CHAPTER 3

PRINCIPAL DESIGN CRITERIA

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LIST OF ACRONYMS

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
AWS	American Welding Society
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
DSC	Dry Shielded Canister
HSM	Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
MW/MTHM	Megawatt/Metric Tor Heavy Metal
NRC	Nuclear Regulatory Commission
NSR	Non-Safety-Related
NUHOMS	Nutech Horizontal Modular Storage®
QA	Quality Assurance
RG	Regulatory Guide
SAR	Safety Analysis Report
SR	Safety-Related
TR	Topical Report
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report

3.0 PRINCIPAL DESIGN CRITERIA

3.1 PURPOSE OF THE CALVERT CLIFFS INDEPENDENT SPENT FUEL STORAGE INSTALLATION

The purpose of the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) is to provide additional interim spent fuel storage capacity to allow continued operation of the Calvert Cliffs Nuclear Power Plant (CCNPP). The ISFSI utilizes the Nutech Horizontal Modular Storage® (NUHOMS) system described in Reference 3.1. The NUHOMS system is comprised of an array of reinforced concrete horizontal storage modules (HSMs), each of which can house a stainless steel, helium filled dry shielded canister (DSC) containing qualified spent fuel assemblies. The DSC shield plug and cover plate assemblies are independently seal welded to assure long-term confinement of the irradiated fuel. A shielded transfer cask is used to transfer the DSC from the spent fuel pool to the HSM. During storage, the HSM provides radiation shielding and passive decay heat removal from the DSC.

3.1.1 MATERIAL TO BE STORED

Table 3.1-1 lists the principal design parameters for the spent fuel assemblies to be stored in the Calvert Cliffs ISFSI. The following subsections delineate the fuel assemblies' physical, reactivity, thermal, and radiological design criteria for storage in the Calvert Cliffs ISFSI. These design criteria compare with Section 3.1.1 of Reference 3.1 as follows:

A. The neutron and gamma source strengths are higher than those in Table 3.1-1 of Reference 3.1. Additional shielding is added to the transfer cask, the DSC end plugs, and the HSM access door to limit the external contact dose rates to less than the acceptance criteria established in Section 10.3.2.6 of Reference 3.1 and this document.

B. The total decay heat power per DSC is equal to that described in Table 1.2-2 of Reference 3.1 and the Updated Safety Analysis Report (USAR), Section 12.8.1.3.

C. As reported in Sections 3.3.4 and 12.3.3.4, plant specific criticality analyses have been performed.

D. The fuel weight per NUHOMS-24P DSC spacer disk and the total weight of the DSC are less than those used in the analysis of Sections 8.1.1.2 and 8.1.1.3 of Reference 3.1. The on-site transfer cask is heavier than that reported in Table 8.1-3 of Reference 3.1. However, the gross weight of the transfer cask, DSC, and fuel is less than that reported in Table 8.1-3 of Reference 3.1. The cask drop height and drop surface conditions along the transfer route to the ISFSI meet the limits described in Table 8.1-3 of Reference 3.1.

3.1.1.1 Physical Characteristics

The physical characteristics of the fuel to be stored in the Calvert Cliffs ISFSI are listed in Table 3.1-1. The key physical parameters listed in Table 3.1-1 of Reference 3.1 are the weight, length, cross-section dimensions, and the axial distance between fuel assembly spacer grids. Each of these Calvert Cliffs parameters are enveloped by the Reference 3.1 values. The DSC internal basket dimensions have been modified to suit Calvert Cliffs fuel.

The assumed maximum initial fill gas pressure in the CCNPP fuel rods is 465 psia. This pressure was used to calculate the number of moles of helium gas available for release from fuel rods into the DSC cavity during accident conditions causing an increase in the DSC internal pressure. The DSC accident pressures were calculated and utilized in the DSC stress analysis.

Table 3.1-1 lists the principal design parameters for fuel to be stored in the CCNPP ISFSI. The helium fill gas from the fuel rods is only a maximum of 30% for a NUHOMS-24P DSC and 35% for a NUHOMS-32P DSC (References 3.46 and 3.47) of the total gases (fission and fill gases from fuel rods, and DSC fill gas) in the DSC and therefore is not a major contributor to the accident DSC internal pressures. The helium fill gas from the fuel rods is a greater percentage of the total gases for the NUHOMS-32P DSC since there are more fuel rods and less DSC fill gas (References 3.46 and 3.47).

The peak fuel clad temperature limit for long term dry storage of spent fuel is calculated using end-of-life pressure in the fuel rods. The maximum initial fill gas pressure has a second order effect on the end-of-life pressure. Also the peak clad temperature is not very sensitive to the end-of-life pressure but is rather strongly dependent on the fuel burnup and cooling time.

Since the maximum helium fill gas pressure has a small impact on the analysis of the CCNPP ISFSI, it is not necessary to include it in Table 3.1-1.

Tables 3.3-3, 3.3-4, and 3.3-5 contain no information related to or affected by the differences in as-manufactured and post-irradiation dimensional envelope.

The value for assembly length provided in Table 3.1-1 (<158") is based on information provided by the fuel vendor, Combustion Engineering, Inc. (CE), which was that assembly length at 55,000 MWD/MTU was predicted to be 157.44”. The value provided on Table 3.1-1 does account for changes after irradiation in the core.

In order that experimental or prototype assemblies/rods be available for further examination/irradiation, it is not planned for these assemblies to be taken to the ISFSI, where they would not be readily accessible. However, should Calvert Cliffs Nuclear Power Plant for some reason, wish in the future to store these assemblies in the ISFSI, they would undergo the same qualification program as every other assembly to be stored in the ISFSI to ensure that all requirements are met.

3.1.1.2 Thermal Characteristics

To determine the cooling time required to limit the total decay heat power per spent fuel assembly to 0.66 kW, a series of ORIGEN2 calculations were performed for typical ranges of discharge fuel assembly burnups and enrichments shown in Table 9.4-1. These calculations were performed in accordance with the criteria in Section 3.1.1.2 of Reference 3.1 and resulted in the required cooling times shown in Table 9.4-1. The results were verified by comparison with other sources and calculational techniques (References 3.2, 3.3, 3.4, and 3.5). The resulting cooling times are also used to determine the reference assembly for radiological design purposes.

3.1.1.3 Radiological Characteristics

The radiological source terms were calculated, using ORIGEN2, for the range of initial enrichments and burnups given in Table 9.4-1. The source terms were calculated with cooling times for each assembly corresponding to 0.66 kW. The fuel assembly chosen to yield the largest source terms (both neutron and gamma) was found to be a 3.4 w/o initial enrichment, 42,000 MWD/MTU element, cooled for 8 years. These source strengths were used for shielding design throughout the ISFSI and are listed in Tables 7.2-1 and 7.2-2. A bounding curve combining both the neutron and gamma source terms as a function of assembly burnup and enrichment is given in Figure 7.2-1.

The corresponding information used in the NUHOMS-32P DSC evaluation is presented in Section 12.7.

3.1.2 GENERAL OPERATING FUNCTIONS

3.1.2.1 Functional Overview of the Facility

The Calvert Cliffs ISFSI is designed to maximize the use of existing plant features and equipment, and to minimize the need to add or modify equipment. The storage facility is located away from the existing plant security boundary in a separate protected area. The only services required from the plant during the ongoing passive storage mode will be security surveillance equipment located in the plant Central Alarm Station and Secondary Alarm Station. The storage facility is included in routine daily security patrols for the plant site. The power provided for the ISFSI security system and lighting is obtained from a retail source. Other support services from the plant are necessary only during DSC transfer and retrieval operations.

As described in Section 1.2, HSMs will be constructed on an as-required basis. Modules will be arranged in sets of two 2x6 arrays with capacity for 24 individual DSCs. Dry shielded canisters will be procured on an as-needed basis and will be delivered either singly or in small numbers according to actual plant needs.

3.1.2.2 Handling and Transfer Equipment

The NUHOMS system components are designed to interface with CCNPP fuel handling/storage equipment and facilities. This includes the Auxiliary Building receiving areas, spent fuel pool, cask washdown pit, processing systems, cask handling crane, spent fuel handling machine, and water and power supplies.

The additional handling and transfer equipment required to support the operation of the ISFSI include the transfer trailer, the cask skid, the skid positioning system, and the hydraulic ram system. Other equipment necessary to operate the system include a tractor for towing the transfer trailer to and from the ISFSI, and a mobile yard crane for raising and lowering the HSM front access door. This equipment is further described in Chapter 4.

3.1.2.3 Waste Processing, Packaging, and Storage Areas

As described in Section 3.1.2.3 of Reference 3.1, the only waste produced during the NUHOMS system operations is generated during fuel loading and subsequent DSC closure operations. The vacuum drying system is used to pump contaminated pool water from the DSC cavity either back into the spent fuel pool or to processing systems.

Likewise, the air and helium evacuated from the DSC during the drying operations will be routed to the Auxiliary Building processing systems or the spent fuel pool.

A limited amount of dry active waste is generated in the Auxiliary Building from protective clothing and consumable materials used during fuel loading, DSC drying, and sealing operations. This material is treated according to standard handling procedures at Calvert Cliffs.

The only other waste generated by the NUHOMS system will be the components of storage themselves, which will be treated and disposed of during facility decommissioning.

TABLE 3.1-1
PRINCIPAL DESIGN PARAMETERS FOR FUEL TO BE STORED

PARAMETER	**VALUE**
Physical Parameters:	
Maximum Assembly Length (with allowance for irradiation growth)	less than 158.0"
Nominal Cross-Section al Envelope	8.115"
Active Fuel Length	136.700"
Number of Fuel Rods/Assembly (including poison/inert replacement rods)	176*
Number of Guide Tubes/Assembly	5
Maximum Assembly Weight	1,450 lbs
Nominal Center-to-Center Distance Between Spacer Grids	18.86"
Thermal Characteristics:	
Decay Heat Power per Assembly	≤ 0.66 kW
Radiological Characteristics:	
Initial Uranium Content	386 Kg/Assembly (NUHOMS-24P Nominal) 400 Kg/Assembly (NUHOMS-32P Maximum)
Initial Fissile Content	≤ 4.5 w/o U^{235}
Total Gamma Source per Assembly	4.27×10^{15} photons/sec
Total Neutron Source per Assembly	2.23×10^{8} neutrons/sec (NUHOMS-24P) 3.30×10^{8} neutrons/sec (NUHOMS-32P)
Specific Power (core avg.)	32.2 MW/MTHM (NUHOMS-24P) 31.1 MW/MTHM (NUHOMS-32P)

For more information see Reference 3.14.

* Fuel Rods/Assembly (32P)

Fuel assemblies to be stored in 32P DSCs may contain up to two vacancies in any column or row; the vacancies do not need to be adjacent. Vacancies that violate this configuration are to be filled with stainless steel replacement rods.

Fuel assemblies to be stored in the 32P DSC may also contain a varying number of irradiated stainless steel replacement rods depending on the rods' exposure and time of cooling as shown in Table 9.4-3. An unlimited number of unirradiated stainless steel rods is permissible.

TABLE 3.1-4
RADIOLOGICAL CRITERIA FOR STORAGE OF MATERIAL IN THE ISFSI

Gamma Source Spectrum

E (Mean)[Mev]	Source Strength [Gamma/Sec Assembly]	Source Strength [Mev/Sec Assembly]
0.01	$9.79x10^{14}$	$9.79x10^{12}$
0.03	$2.18x10^{14}$	$6.54x10^{12}$
0.04	$2.66x10^{14}$	$1.06x10^{13}$
0.06	$1.95x10^{14}$	$1.17x10^{13}$
0.09	$1.20x10^{14}$	$1.08x10^{13}$
0.13	$1.25x10^{14}$	$1.63x10^{13}$
0.23	$9.76x10^{13}$	$2.24x10^{13}$
0.38	$5.13x10^{13}$	$1.95x10^{13}$
0.57	$1.83x10^{15}$	$1.04x10^{15}$
0.85	$2.53x10^{14}$	$2.15x10^{14}$
1.25	$1.28x10^{14}$	$1.60x10^{14}$
1.75	$2.20x10^{12}$	$3.85x10^{12}$
2.25	$1.40x10^{11}$	$3.15x10^{11}$
2.75	$8.03x10^{9}$	$2.21x10^{10}$
3.50	$1.03x10^{9}$	$3.61x10^{9}$
5.00	$9.63x10^{6}$	$4.82x10^{7}$
7.00	$1.11x10^{6}$	$7.77x10^{6}$
9.50	$1.28x10^{5}$	$1.22x10^{6}$
	$4.27x10^{15}$	$1.53x10^{15}$

Neutron Source Spectrum

Energy Range (Mev)			Source Strength (N/sec assembly) NUHOMS-24P DSC	Source Strength (N/sec assembly) NUHOMS-32P DSC
6.36	-	20.0	$6.100x10^{6}$	$9.027x10^{6}$
3.01	-	6.36	$4.486x10^{7}$	$6.638x10^{7}$
1.83	-	3.01	$5.007x10^{7}$	$7.409x10^{7}$
1.11	-	1.83	$4.658x10^{7}$	$6.893x10^{7}$
0.55	-	1.11	$4.287x10^{7}$	$6.344x10^{7}$
0.11	-	0.55	$2.959x10^{7}$	$4.379x10^{7}$
0.00335	-	0.11	$2.999x10^{6}$	$4.438x10^{6}$
< 0.00335			0	0
			$2.23x10^{8}$	$3.30x x10^{8}$

3.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA (References 3.14 and 3.44)

The Calvert Cliffs ISFSI components that are important to safety are the reinforced concrete HSM and its DSC support structure, the DSC and its internal basket assembly, and the transfer cask. Consequently, they are designed and analyzed to perform their intended functions under the extreme environmental and natural phenomena specified in Title 10 Code of Federal Regulations (CFR) Part 72 and American National Standards Institute (ANSI) 57.9. Table 3.2-1 of Reference 3.1 and USAR Section 12.3 summarize the design loadings for the equipment that is important to safety. These tables also summarize the applicable codes and standards used for design. A description of the structural and mechanical safety criteria for the remaining design loads listed in Table 3.2-1 of Reference 3.1 and USAR Section 12.3, such as thermal loads and cask drop loads, are provided in Section 8 of Reference 3.1 and USAR Section 12.8.

Other system components such as the vacuum drying system, the remote closure welding system, and cask support skid and positioning system, the transfer tractor and trailer, and the hydraulic ram are needed for the efficient operation of the NUHOMS system. Failure in any of these components will delay transfer of the loaded DSC from the Auxiliary Building to the ISFSI, but will not expose the public to any additional radiation. Table 3.2-1 and Section 12.3 provide a summary of the design criteria for those components to demonstrate that they meet reasonable industry standards and will ensure safe and efficient operation of the Calvert Cliffs ISFSI.

3.2.1 TORNADO WIND AND TORNADO-GENERATED MISSILE LOADINGS

The ISFSI is constructed within the existing boundaries of the CCNPP. For conservatism, the ISFSI HSM and transfer cask are designed for the Reference 3.1 severe wind and tornado loadings specified in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.76 and NUREG-0800, Section 3.5.1.4. These tornado wind and missile loads envelope the CCNPP values described in the Updated Final Safety Analysis Report (UFSAR). As discussed in Reference 3.1, the NUHOMS transfer cask design was evaluated to ensure that a design basis tornado missile would not breach the DSC pressure boundary or jeopardize the health and safety of the public. The non-safety-related components are designed for extreme operational loads, and have been designed to ensure that failure cannot jeopardize the health or safety of the public and the plant workers.

3.2.1.1 Applicable Design Parameters

The maximum tornado wind speed is 360 mph, the rotational speed is 290 mph, the maximum translational speed is 70 mph, the radius of the maximum rotational speed is 150' with an associated pressure drop of 3.0 psi occurring at a rate of 2.0 psi per second. The maximum transit time based on the 5 mph minimum translational speed specified for Region I was not used since an infinite transit time is conservatively assumed. The tornado-generated missiles are based upon the NUREG-0800 Section 3.5.1.4 (Reference 3.8), III.4 criteria and are the same as those discussed in Section 3.2.1.2 of Reference 3.1.

3.2.1.2 Determination of Forces on Structures

Tornado wind and missile loads were calculated using the same method described in Section 3.2.1.2 of Reference 3.1 for the HSM.

3.2.1.3 Ability of Structures to Perform

As detailed in Section 3.2.1.3 of Reference 3.1, the HSM protects the DSC from adverse environmental effects and is the principal NUHOMS structure exposed to tornado wind and missile loads. The HSM, including its air outlet shielding blocks, is designed to withstand tornados and tornado-generated missiles. 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 Calvert Cliffs HSMs have been constructed outdoors in an open area, there is no possibility of an adjacent building collapsing on an HSM. However, the possibility of blocking the ventilation air openings by a foreign object during a tornado event is considered. The effects of ventilation opening blockage are presented in Sections 8.2.7 and 12.8.2.7.

Tornado wind or missile loads on non-important to safety NUHOMS equipment could damage the equipment. Such damage may be sufficient to temporarily delay the transfer of a loaded DSC from the fuel building to the ISFSI, but at no time could it endanger the health or safety of the public or plant workers. Recovery from damage to the tractor, trailer, cask support skid, positioning system, or yard crane could require repair or replacement of damaged equipment to either allow return of the cask/DSC to the fuel building or completion of the transfer.

3.2.1.4 Tornado Missiles

The effects of tornado missiles defined by RG 1.76 were evaluated and reported in Section 8.2.2.

3.2.2 WATER LEVEL (FLOOD) DESIGN

The Calvert Cliffs ISFSI is located at an Elevation of 114' on a location free from flooding. The maximum postulated flood elevation including wind and wave run-up for the CCNPP is 28' Elevation (mean sea level) as discussed in Section 2.4. Since the maximum postulated flood is 86' below the ISFSI yard grade, the ISFSI is not subject to flooding.

3.2.3 SEISMIC DESIGN

In accordance with Section 3.2.3 of Reference 3.1, the design basis response spectra of NRC RG 1.60 was selected for the Calvert Cliffs ISFSI design earthquake. 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 RG 1.61, Table 1 was assumed to be applicable. Hence, a damping value of 3% of critical damping for the design basis earthquake was used. Similarly, from the same RG table, a damping value of 7% 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 RG 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" for a ground acceleration of 1.0g.

Nuclear Regulatory Commission RG 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 RG Figures 1 and 2 in proportion to the maximum specified horizontal ground acceleration. The maximum horizontal ground acceleration component selected for design in the NUHOMS-24P Topical Report (TR) (Reference 3.1) was 0.25g and the maximum vertical acceleration component selected was two-thirds of the horizontal component which is 0.17g. The design basis seismic peak ground acceleration values used for the design of the ISFSI, as obtained from the CCNPP UFSAR (Reference 3.9) are 0.15g horizontal and 0.10g vertical.

As discussed in Section 3.2.3 of Reference 3.1, various frequency analyses were performed for the different NUHOMS components and structures to establish the amplification factor associated with the design basis response spectra. The results of these analyses indicated that the dominant lateral frequency for a single stand-alone 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 Hertz produces a vertical seismic design acceleration of 0.21g. As discussed in Reference 3.1, the 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 Sections 8.2.3 and 12.8.2.3.

3.2.4 SNOW AND ICE LOADINGS

Horizontal storage module snow and ice loads were conservatively derived from ANSI A58.1-1982 and the maximum 100 year roof snow load of 110 psf assumed as in Section 3.2.4 of Reference 3.1. This exceeds the maximum Calvert Cliffs design basis snow load of 30 psf specified in the UFSAR. For the Calvert Cliffs ISFSI, a total live load of 200 psf was used in the HSM analysis to envelope all postulated live loadings, including snow and ice, as was done in Section 3.2.4 of Reference 3.1. Snow and ice loads for the NUHOMS transfer cask, with a loaded DSC, are negligible due to the curved surface of the cask, the decay heat of the fuel assemblies, and the infrequent short term use of the cask consistent with Reference 3.1.

3.2.5 COMBINED LOAD CRITERIA

3.2.5.1 Horizontal Storage Module

The Calvert Cliffs HSM site-specific load combination and design criteria matrix is the same as that specified in Section 3.2.5.1 and the Safety Evaluation Report for Reference 3.1, except that the exceptions taken to the Reference 3.1 criteria in the associated Safety Evaluation Report are addressed. The matrix is shown in Table 3.2-2.

3.2.5.2 Dry Shielded Canister

The Calvert Cliffs DSC site-specific load combinations and design criteria matrix are shown in Tables 3.2-3 and 12.3-6.

3.2.5.3 NUHOMS Transfer Cask

The Calvert Cliffs transfer cask site-specific load combinations and design criteria matrix is shown in Table 3.2-4.

The top and bottom support rings for the transfer cask were fabricated using ASME SA 182 Type F304N material, which is the forging equivalent of SA 240 Type 304 plate material. The analysis (Reference 3.32) conservatively uses the allowables for SA 240 Type 304 for the SA 182 Type F304N material, which has somewhat higher S_m, S_y, and S_u values per Tables I-1.2, I-2.2, and I-3.2, respectively, of the ASME Code.

The transfer cask shell was fabricated from SA 240 Type 304 plate material and; therefore, fracture toughness is not a concern.

As noted in Reference 3.1, the transfer cask is an atmospheric vessel whose configuration does not meet all of the requirements of the ASME Code. The Calvert Cliffs NUHOMS Transfer Cask is designed to the same consistent set of ASME Code rules reported in the topical report. The design rules of Subarticle NC-3200 and Appendix XIII (Alternative Design Rules for Vessels) were used to develop the transfer cask design criteria. The design basis stress intensities reported in Reference 3.1, Table 8.1-2, were developed from ASME Table I-1.0 and are consistent with the requirements of NC-3200.

The rules employed in the design of the transfer cask are by no means a "least restrictive set." The rules are consistently selected from NC-3200. No mixing of requirements for different types of vessels has been employed. All design, fabrication, inspection, and testing requirements are consistently applied for a Class 2 pressure vessel in strict accordance with the ASME Code. The design of the cask trunnions is performed to ANSI N14.6 which provides design criteria even more restrictive than the ASME Code.

The transfer cask allowable bolt stresses reported in Reference 3.1, Table 3.2-9 were developed from XIII-1180 for vessels designed to NC-3200. As noted in Reference 3.1, Table 3.2-9, the same allowable stress intensity values are used for Service Levels A, B, & C. The reported allowable stresses for Service Level D were developed in accordance with the rules of F-1335. The bolt stress limits of Table NC-3923.1-1 and NC-3923.2(c) are not used in this design. (Also see Table 3.6-1.)

3.2.5.4 NUHOMS System Transfer Equipment

The NUHOMS system transfer equipment consists of the cask support skid, positioning system, trailer, tractor unit, and hydraulic ram. This equipment is non-safety-related and is designed, fabricated, and operated in accordance with applicable industry codes and standards. Chapter 4 provides a discussion of the design loads, codes, and standards for the NUHOMS system transfer equipment.

3.2.6 WELD REQUIREMENTS

3.2.6.1 NUHOMS-24P DSC

The DSC is designed and fabricated to the rules of Subsection NB for a Class 1 component, but is not a certified ASME pressure vessel. Subsection NB contains no prequalified approved weld joint details applicable to the DSC closure welds; therefore, the code was used for guidance in developing sound engineered details for the redundant DSC closure welds.

The redundant closure welds were included in the DSC finite element models to ensure that the calculated stresses represent the expected behavior of the component for each postulated loading event. Weld allowables and the joint efficiency factors were developed by taking the most restrictive interpretation of the Section III rules for welds of this type. In addition, the loading events which induce stress into the DSC primary closure welds only relate to accident type events such as: failure of the inner seal weld plus pressurization of the DSC due to fuel cladding failure; or an accidental drop of the DSC during transportation from the fuel building. The postulated events which produce significant stress in the DSC closure welds are one time events which do not cause repetitive loads and do not act as crack propagating events. Therefore, with no significant cyclical loads present, the conservatively designed partial penetration welds provide reliable closure weld details for the DSC. Details of specific welds are discussed below.

- Reference 3.17, Section C-C: The multi-pass 1/4" fillet weld provides a redundant bottom closure weld, and with a maximum stress of 4.67 ksi does not need to meet unrelated AISC size requirements. (Reference 3.41 pgs 3.93, 3.94)

- Reference 3.17, Detail 2: The full penetration welds of Items 6 to 8 and Items 2 to 8 are included in the DSC finite element analytical model while the partial penetration weld of Items 5 to 8 is only loaded when inserting the DSC into the HSM. (Reference 3.41, pg 3.95)

- Reference 3.17, Note 3: The note applies to the full penetration longitudinal and girth welds of the canister shell as shown in the detail for the shell. The note states that full penetration welds are required at the seams, that the detail of the weld preparation is the responsibility of the fabricator, and it restates the requirement shown on the weld callout that these welds must be 100% radiographed. The 5/8” 60° vee weld between the shell, part number I and the bottom cover plate, part number 2, is not possible to radiograph. As stated in the earlier response to this question, Subsection NB does not contain prequalified weld joint details applicable to the canister closure welds, and the Code was used for guidance in these instances. Accordingly, both the 5/8” 60° vee weld and the 1/4” fillet providing the redundant seal on the opposite face of the shell are required to have multi-level dye penetrant testing since volumetric inspection of these weld joints is not feasible.

- Reference 3.18, Section B-B: The two welds shown on Section B-B serve the function of attaching the drain and fill port block to the DSC shell and provide a seal to ensure that accidental pressurization of the DSC cannot bypass the shield plug to shell closure weld and inadvertently pressurize the top cover plate. All welds are multi-pass and are inspected by the liquid penetrant method and are examined as part of the DSC helium leak check following shield plug welding to ensure no leakage paths exist.

- Reference 3.19, Detail 1: The multi pass partial penetration groove welds connecting the top shield plug to the DSC shell and sealing the Swagelok fittings provide the inner seal weld for the DSC cavity. The weld stresses are calculated (Reference 3.41, pg 3.90) as 1.32 ksi due to pressure and 12.42 ksi due to an accident drop event compared to allowables of 9.35 ksi and 22.4 ksi, respectively.

- Reference 3.20, Detail 4: The 3/8" bevel weld connects the shield plug side casing plate to the inner cover. The only stress in this weld is due to thermal affects and, therefore, it was sized to meet minimum weld requirements.

- Reference 3.21: The 1/8" fillet welds shown on Sections A-A and C-C are used to seal the gap between the spacer disc and support rod to avoid a potential crud trap. The welds have no structural significance and will perform their design function as detailed.

- Reference 3.22: The 1/8" fillet welds connecting the keyway to the shell have no structural function. The keyways at 0° and 180° azimuths are provided to avoid any potential rotation of the DSC basket assembly during transportation of the empty DSC. Failure of any or all of these 1/8" fillet welds would be of no consequence to the safety of the DSC or spent fuel. Increasing the weld sizes could potentially have a detrimental affect on local shell deformations.

- Reference 3.23: The 3/8" bevel partial penetration welds specified for the ram access penetration port were conservatively designed to meet ASME Code requirements for fillet weld allowables (Reference 3.42, pg 4.72). American Welding Society (AWS) D1.1 Table 2.10.3 requires a total effective throat of 1/2" for material 2-1/4" to 6" thick. Detail 3 provides a total effective throat of greater than 3/4" and therefore meets AWS requirements. The transfer cask, including the 3/8” partial penetration weld, uses ASME Boiler and Pressure Vessel Code, Section III, Subsection NC, as the code of design and construction. Neither AISC nor AWS standards are applicable. The ASME Code does not specify minimum weld sizes. The cask and this weldment is designed by analysis to meet the requirements of the Code, Subsection NC, and is therefore acceptable.

- Reference 3.24: The fillet welds were increased to 1/4".

- Reference 3.25: The 3/16" fillet weld was changed to 1/4".

- Reference 3.26: The 3/8" bevel weld was changed to 1/2".

- Reference 3.27: The 1/4" plate used to form the neutron shell casing for the cask top lid has no structural function. Holes are provided in the 1/4" plate to ensure that any off-gas products from the NS-3 do not pressurize the cavity. The welding requirements for these items were therefore specified as seal welds of minimum size to avoid distortion of the relatively light weight (1/4") plate used. The cask fabricator will use the necessary preheat temperatures and weld detail to ensure that he can satisfactorily build the item.

- Reference 3.28: The inner and outer shield plug assemblies are non-structural items provided to limit the radiation dose at the cask bottom surface during DSC transfer operations at the HSM. As such, the weld sizes specified are not critical and the fabricator will provide the minimum weld sizes to fabricate these items. American Institute of Steel Construction Code and AWS D1.1, are codes for welding of structural steel. The codes are intended for structures which perform some kind of load carrying function. This certainly is not the case for the temporary plug assemblies. They carry no load except for their own weight when being placed onto or removed from the cask. Failure of a weld would result in neither an increase in radiation from the cask nor a safety hazard. The weld sizes shown are adequate for the plug assemblies to perform their intended function, and are acceptable.

- Reference 3.29: The 1/8" fillet provides a seal weld to assist in decontamination of the yoke. It serves no structural function and is acceptable as specified.

- Reference 3.30: The 1/2" plug weld shown in Detail 5 is not part of the ASME pressure boundary of the transfer cask shell. The weld is required by the dog leg path introduced into the annulus pressurization part to access the cask/DSC annulus below the annulus seal. The plug weld is only subjected to minor stresses during pressurization of the annulus during fuel load operations. This weld does not need to meet ASME requirements and does not require any backup.

3.2.6.2 NUHOMS-32P DSC

The weld requirements for the NUHOMS-32P DSC are described in Section 12.3.2.6.

TABLE 3.2-1
SUMMARY OF DESIGN CRITERIA FOR NON-IMPORTANT TO SAFETY COMPONENTS

COMPONENT	DESIGN LOAD TYPE	DESIGN PARAMETERS	APPLICABLE CODE
Transfer Trailer	Dead Load	Weight of loaded DSC + Transfer Cask = 215,000 lbs[a] Weight of Skid and Positioning System = 49,700 lbs Trailer Dead Load (Payload Capacity) = 264,700 lbs	N/A
	Operating Loads to Trailer Deck	a) Longitudinal load: 30,000 lb b) Transverse load: 40,000 lb	N/A
Skid Positioning System	Operating Loads	a) Longitudinal Load: 30,000 lb b) Transverse Load: 20,000 lb/cylinder c) Vertical Load: 100 tons	
	Extents of Motion	a) Longitudinal: 35" b) Transverse: +/- 5" c) Vertical: 10"	
Cask Support Skid	Dead Load	Weight of loaded DSC + Cask = 215,000 lb[a] enveloping	AISC Code - 1978
	Operating Loads	a) Positioning system loads: Longitudinal load: 30,000 lb Transverse load: 40,000 lb b) Transportation load, case 1: 1.0g vertical or 1.0g transverse or 1.0g longitudinal c) Transportation load, case 2: 0.5g vertical + 0.5g transverse + 0.5g longitudinal	AISC Code - 1978
Hydraulic Ram System	Normal Operation	a) Operating force: 20,000 lb b) Maximum speed: 36 in/min	N/A
	Jammed Condition Handling	a) Operating force: 80,000 lb b) Maximum Speed: 9 in/min	
Cask Restraints	Normal Operation	Ram hydraulic cylinder normal operating force, 20,000 lb	ANSI/ANS 57.9
	Jammed Condition Handling	Ram hydraulic cylinder maximum operating force, 80,000 lb	ANSI/ANS 57.9
Vacuum Drying System	Normal Operation	DSC vacuum: 3.0 Torr	

For more information see References 3.14 and 3.44.

AISC American Institute of Steel Construction

[a] The values are based upon the NUHOMS-32P DSC design and are bounding for the NUHOMS-24P DSC design.

TABLE 3.2-2
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
5	D + L + H + T + E	L = Live Load
6	D + L + H + T + F	T = Normal Condition Thermal Load
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 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, overall damage, overturning, and sliding effects.
4. The HSM load combinations included an additional 5% dead load. The live load used was 100% of the estimated value. In the calculation of moments and shears for major HSM components, absolute values of maximum moments and shears were added for conservatism.

TABLE 3.2-3
NUHOMS-24P DSC DESIGN LOAD COMBINATIONS

Load Case(1)			Normal Operating Conditions				Off-Normal Conditions				Emergency and Accident Conditions(2)							
Type		I.D.	1	2	3	4	1	2	3	4	1	2	3	4	5	6	1	2
Dead Weight	Empty DSC	DW_1	X															
	DSC w/water	DW_2		X														
	DSC w/fuel	DW_3			X	X	X	X	X	X	X	X	X	X	X	X	X	X
Thermal	Inside HSM: normal	T_{nh}				X					X	X			X			
	Inside Cask: normal	T_{nc}		X	X		X									X		X
	Inside HSM: off-normal	T_{ho}							X				X					
	Inside Cask: off-normal	T_{co}						X		X								
	Inside HSM: Accident	T_{ha}												X				
	Inside Cask: Accident	T_{ca}															X	
Internal Pressure	Normal Operating	P_n			X		X											
	Hydrostatic	P_h		X														
	Off-normal (blowdown)	P_b				X		X	X	X								
	Accident (inner boundary)	P_{a1}									X	X	X	X	X	X		
	Accident (outer boundary)	P_{a2}															X	X
Handling Loads	Normal DSC Transfer	L_n			X	X						X						
	Off-normal (jammed DSC)	L_o					X	X	X						X	X		
Accident Loads	Cask Drop	DL																X
	Seismic	E									X							
ASME BPVC Service Level			A	A	A	A	B	B	B	B	C	C	C	C	C	C	D	D
Load Combination No.			A_1	A_2	A_3	A_4	B_1	B_2	B_3	B_4	C_1	C_2	C_3	C_4	C_5	C_6	D_1	D_2

(1) The Table has been modified to include hydrostatic and blowdown pressure, to distinguish between accident pressure along the inner boundary (ASME Service Level C) and outer boundary (ASME Service Level D), and to delete the flooding accident load for which no analysis is required.

(2) For emergency and accident load combinations, the DSC shall not be allowed to deform to an extent that would prevent retrieval of spent fuel. For Service Level D, the DSC internal components need only comply with deformation limits that will allow the retrieval of spent fuel. In addition, both end plug assemblies shall maintain their ability to provide shielding for personnel during DSC handling operations.

TABLE 3.2-4
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	6	7
Dead/Load/Live Load		X	X	X	X	X	X	X	X	X	X	X	X	X	X
Thermal w/DSC	-3° to 103°F Ambient[1]	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Handling Loads (Critical Lifts)[3]	Vertical	X													
	Tilted		X												
	Horizontal			X											
Handling Loads (Non-Critical)	Transport				X		X		X						
	DSC Transfer					X		X		X					
Seismic									X	X					
Tornado Wind Loads[2]											X				
Tornado-Generated Missile[2]															X
Drop	Vertical (Top & Bottom)											X			
	Corner												X		
	Horizontal													X	
ASME Code Service Level		A	A	A	A	A	B	B	C	C	C	D	D	D	D
Load Combination No.		A_1	A_2	A_3	A_4	A_5	B_1	B_2	C_1	C_2	C_3	D_1	D_2	D_3	D_4

(1) Off-normal temperature based on Table 3.6-2.
(2) Load case is additional to Reference 3.1 requirements.
(3) ANSI 14.6 (Reference 3.45) allowable stresses for all Upper Trunnion critical lifts.

3.3 SAFETY PROTECTION SYSTEMS

3.3.1 GENERAL

The Calvert Cliffs ISFSI is designed for safe and secure long-term containment and storage of spent fuel. The major components which assure that the safety objectives are met are listed in Table 3.3-1. The key elements which require special design consideration are:

A. Double Closure Seal Welds on DSC Ends
B. Minimization of Radiation Exposure During DSC Closure Operations
C. Minimization of the Contamination of DSC Exterior by Pool Water
D. Minimization of Radiation Exposure During DSC Transfer Operations
E. Design of the Transfer Cask and DSC for Postulated Accidents
F. Design of the HSM Passive Ventilation System for Effective Decay Heat Removal to Assure Fuel Cladding Integrity

These items are addressed in Section 3.3.2 of Reference 3.1 and in the following subsections for the NUHOMS-24P DSC and Section 12.3.3 for the NUHOMS-32P DSC.

3.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

3.3.2.1 Confinement Barriers and Systems

The Calvert Cliffs ISFSI design incorporates multiple confinement barriers to ensure there will be no release of airborne radioactive effluent to the environment. The radioactivity which must be confined originates from the spent fuel assemblies and potential DSC and transfer cask exterior contamination from loading operations in the spent fuel pool. The ISFSI multiple radioactivity confinement barriers are listed in Table 3.3-2.

Section 3.3.2.1 of Reference 3.1 provides a detailed discussion of the DSC physical containment barriers. These barriers which include double closure seal welds at the DSC ends provide protection against the design basis accident of DSC leakage as defined in Section 8.2.8. Reference 3.1 includes additional requirements for fabrication and testing of the closure welds which include multiple-pass and multiple level liquid penetrant inspection. This requirement provides additional assurance of leak tightness because it effectively eliminates a pinhole leak which might occur in a single pass weld. The chance of pinholes being in alignment on successive weld passes is negligible. Additionally, helium leak testing is required for the top shielding welds to 10^{-4} atm-cc/sec. Use of a single pass weld and a single liquid penetrant inspection for the interior 1/4" seal weld at the bottom end of the DSC is also acceptable provided a leak test is performed on the closure.

The NRC, in their Safety Evaluation Report for the Calvert Cliffs Updated Safety Analysis Report, specified a requirement for the DSC shell hoop and longitudinal welds that they be tested using proof pressure testing method in accordance with ASME B&PV Code, Section III, NB-6000. This requirement is being met, except for the first ten DSCs, which were loaded with spent fuel before they were proof pressure tested. These canisters were, however, leak tested as follows: The top welds were leak tested with helium in accordance with the Technical Specifications, and the bottom, girth and longitudinal welds were leak tested with soap bubble film per ANSI N14.5. These tests provide adequate assurance that the DSCs provide a leak tight containment. Further proof pressure testing of the in-service canisters is not practical. The NRC concurred with the "as-is" use of similar in-service canisters in their letter to the canister manufacturer (Reference 3.43).

Dry shielded canister exterior contamination is minimized by preventing spent fuel pool water from contacting the DSC exterior. Dry shielded canister loading procedures (Section 4.4.1) require that the annulus between the transfer cask and DSC be filled with demineralized water and sealed prior to immersion in the spent fuel pool. Annulus sealing is accomplished by an inflatable seal between the transfer cask and DSC. The combination of the above operations provides assurance that the DSC exterior surface has less residual contamination than required for shipping cask externals. Surface swipes of the DSC exterior will be taken while in the cask washdown area to assure that the maximum DSC removable contamination does not exceed:

Beta/Gamma Emitters	22,000 dpm/100 cm^2
Alpha Emitters	2,200 dpm/100 cm^2

Transfer cask external contamination is minimized by the use of smooth, easily decontaminated surface finishes to minimize personnel radiation exposures during cask handling operations outside the spent fuel pool. Title 49 CFR 173.443(d), which governs contamination levels for off-site shipment in a closed exclusive use vehicle, was used to develop the cask maximum removable contamination limits as:

Beta/Gamma Emitters	22,000 dpm/100 cm^2
Alpha Emitters	2,200 dpm/100 cm^2

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

Containment of radioactive material associated with spent fuel assemblies is provided by fuel cladding, the DSC stainless steel body, and double seal welded primary and secondary closures. As described in Section 3.3.2.1 of Reference 3.1, there are no credible events which will breach a DSC to provide a possible leakage path to the environment.

3.3.2.2 Ventilation — Off-gas

There are no radioactive effluent releases during normal or off-normal operations at the ISFSI. There are no credible accidents which cause releases of radioactive effluent from the DSC during storage. No off-gas system or radiological effluent release monitoring is required for the ISFSI.

Auxiliary Building processing systems are used during the DSC purge and drying operations. During this operation, the gases purged from the DSC cavity are routed to the Auxiliary Building processing systems or the spent fuel pool.

3.3.3 PROTECTION BY EQUIPMENT AND INSTRUMENTATION SELECTION

3.3.3.1 Equipment

The DSC and the HSM are the only important to safety system components in use during normal and off-normal storage. The transfer cask is used during the DSC transfer operation and is also important to safety. Safety-related plant equipment associated with the operation of the ISFSI includes the Spent Fuel Cask Handling Crane and its lifting yoke, which are used inside the Auxiliary Building and are regulated under the 10 CFR Part 50 plant license. The design criteria for the ISFSI important to safety components are provided in Section 3.2. A discussion of the classification of components as safety-related or important to safety is presented in Section 3.4.

3.3.3.2 Instrumentation

The Calvert Cliffs ISFSI is designed to maintain a safe and secure long-term containment and storage environment for spent fuel using only passive components. Therefore, no important to safety instrumentation is required for operation of the facility.

3.3.4 NUCLEAR CRITICALITY SAFETY

The NUHOMS DSC internals were designed to provide nuclear criticality safety during all phases of NUHOMS operations and storage, including wet loading operations and postulated accident conditions. The Calvert Cliffs site-specific NUHOMS design has been determined to satisfy the requirements of 10 CFR 72.124 for both normal and postulated accident conditions.

The nuclear criticality safety of the NUHOMS-32P DSC is addressed in Section 12.3.3.4. The nuclear criticality safety of the NUHOMS-24P system is addressed in detail in Section 3.3.4 of Reference 3.1. This section reports the results of the Calvert Cliffs site-specific criticality analysis. In addition, discussion is presented regarding particular site-specific issues at Calvert Cliffs and certain areas where the Calvert Cliffs criticality analysis methodology differs from that presented in Reference 3.1.

3.3.4.1 Control Methods for Prevention of Criticality

Criticality control is provided during the cask fuel loading, DSC drying and sealing (wet conditions), and the transfer and storage phases (dry conditions). Control methods for the prevention of criticality under wet conditions consist of the physical properties and irradiation history of the fuel; mechanical control of the fuel assemblies' location by the DSC basket; neutron absorption in the DSC basket structural materials, such as the steel guide sleeves; and Calvert Cliffs administrative controls for fuel identification, verification, and handling. An additional safety margin for subcriticality during wet operations is the presence of soluble boron in the fuel pool and in the water used to fill the DSC. The fuel pool is typically maintained at a boron penetration of 2300 to 2500 ppm. However, any concentration can be maintained based on need.

Rigorous measures are taken to exclude the possibility of introducing moderator into the DSC cavity during the dry operations of transfer and storage. Prior to these operations, the DSC will be vacuum dried, backfilled with helium, double seal welded, and helium leak tested to assure weld integrity. Therefore, under normal operating conditions there is no possibility of a criticality incident. Since the transfer cask and HSM are designed to provide adequate drop and/or missile protection for the DSC, and the DSC basket is designed to keep the fuel assemblies separated from each other even after a drop accident, there is no credible accident scenario which would result in the possibility of the entrance of moderator into the DSC; nor is there a credible accident scenario which would prohibit the canister from being opened and re-flooded.

The NUHOMS-24P system was generically licensed for Babcock & Wilcox (B&W) 15x15 fuel with a maximum equivalent initial enrichment, as determined by a burnup equivalence curve, of 1.45 w/o U^{235} (Reference 3.1). Calvert Cliffs' fuel is CE 14x14 design. This site-specific fuel was determined to be less reactive than B&W 15x15 fuel and the corresponding maximum equivalent initial enrichment was calculated to be 1.80 w/o U^{235}. A new burnup equivalence curve was established for this site-specific design and is presented in Figure 3.3-1. The data used to construct Figure 3.3-1 is presented in tabular form in Table 9.4-2.

3.3.4.2 Design Parameters for Criticality Model

The criticality model was constructed using the same methodology and assumptions used in the Topical Report (Reference 3.1) except as noted below.

The criticality analysis was performed for CE design 14x14 fuel assemblies with geometry and fuel characteristics as shown in Table 3.3-3. The nominal dimensions of the DSC and transfer cask are provided in Table 3.3-4 and the geometry is illustrated in Figure 3.3-2. A summary of the design parameters for the criticality analysis is presented in Table 3.3-5.

The 45 GWD/MTU shown on Table 3.3-5 represents the maximum value on the Y axis (burnup) scale of the burnup equivalence curve given in Figure 3.3-1. This burnup equivalence curve is used to determine the acceptable maximum equivalent U^{235} initial enrichment at various burnups in the criticality analysis of the Calvert Cliffs NUHOMS design. The highest data point in Figure 3.3-1 corresponds to a burnup of 42 GWD/MTU.

All fuel assemblies to be stored in the Calvert Cliffs NUHOMS must meet the equivalent enrichment criterion of less than 1.8 w/o U^{235}, as determined using the equivalence curve in Figure 3.3-1 or an equivalent non-graphical method based on the data listed in Table 9.4-2. The initial fuel enrichments considered in the analysis range from 1.8% to 4.5% U^{235}. The irradiated fuel is assumed to be cooled for 5 years following discharge from the reactor for the purposes of the criticality analysis.

The analyses performed to generate the reactivity equivalence data of Table 9.4-2 assume cooling time $>$ 5.0 years. Use of either a specific ORIGEN analysis or the decay times shown in Table 9.4-1 in conjunction with the reactivity equivalence curve, constitute the mechanism for insuring a K_{eff} value of $<$ 0.95 and a decay heat value $\leq$ 0.66 kW/assembly. This decay heat limitation ensures that the peak fuel clad temperature will remain below 335°C under normal storage conditions. Additionally, the component surface dose rate values will not exceed acceptable limits for any fuel assembly which meets the limiting conditions above.

The peak fuel clad temperature limit of 335°C for Calvert Cliffs fuel (Combustion Engineering 14x14) was calculated using the same bounding conservative design criteria and analysis methods previously reviewed and approved by the NRC for the generic NUHOMS-24P design. The peak clad temperature limit of 340°C reported in the NUTECH topical report is based on Babcock and Wilcox 15x15 fuel which is the basis for the topical report design. The peak clad temperature limit is dependent on end-of-life cladding hoop stress, cooling time, and temperature in storage using the model described in Reference 3.36, which is based on thermal creep. This reference provides curves and algorithms for calculating the acceptable initial storage temperature for a given cladding hoop stress and cooling time.

Using the methodology of Reference 3.36, the resulting maximum storage temperature for the design basis fuel to preclude damage of fuel cladding during long term storage is 335°C for CE 14x14 fuel with a burnup of 50 GWD/MTHM and 12 years cooling time. The maximum allowable cladding temperature limit for lower burnups and shorter times which also result in 0.66 kw of decay heat per assembly are higher than 335°C.

3.3.4.3 Reactivity Equivalence and Criticality Analysis Methods (Reference 3.14)

The reactivity equivalence and criticality analysis methodology is identical to that described in Reference 3.1, with the following differences:

A. The design parameters used for the criticality analysis are those described above in Section 3.3.4.2.

B. The ROCS computer code (Reference 3.12) was used to generate sets of irradiated fuel actinide inventories over the burnup and enrichment ranges of interest. ROCS is a multi-group two-dimensional diffusion code which uses higher order difference methodology to perform burnup calculations on pressurized water reactor fuel assemblies. ROCS is an industry recognized code which has been accepted by the NRC for use in core reload analyses.

   Irradiated fuel assembly K_{inf} data generated by stand alone ROCS calculations are considered reliable for use as benchmark standards since the code has been used extensively in reactor physics and calculations and its ability to produce accurate irradiated fuel assembly neutron cross-sections is well established by comparisons of calculated and measured operating conditions.

Start-up data for eleven cores covering four cycles of fuel management are used to validate the ROCS code (Reference 3.31). Measured data obtained during reactor start-up are considered the most reliable because they consist of well controlled conditions. The measured and ROCS predicted hot, zero power, xenon free, all rods out critical boron concentrations for each cycle are compared. Over the 11 points of the data base, the comparison indicates that ROCS underestimates critical boron concentrations by an average of 14 ppm, with tolerance levels of ±29 ppm at a 95/95 probability/confidence level. In terms of reactivity, this corresponds to an underprediction of 0.18%$\Delta\rho$ with two sided tolerance limits of ±0.37%$\Delta\rho$.

Additionally, a comparison of ROCS calculated versus measured reactivity has been performed for various reactor sites including a large data base of 1281 data points. This large data base consists of measurements made over a wide range of reactor core critical conditions characterized by core average exposure, power level, inlet temperature, soluble boron concentration and control rod insertion. This large data base has been used to establish a ROCS bias of -0.25%$\Delta\rho$ with tolerance limits of ±0.39%$\Delta\rho$ at a 95/95 probability/confidence level. This bias and uncertainty are in good agreement with the hot zero power results, demonstrating that Doppler and thermal-hydraulic reactivity effects, as well as fission product worth, are correctly treated throughout the irradiation cycle by the ROCS method.

Review of the benchmark results provided in the CCNPP criticality analysis demonstrates that the criticality equivalence method used conservatively overpredicts K_{eff} for systems containing irradiated fuel. The CSAS4 method comparison to stand-alone ROCS K_{inf} results indicates that the CCNPP criticality analysis method over predicts K_{inf} by approximately 0.027 to 0.033 $\Delta K/K$ over the burnup range of interest. This positive bias more than offsets the slight negative bias of the ROCS method documented in the ROCS topical report validation.

The validity of ROCS calculated actinide concentrations is verified in a comparison of calculated and measured nuclide content for three- and four-cycle Calvert Cliffs Unit 1 fuel.

The uncertainties in isotopic composition of fuel have been sufficiently quantified and conservatively accounted for in the

CCNPP criticality analysis. These, coupled with other conservatism inherent in the design of the DSC which takes credit for fuel burnup and system operational controls are adequate to ensure that the acceptance criteria for subcriticality are met for all plausible conditions.

The ROCS data is corrected, as recommended by CCNPP ROCS code analysis, to reflect an average fuel temperature of 1100°F (versus the 1500°F temperature actually used in the ROCS calculations). Note that an average fuel temperature of 68°F is used in the CSAS4 comparison calculations.

Several factors likely contribute to the level of disparity observed between the ROCS stand-alone and CSAS4 method calculated K_{inf} values. The use of 68°F for fuel and moderator temperature in the CSAS4 calculations contributes approximately +1.1%ΔK to the CSAS4 results. It is also reasonable to assume that not including SM-151 and volatile fission product absorbers as well as the collective contribution of relatively minor neutron absorbers (treated as lumped fission product absorbers in ROCS) in the CSAS4 calculations contribute significantly to the CSAS4 overprediction. Other explanations and/or contributing factors are possible. However, the complex nature of the subject calculations do not allow a more definitive explanation for the disparity observed between the two calculational methods.

The CSAS25 K_{inf} overpredicts the ROCS K_{inf} by almost four percent. Since this magnitude of overprediction is consistent over the wide range of K_{inf} cases analyzed, one would expect this trend to apply for slightly lower values of ROCS K_{inf} as well.

A series of calculations using the CSAS25/SAS2/ROCS method for analyzing irradiated fuel reactivity was performed which attempted to duplicate the geometry, materials, and conditions modeled by ROCS in the fuel reload data base calculations from which the actinide number densities for the CCNPP criticality analysis were obtained. These results indicate that the CSAS25 method overpredicts K_{inf} relative to ROCS calculated results by an average of 0.02754ΔK/K for zero burnup conditions and 0.03301ΔK/K over the burnup range indicated (this range becomes 0.01444ΔK/K and 0.02275ΔK/K, respectively, when fuel and moderator temperatures consistent with ROCS values are used in the CSAS4 calculations). Since the ROCS reactivity calculation method has been validated by its successful application in core and fuel reload analyses, the CSAS25 method's positive bias to ROCS over the entire burnup range considered in the CCNPP criticality analysis demonstrates the conservative nature of the CSAS results in this application.

The uncertainties presented the CCNPP criticality analysis are based on comparisons between calculations. Fuel assembly burnable poison and fuel cycle boron concentration assumptions were considered and calculational biases added to ensure worst-case treatment of these variable fuel cycle conditions. SAS2 fuel assembly model approximation effects were considered and SAS2 comparison calculations were performed and determined to be negligible.

The reactivity bias associated with shimmed assemblies is assumed to be 1%, and is added into the reactivity equivalency calculation. The shimmed assembly bias is based on CCNPP fuel reload calculations. The information provided in Attachment 2 to the Calvert Cliffs criticality analysis states that fuel assemblies which contain fixed, burnable neutron poisons overpredict K_{eff} by approximately 1% over the burnup range of interest.

The effect of the fuel cycle boron concentration on irradiated fuel actinide and fission product inventory (and subsequently K_{eff}) concentrations is assumed to be a 2% bias over the nominal case 450 ppm concentration and is also incorporated into the equivalence calculation. This bias is based on the K_{eff} results for 0 ppm and 800 ppm fuel cycle boron concentrations at various burnup levels. The 800 ppm case produced the largest bias over the nominal case and was used in reactivity equivalencing.

Because of the high solubility of boron, especially at higher temperatures, for levels less than 13,000 ppm, the boron concentration in the liquid phase is not affected by boiling. Boron remains in the liquid phase of any two-phase mixture (i.e., the two-phase mixture is composed of pure unborated steam and borated saturated liquid water). Therefore, boron both a) remains in the liquid phase, and b) is carried into the two-phase mixture through its continued presence in the liquid phase component of any steam/water mixture.

Dry shielded canister cavity pressure during routine loading/unloading operations does not vary significantly from atmospheric pressure. For moderator densities in excess of 0.9579 gm/ml, the moderator is a single phase liquid. For moderator densities less than 0.0005978 gm/ml, the moderator is a single phase gas (steam). For moderator densities between 0.0005978 and 0.9579 gm/ml, the moderator exists as a two-phase mixture of saturated liquid water containing soluble boron and saturated steam. For any finite volume of a two-phase steam/water mixture, the total mass of H_2O in the volume is equal to the mass of H_2O in the steam or vapor phase plus the mass of H_2O in liquid phase. Similarly, the total boron in any finite volume is equal to the mass of boron in the vapor phase plus the mass of boron in the liquid phase. It follows that the effective boron concentration for any two-phase saturated borated

steam/water mixture can be calculated using a "two-phase physical model":

$$ND_m = \frac{D_m \; x \; 0.6023 \; x \; [(MF_s \, x \, PPMB_s) + (MF_1 \, x \, PPMB_1)]}{MW_b \, x \, 10^6}$$

Where:

ND_m	=	Effective boron number density of mixture (atoms/barn-cm)
D_m	=	Steam/liquid mixture effective density (gm/ml)
	=	MF_s x D_s + MF_1 x D_1
MF_s	=	Mass fraction of steam phase in mixture
	=	$(1/D_m - 1/D_1) / (1/D_s - 1/D_1$
MF_1	=	Mass fraction of liquid phase in mixture
	=	$1 - MF_s$
D_s	=	Steam phase density (gm/ml)
	=	0.0005978 (assume saturated)
D_1	=	Liquid phase density (gm/ml)
	=	0.9579 (assume saturated)
0.6023	=	Avogadro's number multiplied by $1.0x10^{24}$ cm^2/barn (atoms-cm^2/mole-barn)
$ppmB_s$	=	Parts-per-million boron concentration in steam
$ppmB_1$	=	Parts-per-million boron concentration in liquid
MW_b	=	Molecular weight of boron (gm/mole)
10^6	=	ppm unit conversion factor

Assuming the liquid phase boron concentration remains constant at 1800 ppmb (i.e., the liquid phase boron concentration level does not rise due to concentrating effects of boiling) and the steam phase contains no boron (i.e., no entrainment of soluble boron), the "two-phase physical model" described by the above equation is used to calculate the effective B-10 number densities of steam/water mixtures (i.e., ND_m values multiplied by the atomic fraction of boron that is B-10, or 0.19764) for the entire range of possible mixture density conditions.

The comparison of B-10 number densities calculated using the physically based two-phase model and the uniform moderator density model demonstrates that the moderator densities at which optimum moderation occurs in the CCNPP boron credit calculations (i.e., between 0.2 and 0.4 gm/ml) are too high for the presence of steam bubble voids to significantly lower the effective boron concentration of the water/steam mixtures below the values calculated by the "uniform moderator density model" and used in the K_{eff} calculations.

The uniform moderator density/boron concentration assumption used in the CCNPP criticality analysis is identical to the method used and approved in the NUTECH topical report and the presence of steam bubbles do not adversely affect the K_{eff} of the system above the values calculated in the CCNPP criticality analysis. This conclusion also holds for higher pressures corresponding to the static head present in submerged elevations of a flooded DSC. In addition, pure steam (i.e., no boron) conditions have been analyzed and demonstrated not to be a concern.

Although Reference 3.1 utilized the CASMO computer code, extensive validation and successful use in both incore and excore applications adequately demonstrate that licensed incore fuel management neutronics codes such as CASMO and ROCS accurately predict the reactivity of fuel arrays containing irradiated fuel under cold conditions representative of excore storage and handling conditions.

The ROCS computer code incorporates a high-order nodal solution to the diffusion equation and the CASMO code employs a multigroup solution using transport theory. Both codes are licensed industry recognized tools for performing incore neutronics calculations. Details regarding methodology and assumptions as well as each code's ability to accurately predict the reactivity of irradiated fuel is documented in their respective topical reports (References 3.12 and 3.37).

CASMO actinide inventory data for irradiated fuel along with ORIGEN fission product inventories are used in Reference 3.1 criticality analysis. This method was shown to be a conservative predictor of irradiated fuel reactivity when compared to stand-alone CASMO calculated K_{inf} values. A similar series of comparison calculations was performed to validate the CCNPP reactivity equivalence analysis method. The use of ROCS irradiated fuel actinide data and stand-alone K_{inf} values for method verification in the CCNPP criticality analysis is further justified by its extensive use in CCNPP reload calculations involving the CE 14x14 fuel design specifically.

Both CASMO and nodal solution codes like ROCS are currently being used to perform excore criticality analysis including burnup-credit applications (References 3.38, 3.39, and 3.40).

C. The Criticality Safety Analysis Sequence No. 4 (CSAS4), included in the SCALE-3 package of codes (Reference 3.13), was used in calculating effective neutron multiplication factors. This analysis sequence includes the updated Monte-Carlo Code (KENO-Va) which precludes the necessity for explicitly including metal reflector positioning bias and uncertainty due to enhanced geometrical modeling capability.

KENO-Va is an extension of the KENO-IV Monte Carlo criticality program and was developed for use in the SCALE system. With the exception of enhanced geometry capabilities and output features, the two codes remain functionally identical in terms of neutronics calculations. The new geometry features include the "arrays of arrays" option, the "holes" option, and variable chords for hemicylinders and hemispheres. In addition, identical cross-section data is employed in both the NUTECH topical report and CCNPP Safety Analysis Report criticality analyses (i.e., the 123GROUPGMTH neutron cross-section library provided with the SCALE package).

A diverse subset of the critical experiments used to benchmark the Reference 3.1/KENO-IV criticality analysis method was also analyzed by the BGE/KENO-Va method. Comparison of the common critical experiments for each version of KENO yields an average K_{eff} of 0.99620 for the KENO-Va analysis and an average K_{eff} of 0.99550 for the KENO-IV results.

For the unirradiated, nominal case analyzed in the CCNPP criticality calculation, 40,100 neutron generations were followed using KENO-Va. Reference 3.1 analysis used KENO-IV with 50,100 neutron generations. Differences in the number of neutron histories followed is somewhat arbitrary. In general, the number of histories followed is minimized to reduce computer run time. However, the number of neutron histories followed in each analysis case is selected to ensure problem convergence and provide low monte-carlo statistical uncertainty.

D. Credit was taken for 33 major fission product poisons identified as stable sources of negative reactivity.

3.3.4.4 Normal Conditions

The calculated worst-case reactivity of a fully loaded Calvert Cliffs NUHOMS-24P DSC is 0.93144. This value is less than the ANSI/American Nuclear Society (ANS)-8.17-1984 criterion of $K_{eff} \leq 0.95$. It was calculated with a 95% probability at a 95% confidence level for a system moderated by pure, unborated water and contains all applicable uncertainties and biases including the effects of:

- Stainless steel guide sleeve thickness
- Fuel assembly pitch
- Cell ID
- Cell bowing
- Assembly positioning
- Optimum moderator density
- Irradiated fuel reactivity analysis methodology

The following equation is used to develop the final K_{eff} result for the DSC fuel storage array in the manner presented in Reference 3.1:

$$K_{eff} = k_{nominal} + B_{method} + B_{moderator} + [(k_{s\text{-}nominal})^2 + (k_{s\text{-}method})^2 + (k_{s\text{-}moderator})^2 + (k_{s\text{-}mechanical})^2]^{0.5}$$

Where:

$k_{nominal}$	=	Nominal case K_{eff}	=	0.86983
B_{method}	=	Method bias	=	0.00492
$B_{moderator}$	=	Moderator density bias	=	0.02283
$k_{s\text{-}nominal}$	=	95/95 uncertainty in the nominal case K_{eff}	=	0.00552
$k_{s\text{-}method}$	=	95/95 uncertainty in the method bias	=	0.01350
$k_{s\text{-}moderator}$	=	95/95 uncertainty in the moderator bias	=	0.00596
$k_{s\text{-}mechanical}$	=	95/95 uncertainty resulting from mechanical uncertainties	=	0.029968

Substituting the appropriate values in the order listed:

$$K_{eff} = 0.86983 + 0.00492 + 0.02283 + [(0.00552)^2 + (0.0135)^2 + (0.00596)^2 + (0.029968)^2]^{0.5}$$

$$K_{eff} = 0.93144$$

The mechanical uncertainties and biases incorporated in the CCNPP criticality analysis are based on a DSC basket design which is slightly different than the design presented in Reference 3.1. Also, the fuel type analyzed in the CCNPP analysis is CE 14x14 while Reference 3.1 uses Babcock & Wilcox 15x15. The mechanical biases and uncertainties are calculated in the same manner as those presented in Reference 3.1, but the magnitude of each of these values is dependent on fuel type and cell geometry.

A fissile array reflector bias and uncertainty is not included in the CCNPP analysis since a hemispherical reflector can be accurately modeled using KENO-Va and worst-case reflector effects were included in the nominal case model.

In the irradiated fuel reactivity equivalence calculations, the CCNPP analysis includes biases to account for 1) possible burnable poison rod effects on the reactivity of irradiated CE 14x14 fuel assemblies, and 2) positive reactivity effects resulting from the use of 0 ppm and 450 ppm average reactor coolant boron concentrations in the ROCS and SAS2 fuel cycle calculations, respectively.

Sensitivity calculations performed by CCNPP indicate that for some special cases involving high burnup fuel (17.5 - 60 GWD/MTU), arrays of shimmed CE 14x14 fuel assemblies may be as much as 1% DK more reactive than arrays of irradiated fuel assemblies containing no integral burnable poison rods. Sensitivity calculations performed as part of Reference 3.1 criticality analysis did not indicate a positive effect from the use of burnable poisons and the application of bias was not necessary. This is likely due to the differences in fuel assembly design, enrichments, etc., and the methods used to implace burnable poisons in each design.

Sensitivity calculations performed as part of the CCNPP criticality analysis indicate that assuming higher average reactor coolant boron concentrations than used in developing irradiated fuel actinide and fission product number densities resulted in a positive reactivity effect. Assuming a maximum average reactor coolant concentration of 800 ppm boron resulted in the application of a 2% DK positive bias since the nominal case ROCS and SAS2 calculations assumed 0 ppmb and 450 ppmb, respectively. Since a worst-case reactor coolant boron concentration was used in Reference 3.1 burnup credit criticality analysis, no such bias was necessary.

The method bias and uncertainty values in the CCNPP criticality analysis are different from Reference 3.1 since the CSAS2/KENO-IV method was used in Reference 3.1 and the CSAS4/KENO-Va method was used in the Calvert Cliffs analysis.

3.3.4.5 Off-Normal Conditions (Reference 3.14)

The three postulated off-normal conditions will not result in a DSC storage array with a reactivity higher than that allowed by ANSI/ANS-8.17-1984. The off-normal conditions considered include:

1. The misloading of one highly enriched, unirradiated fuel assembly into the DSC,
2. The misloading of 24 highly enriched, unirradiated fuel assemblies into the DSC, and
3. Optimum moderation.

Misloading of a Single High Enrichment Assembly — No Boron Credit

The calculated worst-case K_{eff} value for a misloading of one 4.5 w/o enriched, unirradiated fuel assembly in a DSC containing 23 1.8 w/o equivalent enriched assemblies and pure, unborated water at optimum moderator density is 0.97968. This calculated value includes geometrical and material uncertainties and biases at a 95% probability, 95% confidence level as required by ANSI/ANS-57.2-1983, to demonstrate criticality safety. This value is less than the ANSI/ANS-57.2-1983 criterion of $K_{eff} \leq 0.98$.

The following equation is used to develop the final K_{eff} result for the DSC fuel storage array loaded with 23 design basis fuel assemblies and one highly enriched, unirradiated fuel assembly with unborated water:

$$K_{eff} = k_{nominal} + B_{method} + B_{moderator} + [(k_{s\text{-}nominal})^2 + (k_{s\text{-}method})^2 + (k_{s\text{-}moderator})^2 + (k_{s\text{-}mechanical}{}^2]^{0.5}$$

Where:

$k_{nominal}$	=	Nominal case K_{eff}	=	0.94349
B_{method}	=	Method bias	=	0.00492
$B_{moderator}$	=	Moderator density bias	=	0.0
$k_{s\text{-}nominal}$	=	95/95 uncertainty in the nominal case K_{eff}	=	0.00550

A series of sensitivity calculations were performed to determine that a slightly higher peak reactivity occurs for water densities between 0.5 - 0.7 g/cc. This positive reactivity effect was conservatively considered as a "worst-case" condition in this analysis and was included in both the 0.93144 (normal) and 0.97968 (off-normal) values reported above.

$k_{s\text{-}method}$	=	95/95 uncertainty in the method bias	=	0.01350
$k_{s\text{-}moderator}$	=	95/95 uncertainty in the moderator bias	=	0.0
$k_{s\text{-}mechanical}$	=	95/95 uncertainty resulting from mechanical uncertainties	=	0.02766

Substituting the appropriate values in the order listed:

$$K_{eff} = 0.94349 + 0.00492 + 0.0 + [(0.00550)^2 + (0.0135)^2 + (0.0)^2 + (0.02766)^2]^{0.5}$$

$$K_{eff} = 0.97968$$

<u>Misloading of 24 High Enrichment Assemblies — With Boron Credit</u>

The criticality analysis has demonstrated that the Calvert Cliffs ISFSI system, like the NUHOMS-24P generic design, meets the double contingency requirement of ANSI/ANS-57.9-1984 for simultaneous optimum moderation and a misload of 24 unirradiated highly enriched fuel assemblies in borated water.

The calculated worst-case K_{eff} value for the incredible event of misloading of 24 4.5 w/o enriched, unirradiated fuel assemblies in a DSC containing 1800 ppm borated water at optimum density is 0.94481. This calculated value includes geometrical and material uncertainties and biases at a 95% probability, 95% confidence level as required by ANSI/ANS-57.2-1983, to demonstrate criticality safety. This value is less than the ANSI/ANS-57.2-1983 criterion of $K_{eff} \leq 0.98$.

The boric acid concentration in the spent fuel pool is typically much higher than 1800 ppm, usually about 2600 ppm. The Fuel Handling Procedure which governs fuel placement into the DSC verifies that the pool concentration is at least 1800 ppm, with the Safety Analysis as a Basis of that requirement. With that as a Basis, the procedural requirement cannot become less restrictive without verifying consistency with the Safety Analysis.

The following equation is used to develop the final K_{eff} result for the DSC fuel storage array loaded with 24 highly enriched, unirradiated fuel assemblies and 1800 ppm borated water:

$$K_{eff} = k_{nominal} + B_{method} + B_{moderator} + B_{spacer} + [(k_{s\text{-}nominal})^2 + (k_{s\text{-}method})^2 + (k_{s\text{-}moderator})^2 + (k_{s\text{-}spacer})^2 + (k_{s\text{-}mechanical})^2]^{0.5}$$

Where:

$k_{nominal}$	=	Nominal case K_{eff}	=	0.91324
B_{method}	=	Method bias	=	0.00492
$B_{moderator}$	=	Moderator density bias	=	0.0
B_{spacer}	=	Bias accounting for the positive reactivity effects of borated water displacement by DSC basket spacer disk	=	0.0
$k_{s\text{-}nominal}$	=	95/95 uncertainty in the nominal case K_{eff}	=	0.00550
$k_{s\text{-}method}$	=	95/95 uncertainty in the method bias	=	0.01350
$k_{s\text{-}moderator}$	=	95/95 uncertainty in the moderator bias	=	0.0
$k_{s\text{-}spacer}$	=	95/95 uncertainty in the moderator bias	=	0.0
$k_{s\text{-}mechanical}$	=	95/95 uncertainty resulting from mechanical uncertainties	=	0.02231

Substituting the appropriate values in the order listed:

$$K_{eff} = 0.91324 + 0.00492 + 0.0 + 0.0 + [0.00550)^2 + 0.0135)^2 + (0.0)^2 + (0.0)^2 + (0.02231)^2]^{0.5}$$

$$K_{eff} = 0.94481$$

Note that the example given uses a worst-case moderator density of 0.3 g/cc. At this moderator density, the steel spacer disk provides more absorption than the borated moderator, hence B_{spacer} and $k_{s\text{-}spacer}$ are both 0.0. $B_{moderator}$ and $k_{s\text{-}moderator}$ are also both 0.0 since this calculation was performed for 0.3 g/cc moderator density.

<u>Optimum Moderation</u>

Optimum moderation effects were calculated and included in all reported system reactivities.

It was noted for the nominal case that the system was slightly overmoderated and a 95/95 bias of 0.02283 +/- 0.00596 was added to account for the positive reactivity noted in the range of moderator density from 0.5 to 0.7 g/cc of pure water.

For the single misloaded assembly off-normal event, it was noted that the system reactivity was essentially driven by the single assembly and no positive reactivity insertion was noted for partial density pure water moderator. Instead, a monotonically decreasing negative reactivity insertion was noted for reduced moderator density. A negative bias of approximately 0.02 delta k was observed in the range of 0.5 to 0.7 g/cc, however no credit was taken.

For the case where all 24 assemblies are unirradiated, high enrichment assemblies, there was substantial reactivity insertion on the order of 0.13 delta k due to diminished density of the borated moderator. In this case, the peak was observed at 0.3 g/cc borated moderator density.

Since all reported reactivities include an allowance for optimum moderation, and all reported reactivities are within the design limits specified by ANSI/ANS-57.2-1983, a criticality event due to moderator density alone is not credible for Design Events I, II, or III. For a Design Event IV or V condition, which is postulated to be a misload of all 24 assemblies, optimum moderation will not result in criticality, provided that at least 1800 ppm of boron is present in the DSC fill water. Therefore subcriticality is assured, even in the event that a flooded DSC was to remain out of the pool long enough for boiling to occur.

3.3.4.6 Criticality Analysis Method Verification

The analysis method which ensures a subcriticality margin of greater than 5% under all normal conditions uses the Criticality Analysis Sequence No. 4 (CSAS4) and the 123GROUPGMTH master cross-section library included in the SCALE-3 system of codes (Reference 3.13).

A set of 21 critical experiments shown in Table 3.3-6 have been analyzed using the CSAS4/123GROUPGMTH reactivity calculation method to demonstrate its applicability and to establish method bias and variability. The experiments analyzed represent a diverse group of water moderated, heterogeneous oxide fuel arrays separated by various materials (stainless steel, Boral, water, etc.) that are representative of light water reactor shipping and storage conditions. The average K_{eff} of the benchmarks is 0.99620 with a method bias of 0.00492 delta k and a 95/95 uncertainty of 0.01350 delta k, given a sample size of 21 experiments.

3.3.5 RADIOLOGICAL PROTECTION

The Calvert Cliffs ISFSI is designed to maintain on-site and off-site doses as low as reasonably achievable (ALARA) during transfer operations and long-term storage conditions. Independent Spent Fuel Storage Installation transfer procedures, shielding design, and access controls provide the necessary radiological protection to assure radiological exposures to station personnel and the public will be maintained ALARA. Further details on collective on-site and off-site doses resulting from ISFSI operations and the ISFSI ALARA evaluation are provided in Chapter 7.

3.3.5.1 Access Control

The Calvert Cliffs ISFSI is located within the owner controlled area of the CCNPP. A separate protected area consisting of a double fenced, double gated, lighted area will be installed around the ISFSI. Access is controlled by locked gates. Guards are stationed at the ISFSI when the gates are open. As described in the Security Plan, remote sensing devices will be employed to detect unauthorized access to the facility. In addition to the controlled access, the HSM steel doors will be tack welded after insertion of a loaded DSC. The HSM steel doors weigh approximately 6 tons and require heavy equipment for removal. This ensures that there is ample time to respond to an unauthorized entry into the ISFSI before access can be gained to any radiological material.

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 HSM is designed to limit the maximum surface dose to 100 mrem/hr at the penetrations with a nominal external surface dose (gamma and neutron) of 20 mrem/hr. The transfer cask and the DSC top shielded end plug are designed to limit the surface dose (gamma and neutron) to less than 200 mrem/hr. Temporary neutron shielding may be placed on the DSC shield plug and top cover plate during closure operations. Information on collective doses is provided in Chapters 7 and 12.

3.3.5.3 Radiological Alarm Systems

No radiological alarms are required at the ISFSI.

3.3.6 FIRE AND EXPLOSION PROTECTION

The ISFSI HSM and DSC contain no flammable material and the concrete and steel used for their fabrication can withstand any credible fire hazard. There is no fixed fire suppression system within the boundaries of the ISFSI. The facility is located such that plant personnel can respond to any fire emergency using portable fire suppression equipment. The effects of a forest fire around the facility are discussed in Sections 8.2.10 and 12.8.

Independent Spent Fuel Storage Installation initiated explosions are not considered credible since no explosive materials are present in the fission product or cover gases. Externally initiated explosions are considered to be bounded by the design basis tornado-generated missile load analysis presented in Section 8.2.2. Liquid natural gas explosions are discussed in Section 8.2.11.

The HSM is designed to withstand a peak overpressure of 12 psi (factor of safety of 1.0) which is equivalent to a 750 mph wind. The transfer cask is designed to withstand a 6.6 psi overpressure without tipping (factor of safety of 1.0) which is equivalent to a 638 mph wind.

Portable fire suppression equipment will be provided within the fenced boundary whenever a motor vehicle is stationed on the ISFSI site. In addition, as diesel fuel does not present a significant fire or explosion hazard, signs will be posted at the ISFSI entrance stating that only diesel powered vehicles will be allowed in the site.

Even though several vehicles may be present at any one time, the maximum credible fuel spill within the ISFSI boundary would involve all of the fuel on a single vehicle, for a maximum spill of 100 gallons of diesel fuel (the capacity of both fuel tanks on the tow vehicle). Diesel fuel has a flash point of 120°F.

Due to the positive drainage of the ISFSI approach slabs, a spill large enough to cause puddling would also tend to drain toward the site storm drainage system and thus away from HSMs.

This drainage, coupled with the expected rapid detection of any fire by the fuel transfer personnel and the on-site presence of fire fighting personnel and equipment will tend to severely limit the spread and severity of any fire. In addition, off-site fire fighting assistance is available, if required. The damage caused by any fire is expected to be negligible given the massive nature of the HSMs.

A spill too small to cause puddling would be very difficult to ignite due to the relatively high flash point of diesel fuel. In any case such a small fire would be unlikely to pose a credible threat to the ISFSI.

If a fire was to occur, a post-fire recovery plan would be formulated. The exact scope of the plan would vary depending on the size and intensity of the fire. As a minimum it would include:

a. Complete radiological survey of potentially affected HSMs (including particulate and gaseous activity)

b. Inspection of affected HSM surfaces

c. Interior inspection of HSMs if damage is reasonably suspected

d. Repair of concrete and steel structures as required

e. Root cause analysis and implementation of corrective actions as required

3.3.7 MATERIALS HANDLING AND STORAGE

3.3.7.1 Spent Fuel Handling and Storage

As described in Section 3.3.7.1 of Reference 3.1, all spent fuel handling outside the spent fuel pool is done with the fuel assemblies in the DSC. The DSC provides a double welded containment vessel with an internal basket arrangement to maintain subcriticality and cladding integrity at all times.

For long-term storage, HSM passive ventilation maintains the maximum normal operating fuel clad temperature to 620°F or less (assuming 103°F ambient temperature). During short-term conditions, such as DSC draining and drying, transfer of the DSC to/from the HSM and off-normal and accident temperature excursions, the fuel cladding temperature reaches a maximum value of 838°F which is significantly less than the maximum allowable value of 1,058°F.

Permissible contamination levels for the external DSC and transfer cask surfaces are discussed in Section 3.3.2.1. Any contamination of the DSC interior will remain confined by the double seal welded containment vessel.

3.3.7.2 Radioactive Waste Treatment

No radioactive waste will be generated during the storage life of the DSC. Radioactive wastes generated in the Auxiliary Building during DSC closure operations (contaminated water from the spent fuel pool and potentially contaminated air and helium from the DSC) will be treated using existing plant systems and procedures as described in Chapter 6.

3.3.7.3 On-site Waste Storage

The requirements for on-site waste storage will be satisfied by existing facilities for handling and storage of waste from the spent fuel pool and dry active waste. These requirements are described in Chapter 6.

3.3.8 INDUSTRIAL AND CHEMICAL SAFETY

No hazardous chemicals or chemical reactions are involved in the fuel loading or storage process. Industrial safety relating to handling of the cask and DSC will be addressed by procedures which meet Occupational Safety and Health Administration requirements.

TABLE 3.3-1
MAJOR COMPONENTS AND FUNCTIONS

COMPONENT	FUNCTION
Transfer Cask	On-site Fuel Transport, Shielding
Dry Shielded Canister NUHOMS-24P Guide Sleeves Spacer Disks Support Rods End Shield Plugs DSC Body End Cover Plates NUHOMS-32P Guide Sleeves Egg-Crate Plates Peripheral Steel Rails with Aluminum Inserts End Shield Plugs DSC Body End Cover Plates	Criticality Control, Fuel Support, Cover Gas Containment, Radioactive Material Confinement, Shielding (Lead Plugs)
Horizontal Storage Module Concrete Shielding DSC Support Assembly HSM Passive Ventilation System	Shielding, DSC Support, DSC Tornado Missile Protection, DSC Cooling
Foundation	HSM Foundation Support
Transfer Components Cask Support Skid and Positioning System Transfer Trailer Hydraulic Ram Optical Alignment System Tractor	Transfer Cask Movement, DSC Transfers

TABLE 3.3-2
RADIOACTIVE MATERIAL CONFINEMENT BARRIERS

RADIOACTIVE SOURCE	CONFINEMENT BARRIERS
Spent Fuel Pool Water	Demineralized Water in DSC/Transfer Cask Annulus
	Mechanically Sealed Annulus between DSC and Transfer Cask
Irradiated Fuel and Fission Gases	Fuel Cladding
	DSC Containment Pressure Boundary
	Seal Welded Primary Closure (Top Shield Plug Assembly) of DSC
	Seal Welded Secondary Closure of DSC

TABLE 3.3-3
CE 14x14 FUEL PARAMETERS

FUEL ASSEMBLY PARAMETER	INCHES
Fuel Clad OD	0.44
Fuel Pellet OD	0.3765*
Clad Thickness	0.028*
Fuel Rod Pitch	0.58
Guide Tube OD	1.115
Guide Tube Thickness	0.04
Active Fuel Height	136.7
Fuel Rods/Assembly	176**
No. of Guide Tubes	5

Reference: J. W. Roddy, "Physical and Decay Characteristics of Commercial LWR Spent Fuel," ORNL/TM-9591/V1&R1, January 1988.

For more information see Reference 3.14.

* The fuel pellet OD and clad thickness varied slightly for Fuel Batches A, B, and C in Units 1 and 2. These variances do not affect the results of design basis analysis.

** Fuel Rods/Assembly (32P)

Fuel assemblies to be stored in 32P DSCs may contain up to two vacancies in any column or row; the vacancies do not need to be adjacent. Vacancies that violate this configuration are to be filled with stainless steel replacement rods.

Fuel assemblies to be stored in the 32P DSC may also contain a varying number of irradiated stainless steel replacement rods depending on the rods' exposure and time of cooling as shown in Table 9.4-3. An unlimited number of unirradiated stainless steel rods is permissible.

TABLE 3.3-4
NUHOMS-24P FUEL BASKET AND CASK DIMENSIONS

Geometry Description	Nominal Dimensions (Inches)
Guide Sleeve ID	8.70
Guide Sleeve Thickness	.1050
Ligament Thickness	1.5
	1.25
	1.0
C-C Spacings	10.360
	10.285
	10.035
Support Plate Opening ID	9.100
Cell Height	136.7
DSC Shell ID	66.0
DSC Shell Thickness	0.625
Cask Inner Shell ID	68.0
Cask Inner Shell Thickness	.75
Pb Shield Thickness	4.0
Cask Outer Shell Thickness	1.5

For more information see Reference 3.14.
See Table 12.3-1 for NUHOMS-32P dimensions.

TABLE 3.3-5
DESIGN PARAMETERS FOR CRITICALITY ANALYSIS OF THE NUHOMS-24P DSC

PARAMETERS	DESIGN VALUE
FUEL ASSEMBLIES	
Number/Type	24/CE design 14x14
Rod Array	14x14
Number of Fuel Rods	176
Number of Control Rod Guide Tubes	5
Number of Instrument Tubes	1 (1 of the 5 Guide Tubes)
Rod Pitch (inches)	0.580
Burnup Credit	0 - 45 GWD/MTU
FISSILE CONTENT (% initial U equivalent)	
U^{235}	1.8 - 4.5
FUEL PELLETS	
Density	95% Theoretical
Diameter (inches)	0.3765*
FUEL ROD CLADDING	
Material	Zircaloy - 4
Thickness (inches)	0.028*
Outside Diameter (inches)	0.440
CONTROL ROD GUIDE TUBES	
Material	Zircaloy - 4
Thickness (inches)	0.040
Outside Diameter (inches)	1.115
INSTRUMENT TUBE	
Material	Zircaloy - 4
Thickness (inches)	0.040
Outside Diameter (inches)	1.115
DSC GUIDE SLEEVES	
Material	Stainless Steel
Thickness (inches)	0.1050
DSC FILL MATERIAL	
Material	Pure Water
Density (g/cm^3)	0.0 - 1.0
DSC SHELL	
Material	Stainless Steel
Thickness (inches)	0.625
Outside Diameter (inches)	67.25
CASK	
Material	Stainless Steel/Lead
Thickness (inches)	6.25[a]
Outside Diameter (inches)	80.5[a]

For more information see Reference 3.14.

[a] Exclusive of the Cask Neutron Shield

* The fuel pellet OD and clad thickness varied slightly for Fuel Batches A, B, and C in Units 1 and 2. These variances do not affect the results of design basis analysis.

See Table 12.3-2 for NUHOMS-32P parameters.

TABLE 3.3-6
RESULTS FOR 21 B&W CRITICAL BENCHMARK EXPERIMENTS
FOR THE NUHOMS-24P DSC

Core	K_{eff}	SIGMA	Measured	SIGm
1	0.99929	+/- 0.00325	1.00020	+/- 0.0005
2	0.99793	+/- 0.00312	1.00010	+/- 0.0005
3	0.99159	+/- 0.00326	1.00000	+/- 0.0006
4	1.00466	+/- 0.00359	0.99990	+/- 0.0006
5	1.00283	+/- 0.00387	1.00000	+/- 0.0007
6	1.00734	+/- 0.00401	1.00970	+/- 0.0012
7	0.99108	+/- 0.00385	0.99980	+/- 0.0009
8	1.00230	+/- 0.00422	1.00830	+/- 0.0012
9	0.99028	+/- 0.00425	1.00300	+/- 0.0009
10	0.98536	+/- 0.00341	1.00010	+/- 0.0009
11	0.99951	+/- 0.00314	1.00000	+/- 0.0006
12	0.98885	+/- 0.00373	1.00000	+/- 0.0007
13	1.00092	+/- 0.00405	1.00000	+/- 0.0010
14	1.00607	+/- 0.00373	1.00010	+/- 0.0010
15	0.99559	+/- 0.00382	0.99980	+/- 0.0016
16	0.97893	+/- 0.00397	1.00010	+/- 0.0019
17	0.99803	+/- 0.00317	1.00000	+/- 0.0010
18	0.99755	+/- 0.00340	1.00020	+/- 0.0011
19	0.99557	+/- 0.00331	1.00020	+/- 0.0010
20	0.99341	+/- 0.00411	1.00030	+/- 0.0011
21	0.99110	+/- 0.00412	0.99970	+/- 0.0015

3.4 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Table 3.4-1 provides a list of major ISFSI components and their classification. Components are classified not only with respect to the criteria in 10 CFR Part 72 but, for equipment used inside of the CCNPP, with respect to the criteria in 10 CFR Part 50. This is necessary because 10 CFR Parts 72 and 50 use different criteria in determining what structures, systems, and components are safety significant.

Structures, systems, and components "Important to Safety" are defined in 10 CFR 72.4 as those features of the ISFSI whose function is (1) to maintain the conditions required to store spent fuel safely, (2) to prevent damage to the spent fuel container during handling and storage, or (3) to provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

"Safety-Related" equipment is defined in 10 CFR 50.49 as that which is relied upon to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the 10 CFR Part 100 guidelines.

3.4.1 TRANSFER CASK

The transfer cask is "important to safety" because it protects the spent fuel container during handling. The transfer cask is also considered "safety-related" since dropping a loaded transfer cask (weighing 109.25 tons) has the potential for creating unanalyzed accident situations in the power plant. The transfer cask is described in Section 4.7.3.3 and is designed, constructed, and tested in accordance with a Quality Assurance (QA) program that meets the requirements as defined by 10 CFR Part 50, Appendix B and the Quality Assurance Topical Report.

3.4.2 DRY SHIELDED CANISTER

The DSC is "important to safety" because it maintains the conditions needed to store fuel safely. The DSC is considered "safety-related" since it performs criticality control and primary fuel assembly support functions required to maintain the assumed fuel geometry. Unexpected criticality inside a DSC could lead to off-site doses comparable with the limits in 10 CFR Part 100. The DSC is designed to remain intact under all accident conditions identified in Chapter 8 with no loss of function. The DSC was designed, constructed, and tested in accordance with a QA program that meets the requirements defined by 10 CFR Part 50, Appendix B and the Quality Assurance Topical Report. The welding materials required to make the closure welds on the DSC top shield plug assembly and DSC top cover plate are purchased to the same American Society of Mechanical Engineers (ASME) Code criteria as the DSC (Section NB Class 1).

3.4.3 HORIZONTAL STORAGE MODULE

The HSM which consists of the concrete shielding, the DSC support assembly, and the foundation is considered "important to safety" because it protects the spent fuel container during storage. This component is not used in the Power Plant and therefore is not classified under 10 CFR Part 50. The concrete HSM is designed in accordance with American Concrete Institute (ACI) 349-85 and the level of testing, inspection, and documentation provided during construction and maintenance is in accordance with the CCNPP QA Program, described in Chapter 11.

3.4.4 TRANSFER COMPONENTS

The remaining DSC transfer components (i.e., ram, skid, trailer) are necessary for the successful loading of the DSC into the HSM. However, the analyses described in Chapter 8 demonstrate that the performance of these items is not required to provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. Therefore, these components are neither considered "important to safety" nor do they meet the 10 CFR Part 50 definitions of "safety-related" components. Therefore, transfer components are considered "non-safety-related." These components are designed, constructed, and tested in accordance with good industry practices.

3.4.5 OTHER COMPONENTS

The lifting yoke used for movement of the transfer cask within the Auxiliary Building is designed and procured as a "safety-related" component for the reasons given for the classification of the transfer cask as "safety-related." The lifting yoke is controlled by NUREG-0612 and is designed to ANSI N14.6-1986 criteria for non-redundant yokes.

The vacuum drying system and the automatic welding system are neither "important to safety" nor "safety-related" for the reasons given in Section 3.4.4. Failure of any part of these systems can only result in a delay of operations, and not in a hazardous situation to the public or operating personnel.

TABLE 3.4-1
MAJOR COMPONENTS AND CLASSIFICATION

COMPONENT	10 CFR PART 72 CLASSIFICATION	10 CFR PART 50 CLASSIFICATION	CCNPP QA PROGRAM
Transfer Cask	Important to Safety[b]	Safety-Related[a]	SR
Dry Shielded Canister (NUHOMS-24P)	Important to Safety	Safety-Related	SR
Basket			
Spacer Disks			
Support Rods			
End Shield Plug/Support (top and bottom)			
DSC Body			
End Closure Plates			
Dry Shielded Canister (NUHOMS-32P)	Important to Safety	Safety-Related	SR
Basket			
Egg-Crate			
End Shield Plug/Support (top and bottom)			
DSC Body			
End Closure Plates			
Lifting Yoke		Safety-Related	SR
Horizontal Storage Module	Important to Safety		Augmented Quality[c]
Concrete Shielding			
DSC Support Assembly			
Foundation			
Transfer Components	Not Important to Safety	Non-Safety-Related	NSR
Transfer Trailer/Skid			
Ram Assembly			
Vacuum Drying System	Not Important to Safety	Non-Safety-Related	NSR
Automatic Welding System	Not Important to Safety	Non-Safety-Related	NSR

For more information see Reference 3.16.

TABLE 3.4-1
MAJOR COMPONENTS AND CLASSIFICATION

(a) "Safety-Related" equipment is defined in 10 CFR 50.49 as that which is relied upon to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the 10 CFR Part 100 guidelines.

(b) Structures, systems, and components "Important to Safety" are defined in 10 CFR 72.4 as those features of the ISFSI whose function is (1) to maintain the conditions required to store spent fuel safely, (2) to prevent damage to the spent fuel container during handling and storage, or (3) to provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

(c) Augmented Quality components are subject to application of certain QA standards to meet regulatory or CCNPP requirements. The HSM is subject to application of the 10 CFR Part 72, Subpart G QA program described in Section 11.2 during the construction phase. As with the SR components, it is controlled by the 10 CFR Part 50, Appendix B QA program during the operational phase and the decommissioning phase.

3.5 DECOMMISSIONING CONSIDERATIONS

Decommissioning of the ISFSI will be performed in a manner consistent with the decommissioning of the CCNPP. It is anticipated that the DSCs will be transported intact to a federal repository when such a facility is operational. However, should the storage facility not accept the DSCs intact, the NUHOMS system allows the DSCs to be brought back into the spent fuel pool and the fuel to be placed into the spent fuel racks for loading into transport casks provided by the Department of Energy.

All components of the NUHOMS system are manufactured of materials similar to those found in the existing CCNPP (i.e., reinforced concrete, stainless steel, lead). These components will be decommissioned by the same methods in place to handle similar materials within the plant.

3.6 SUMMARY OF DESIGN CRITERIA

A summary of the design criteria used for the ISFSI important to safety components for normal operation is given in Tables 3.6-1 (NUHOMS-24P DSC) and 12.3-3 (NUHOMS-32P DSC). Summaries of design criteria for off-normal and accident conditions are given in Tables 3.6-2 and 3.6-3 (NUHOMS-24P), and Tables 12.3-4 and 12.3-5 (NUHOMS-32P DSC), respectively.

TABLE 3.6-1
NUHOMS-24P SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
Horizontal Storage Module	Dead Load	TR 8.1.1.5	Dead weight including loaded DSC	ANSI 57.9-1984 ACI 349-85 and ACI 349R-85
	Load Combination	TR Table 3.2-5	Load combination methodology	ANSI 57.9-1984 Sec 6.17.1.1
	Design Basis Operating Temperature	SAR 8.1.1	DSC with spent fuel rejecting 15.8 kW decay heat. Ambient air temperature range -3°F to 103°F	ANSI 57.9-1984
	Normal Handling Loads	TR 8.1.1.4	Hydraulic ram load: 20,000 lb (25% loaded DSC weight)	ANSI 57.9-1984
	Snow and Ice Loads	TR 3.2.4	Design load: 200 psf (included in live load)	ANSI 57.9-1984
	Live Loads	TR 8.1.1.5	Design load: 200 psf	ANSI 57.9-1984
	Shielding	SAR 7.1.2	Average contact dose rate on HSM exterior surface < 20 mrem/hr.	ANSI 57.9-1984
Dry Shielded Canister	Dead Loads	SAR 8.1.1	Weight of loaded DSC: 65,000 lb nominal, 80,000 lb enveloping	ANSI 57.9-1984
	Design Basis Internal Pressure Load	SAR 8.2.7.2	DSC internal pressure 9.6 psig	ANSI 57.9-1984
	Structural Design	TR Table 3.2-6	Service Level A and B	ASME B&PV Code Sec III, Div 1, NB, Class 1
	Design Basis Operating Temperature Loads	SAR 8.1.1.1	DSC decay heat 15.8 kW. Ambient air temperature -3°F to 103°F	ANSI 57.9-1984
	Operational Handling	TR 8.1.1.1	Hydraulic ram load: 20,000 lb enveloping	ANSI 57.9-1984
	Criticality	TR 3.3.4	K_{eff} less than 0.95 K_{eff} less than 0.98 for optimum moderation	ANSI 57.9-1984

TABLE 3.6-1
NUHOMS-24P SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
DSC Support Assembly	Dead Loads	TR 8.1.1.4	Loaded DSC + self weight: 85,000 lb	ANSI 57.9-1984 AISC Code
	Operational Handling	TR 8.1.1.4	DSC reaction load with hydraulic ram load: 20,000 lb	ANSI 57.9-1984
Transfer Cask	Normal Operating Condition	TR Table 3.2-8	Service Level A and B	ASME B&PV Code Sec III, Div 1, Class 2, NC-3200
Structure: Shell, Rings, etc.	Dead Loads	TR 8.1.1.9	a) Vertical orientation, self weight + loaded DSC + water in cavity: 200,000 lb enveloping	ANSI 57.9-1984
			b) Horizontal orientation, self weight + loaded DSC on transfer skid: 193,000 lb nominal 200,000 lb enveloping	ANSI 57.9-1984
	Snow and Ice Loads	TR 3.2.4	External surface temperature of cask will preclude buildup of snow and ice loads when in use: 0 psf	10 CFR 72.122
	Design Basis Operating Temperature Loads	SAR 8.1.1.1	Loaded DSC rejecting 15.8 kW decay heat. Ambient air temperature range -3°F to 103°F	ANSI 57.9-1984
	Shielding	SAR 7.1.2	Contact dose rate less than 200 mrem/hr.	ANSI 57.9-1984
Transfer Cask Upper Trunnions	Operational Handling	TR 8.1.1.9	a) Upper lifting trunnions while in Auxiliary Building: i) Stress must be less than yield stress for 6 times critical load of 115,000 lb/trunnion nominal	ANSI N14.6-1978
		TR 8.1.1.9	ii) Stress must be less than ultimate stress for 10 times critical load	

TABLE 3.6-1
NUHOMS-24P SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
		TR App. C	b) Upper lifting trunnions for on-site transfer: i) Dead Load +/- 1g vertically ii) Dead Load +/- 1g axially iii) Dead Load +/- 1g laterally iv) Dead Load (+/- 1/2g vertically +/- 1/2g axially + 1/2g laterally)	ASME B&PV Code Sec III, Div 1, Class 2, NC-3200
Lower Trunnions	Operational Handling	TR 8.1.1.9	Lower support trunnions weight of loaded cask during downloading and transit to HSM	ASME B&PV Code Sec III, Div 1, Class 2, NC-3200
Shell	Operational Handling	TR 8.1.1.9	Hydraulic ram load due to friction of extracting loaded DSC: 20,000 lb enveloping	ANSI 57.9-1984
Bolts	Normal Operation	TR Table 3.2-9	Service levels A, B, and C Avg stress less than 2 S_m Max stress less than 3 S_m	ASME B&PV Code Section III, Div 1, Class 2, NC-3200

For more information see Reference 3.14.

B&PV Boiler & Pressure Vessel
SAR Safety Analysis Report
(ASME B&PV Code-1983, with Addenda up to 1985)

TABLE 3.6-2
NUHOMS-24P SUMMARY OF DESIGN PARAMETERS FOR OFF-NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
Horizontal Storage Module	Off-Normal Temperature	SAR 8.1.1.1	-3°F to 103°F ambient temperature	ANSI 57.9-1984
	Jammed Condition Handling	TR 8.1.2.1	Hydraulic ram load equal to 100% of DSC: 80,000 lb nominal	ANSI 57.9-1984
	Load Combination	TR Table 3.2-5	Load combination methodology	ANSI 57.9-1984 Sec 6.17.1.1
Dry Shielded Canister	Off-normal Temperature	SAR 8.1.1.1	-3°F to 103°F ambient temperature	ANSI 57.9-1984
	Off-normal Pressure	SAR 8.2.7.2	DSC internal pressure less than 9.6 psig	ANSI 57.9-1984
	Blowdown Pressure	SAR 8.1.1.1	DSC internal pressure: 40.0 psig	10 CFR 72.122(b)
	Jammed Condition Handling	TR 8.1.2.1	Hydraulic ram load equal to 80,000 lb nominal	ANSI 57.9-1984
	Structural Design Off-Normal Conditions	TR Table 3.2-6	Service Level C	ASME B&PV Code Sec III, Div 1, NB, Class 1
DSC Support	Jammed Handling Condition	TR 8.1.2.1	Hydraulic ram load: 80,000 lb nominal	ANSI 57.9-1984
	Load Combination	TR Table 8.2-11	Load combination methodology	ANSI 57.9-1984
Transfer Cask	Off-normal Temperature	SAR 8.1.1.1	-3°F to 103°F ambient temperature	ANSI 57.9-1984
	Jammed Condition Handling	TR 8.1.2.1	Hydraulic ram load: 80,000 lb nominal	ANSI 57.9-1984
	Structural Design Off-Normal Conditions	TR Table 3.2-8	Service Level C	ASME B&PV Code Sec III, Div 1, Class 2, NC-3200

TABLE 3.6-2
NUHOMS-24P SUMMARY OF DESIGN PARAMETERS FOR OFF-NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
	Bolts, Off-Normal Conditions	TR Table 3.2-9	Service Level C Avg stress less than 2 S_m Max stress less than 3 S_m	ASME B&PV Code Sec III, Div 1 Class 2, NC-3200

TABLE 3.6-3
NUHOMS-24P SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
Horizontal Storage Module	Design Basis Tornado	TR 3.2.1	Max velocity 360 mph Max wind pressure 304 psf	RG 1.76 ANSI 58.1 1982
	Load Combination	TR Table 3.2-5	Load Combination Methodology	ANSI 57.9-1984 Sec 6.17.1.1
	Design Basis Tornado Missiles	TR 3.2.1.2	Max velocity 126 mph Types: Automobile, 3,967 lb 8" diam shell, 276 lb 1" solid sphere	NUREG-0800 Sec 3.5.1.4
	Flood	SAR 2.4.1.2	Dry Site	
	Seismic	SAR 3.2.3	Horizontal ground acceleration 0.15g (both directions) Vertical ground acceleration 0.10g 7% critical damping	NRC RGs 1.60 and 1.61
	Accident Condition Temperature	SAR 8.2.7.2	HSM vents (inlet/outlet) blocked for 48 hrs or less. HSM inside surface temp: 391°F	ANSI 57.9-1984
	Fire	SAR 8.2.10	1 hour forest fire 65' from HSM	
	Explosions	SAR 8.2.11	Probability of liquefied natural gas spill affecting HSM $< 10^{-7}$	NUREG-0800 Section 2.2.3
Dry Shielded Canister	Accident Drop	TR 8.2.5	Equivalent static deceleration: 75g vertical end drop 75g horizontal side drop 25g corner drop with slap down (corresponds to an 80" drop height) Structural damping during drop: 10%	RG 1.61
	Flood	TR 3.2.2	Maximum water height: 50'	10 CFR 72.122(b)
	Seismic	SAR 8.2.3.2	Horizontal acceleration: 1.5g Vertical acceleration: 1.0g 3% critical damping	NRC RGs 1.60 and 1.61

TABLE 3.6-3
NUHOMS-24P SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
	Accident Internal Pressure (HSM vents blocked)	SAR 8.2.7.2	DSC internal pressure: 50 psig based on 100% fuel clad rupture and fill gas release, and ambient air temp. = 103°F. DSC shell temperature: 460°F Vent block time = 48 hours	10 CFR 72.122(b)
	Accident Conditions	SAR Table 3.2-3	Service Level C/D	ASME B&PV Code Sec III, Div 1, NB, Class 1
	Reflood Pressure	SAR 8.2	DSC internal pressure: 40.0 psig	10 CFR 72.122(i)
DSC Support Assembly	Seismic	SAR 8.2.3.2	DSC reaction loads: Horizontal acceleration: 0.61g Vertical acceleration: 0.39g 7% critical damping	NRC RGs 1.60 and 1.61
	Load Combination	TR Table 8.2-11	Load combination methodology	ANSI 57.9-1984 Sec 6.17.3.2.1
Transfer Cask	Design Basis Tornado	TR 3.2.1	Max wind velocity: 360 mph Max wind pressure: 397 psf	NRC RG 1.76, ANSI 58.1-1982
	Design Basis Tornado Missiles	TR 3.2.1	Automobile, 3967 lb 8" diameter shell, 276 lb	NUREG-0800 Sec 3.5.1.4
	Flood	TR 3.2.2	Cask use to be restricted by administrative controls	10 CFR 72.122
	Seismic	TR 3.2.3	Horizontal ground acceleration: 0.25g (both directions) Vertical acceleration: 0.17g 3% critical damping	NRC RGs 1.60 and 1.61

TABLE 3.6-3
NUHOMS-24P SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
	Accident Drop	TR 8.2.5	Equivalent static deceleration: 75g vertical end drop 75g horizontal side drop 25g corner drop with slapdown (corresponds to an 80" drop height)	10 CFR 72.122(b)
			Structural damping during drop 10%	RG 1.61
	Bolts, Accident Drop	TR Table 3.2-9	Service Level D	ASME B&PV Code Sec III, Div 1, Class 2, NC-3200
	Structural Design, Accident	TR Table 3.2-8	Service Level D	ASME B&PV Code Sec III, Div 1, Class 2, NC-3200
	Internal Pressure	--	Not applicable because DSC provides pressure boundary	10 CFR 72.122(b)

For more information see Reference 3.14.

3.7 REFERENCES

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3.2 J. C. Ryman, O. W. Herman, C. C. Webster, C. V. Parks, "Fuel Inventory and After Heat Power Studies of Uranium-Fuel Pressurized Water Reactor Fuel Assemblies Using the SAS2 and ORIGEN-S Modules of SCALE with an ENDF/B-V Updated Cross-Section Library," NUREG/CR-2397 (ORNL-CSD-90), U.S. Nuclear Regulatory Commission and Oak Ridge National Laboratory, 1980

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3.4 J. L. Russell, "Technical Support for Radiation Standards for High-Level Radioactive Waste Management," Contract No. 68-01-4470, Arthur D. Little, Inc., 1977

3.5 American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, American Nuclear Society, LaGrange Park, Illinois, 1979

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3.9 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket No. 50-317 and 50-318, Baltimore Gas and Electric Company

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3.13 SCALE-3: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, NUREG/CR-0200, Revision 3, U.S. Nuclear Regulatory Commission, 1984

3.14 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated December 20, 1990, Response to NRC's Comments on the Safety Analysis Report (SAR) for BGE's License Application for Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI)

3.15 Deleted

3.16 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated December 19, 1991, Response to NRC's Request for Additional Information Regarding BGE's Quality Assurance (QA) Program for Calvert Cliffs ISFSI

3.17 CCNPP Drawing 84-003-E, NUHOMS-24 ISFSI DSC Shell Assembly

3.18 CCNPP Drawing 84-004-E, NUHOMS-24P ISFSI DSC Shell Assembly, Sheet 1 of 2

3.19 CCNPP Drawing 84-006-E, NUHOMS-24P ISFSI DSC Main Assembly, Sheet 1 of 3

3.20 CCNPP Drawing 84-007-E, NUHOMS-24 ISFSI DSC Main Assembly, Sheet 2 of 3

3.21 CCNPP Drawing 84-001-E, NUHOMS-24P ISFSI DSC Basket Assembly

3.22 CCNPP Drawing 84-005-E, NUHOMS-24P ISFSI DSC Shell Assembly, Sheet 2 of 2

3.23 CCNPP Drawing 84-021-E, NUHOMS-24P ISFSI Onsite Transfer Cask Structural Shell

3.24 CCNPP Drawing 84-025-E, NUHOMS-24P ISFSI Onsite Transfer Cask Inner and Outer Shell Assembly

3.25 CCNPP Drawing 84-026-E, NUHOMS-24P ISFSI Onsite Transfer Cask Inner and Outer Shell Assembly

3.26 CCNPP Drawing 84-027-E, NUHOMS-24P ISFSI Onsite Transfer Cask Main Assembly

3.27 CCNPP Drawing 84-028-E, NUHOMS-24P ISFSI Onsite Transfer Cask Main Assembly

3.28 CCNPP Drawing 84-030-E, NUHOMS-24P ISFSI Onsite Transfer Cask Accessories

3.29 CCNPP Drawing 84-036-E, ISFSI Cask Lifting Yoke Assembly

3.30 CCNPP Drawing 84-022-E, NUHOMS-24P ISFSI Onsite Transfer Cask Structural Shell Assembly

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3.38 T. L. Sanders, et al, "Feasibility and Incentives for the Consideration of Spent Fuel Operating Histories in the Criticality Analysis of Spent Fuel Shipping Casks," SAND87-0151, August 1987

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3.43 Letter from T. J. Kobetz (NRC) to V. Franceschi (Vectra), dated February 25, 1997, Clarification of Item 3 of Confirmatory Action letter dated July 7, 1995

3.44 CCNPP Calculation No. CA06329, NUHOMS-32P – Transfer Cask Structural Analysis

3.45 ANSI 14.6 – American National Standard for Radioactive Materials, “Special Lifting Devices for Shipping Containers Weighting 10,000 Pounds (4500 kg) or More”

3.46 CCNPP Calculation No. CA03947, Dry Storage DSC Internal Pressure

3.47 CCNPP Calculation No. CA06300, Maximum Operating Pressure, Storage and Transfer