Case 3 was first satisfied without the tornado missile load. Missile loads were analyzed for local damage, over all damage, overturning and sliding effects.

Table 3.2-5a

DSC DESIGN LOAD COMBINATIONS

Load Case		Normal Operating Conditions				Off-Normal Conditions				Accident Conditions								
		1	2	3	4	1	2	3	4	1	2	3	4	5	6	7	8	9
Dead Weight	Vertical, DSC Empty	X																
	Vertical, DSC w/Water		X															
	Horizontal, DSC w/Fuel			X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Thermal	Inside HSM: 0° to 100°F				X							X	X	X			X	
	Inside Cask: 0° to 100°F		X	X		X					X							X
	Inside HSM: -40° to 125°F							X							X			
	Inside Cask: -40° to 125°F						X		X									
	Inside Cask: Accident									X								
	Inside HSM: Accident															X		
Internal Pressure	Normal Pressure		X	X		X							X					
	Off-Normal Pressure				X		X	X	X									
	Accident Pressure									X	X	X		X	X	X	X	X
Handling Loads	Normal DSC Transfer			X	X									X				
	Jammed DSC Loads					X	X	X									X	X
Cask Drop (End, Side or Corner Drop)											X							
Seismic												X						
Flooding													X					
ASME Code Service Level		A	A	A	A	B	B	B	B	D	D	C	C	C	C	C	C	C
Load Combination No.		A1	A2	A3	A4	B1	B2	B3	B4	D1	D2	C1	C2	C3	C4	C5	C6	C7

Table 3.2-5b

TRANSFER CASK LOAD COMBINATIONS

Load Case		Normal Operating Conditions					Off-Normal Conditions		Accident Conditions				
		1	2	3	4	5	1	2	1	2	3	4	5
Dead Load/Live Load		X	X	X	X	X	X	X	X	X	X	X	X
Thermal w/DSC	0° to 100°F Ambient	X	X	X	X	X			X	X	X	X	X
	-40° to 125°F Ambient						X	X					
Handling Loads (Critical Lifts)	Vertical	X											
	Tilted		X										
	Horizontal			X									
Handling Loads (Non-Critical)	Transport				X		X		X				
	DSC Transfer					X		X		X			
Seismic									X	X			
Drop	Vertical										X		
	Corner											X	
	Horizontal												X
ASME Code Service Level		A	A	A	A	A	B	B	C	C	D	D	D
Load Combination No.		A1	A2	A3	A4	A5	B1	B2	C1	C2	D1	D2	D3

Table 3.2-6

STRUCTURAL DESIGN CRITERIA FOR DSC

Item	Stress Type	Stress Values (1)		
		Service Levels A & B	Service Level C	Service Level D
DSC (2) and Internals	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	Smaller 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 Fillet Welds (3)	Primary	0.50 S_m	Greater of 0.65S_m or 0.50 S_y	Smaller of 1.2 S_m or 0.35 S_u
	Primary + Secondary	0.75 S_m	Smaller of 0.9 S_m or 0.75 S_y	N/A

Notes

1. Values of S_y, S_m, and S_u versus temperature are given in Table 8.1-2.
2. Includes full penetration welds.
3. An efficiency factor of 0.5 has been applied for nonvolumetric inspected welds.

Table 3.2-7

STRUCTURAL DESIGN CRITERIA FOR DSC SUPPORT ASSEMBLY

Stress Type	Stress Value [1]
Tensile	0.60 Sy
Compressive	(See Note 2)
Bending	0.60 Sy (4)
Shear	0.40 Sy
Interaction	(See Note 3)

Notes:

1. Values of Sy versus temperature are given in Table 8.1-2.
2. Equations 1.5-1, 1.5-2 or 1.5-3 of the AISC Code (3.45) are used as appropriate.
3. Interaction equations per the AISC Code are used as appropriate.
4. If the requirements of Paragraph 1.5.1.4.1 are met, an allowable bending stress of 0.66 Sy is used.

Table 3.2-8

STRUCTURAL DESIGN CRITERIA FOR ON-SITE TRANSFER CASK

Item	Stress Type	Stress Values(1)		
		Service Levels A & B	Service Level C	Service Level D
Transfer Cask Structural Shell	Primary Membrane	S_m	1.2 S_m	Smaller of 2.4 S_m or 0.7 S_u
	Primary Membrane + Bending	1.5 S_m	1.8 S_m	Smaller of 3.6 S_m or S_u
	Primary + Secondary	3.0 S_m	N/A	N/A
Trunnions(2)	Membrane and Membrane + Bending	Smaller of $S_y/6$ or $S_u/10$	N/A	N/A
	Shear	Smaller of 0.6 $S_y/6$ or 0.6 $S_u/10$	N/A	N/A
Fillet Welds(3)	Primary	0.5 S_m		Smaller of 1.2 S_m or 0.35 S_u
	Secondary	0.75 S_m		N/A

Notes:

1. Values of S_y, S_m, and S_u versus temperature are given in Table 8.1-2.

2. These allowables apply to the upper lifting trunnions for critical lifts governed by ANSI N14.6. 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.

3. An efficiency factor of 0.5 has been applied for nonvolumetric inspected welds.

Table 3.2-9

STRUCTURAL DESIGN CRITERIA FOR BOLTS

Service Levels A, B, and C	
Average Service Stress	$< 2\ S_m$
Maximum Service Stress	$< 3\ S_m$
Service Level D	
Average Tension	Smaller of S_y or $0.7\ S_u$
Tension + Bending	S_u
Shear	Smaller of $0.6\ S_y$ or $0.42\ S_u$
Interaction	Interaction equation of Appendix F (F-1335.3) of ASME Code (3.14)

FIGURE 3.2-1 HAS BEEN DELETED

3.3 Safety Protection System

3.3.1 General

The NUHOMS system is designed for safe and secure, long-term containment and storage of SFAs. The components, structures, and equipment which are designed to assure that this safety objective is met are shown in Table 3.3-1. The key elements of the NUHOMS system 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 to form a pressure retaining containment boundary and to maintain a helium atmosphere.

3. Minimizing personnel radiation exposure during DSC loading and retrieval operations.

4. Design of the transfer 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.

These items are addressed in the following subsections.

3.3.2 Protection by Multiple Confinement Barriers and Systems

3.3.2.1 Confinement Barriers and Systems The radioactive material which the NUHOMS system confines is the SFAs themselves and the associated contaminants. This radioactivity is confined by the multiple barriers listed in Table 3.3-2.

During fuel loading operations,the radioactive material in the plant's fuel pool is prevented from contacting the DSC exterior by filling the cask/DSC annulus and DSC with uncontaminated, demineralized water prior to placing the cask and DSC in the fuel pool. This places uncontaminated water in the annulus between the DSC and cask interior. In addition, the cask/DSC annulus opening at the top of the cask is sealed using tape, or other mechanical means, to prevent pool water from displacing the uncontaminated water. This procedure minimizes the likelihood of contaminating the DSC exterior surface. The combination of the above operations assures that the DSC surface has less residual contamination than required for shipping cask externals (see Section 3.3.7.1). The attainment of this level of contamination is assured by taking a surface swipe of the upper one foot of the DSC while resting in the cask on the fuel building drying pad.

Once inside the DSC, the SFAs are confined by the DSC shell and by multiple barriers at each end of the DSC. As listed in Table 3.3-2, the fuel cladding is the first barrier for confinement of radioactive materials. The fuel cladding is protected by maintaining the cladding temperatures during storage below those which may cause degradation of the cladding. In addition, the SFAs are stored in a helium atmosphere to prevent degradation of the fuel, specifically cladding rupture due to oxidation and its resulting volumetric expansion of the fuel. Thus, a helium atmosphere for the DSC is incorporated in the design to protect the fuel cladding integrity by inhibiting the ingress of oxygen inside the DSC.

Helium will leak through valves, mechanical seals, and escape through very small passages because of its small atomic diameter and because it is an inert element and exists in a monatomic species. Negligible leak rates can be achieved with careful design of vessel closures. Helium will not, to any practical extent, diffuse through stainless steel (3.33). For this reason the DSC has been designed as a weld-sealed containment pressure vessel with no mechanical or electrical penetrations.

The DSC itself has a series of barriers to ensure the confinement of radioactive materials. All closure welds are multiple-pass welds. This effectively eliminates a pinhole leak which might occur in a single pass weld, since the chance of pinholes being in alignment on successive weld passes is negligible. Furthermore, the DSC cover plates are sealed by separate, redundant closure welds. All the DSC pressure boundary welds are inspected according to the appropriate articles of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB. This criteria insures that the weld metal is as sound as the parent metal of the pressure vessel.

Pressure monitoring instrumentation is not used since penetration of the pressure boundary would be required. The penetration itself would then become a potential leakage path and by its presence compromise the integrity of the DSC design. The DSC shell and welded cover plates provide total confinement of the radioactive materials. Once the DSC is sealed, there are no credible events which would fail the DSC cylindrical shell or the double closure plates which form the DSC containment pressure boundary. This is further discussed in Section 8 of this report.

3.3.2.2 Ventilation - Offgas The NUHOMS 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 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 Sections 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 the DSC. Therefore, there are no off-gas or monitoring system requirements for the HSM. The only time an off-gas system is required is during the cask-DSC drying phase. During this operation, the plant's radwaste system will be used for the air and helium which is evacuated from the DSC.

3.3.3 Protection by Equipment and Instrumentation Selection

3.3.3.1 Equipment The HSM, DSC, and on-site transfer cask are the equipment important to safety. Other equipment important to safety associated with the NUHOMS system is the equipment required for handling operations within the plant's fuel building.

3.3.3.2 Instrumentation The NUHOMS system is a totally passive system. No safety-related instrumentation is necessary. The maximum temperatures and pressures were conservatively bounded by analyses (see Section 8.1.3). Therefore, there is no need for monitoring the internal cavity of the DSC for pressure or temperature during normal operations. The DSC was conservatively designed to perform its containment function during all worst case normal, off-normal, and postulated accident conditions.

3.3.4 Nuclear Criticality Safety for NUHOMS-24P System

3.3.4.1 Control Methods for Prevention of Criticality The NUHOMS DSC internals were designed to provide nuclear criticality safety during wet loading operations. Control methods for the prevention of criticality consist of the material properties of the fuel, administrative procedures (i.e., a site-specific system using records or tests to document initial enrichment and burn-up), geometrical arrangement and the inherent neutron absorption in the stainless steel guide sleeve assemblies.

Fuel loading and handling in an unborated pool is the worst case for nuclear criticality (i.e. the most reactive state). Analyses with the CSAS2 criticality analysis sequence included in the SCALE-3 package of computer codes (3.44) performed by Duke Power Company Design Engineering were used to show that subcritical conditions are maintained during these handling operations. After fuel loading and DSC drying, there is no moderator present and, hence, subcriticality is assured. Furthermore, the seal welded DSC body assures that no water can leak into the DSC during storage or transfer under any accident conditions.

Design Parameters for Criticality Model The geometry and fuel characteristics of the criticality model are shown in Table 3.3-3. Figure 3.3-1 shows the actual geometry of the DSC and the geometry of the CSAS2 model. Figure 3.3-2 describes the modeling of the fuel assembly guide sleeves with the heterogeneous fuel assembly region inside.

The NUHOMS DSC internals design relies on administrative procedures to allow only fuel assemblies of less than a predetermined residual reactivity to be placed in the DSC for storage. The predetermined residual reactivity limit was selected to correspond roughly to a fuel assembly at 80 percent of what is typically considered full burnup. The concept of reactivity equivalency was used to develop a curve of constant reactivity through the enrichment/burnup space assuming the DSC was fully loaded with spent B&W 15x15 fuel assemblies. The resulting curve of reactivity equivalence for the DSC is presented in Figure 3.3-3. The reactivity equivalence curve extends from a zero burnup, initial enrichment equivalent point of 1.45 w/o ^{235}U to a high enrichment endpoint corresponding to 4.00 w/o ^{235}U initial enrichment irradiated for approximately 37,000 MWD/MTHM. The reactivity equivalence curve presented in Figure 3.3-3 will be used to determine the acceptability of storing specific fuel assemblies in the NUHOMS DSC.

The selection of a 15x15 fuel assembly for criticality calculation has been shown by many analyses to be conservative under a variety of conditions when compared to other fuel assemblies (i.e., 14x14, 16x16, and 17x17) (3.28, 3.29, 3.30, 3.31).

3.3.4.2 Reactivity Equivalence and Criticality Analysis Methods

The analysis methods which ensure criticality safety under all DSC fuel transfer, handling or storage conditions uses:

1. The CASMO-2 computer code developed by Studsvik Energiteknik AB and supported by the Electric Power Research Institute (3.38),

2. The Shielding Analysis Sequence No. 2 (SAS2) included in the SCALE-3 package of codes developed by Oak Ridge National Laboratories (3.44), and

3. The Criticality Safety Analysis Sequence No. 2 (CSAS2) included in the SCALE-3 package of codes.

The CASMO-2 computer code is a multigroup two-dimensional transport theory code for burnup calculations on PWR or BWR assemblies. CASMO-2 is an industry recognized code which has been accepted by the NRC (3.39). Its ability to predict isotopic generation and depletion as well as neutron multiplication is well established in benchmark calculations (References 3.48, 3.49) and through its successful application in numerous reactor physics and core reload design calculations.

The SAS2 sequence in SCALE-3 was used to generate sets of spent fuel fission product inventory data at various burnup levels for the initial enrichment points considered. Only nonvolatile fission products identified as major neutron absorbers in Reference 3.40 were included in spent fuel criticality/ equivalencing calculations.

SAS2 is an industry recognized code which employs ORIGEN-S to perform fuel burnup, depletion and decay calculations. SAS2 has been extensively tested for use in spent fuel isotopic inventory and decay heat source term development work (3.2). SAS2 was used with the 27GROUPNDF4 cross-section data library included in SCALE-3 to generate the fission product data used in the criticality analyses.

CSAS2 and the 123GROUPGMTH master cross-section library included in SCALE-3 were used in calculating an effective neutron multiplication factor, K_{eff}, for each initial enrichment/burnup combination considered. The CSAS2 analysis sequence used two cross-section processing codes (NITAWL and BONAMI), and a three-dimensional Monte-Carlo code (KENO-IV) for calculating the K_{eff} values for fully loaded DSC fuel arrays. The actinide data generated by CASMO-2 and the fission product data generated by SAS2 were used as input to CSAS2 for specifying spent fuel compositions.

Several sets of K_{eff} values were calculated corresponding to a nominal case DSC model fully loaded with spent fuel at a number of different burnup levels for several initial fuel enrichments over the 1.45 to 4.0 w/o ^{235}U range. Appropriate burnup related uncertainties were added and the resulting K_{eff} data analyzed using least-squares methods to determine the burnup level necessary in each initial enrichment case to obtain an acceptable K_{eff} value, allowing for additional uncertainties. A curve of reactivity equivalence was then constructed through these initial enrichment/burnup points and extended to a zero burnup intercept point. Additional uncertainties related to mechanical tolerances, fuel assembly positioning, moderator density, and reflector effects were analyzed at the zero burnup intercept point of 1.45 w/o ^{235}U. The final K_{eff} value assigned to the DSC represents a maximum at a 95/95 tolerance level assuming a full loading of design basis fuel with an initial enrichment and burnup combination found anywhere along the curve of equal reactivity presented in Figure 3.3-3.

3.3.4.3 Criticality Evaluation This section presents the analyses which demonstrate the acceptability of storing qualified fuel in the DSC under normal fuel loading, handling, and storage conditions. A nominal case model is described and a neutron multiplication factor, K_{eff}, presented. Uncertainties were addressed and applied to the nominal calculated K_{eff} value. The final K_{eff} value produced represents a maximum with a 95 percent probability at a 95 percent confidence level as required by ANSI/ANS-57.2-1983 (3.46) to demonstrate criticality safety.

The following assumptions were used in the criticality evaluation:

1. Credit was taken for fuel burnup as allowed by ANSI/ANS 8.17-1984 (3.47).

2. Credit was taken for the inherent neutron absorption capability of the stainless steel guide sleeve assemblies of the DSC basket as shown on the design drawings in Appendix E of this report.

3. No burnable poisons, control rods, or supplemental neutron poisons were assumed to be present.

4. All assemblies were assumed to be nonirradiated 1.45 w/o ^{235}U enriched Babcock and Wilcox 15x15 type or irradiated Babcock and Wilcox 15x15 assemblies of equal or less reactivity when a fully loaded DSC was considered.

5. The DSC spent fuel storage array was modeled as finite in lateral extent and infinite in axial extent except in axial burnup variation sensitivity calculations which assumed water reflection at both axial ends of the fuel array.

6. Geometrical and material uncertainties due to mechanical tolerances were treated by either using worst-case configurations or by performing sensitivity calculations and obtaining appropriate uncertainty values. The uncertainties considered included:

   a. Stainless steel guide sleeve wall thickness
   b. Center-to-center spacing
   c. Cell ID
   d. Cell bowing
   e. Assembly positioning
   f. Metal reflector positioning

7. Each fuel assembly was treated as a heterogeneous system with the fuel pins, control rod guide tubes, and instrument guide tube modeled explicitly as illustrated by Figure 3.3-2.

8. The moderator was pure unborated water at a uniform density of 0.9982 gms/cc. Uncertainties resulting from moderator density variation effects were considered by performing sensitivity calculations. A conservatively calculated bias was applied to assure that the final K_{eff} value assigned to the array is a maximum at any water density between 1.0E-4 and 1.0 gms/cc.

9. A uniform axial burnup profile was assumed in CSAS2 irradiated fuel equivalence calculation cases. Consideration was given to variations in axial burnup as required by ANSI/ANS-8.17-1984; sensitivity calculations were performed and an appropriate bias was applied.

10. Irradiated fuel was assumed to be cooled 7.5 years following discharge from the reactor.

11. For irradiated fuel equivalence calculations, credit was taken for 34 major fission product absorbers identified as stable sources of negative reactivity in Reference 3.41. Quantitative estimates of negative reactivity credit taken for the major fission product absorbers at several representative initial enrichment points along Figure 3.3-3 reactivity equivalence curve are provided in Table 3.3-5.

Table 3.3-3 provides the nominal dimensions of the DSC and transfer cask geometry illustrated in Figure 3.3-1. Figure 3.3-1 also provides an illustration of the fuel array and reflectors modeled in CSAS2 for the nominal case.

A. Nominal Case

Referring to Figures 3.3-1, 3.3-2, and Table 3.3-3, the following zones were explicitly modeled in the nominal case CSAS2 analysis:

1. Nominal DSC basket geometry parameters
2. 0.369 inch OD fuel pellets (208 rods)
3. 0.430 inch OD Zircaloy fuel clad (208 rods)
4. Void gap between fuel and clad
5. 0.530 inch OD control rod guide tubes (16 tubes)
6. 0.493 inch OD instrument tube

The nominal case CSAS2 calculation in which 50,100 neutron histories were followed, resulted in a K_{eff} of 0.87170 with a 95 percent probability at a 95 percent confidence level uncertainty of +/- 0.00488.

B. Criticality Analysis Method Variability

In addition to UO_2 experiments, the CSAS2/123GROUPGMTH method has been validated against PuO_2 - UO_2 mixed oxide experiments (3.42) and K_{inf} results generated by CASMO-2E for irradiated B&W 15x15 fuel assembly arrays (3.43). The results of these additional validation runs indicate that the CSAS2/123GROUPGMTH method conservatively overpredicts K_{eff} for systems containing Pu and/or fission products.

C. Additional Biases and Uncertainties

The 95/95 uncertainty in the nominal case analysis is 0.00488 ΔK. A statistical bias of +0.00488 and a 95/95 uncertainty of 0.01161 ΔK is associated with the CSAS2 method used. In addition to these uncertainties, there are other considerations which may effect the final K_{eff} value assigned to the array. These considerations were treated as either worst-case in the nominal run or sensitivity runs were performed to determine the ΔK associated with a variable parameter (e.g., guide sleeve thickness).

[ ]

D. Worst-Case Maximum DSC K_{eff}

The worst-case maximum DSC array K_{eff} was determined by combining the nominal case results with the uncertainties and biases developed from the method benchmark calculations and the sensitivity studies performed for the DSC fuel storage array. [

The main conclusion of the criticality analysis is that the calculated worst-case K_{eff} value for a fully loaded DSC flooded with pure unborated water of a uniform optimum density, including uncertainties, is 0.94782. Conclusions regarding specific aspects of the methods used or the analyses presented can be drawn from the quantitative results presented in the Tables.

The resulting final K_{eff} value represents a maxiumum with a 95 percent probability at a 95 percent confidence level.

The following equation was used to develop the final K_{eff} result for the DSC fuel storage array:

$$K_{eff} = \text{K-nominal} + \text{B-method} + \text{B-axial} + \text{B-mod} + \text{B-ref} + [(\text{ks-nominal})^2 + (\text{ks-method})^2 + (\text{ks-axial})^2 + (\text{ks-mechanical})^2 + (\text{ks-reflector})^2 + (\text{ks-burnup})^2 + (\text{ks-mod})^2]^{1/2}$$

Where:

K-nominal	=	Nominal case K_{eff}
B-method	=	Method bias
B-axial	=	Bias accounting for non-uniform axial burnup in PWR fuel assemblies (irradiated fuel cases only)
B-mod	=	Bias accounting for worst-case moderator density conditions
B-ref	=	Bias accounting for worst-case metal reflector positioning
ks-nominal	=	95/95 uncertainty in the nominal case K_{eff} value
ks-method	=	95/95 uncertainty in the method bias
ks-axial	=	95/95 uncertainty in the non-uniform axial burnup bias
ks-mechanical	=	95/95 uncertainty resulting from material and construction tolerances and positioning uncertainties
ks-reflector	=	95/95 uncertainty in the bias accounting for worst case metal reflector model assumptions

ks-burnup = Uncertainty in the equivalencing method results

ks-mod = 95/95 uncertainty in the moderator density effects bias

Substituting the appropriate values for the nominal, zero-burnup case:

$$K_{eff} = 0.94782$$

All values of burnup used to develop the Figure 3.3-4 equivalence curve were selected to maintain K_{eff} equal to or below 0.94782. Components of the final K_{eff} result for all cases analyzed are provided in Table 3.3-11.

3.3.4.4 Off-Normal Conditions

Postulated off-normal conditions will not result in a DSC storage array reactivity which exceeds the K_{eff} value calculated and presented in Section 3.3.4.3.

Off-normal conditions considered include:

1. The misloading of one or more high enrichment nonirradiated fuel assemblies into the DSC, and

2. Optimum moderation.

Misloading a High Enrichment Assembly

Misloading one or more fuel assemblies which do not qualify as acceptable for storage in the DSC according to the burnup equivalence curve shown in Figure 3.3-3 will not result in a K_{eff} value greater than the 0.95 criterion. The double contingency principle of ANSI/ANS 8.17-1984 can be applied to take credit for dissolved boron in this case. The approximate 0.34 ΔK negative reactivity provided by 2,000 ppm boron would more than compensate for the additional reactivity added by the misloading of one or more unqualified, high enrichment fuel assemblies.

Optimum Moderation

Optimum moderation conditions were considered and a conservative bias was applied in the normal case analysis presented in Section 3.3.4.3. Therefore, the presence of a pure water moderator of optimum density will not result in a DSC storage array reactivity which exceeds the K_{eff} value calculated and presented in Section 3.3.4.3.

3.3.4.5 Safety Criteria Compliance

The calculated worst-case K_{eff} value for a fully loaded DSC flooded with pure unborated water of a uniform optimum density is 0.94782. This calculated maximum K_{eff} value includes consideration of geometrical, material and burnup uncertainties and biases at a 95/95 tolerance level as required by ANSI/ANS 57.2-1983 to demonstrate criticality safety.

Additionally, off-normal conditions potentially resulting in reactivity increases over the normal conditions considered have been addressed.

The analyses presented in this report demonstrate that the ANSI/ANS 57.2-1983 criteria limiting K_{eff} to 0.95 is satisfied under all postulated conditions.

3.3.5 Radiological Protection

3.3.5.1 Access Control Access controls for a NUHOMS HSM installation is site specific. The details of access control and the division of the ISFSI site into radiologically protected areas will be contained in the site license applications.

3.3.5.2 Shielding For the NUHOMS system, shielding is provided by the HSM, transfer cask, and shielded end plugs of the DSC. The NUHOMS HSM is designed to limit the maximum surface dose to 100 mrem/hour with an average external surface dose (gamma and neutron) of less than 20 mrem/hour. Although the dose rates for the HSM are extremely low, additional reductions in the dose rates may be obtained by increasing material thicknesses. The NUHOMS transfer cask and the DSC top shielded end plug are designed to limit the surface dose (gamma and neutron) to less than 200 mrem/hour. Temporary neutron shielding is required to be placed on the DSC shield plug and top cover plate during closure operations. Similarly, additional temporary shielding may be used to further reduce surface doses. Radiation zone maps of the HSM , cask, DSC surfaces and the area around these components are provided in Section 7.

3.3.5.3 Radiological Alarm Systems There are no radiological alarms required on the NUHOMS HSM or DSC. Instruments and alarms which may be required for the HSM installation site will be defined on a site specific basis.

3.3.6 Fire and Explosion Protection

The NUHOMS HSM and DSC contain no flammable material and the concrete and steel used for their fabrication could withstand any reasonable fire hazard. No specific fire protection system is required (other than an outside fire hydrant located nearby or hand held fire extinguishers). The specific fire protection system requirements will be assessed in site license applications.

Loading due to an internal explosion is not considered since no explosive gases are present within the DSC. The DSC has been calculated to withstand an external design pressure equivalent to immersion of greater than 50 feet of water.

3.3.7 Materials Handling and Storage

3.3.7.1 Spent Fuel Handling and Storage All spent fuel handling outside the plant's fuel pool will be done with the fuel assemblies in the DSC. Subcriticality during all phases of handling and storage is discussed in Section 3.3.4. The criterion for a safe configuration is an effective mean plus two-sigma neutron multiplication factor (K_{eff}) of 0.95. Section 3.3 calculations show that the expected K_{eff} value is below this limit.

For the purpose of establishing a conservative basis for this generic report, the criterion for spent fuel decay heat removal during long term storage conditions is to maintain the maximum cladding temperature to 340°C (644°F) or less. Higher cladding temperatures may be sustained for brief periods without affecting cladding integrity, however. During short term conditions such as DSC drying, transfer of the DSC to and from the HSM, and off-normal and accident temperature excursions, the maximum fuel cladding temperature is limited to 570°C (1,058°F) or less. This value is based on the results of experiments which have shown that Zircaloy clad rods subjected to short term temperature excursions below 760°C did not show indications of failure (3.20).

In order to substantiate the design basis initial storage cladding temperature limit of 340°C, a study was performed which included variations of several fuel parameters as follows:

Burnup	25,000-40,000 MWD/MTU
Initial Rod Fill Pressure	375-480 psia
Post-Irradiation Cooling Time	7-15 years

The study concluded that the design basis cladding temperature limit of 340°F is conservative and bounds the findings of current research efforts which studied the behavior of zircaloy cladding during long term dry storage (3.21). The cited research effort defines a methodology to determine an acceptable initial storage temperature which ensures that cladding degradation does not occur. A simple relationship for calculating the cladding stress is provided given the rod diameter, thickness, and pressure. The curves in the reference are then used to determine the acceptable initial storage temperature for a given cladding stress and cooling time. Applying this methodology for a range of rod fill pressures, burnups and ten years or less cooling time listed above, a minimum initial storage temperature of 344°C was determined. For this bounding generic analysis, a long term fuel cladding temperature storage limit of 340°C is conservatively used.

The following DSC external surface removable contamination limits should be used as a guide:

Beta/Gamma Emitters	10^{-4}	μ Ci/cm^2
Alpha Emitters	10^{-5}	μ Ci/cm^2

Other limits may be used on a site specific basis. External surface contamination on the transfer cask should be in accordance with plant specific technical limits. The DSC is sealed by double welds prior to storage so that any contamination of the DSC interior or its contents will remain confined during transfer and storage.

3.3.7.2 Radioactive Waste Treatment No radioactive waste will be generated during the storage life of the NUHOMS DSC. Radioactive wastes generated at the reactor during loading operations (contaminated water from the spent fuel pool and potentially contaminated air and helium from the DSC) will be treated using existing plant system and procedures which are site specific.

3.3.7.3 On-site Waste Storage The requirements for on-site waste storage facilities are site specific. The requirements for those facilities, in general, will be satisfied by existing facilities for handling and storage of waste from the spent fuel pool.

3.3.8 Industrial and Chemical Safety

No hazardous chemicals or chemical reactions are involved in the NUHOMS system loading and storage process. Industrial safety relating to handling of the cask and DSC will require that procedures acceptable to the Occupational Safety and Health Administration (OSHA) be developed and applied on a site specific basis.

Table 3.3-1

NUHOMS SYSTEM COMPONENTS IMPORTANT TO SAFETY

1. On-site Transfer Cask

   1a. Shielding materials
   1b. Structural shell and cover plates
   1c. Upper and lower trunnions

2. Dry Shielded Canister

   2a. Internal basket assembly
   2b. Shielded end plugs
   2c. DSC containment pressure boundary

3. Horizontal Storage Module

   3a. DSC support assembly
   3b. HSM reinforced concrete and structural steel
   3c. HSM passive ventilation system

Table 3.3-2

RADIOACTIVE MATERIAL CONFINEMENT BARRIERS
FOR NUHOMS SYSTEM

Confinement Barriers and Systems

1. Fuel Cladding
2. DSC Containment Pressure Boundary
3. Inner Seal Weld Primary Closure of DSC
4. DSC Shielded End Plugs
5. Outer Seal Weld Secondary Closure of DSC
6. DSC Cover Plates

Table 3.3-3

DESIGN PARAMETERS FOR CRITICALITY ANALYSIS OF THE DSC

Parameter	Design Value
FUEL ASSEMBLIES	
Number/Type	24/PWR
Rod Array	15x15
Number of Fuel Rods	208
Number of CR Guide Tubes	16
Number of Instrument Tubes	1
Rod Pitch (inch)	0.568
Burnup Credit	Yes
FISSILE CONTENT (% initial U equivalent)	
^{235}U	1.45
^{238}U	98.55
FUEL PELLETS	
Density (g/cm^3)	10.14
Diameter (inch)	0.369
FUEL ROD CLADDING	
Material	Zircaloy-4
Thickness (inch)	0.0265
Outside Diameter (inch)	0.430
CONTROL ROD GUIDE TUBES	
Material	Zircaloy-4
Thickness (inch)	0.016
Outside Diameter (inch)	0.530
INSTRUMENT TUBE	
Material	Zircaloy-4
Thickness (inch)	0.026
Outside Diameter (inch)	0.493
DSC GUIDE SLEEVES	
Material	304 Stainless Steel
Thickness (inch)	
12 Outer	0.0598
12 Inner	0.1046
Outside Width (inch)	
12 Outer	9.120
12 Inner	9.109

Table 3.3-3

DESIGN PARAMETERS FOR CRITICALITY ANALYSIS OF THE DSC

(Concluded)

Parameter	Design Value
OVERSLEEVES	
Material	304 Stainless Steel
Thickness (inch)	0.125
DSC FILL MATERIAL	
Material	Unborated Water
Density (g/cm^3)	10^{-4} - 1.0
DSC SHELL	
Material	304 Stainless Steel
Thickness (inch)	0.625
Outside Diameter (inch)	67.25
CASK	
Material	Steel/Lead
Thickness (inch)	0.5-3.5-1.5-3.0-0.13
Outside Diameter (inch)	85.3

Table 3.3-4

RESULTS SUMMARY FOR REACTIVITY EQUIVALENCE BURNUP CASES - NOMINAL CASE DSC MODEL FULLY LOADED WITH IRRADIATED FUEL

Table 3.3-5

SUMMARY OF SELECTED IRRADIATED FUEL EQUIVALENCE

CALCULATION RESULTS COMPONENTS

Table 3.3-6

BENCHMARK CRITICAL EXPERIMENTS

Table 3.3-7

CSAS2 PuO2 - UO2 CRITICAL EXPERIMENT BENCHMARK CALCULATIONS(1)

Table 3.3-8

CSAS2/SAS2/CASMO-2E IRRADIATED FA REACTIVITY CALCULATIONS METHOD VALIDATION
BY COMPARISON TO CASMO-2 K-inf RESULTS

Job Name	Case No.	Init. Enrich. (% U235)	Burnup (GW/MTU)	Mod. Temp. (°K)	CASMO-2E K-inf	CSAS2 K-inf	CSAS2 Bias (1) (ΔK/K)
CAS2EY3	113	2.91	0	293	1.37215	1.39212	0.01455 +/- 0.00292
CAS2EY3	114	2.91	0	422	1.36068	1.38186	0.01557 +/- 0.00304
CAS2EY3	122	2.91	20	422	1.13572	1.16556	0.02627 +/- 0.00251
CAS2EY3	123	2.91	30	422	1.04895	1.07833	0.02801 +/- 0.00258
CAS2EEC	114	3.20	0	422	1.38361	1.40239	0.01357 +/- 0.00274
CAS2EEC	123	3.20	30	422	1.07527	1.10647	0.02902 +/- 0.00281
CAS2EEC	124	3.20	40	422	0.99756	1.02617	0.02868 +/- 0.00193
CAS2E5L	114	3.41	0	422	1.39818	1.41200	0.00988 +/- 0.00303
CAS2E5L	123	3.41	30	422	1.09350	1.12349	0.02743 +/- 0.00257
CAS2E5L	124	3.41	40	422	1.01482	1.04365	0.02841 +/- 0.00222

Note:

1. Calculated (CSAS2 K-inf - CASMO2 K-inf) / (CASMO2 K-inf).

Table 3.3-9

AXIAL BURNUP VARIATION SENSITIVITY RESULTS SUMMARY -
DSC FULLY LOADED WITH
4 W/O IRRADIATED FUEL FLOODED WITH PURE WATER

Table 3.3-10

MODERATOR DENSITY SENSITIVITY RESULTS SUMMARY
FOR 1.45 W/O NON-IRRADIATED FUEL

Table 3.3-11

DSC CRITICALITY ANALYSIS SUMMARY OF FINAL K_{eff}

RESULT COMPONENTS FOR SELECTED POINTS ON

REACTIVITY EQUIVALENCE CURVE

Figure 3.3-1

NUHOMS-24P DSC AND KENO MODEL GEOMETRY

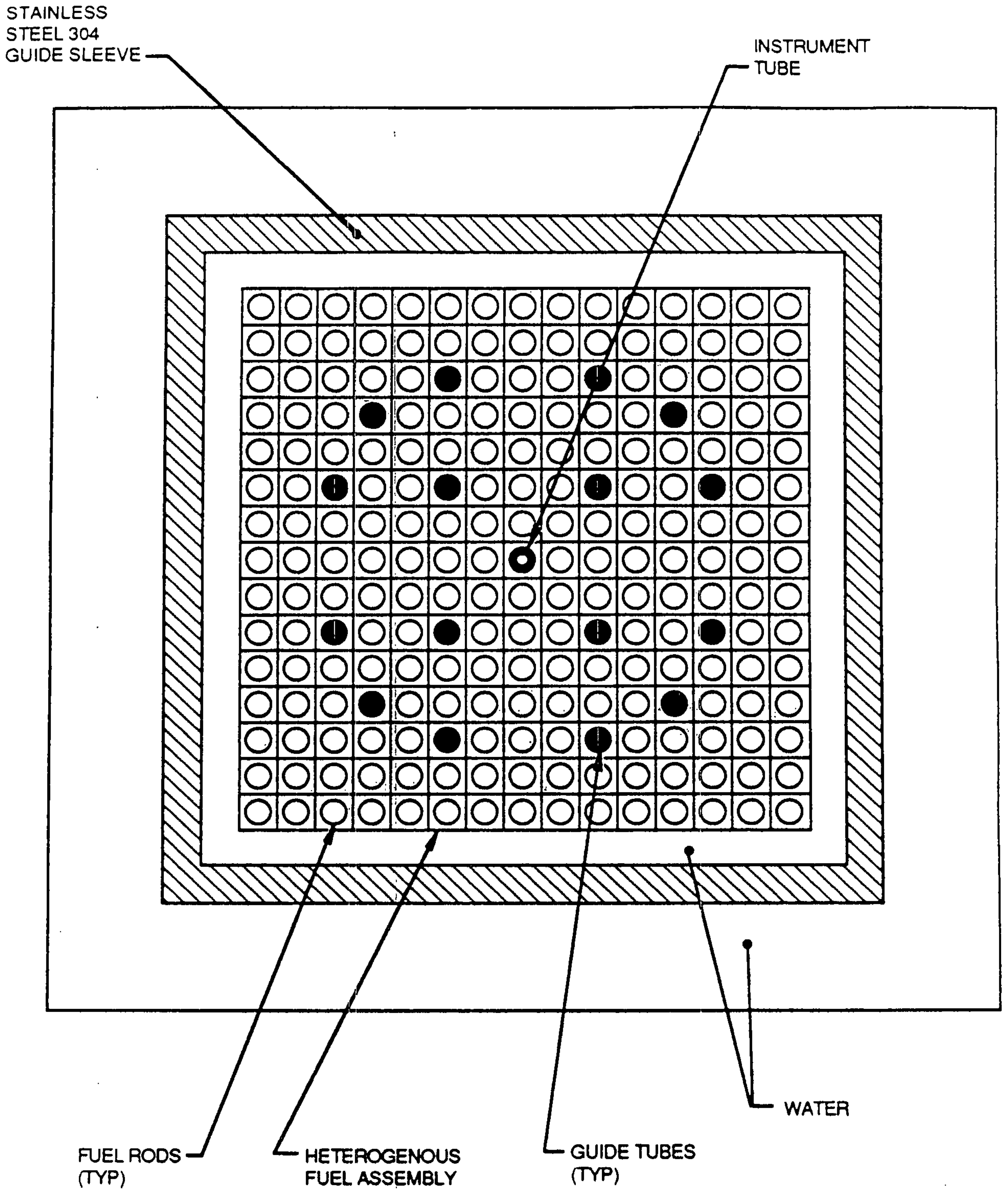


Figure 3.3-2

KENO MODEL FOR NUHOMS-24P FUEL

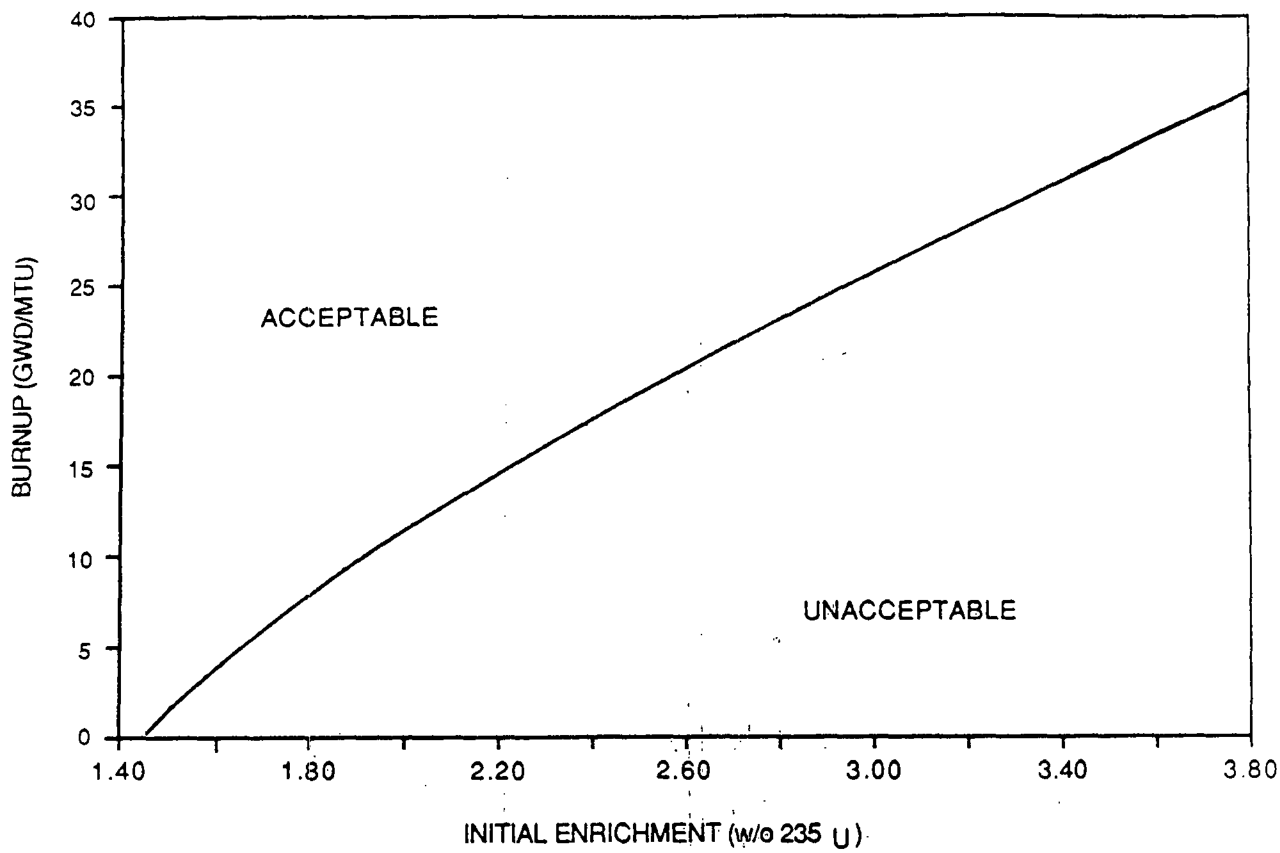


Figure 3.3-3

NUHOMS 24 ELEMENT DSC BURNUP EQUIVALENCE CURVE

Figure 3.3-4

KENO GEOMETRY OF 1/8 DSC ARRAY MODEL
USED TO ANALYZE AXIAL BU VARIATION
EFFECTS ON CALCULATED REACTIVITY

Figure 3.3-5

RELATIVE AXIAL BURNUP VS. FUEL HEIGHT
USED IN AXIAL BU SENSITIVITY STUDY

3.4 Classification of Structures, Components, and Systems

Safety during dry spent fuel storage using the NUHOMS system is assured by the two major components: the DSC and the HSM. The DSC provides for containment of all radioactive material, criticality control, decay heat removal, and axial shielding. The HSM provides shielding, a passive decay heat removal system, and protection against potentially hazardous natural phenomena. A detailed breakdown of the safety functions provided by the primary components of the NUHOMS system, and their relative importance and quality requirements are described in Section 3.3.

3.5 Decommissioning Considerations

The DSC is designed with consideration for interfacing with a transportation system planned to transport canistered intact fuel assemblies to either a monitored retrievable storage facility (MRS) or a geologic repository. However, if the fuel must be removed from the DSC, the DSC itself will be contaminated internally by crud from the spent fuel and may be slightly activated by spontaneous neutron emissions from the spent fuel. The DSC interior could be cleaned to remove surface contamination and disposed of as low-level waste. Alternatively, if the contamination and activation levels of the DSC are small enough (to be determined on a case-by-case basis), it may be possible to decontaminate the DSC and dispose of it as scrap.

While the intended logistics for the NUHOMS system include the eventual disposal of each DSC following fuel removal, current shield plug and cover weld designs do not preclude future development of a non-destructive plug removal technique that allows for reuse of the DSC shell/basket assembly. Economic and political conditions existing at the time of fuel removal would be assessed prior to making a decision to reuse the DSC.

The exact decommissioning plan to be applied will be dependent on the requirements of a specific utility and should be addressed on a site specific basis. However, because of the minimal contamination of the outer surface of the DSC, only small amounts of contamination may occur on the internal passages of the HSM.

3.6 REFERENCES

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4.0 INSTALLATION DESIGN

4.1 Summary Description

This chapter provides a more detailed description of the NUHOMS system including the HSM, DSC, and on-site transfer cask.

4.1.1 Location and Layout of Installation

The details of a NUHOMS ISFSI layout are site specific. The following guidelines are provided for site layout as a means of providing a basis for generic analysis of the anticipated HSM arrangements:

1. HSM units may be constructed as side-by-side modules in a single row ranging in size from a single stand alone HSM to a 1x10 array of HSMs. Adjoining HSM units shall have an expansion joint between them to permit thermal expansion. Common interior walls of the HSMs unit are 2'-0" thick. Outside end walls of an HSM unit are 3'-0" thick. The outside rear wall of a single module row HSM unit is 3'-0" thick. All remaining features of the HSMs shall be as defined by the Appendix E design drawings for the HSM.

2. HSM units may be constructed as a double module row of back-to-back HSMs ranging in size from a 2x1 module array to a maximum array size of 2x10. Adjoining HSM units will have an expansion joint between them to permit thermal expansion. Common interior walls of the HSM unit are 2'-0" thick. Outside end walls of an HSM unit are 3'-0" thick. All remaining features of the HSMs shall be as defined by the Appendix E design drawings for the HSM.

3. A concrete slab with access space in front of the HSM's is needed which ranges in size from 30' to 50' wide, depending on the site specific layout and transfer equipment designs.

4.1.2 Principal Features

The principal features of a NUHOMS ISFSI installation are described in Section 1.3.2 The details of a plant's ISFSI are site specific and will be addressed in site license applications.

4.1.2.1 Site Boundary This information is site specific and will be addressed in site license applications.

4.1.2.2 Controlled Area This information is site specific and will be addressed in site license applications.

4.1.2.3 Site Utility Supplies and Systems This information is site specific and will be addressed in site license applications.

4.1.2.4 Storage Facilities The NUHOMS system for dry storage consists of two components, the DSC and HSM, which act in concert to provide shielding, confinement, and cooling for spent fuel. The location and arrangement of the HSM facilities and DSC handling systems is site specific and will be addressed in site license applications.

4.1.2.5 Stack There is no stack in the NUHOMS system.

4.2 Storage Structures

4.2.1 Structural Specifications

Fabrication and construction specifications will be developed in accordance with 10CFR72 (4.1) and industry codes and standards. The codes and standards used for the NUHOMS-24P components, equipment, and structures are identified throughout the TR. They are summarized as follows:

Component, Equipment, Structure	Code of Construction
DSC	ASME Code, Section III, Subsection NB 1983 Edition with Winter 1985 Addenda
Transfer Cask	ASME Code, Section III, Subsection NC 1983 Edition with Winter 1985 Addenda
HSM	ACI-318-83 Code
DSC Supports	AISC Code Eighth Edition
Transfer Equipment	AISC, ANSI, AWS and/or other applicable standards.

4.2.2 Installation Layout

The layout and location of the NUHOMS HSMs will be submitted by the utility applying for an ISFSI license under 10CFR72.

4.2.3 Individual Unit Description

4.2.3.1 Dry Shielded Canister The DSC is a high integrity stainless steel, welded pressure vessel that provides confinement of radioactive materials, encloses the fuel in a helium atmosphere, and, together with the transfer cask, provides biological shielding during draining, drying, closure and transfer operations. The NUHOMS-24P DSC design is illustrated in Figure 1.3-1. Detailed design drawings for the DSC are contained in Appendix E of this report.

The DSC cylindrical shell is constructed from a rolled and butt-welded stainless steel plate material. Stainless steel structural plates and fabricated casings for the poured lead shielding material form the DSC top and bottom end assemblies. These assemblies are double seal welded to the DSC shell to form the containment pressure boundary.

The DSC shell, and top and bottom end assemblies enclose a stainless steel basket assembly which serves as the structural support for the SFAs. The basket assembly consists of 24 stainless steel guide sleeves and eight spacer discs located at the grid spacer locations of the fuel assembly. The spacing of these plates depends on the type of fuel being stored. Criticality control is achieved as described in Section 3.3. The spacer disks maintain the cross-sectional spacing of the fuel assemblies and provide lateral support for the fuel assemblies and the guide sleeves. The spacer disks are held in place by four support rods which maintain longitudinal separation during a postulated cask drop accident.

The lead shielded end plug assemblies at each end of the DSC provide biological shielding when the DSC is in the transfer cask or in the HSM. The steel plates forming the casings of the end plugs are designed to retain the lead in a postulated cask drop accident. The top end plug assembly is captured between an inner structural plate and an outer cover plate, which provide a redundant pressure retaining function. The bottom end plug and cover plate assembly, the internal basket assembly, and the shielded end plug and cover plate assembly are fabricated and assembled in the fabricator's shop. The top shielded end plug and top cover plate assembly are installed at the plant after the fuel assemblies have been loaded into the DSC internal basket. Nominally, a diametral gap of 0.25" exists between the top shield plug and the DSC shell. The minimum radial gap at any point on the circumference of the top shield plug/DSC shell annulus is controlled during fabrication to be not less than .06". This gap is adequate to allow the plug to be freely inserted or removed from the DSC shell assembly with the DSC submerged in the fuel pool.

The top surface of the upper shielded end plug is welded to the DSC shell to form the primary pressure boundary at the upper end of the DSC. A secondary pressure boundary is provided by the top cover plate which is also welded to the DSC shell. All closure welds are multiple-pass welds. This effectively eliminates a pinhole leak which might occur in a single-pass weld, since the chance of pinholes being in alignment on successive weld passes is negligibly small. In addition, the DSC end plates are sealed by separate, redundant closure welds. The circumferential and longitudinal shell plate weld seams are fabricated using multi-pass full penetration butt welds. These welds are fully radiographed and inspected according to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, to insure that the integrity of the welded joint is as sound as the parent metal itself. The remaining pressure boundary welds will be tested to the same code standards, either ultrasonically or multilevel dye penetrant examination.

These stringent design and fabrication requirements insure that the containment pressure retaining function of the DSC is maintained. It should be noted that pressure monitoring instrumentation is not used since penetration of the pressure boundary would be required. The penetration itself would then become a potential leakage path and, by its presence, compromise the integrity of the DSC design.

The top shielded end plug support ring assembly includes a drain and fill port similar in construction to its mating component. The design incorporates two small diameter tubing penetrations into the DSC cavity for draining and filling operations. One penetration, the vent port is terminated at the bottom of the end plug assembly. The other port is attached to a siphon tube, which continues to the bottom of the DSC cavity. The drain and fill port includes a two plane, dog-leg type offset to prevent streaming. The drain and fill ports terminate in normally closed quick-connect fittings. Both ports are used to remove water from the DSC during the drying and sealing operations.

The drying and sealing operations are described in Section 4.7.3. Transfer of the DSC to the HSM by the hydraulic ram is done by grappling the ring plate assembly welded to the bottom cover plate of the DSC. This thick plate prevents significant deformation and bending stress that might occur in the DSC during handling. When the DSC is transferred by the hydraulic ram, the load is sustained by this cover plate.

Four lifting lug plates are provided on the interior of the DSC shell to facilitate placement of the empty DSC into the transfer cask prior to fuel loading. The DSC is lowered into the transfer cask cavity using the fuel building crane or another suitable crane at the utility's option. Shackles and rope slings are used to rig the DSC lifting lugs to the crane hook.

The transfer cask bottom cover plate assembly includes a shield plug which fits into the DSC grapple ring. DSC orientation during insertion into the transfer cask is achieved by alignment of match marks on the DSC and the cask. In addition, a key-way detail is used for the DSC basket assembly to ensure that the fuel matrix orientation with the DSC shell and transfer cask is maintained.

Frictional loads during DSC transfer are reduced by the application of a dry film lubricant to the lower portion of the

DSC shell surface which is in contact with the cask liner during horizontal DSC transfer and the DSC support rail surfaces inside the HSM. The lubricant chosen for this application is Everlube 823 (4.3), a tightly adhering inorganic lubricant with an inorganic binder. Everlube 823 is a dry film lubricant which provides a thin, clean, dry, permanently bonded layer of lubricating solids that is intended to reduce wear, and prevent galling in metals. It is shop applied as a thin sprayed coating, like paint, and is baked on, using a carefully controlled curing schedule. Everlube 823 is not affected by water and was designed to be highly resistant to aggressive chemicals such as fuming nitric acid and hydrazine. Everlube 823 maintains its lubricity even when exposed to these chemical agents. This product was designed for radiation service and has a coefficient of sliding friction of 0.08 against stainless steel.

Following DSC fabrication, leak tests of the DSC shell assembly as well as the drain and fill port subassembly will be performed in accordance with the ASME Code, Section III, Subarticle NB-5300 and Section V, Article 10. The leak tests provision of the ASME Code meet the intent of N14.5-1977, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment."

The principal material of construction for the NUHOMS-24P system DSC is Type 304 stainless steel. All structural component parts of the DSC are fabricated from this material. Common lead is used for shielding of the DSC top and bottom ends, as previously described. The DSC cylindrical shell and DSC containment pressure boundary component parts are ASME SA240 Type 304 material (4.5).

<u>4.2.3.2 The Horizontal Storage Module</u> The design of the NUHOMS-24P HSM has been developed in accordance with the applicable codes and a quality assurance program suitable for design of structures important to safety, as documented in Section 3. The design of the NUHOMS-24P HSM is fundamentally the same as that reviewed and approved by the NRC for the NUHOMS-07P design (TR No. NUH-001, Revision 1A). The width and height of the HSM cavity have been increased proportionally to accommodate the larger NUHOMS-24P DSC. All other features of the HSM are the same.

The HSM is a massive reinforced concrete structure that provides protection for the DSC against tornado missiles and other potentially adverse natural phenomena. The HSM also serves as the principal biological shield for the spent fuel during storage. The NUHOMS-24P HSM design is illustrated in Figure 1.3-1a. Design drawings for the HSM are contained in Appendix E of this report. Specific design details for the HSM may vary with architect/engineer or utility standards for reinforced concrete construction and will be addressed in site license applications. Representative design details for the HSM are depicted in the illustrations and drawings referred to herein.

The HSM contains a shielded air inlet opening in the lower front wall of the structure to admit ambient ventilation air into the HSM. The air inlet opens into a plenum formed by an interior shielding slab and a partial-height wall. Ventilating air exits the plenum from two vertical and one horizontal openings. The cooling ventilation air flows around the DSC (see Figure 1.3-2) to the top of the HSM. Air warmed by the DSC is exhausted through two shielded vent openings in the HSM roof. This passive system provides an effective means for spent fuel decay heat removal. A heat shield, Figure 4.2-7, is provided between the DSC and HSM concrete to control the peak concrete temperatures.

The DSC rests on a support rail assembly inside the HSM, which is anchored to embedments in the HSM interior walls. The support assembly is fabricated from structural steel. The DSC support assembly member sizes are W10x68 for the cross beams and WT6x115 for the support rails, as shown in Figure 4.2-1. The support assembly is shimmed level and bolted to the supports embedded in the HSM walls. The DSC support system connection details are shown in Figure 4.2-2. The support rails extend into the HSM access opening, which is formed by casting in place a steel sleeve which is slightly larger in diameter than the DSC. The HSM access opening sleeve details are shown in Figure 4.2-3. The HSM access opening sleeve has a stepped flange sized to facilitate docking of the transfer cask for DSC transfer. This configuration was chosen to minimize streaming of radiation through the HSM opening during DSC transfer.

In the HSM access opening, the support rails will be shimmed to align and level the rails which are then welded to the access opening sleeve. Thermal expansion of the support rails is permitted by using slotted bolted connections to attach the rails to the structural cross-members. The top surfaces of the rails, on which the DSC slides, are coated with a dry film lubricant (Everlube 823, see previous discussion for DSC) which is designed for a radiation environment. The support assembly surfaces will also be coated with a sacrificial anodic coating (zinc paint), galvanizing, or hard plating for corrosion protection. The ambient atmospheric conditions will depend on the site location. Inside the HSM, the heat rejected from the DSC will have a drying effect, thus the HSM atmosphere will be benign in terms of corrosion. Decay heat will warm the air, thus preventing the accumulation or condensation of moisture inside the HSM.

The DSC is prevented from sliding along the support rails during a postulated seismic event by a seismic restraining assembly. The assembly consists of rail stops attached to the rear ends of the DSC support rails, and a steel retaining block positioned in the front access opening of the HSM. The DSC seismic restraint detail is shown in Figure 4.2-4. The restraint has a plate which bears on the end of the DSC and transfers axial seismic loads to a shear key arrangement with the HSM access opening sleeve. The seismic restraint is set in place between the DSC support rails following transfer of the DSC to the HSM.

Clearance between the DSC seismic restraint and the DSC is designed for the maximum DSC thermal growth which occurs during the postulated HSM blocked vent case, as discussed in Section 8.2-7. During normal storage there will be a small (1/8 to 1/4 inch) gap which will allow movement of the DSC relative to the HSM. This motion produces a small increase in the DSC axial force due to seismic loads, and has been included in the design of the DSC seismic restraint shear key arrangement.

The HSM wall and roof thicknesses were primarily determined by shielding requirements and are demonstrated to adequately protect the DSC against tornado missiles and other adverse natural phenomena. The tornado generated missile effects are considered to bound any other reasonable impact-type accident. The required HSM wall thickness for individual modules and HSM arrays are specified on the Appendix E drawings and discussed in Section 4.1.1.

The entrance to the HSM is covered by a thick steel plate (a vertically sliding door). The HSM door frame details are shown in Figure 4.2-5. The door assembly also includes a solid neutron-absorbing material which acts as a combined gamma and neutron shield as shown on the Appendix E drawings. For security purposes, the HSM door will be tack welded closed to its frame.

During DSC loading/unloading operations, the transfer cask is docked in the HSM door opening recess and mechanically secured to tiedown embedments provided in the front wall of the HSM. The cask restraint system used for this purpose is shown in Figure 4.2-6. The tiedown embedments are equally spaced above and below the HSM horizontal center line on either side of the HSM access opening. The HSM embedments are designed as conventional embedded hooks in accordance with the ACI 349-85 Code. The transfer cask restraint system is designed for loads which occur during normal DSC transfer operations and during an off-normal jammed DSC event.

The HSM air vent inlet and outlets will be covered with stainless steel wire bird screen to prevent foreign material from entering the HSM. Periodic surveillance constitutes the only required maintenance activity in the NUHOMS storage system.

The design flexibility of the HSMs permits a utility to choose the most economical arrangement of HSMs which best meets plant specific conditions and requirements. This TR presents details for a single stand-alone module as this is the governing design case for the postulated environmental loads such as earthquake, flooding, and tornado loads. Analyses and economic evaluations have shown that the probable maximum number of HSMs to be constructed at one time as a unit is a 2x10 array of HSMs for the storage of 20 DSCs. Thermal loads provide the critical load cases for the HSM structural design for this bounding size HSM unit. The reinforcing steel design for all smaller HSM unit sizes, i.e., a single module up to a 2x10 array of modules, have been conservatively detailed based on the design analysis performed for the bounding HSM unit sizes.

A typical reinforcing steel layout for the HSM basemat, walls, and roof is shown in Figure 8.1-9. The reinforcement sizing and placement specified will be used for HSM array configurations ranging in size from a single stand alone module to a 2x10 array of HSMs. Construction details, such as concrete joints and reinforcing bar lap splices, will be shown on plant specific construction drawings and will vary with the number of HSMs being constructed at one time.

The HSM design documented in this TR is constructed of 5,000 psi (min.) normal weight (145 pounds per cubic foot minimum density) concrete with Type II Portland cement meeting the requirements of ASTM C150 (4.6). The aggregate will meet the specifications of ASTM C33 (4.6). The concrete is reinforced by ASTM A615 Grade 60 (4.7) deformed bar placed vertically and horizontally in each face of the walls and roof. Concrete and reinforcing steel of varied strengths and properties may be used in site specific designs. These will be addressed in site license applications.

For NUHOMS ISFSIs with a storage capacity larger than a 2x10 HSM unit, an expansion joint is provided between the adjacent HSM units to permit thermal expansion. Therefore, the HSM structural design presented in this TR is applicable to any size NUHOMS ISFSI comprised of HSM units which are 2x10 module arrays or smaller.

4.2.3.3 NUHOMS-24P Transfer Cask The NUHOMS-24P transfer cask is a nonpressure-retaining cylindrical vessel with a welded bottom assembly and bolted top cover plate. The transfer cask is designed for on-site transport of the DSC to and from the plant's spent fuel pool and the HSM. The transfer cask provides the principal biological shielding and heat rejection mechanism for the DSC and SFAs during transfer from the fuel pool, drying operations, DSC closure, and transport to the HSM. The transfer cask also provides primary protection for the loaded DSC during off-normal and drop accident events postulated to occur during the transport operations. The NUHOMS-24P transfer cask design is illustrated in Figure 1.3-2a. Detailed design drawings of the transfer cask are contained in Appendix E of this report.

The transfer cask cross section is constructed from three concentric cylinders to form an inner and outer annulus which are filled with lead and a water-based liquid. The three cylinders are welded to heavy forged ring assemblies at the top and bottom ends of the cask as shown in Figure 1.3-2a. The inner stainless steel liner is polished to a smooth finish to minimize sliding friction during horizontal transfer of the DSC which is coated with a dry film lubricant, Everlube 823 to and from the HSM. The transfer cask structural shell is fabricated from SA516 Grade 70 carbon steel to provide good impact toughness properties for postulated drop events. These include the cylindrical shell and the bolted top cover plate. All steel surfaces exposed to fuel pool water are stainless steel. The transfer cask carbon steel structural shell and top cover plate are coated with a durable epoxy paint which is shop applied in accordance with the manufacturer's standards. This paint is suitable for immersion service with a continuous temperature of 250°F with intermittent temperatures to 400°F.

The preferred method used for pouring the transfer cask lead shielding will vary between fabricators. Only one transfer cask will be provided for each plant. Transfer casks for different plants may be supplied by different fabricators. Each fabricator is required to submit detailed procedures for the lead pours consistent with the requirements delineated on the Appendix E drawings. These procedures include specific locations and sealing of pour holes, temporary bracing, and controlled cooling methods for the lead, all of which must meet ASME Code requirements.

The transfer cask liquid neutron shield includes suitable provisions to minimize the potential for freezing or boiling and corrosion. The neutron shield contains an ethyelene-glycol-water mixture (approximately 50-50) to prevent freezing at -40°F or boiling at the maximum postulated liquid neutron shield temperature of 260°F. An expansion tank and relief valve are provided for the neutron shield cavity for volumetric expansion and internal pressure control. The expansion tank system is illustrated in Figure 4.2-8. The expansion tank is constructed from a standard pipe section with end caps on either end. The expansion tank is connected to the vent port for the cask neutron shield with tubing. The expansion tank is sized to accommodate the increased liquid volume plus the volume of air which is initially present in the system.

As compared to a solid material, the use of a liquid neutron shield has the added advantage of better shielding and heat transfer characteristics (i.e., higher thermal conductivity) at a reduced weight. A utility may elect to substitute an equal volume of solid neutron shield material such as Boro-Silicon in the cavity of the liquid neutron shield for the NUHOMS-24P transfer cask. The effects of this design alternative on the safety analysis presented herein shall be addressed in site specific license applications. Solid neutron shielding materials are already incorporated into the top and bottom closure heads to minimize maintenance while providing effective radiological protection.

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

The cask bottom cover plate assembly is a water tight closure used during fuel loading in the fuel pool, during DSC closure operations in the cask decon area, and during cask handling operations in the fuel building. Prior to cask transport from the plant's fuel building to the HSM, the bottom cover plate of the cask is removed and a temporary neutron/gamma shield plug is attached. An illustration of the temporary shield plug design is shown in Figure 4.2-9. The temporary shield plug is a two piece construction with a center cover which is removed from ram insertion. The temporary shield plug is designed such that the contact dose rate is 100 mrem/hour or less.

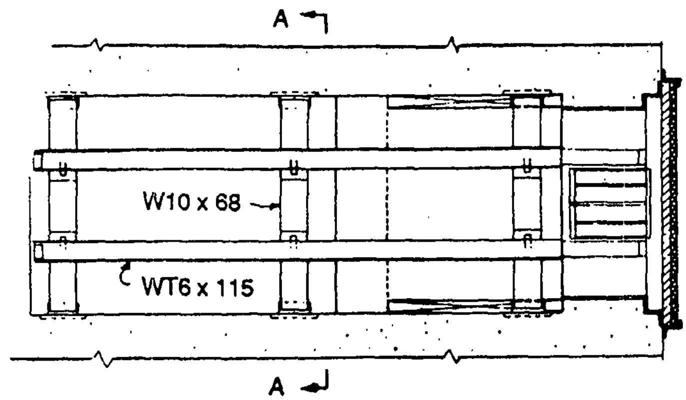
Alignment of the DSC with the transfer cask is achieved by the use of permanent alignment marks on the DSC and transfer cask top surfaces. These will permit orienting the DSC to the required tolerances for fuel loading. The circular projection on the transfer cask bottom cover plate is dimensioned to ensure that the DSC does not contact any surface of the bottom cover plate assembly and, therefore, removal of the bottom cover plate will not require the use of force.

The yoke design used to lift the cask is a non-redundant two point lifting device with a single pinned connection to the crane hook. Thus, 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. Also, the cask lifting trunnions are designed in accordance with ANSI-14.6 which imposes additional factors of safety over and above those already included in the ASME Code used for the cask design.

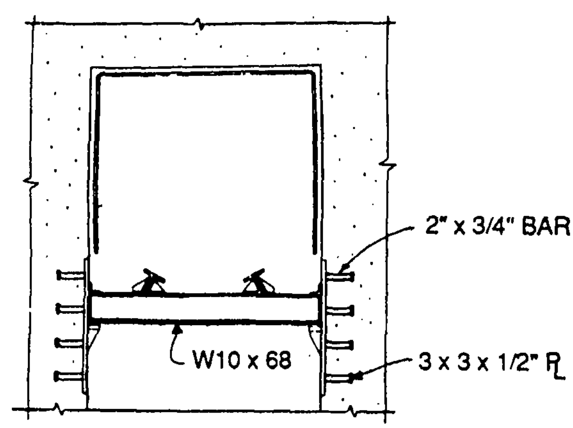
As shown in Figure 4.2-10, the cask upper flange is designed to allow an inflatable elastomeric seal to be inserted between the cask liner and the DSC. The seal is fabricated from fabric reinforced elastmeric material rated for temperatures well above boiling. The seal is placed after the DSC is located in the cask and serves to isolate the clean water in the annulus from the water in the spent fuel pool. After installation, the seal is inflated to prevent contamination of the DSC exterior surfaces by waterborne particulates.

The structural materials and fabrication of the NUHOMS-24P transfer cask meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC for Class 2 vessels except that no pressure test will be performed and no N-stamp is required. The upper lifting trunnions are conservatively designed in accordance with the ANSI N14.6 (4.9) requirements for critical loads. All structural welds are radiographed, ultrasonically examined, or tested by the dye penetrant method as appropriate for the weld configuration. These stringent design and fabrication requirements will insure the structural integrity of the transfer cask and the performance of its intended safety function.

Impact loads which would result from a postulated drop accident are primarily resisted by the structural elements of the transfer cask including the cylindrical shell, the bottom ring and cover plate, and the top flange and cover plate. Based upon plant specific surface conditions along the transfer route from the fuel building to the HSM installation, an evaluation of potentially impacted surfaces is needed to ensure that the design basis deceleration magnitudes for the cask are not exceeded, as discussed in Section 8.2 of this report.



PLAN



SECTION A-A

Figure 4.2-1

DSC SUPPORT ASSEMBLY DETAILS

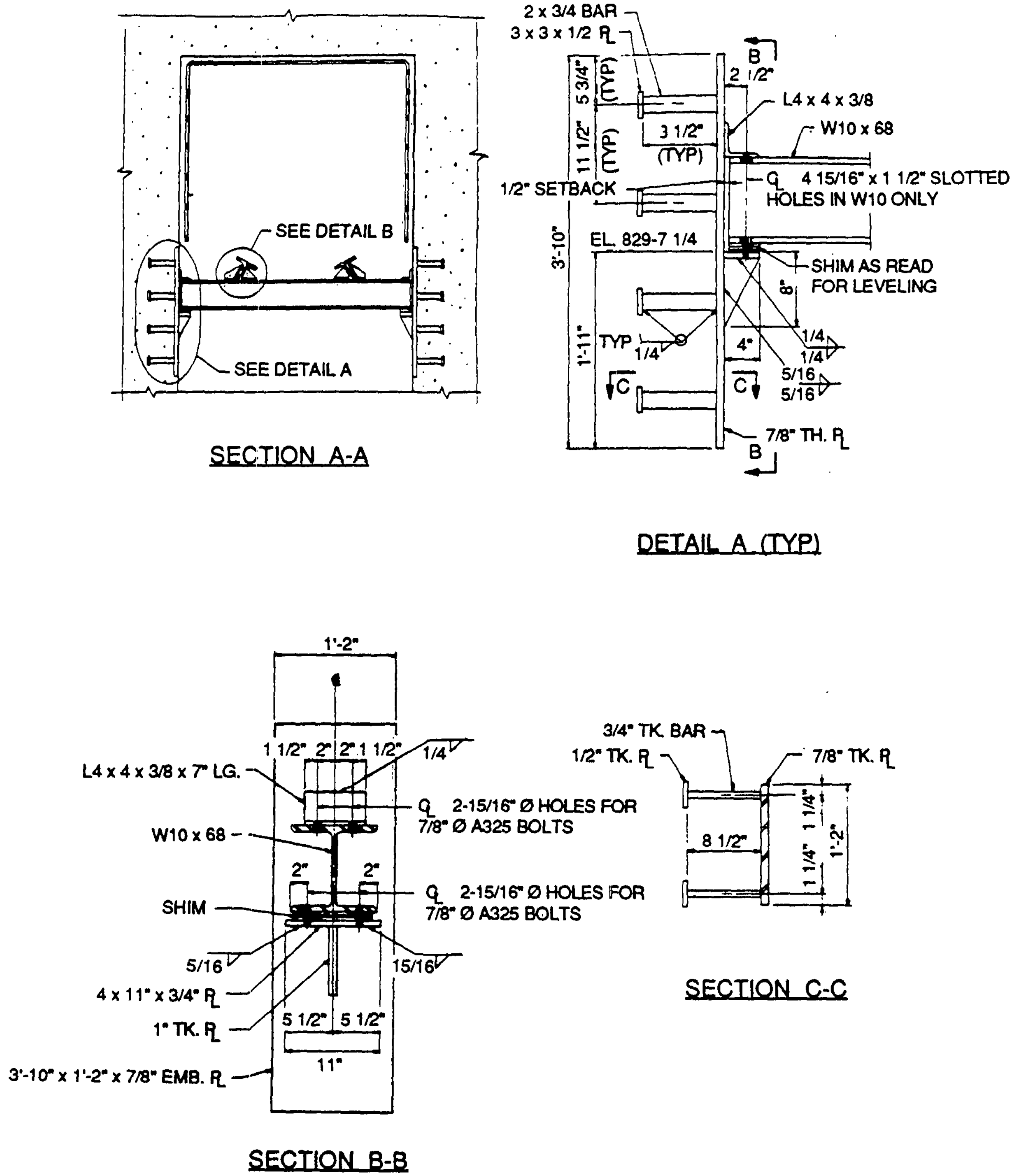


Figure 4.2-2

DSC SUPPORT ASSEMBLY CONNECTION DETAILS

Figure 4.2-2a

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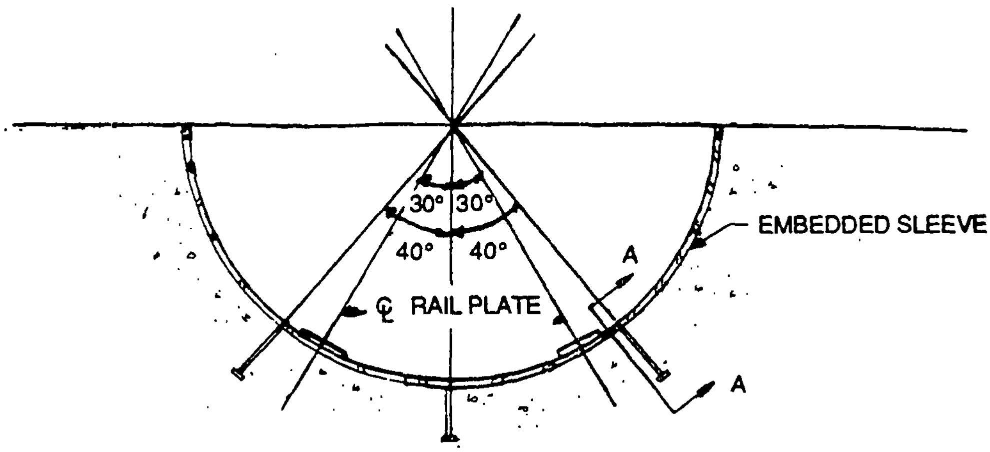

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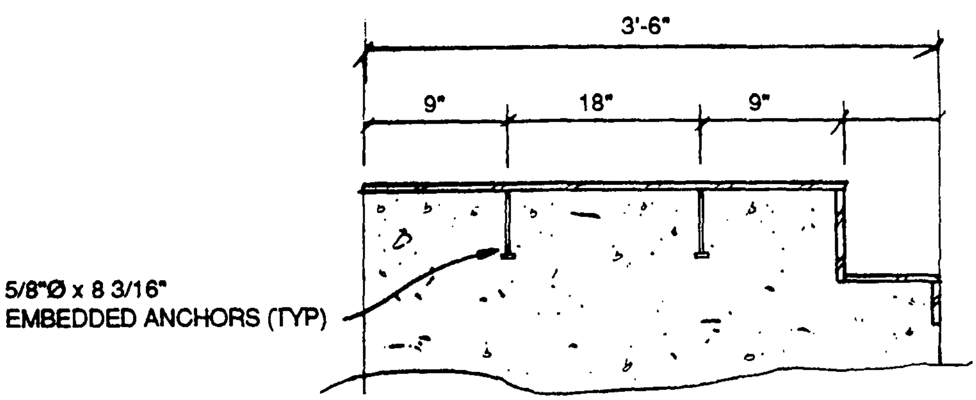


SECTION A-A

Figure 4.2-3

HSM ACCESS OPENING SLEEVE DETAILS

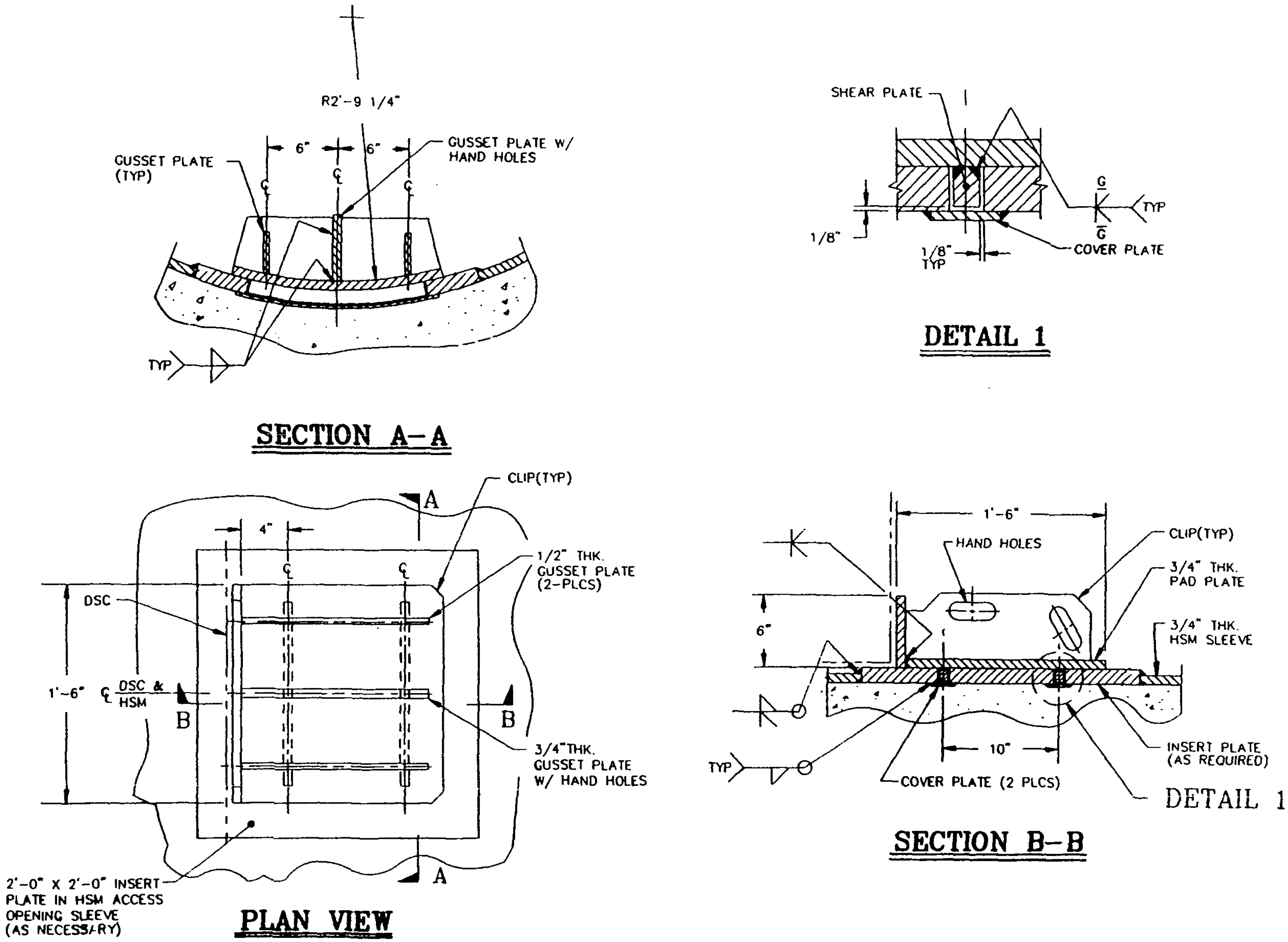


Figure 4.2-4

HSM SEISMIC RESTRAINT DETAIL

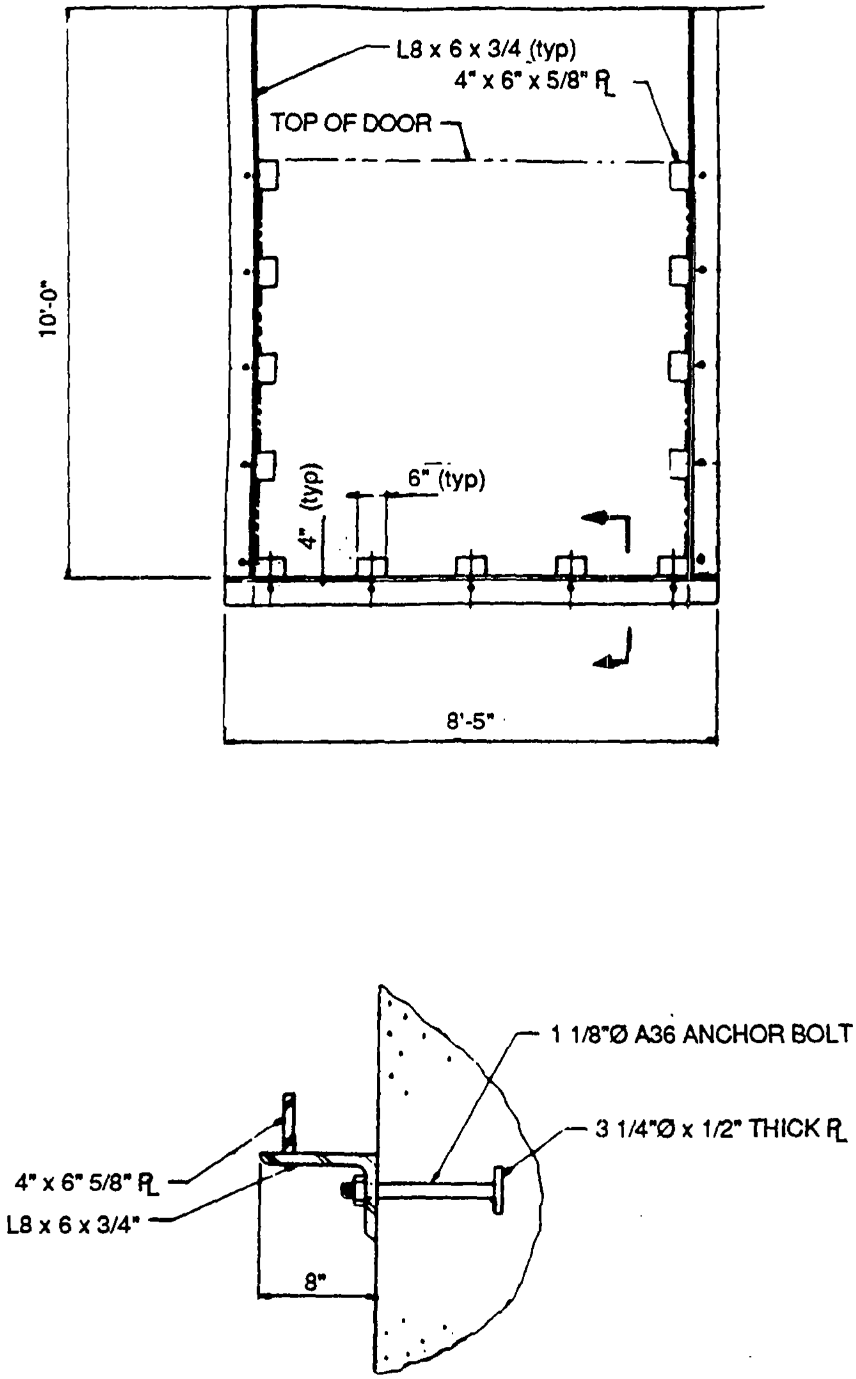


Figure 4.2-5

HSM DOOR FRAME DETAILS

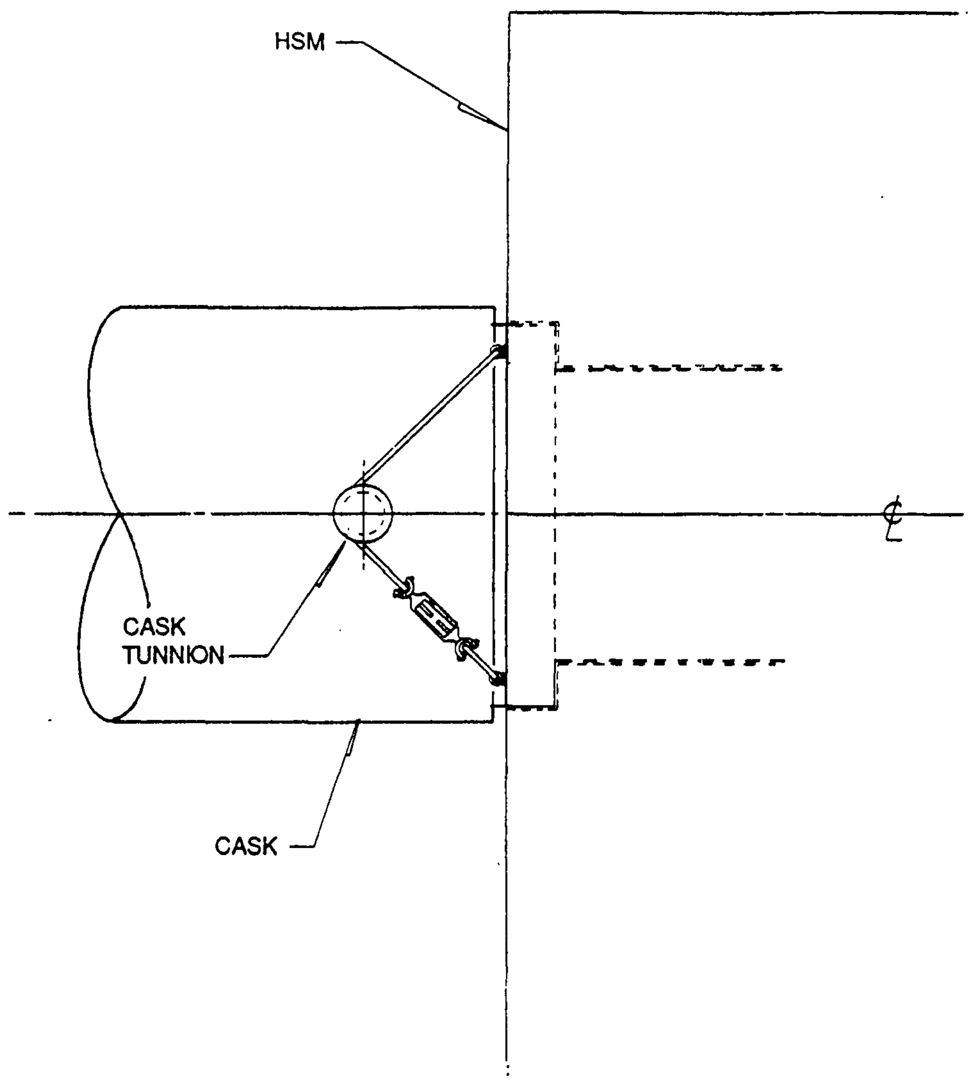


Figure 4.2-6

ELEVATION VIEW OF CASK RESTRAINING SYSTEM

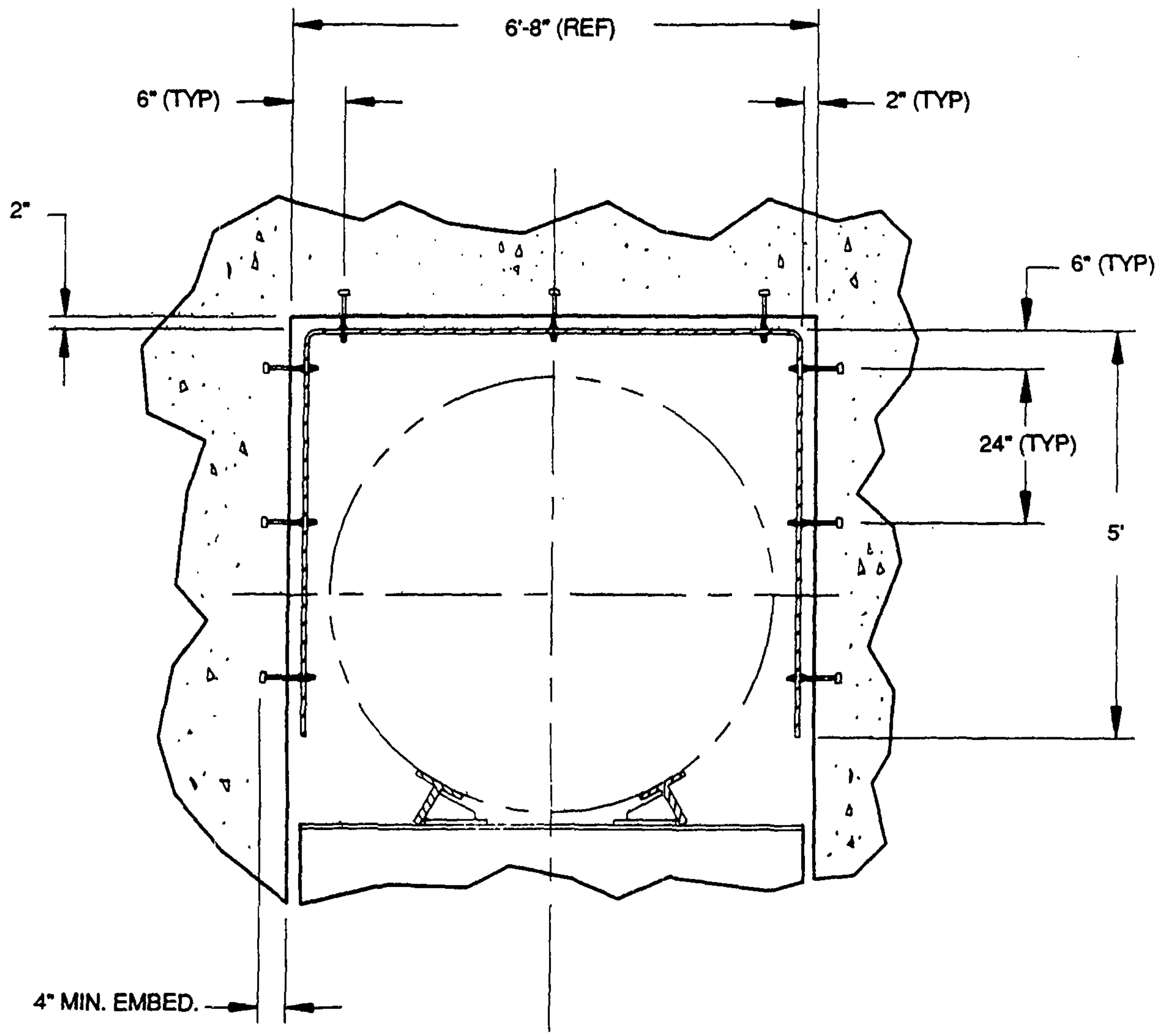


Figure 4.2-7

HSM HEAT SHIELD DETAIL

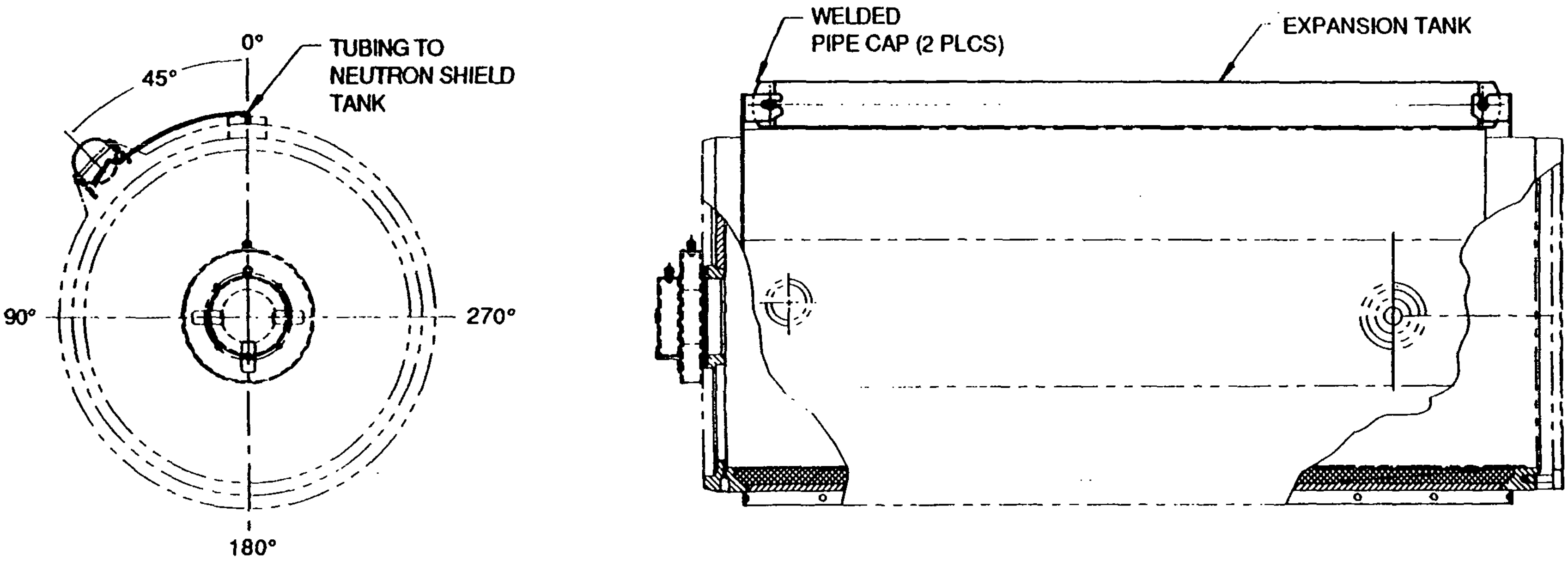


Figure 4.2-8

TRANSFER CASK NEUTRON SHIELD EXPANSION TANK

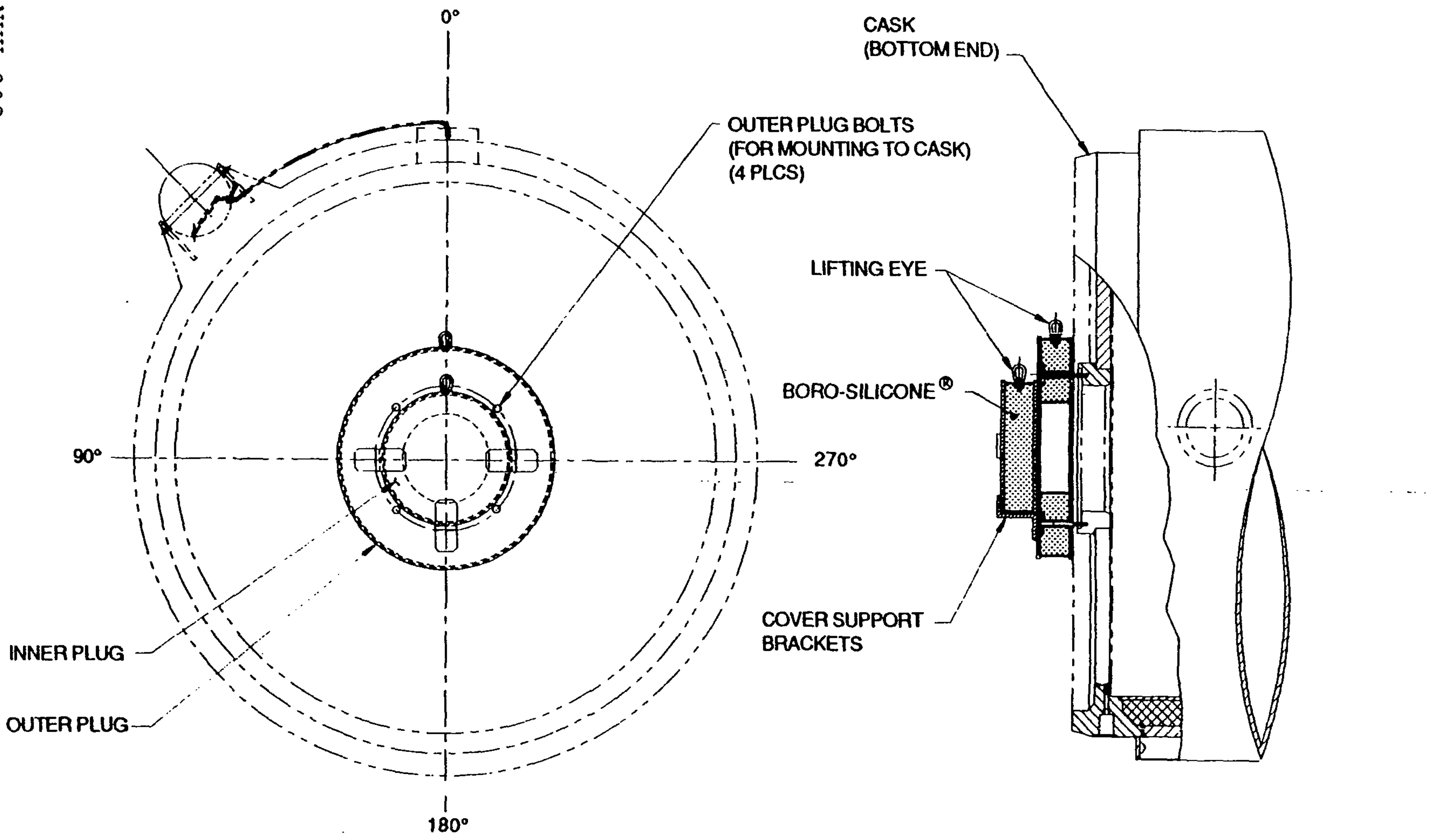


Figure 4.2-9

TRANSFER CASK TEMPORARY SHIELD PLUG

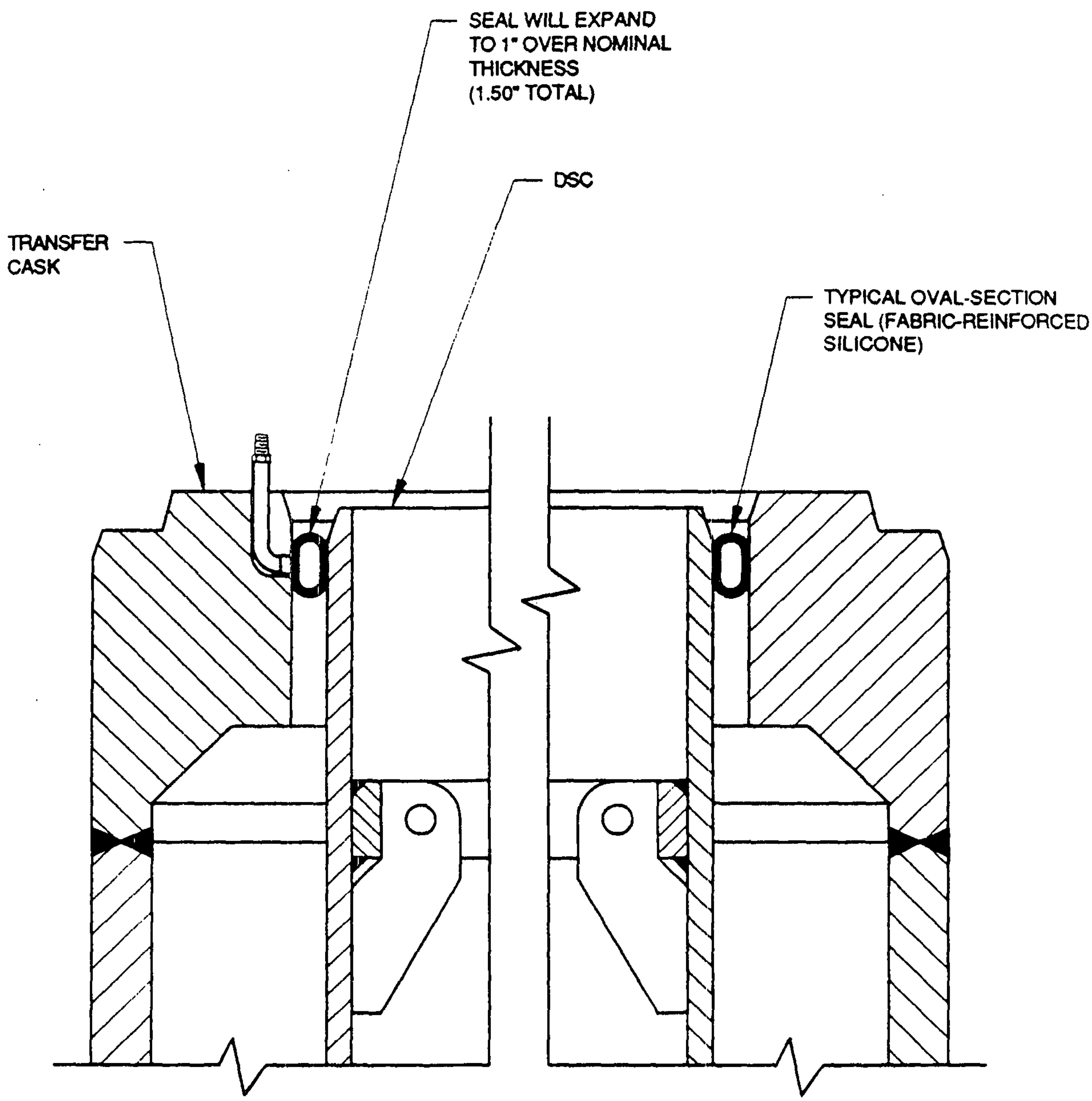


Figure 4.2-10

DSC/TRANSFER CASK ANNULUS SEAL DETAIL

4.3 Auxiliary Systems

The NUHOMS HSM is a self-contained, passive storage facility which requires no support systems.

4.3.1 Ventilation Systems

Spent fuel confined in the NUHOMS DSC is cooled by conduction and radiation within the DSC, and conduction, convection and radiation from the DSC surface. An air inlet near the bottom of the HSM front wall and outlets in 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.

4.3.1.1 Offgas Systems Any offgas systems required during the DSC drying and backfilling operations utilize existing plant systems which are site specific and will be addressed in the site license applications.

4.3.2 Electrical Systems

No electrical systems are required for the HSM or DSC during long term storage. Nonessential electrical power is used during DSC closure operations and during DSC transfer operations to the HSM. The required electrical power will be obtained from existing plant systems which are site specific and will be addressed in site license applications.

4.3.3 Air Supply System

The plant's existing nonessential air supply system will be needed to supply clean compressed air at a minimum pressure of 25 psig.

4.3.4 Steam Supply and Distribution System

There are no steam systems required.

4.3.5 Water Supply System

The NUHOMS-24P system requires borated water for the DSC cavity compatible with the plant's existing fuel pool. Demineralized water is needed for filling the DSC/cask annulus, for washdown operations, and for the transfer cask neutron shield cavity which is mixed with a predetermined amount of ethelyne glycol. This water will be supplied by existing plant systems.

4.3.6 Sewage Treatment System

The sewage treatment system requirements are site specific and will be addressed in site license applications.

4.3.7 Communications and Alarm System

Communications and alarm systems required for a NUHOMS ISFSI will be addressed in site license applications.

4.3.8 Fire Protection System

No fire detection or suppression system is required for a NUHOMS ISFSI. The HSM contains no combustible materials. The HSM site should be located well away from fuel tanks, gas pipelines, or other fire hazards. The provisions needed for fire suppression should be provided consistent with existing plant requirements. Fire safety will be addressed in site license applications.

4.3.9 Maintenance Systems

The NUHOMS System is designed to be totally passive with minimal maintenance requirements. During fuel storage, the system requires only periodic inspection of the air inlets and outlets to ensure that no blockage has occurred. The inspection will be done visually and any debris removal will be performed by hand or with hand held tools.

The transfer cask is designed to require only minimal maintenance. Transfer cask maintenance is limited to periodic inspection of critical components and replacement of damaged or nonfunctioning components. A detailed discussion of these requirements is provided in Section 4.5.

4.3.10 Cold Chemical Systems

There are no cold chemical systems.

4.3.11 Air Sampling System

There are no special air sampling systems required other than those that may be part of any existing plant offgas systems (see Section 4.3.1.1).

4.4 Decontamination System

4.4.1 Equipment Decontamination

Decontamination hardware and procedures are site specific. The principal planned decontamination activity is the prevention/ removal of contamination from the outside surface of the transfer cask. Such contamination is due to immersion in spent fuel pool water. To prevent exterior contamination by pool water, the annulus between the DSC and cask will be filled with clean demineralized water prior to insertion into the pool. The annulus will then be sealed with a mechanical seal and/or suitable tape.

Upon withdrawal from the fuel pool, the exterior surface of the transfer cask will be decontaminated to levels consistent with existing plant technical specifications prior to the transfer operation to the HSM. After decontamination of the cask outer surface, the mechanical seal or tape will be removed and the water in the cask/DSC annulus drained by means of the cask drain. The DSC exterior surface should be checked for smearable contamination to a depth of about one foot below the top surface. If no smearable contamination has penetrated to this depth, the DSC exterior will be presumed to be clean throughout its length. If smearable contamination exceeds plant technical specification limits, then the annulus should be flushed with clean demineralized water until acceptable smearable contamination levels are obtained.

4.4.2 Personnel Decontamination

The personnel decontamination system will be site specific and should be described in site license applications.

4.5 Transfer Cask Repair and Maintenance

The transfer cask is designed to minimize maintenance and repair requirements. The actual maintenance procedures are plant specific and will be addresssed in the site specific applications. However, the following items should be included:

4.5.1 Routine Inspection

The following inspections should be performed prior to each use of the transfer cask:

1. Visual inspection of the cask exterior for cracks, dents, gouges, tears, damaged bearing surfaces, and leakage from neutron shield fittings.

2. Visually inspect all threaded parts and bolts for burrs, chafing, distortion or other damage.

3. Check all quick-connect fittings to ensure their proper operation.

4. Visually inspect the interior surface of the cask for any indications of excessive wear to bearing surfaces.

4.5.2 Annual Inspection

The following inspections and tests shall be performed on an annual basis:

1. Test the neutron shield pressure relief system.

2. Inspect the cask lifting points and cask lifting yoke.

Any parts which fail these tests shall be repaired or replaced as appropriate.

Any indications of damage, failure to operate, or excessive wear should be evaluated to ensure that the safe operation of the cask is not impaired. Damage which impairs the ability of the cask to properly function should be repaired or replaced. This work may be performed on site, depending upon the skill level of the site work force, or at an approved vendor's facility. Repairs should be performed in accordance with the manufacturer's or cask designer's recommendations.

4.6 Cathodic Protection

The NUHOMS storage system 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 steel DSC requires no corrosion protection for any foreseeable event. The DSC support assembly components are Type A36 carbon steel and should require no protection against the expected environment. However, these components will be coated or hard plated for corrosion protection. The HSM heat shield is stainless steel and requires no corrosion protection.

The carbon steel used in the transfer cask is protected from corrosion by suitable coatings and stainless steel material is used for all surfaces exposed to pool water or liquid in the neutron shield cavity. Although the top head is not exposed to pool water, it is also protected by a suitable coating.

4.7 Fuel Handling Operation Systems

The fuel handling systems within the plant fuel building used to operate the NUHOMS design are site specific. The basis and engineering design required for these systems (including detailed structural specifications, installation layout and unit descriptions) will be included in an application for a site license. However, certain generic aspects of the handling system are described in the sections below.

4.7.1 Structural Specifications

Fabrication and construction specifications for fuel handling equipment will be site specific. However, some components of the handling systems, the DSC and the HSM, will be the same for all sites. The criteria for other components (cask, trailer, skid, and hydraulic ram) were discussed in Section 3. The specific components of the handling system to be covered by a quality assurance program will be site specific.

4.7.2 Installation Layout

The installation plans and sections of a NUHOMS HSM unit will be site specific as discussed in Section 4.1. However, illustrations of the DSC and the HSM are provided in Figures 1.3-1 and 1.3-1a.

Specific confinement features of the NUHOMS system were discussed in Section 3.3.2. General layout criteria (such as fence location, distance to site boundary, distance to personnel, etc.) which have radiological dose impact will be site specific. The NUHOMS system includes no ventilation, piping encasements, liners or protective coatings which affect containment of radioactive materials. The DSC containment pressure vessel provides the ultimate containment of radioactive material (see Section 3.3.2).

<u>4.7.2.1 Building Plans</u> Plans for a NUHOMS-24P ISFSI are provided in Figure 1.3-7. The building plans for individual plants are site specific and will be prepared in accordance with industry codes and standards.

<u>4.7.2.2 Building Sections</u> Generic sections for a NUHOMS-24P ISFSI are provided in Appendix E. The building plans for individual plants are site specific and will be prepared in accordance with industry codes and standards.

4.7.2.3 Confinement Features The confinement features of the NUHOMS system are described in Section 3.3.2.

4.7.3 Individual Unit Descriptions

Records of the SFAs which are candidates for dry storage will be reviewed to ensure conformance with the required fuel characteristics delineated in Section 3.1.1, and to ensure mechanical and structural integrity prior to placement in the DSC. If plant records indicate that the structural integrity of the fuel assemblies is adequate, inspection of the fuel assembly may not be required. The fuel assembly identification number must always be visible and recorded prior to placement into the DSC in order to maintain fuel accountability.

The SFAs will be placed into the DSC while the DSC is resting in the cask cavity in the spent fuel pool. The general layout of the spent fuel pool area is site specific and will be identified and discussed in site license applications. Each time a DSC containing spent fuel is moved or handled, the DSC will reside inside the cavity of a shipping or transfer cask.

Prior to placement of the DSC into the fuel pool, the top shield plug will be placed on the DSC as follows:

1. The shield plug is rigged and placed on the DSC to recheck proper fit-up with the DSC shell assembly. During the fit-up operation the shield plug lifting cables, attached to the cask lifting yoke, may require adjustment or tensioning to ensure that the plug fits squarely into the DSC shell assembly when suspended from its rigging. Once the rigging has been adjusted, the plug should fit true in subsequent lift and placement operations in the pool.

2. At many plants, the cask decontamination area and fuel pool are not in the same area/building. By moving the lead plug and DSC/transfer cask in one movement, the need for a number of crane movements is minimized and the fuel loading process is expedited. In addition, at some plants, because of space limitations in the areas surrounding the fuel pool, the cask lid or shield plug and cask yoke may be left in the pool during the fuel loading process.

Placement of the top shield plug into the DSC in the fuel pool following completion of the fuel loading will be performed in gradual movements to ensure that misalignment or damage to components does not occur. Because of the extremely slow speed of the crane hoist during shield plug placement, the water flow rate out of the cask is not excessive. A gap exists between the top shield plug and DSC shell which will permit unresticted flow of the displaced water out of the DSC. This radial gap ranges in size based on fabrication tolerances from a minimum of .06" to a maximum of .19". The vent and siphon ports may be left open for this operation, however, this is not considered to be necessary.

Connected with the generic operating procedures outlines in Section 5, the following description is provided to illustrate how the top shield plug installation operation can be accomplished. Detailed operating procedures will be developed on a site specific basis.

After the spent fuel assemblies have been placed into the DSC, install the shield plug as follows:

1. Position the shield plug over the DSC so that the shield plug aligns with the DSC. Tag lines may be used for precise alignment. Underwater CCTV cameras may be used to verify alignment.

2. Lower the crane hook until the shield plug is seated on the DSC and sufficient slack exists in the lifting cables to permit attachment of the yoke to the cask.

3. During placement of the shield plug, the following observations should be made to ascertain that the shield plug is seated properly:

   a. Observe the shield plug and DSC to see that the shield plug is seating into the DSC shell uniformly.

   b. Ascertain that all shield plug lifting cables slacken simultaneously.

   c. After shield plug placement, an underwater television camera, periscope, or other method may be used to inspect the height of the DSC shell above the lead plug to make sure that it is uniform and that the DSC drain and fill port is flush with the shield plug.

4. Corrective action which may be taken if the top shield plug is cocked:

   a. Slowly raise the shield plug from the DSC. When it clears the DSC shell, laterally move the crane trolley toward the low side of the cocked shield plug.

b. Re-seat the shield plug and inspect for correct seating. Repeat this operation if necessary, making slight lateral adjustments to the crane trolley.

c. If this fails, it may be necessary to adjust the shield plug rigging. This is accomplished by bringing the shield plug out of the pool and placing it on a stand or the floor. Adjust the length of the cables as necessary.

During the drying operation, the water will be removed by supplying compressed helium into the DSC through the vent port, thus water will be forced out through the siphon ports. The inlets to the siphon and vent ports have double-ended quick release connections (see Figure 4.7-1). The quick-release connections will remain in the closed position unless a connection is made. These quick-release valves insure that no accidental release of radioactive material will occur through the siphon or vent ports.

The basic system for both the siphon and vent ports used during the drying process will consist of a hose with a valve on each end of the hose and quick release connections on the intake and outlet openings of the valves. One of the valves will be connected to the siphon or vent port. This valve is designated as Valve No. 1 of the draining and drying system (see Figure 4.7-1). The valve at the other end of the hose will be designated as Valve No. 2 of the draining or drying or siphon tube system. All auxiliary equipment and radioactive waste systems will be connected to Valve No. 2. The type or method of connection will be dependent on the equipment used at the plant and will be discussed in the application for the utility's ISFSI license.

Valve No. 1 for both the vent and the siphon ports will remain open during most of the drying operation. The flow will be regulated with Valve No. 2. Valve No. 1 will act as a redundant valve and will only be used if Valve No. 2 should fail or a leak in the hose should develop. By using Valve No. 2 as the flow regulator as well as the connection points between the auxiliary system and the DSC, the amount of time which personnel will be exposed to the radiation at the DSC head will be minimized. Once the connection between Valve No. 1 and the siphon or vent tube is made, additional shielding may be placed over the lead shield plug to further reduce personnel radiation exposure.

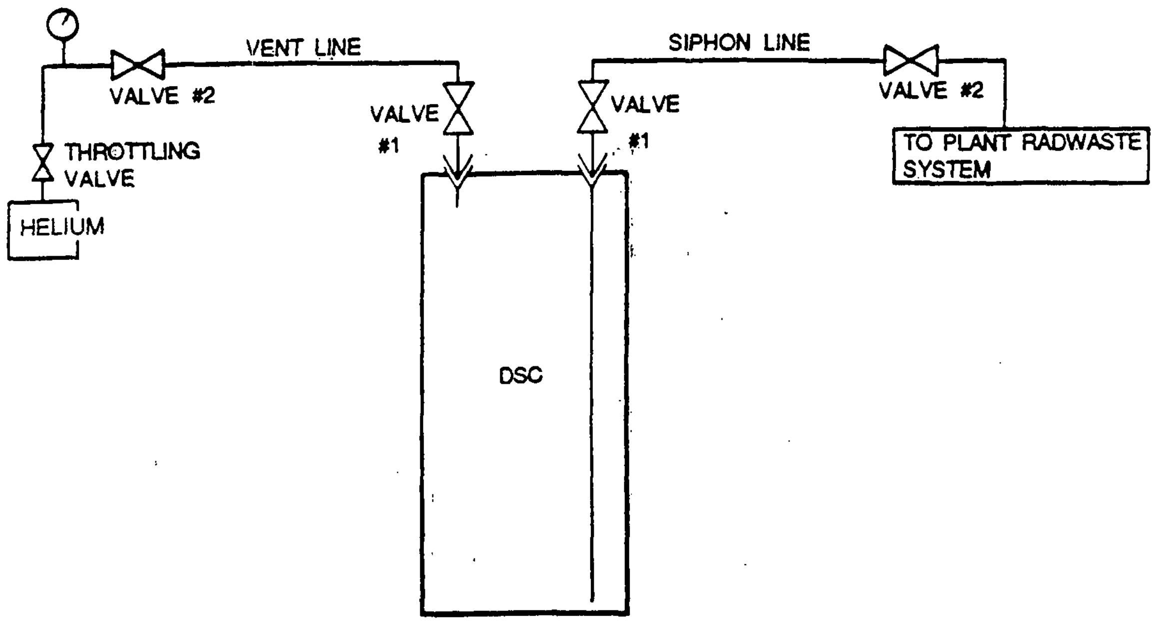
All discharges from the DSC cavity, whether gas or water, will be routed to the plant's existing radioactive waste system. Borated water may be routed back to the plant's fuel pool as appropriate.

Therefore, all radioactive materials or particles will be confined within a closed controlled system.

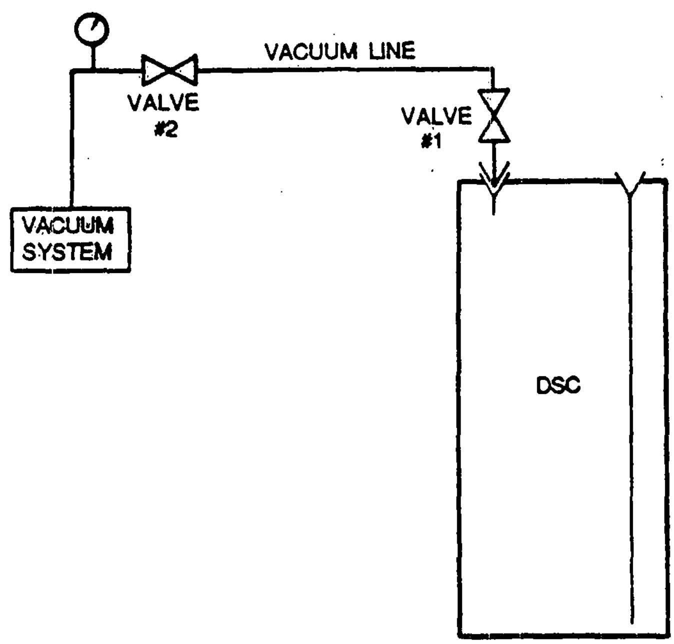
Once all the water has been forced out of the DSC cavity with compressed helium, the remaining moisture contained within the cavity will be removed with a vacuum drying system. The vacuum drying system will evacuate the DSC cavity and lower the moisture content to an acceptable level.

The intake valve on the vacuum drying system will be connected to Valve No. 2 of the siphon port piping system. A hose will be connected from the discharge outlet of the vacuum drying system to the plant's radioactive waste system. A particle filter will be located on the intake of the vacuum drying system. The filter will be used to capture any radioactive particles that may be contained within the gas and prevent contamination of the vacuum drying system. This vacuum drying system will be completely closed so that all radioactive material will be confined within a controlled system.

During the drying and final sealing operations of the DSC, the fillet seal weld on the top shield plug will confine any radioactive particles in the DSC cavity. The primary pressure boundary is formed by welding the top lead plug assembly to the DSC shell either manually or by using remote automatic welding equipment. The siphon port line will remain open and vented at atmosphere pressure to the plant's radwaste system during welding of the top shield plug. Fabricated plugs will be placed over the siphon and vent port openings and welded into place. Once the DSC has been dried and backfilled with helium, the top cover plate will be lowered onto the DSC. Using automatic or manual welding processes, the cover plate will be welded in place. These welded joints act as barriers for confining all radioactive material within the DSC throughout the service life of the DSC.



CONFIGURATION FOR DSC DRAINING



CONFIGURATION FOR DSC VACUUM DRYING

Figure 4.7-1

DSC DRAINING AND DRYING SYSTEM

4.8 References

4.1 Code of Federal Regulations, Title 10, Part 72, January 1, 1982.

4.2 Deleted.

4.3 Product Specification, E/M Lubricants, Inc., West Lafayette, Indiana.

4.4 Deleted.

4.5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 1, Appendices, 1980 Edition.

4.6 American Society for Testing and Materials, Annual Book of ASTM Standards, Section 4, Volume 04.02, 1983.

4.7 American Society for Testing and Materials, Annual Book of ASTM Standards, Section 1, Volume 01.04, 1983.

4.8 Deleted.

4.9 "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials", ANSI N14.6-1978, American National Standards Institute, Inc., New York, New York (1978).

5.0 OPERATION SYSTEM

This section presents the general operating procedures for the NUHOMS-24P system. These procedures include the preparation of the DSC for fuel loading, drying of the DSC cavity, transport to the HSM, DSC transfer into the HSM, monitoring operations, and DSC retrieval from the HSM. The NUHOMS-24P transfer equipment, and the plant systems and equipment used for these operations will be site specific. They are described here to point out how such operations would be performed. The operating procedures described in this section are based on the use of the NUHOMS system with the NUHOMS-24P on-site transfer cask.

The system operations and procedures described in this section are typical for a NUHOMS-24P installation. Detailed systems operating procedures will be developed on a site specific basis as part of preparing site license applications.

The procedures discussed in the following pages describes the operations for the NUHOMS-24P system shown in Figure 1.1-1. As can be seen from Figure 1.1-1, this system has the hydraulic ram at the rear of the transfer cask and is located on a support structure which is attached to the rear of the transfer cask.

5.1 Operation Description

The following sections outline the typical operating procedures for the NUHOMS-24P system. These generic NUHOMS procedures were developed to minimize the amount of time required to complete operations, to minimize personnel exposure, and assure that all operations required for DSC loading, closure, transfer, and storage are performed safely. Detailed site specific procedures, describing the specific NUHOMS transfer equipment selected, will to be developed for each site license application. The generic procedures are discussed here to point out how the NUHOMS system operations could be accomplished and are only intended to serve as a guide. Approval for their actual use is not being sought.

5.1.1 Narrative Description

The following steps describe the recommended generic operating procedures for the NUHOMS-24P system. A flowchart of these operations is provided in Section 5.1.2 (Figure 5.1-3).

5.1.1.1 Preparation of the Transfer Cask and DSC

1. Prior to the start of the operation, the candidate fuel assemblies to be placed in dry storage will be visually examined to insure that no gross defects exist and that the fuel assembly structure is intact. Verification of fuel integrity may also be accomplished using suitable plant records. The assemblies will also be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria described in Section 3.

2. Prior to the loading of fuel, the cask will be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Section 3.3.7.

3. Place the transfer cask in the vertical position in the cask decon area.

4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.

5. Remove the cask top cover plate and examine the cask cavity for any physical damage. Check the liquid level in the cask

neutron shield and the expansion tank system to ensure that it is in proper working order.

6. Examine the DSC for any physical damage which might have occurred since the receipt inspection was performed. If any damage is detected, the damage will be repaired. The DSC will be cleaned and any loose debris removed.

7. Using a crane, lower the DSC into the cask cavity by the internal lifting lugs and rotate the DSC to match the cask and DSC alignment marks.

8. Fill the cask-DSC annulus with clean, demineralized water. Fill the DSC cavity with borated water from the fuel pool or an equivalent source of borated water.

9. Place the inflatable seal into the upper cask liner recess and seal the cask-DSC annulus by pressurizing the seal with compressed air. Apply a backup seal of suitable tape as necessary.

10. Place the top shield plug assembly onto the DSC. Examine the top shield plug to ensure a proper fit.

11. Position the cask lifting yoke and engage the cask lifting trunnions and the lid lifting cables to the DSC top shield plug assembly. Adjust the lid lifting cables as necessary to obtain even table tension.

12. Visually inspect the yoke lifting arms to insure that the lift arms are properly positioned and engaged on the cask lifting trunnions.

13. Move the scaffolding away from the cask as necessary.

14. Lift the cask just far enough to allow the weight of the cask to be distributed onto the yoke lifting arms. Inspect the lifting arms to insure that they are properly positioned on the cask trunnions.

15. Secure a sheet of suitable material to the bottom of the transfer cask to minimize the potential for ground-in contamination.

16. Prior to the cask being lifted into the fuel pool, the water level in the pool should be adjusted as necessary to accommodate the cask DSC volume. If the borated water placed in the DSC cavity was obtained from the fuel pool, a level adjustment may not be necessary.

5.1.1.2 Fuel Loading

17. Lift the cask with DSC into the fuel pool.

18. Lower the cask into the fuel pool until the bottom of the cask is at the height of the fuel pool railing. As the cask is lowered into the pool, spray the exterior surface of the cask with demineralized water.

19. Lower the cask into the location in the fuel pool designated as the cask loading area.

20. Disengage the lifting yoke from the cask lifting trunnions and move the yoke and the top shield plug clear of the cask. Spray the lifting yoke and top shield plug assembly with clean demineralized water if it is raised out of the fuel pool.

21. Move a candidate fuel assembly from a fuel rack to the visual inspection area, if inspection is necessary.

22. Record the fuel assembly identification number from the fuel assembly and check this identification number against the storage documents which indicate which fuel assemblies are suitable for dry storage.

23. Position the fuel assembly for insertion a DSC guide sleeve (repeat steps 21 through 23 once for each SFA loaded into the DSC).

24. After all the SFAs have been placed into the DSC, position lifting yoke and the top shield plug assembly and lower the shield plug onto the DSC.

25. Visually verify that the top shield plug assembly is properly seated onto the DSC.

26. Engage the lifting yoke to the cask trunnions.

27. Raise the transfer cask to the pool surface. Prior to raising the top of the cask above the water surface, stop vertical movement.

28. Inspect the top shield plug assembly to insure that it is properly seated onto the DSC. If not, lower the cask and reposition the top shield plug assembly. Repeat steps 27 and 28 as necessary.

29. Continue to raise the cask from the pool and spray the exposed portion of the cask with demineralized water.

30. Drain any excess water from the top of the DSC shield plug assembly back to the fuel pool.

31. Check radiation levels at the center and perimeter of the top shield plug assembly and around the exposed surface of the cask.

32. Lift the cask from the fuel pool. As the cask is raised from the pool, continue to spray the cask with demineralized water.

33. Move the DSC cask to the cask decon area.

5.1.1.3 Cask/DSC Drying Process

34. Check radiation levels along the surface of the cask. The cask exterior surface should be decontaminated as necessary. Temporary shielding may be installed as necessary to minimize personnel exposure.

35. Place scaffolding around the cask so that any point on the surface of the cask is easily accessible to personnel.

36. Disengage the lid lifting cables from the top shield plug and remove the eyebolts. Disengage the lifting yoke from the trunnions.

37. Decontaminate the top shield plug surface and the exposed DSC shell surface.

38. Connect the cask drain line, open the cask cavity auxiliary port and allow water from the cask-DSC annulus to drain out so that the water level is approximately ten inches below the top edge of the DSC shell.

39. Dry the top lead plug surface and exposed interior of the DSC shell above the top lead plug.

40. Check radiation levels along surface of the top lead plug. Temporary shielding may be installed as necessary to minimize personnel exposure.

41. Attach a self priming pump to the DSC siphon port, open the vent port and drain approximately 60 gallons from the DSC to the fuel pool. This will lower the water level about four inches below the top shield plug.

42. Remove the pump.

43. Drain and dry any residual water from top shield plug and exposed surface of DSC shell.

44. Open the valves on the DSC vent port, allowing pressure to remain atmospheric. Seal weld the top shield plug to the DSC shell.

45. Perform dye penetrant weld examination of the seal weld pass and the cover pass.

46. Assemble the DSC draining and drying system as shown in Figure 4.7-1.

47. Connect a pressure gauge to the DSC vent port.

48. Engage the compressed helim supply and open the valve on the vent port and allow compressed helim to force the water from the DSC.

49. Once the water stops flowing from the DSC, close on the DSC siphon port.

50. Seal weld the prefabricated plug over the siphon port and perform a dye penetrant examination of the root and cover passes.

51. Connect the hose from the vent port to the intake of the vacuum pump. A hose should be connected from the discharge side of the vacuum system to the site radioactive waste system. Connect the vacuum system to a helium source.

52. Open valve, start the vacuum system and draw a vacuum of 3 torr on the DSC cavity. The cavity pressure should be reduced in steps of 100 torr, 50 torr, 25 torr, 15 torr, 10 torr, 5 torr, and 3 torr. After pumping down to each level, the pump is valved off and stopped and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is reactivated and the pressure reduced to the next step. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase.

53. Open the valve to the vent port and allow the helium to flow into the DSC cavity.

54. Pressurize the DSC with helium.

55. Leak test the top shield plug weld for helium leaks.

56. If a leak is found, repair the weld.

57. If no leak is detected, release the pressure.

58. Evacuate the helium from the DSC cavity to 3 torr using the vacuum pump.

59. Open the valve on the vent port and allow a predetermined quantity of helium to flow into the DSC cavity to pressurize the DSC.

60. Close the valves on the helium source.

5.1.1.4 DSC Sealing Operations

61. Remove the draining and drying system from the top of the DSC. Seal weld the prefabricated plug over the vent port and perform a dye penetrant weld examination on the root and cover passes.

62. Place the top cover plate onto the DSC. Remove the seal from the cask/DSC annulus if necessary.

63. Seal weld the top cover plate onto the DSC shell and perform the necessary nondestructive examination on the weld.

64. Remove the seal from cask/DSC annulus if still in place. Using the crane, rig the cask top cover plate and lower the cover plate onto the transfer cask.

65. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern. Drain the remaining water from the cask/DSC annulus.

5.1.1.5 Transport of Cask to Transfer Trailer Skid and HSM

66. Reattach the lifting yoke to the crane hook, as necessary.

67. Engage the lifting yoke and move the scaffolding away from the cask, if necessary. Using the crane, lift the cask to the cask support skid on the transport trailer.

68. The transport trailer should be positioned so that cask support skid is accessible to the crane with the trailer supported on the vertical jacks.

69. Position the cask lower support trunnions onto the cask support skid pillow block supports.

70. Move the crane forward while simultaneously lowering the cask until the cask is just above the support skid upper trunnion supports.

71. Inspect the positioning of the cask to insure that the cask and trunnion supports are properly positioned.

72. Lower the cask onto the trunnion supports until the weight of the cask is distributed to the trunnion supports.

73. Inspect the trunnions to insure that they are properly seated onto the trunnion supports and install the top portion of the pillow block supports.

74. Install the cask tie-downs to the support skid. Remove the bottom ram access cover plate from the cask and install the temporary shield plug.

5.1.1.6 Loading of DSC into the HSM

75. Using a suitable prime mover, transport the cask to the HSM along the designated transfer route.

76. Prior to transferring a DSC to an HSM, remove the HSM door using a porta-crane and inspect the cavity of the HSM to insure that no debris or animals are within the HSM. Doors on adjacent HSMs should be in place.

77. Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage. Replace the screens if necessary. Leave the door on the HSM open.

78. Position the transport trailer to within a few feet of the HSM.

79. Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

80. Using a porta-crane, remove the cask top cover plate.

81. Back the cask to within a few inches of the HSM.

82. Using the optical alignment system and the targets on the cask and HSM, adjust the position of the cask until the cask is properly positioned with respect to the HSM. The position of the cask may be adjusted vertically using jacks

along the bottom of the trailer. Horizontal positioning can be done by using the skid positioning system.

83. Using the skid positioning system, fully insert the cask into the HSM access opening.

84. Secure the cask trunnions to the front wall of the HSM using the cask restraint system.

85. Reinspect the position of the transfer cask and secure the cask support skid to the trailer deck.

86. Setup and align the ram system with the cask using the optical alignment system. Remove the center portion the ram access penetration temporary shield plug from the cask. Extend the ram through the bottom cask opening into the DSC grapple ring.

87. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.

88. Recheck all supports and ready all systems for DSC transfer.

89. Activate the hydraulic ram to initiate DSC transfer. If the ram fails to extend when the load on the hydraulic system exceeds twenty five percent of the maximum DSC load, disengage the hydraulic ram. Check the orientation of the cask with respect to the HSM and reorient if necessary. Repeat this step until the DSC is completely within the HSM.

90. When the DSC reaches the support rail stops at the back of the HSM, disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

91. Retract and disassemble the hydraulic ram system. Remove the cask restraint system from the HSM.

93. Pull trailer and skid away from the HSM a few inches.

94. Lower the HSM door to within a few feet of the closed position and install the DSC seismic restraint.

95. Lower the door of the HSM to the closed position and tack weld it in place.

96. Replace the transfer cask top and bottom cover plates.

97. Tow the trailer and cask to the designated equipment storage area.

98. Close and lock the ISFSI access gate and activate security measures.

5.1.1.7 Monitoring Operations

99. Every 24 hours, personnel will inspect the lock on the gate to insure that no entry into the area has been attempted. Any signs of tampering or entry will be investigated immediately. Exact security monitoring procedures will be site specific.

100. On a 24 hour basis, site personnel will visually inspect the HSM air inlets and outlets, as appropriate for site conditions, to insure that no debris is obstructing the HSM vents. If damage has occurred to the air inlet screen, the screen will be replaced.

5.1.1.8 Unloading the DSC From the HSM

101. Place the cask onto the transport trailer and support skid and tow the trailer to the HSM.

102. Back the trailer to within a few feet of the HSM, remove the cask top cover plate, and install the temporary shield plug over the ram access penetration.

103. Remove the tack welds from the door of the HSM raise the door a few feet and remove the DSC seismic restraint. Remove the HSM door using a porta-crane.

104. Using visual alignment marks, roughly align the cask with respect to the HSM and position the skid until the cask is inserted into the HSM.

105. Using the optical alignment system, align the cask with respect to the HSM. Install the cask skid tie-down assembly.

106. Install and align the hydraulic ram system with the cask.

107. Extend the ram through the cask into the HSM until it is inserted in the DSC grapple ring.

108. Activate the arms on the ram grapple mechanism with the DSC grapple ring.

109. Retract ram and pull the DSC into the cask.

110. Retract the ram grapple arms.

111. Retract the ram out of the cask.

112. Replace the ram access penetration temporary shield plug.

113. Remove the cask skid tie-down assembly.

114. Slowly pull the trailer forward a few feet away from the HSM.

115. Replace the door to the HSM.

116. Place the cask cover plate onto the cask and bolt it in place.

5.1.1.9 Removal of Fuel from the DSC When the DSC has been removed from the HSM, there are several potential options for disposition of the fuel. It may be possible to ship the intact DSC to a reprocessing facility, monitored retrievable storage facility or waste repository in a shipping cask licensed under 10CFR71.

If it becomes necessary to remove fuel from the DSC prior to off-site shipment, there are two basic options available at the ISFSI or reactor site. The fuel assemblies could be removed and reloaded into a shipping cask using dry transfer techniques or if the applicant so desires, the loading sequence could be reversed and the plant's spent fuel pool utilized. Either approach would be site specific, but to complete the sequence of operations begun in this section, procedures for unloading of the DSC in a fuel pool are presented. In either case, wet or dry procedures will be essentially identical through the DSC weld removal or up to Step 141 (beginning of preparation to place the cask in the fuel pool).

117. The cask may now be transported to the cask handling area inside the plant's fuel building.

118. Position and ready the trailer for access by the crane and install the cask bottom cover plate.

119. Attach the lifting yoke to the crane hook.

120. Lower the lifting yoke onto the trunnions of the cask.

121. Visually inspect the yoke lifting arms to insure that they are properly aligned and engaged onto the cask trunnions.

122. Lift the cask approximately one inch off the trunnion supports. Visually inspect the yoke lifting arms to insure that the lifting arms are properly positioned on the trunnions.

123. Move the crane forward in a horizontal motion while simultaneously raising the crane hook vertically and lift the cask off the trailer. Move the cask to the cask decon area.

124. Lower the cask into the cask decon area in the vertical position.

125. Wash the cask to remove any dirt which may have accumulated on the cask during the DSC loading and transfer operations.

126. Place scaffolding around the cask so that any point on the surface of the cask is easily accessible to handling personnel.

127. Unbolt the cask lid and remove the bolts.

128. Connect the lid lifting cables to the the cask top cover plate and lift the cover plate from the cask. Set the cask cover plate aside and disconnect the lid lifting cables.

129. Install temporary shielding to reduce personnel exposure as required. Fill the cask/DSC annulus with clean demineralized water and seal the annulus.

The process of DSC unloading is similar to that used for DSC loading. DSC opening operations described below will be carefully controlled in accordance with plant procedures. This operation will be performed under the site's standard health physics guidelines for welding, grinding, and handling of potentially highly contaminated equipment. These will include the use of prudent housekeeping measures and monitoring of airborn particulates. Procedures may require personnel to perform the work using respirators or supplied air.

If the work is performed outside the fuel building, a tent may be constructed over the work area which may be kept under a negative pressure to control airborne particulates. Any radioactive gas release will be Kr-85, which is not readily captured. Whether the krypton is vented through the plant stack or allowed to be released directly depends on the plant operating requirements.

Following opening of the DSC systems and vent port the cask and DSC are filled with water prior to the placement in the fuel pool to prevent a sudden inrush of pool water. Cask insertion into the pool is slow. Fuel unloading procedures will be governed by the plant operating license under 10CFR50. The generic procedures for these operations are as follows:

130. Locate the DSC drain and fill port using the indications on the top cover plate. Place a portable drill press on the top of the DSC. Position the drill with the siphon tube quick connect.

131. Place an exhaust hood or tent over the DSC, if necessary. The exhaust should be filtered or routed to the site radwaste system.

132. Drill a hole through the DSC top cover plate to the siphon tube quick connect.

133. Drill a second hole through the top cover plate to the vent tube quick connect.

134. Fill the DSC with borated water from the fuel pool through the system port with the vent port open and routed to the plant's off-gas system.

135. Place welding blankets around the cask and scaffolding.

136. Using an air arc-gouging or mechanical grinding system, remove the weld seal from the top cover plate and DSC shell. A fire watch should be placed on the scaffolding with the welder, as appropriate. The exhaust system should be operating at all times.

137. The material or waste from the carbon cutting or grinding process should be treated and handled in accordance with the plant's low level waste procedures unless determined otherwise.

138. Remove the top of the tent, if necessary.

139. Remove the exhaust hood, if necessary.

140. Remove the DSC top cover plate.

141. Reinstall tent and temporary shielding, as required. Remove the seal weld from the top shield plug to the DSC shell in the same manner as the top cover plate.

142. Clean the cask surface of dirt and any debris which may be on the cask surface as a result of the weld removal operation. Any other procedures which are required for the operation of the cask should take place at this point as necessary.

143. Engage the yoke onto the trunnions, install eyebolts into the top shield plug assembly and connect the lid lifting cables to the eyebolts.

144. Visually inspect the lifting arms or the yoke to insure that they are properly positioned on the trunnions.

145. The cask should be lifted just far enough to allow the weight of the transfer cask to be distributed onto the yoke lifting arms. Inspect the lifting arms to insure that they are properly positioned on the trunnions.

146. Install suitable protective material onto the bottom of the transfer cask to minimize cask contamination. Move the cask to the fuel pool.

147. Prior to lowering the cask into the pool, adjust the pool water level, if necessary to accommodate the volume of water which will be displaced by the cask during the operation.

148. Lower the cask into the water.

149. Move the cask over to the cask loading area in the pool.

150. Lower the cask onto the pool floor. As the cask is being lowered, the exterior surface of the cask should be sprayed with clean demineralized water.

151. Disengage the lifting yoke from the cask and lift the top shield plug assembly from the DSC.

152. Remove the fuel from the DSC and place the fuel into the spent fuel racks.

153. Lower the top shield plug assembly onto the DSC.

154. Visually verify that the top shield plug assembly is properly positioned onto the DSC.

155. Engage the lifting yoke onto the cask trunnions.

156. Visually verify that the yoke lifting arms are properly engaged with the cask trunnions.

157. Lift the cask off the pool floor by a small amount and verify that the lifting arms are properly attached to the trunnions.

158. Lift the cask to the pool surface. Prior to raising the top of the cask to the water surface, stop vertical movement and inspect the top shield plug assembly to ensure that it is properly positioned.

159. Spray the exposed portion of the cask with demineralized water.

160. Visually inspect the top shield plug assembly of the DSC to insure that it is properly seated onto the cask. If the top shield plug assembly is not properly seated, lower the cask back to the pool floor and reposition the assembly.

161. Drain any excess water from the top of the top shield plug assembly into the fuel pool.

162. Lift the cask from the pool. As the cask is rising out of the pool, spray the cask with demineralized water.

163. Move the cask to the cask decon area.

164. Check radiation levels along the surface of the cask. The cask exterior surface should be decontaminated if necessary.

165. Place scaffolding around the cask so that any point along the surface of the cask is easily accessible to personnel.

166. Assemble the DSC draining and drying system as shown in Figure 4.7-1.

167. Connect the DSC draining and drying system to the vent port quick release connection with the system open to atmosphere. Also connect the DSC draining and drying system to the siphon tube port and connect the other end of the system to the self priming pump. The pump discharge should be routed to the plant radwaste system.

168. Open the valves on the vent port and siphon port of the system.

169. Activate the self priming pump.

170. Once the water stops flowing out of the DSC, deactivate the pump.

171. Close the valves on the system.

172. Disconnect the system from the vent and siphon ports.

173 The top shield plug assembly may be welded into place if desired.

174. Decontaminate the DSC, as necessary, and handle in accordance with low-level waste procedures.

5.1.2 Process Flow Diagram

Process flow diagrams for the handling operation are presented in Figures 5.1-3 and 5.1-4. The location of the various operations may vary with specific site requirements. Therefore, the flow diagram may need to be modified to meet the physical capabilities of the utility and should be submitted by the utility in application for an ISFSI license.

5.1.3 Identification of Subjects for Safety Analysis

5.1.3.1 Criticality Control Subcriticality is assured as discussed in Section 3 of this report.

5.1.3.2 Chemical Safety There are no chemicals used in the NUHOMS system that require special precautions.

5.1.3.3 Operation Shutdown Modes NUHOMS is a totally passive system and, therefore, this section is not applicable.

5.1.3.4 Instrumentation Table 5.1-1 shows the typical instruments which might be used to measure conditions or control the operations during the DSC handling operations. The instruments are standard industry components and the exact instrumentation requirements will be selected on a site specific basis.

5.1.3.5 Maintenance Techniques NUHOMS is a totally passive system and therefore will not require maintenance. However, to insure that the ventilation airflow is not interrupted, the HSM will be periodically inspected to insure that no debris is in the airflow inlet or outlet.

Table 5.1-1

INSTRUMENTATION USED DURING THE DSC HANDLING OPERATIONS

	Instruments	Function
1)	Gross Gamma/Beta/Neutron Detectors	Measure doses at top of DSC top shield plug
2)	Pressure and Vacuum Gauges	Measure helium, air, water and vacuum pressures inside DSC
3)	Hydraulic Pressure Gauges and Ram Cut-off Switch	Measure and limit hydraulic ram power to less than twenty five percent of loaded DSC weight
4)	Optical Alignment System	Align cask, ram and HSM

FIGURES 5.1-1 AND 5.1-2 HAVE BEEN DELETED

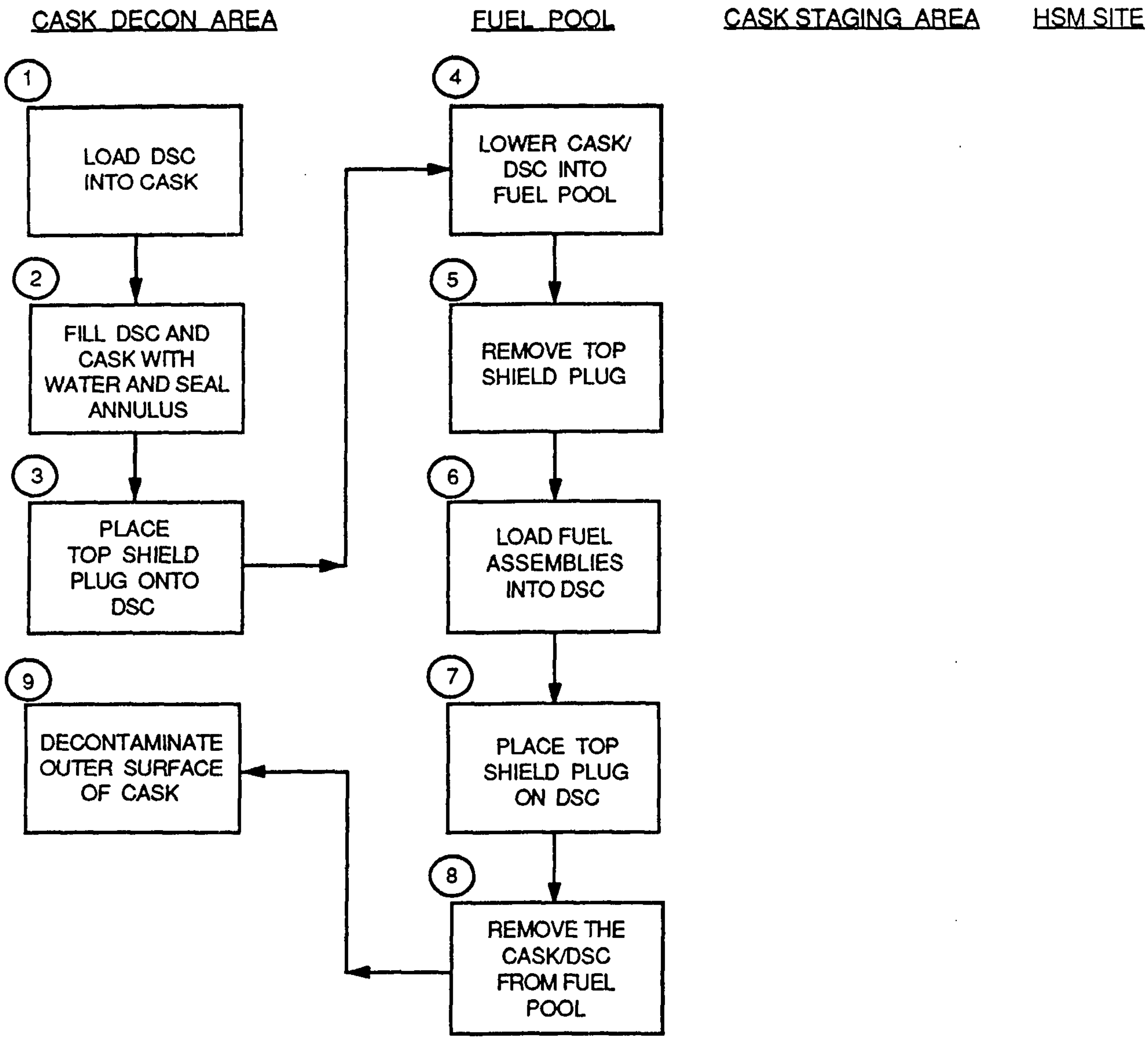


Figure 5.1-3

NUHOMS SYSTEM LOADING OPERATIONS FLOW CHART

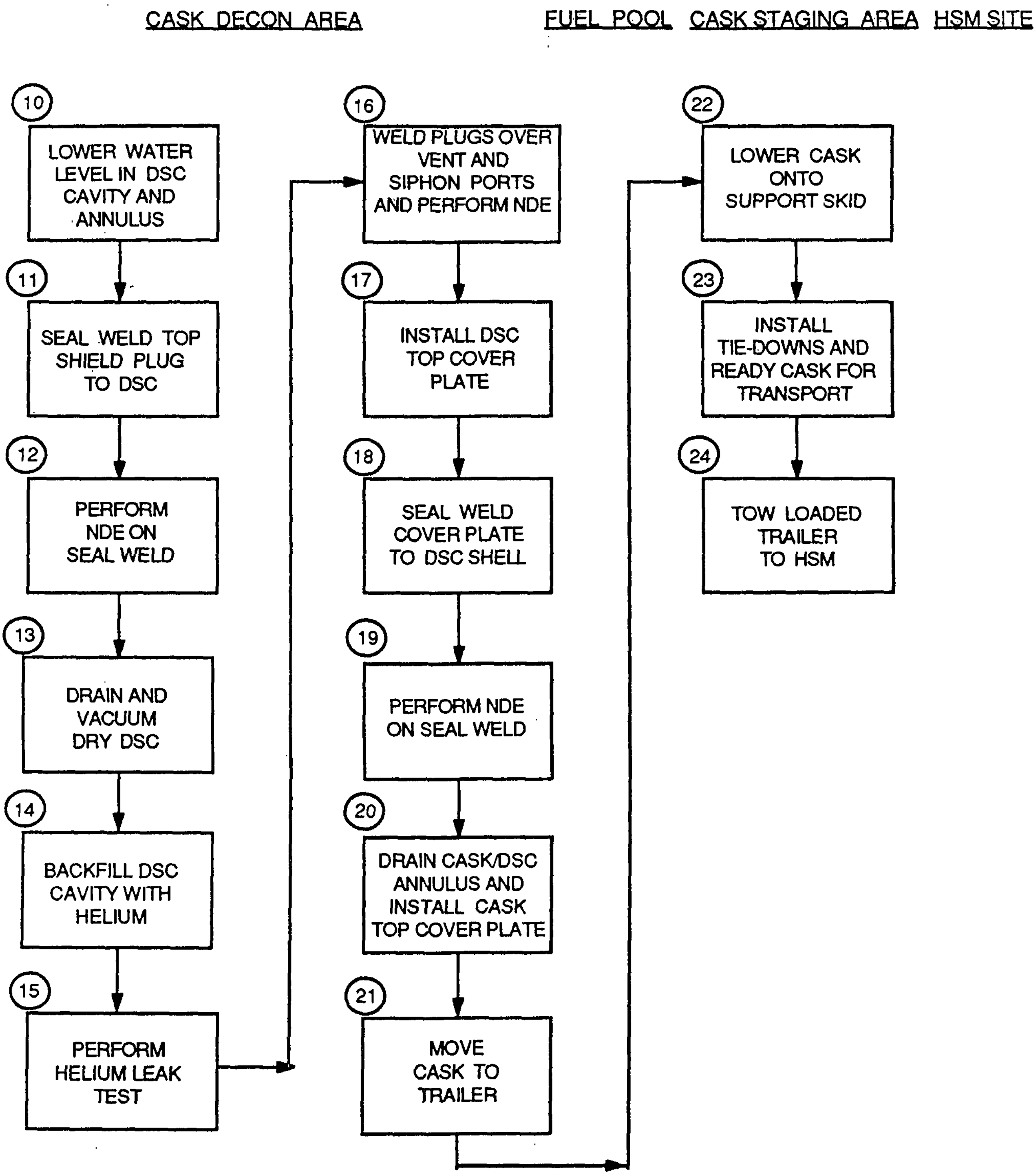


Figure 5.1-3

NUHOMS SYSTEM LOADING OPERATIONS FLOW CHART
(Continued)

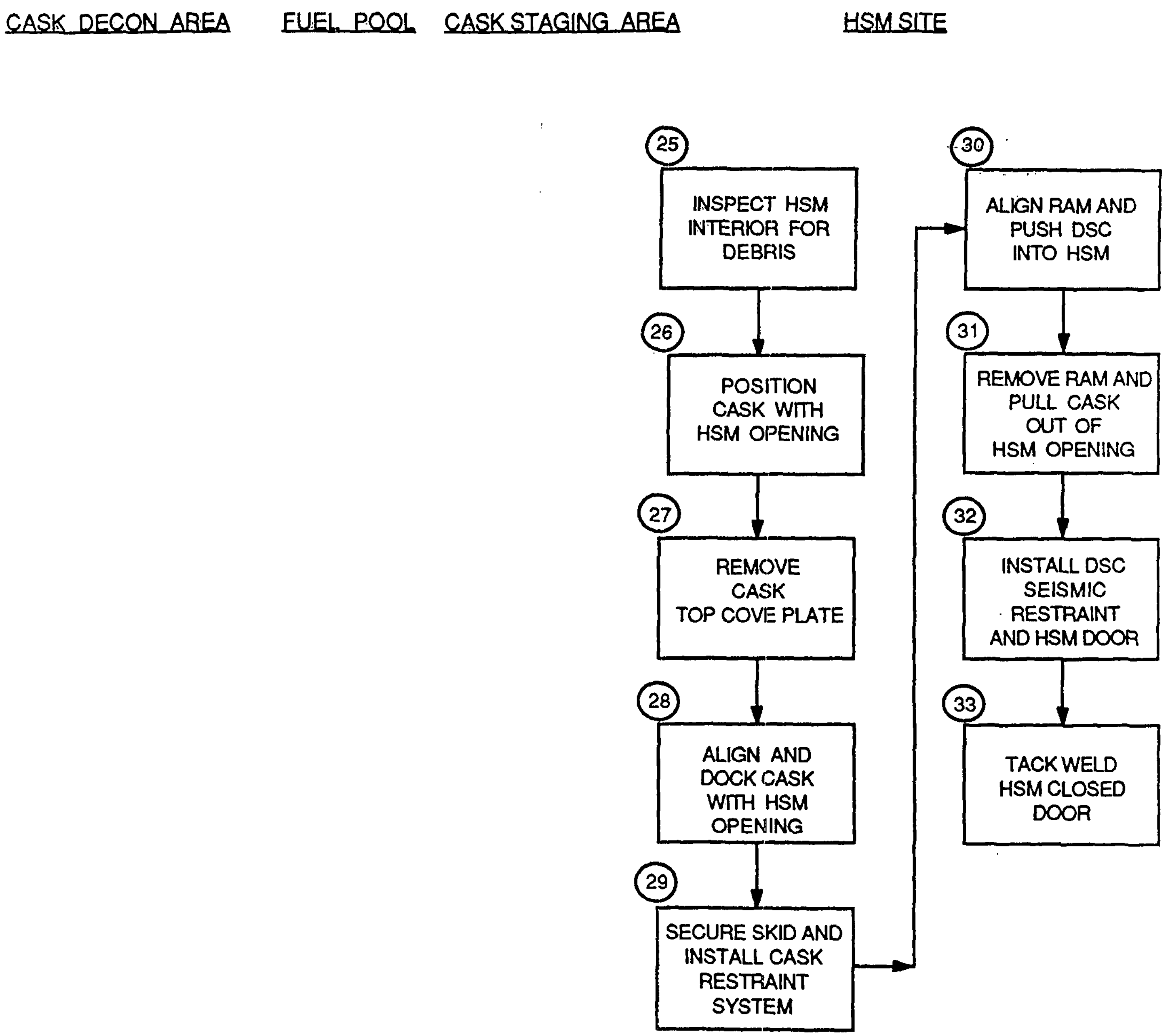


Figure 5.1-3

NUHOMS SYSTEM LOADING OPERATIONS FLOW CHART
(Concluded)

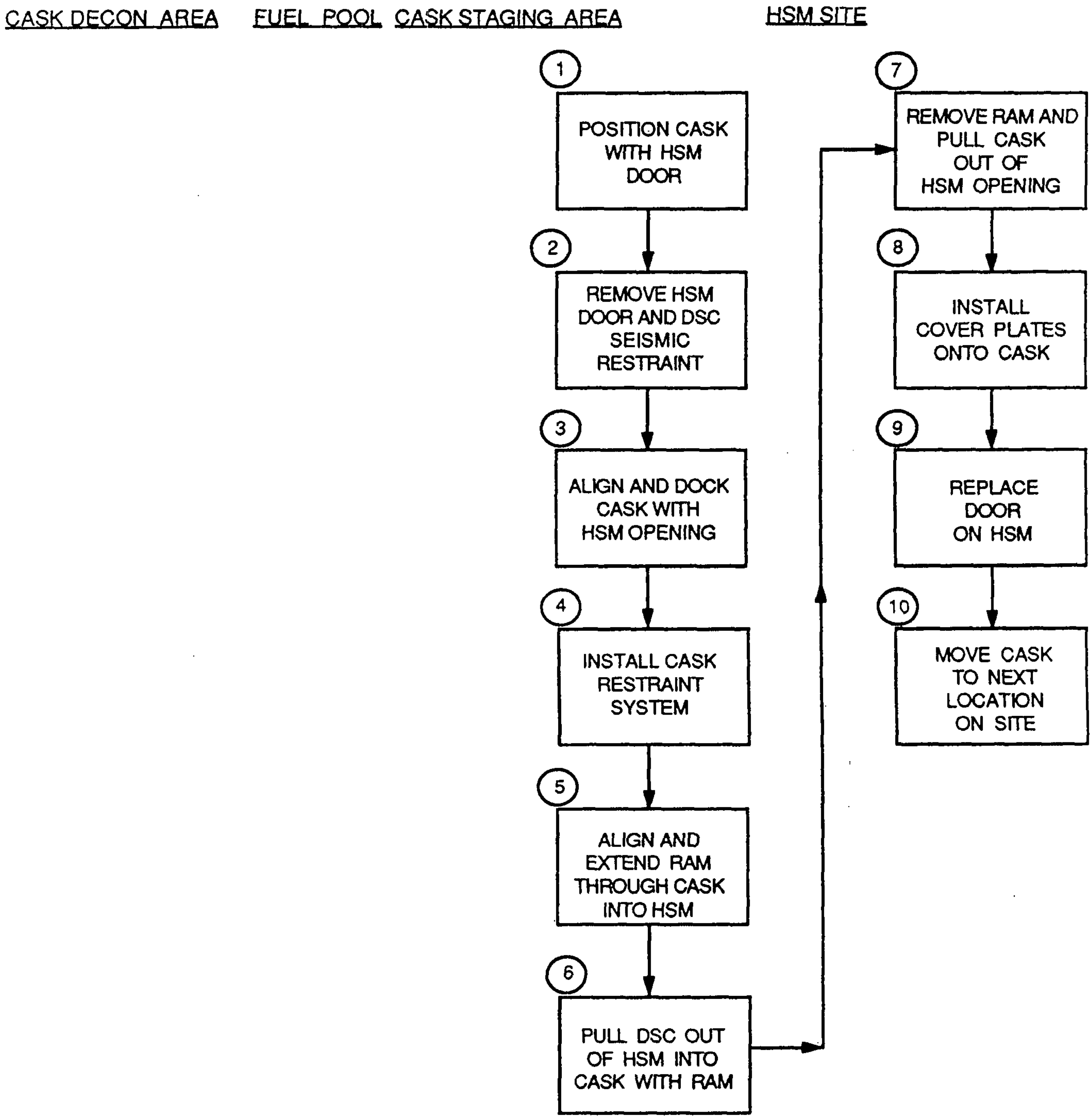


Figure 5.1-4

NUHOMS SYSTEM RETRIEVAL OPERATIONS FLOW CHART

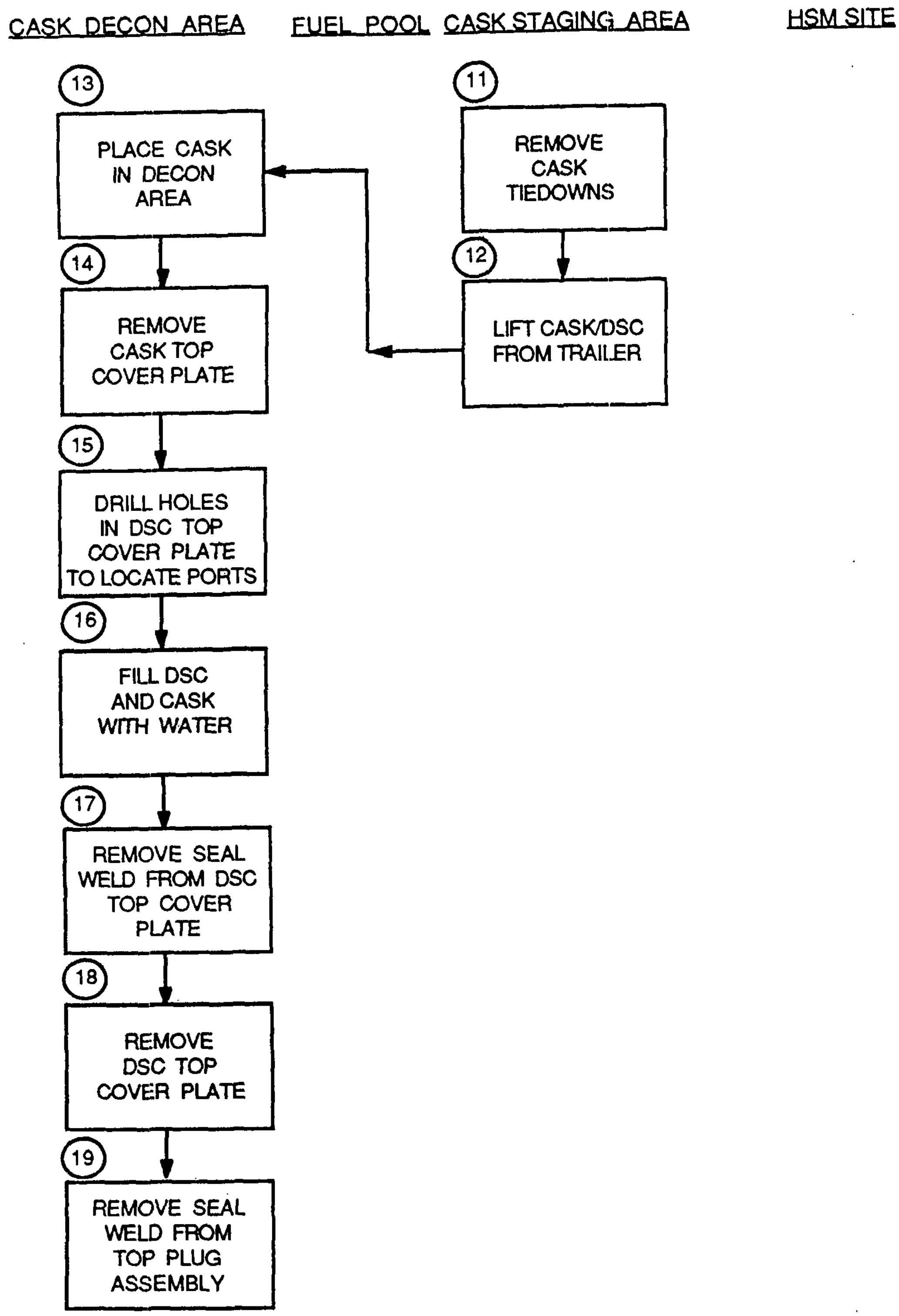


Figure 5.1-4

NUHOMS SYSTEM RETRIEVAL OPERATIONS FLOW CHART
(Continued)

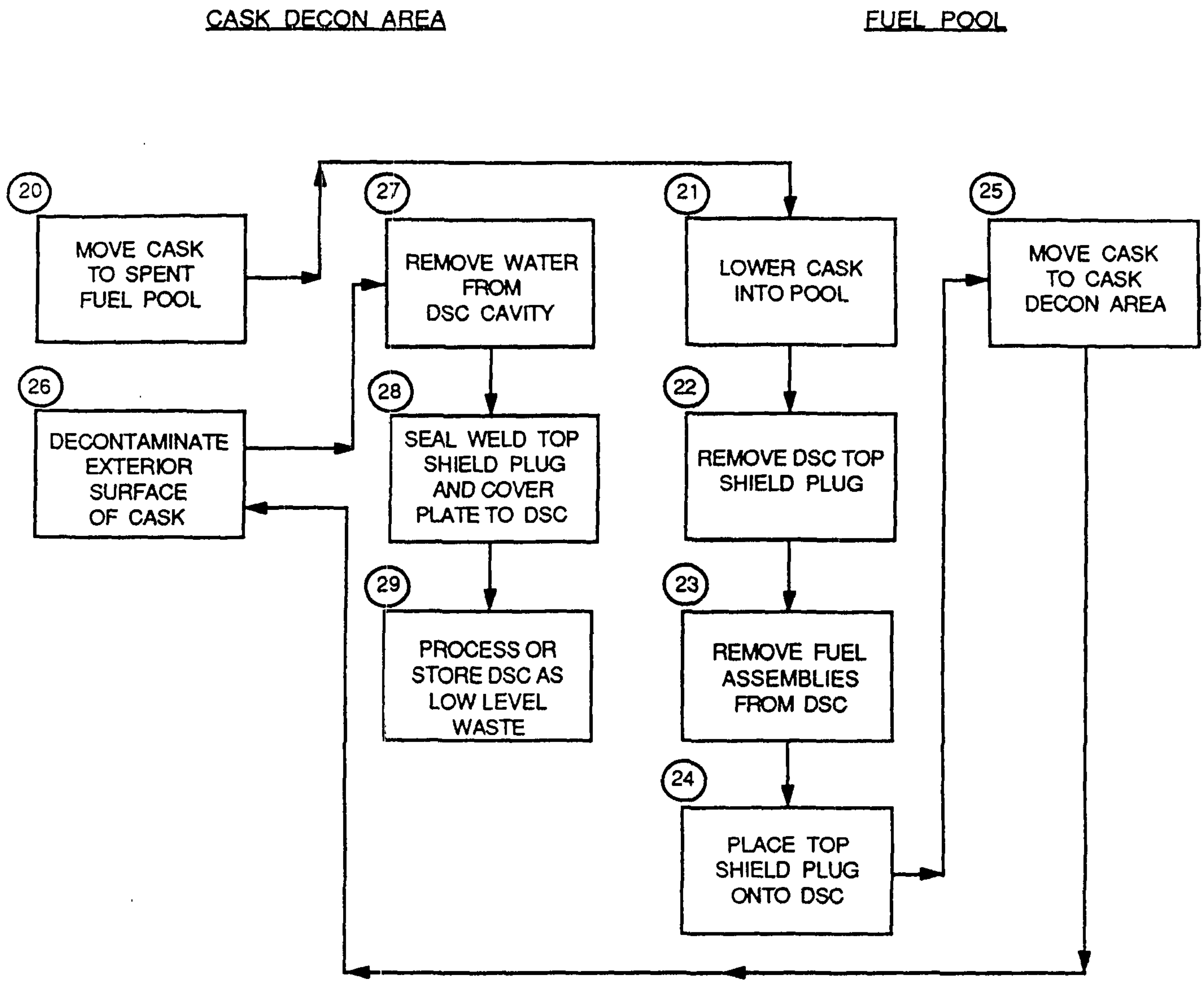


Figure 5.1-4

NUHOMS SYSTEM RETRIEVAL OPERATIONS FLOW CHART
(Concluded)

5.2 Fuel Handling Systems

5.2.1 Spent Fuel Handling and Transfer

NUHOMS is a modular storage system which provides for the dry storage of spent fuel in a horizontal position. NUHOMS is a system designed to be installed at any reactor site or other licensed independent sites. It will use the existing plant systems for handling spent fuel and casks. This section will describe spent fuel handling systems that are unique to NUHOMS and used during the DSC transfer, loading, and retrieving operations. However, these systems are site specific. A conceptual transfer system is described in this section to illustrate the hardware and procedures required for operation of the NUHOMS system.

5.2.1.1 Functional Description Figures 5.1-3 and 5.1-4 present the flow diagrams for the DSC transfer, loading and retrieval operations.

Transfer System The transfer system is composed of a suitable shipping or on-site transfer cask, positioning skid, trailer, hydraulic ram, alignment system, and miscellaneous auxiliary systems as described in Section 1.3.1 and shown in Figures 1.3-2a through 1.3-5.

NUHOMS-24P Transfer Cask The NUHOMS-24P transfer cask is used to transfer a loaded twenty four element DSC to and from the HSM. The cask provides biological shielding during the transfer, loading, and retrieval operations. During transfer of the DSC to the HSM, the top six inches of the cask is seated within the HSM access opening sleeve. A description of the NUHOMS-24P transfer cask design criteria and capabilities are provided in Sections 3, 4, and 8 of this report.

Cask Support Skid The purpose of the skid is to transport the cask, in a horizontal position, to the HSM and maintain the cask alignment during the loading and retrieval operations. The skid will be mounted on Lubrite bearing plates or an engineered equivalent and temporarily fastened to the transport trailer during transport. These rollers will allow for the skid to be moved in the longitudinal and transverse directions with respect to the trailer, to allow the DSC to be precisely aligned with the centerline of the DSC support assembly. The design of the skid will vary with the type of cask used during the operation. Section 3.1.2.2 establishes the criteria for design of the skid.

Transport Trailer The function of the transport trailer is two-fold, 1) to transport the loaded cask in the horizontal position to the HSM and 2) to approximately align the cask with the HSM opening. The trailer shown in Figure 1.3-3 is a typical trailer capable of handling a 120 ton load.

Optical Alignment System Once the loaded trailer has been backed up to the HSM front access the cask will be aligned with the HSM. The alignment system consists of a precision transit level and optical targets on the cask and HSM or other proven methods. Once the cask is aligned with the HSM, the jack system and cask restraining system will insure that the alignment is maintained throughout the transfer or retrieval operation.

Jack Support System The tires on the trailer are pneumatic so as the DSC is being transferred into or out of the HSM, the transfer of the load may cause the alignment to be altered or cause the DSC to bind in the cask or HSM. To ensure that the alignment is maintained throughout the transfer or retrieval operation, jacks will be placed at four locations on the trailer. The actual design of the jack support system will be dependent upon the skid and trailer. The design criteria for the jack support system are established in Section 3.1.2 of this report.

Cask Restraining System During the transfer or retrieval operations, the transfer of the DSC could cause the cask to move in its axial direction. This motion could cause the alignment to be altered or shielding by the HSM and cask to be jeopardized. To insure that the cask does not move in the axial direction, a cask restraining system will be attached from the HSM to the cask. Figure 4.2-6 shows the conceptual design for this system.

Ram and Grappling Apparatus The ram is a hydraulic cylinder which extends from the back of the cask through the length of the cask (Figure 1.3-5). The grappling apparatus is mounted on the front of the piston. Figure 1.3-5 shows a conceptual drawing of the grappling apparatus. The hydraulics for the grappling apparatus are then activated and the arms move out between the end of the DSC and grapple ring. Once the arms are engaged, the ram is extended, pushing the DSC out of the cask and into the HSM. For retrieval of the cask the process is reversed. The DSC slides along the cask inner liner and onto the T-section support rails inside the HSM.

DSC Support Rails During the transfer operation, the DSC will slide out of the cask and onto the T-section support rails which are inside the HSM. The T-section support rails serve as both the sliding surfaces during the transfer operation as well as supports during the storage of the DSC. The surface of the T-section support rails which comes in contact with the surface of the DSC will be coated with the solid film lubricant Everlube 823 or equivalent.

5.2.1.2 Safety Features Except for the transfer of the DSC from the cask to the HSM, the loaded DSC will always be seated inside the cask cavity. The safety features used in handling the cask will be unique to the type of cask used during the loading and retrieval operation and will be included in the site license application.

To ensure that the minimum amount of force is applied to the DSC during the transfer operation, the inside surface of the cask and T-section support rails which are in contact with the DSC will be coated with the solid film lubricant Everlube 823 or equivalent. A low coefficient of friction will minimize the amount of force applied to the DSC, thus minimizing the possibility of damage to the DSC.

If the motion of the DSC is impeded during the transfer operation and the ram continues to travel, the force exerted by the ram could damage the DSC. To eliminate the occurrence of such an event, the amount of force which the ram may exert will be limited to 25% of the loaded DSC weight unless overridden by the systems operator. The stresses which will develop in the DSC due to this force are substantially less than the yield strength of the DSC material and therefore, the integrity of the container and seal weld will not be jeopardized.

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 Section 5.1.

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.

Figures 5.2-1 through 5.2-3

DELETED

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

The only component of the NUHOMS system which could possibly be damaged during a postulated worst case tornado missile impact is the HSM air outlet shielding blocks. The consequence of damaging one or more shielding blocks is an increase in the skyshine scattered dose in the vicinity of the HSM.

In order to reduce the scattered dose, the damaged shielding blocks could be repaired or portable shielding could be placed over the air outlets. The utility may wish to retain spare shielding blocks on site. The quantity is site specific and may depend on the number of HSMs in the installation, frequency of tornadoes, etc. The utility may also wish to retain forms for casting shielding blocks or plan to use existing portable shielding.

5.4 Operation Support System

NUHOMS is a self contained system and requires no effluent processing systems during operations.

5.4.1 Instrumentation and Control Systems

There are no instrumentation and control systems used during storage. The instrumentation and controls necessary during fuel handling and HSM loading were described in Section 5.1.3.4.

5.4.2 System and Component Spares

Since there are no instrumentation and control systems during storage, no system and component spare parts are required.

5.5 Control Room and/or Control Areas

There are no control room or control areas for the NUHOMS system.

5.6 Analytical Sampling

There is no analytical sampling used with the NUHOMS system.

5.7 References

5.1 Deleted.

5.2 Deleted.

5.3 Deleted.

6.0 WASTE CONFINEMENT AND MANAGEMENT

No contaminated wastes are generated during the storage of spent fuel using the NUHOMS system. The handling procedures and systems used in handling the offgas or liquid waste generated during the decontamination, drying, sealing and unsealing operations which utilize existing plant systems are site specific and therefore should be addressed in the utility's application for an ISFSI license.

7.0 RADIATION PROTECTION

7.1 Ensuring That Occupational Radiation Exposures Are As-Low-As-Reasonably-Achievable

7.1.1 Policy Considerations

Management policy and organizational structure related to ensuring that occupational exposures to radiation are as-low-as-reasonably-achievable (ALARA) are necessarily site specific. This information will be provided in specific site license applications.

7.1.2 Design Considerations

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

1. Thick concrete walls on the HSM to limit the contact dose to site workers to below an average of 20 mrem/hr.

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

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

4. Use of a shielded transfer cask for DSC handling and transfer operations which limits the contact dose to 200 mrem/hr or less.

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

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

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

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

Table 7.2-2

GAMMA ENERGY SPECTRUM AND

FLUX-TO-DOSE CONVERSION FACTORS

Upper Energy Level[1] (MeV)	Group[2] Fraction (Unitless)	Flux-to-Dose Conversion Factor[1] (mrem/hr per Photon/cm^2-sec)
10.0	-	9.792 E-3
8.0	-	8.280 E-3
6.5	3.92 E-10	6.840 E-3
5.0	1.18 E-9	5.760 E-3
4.0	1.56 E-7	4.752 E-3
3.0	9.12 E-7	3.960 E-3
2.5	4.04 E-5	3.492 E-3
2.0	-	2.988 E-3
1.66	1.92 E-3	2.412 E-3
1.33	2.71 E-2	1.908 E-3
1.00	-	1.602 E-3
0.80	4.43 E-1	1.260 E-3
0.60	-	9.216 E-4
0.40	1.96 E-2	6.372 E-4
0.30	-	4.392 E-4
0.20	5.01 E-2	2.376 E-4
0.10	1.13 E-1	1.404 E-4
0.05	3.45 E-1	3.024 E-4

Notes:

1. The spectrum and flux-to-dose conversion factors were obtained from Reference 7.7.

2. Group fractions are based on total gammas obtained from methodology defined in Reference 7.6. "-" indicates less than 1.0E-3.

7.3 Radiation Protection Design Features

7.3.1 Installation Design Features

The design considerations listed in Section 7.1.2 ensure that occupational exposures to radiation are ALARA and that a high degree of integrity is achieved through the confinement of radioactive materials inside the DSC. Applicable portions of Regulatory Position 2 of Regulatory Guide 8.8 (7.5) have been used as guidance.

1. Access control for radiation areas is site specific and will be addressed in site specific license applications.

2. Radiation shielding substantially reduces the exposure of personnel during system operations and storage.

3. The NUHOMS system is a passive storage system; no process instrumentation or controls are necessary during storage.

4. Airborne contaminants and gaseous radiation sources are confined by the high integrity double seal welded DSC assembly.

5. No crud is produced by the NUHOMS system.

6. The necessity for decontamination is reduced by maintaining the cleanliness of the DSC and transfer cask during fuel loading and unloading operations (see Section 5.1); the DSC and transfer cask surfaces are smooth, nonporous, and are generally free of crevices, cracks, and sharp corners.

7. No radiation monitoring system is required during storage.

8. No resin or sludge is produced by the NUHOMS system.

Figure 1.3-7 in Section 1.3 is a layout and arrangement drawing for a typical 20-year NUHOMS storage array. Radiation sources are contained within DSCs which are stored in concrete HSMs. The radioactive sources for this NUHOMS installation are described in detail in Section 7.2.1. Design radiation dose rates may be found in Section 10.

The NUHOMS system is a passive storage system which uses ambient air for decay heat removal. Each HSM is capable of providing sufficient ventilation and natural circulation to assure adequate cooling of the DSC and its contents so that fuel cladding integrity is maintained. The convective cooling system is completely passive and requires no filtration system.

7.3.2 Shielding

7.3.2.1 Radiation Shielding Design Features Radiation shielding is an integral part of both the DSC and HSM designs. The features described in this section assure that doses to personnel and the public are ALARA.

The DSC is a cylindrical pressure vessel constructed from stainless steel and lead. Details of the DSC and relevant dimensions can be found on the drawings in Appendix E of this report. Two lead-filled end plugs and three steel plates provide axial shielding at the ends of the DSC. During DSC handling operations, additional shielding is provided by the NUHOMS-24P transfer cask.

Two small penetrations in the top lead plug provide a means for draining water, vacuum drying and helium backfilling the DSC. The penetrations are located on the perimeter of the DSC away from fuel assemblies and contain sharp bends to minimize radiation streaming. Appendix A shows the relevant dimensions of the DSC shielding materials located at the top and bottom ends of the DSC. Figure 1.3-1 shows the physical arrangements of the DSC shielded end plug assemblies.

The transfer cask is a cylindrical shielded vessel or steel and various shielding materials. Details of the transfer cask and relevant dimensions can be found in Appendix E of this report. Radial shielding is provided by a stainless steel inner liner, a lead shield, and carbon steel structural shell. Neutron shielding in the radial direction is provided by an outer metal jacket which forms an annulus with the cask structural shell. The annulus will be filled with a ethylene glycol-water mixture to provide additional neutron shielding. The steel transfer cask top and bottom cover plates, as well as solid neutron shielding material (high hydrogen content), provide additional shielding in the axial direction. Figure 1.3-2a shows the physical arrangement of the transfer cask top and bottom end assembly.

The HSM provides additional shielding during storage of the DSC. Details of the HSM and relevant dimensions can be found in Appendix E of this report. Thick, portland cement, concrete walls and roof provide neutron and gamma shielding. The HSM's front access opening is covered by a thick composite door including steel and solid neutron shield material.

Penetrations in the HSM allow convective air cooling of the DSC and HSM internals. An inlet vent located in the lower front wall of the HSM draws air into a shielded box-like plenum inside the HSM. The HSM outlet vents are located at each end of the HSM roof. The roof openings to the HSM are placed above the DSC shielded end plugs and not directly over the active fuel region. Concrete shielding caps over the HSM outlet vents assure that radiation streaming is reduced. The features of the HSM design are illustrated in Figure 1.3-1a.

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

7.3.2.2 Shielding Analysis This section describes the radiation shielding analytical methods and assumptions used in calculating NUHOMS system local dose rates during the handling and storage operations. The dose rates of interest were calculated at the locations listed in Table 7.3-2. Figure 7.3-3 shows these locations on the HSM, DSC and transfer cask. The two computer codes used for analysis are described below, each with a short description of the input parameters generic to its use. Descriptions of the individual analytical models used in the analysis are also described. Shielding analysis results are summarized in Table 7.3-2.

Computer Codes ANISN (7.1), a one-dimensional, discrete ordinates transport computer code, was used to obtain neutron and gamma dose rates at the outer HSM walls, and at the outside surface of the loaded transfer cask in the radial direction. ANISN was also used to obtain the neutron dose rates at the shielded end plugs of the DSC and transfer cask, and outside the HSM access door. The CASK cross section library, which contains 22 neutron energy groups and 18 gamma energy groups, was applied in an S_8P_3 or $S_{16}P_3$ approximation for cylindrical or slab geometry, respectively (7.7). Calculated radiation fluxes were multiplied by flux-to-dose conversion factors (Table 7.2-1 and Table 7.2-2) to obtain final dose rates. The ANISN calculations used the coupled neutron and gamma libraries. Therefore, dose rates from both primary and secondary gammas were calculated in each run.

QAD-CGGP (7.2), a three-dimensional point-kernel code, was used for gamma shielding analysis of the HSM door, the DSC and cask end sections, the DSC-cask annular gap, and the HSM air vent penetrations. Mass attenuation and buildup factors were obtained from QAD-CGGP's internal library. The gamma energy spectrum was determined in the same manner as the ANISN analysis.

Since QAD-CGGP calculates dose rates from primary gammas only, the primary gamma source strength in the active fuel region was increased for calculations in the axial direction of the DSC. This was done as a way to include the dose rate effect due to secondary gammas primarily generated in the metal end plugs of the DSC and additional activation products located in the fuel assembly end fittings. Earlier calculations indicated that secondary gammas contribute only one percent of the total gamma radiation dose rate in the HSM concrete, so can be neglected for QAD-CGGP model for the HSM concrete.

In order to justify increasing the primary gamma source strength when using QAD-CGGP for axial calculations, a set of benchmarking runs were performed, where QAD-CGGP was used to model an actual metal cask containing spent PWR fuel where the geometry of the cask and the contained fuel was similar to that of the NUHOMS-24P design and actual measured dose rates at the cask ends were available (7.10). It was found that increasing the primary gamma source strength in the active fuel region for the QAD-CGGP runs resulted in the maximum calculated dose rates at the cask ends meeting or slightly exceeding the maximum measured dose rates (average calculated dose rates exceeded average measured dose rates across the entire cask ends). Therefore, increasing the gamma source strength in the active fuel region when using QAD-CGGP for estimating gamma dose rates in the DSC axial direction is expected to result in conservative values.

Manual albedo calculations were used in conjunction with the fluxes calculated by QAD-CGGP and ANISN to provide upper bounds on the reflected dose rate at the air inlet and exits of the HSM and the DSC-cask annular gap. The albedo method used is described in References 7.8 and 7.9.

<u>HSM Surface Dose Rates</u> The ANISN analytical model used to determine neutron and gamma dose rates outside the thick HSM walls (or roof) is presented in Appendix A. The DSC/HSM was represented by a cylindrical model which includes a homogenized, isotropic, self-shielding source region, the DSC wall, an air gap between the DSC and the concrete wall and the thick concrete wall or roof. The effective radius of the source region was chosen to be the inside diameter of the DSC. The mesh size in each material region was chosen to be on the order of one mean free path of neutrons through that material. A buckling factor correction for the infinite length model was made to estimate the neutron dose rate at the active fuel region midplane.

<u>NUHOMS System Axial Dose Rates</u> An ANISN model of an infinite slab was also used to calculate the neutron dose rates in the axial direction (e.g., at the DSC top and bottom cover plate surfaces).

Appendix A illustrates the analytical QAD-CGGP models for the top and bottom axial dose rate calculations, respectively. A simple 3-D slab shield geometry and cylindrical source mesh were constructed. The results were extended to include the dose rate outside the thick HSM door.

Air Outlet (HSM) Penetration Dose Rates - Gamma dose rates at the HSM exhaust vent penetration were calculated by using QAD-CGGP to establish a dose rate profile along the irradiated vent wall, and then applying manual albedo calculations to determine the flux at the HSM roof plane. The streaming was modelled as a broad-beam source incident on a two-dimensional rectangular duct. Empirical differential-dose albedo data for concrete (7.8) were integrated over the irradiated portion of the vent wall to obtain the reflected dose rate.

The QAD-CGGP analytical model is shown in Appendix A. Two discrete sources were modelled to obtain the gamma dose rate profile: the spent fuel region, and the activated nozzle region. The secondary gamma source was found to have a negligible effect on the penetration dose rates due to the large size of the primary gamma source and the self shielding effect of lead, and so it was not included. Additional shielding provided by the concrete air outlet vent cap was analyzed to estimate the external surface dose rate above this penetration.

The neutron dose rate at the air outlet vent was estimated using ANISN and manual albedo calculations. ANISN results were used to obtain fluxes incident along the air outlet vent. The source was assumed to be a broad beam of monodirectional particles at predominant energy groups. Albedo factors were obtained from References 7.8 and 7.9.

Cask-DSC Annular Gap Dose Rate The exact annulus size depends on the design of the DSC and the transfer cask. An evaluation of radiation streaming through a bounding [ ] inch annular gap (maximum possible) between the DSC and the transfer cask was performed. Manual albedo calculations were performed to assess the annular gap streaming. Neutron and gamma fluxes, as determined by ANISN outside the DSC at the fuel midplane, were assumed to be constant over the cask inner wall surface. The worst-case scenario was examined where the DSC is placed off-center in the cask resulting in no clearance on one side and [ ] clearance on the other. The results are given in Table 7.3-2.

Transfer Cask Surface Dose Rates The ANISN model used to determine combined radiation dose rates on and away from the transfer cask surface during handling operations is shown in Appendix A. The analytical modelling methodology is similar to the ANISN model of the HSM previously described.

Shielding Analysis Results Results from the major NUHOMS systems shielding analyses described above are presented in Table 7.3-2.

7.3.3 Ventilation

The HSM has a ventilation system to provide for natural circulation cooling of the DSC. No off-gas treatment system is required due to the low exterior contamination level of the DSC (see Section 3.3.7).

The NUHOMS system is designed to prevent release of radioactive material during normal storage of the DSC in an HSM. No additional design features or equipment would result in a significant reduction in a postulated release of radioactive materials. Furthermore, no credible site accident would result in a release of radioactive materials to the environment due to the key features of the NUHOMS system design. These include:

1. The use of a high integrity DSC with redundant seal welds at each end,
2. The passive nature of the system such as the HSM natural convection cooling system which ensures that fuel cladding integrity is maintained, and
3. The operational limits and controls placed on DSC loading and handling operations.

7.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Information on area radiation and airborne radioactivity monitoring instrumentation is site specific and will be covered in site specific license applications.

TABLE 7.3-1 HAS BEEN DELETED

Table 7.3-2

SHIELDING ANALYSIS RESULTS

Location	Neutron Dose Rate (mrem/hr.)		Gamma Dose Rate (mrem/hr) Primary and Secondary		Total Dose Rate (mrem/hr.)
	Direct	Reflected	Direct	Reflected	
DSC in HSM					
1. HSM Wall or Roof	0.1	(2)	6.5	(2)	6.6
2. HSM Air Outlet Shielding Cap	0	0.2	0	50.0	50.2
3. HSM Air Outlet (No Shielding Cap)	0.7	15.0	265.0	3270.0	3550.7
4. Center of Door	37.0	(2)	7.6	(2)	44.6
5. Center of Opening	430.0	(2)	330.0	(2)	760.0
6. Center of Air Inlet	0.1	2.0	6.5	87.8	96.4
7. 2 Meters From HSM Door	20.0	(2)	4.0	(2)	24.0
DSC IN CASK					
1. Centerline Top of DSC Plug (1)	5.3	(2)	4.7	(2)	10.0
2. Top of DSC Cover Plate (with water in annulus and with 2 inches of temporary neutron shielding)					
2.1 Centerline	49.0	(2)	65.0	(2)	114.0
2.2 Outer Edge (3)	39.0	(2)	52.0	100.0	191.0
3. Transfer Cask Surface					
3.1 Radial	48.0	(2)	120.0	(2)	168.0
3.2 Top axial	15.0	(2)	1.0	(2)	16.0
3.3 Bottom axial	32.0	(2)	16.0	(2)	48.0

Table 7.3-2

SHIELDING ANALYSIS RESULTS
(Concluded)

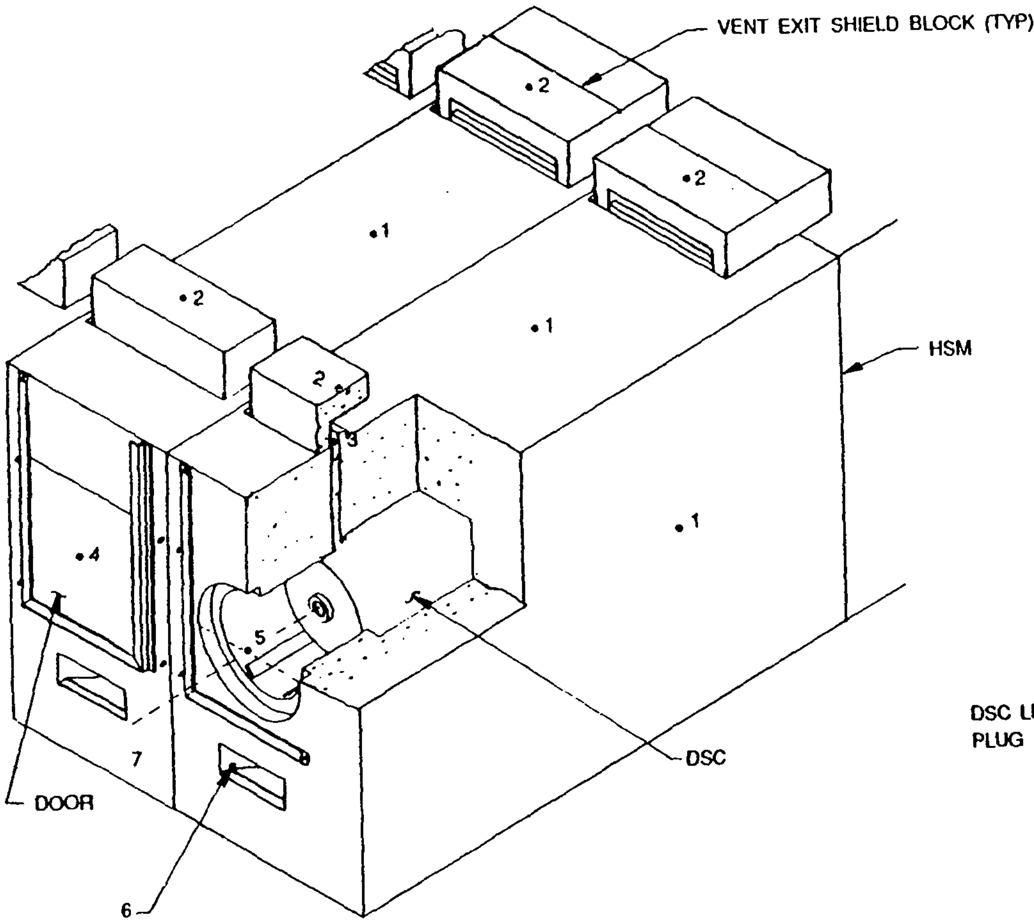
Notes:

1. The DSC/cask annulus is filled with water and additional neutron shielding material is utilized as required. In addition, all but the top six inches of the DSC inner cavity is assumed to be filled with water for this operation.

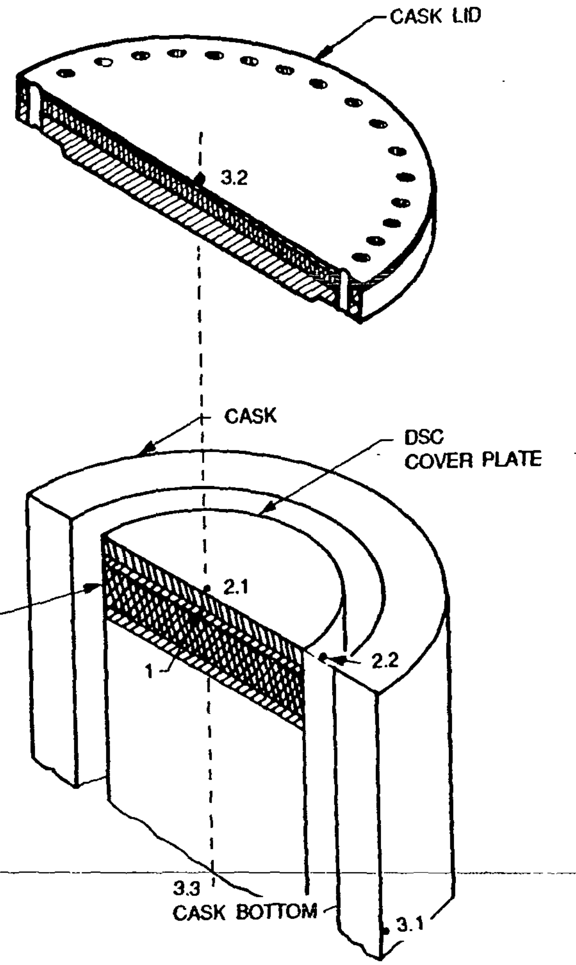
2. The reflected dose at these locations is negligible.

3. The same gap dose rate applies for case where only top lead plug is on DSC. The dose rates reported are with water in the DSC/cask annulus (however, no water was assumed to be in the DSC).

FIGURES 7.3-1 AND 7.3-2 HAVE BEEN DELETED



DSC IN HSM DOSE RATE CALCULATION LOCATIONS



DSC IN CASK DOSE RATE CALCULATION LOCATIONS

Figure 7.3-3

LOCATIONS OF REPORTED DOSE RATES

(TABLE 7.3-2)

- FIGURE 7.3-4 IS INCLUDED IN APPENDIX A
- FIGURES 7.3-5 THROUGH 7.3-7 HAVE BEEN DELETED
- FIGURE 7.3-8 IS INCLUDED IN APPENDIX A
- FIGURES 7.3-9 AND 7.3-10 HAVE BEEN DELETED
- FIGURE 7.3-11 IS INCLUDED IN APPENDIX A

7.4 Estimated On-Site Collective Dose Assessment

7.4.1 Operational Dose Assessment

This 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. Section 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 curves 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.

The cumulative dose rates to personnel for the NUHOMS-24P system are similar to those of the NUHOMS-07P system contained in the respective Topical Report. Since fuel handling, DSC loading, draining, drying, sealing and handling operations are plant unique, cumulative dose calculations for personnel will be needed for site specific license applications.

7.4.2 Storage Term Dose Assessment

Figure 7.4-1 shows graphs of the dose rate (mrem/hr) versus distance from the face of a single NUHOMS-24P and a 2x10 array of NUHOMS-24P HSMs. The curve was constructed from the shielding analysis described in the previous sections. Direct neutron and gamma flux, as well as the air-scattered radiation from the module surfaces are considered. The surface radiation sources used for the direct and air scattered dose calculations are shown in Figure 7.4-2. The energy distribution of the neutron and gamma fluxes was taken from the applicable calculation as described in the previous sections. Air-scattered dose rates are determined with the computer code SKYSHINE-II (7.4); direct dose rates are calculated using the computer codes MICROSHIELD (7.11) and ANISN (7.1). The direct flux from the "hidden" row of modules is considered completely shielded by the front row. Initial loading of all HSMs with the design basis ten year post-spent fuel is assumed.

Occupancy times for on-site personnel near the HSM array and the HSM loading sequence are necessarily site specific. The collective on-site dose assessment due to the loading and the storage phase should be evaluated, (using Figure 7.4-1), by utilities submitting site specific safety analysis reports.

Table 7.4-1

SUMMARY OF OPERATIONS FOR PERSONNEL DOSE CALCULATIONS[1]

Operation	Number of Personnel	Time[3] (Hours)	Average Distance From Cask Surface (Feet)
Location: Fuel Pool			
Load Fuel into DSC	4	10.0	30.0
Place Shielded End Plug onto Cask	2	0.50	1.5
Location: Cask Decon Pit			
Decontaminate Outer Surface of Cask	4	2.0	1.5
Place Scaffolding Around Cask	4	0.50	4.0
Lower Water Level in DSC Cavity[2]	2	0.25	4.0
Set up Automatic Welder to Seal Weld Lead Plug to DSC	2	3.00	1.5
Perform Dye Penetrant Examination	1	1.50	1.5
Remove Remaining Water and Vacuum Dry DSC Cavity[2]	2	24.00	4.0
Backfill DSC Cavity with Helium	2	0.50	4.0
Perform Helium Leak Test	1	0.50	1.5
Seal Weld Plug Penetration	1	0.50	1.5
Perform Dye Penetrant Examination On Plug Penetration Welds	1	0.25	1.5
Install Top Cover Plate	2	0.25	1.5

Table 7.4-1

SUMMARY OF OPERATIONS FOR PERSONNEL DOSE CALCULATIONS[1]
(Continued)

Operation	Number of Personnel	Time[3] (Hours)	Average Distance From Cask Surface (Feet)
Set Up Automatic Welder to Weld Top Cover Plate to DSC	2	3.00	1.5
Perform Dye Penetrant Examination on Weld	1	2.00	1.5
Install Cask Head and Bolt into Place	2	1.00	1.5
Drain Cask/DSC Annulus	2	0.50	1.5
Remove Scaffolding from Around Cask	4	0.50	4.0
Transport Cask to Skid and Trailer	4	2.00	8.0
<u>Location</u>: Trailer/HSM			
Attach Skid-Tiedown to Cask	2	0.25	2.0
Transport Cask to HSM	------Site Specific------		
Remove Cask Head, Bottom Cover Plate and Position Ram	2	1.00	1.5
Align Cask with HSM and Install Cask Restraints	4	2.00	3.0
Transfer DSC from Cask to HSM	4	0.50	3.0
Close and Tack Weld Front Door	1	1.00	4.5

Table 7.4-1

SUMMARY OF OPERATIONS FOR PERSONNEL DOSE CALCULATIONS[1]
(Concluded)

Notes:

1. Cumulative doses for these operations are expected to be similar to those presented in the NUHOMS-07P Topical Report.

2. Operations and man-hours are provided here as an example. Cumulative dose calculations will be needed for site specific license applications.

3. These are the estimated times to complete the specific activities. The quantity of time personnel are present within the radiation field may be significantly less.

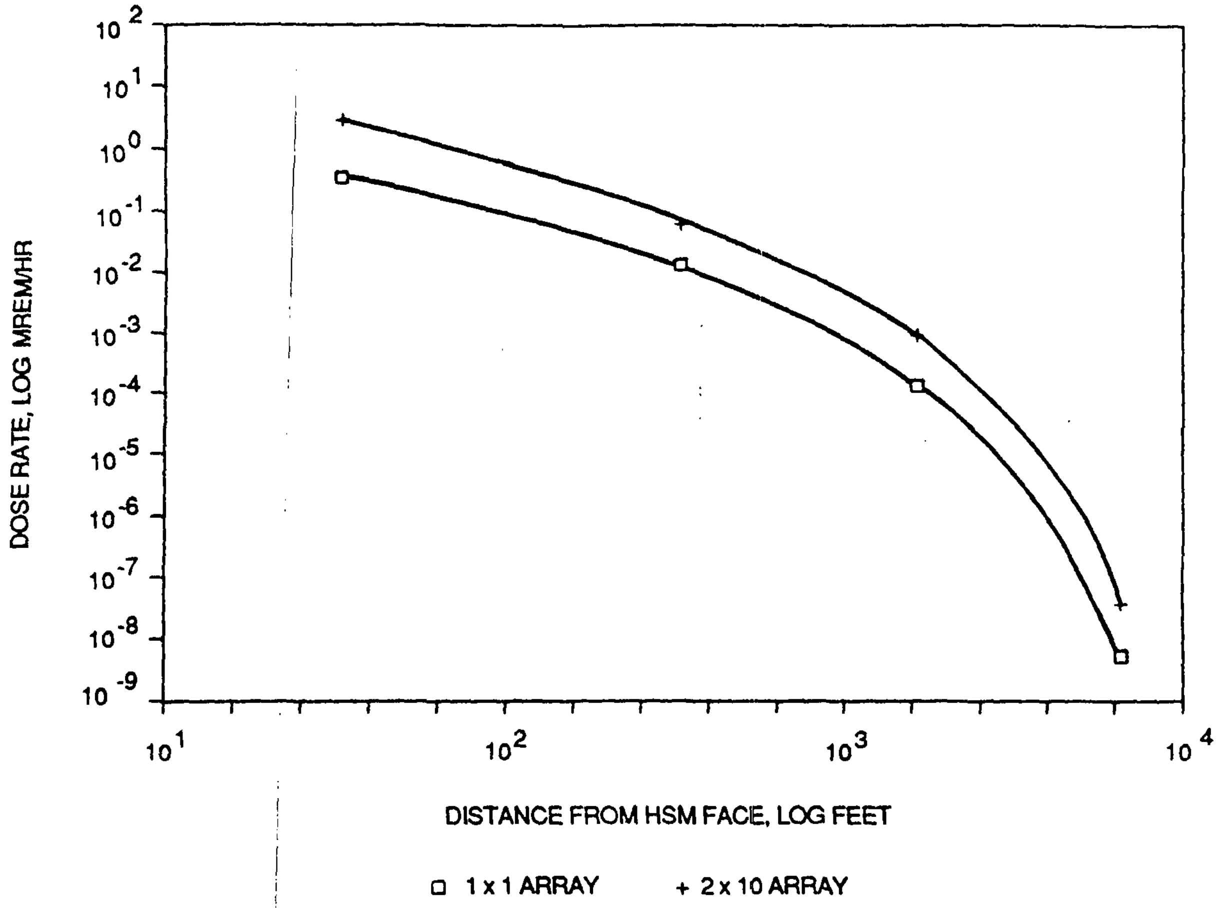
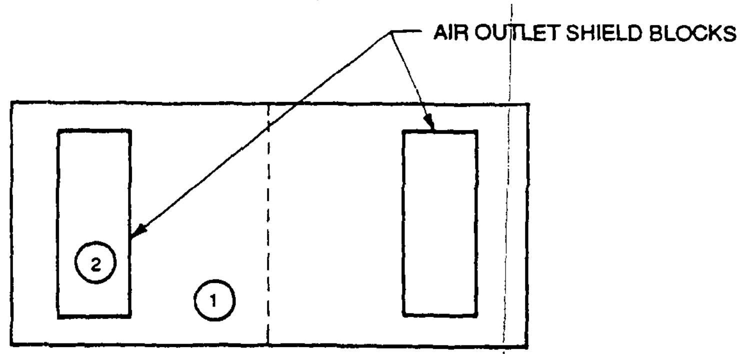
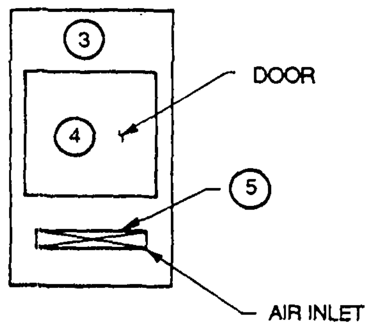


Figure 7.4-1

DOSE RATE VS. DISTANCE FROM SURFACE OF HSM



HSM ROOF PLAN



HSM FRONT PLAN

LOCATION	AREA (ft^2)	NEUTRON DOSE RATE (mrem / hr)	GAMMA DOSE RATE (mrem / hr)	TOTAL DOSE RATE (mrem / hr)
ROOF				
1**	166.67	0.1	6.5	6.6
2*	33.33	0.2	50.0	50.2
AREA WEIGHTED AVG		0.1	13.7	13.8
AVG				
FRONT				
3**	116.67	0.1	6.5	6.6
4	26.00	37.0	7.6	44.6
5*	6.00	2.1	94.3	96.4
AREA WEIGHTED AVG		6.6	10.2	16.8

* DIRECT AND REFLECTED DOSES

** DOSE RATES FOR ALL OF AREA 1 ON BOTH THE ROOF AND FRONT WALL ARE ESTIMATED TO BE THE SAME AS THE DOSE RATE CALCULATED AT THE CENTER OF THE ROOF.

Figure 7.4-2

RADIATION ZONE MAP OF HSM SURFACE DOSE RATES

7.5 Health Physics Program

The health physics program for a NUHOMS installation is site specific and shall be described in site license applications.

7.6 Estimated Off-Site Collective Doses

During normal operation there are no effluent streams from the HSM. The program and the analytical approach taken to monitor the radioactive material content of the effluent streams during fuel handling will be site specific.

The only off-site dose due to the NUHOMS installation will be from direct and skyshine radiation. Figure 7.4-1 shows the contribution of each of these sources as a function of distance from the center of a single NUHOMS-24P HSM and also from a 2x10 array of NUHOMS-24P HSMs installation. Estimates of the actual contribution at a plant site will be provided in a site specific license.

7.7 References

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7.2 Oak Ridge National Laboratory, "QAD-CGGP, Point-Kernel Gamma Ray Shielding Code," CCC-493, Oak Ridge National Laboratory (1986).

7.3 Deleted.

7.4 C. M. Lampley, "The SKYSHINE-II Procedure: Calculation of the Effects of Structure Design on Neutron, Primary Gamma-Ray and Secondary Gamma-Ray Dose Rates in Air," NUREG/CR-0781, RRA-T7901, USNRC (1979).

7.5 U. S. Nuclear Regulatory Commission, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be As Low As Reasonably Achievable," Regulatory Guide 8.8, Revision 3, (1978).

7.6 SCALE-3: A Modular Code System for Performing Standarized Computer Analysis for Licensing Evaluation, ORNL, Revision 3, December 1984.

7.7 Radiation Shielding Information Center, "CASK: 40 Group Neutron and Gamma Ray Cross Section Data," DLC-23, September 1978.

7.8 American Nuclear Society Standards Committee Working Group ANS-6.4, "American National Standard Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," ANSI/ANS-6.4-1977, American Nuclear Society, 1978.

7.9 W. E. Selph, "Neutron and Gamma-Ray Albedos," ORNL-RSIC-21, Radiation Shielding Information Center, Oak Ridge National Laboratory, February 1968.

7.10 Electric Power Research Institute, "The TN-24P PWR Spent-Fuel Storage Cask: Testing and Analysis", EPRI NP-5128, April 1987.

7.11 Grove Engineering, Inc., "Microshield User's Manual, A Program for Analyzing Gamma Radiation Shielding", Ver. 2.0, 1985.

8.0 ANALYSIS OF DESIGN EVENTS

In previous chapters of this report, features of the NUHOMS system 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 identify and analyze a range of credible and hypothetical accidents.

In accordance with NRC Regulatory Guide 3.48 (8.1), the design events identified by ANSI/ANS 57.9-1984, (8.2) form the basis for the accident analyses performed for the NUHOMS-24P system. Four categories of design events were defined. These events provide a means of establishing design requirements to satisfy operational and safety acceptance criteria. Design event Types I and II covering normal and off-normal events are addressed in Section 8.1. Design event Types III and IV covering a range of postulated accident events are addressed in Section 8.2 of this report.

It is important to note, that given the generic nature of this report, the majority of the analyses presented throughout this section are based on bounding conservative assumptions and methodologies, with the objective of establishing upper bound values for the responses of the primary components and structures of the NUHOMS-24P system for the design basis events. Because of the conservative approach adopted, the reported temperatures and stresses in this section do not necessarily reflect the actual temperatures or states of stress for the various operating and postulated accident conditions. More rigorous and detailed analyses and/or more realistic assumptions and loading conditions would result in temperatures and states of stress which are significantly lower than the reported values.

8.1 Normal and Off-Normal Operations

Normal operating design conditions consist of a set of events that occur regularly, or frequently, in the course of normal operation of the NUHOMS system. These normal operating conditions are addressed in Section 8.1.1. Off-normal operating design conditions are events that could occur with moderate frequency, possibly once during any calendar year of operation. These off-normal operating conditions are addressed in Section 8.1.2. The thermal-hydraulic, structural, and radiological analyses, associated with these events are presented in the sections which follow.

8.1.1 Normal Operation Structural Analysis

Table 8.1-1 shows the normal operating loads for which the NUHOMS safety-related components are designed. The table also lists the individual NUHOMS components which are affected by each loading. The magnitude and characteristics of each load are described in Section 8.1.1.1.

The method of analysis, and the analytical results for each load, are described in Sections 8.1.1.2 through 8.1.1.9. The mechanical properties of materials employed in the structural analysis of the NUHOMS system components are presented in Table 8.1-2.

8.1.1.1 Normal Operating Loads The normal operating loads for the NUHOMS system components are:

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

These loads are described in detail in the following paragraphs.

A. Dead Weight Loads

Table 8.1-3 shows the weights of various components of the NUHOMS-24P system. The dead weight of the component materials was determined based on nominal component dimensions.

A density value of 0.283 pound per cubic inch for carbon steel, 0.285 pound per cubic inch for stainless steel, 0.408 pound per cubic inch for lead shielding, and 0.064 pound per cubic inch for solid neutron shielding material were used in the dead weight calculations.

A nominal concrete density of 140 to 145 pounds per cubic foot was selected as a design basis for the shielding and thermal evaluations. A maximum nominal density of 150 pounds per cubic foot was assumed for the dead weight evaluation of the HSM.

B. Design Basis Internal Pressure

The range of DSC internal pressures for operating conditions and postulated accident conditions and the associated helium gas temperatures are shown in Table 8.1-4. The DSC internal pressure, with the fuel cladding intact, for normal and off-normal operating conditions ranges from 4.0 psig to 9.7 psig for the seasonal average temperature range of -40°F to 125°F. The DSC internal pressure for off-normal or postulated accident conditions ranges from 32.3 psig. to 49.1 psig. These bounding design basis pressures were determined by assuming cladding failure in 100% of the possible 4992 spent fuel rods being stored in the DSC. The cladding failure was assumed to release all of the fuel rod fill gas and 30% of the fission gas generated in fuel assemblies irradiated to 40,000 MWD/MTU. This postulated worst case condition was included in the design basis for the DSC as a conservative means of providing over-pressure protection for the DSC pressure boundary. For normal operating conditions, the gas inside the DSC was assumed to be equilibrated thermodynamically at the average gas temperature which occurs at a maximum normal ambient temperature of 100°F. Similar assumptions were made for off-normal and accident conditions to determine the DSC internal pressure occurring with a gas temperature at an extreme ambient temperature of 125°F. The effects of postulated accident pressures are described in Section 8.2 of this report.

C. Design Basis Thermal Loads

The NUHOMS-24P transfer cask, DSC, and HSM are subjected to the thermal expansion loads associated with normal operating conditions. The range of average daily ambient temperatures used for the design of the transfer cask, DSC, and HSM for normal operating conditions is 0°F to 100°F. The normal operating seasonal average daily ambient temperature fluctuates from 0°F minimum (winter) to 100°F maximum (summer) and is conservatively assumed to occur for a sufficient duration to establish steady state conditions for the transfer cask, DSC and HSM. These minimum and maximum steady-state long-term ambient design temperatures, envelop the 24 hour average seasonal ambient temperature at any location within the United States (8.46).

The long-term average normal ambient temperature for the 50 year design life of the system is assumed to be 70°F. This base case average ambient temperature bounds practically all reactor sites within the United States (8.27). The few exceptions will be addressed in site license applications as necessary.

The range of ambient temperature cases analyzed are further defined in Section 8.1.3. The resulting temperature distributions in the HSM, DSC and NUHOMS-24P transfer cask were determined by performing thermal analyses for these ambient conditions. The thermal analyses, described in Section 8.1.3, provide temperature distributions for the HSM, DSC, and NUHOMS-24P transfer cask such as those shown in Tables 8.1-9b, 8.1-11 through 8.1-14 and Figures 8.1-1 through 8.1-3c. These temperature distributions were developed for the range of normal ambient temperatures specified above and are applicable to any size HSM array. The corresponding HSM structural analysis results for thermal loads are applicable to HSM arrays ranging in size from a single stand-alone module to a 2x10 array.

Figure 8.1-1 shows the temperature distribution around the circumference of the DSC (when implaced in an HSM) and the temperatures at selected locations throughout the spent fuel assemblies, guide sleeves and helium gas regions inside the DSC for the base case 70°F lifetime average ambient temperature. The maximum temperatures of the centermost fuel rods for each fuel assembly for the DSC are also shown. The analysis and HEATING6 (8.5) results for the 70°F ambient temperature base case are discussed in Section 8.1.3. The DSC and fuel assembly temperature distributions when implaced in an HSM were also evaluated for the minimum average daily (winter) and maximum average daily (summer) ambient conditions. The resulting temperature distribution for the 100°F ambient temperature case are shown in Figure 8.1-1a. The fuel assembly temperatures for the minimum average daily (winter) conditions are enveloped by this case and were, therefore, not evaluated further.

Figure 8.1-2 shows the temperature distribution in a spacer disk located mid-length along the axis of the DSC for the 100°F ambient temperature case. This temperature distribution was determined by averaging the temperatures from a two-dimensional HEATING6 calculation of the heat transfer across a spacer disk (steel assumed to be between the guide sleeves and DSC shell) and the higher temperatures of the helium on either side of the spacer disk (from the analysis results with helium assumed to be between the guide sleeves and the DSC shell). This accounts for the effects of helium heating the surfaces of the spacer disk and provides a more conservative temperature distribution for use in the thermal stress analysis of the DSC spacer disk.

Figures 8.1-3, and 3a show the temperatures at various locations in the HSM for the 70°F, and 100°F normal operating ambient temperature cases. Solar heat loads were conservatively neglected for the 0°F ambient temperature case. The temperature distributions shown were derived from two dimensional HEATING6 analyses described in Section 8.1.3.

The maximum calculated temperatures for the various structural sections of the HSM for normal, operating conditions are summarized in Table 8.1-12. A more detailed tabulation of the HSM thermal results used for the structural design of a single HSM and a 2x10 HSM array are shown in Table 8.1-9b. The HSM reinforced concrete design is controlled by the accident thermal gradients described in Section 8.2.

The effect of DSC's being emplaced at varying locations throughout an HSM array were investigated using the load case matrix shown in Table 8.1-9c. The appropriate peak temperatures and thermal gradients from Table 8.1-9b were applied to the HSM computer models and the resulting forces and moments calculated. A sufficient range of thermal load cases were included in this investigation to demonstrate that the Table 8.1-9c load cases produced the worst distribution of DSCs within an HSM array.

Figure 8.1-3b shows the temperature distribution in the NUHOMS-24P transfer cask during transfer of a loaded DSC for normal operating conditions. This distribution assumes a 100°F ambient temperature with solar heating of the transfer cask outer surface. Similarly, Figure 8.1-3c shows the transfer cask temperature distribution with an ambient temperature of 0°F. Solar heat loads were conservatively neglected for this case. The DSC was assumed to remain in the transfer cask for a sufficient length of time to reach steady state conditions.

The temperature distributions derived from the three normal operating cases (0°F, 70°F, 100°F) were considered in the structural analysis of the DSC, HSM and transfer cask discussed in Sections 8.1.1.2. through 8.1.1.9. The temperature distributions for each component were used to determine the effects of thermal stresses and thermal cycling on the NUHOMS components. These results were also used to evaluate the effects of creep on the HSM reinforced concrete.

The thermophysical properties of materials used in the thermal and stress analyses of the NUHOMS system components are presented in Tables 8.1-5 and 8.1-6.

D. Operational Handling Loads

The most significant operational loading condition for the NUHOMS system components is sliding of the DSC from the NUHOMS-24P transfer cask into the HSM. Sliding is achieved by the push/pull forces induced by the hydraulic ram system. These forces are applied to the grapple ring assembly which is an integral part of the DSC bottom end assembly. The forces induced by the ram system are reacted by friction forces which develop between the sliding surfaces of the DSC, the transfer cask, and HSM support rails.

Based on the surface finish and the contact angle of the DSC support rails inside the HSM described in Section 4, a bounding coefficient of friction was conservatively assumed to be 0.25. Therefore, the nominal ram load required to slide the DSC under normal operating conditions is:

$$P = \frac{0.25\ W}{\cos\theta} = 0.29\ W \qquad (8.1\text{-}1)$$

Where: P = Push/Pull Load

W = Loaded DSC Weight

θ = 30 degree, Angle of the T-Section Support Rail

The DSC bottom cover plate and grapple ring assembly were designed to withstand a normal operating push/pull force equal to approximately 25% of the loaded DSC weight (conservatively assumed to be 20,000 lb.). To insure retrievability for a postulated jammed DSC condition, the ram is sized with a capacity for an enveloping load of 80,000 lb., as described in Section 8.1.2 of this report. These loads also bound those friction forces which would develop between the sliding surfaces of the DSC and transfer cask.

E. Design Basis Live Loads

As discussed in Section 3.2.4, a live load of 200 pounds per square foot was conservatively selected to envelope all postulated live loads acting on the HSM, including the effects of snow and ice. Live loads which may act on the transfer cask are negligible, as discussed in Section 3.2.4.

8.1.1.2 Dry Shielded Canister Analysis

Stresses were evaluated in the DSC due to:

1. Dead Weight
2. Design Basis Normal Operating Internal Pressure Loads
3. Normal Operating Thermal Loads
4. Normal Operation Handling Loads

The methodology used to evaluate the effects of these normal loads is addressed in the following paragraphs. Table 8.1-7 summarizes the resulting stresses for normal operating loads.

A. DSC Dead Load Analysis

For the dead load analysis, the most limiting conditions were considered. Both the beam behavior and shell bending behavior of the DSC seated on the HSM support rails were evaluated. By conservatively considering the DSC to be supported at the three support system cross member locations, and distributing the total DSC weight of approximately 72,000 pounds along its length, the maximum positive and negative beam bending stresses in the DSC shell were derived. The analysis of the DSC cylindrical shell acting as a simply supported beam resulted in a maximum membrane plus bending stress intensity of 0.2 ksi which is insignificant compared to the allowable membrane stress intensity of 18.7 ksi.

For the evaluation of local shell bending stresses, it was conservatively assumed that the total dead weight of the DSC, less the weight of the shielded end plugs, was uniformly supported by the continuous T-section support rails. The geometric boundary conditions assumed in this analysis are depicted in Figure 8.1-4.

The correlation from Bednar (8.52) was used in the evaluation of this condition as follows:

$$S_{bx} = 1.75 \sqrt{Rt} \frac{f_3}{t^2} \quad \text{(Ref. Bednar, page 193)} \quad (8.1\text{-}2)$$

Where:

S_{bx} = Local Shell Bending Stress Intensity, = 3.7 Ksi

t = 0.625 in., DSC Shell Thickness

R = 33.625 in., DSC Outside Radius

f_3 = 179 lb./in., Weight of loaded DSC less shielded end plugs per unit length for each support rail.

The calculated dead weight stresses are tabulated in Table 8.1-7. The stresses induced by the cover plates and shielded end plugs are negligible compared to the weight of the DSC internal basket and fuel assemblies and do not warrant further analysis.

B. DSC Normal Operating Design Basis Pressure Analysis

For the evaluation of DSC stresses due to design basis normal operating internal pressures, two analytical models, one each for the DSC top and bottom ends, were developed using the ANSYS computer program. A segment of the DSC shell, the cover plates, and the shielded end plug assemblies were included in the analytical models, as shown in Figures 8.1-5 and 8.1-6. The design basis internal pressures shown in Table 8.1-4 were applied to the analytical models as a uniform internal pressure loading and the DSC stresses calculated. The resulting maximum stress intensities are reported in Table 8.1-7.

C. DSC Normal Operating Thermal Analysis

The thermal analysis of the DSC is presented in Section 8.1.3 of this report. The results of this analysis show that for the range of normal operating ambient temperature conditions, no significant DSC through wall thermal gradients exist. There is also sufficient space provided in the axial direction between the internal basket assembly and the inner surfaces of the DSC cover plates for free thermal expansion. Similarly, sufficient radial gap is provided between the spacer disks and inner DSC shell to permit free thermal expansion. As a result, no thermal stresses are induced in the DSC or the basket assembly. This design feature also acts to minimize the effects of thermal cycling and fatigue on the DSC.

The effects of thermal loads due to differential expansion of the spacer disk were evaluated using a finite element model of the spacer disk. The geometry of the spacer disk model is shown in Figure 8.1-7 and is comprised of 590 nodes, and 392 plate bending and stretching elements. The ANSYS (8.48) computer code was employed in this analysis. The spacer disk normal operating temperature profile from the DSC heat transfer analysis results discussed in Section 8.1.3 was utilized.

The temperature distribution shown in Figure 8.1-2 was imposed on the analytical model of the spacer disk. The maximum stress determined from this analysis was 46.5 ksi. The resulting radial growth of 0.04 inches is much less than the gap of 0.25 inches between the spacer disk and the inside surface of the DSC shell.

For normal operating conditions, a thermal stress analysis of the DSC shell was performed to establish the DSC shell stresses induced by circumferential variations in shell temperatures. The 70°F ambient case DSC shell temperature varies circumferentially, as shown in Figure 8.1-1. The thermal bending and membrane stresses due to these temperature variations were evaluated using an ANSYS plane strain model of the DSC shell. The analytical model consists of a unit length segment of the DSC shell near its mid-length. The circumferential temperature distribution was imposed on the DSC shell. A maximum combined membrane and bending stress intensity of 17.5 ksi was calculated in the DSC shell. The analysis results for this case are considered in the formulation of normal operating load combination results in Section 8.2.10.

An additional analysis (for normal operating conditions) was performed to evaluate the effects of temperature variations at the interface of the DSC top and bottom end plugs and the DSC shell. This analysis was performed by using the DSC top and bottom end assembly models described previously. Through-wall temperature distributions across the plug assemblies as well as the maximum DSC shell circumferential temperature variations were conservatively imposed on the models to determine the maximum stresses. The analysis showed a maximum stress of 1.9 ksi in the DSC shell and 1.8 ksi in the bottom cover plate.

The analysis results for this case are considered in the formulation of normal operating load combination results in Section 8.2.10.

D. DSC Operational Handling Load Analysis

The applied force from the hydraulic ram, specified in Section 8.1.1.1, was applied to the analytical model of the DSC bottom end assembly at the grapple ring location. The primary bending stresses in the bottom cover plate and secondary bending stresses at local discontinuities in the DSC were calculated using this ANSYS finite element model which is shown in Figure 8.1-6. A uniform annular line load was applied to the grapple ring plate which is attached to the bottom cover plate to evaluate the effect of the ram forces. The resulting ring plate stresses were increased to account for the actual grapple mechanism load which is only applied at two circumferential locations on the grapple ring plate. The results of this analysis are tabulated in Table 8.1-7.

Other components of the DSC grapple ring assembly were also designed to withstand the design basis handling loads. These components, consist of a ring plate, a cylinder attached to the bottom cover plate and the associated attachment welds. The resulting stresses in these components are well within the ASME Code acceptable limits.

E. Evaluation of the Results

The maximum calculated DSC shell stresses induced by normal operating load conditions are shown in Table 8.1-7. The calculated stresses for each load case were combined in accordance with the load combinations presented in Table 3.2-5a. The resulting stresses for the controlling load combinations are reported in Section 8.2.10 with the ASME Code allowable stresses.

8.1.1.3 DSC Internal Basket Analysis

The DSC internal basket assembly can be easily designed to store various types of PWR fuel assemblies. As discussed in Section 3.1.1 of this report, the physical parameters selected for this generic analysis conservatively envelope those of PWR assemblies of this type. The spacer disks of the DSC basket assembly are located axially to coincide with the grid spacers of the fuel assemblies. The weight of each spent fuel assembly is, therefore, directly transmitted to the spacer disk for any load applied perpendicular to the DSC axis. Thus, the other components of the basket (i.e., the 24 guide sleeves and the four three-inch diameter support rods) do not bear any normal operation loads during storage in the HSM except for their own weights. The stresses induced by the weights of these components were evaluated by simple beam theory and found to be negligible. The normal operating loads which induce stresses in the spacer disks are the dead weight loads and differential thermal expansion effects.

A. DSC Internals Dead Weight Analysis

The spacer disk dead weight stresses were obtained directly from the analytical model described in Section 8.2.5. The horizontal drop load analysis results (evaluated for 75g decelerations) reported in Section 8.2.5 were divided by a factor of 75 to obtain the dead weight stresses for the spacer disk.

B. DSC Internals Thermal Analysis

The effects of axial and radial thermal expansion were evaluated for the DSC internal basket components. As described in Section 8.1.1.2, Paragraph C, adequate space exists in the cavity of the DSC, between the spent fuel assemblies and the shielded end plug assemblies, for free thermal expansion. To verify that adequate provision for free axial expansion of the spent fuel assemblies and other internal components of the basket were included, the differential expansion of each DSC component was calculated. The spent fuel assemblies were assumed to be at their long term storage temperature limit of 644°F (340°C) and the DSC shell at its long term average normal operating temperature (70°F ambient air) of 216°F. Calculated results for varying ambient temperatures show similar results as the ΔT between the spent fuel and DSC does not change appreciably. The length of the spent fuel assembly when hot is:

$$L_H = (L_Z\alpha_Z + L_S\alpha_S)\ \Delta T + L_T \qquad (8.1\text{-}28)$$

Where:

L_H = Hot length of spent fuel assembly, in.

L_Z = Length of Zircaloy guide tube = 155.31 in.

α_Z = Zircaloy coefficient of thermal expansion = 3.5E-6 in./in.°F.

L_S = Length of stainless steel per fuel assembly is 15.44 in.

α_s = Stainless steel coefficient of thermal expansion is 10.6E-6 in./in.°F at 650°F

ΔT = 644° - 70° = 574°F

L_T = Total length of fuel assembly including control components at room temperature = 170.75 in.

Therefore: L_H = 171.16 in.

Allowing 1.625 inch for irradiation growth of the spent fuel assembly, the total assembly length including thermal expansion is 172.79 inches. The minimum length of the DSC cavity at room temperature is 172.95 inches when considering dimensional tolerances. The minimum height of the DSC cavity (L_H) at 216°F is:

$$L_H = L_C \alpha \Delta T + L_C \qquad (8.1\text{-}29)$$

Where: L_C = 172.95 in., Minimum DSC length at room temperature

ΔT = 216° - 70° = 142°F

α = 9.19E-6 in./in.°F at 230°F

Therefore: L_H = 173.18 in.

The minimum clearance between the end of the spent fuel assemblies and the inner surface of the shielded end plug assembly for the base case ambient conditions (70°F) and the associated normal operating spent fuel temperature is about 0.4 inch. Thus, adequate clearance has been provided between the spent fuel assemblies and the DSC internal basket assembly to permit free thermal expansion under the maximum differential temperatures expected during normal operation.

8.1.1.4 DSC Support Assembly Analysis

The general description of the DSC support assembly inside the HSM is provided in previous sections of this report. The DSC support assembly is shown in Figures 4.2-1 and 4.2-2. The DSC support assembly cross member end connections consist of W10x68 beams resting on beam seat connections anchored to embedded plates cast in place with the HSM walls. The DSC support assembly design uses bolted and welded connection details. Slotted holes are used in the cross member connections for easy installation and to allow free thermal growth. Normal operating condition loads on the cross member end connections consist of the DSC dead weight, the support assembly dead weight, and the

DSC operational handling loads. Minimum frictional loads are experienced by the support rail end connections as the WT6x115's are welded to the inside surface of the steel penetration sleeve located in the access opening to the HSM. The resulting friction loading which develops between the sliding surfaces of the DSC shell and the top flange of DSC support rails is transferred axially by the support rails to the HSM penetration sleeve to which the rails are welded.

The various components of the DSC support assembly are subjected to normal operating loads including dead weight, thermal, and operational handling loads. A linear elastic beam model of the assembly was developed to evaluate axial, shear, and bending stresses in the DSC support assembly members. The geometry and boundary conditions for the analytical model are shown in Figure 8.1-8. The model consists of 28 nodes and 29 beam elements. Rigid members were used to approximate the connection between the T-section support rails and the cross member support beams to maintain the geometric relationship between these members for load transfer. Restraints at the appropriate member end locations were utilized to simulate the DSC support assembly connections to the HSM.

A. DSC Support Assembly Dead Weight Analysis

For the dead weight analysis, the total weight of the DSC was applied to the T-section support rails. The dead weight of the DSC support assembly was also included in this analysis.

B. DSC Support Assembly Operational Handling Analysis

For the NUHOMS-24P system operational handling loads, a sliding force of 20,000 pounds was applied axially to the DSC support rails to account for the sliding friction between the DSC shell and the support rails.

C. DSC Support Assembly Thermal Analysis

Slotted holes are used for the DSC support assembly bolted connections to permit free thermal expansion. Adequate space is provided between the DSC support assembly cross members and the HSM wall connections, to prevent temperature expansion loads from developing. The allowable stresses for the DSC support assembly material (A36 carbon steel) were selected at 600°F to conservatively envelope the range of design basis ambient temperature cases.

The bolts are installed "snug tight" in accordance with AISC (8.45) requirements with a lock nut added to ensure that the bolts remain in place. The connection details permit free thermal expansion of the DSC support assembly and thus prevent thermal stresses from occurring. The bolted connections are bearing connections and do not take credit for frictional forces to fulfill their design function.

D. Evaluation of DSC Support Assembly Results

The results of the DSC support assembly analysis are tabulated in Table 8.1-8. Maximum normal operating loads calculated for the DSC support system end connections are listed in Table 8.1-9. Also shown in Table 8.1-9, are the calculated end connection loads which result from the effects of postulated vertical and horizontal seismic accelerations. The maximum calculated DSC support assembly deflections for normal and off-normal operating loads are shown in Table 8.1-9a. Specific information on the DSC support assembly seismic loads can be found in Section 8.2.3. Details of the end connection analysis are presented in Section 8.2.10 using the aforementioned design load analysis results.

8.1.1.5 HSM Loads Analysis

As discussed in Section 4, the NUHOMS modular storage concept has the flexibility of arranging modules in an interconnected array with the adjoining HSM concrete poured monolithically or constructed as individual units. The exact number of HSMs in an array is dependent on plant specific needs and economic trade-offs and can range in size from a single stand-alone HSM to a 2x10 array of HSMs.

In order to qualify the design for a range of NUHOMS-24P ISFSI applications, both a single free standing HSM and HSMs joined together to form arrays up to 2x10 were evaluated. The HSM structural analyses included an evaluation of normal operating, off-normal, and postulated accident loads for these HSM configurations. The frame action of the HSM walls and roof slab was considered to be the primary structural system for transverse loads. The selection of single and multiple module configurations provides a conservative methodology for evaluating the response of the HSM structural elements under various static and dynamic loads.

The HSM external walls and roof slab thicknesses were established on the basis of radiological shielding requirements. As such, all other thickness requirements such as the minimum barrier thickness requirements for tornado generated missiles specified by NUREG-0800, Section 3.5.3 (8.19) are bounded. The ultimate strength method was used to evaluate stresses in the HSM reinforced concrete walls, roof, and floor slab. The HSM reinforcement was designed to meet the minimum flexural and shear reinforcement requirements of ACI 349-85 (8.20). The available design strength exceeds that required for the factored design loads specified in Section 3.2 of this report. A typical reinforcement layout for a single free standing HSM is shown in Figure 8.1-9 and is comparable to those previously reviewed and approved for the NUHOMS-07P design. HSM construction details such as construction joints and reinforcing bar splices will be detailed on the construction drawings for a plant's ISFSI.

The HSM structural analyses considered HSM ISFSI's ranging in size from a single stand-alone module up to the maximum array size of 2x10. The reinforced concrete HSM was evaluated assuming simple frame behavior using the analytical models shown in Figures 8.1-10 and 8.1-10a. The various normal operating loads were applied to the analytical models and the HSM internal forces and moments calculated by performing a linear elastic finite element analysis. The HSM finite element model results are applicable to both side-by-side or back-to-back model arrangements. A range of plausible DSC loading patterns were analyzed (i.e., HSM's containing DSC's) for each array size to establish the worst case design loadings. The analyses showed that the single HSM provides the governing case for load combinations containing tornado wind and missile loads, seismic loads and flooding conditions. The postulated failure mode for each of these cases is sliding or overturning of the single stand-alone HSM unit which enveloped that of the 2x10 array. The analyses also showed that the thermal loads for a 2x10 array control the reinforcement requirements for the HSM walls, roof, and foundation members for all intermediate array sizes including those of a single HSM. A description of the individual loads and load analyses are provided in the following sections.

The HSM model used for the analysis of a single, stand-alone HSM for the bounding environmental loads (design basis tornado, earthquake, and flooding effects) is shown in Figure 8.1-10. The load inputs for this analysis are described in Sections 8.2.2 through 8.2.4. The computer model used for the thermal analysis of a 2x10 array of HSMs (maximum HSM unit size) is shown in Figure 8.1-10a. The load inputs for this analysis are described in Section 8.1.1.5, Paragraph C, Section 8.1.2.2, Paragraph A, and Section 8.2.7.2. Since the maximum 2x10 HSM array size was the controlling array size for the thermal load conditions. The remaining HSM loads such as dead loads, live loads, and creep effects, were evaluated using the computer model for the 2x10 array. The output from these models is summarized in Tables 8.1-10, 8.2-3, and 8.2-10. The resulting design analysis of the HSM to determine the reinforcement requirements is presented in Section 8.1.1.5, Paragraph E.

A. HSM Dead Load and Live Load Analyses

The dead weight of the HSM plus the loaded DSC and the DSC support assembly weights were applied to the analytical frame model as shown in Figure 8.1-11. The HSM models shown in Figures 8.1-10 and 8.1-10a represent a one foot thick slice of the HSM. The concentrated loads shown on Figure 8.1-11 conservatively envelope the dead weight of the concrete walls and DSC support structure reaction loads. The total weight of the concrete wall plus the internal concrete plenum were applied to the top of the wall (7.4k) and the average DSC support reaction (80 k divided by 6 supports = 13.3k) was applied at the embedded support elevation. The applied DSC support reaction moment (286.7 k.in) was calculated from the center of the wall plus 3.5 inches for the beam seat angle. The resulting calculated maximum dead load shears and moments are tabulated in Table 8.1-10. A live load of 200 psf was applied to the HSM roof to conservatively envelope all postulated live loads, including snow and ice. The resulting calculated maximum live load shears and moments are tabulated in Table 8.1-10.

B. Concrete Creep and Shrinkage Analyses

ACI 349-85 Section 9.2.2 states that "where the effects of... creep or shrinkage may be significant, they shall be included with the dead load...". Since creep is mainly dependent on elastic strain due to dead loads, the loading contribution from creep is minimal as the dead loads are small in relation to the capacity of the massive HSM frame formed by the walls and roof. The creep strain was conservatively calculated using the ultimate creep strain value suggested by Wang and Salmon (8.21).

$$\epsilon_c' = C_u \epsilon_c \qquad (8.1\text{-}30)$$

Where: ϵ_c' = Creep strain in./in.

C_u = Ultimate creep strain or ratio of creep to initial strain from dead weight = 2.35

ϵ_c = Initial strain from dead weight = 3E-5 in./in.

Therefore:

$$\epsilon_c' = 4.6\text{E-}5 \text{ in./in.}$$

Shrinkage of the HSM concrete was conservatively calculated using an ultimate shrinkage strain value suggested by Wang and Salmon. Additionally, since shrinkage is significantly affected by the surface area to volume ratio, the ultimate shrinkage strain value was reduced according to the method recommended by Fintel (8.22). The combined shrinkage strain is:

$$\epsilon_s = \epsilon_{sh\mu}\alpha \qquad (8.1\text{-}31)$$

Where:

ϵ_s = Shrinkage strain (in./in.)

$\epsilon_{sh\mu}$ = Ultimate shrinkage strain = 8E-4 in./in.

α = Volume to surface area reduction = 0.5 (conservative)

Therefore: ϵ_s = 4E-4 in./in.

For determination of moments and shears in the HSM modules due to creep and shrinkage, the total strain was converted to an axial change in length across the roof of a single HSM.

$$\Delta L = L\ (\epsilon_s + \epsilon_c')$$

Where:

ΔL = Axial length change (in.)

L = Length from center to center of the HSM walls = 104 in.

ϵ_s = 4E-4 in./in.

ϵ_c' = 4.8E-5 in./in.

The axial change in length of the HSM roof, ΔL = 0.044 inch, per module, was applied to the HSM analytical models and the resulting induced shears and moments calculated. For load combinations in which creep and shrinkage increased the calculated shears and moments the effects were combined with the calculated dead load values, in accordance with ACI 349-85, and as reported in Table 8.1-10. This analysis is conservative since shortening due to creep and shrinkage occurs gradually over a period of time, and the effects will be lessened by plastic creep flow and microcracking of the members. The PCI design handbook (8.23) suggests that the calculated creep and shrinkage shortening values be reduced by a factor of three to five for design. This analysis conservatively ignored this permissible reduction.

C. HSM Thermal Analysis

To evaluate the effects of thermal loads on the HSM, heat transfer analyses for a range of normal ambient temperatures were performed and the limiting thermal gradients and temperature values at various locations in the HSM determined. A more detailed description of the heat transfer analyses is provided in Section 8.1.3. Structural analyses of the HSM for the maximum calculated floor, wall and roof temperature loads listed in Table 8.1-9b were performed for the enveloping 2x10 HSM array using the analytical model shown in Figure 8.1-10a. As shown in Table 8.1-9c a range of DSC loading patterns were evaluated to maximize the HSM floor, wall and roof shears and moments. The results of these analyses are summarized in Table 8.1-10. The basis for the HSM thermal analysis is discussed further in the following paragraphs.

ACI 349-85, Appendix A, provides a general methodology for designing reinforced concrete structures subjected to thermal loads. The commentary to this Appendix, Section A.3.3, defines a range of approaches utilized in analysis of thermal loads. The most conservative approach neglects the self-relieving nature of the thermal loads (relief is obtained by the occurrence of thermal cracking when the concrete modulus of rupture is reached). For the thermal analysis of the HSM for normal operating conditions, the thermal loads were calculated for the full uncracked cross section of the HSM walls, roof and floor.

The commentary to ACI 349-85, Appendix A also contains a method for evaluating the temperature distribution across a section. The steady state non-linear temperature distribution through the thickness was converted into an equivalent linear temperature distribution. The linear temperature distribution was then separated into a pure through thickness gradient (ΔT) and a uniform ΔT equal to the difference between the mean and stress-free temperature. The calculated thermal gradient and mean temperature difference were conservatively applied to the linear elastic finite element beam model of the HSM as discussed above.

To account for the seasonal effects of ambient temperature fluctuations on the outside surface of the HSM, an average daily ambient temperature range of 0°F (winter) to 100°F (summer) was considered in the heat transfer analysis for normal operating conditions. Analyses were performed for ambient temperatures of 0°F, 70°F, and 100°F, to determine the limiting design conditions for the HSM. As described in Table 8.1.9b the temperature gradients for varying DSC positions in the HSM array were simulated by adjusting the boundary conditions of the HSM HEATING6 model. For the HSM roof slab the results of the HSM heat transfer analysis for normal operating conditions for a life time average ambient temperature of 70°F and with solar heating effects included are shown in Figure 8.1-3. For the HSM roof slab the results indicate an average concrete temperature of 128°F and a maximum local inside surface temperature of 144°F. Maximum thermal gradients of 32°F for the roof slab and 30°F for the HSM walls were also observed.

A maximum inside surface temperature of 144°F was observed near the center of the HSM roof slab. Assuming direct conduction paths through the heat shield anchor bolts embedded in the HSM concrete, the local concrete temperature around the bolts would be 169°F. These concrete temperatures are within the ACI 349-85, Section A.4.1 acceptable range, of 150°F to 200°F.

The results of the heat transfer analysis for the 100°F ambient temperature case (maximum average summer temperature), with a solar heat flux (62 Btu/hr. ft.^{2}) indicated an average concrete temperature of 160°F and a maximum local inside surface temperature of 179°F as shown in Figure 8.1-3a. The thermal gradients through the concrete thickness, which are the primary cause of HSM thermal stresses, were 39°F for the roof slab and 26°F for the HSM side walls. The results of the thermal analysis for normal operating conditions are tabulated in Table 8.1-12. The HSM concrete temperatures associated with the extreme ambient condition of 125°F, and the postulated accident condition of a total ventilation air inlet and outlet blockage were also evaluated, as discussed in subsequent report sections. The maximum local concrete surface temperature of the HSM roof for this short duration condition was 368°F.

For conservatism and consistency with the philosophy of the ACI 349-85, Section A.4.3, the strength properties of the concrete and reinforcing steel used in the HSM structural analysis were consistent with the postulated temperature range for each load case. For all normal operating load cases the concrete and reinforcing properties were assumed to be equal to the specified values (f'_c = 5000 psi for concrete and f_y = 60,000 psi for rebar). For the 125°F extreme ambient, off-normal, and postulated accident conditions the material properties were assmed to be equal to those specified in Table 8.1-2 at 400°F i.e., f'_c = 4500 psi and f_y = 51,000 psi. The use of material properties at 400°F is extremely conservative since the maximum concrete temperatures calculated are well below this value even for short term conditions which may result from postulated accident events such as blockage of the HSM vents. Temperature dependent mechanical properties of concrete and reinforcing steel utilized for the HSM are presented in Table 8.1-2.

The question of adverse effects on reinforced concrete in relation to its strength properties, its deterioration and subsequent spalling and crack formation, due to sustained elevated temperatures is discussed in detail in Appendix D of this report, and is summarized in the following paragraphs. As concluded in Appendix D, and based on the criteria specified by ACI 349-85 and tests performed in recent years, no adverse effect on reinforced concrete strength, particularly in terms of concrete deterioration and spalling is anticipated for the range of HSM concrete temperatures calculated. The effect of elevated short and long term temperatures on normal concrete structures are discussed further in the following paragraphs.

The effects of elevated short and long term temperatures on concrete structures has been the subject of much research in the U.S. and European communities. The findings of these studies and tests have been reported in a number of publications. A review of some reports, particularly references (8.11), (8.22), and (8.24) show that the physical, chemical, and mechanical properties of concrete are not significantly affected at temperatures below 212°F. When heated beyond 212°F, unsealed concrete begins to lose its internal free moisture, causing loss of weight and shrinkage at higher temperatures. As this process continues the chemically bonded water within the cement paste is released which further accelerates the loss of weight and volume. The chemical and physical change in concrete affects its mechanical properties. Mechanical properties primarily affected by long-term sustained temperatures above 212°F are modulus of elasticity, compressive strength, tensile strength and the Poisson ratio. Data on these properties for normal Portland cement concrete as a function of temperature were extracted from various publications.

Figure 8.1-13 gives a comparison of strength losses with sustained high temperatures. Figure 8.1-14 shows the strength behavior of three types of aggregate commonly used. As can be observed from these curves, the compressive strength of concrete for a short duration temperature rise is unaffected. However, long-term heating at temperatures above 250°F reduces the compressive strength of the concrete. For the operating temperature range of the NUHOMS-24P HSM shown in Table 8.1-9b, no loss in compressive strength is expected. During a postulated blocked vent accident the HSM concrete is heated beyond the normal range in localized areas for a short period of time. The concrete strength properties used in analysis of the off-normal and accident conditions were reduced accordingly to reflect the results of these findings.

In Figures 8.1-15 and 8.1-16, the effects on modulus of elasticity of Portland cement concrete exposed to short and longterm elevated temperatures are shown. These figures indicate substantial loss of elastic modulus at temperatures beyond 400°F. No loss in properties is expected to occur for temperatures below 150°F which bounds the normal operating temperature conditions for the HSM. The modulus of elasticity determines the flexural rigidity of the structure, and substantial loss of modulus of elasticity will cause excessive flexural deflections in long span members. Since the NUHOMS concrete walls and roof slab are deep, short span members, flexural deflections are negligible. Furthermore, the decrease in modulus of elasticity will decrease the calculated concrete thermal stresses; however, for sake of conservatism, this decrease was not considered in the analysis.

Other effects of the mechanical properties of concrete (i.e., reduction in tensile strength, creep, and shrinkage due to elevated temperatures) were investigated and found to have an insignificant effect on the design of the HSM.

Table 8.1-12 shows that the maximum HSM temperature for the life time average ambient temperature of 70°F is less than the ACI limit of 150°F. As discussed above no reduction in concrete strength results from a short term temperature rise such as would occur for maximum and extreme ambient temperature conditions, or blockage of the HSM vents. Even for a long term temperature increase of up to 250°F, the resulting reduction in concrete strength is minimal. As can be seen from Table 8.1-12, the HSM concrete temperatures are much less than 250°F for all cases but the enveloping blocked vent case, which is postulated to occur for a period of 48 hours or less.

Coupled with the conservative reductions in concrete material strength used in the HSM design calculations for the 125°F off-normal and accident conditions, the design criteria utilized is adequate to ensure that the NUHOMS-24P HSM will perform its intended safety function for all design conditions. In addition, the NUHOMS-24P HSM concrete temperatures have decreased by an average of 15% compared with those previously calculated for the NUHOMS-07P.

D. Radiation Effect on HSM Concrete

The accumulated neutron flux over the 50 year service life of the HSM is estimated to be 1.2E14 neutrons/cm^2. From the study by Hilsdorf, Kropp, and Koch (8.25), the compressive strength and modulus of elasticity of concrete will not be affected by a neutron flux of this magnitude.

The gamma energy flux deposited in the HSM concrete is 6.8E10 MeV/cm^2-sec. or 1.1E-2 watt/cm^2. According to ANSI/ANS-6.4-1977 (8.26), the temperature rise in concrete due this level of radiation is negligible.

E. HSM Design Analysis

For comparison with the normal operating condition loads factored to include the ACI Codes strength reduction factors the flexural and shear strength capacities of a typical concrete section were calculated using the ultimate strength method of ACI 349-85. The results of the analyses and comparison with the HSM bending and shear capacities are shown in Table 8.1-10.

The ultimate moment capacity of a typical 12 inch wide section of the HSM wall, roof, or floor for normal operating load combinations is:

$$M_u = \phi A_s f_y \left(d - \frac{A_s f_y}{1.7 f'_c b} \right) \qquad (8.1\text{-}32)$$

Where:

M_u = Ultimate moment capacity k-in./ft.

b = 12 in., Width of section

d = 32.0 in., Depth of the section minus three inches of cover

ϕ = 0.9, Flexural reduction factor

f'_c = 5000 psi, Design compressive strength of concrete at 150°F

A_s = 2.54 in.^{2}, Area of reinforcing steel (#10 bars at six inch spacing)

f_y = 60,000 psi, Steel design strength at 150°F

Therefore: M_u = 4180 k-in./ft. for the 3'-0" thick HSM roof and end walls and 2540 k-in./ft. for the 2'0" thick interior walls in an HSM array.

The ultimate shear capacity of the HSM concrete for a typical 12 inch wide strip of the 3'-0" thick HSM roof and end walls is:

$$V_u = 2\phi \sqrt{f'_c}\, bd$$

Where all parameters as defined previously except:

ϕ = 0.85; Shear reduction factor

Therefore: V_u = 46.1 kips

For concrete sections containing shear reinforcement, the allowable shear capacity (V_c) was calculated in accordance with Section 11.8.6 of ACI 349 using the formula:

$$V_c = \left(3.5 - 2.5 \frac{M_u}{V_u d} \right) \left(1.9\sqrt{f'_c} + 2500\rho_w \frac{V_u d}{M_u} \right) b_w d \qquad (8.1\text{-}32a)$$

The capacities of the HSM floor and interior walls were calculated in a similar manner.

8.1.1.6 HSM Door Analyses

The access opening for transferring the DSC into and out of the HSM is protected by a three inch thick solid steel door plus two inches of solid neutron shielding material. Steel angle sections are attached to embedments in the HSM front wall to form guides for sliding the HSM door vertically. The door and support frame are shown in Figure 4.2-5.

For normal system operation, the only loading the door assembly is subjected to is the dead weight and handling loads resulting from opening and closing of the door during DSC transfer operations. To approximate the worst normal operating loads, it was assumed that three times the dead weight of the door acts on the bottom angle section of the door frame. The factor of 3.0 applied to the dead weight of the HSM door accounts for the additional inertia forces and sudden lifting loads associated with opening and closing of the door. This factor was arrived at by taking guidance from CMAA Specification #70 prepared by the Crane Manufacturers Association of America, Inc. (1983). In CMMA-70, the maximum safety factor specified to account for inertia forces and sudden lifting loads is 1.5. An added safety factor of 2 was conservatively applied to arrive at the factor of 3.0 used in the analysis. The corresponding stress analysis of the HSM door angle section are as follows.

The maximum bending moment on the angle section supporting the door is:

$$M_{max} = bwL \qquad (8.1\text{-}33)$$

Where:

M_{max} = Maximum bending moment (lb.-in.)

b = Length of angle resisting bending = 88 in.

w = Weight of door per inch times 3 = 274 lb./in.

L = Moment arm to base of angle = 3.13 in.

Therefore:

$$M_{max} = 75{,}500 \text{ lb.-in.}$$

The section modulus provided is:

$$S = \frac{bh^2}{6} = 8.25 \text{ in.}^3$$

Where: S = Section modulus at line of bending (in.^{3})

h = Thickness of angle = 0.75 in.

The bending stress on the angle is:

$$f_b = \frac{M_{max}}{S} = 9.2 \text{ ksi}$$

The allowable bending stress for a rectangular A-36 steel section is:

$$F_b = 0.66\ S_y = 0.66 \times 36.0 = 24.0 \text{ ksi}$$

Where: F_b = Allowable bending stress per AISC

S_y = Yield strength of A-36 steel at 100°F = 36.0 ksi

The maximum normal operating shear stress on the angle can be found from:

$$f_v = \frac{3V_{max}}{2A_v} = 0.55 \text{ ksi} \qquad (8.1\text{-}34)$$

Where: f_v = Maximum shear stress

V_{max} = Weight of door times 3 = 24.1 kips

A_v = Shear area resisting load = 66 sq. in.

The allowable shear stress for the angle is:

$$F_v = 0.4S_y = 0.4 \times 36.0 = 14.40 \text{ ksi}$$

Where: F_v = Maximum allowable shear stress per AISC

Clearly the normal operating loads on the door frame are much lower than the design allowables.

Similarly, the embedded anchor bolts for the HSM door frame were designed in accordance with ACI 349-85, Appendix B. The governing design load combination for the HSM door embedded anchor bolts is the dead load plus seismic load combination. The dead load consists of the weight of the door and the seismic load consists of the seismic accelerations acting on the door. The dead load of the door produces shear on the anchor bolts and the seismic loads produce shear and tension on the anchor bolts. The maximum tensile stress on the anchor bolts is 0.3 ksi while the maximum shear stress on the anchor bolts is 2.3 ksi. The ACI 349 allowable tensile stress and shear stress is 23.9 ksi and 14.6 ksi respectively.

8.1.1.7 HSM Heat Shield Analysis

The general arrangement of the HSM heat shield assembly is shown on the drawings in Appendix E of this report. (Additional details are provided in Figure 4.2-7.) This assembly was designed to reduce the HSM surface temperatures due to the heat rejected by the spent fuel assemblies during normal operating conditions, off-normal conditions, and postulated accident conditions. The only loading that the heat shield assembly is subjected to during normal operation is its own dead weight. Since slotted holes are used at the bolted connections to the HSM no thermal expansion loads will be experienced by the stainless steel heat shield.

The only loading that the HSM heat shield assembly experiences during off normal conditions or a postulated accident is the inertia force associated with a seismic event. The heat shield and the embedded anchor bolts were analyzed for normal , off-normal, and accident loads. The heat shield was conservatively analyzed as a series of simply supported beams with a span equal to the distance between two adjacent bolts in the longitudinal direction. Also, in lieu of a frequency analysis to establish the dynamic amplification factor (DLF), the maximum DLF of 4.25 for 2% damping was selected from the design response spectra curve.

Using these conservative assumptions, the maximum bending stress obtained for the combined effect of dead weight and seismic load was 5.5 ksi, which is well within the allowable limits of 0.75 F_y or 20 ksi at 600°F operating temperature. Tensile and shear stresses of 0.91 ksi and 0.76 ksi, respectively, were calculated for the anchor bolts. These values are well within the allowable limits for the bolts. The calculated stresses show that the heat shield will be capable of withstanding any normal, off-normal, or postulated accident condition.

8.1.1.8 HSM Seismic Restraint for DSC

The general arrangement of the HSM seismic restraint for the DSC is shown on the drawings in Appendix E of this report. Additional details are provided in Figure 4.2-4 and Section 8.2.3.2.

8.1.1.9 NUHOMS-24P Transfer Cask Analysis

The NUHOMS-24P transfer cask was evaluated for normal operating condition loads including:

1. Dead Weight Load
2. Thermal Loads
3. Handling Loads

The NUHOMS-24P transfer cask is shown on the design drawings contained in Appendix E of this report. Table 8.1-10a summarizes the transfer cask calculated stresses for the normal operating loads. The methodology used to evaluate the transfer cask for the effects of normal operating loads is described in the following paragraphs. The analytical results and comparisons with the acceptance criteria defined in Section 3.2 are also presented in this section.

A. Transfer Cask Dead Weight Analysis

The effects of dead weight for a loaded NUHOMS-24P transfer cask were evaluated for two cases. The first case evaluated was for the transfer cask hanging vertically from the two lifting trunnions, and loaded to its maximum capacity of 90,000 lbs. Including the self weight of the transfer cask, this gives a total dead weight of 190,000 lbs. This load is the same as the normal handling loads evaluated in Paragraph B below.

The second dead weight load case evaluated for the transfer cask includes the loaded transfer cask resting in a horizontal position on the transport skid/trailer. In this position, the weight of the cask is shared between the lower tilting trunnions and the upper lifting trunnions resting in the pillow block supports of the support skid. The maximum dead load stresses are shown in Table 8.1-10a. The local stresses around the trunnions are included in the normal handling load case described in Paragraph B.

B. Transfer Cask Normal Handling Loads Analysis

The major components of the NUHOMS-24P transfer cask affected by the normal handling loads are the structural shell including the top and bottom cover plates, the upper and lower trunnions, the upper trunnion assembly insert plates, and the structural shell local to the trunnions. As described for the dead weight analysis, there are two normal operating cask handling cases which form the design basis for the NUHOMS-24P transfer cask. These cases are illustrated in Figure 8.1-17a are summarized as follows:

(i) The transfer cask is oriented in the vertical position, loaded to its maximum weight of 190,000 lbs, hanging by the upper lifting trunnions, and being moved in an area of the plant which requires conformance with the requirements of ANSI N14.6 (8.53). The upper trunnions and their attachment weld allowable stresses are restricted to less than one sixth of the material yield strength, or one tenth of the material ultimate strength for critical lists governed by ANSI N14.6. Allowable stresses for the remaining transfer cask components including the lower tilting trunnions are governed by the requirements of the ASME Code. The cask handling load is assumed to be shared equally between the upper two trunnions. An additional load factor of 15% is conservatively applied to account for the inertial effects of crane hoist motions in accordance with CMAA #70 recommendations. The transfer cask was 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 was conservatively assumed to be 100 kips per trunnion plus the 15% allowance, or 115 kips, acting vertically with a moment arm measured, from the center of the yoke lifting plate to the middle surface of the transfer cask structural shell.

As shown by the calculations presented in Appendix C.1, the maximum calculated upper trunnion stresses for this load case are 6.3 ksi at the junction between the trunnion shoulder and the trunnion sleeve attached to the structural shell insert plate. This compares with the ANSI N14.6 allowable stress of 13.5 ksi for the SA564 Gr. 630 PH trunnion material. The maximum weld stress is 7.0 ksi. The ANSI N14.6 allowable weld stress is 8.0 ksi. As shown in Appendix C.1, the maximum calculated stress in the lower trunnion is 5.6 ksi, and the maximum weld stress is 12.6 ksi. These stresses compare with the ASME Code allowable value of 20 ksi.

The maximum stress in the transfer cask structural shell occurs in the thickened insert plate at the junction with the two inch thick sleeve. As described in Appendix C.1, stresses in the insert plate and the structural shell were calculated using the WRC Bulletin No. 297 (8.54) method. The maximum calculated stress intensity in the cask structural shell is 40.9 ksi compared with an ASME Code allowable stress intensity value of 67.5 ksi.

(ii) During transport of the DSC from the plant's fuel building to the HSM storage location, the NUHOMS-24P transfer cask is oriented in a horizontal position, and is firmly secured to the transport skid/trailer. During this condition the cask/DSC is loaded with fuel with the DSC top shield plug and the DSC and cask top cover plates in place. The resulting trunnion loads were developed by taking the summation of moments about a horizontal axis to account for the fact that the upper trunnions are closer to the horizontal center of gravity of the cask and thus carry a greater part of the total cask weight compared with the lower tilting trunnions. The transfer cask is supported in pillow block supports at two locations; the lower tilting trunnions near the bottom end of the cask, and the lifting upper trunnions supports near the top end of the cask. The allowable stresses for the onsite transfer load cases are governed by the ASME Code. The maximum postulated ASME Code upper lifting trunnion load is 118 kips while the critical load for the structural shell insert plate is a combination of the 59 kips dead load acting vertically, plus a postulated lateral load of ± 1g or 118 kips acting radial to the shell. The loads from this case envelope the design basis transport operation loads of ±0.5g simultaneously applied in three directions to account for vibratory motion loads and start/stop loads which may occur during transport. The design loads for the lower tilting trunnion were developed in a similar manner and are given in Appendix C.1.

During transfer of a DSC from the transfer cask to and from the HSM a mechanical connection is made between the cask and HSM to prevent any relative motion. This connecting device functions by firmly securing the transfer cask lifting trunnions to embedded anchor points in the HSM front wall. The maximum load exerted on the transfer cask lifting trunnion is equal to one half the maximum hydraulic ram load, or 40 kips. This load magnitude is much less than the design basis handling loads described above and is therefore enveloped by the calculated stresses reported for that case.

The maximum calculated upper lifting trunnion stress for the transport load case is 3.6 ksi and occurs at the junction of the trunnion shoulder and the sleeve. This compares with an ASME Code allowable stress of 33.8 ksi. The maximum calculated weld stress is 6.8 ksi compared with an ASME Code allowable stress, of 45 ksi. The maximum calculated lower tilting trunnion stress is 5.6 ksi compared to the ASME Code allowable of 20 ksi. The maximum weld stress intensity was 12.6 ksi and the maximum cask structural shell stress intensity was 67.0 ksi compared with ASME Code allowable values of 20.0 ksi and 67.5 ksi respectively.

C. Transfer Cask Normal Operating Thermal Analysis

The heat transfer analyses of the NUHOMS-24P transfer cask are presented in Section 8.1.3 of this report. Analyses were performed for the normal operating ambient temperature range of 0°F to 100°F to establish the through wall thermal gradients shown in Figures 8.1-3b and 8.1-3c. A circumferential temperature gradient was also calculated with the transfer cask secured to the transport trailer/skid. This gradient includes a temperature increase for solar heat flux.

The nominal room temperature gaps, of 0.5 inch axially and 0.375 inch radially, between the DSC and the transfer cask inner cavity were established to ensure that for the worst case tolerance buildup and differential temperatures that the DSC will slide in/out of the transfer cask without binding. Thermal stresses due to the differential expansion of the dissimilar materials; namely stainless steel, carbon steel, lead, and solid neutron shielding material, were evaluated. The analyses for the normal operating thermal loads are summarized in the paragraphs which follow.

Calculations for the combined effect of the worst case radial thermal gradients, and the circumferential temperature variation, were performed using the combined transfer cask/DSC axisymmetric finite element ANSYS models shown on Figures 8.2-6 and 8.2-7 and described in Section 8.2.5.2, Paragraph B (i). In addition, the ANSYS models were also used to evaluate the effects of differential expansion of the dissimilar materials. The temperatures associated with the radial and axial thermal gradients were input to the analytical model as discrete element temperatures and the resulting induced thermal stresses calculated. Transfer cask stresses due to axial growth of the cask are minimized by the design of the transport trailer/skid pillow block support system.

The results of these analyses are shown in Table 8.1-10a.

D. Transfer Cask Analyses of Results

The results of the transfer cask analyses for normal operating loads were combined to obtain stresses for the associated load combinations which are compared to the appropriate allowable stresses, as discussed in Section 8.2.10.

8.1.2 Off-Normal Load Structural Analysis

Table 8.1-1a shows the off-normal operating loads for which the NUHOMS safety-related components are designed. This section describes the design basis off-normal events for the NUHOMS-24P system and presents analyses which demonstrate the adequacy of the design safety features of a NUHOMS system.

For an operating NUHOMS system, off-normal events may involve the site specific aspects of fuel loading, cask handling, trailer towing, alignment and other operational events. For this topical report two off-normal events are defined which bound the range of off-normal conditions.

The limiting off-normal events are defined as a jammed DSC during loading or unloading from the HSM and the extreme ambient temperatures of -40°F (winter) and +125°F (summer). These events envelope the range of expected off-normal structural loads and temperatures acting on the DSC, transfer cask, and HSM.

8.1.2.1 Jammed DSC During Transfer

The interfacing dimensions of the top end of the transfer cask and the HSM access opening sleeve are specifically designed so that docking of the transfer cask with the HSM is not possible should gross misalignments between the transfer cask and HSM exist. Furthermore, beveled lead-ins are provided on the ends of the transfer cask, DSC, and DSC support rails to minimize the possibility of a jammed DSC during transfer. Nevertheless, it is postulated that if the transfer cask is not accurately aligned with respect to the HSM, the DSC binds or becomes jammed during transfer operations. Based on the dimensions of the DSC, transfer cask, and HSM, the maximum misalignment of the sliding surfaces is limited by operating procedures to 0.125 inches or less. Assuming a worst case misalignment in positioning and docking the transfer cask with the HSM access opening sleeve, the maximum possible misalignment which would permit transfer of the DSC to occur is 0.25 inches. Although unlikely, any greater misalignment may cause axial sticking and/or a rotation of the DSC to occur which may result in a binding condition.

A. Detection of the Event

When a jam of the DSC occurs during transfer the hydraulic pressure in the ram will begin to increase. When the hydraulic pressure corresponds to a force on the DSC equal to 25% of the DSC loaded weight, the DSC will be presumed to be jammed. The maximum ram push/pull forces are limited automatically by features in the ram system design to a maximum load equal to 25% of the DSC loaded weight. Override controls are available to the operator to increase the ram force up to its maximum design load, equal to the DSC loaded weight, or to interrupt the transfer operation at any time.

B. Axial Sticking of the DSC

The DSC has three-to-one beveled lead-ins on each end which are designed to avoid binding or sticking on small (<0.25 inch) obstacles. The transfer cask and the DSC support rails inside the HSM, are also designed with lead-ins to minimize binding or obstruction during DSC transfer. The off-normal handling load event postulated to occur assumes that the leading edge of the DSC becomes jammed against some immovable feature because of gross misalignment of the transfer cask.

During the transfer operation, the force exerted on the transfer cask and the DSC by the hydraulic ram is that required to first overcome static, then sliding friction of the DSC and the transfer cask or DSC support rail sliding surfaces. If motion is prevented, the hydraulic pressure will increase, thereby increasing the force on the DSC until the system pressure limit is reached. This limit will be controlled so that adequate force is available to overcome variations in surface finish, etc., but is sufficiently low to ensure that component damage does not occur. To overcome potentially higher resistance loads due to binding of the DSC in either the transfer cask or the HSM, the maximum ram force is designed to be equal to the weight of the loaded DSC. This force corresponds to a coefficient of friction of one and is the design basis for the hydraulic ram system. This postulated loading condition is illustrated in Figure 8.1-18.

The resulting ram load acting on the DSC grapple ring assembly and bottom cover plate were analyzed as follows:

The DSC bottom cover plate and the grapple ring assembly were subjected to a maximum force of 80,000 pounds. The method of analysis was the same as described in Sections 8.1.1.1 and 8.1.1.2. The maximum bending stress intensity calculated for the DSC bottom cover plate is 6.5 ksi. This stress is well within the ASME Code allowable limit.

It was conservatively assumed that the force created by the jammed DSC condition will produce a force-couple of magnitude F x R, where: F is the imposed force of 80,000 pounds and R is 33.625 inches, the outside radius of the DSC shell. Thus, a moment of 2670 in.-kip is produced in the DSC shell. The gross section modulus available to resist this bending moment is 2165 in^3. Thus, the DSC shell stress due to the 2690 in.-kip moments is:

$$S_{mx} = \frac{M}{S} \tag{8.1-35}$$

Where: M = 2690 in.-kip, Bending moment

S = 2165 $in.^3$, DSC section modulus

Therefore: S_{mx} = 1.24 ksi

This magnitude of stress is negligible when compared to the allowable membrane stress of 22.4 ksi.

For a jammed DSC, a ram load of 80,000 pounds was postulated to act on the DSC support rail inside the HSM in the most critical location. At the same time, a concentrated force of one half of the DSC weight was assumed to act vertically at mid span of the DSC support rail member. The results of this analysis are reported in Tables 8.1-7a and 8.1-8.

C. Binding of the DSC

If axial alignment within system operating specifications is not achieved, it may be possible to pinch the DSC shell as shown in Figure 8.1-19. The pinching force acting on the DSC body and the transfer cask inner liner is directly proportional to the angle of rotation. The maximum possible inclination angle established by various conservative geometric and operational assumptions is less than one degree. If this angle is conservatively assumed to be one degree, then the pinching force will be the product of the maximum ram loading of 80,000 pounds and the sine of the angle, or 1,400 pounds. This force is assumed to be distributed around the circumference of the DSC shell and either the transfer cask or HSM sleeve as a cosine distribution.

The 1,400 pound load was conservatively assumed to be applied as a point load at a location away from the ends of the cask or DSC. The resulting maximum stresses are given by Table 31, Case 9a of Roark (8.16) as:

Membrane stress: $\sigma = \dfrac{0.4P}{t^2}$

Bending stress: $\sigma' = \dfrac{2.4P}{t^2}$

Therefore, the maximum stress is:

$$\sigma + \sigma' = \frac{2.8P}{t^2}$$

For the DSC shell, t = 0.625 inch. For the cask inner liner, t = 0.5 inch. Substituting for t and P equal to 1400 pounds, the maximum extreme fiber stresses in the DSC shell and cask inner liner are 10.0 ksi and 15.7 ksi respectively. These bounding stresses are well within the ASME Code Service Level C allowable of 39.1 ksi for an off-normal jammed DSC event.

The tangential component of ram loading under the assumed condition is less than the force of the jammed condition calculated previously in paragraph B and as such is not considered further. The stresses on the DSC are given in Table 8.1-7a and for the DSC support assembly inside the HSM for the jammed condition are reported in Table 8.1-8 of this report.

D. Consequences of Jammed DSC

In both scenarios for a jammed DSC, the stress on the DSC shell and transfer cask inner liner were shown to be much less than the ASME Code allowable stress. Therefore, permanent deformation of the DSC body and cask inner liner will not occur. There is no potential for breach of the DSC containment pressure boundary and therefore, no potential for release of radioactive material.

E. Corrective Action

In both cases, the required corrective action is to reverse the direction of the force being applied to the DSC by the ram, and return the DSC to its previous position. Since no permanent deformation of the DSC or transfer cask inner liner has occurred, the sliding transfer of the DSC to its previous position will be unimpeded. The transfer cask alignment should be rechecked, and the cask repositioned as necessary before attempts at transfer are renewed.

8.1.2.2 Off-Normal Thermal Loads Analysis

As described previously, the NUHOMS-24P system is designed for use at all reactor sites within the continental United States. Therefore, off-normal ambient temperatures of -40°F (extreme winter) to 125°F (extreme summer) was conservatively chosen. In addition, even though these extreme temperatures would likely occur for a short period of time, it was conservatively assumed that these temperatures occur for a sufficient duration to produce steady state temperature distributions in each of the affected NUHOMS-24P components. Each site license applicant should verify that this range of ambient temperatures envelopes the design basis ambient temperatures for that site. The NUHOMS-24P system components affected by the postulated extreme ambient temperatures are the transfer cask and DSC during transfer from the plant's fuel building to the HSM site, and the HSM during storage of a DSC.

The thermal stresses in the various NUHOMS-24P system components due to the off-normal temperatures were calculated in the same manner as described for the normal operating thermal loads. A description of these methods is provided in Sections 8.1.1.2 for the DSC shell, 8.1.1.3 for the DSC internals, and 8.1.1.8 for the NUHOMS-24P transfer cask.

A. HSM Off-Normal Thermal Analysis

As described in Section 8.1.3.1, the maximum HSM temperatures were calculated for the off-normal extreme ambient temperatures of -40°F and 125°F. The resulting maximum HSM concrete temperature calculated for off-normal conditions is 215°F. The maximum calculated temperature gradient was 31°F for the HSM roof and 32°F for the HSM walls. These values are comparable to the maximum temperature gradients calculated for the 100°F normal operating ambient temperature. The short duration peak concrete temperature exceeds the ACI 349-85 recommendations of 200°F by a small amount. However, as discussed in Section 8.1.1.5 the small reduction in concrete strength properties is offset by a compensating reduction in overall structural stiffness, and the mechanical properties for the 125°F off-normal event were conservatively reduced by 10% equivalent to the properties at 400°F. The HSM reinforced concrete design developed for the range of normal operating ambient temperatures is more than adequate for the off-normal temperature conditions.

The DSC support assembly is designed with slotted holes as described in Section 8.1.1.4.C, and therefore the increase in temperature has no affect on the DSC support structure.

B. DSC Off-Normal Thermal Analysis

The off-normal thermal gradients and maximum temperatures for -40°F and 125°F ambient air were developed for the DSC resting in the cavity of NUHOMS-24P transfer cask as described in Section 8.1.1.3. The maximum off-normal calculated surface temperature for the DSC shell was 392°F for an extreme ambient temperature of 125°F. The corresponding internal pressure for the DSC is shown in Table 8.1-4.

The off-normal thermal gradients and maximum temperatures were also developed for the DSC resting in the HSM, as described in Section 8.1.3.2. The maximum off-normal surface temperature calculated for the DSC shell was 305°F for an extreme ambient air temperature of 125°F.

C. NUHOMS-24P Transfer Cask Off-Normal Thermal Analysis

As described in Section 8.1.3.3 the maximum temperatures and associated through wall thermal gradients were calculated for a loaded NUHOMS-24P transfer cask for the off-normal ambient temperatures of -40°F and the 125°F. The temperature gradient for the 125°F extreme ambient temperature case includes an enveloping solar heat flux of 127 Btu/hr.ft.^{2} (8.59) on the top half of the cask outer surface. This results in a maximum calculated temperature of 248°F on the exterior of the transfer cask and a maximum through wall temperature gradient of 10°F for the bounding postulated off-normal cases.

The results of the off-normal thermal analyses shown in Table 8.1-7a and 8.1-10b for each of the NUHOMS-24P system components were combined with the appropriate results from other analyses for the associated load combinations. The resulting stresses and comparisons with allowable stresses are discussed in Section 8.2.10.

8.1.3 Thermal-Hydraulic Analysis

This section of the report describes the thermal analysis of the NUHOMS-24P HSM, DSC and transfer cask. The analytical models of the HSM, the DSC and the transfer cask are described and the calculation results are summarized. The thermophysical properties of the NUHOMS system components used in the thermal analysis are listed in Tables 8.1-5 and 8.1-6. The following evaluations were performed for the NUHOMS-24P system:

1. Thermal Analysis of the HSM
2. Thermal Analysis of the DSC in the HSM
3. Thermal Analysis of the DSC in the NUHOMS-24P Transfer Cask

The NUHOMS-24P components were evaluated for a range of design basis ambient temperatures as follows:

1. Normal Operating Conditions: The system components were evaluated for average ambient temperatures in the range of 0°F minimum (winter) to 100°F maximum (summer). Ambient temperatures within this range were assumed to occur for a sufficient duration to cause a steady-state temperature distribution in the NUHOMS-24P components. For the evaluation of thermal cycling and material properties, fluctuations in the ambient temperature from winter to summer conditions are assumed to occur once per year for the HSM, and six times per year for the transfer cask. The lifetime average ambient temperature for the 50 year service life is taken as 70°F. The "stress-free" temperature for material properties is also assumed to be 70°F.

2. Off Normal and Accident Conditions: The system components were evaluated for the extreme ambient temperatures of -40°F (winter) and 125°F (summer). Should these extreme conditions ever occur, they would be expected to last for a very short duration of time. Nevertheless, these ambient temperatures are conservatively assumed to occur for a sufficient duration to cause a steady-state temperature distribution in the NUHOMS-24P components (a few hours for the transfer cask, several days for the HSM). In addition, for postulated accident conditions the HSM

ventilation inlet and outlet openings are assumed to be completely blocked for a 48 hour period concurrent with the extreme ambient conditions (125°F).

8.1.3.1 Thermal-Hydraulics of the HSM

A. Principles of HSM Cooling System

The HSM is cooled by a natural draft of air entering through the air inlet opening located in the lower front wall of the HSM, and exiting through two air outlet openings located in the front and back of the HSM roof. Cooler air at the prevailing ambient conditions is drawn into a shielded plenum inside the the HSM. The cooler air exits the internal plenum and flows from the bottom of the HSM along the outer DSC surface where it is warmed by the decay heat of the spent fuel inside the DSC. The warmed air flows along the ceiling of the HSM and exits through the two outlet openings in the HSM roof. The HSM drawings contained in Appendix E and Figure 8.1-21 illustrate the HSM vent geometries and flow paths for ventilation air.

The HSM roof is the primary concrete surface conducting heat to the outside environment. For analytical purposes, the interior or common walls of an HSM centered in a group of HSMs, each loaded with a DSC, were assumed to be insulated. For the thermal analysis of an interior HSM with no DSC present in the adjacent HSMs, a two foot thick wall was modeled, the outer surface of which was assumed to be exposed to the prevailing ambient conditions. For the thermal analyses of a single free standing HSM, or an HSM at the end of a multiple module array, a three foot thick wall was modeled, the outer surface of which was assumed to be exposed to the prevailing ambient conditions. The HSM foundation slab is in contact with soil, which was assumed to be at a constant temperature at a combined depth of ten feet. Infiltration, or heat radiation from the HSM access opening door, was conservatively neglected.

The temperature difference (ΔT) and the height difference (Δh) between the bottom of the DSC and the HSM vent outlets creates a "stack effect," to drive air through the HSM. The ventilation air has sufficient velocity to provide adequate cooling for the DSC so that the spent fuel cladding temperature remains below acceptable limits. The ventilation flow paths inside the HSM were designed so that the pressure difference due to the stack effect (ΔP_s) will be greater than the pressure losses due to friction, vent area changes and flow direction changes (ΔP_f).

The pressure loss due to friction was calculated by summing the individual losses (K/A^2, where K is the loss coefficient and A is the flow area) through the air inlet opening, the shielded plenum outlets, the air outlet openings, and the flow paths through the HSM. Standard loss coefficients for entrances, exits, screens, elbows, slots, friction, flow over cylinders, flow between parallel plates, flowpath expansions and contractions were taken from References 8.28 and 8.44. The pressure drop due to the flow losses was determined by:

$$\Delta P_f = \frac{\dot{m}^2}{2 g_c \bar{\rho}} \sum_i \frac{K_i}{A_i^2} \qquad (8.1\text{-}37)$$

Where: $\dot{m}$ = Mass flow rate (lbm/sec)

g_c = Gravitational constant

$\bar{\rho}$ = Average density

The pressure drop from the stack effect was calculated as follows:

$$\Delta P_s = \frac{g \bar{\rho} \Delta T h}{g_c \bar{T}} \qquad (8.1\text{-}38)$$

Where: ΔT = Temperature difference (°R)

h = Height (ft.)

g = Local acceleration due to Gravity = g_c

$\bar{T}$ = Average temperature (°R)

The above equations were solved iteratively to determine values of ΔT and $\dot{m}$ at specific values of (K/A^2) for $\Delta P_f \leq \Delta P_s$. This flow rate calculation conservatively accounted for possible flow separation at mid-length of the DSC which then flows toward the DSC ends and out through exhaust vents. Hence, the calculation conservatively accounted for potential low flow regions by using a reduced bulk air temperature based on conservative flow correlations. Based on this analysis, the geometry of the flow areas for the HSM were established as shown on the drawings in Appendix E. Using the calculated values of $\dot{m}$, the HSM bulk air temperatures surrounding the DSC were determined assuming isotropic heat flow from the DSC surface and integrating the energy equation around the DSC. The natural circulation cooling flow over the DSC surface for the NUHOMS-24P design is substantially higher (approximately 3 times) than the NUHOMS-07P design due to the increased capacity of the HSM vents and the increased driving force of the DSC. The results of this analysis are shown in Table 8.1-11 for a range of ambient conditions.

The resulting bulk air temperatures for the range of ambient conditions were used in the subsequent HSM analyses to calculate the temperatures throughout the HSM and DSC shell. In the HSM HEATING6 model, Boundary Type 1 (surface-to-boundary) was used to describe the natural circulation heat transfer between the DSC and the adjacent cooling air at the bulk air temperatures. The DSC temperatures were used as boundary conditions to determine the temperature distributions for the DSC internals and the spent fuel assembly regions. These calculations are described in the following subsections.

Initial calculations showed that the inside surface of the HSM concrete walls above the DSC centerline and the HSM roof were heated by radiation and convection heat transfer to over 200°F. While these temperatures are sufficiently low so that the HSM concrete material properties are not adversely affected, it was determined that additional design margins could be obtained by shielding the HSM walls and roof from the radiation heat transfer by placing a thin metal heat shield around the upper half of the DSC. The location and geometry of the heat shield is shown on the HSM drawings contained in Appendix E. The heat shield protects the HSM concrete walls and ceiling from direct thermal radiation emanating from the DSC surface and significantly increases the combined surface area for convection cooling inside the HSM. The concrete walls and ceiling continue to be subjected to thermal radiation from the back side of the heat shield, however, the radiation is emanated at substantially lower temperatures than the direct thermal radiation from the DSC surface.

B. Computer Program:

The HEATING6 computer program was used for the heat transfer analysis of the HSM and DSC. The HEATING6 program is known as "The HEATING Program," where HEATING is an acronym for Heat Engineering and Transfer In Nine Geometries. HEATING6 was designed to be a functional module within the SCALE system of computer programs (8.15) for performing standardized analysis for licensing evaluations of nuclear systems. Thus its features were designed to perform thermal analyses on problems arising in licensing evaluations, and its input format was designed to be compatible with that of other functional modules within the SCALE system. HEATING6 may also be used as a stand-alone, heat conduction code.

HEATING6 solves steady-state and/or transient heat conduction problems in one-, two-, or three-dimensional Cartesian or cylindrical coordinates or in one-dimensional spherical coordinates. The thermal conductivity, density, and specific heat may be both spatially and temperature-dependent. In addition, the thermal conductivity may be anisotropic. Selected materials may undergo a change of phase for transient calculations involving one of the explicit procedures. The heat generation rates may be dependent on time, temperature and position. Boundary temperatures may be dependent on time and position. Boundary conditions which may be applied along surfaces of an analytical model include specified temperatures or any combination of prescribed heat flux, forced convection, natural convection, and radiation. Models are also available to simulate the thermal fin efficiency of certain finned surfaces. In addition, one may specify radiative heat transfer across gaps or regions which are embedded in the model. The boundary condition parameters may be time- and/or temperature-dependent. The mesh spacing may be variable along each axis.

The HEATING6 thermal calculations performed for the HSM and DSC employed the optional direct solution technique of the program. This technique generally required from three to five iterations per calculation to obtain results with a convergence of better than 0.1% on the temperatures at each node in the analytical model.

For the DSC thermal calculations one hundred percent of the heat was assumed to be generated in the active fuel length of the spent fuel rods with no heat generation in the remaining portion of the fuel rods or the non-fuel bearing components. This conservative approach provides bounding values of spent fuel cladding temperatures for storage. The amount of heat rejected to the HSM concrete by the spent fuel due to direct gamma radiation or neutron emission is a very small fraction of the total heat generated by fission product decay.

C. Thermal Model of the HSM

The HEATING6 thermal model of the HSM is depicted in Figure 8.1-22. The model represents the symmetric right half of a HSM and DSC cross section at an axial location of maximum decay heat power. A bounding decay heat power level for 40,000 MWD/MT, 4.0% initial enrichment fuel with a 10 year cooling time of 0.66 kw/assembly resulting in a total of 15.8 Kw/DSC was used. A heat flux of 1.51 BTU/hr-in.^{2} (0.0251 BTU/min.-in.^{2}) was determined by distributing the total decay heat power of 15.8 Kw over the inner radial surface area of the DSC cylindrical shell with a length equal to the fuel assembly fuel pin length. The heat rejection through the fuel assembly end fittings and the DSC shielded end plugs and cover plates were conservatively neglected.

The HEATING6 analytical model of the HSM consists of a typical HSM cross-section of unit thickness. In addition, symmetry or an insulated boundary was assumed along the vertical centerline of the HSM, as shown in Figure 8.1-22. The HEATING6 model includes 21 regions for the concrete wall, roof slab, and foundation slab of the HSM. The soil below the foundation slab was modelled as a seven foot thick region with a constant temperature boundary at the edge of this region. Sufficient nodal refinement was used in the HSM analytical model to obtain accurate temperature distributions through the thickness of the HSM walls, roof and foundation slab.

For the thermal analysis of a typical HSM containing a loaded DSC located in the interior of a multiple module array with a DSC present in the two adjacent HSMs, a 1.0 foot thick wall with insulated boundary conditions was modeled. For the thermal analysis of an interior HSM with no DSC present in the adjacent HSMs, a two foot thick wall was modeled, the outer surface of which is assumed to be exposed to the prevailing ambient conditions. For the thermal analysis of a single free standing HSM, or an HSM at the end of a multiple module array, a three foot thick wall was modeled, the outer surface of which was assumed to be exposed to the prevailing ambient conditions. For summer ambient conditions, a solar heat flux of 62 BTU/hr.-ft.^{2} for normal conditions, and 127 BTU/hr.-ft.^{2} for off-normal and accident conditions, was conservatively applied to the roof surface and the outer end wall surface of the HSM. The enveloping solar heat flux of 127 Btu/hr-ft^2°F for the extreme off-normal case is based on Reference 8.59. It was calculated for the worst case location in the contiguous United States using the maximum day long solar irradiation value specified for a horizontal surface in the worst month with the maximum clearness correction. Similarly, the solar heat flux of 62 Btu/hr-ft^2°F for the normal case is an enveloping value for the southeastern United States and is approximately half of the extreme off-normal value. Solar heat loads were conservatively neglected for the HSM thermal analysis for winter ambient conditions for normal, off-normal and accident conditions.

The DSC cylindrical shell is approximated in the model by 40 rectangular regions with the thickness and properties of the stainless steel DSC shell. The approximation using rectangular regions is necessary since HEATING6 restricts the user to one geometry type in the same analytical model. To improve the approximation, the narrow sides on the modeled DSC regions are assumed to be insulated while the longer sides of the modeled DSC regions have the same surface area as the outer surface of the DSC cylindrical shell. The analytical model of the HSM includes four regions for the metal heat shield located between the top and sides of the DSC, and the roof and walls of the HSM, as shown in Figure 8.1-22. The analytical model also includes 45 regions to model the air gaps between the DSC, the heat shield, and the HSM.

For the HSM thermal analysis, a value of 0.0251 Btu/Min-in^2 was calculated for the decay heat flux through the DSC shell using the DSC internal cavity length of 173 inches. Use of the DSC cavity length and exclusion of the 1.08 axial peaking factor for the HSM thermal analysis is based on the test data contained in Reference 7.10. The reference test data for cylindrical casks shows that the measured surface temperature profiles are relatively flat over the entire length, indicating that the heat flux was nearly uniform over the surface and axial peaking was not affecting the surface temperatures distribution. One reason for the relatively flat temperature profiles is the high thermal conductivity of the DSC shell material and relatively open design of the DSC basket assembly. The resulting heat flux is therefore more representative of the manner in which heat is actually rejected to the HSM air space by the DSC. The active fuel length of 144 in. and the peaking factor of 1.08 were conservatively used in the thermal analysis of the DSC internals for the evaluation of local effects such as the peak fuel clad temperature. The outer surface of the DSC shell dissipates heat to the HSM through both convection and radiation. The air surrounding the DSC was modeled as a gap filled with gas (air), thus providing a mechanism for heat transfer from all HSM interior surfaces and the DSC outer surface. Due to a limitation in the HEATING6 code, conduction in the air gap could not be included, however, this is a minor effect.

Convection heat transfer from the DSC and HSM surfaces was modeled by inputting a constant air temperature for the eight air gap regions between the DSC and HSM. These temperatures were also used to calculate the heat transfer coefficients for these gas regions. The bulk air temperatures used for each ambient temperature case are shown in Table 8.1-11. With these temperatures and the equations for the heat transfer coefficients described below, the HEATING6 program calculates the temperatures of the DSC exterior surface and the HSM interior and exterior surfaces.

The spent fuel assembly decay heat is removed from the DSC outer surface through convection. A heat transfer coefficient h_{can} was used which corresponds to the heat transfer coefficient for natural circulation of air over a horizontal cylinder. Horizontal slab surfaces with convection on their lower surface, such as the HSM ceiling, were assumed to be cooled by natural convection with a heat transfer coefficient of h_{ceil}. Horizontal surfaces with convection on their upper surfaces, such as the HSM roof outer surface, were assumed to be cooled by natural convection with a heat transfer coefficient of h_{plate}.

Both sides of the metal heat shield and the HSM concrete walls were assumed to be cooled by air with a heat transfer coefficient of h_{wall}. Radiation heat transfer is modeled between the DSC outer surface and heat shield, between the DSC outer surface and the HSM floor, and between the heat shield and the HSM concrete walls and ceiling. The external surface of the HSM roof was assumed to be cooled by external air with a heat transfer coefficient of h_{plate}, and by radiation cooling to ambient air. The formulas used for the calculation of the heat transfer coefficients for natural convection are as follows [all in BTU/(hr. ft.^{2} °F)] (8.28):

$$h_{can} = 0.22\ (\Delta T)^{1/3} \qquad (8.1\text{-}39)$$

$$h_{ceil} = 0.12\ (\Delta T/L)^{1/4} \qquad (8.1\text{-}39a)$$

$$h_{plate} = 0.22\ (\Delta T)^{1/3} \qquad (8.1\text{-}39b)$$

$$h_{wall} = 0.19\ (\Delta T)^{1/3} \qquad (8.1\text{-}40)$$

Where: $\Delta T = T_{surface} - T_{air}$

L = 1/2 HSM ceiling width

The value of length scale (L) was assumed to be half of the HSM ceiling width. This length was conservatively chosen based on the assumption that the average distance the air has to travel over the DSC surface before exiting through the HSM air outlet vents will be at least one half of the HSM ceiling width, or 40 inches. Reference 8.60 recommends that the average of length and width of a horizontal plate be used for L. Use of one half of the HSM ceiling width is conservative since the average of the length and width of the HSM ceiling is greater than one half of the HSM ceiling width.

The heat transfer coefficients were updated by HEATING6 following each iteration using the resulting average temperature of the corresponding surface node. A sufficient number of iterations were performed until the temperatures differ by less than 0.1% from the previous temperature calculated in two consecutive iterations indicating that stable convergence has been achieved. The remaining thermal-hydraulic parameters used in the HSM heat transfer calculations are given in Tables 8.1-5 and 8.1-6.

The results of the HEATING6 analysis for the HSM are in the form of temperature distribution profiles. Figure 8.1-23 shows an example of the HEATING6 results for the HSM. The resulting temperature profile shows the steady state temperature distribution on the outer surface of the DSC, and at various locations throughout the HSM.

The calculated HSM wall and roof temperature gradients were used in the reinforced concrete structural analysis for long term thermal loads which occur during normal operating conditions, and the short term thermal loads occurring during off-normal and postulated accident conditions. The HSM thermal analysis results were also used to obtain steady state temperature distributions for the outer surface of the DSC for the range of design basis ambient conditions. These steady state surface temperatures were used as a temperature boundary condition for the DSC model, described in Section 8.1.3.2.

D. Description of the Cases Evaluated for the HSM

The HSM thermal analyses were performed for the design basis ambient air temperatures defined in Section 8.1.3. These include a total of seven cases with ambient air entering and/or surrounding the HSM at the following temperatures:

1. 0°F (minimum winter average), 70°F (lifetime average), and 100°F (maximum summer average) for normal operating conditions which can be expected to occur for long periods of time,

2. -40°F (extreme winter minimum), and 125°F (extreme summer maximum) for off-normal conditions which can be expected to occur for short periods of time, and

3. -40°F and 125°F extreme ambient temperatures with the HSM inlet and outlet vents postulated to be blocked for a period of 48 hours. This design basis condition is designated as an accident condition assumed to occur once in service life of facility.

The results of these calculations are summarized in Table 8.1-12 which shows that the highest temperature regions for the DSC occur on its top surface. The temperature at this location varies between 142°F (extreme winter) to 304°F (extreme summer). Similarly the HSM ceiling temperature above the DSC varies from 0°F to 215°F. During normal operation, the maximum temperature at the top of the DSC is 279°F (for 100°F ambient air). During normal operation, the maximum HSM temperature is 179°F (for 100°F ambient air). The bulk air temperature inside the HSM is -1°F (extreme winter) and 180°F (extreme summer). The HSM ventilation outlet air temperatures also vary from -1°F to 180°F.

8.1.3.2 Thermal Analysis of the DSC Inside the HSM

For the DSC thermal analyses, the internal basket assembly of the DSC was modeled in detail. A worse case, two-dimensional slice of the DSC and fuel cross sections was modeled. Heat transfer effects along the axis of the DSC (third dimension) were conservatively neglected. The DSC was assumed to be cooled through natural convection with the DSC surface specified as a temperature boundary condition equal to that calculated in the HSM thermal analysis. The fuel region inside the DSC was modeled as heat source equal to 1.08 times the nominal decay heat power of 0.66 kw/assembly.

The steady state outer surface temperatures for the DSC resting inside the HSM were calculated in the HSM thermal analysis, described in Section 8.1.3.1. The analytical results for each HSM analysis case were used to obtain average DSC surface temperatures for each region in the analytical model representing the DSC cylindrical shell. These surface temperatures were used as boundary conditions for the DSC thermal analysis and were assumed to remain constant.

A. Thermal Model of the DSC

The HEATING6 computer program was used to perform the thermal analysis of the DSC internal basket assembly and spent fuel assembly regions. The analytical model of the DSC contains 115 regions and is shown in Figure 8.1-24, with the individual regions indicated by number. The model includes 12 regions inside the guide sleeves for the spent fuel assemblies, 48 regions for the guide sleeves, 35 regions for the space between the adjacent guide sleeves and the DSC shell, and 20 regions for the DSC shell. The space between the guide sleeves and DSC shell were assumed to be filled with helium. In order to facilitate the thermal stress analysis of the DSC spacer disks, a similar model of the DSC was used with steel in the spaces between the guide sleeves and the DSC shell.

The heat generated in the fuel region was assumed to be transferred to the guide sleeves by conduction and radiation. Heat is transferred through the guide sleeve walls by conduction. For the narrow channels between adjacent guide sleeves, heat transfer was assumed to occur through conduction and radiation. Convection was conservatively neglected as the Grashof number which corresponds to convection between two parallel plates in an enclosed space is small for the DSC basket geometry. In the physical system, macroscopic convection in these regions and conduction in the axial direction would provide an additional mechanism for heat removal from the DSC, however, these conservatisms apparently were neglected. For the space between the horizontal and vertical surfaces of the outer guide sleeve and the DSC shell, heat was assumed to be transferred through conduction, convection and radiation. The calculation of apparent conductivities for these regions is contained in Appendix B. Heat transfer through the DSC shell is achieved by conduction.

The decay heat power of 0.66 kw for each spent fuel assembly was applied as a heat flux uniformly distributed over the the fuel regions inside the 24 guide sleeves. The resulting volumetric heat density, including a peaking factor of 1.08, which was applied over the active fuel length of 144 in. is 3.55E-3 Btu/min.-in.^{3}.

An effective thermal conductivity for the fuel region inside the DSC guide sleeve was determined to account for the different materials (UO_2, zircaloy and helium) and to include the combined effects of radiation, conduction, and convection. Appendix B describes the derivation of the effective thermal conductivity used to model the fuel region. The derivation of the apparent conductivities for the gas regions between the guide sleeves and the inner surface of the DSC shell are also described in Appendix B. Effective thermal conductivities were developed for the condition with a vacuum inside the DSC during the drying and helium backfilling operations as shown in Appendix B. A thermal emissivity of 0.587 was used for the radiation between all stainless steel surfaces (8.8).

The resulting calculated temperature profiles for the DSC were used for the evaluation of fuel cladding temperatures, helium temperatures, guide sleeve temperatures, and other DSC internal component temperatures. These temperatures were also used to evaluate the thermal stresses in the DSC shell and the spacer disks.

The maximum DSC shell temperature under all normal, off-normal or accident conditions is 513°F as reported in Table 8.1-13. This maximum temperature occurs on the DSC shell surface at mid-length of the DSC. The temperature at the ends of the DSC for this condition is at least 100°F lower than the maximum DSC shell temperature. Conservatively assuming that the 513°F temperature also exists on the DSC end surfaces, the worst case temperature at the DSC lead shield plug will be well below the 621°F melting point of lead (Reference 8.4). Therefore, melting of the DSC lead shield plugs will not occur for the worst case accident scenarios and was not evaluated further.

B. Evaluation of DSC inside the HSM

The DSC and fuel assembly heat transfer analyses with the DSC inside the HSM were performed for the design basis ambient air temperature cases defined in Section 8.1.3. These include a total of seven cases corresponding to ambient air entering the HSM at:

1. 0°F, 70°F, and 100°F for normal operating conditions,

2. -40°F and 125°F for off-normal conditions, and

3. -40°F and 125°F ambient conditions with the HSM inlet and outlet vents blocked for 48 hours.

Since the principal cases of interest are those which maximize the fuel cladding temperatures, the DSC thermal analyses were limited to evaluating summer ambient conditions as described in Section 8.1.3.1. From these analysis results, key temperatures were extracted and are summarized in Table 8.1-13. The results obtained from the HEATING6 analysis are in the form of temperature profiles for the DSC cross-section.

The back wall of the concrete HSM facing the DSC does not require a heat shield. As noted above, the maximum temperature of the DSC end plugs is substantially less than the DSC shell at mid-length. In addition, ventilation air flows in the gap between the DSC and the HSM back wall act to cool the concrete and DSC end assembly. The resulting HSM back wall temperatures are bounded by the HSM side wall and/or ceiling concrete temperatures. The maximum temperature of the HSM back wall concrete is less than 120°F for the 70°F ambient case which is well within acceptable limits.

From the 70°F ambient DSC temperature profile in Figure 8.1-25, it can be observed that the maximum temperatures occur for the fuel regions in the centermost guide sleeves just above the horizontal center line of the DSC. The maximum temperature occurs slightly above the midplane because the lower half of DSC shell is at a lower temperature than is the upper half. Also the DSC temperature distribution is not symmetric for the fuel assemblies located further from the central region of the DSC.

For the fuel assemblies located towards the outer edge of the DSC (i.e., fuel assemblies #1, 6, 7, 8, 9, and 10 in Figure 8.1-1), the boundary temperature is higher than the central region. The reason for this temperature distribution is that the heat flow is from the fuel assemblies/guide sleeves located in the center of the DSC to the fuel assemblies/guide sleeves located on the perimeter of the DSC. Heat is then removed from the DSC outer surface by natural convection and radiation. As a result, the outer-most fuel assemblies experience temperatures which are far below the bounding fuel clad temperature for the central fuel assemblies with the guide sleeve temperatures elevated above those of the corresponding fuel assemblies.

The fuel assemblies locate toward the center of the DSC (i.e., fuel assemblies #3 and #4 in Figure 8.1-1) show temperature profiles with peak temperatures at the center fuel region. This type of temperature distribution was also observed in the test results and was predicted by the COBRA-SFS Code as documented in Reference 7.10. As shown in Figure 5-29 of the reference report, the temperature distribution for the central fuel assembly is symmetrical with the maximum temperature occurring at the center. For the fuel assemblies located on the boundary of the DSC basket there is less symmetry.

For variations in ambient air temperatures for normal operating conditions, the maximum calculated fuel clad temperatures vary from 643°F (339°C), for the 70°F ambient air to 660°F (349°C), for 100°F ambient air. The maximum cladding temperature of 643°F (339°C), corresponding to 70°F lifetime average ambient temperature, is below the design basis initial storage temperature limit of 644°F (340°C) defined in Section 3.3.7.1 for long term dry storage. For extreme ambient conditions, or short term operating conditions, the maximum fuel cladding temperature ranges from 668°F (353°C) to 789°F (421°C). These values are well below the short term temperature limit of 1058°F (570°C) defined in Section 3.3.7.1. The corresponding DSC internal pressures are listed in Table 8.1-4.

A parametric study of temperature versus time was conducted in order to evaluate the effects of changing decay heat with spent fuel cooling time. In addition to the base case of ten years, cooling times of eight years and sixteen years were evaluated. The decay heat generation rates, the DSC shell boundary temperatures, the effective thermal conductivity of the fuel region, and the effective thermal conductivity of the helium were modified as appropriate in the analyses for spent fuel cooling times of eight and sixteen years. The maximum fuel cladding temperatures, HSM concrete temperatures, HSM bulk air temperatures, and the helium temperature and pressure inside the DSC, were obtained as a function of cooling time by applying the appropriate decay heat flux for the corresponding cooling time. Results of these analyses are presented in Figures 8.1-25a through 8.1-30.

8.1.3.3 Thermal Analysis of the DSC Inside the NUHOMS-24P Transfer Cask

A. DSC in Cask During Transport

The cylindrical NUHOMS-24P transfer cask and DSC shell were modeled as long composite cylinders with the cross section configuration shown in Figure 8.1-25a. Full contact between all cask materials (steel, lead, and liquid neutron shield) is assumed. When the NUHOMS-24P transfer cask is in the horizontal position, the DSC outer surface was assumed to be in contact with the transfer cask inner liner. Two separate models were developed to determine the radial and circumferential temperature distribution at various composite regions of the transfer cask. The first model is for the bottom half of the transfer cask cross-section where the DSC outer surface was assumed to be in contact with the transfer cask inner liner. The second model was for the top half of the transfer cask cross-section with the gap between the DSC outer surface and the top region of the transfer cask inner liner at its maximum value.

The transfer cask was evaluated for a range of ambient temperatures including the normal, off-normal, and postulated accident conditions which are described in Section 8.1.3.1. The ambient conditions include:

1. 0°F to 100°F for normal operating conditions, and

2. -40°F and 125°F for off-normal and accident conditions.

The total decay heat power for the 24 spent fuel assemblies of 15.8 Kw/DSC was uniformly distributed over the inner radial surface area of the DSC shell. The resulting decay heat power is 58,600 BTU/hr. The solar heat flux was conservatively neglected for winter ambient conditions to maximize the transfer cask through wall temperature gradient.

The temperatures of the DSC shell outer surface, through the wall thickness of the transfer cask, and the transfer cask outer surface were determined by performing a heat balance analysis for the applied decay heat power and the ambient air conditions, including the effects of solar heating as applicable. The enveloping solar heat flux of 127 Btu/hr.-ft.^{2}°F for the extreme off-normal case and 62 Btu/hr.-ft.^{2}°F for the normal case (8.59) were used. A steady state heat balance analysis was performed to determine the temperatures at which the heat flow equals the convection and radiation heat loss from the outer surface of the transfer cask as follows (8.8):

$$\dot{Q} = \left\{ 0.18\ [T_{out} - T_{air}]^{4/3} + 0.1714\text{x}10^{-8}\ \epsilon\ [T_{out}^{\ 4} - T_{air}^{\ 4}] \right\} A_o \qquad (8.1\text{-}41)$$

Where: ϵ = 0.587 (for cask emissivity)

0.1714×10^{-8} = Stefan-Boltzman constant

$A_o = 2\pi R_{DSC}$ x length

T_{out} = Cask outer surface temperature

The conduction heat transfer from the transfer cask outer surface was conservatively neglected. Equation (8.1-41) was solved iteratively for each ambient air condition to obtain

$$\Delta T = T_{out} - T_{air}$$

To obtain the surface temperatures, T_{in} on the inner liner of the transfer cask the steady state heat transfer relationship for a composite cylinder was used (8.57).

$$(T_{in} - T_{out}) = \frac{\dot{Q}}{2\pi L} \sum_{n=1}^{n} \frac{\ln\left(\frac{r_{n+1}}{r_n}\right)}{K_n} \qquad (8.1\text{-}42)$$

Where r_1, r_2, r_3 r_4, r_5, r_6 and r_7 are the radii for the regions through the cask wall as shown Figure 8.1-25a. Radiation through the 50% by weight ethylene glycol, 50% water filled neutron shield cavity was conservatively neglected.

In the top half model of the NUHOMS-24P transfer cask shown in Figure 8.1-25a, to calculate the DSC outer surface temperature (T_{DSC}), conduction, convection and radiation heat transfer from the DSC inner surface through the DSC shell and the air gap to the transfer cask inner liner were evaluated. The heat loss through the gap was calculated using the 0.75 inch maximum air gap.

$$\dot{Q} = \frac{0.1714 \times 10^{-8} A_{DSC} \left(T_{DSC}^4 - T_{in}^4\right)}{1/\epsilon_1 + \frac{r_1}{r_2}\left(1/\epsilon_2 - 1\right)} + \frac{K_e 2\pi L}{\ln\left(\frac{r_2}{r_1}\right)} \left(T_{DSC} - T_{in}\right) \qquad (8.1\text{-}43)$$

Where: ϵ_2 = $0.587 = \epsilon_1$ for stainless steel surfaces

r_1 = Outside radius of DSC

r_2 = Inside radius of the cask

A_{DSC} = DSC outside surface area = $2\pi r_1 L$

K_e = Effective thermal conductivity which is evaluated as follows (8.8)

$$K_e = 0.135\, K_f \left(\frac{Pr^2 Gr}{1.36 + Pr}\right)^{0.278}$$

$$\text{for } 10^3 < \frac{Pr^2 Gr}{1.36 + Pr} < 10^8 \qquad (8.1.44a)$$

and $$K_e = K_f \text{ for } \frac{Pr^2 Gr}{1.36 + Pr} < 10^3 \qquad (8.1\text{-}44b)$$

Where: K_f = Thermal conductivity of fluid in the gap

Gr = Grashof number

Pr = Prandtl number

To estimate K_e, the Prandtl number, Grashof number and the fluid thermal conductivity were evaluated at the average fluid temperature in the gap.

An iterative solution of equations 8.1-43 and 8.1-44a or 8.1-44b provided the DSC outer surface temperature value (T_{DSC}) for the top half of the DSC.

For the bottom half model of the NUHOMS-24P transfer cask shown in Figure 8.1-25a, complete contact between the DSC outer surface and the lower region of the transfer cask inner liner was assumed. To maintain symmetry and to keep the outside diameter the same as the top half model of the transfer cask the thickness of the transfer cask inner liner was adjusted in the model. Equation 8.1-42 was then used to determine the DSC outer surface temperature value (T_{DSC}) for the bottom half of the DSC.

An analysis was performed for normal, off-normal, and accident summer ambient conditions, including the effect of a solar heat flux applied to the cask outer surface. The resulting through wall thermal gradients for the transfer cask for the normal, off-normal, and accident conditions are summarized in Table 8.1-14. The accident condition considered was a complete loss of liquid neutron shield with the off normal ambient condition of 125°F. These temperature gradients were used to perform a thermal stress analysis of the transfer cask as discussed in 8.1.1.8.

The resulting temperatures for the DSC outer surface were used as temperature boundary conditions for the heat transfer analyses of the DSC internals to confirm that short term fuel clad temperatures during transport from the plant's fuel handling building to the HSM remain below 570°C. The analytical models and the methodology used to perform the heat transfer analyses to determine the temperatures of the DSC internals and the fuel cladding are discussed above.

B. DSC in Cask During Draining and Drying

The methodology used to evaluate the heat transfer effects which occur during transport of the DSC (inside the transfer cask) from the plant's fuel handling building to the HSM, where the DSC is transferred for storage, are discussed in the previous paragraphs. Other conditions during the NUHOMS system also result in heat transfer effects on the NUHOMS system components. These include placement of the DSC and transfer cask in the plant's fuel pool, loading of spent fuel into the DSC, seal welding of the DSC, draining and vacuum drying of the DSC, and backfilling the DSC with helium. Of these conditions, vacuum drying is the most severe since heat conduction in the cavity of the DSC filled with air is minimized.

For the DSC inside the NUHOMS-24P transfer cask, increased temperature conditions will be encountered during the vacuum drying process. An analysis of the transfer cask and DSC was performed to determine the temperature distribution and the maximum fuel cladding temperature for this condition. The analytical methods used for this analysis are similar to those discussed in Sections 8.1.3.2 and 8.1.3.3. The cask and DSC are oriented vertically for this operation and the DSC cavity is dried by pulling a vacuum. The resulting through wall thermal gradients in the transfer cask were bounded by those calculated in Section 8.1.3.3 and as such were not evaluated further. The maximum fuel cladding temperature calculated for this condition was 770°F (410°C) which is well below the 570°C short term temperature limit.

C. Cask Liquid Neutron Shield Expansion

The expansion of the liquid in the transfer cask neutron shield jacket is accommodated by an expansion tank. The expansion tank design is shown in Figure 4.2-8.

The volume of the expansion tank was conservatively calculated assuming that the initial temperature of the liquid in the transfer cask neutron shield cavity is the minimum normal operational fuel building temperature of 50°F. The thermal analysis of the DSC inside the NUHOMS-24P transfer cask shows that the maximum temperature of the liquid in the neutron shield jacket under the worst case extreme ambient condition was estimated to be 260°F. This worst case temperature difference was used to calculate the increase in the volume of liquid in the transfer cask. The expansion tank volume was then calculated to accommodate this increased liquid volume plus the volume of the air initially present in the expansion tank.

Table 8.1-1

NUHOMS-24P NORMAL OPERATING LOADING IDENTIFICATION

Load Type	Component Loaded				
	DSC Shell Assembly	DSC Internals	DSC Support Assembly	Reinforced Concrete HSM	NUHOMS-24P Transfer Cask
Dead Weight	X	X	X	X	X
Internal Pressure	X				
Normal Thermal	X	X	X	X	X
Normal Handling	X		X	X	X
Live Loads				X	X

Table 8.1-1a

NUHOMS-24P OFF-NORMAL OPERATING LOADING IDENTIFICATION

Load Type	Component Loaded				
	DSC Shell Assembly	DSC Internals	DSC Support Assembly	Reinforced Concrete HSM	NUHOMS-24P Transfer Cask
Dead Weight	X	X	X	X	X
Internal Pressure	X				
Off-Normal Thermal	X	X	X	X	X
Off-Normal Handling	X		X	X	X

Table 8.1-2

MECHANICAL PROPERTIES OF MATERIALS

Material	Temperature (°F)	Stress Properties (1) (ksi)			Elastic Modulus (1) (x1.0E3 ksi) (E)	Instantaneous Coefficient of Thermal Expansion (1) (μ in./in.-°F)
		Stress (6) Intensity (S_m)	Yield Strength (S_y)	Ultimate Strength (S_u)		
Stainless Steel ASME SA240 Type 304 and SA479 Type 304	70	-	30.0	75.0	28.3	8.46
	100	20.0	30.0	75.0	-	8.63
	200	20.0	25.0	71.0	27.6	9.08
	400	18.7	20.7	64.4	26.5	9.80
	500	17.5	19.4	63.5	25.8	10.10
	600	16.4	18.2	63.5	25.3	10.38
	800	15.2	16.8	62.7	24.1	10.79
Carbon (7) Steel ASTM A36	70	-	36.0	58.0	29.5	6.41
	100	14.5	36.0	-	-	6.53
	200	14.5	32.8	-	28.8	6.93
	400	14.5	30.8	-	27.7	7.66
	500	14.5	29.1	-	27.3	8.03
	600	14.5	26.6	-	26.7	8.35

Table 8.1-2

MECHANICAL PROPERTIES OF MATERIALS
(Continued)

Material	Temperature (°F)	Stress Properties (1) (ksi)			Elastic Modulus (1) (x1.0E3 ksi) (E)	Instantaneous Coefficient of Thermal Expansion (1) (μ in./in.-°F)
		Stress Intensity (S_m)	Yield Strength (S_y)	Ultimate Strength (S_u)		
Carbon Steel Plate ASME SA516 Grade 70	70	23.3	38.0	70.0	29.5	5.42
	100	23.3	38.0	70.0	29.3	5.65
	200	23.1	34.6	70.0	28.8	6.39
	400	21.7	32.6	70.0	27.7	7.60
	500	20.5	30.7	70.0	27.3	8.07
	600	18.7	28.1	70.0	26.7	8.46
Transfer Cask Lifting Trunnions SA-564 Gr. 630 PH (5)	70	45.0	105.0	135.0	28.3	5.89
	100	45.0	105.0	135.0	28.1	5.89
	200	45.0	97.1	135.0	27.6	5.90
	400	43.8	89.8	131.4	26.5	5.91
	500	42.8	87.0	128.5	25.8	5.91
	600	42.1	87.7	126.7	25.3	5.96

Table 8.1-2

MECHANICAL PROPERTIES OF MATERIALS
(Continued)

Material	Temperature (°F)	Stress Properties [(1)] (ksi)			Elastic Modulus [(1)] (x1.0E3 ksi) (E)	Instantaneous Coefficient of Thermal Expansion [(1)] (μ in./in.-°F)
		Stress Intensity (S_m)	Yield Strength (S_y)	Ultimate Strength (S_u)		
Transfer Cask Lifting Trunnion Sleeves SA533 Gr B Cl 2	70	30.0	70.0	90.0	29.2	7.02
	100	30.0	70.0	90.0	29.0	7.13
	200	30.0	65.5	90.0	28.5	7.45
	400	30.0	63.2	90.0	27.4	8.01
	500	30.0	62.3	90.0	27.0	8.25
	600	30.0	61.4	90.0	26.4	8.46

Table 8.1-2

MECHANICAL PROPERTIES OF MATERIALS
(Continued)

Material	Temperature (°F)	28 Day Compressive Strength (ksi)	Modulus of Elasticity (1.0E3 ksi)
Concrete Normal Wt. 5000 psi Strength (2)	100	5.0	4.0
	200	5.0	3.6
	300	4.8	3.3
	400	4.5	3.0

Material	Temperature (°F)	Yield Strength (ksi)	Modulus of Elasticity (1.0E3 ksi)
Reinforcing (2) Steel ASTM A615 Grade 60	100	60.0	29.0
	200	57.0	28.4
	300	54.0	27.8
	400	51.0	27.3

Solid Neutron Shielding Material (4)	Poisson Ratio	Compressive Strength (ksi)	Modulus of Elasticity (1.0E3 ksi)
BISCo NS-3	0.2	3.9	0.16
Boro-Silicone	N/A	0.45	N/A

Table 8.1-2

MECHANICAL PROPERTIES OF MATERIALS
(Continued)

Material	Temperature (°F)	Allowable Stress Values for Class 2 Components (S)(1) (ksi)	Yield Strength(1) (ksi)
HSM Structural Bolting Material ASTM A325	100	20.2	81.0
	200	20.2	73.9
	400	20.2	69.3
NUHOMS-24P Transfer Cask Bolting Materials ASME SA-193 Grade B7	-40	25.0	105.0
	+70	25.0	105.0
	+100	25.0	98.0
	+200	25.0	94.1
	+400	25.0	91.5
	+500	25.0	88.5
	+600	25.0	85.3

Table 8.1-2

MECHANICAL PROPERTIES OF MATERIALS
(Concluded)

Material	Yield Strength (ksi)	Tensile Strength (ksi)	Modulus of Elasticity ($1x10^6$ psi)	Coefficient of Linear Expansion (μ in./in.-°F)	Approximate Melting Point (°F)
Common (3) Lead	---	2.5	2	16.4	621

Notes:

1. Steel data and thermal expansion coefficients were obtained from ASME Boiler and Pressure Vessel Code, Section III-1 Appendices. (8.3)

2. Concrete and reinforcing steel data were obtained from Handbook of Concrete Engineering, by Mark Fintel. (8.22)

3. Lead data was obtained from CRC Handbook of Tables for Applied Engineering Science, 2nd Edition, pp. 111 and 118. (8.4)

4. Data obtained from manufacturers published information.

5. Age hardened at 1150°F in accordance with note (5) of ASME Code, Appendix I, Table I-1.4.

6. For ASTM A36, the values in this column are the allowable stress values (S) for component supports.

7. Allowable stress values (S) and the yield strength (S_y) for A36 steel are given in Table I-12.1 and Table I-13.1, respectively, of the ASME Boiler and Pressure Vessel Coe, Section III, Division 1, Appendix I.

Table 8.1-3

ESTIMATED COMPONENT WEIGHTS

	Component Description	Calculated Weight (Pounds)
1.	Dry Shielded Canister Shell Assembly	15,740
2.	DSC Top Shield Plug Assembly	5,620
3.	DSC Internal Basket Assembly	10,090
	Total DSC Dry Weight	31,450
4.	24 15x15 PWR Spent Fuel Assemblies	40,370
5.	Weight of Water in DSC Cavity	16,240
	Total Wet DSC Loaded Weight	88,060
	Total Dry DSC Loaded Weight	71,820
6.	NUHOMS-24P Transfer Cask Empty Weight	103,000
7.	NUHOMS-24P Transfer Cask Max. Loaded Weight	190,000(2)
8.	HSM Single Module Weight (Empty)	501,000

Notes:

1. Includes weight of cask top cover plate assembly.

2. Weight includes: DSC dry weight plus fuel, plus water in DSC and cask canisters less DSC and cask top cover plate assemblies.

Table 8.1-4

DSC OPERATING AND ACCIDENT PRESSURES

Case	Ambient Air Temperature (°F)	Average Helium Temperature (°F)	Helium (1) Pressure (psia/psig)	Partial (2) Pressure Fission and Fill Gas (psia)	Design Basis Accident Pressure	
					(psia)	(psig)
1	-40	320	18.7/4.0	28.3	47.0	32.3
2	0	340	19.2/4.5	29.0	48.2	33.5
3	70	375	20.0/5.3	30.3	50.3	35.6
4	100	400	20.6/5.9	31.2	51.8	37.1
5	125	410	20.8/6.1	31.6	52.4	37.7
6	Blocked HSM Vents (125)	560	24.4/9.7	37.0	61.4	46.7
7	Loss of Cask Neutron Shield (125)	600	25.4/10.7	38.5	63.9	49.1
8	DSC in Cask (100)	560	24.4/9.7	37.0	61.4	46.7

Notes:

1. Operating Pressure with all fuel cladding intact.
2. Design Basis Accident pressure with 100% of fuel rod fill gas and 30% of fission gas assumed to be released.

Table 8.1-5

THERMOPHYSICAL PROPERTIES OF MATERIALS

Material	Effective Thermal Conduct. (k) (Btu/h-ft-°F)	Density (lb./ft.^{3})	Specific Heat (C_p) (Btu/lb.-°F)	Emissivity (e) Fraction
Carbon Steel	Table 8.1-6 (8.57)	490 (8.57)	0.11 (8.57)	--
Concrete	Table 8.1-6 (8.11)	140 (8.11)	0.25 (8.7)	0.80 (8.11)
Stainless Steel	Table 8.1-6 (8.3)	493 (8.56)	Table 8.1-6 (8.4)	0.587 (8.8)
Lead	Table 8.1-6 (8.57)	705 (8.57)	0.03 (8.57)	--
50 weight% Ethylene Glycol (With Convection)	Table 8.1-6 (8.7)	Table 8.1-6 (8.7)	Table 8.1-6 (8.7)	--
Soil	0.5 (8.28)	112.5 (8.28)	0.225 (8.28)	--
Boro-silicon	1.4 (8.61)	99.0 (8.61)	0.24 (8.61)	--

Note:

1. Numbers in parenthesis are references.

Table 8.1-6

TEMPERATURE DEPENDENT THERMOPHYSICAL PROPERTIES

Temperature (°F)	Density (lb./cu.ft.)	Specific Heat (Btu/lb.-°F)	Thermal Conductivity (Btu/h-ft.-°F
Stainless Steel 304			
-60		0.100	7.7
140	493	0.115	8.95
640		0.135	11.50
1000		0.143	13.20
1640		0.156	15.6
Carbon Steel			
32	490	0.11	26.5
212			26.0
572			25.0
932			22.0
Lead			
32	705	0.03	20.10
212			19.00
572			18.00
Concrete			
100	140	0.25	1.17
200			1.14
500			1.04
1000			0.80
50 Weight % Ethylene Glycol (with Convection)			
100	65.61	0.81	6.36
175	64.23	0.86	7.68
200	63.30	0.87	8.07
256	62.37	0.90	8.69
Helium (8.7)			
45	0.01116	1.24	0.0831
80	0.01016	1.24	0.0866
260	0.00762	1.24	0.1037
350	0.00685	1.24	0.11252
495	0.00578	1.24	0.1283
1520	0.00277	1.24	0.2248

Table 8.1-6

TEMPERATURE DEPENDENT THERMOPHYSICAL PROPERTIES
(Concluded)

Temper-ature °F	Density (lb./cu. ft.) ρ	Viscosity (sq. ft./s) ν	Specific Heat (Btu/lb.-°F) C_p	Thermal Conduc-tivity (Btu/h-ft.-°F) k	Prandtl Number Pr
Air (8.57)					
0	0.086	0.00013	0.239	0.0133	0.73
32	0.081	0.000145	0.240	0.0140	0.72
100	0.071	0.000180	0.240	0.0154	0.72
200	0.060	0.000239	0.241	0.0174	0.72
300	0.052	0.000306	0.243	0.0193	0.71
400	0.046	0.000378	0.245	0.0212	0.689
500	0.0412	0.000455	0.247	0.0231	0.683
1000	0.0271	0.000917	0.262	0.0319	0.713

Table 8.1-7

MAXIMUM DSC STRESSES

FOR NORMAL LOADS

DSC Components	Stress Type	Stress (ksi)[1]			
		Dead Weight	Internal Pressure	Thermal	Normal Handling
DSC Shell	Primary Membrane	0.1	0.5	N/A	0.2
	Membrane + Bending	0.2	6.2	N/A	1.8
	Primary + Secondary	3.7	6.2	17.5	1.8
Inner Top Cover Plate	Primary Membrane	0.1	0.0	N/A	0.1
	Membrane + Bending	0.5	4.6	N/A	0.3
	Primary + Secondary	0.2	3.4	0.3	N/A
Outer Top Cover Plate	Primary Membrane	0.1	N/A	N/A	0.1
	Membrane + Bending	0.4	4.6	N/A	0.3
	Primary + Secondary	0.2	3.4	1.0	N/A

Note:

1. Values shown are maximum irrespective of location.

Table 8.1-7

MAXIMUM DSC STRESSES

FOR NORMAL LOADS
(Concluded)

DSC Components	Stress Type	Stress (ksi)[1]			
		Dead Weight	Internal Pressure	Thermal	Normal Handling
Bottom Cover Plate	Primary Membrane	0.1	0.0	N/A	0.7
	Membrane + Bending	0.3	1.0	N/A	1.6
	Primary + Secondary	0.3	0.5	1.7	0.8
Spacer Disk	Primary Membrane	0.5	0.0	N/A	0.0
	Primary + Secondary	0.3	N/A	46.5	N/A

Note:

1. Values shown are maximum irrespective of location.

Table 8.1-7a

MAXIMUM DSC STRESSES

FOR OFF-NORMAL LOADS

DSC Components	Stress Type	Stress (ksi)[1]		
		Internal Pressure	Thermal	Off-Normal Handling
DSC Shell	Primary Membrane	0.5	N/A	1.2
	Membrane + Bending	6.2	N/A	5.9
	Primary + Secondary	6.8	20.9	7.0
Inner Top Cover Plate	Primary Membrane	0.0	N/A	0.0
	Membrane + Bending	4.6	N/A	0.0
	Primary + Secondary	3.4	0.3	0.0
Outer Top Cover Plate	Primary Membrane	0.0	N/A	0.0
	Membrane + Bending	4.6	N/A	0.0
	Primary + Secondary	3.4	1.0	0.0

Note:

1. Values shown are maximum irrespective of location.

Table 8.1-7a

MAXIMUM DSC STRESSES

FOR OFF-NORMAL LOADS

(Concluded)

DSC Components	Stress Type	Stress (ksi)(1)		
		Internal Pressure	Thermal	Off-Normal Handling
Bottom Cover Plate	Primary Membrane	0.0	N/A	0.0
	Membrane + Bending	1.0	N/A	6.5
	Membrane + Secondary	0.5	1.7	3.1
Spacer Disk	Primary Membrane	0.0	N/A	0.0
	Primary + Secondary	0.0	46.5	N/A

Note:

1. Values shown are maximum irrespective of location.

Table 8.1-8

MAXIMUM DSC SUPPORT ASSEMBLY STRESSES FOR

NORMAL AND OFF-NORMAL LOADS

Component	Load Type	Calculated Stress		
		Axial (ksi)	Bending (ksi)	Shear (ksi)
WF-Section Cross Members	Dead Weight	0.4	7.0	4.8
	Normal DSC Handling Loads	0.0	4.9	0.6
	Off Normal DSC Handling Loads	0.5	16.6	5.2
WT-Section Support Rail	Dead Weight	0.1	1.2	1.5
	Normal DSC Handling Load	0.1	2.5	0.1
	Off-Normal DSC Handling Loads	0.9	21.5	1.8

Notes:

1. Maximum stresses reported irrespective of location.
2. Dead weight and normal handling load allowables are based on 600°F which conservatively envelopes all ambient cases. Off-normal handling load allowables are based on 600°F. A 50% increase in normal AISC allowables was assumed for off-normal conditions.

Table 8.1-9

MAXIMUM NORMAL AND OFF-NORMAL LOADS
FOR DSC SUPPORT ASSEMBLY END CONNECTIONS

Loading	F_x (k)	F_y (k)	F_z (k)	M_z (k-in.)
Dead Weight	3.6	23.7	6.7	2.3
Normal DSC Handling Loads	3.7	1.1	0.0	0.5
Off-Normal DSC Handling Loads (See Note 4)	19.6	25.5	7.3	5.2
Design Loads	14.8	25.6	12.8	5.2

Notes:

1. Maximum Loads shown are irrespective of locations.
2. Global coordinate system used as shown on Figure 8.1-8.
3. All F_y loads are downward. F_x, F_z, and M_z values are reversible.
4. Design loads for this case are 1.5 times the listed values. Embedment design loads were established by factoring the given loads by 1.7 times.

Table 8.1-9a

MAXIMUM DSC SUPPORT ASSEMBLY
VERTICAL DISPLACEMENTS FOR
NORMAL AND OFF-NORMAL LOADS

Components	Load Type	Maximum Vertical Displacements (In.)
WF-Section Cross Members	Dead Weight $DW_s + DW_c$	.0323
	Normal DSC Handling Loads $DW_s + HL_n$	.0026
	Off-Normal DSC Handling Loads $DW_s + DW_c + HL_o$	.0346
WT-Section Support Rails	Dead Weight $DW_s + DW_c$	.0412
	Normal DSC Handling Loads $DW_s + HL_n$	.0497
	Off-Normal DSC Handling Loads $DW_s + DW_c + HL_o$	.0443

Table 8.1-9b

THERMAL LOAD CASE DEFINITIONS

FOR HSM STRUCTURAL ANALYSIS

Thermal Condition	Case No.	Ambient Temp.	Maximum Inner Surface Temperature			Maximum Outer Surface Temperature			Maximum Thermal Gradient		
			Roof	Wall	Floor	Roof	Wall	Floor	Roof	Wall	Floor
Normal Operating (T_o)	1	70	144	120	135	112	107	114	32	13	21
	2	70	142	107	133	111	77	107	31	30	26
	3	70	142	110	133	111	84	108	31	26	25
	4	100	179	152	167	140	139	140	39	13	27
	5	0	48	33	57	11	8	41	37	25	16
Off-Normal (T_a)	1	125	215	178	194	187	165	162	28	13	32
	2	-40	0	-10	14	-30	-33	6	30	23	8
	1A	125	214	164	191	186	132	155	28	32	36
	2A	-40	2	1	15	-29	-10	11	31	11	4
Accident (T_a)	1	125	368	333	395	187	193	199	181	140	196
	2	-40	173	146	200	-5	13	3	178	133	197
	3	125	368	336	395	187	170	199	181	166	196
	4	-40	173	147	200	-5	-10	3	178	157	197
	5	125	368	381	395	187	323	199	181	58	196
	6	-40	173	186	200	-5	127	3	178	59	197

Notes:
1. All temperatures are in degrees Fahrenheit.
2. Temperatures for two foot side wall insulated cases are wall centerline values.

Table 8.1.9b

THERMAL LOAD CASE DEFINITIONS

FOR HSM STRUCTURAL ANALYSIS
(Continued)

Normal Conditions

1. Interior module with 70°F ambient temperature, 2 ft. walls insulated (DSC in adjacent HSMs), vents open, solar heat flux of 62.0 Btu/hr-ft^2

2. Interior module with 70°F ambient temperature, 2 ft. walls uninsulated (no DSC in adjacent HSMs), vents open, solar heat flux of 62.0 Btu/hr-ft^2

3. End module with 70°F ambient temperature, 3 ft. end wall uninsulated, vents open, solar heat flux of 62.0 Btu/hr-ft2

4. Interior module with 100°F ambient temperature, 2 ft. walls insulated, vents open, solar heat flux of 62.0 Btu/hr-ft^2

5. Interior module with 0°F ambient temperature, 2 ft. walls uninsulated, vents open, solar heat flux neglected

Off-Normal Conditions

1. Interior module with 125°F ambient temperature, 2 ft. walls insulated (DSC in adjacent HSMs), vents open, solar heat flux of 127 Btu/hr-ft.^{2}

2. Interior module with -40°F ambient temperature, 2 ft. walls uninsulated (no DSC in adjacent HSMs), vents open, neglect solar heat flux

1A. Interior module with 125°F ambient temperature, 2 ft. walls uninsulated (no DSC in adjacent HSMs), vents open, solar heat flux of 127 Btu/hr-ft.^{2}

2A. Interior module with -40°F ambient temperature, 2 ft. walls insulated, vents open, solar heat flux neglected

Table 8.1.9b

THERMAL LOAD CASE DEFINITIONS

FOR HSM STRUCTURAL ANALYSIS
(Concluded)

Accident Conditions

1. Interior module with 125°F ambient temperature, 2 ft. walls uninsulated (no DSC in adjacent HSMs), vents blocked for 48 hours, solar heat flux of 127 $Btu/hr\text{-}ft^2$

2. Interior module with -40°F ambient temperature, 2 ft. walls uninsulated, vents blocked for 48 hours, solar heat flux neglected

3. End module with 125°F ambient temperature, 3 ft. end wall uninsulated, vents blocked for 48 hours, solar heat flux of 127 Btu/hr-ft2

4. End module with -40°F ambient temperature, 3 ft. end walls uninsulated, vents blocked for 48 hours, solar heat flux neglected

5. Interior module with 125°F ambient temperature, 2 ft. walls uninsulated (DSC in adjacent HSMs), vents blocked for 48 hours, solar heat flux of 127 $Btu/hr\text{-}ft^2$

6. Interior module with -40°F ambient temperature, 2 ft. walls insulated, vents blocked for 48 hours, solar heat flux neglected.

Table 8.1-9c

THERMAL LOAD CASES FOR HSM 2x10
ARRAY STRUCTURAL DESIGN

Load Case	Ambient Temperature (°F)	Module Number									
		1	2	3	4	5	6	7	8	9	10
Normal Conditions	70	X	X	X	X	X	X	X	X		X
	100	X	X	X	X	X	X	X	X	X	X
	0		X		X		X		X		X
Off-Normal Conditions	125		X		X		X		X		X
	125	X	X	X	X	X	X	X	X	X	X
	-40		X		X		X		X		X
	-40	X	X	X	X	X	X	X	X	X	X
Accident Conditions	125		X		X		X		X		X
	125		X		X		X		X		X
	-40		X		X		X		X		X
	-40		X		X		X		X		X

Notes:

1. "X" represents the location of a DSC.

2. X represents the location of HSM with blocked vents.

3. The HSM temperature and thermal gradients for each load case are given in Table 8.1-9b.

Table 8.1-10

MAXIMUM HSM REINFORCED CONCRETE BENDING MOMENTS AND SHEAR FORCES FOR NORMAL AND OFF-NORMAL LOADS

Structural Section	Force Component	HSM Internal Forces (in., kips)					Ultimate Capacity (in, kips) (3)
		Dead Weight	Creep Effects	Live Loads	Normal Thermal (5)	Off-Normal Thermal (5)	
Floor Slab	Shear	1.1	6.73	.12	23.1	30.4	43.8
	Moment	16.5	244.9	4.3	336	489	2840
Inner Wall	Shear	1.7	1.8	0	7.8	11.2	27.4
	Moment	106.7	153.5	.35	217	238	1730
End Wall	Shear	2.6	5.1	.11	14.6	19.5	43.8
	Moment	164	463.3	12.3	490	613	3570
Roof Slab	Shear	2.6	9.32	.73	25.4	35.6	43.8
	Moment	75.9	387.1	10.1	575	717	3570

Notes:

1. Values shown are maximums irrespective of location.
2. Concrete and reinforcing steel properties were taken at 400°F to conservatively envelope all ambient cases.
3. Ultimate Shear Capacity based on ACI 349-85 Equation 11-28 (Section 11.8.5), $V_c = \phi\, 2\sqrt{f'_c}\, b_w d$
4. Ultimate capacities are reported for a 12 in. section of HSM using f'_c and f_y values at 400°F.
5. Maximum moments are based on cracked section properties.

Table 8.1-10a

MAXIMUM NUHOMS-24P TRANSFER CASK STRESSES
FOR NORMAL LOADS

NUHOMS-24P Transfer Cask Components	Load Type / Stress Type	Stress (ksi)[1]		
		Dead Weight	Thermal	Normal Handling
Transfer Cask Structural Shell	Primary Membrane	0.7	N/A	0.5
	Membrane + Bending	0.8	N/A	30.3
	Primary + Secondary	0.8	20.3	35.6
Top Cover Plate	Primary Membrane	0.2	N/A	N/A
	Membrane + Bending	0.6	N/A	6.3
	Primary + Secondary	0.5	7.4	N/A
Bottom End Assembly	Primary Membrane	1.3	N/A	N/A
	Membrane + Bending	1.4	N/A	8.9
	Primary + Secondary	0.5	5.3	N/A

Note:

1. Values shown are maximum irrespective of location.

Table 8.1-10b

MAXIMUM NUHOMS-24P TRANSFER CASK STRESSES

FOR OFF-NORMAL OPERATING LOADS

DSC Components	Load Type / Stress Type	Stress (ksi) (1)	
		Seismic (2)	Thermal
Transfer Cask Structural Shell	Primary Membrane	0.5	N/A
	Membrane + Bending	30.3	N/A
	Primary + Secondary	35.6	20.3
Top Cover Plate	Primary Membrane	N/A	N/A
	Membrane + Bending	6.3	N/A
	Primary + Secondary	N/A	7.4
Bottom End Assembly	Primary Membrane	0.0	N/A
	Membrane + Bending	8.9	N/A
	Primary + Secondary	N/A	4.7

Notes:

1. Values shown are maximums irrespective of location.

2. Seismic loads assumed are for ± 0.5g applied simultaneously in three directions.

Table 8.1-11

HSM BULK AIR TEMPERATURE

HSM INLET AIR TEMPERATURE (°F)	TEMPERATURES IN REGIONS (°F)							
	1*	2	3	4	5	6	7	8
-40	-35	-30	-25	-20	-16	-11	-6	-1
0	5	11	16	21	27	32	37	43
70	76	82	89	95	101	107	113	119
100	107	113	120	126	133	139	146	152
125	132	139	146	152	159	166	173	180

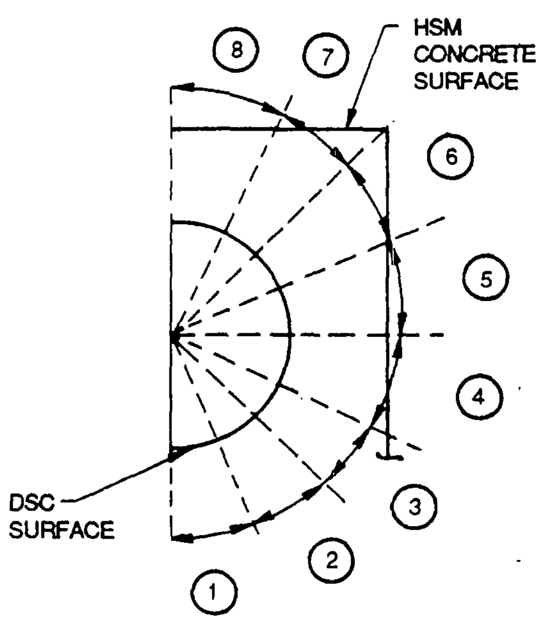


Table 8.1-12

HSM THERMAL ANALYSIS RESULTS SUMMARY

Case	HSM Air Temperature (°F)		Maximum DSC Outer Surface Temperature (°F)			Maximum Concrete Temperature (°F)			
						Roof		Side Wall	Floor
	In	Out	Bottom	Side	Top	Inside	Outside		
1	-40	-1	113	145	142	0	-30	-10	14
2	0	43	148	178	181	48	11	33	57
3	70	119	211	230	250	144	112	120	135
4	100	152	238	252	279	179	140	152	167
5	125	180	260	269	304	215	187	178	194
6	None (All vents plugged for 48 hours with outside air at -40°F)	None	332	332	332	173	-5	146	200
7	None (All vents plugged for 48 hours with outside air at 125°F)	None	445	445	455	368	187	333	395

Table 8.1-13

DSC THERMAL ANALYSIS RESULTS SUMMARY

Case	HSM Vent Air Inlet Temperature (°F)	Max. DSC Shell Temperature (°F)	Max. Fuel Cladding Temperature (°F/°C)	Average Helium Temperature (°F)	Fuel Cladding Acceptance Criteria (°F/°C)
1	70	247	643/339	375	644/340
2	125	305	668/353	410	1058/570
3	N/A (HSM Vents Plugged for 48 Hours with Ambient Air at 125°F)	455	757/403	560	1058/570
(1) 4	N/A (DSC in Cask with Internal Vacuum)	378	770/410	N/A	1058/570
(1) 5	N/A (DSC in Cask with Loss of Neutron Shield)	513	789/421	600	1058/570

Note:

1. Described in Section 8.1.3.3

Table 8.1-14

NUHOMS-24P TRANSFER CASK THERMAL ANALYSIS RESULTS SUMMARY

Case	Ambient Air Temperature (°F)	Max. Inner Liner Temp. Temperature (°F)	Max. Exterior Cask Temperature (°F)
1	-40	104	92
2	0	136	123
4	100	227	216
5	125	258	248
6	125 (Loss of Neutron Shield)	355	223
7	70 (Vacuum in DSC, Cask Vertical)	189	177

FUEL ASSEMBLY ID NUMBER	MAX. FUEL CLAD TEMP. (°F/°C)
1	425/218
2	572/300
3	643/339
4	643/339
5	573/301
6	433/223
7	451/233
8	564/296
9	564/296
10	451/233
11	415/213
12	411/211

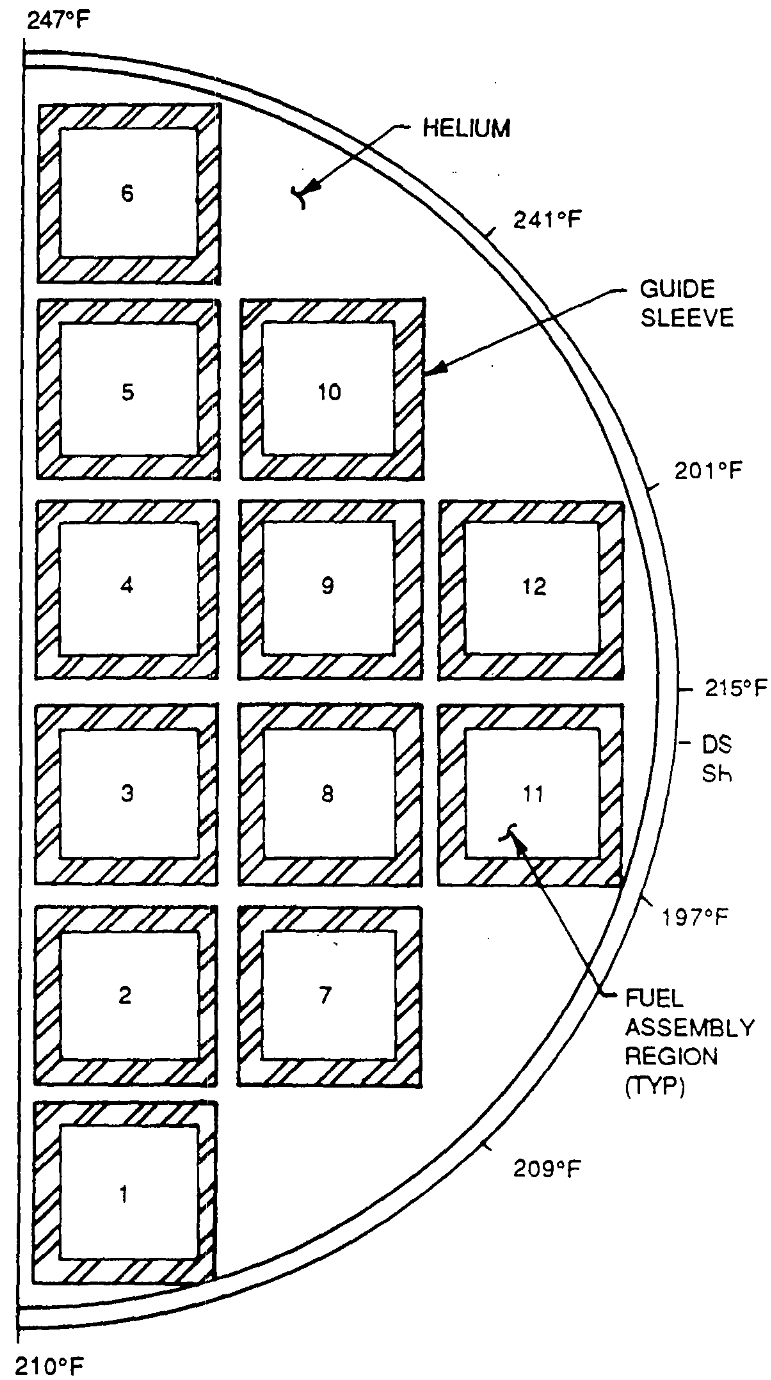


Figure 8.1-1

DSC INTERNAL TEMPERATURE DISTRIBUTION FOR 70°F AMBIENT

FUEL ASSEMBLY ID NUMBER	MAX. FUEL CLAD TEMP. (°F/°C)
1	436/224
2	585/307
3	660/349
4	660/349
5	594/312
6	456/236
7	472/244
8	585/307
9	585/307
10	481/249
11	430/221
12	430/221

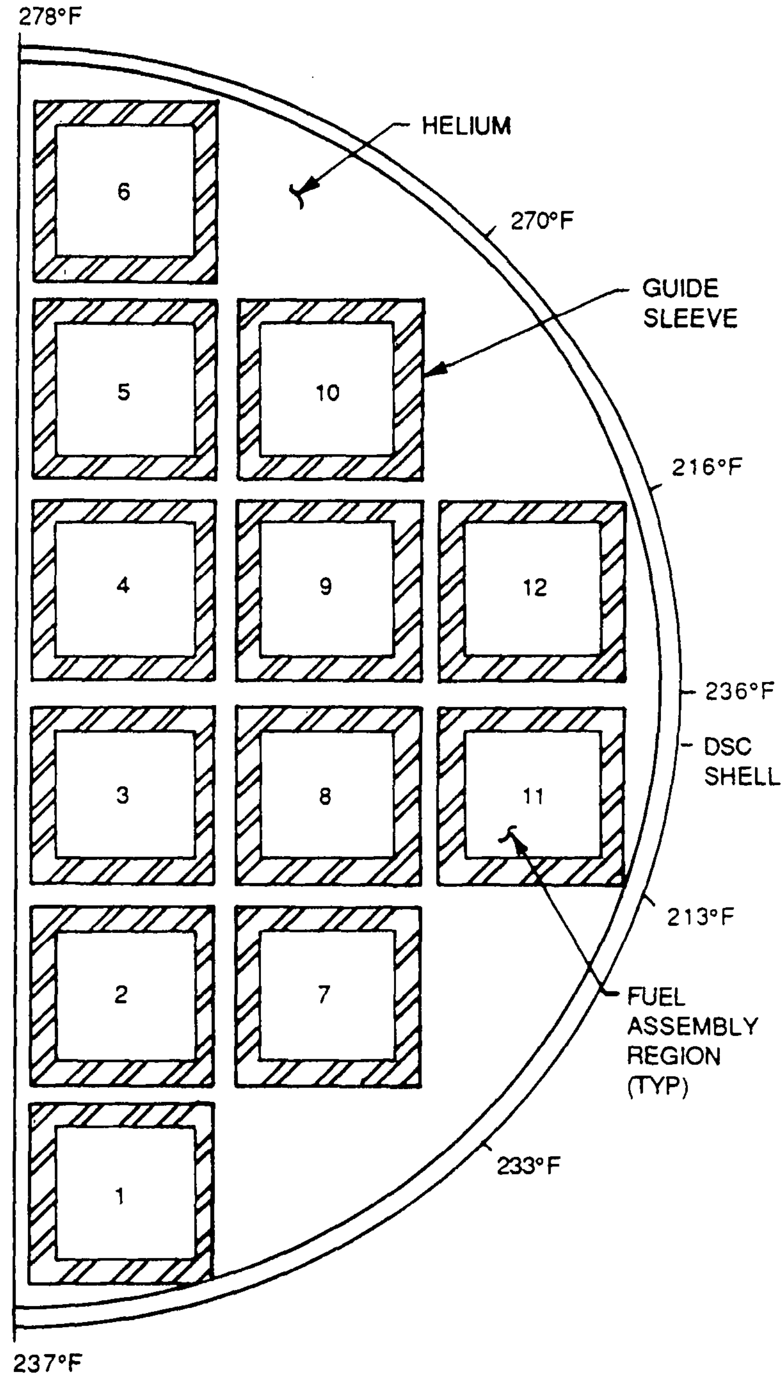


Figure 8.1-1a

DSC INTERNAL TEMPERATURE DISTRIBUTION FOR 100°F AMBIENT

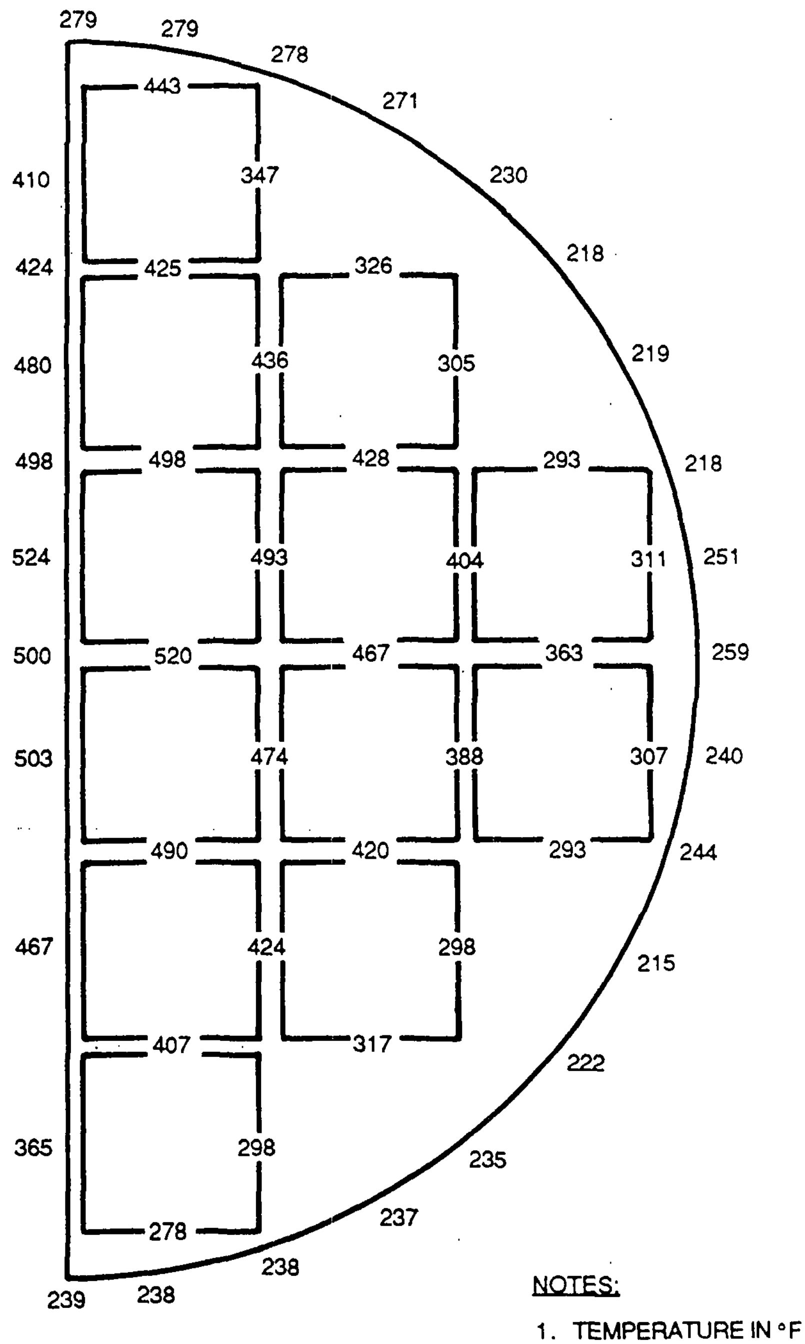
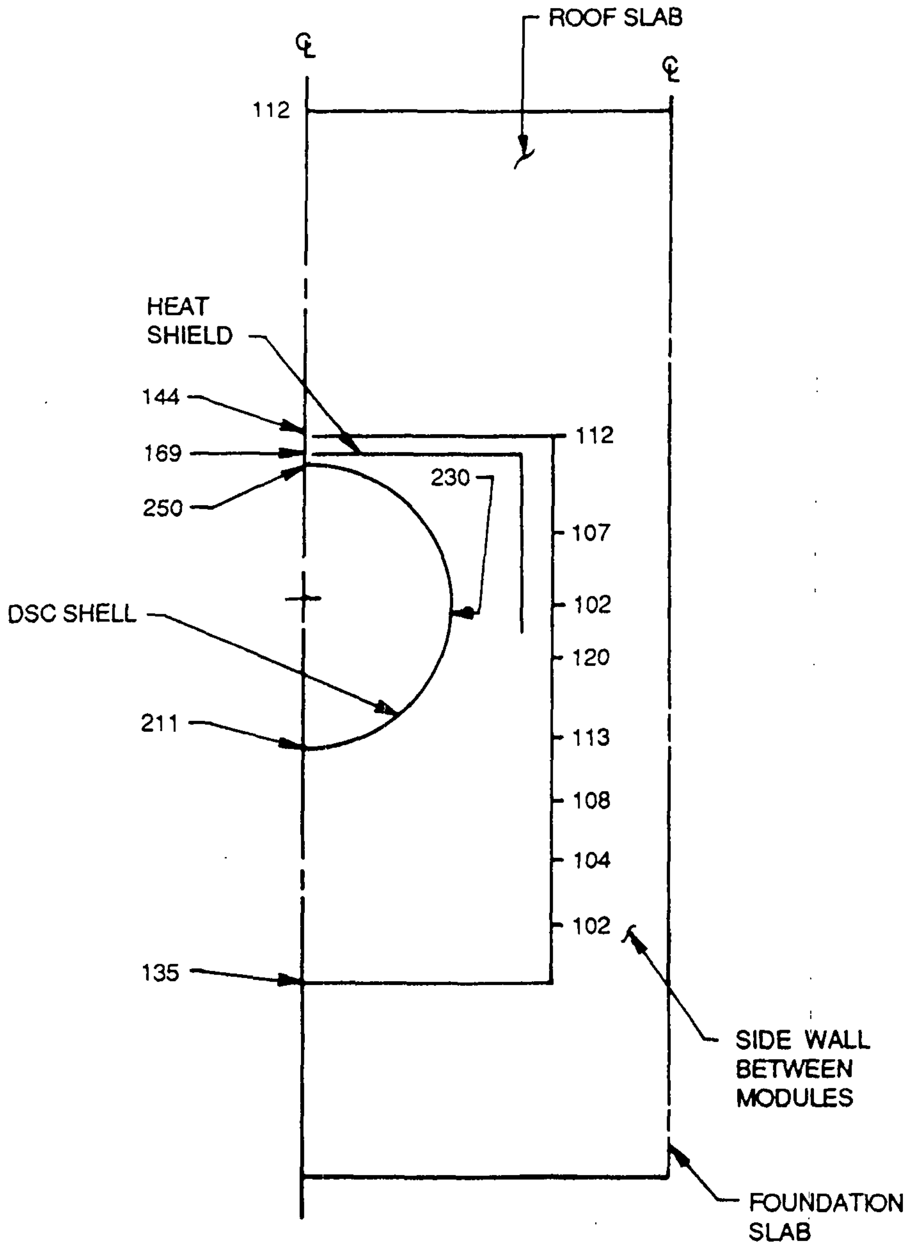


Figure 8.1-2

DSC SPACER DISK TEMPERATURE DISTRIBUTION FOR 100°F AMBIENT

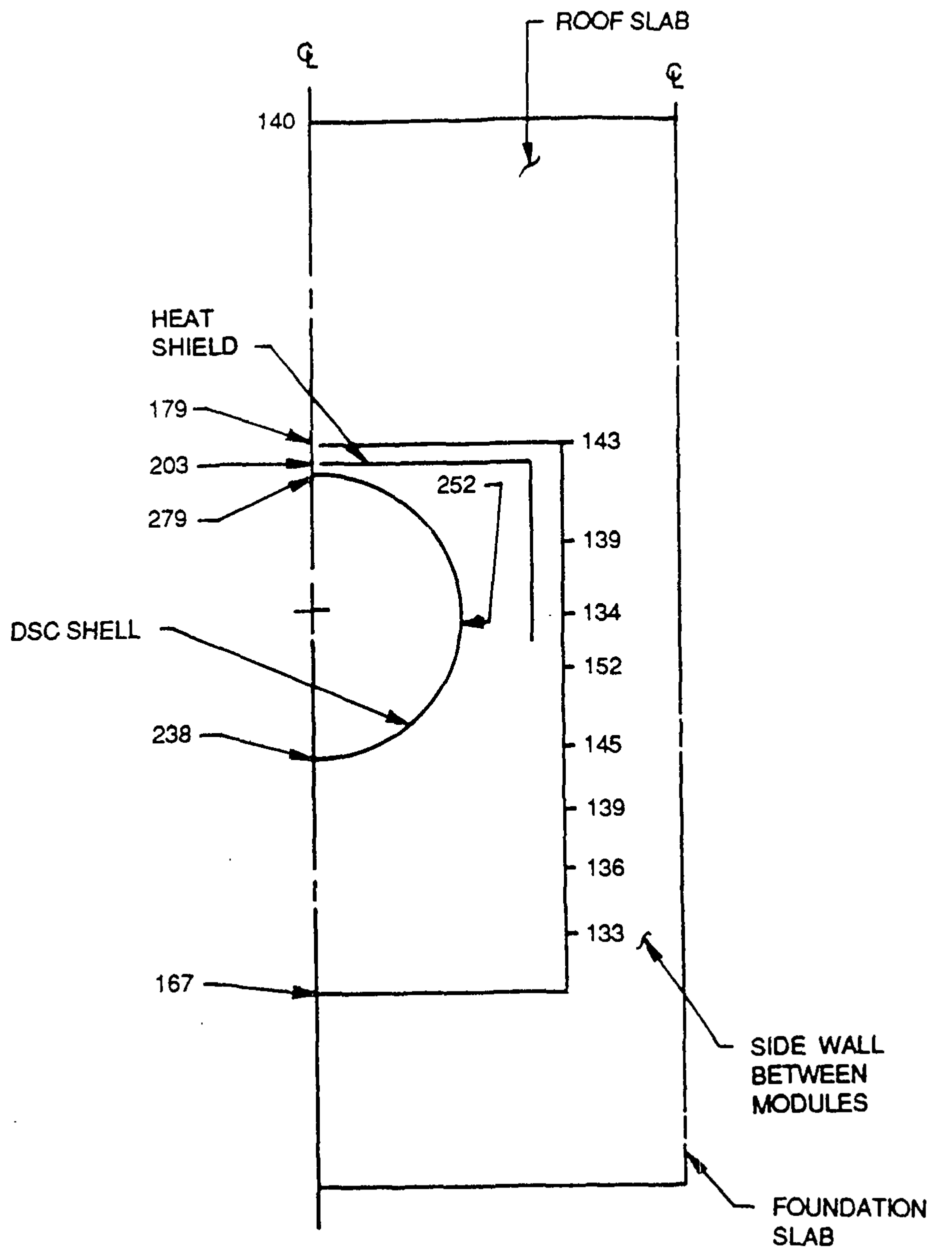


NOTES:

1. TEMPERATURE IN °F
2. RESULTS SHOWN FOR MODULE LOCATED AT CENTER OF MODULE ARRAY

Figure 8.1-3

HSM TEMPERATURE DISTRIBUTION FOR 70°F AMBIENT



NOTES:

1. TEMPERATURE IN °F
2. RESULTS SHOWN FOR MODULE LOCATED AT CENTER OF MODULE ARRAY

Figure 8.1-3a

HSM TEMPERATURE DISTRIBUTION FOR 100°F AMBIENT

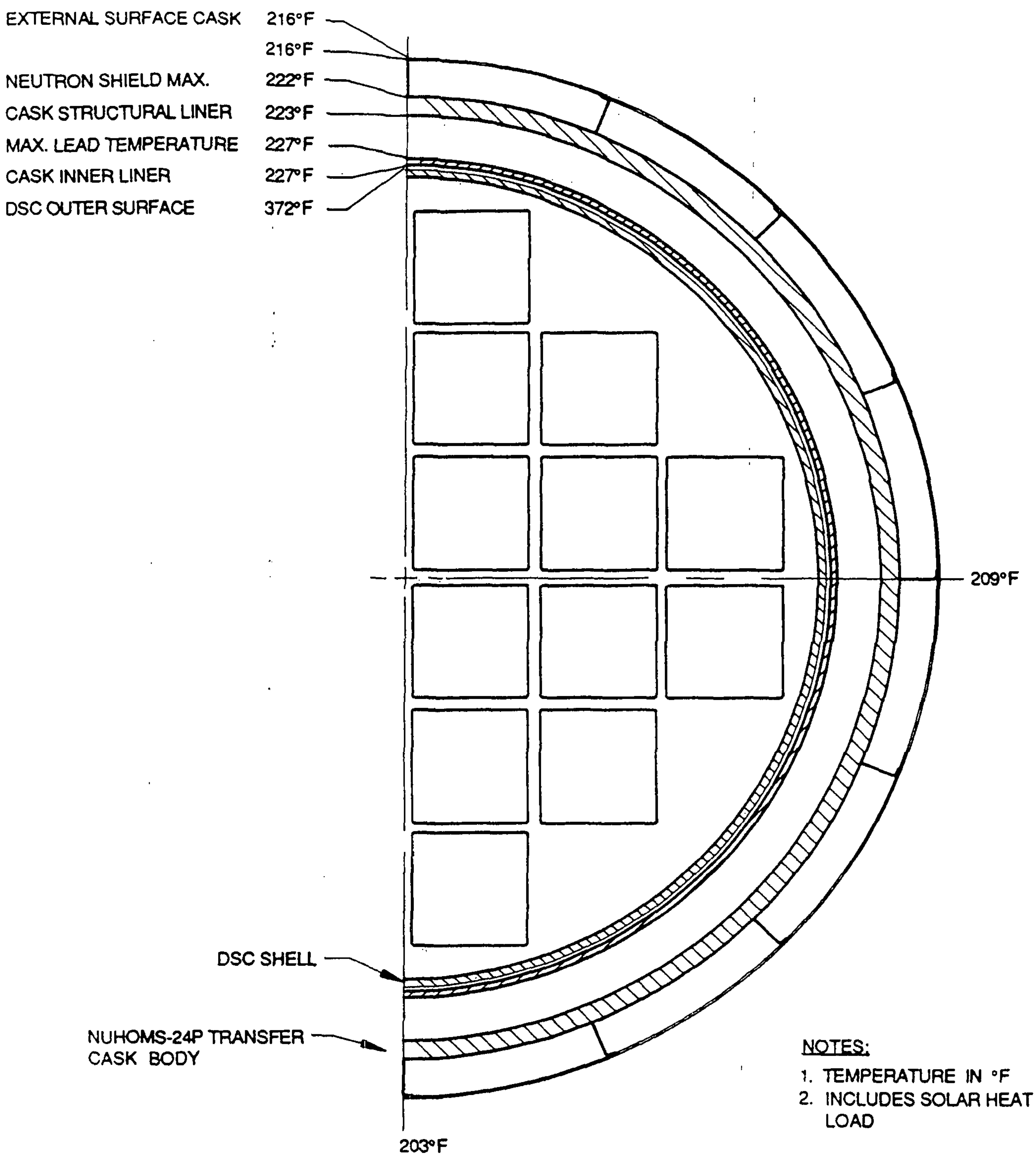
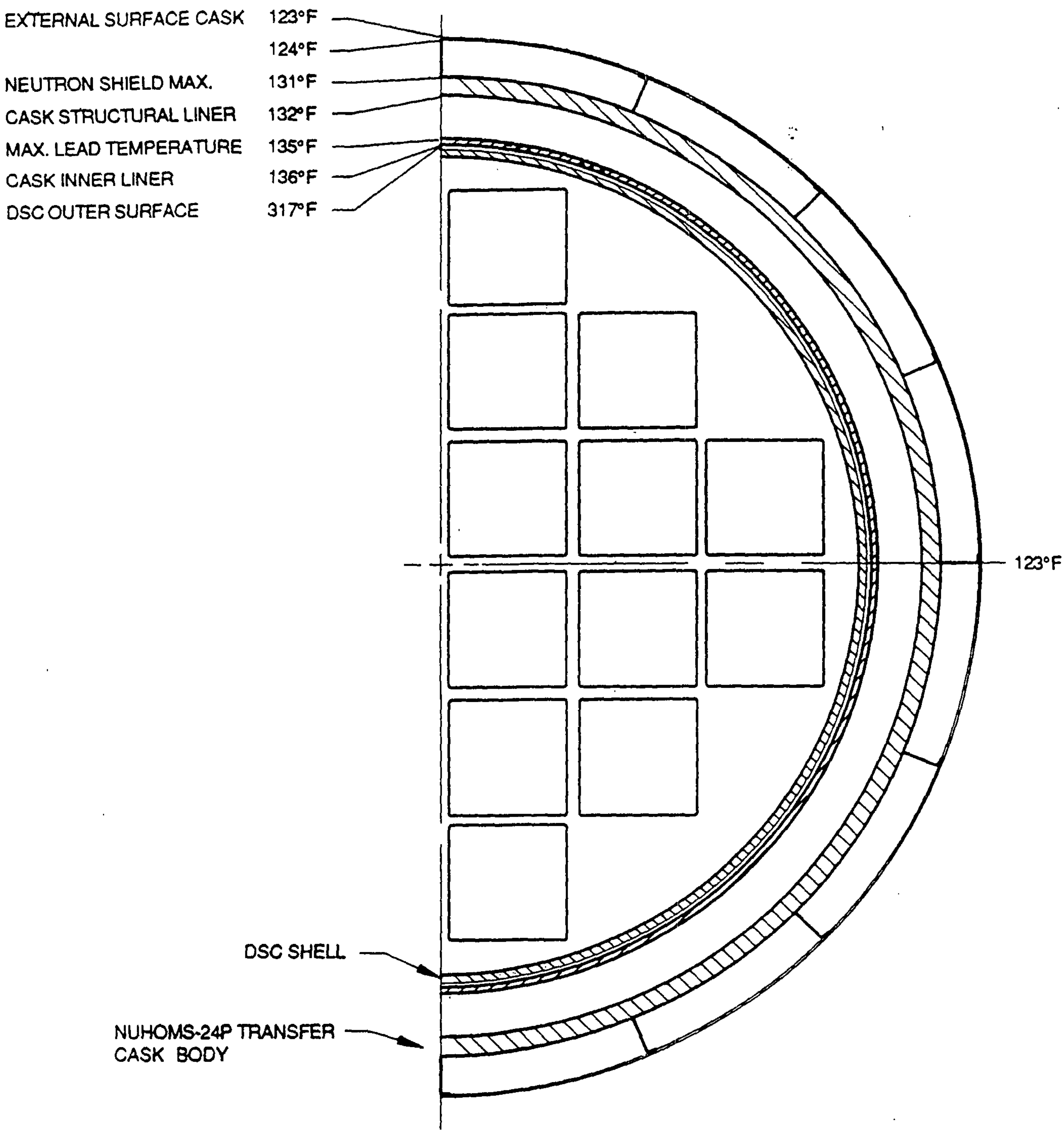


Figure 8.1-3b

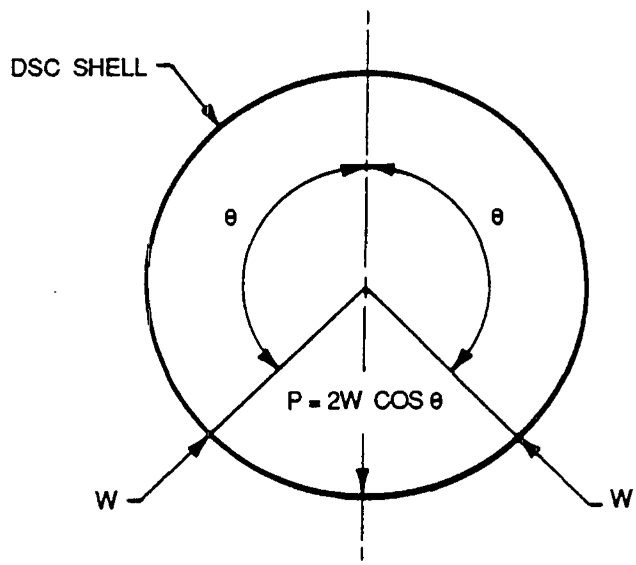
NUHOMS-24P TRANSFER CASK TEMPERATURE DISTRIBUTION FOR 100°F AMBIENT



NOTES:
1. THERE IS NO SOLAR HEAT INPUT

Figure 8.1-3c

NUHOMS-24P TRANSFER CASK TEMPERATURE DISTRIBUTION
FOR 0°F AMBIENT



KEY:
P = DEAD WEIGHT OF LOADED DSC
W=DSC SUPPORT RAIL REACTION

Figure 8.1-4

GEOMETRIC BOUNDARY FOR DSC

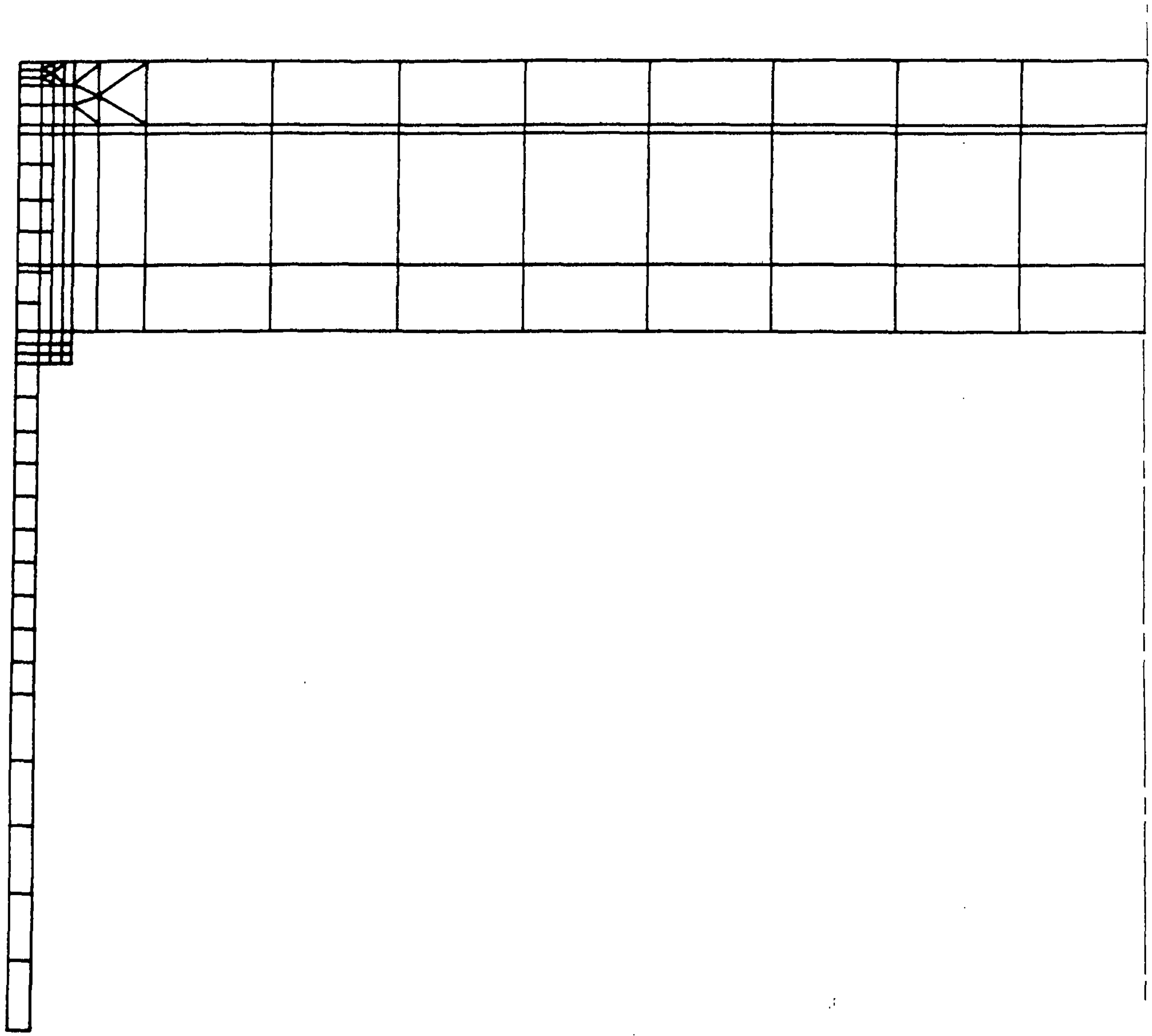


Figure 8.1-5

AXISYMETRIC MODEL OF DSC TOP END

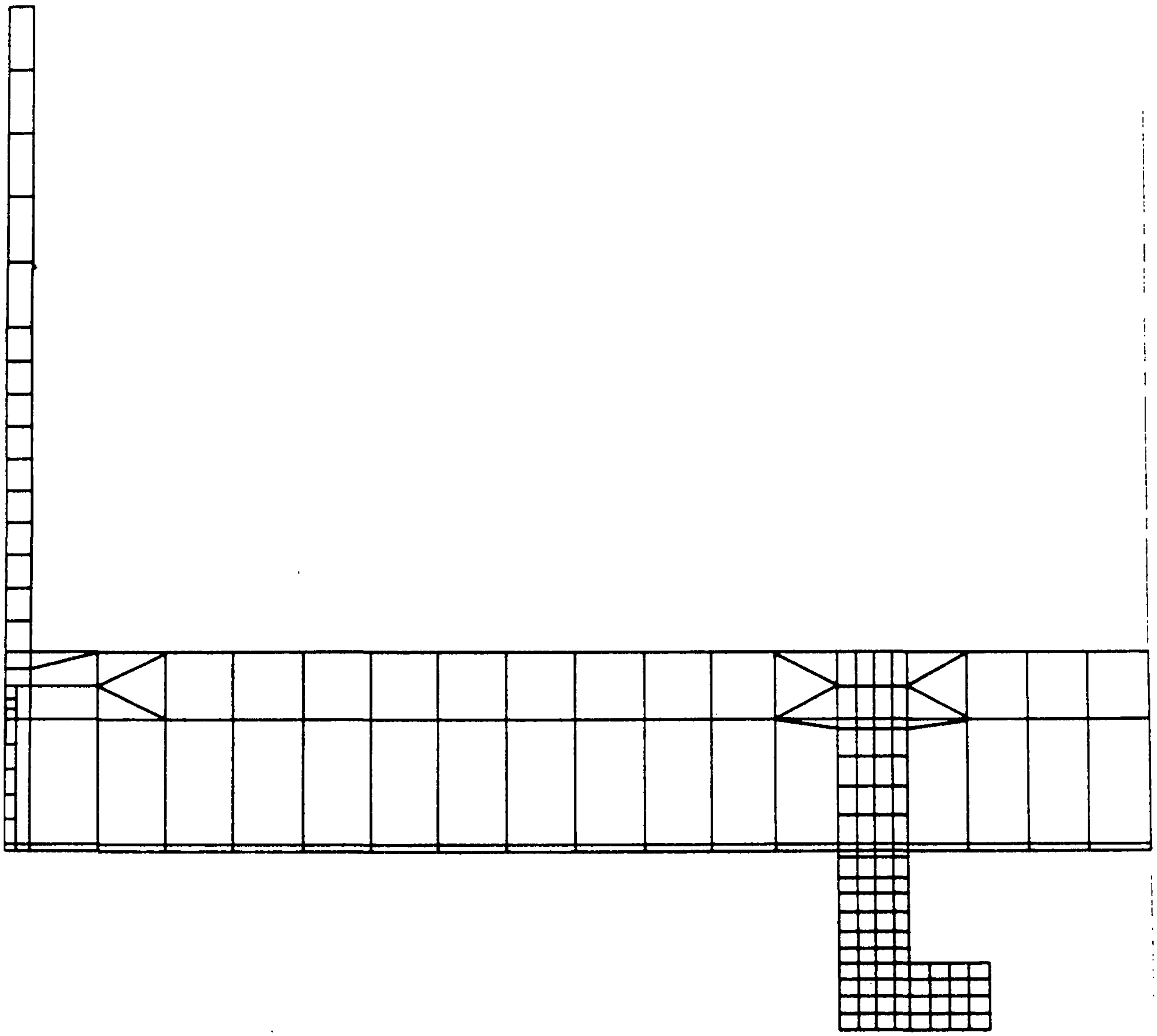


Figure 8.1-6

AXISYMETRIC MODEL OF DSC BOTTOM END

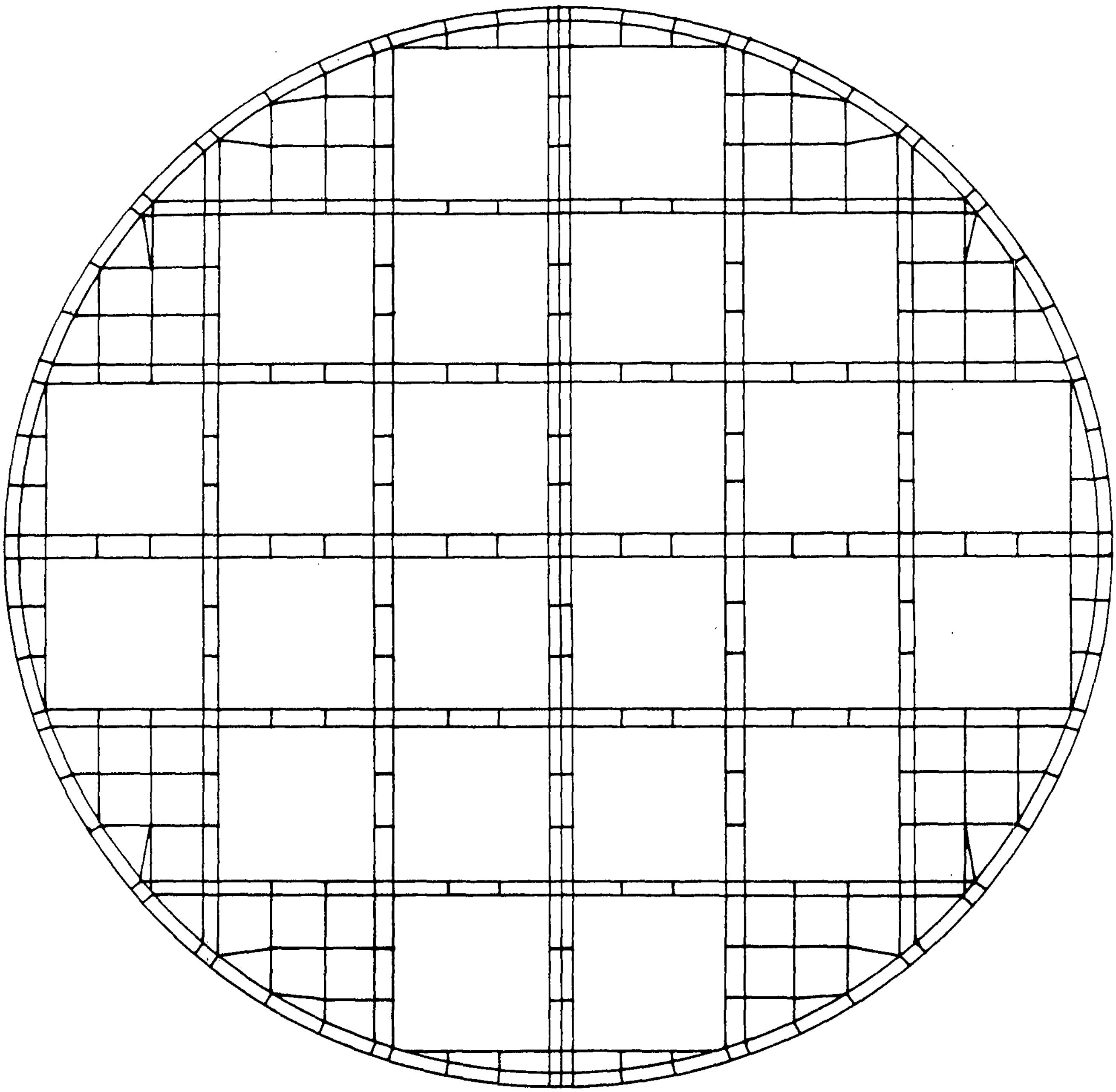


Figure 8.1-7

DSC SPACER DISK THERMAL MODEL

Figure 8.1-7a

(DELETED)

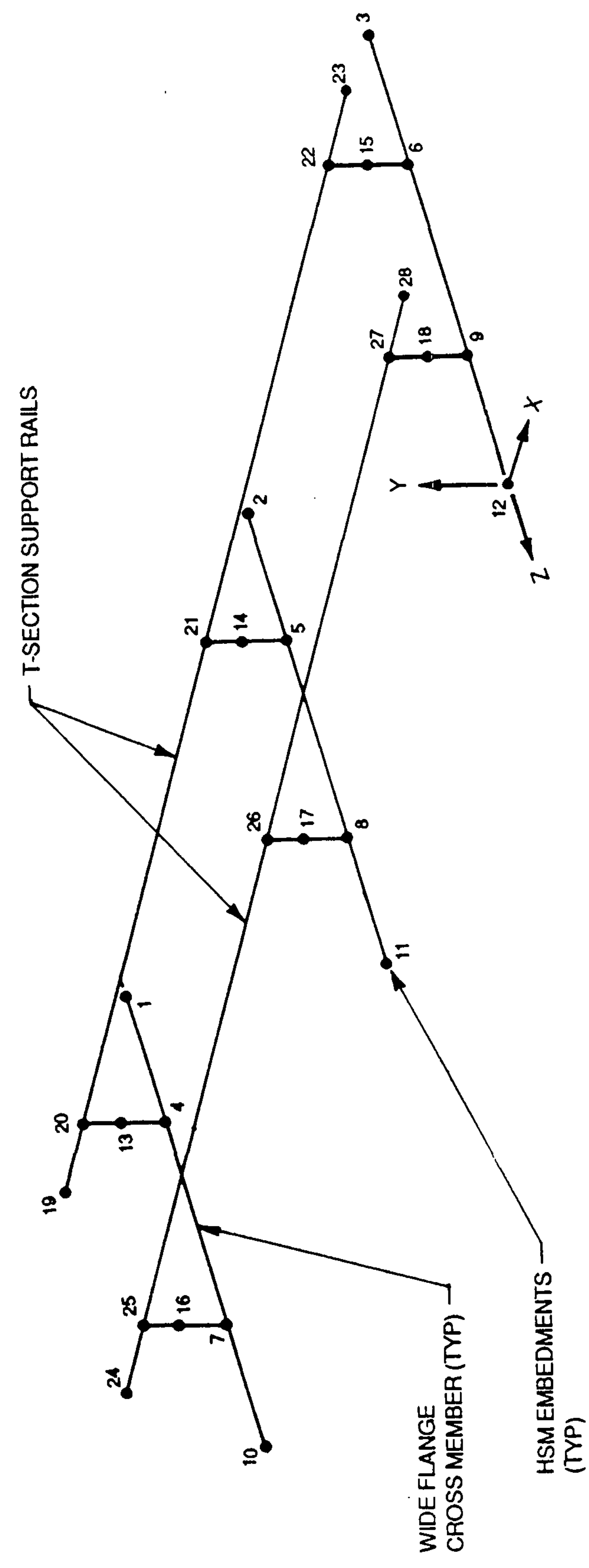


Figure 8.1-8

DSC SUPPORT STRUCTURE ANALYTICAL MODEL

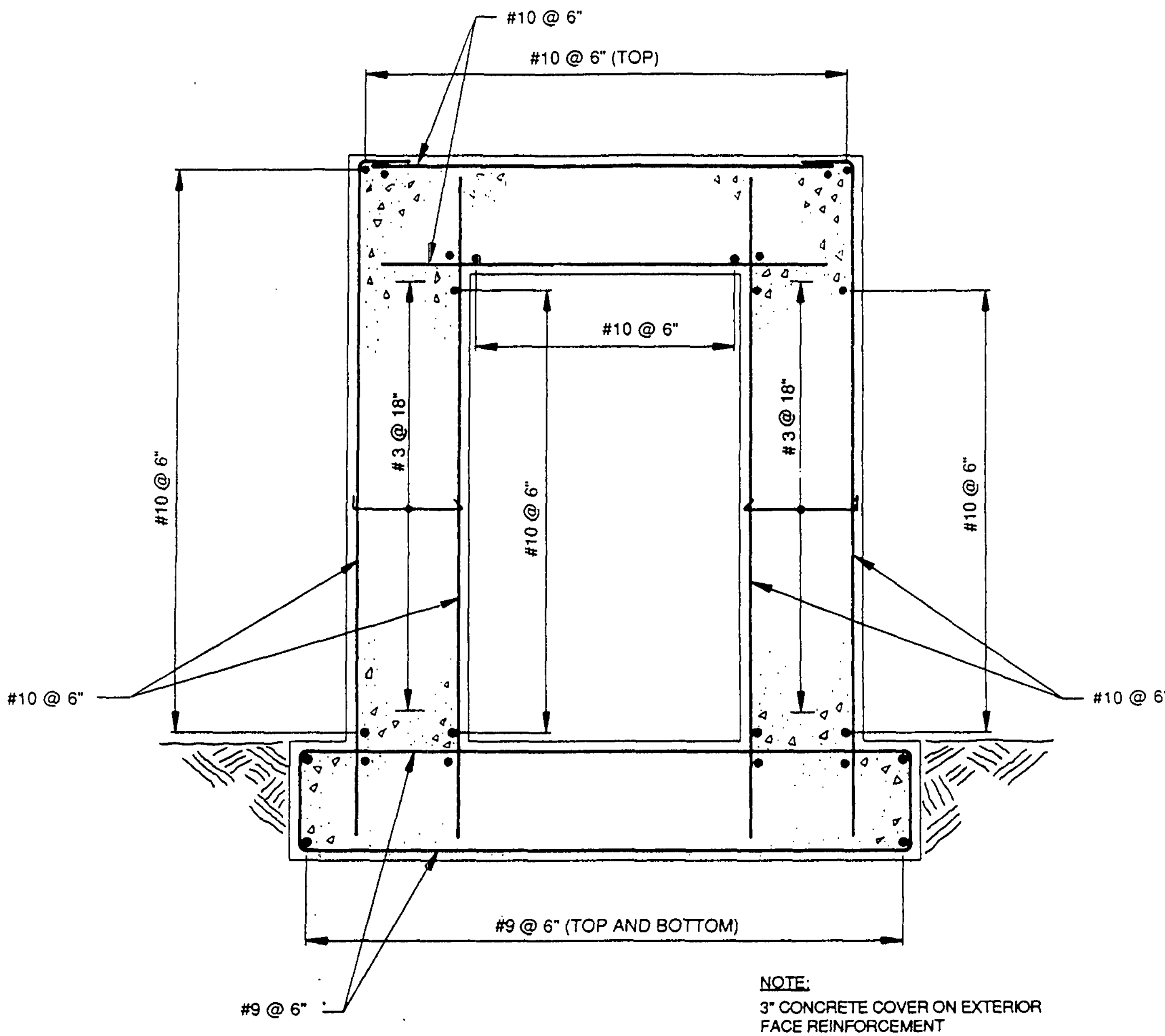


Figure 8.1-9

TYPICAL HSM REINFORCEMENT

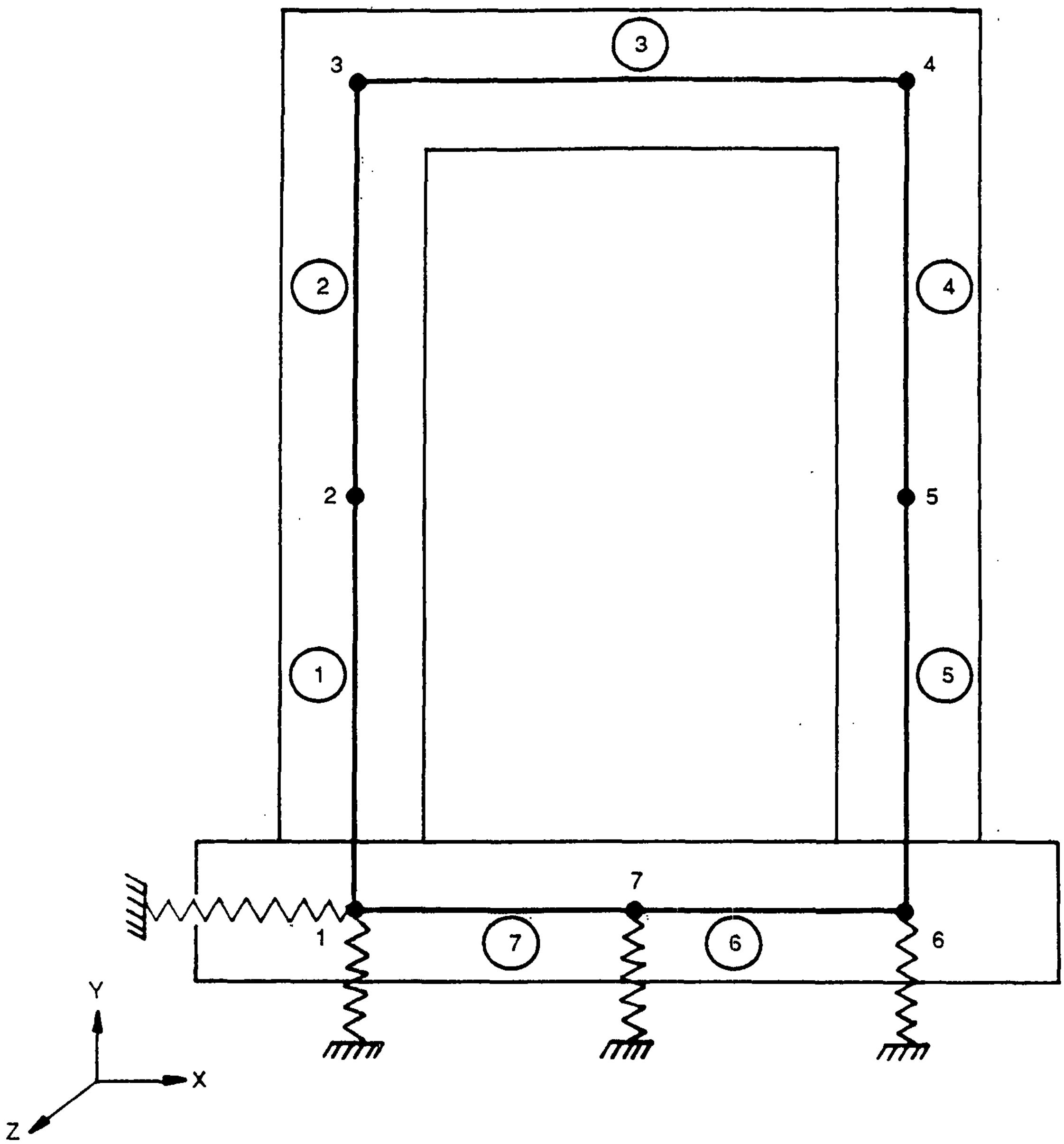


Figure 8.1-10

BEAM MODEL FOR SINGLE HSM

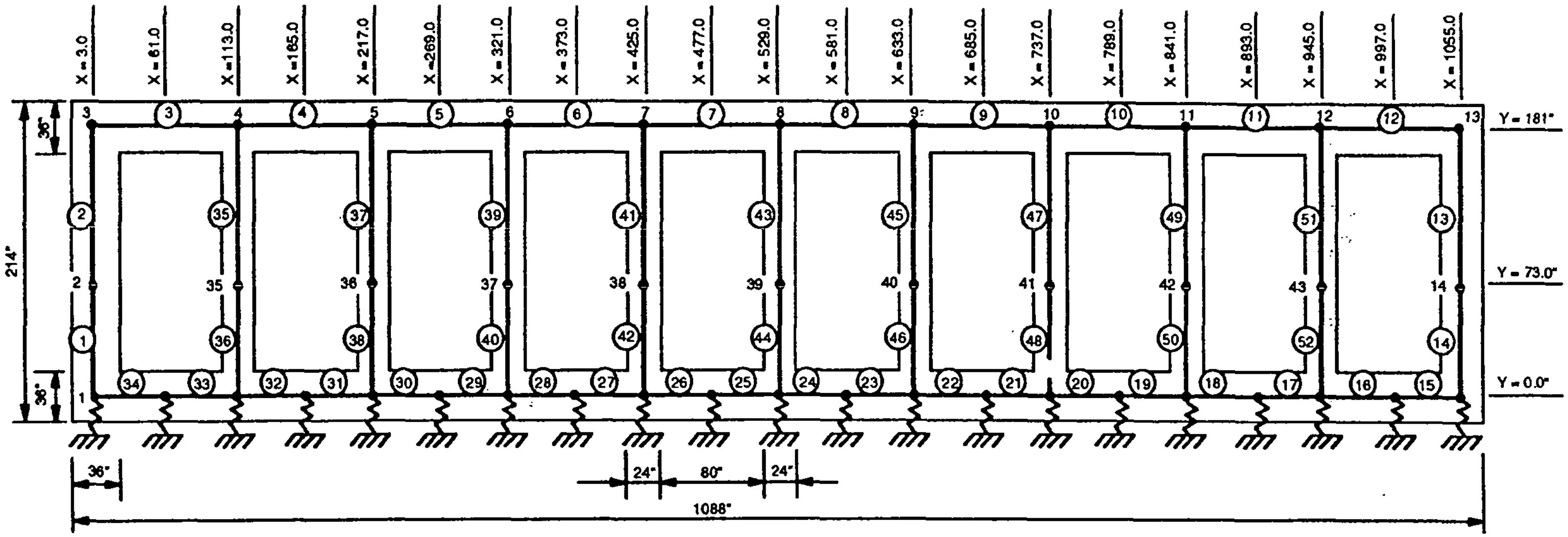
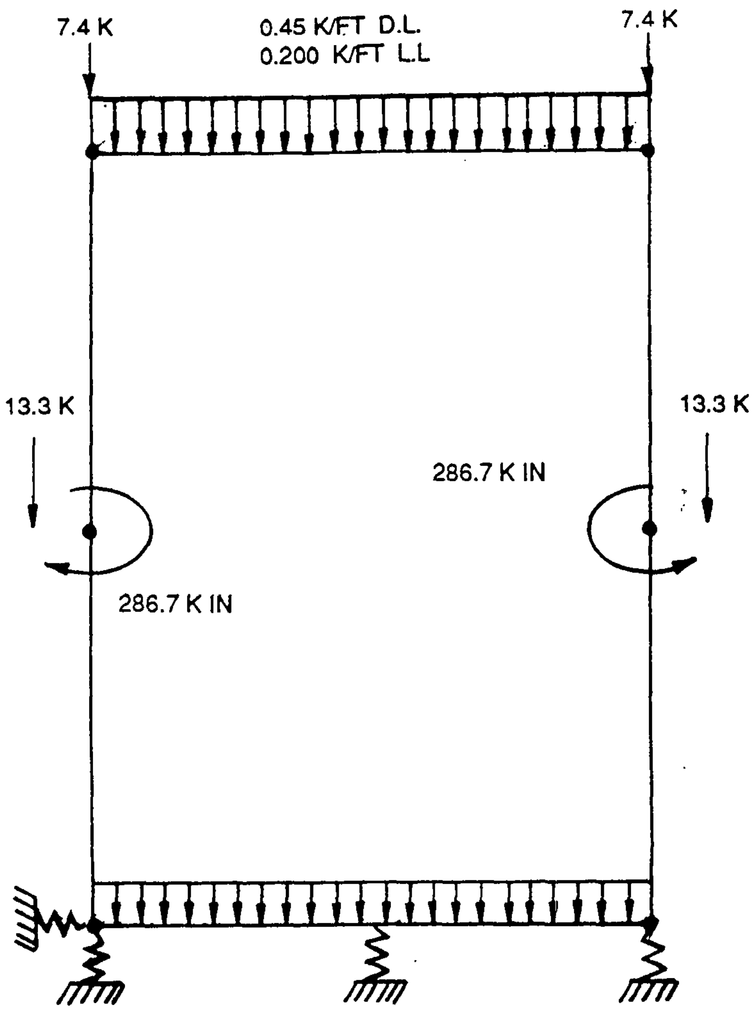


Figure 8.1-10a

BEAM MODEL FOR 2x10 ARRAY OF HSMs



NOTE:

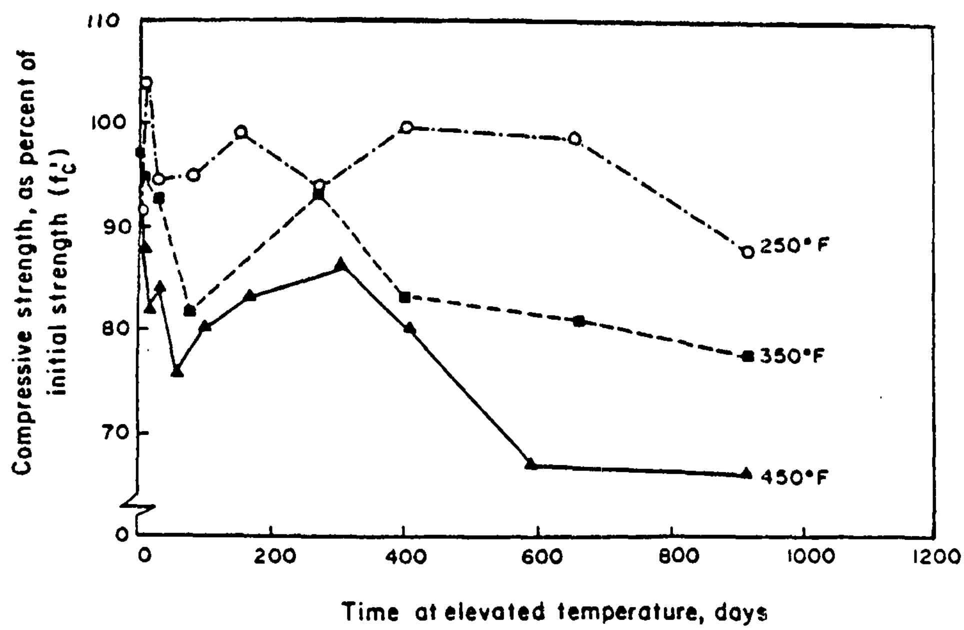
SEE SECTION 8.1.1.5, PARAGRAPH A FOR LOAD VALUE DESCRIPTION.

Figure 8.1-11

HSM DEAD/LIVE LOAD DISTRIBUTION

Figure 8.1-12

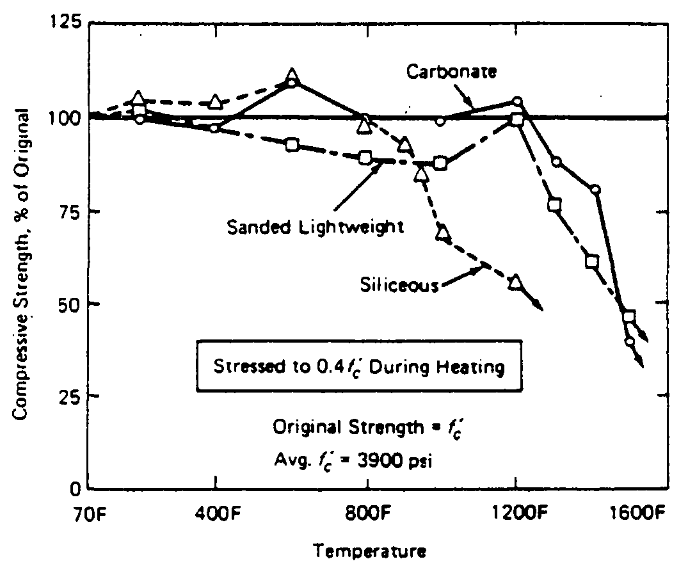
(DELETED)



(Ref. 8.24)

Figure 8.1-13

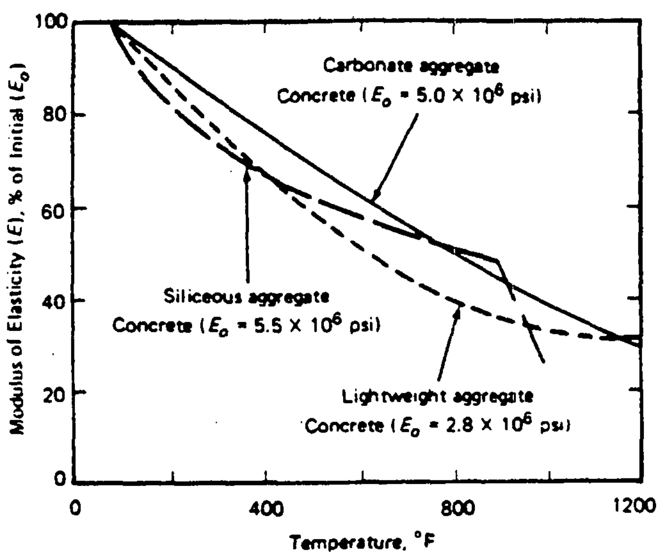
VARIATION IN STRENGTH OF BASALT AGGREGATE CONCRETE WITH TEMPERATURE AND LENGTH OF EXPOSURE



(Ref. 8.22)

Figure 8.1-14

COMPRESSIVE STRENGTH OF CONCRETE AT HIGH TEMPERATURES

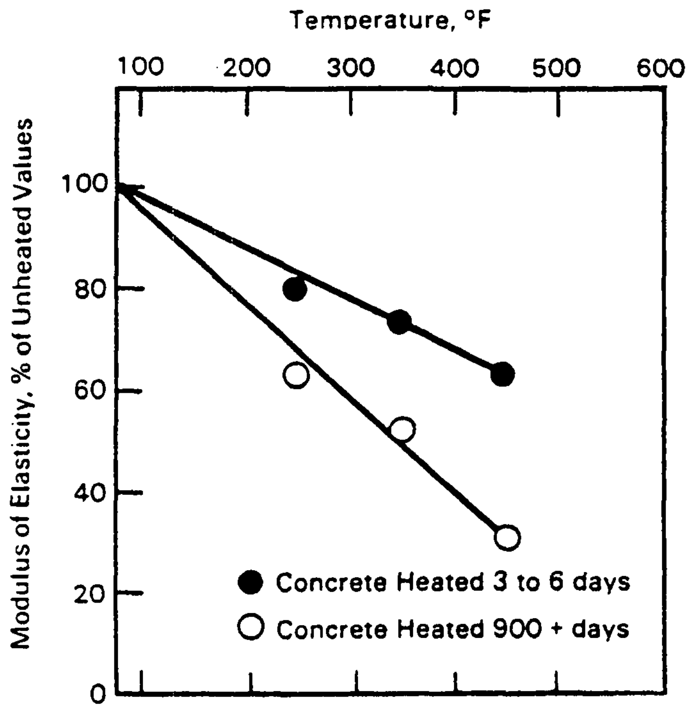


(Ref. 8.24)

Figure 8.1-15

MODULUS OF ELASTICITY OF CONCRETE

AT HIGH TEMPERATURES



(Ref. 8.22)

Figure 8.1-16

MODULUS OF ELASTICITY FOR PORTLAND CEMENT/BASALT AGGREGATE CONCRETE SUBJECTED TO SHORT AND LONG-TERM HEATING

Figure 8.1-17

(DELETED)

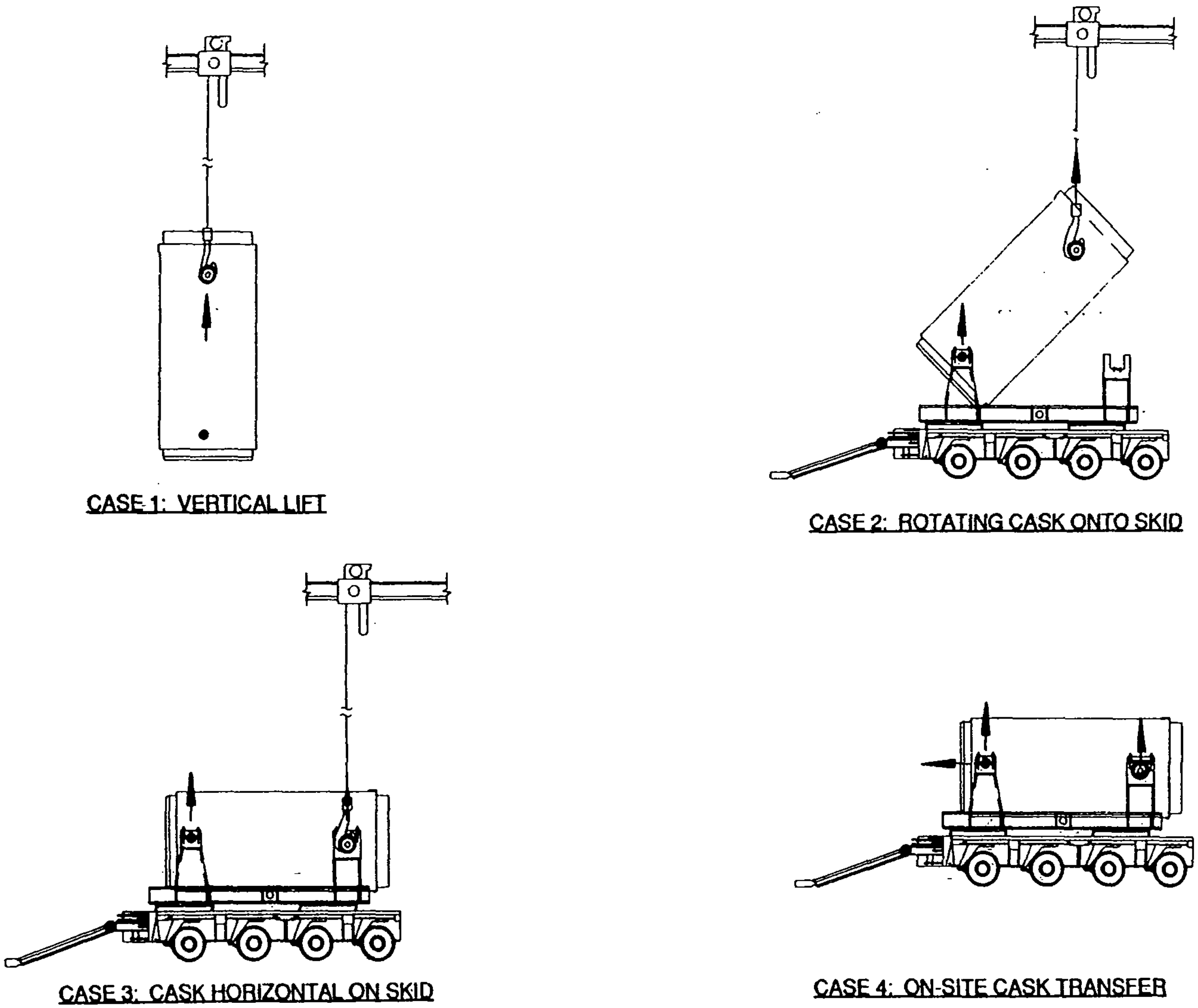


Figure 8.1-17a

NUHOMS-24P TRANSFER CASK HANDLING DESIGN/LOADING

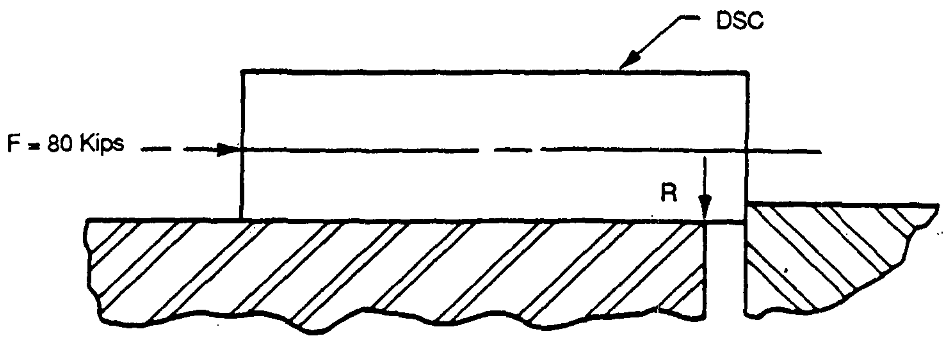


Figure 8.1-18

DSC AXIAL JAM ACCIDENT

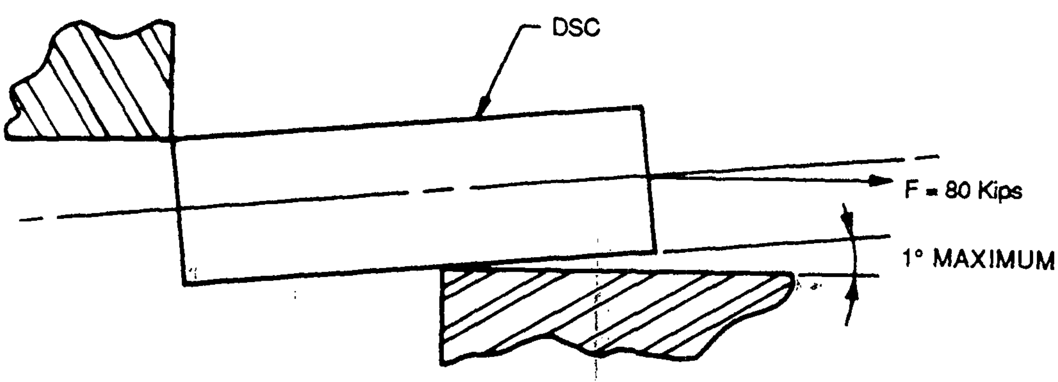
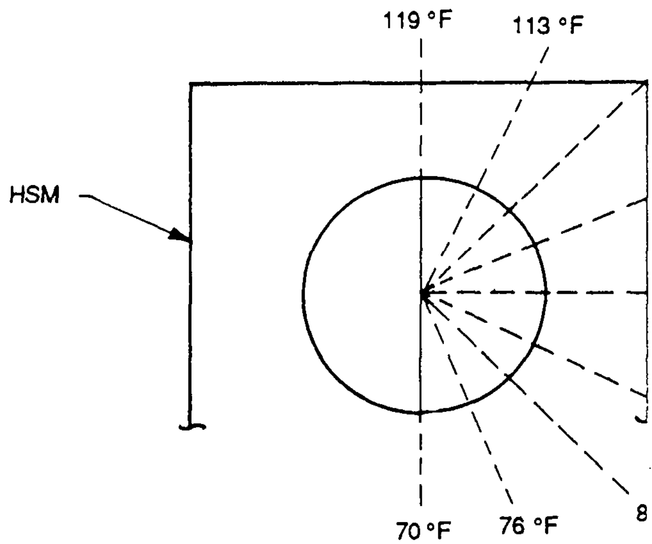


Figure 8.1-19

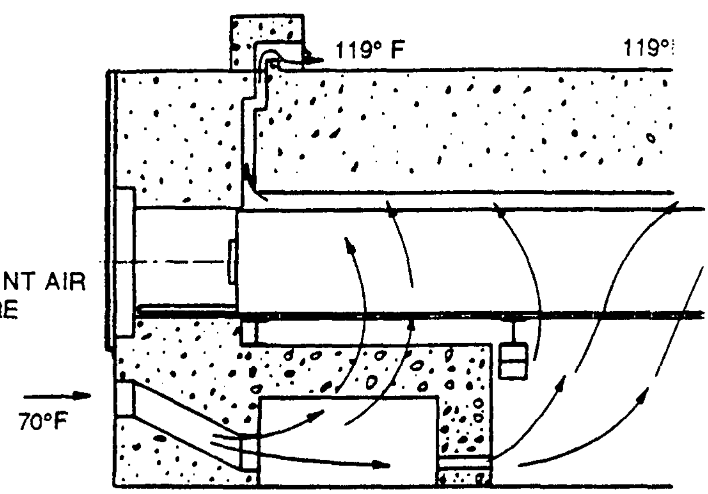
DSC BINDING (PINCHING) ACCIDENT

Figure 8.1-20

(DELETED)



BULK AIR TEMPERATURES AROUND TH
DSC INSDE THE HSM



AIR FLOW PATHS

Figure 8.1-21

HSM BULK AIR TEMPERATU

Figure 8.1-22

HEATING6 MODEL OF HSM

Figure 8.1-23

HSM HEATING6 RESULTS FOR 70^{O}F AMBIENT

Figure 8.1-23

<u>HSM HEATING6 RESULTS FOR 70°F AMBIENT</u>

Sheet 2

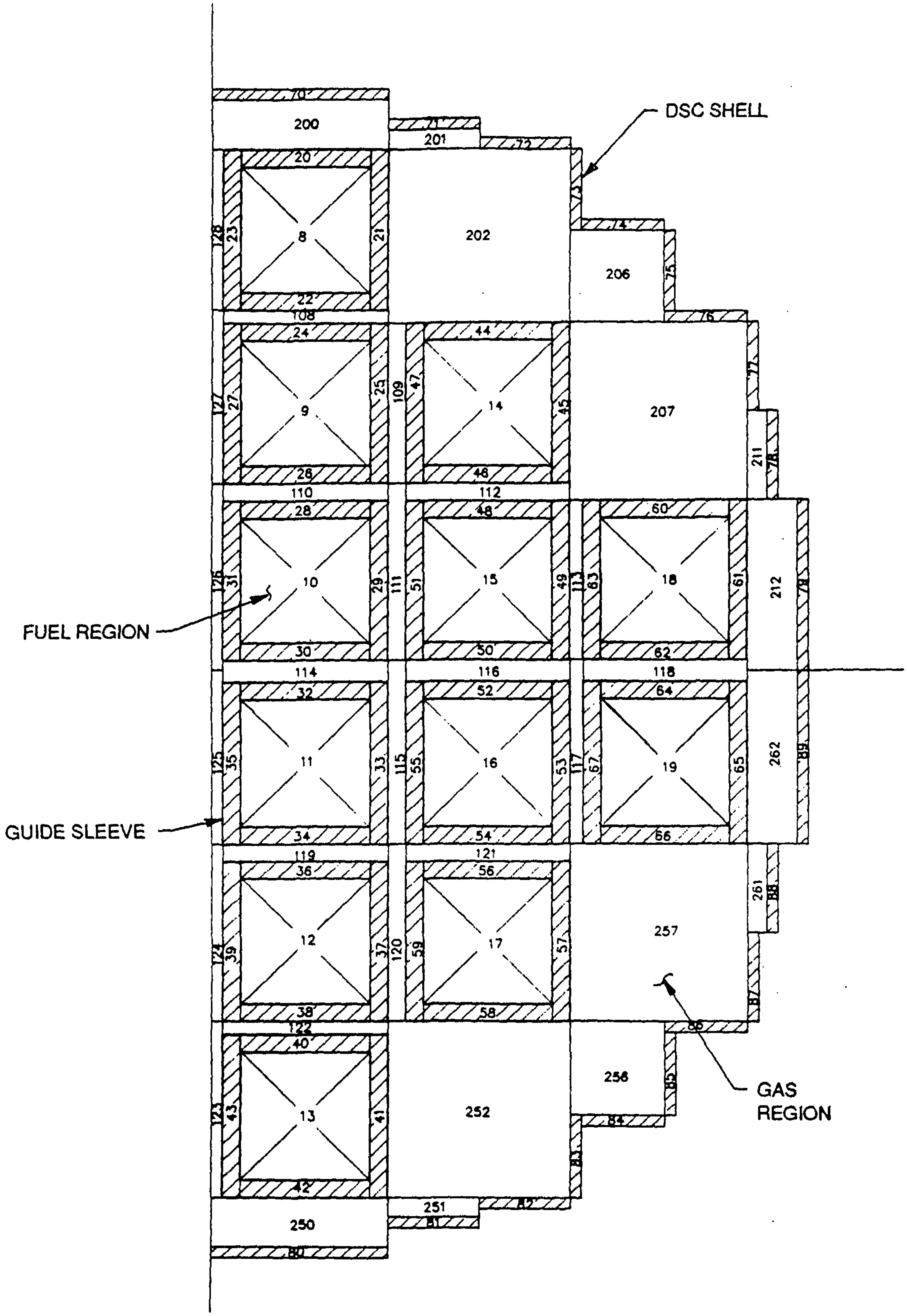


Figure 8.1-24

HEATING6 MODEL OF DSC

Figure 8.1-25

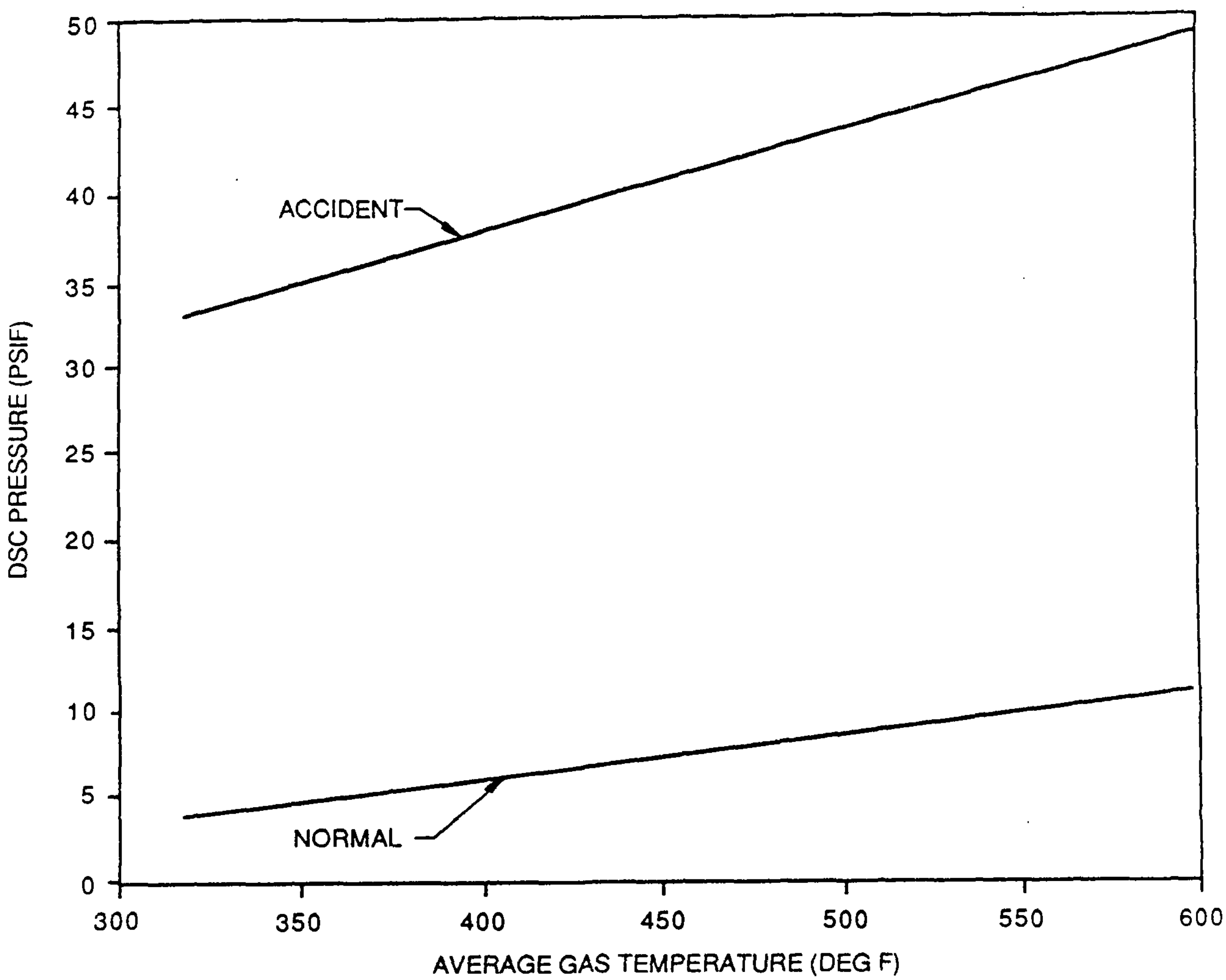
DSC HEATING6 RESULTS FOR 70°F AMBIENT
Sheet 1

Figure 8.1-25

DSC HEATING6 RESULTS FOR 70°F AMBIENT
Sheet 2

Figure 8.1-25a

NUHOMS-24P TRANSFER CASK THERMAL MODEL
FOR TOP AND BOTTOM OF THE CASK



NOTES:

1. NORMAL ASSUMES FUEL ASSEMBLIES ARE INTACT
2. ACCIDENT ASSUMES 100% CLADDING RUPTURE AND FILL GAS RELEASE, AND 30% FISSION GAS RELEASE

Figure 8.1-25b

DSC INTERNAL PRESSURE

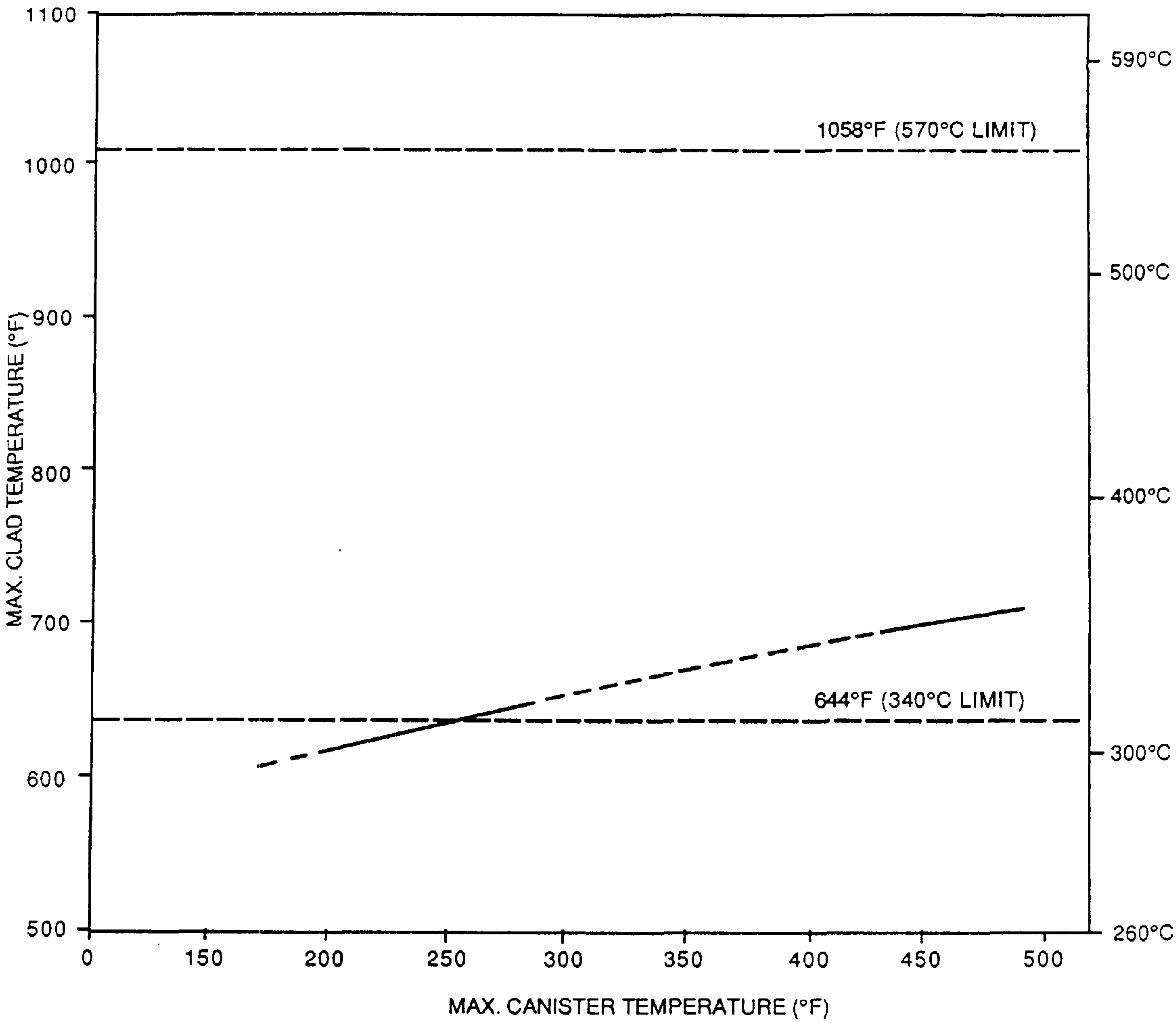


Figure 8.1-26

MAXIMUM FUEL CLAD TEMPERATURE VS.
DSC SURFACE TEMPERATURE

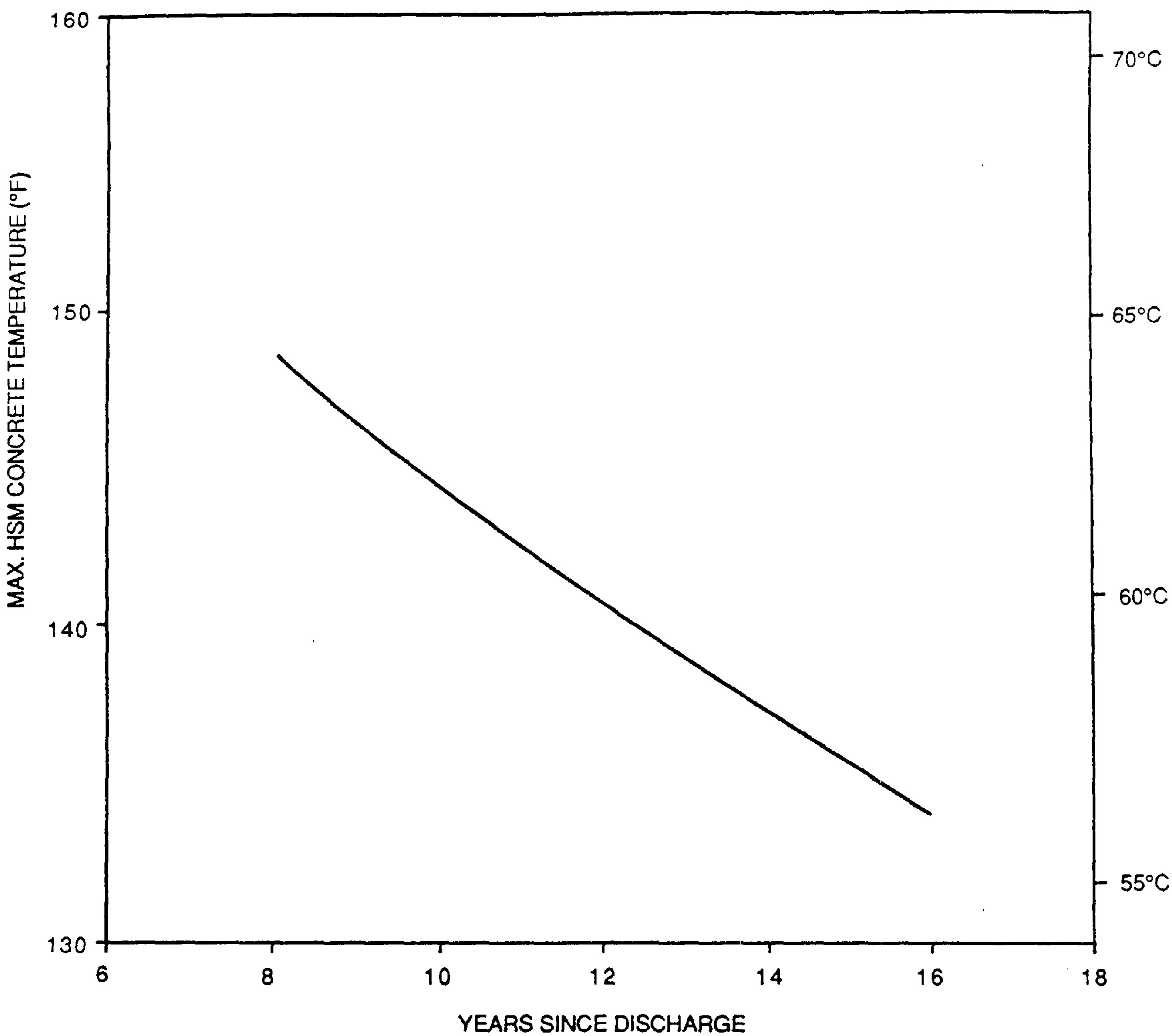


Figure 8.1-27

MAXIMUM CONCRETE TEMPERATURE VS.

YEARS SINCE DISCHARGE

(70°F AMBIENT TEMPERATURE)

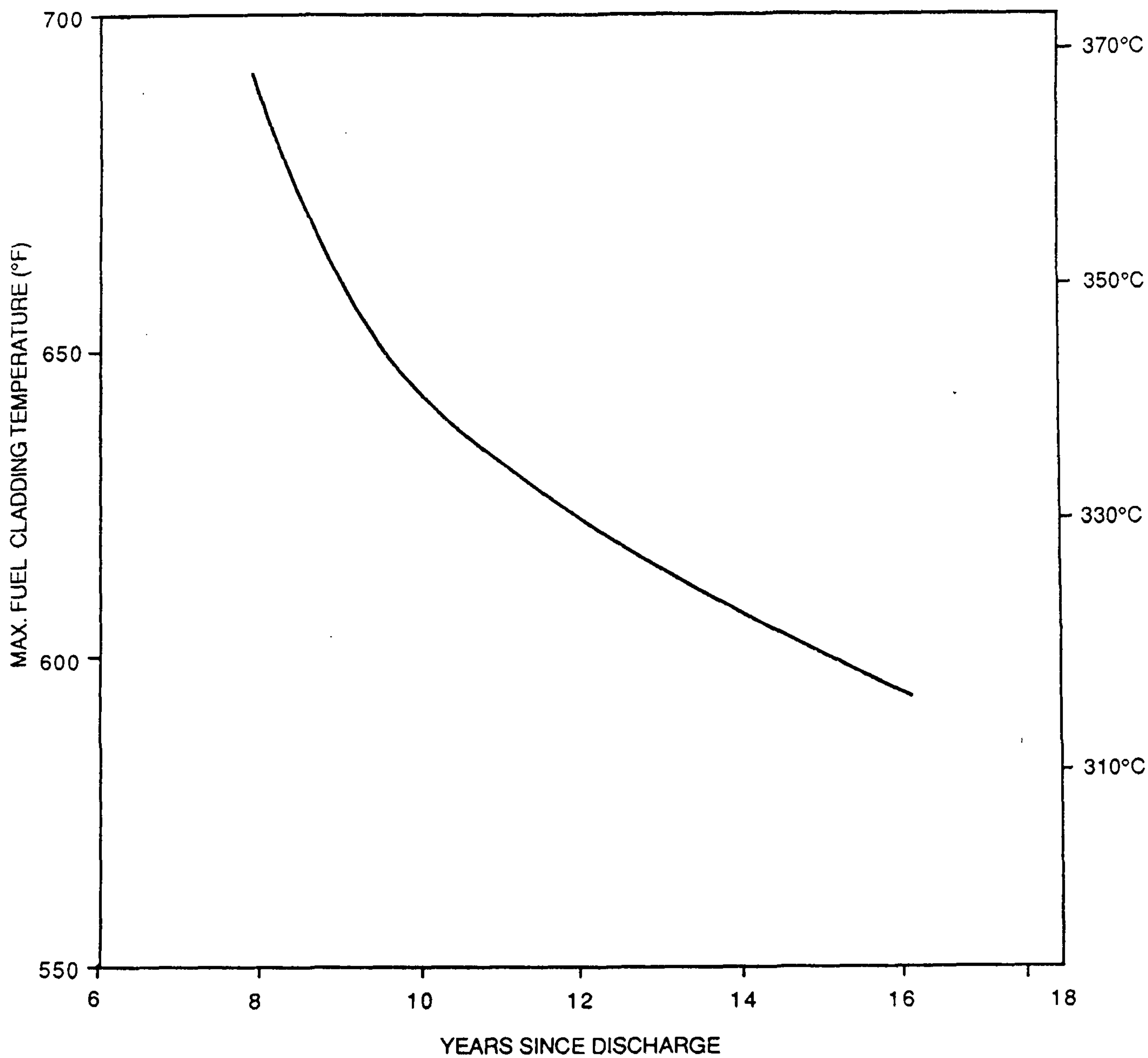


Figure 8.1-28

MAXIMUM FUEL CLAD TEMPERATURE VS.

YEARS SINCE DISCHARGE

(70°F AMBIENT TEMPERATURE)

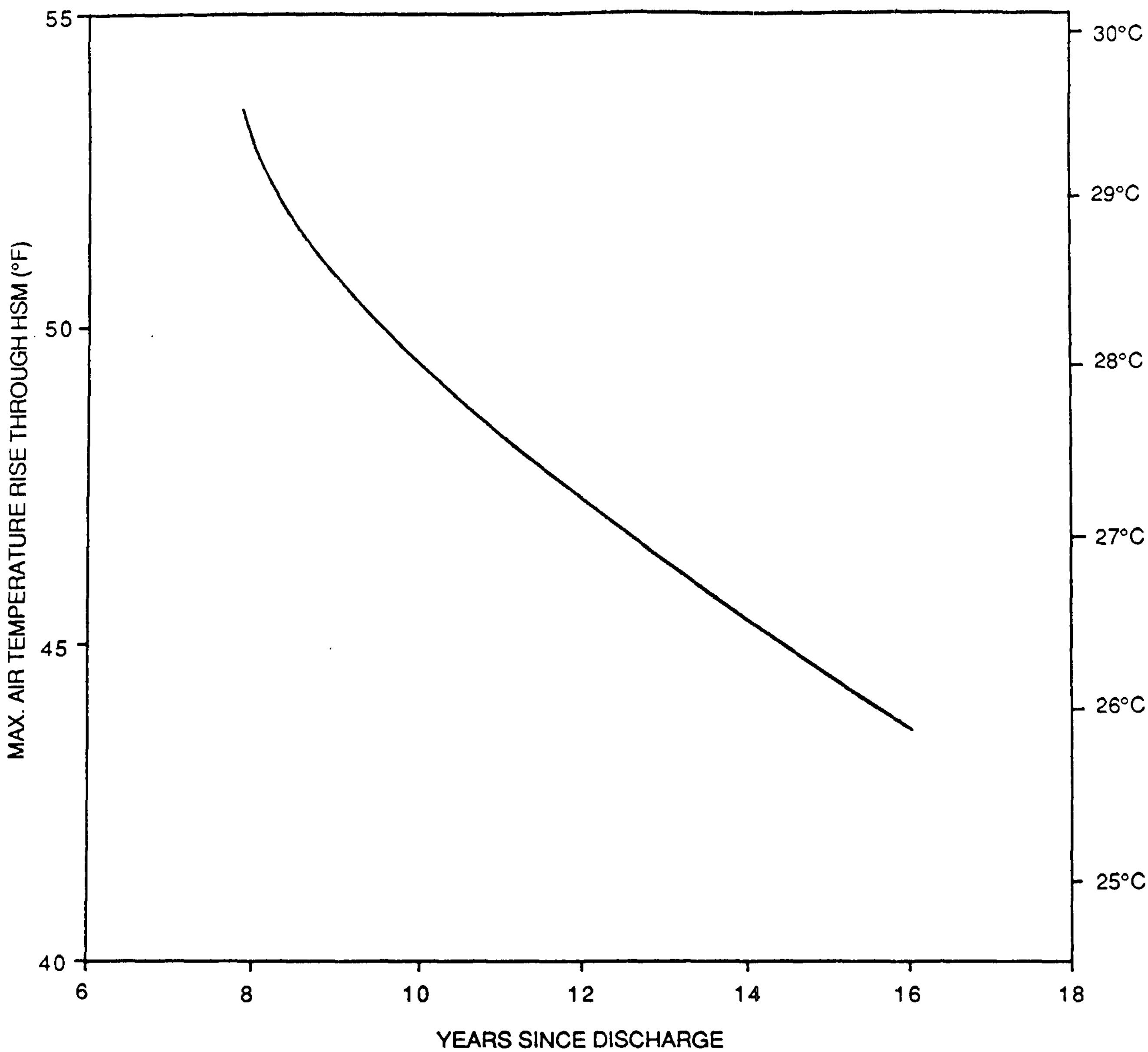


Figure 8.1-29

MAXIMUM AIR TEMPERATURE RISE THROUGH HSM VS. YEARS SINCE DISCHARGE (70°F AMBIENT TEMPERATURE)

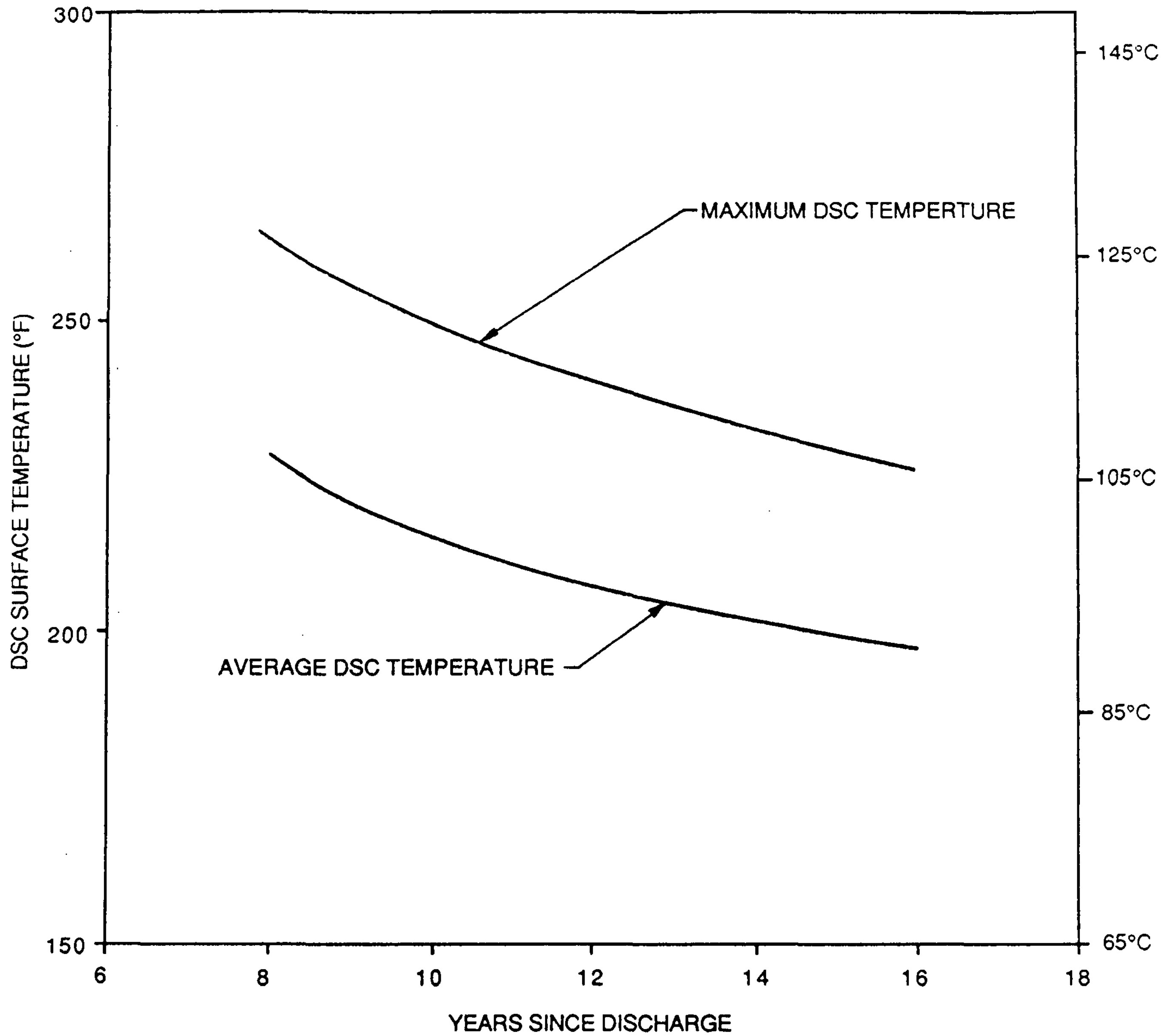


Figure 8.1-30

DSC SURFACE TEMPERATURE VS. YEARS SINCE DISCHARGE

8.2 Accident Analyses

The accident events for design specified by ANSI/ANS 57.9-1984, and other credible accidents which could affect the safe operation of the NUHOMS system are addressed in this section. Analyses are provided for a range of postulated accidents, including those with the potential to result in doses greater than 25 mrem outside the owner controlled area in accordance with 10CFR72. The postulated accidents considered in the analysis and the associated NUHOMS-24P components affected by each accident condition are shown in Table 8.2-1.

In the following sections, each accident condition is analyzed to demonstrate that the requirements of 10CFR72.72 are met and that adequate safety margins exist for the NUHOMS-24P system design. Radiological calculations were performed to confirm that on-site and off-site dose rates are within acceptable limits. The resulting accident condition stresses in the NUHOMS components were evaluated and compared with the applicable code limits set forth in Section 3.2 of this report. Where appropriate, these accident condition stresses were combined with those of normal operating loads in accordance with the load combination definitions in Tables 3.2-5, 3.2-5a, and 3.2-5b. Load combination results for the HSM, DSC, and transfer cask as well as the evaluation for fatigue effects are presented in Section 8.2.10.

The postulated accident conditions addressed in this report section include:

1. Loss of HSM air outlet shielding blocks.
2. Tornado winds and tornado generated missiles.
3. Design basis earthquake.
4. Design basis flood.
5. Accidental transfer cask drop with loss of neutron shield.
6. Lightning.
7. Debris blockage of HSM ventilation air inlets and outlets.
8. Postulated DSC leakage.
9. Pressurization due to fuel cladding failure within the DSC.

For each postulated condition, the accident cause, the structural, thermal, and radiological consequences, and the recovery measures required to mitigate the accident are discussed.

8.2.1 Loss of HSM Air Outlet Shielding Blocks

This postulated accident is the loss of both air outlet shielding blocks (front and rear) from the roof of an HSM. All other components of the NUHOMS system are assumed to be functioning normally.

8.2.1.1 Cause of Accident The HSM air outlet shielding blocks are designed to remain intact for all postulated events except the direct impact of a large tornado generated missile, which is highly unlikely. For this event the shielding blocks are conservatively assumed to be impacted by a tornado missile which impairs the blocks from performing their shielding function. For the sake of this conservative generic analysis, it is further assumed that both shielding blocks on a single HSM are completely lost.

8.2.1.2 Accident Analysis There are no structural or thermal consequences which affect the safe operation of the NUHOMS system resulting from the loss of the HSM air outlet shielding blocks. The ventilation air flow resistance is reduced without the shield blocks and, hence, the air flow will increase slightly and provide added heat removal capacity for the DSC. The radiological consequences of this accident are described in the next section.

8.2.1.3 Accident Dose Calculations The off-site radiological effects which result from a loss of the HSM vent outlet shielding blocks is an increase in the air scattered (skyshine) doses. On-site radiological effects result from an increase in direct radiation during recovery operations performed on the HSM roof, and skyshine radiation. The calculation of these doses during normal conditions is described in Section 7.3.2.2. Removal of the shield blocks results in a local increase in the surface dose of 3600 mrem/hour inside the HSM vent outlet openings. This increased surface dose was used in the analytical models described in Section 7.4.2 to calculate the direct and scattered doses as a function of distance from the HSM. Table 8.2-2 shows comparisons of the increased dose rate as a function of distance due to a loss of the HSM shielding blocks. The dose increase to a person located 100 meters away from the NUHOMS installation for eight hours a day for seven days (estimated recovery time) would be 30 mrem. The increased dose to an off-site person for 24 hours a day for seven days located 2000 feet away would be 2E-4 mrem.

8.2.1.4 Recovery To recover from an accident resulting in a loss of the HSM shielding blocks, replacement shielding blocks are either cast in place or are precast and installed on the HSM roof. Spare shielding blocks may be kept on site as a contingency measure by the licensee. A precast shield block may be transported to the HSM using a truck-mounted crane which is also used to lift the replacement shield block into position. The replacement shield block is then bolted in place. It is estimated that the entire operation could be completed in less than 30 minutes, of which a mechanic will be on the HSM roof for approximately 15 minutes. During this time he will receive a dose of less than 50 mrem. An additional dose to the mechanic and to the crane operator on the ground while putting the shield block in place will be less than 10 mrem each (assuming an average distance of 10 feet from the center of the HSM front wall). Alternatively, license applicants may, on a plant specific basis, specify a shielding block design which is integral with the HSM reinforced concrete and is specifically designed to resist tornado missiles.

8.2.2 Tornado Winds/Tornado Missile

8.2.2.1 Cause of Accident In accordance with ANSI-57.9 and 10CFR72.72, the NUHOMS-24P HSM and transfer cask are designed for tornado effects including tornado wind loads. In addition, the HSM is also designed for tornado missile effects, although not specifically required by ANSI-57.9 and 10CFR72.72. For the purpose of this conservative generic evaluation, the NUHOMS system is designed to be located anywhere within the United States. Consequently, the most severe tornado wind loadings specified by NUREG-0800 (8.19) and (8.30) and NRC Regulatory Guide 1.76 (8.31) were selected as a design basis for this postulated accident.

8.2.2.2 Accident Analysis The applicable design parameters for the design basis tornado (DBT) are specified in Section 3.2.1. The determination of the tornado wind and tornado missile loads acting on the HSM are detailed in Section 3.2.2. For the purpose of this conservative generic analysis, the tornado loads are assumed to be acting on a single free standing HSM with no shear ties between the HSM walls and foundation slab. This case conservatively envelopes the effects of wind on a 2x10 HSM array. The NUHOMS-24P transfer cask is designed for the tornado wind loads defined in Section 3.2.2.

A. Effect of DBT Wind Pressure Loads on HSM

As described in Section 3.2.2, the maximum DBT generated wind loads are 397 lb./ft.^{2} and 196 lb./ft.^{2} on the windward and leeward walls of the HSM, and a suction of 357 lb./ft.^{2} on the roof of the HSM. For conservatism, the design basis operating wind pressure loads are assumed to be equal to those calculated for the DBT in the formulation of HSM load combination results.

The DBT pressures were applied to the HSM as uniformly distributed loads. The rigidity of the HSM in the transverse direction (frame action of a single HSM) was the primary load transfer mechanism assumed in the analysis. The bending moments and shears at critical locations in the HSM walls and roof were calculated by performing a linear finite element analysis. The resulting moments and shears are tabulated in Table 8.2-3 and were included in the formulation of load HSM combination results reported in Section 8.2.10.

An analysis was also performed to evaluate the effect of overturning and sliding of a single, un-anchored HSM due to a postulated DBT. For the DBT wind overturning analysis, the overturning moment and the resulting stabilizing moments are calculated.

(i) HSM Overturning Analysis

The stabilizing moment (M_{st}) for a single HSM is:

$$M_{st} = Wd \qquad (8.2\text{-}1)$$

Where: W = 573 kips, weight of HSM plus DSC (excluding foundation slab)

d = 76 in. (6.33 ft.), horizontal distance between center of gravity of HSM to the outer edge of the wall.

Therefore: M_{st} = 43,500 K-in.

and the overturning moment (M_{to}) for a single HSM due to DBT wind pressure is:

$$M_{to} = (W_1 + W_2)\, A_w h/2 + W_3 A_r d \qquad (8.2\text{-}2)$$

Where: W_1 = 0.397 K/ft.^{2}, wind load, windward wall

W_2 = 0.196 K/ft.^{2}, wind load, leeward wall

h = 180 in. (15.0 ft.) wall height

W_3 = 0.357 K/ft.^{2}, wind uplift on roof

A_r = 279 ft.^{2}, roof area

A_w = 330 ft.^{2}, wall area

Therefore: M_{to} = 25,200 K-in.

Since the overturning moment is smaller than the stabilizing moment, an unanchored HSM will not overturn. The resulting factor of safety against overturning effects for DBT wind loads is 1.7.

The 356 lb./ft.^{2} DBT negative pressure acting on the HSM door results in a total load of 20 kips, which is reacted by the HSM door frame anchor bolts. Each of the fifteen 1-1/8" diameter door frame anchor bolts has a tensile load capacity which easily exceeds this total load. All other loads acting on the HSM door assembly envelop DBT negative pressure effects. Therefore DBT negative pressure loads have a negligible effect on the HSM door assembly design.

(ii) HSM Sliding Analysis

To evaluate the potential for sliding of a single, unanchored HSM, the sliding force generated by the postulated DBT wind pressure was compared to the sliding resistance provided by friction between the base of the HSM walls and the HSM foundation slab.

The force (F_{sl}) required to slide a single HSM is:

$$F_{sl} = [W - W_3A_r]\ \mu \qquad (8.2\text{-}3)$$

Where: μ = 0.6, coefficient of friction (ACI 318-83)(8.47)

W, W_3 and A_r are defined above.

Substituting gives: F_{sl} = 284 K

The sliding force (F_{hw}) generated by DBT wind pressure for a single HSM is:

$$F_{hw} = (W_1 + W_2)\ A_w$$

Where: W_1, W_2, and A_w are as defined previously

Substituting gives: F_{hw} = 196K

Since the horizontal force generated by the postulated DBT is smaller than the force required to slide a single free standing HSM, the HSM will not slide. The factor of safety against sliding of the HSM due to DBT wind loads is 1.5. To provide additional design margin, the base of the HSM walls may be tied to the HSM foundation slab with dowels or other means. The type of mechanical tie to be installed is necessarily site specific, and depends upon the type of construction technique employed by the individual license applicant. It is envisioned that for cast in-place construction, a roughened surface or reinforcing dowels would be employed.

B. Effects of DBT Wind Pressure Loads on Transfer Cask

The NUHOMS-24P transfer cask design was evaluated for the effects of tornado wind loads in accordance with 10CFR72.72 and ANSI 57.9 criteria. This evaluation was performed for the transfer cask mounted horizontally on the transport trailer/skid. Both overall stability and maximum cask stresses were evaluated.

The critical overturning case for the transfer cask stability occurs when the wind loads are applied perpendicular to the cask/skid/trailer.

The stabilizing moment (M_{st}) is given by:

$$M_{st} = W_t d$$

Where: W_t = 225 kips, minimum weight of cask/skid/trailer

d = 66 in., half wheel base of trailer

Therefore: M_{st} = 14,800 K-in.

Conservatively assuming that the combined cask/skid/trailer has a solid vertical projected area, and ignoring the reduction in total wind pressure due to the open areas and shape factors, the maximum overturning moment (M_{to}) for the cask/skid/trailer due to DBT wind pressure is:

$$M_{to} = (W_1 + W_2)\ (A)\ h/_2$$

Where: A = 232 ft.^{2}, combined vertical projected area of cask/skid/trailer

W_1 = 0.397 k/ft.^{2}, wind load windward side

W_2 = 0.196 k/ft.^{2} wind load leeward side

h = 146 in., height to top of cask during normal transfer operations

Therefore: M_{to} = 10,000 k-in.

Since the overturning moment is smaller than the stabilizing moment, the cask/skid/trailer will not overturn. The resulting factor of safety against the overturning effects for DBT wind loads is about 1.5.

The maximum stresses induced in the transfer cask structural shell by DBT wind pressure loads were very conservatively calculated using the correlation presented in Roark (8.16) Table 31 Case 9.c. The wind pressure loads were applied as a line load to the cylindrical shell. Substituting the cask physical dimensions and an equivalent line load of 0.24 k/inch (397 psf x 85.3/144) into the correlation, the maximum calculated shell stress is 3.8 ksi. Similarly, the maximum tornado wind load pressure stresses induced in the top and bottom cover plates were calculated using the Roark correlations given in Table 24 Cases 10a and 10b for the simply supported (bolted) top cover and fixed (welded) bottom cover plates. The maximum calculated DBT wind pressure stress calculated for these items was 0.5 ksi. Since the resulting DBT transfer cask stresses are a small fraction of the ASME Code Level A allowables, DBT wind loads are not considered further.

C. HSM Missile Impact Analysis

The outer walls and roof slab of the massive reinforced concrete HSM are 36 inches thick. The walls and roof were designed to provide adequate biological shielding and easily meet the minimum acceptable barrier thickness requirements for local damage against tornado generated missiles, specified in Table 1 of Section 3.5.3, NUREG-0800. Nevertheless, in order to demonstrate the adequacy of the HSM design for tornado missiles, a bounding analysis of the HSM was performed. The items evaluated include the resistance to penetration, spalling, scabbing and perforation for a postulated missile impact. For these analyses, a rigid penetration resistance missile consisting of a 276 pound, eight inch diameter blunt nosed hardened steel object was postulated. The method of analysis was based on the modified NDRC formula as recommended in Section 3.5.3 of NUREG-0800.

The three inch thick steel door which covers the access opening of the HSM was also evaluated for DBT missile penetration resistance. The results of this evaluation indicate that the HSM access door provides sufficient capacity to preclude penetration. The only component, of the reinforced concrete HSM which is not specifically designed to withstand the tornado generated missiles are the shielding blocks which cover the air outlet vents. The effects of a loss of the HSM shielding blocks were evaluated in Section 8.2.1 of this report. The license applicant may also elect to design the shielding blocks to withstand a DBT missile, as discussed in Section 8.2.1.

The DBT missile penetration resistance analyses for the HSM are presented in the following paragraphs.

(i) Missile Impact Penetration Resistance Analysis

The modified National Defense Research Committee (NDRC) formula from Kennedy, Holmes and Narver (8.32) was used to predict the HSM wall penetration depth for a postulated DBT missile.

$$x = \sqrt{4KNWd^{-0.8}\left[\frac{V}{1000}\right]^{1.8}} \qquad (8.2\text{-}5)$$

When: $\frac{x}{d} \leq 2.0$

Where: x = Total penetration depth (in.)

d = 8 in., Projectile diameter

K = $180/\sqrt{f'_c}$, Concrete penetrability factor = 2.88

N = 0.84 (blunt nosed), Projectile shape factor

f'_c = 5000 psi, Concrete compressive design strength at 150°F

W = Projectile weight = 276 lbm.

V = Striking velocity = 184.8 ft./s

Therefore: x = 4.6 inches

The perforating thickness, or maximum thickness that the postulated DBT missile will completely penetrate, was calculated using the correlation:

$$\frac{e}{d} = 1.32 + 1.24\left(\frac{x}{d}\right) \text{ for } 1.35 \leq \frac{e}{d} \leq 13.5$$

Substituting yields: (8.2-6)

$$e = 16.3 \text{ in., } \left(\frac{e}{d} = \frac{16.3}{8} = 2.03 > 1.35 \text{ and } < 13.5\right)$$

Therefore; e, the maximum perforation thickness, is conservative.

The minimum thickness necessary to prevent scabbing of material from the rear face of the target was calculated using:

$$\frac{s}{d} = 2.12 + 1.36 \left(\frac{x}{d}\right) \quad \text{for } \left(0.65 \le \frac{x}{d} \le 11.8\right) \qquad (8.2\text{-}7)$$

Substituting yields:

$$s = 23.2 \text{ in.}$$

Where: s = Scabbing thickness (in.)

Code requirements for nuclear safety related concrete structures (ACI 349-85) require a minimum of 20% additional wall thickness to prevent perforation and scabbing. Scabbing effects control the minimum required wall thickness. Therefore, the minimum wall thickness required to provide adequate protection for the enveloping DBT missile is:

$$1.2s = 27.9 \text{ in.}$$

The specified minimum wall thickness for exterior HSM walls is 36 inches. Consequently, there is adequate protection against local DBT missile impact damage.

(ii) Local Barrier Impingement Analysis

A three inch steel door is used to cover the HSM access opening after the DSC is in place. The HSM door has been analyzed to verify its adequacy for local barrier impingement of a DBT missile. The 276 pound, eight inch diameter artillery shell was used for this calculation as it envelopes effects caused by the postulated one inch diameter solid steel sphere. The minimum thickness of a steel plate capable of being perforated by the postulated DBT missile is given in the McDonalds, Mehta, and Minor paper (8.33) as:

$$T = \left(\frac{0.5 M_m V_s^2}{672\, d_m}\right)^{2/3} = 0.52 \text{ in.} \qquad (8.2\text{-}8)$$

Where: T = Perforation thickness (in.)

M_m = Mass of missile = $\frac{W}{g}$ = 8.58 slugs

W = Weight = 276 lb.

g = 32.2 ft./s^2

V_s = Missile strike velocity = 184.8 ft./s

d_m = Diameter of missile = 8 in.

The three inch thick steel HSM door specified exceeds the minimum required perforation thickness of 0.52 inch by a wide margin.

(iii) Massive Missile Impact Analysis

To evaluate the potential for damage due to a postulated massive DBT missile, the force determined from the massive missile impact (i.e., a 3,976 pound automobile with 20 square foot frontal area traveling at 184.8 feet per second) was applied to the side wall of the HSM. The magnitude of this force was determined in accordance with the procedure established by Williamson and Alvy (8.34), as recommended by NUREG-0800, Section 3.5.3. The basic assumption adopted by this procedure is that the massive missile strikes a flat surface and results in little or no penetration of the target. However, the missile is sustained within the impacted structure by allowing permanent deformation to take place (plastic impact). The equation to calculate an equivalent static impact force developed by Alvy and Williamson is:

$$q_y = \frac{\mu}{(2\mu-1)}\ mg \left[1 + \sqrt{1 + \left(\frac{2\mu-1}{\mu^2}\right) \left(\frac{2\pi V}{gT}\right)^2} \right] \qquad (8.2\text{-}9)$$

Where: q_y = Equivalent static impact force, lb.

μ = 10.0, Concrete ductility factor, (ACI 349-85, Appendix C)

m = $3967/g \ \frac{\text{lb-sec.}^2}{\text{ft.}}$, Mass of missile

g = 32.2 ft./s^2

V = 184.8 ft./s, Initial striking velocity of missile

T = 0.04 sec., Natural period of vibration

The natural period of vibration was established from the frequency analysis of a single HSM which is reported in Section 8.2.3. Substituting into Equation 8.2-9 yields a q_y of 823 kips.

The most damaging impact location for the automobile to strike would be the center of the side wall of a single, unanchored

HSM. To calculate the maximum bending moment in the wall, a rectangular plate with all edges simply supported was conservatively assumed. The plate dimensions were taken to the center lines of the HSM roof and the end walls. A 20 sq. ft. rectangular area of uniform loading was applied at the center of the plate. By using a Roark correlation, the maximum stress is:

$$S_b = \frac{\beta W}{t^2} = 0.63 \text{ ksi (Roark, Table 26, Case 1c)} \quad (8.2\text{-}10)$$

Where: S_b = Bending stress (psi)

β = 1.0, Plate geometry coefficient

W = q_y = 823K, Total static force

t = 36 in. Thickness

The maximum bending moment causing a stress of 0.63 ksi is:

$$M = \frac{S_b t^2}{6} = 137. \text{ K-in./in.} \quad (8.2.11)$$

and for a typical 12 inch wide strip of the HSM side wall, the maximum bending moment caused by a postulated massive missile impact is:

$$M_T = 12M = 1{,}640 \text{ k-in.}$$

The ultimate moment (M_u) for a 12 inch section of the HSM wall at 150°F is 4180 K-in. Therefore, the requirements of ACI 349-85, Section 9.2.6 are met and massive missile moments and shears need not be included in the HSM load combinations presented in Section 8.2.10.

The maximum sliding force due to a DBT generated massive missile (823 K) is greater than resistance to sliding provided by a single, unanchored HSM (284 K). Therefore, in the event that an applicant chooses to construct a single free standing HSM, as opposed to an array of interconnected HSMs, suitable dowels and/or shear keys between the walls and the foundation slab must be provided to resist the worst case postulated DBT massive missile impact. These considerations will be addressed in the site license application as necessary. For the more typical case where an array of four or more HSM's are constructed to form an HSM unit, the resistance to sliding increases to over 1100 kips. As a result no additional measures are required to resist sliding of the HSM due to a postulated DBT massive missile impact.

8.2.2.3 Accident Dose Calculations The only components of the NUHOMS system which are not specifically designed to withstand tornado generated missiles are the air outlet shielding blocks located on the roof of the HSM and the NUHOMS-24P transfer cask. The consequence of losing the shielding blocks during this accident is presented in Section 8.2.1 of this report.

8.2.3 Earthquake

8.2.3.1 Cause of Accident As discussed in Section 3.2.3.1, determination of the seismic forces acting on the NUHOMS system components is site specific. However, for the purpose of this conservative generic evaluation, the design response spectra of NRC Regulatory Guide 1.60 (8.35) were selected for the seismic analysis of the NUHOMS system components.

8.2.3.2 Accident Analysis As discussed in Section 3.2.3.1, the maximum horizontal ground acceleration of 0.25g and the maximum vertical ground acceleration of 0.17g were utilized for the design basis seismic analysis of the NUHOMS components. Based on NRC Reg. Guide 1.61 (8.36), a damping value of three percent was used for the DSC seismic analysis. Similarly, a damping value of seven percent for miscellaneous steel and concrete were utilized for the HSM. An evaluation of the fundamental frequency of the HSM was performed to determine the dynamic amplification factors associated with the design basis seismic response spectra for the NUHOMS HSM and DSC. The lowest structural frequency calculated for a single unanchored HSM in the lateral direction was 25 Hz. Tables 1 and 2 of NRC Regulatory Guide 1.60 require an amplification factor of 1.27 for this structural frequency, which results in a horizontal acceleration of 0.32g and a vertical acceleration of 0.22g for design of the HSM. The maximum vertical acceleration was conservatively determined using the lateral fundamental frequency of the HSM.

A. DSC Seismic Evaluation.

The maximum calculated seismic input accelerations for the DSC at the DSC support structure elevation inside the HSM are 0.40g horizontally and 0.27g vertically. An analysis using these seismic loads showed that the DSC will not lift off of the support rails inside the HSM. The resulting stresses in the DSC shell due to vertical and horizontal seismic loads were also determined and included in the appropriate load combinations. The seismic evaluation of the DSC is described in the paragraphs which follow. The DSC support structure was also subjected to the calculated DSC seismic reaction loads as discussed in Paragraph C below.

(i) DSC Natural Frequency Calculation

Two natural frequencies, each associated with a distinct mode of vibration of the DSC were evaluated. These two modes are the DSC shell cross-sectional ovalling mode, and the mode with the DSC shell bending as a beam.

a. DSC Shell Ovalling Mode

The natural frequency for the DSC shell ovalling mode was determined from the Blevins (8.37) correlation as follows.

$$f = \frac{\lambda_i}{2\pi R}\sqrt{\frac{E}{\mu\,(1 - \nu^2)}} \quad \text{(Blevins, Table 12-1, Case 3)} \qquad (8.2\text{-}12)$$

Where: R = 33.31 in., DSC mean radius

E = 26.5E6 psi, Youngs Modulus

ν= 0.3, Poisson's ratio

$$\lambda_i = 0.289\,\frac{t}{R}\sqrt{\frac{i(i^2 - 1)}{1 + i^2}} \qquad (8.2\text{-}13)$$

t = 0.625 in., Thickness of DSC shell

μ = 0.288/g, Steel mass density

The lowest natural frequency corresponds to the case when i=2.

Hence: λ_2 = 0.0146 sec.

Substituting gives: f = 13.8 Hertz

The resulting spectral accelerations in the horizontal and vertical directions for this DSC ovalling frequency are 1.0g and 0.68g.

b. DSC Beam Bending Mode

The DSC shell was conservatively assumed to be simply supported at the two ends of the DSC. The beam bending mode natural frequency of the DSC was calculated from the Blevins correlation:

$$f_i = \frac{\lambda_i^2}{2\pi L^2}\sqrt{\frac{EI}{m}} \quad \text{(Blevins, Table 8.1, Case 5)} \qquad (8.2\text{-}14)$$

E = 26.5E6 psi, Young's Modulus

I = 72,900 in.^{4}, DSC moment of inertia

L = 186.5 in., Total length of DSC

m = 72,000/186.5g = 386/g lb./in.

λ = $i\pi$; for lowest natural frequency, $i = 1$

Substituting yields: f_1 = 62.8 Hertz.

The DSC spectral accelerations at this frequency correspond to the zero period acceleration. These seismic accelerations are bounded by those of the ovalling mode frequency which were used in the subsequent stress analysis of the DSC shell.

(ii) DSC Seismic Stress Analysis

With the DSC resting on the T-section support rails inside the HSM, the stresses induced in the DSC shell were calculated due to the 1.0g horizontal and 0.68g vertical seismic accelerations applied as equivalent static loads. These values were conservatively increased by a factor of 1.5 to account for the effects of possible multimode excitation. The DSC stresses due to the resulting 1.0g vertical acceleration were calculated by factorin the dead load analysis results reported in Section 8.1. The maximum DSC shell stress intensity obtained from this analysis was 3.8 ksi. For the stress evaluation of the DSC shell due to seismic accelerations in the lateral direction, the resulting equivalent static acceleration of 1.5g was conservatively assumed to be resisted by one of the two T-section support rails inside the HSM. Local bending stresses in the DSC shell at the T-section support rail location were calculated to be 17.2 ksi. The DSC shell stresses obtained from the analyses of vertical and horizontal seismic loads were summed absolutely. The magnitude of the combined shell stress is 21.0 ksi.

As stated, in Section 4.2.3.2, a seismic restraint is included in the design of the DSC support system inside the HSM to prevent sliding of the DSC in the axial direction during a postulated seismic event. The stresses induced in the DSC shell due to the restraining action of this assembly for a horizontal seismic load, applied along the axis of the DSC, were evaluated and found to be negligible.

The stability of the DSC against lifting off one of the T-section support rails during a seismic event was evaluated by performing a rigid body analysis, using the 0.40g horizontal and 0.27g vertical input accelerations. The factor of 1.5 used in the DSC analysis to account for multimode behavior need not be included in the seismic accelerations for this analysis, as the potential for lift off is due to rigid body motion, and no frequency content effects are associated with this action. The horizontal equivalent static acceleration of 0.40g was applied laterally to the center of gravity of the DSC. The point of rigid body rotation of the DSC was assumed to be the center of the T-section support rail, as shown in Figure 8.2-1. The applied moment of 840 kip-in. was calculated by summing the overturning moments. The stabilizing moment, acting to oppose the applied moment, was calculated by deducting the effects of the upward vertical seismic acceleration of 0.27g from the total weight of the DSC and summing moments at the rail support. The resulting stabilizing moment is 880 kip-in. Since the calculated stabilizing moment is greater than that of the applied moment, the DSC will not lift off the DSC support system inside the HSM.

Referring to Figure 8.2-1, the margin of safety associated with DSC lift off was calculated as follows:

$$M_{am} = yF_H \tag{8.2-17}$$

and

$$M_{sm} = (F_{v1} - F_{v2})x \tag{8.2-18}$$

Where: M_{am} = The applied seismic moment

M_{sm} = The stabilizing moment.

All other variables are defined in Figure 8.2-1.

Substituting yields: $M_{am} = 840$ K-in.

and $M_{sm} = 880$ K-in.

Thus, the margin of safety (SF) against DSC lift off from the DSC support rails inside the HSM obtained from this bounding analysis is:

$$SF = \frac{M_{sm}}{M_{am}} = 1.05 \tag{8.2-19}$$

B. HSM Seismic Evaluation

To evaluate the seismic response of the HSM, an equivalent static analysis was performed. Seismic loads in the transverse direction were assumed to be resisted by frame action of the HSM. The stiffening effects of the longitudinal shear walls was conservatively omitted from this analysis. Accordingly, the HSM was modeled as a frame (including the foundation slab), and the calculated horizontal acceleration of 0.32g applied to the frame members as equivalent inertia loads. Similarly, the calculated vertical seismic load of 0.22g was applied to account for vertical seismic effects. The horizontal seismic loads in the axial direction of the HSM are resisted by thick shear walls. Bending stresses resulting from the horizontal and vertical seismic analyses were increased by a factor of 1.5 to conservatively account for possible multimode excitation effects of the HSM. The results were included in the load combinations with the appropriate strength reduction factors. The factors used for the HSM are presented in Section 3.2.5 of this report. The load combination results for normal, off-normal, and accident conditions are presented in Section 8.2.10.

An analysis was also performed to establish the worst case factor of safety against overturning and sliding for a single, unanchored module. This analysis consisted of comparing the stabilizing moment produced by the weight of the HSM, reduced by 22 percent to account for the upward vertical seismic acceleration, against the overturning moment produced by applying the 0.32g load at the centroid of the HSM. For sliding of an unanchored HSM, the horizontal force of 0.32g acceleration was compared against the frictional resisting force of the foundation slab. In this manner, the factor of safety against sliding was established. The concrete coefficient of friction was taken as 0.6 as defined in Section 11.7.4.3 of ACI 318-83 (8.47).

The details of the seismic evaluation of the HSM are described in the paragraphs which follow.

(i) HSM Frequency Analysis

To determine the HSM fundamental frequency, the STRUDL-DYNAL (8.49) finite element model shown in Figure 8.1-10 was developed. The lowest structural frequency calculated for a single free standing HSM was 25 Hz. The corresponding horizontal and vertical spectral accelerations at this frequency are 0.32g and 0.22g.

(ii) HSM Seismic Response Analysis

An equivalent static horizontal seismic acceleration of 0.32g was applied to the HSM with a factor of 1.5 to account for the possible effects of multimode excitations. The HSM concrete mass was applied as a uniformly distributed load on the walls, roof and foundation slabs. The value of the distributed load was calculated for the concrete thickness times 0.48 to provide an equivalent static load. The mass of the lower wall was applied at the base of the HSM. The mass of the DSC, and the DSC support assembly, were applied at the support assembly elevation in the HSM. The resulting forces and moments in the HSM walls and roof of a single HSM, due to the horizontal seismic accelerations, were calculated using the linear finite element model shown in Figure 8.1-10, and the computer program STRUDL (8.49).

The forces and moments due to horizontal seismic accelerations acting in the axial direction of the HSM are very small as the thick side walls directly resist the resulting lateral shearing force. The calculated forces and moments due to the horizontal seismic accelerations in the two orthogonal directions obtained from the frame model analysis were conservatively summed absolutely to establish the seismic forces and moments used for the design of the HSM reinforced concrete.

Similarly, a maximum vertical seismic acceleration of 0.22g was applied to the HSM model with a factor of 1.5 included to account for multimode effects. This loading equates to 34% of the previously described dead weight analysis for the HSM. The resulting maximum shear and bending moments are also 34% of those for the dead weight analysis. The resulting shears and moments due to vertical seismic loads were summed absolutely with those of the horizontal seismic analysis and are reported in Table 8.2-3.

<u>(iii) HSM Overturning Due to Seismic</u>

Should a single unanchored HSM be constructed, as discussed in Section 8.2.2.2, consideration of anchorage to the foundation slab is required to resist the tornado massive missile loads. The following conservative analysis was performed to show that a single, unanchored HSM will not overturn due to seismic loads. As defined by equation 8.2-1, the HSM stabilizing moment (M_{st}) is 43,500 K-in.

The seismic overturning moment is:

$$M_{ot} = Wa_v d + Wa_h h = 28,300 \text{ K-in.}$$

Where: M_{ot} = Overturning moment

a_v = 0.22g, Maximum vertical seismic acceleration

a_h = 0.32g, Maximum horizontal seismic acceleration

h = 102 in., Vertical height from HSM center of gravity to base

The result of this analysis indicates that a single unanchored HSM will not overturn during a seismic event which subjects the HSM to 0.32g horizontal and 0.22g vertical accelerations. The margin of safety against overturning is 1.5.

(iv) HSM Sliding Due to Seismic

To show that an unanchored HSM will not slide due to the postulated horizontal and vertical seismic accelerations, the following conservative analysis was performed. The friction force resisting sliding (F_{s1}) is:

$$F_{s1} = W\mu g = 268 \text{ kips} \qquad (8.2\text{-}24)$$

W = HSM loaded weight = 573 kips

μ = Coefficient of friction between the HSM concrete walls and the floor slab foundation = 0.6

g = Net downward gravitational force = (1 - 0.22)g or 0.78g

The applied horizontal seismic force is:

$$F_{hs} = Wa_H = 183 \text{ kips}$$

Where: F_{hs} = Induced horizontal seismic force

a_H = 0.32g, Horizontal seismic acceleration

The force required to slide the HSM is larger than the resulting lateral seismic force and therefore, the HSM will not slide. The factor of safety against sliding is 1.5.

C. DSC Support Assembly Seismic Evaluation

(i) DSC Support Assembly Natural Frequency

The lowest structural frequency of the DSC support assembly inside the HSM corresponds to strong axis bending of the WF-section cross member support beam. In the long axis of the HSM the natural frequency of the T-section guide rail is enveloped by the DSC beam bending mode frequency of 62.6Hz calculated previously. The WF-section cross member frequency is derived by:

$$f_1 = \frac{1}{2\pi}\sqrt{\frac{K}{m}} \qquad (8.2\text{-}25)$$

Where: f_1 = Wide flange cross member lowest natural frequency, Hz

$K = \dfrac{48EI}{a(3L^2 - 4a^2)}$, Stiffness of pinned end beam due to two equal concentrated masses symmetrically placed at the T-section support rail locations

E = 26.5E6 psi, Modulus of elasticity @ 600°F

I = 394 in.^{4}, Cross member moment of inertia

a = 22.5 in., Distance from end of cross member to center of DSC concentrated mass (both ends)

L = 80 in., Unsupported length of cross member

$M = \frac{38,000}{g} \quad \frac{\text{lb.-s}^2}{\text{in.}}$, Proportional mass of DSC plus support structure applied to the center cross member at the T-section support rail locations

g = 386.4 in./s^2

Substituting yields: f_1 = 18.3 Hz

(ii) DSC Support Assembly, Vertical Seismic Analysis

The DSC support assembly vertical acceleration was generated for 7% damping and a 0.17g vertical ground acceleration using the Reg. Guide 1.60 response spectra. This yields a calculated vertical acceleration of 0.27g at 18.3 Hz. To determine the stresses in the DSC support assembly due to the resulting vertical seismic acceleration, the stresses previously calculated for the dead weight analysis of the DSC support assembly were factored by 0.40. This includes a conservative dynamic amplification factor of 1.5. For the WF-section cross member support beams, the maximum bending stress is 10.6 ksi and the corresponding maximum shear stress is 7.3 ksi. Similarly, the maximum stresses in the T-section support rails are 16.8 ksi and 2.3 ksi, respectively. These compare with Code allowables of 26.3 ksi for bending and 16.0 ksi for shear and, as a result, have a considerable design margin.

(iii) DSC Support Assembly Horizontal Seismic Analysis

The DSC support assembly horizontal acceleration was calculated using the Reg. Guide 1.60 spectra with a 0.25g horizontal ground acceleration and 7% damping. The resulting maximum spectral acceleration was 0.60g which includes a conservative dynamic amplification factor of 1.5. The calculated acceleration was applied as an equivalent static force, in the transverse direction, of the T-section support rails. The effective mass of the DSC was distributed along the length of the T-section support rails. The STRUDL (8.49) finite element model of the DSC support assembly, described in Section

8.1.1.4, was used in this analysis. The resulting maximum bending stress in the WF-section cross member is 3.6 ksi and the corresponding maximum shear stress is 2.5 ksi. Similarly the maximum bending stress in the T-section guide rails is 9.8 ksi and the shear stress is 1.3 ksi. The corresponding Code allowable stresses are 26.3 ksi and 16.0 ksi respectively.

The effect of concentrated bolt forces was included in the design of the DSC support structure connection details. Each WT6x115 and connections of the WT6x115 to the W10x68 cross members, were designed for a horizontal load equal to one half of the total DSC dead weight. This condition envelopes all other loading conditions for the individual bolts or structural elements of the DSC support assembly.

For the DSC support assembly seismic analysis, the stresses due to a horizontal acceleration in the axial direction were obtained by factoring the results from the normal DSC handling axial load case. The resulting maximum bending and shear stress in the WF-section cross member are 7.1 ksi and 1.9 ksi, respectively. Similarly, the maximum bending and shear stress in the T-section guide rails are 3.6 ksi and 0.1 ksi respectively.

The stresses in the WF-section cross members and the T-section guide rails due to the vertical and two horizontal seismic accelerations were combined absolutely and included in the subsequent load combination results reported in Section 8.2.10.

(iv) DSC Seismic Restraint Analysis

The DSC seismic restraint detail, located inside the HSM access opening, is shown in Figure 4.2-4. The restraint has a plate which bears on the end of the DSC and transfers axial seismic loads to a shear key arrangement built into the HSM access opening sleeve. The seismic restraint is set in place between the DSC support rails following transfer of the DSC to the HSM.

The clearance between the DSC seismic restraint and the DSC is designed for the maximum DSC thermal growth which occurs during the postulated HSM blocked vent case, as discussed in Section 8.2.7. During normal storage there will be a small (1/8 to 1/4 inch) gap which will allow movement of the DSC relative to the HSM. This motion produces a small increase in the DSC axial force due to seismic loads, and has been included in the design of the DSC seismic restraint shear key arrangement.

The DSC seismic restraint and it's shear key attachment to the HSM access opening sleeve were designed for a lateral force equal to the mass of the DSC times the horizontal acceleration. An impact factor of 1.5 was also applied to this force to account for the impact due to the small gap between the DSC and the seismic restraint. The total design lateral force, therefore, is equal to:

$$F_{seismic} = \text{DSC mass x horizontal acceleration x 1.5}$$

$$= \text{80 kips x .48g x 1.5}$$

$$= \text{58 kips}$$

The controlling design element in the seismic restraint is the restraint to HSM sleeve shear key. The maximum shear stress on this member is 12.8 ksi compared to the allowable shear stress of 14.2 ksi.

D. NUHOMS-24P Transfer Cask Seismic Evaluation

The effects of a seismic event occurring when a loaded DSC is resting inside the NUHOMS-24P transfer cask were conservatively postulated for two conditions which affect the transfer cask. All other conditions which exist during DSC loading or transport operations are enveloped by the two cases postulated. The first case postulates a fully loaded NUHOMS-24P transfer cask standing vertically in the plant's cask decontamination area during closure of the DSC. For this condition it is required that the transfer cask remain upright. The rigid body horizontal acceleration required to overturn a loaded NUHOMS-24P transfer cask at a minimum gross weight of 190 kips is at least 0.40g. Each licensing applicant shall ensure that the transfer cask is not subjected to accelerations greater than this magnitude while in the plant's decontamination area, or provide sufficient lateral restraint to prevent cask overturning.

The second case postulates a seismic event occurring during transport of a loaded DSC, resting inside the transfer cask, in a horizontal position, secured to the transport skid/trailer. This load case is conservatively enveloped by the postulated normal transport load accelerations of $\pm 0.5g$ acting in the vertical, axial, and transverse directions, applied simultaneously at the center of gravity of the transfer cask, as specified in Section 8.1.1.8. These accelerations envelope those which would result from a seismic event in the highly unlikely event that a design basis earthquake would occur during transport of the loaded DSC to or from the HSM. Therefore, the calculated stress intensities for the normal transport loads case of 17.9 ksi for the cask structural shell, and 2.0 ksi for the trunnions, were conservatively used as the maximum seismic stresses in the load combination results reported in Section 8.2.10.

The stabilizing moment to prevent overturning of the cask/ trailer assembly due to the 0.25g horizontal and 0.17g vertical seismic ground accelerations was calculated and compared to the dead weight stabilizing moment. The results of this analysis showed that there is a factor of safety of at least 2.0 against overturning which ensures that the cask/trailer assembly will has sufficient margin for the design basis seismic loading.

8.2.3.3 Accident Dose Calculations The NUHOMS system components were conservatively designed and analyzed to withstand the forces generated by a postulated design basis earthquake accident. Hence, there are no dose consequences resulting from an earthquake.

8.2.4 Flood

8.2.4.1 Cause of Accident Flooding conditions simulating a range of flood types, such as tsunami and seiches as specified in 10CFR72.72(b) were considered. In addition, floods resulting from other sources, such as high water from a river or a broken dam, are postulated as the cause of the accident.

8.2.4.2 Accident Analysis Since the source of flooding is site specific, the exact source, or quantity of flood water, cannot be established. However, for the purpose of this generic evaluation of the NUHOMS-24P DSC and HSM, bounding flooding conditions were specified which envelop those which are postulated for most plant sites. As described in Section 3.2, the design basis flooding load was specified as a 50 foot static head of water and a maximum flow velocity of 15 feet per second. Each license applicant should confirm that this represents a bounding design basis for their specific plant site.

A. HSM Flooding Analysis

Since the HSM is open to atmosphere, external pressure due to flooding is not a design load for the HSM.

The maximum drag force acting on the HSM due to a 15 fps flood water velocity was calculated by the Streeter and Wylie (8.38) correlation:

$$F = \frac{v^2}{2g} C_D A \rho_w = 6,600 \text{ lbs./ft. width of HSM} \quad (8.2\text{-}26)$$

Where: v = 15 fps, Flood water velocity

C_D = 2.0, Drag coefficient for flat plate

A = 15.0 ft.^{2}, HSM area per foot length

ρ_w = 62.4 lb./ft.^{3}, Flood water density

F = Drag force (lb.)

g = 32.2 ft./s^2

The resulting flood induced force was calculated to be 6,600 lbs./ft. which is assumed to be applied normally to the side wall of an HSM. This loading produces a maximum flood induced moment of 144 K-in./ft. This compares with the calculated ultimate strength capacity of the HSM walls at 150°F of 4,180 K-in./ft., as shown in Section 8.2.2.2.

(i) HSM Overturning Analysis

The factor of safety against overturning, for the postulated flooding conditions specified in Section 3.2, was calculated by summing moments about the bottom outside corner of a single, unanchored HSM. A net weight of 348 kips for a loaded HSM, including buoyancy effects, was used to calculate the stabilizing moment resisting the overturning moment applied to the HSM by the flood water drag force. The stabilizing moment is:

$$M_{st} = 348 \times 6.33 \text{ ft.} = 2200 \text{ K-ft.} \qquad (8.2\text{-}27)$$

The maximum drag force due to the postulated water current velocity of 15 fps was derived from Equation 8.2-26 as 6.6 k/ft. acting over the entire height and width of a side wall of a single HSM. Therefore, the overturning moment due to the postulated flood current is:

$$M_{ot} = 6.6 \text{ k/ft.} \times 22 \text{ ft.} \times 7.5 \text{ ft.} = 1090 \text{ K-ft.} \qquad (8.2\text{-}28)$$

The factor of safety against overturning for a single, unanchored HSM due to the postulated design basis flood water velocity is given by:

$$F \text{ of } S = \frac{M_{st}}{M_{ot}} = 2.0$$

(ii) HSM Sliding Analysis

The factor of safety against sliding of an unanchored single HSM due to the maximum postulated flood water velocity of 15 fps was calculated using methods similar to those described above. The effective weight of the HSM including the DSC acting vertically downward, less the effects of buoyancy acting vertically upward is 348 k. The friction force resisting sliding of the HSM is equal to the product of the net weight of the HSM and DSC and the coefficient of friction for concrete placed against a roughened concrete surface such as that between the HSM walls and floor slab, which is 0.6 as specified in ACI 318-83 (8.47). Therefore, the force resisting sliding of the HSM is 0.6 x 348 or 209 kips. As shown in the previous flooding calculations the drag force acting on a single HSM is 6.6 kips/ft., or 145 kips total acting on the side wall of a single HSM, due to a flood velocity of 15 fps. The resulting factor of safety against sliding of a single free standing HSM due to the design basis flood water velocity is 1.4.

B. DSC Flooding Analyses

The DSC was evaluated for the design basis fifty foot hydrostatic head of water producing external pressure on the DSC shell and outer cover plates. To conservatively determine design margin which exists for this condition, the maximum allowable external pressure on the DSC shell was calculated for Service Level A stresses using the methodology presented in NB-3133.3 of the ASME Code (8.3). The resulting allowable pressure of 63.6 psi is 2.9 times the maximum external pressure of 21.7 psi due to the postulated fifty foot flood height. Therefore, buckling of the DSC shell will not occur under the worst case external pressure due to flooding. The maximum DSC shell primary membrane stress intensity for the 21.7 psi external pressure is 1.2 ksi which is considerably less than the Service Level C allowable primary membrane stress of 22.4 ksi. The maximum stress in the flat heads of the DSC occurs in the top cover plate. The maximum bending stress was calculated using Roark (8.16) for a simply supported circular plate as:

$$M_c = \frac{qa^2(3+\nu)}{16} \quad \text{(Table 24, Case 10 with } r_o = 0\text{)}$$

Where: M_c = Maximum moment at center of plate

q = Uniform pressure load = 21.7 psi

ν = Poisson's ratio = 0.3

Substituting yields: M_c = 5060 in.-lbs.

$$S_{bx} = \frac{6M_c}{t^2}$$

Where: t = Plate thickness = 1.5 inches

Therefore: $S_{bx} = 13.5$ ksi

This value is considerably less than the ASME Service Level C allowable of 31.1 ksi for primary bending. These stresses were combined with the appropriate loads to formulate load combinations. The resulting total stresses for the DSC are reported in Section 8.2.10.

8.2.4.3 Accident Dose Calculations The radiation dose due to flooding of the HSM is negligible. The radioactive material inside the DSC will remain sealed in the DSC and, therefore, will not contaminate the encroaching flood water. The minimal amount of contamination which may be on the outside surface of the DSC (see Section 3.3.7.1) is not sufficient to be a radiological hazard if it were to be washed off the DSC outer surface.

8.2.4.4 Recovery

Because of the location and geometry of the HSM vents, it is unlikely that any significant amount of silt would enter an HSM should flooding occur. Any silt deposits would be removed using a pump suction hose or fire hose inserted through the inlet vent to suck the silt out, or produce a high velocity water flow to flush the silt through the drain in the lower front wall of the HSM.

8.2.5 Accidental Cask Drop

This section addresses the structural integrity of the NUHOMS-24P transfer cask, the NUHOMS-24P DSC and its internal basket assembly when subjected to postulated cask drop accident conditions.

8.2.5.1 Cause of Accident

A. Cask Handling and Transfer Operation

As described in Section 5.0 of this topical report, all handling operations involving hoisting and movement of the NUHOMS-24P transfer cask and DSC are performed inside the plant's fuel handling building. These include utilizing the crane for placement of the DSC into the cavity of the transfer cask, lifting the transfer cask/DSC into and out of the plant's spent fuel pool, and placement of the transfer cask/DSC onto or off of the transport skid/trailer. An analysis of the plant's lifting devices used for these operations, such as the crane and lifting yoke, is needed to address a postulated drop accident for the transfer cask and its contents. The postulated drop accident scenarios should be consistent with those currently addressed in the plant's licensing basis for handling of a shipping cask. Such postulated accidents are plant specific; they will be addressed in the site license application, and do not form a design basis for this generic topical report.

Once the transfer cask is loaded onto the transport skid/trailer and secured, it is pulled to the HSM site by a tractor unit. A predetermined route will be chosen to minimize the potential hazards which could occur during transport. This movement will normally be performed at very low speeds. System operating procedures and technical specification limits defining the safeguards to be provided will ensure that the system design margins are not compromised. As a result, it is highly unlikely that any plausible incidents leading to a transfer cask drop accident could occur. Similarly, at the plant's HSM site, the transport skid/trailer is backed-up to, and aligned with, the HSM using hydraulic positioning equipment. The transfer cask is then docked with, and secured to, the HSM access opening. The loaded DSC is transferred to or from the HSM using a hydraulic ram system. The hold down mechanisms which secure the transfer cask to the transport skid/trailer remain in place at all times during the DSC transfer. As a result, there is no reasonable way during these operations for a cask drop accident to occur.

B. Cask Drop Accident Scenarios

In spite of the incredible nature of any scenario which could lead to a drop accident for a NUHOMS-24P transfer cask, a conservative range of drop scenarios have been developed and evaluated. These bounding scenarios assure that the integrity of the DSC and spent fuel cladding is not compromised. Analyses of these scenarios demonstrate that the transfer cask will maintain the structural integrity of the DSC pressure containment boundary. Therefore, there is no potential for a release of radioactive materials to the environment due to a cask drop. The range of drop scenarios conservatively selected for design are illustrated in Figure 8.2-2 and include the following cases:

1. A horizontal side drop or slap down from a height of 80 inches.

2. A vertical end drop from a height of 80 inches onto the top or bottom of the transfer cask (two cases).

3. A corner drop from a height of 80 inches at an angle of 30° to the horizontal, onto the top or bottom corner of the transfer cask (two cases).

The height of 80 inches was chosen as this envelopes the maximum vertical height of the NUHOMS-24P transfer cask when secured to the transport skid/trailer assembly. The angle of inclination for the corner drop of 30° represents the maximum impact angle possible that the transfer cask can rotate downward with the transfer cask initially supported horizontally on the transport skid trailer at a height of 80 inches.

C. Cask Drop Accident Load Definitions

The various parameters which are needed to completely define and evaluate the transfer cask impact time history loading associated with each postulated drop accident scenario are necessarily site specific. In particular, the energy absorbing capacity of the transfer cask can be determined, given the mechanical properties of the surface onto which the cask is dropped and the maximum height of the drop. In addition, the maximum decelerations that will be experienced by the DSC and its internals can be determined.

The period of a typical impact time history due to a drop is of very short duration and can be approximated by a very short triangular impulse. The frequency analysis performed on various DSC components showed that the highest structural period of vibration (lowest fundamental frequency) is much greater than the time duration of a typical impact load function for a postulated cask drop event. Thus, the dynamic effects of the impact load do not produce a structural response which exceeds the equivalent static response utilized, since the dynamic amplification factor for the impact load function is less than unity. Therefore, the linear elastic static equivalent analysis performed for the NUHOMS-24P components, as discussed in subsequent paragraphs, is much more conservative than a non-linear dynamic analysis. This indicates that the spacer disks and other NUHOMS system components have a large reserve capacity for higher drop load decelerations if inelastic behavior of the materials is considered.

In order to form a basis for this generic evaluation, conservative static equivalent deceleration values have been established for each cask drop scenario as design criteria for the NUHOMS-24P transfer cask, and DSC. Each site license applicant will need to ensure, that the physical properties of a postulated drop surface are known, and that the postulated drop heights have been predetermined. This will ensure that the NUHOMS-24P transfer cask and DSC will not be subjected to deceleration loads which exceed those used for this generic evaluation.

EPRI Report NP-4830, "The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent Fuel Casks," (8.55) provides expected decelerations for postulated cask side and end drops for typical ISFSI's licensed to 10CFR72 requirements. The report establishes the maximum expected decelerations for a range of surface conditions and drop heights up to 80 inches. For the NUHOMS-24P transfer cask weight and dimensions, the expected maximum decelerations for drops onto a 36 inch thick under-reinforced concrete slab are 59g for an end drop and 49g for a side drop. Corner drops are not explicitly covered in the EPRI report. However, based on the information presented in the document and other supporting calculations, the maximum decelerations for a corner drop were determined to be significantly lower than those for side and end drops. Based on this information, a static equivalent deceleration of 75g was conservatively chosen as the generic design basis for the postulated horizontal and vertical orientation drop accidents. Similarly, a static equivalent deceleration of 25g was conservatively used for the postulated corner drop accidents. These deceleration magnitudes were established to provide bounding design loads for the DSC, and the NUHOMS-24P transfer cask for primary and secondary slap-down drop conditions.

Cask decelerations of 75g from a vertical end drop or a horizontal side drop accident will not compromise the fuel cladding integrity for spent fuel assemblies typically stored in a NUHOMS-24P system. As shown by LLNL Report UCID-21246, "Dynamic Impact Effects on Spent Fuel Assemblies," (8.58), B&W 15x15 fuel assemblies will maintain their structural integrity for an end drop deceleration of 147g, or a side drop deceleration of 101g. Of the fuel assemblies investigated, the Westinghouse 17x17 had the lowest axial and side drop capacities of 82g and 63g, respectively. Therefore, the side drop and end drop load magnitude of 75gs defined for the NUHOMS-24P components is comparable to the capacity of the spent fuel assemblies to withstand a postulated drop accident.

D. Cask Drop Surface Conditions

Because of the passive nature of the NUHOMS system operations and the protective measures taken during transport of the transfer cask to and from the HSM, it was concluded that a postulated cask drop accident is much less plausible during transport operations than those which takes place at the HSM site. Site conditions away from the HSM storage pad will typically be relatively thin (12 inch or less) concrete slabs, asphalt road surfaces or compacted gravel. The target hardness numbers as defined by Reference 8.55 for these surfaces will be small compared with that of a 36 inch thick slab. Therefore, the expected cask decelerations for a cask drop accident will be substantially less than the assumed 75g end drop or side drop, and the 25g corner drop design basis loadings.

Furthermore, the impact of an object as massive and stiff as the transfer cask, will tend to punch through lightly reinforced concrete slabs because of the very high shear stresses induced over small areas. Punching shear failures would be expected to occur for deceleration values ranging from as low as 0.5gs for a corner drop, to 2.6gs for a side drop. Also, it is more likely that the surface conditions at the HSM site will be more rigid than those which exist along the designated transfer route. For these reasons, the cask drop scenarios postulated and evaluated by site license applicants should focus on conditions which exist at the HSM site location.

Based on the data provided in EPRI Report NP-4830 (8.55) it is highly unlikely that any surface conditions encountered will produce decelerations which are not enveloped by the assumed design values. However, should site specific surface conditions at a location where it is possible to postulate a cask drop accident, exceed the maximum target hardness number (8.55) of 400,000, then additional measures, may need to be evaluated. These include the option of utilizing a prepared surface of redwood timbers, or integral cask impact limiters which limit the static equivalent drop load magnitudes to the design limits of the DSC and transfer cask.

As an alternative, the license applicant may elect to perform more rigorous analyses, such as a dynamic transient analysis which includes material plasticity, and geometric nonlinearities, to demonstrate the adequacy of the designs for site specific conditions. For determination of the drop deceleration magnitudes used in this generic evaluation, it was conservatively assumed that the DSC and transfer cask were rigid with no credit taken for the strain energy absorbed through structural deformations which will act to reduce the equivalent drop decelerations. Furthermore, the DSC was designed for the same drop decelerations as the transfer cask with no credit taken for reductions which will occur due to the energy absorbed through the cask deformations and the strain energy deformations of the DSC itself. Additionally, penetration and spalling of the impacted surface will contribute to reducing the equivalent cask deceleration.

8.2.5.2 Accident Analysis

A. DSC Horizontal Side Drop Analyses

The principal DSC subassemblies affected by the postulated horizontal cask drop are the basket assembly including the spacer disks and the guide sleeves which contain the spent fuel assemblies. The DSC basket assembly is designed so that the spacer disks coincide with the fuel assembly grid spacers. In this manner, the weight of each fuel assembly is transmitted directly to the spacer disk. The guide sleeves serve only as guides for the placement and positioning of the spent fuel assemblies and are not considered principal load bearing members.

(i) DSC Spacer Disk Stress Analysis

The ANSYS (8.48) finite element model shown in Figure 8.2-3, for one half of a typical DSC spacer disk, was developed to calculate the contact area between the spacer disk and DSC shell. A 75g drop deceleration loading for each horizontal spacer disk ligament was calculated and applied as an equivalent static load. The magnitude of the static equivalent applied load on the spacer disk was obtained by multiplying the distributed weights imposed on each guide sleeve and the self weight of the guide sleeve and spacer disk by the maximum deceleration value of 75g. The total equivalent static load imposed on the spacer disk is 591 kips. The distribution of the equivalent static drop loads used for the spacer disk analysis is shown in Figure 8.2-4. Gap elements were used between the outer edge of the spacer disk and the inner surface of the DSC shell to determine the contact area which results during a postulated horizontal cask drop.

A linear elastic stress analysis of a DSC spacer disk was performed for the postulated horizontal drop loads using the analytical model of a complete spacer disk shown in Figure 8.2-5. The contact area between the DSC shell and the spacer disk calculated using the half model analysis was input as a boundary condition. A maximum membrane stress intensity of 36.4 ksi due to axial compression of the vertical ligaments of the spacer disk was computed. The maximum stress intensity due to ligament bending was 26.1 ksi. The resulting stresses are within the ASME Code Service Level D compressive stress limit for accident conditions, of 44.88 ksi for membrane stresses and 64.4 ksi for membrane plus bending.

(ii) DSC Spacer Disk Stability Analysis

In addition, an ANSYS bifurcation buckling analysis of the entire spacer disk was performed to evaluate the global buckling behavior and stability of the spacer disk. The spacer disk model shown in Figure 8.2-5 was used to perform this analysis. The spacer disk analytical model permits out-of-plane deformations, and was assumed to be supported both in-plane at the perimeter of the spacer disc which 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 was the controlling buckling mode for the spacer disk. A factor of safety of 1.80 against collapse of the spacer disk was calculated for the postulated 75g horizontal drop.

(iii) DSC Spacer Disk Deformations

The maximum elastic deflection observed in the spacer disk analysis was 0.050 inches and occurs at mid-span of the spacer disk horizontal ligaments. This magnitude of deflection, is much less than the nominal gap between the fuel assembly and the guide sleeve is sufficiently small to ensure that the fuel assembly integrity is maintained, and to permit fuel assembly retrieval from the DSC. This deformation is based on an elastic analysis, and since the stress levels at some locations exceed the minimum yield stress, some plastic deformation may occur.

To determine the magnitude of plastic deformation of the spacer disk ligaments which may exist due to the postulated horizontal drop load, a plastic analysis of the individual ligaments was performed. By assuming a three hinge beam collapse mechanism subjected to a uniform load, a deformation of 0.022 inches was obtained. Conservatively adding this upper bound plastic deformation to the predicted elastic deformation results in a total deformation of 0.072 inches which demonstrates that the gap between the guide sleeve and the fuel assembly remains open. This magnitude of deformation would not inhibit the retrieval of the spent fuel assemblies if necessary, following a postulated drop accident.

(iv) Guide Sleeve Analysis

The DSC guide sleeves were evaluated for the horizontal drop analysis by considering the maximum span between spacer disks as a simply supported beam. The maximum deceleration value of 75g was statically imposed on the entire length of the guide sleeve acting as a beam, and the maximum stresses obtained. The results of this stress analysis shows that the maximum membrane plus bending stress intensity for the guide sleeve is 2.0 ksi. This value is well within ASME Code acceptable limits and no permanent deformation of the guide sleeves occur.

The maximum membrane stresses in the spacer disk and the guide sleeve due to the postulated horizontal drop accident are tabulated in Table 8.2-7 of this report. Other components of the DSC are not as significantly affected by the postulated horizontal drop loads. As a check, stresses in these components, i.e., DSC shell, cover plates, lead plug, and the support rods, were evaluated as part of the NUHOMS-24P transfer cask analysis.

B. NUHOMS-24P Transfer Cask Horizontal Drop Analyses

An analysis was performed to evaluate the NUHOMS-24P transfer cask for a postulated horizontal drop accident with a static equivalent deceleration of 75g's. The structural capacity of the NUHOMS-24P transfer cask neutron shield was neglected for the horizontal drop accident analysis. In reality, the neutron shield subassembly would provide additional energy absorbing capacity which would further reduce the equivalent drop deceleration magnitude. No credit was taken for the energy absorbing capacityof the neutron shield in performing the cask analysis. It was also conservatively assumed, for the purpose of this generic analysis, that the cask is rigid.

(i) Cask/DSC Analytical Model

Two ANSYS axisymmetric finite element models were utilized for the transfer cask drop analysis, including; one for the cask top region, and the other for the cask bottom region, as shown in Figures 8.2-6 and 8.2-7. Each of the two analytical models consists of the principal load carrying members of the transfer cask, as well as those which contain a significant mass of material. These include the cask structural shell, the radial lead shielding material, the inner liner, and the cask top or bottom cover plates. The DSC shell, lead shield plug assemblies and cover plates were also included in the transfer cask analytical models to provide a more accurate means of applying loads, to evaluate load sharing, and to ensure displacement compatibility between the transfer cask and DSC. The nodal degrees of freedom between the DSC shell and the inner surface of the transfer cask were decoupled in the tangential direction such that the DSC shell could move independently of the cask inner surface in this direction. However, they were coupled in the normal direction such that the DSC outer surface bears on the cask inner surface during a postulated drop accident.

(ii) Cask/DSC Loading Application

The loading due to the transfer cask horizontal drop is non-axisymmetric since it is reacted by a portion of the shell circumference. The loading is assumed to be uniform along the length of the cask. In order to apply this non-axisymmetric loading to the axisymmetric models shown in Figures 8.2-6 and 8.2-7, the loading was resolved into Fourier harmonics using the ANSYS PREP 6 routine. As shown in Appendix C.3, the first twelve Fourier harmonics were chosen to represent the impact force. These Fourier harmonics, expressed in terms of pressure loading, were applied to the exterior nodes of the impacted surface of the cask structural shell.

The DSC loading of the cask includes the DSC and its internals, factored by the equivalent static deceleration value of 75g. These loads were conservatively applied to the transfer cask inner liner at the spacer disk and end plug locations. As for the cask impact force, this loading acts over a portion of the DSC circumference, and is therefore non-axisymmetric. The loading was resolved into Fourier harmonics and the first eight harmonics were selected for application to the axisymmetric model. This loading assumed that the contact surface along DSC the circumference was similar to that obtained from the spacer disk horizontal drop analysis. This assumption does not have a significant impact on the outcome of the analysis since the stresses arise primarily from bearing.

The cask weight, factored by the deceleration values, was applied to the interior nodes of the cask analytical models with its appropriate harmonic components. The cask weight was assumed to have the same circumferential contact surface as the DSC. A detailed description of load development and application of the loads to the axisymmetric model is provided in Appendix C.3.

(iii) Cask/DSC Stress Analysis

The results of the transfer cask analysis for a postulated horizontal drop accident show that the maximum stress in the cask structural shell is 9.1 ksi. Similarly, the maximum stress in the cask inner liner is 2.3 ksi. These stresses are well within the ASME Code Service Level D allowables. The calculated transfer cask and DSC stresses for the horizontal drop are tabulated in Tables 8.2-7 and 8.2-7a.

C. DSC Vertical Drop Analyses

For this drop accident case, the transfer cask is assumed to be oriented vertically and dropped onto a uniform unyielding surface. The vertical cask drop evaluation conservatively assumed that the transfer cask could be dropped onto either the top or bottom surfaces. No credit was taken for the energy absorbing capacity of the cask top or bottom cover plates assemblies during the drop. Therefore, the DSC was analyzed as though it is dropped on to an unyielding surface. The principal components of the DSC and internals affected by the vertical drop are the DSC shell, the top cover plate, the lead shield plugs, the bottom cover plates and the four basket support rods.

The end drop with the bottom end of the DSC oriented downward is the more credible of the two possible vertical orientations. Nevertheless, an analysis for the DSC top end drop accident was also performed. For a postulated vertical drop, membrane stresses in the DSC shell and local stresses at the cover plate weld region discontinuities were evaluated.

(i) DSC Stress Analysis

The ANSYS analytical models for the top and bottom portions of the DSC as described in Section 8.1 and shown in Figures 8.1-5 and 8.1-6, were used to determine the vertical drop accident stresses in the DSC shell, the cover plates, and the lead shield plugs. For both models, the nodal points on the dropped end were restrained in the vertical direction. As discussed previously, and in Appendix C.2, an equivalent static linear elastic analysis, was conservatively used for the vertical drop analyses. Inertia loadings based on forces associated with the 75g deceleration were statically applied to the models. Analyses showed that the maximum stress in the DSC top cover plates occur during a bottom end drop and vice versa. This occurs because during a bottom end drop, the top plates experience bending under their own weight and that of the lead shield plugs. During a top end drop, these cover plates are assumed to be supported by the unyielding target surface and are only subjected to a uniform bearing load imposed by the DSC internals. The same is true for the DSC bottom cover plate. Hence; for the DSC top end model only a bottom end drop was considered while for the DSC bottom end model, only the top end drop was considered.

The results of these analyses produced a maximum membrane stress of 6.2 ksi in the DSC shell. A maximum primary membrane plus bending stress intensity of 33.8 ksi was calculated for the top cover plates and 19.1 ksi for the bottom cover plate. A summary of the calculated stresses for the main components of the DSC and the associated welds is provided in Table 8.2-7.

(ii) DSC Shell Stability Analysis

The stability of the DSC shell for a postulated vertical drop impact was also evaluated. This evaluation was performed using the correlation from Roark.

$$S_{cr} = 0.3 \frac{Et}{R} \quad \text{(Roark, Table 35, Case 15)} \qquad (8.2\text{-}37)$$

Where: S_{cr} = The critical compressive buckling stress

E = 26.5 E6 psi, Modulus of elasticity

t = 0.625 in., Shell thickness

and R = 33.36 in. Mean radius of DSC

Therefore: S_{cr} = 149 ksi

The critical compressive buckling stress provides a factor of safety against buckling of 24. Therefore, buckling of the DSC shell for a 75g vertical deceleration load will not occur.

(iii) DSC Support Rods Analysis

The four DSC spacer disk support rods were also analyzed for the postulated 75g vertical drop. These rods extend the full length of the DSC cavity with adequate clearance for differential thermal expansion. Each of the eight spacer disks are welded to the support rods. The main functions of these rods are to maintain the position of the spacer disks and provide added support for all imposed vertical loads. The support rods are designed to resist the weight of the spacer disks for the postulated vertical drop accidents. The bottom spacer disk and the support rods also provide axial support for the guide sleeves through the associated attachment welds. The guide sleeves are raised above the bottom cover plate by a small amount to facilitate fabrication and draining of the DSC.

For postulated vertical drop accidents, the lower spacer disk would deflect a small amount and come in contact with the bottom cover plate, thus, providing additional support for the guide sleeves for this condition. The most critical segment of the support rod is the cantilever segment above the top spacer disk for the top end drop case. For this analysis, the weight imposed on a single rod is the weight of eight spacer disks divided by four, the weight of the guide sleeves, plus the self weight of one support rod. The axial stress on the top segment of the rod was determined by:

$$S_{mx} = \frac{Wa}{A} \qquad (8.2\text{-}38)$$

Where: S_{mx} = Axial stress

W = 3075 lb., Total weight on one rod

a = 75g, maximum vertical deceleration

A = 7.07 sq. in., Cross sectional area of rod

Therefore: S_{mx} = 32.6 ksi.

The theoretical critical buckling stress of the rod is:

$$S_{cr} = \frac{C\pi^2 E}{\left(\frac{L}{r}\right)^2} \qquad (8.2\text{-}39)$$

Where: S_{cr} = Critical buckling stress limit, psi

C = 0.25 Coefficient of constraint

E = 26.5 E6 psi, Modulus of elasticity

L = 14.3 in., Maximum unsupported length

r = 0.75 in., Radius of gyration

S_{cr} = 180 ksi

Therefore, the axial stress in the support rods is well within the critical buckling stress limit and the compressive stress allowable of 45.1 ksi. The allowable compressive stress was established using the methodology contained in the ASME Code, Section III, Appendix XVII, and Appendix F, in which the effect of the slenderness ratio and plasticity of the rods were considered.

D. NUHOMS-24P Transfer Cask Vertical Drop Analysis:

A vertical drop onto the bottom surface of the transfer cask is highly unlikely. A drop onto the transfer cask top surface is even more unlikely. At no time during the NUHOMS-24P system operations, outside the fuel building, is the transfer cask in a vertical orientation. The transfer cask remains secured to the transport trailer skid in a horizontal position at all times, except for handling operations inside the plant's fuel building which are not formally covered by this 10CFR72 topical report.

All conceivable scenarios leading to a drop accident onto the transfer cask bottom surface are highly remote, with the most probable postulated to occur in the plant's fuel handling building. A drop onto the transfer cask top surface, is not possible even during placement or removal from the skid/transport trailer. However, both of these vertical drop orientations are conservatively postulated for the NUHOMS-24P transfer cask with specific analyses performed to confirm the structural integrity of the transfer cask and the loaded DSC resting in the cask cavity.

(i) Transport Cask Analysis Methodology

The ANSYS axisymmetric finite element models used to perform these analyses were described in Section 8.2.5.2. The loadings due to the postulated vertical drops were applied to the transfer cask analytical models in a symmetric manner. The individual models of the top and bottom cask regions shown in Figures 8.2-6 and 8.2-7 were used for these analyses. The respective top or bottom impacted surface of the transfer cask was assumed to be uniformly supported vertically and the 75g equivalent static decelerations were applied to the models.

(ii) Transfer Cask Stress Analysis

For the top end drop analysis, the stresses in the cover plates are relatively small since they arise primarily from bearing of the DSC and its contents on the cask top cover plates. The most critical vertical drop direction for the transfer cask top region was the bottom end drop, since this produced the maximum bending stress in the top cover plate. The maximum primary bending stress in the cover plate is 24.2 ksi. The maximum local membrane stresses in the cask structural shell and inner liner are 9.6 ksi and 12.9 ksi, respectively. Similarly for the transfer cask bottom region, the most critical drop direction was the top end drop producing a maximum bending stress of 22.9 ksi in the cask bottom cover plate. These stresses are well below the appropriate ASME Code Service Level D allowables.

E. NUHOMS-24P Transfer Cask/DSC Corner Drop Analyses

The possibility of a drop onto the top or bottom end corners of the NUHOMS 24-P transfer cask is extremely remote due to the limited cask handling operations of the NUHOMS system, as discussed previously. Nevertheless, for the purpose of this generic evaluation, a cask corner drop is conservatively postulated to occur onto a concrete surface with an equivalent static deceleration of 25g. The orientation of the drop is shown in Figure 8.2-2 as occurring at 30° to the horizontal. This is the largest drop orientation angle which can occur as the center of gravity of the cask passes beyond the back end of the transport trailer and pitches downward. The derivation of this load definition is contained in Appendix C.2.

It is probable that the cask support skid would remain firmly attached to the cask and would absorb considerable energy upon impact, thus reducing the transfer cask deceleration. In addition, this would further reduce the angle of the impact and the drop height. The combined support skid and transfer cask would act as a substantial energy absorbing mechanism thus significantly reducing the effects of impact loads on the DSC and the spent fuel assemblies. Also, for the postulated case of the cask sliding forward, the cask and skid may initially impact the tractor unit, prior to pitching onto the ground, with significant reductions in the resulting impact velocity and the energy imparted to the transfer cask and its contents.

(i) Cask/DSC Analysis Methodology

The combined transfer cask/DSC ANSYS linear elastic axisymmetric models used in the side drop and the end drop analyses as shown in Figures 8.2-6, and 8.2-7, were used for the corner drop analyses. The postulated transfer cask corner drop accident results in a very complex loading function because it involves both symmetric and asymmetric load components in both the vertical and horizontal directions. The analysis involved the development of the impact force as well as the content loading and applying these loads to the axisymmetric model as Fourier harmonics. A complete description of the load development and application of the loads to the ANSYS models is provided in Appendix C.2.

(ii) Cask/DSC Stress Analysis

The resulting local primary membrane and primary bending stresses in the transfer cask due to both the postulated top and bottom corner drop analysis are tabulated in Table 8.2-7a. The resulting stresses in the DSC due to a cask corner drop were evaluated and found to be enveloped by those calculated for the 75g end and side drop analyses. As can be seen from the results, the DSC and transfer cask stress intensities are within the appropriate ASME Code Service Level D allowable limits.

8.2.5.3 Accident Dose Calculations for Loss of Neutron Shield

The design basis cask drop analyses have shown that all components important to safety, including the NUHOMS-24P transfer cask, the DSC and its internal basket assembly, will maintain their structural integrity. For the purpose of this conservative generic analysis, it is assumed that the transfer cask neutron shield will be breached as a result of a postulated drop accident, and the shielding effect of the liquid will be lost. The effect of this will increase the cask surface contact dose from 170 mrem/hour to 837 mrem/hour. The only potential off-site dose consequences would be additional direct and air scattered radiation if the accident were to occur sufficiently close to the site boundary. It is assumed that eight hours would be required to either recover the neutron shield or to add temporary shielding while arranging recovery operations. As a result, it is estimated that on-site workers at an average distance of fifteen feet would receive an additional dose rate of 80 mrem/hr.

Off-site individuals at a distance of 2000 feet would receive an additional dose of 5.7E-4 mrem for the assumed eight hour exposure. This increase is well within the limits of 10CFR72 for an accident condition. Also, this does not preclude handling operations for recovery of the cask and its contents. Water bags or other neutron absorbing material could be wrapped around the cask to reduce the surface dose to an acceptable limit for recovery operations thus minimizing exposure of personnel in the vicinity. The actual local and off-site dose rates, recovery time and operations needed to retrieve the cask, and the required actions to be performed following the event will depend upon the severity of the event and the resultant cask and trailer/skid damage.

8.2.5.4 Recovery For drop heights of less than fifteen inches the transfer 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 plant's fuel building. There the DSC will be inspected for damage, and the DSC opened and the fuel removed for inspection, as necessary. Removal of the transfer 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 decontamination pit after recovery of the transfer cask. Removal of the DSC cover plates and lead plug assembly are described in Section 5.0 of this report.

Following recovery of the transfer cask and unloading of the DSC, the transfer 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.

As described in Section 8.1.3.3, the maximum fuel cladding temperatures are well below the short term allowable maximum temperature of 570°C for this event. The maximum transfer cask exterior surfaces will increase to 223°F. The transfer cask will be roped off to ensure the safety of the site personnel.

8.2.6 Lightning

8.2.6.1 Postulated Cause of Event The likelihood of lightning striking the HSM and causing an off-normal condition is not considered to be a credible event since a lightning protection system is provided for the HSM. Lightning protection for the HSM will be provided by either taller grounded structures in the vicinity of the HSM, or by a lightning protection system installed on the HSM. A lightning protection system installed on the HSM will be complete with air terminals, conductors, ground terminals, and other fittings necessary to satisfy the Lightning Protection Code NFPA 78-1986 (8.40). Lightning protection system requirements will be site specific and will depend upon the frequency of occurrences of lightning storms in the proposed ISFSI location and the degree of protection offered by other grounded structures in the proximity of the HSMs.

8.2.6.2 Analysis of Effects and Consequences Should lightning strike in the vicinity of the HSM the normal storage operations of the HSM will not be affected. The current discharged by the lightning will follow the low impedance path offered by the surrounding structures or lightning protection system. Therefore, the HSM will not be damaged by the heat or mechanical forces generated by current passing through the higher impedance concrete. Since the HSM requires no equipment for its continued operation, the resulting current surge from the lightning will not affect the normal operation of the HSM.

Since no off-normal condition will develop as the result of lightning striking in the vicinity of the HSM, no corrective action would be necessary. Also, there would be no radiological consequences.

8.2.7 Blockage of Air Inlets and Outlets

This accident conservatively postulates the complete blockage of the HSM ventilation air inlets and outlets.

8.2.7.1 Cause of Accident Since the NUHOMS HSMs are located outdoors, there is a remote probability that the ventilation air inlets and outlets could be blocked by debris from such unlikely events as floods and tornados. The NUHOMS design features such as the perimeter security fence and spatial separation of the air inlets and outlets reduces the probability of occurrence of such an accident. Nevertheless, for the purpose of this conservative generic analysis, such an accident is postulated to occur and has been analyzed.

8.2.7.2 Accident Analysis The structural consequences due to the weight of the debris blocking the air inlets and outlets are negligible and are bounded by the HSM loads induced for a postulated tornado (Section 8.2.2) or earthquake (Section 8.2.3).

The thermal effects of this accident results from the increased temperatures of the DSC and the HSM due to blockage of the ventilation air inlets and outlets. The heat generated in the DSC is conservatively assumed to be contained entirely within the DSC and HSM during the course of this postulated accident. The accident duration is assumed to be 48 hours, at which time the air inlet and outlet obstructions would be cleared by site personnel and natural circulation air flow restored to the HSM.

Based on the NUHOMS-07P design, it was concluded that heat-up of the spent fuel, DSC and HSM are limited by the heat up of the HSM. The spent fuel assemblies and the DSC quickly rise in temperature to the level required to radiate and conduct the 15.8 kw of decay heat to the HSM internal surfaces. However, the HSM surface heat up is limited by the heat up of the entire HSM. Because the heat up rate of the HSM is much lower than that of the DSC, or the spent fuel, the DSC can be assumed to be at steady state at any instant in time and transferring 15.8 kw of heat to the HSM. The HSM internal surface temperatures are limited by the heat up of the total quantity of concrete behind the surface. Therefore, applying a constant heat flux to the HSM concrete and calculating the time dependent temperature distributions through the concrete, the surface temperature of the concrete as a function of time was obtained. Using the calculated HSM surface temperatures, the maximum DSC and fuel cladding temperatures were determined.

A thermal transient analysis of the HSM for the blocked vent condition was performed using a control volume model of the HSM internal air space and the surrounding concrete walls, roof, and floor. The first law of thermodynamics was used to obtain an energy balance equation which was solved using a forward finite differencing scheme with a sufficiently small time step to ensure the accuracy of the solution. The initial conditions for the analysis correspond to the steady state temperatures calculated for the off-normal analysis cases with ambient temperatures of -40°F and 125°F. The heat source included in the analysis is the 15.8 Kw decay heat rejected from the surface of the DSC. The solution was carried out to 48 hours at which time it is assumed that corrective action would be completed and natural circulation air flow restored to the HSM. The change in temperature with time after vent blockage for the HSM roof interior surface is shown in Figure 8.2-12.

At the end of the 48 hours required to clear the inlets and outlets, the maximum HSM inside surface is 395°F, and the DSC surface temperature is 455°F to radiate the decay heat from the spent fuel assemblies. Using the HEATING6 model of the DSC and its internals described in Section 8.1.3, the maximum fuel cladding temperature for this case was determined to be 757°F (403°C). The fuel clad temperature at the start of the blocked vent transient for the 125°F extreme ambient temperature case was calculated to be 668°F (353°C). The maximum fuel clad temperature as a function of time is estimated to be linear between these two temperature values. The resulting temperatures are well below the fuel cladding short term temperature limit of 570°C.

These temperatures are below the levels for which safety impairing damage would occur to the HSM or DSC. The short time exposure of the DSC and the spent fuel assemblies to the elevated temperatures will not cause any damage or result in the release of radioactivity. The maximum DSC internal pressure during this event is 9.7 psig (assuming that no fission and fill gas is released). If the fission and fill gases were released from the spent fuel rods, the DSC internal pressure would be 46.7 psig as shown in Table 8.1-4. This pressure is considered the design basis accident pressure for analysis of the DSC.

The thermal-induced stresses for the blocked vent case were calculated using the 2x10 HSM array structural model shown in Figure 8.1-10a as discussed in Section 8.1.1.5, paragraphs C and E. The location of the module in the array with vents assumed to be blocked was varied to ensure that the most conservative induced forces and moments were calculated. The non-linear transient thermal gradients were converted to equivalent linear gradients in accordance with the guidelines of ACI 349 Appendix A. The worst case equivalent linear thermal gradients were then applied to the Figure 8.1-10a computer model to calculate the elastic forces and moments induced. The resulting elastic forces and moments were modified to account for the concrete cracked section properties in accordance with ACI 349 Appendix A, and combined with the calculated forces and moments from other loads.

<u>8.2.7.3 Accident Dose Calculations</u> There are no off-site dose consequences as a result of this accident. The only significant dose increase is that related to the recovery operation where it is conservatively estimated that the on-site workers will receive an additional dose of no more than 700 mrem during the eight hour period it is estimated may be required for removal of the debris from the air inlets and outlets.

<u>8.2.7.4 Recovery</u> Debris removal is all that is required to recover from a postulated blockage of the HSM ventilation air inlets and outlets. Cooling will begin as soon as the debris is removed from the inlets and outlets. The amount and nature of debris can vary, but even in the most extreme case, manual means

or readily available equipment (front-end loader, etc.) can be used to remove debris.

The debris is conservatively assumed to remain in place for 48 hrs as described in Section 8.2.7.2. The last eight hours of this period are assumed to be the time required to completely remove all debris and the natural circulation air flow to be restored.

8.2.8 DSC Leakage

The DSC shell is designed as a pressure retaining containment boundary to prevent leakage of contaminated materials, as discussed in Section 8.1.1.1, paragraph B. 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, for the purpose of this conservative generic evaluation, a total and complete instantaneous leak from the DSC was postulated.

This event postulates an instantaneous release directly to the environment of 30% of all fission gasses contained in all the spent fuel rods in all 24 PWR fuel assemblies. This accident conservatively assumes that all spent fuel rods are ruptured and that concurrent DSC leakage occurs. All other components of the NUHOMS system remain intact.

8.2.8.1 Cause of Accident There is no credible event that could result in the rupture of any spent fuel rods concurrent with leakage from the DSC due to the passive nature of the NUHOMS system, and the various design features which ensure that the integrity of the DSC shell containment pressure boundary and spent fuel cladding are maintained. Nevertheless for the purpose of this conservative generic evaluation of the NUHOMS system, this accident assumes that the spent fuel rods and the DSC are ruptured due to an event of unspecified origin.

8.2.8.2 Accident Analysis There are no structural or thermal consequences resulting from the DSC leakage accident described above. The radiological consequences of this accident are described in Section 8.2.8.3.

8.2.8.3 Accident Dose Calculations The postulated accident assumes that one DSC is ruptured and that all of the spent fuel rod cladding fails simultaneously such that 30% of all the fission gasses in the spent fuel assemblies (mainly Kr-85) are instantaneously released to the atmosphere. The whole body dose and skin dose at 1000 feet from the storage site under the worst meteorologic conditions were calculated and are listed in Table 8.2-8. From this table it can be seen that the resultant accident dose is well within the 10CFR72.68 limit, which restricts the maximum whole body or organ dose beyond the owner controlled area from any design basis accident to be less than five Rem. Doses (on-site and off site) will require a site specific evaluation. Table 8.2-8 shows that typical site boundary doses are below the 10CFR72 limits.

8.2.9 Accident Pressurization of DSC

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

8.2.9.1 Accident Analysis

The bounding internal pressurization of the DSC for the purpose of this conservative generic evaluation is postulated to result from cladding failure of the spent fuel, and the consequent release of spent fuel rod fill gas and free fission gas. Fission gas release fractions are not so easily estimated however. A recent report on ISFSI facility evaluations (8.43) uses a release fraction of 8% as a nominal case and 30% as an upper bound.

For design purposes, and as a means of providing over pressure protection for the DSC, it was conservatively assumed that all fuel rods in the DSC suffer cladding failure, as discussed in Section 8.1.1.1, paragraph B. It was further assumed that the fission gas release fraction is 30%. This results in release of 10,000 $in.^3$ of fission gas (interpolated from the data provided in Reference 8.42) assuming the original fuel rod fill gas pressure was 480 psig (10,866 $in.^3$ per assembly at 32°F and one atm.) As shown in Table 8.1-4 and Figure 8.1-25b, the resulting DSC pressure is 37.7 psig when the outside ambient air temperature is 125°F. When the ambient air temperature is -40°F, the DSC pressure is 32.2 psig. The limiting postulated accident for DSC pressurization is the loss of the transfer cask neutron shield during transfer operations. As shown in Table 8.1-4 and Figure 8.1-25b the helium gas temperatures inside the DSC will rise to 600°F producing a DSC internal pressure of 49.1 psig. The stress analysis of the DSC shell assembly for an internal pressure of 49.1 psig is described in Section 8.1.1.1. The maximum DSC shell membrane stress intensity calculated was 2.6 ksi.

<u>8.2.9.2 Accident Dose Calculations</u> There are no dose consequences as the result of the accidental pressurization of the DSC.

8.2.10 Load Combinations

The load categories associated with normal operating conditions, off-normal conditions and postulated accident conditions have been described and analyzed in previous sections of this report. The load combination results for the NUHOMS components important to safety are presented in this section. Fatigue effects on the NUHOMS-24P transfer cask and the DSC are also addressed in this section.

8.2.10.1 DSC Load Combination Evaluation As described in Section 3.2 of this report, the stress intensities in the DSC at various critical locations for the appropriate normal operating condition loads were combined with the stress intensities experienced by the DSC during postulated accident conditions. It was assumed that only one postulated accident event occurs at any one time. Since the postulated cask drop accidents are by far the most critical, the load combinations for these events envelope all other accident event combinations. Tables 8.2-9 through 8.2-9c tabulate the maximum stress intensity for each component of the DSC calculated for the enveloping normal operating, off-normal, and accident load combinations. For comparison the appropriate ASME Code allowables are also presented in these tables.

8.2.10.2 DSC Fatigue Evaluation Tables 8.2-9 through 8.2-9c present the calculated enveloping normal, off-normal and accident stress intensities for the DSC components. Fatigue effects on the DSC were addressed using the criteria contained in NB-3222.4 of the ASME Code (Reference 8.3). Fatigue effects need not be specifically evaluated provided the six criteria contained in NB-3222.4(a) are met. As demonstrated in Appendix C.4.1, an evaluation using these six criteria was performed to show that the ASME Code fatigue requirements are satisfied for the DSC.

8.2.10.3 NUHOMS-24P Transfer Cask Load Combination Evaluation
As described in Section 3.2 of this report, the NUHOMS-24P transfer cask calculated stresses due to normal operating loads were combined with the appropriate calculated stresses from postulated accident conditions at critical stress locations. It was assumed that only one postulated accident can occur at a time. Also, since the postulated drop accidents produce the highest calculated stresses, the load combination of dead load plus drop accident envelopes the stresses induced by other ostulated accident scenarios. The maximum calculated stress intensities for the transfer cask normal operating, off-normal, and accident load combinations are tabulated in Tables 8.2-13

through 8.2-15, with the corresponding ASME Code allowables for comparison.

8.2.10.4 NUHOMS-24P Transfer Cask Fatigue Evaluation As for the the DSC, fatigue effects on the transfer cask were addressed using the criteria provided in NC-3219.2 of the ASME Code (Reference 8.3). As described in Appendix C.4.2 the code specified criteria given in this section were evaluated relative to the transfer cask to demonstrate that fatigue requirements have been satisfied.

8.2.10.5 HSM Load Combination The maximum bending moments and shear forces induced in the HSM for the individual normal and off-normal loads are listed in Table 8.1-10. Similarly, the maximum moments and shears induced in the HSM for the individual accident loads are listed in Table 8.2-3. As described in Section 3.2.5.1, the load combination procedure of Section 6.17.3.1 of ANSI 57.9 (8.2) was used to combine the factored normal operation, off normal, and postulated accident loadings imposed on the reinforced concrete HSM. Many of the general event combinations, shown in Table 3.2-5, are enveloped by others which contain the same load factors with additional applied load cases. As stated earlier in this report, the effects of shrinkage and creep were included with the dead load in those load combinations where these effects increased the calculated stresses.

The maximum calculated bending moments and shears for each load combination are tabulated in Table 8.2-10. The tabulated results represent the bounding shears and moments for either the single free standing HSM or a 2x10 array of HSMs. For comparison, the minimum ultimate moment and shear capacity of the HSM calculated using concrete properties at 400°F are also shown in Table 8.2-10. Comparison of the maximum bending moment and shear for each load combination with the ultimate capacity shows that the design strength of the HSM is greater than the strength required for the most critical load combination.

8.2.10.6 Thermal Cycling of the HSM As stated earlier, the largest mean daily change of temperature in the United States of 45°F occurs in Reno, Nevada. Because of the massive concrete sections used in the HSM a period of about one week is needed to obtain steady temperatures and a steady state thermal gradient. For conservatism it was assumed that the 45°F maximum daily change could produce a steady state gradient every day for 50 years, for a total of 18250 thermal cycles. The maximum moment caused by the worst case steady state normal operating thermal loads, is 575 k-in. This loading is 16% of the ultimate strength. Assuming this load amplitude is cycled daily, and referring to the S-N curve of Figure 6-41 of reference 8.22, the number of cycles before failure will occur is greater than 10,000,000. Since this value is far greater than the postulated worst case 18,250 cycles, thermal cycling has negligible effect on the HSM reinforced concrete.

8.2.10.7 DSC Support Assembly Load Combination The applicable loads for the DSC support assembly inside the HSM include the DSC and support assembly dead weight, seismic loads, and DSC handling loads. Three load combinations were evaluated. Load Combination one consists of the DSC plus the support assembly dead weight, plus the DSC handling loads for a typical normal operating load case.. Load Combination two includes the dead weight of the support structure plus DSC handling loads in the jammed condition representing an off-normal loading. The third load combination included the total dead weight plus design basis seismic loads for an accident event. The resulting maximum stresses were compared to AISC code allowables are shown in Table 8.2-11.

The same load combinations were used for the DSC support structure connecting elements. The maximum support loads for the design basis load combinations are shown in Table 8.2-12. All end connection components are designed to meet the AISC Code requirements for these design loads. The structural seated beam support design is based on the requirements of the AISC code, and the embedments are designed in accordance with the requirements of ACI 349-85.

Table 8.2-1

NUHOMS-24P ACCIDENT LOADING IDENTIFICATION

Accident Load Type	Section Reference	NUHOMS-24P Component Loads				
		DSC Shell Assembly	DSC Internal Basket	DSC Support Assembly	HSM	NUHOMS-24P Transfer Cask
Loss of HSM Air Outlet Shielding Blocks	8.2.1	(radiological consequence only)				
Tornado Wind	8.2.2				X	X
Tornado Missiles	8.2.2				X	
Earthquake	8.2.3	X	X	X	X	X
Flood	8.2.4	X			X	
Accidental Cask Drop	8.2.5	X	X			X
Loss of Liquid Neutron Shield	8.2.5					X
Lightning	8.2.6				X	
Blockage of HSM Air Inlets and Outlets	8.2.7	X	X	X	X	
DSC Leakage	8.2.8	(radiological consequence only)				
DSC Accident Internal Pressure	8.2.9	X				
Load Combinations	8.2.10	X	X	X	X	X

Table 8.2-2

COMPARISON OF TOTAL DOSE RATES
FOR HSM WITH AND WITHOUT
AIR OUTLET SHIELDING BLOCKS

Distance (meters) from Nearest HSM Wall, 2x10 Array	Normal Case Dose Rate* (mrem/hr.) (With Shield Blocks)	Accident Case Dose Rate* (mrem/hr.) (Without Shield Blocks)
10	2.85	21.9
100	0.0587	0.533
500	8.97E-4	2.14-23
2000	3.77E-8	9.62E-7

* Air scattered plus direct radiation.

Table 8.2-3

MAXIMUM HSM REINFORCED CONCRETE BENDING MOMENTS AND SHEAR FORCE FOR ACCIDENT LOADS

Structural Section	Force Component	HSM Internal Forces (in, kips)					Ultimate Capacity (in, kips)
		Tornado Winds	Tornado Missile	Seismic	Flooding	Blocked Vents Thermal (7)	
Floor Slab	Shear	3.7	N/A	18.7	3.2	52.5	70.4 (6)
	Moment	134	N/A	686	114	1340	2840
Inner Wall	Shear	0.5	N/A	9.8	0.3	15.9	27.4
	Moment	33	N/A	476	20	653	1730
End Wall	Shear	3.7	974	13.9	3.4	56.2	2150 (4) / 60.9
	Moment	129	1630	769	104	2520	3570
Roof Slab	Shear	0.4	974	7.3	0.4	77.2	2150 (4) / 97.8 (6)
	Moment	19	1630	487	24	2750	3570

Notes:

1. Maximum loads shown are irrespective of location.
2. Forces and capacities are calculated per 12 in. section of HSM with the exception of tornado missile case.
3. Concrete and reinforcing steel properties are at 400°F to conservatively envelope all ambient cases.
4. The maximum shear and ultimate shear capacity calculated for the DBT Missile are based on the punching shear at the periphery of the impacted area.
5. Seismic loading is based on a vertical acceleration of 0.22g and a horizontal acceleration of 0.32g. To account for possible multiple excitations, a factor of 1.5 was included in the seismic analysis results.

Table 8.2-3

MAXIMUM HSM REINFORCED CONCRETE BENDING

MOMENTS AND SHEAR FORCE FOR ACCIDENT LOADS

(Concluded)

Notes:

6. Ultimate shear capacity was calculated using Equation 11-29 of ACI 349-85 except for the tornado missile ultimate shear capacity which is discussed in Note 4. The values of V_u and M_u used in Equation 11-29 were the maximum calculated values from the blocked vent load cases.

7. Maximum moments are calculated using cracked section properties.

Table 8.2-4 through Table 8.2-6

(DELETED)

Table 8.2-7

MAXIMUM DSC STRESSES FOR DROP ACCIDENT LOADS

DSC Components	Stress Type	Calculated Stress (ksi)	
		Vertical	Horizontal
DSC Shell	Primary Membrane	6.2	9.2
	Membrane + Bending	19.3	12.4
Inner Top Cover Plate	Primary Membrane	0.0	14.0
	Membrane + Bending	33.8	15.9
Outer Top Cover Plate	Primary Membrane	0.0	9.5
	Membrane + Bending	20.1	14.6
Bottom Cover Plate	Primary Membrane	0.0	9.5
	Membrane + Bending	19.1	14.6
Spacer Disk	Primary Membrane	1.0	36.4
	Membrane + Bending	22.4	26.1
Guide Sleeve	Primary Membrane + Bending	0.0	2.0
Support Rods	Primary Membrane	32.6	0.0
Top End Structural Weld	Primary	5.0	9.5
Bottom End Structural Weld	Primary	4.1	9.5

Notes:

1. Values shown are maximums irrespective of location.
2. DSC was also included in corner drop analysis for cask, however, stresses for above cases are enveloping.

Table 8.2-7a

MAXIMUM NUHOMS-24P TRANSFER CASK STRESSES
FOR DROP ACCIDENT LOADS

NUHOMS-24P Transfer Cask Components	Stress Type	Stress (ksi)[1]		
		Vertical	Horizontal	Corner[2]
Transfer Cask Structural Shell	Primary Membrane	9.6	2.2	4.6
	Membrane + Bending	10.2	9.1	13.9
Top Closure Plate	Primary Membrane	24.2	2.8	2.7
	Membrane + Bending	24.2	2.8	14.1
Bottom Cover Plate	Primary Membrane	22.9	2.8	0.0
	Membrane + Bending	22.9	2.8	33.1

Notes:

1. Values shown are maximums irrespective of location.

2. DSC was also included in corner drop analysis. DSC stresses for this case are enveloped by those for horizontal and vertical drop loads shown in Table 8.2-7.

Table 8.2-8

DOSE AT 300m FROM HSM SITE
DUE TO Kr-85 RELEASE FROM
POSTULATED DSC RUPTURE

Dose Type	χ /Q* (Sec./m^3)	Dose (Rem)
Whole Body	5.0E-3	3.8E-2
Skin	5.0E-3	4.8E-0

* Taken from Reg. Guide 1.4 Ground Level Release Data

Table 8.2-9

DSC ENVELOPING LOAD COMBINATION

RESULTS FOR NORMAL AND OFF-NORMAL LOADS

(ASME SERVICE LEVELS A AND B)

DSC Components	Stress Type	Controlling Load Combination (1)	Stress (ksi)	
			Calculated	Allowable (2)
DSC Shell	Primary Membrane	B2	1.8	18.7
	Membrane + Bending	B3	7.4	28.0
	Primary + Secondary	B3	38.7	56.1
Bottom Cover Plate	Primary Membrane	A3	0.8	18.7
	Membrane + Bending	B3	7.5	28.0
	Primary + Secondary	B3	5.3	56.1
Inner Top Cover Plate	Primary Membrane	A3	0.2	18.7
	Membrane + Bending	A2	5.1	28.0
	Primary + Secondary	B3	0.3	56.1
Outer Top Cover Plate	Primary Membrane	A3	0.2	18.7
	Membrane + Bending	A2	5.6	28.0
	Primary + Secondary	A4	4.8	56.1
Spacer Disk	Primary Membrane	A3	0.5	18.7
	Membrane + Bending	A3	0.5	28.0

Notes:

See Table 8.2-9c for notes.

Table 8.2-9a

DSC ENVELOPING LOAD COMBINATION RESULTS

FOR ACCIDENT LOADS

(ASME SERVICE LEVEL C)

DSC Components	Stress Type	Controlling Load Combination (1)	Stress (ksi)	
			Calculated	Allowable (2)
DSC Shell	Primary Membrane	C6	3.8	22.4
	Membrane + Bending	C6	12.4	31.0
Bottom Cover Plate	Primary Membrane	C6	0.1	22.4
	Membrane + Bending	C6	11.4	31.0
Inner Top Cover Plate	Primary Membrane	C6	0.1	22.4
	Membrane + Bending	C6	23.2	31.0
Outer Top Cover Plate	Primary Membrane	C6	0.1	22.4
	Membrane + Bending	C6	23.2	31.0
Spacer Disk	Primary Membrane	C1	0.5	22.4
	Membrane + Bending	C1	0.3	31.0

Notes:

See Table 8.2.9c for notes.

Table 8.2-9b

DSC ENVELOPING LOAD COMBINATION

RESULTS FOR ACCIDENT LOADS

(ASME SERVICE LEVEL D) (3)

DSC Components	Stress Type	Controlling Load Combination (1)	Stress (ksi)	
			Calculated	Allowable (2)
DSC Shell	Primary Membrane Membrane + Bending	D2 D2	11.9 25.9	44.9 64.4
Bottom Cover Plate	Primary Membrane Membrane + Bending	D2 D2	9.5 28.4	44.9 64.4
Inner Top Cover Plate	Primary Membrane Membrane + Bending	D2 D2	14.0 57.5	44.9 64.4
Outer Top Cover Plate	Primary Membrane Membrane + Bending	D2 D2	9.5 43.5	44.9 64.4
Spacer Disk	Primary Membrane Membrane + Bending	D2 D2	36.4 26.4	44.9 64.4
Guide Sleeve	Membrane + Bending	D2	2.0	64.4
Support Rods (4)	Primary Membrane	D2	32.6	45.0
Top End Structural Weld	Primary	D2	15.5	44.9
Bottom End Structural Weld	Primary	D2	12.0	44.9

Notes:

See Table 8.2-9c for notes.

Table 8.2-9c

DSC ENVELOPING LOAD COMBINATION TABLE NOTES

1. See Table 3.2-5a for load combination momenclature.
2. See Table 3.2-6 for allowable stress criteria. Material properties were obtained from Table 8.1-2 at a design temperature of 400°F.
3. In accordance with the ASME Code, thermal stresses need not be included in Service Level D load combinations.
4. Compressive stress allowable of the support rods is based on the criteria specified in Appendix XVII and Appendix F of the ASME Code.

Table 8.2-10

HSM ENVELOPING LOAD COMBINATION RESULTS

Load[1] Combination	Loading Combination Description	Maximum Loading		Capacities[7]	
		V_{max} (k)	M_{max} (k.in.)	V_u (k)	M_u (k.in.)
1	1.4D + 1.7L	4.8	233	43.8	3570
2	1.4D + 1.7L + 1.7H	4.8	233	43.8	3570
3	0.75(1.4D + 1.7L + 1.7H + 1.7T + 1.7W)	34.5	867	43.8	3570
4	0.75(1.4D + 1.7L + 1.7H + 1.7T)	34.5	867	43.8	3570
5	D + L + H + T + E	42.8	1220	43.8	3570
6	D + L + H + T + F	40.9	1100	43.8	3570
7	D + L + H + T_a	79.4	2800	104.[8]	3570

D = Dead Weight, E = Earthquake Load, F = Flood Induced Loads, H = Lateral Soil Pressure Load, L = Live Load, T = Normal Condition Thermal Load, T_a = Off-normal or Accident Condition Thermal Load, W = Tornado Wind and Missile Loads

Notes:

1. Load combinations are based on ANSI-57.9 as shown in Table 3.2-5.
2. Maximum loads shown are irrespective of locations.
3. Thermal accident load (T_a) are based on 125°F ambient with air inlets and outlets blocked (See Section 8.1.2.2).
4. V_{max}, V_u, M_{max}, and M_u calculated per 12" section of HSM.
5. Results of load combinations 3 through 7 are based on cracked section. Others based on uncracked sections.
6. Material properties taken at 400°F for all load combinations.
7. V_u values based on allowable shear for deep flexural members, ACI 349-85, Section 11.8.
8. The shear capacity V_c is calculated using Equation 11-29 of ACI 349-85.

Table 8.2-11

DSC SUPPORT ASSEMBLY ENVELOPING

LOAD COMBINATION RESULTS

Component	Load Combination	Calculated Stress			AISC Allowable Stress		
		Axial (ksi)	Bending (ksi)	Shear (ksi)	Axial (ksi)	Bending (ksi)	Shear (ksi)
W10x68 Cross Beam	Normal Operation $DW_s + DW_c + HL_f$	0.4	7.2	4.8	14.8	17.6	10.6
	Off-Normal Operation $DW_s + HL_j$	0.01	12.6	1.4	14.8	17.6	10.6
	Accident $DW_s + DW_c + DBE$	1.0	11.3	7.8	22.3	26.3	16.0
WT6x115 Support Rail	Normal Operation $DW_s + DW_c + HL_f$	0.2	12.5	1.5	13.8	17.6	10.6
	Off-Normal Operation $DW_s + HL_j$	0.8	6.4	0.3	13.8	17.6	10.6
	Accident $DW_s + DW_c + DBE$	0.5	22.7	2.9	20.7	26.3	16.0
KEY: DW_s = Dead Weight Support Assembly, HL_j = Off-normal Handling Loads-Jammed, DW_c = Dead Weight Canister, HL_f = Normal Loads Friction, DBE = Seismic Loads							

Notes:

1. Maximum stresses reported irrespective of location.
2. Allowable stresses taken at 600°F to conservatively envelope all ambient temperature cases.
3. Allowables for $DW_s + DW_c$ + DBE increased by 50%.
4. Seismic stresses are reported for the calculated response of three components fo earthquake summed absolutely.

Table 8.2-12

DSC SUPPORT ASSEMBLY ENVELOPING LOAD COMBINATION RESULTS
CROSS MEMBER END CONNECTION LOADS

Load Combination	Maximum End Loads			
	F_x (k)	F_y (k)	F_z (k)	M_x (k-in.)
Normal Operation $DW_s + DW_c + HL_f$	8.52	23.6	6.65	2.64
Off-Normal Operation $DW_s + HL_j$	14.80	3.10	0.12	1.8
Accident $DW_s + DW_c + DBE^{(4)}$	13.35	38.00	19.20	3.72
Design Loads	14.80	25.63	12.80	5.20
Key: DW_s = Dead Weight Support Assembly, HL_J = Off-normal Handling Loads - Jammed, DW_c = Dead Weight Canister, HL_f = Normal Handling Loads - Friction, DBE = Seismic Loads				

Notes:

1. Maximum loads reported irrespective of location.

2. Global coordinate system used as shown on Figure 8.1-8.

3. All F_y loads are downward. F_x, F_z, and M_z loads are reversible.

4. These loads are compared to 1.5 times the basic AISC allowables for the connection design. For comparison with the design loads, divide values given by 1.5 to account for increase in allowable stresses. For the design of DSC support embedments the calculated loads were conservatively increased by a factor of 1.7 to give ACI 349-85 design ultimate loads required.

Table 8.2-13

NUHOMS-24P TRANSFER CASK ENVELOPING LOAD COMBINATION RESULTS FOR NORMAL AND OFF-NORMAL LOADS (ASME SERVICE LEVELS A AND B)

Transfer Cask Component	Stress Type	Controlling Load Combination (1)	Stress (ksi)	
			Calculated	Allowable (2)
Structural Shell	Primary Membrane	A4	8.2	21.7
	Membrane + Bending	A4	31.1	32.6
	Primary + Secondary	A4	56.0	65.1
Top Cover Plate	Primary Membrane	A1	0.2	21.7
	Membrane + Bending	A4	6.5	32.6
	Primary + Secondary	A4	7.4	65.1
Bottom Cover Plate	Primary Membrane	A1	0.2	21.7
	Membrane + Bending	A4	14.4	32.6
	Primary + Secondary	A4	5.3	65.1

Notes:

1. See Table 3.2-5b for load combination nomenclature.

2. See Table 3.2-8 for allowable stress criteria. Material properties were obtained from Table 8.1-2 at a design temperature of 400°F.

Table 8.2-14

NUHOMS-24P TRANSFER CASK ENVELOPING LOAD COMBINATION

RESULTS FOR ACCIDENT LOADS

(ASME SERVICE LEVEL C)

Transfer Cask Component	Stress Type	Controlling Load Combination (1)	Stress (ksi)	
			Calculated	Allowable (2)
Structural Shell	Primary Membrane	C1	1.2	26.0
	Membrane + Bending	C1	31.1	39.1
Top Cover Plate	Primary Membrane	C1	0.1	26.0
	Membrane + Bending	C1	6.5	39.1
Bottom Cover Plate	Primary Membrane	C1	0.1	26.0
	Membrane + Bending	C1	14.4	39.1

Notes:

1. See Table 3.2-5b for load combination nomenclature.

2. See Table 3.2-8 for allowable stress criteria. Material properties were obtained from Table 8.1-2 at a design temperature of 400°F.

Table 8.2-15

NUHOMS-24P TRANSFER CASK ENVELOPING LOAD COMBINATION

RESULTS FOR ACCIDENT LOADS

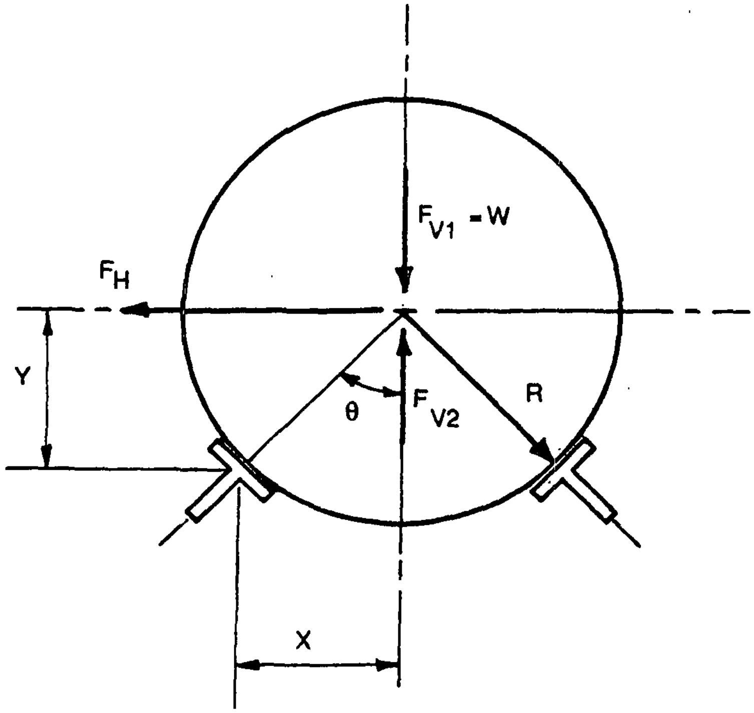
(ASME SERVICE LEVEL D)

Transfer Cask Component	Stress Type	Controlling Load Combination (1)	Stress (ksi)	
			Calculated	Allowable (2)
Structural Shell	Primary Membrane	D1	9.7	49.0
	Membrane + Bending	D2	14.3	70.0
Top Cover Plate	Primary Membrane	D1	24.4	49.0
	Membrane + Bending	D1	24.4	70.0
Bottom Cover Plate	Primary Membrane	D1	23.1	49.0
	Membrane + Bending	D2	33.3	70.0

Notes:

1. See Table 3.2-5b for load combination nomenclature.

2. See Table 3.2-8 for allowable stress criteria. Material properties were obtained from Table 8.1-2 at a design temperature of 400°F.



WHERE:

R	=	33.625 in., DSC outer radius
θ	=	30°
X	=	R Sin θ = 16.8 in
Y	=	R Cos θ = 29.1 in
F_{V1}	=	W = 72,000 lb., weight of DSC
F_{V2}	=	0.27g = 19,400 lb., upward vertical seismic load
F_H	=	0.40g = 28,800 lb., horizontal seismic load

Figure 8.2-1

DSC LIFT-OFF MODEL

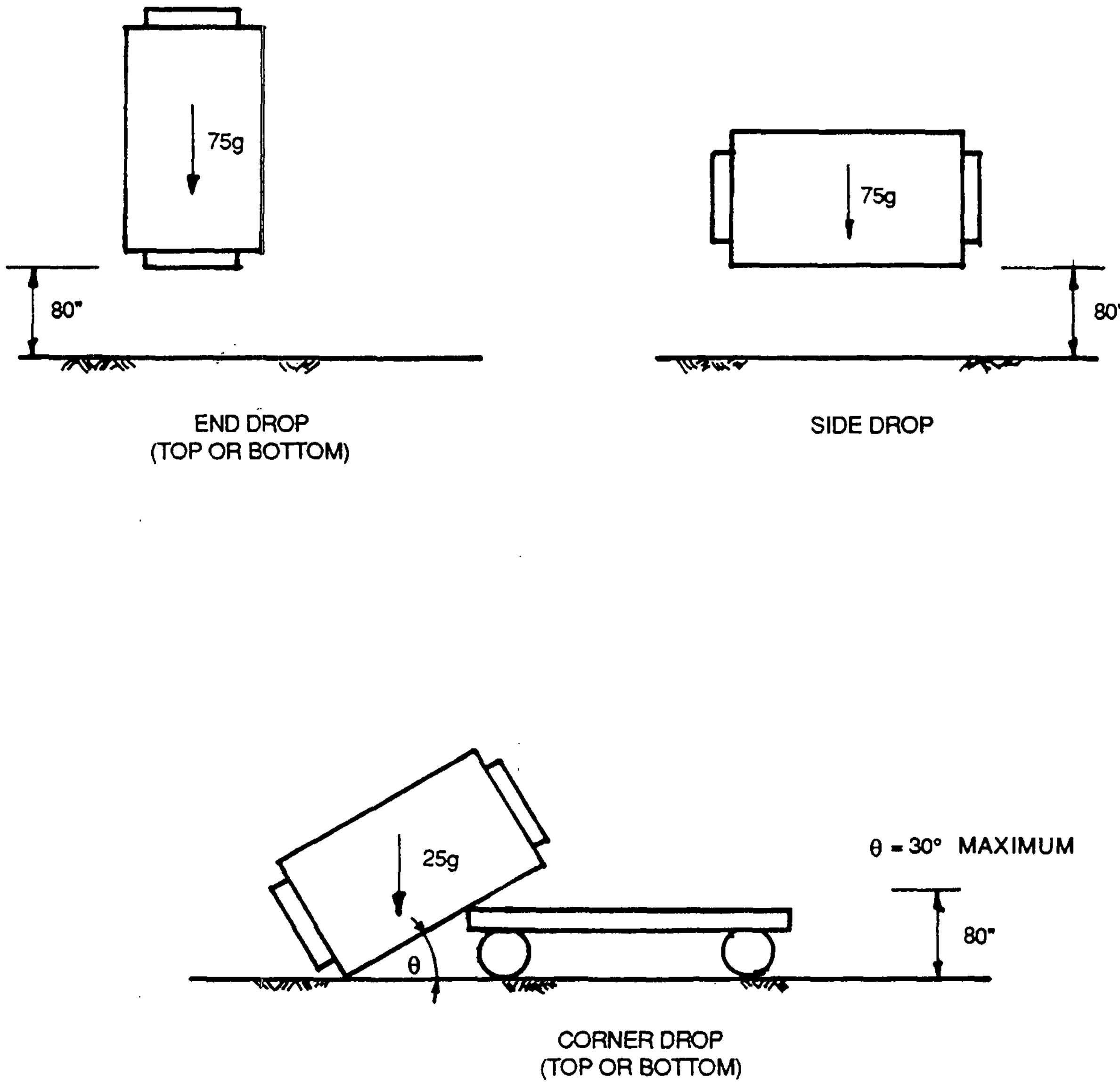


Figure 8.2-2

NUHOMS-24P TRANSFER CASK POSTULATED

DROP ACCIDENT SCENARIOS

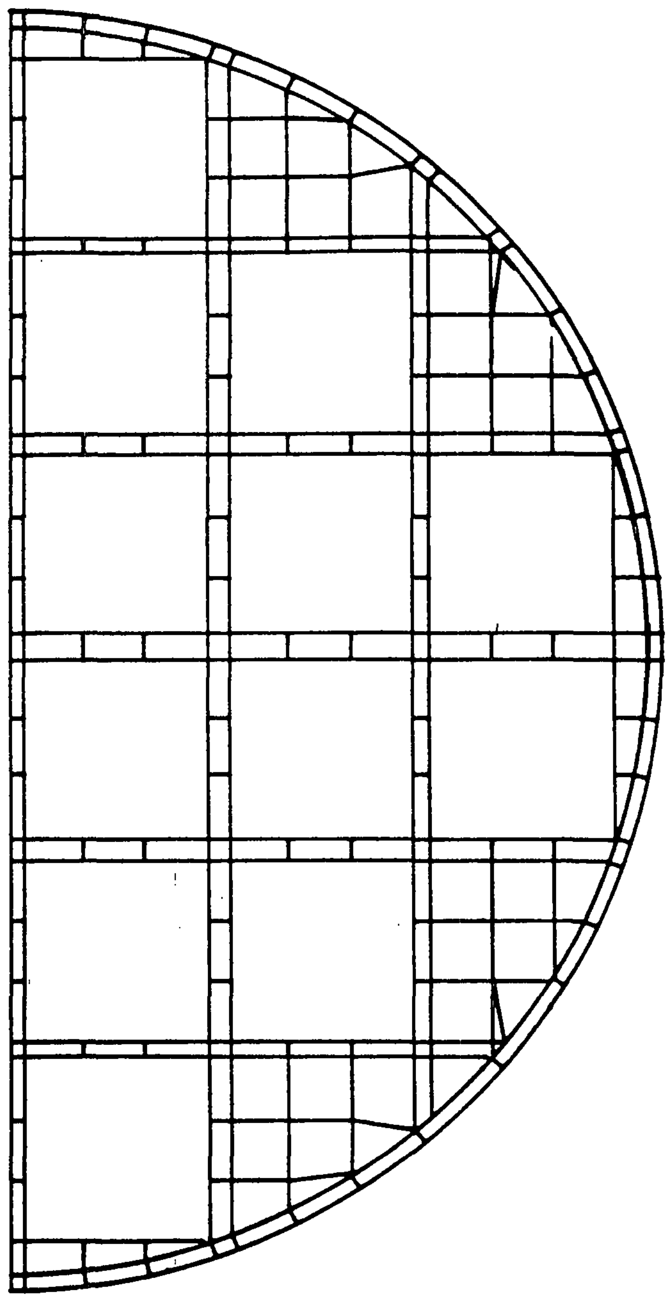


Figure 8.2-3

DSC HALF SPACER DISK MODEL

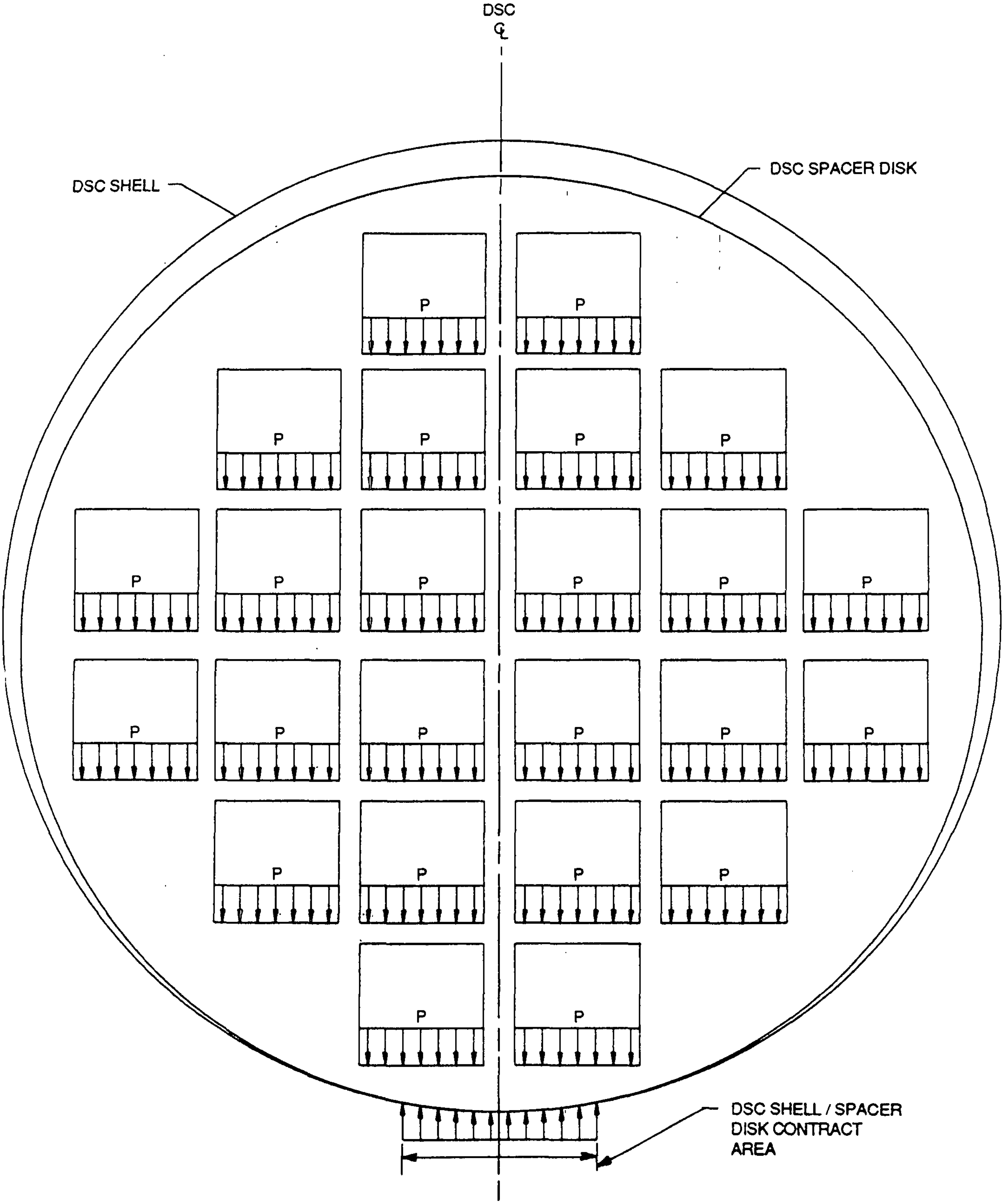


Figure 8.2-4

DSC SPACER DISK LOADING FOR HORIZONTAL DROP ACCIDENT

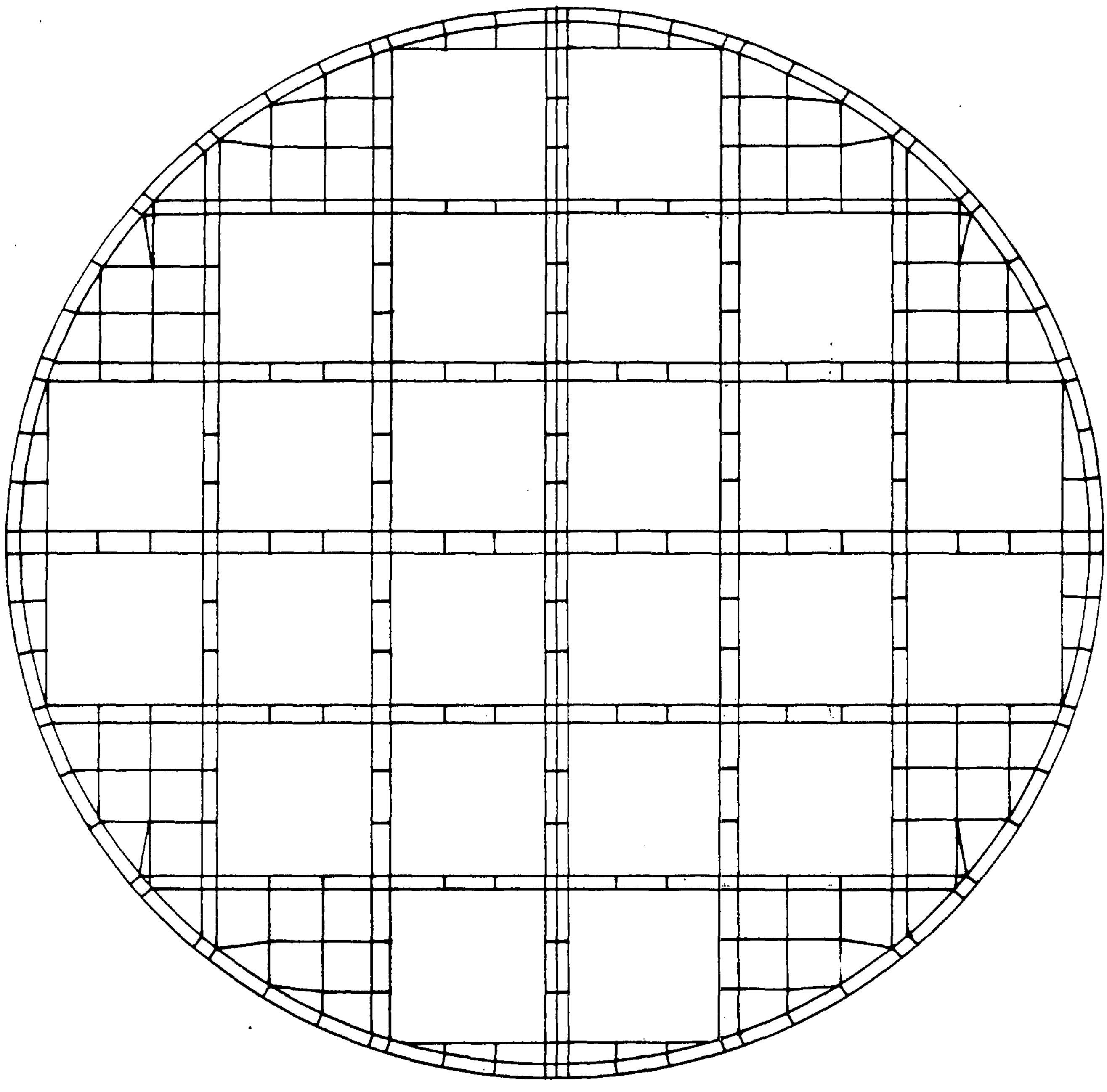


Figure 8.2-5

DSC SPACER DISK DROP ACCIDENT MODEL

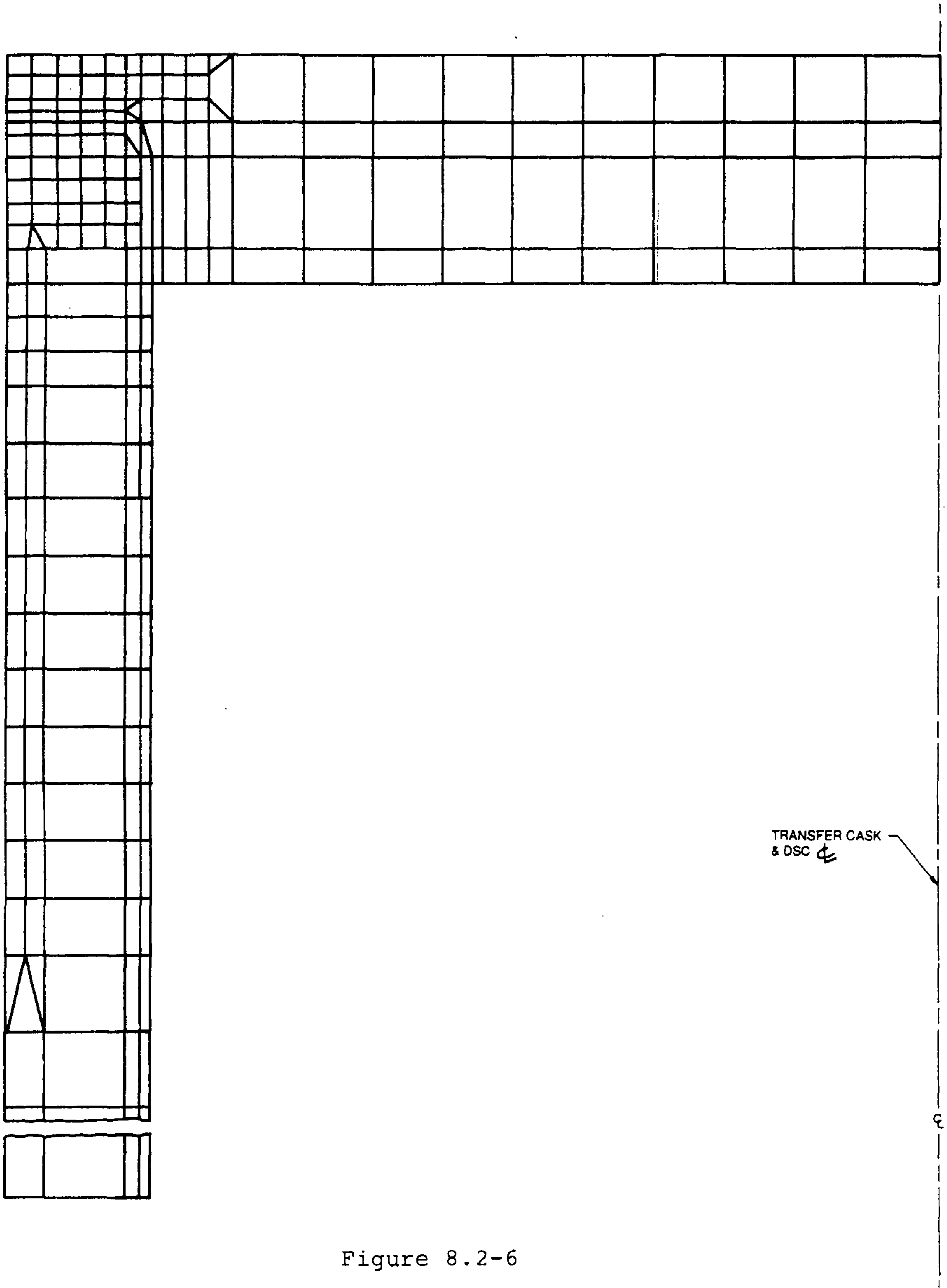


Figure 8.2-6

NUHOMS-24P TRANSFER CASK AND DSC TOP DROP MODEL

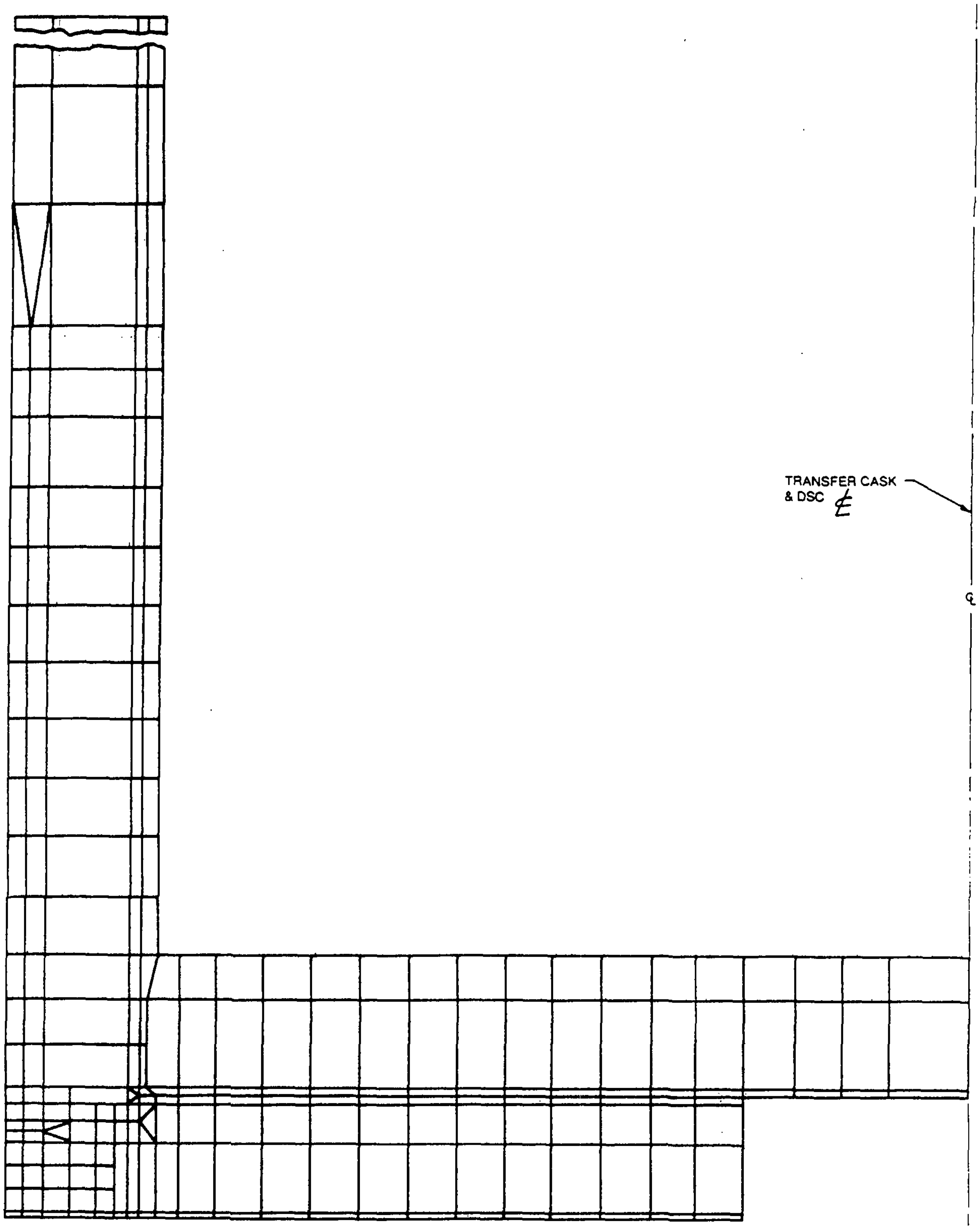


Figure 8.2-7

NUHOMS-24P TRANSFER CASK AND DSC BOTTOM DROP MODEL

Figures 8.2-8, 8.2-9, 8.2-10, and 8.2-11

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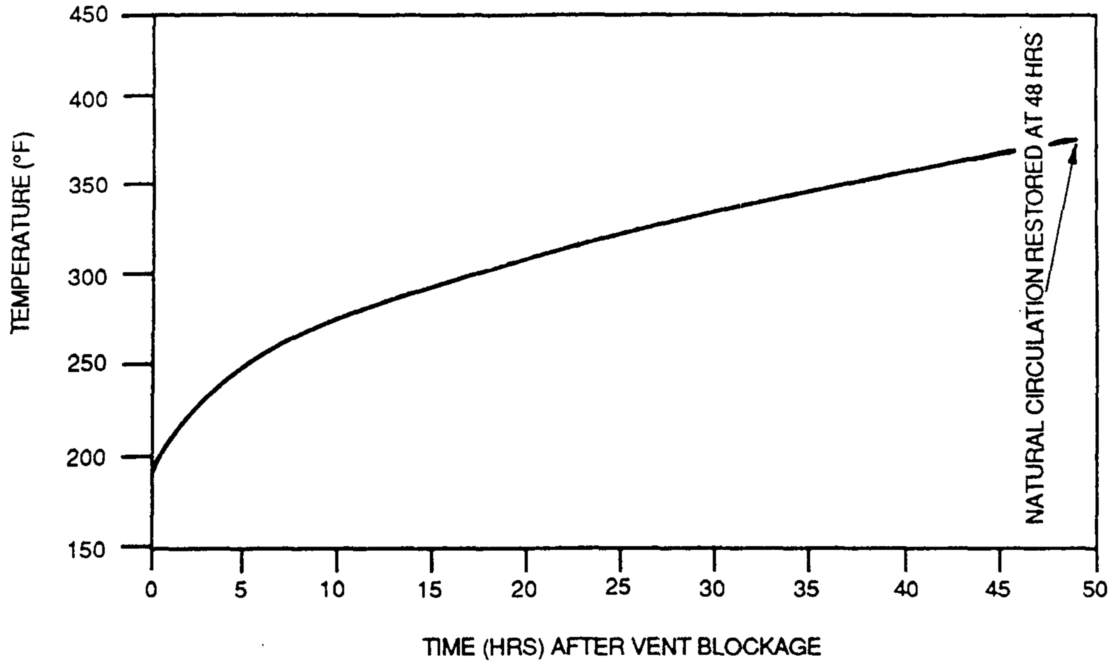

Figure 8.2-12

HSM INTERNAL CONCRETE TEMPERATURES

FOLLOWING VENT BLOCKAGE

Figure 8.2-13 and Figure 8.2-14

DELETED

8.3 Site Characteristics Affecting Safety Analysis

This information will be provided in site license applications.

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9.0 CONDUCT OF OPERATIONS

9.1 Organizational Structure

Organizational structure should be provided by the entity applying for an ISFSI site license.

9.2 Pre-operational Testing and Operation

An intensive pre-operational testing program is being carried out at Carolina Power and Light's Robinson plant. This program is site specific for the NUHOMS-07P system and has been developed jointly by Carolina Power and Light, the Department of Energy, the Electric Power Research Institute and NUTECH.

The pre-operational testing being performed at the H. B. Robinson plant for the NUHOMS-07P system will provide sufficient data to demonstrate that the analytical methods described in this report provide conservative thermal and radiological results. Therefore, the only pre-operational testing required for the NUHOMS-24P is to verify the adequacy of the alignment, handling systems, and procedures for the transfer equipment as noted in Table 9.2-1.

Table 9.2-1

SUMMARY OF ANTICIPATED PRE-OPERATIONAL TESTS

Components Under Test	Features Tested	Method of Test	Purpose of Test
Skid/Trailer/ Cask/HSM/Ram	Alignment	Cold Test*	Verify Alignment Capability
Skid/Trailer/ Cask/HSM/Ram/ DSC	DSC Insertion and Retrieval	Cold Test*	Verify Ease of Transfer and Determine Maximum Tolerable Misalignment

*Loaded with non-radioactive material.

9.3 Training Programs

Training program plans should be submitted in a site specific license application.

9.4 Normal Operations

9.4.1 Procedures

Detailed written procedures which indicate the utility's commitment to safe operation of a NUHOMS system is specific to the utility and should be included in the utility's application for an ISFSI license.

9.4.2 Records

The management system for maintaining records is site specific and should be included in the utility's application for an ISFSI license.

9.5 Emergency Planning

Emergency plans concerning a NUHOMS installation are site specific and should be submitted in an application for an ISFSI license.

9.6 Decommissioning Plan

The decommissioning plans for an ISFSI depend upon the size of the facility, the projected length of time fuel is to be stored, and various site characteristics. The NUHOMS system is a dry containment system that effectively confines all contamination within the DSC. When the DSC is removed from the HSM, the HSM can be manually decontaminated of trace activity, if necessary, and removed using commercial demolition methods.

Removal of fuel assemblies from the DSC can be done in the site's spent fuel pool, as described in Section 5, or the DSC could also be qualified for off-site shipment in a suitable transportation cask licensed to 10CFR71. If such transport is made, the DSC could be disposed of as-is at the final spent fuel repository. If the DSC is not compatible with the final repository handling systems, fuel transfer to a suitable container can be performed in any suitable large hot cell or off-site fuel pool.

The cost of decommissioning depends on the method selected and should be estimated by the applicant for a site license.

10.0 OPERATING CONTROLS AND LIMITS

10.1 Proposed Operating Controls and Limits

The NUHOMS system is totally passive during long term storage and requires very few operating controls. The controls that are necessary for the system are for fuel selection. The general areas where controls and limits are necessary for safe operation of the NUHOMS system are shown in Table 10.1-1. The conditions and other items to be controlled were selected based on the safety assessments for normal, off-normal, and accident conditions provided in Section 8.

This topical report provides specifications in the areas of fuel characteristics; DSC drying, backfill pressure, sealing, and transfer; HSM design and surveillance; and the NUHOMS-24P transfer cask design, operation and surveillance. Site specific specifications in these areas and for DSC fuel loading, cask handling, trailer towing, administrative controls and training will be required in the site license applications.

Table 10.1-1

GENERAL AREAS WHERE CONTROLS AND LIMITS ARE NECESSARY

AREAS FOR OPERATING CONTROLS AND LIMITS	CONDITIONS OR OTHER ITEMS TO BE CONTROLLED
1. Fuel Characteristics	Burnup/Initial Enrichment Decay Heat Power Gamma Source Strength Neutron Source Strength
2. On-site Transfer Cask	Surface Dose Rates Transfer Route Selection
3. Dry Shielded Canister	Dye Penetrant Test of Closure Welds Surface Dose Rate of Lead Shield Plug
3.1 Drying	Vacuum Pressure
3.2 Backfilling	Helium Pressure and Content Helium Leakage
4. Trailer Towing	- Site Specific -
5. Alignment	- Site Specific -
6. DSC in HSM	Surface Dose Rates Air Inlets Air Outlets Center of Door Roof Side Walls Maximum Air Exit Temperatures
7. Surveillance	Inspection of HSM Air Inlets and Outlets
8. Administrative Controls	- Site Specific -
9. Training	- Site Specific -

10.2 Development of Operating Controls and Limits

This section provides an overview and general bases for the operating controls and limits specified in this topical report for the NUHOMS-24P system. These specifications cover the generic issues associated with the operation of a NUHOMS installation to ensure the protection of the public's health and safety. Section 10.3 provides a full description and discussion of these specifications. Any additional site specific operating controls and limits will be supplied by site license applicants.

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. However, all the monitoring instruments and limiting control settings for the NUHOMS system are associated with the handling operations and are site specific. The only remaining generic functional limit is that of defining the types of fuel to be stored. Other operational limits are discussed in Section 10.2.2.2, Technical Conditions and Characteristics.

The functional limits for fuel to be stored in the NUHOMS system is provided in Section 10.3.1.

10.2.2 Limiting Conditions for Operation

10.2.2.1 Equipment Limiting conditions for operation of equipment, systems, and components (in terms of lowest acceptable level of performance, minimum number of components available, etc.) will be concerned with operating systems (i.e., transfer cask, cranes, hydraulic rams, trailers, etc.). As stated previously, these systems will be site specific.

During storage, only the DSC and HSM are important to safety. The controlling limits for their operational condition are discussed in the next section.

10.2.2.2 Technical Conditions and Characteristics The following technical conditions and characteristics are required for the NUHOMS system:

1. DSC Vacuum Pressure During Drying
2. DSC Helium Backfill Pressure
3. DSC Helium Leakage Rate of Closure Weld

4. DSC Dye Penetrant Test of Closure Welds
5. Dose Rate at End of DSC Lead Shield Plug
6. Surface Dose Rates on HSM while DSC is in Storage
7. Maximum Air Exit Temperature for HSM

A description of the bases for selecting the above conditions and characteristics are described in the Bases Sections for each of the individual specifications. However, the overall technical and operational considerations are to:

1. Assure proper internal DSC atmosphere to promote heat transfer, minimize the occurrence of uranium dioxide oxidation, and minimize the likelihood of the uncontrollable release of radioactive material (Items 1-4 above).

2. Assure as-low-as-reasonably-achievable dose rates in areas where operators must work (Items 5 and 6 above)

3. Assure the long term average fuel clad temperature is maintained at or below 340°C (644°F) during normal storage operation (Item 7).

Through the analysis and evaluation provided in Section 8, this topical report has shown that if the above seven technical conditions and characteristics are met, no significant public or occupational health and safety hazards will exist.

10.2.3 Surveillance Requirements

Analysis has shown that the DSC and the HSM can fulfill their safety functions during all normal and off-normal operating conditions and short term accident conditions as described in Section 8. The surveillance required during spent fuel and DSC loading and handling have been summarized in this report and should be described in detail in other plant specific specifications. The only other surveillance required during long-term storage is the periodic checking (once every 24 hours) of the HSM air inlets and outlets to insure they are clear of obstructions.

10.2.4 Design Features

The following design features are important to the safe operation of the NUHOMS system and require design controls and limits:

DSC

1. Material Specification for Structural Integrity
2. Dimensional Control for Subcriticality

HSM

1. Material Specification for Structural Integrity and Shielding
2. Air Inlet and Outlet Openings sized for Fuel Cladding Integrity

On-Site Transfer Cask

1. Surface Dose Rates
2. Transfer Route Selection
3. Wind Speed During Transfer Operations

Component dimensions are not specified here because the combination of materials, dose rates, criticality control, and system fit-up define the operable limits for dimension (i.e., thickness of lead, thickness of concrete, DSC pressure plates, body thicknesses, etc.). The values for these design parameters were described in previous sections.

The combination of the above controls and limits and those discussed in the previous subsections of Section 10.2 define requirements for the NUHOMS system components that provide radiological protection and structural integrity during normal storage and postulated accident conditions.

10.2.5 Administrative Controls

Site specific license applications will contain a full description and discussion of the administrative systems and procedures, recordkeeping, review, audit, and reporting practices necessary to ensure that the operation of a NUHOMS installation is performed in a safe manner.

10.3 Operational Control and Limit Specification

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

The following specifications are contained in this section:

10.3.1.1 Spent Fuel Specifications

10.3.1.1 Spent Fuel Specifications

1.	Title:	Fuel Specifications	
2.	Specifications:	Type	PWR fuels
		Fuel Cladding	Zircaloy-clad fuel with no known or suspected cladding damage.
		Burn-up/Initial Enrichment*	Burnup and Initial Enrichment for a fuel assembly must be within the acceptable region specified in Figure 10.3-1.
		Decay Heat Per* Assembly	≤ 0.66 kw
		Neutron Source* Per Assembly	≤1.548E8 n/sec with spectrum bounded by that in Section 7.
		Gamma Source* Per Assembly	≤4.62E15 photon/sec with spectrum bounded by that in Section 7.
3.	Applicability:	This specification is applicable to all spent fuel to be stored in the NUHOMS-24P system.	
4.	Objective:	This specification was derived to insure that the peak fuel rod temperatures, surface doses, and nuclear subcriticality are below the design values.	
5.	Action:	If this specification is not met, additional analysis and/or data must be	

* Any fuel not specifically meeting the above requirements may be stored in the NUHOMS system provided the safe storage criteria presented in Section 3 are satisfied.

presented demonstrating that the nonconformance does not exceed other safe operating limits before the spent fuel can be placed in the DSC for storage.

6. Surveillance: The fuel assembly parameters specified above must be verified prior to fuel loading. No other additional surveillance is required.

7. Bases: The fuel parameters specified above were selected to bound representative PWR fuels. The NUHOMS-24P system was designed to provide adequate radiological and structural margins for safe operation and response to accident conditions based on these fuel design parameters.

10.3.2 Limiting Conditions for Operation

The following specifications are contained in this section:

10.3.2.1 DSC Vacuum Pressure During Drying

1.	Title:	DSC Vacuum Pressure During Drying
2.	Specification:	Vacuum Pressure: 3 Torr Time at Pressure: Not less than 30 minutes following stepped evacuation (see Bases).
3.	Applicability:	Applicable to all DSCs.
4.	Objective:	To minimize moisture content.
5.	Action:	If the required vacuum pressure cannot be obtained: a. Check and repair the vacuum tubing, hoses, and fittings. b. Check and repair or replace the vacuum pump. c. Check and repair the fillet weld between the upper lead shield plug assembly and the DSC body. If the vacuum line appears to be ice blocked (see Surveillance), wait one hour before restarting pump.
6.	Surveillance:	No maintenance or tests are required. Surveillance of the vacuum gauge is required during vacuum drying operation to verify that the vacuum line is not blocked by ice. The pressure at each plateau should rise after the pump is valved off, indicating that the vacuum indicated on the gauge exists in the DSC. Failure to rise may indicate that the line is blocked by ice.
7.	Bases:	A stable vacuum pressure of less than three torr indicates that all liquid water has evaporated in the DSC cavity, and that the resulting inventory of oxidizing gasses in the DSC is less than 0.25% vol.%. A stepped pumpdown is specified to prevent ice formation in the

DSC and the vacuum lines. The stepped pumpdown procedure is based on experience with similar fuel storage baskets. The DSC pressure will be reduced in stages to 100 torr, 50 torr, 25 torr, 15 torr, 10 torr, five torr, and three torr. Some steps may need to be repeated, based on observation of the system pressure.

This limit is a minimum requirement to ensure that oxidizing gas levels in the DSC do not exceed prudent values. Note that the operating procedures discussed in Chapter 5 specify a subsequent dilution with dry helium and second evacuation to three torr to ensure purity of the cover gas.

10.3.2.2 DSC Helium Backfill Pressure

1.	Title:	DSC Helium Backfill Pressure
2.	Specifications:	Helium 2.5 psig ± 2.5 psig backfill pressure
3.	Applicability:	This specification is applicable to all DSCs.
4.	Objective:	To assure that (1) the atmosphere surrounding the spent fuel is a nonoxidizing inert gas; (2) the atmosphere is favorable for the dissipation of decay heat; (3) the DSC does not become overpressurized.
5.	Action:	If the required pressure cannot be obtained: a. Check and repair or replace the pressure gauge. b. Check and repair or replace the pressure tubes, connections, and valves. c. Check and repair or replace helium source. d. Check and repair the fillet weld on upper lead plug. If pressure exceeds the criterion, release a sufficient quantity of helium to the site radioactive waste system, to lower the DSC cavity pressure.
6.	Surveillance:	No maintenance or tests are required during the normal storage operations. Surveillance of the pressure gauge is required during the helium backfilling operation. Alternate methods such as filling the DSC with a measured quantity of helium, 850 gm ± 120 gm, are acceptable. Exact methods for verification of the quantity of helium backfill will be site specified.
7.	Bases:	The value of 2.5 psig was selected to assure that the pressure within the DSC is within the design limits during any expected normal storage operating condition.

10.3.2.3 DSC Helium Leakage Rate of Seal Weld

1.	Title:	DSC Helium Leakage Rate of Primary Seal Weld
2.	Specification:	Leakage rate of primary weld 10^{-4} atm cc/sec.
3.	Applicability:	This is applicable to all DSCs.
4.	Objective:	To reduce the chance of radioactive gas release and assure that the atmosphere surrounding the fuel assemblies is an inert gas.
5.	Action:	If leakage rate test of primary weld exceeds 10^{-4} atm - cc/sec: a. Check and repair the weld. b. Check end-plug assembly for any cracks.
6.	Surveillance:	No maintenance or tests are required during normal storage operations. After welding operation has been completed, perform a leak test with a helium sniffer or alternate acceptable method.
7.	Bases:	This is the lowest rate measurable for use with portable helium leak detectors. If a pressure of 1.5 atm developed within the DSC cavity for a period of 10 years, a leak rate of 10^{-4} atm -cc/sec. would allow 47,300 cm^3 of helium to escape. This would be insignificant compared to the 6.75E6 cm^3 of helium in the DSC initially.

10.3.2.4 DSC Dye Penetrant Test of Closure Welds*

1. Title: DSC Dye Penetrant Test of Closure Weld

2. Specification: The acceptance standards for liquid penetrant examination contained in the ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsection NB-5350 (1983) Liquid Penetrant Acceptance Standards or equivalent shall apply.

   Evaluation of Indications:

   a. Mechanical discontinuities at the surface shall be indicated by bleeding out of the penetrant. However, localized surface imperfections such as may occur from machining marks, surface conditions or an incomplete bond between base metal and cladding may produce similar indications which are nonrelevant to the detection of unacceptable discontinuities.

   b. Any indication which is believed to be nonrelevant shall be regarded as a defect and shall be re-examined to verify whether or not actual defects are present. Surface conditioning may precede the re-examination. Nonrelevant indications and broad areas of pigmentation which would mask indications of defects are unacceptable.

   c. Relevant indications are those which result from mechanical discontinuities. Linear indications are those indications in which the length is more than three times the width. Rounded indications are indications which are circular or elliptical with the length less than three times the width.

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* Alternate inspection techniques such as ultrasonic examination may be specified.

Acceptance Standards:

a. Only indications with major dimensions greater than 1/16 inch (1.6 mm) shall be considered relevant.

b. Unless otherwise specified in this Subsection, the following relevant indications are unacceptable:

   1. Any cracks or linear indications.

   2. Rounded indications with dimensions greater than 3/16 inch (4.8 mm).

   3. Four or more rounded indications in a line separated by 1/16 inch (1.6 mm) or less edge-to-edge.

   4. Ten or more rounded indications in any 6 square inches (3870 mm^2) of surface with the major dimension of this area not to exceed six inches (152 mm) with the area taken in the most favorable location relative to the indications being evaluated.

3. Applicability: This is applicable to all DSCs.

4. Objective: To ensure that the DSC is adequately sealed and to ensure that an uncontrolled release from the DSC is prevented.

5. 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 specification.

6. Surveillance: No additional surveillance is required

7. Bases: Article NB-5000 Examination
ASME Boiler and Pressure Vessel Code
Section III - Division I
Subsection NB (1983)

10.3.2.5 Dose Rate at End of DSC Lead Shield Plug

1. Title: Dose Rate at End of DSC Lead Shield Plug

2. Specification: Dose rates at the following locations:

   Center of Lead Shield Plug 100 mrem/hr. with Water in Cavity of DSC
   Center of DSC top cover plate 200 mrem/hr. with Temporary Shielding In-place

3. Applicability: This specification is applicable to all DSCs.

4. Objective: To maintain as-low-as-reasonably-achievable dose rates in the areas where DSC welding and drying operations must be performed.

5. Action: If the specified dose rates are exceeded the following actions should be taken:

   a. Visually inspect placement of the lead shield plug. Re-install or adjust position of lead shield plug.

   b. Install temporary shielding if necessary.

   c. Review spent fuel inventory to ensure that assemblies placed within the DSC conform with technical specifications contained in 10.3.1.1.

6. Surveillance: The dose rates must be measured prior to seal welding lead shield plug to the DSC shell and welding the top coverplate to the DSC shell.

7. Bases: The dose rates selected for this specification were chosen to provide as-low-as-reasonably-achievable doses to the worker who must perform the drying and welding operations. Dose rates in the range of 100 to 200 mrem/hour are typical of the dose rates on contaminated equipment, low-level waste drums, and other items which are contact handled, operated, and maintained. Additionally, the analyses provided in Section 7 show that the actual dose rates are lower than those specified above.

10.3.2.6 Surface Dose Rates of the HSM While the DSC is in Storage

1.	Title:	Surface Dose Rates of the HSM While the DSC is in Storage
2.	Specification:	Surface dose rates at the following locations shall be less than:

a.	Outside of HSM door on centerline of DSC	100 mrem/hr.
b.	Center of air inlets	200 mrem/hr.
c.	Center of air outlet shielding cap	125 mrem/hr.
d.	Exterior Side Walls	20 mrem/hr.

3.	Applicability:	This specification is applicable to initially loaded HSM and DSC.
4.	Objective:	The objective of this specification is to maintain as-low-as-reasonably-achievable dose rates at locations on the HSMs where surveillance is needed, and to reduce off-site exposures to as-low-as-reasonably achievable
5.	Action:	If the dose rates are exceeded, the DSC must be removed or additional shielding must be installed to reduce the dose rates to the specified levels. If additional shielding is used the outlet air temperature must be measured after the shielding is located to verify that the air flow has not been hindered.
6.	Surveillance:	The HSM, initially loaded, shall be checked to verify that this specification has been met after the DSC is placed in storage and the HSM door closed. Standard industry surveillance techniques should be used.

7. Bases:

The dose rates stated in this specification were selected to maintain as-low-as-reasonably-achievable exposures off-site and to personnel verifying the air vent opening on the HSM. As stated in Specification 10.3.2.5, these dose rates are within industry accepted standards for contact handling, operation and maintenance of radioactive material. Maintenance personnel will have to use their hands and hand held tools to remove any potential debris from air vent openings. At 200 mrem/hour the dose for a one hour job of unblocking the air inlets (or outlets) would be less than 200 mrem (whole body) and hence would be only 4% of the total yearly burden. Furthermore, analysis provided in Section 7 showed that the actual dose rates around the HSM surface all be well below the listed specified above.

10.3.2.7 Maximum Air Exit Temperature

1.	Title:	Maximum Air Exit Temperature
2.	Specification:	Maximum air temperature rise 60°F.
3.	Applicability:	This specification is applicable to all HSMs.
4.	Objective:	To ensure that HSM air inlets and outlets are unobstructed so that the integrity of fuel cladding is maintained.
5.	Action:	If the temperature rise is greater than 60°F, the air inlets and exits should be checked for blockage. If the blockage is cleared and the temperature is still too great, the DSC may be removed from the HSM or additional information and analysis provided that will prove the existing condition does not represent an unsafe condition.
6.	Surveillance:	The temperature rise shall be measured 24 hours after the DSC is placed into the HSM. If the temperature rise is within the specifications, then the HSM and DSC are performing as designed and no further temperature measurements are required.
7.	Bases:	The 60°F temperature rise was selected to ensure the fuel clad temperatures are maintained at or below acceptable storage limits.

10.3.2.8 Alignment of Transfer Cask and HSM for the DSC Transfer Operation*

1.	Title:	Alignment of Transfer Cask and HSM for the DSC Transfer Operation
2.	Specification:	The cask must be aligned with respect to the HSM so that the longitudinal centerline of the DSC in the cask is within ± 0.125 inch of its true position when the cask is within the HSM front access sleeve.
3.	Applicability:	DSC transfer operation
4.	Objective:	To ensure smooth transfer of the DSC from the cask to HSM and back.
5.	Action:	The transfer operation must not proceed unless this specification is met.
6.	Surveillance:	The alignment procedures are site specific. Alignment techniques must be amenable to independent checking and verification.
7.	Bases:	The basis for the true position alignment tolerance is the clearance between the DSC and the transfer corridor, which consists of the cask cavity, the HSM entrance, and the DSC support assembly in the HSM.

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* Note: This specification is provided only as an example. The exact specification depends on the transfer cask selected and the clearance between the DSC and the transfer cask. The specification described above is for the NUHOMS-24P transfer cask.

10.3.2.9 Fuel Assembly Retrieval and Inspection

1. Title: Fuel Assembly Retrieval and Inspection

2. Specification: Fuel assemblies must be retrieved and inspected for damage subsequent to any cask drop of fifteen inches or greater.

3. Applicability: This is applicable to all loaded DSCs.

4. Objective: To assure integrity of the fuel assemblies following moderate or severe drop accidents.

5. Action: If the loaded DSC is dropped, and examination suggests that the drop height is fifteen inches or greater through air, then the DSC, transfer cask or fuel may have been damaged and an examination of these items should be conducted.

6. Surveillance: Not applicable.

7. Bases: The height chosen for this specification was selected to provide reasonable assurance that damage to the cask and DSC does not occur. Although analyses performed for drop accidents at various orientations indicate much greater resistance to damage, this specification's intent is to provide guidance in the event of drop accidents where there is little or no visible damage to the cask or DSC.

10.3.3 Surveillance Requirements

The following specifications are provided in this section:

10.3.3.1 Surveillance of the HSM Air Inlets and Outlets

10.3.3.1 Surveillance of the HSM Air Inlets and Outlets

1.	Title:	Surveillance of the HSM Air Inlets and Outlets
2.	Specifications:	Normal visual inspection: Once every 24 hours
3.	Applicability:	Every HSM
4.	Objective:	To assure that HSM air inlets and outlets are not blocked for more than 24 hours and to assure that complete blockage of all inlets and outlets does not occur for periods exceeding 48 hours.
5.	Action:	If the air inlets and outlets are plugged they should be cleared by hand or hand held tools. If the screen is damaged it should be replaced.
6.	Surveillance:	This is a surveillance specification.
7.	Bases:	Analyses presented in Section 8 demonstrated that the ability of the system to safely function is not jeopardized if obstructions in the air inlets or outlets impairs air flow through the HSM for periods of less than 48 hours.

10.3.4 Limiting and Operating Conditions for Transfer Cask Containing Fuel

The following specifications are provided in the section.

10.3.4.1 Maximum Surface Dose Rate on Transfer Cask

10.3.4.2 Transfer Route Selection

10.3.4.1 Maximum Surface Dose Rate on Transfer Cask

1.	Title:	Maximum Surface Dose Rate on Transfer Cask
2.	Specification:	Surface dose rates at the following locations Transfer cask lid — 250 mrem/hr. Body of transfer cask with neutron shield filled with water — 250 mrem/hr. Bottom of transfer cask with bottom cover plate installed — 250 mrem/hr.
3.	Applicability:	This specification is applicable to the NUHOMS-24P transfer cask
4.	Objective:	To maintain as-low-as-reasonably-achievable dose rate during DSC transfer operations.
5.	Action:	If the specified dose rates are exceeded, place temporary shielding around transfer cask and review the plant records of the fuel assemblies which have been placed into the DSC to insure that they conform to the requirements of Section 10.3.1.1:
6.	Surveillance:	The dose rates must be measured as soon as possible after the transfer cask is removed from the spent fuel pool.
7.	Bases:	The dose rates selected for this operation were chosen to provide as-low-as-reasonably-achievable doses to personnel during transfer operations.

10.3.4.2 Transfer Route Selection

1.	Title:	Transfer Route Selection
2.	Specification:	The surface within a eight foot proximity of the transfer trailer roadway shall be at the same elevation.
3.	Applicability:	This specification is applicable to DSC transfer utilizing the NUHOMS-24P transfer cask.
4.	Objective:	Ensure that a potential drop height of 80 inches is not exceeded.
5.	Action:	Prior to the initial transfer of fuel to the HSM, the proposed transfer route shall be visually inspected to ensure that eight feet on either side of the transfer trailer is at the same elevation or higher as the roadway.
6.	Surveillance	No surveillance required.
7.	Bases:	Analysis presented in Section 8.2.5 demonstrates that a drop from a height less than 80 inches does not jeopardize the ability of the transfer cask or DSC to safely function.

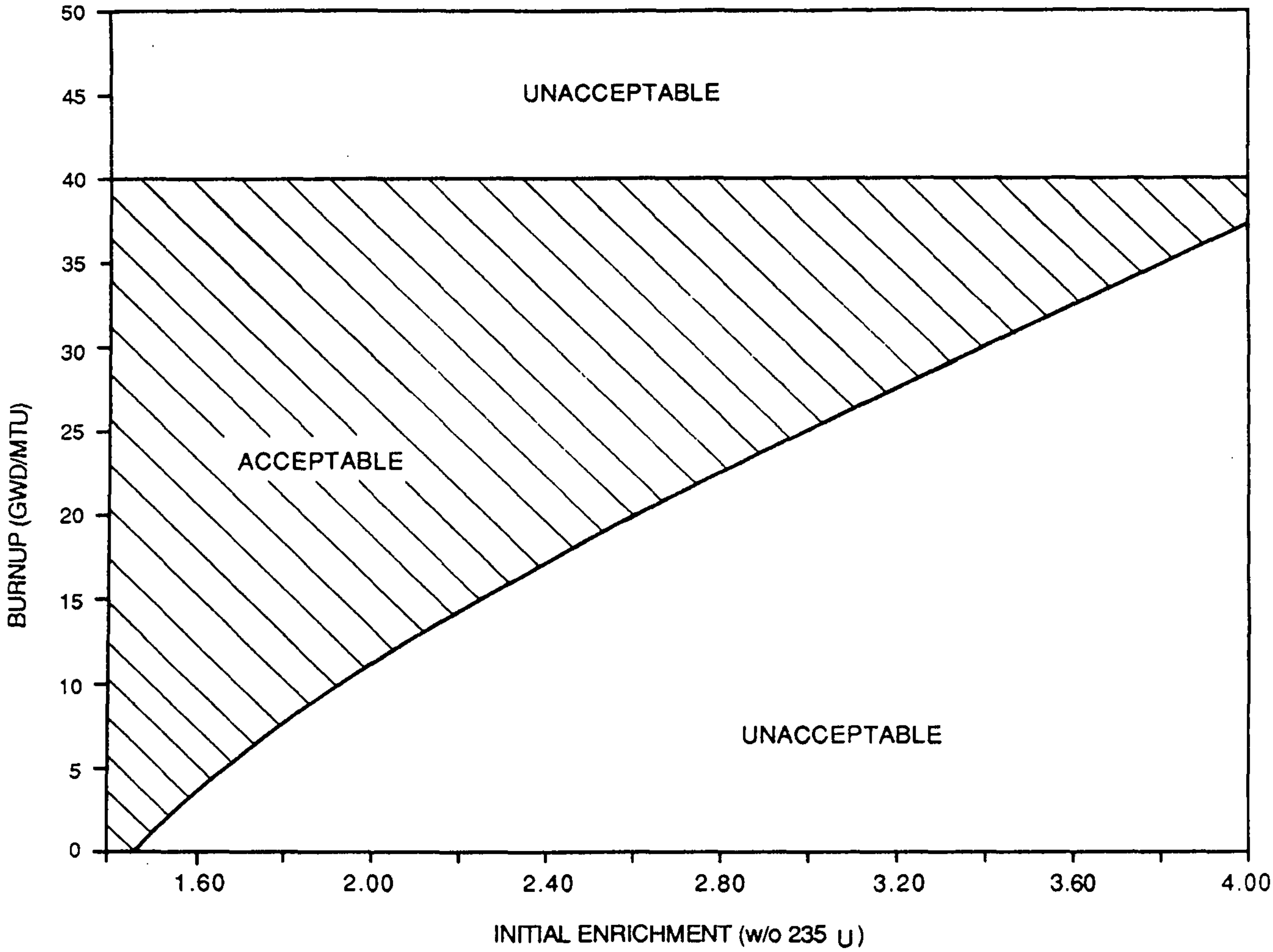

Figure 10.3-1

FUEL ACCEPTANCE CRITERIA

11.0 Quality Assurance

The Quality Assurance Program to be applied to the quality-related activities associated with the NUHOMS®-24P design is the same as that previously described in Section 11 of the NUHOMS-07P Topical Report (NUH-001, Revision 2A).

ATTACHMENT (12)

TRANSNUCLEAR and PACIFIC NUCLEAR

PROPRIETARY AFFIDAVITS

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

Transnuclear, Inc.)
State of Maryland) SS.
County of Howard)

I, Jayant Bondre, depose and say that I am a Vice President of Transnuclear, Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in Enclosure 1 and as listed below:

1. TN Calculation 1095-6, "NUHOMS-32P Transfer Thermal Analysis 103°F Ambient", Revision 1
2. TN Calculation 1095-16, "Transfer Thermal Analysis -3°F Ambient", Revision 0
3. TN Calculation 1095-35, "NUHOMS 32P- Transfer Cask Structural Analysis" Revision 2

These documents have been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Transnuclear, Inc. in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure are portions of certain NUHOMS 32P dry storage canister structural and thermal analyses which are owned and have been held in confidence by Transnuclear, Inc.

2) The information is of a type customarily held in confidence by Transnuclear, Inc. and not customarily disclosed to the public. Transnuclear, Inc. has a rational basis for determining the types of information customarily held in confidence by it.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear, Inc. because the information consists of descriptions of the design and analysis of dry spent fuel storage systems, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Transnuclear, Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear, Inc.'s product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

Jayant Bondre
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this 18th day of February, 2011.

Notary Public

My Commission Expires ___/___

Lauren McKee
NOTARY PUBLIC
Anne Arundel County, Maryland
My Commission Expires 2/12/2015

LAUREN MCKEE
NOTARY
PUBLIC
ANNE ARUNDEL COUNTY, MD

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

Transnuclear, Inc.)
State of Maryland) SS.
County of Howard)

I, Jayant Bondre, depose and say that I am a Vice President of Transnuclear, Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in Enclosure 1 and as listed below:

1. TN Calculation 1095-49, "NUHOMS-32P – Radiation Dose Rates for Loading and Transfer" Revision 0

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Transnuclear, Inc. in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure are portions of certain NUHOMS 32P dry storage canister radiation dose rate analyses which are owned and have been held in confidence by Transnuclear, Inc.

2) The information is of a type customarily held in confidence by Transnuclear, Inc. and not customarily disclosed to the public. Transnuclear, Inc. has a rational basis for determining the types of information customarily held in confidence by it.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear, Inc. because the information consists of descriptions of the design and analysis of dry spent fuel storage systems, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Transnuclear, Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear, Inc.'s product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

Jayant Bondre
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this ___ day of March, 2011.

ZOIA M. DOUGHERTY
COMMISSION EXPIRES
OCT
17
200_
HOWARD COUNTY, MD
NOTARY PUBLIC

Notary Public

My Commission Expires ________

AFFIDAVIT PURSUANT
TO 10 CFR 2.790

Pacific Nuclear Fuel Services, Inc.)
State of Washington) SS.
County of King)

I, William J. McConaghy depose and say that I am a Vice President of Pacific Nuclear Fuel Services, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

Approved Topical Report for the NUTECH Horizontal Modular Storage (NUHOMS®) System for Irradiated Nuclear Fuel; NUH-002, Revision 2A, Proprietary Information (this information is appropriately identified in Revision 1A of NUH-002).

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Pacific Nuclear Fuel Services in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure are designed drawings and descriptions of the design and analysis of a concrete modular storage system, which are owned and has been held in confidence by Pacific Nuclear Fuel Services.

2) The information is of a type customarily held in confidence by Pacific Nuclear Fuel Services and not customarily disclosed to the public. Pacific Nuclear Fuel Services has a rational basis for determining the types of information customarily held in confidence by it.

3) The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

4) The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

5) Public disclosure of the information is likely to cause substantial harm to the competitive position of Pacific Nuclear Fuel Services because:

a) A similar product is manufactured and sold by competitors of Pacific Nuclear Fuel Services.

b) Development of this information by Pacific Nuclear Fuel Services required thousands of man-hours and hundreds of thousands of dollars. To the best of my knowledge and belief, a competitor would have to undergo similar expense in generating equivalent information.

c) In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of a design and analysis of a dry spent fuel storage system.

d) The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e) The information consists of description of the design and analysis of a dry spent fuel storage system, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Pacific Nuclear Fuel Services, take marketing or other actions to improve their product's position or impair the position of Pacific Nuclear Fuel Services' product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

f) In pricing Pacific Nuclear Fuel Services' products and services, significant research, development, engineering, analytical, licensing, quality assurance and other costs and expenses must be included. The ability of Pacific Nuclear Fuel Services' competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

Further the deponent sayeth not.

William J. McConaghy

William J. McConaghy, P.E.
Vice President
Pacific Nuclear Fuel Services, Inc.

Subscribed and sworn to before me this 15th day of July, 1991.

Judi A. Wilson

Notary Public

OFFICIAL SEAL
JUDI A. WILSON
NOTARY PUBLIC CALIFORNIA
SANTA CLARA COUNTY
My Comm Expires Oct 14 1994