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10 CFR 72.56

July 25, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Independent Spent Fuel Storage Installation
License No. SNM-2505
NRC Docket No. 72-8

Subject: Request for Supplemental Information – Amendment Request No. 11 to Materials License No. SNM-2505 for the Calvert Cliffs Specific Independent Spent Fuel Storage Installation – Acceptance Review

References:
1. Letter from G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated March 26, 2014, License Amendment Request: High Burnup NUHOMS®-32PHB Dry Shielded Canister
2. Letter from J. M. Goshen (NRC) to G. H. Gellrich, dated June 23, 2014, Amendment Request No. 11 to Materials License No. SNM-2505 for the Calvert Cliffs Specific Independent Spent Fuel Storage Installation – Acceptance Review - Request for Supplemental Information (TAC No. L24912)

Reference 1 submitted a license amendment request for the Calvert Cliffs Nuclear Power Plant site-specific independent spent fuel storage installation. The amendment, if approved, would authorize the storage of Westinghouse and Areva Combustion Engineering 14X14 fuel in the NUHOMS®-32PHB Dry Shielded Canister system.

The Nuclear Regulatory Commission staff performed an acceptance review of the application to determine if the application contained sufficient technical information in scope and depth to allow the staff to complete the detailed technical review. The Nuclear Regulatory Commission staff has requested supplemental information to complete the acceptance review (Reference 2). Responses to the requested information are contained in Attachment (1). Some of the

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information requested includes calculations from Transnuclear, Inc. Some calculations are proprietary to Transnuclear, Inc., therefore, they are accompanied by an affidavit signed by Transnuclear, the owner of the information (Attachment 2). The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission, and address, with specificity, the considerations listed in 10 CFR 2.390(b)(4). Accordingly, it is requested that the information that is proprietary to Transnuclear, Inc. be withheld from public disclosure. Non-proprietary versions of the proprietary information are not available at this time.

The supplemental information does not change the environmental assessment provided in Reference 1 and the categorical exclusion set forth in 10 CFR 51.22(c)(11) is still valid. There are no regulatory commitments contained in this correspondence.

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 25, 2014.

Respectfully,

George Gellrich

George H. Gellrich
Site Vice President

GHG/PSF/bjd

Attachments: (1) Request for Supplemental Information
(2) Affidavit

Enclosures: 1. Draft Revision XX of the ISFSI USAR
2. Drawings (11x17)
3. ANSYS File List
4.–26. Requested Calculations

Proprietary calculations are noted on the cover page of the Enclosure.

cc: **(Without Enclosures)**
NRC Project Manager, Calvert Cliffs
NRC Regional Administrator, Region I
NRC NMSS Project Manager, Calvert Cliffs
NRC Resident Inspector, Calvert Cliffs
S. Gray, MD-DNR

ATTACHMENT (1)

REQUEST FOR SUPPLEMENTAL INFORMATION

Reference 1 submitted a license amendment request (LAR) for the Calvert Cliffs Nuclear Power Plant (CCNPP) site-specific independent spent fuel storage installation (ISFSI). The amendment, if approved, would authorize the storage of Westinghouse and Areva Combustion Engineering 14X14 fuel in the NUHOMS®-32PHB Dry Shielded Canister (DSC) system.

The Nuclear Regulatory Commission (NRC) staff performed an acceptance review of the application to determine if the application contained sufficient technical information in scope and depth to allow the staff to complete the detailed technical review. The NRC staff has requested supplemental information to complete the acceptance review. In addition, the NRC has provided three observations that do not require a response in order for the NRC to complete it's acceptance review. However, responses to the observations have also been provided. Responses to the requested information and observations are provided below.

References

1. Letter from G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated March 26, 2014, License Amendment Request: High Burnup NUHOMS®-32PHB Dry Shielded Canister
2. Letter from G. H. Gellrich (CCNPP) to Document Control Desk, dated September 17, 2010, Site-Specific Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application
3. Letter from G. H. Gellrich (CCNPP) to Document Control Desk, dated June 14, 2013, Response to Request for Additional Information, RE: Calvert Cliffs Independent Spent Fuel Storage Installation License Renewal Application (TAC No. L24475)

Chapter 5 Structural Evaluation

NRC Question Chapter 5-1:

5-1 Provide the List of Effective Pages (LOEP) and all revision pages for Revision 22 to the Calvert Cliffs ISFSI Updated Safety Analysis Report.

The staff notes that Section 13.0 of the Calvert Cliffs ISFSI USAR, Revision 22 Draft A is not the entire revision. Section 13.0 states that "Chapter 1 is revised to include information for the NUHOMS-24P, NUHOMS-32P and the NUHOMS 32PHB DSCs." Based on Amendment No. 6, there should be changes to Chapters 1-11 as a result of the addition of the NUHOMS-32PHB; therefore, the entire revision is needed for evaluation.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 5-1:

The draft List of Effective Pages and all draft Revision XX pages for the Calvert Cliffs ISFSI Updated Safety Analysis Report (USAR) are provided in Enclosure 1. Note that the proposed revision number that was used in Reference 1 has been removed and replaced with XX.

Updated Safety Analysis Report updates are submitted to the NRC on at least a biennial basis as required by 10 CFR 72.70(c)(6). The next USAR update (Revision 22) will be submitted on or before September 18, 2014 to comply with the biennial update requirement. This update will include all changes made to the ISFSI licensing basis since the last update in September 2012. It will not reflect the changes needed to implement this amendment, since they are not yet approved. The draft Revision XX that is included as Enclosure 1 includes the changes necessary to implement this LAR as well as the Revision 22 changes that will be submitted by September 18, 2014.

NRC Question Chapter 5-2:

5-2 *Provide a one-to-one mapping of the elements of the NUHOMS-24P, -32P, and -32PTH DSCs (including basket and fuel) to those of the NUHOMS-32PHB DSC that are being cited in lieu of analysis along with the justification and the section of the applicable USAR, FSAR, TSAR, and CoC with revision, as appropriate.*

The staff notes that there are similarities between the proposed NUHOMS-32PHB and previous models associated with Calvert Cliffs Nuclear Power Plant, CoC-1004, and CoC-1030. It is unclear how the licensee is comparing the proposed design with those of previously approved models. For example, Section 2.0 of Attachment (1) compares the NUHOMS-32PHB to the NUHOMS-32P and lists several design differences. The Comparison Matrix of Enclosure (1) shows that the cladding material is Zircaloy-4 for the NUHOMS-32P and M5 for the NUHOMS-32PHB and the NUHOMS-32PTH, leaving the impression that the licensee intends to use the NUHOMS-32PTH as the basis for the structural analysis of the cladding for the NUHOMS-32PHB.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 5-2:

The matrix provided as Enclosure 1 in Reference 1 was submitted to compare the design elements of the NUHOMS-32PHB DSC the high burnup horizontal storage module (HSM-HB) and the transfer cask to previously approved models for illustrative purposes only. The proposed elements in Reference 1 are stand-alone designs, and as such have been analyzed in accordance with applicable regulatory requirements and guidance. The primary calculations for these design elements are listed in Section 13.13 of the draft ISFSI USAR (provided in an Enclosure to Reference 1). These design calculations have been submitted or are being submitted in the Enclosures to this letter, as tabulated below.

Calculation Referenced in Chapter 13	Title	Location
NUH32PHB-0101	DESIGN CRITERIA DOCUMENT (DCD) FOR NUHOMS 32PHB SYSTEM FOR STORAGE	Enclosure 4
NUH32PHB-0111	DESIGN REPORT FOR 32PHB DSC	Enclosure 5
NUH32PHB-0201	NUHOMS 32PHB WEIGHT CALCULATION OF DSC/TC SYSTEM	ML12173A186
NUH32PHB-0204	NUHOMS 32PHB CANISTER STRUTURAL EVALUATION FOR STORAGE AND ONSITE TRANSFER LOADS	Enclosure 6
NUH32PHB-0205	NUHOMS 32PHB BASKET EVALUATION FOR STORAGE AND TRANSFER LOADS	Enclosure 7
NUH32PHB-0206	NUHOMS 32PHB TRANSFER CASK-LOCAL SHELL STRESSES AT TRUNNION LOCATIONS	Enclosure 8
NUH32PHB-0207	FUEL END DROP ANALYSIS FOR NUHOMS 32PHB USING LS-DYNA	Enclosure 9
NUH32PHB-0208	HSM-HB STRUCTURAL ANALYSIS FOR NUHOMS 32PHB SYSTEM	Enclosure 10

Calculation Referenced in Chapter 13	Title	Location
NUH32PHB-0209	CCNPP-FC TRANSFER CASK IMPACT ONTO THE CONCRETE PAD, LS-DYNA ANALYSIS (80 INCH SIDE, CORNER, AND END DROPS)	Enclosure 11
NUH32PHB-0210	NUHOMS 32PHB CANISTER, BASKET AND FUEL ASSEMBLIES DYNAMIC LOAD FACTORS	Enclosure 12
NUH32PHB-0211	RECONCILIATION FOR TRANSFER CASK CCNPP-FC STRUCTURAL EVALUATION	Enclosure 13
NUH32PHB-0212	CCNPP-FC TRANSFER CASK STRUCTURAL EVALUATION – ACCIDENT CONDITIONS, 75G SIDE DROP AND 75G TOP END DROP CASES	Enclosure 14
NUH32PHB-0213	NUHOMS 32PHB LIFTING LUG ANALYSIS	Enclosure 15
NUH32PHB-0214	NUHOMS 32PHB RECONCILIATION FOR CIVIL STRUCTURES	Enclosure 16
NUH32PHB-0400	BENCHMARKING OF ANSYS MODEL OF THE OS200FC TRANSFER CASK	Enclosure 17
NUH32PHB-0401	THERMAL EVALUATION OF NUHOMS 32PHB TRANSFER CASK FOR NORMAL, OFF NORMAL AND ACCIDENT CONDITIONS WITH FORCED COOLING (STEADY STATE)	ML12093A107
NUH32PHB-0402	THERMAL EVALUATION-NUHOMS 32PHB TRANSFER CASK FOR NORMAL, OFF NORMAL AND ACCIDENT CONDITIONS	ML12093A104
NUH32PHB-0403	THERMAL EVALUATION FOR NUHOMS 32PHB CANISTER FOR STORAGE AND TRANSFER CONDITIONS	ML12173A182
NUH32PHB-0404	INTERNAL PRESSURE FOR NUHOMS 32PHB DSC FOR STORAGE AND TRANSFER CONDITIONS	Enclosure 18
NUH32PHB-0405	THERMAL EXPANSION OF THE NUHOMS-32PHB SYSTEM FOR TRANSFER AND STORAGE CONDITIONS	Enclosure 19
NUH32PHB-0406	THERMAL EVALUATION-NUHOMS 32PHB TRANSFER CASK FOR NORMAL, OFF- NORMAL AND ACCIDENT CONDITIONS (HEAT LOAD <29.6KW)	ML12093A103
NUH32PHB-0407	EFFECTIVE THERMAL PROPERTIES OF BOUNDING CE 14X14 FUEL ASSEMBLY FOR 32PHB DSC	Enclosure 20
NUH32PHB-0408	THERMAL ANALYSIS OF NUHOMS 32PHB DSC FOR VACUUM DRYING OPERATIONS	ML12093A102
NUH32PHB-0409	FOREST FIRE THERMAL EVALUATION FOR CCNPP ISFSI	Enclosure 21
NUH32PHB-0410	RECONCILIATION OF THERMAL ANALYSIS RESULTS FOR 32PHB DSC STORAGE IN HSM-HB MODULE	Enclosure 22

Calculation Referenced in Chapter 13	Title	Location
NUH32PHB-0502	CALVERT CLIFFS NUHOMS 32PHB RADIATION DOSE RATES FOR LOADING AND TRANSFER	Enclosure 23
NUH32PHB-0503	HSM-H SHIELDING ANALYSIS FOR 32PHB SYSTEM	Enclosure 24
NUH32PHB-0505	SITE DOSE ANALYSIS FOR NUHOMS 32PHB SYSTEM	Enclosure 25
NUH32PHB-0600	CRITICALITY EVALUATION FOR NUHOMS 32PHB SYSTEM	Enclosure 4 of Reference 1
NUH32PHB-0603	USL EVALUATION FOR NUHOMS 23PHB SYSTEM	Enclosure 5 of Reference 1
11562-019-ST-05	ISFSI HSM-HB PAD AND APPROACH SLAB DESIGN	Enclosure 26

NRC Question Chapter 5-3:

5-3 *Provide a one-to-one mapping of the elements of the CCNPP HSM, HSM-H and the NUHOMS-HD to those of CCNPP HSM-HB that are being cited in lieu of analysis along with the justification and the section of the applicable USAR, FSAR, TSAR, and CoC with revision, as appropriate.*

The staff notes that there are similarities between the proposed CCNPP HSM-HB and previous models associated with Calvert Cliffs Nuclear Power Plant, CoC-1004, and CoC-1030. It is unclear how the licensee is comparing the proposed design with those of previously approved models. For example, Section 2.0 of Attachment (1) compares the HSM-HB module to the HSM modules used to house the NUHOMS-24P and -32P and lists several design differences. The Comparison Matrix of Enclosure (1) lists dissimilar design parameters for Tornado Missile loading for the HSM-HB and the HSM modules. Similar design parameters are listed for the HSM-HB, HSM-H and HSM-HD modules, indicating that the licensee intends to use either the HSM-H or HSM-HD modules as the basis for the Tornado Missile loading analysis for the HSM-HD module.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 5-3:

See Response to Question 5-2.

NRC Question Chapter 5-4:

5-4 *Provide a one-to-one mapping of the elements of the CCNPP TC and the NUHOMS OS197FC TC to those of CCNPP FC-TC that are being cited in lieu of analysis along with the justification and the section of the applicable USAR, FSAR, TSAR and CoC with revision, as appropriate.*

The staff notes that there are similarities between the proposed CCNPP FC-TC and previous models associated with Calvert Cliffs Nuclear Power Plant, CoC-1004 and CoC-1030. It is unclear how the licensee is comparing the proposed design with those of previously approved models. For example, Attachment (1) states that the same transfer cask is used for the NUHOMS-32PHB, -24P and -32P DSCs, but the forced-cooling configuration will be used when handling the NUHOMS-32PHB. The Comparison Matrix *of Enclosure (1) lists dissimilar design parameters for Tornado Missile loading for the CCNPP FC-TC and the CCNPP TC. Similar design parameters are listed for the CCNPP FC-TC and the OS197FC TC (NUHOMS-32PTH DSC), indicating that the licensee intends to use the OS197FC TC as the basis for the Tornado Missile loading analysis for the CCNPP FC-TC.*

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 5-4:

See Response to Question 5-2.

NRC Question Chapter 5-5:

5-5 *Provide the structural calculations (-02XX) referenced in Section 13.13 of Enclosure 2 of Attachment (1), Draft USAR Chapter for NUHOMS-32PHB DSC.*

These references are for the NUHOMS-32PHB, the associated HSM-HB and the transfer cask in forced circulation and are required to conduct a review of the amendment request.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 5-5:

The calculations are listed in the Response to Question 5-2 and are enclosed with this letter.

NRC Question Chapter 5-6:

5-6 *Provide the safety analysis for the ISFSI pad(s) expanding the previously analyzed total allowable deployment number of 120 in the Calvert Cliffs ISFSI USAR to the proposed 132 HSMs.*

The staff noted that a detailed configuration of the new 1X12 array pad that is required for a license amendment is not included in the application.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 5-6:

The analysis of the ISFSI pads is enclosed as listed in the Response to Question 5-2.

NRC Question Chapter 5-7:

5-7 *Provide a new Enclosure (3) to Attachment (1) with readable drawings.*

The staff notes that the drawing package of Enclosure (3) is in accordance with Regulatory Guide 3.48, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage); however, many of the details or the drawings are illegible.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question Chapter 5-7:

Larger drawings (11x17) are attached in Enclosure 2.

Chapter 6 Thermal Evaluation

<u>*NRC Question Chapter 6-1*</u>*:*

6-1 Provide the information identified in NUREG-1567, Sections 6.4 and 6.5. The application requests the staff's approval of the use of new HSM-HBs and NUHOMS-32PHB DSCs. However, details of the HSM-HB and NUHOMS-3PHB DSCs design and analysis were not provided in the application. Alternatively, provide all calculation packages referenced in the application and revise the comparison matrix to cross-reference the NUREG-1567 acceptance criteria with specific Sections, Tables, Figures, and/or Appendices of the application, ISFSI USAR, and/or calculations packages. It may be necessary to also cross-reference portions of the NUREG-1536 (as referenced in NUREG -1567) acceptance criteria based on the new canister design and HSM proposed below.

The application should include sufficient information to ensure that a thermal evaluation can be completed. Without the information identified in NUREG-1567, Sections 6.4 and 6.5, the staff cannot perform a complete thermal review for the following requests to:

a. Approve a new NUHOMS®-32PHB canister design and a new HSM-HB) for use at the Calvert Cliffs ISFSI

b. Expansion of the ISFSI total capacity from 120 horizontal storage modules (HSMs) to 132 HSMs on the existing site

c. Approve the modified or new Technical Specifications (TS): 3.1.1(5), 3.3.2.1, 3.3.3.1, and 3.4.1.1.

For example, Sections 6.4.1, 6.4.2, 6.4.3, 6.4.4, 6.4.5, 6.5.1, 6.5.2, 6.5.3, 6.5.4 and 6.5.5 are relevant considering that the amendment is based on: 1. new fuel content (high burnup fuel) with increased maximum fuel assembly heat loads, 2. a new dry shielded canister, 3. a new modular high burnup horizontal storage module, 4. an increased number of horizontal storage modules, 5. Modified or new TS: 3.1.1(5), 3.3.2.1, 3.3.3.1, and 3.4.1.1.

This information is needed to demonstrate compliance with 10 CFR 72.24 and 10 CFR 72.56.

<u>CCNPP Response to Question 6-1</u>:

The following sections provide the requested information, or pointers to the calculations that provide that information specified in NUREG-1567, Sections 6.4 and 6.5. This information may be used to supplement the comparison matrix provided in Enclosure 1 of Reference 1. Calculations referenced in the matrix and in Section 13.13 of the draft USAR are provided as enclosures to this letter (see the Response to Question 5-2).

Sections 6.4.1 and 6.5.1 Decay Heat Removal Systems

In addition to the description of the NUHOMS-32PHB DSC, HSM-HB, and Forced Cooling modification to the CCNPP Transfer Cask provided in Attachment (1) Section 2 of Reference 1, the proposed additional description of these systems in the ISFSI USAR Section 1.3.1 is provided as described in the response to Question 5-1.

In general, the decay heat removal systems for storage of the NUHOMS-32PHB DSC functions in the same manner as the existing NUHOMS-32P DSC system employed at Calvert Cliffs ISFSI.

The primary change to the decay heat removal system for loading and transfer of the DSC is the introduction of time limits on some steps for DSC heat loads higher than the current licensed heat load of 21.12 kW. These steps are 1) the time to complete the NUHOMS-32PHB DSC blowdown and vacuum drying process if nitrogen is used for blowdown (new TS 3.3.3.1), and 2) the transfer of the DSC from the cask handling area to the HSM (new TS 3.3.2.1). These are discussed in further detail in Section 3.L and 3.K respectively of Reference 1. In both cases, these time limits are introduced primarily to ensure that the fuel cladding temperature limits for normal operation discussed below are not exceeded. In the case of the latter time limit, a forced air cooling mode will be added to the site Transfer Cask, as described in Section 2 and the draft USAR Section 13.3.2.5.3 provided in Enclosure 2 of Reference 1, to support the required action of new TS 3.3.2.1 to maintain fuel cladding temperatures within the short-term limit if the transfer times will exceed those specified in that TS.

Sections 6.4.2 and 6.5.2 Material Temperature Limits

1. General

   In addition to temperature limits for the fuel cladding and concrete discussed below, temperature limits are also specified for the lead gamma shielding in the NUHOMS-32PHB DSC and Transfer Cask, and NS-3 solid neutron shielding in the Transfer Cask. The maximum lead temperature limit for normal, off-normal and accident conditions is 620°F. This is also the design basis lead temperature limit used for the Calvert Cliffs NUHOMS-24P and NUHOMS-32P DSCs and is based on the melting point of American Society for Testing and Materials B29 lead. The long-term, bulk average temperature limit for the NS-3 used in evaluation of normal/off-normal transfer events for the NUHOMS-32PHB DSC is 280°F. This is also the same as the current design basis limit for the Calvert Cliffs NUHOMS-24P and NUHOMS-32P DSCs in the Transfer Cask described in ISFSI USAR Section 8.1.3.3. For the NUHOMS-32PHB DSC, a short-term NS-3 temperature limit for accident conditions of 1,300°F has been introduced.

2. Fuel Cladding

   Fuel cladding temperature limits for normal, off-normal, and accident conditions were discussed in Reference 1 in Attachment (1), Sections 3.F, 3.K, 3.L, and 3.M. Specifically, the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad), and should not exceed 570°C (1058°F) for off-normal and accident conditions. The basis for these temperature limits is NRC Interim Staff Guidance No. 11 Revision 3, which provides acceptance criteria for the storage of spent fuel assemblies with average burnups exceeding 45 GWd/MTU.

3. Concrete

   A concrete surface temperature limit of 350°F is used to determine the depth of concrete spalling during a forest fire accident. This limit is based on American Concrete Institute 349-85 limit for short-term concrete surface temperatures for safety-related nuclear structures, and is the same as the current design basis limit for this event for storage of NUHOMS-24P and NUHOMS-32P DSCs.

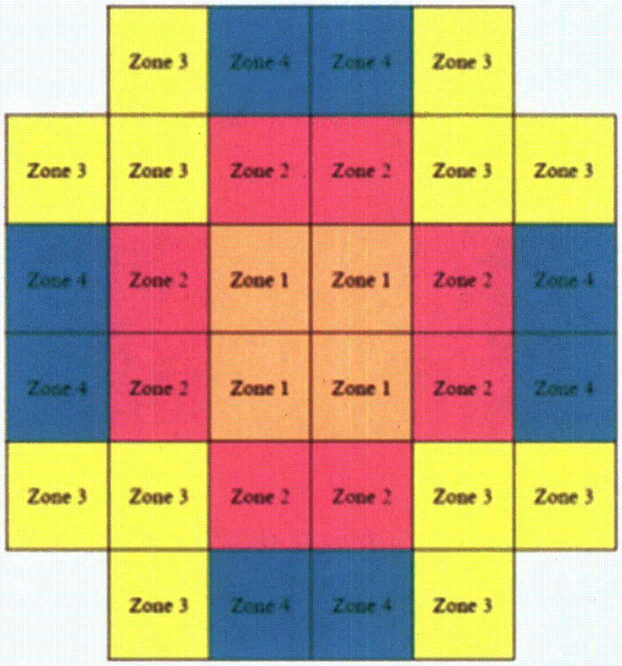
   For the blocked vent accident, a design basis local maximum concrete temperature limit of 395°F has traditionally been used for NUHOMS-24P and NUHOMS-32P DSC storage at the Calvert Cliffs ISFSI. As shown on page 7 of Enclosure 1 of Reference 1, the local maximum concrete temperature in a 36-hour blocked vent accident is predicted to exceed this value for storage of the highest heat load NUHOMS-32PHB DSC in an HSM-HB. Qualification testing of the HSM-HB concrete will be performed to verify that there are no significant signs of spalling or cracking and that the concrete compressive strength is greater than that assumed in the structural analysis (NUH32PHB-0208). The test will be performed at or above the calculated peak temperature and for a period no less than the 36-hour duration of the blocked vent accident. The option to perform such qualification testing is similar to License Condition 6 for the HSM-H under Amendment 10 of CoC 1004.

4. Extreme Low Temperatures

The ISFSI USAR Table 1.2-1 and Section 2.3.1.3.1 establishes an off-normal minimum design basis ambient temperature for the Calvert Cliffs ISFSI of -3°F. Structural analyses of the HSM-HB concrete and steel components documented in calculation NUH32PHB-0208 demonstrate stresses are within allowable values for off-normal ambient temperatures as low as -40°F, which bounds the Calvert Cliffs site-specific value. As indicated in Reference 1, Enclosure 2 Section 13.8.1.3.B, the off-normal minimum ambient temperature considered in thermal evaluations of the NUHOMS-32PHB DSC that are detailed in calculation NUH32PHB-0403 and NUHOMS-32PHB-0410 is -8°F, which also bounds the Calvert Cliffs site-specific value.

Sections 6.4.3 and 6.5.3 Thermal Loads and Environmental Conditions

The design basis thermal loads being requested for approval for the NUHOMS-32PHB DSC were discussed in Sections 3.G, 3.K and 3.L of Attachment (1) of Reference 1. These sections detail four possible TS thermal configurations for the NUHOMS-32PHB DSC, which are summarized in the figure and table below. In general, thermal analyses used to demonstrate compliance with material temperature limits are performed only for the bounding thermal load. The exception is thermal analyses which establish the time limits for transfer or vacuum drying, which consider the lower thermal load configurations.



Heat Zone	No. of FA	Configuration 1 (kW/FA)	Configuration 2 (kW/FA)	Configuration 3 (kW/FA)	Configuration 4 (kW/FA)
1	4	0.8	0.8	0.72	0.66
2	8	1.0	0.8	0.72	0.66
3	12	1.0	0.8	0.72	0.66
4	8	0.8	0.8	0.72	0.66
Total Heat Load, kW		29.6	25.6	23.04	21.12
Vacuum Drying Time Limit (h) (N_2 Blow Down Only)		32	40	56	56*
Transfer Time Limit (h)		20	48	72	∞

* Note that this time limit is highly conservative for configuration 4 since steady state analyses for the NUHOMS-32P DSC detailed in ML091680551 support an ∞ vacuum drying time limit.

In addition, the proposed draft additions to ISFSI USAR Table 9.4-1 detailing the required fuel assembly cooling times to reach the indicated heat loads have now also been provided with the response to Question 5-1. The NUHOMS-32PHB DSC cooling times are based on thermal and radiological source term calculations performed using the SAS2H/ORIGEN-S modules of the SCALE 4.4a computer code package, which is the same method approved by the NRC for the NUHOMS-32P DSC (see ML091680542 and ML102571637).

The range of ambient air temperatures used to evaluate the design of the NUHOMS-32PHB DSC, its transfer in the Transfer Cask as modified for forced cooling, and storage in the HSM-HB module are -8 to 104°F, which are bounding for those described in the ISFSI USAR Section 2.3. Similarly, solar heat flux of 127 Btu/hr-ft^2 maximum and 82 Btu/hr-ft^2 average utilized for evaluation of the transfer and storage of the NUHOMS-32PHB DSC at the ISFSI are the same as the current design basis for the ISFSI described in USAR Table 1.2-1. This information was also previously provided in Reference 1 on Table 8 in Section 3.K, on Page 6 of Enclosure 1 and in Sections 13.3.3.7, 13.8.1.3, 13.8.2.9.2, and Table 13.3-3 of Enclosure 2.

Sections 6.4.4 and 6.5.4 Analytical Methods, Models, and Calculations

Analytical methods and models for thermal evaluation of the NUHOMS-32PHB DSC, its transfer in the Transfer Cask as modified for forced cooling, and storage in the HSM-HB module are detailed in the NUH32PHB-04xx series calculations. See Response to Question Response 5-1 for copies of these calculations. The thermal calculations were performed using Version 10.0 of the ANSYS code, and copies of input and output files are provided for the Response to Question 6-4.

Sections 6.4.5 and 6.5.5 Fire and Explosion Protection

As discussed in Table 13.3-5 and Section 13.3.3.6 of Enclosure 2 of Reference 1, the discussion presented in ISFSI USAR Section 3.3.6 is applicable to storage and transfer of the NUHOMS-32PHB DSC.

For the transfer fire accident analysis, a diesel fuel pool of 190" in diameter, which is the approximate length of the Transfer Cask, is conservatively assumed to engulf the entire cask. A maximum fuel spill of 100 gallons of diesel fuel which is the maximum capacity of both fuel tanks within the tow vehicle is considered in Section 3.3.6 of ISFSI USAR. For this postulated fire accident with a conservative volume of 200 gallons of diesel fuel spill, the thickness of the fuel pool would be 1.63". This pool is assumed to burn at a minimum burning rate of 0.15 in/min. The 1.63" thick fuel pool would burn for 11 minutes. For conservatism a 15-minute fire duration is considered. Maximum component temperatures from this fire are determined in calculations NUH32PHB-0402 and NUH32PHB-0403 for the Transfer Cask and NUHOMS-32PHB DSC, respectively. Section 13.8.2.9.2 in Enclosure 2 of Reference 1 summarizes the maximum pressure evaluation results from calculation NUH32PHB-0404.

The Calvert Cliffs site-specific forest fire evaluation is discussed in Section 13.8.2.10 of Enclosure 2 of Reference 1. Additional detail on the forest fire accident thermal evaluation can be obtained from calculation NUH32PHB-0409. Section 13.8.2.11 of Enclosure 2 of Reference 1 also addresses the site-specific LNG plant fire/explosion evaluation.

Finally, the Aircraft Hazards Assessment discussed in ISFSI USAR Section 2.2.1 was also reviewed. The analysis shows 55,448 square feet was assumed for this analysis (116' x 478'), which is the area of the 120 HSMs plus the approach slabs in front of them. The expansion to 132 HSMs remains within the same foot print utilized to determine the above area. The risk of aircraft impact for the ISFSI of 5.8 E-8 per year considers only the probability of aircraft impact. Therefore conclusions of the aircraft hazards analysis continue to apply.

<u>*NRC Question Chapter 6-2:*</u>

6-2 Clearly indicate in the application which, if any, referenced documents are incorporated by reference. Also provide all documents considered to be incorporated by reference.

The application states, "A referenced document shall be considered to be a part of the USAR only if it is clearly annotated as being "incorporated by reference" in Chapter 13 of this report. Documents that are incorporated by reference are subject to the same administrative controls and regulatory requirements as the USAR."

The staff did not find any referenced documents that were incorporated by reference and therefore did not review any referenced documents as part of the application.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 6-2:

The CCNPP USAR was developed in accordance with Regulatory Guide 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)," Revision 1, August 1989. The Regulatory Guide provides guidance for information that should be incorporated by reference. That guidance (section 1.5) indicates that topical reports, submitted separately to the NRC by other organizations, are appropriate material for incorporation by reference. In addition, documents submitted in other applications may be incorporated by reference if appropriate.

As noted in the CCNPP USAR, Section 1.5, two topical reports addressing the design of the 24P DSC have been incorporated by reference. Since the time of initial licensing of the ISFSI, no other documents have been incorporated by reference. Because the ISFSI is licensed under a site-specific license, topical reports do not address CCNPPs site-specific designs. Each design is evaluated based on previous designs licensed for the CCNPP site and site-specific calculations are performed to demonstrate the safety of the design. The results of these calculations are presented in the USAR, in keeping with the level of detail described in Regulatory Guide 3.48. The calculations supporting an ISFSI Technical Specification change request have often been provided to the NRC to support their review. However, the actual calculation is not incorporated by reference into the ISFSI USAR. Important assumptions, analysis methods and results are provided in appropriate USAR chapters. The calculation is referenced at the end of the chapter to make retrieval easier for the design organization.

In the initial submittal and this supplement numerous design calculations have been provided to support the NRC review. These calculations are not incorporated by reference in the USAR. They are described in appropriate detail in the corresponding sections of the USAR. The calculations are controlled at CCNPP in accordance with the same design processes used to control design calculations for the CCNPP Part 50 licensed power plant. If additional calculations are needed to support the review, they will be provided.

NRC Question Chapter 6-3:

6-3 Clarify if the application requires the evaluation of upgrading portions of the Calvert Cliffs ISFSI.

A previous submittal of this application requested to: "Upgrade portions of the Calvert Cliffs ISFSI to allow use of the prefabricated high burnup horizontal storage modules (HSM-HB) for future expansion." It is not clear from the application if this is still being requested.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 6-3:

In this request we are asking for approval to store the 32PHB DSC in HSM-HBs. Twenty four HSM-HBs have been constructed on site to prepare for storage of the 32P and 32PHB DSCs. Calculations have been and are being provided to the NRC staff that address the design aspects (e.g., thermal, weight, etc.) of the HSM-HBs that are impacted by the design of the 32PHB DSC.

NRC Question Chapter 6-4:

6-4 *Provide ANSYS input and output files to support the thermal evaluation and the licensing requests.*

The staff cannot begin to perform a thermal acceptance review, or a complete thermal review without ANSYS input and output files that support the requests summarized in RSI 1. The staff specifically prefers text based files (i.e. .inp) with an appropriate level of comments to allow for a timely technical review.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 6-4:

The requested ANSYS files are listed in Enclosure 3.

NRC Question Chapter 6-5:

6-5 *Revise the application to justify the design similarity claims.*

The application frequently refers to how the NUHOMS-32PHB canister design and the HSM-HB are similar to previous designs, but it is not clear what previously submitted documentation the staff should refer to determine if the similarity claims are justified. The applicant needs to identify the similarities of these two designs and demonstrate that the analyses the application references to are applicable to the new design.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 6-5:

The intent of discussing similarities between the 32PHB DSC and HSM-HB designs and other approved designs was to show the NRC staff that the proposed designs were similar to existing approved designs and were not completely new designs. Independent calculations for the design elements of the 32PHB DSC, the HSM-H and the transfer cask have been performed. Those are stand-alone designs. It was not the intent of the application to cite other designs or standards in lieu of analysis. We apologize for any misunderstanding about the intent of the comparison with existing approved designs. The primary analyses used for the design are listed in Section 13.13 of the draft ISFSI USAR provided in Reference 1. Those calculations are listed in the Response to Question 5-2.

NRC Question Chapter 6-6:

6-6 *Provide legible licensing drawings.*

The staff cannot read the licensing drawings. Because the licensing drawings were not legible, the staff could not review them for the acceptance review.

This information is needed to demonstrate compliance with 10 CFR 72.24.

CCNPP Response to Question 6-6:

Larger drawings (11x17) are provided in Enclosure (2).

Chapter 9 Confinement Evaluation

NRC Question Chapter 9-1:

9-1 Provide readable drawings so that a review and understanding of the DSC can be performed.

a) Enclosure 3 of the application included 13 drawings that were difficult to read. Provide larger drawings or drawings with higher resolution.

b) Some of the drawings in Enclosure 3 were not included. For example, NUH32PHB-30-1 indicates there are 2 sheets, but only sheet 1 of 2 was included.

c) Provide drawings listed in Section 6.6.1 of Appendix 6.6 of NUH32PHB-0600.

This information is needed to determine compliance with 10 CFR 72.24.

CCNPP Response to Question 9-1:

Larger drawings (11x17) are provided in Enclosure 2. Sheet 2 of 2 for NUH32PHB-30-1 is also included.

NRC Question Chapter 9-2:

9-2 Provide a drawing and explicit description of the confinement boundary.

a) Per Section 9.5.1 of NUREG-1567, details of the confinement boundary should be provided, including a clear drawing and description of the entire confinement boundary. In addition, drawings or sketches defining "top shield plug closure weld" (page 5), "top cover plate weld" (page 5), and "double seal welded primary and secondary closures" at the top and bottom ends of the DSC (page 5 and 13.3-8) would aid in understanding the confinement boundary.

b) Note 1 in Drawing NUH32PHB-30-20 indicates the presence of an O-ring. The details of the O-ring groove and O-ring, including its purpose, should be provided. In addition, specify if there are O-ring degradation issues over the license period.

This information is needed to determine compliance with 10 CFR 72.24.

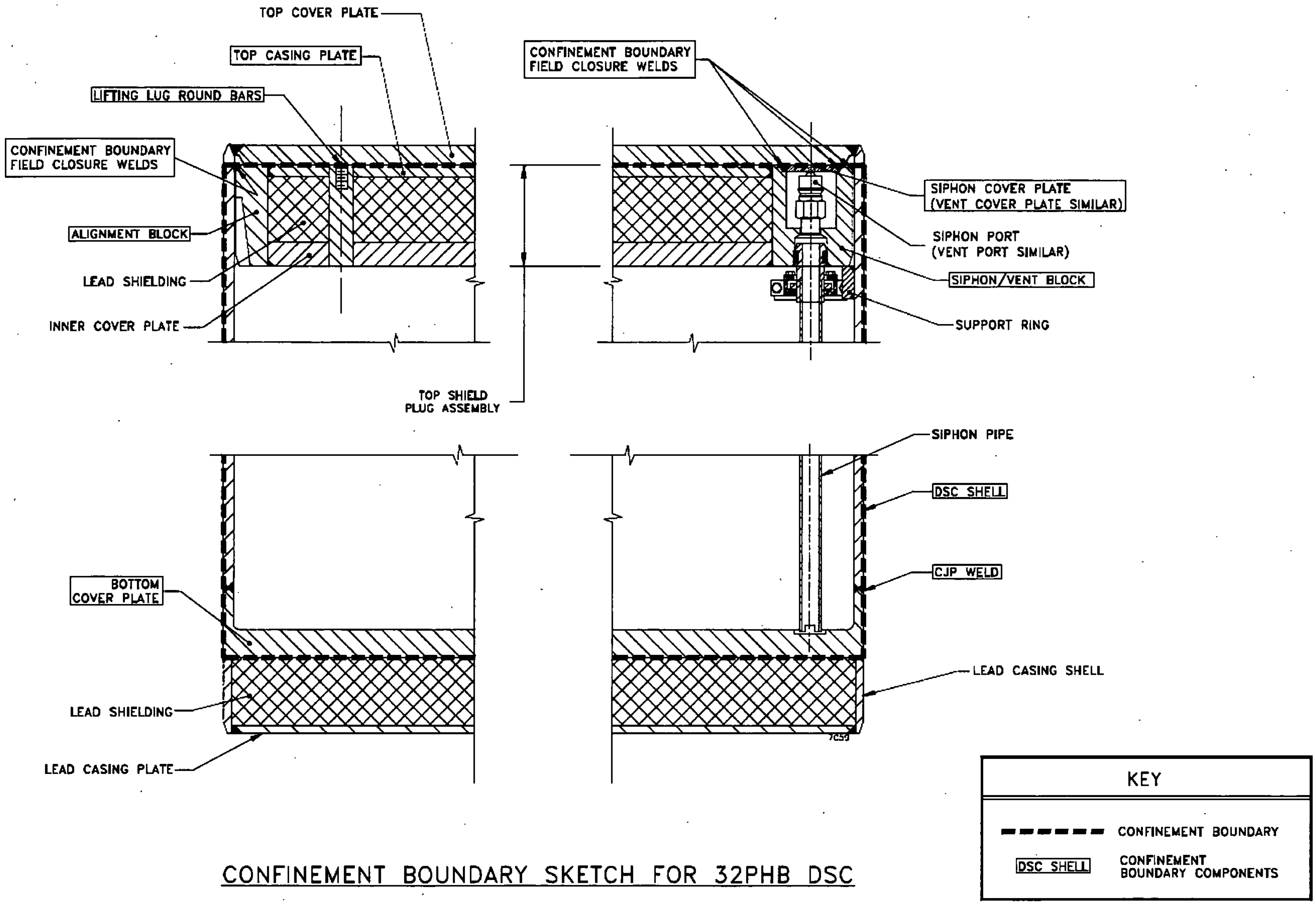
CCNPP Response to Question 9-2:

The physical design of the confinement boundary for the NUHOMS-32PHB DSC is identical to that of the NUHOMS-32P DSC currently licensed for use at Calvert Cliffs ISFSI. The confinement boundary components are called out in Transnuclear drawings NUH32PHB-30-1 and NUH32PHB-30-4 as follows:

1. Item 52: Siphon/Vent block (component of top shield plug assembly Item 50)
3. Item 53: Alignment block (component of top shield plug assembly Item 50)
4. Item 54: Top casing plate (component of top shield plug assembly Item 50)
5. Item 58: Lifting lug round bars (component of top shield plug assembly Item 50)
6. Item 2: DSC Shell
7. Item 10: Bottom cover plate

The associated shop and field welds between the above components are also part of the confinement boundary and are constructed to meet the American Society of Mechanical Engineers Section III, Subsection NB requirements. A sketch is provided below summarizing the above discussion.

The O-Rings identified as item 20 on Note 1 of Transnuclear drawing NUH32PHB-30-20 can be seen in detail along with the groove in which they are located on Section B-B of Transnuclear drawing NUH32PHB-30-2. They are located on the siphon tube adaptor and their purpose is to provide a seal between the adapter and the siphon block on the top shield plug to support blowdown of the DSC. They are not part of the confinement boundary, are classified as not important to safety and as indicated on Table 3.3-1 of Attachment (1) of Reference 2, the siphon tube does not have an intended function for license renewal. They are therefore not subject to aging management.



CONFINEMENT BOUNDARY SKETCH FOR 32PHB DSC

NRC Question Chapter 9-3:

9-3 *Provide details that show the design and operation of the DSC will keep the fuel and cladding from degrading.*

a) *Per Section 9.4.4.1 of NUREG-1567, the fuel and cladding must be protected from degradation. Aspects of the design and operation of the DSC that protect the fuel and cladding should be provided, especially considering that the DSC is a new design and a new cladding material would be stored within the DSC.*

b) *Per Section 9.5.4.1 of NUREG-1567, the application should indicate the design features and procedures for drying, evacuation, and backfilling that meet the oxidizing criteria within the DSC.*

This information is needed to determine compliance with 10 CFR 72.24.

CCNPP Response to Question 9-3:

The procedures and equipment used to blowdown, vacuum dry, and inert the NUHOMS-32PHB DSC are the same as those for the NUHOMS-32P DSC as described in USAR Sections 3.3.2 and 4.3.1. Reference 1 did not request changes to ISFSI TSs 2.2.1 and 2.2.2, and therefore the acceptance criteria for vacuum drying and final helium pressurization of the NUHOMS-32PHB DSC remain the same as those used for the NUHMOS-24P DSC and NUHOMS-32P DSC.

NRC Question Chapter 9-4:

9-4 *Provide an evaluation and supporting documents, e.g. stainless steel corrosion and SCC AMP, etc. that ensures confinement integrity would be maintained by the NUHOMS-32PHB DSC for the 40 year license period, especially due to the effects of chlorine-induced stress corrosion cracking (CISCC).*

NRC Information Notice 2012-20 has indicated that CISCC can affect the integrity of stainless steel vessels/piping via through-wall cracks over time. An analysis and discussion are necessary that confirm the NUHOMS-32PHB DSC, which is constructed from stainless steel, will maintain confinement.

This information is required to evaluate compliance with 10 CFR 72.24 (d) and 10 CFR 72.122(b)(1).

CCNPP Response to Question 9-4:

The aging management discussion for the NUHOMS-32PHB DSC was provided on pages 3 and 4 of Attachment (1) of Reference 1. As indicated therein, "Calvert Cliffs intends to include the HSM-HB and NUHOMS®-32PHB DSCs in the ISFSI Aging Management Program (AMP) described in Reference 1. This includes placing those components into the next lead canister inspection as noted by NUREG-1927, Appendix E." Reference 1 was the ISFSI License Renewal Application (see Reference 2 of this letter). In Reference 3, Calvert Cliffs submitted a proposed CISCC AMP with the intention that the population of canisters selected for inspection would bound all DSCs licensed at the Calvert Cliffs ISFSI [see discussion on page 11 of Attachment (1) of ML13170A574]. This proposed CISCC AMP is currently the subject of additional NRC RAIs received on June 23, 2014, with a planned response date of September 27, 2014. Any modifications to the previously proposed CISCC AMP submitted at that time will also be applicable to the NUHOMS-32PHB DSC when that design is incorporated into the Calvert Cliffs site-specific ISFSI license SNM-2505.

Finally, it should be noted that Reference 2 requested that the Calvert Cliffs site-specific ISFSI license SNM-2505 be extended from 20 years to 60 years (extended license would expire November 2052). The discussion of the expected service life for the NUHOMS-32PHB DSC being less than 40 years given that extended license period provided on page 4 of Attachment (1) of Reference 1 should not be interpreted as a request for a separate 40-year license for this DSC design.

Observations

NRC Observation 4-1:

4-1 *Provide and augment the aging management programs (AMP) supplied in the CC ISFSI license renewal application that addresses the requested increase in fuel burnup, aging related degradation mechanisms, and new equipment requested in LAR 2505-11.*

The CC ISFSI license renewal application is currently undergoing staff review. The application and subsequent staff evaluation does not contain AMPs for the new system, structures, and components requested in this amendment application. The requested AMPs are necessary because LAR 2505-11 will be processed in conjunction with or after the CC ISFSI license renewal application which implies that the total licensing period for the requested system in LAR 2505-11 is 60 years.

CCNPP Response to Observation 4-1:

See the response to Question 9-4.

NRC Observation 6-1:

6-1 *Address the use of forced cooling during transfer operations.*

Forced cooling during transfer operations is briefly mentioned in Section 13.3.2.5.3 of the application, specifically, "When utilized with the NUHOMS-32PHB DSC, the transfer cask is in the forced-cooling configuration for heat loads greater than Table 7-1 of Reference 13.26, Section 7." It appears that forced cooling will be used during each transfer operation for heat loads greater than Table 7-1 of Reference 13.26, Section 7.

CCNPP Response to Observation 6-1:

In addition to the information provided in Section 3.K of Attachment (1) of Reference 1, and the draft ISFSI USAR section discussed above, discussion of forced cooling has been included in other sections of the draft ISFSI USAR provided in Enclosure 1. The transfer cask will be placed into the forced cooling configuration for any NUHOMS-32PHB DSC with a heat load > 21.12 kW when it is loaded onto the transfer skid. However, operation of the blowers is not required initially, and is only one of three possible options when in the horizontal position for satisfying the action statement in new TS 3.3.2.1 within the time limits indicated.

NRC Observation 9-1:

9-1 *Confirm that the entire confinement boundary is helium leak tested to the "leaktight" criteria or provide a confinement dose analysis for the NUHOMS-32PHB DSC.*

Section 13.3.3.2 and Section 13.8.2.8 mention that the NUHOMS-32PHB DSC is leak tested to "leaktight" criteria (per ANSI N14.5) after loading and therefore does not require a confinement dose analysis (page 15 of application). Although page 15 of the application indicates that the top shield plug closure weld and the siphon and vent port cover welds *would be tested to 1E-7 ref-cc/sec, the leak test rates of longitudinal and circumferential shell welds, shell to baseplate weld, base metal, etc., were not specified in the application. It is necessary to test the entire confinement boundary to the "leaktight" criterion in order to forego a confinement dose analysis. If not tested to the "leaktight" criterion, it is necessary to test the entire confinement boundary to the allowable reference leakage rate. The allowable reference leakage rate would be determined from a confinement dose analysis using release fractions that have been justified for high burnup fuel. Additional guidance on this issue can be found in Interim Staff Guidance-25.*

CCNPP Response to Observation 9-1:

Design Specification SP-0564D delineates the design requirements for a NUHOMS-32PHB DSC.

The Design Specification addresses the testing of the closure welds in section 3.7.10. It says, "To reduce the likelihood or an accidental release of contaminants and loss of the Helium atmosphere, welded joints and seals shall be used in the design of the DSC. These permanent joints/ seals shall be placed in a redundant manner to ensure that an uncontrolled release of radioactive material or loss of Helium will not occur, should a single joint /seal fail. Closure weld joint inspection shall be by radiographic, multi-level liquid penetrate, and Helium leak detection methods as described in the ISFSI USAR and in Section 2.0 of this Specification." Section 2.0 of the Design Specification provides a list of applicable codes and standards.

The DSC design includes redundant seal welds to join the shell to the top and bottom end plug and cover plate assemblies, which form the containment boundary. The bottom end containment boundary welds are made during shop fabrication of the DSC. The top and containment boundary welds are made at CCNPP following fuel loading. The vent and siphon port penetrations are seal welded at CCNPP after DSC draining, drying and back filling operations are complete.

ATTACHMENT (2)

AFFIDAVIT

AFFIDAVIT PURSUANT TO 10 CFR 2.390

AREVA Inc.)
State of Maryland) SS.
County of Howard)

I, Paul Triska, depose and say that I am a Vice President of AREVA Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is for the following Design Calculations:

- NUH32PHB-0204 R-0
- NUH32PHB-0205 R-1
- NUH32PHB-0206 R-0
- NUH32PHB-0207 R-1
- NUH32PHB-0208 R-0
- NUH32PHB-0209 R-0
- NUH32PHB-0210 R-0
- NUH32PHB-0211 R-1
- NUH32PHB-0213 R-0
- NUH32PHB-0214 R-0
- NUH32PHB-0410 R-1
- NUH32PHB-0502 R-2
- NUH32PHB-0503 R-2
- NUH32PHB-0505 R-1

These documents have been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by AREVA Inc. in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the document described above, should be withheld.

1) The information, which is owned and have been held in confidence by AREVA Inc., sought to be withheld from public disclosure involves details regarding AREVA Inc.'s approach to its own dry fuel storage systems and Calvert Cliffs Nuclear Power Plant (CCNPP) intended use of the 32PHB dry spent fuel storage system.

2) The information is of a type customarily held in confidence by AREVA Inc. and not customarily disclosed to the public. AREVA Inc. has a rational basis for determining the types of information customarily held in confidence.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of AREVA Inc. because the information consists of details regarding AREVA Inc.'s approach to its own dry fuel storage systems and Calvert Cliffs Nuclear Power Plant (CCNPP) intended use of the 32PHB dry spent fuel storage system, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with AREVA Inc., take marketing or other actions to improve their product's position or impair the position of AREVA Inc.'s product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

Paul A Triska

Paul Triska
Vice President, AREVA Inc.

Subscribed and sworn to me before this 25th day of July, 2014.

Patricia Hoch

Notary Public

My Commission Expires 11 / 17 / 2014

Patricia Hoch
Notary Public
Howard County, MD
My Commission Expires Nov. 17, 2014

ENCLOSURE 1

DRAFT REVISION XX OF THE ISFSI USAR

Calvert Cliffs Nuclear Power Plant
July 25, 2014

UPDATED SAFETY ANALYSIS REPORT

REVISION SUBMITTAL DATES

REVISION	DATE
0	December 21, 1989
1	July 16, 1992
2	November 24, 1993
3	September 20, 1994
4	September 20, 1995
5	September 18, 1996
6	September 19, 1997
7	September 18, 1998
8	September 17, 1999
9	September 20, 2000
10	September 20, 2001
11	September 20, 2002
12	September 17, 2003
13	September 13, 2004
14	October 10, 2005
15	August 16, 2006
16	September 13, 2007
17	September 8, 2008
18	September 15, 2009
19	September 9, 2010
20	September 8, 2011
21	September 18, 2012
22	November 00, 2014

CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

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1.1-2	XX		
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1.3-1	XX		
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1.3-6	XX		
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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

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INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

LIST OF ACRONYMS

BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
DSC	Dry Shielded Canister
HRC	Hydraulic Ram Cylinder
HSM	Horizontal Storage Module
HSM-HB	High Burnup Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
NUHOMS	Nutech Horizontal Modular Storage
SPMT	Self-Propelled Modular Transporter
TCSS	Transfer Cask Support Skid

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

1.1 INTRODUCTION

Baltimore Gas and Electric Company (BGE) began commercial operation of the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2 on May 8, 1975 and April 1, 1977, respectively. Since then, these two 2700 MWT units have generated millions of KWH in a safe and reliable manner. In so doing, these units have discharged more than 1350 spent fuel assemblies. These assemblies are currently stored in a common storage pool. The need to provide additional on-site storage facilities to permit continued operation is discussed in Chapters 9, 10, and 11 of the Environmental Report.

In order to provide spent fuel storage until the Department of Energy begins to accept title to spent fuel under the requirements of the Nuclear Waste Policy Act of 1982, as amended in 1987, BGE has built and will operate an Independent Spent Fuel Storage Installation (ISFSI) in compliance with Title 10 Code of Federal Regulations Part 72. Baltimore Gas and Electric Company has chosen the Nutech Horizontal Modular Storage® (NUHOMS)-24P dry storage system designed by Transnuclear West (formerly Nutech Engineers, Inc.) to be used for the Calvert Cliffs ISFSI. The NUHOMS-24P system is more fully described in Reference 1.2. The location of the ISFSI on the Calvert Cliffs site is shown on Figure 1.1-1.

Calvert Cliffs Nuclear Power Plant has reanalyzed the ISFSI to use Transnuclear NUHOMS-32P and NUHOMS-32PHB Dry Shielded Canisters (DSCs) to optimize its dry spent fuel storage capacity. The NUHOMS-32P and NUHOMS-32PHB DSC system stores eight more spent fuel assemblies than the NUHOMS-24P DSC using the same external and internal shell dimensions. The NUHOMS-32P and NUHOMS-32PHB DSC storage capacities are optimized by reducing the space between the locations of each fuel assembly and by slightly reducing the size of the storage locations.

Chapter 12 is a dedicated discussion of the use of the NUHOMS-32P DSC design and Chapter 13 is a dedicated discussion of the use of the NUHOMS-32PHB DCS design at CCNPP. Unless otherwise explicitly stated the information on the NUHOMS DSCs provided throughout this Updated Safety Analysis Report is applicable to ~~both~~ the NUHOMS-24P DSC, ~~and~~ the NUHOMS-32P DSC and the NUHOMS-32PHB DSC.

The major difference between the NUHOMS-32P DSC and the NUHOMS-24P DSC is the internal basket assembly. The DSC is loaded into a transfer cask for transporting to and from the horizontal storage module (HSM). The same transfer cask is used for on-site transfer of either a NUHOMS-24P DSC or a NUHOMS-32P DSC. Likewise, the same HSM design is used for the storage of either a NUHOMS-24P DSC or a NUHOMS-32P DSC, however, the latter may also be stored in a modular high burnup horizontal storage module (HSM-HB).

When moving a NUHOMS-32PHB DSC the transfer cask is modified to allow for forced cooling. Also, the NUHOMS-32PHB DSCs are only stored in the HSM-HBs.

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

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

The NUHOMS system is a totally passive installation that is designed to provide shielding and safe confinement of irradiated fuel. The DSC and HSM/HSM-HB have been designed to withstand certain accidents.

The fuel assemblies to be stored in the ISFSI are located in the Calvert Cliffs spent fuel pool and were irradiated only in the Calvert Cliffs reactors. Twenty-four fuel assemblies are stored in each NUHOMS-24P DSC, 32 fuel assemblies are stored in each NUHOMS -32P DSC or NUHOMS-32PHB and one DSC is stored in each concrete module. The license allows construction and operation of a total of 132 modules. Of these, 72 HSMs were poured in place and the remaining 40 HSM-Hs will be of modular construction. The NUHOMS-32PHB will not be stored in the poured in place HSMs, but are to be stored in the HSM-HBs only. These modules will be built incrementally, as needed, to match CCNPP's requirements for additional storage. The first 72 modules built were poured in place (HSM) and the remaining will be of modular construction (HSM-HB). Operation of the facility will continue for up to 20 years under the initial license and continue under license renewal as necessary until a permanent facility is available for spent fuel disposal. As defined in Table 1.2-2 of Reference 1.2, the minimum design life of the facility is 50 years.

1.2 GENERAL DESCRIPTION OF INSTALLATION

1.2.1 GENERAL DESCRIPTION

The ISFSI provides for the horizontal dry storage of irradiated fuel assemblies in a concrete module. The principal components are a concrete HSM and a stainless steel DSC with an internal basket which holds the fuel assemblies. Each HSM contains one DSC. Each NUHOMS-24P DSC contains 24 fuel assemblies. Each NUHOMS-32P or NUHOMS-32PHB DSC contains 32 fuel assemblies.

Despite Department of Energy's obligations under the Nuclear Waste Policy Act of 1982, as amended, to begin accepting fuel on January 31, 1998, BGE's current best estimate for the earliest date to ship spent fuel for permanent disposal has not been determined at this date. The license allows for construction and use of up to 132 HSMs. The provision for 132 HSMs will provide the minimum storage capacity needed to carry Calvert Cliffs to approximately 2026.

The initial phase of construction includes 72 HSMs. Additional modules can be added as required on separate foundations without impact to the preceding or subsequent modules. Analyses for structural and foundation requirements provide for constructing poured in place modules in a 2x6 array. Analyses for structural and foundation requirements provide for constructing the pad to accommodate HSM-HB modules in two 2x12 arrays and one 1x12 array. The layout of the ISFSI is shown on Figure 1.2-1.

In addition to these primary components, the Calvert Cliffs ISFSI also requires transfer equipment to move the DSCs from the spent fuel pool (where they are loaded with spent fuel) to the HSMs where they are stored. This transfer system consists of a transfer cask, a hydraulic ram, a cask skid mounted on either a self-propelled modular transporter (SPMT) or a trailer pulled with a truck. This transfer system interfaces with the existing Calvert Cliffs spent fuel pool, the cask handling crane, and the site layout (i.e., roads and topography).

1.2.2 PRINCIPAL SITE CHARACTERISTICS

The ISFSI is located on the CCNPP site near Lusby, MD. Baltimore Gas and Electric Company owns and operates two 2700 MWT nuclear generating units on the Calvert Cliffs site. The ISFSI is located outside the protected area, but within the owner controlled area approximately 2300' southwest of the plant (Figure 1.1-1).

1.2.3 PRINCIPAL DESIGN CRITERIA

The principal design criteria and parameters for the Calvert Cliffs ISFSI are shown in Table 1.2-1 and Table 1.2-3. A detailed description of the criticality safety, shielding, structural, and decay heat removal features of the storage system is presented in the following Chapters and in Reference 1.2.

Design features of the NUHOMS system important to safe operation are outlined in Reference 1.2 and USAR, Chapters 12 and 13. Changes to any of these design features will be implemented only after conducting a safety review in accordance with 10 CFR 72.48 or a license amendment is granted by the NRC.

1.2.4 OPERATING AND FUEL HANDLING SYSTEMS

The major operating systems of the ISFSI are those required for fuel handling in the Auxiliary Building and transport of the transfer cask and DSC from the spent fuel pool to the ISFSI. The primary design parameters for these systems are listed in Table 1.2-2 and Table 1.2-4. The majority of the fuel handling operations involving the transfer cask which take place in the Auxiliary Building (i.e., fuel loading, drying, trailer loading, etc.) utilize standard techniques at Calvert Cliffs for spent fuel shipment. The remaining operations (canister seal welding, transfer cask-HSM alignment, and DSC transfer) are unique to the ISFSI.

1.2.5 SAFETY FEATURES

The principal safety features of the ISFSI are inherent in the design of the DSC and the HSM. These safety features include protection of the spent fuel from the consequences of extreme environmental phenomena, redundant DSC closure welds to ensure containment, and a range of operational design features to maintain occupational dose as low as reasonably achievable. Additional details of the safety features of the NUHOMS System are presented in Reference 1.2 and USAR, Chapters 12 and 13.

1.2.6 RADIOACTIVE WASTE AND AUXILIARY SYSTEMS

No radioactive waste is generated during normal storage operations and, because of the passive nature of the ISFSI, no auxiliary systems are required for storage. The DSC Vacuum Drying System, used during initial canister closure operations, is an auxiliary system which pumps contaminated water from the DSC to plant processing systems or back to the spent fuel pool. It is also used to evacuate the DSC and backfill it with helium. The existing Calvert Cliffs Auxiliary Building processing systems are used to handle water and gasses which are drained and vented from the cavity of the DSC during the drying process.

TABLE 1.2-1
DESIGN PARAMETERS FOR THE CALVERT CLIFFS ISFSI USING A NUHOMS-24P OR NUHOMS-32P DSC

GENERAL DESIGN REQUIREMENTS	
Capacity (Fuel Assemblies/Canister)	NUHOMS-24P DSC 24 Pressurized Water Reactor Assemblies NUHOMS-32P DSC 32 Pressurized Water Reactor Assemblies
Reference Fuel Assembly Parameters:	
Burnup: Max. Assembly Average	47,000 MWD/MTU (NUHOMS-24P) 52,000 MWD/MTU (NUHOMS-32P)
Initial Enrichment (Maximum)	4.5 w/o U^{235}
Initial Uranium Content	NUHOMS-24P 386 kg/Assembly (Nominal) NUHOMS-32P 400 kg/Assembly (Maximum)
Decay Heat Power (Maximum)	0.66 kW/Assembly
Cooling Time	As Required for Decay Heat Limit
Fuel Rod Array	Combustion Engineering 14x14
Assembly Weight (Maximum)	1,450 lbs
Maximum Assembly Envelope	8.25 inches by 8.25 inches
Effective Multiplication Factor:	
Normal	NUHOMS-24P $K_{eff} < 0.95$ NUHOMS-32P $K_{eff} < 0.95$
Off-Normal	NUHOMS-24P $K_{eff} < 0.98$ NUHOMS-32P $K_{eff} < 0.95$
Internal DSC Atmosphere	Helium ≥ 99.995% pure
Ambient Temperature	Range -3°F to 103°F
Solar Heat Load: Maximum	127 Btu/hr-ft^2
Average	82 Btu/hr-ft^2
Maximum Dose at HSM Surface During Storage (Away from Openings)	20 mrem/hr
Maximum Dose at HSM Door and Penetrations	100 mrem/hr
Peak Long-Term Clad Temperature	635°F
Peak Short-Term Clad Temperature	1,058°F
Credit for Burnup Criticality Analysis	NUHOMS-24P Based on 1.8% equivalent initial enrichment NUHOMS-32P – N/A[(a)]
Maximum Assembly Length (Includes Radiation Growth)	less than 158.0"
Active Fuel Length	136.7"

(a) See Section 12.3.3.4.

TABLE 1.2-2
PRIMARY DESIGN PARAMETERS FOR THE NUHOMS-24P OR NUHOMS-32P DSC TRANSFER SYSTEM

[References 1.4, 1.7, and 1.8]

SYSTEM	PARAMETER	VALUE
Transfer Cask	Nominal Cavity Diameter	68"
	Nominal Cavity Length	173.5"
	Payload	95,000 lbs (Maximum)
	Decay Heat Rejection (Maximum)	21.12 kW (0.66 kW/Assembly) (NUHOMS-32P DSC)
	Shielding (Surface Dose, Combined Neutron and Gamma, Away from Penetrations)	200 mrem/hr (Maximum)
Transfer Cask Handling	Liftable by Yoke Using Crane Rotates on Lower Trunnions Vertical to Horizontal when Lowered by Crane	109.25 tons Gross Lift (Maximum)
Transfer Cask Skid	Payload	215,000 lbs (Maximum)
Transfer Trailer	Payload (Cask + Skid)	240,000 lbs (Maximum)
	Dead Weight	40,000 lbs (Maximum)
	Gross Vehicle Weight	280,000 lbs (Maximum)
	Limiting Cask Height	80" (Maximum)

TABLE 1.2-3
DESIGN PARAMETERS FOR THE CALVERT CLIFFS ISFSI USING A NUHOMS-32PHB DSC

GENERAL DESIGN REQUIREMENTS	
Capacity (Fuel Assemblies/Canister)	32 Pressurized Water Reactor Assemblies
Reference Fuel Assembly Parameters:	
Burnup: Max. Assembly Average	62,000 MWD/MTU
Initial Enrichment (Maximum)	5.0 w/o U^{235}
Initial Uranium Content	475 kg/Assembly
Decay Heat Power (Maximum)	1.5 kW/Assembly
Cooling Time	As needed to reach 1.5 kW
Fuel Rod Array	Combustion Engineering 14x14. VAP 14x14 AREVA 14x14
Assembly Weight (Maximum)	1450 lbs
Maximum Assembly Envelope	8.25" by 8.25"
Effective Multiplication Factor:	
Normal	$K_{eff} < 0.95$
Off-Normal	$K_{eff} < 0.95$
Internal DSC Atmosphere	Helium
Ambient Temperature	Range -3°F to 103°F
Solar Heat Load: Maximum	127 Btu/hr-ft^2
Average	82 Btu/hr-ft^2
Maximum Dose at HSM-H Surface During Storage (Away from Openings)	20 mrem/hr
Maximum Dose at HSM Door and Penetrations	100 mrem/hr
Peak Long-Term Clad Temperature	≤ 752°F (400°C)
Peak Short-Term Clad Temperature	≤ 71,058°F (570°C)
Credit for Burnup Criticality Analysis	N/A
Maximum Assembly Length (Includes Radiation Growth)	< 158.0"
Active Fuel Length	136.7"

TABLE 1.2-4
PRIMARY DESIGN PARAMETERS FOR THE NUHOMS-32PHB DSC TRANSFER SYSTEM

SYSTEM	PARAMETER	VALUE
Transfer Cask	Nominal Cavity Diameter	68"
	Nominal Cavity Length	173.5"
	Payload	215,000 lbs
	Decay Heat Rejection	33.0 kW [1.5 kW/Assembly – (Maximum)]
	Shielding (Surface Dose, Combined Neutron and Gamma, Away from Penetrations)	250 mrem/hr (Maximum)
Transfer Cask Handling	Liftable by Yoke Using Crane Rotates on Lower Trunnions Vertical to Horizontal when Lowered by Crane	109.25 tons gross lift (Maximum)
Transfer Cask Skid	Payload	215,000 lbs (Maximum)
Self-Propelled Modular Transporter (SPMT)	Payload (Cask + Skid)	240,000 lbs (Maximum)
	Dead Weight	30,000 lbs
	Gross Vehicle Weight	279,000 lbs
	Limiting Cask Height	80" (Maximum)

1.3 GENERAL SYSTEMS DESCRIPTIONS

The following subsections briefly describe the principal systems and components and their operation. The major systems, subsystems, and components of the Calvert Cliffs ISFSI are shown in Table 1.3-1 and Table 1.3-2.

1.3.1 SYSTEMS DESCRIPTIONS

The components of storage at the ISFSI are the DSC and the HSM/HSM-HB. Additional systems required for the DSC closure and transfer include the transfer cask, the skid and skid positioning system, the trailer or the SPMT, the hydraulic ram system, and the DSC vacuum drying system.

1.3.1.1 Dry Shielded Canister Design

1.3.1.1.1 NUHOMS-24P DSC

The design of the generic NUHOMS-24P DSC is described in detail in Reference 1.2. The Calvert Cliffs DSC is very similar to the referenced design with revisions as necessary to accommodate a slightly different fuel assembly design. The main component of construction of the DSC is a stainless steel cylindrical containment vessel.

The component subassemblies of the NUHOMS-24P DSC are listed in Table 1.3-1 and shown on Figure 1.3-1. The internal basket assembly is comprised of 24 guide sleeves supported by spacer disks at intervals corresponding, for the most part, to the fuel assembly spacer grids. For a few of the fuel assemblies, the spacer grids were found not to be in complete alignment with the NUHOMS-24P DSC spacer disks. Such misalignments were evaluated structurally, and found to be able to withstand normal and cask drop loads. Support rods maintain the spacer disk location. All canister structural components are fabricated from type 304 stainless steel, except the spacer disks and support rods may be fabricated from aluminum coated carbon steel. Lead gamma shielding is used in both the top and bottom end shield plugs.

The principal differences between the Calvert Cliffs NUHOMS-24P DSC and the generic DSC design are: the addition of one spacer disk for a total of nine to accommodate the Calvert Cliffs fuel which has nine spacer grids; thinner spacer disks with wider ligaments; an additional 1/2" of lead in both shield plugs; and a shorter overall length accounting for the shorter fuel assembly design.

Criticality safety for the NUHOMS-24P DSC, during wet loading operations, is maintained through the geometric separation of the fuel assemblies within the internal basket assembly, the inherent neutron absorption capability of the stainless steel guide sleeves, and the proper selection of sufficiently depleted fuel assemblies. The NUHOMS-24P DSC has only been analyzed for storage in the HSM.

1.3.1.1.2 NUHOMS-32P DSC

The NUHOMS-32P DSC design increases the number of stainless steel guide sleeves to 32 (one for each spent fuel assembly) and uses an egg-crate design made of stainless steel and aluminum (borated and unborated plates) to support the guide sleeves. This egg-crate design is similar to the Transnuclear TN-68 basket assembly currently in use at a number of nuclear plants. Both the guide sleeves and the egg-crate components run the full length of the DSC cavity. This allows the guide sleeves to be in contact with the egg-crate components over the whole length of the DSC cavity versus only at spacer discs in the NUHOMS-24P DSC design. As with the NUHOMS-24P DSC design, the basket assembly is not attached to the DSC shell walls or cover plates.

Other differences are the relocation of the vent and siphon ports. They have been moved from the DSC shell wall (in NUHOMS-24P DSC) to the DSC top shield plug (in NUHOMS-32P DSC) to improve the welding, blowdown, and vacuum drying operations. The NUHOMS-32P DSC lifting fixture device is different than the lifting eyes of the NUHOMS-24P DSC (Reference 1.9). The top shield plug for the NUHOMS-32P DSC is different than the NUHOMS-24P DSC top shield plug. The design change to the top shield plug involves a reduction in the lead shield thickness, and an increase in the outer steel plate thicknesses.

Criticality Safety for the NUHOMS-32P DSC is maintained through fixed neutron absorbers in the NUHOMS-32P basket and increasing the soluble boron in the spent fuel pool water to a concentration of 2,450 ppm.

All the major steps for loading and unloading a DSC (welding, vacuum drying, etc.) are the same for the NUHOMS-24P DSC and the NUHOMS-32P DSC systems. The DSC is loaded into a transfer cask for transporting to and from the HSM/HSM-HB. The NUHOMS-32P DSC may be stored in either the HSM or HSM-HB.

1.3.1.1.3 NUHOMS-32PHB DSC

The design of the NUHOMS-32PHB is very similar to the design of the NUHOMS-32P with revisions as necessary to accommodate higher burnup discharged spent fuel assemblies. The NUHOMS-32PHB basket periphery area will contain solid aluminum rails to increase the heat transfer properties the DSC.

Criticality safety for the NUHOMS-32PHB is similar to that of the NUHOMS-32P and is maintained through fixed neutron absorbers in the NUHOMS-32PHB basket. Also, the soluble boron concentration in the spent fuel pool water will be maintained to at least 2,450 ppm during cask loading operations in the spent fuel pool.

The major steps for loading and unloading a DSC (welding, vacuum drying, etc.) are the same for the NUHOMS-24P DSC, the NUHOMS-32P DSC and the NUHOMS-32PHB DSC systems. The DSC is loaded into a transfer cask for transporting to and from the HSM. However, the NUHOMS-32PHB is placed into a HSM-HB module for storage.

The NUHOMS-32PHB will not be loaded into one of the HSMs that were constructed before 2008.

1.3.1.2 Horizontal Storage Module

The Calvert Cliffs ISFSI employs poured in place HSMs constructed in units of 12 configured in a 2x6 array for the first 72 modules, and modular construction HSM-HBs configured in two 2x12 arrays and one 1x12 array for the remaining modules.

1.3.1.2.1 HSM Modules

The HSM major design features are similar to the design presented in Reference 1.2. Major design features include items such as the overall module layout, size, wall thicknesses, DSC support rails layout and location, and air inlet and outlet configurations and sizes. There are differences in the HSM design details compared to those presented in Reference 1.2, including the following:

A. The DSC rail support beam at the front of the module in the Reference 1.2 design was eliminated for simplification of the rail support scheme. The front end of the rails were changed to be supported directly by the front wall of the module using anchored angles similar to those used to support the rail support beams at the middle and rear of the module.

B. The amount of shear reinforcement was changed to be consistent with the specific design parameters applicable to the Calvert Cliffs ISFSI HSMs.

C. The foundation size was reduced to simplify construction.

D. The DSC seismic restraint was redesigned to make it significantly lighter than the Reference 1.2 design and therefore easier to handle. This reduces personal radiation exposure for placement of the restraint in the module after insertion of the DSC.

E. The DSC support rails were redesigned from WT 6x115 to WF 8x40 to employ a more efficient section which reduces weight, resulting in reduced material and construction costs.

F. The DSC support assembly cross-member section was redesigned from W 10x68 to W 8x48 to employ a more efficient, lighter weight section.

G. The module rebar design was revised to eliminate unnecessary rebar.

H. The HSM door design was revised to incorporate additional radiological shielding material.

The HSMs are constructed in place at the ISFSI with pairs of 2x6 arrays placed end to end. The arrangement of the HSMs at the ISFSI is shown in Figure 1.2-1. Each array of 12 HSMs is constructed on a common reinforced concrete foundation slab. The HSM is designed to provide neutron and gamma shielding to achieve a nominal 20 mrem/hr contact dose rate. Nominal contact dose rates at the HSM access door and vents are designed to be less than 100 mrem/hr.

Three foot thick end walls provide shielding on the sides of each HSM array. The front walls of the HSMs are thickened to 3-1/2'. Two foot thick interior common walls provide shielding between modules to prevent scatter in adjacent modules during DSC loading and retrieval. The roof slab for the HSMs is 3' thick. An internal slab and roof caps are provided to shield the ventilation inlet and outlet openings.

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

A. Winds and Tornado (including missile impact) — Regulatory Guide 1.76

B. Seismic — CCNPP Updated Final Safety Analysis Report, Section 2.6

C. Flood — CCNPP Updated Final Safety Analysis Report, Section 2.5

D. Snow and Ice — American National Standards Institute A58.1-1982

E. Combined Loads (dead weight, live loads, thermal loads, creep effects) — American Concrete Institute 349-85.

1.3.1.2.2 HSM-HB Modules

The HSM-HB to be used at CCNPP is a modified version of the HSM-H described in the Updated Final Safety Analysis

Report for the Standardized NUHOMS® and the NUHOMS® HD systems (References 1.10 and 1.11). For CCNPP, the door and the spacer are modified to accommodate the diameter and length of the 32P DSC. While the HSM-HB module design is similar to the design of the poured inplace HSM, the following is a summary of the design differences which provide improved heat rejection and shielding capabilities:

- Use of a thicker roof (3'8" vs 3') provides improved shielding,
- Door is inset in DSC opening, with increased thickness to provide improved shielding,
- Use of slotted plates and holes in the DSC support rails to increase airflow at the bottom portion of canister,
- Increased dose rates near inlet vent, decreased in all other analyzed locations,
- Increased height of the module to increase module cavity and stack height and to minimize air flow resistance in the module cavity,
- Optimized DSC support structure minimizes airflow resistance,
- Outlet vents repositioned from top front and back to top sides (opening shared by adjacent modules),
- Inlet vents repositioned from front bottom center to front bottom sides (opening shared by adjacent modules). Use of operational attenuation pipes improves inlet vent shielding.

The HSM-HBs will be prefabricated and assembled at the Calvert Cliffs ISFSI site whereas the current HSMs were poured in place. Each prefabricated HSM-HB is comprised of a base unit and roof unit assembled together to form a single module. The DSC is supported inside the HSM-HB by the DSC support structure. The DSC support structure (rail support assembly) is comprised of two rail sections, two slotted plates and two rail support plates. The rail support assembly provides support for the DSC during storage and acts as a sliding surface during DSC insertion and retrieval. The modules sit on a reinforced concrete basemat at the ISFSI site.

Each HSM-HB is placed in contact with an adjacent HSM-HB to form a two 2x12 arrays and one 1x12 array. The air inlet vents extend through the front on both sides of the front wall. The front wall and the rear wall of the base unit provide support for the rails and the rail extension flanges. The roof unit rests on the front, rear, and side walls of the base unit. The air outlet vents are provided in the roof unit. Similar to the current poured in place HSMs, flat panels are used as heat shields on the interior walls of the HSM-HB. The heat shields provide thermal protection for the HSM-HB concrete. Finally, attenuation pipes are added, as an option, to the inlet vent screens to improve the shielding capabilities of the module.

The HSM-HB shield door consists of a rectangular steel plate at the front attached to a circular reinforced concrete block at the rear. Both the steel plate and concrete blocks fit the circular opening in the front wall. Studs are welded to the circular steel plate which provides anchor's to the rear reinforced concrete block. The concrete door provides missile protection and shielding. End shield walls are provided at the ends of a module array to provide the required missile and shielding protection.

During DSC insertion/retrieval operations, the transfer cask is docked with the HSM-HB docking surface and mechanically secured to the embedments provided in the front wall. The embedments are equally spaced on either side of the HSM-HB access opening.

Storage of the NUHOMS-32P DSC in the HSM-HB is subject to the same controls and limitations as those of the HSM.

1.3.1.3 Transfer Cask

The transfer cask used with the ISFSI provides radiological shielding during the DSC closure operations and during transfer of the DSC to the HSM/HSM-HB. To ensure structural integrity, the transfer cask also provides protection of the DSC against potential natural and operational hazards during transport and transfer of the DSC to the HSM/HSM-HB. Both solid neutron and lead gamma shielding are incorporated into the transfer cask design. Figure 1.3-2 shows the major components of the transfer cask. The Calvert Cliffs transfer cask has a solid hydrogenous neutron shield in the outer annulus of the cask, and as a result the liquid neutron shield expansion tank of Reference 1.2 was deleted.

1.3.1.4 Transfer Equipment

1.3.1.4.1 Transfer Trailer

The transfer trailer is used to transport the transfer cask skid and the loaded transfer cask from the Auxiliary Building to the ISFSI. The transfer trailer is an industrial heavy-haul trailer with pneumatic tires, hydraulic suspension and steering, and brakes on all wheels. The approach slab has adequate space for turning the transport trailer and tow vehicle. Four hydraulic jacks are incorporated into the transfer trailer design to provide vertical elevation adjustment for alignment of the cask at the HSM. The transfer trailer is shown in Figure 1.3-3. It is pulled by a conventional tractor.

1.3.1.4.2 SPMT

The Doerfer Companies Wheelift SPMT, complete with Transfer Cask Support Skid, is designed to accept a Transfer Cask filled with a Dry Shield Canister that is lowered onto the Transfer Cask Support Skid (TCSS) in a vertical orientation. In the Down-ending Position (see Figure 1), the SPMT is lowered so that the TCSS rests on the pavement.

The transporter is composed of four major systems: controls, power generation, hydraulic, and the drive system. Using these systems, an operator can manually position the vehicle to lift a maximum load of 150 tons. Operation is powered either by using building power through an umbilical cable ("Shore") or by relying on diesel to fuel the engine ("Gen") for electrical generation.

This transporter is designed to operate on smooth surfaces, such as concrete or asphalt, with a grade of no more that 6% (loaded or unloaded).

The transporter is equipped with twelve articulating axle Wheelift assemblies, each of which is comprised of a large-bore hydraulic cylinder, two servo motors and gearboxes, and two 18" diameter x 12" wide urethane drive wheels coupled by chains and sprockets to the gearboxes. The center of the assembly contains a custom lift column designed to provide on-center rotation as well as lift and suspension functions. The vehicle has 10" vertical stroke provided by the hydraulic cylinders to accommodate the clearance needed to drive under the load, and then lift and carry the load. Steering is accomplished by varying torques and speeds of the two servo motors on any given wheel set. Hard stops limit the module's rotation to approximately +140° to -50°, while software limits further reduce the rotational angle to preset limits depending on the mode of travel. The motors are designed to propel the module at speeds ranging from 0.0 to 110 fpm.

1.3.1.5 <u>Cask Skid and Positioning System</u>

1.3.1.5.1 Transfer Trailer Cask Skid and Positioning System

The transfer cask skid is essentially identical in design and operation to previous NUHOMS-24P system transfer cask support skids. The skid is supported on lubricated bearing plates attached to the trailer deck and can be moved horizontally on the bearing plates by the hydraulic actuators of the skid positioning system. The skid is secured to the trailer deck in a travel lock position during cask loading and transport operations. The transfer cask skid is shown in Figure 1.3-4.

1.3.1.5.2 SPMT Skid and Positioning System

The TCSS is a large weldment with a capacity of 118 tons, designed to hold the Transfer Cask and Dry Shield Canister during transfer, loading, and unloading operations. The TCSS includes aluminum bronze trunnion inserts and a Gleason Reel for electrical cable storage. This framework also supports the Hydraulic Ram Cylinder (HRC) Assembly and a Hydraulic Tilt Cylinder Assembly. Each of these assemblies has chamfered locator blocks to guide each assembly into proper position when loading/connecting the hydraulic ram assembly to the TCSS.

1.3.1.6 Hydraulic Ram System

1.3.1.6.1 Transfer Trailer Hydraulic Ram System

The hydraulic ram consists of a double acting hydraulic cylinder with a capacity of 80,000 lb in either push or pull and stroke of 21'. The ram is supported during operation by a frame assembly attached to the bottom of the transfer cask and a tripod assembly resting on the concrete slab. The operational loads of the hydraulic ram are grounded through the transfer cask. The hydraulic ram system includes a grapple at the end of the piston which is used to engage a grapple ring on the DSC for retrieval operations. Figure 1.3-5 shows the hydraulic ram system.

1.3.1.6.2 SPMT Hydraulic Ram System

Mounted to the TCSS, is a hydraulic and electric line connection plate. This connection plate houses the quick disconnect fittings for the Ram and Tilt hydraulic cylinders and for the Grapple assembly cable.

The HRC Assembly consists of the HRC and the Grapple Assembly. The HRC is a telescopic three stage double acting cylinder that can advance approximately 21 feet.

The Grapple Assembly consists of a 130 vdc motor, lead screw, toggle mechanism, two grapple fingers, and quick disconnect mounting. The grapple fingers are actuated by the dc motor and lead screw. The Grapple Assembly is used to secure the HRC to the DSC.

1.3.1.7 Vacuum Drying System

The vacuum drying system removes water and air from the DSC and fills it with helium. The vacuum drying system has four operational modes: water removal, helium or air forced water removal, vacuum pumping, and helium backfilling.

1.3.1.8 Closure Welding System

The DSC closure welds on the shield plug and the top cover plate are normally placed by a fully remote, automatic welding system. The system includes modular components and is designed for rapid setup. Welding operations are remotely controlled by an operator who views the progress of the weld through closed circuit television. The welding head is designed to permit rapid replacement with either a UT probe, or a plasma gouging torch which can be used to remove the shield plug and top cover plate closure welds. Manual welding may also be used for closure welds. The allowed duration of manual welding is limited by the ambient dose rate at the location of the welding (Reference 1.6).

1.3.1.9 System Operation

See Chapter 5 for a detailed description of the Calvert Cliffs ISFSI System Operation.

TABLE 1.3-1
MAJOR SYSTEMS, SUBSYSTEMS, AND COMPONENTS OF THE CALVERT CLIFFS ISFSI FOR A NUHOMS-24P OR NUHOMS-32P DSC

Dry Shielded Canister

- NUHOMS-24P DSC Basket
  - Guide Sleeves (24)
  - Spacer Disks (9)
  - Support Rods (4)
- DSC Shell With Bottom Shield Plug
- Shield Plug (Top)
- Cover Plates (Top and Bottom)
- Siphon and Vent Ports
- Ram Grapple Ring

- NUHOMS-32P DSC Basket
  - Guide Sleeves (32)
  - Egg-Crate
  - Peripheral Steel Rails
- DSC Shell With Bottom Shield Plug
- Shield Plug (Top)
- Cover Plates (Top and Bottom)
- Siphon and Vent Ports
- Ram Grapple Ring

Horizontal Storage Module

- Reinforced Concrete Walls, Roof, Basemat, and Foundation
- DSC Structural Steel Support Assembly
- DSC Seismic Retainer
- Cask Docking Flange and Tie-Down Restraints
- Heat Shield
- Shielded Front Access Door and Door Supports
- Ventilation Air Openings (One Inlet, Two Outlets)
- Shielded Ventilation Air Inlet Plenum
- Ventilation Air Outlet Shielding Blocks
- Lightning Protection System

Cask Lifting Yoke

Transfer Cask

- Cask Structural Shell Assembly
- Bolted Top Head Assembly
- Upper Lifting Trunnions
- Lower Tilting Trunnions
- Lead Gamma Shielding
- Solid Neutron Shielding
- Ram Access Penetration Cover Plate
- Ram Access Penetration Shield Plug Assembly
- Ram Mounting Frame

Transfer Trailer* and Skid

- Heavy Industrial-Grade Trailer
- Cask Support Skid
- Skid Positioning and Alignment System

Hydraulic Ram System

- Hydraulic Cylinder
- Rear Tripod Support Frame
- Grapple Assembly

Vacuum Drying System

Automated Remote Closure Welding System

* The SPMT may be used in place of the Transfer Trailer.

TABLE 1.3-2
MAJOR SYSTEMS, SUBSYSTEMS, AND COMPONENTS OF THE CALVERT CLIFFS ISFSI FOR A NUHOMS-32PHB DSC

Dry Shielded Canister
- NUHOMS-32P DSC Basket
- Guide Sleeves (32)
- Egg-Crate
- Peripheral Steel Rails
- DSC Shell With Bottom Shield Plug
- Shield Plug (Top)
- Cover Plates (Top and Bottom)
- Siphon and Vent Ports
- Ram Grapple Ring

Horizontal Storage Module (HSM-H)
- Reinforced Concrete Walls, Roof, Basemat, and Foundation
- DSC Structural Steel Support Assembly
- DSC Seismic Retainer
- Cask Docking Flange and Tie-Down Restraints
- Heat Shield
- Shielded Front Access Door and Door Supports
- Ventilation Air Openings (One Inlet, Two Outlets)
- Shielded Ventilation Air Inlet Plenum
- Ventilation Air Outlet Shielding Blocks
- Lightning Protection System

Cask Lifting Yoke

Transfer Cask
- Cask Structural Shell Assembly
- Bolted Top Head Assembly
- Upper Lifting Trunnions
- Lower Tilting Trunnions
- Lead Gamma Shielding
- Solid Neutron Shielding
- Ram Access Penetration Cover Plate
- Ram Access Penetration Shield Plug Assembly
- Ram Mounting Frame

Self-Propelled Horizontal Cask Transporter and Skid*
- Self-Propelled Horizontal Cask Transporter
- Cask Support Skid
- Skid Positioning and Alignment System

Hydraulic Ram System
- Hydraulic Cylinder
- Rear Tripod Support Frame
- Grapple Assembly

Vacuum Drying System

Automated Remote Closure Welding System

* The Self-Propelled Horizontal Cask Transporter and Skid may be used with the NUHOMS-24P and -32P DSC.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The prime contractor for design, analysis, and component supply for the Calvert Cliffs ISFSI was Transnuclear (formerly Nutech Engineers, Inc.), of Columbia, MD. The ISFSI is owned by CCNPP, LLC and operated by Exelon Generation. Construction of the ISFSI is the responsibility of an approved construction contractor. For the poured in place HSMs, licensing support, geotechnical engineering, and Quality Assurance Program revisions were performed by Duke Engineering & Services, Inc., utilizing Duke Power Company personnel experienced on the Oconee Nuclear Station ISFSI. Subsurface investigations at the ISFSI were performed by Law Engineering Testing Company.

Geotechnical, civil, structural, mechanical, and electrical engineering support for the installation of the HSM-HB modules was provided by Sargent & Lundy LLC and MACTEC Engineering and Consulting, Inc. (formerly Law Engineering Testing Company). Subsurface analysis for the installation of the HSM-HB modules was performed by Sargent & Lundy LLC.

1.5 MATERIAL INCORPORATED BY REFERENCE

The Topical Reports for the Nutech Horizontal Modular Storage Systems for Irradiated Nuclear Fuel, (NUH-002, Revision 1A, July 1989) Reference 1.2, and (NUH-001, Revision 1A, June 1986) Reference 1.3, are hereby incorporated into this document by reference and referred to by Section in Table 1.5-1.

TABLE 1.5-1
SECTION REFERENCE FOR NUHOMS TOPICAL REPORTS

TITLE	REPORT NO.	SUBMITTAL DATE	SAR SECTIONS IN WHICH REFERENCED
Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, Nutech Engineers, Inc.	NUH-002, Revision 1A, July 1989	July 14, 1989	1.1, 1.2, 1.3, 1.5, 1.6, 3.1, 3.2, 3.3, 4.1, 4.2, 4.7, 7.1, 7.2, 7.3, 7.4, 8.1, 8.2
Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Fuel, NUHOMS-07P, Nutech Engineers, Inc.	NUH-001, Revision 1A, June 1986	June 27, 1986	7.1

1.6 REFERENCES

1.1 Deleted

1.2 Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, Nutech Engineers, Inc., NUH-002, Revision 1A, July 1989

1.3 Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Fuel, NUHOMS-07P, Nutech Engineers, Inc., NUH-001, Revision 1A, June 1986

1.4 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)

1.5 Deleted

1.6 CCNPP Calculation CA05924, Calvert Cliffs ISFSI/NUHOMS-24P Radiation Dose Rates for Cask Loading and Transfer

1.7 CCNPP Calculation CA06297, Transfer Thermal Analysis, 103°F Ambient

1.8 CCNPP Calculation CA06329, NUHOMS-32P – Transfer Cask Structural Analysis

1.9 CCNPP Drawing 84227SH0001, NUHOMS-32P DSC Parts List

1.10 Transnuclear, Inc., "Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1004, Transnuclear Document No. NUH-003, Revision 11

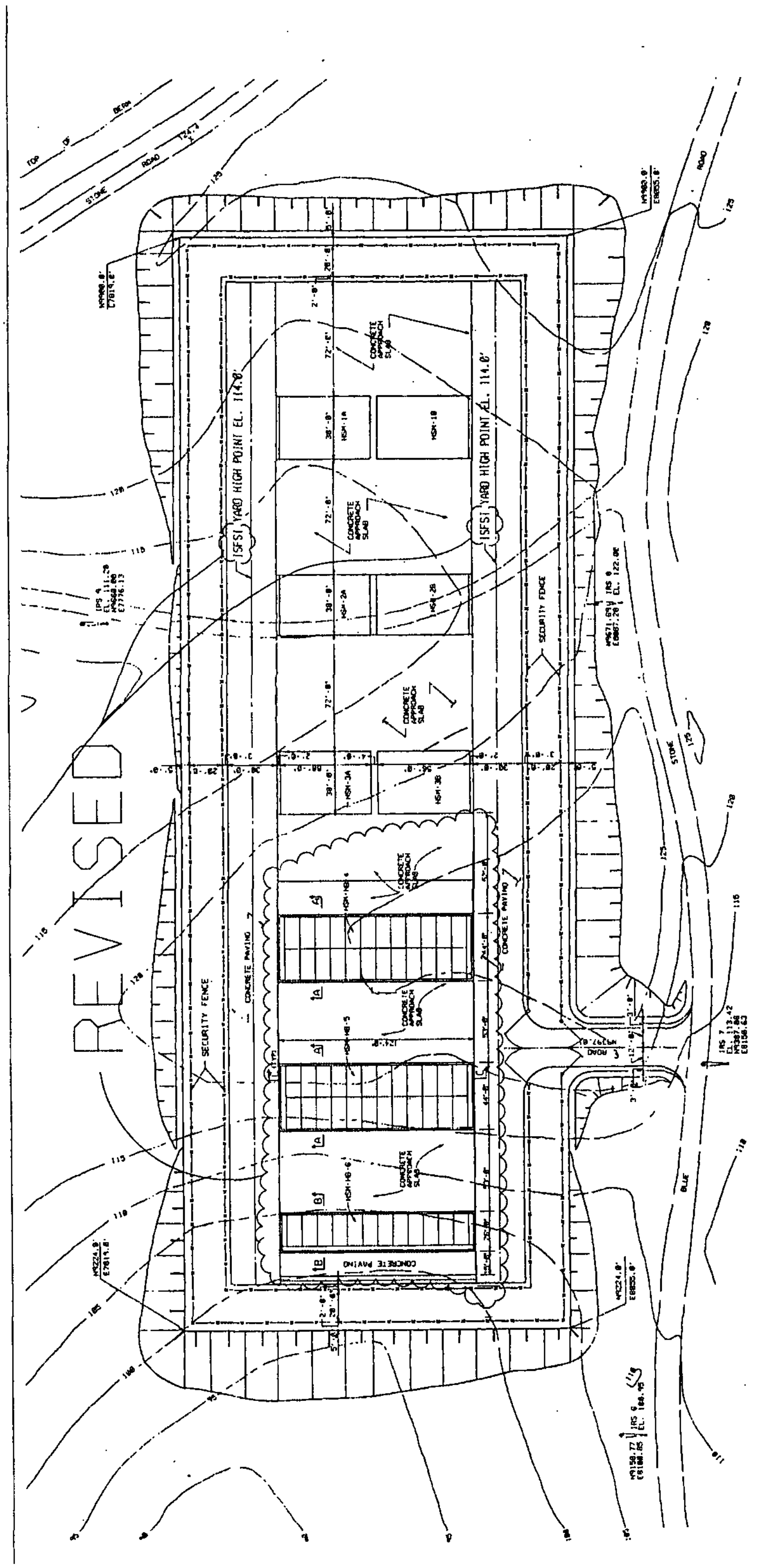
1.11 Transnuclear Inc., "Updated Final Safety Analysis Report for the NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1030, Revision 2

:liffs Site Plan
ffs Nuclear Power Plant
/, Maryland

plan showing the relative location
ith respect to the power plant.

Revision 3

DRAFT



Calvert Cliffs Independent Spent Fuel Storage Installation	GENERAL LOCATION	USAR Figure 1.2-1 Rev. XX

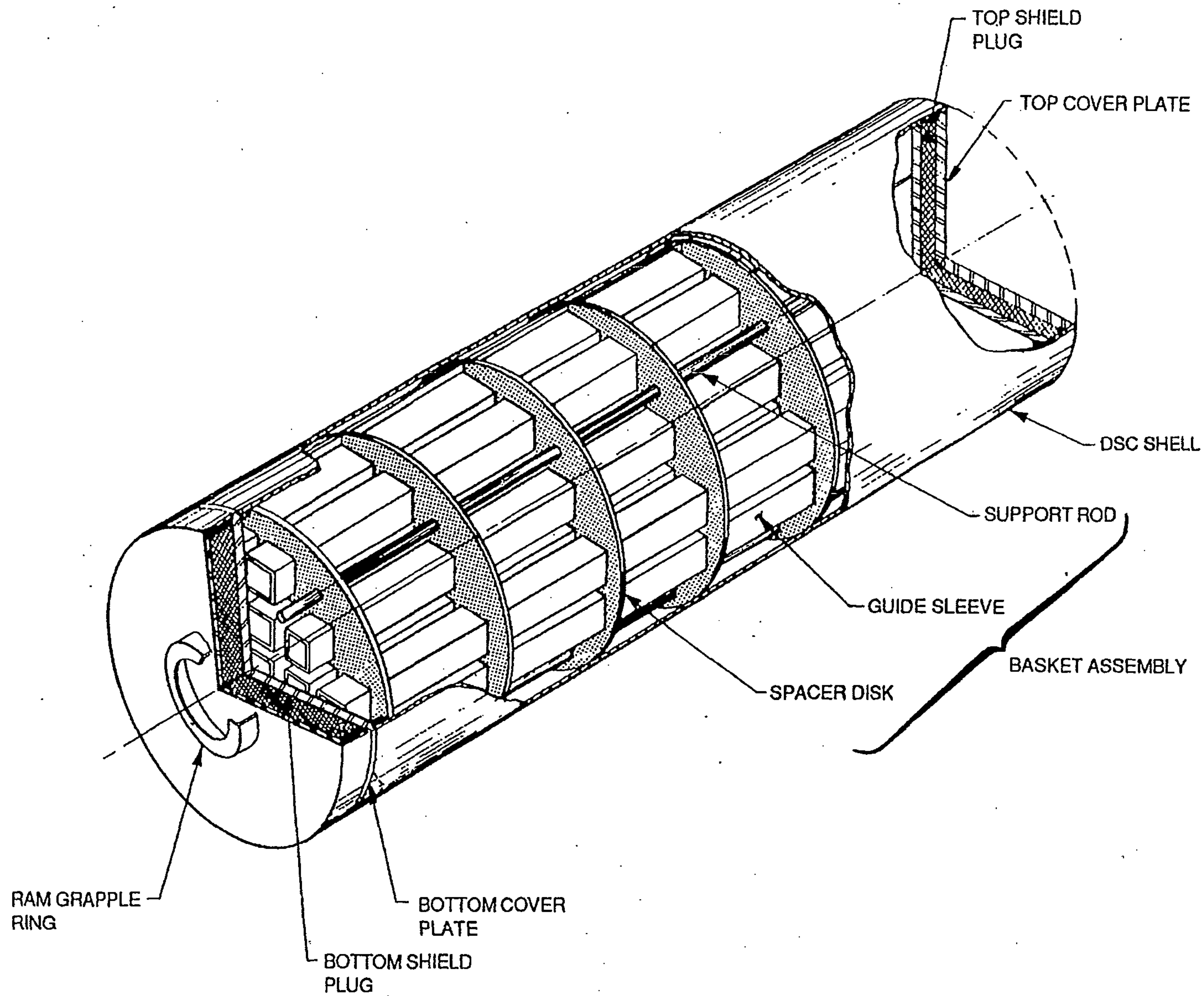
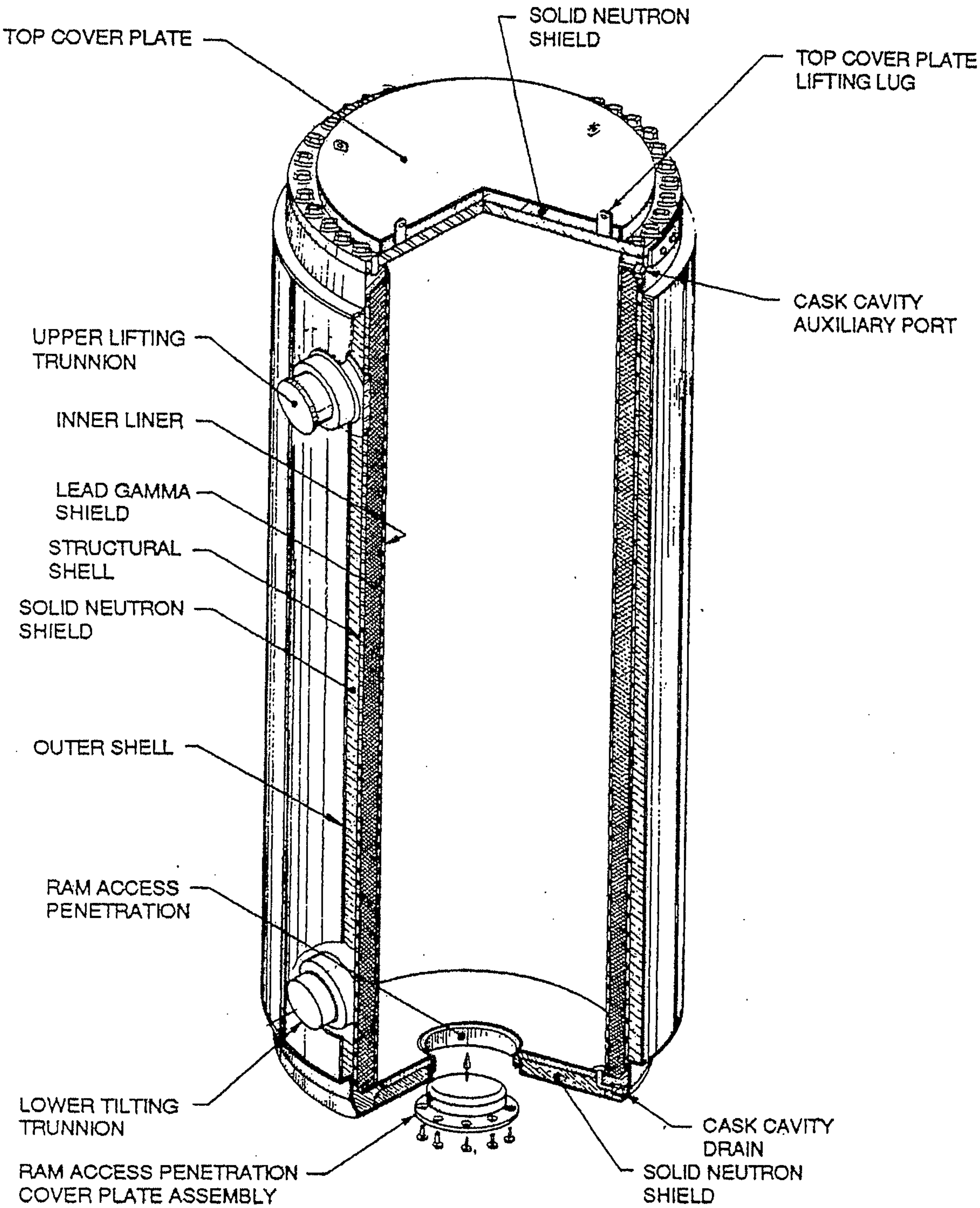
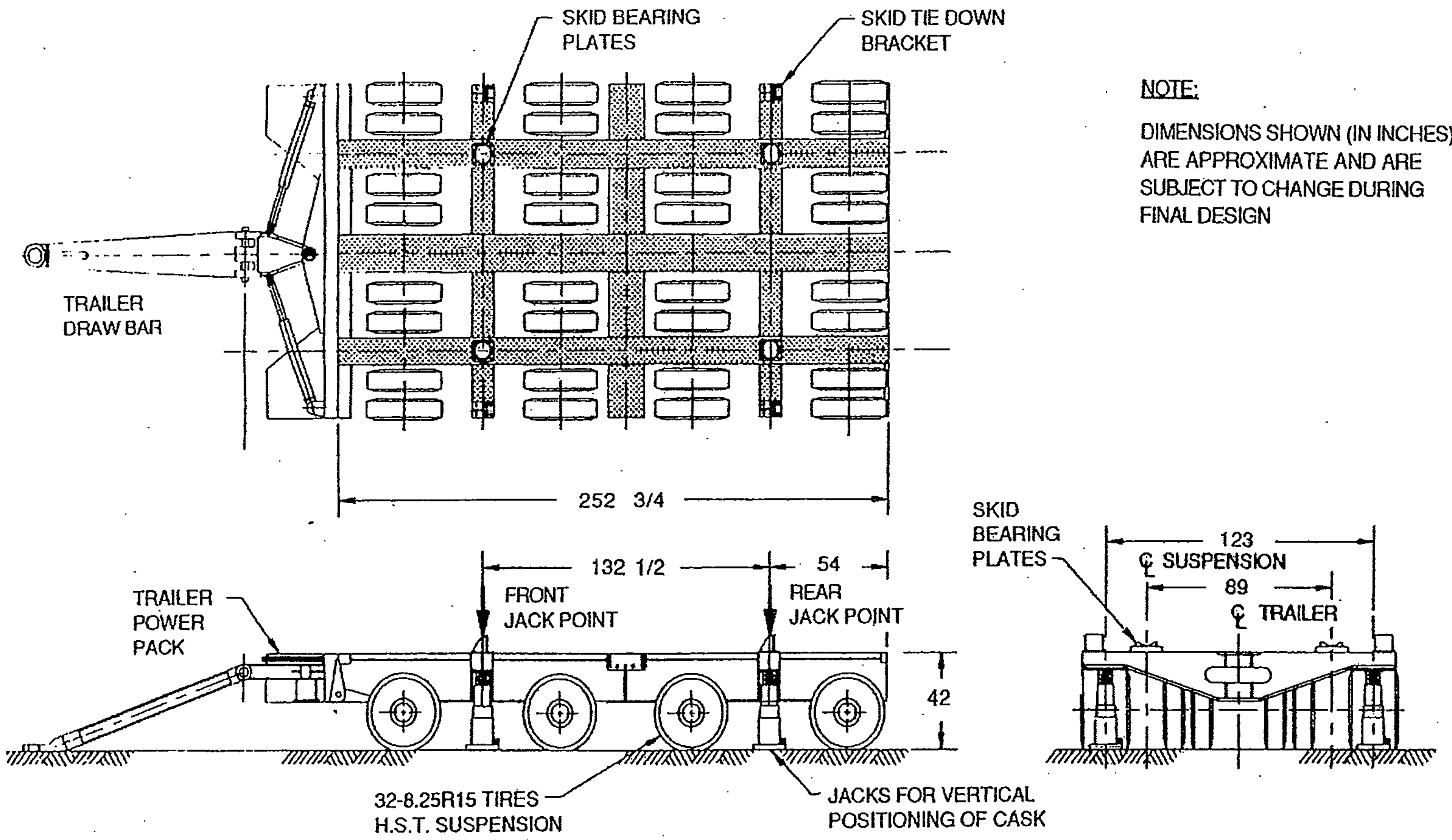


Figure 1.3-1

NUHOMS®-24P Dry Shielded Canister Assembly Components



Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	NUHOMS ON-SITE TRANSFER CASK	USAR Figure 1.3-2 Revision 15



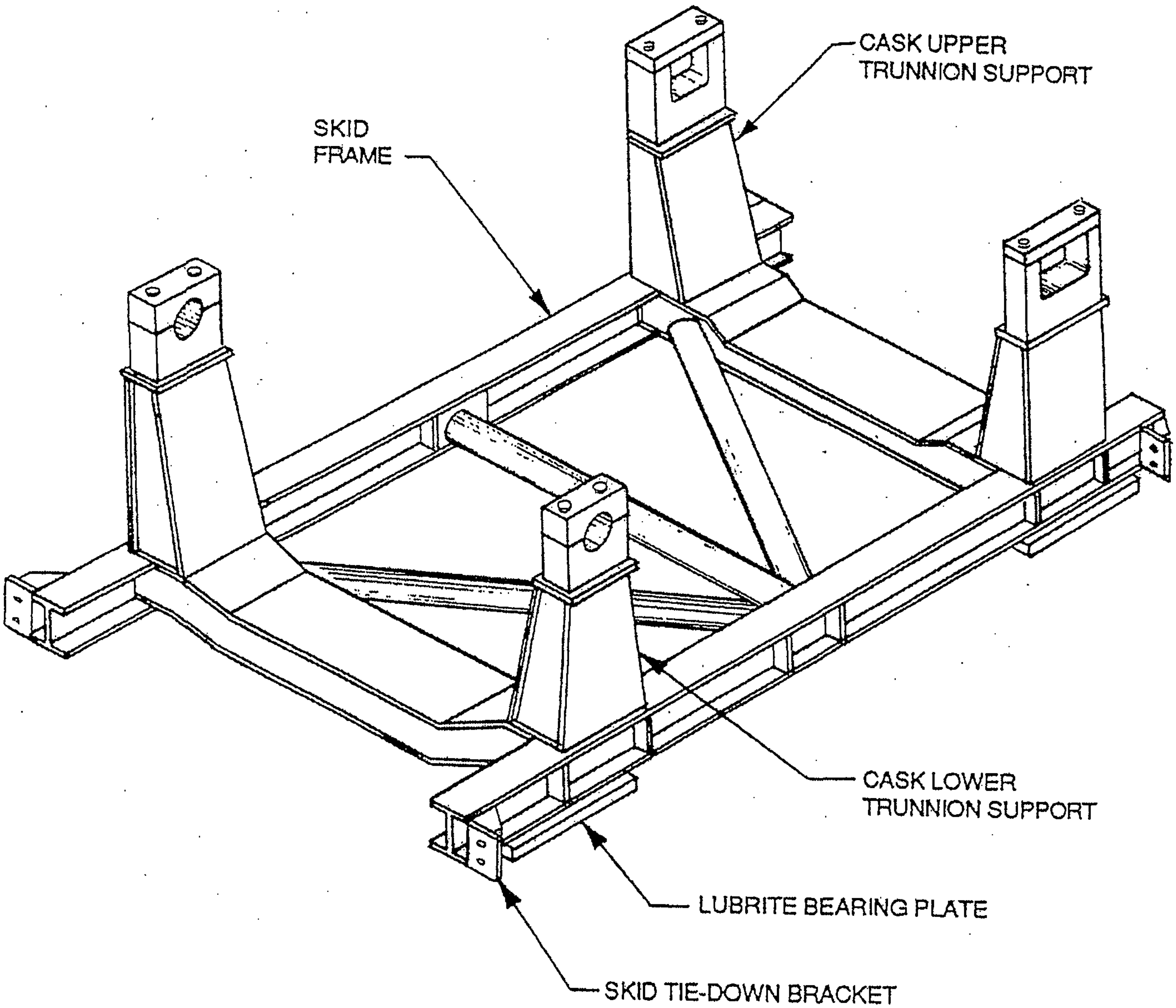
NOTE:

DIMENSIONS SHOWN (IN INCHES) ARE APPROXIMATE AND ARE SUBJECT TO CHANGE DURING FINAL DESIGN

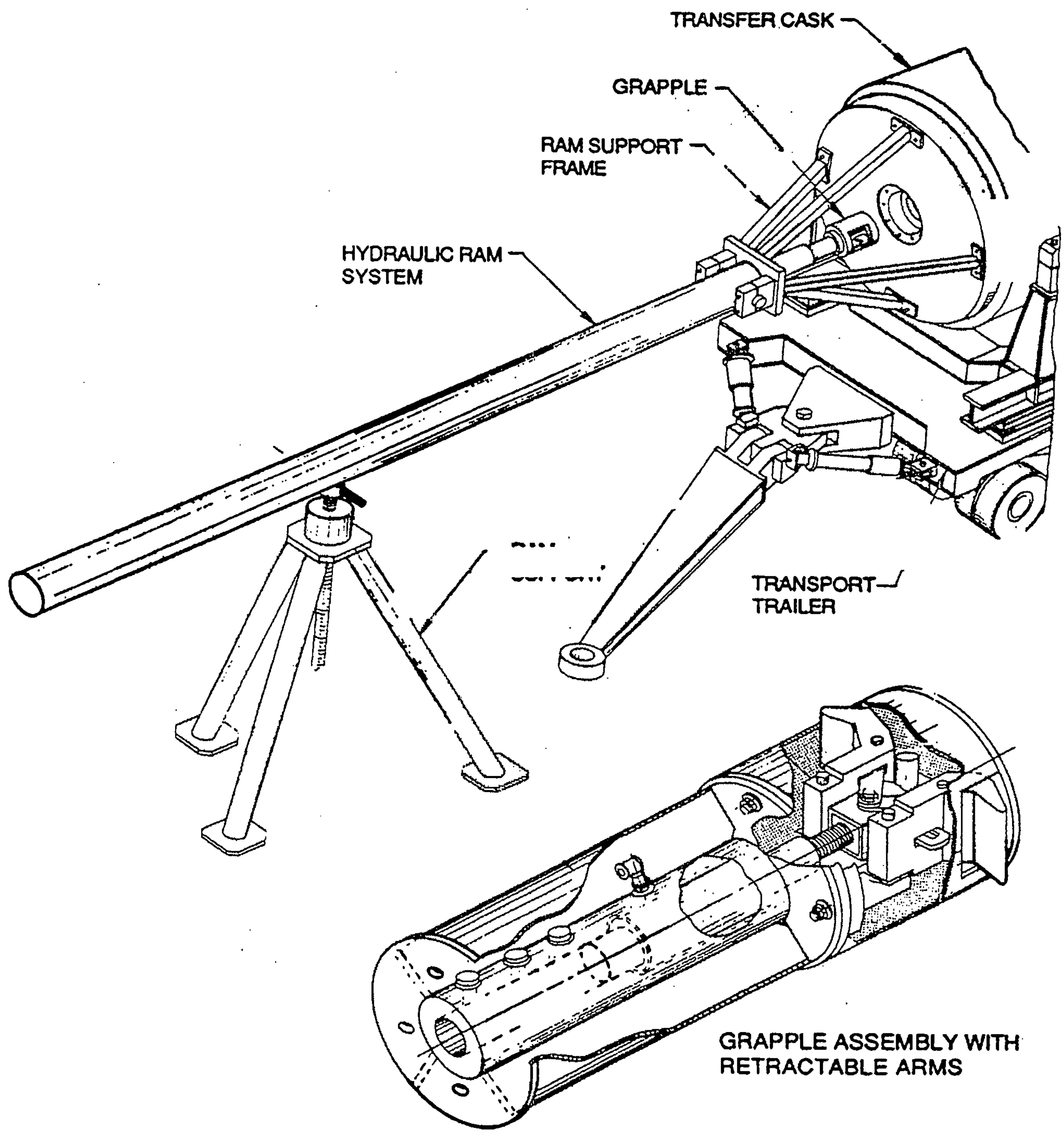
Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation

NUHOMS TRANSFER TRAILER
[See Reference 1.4]

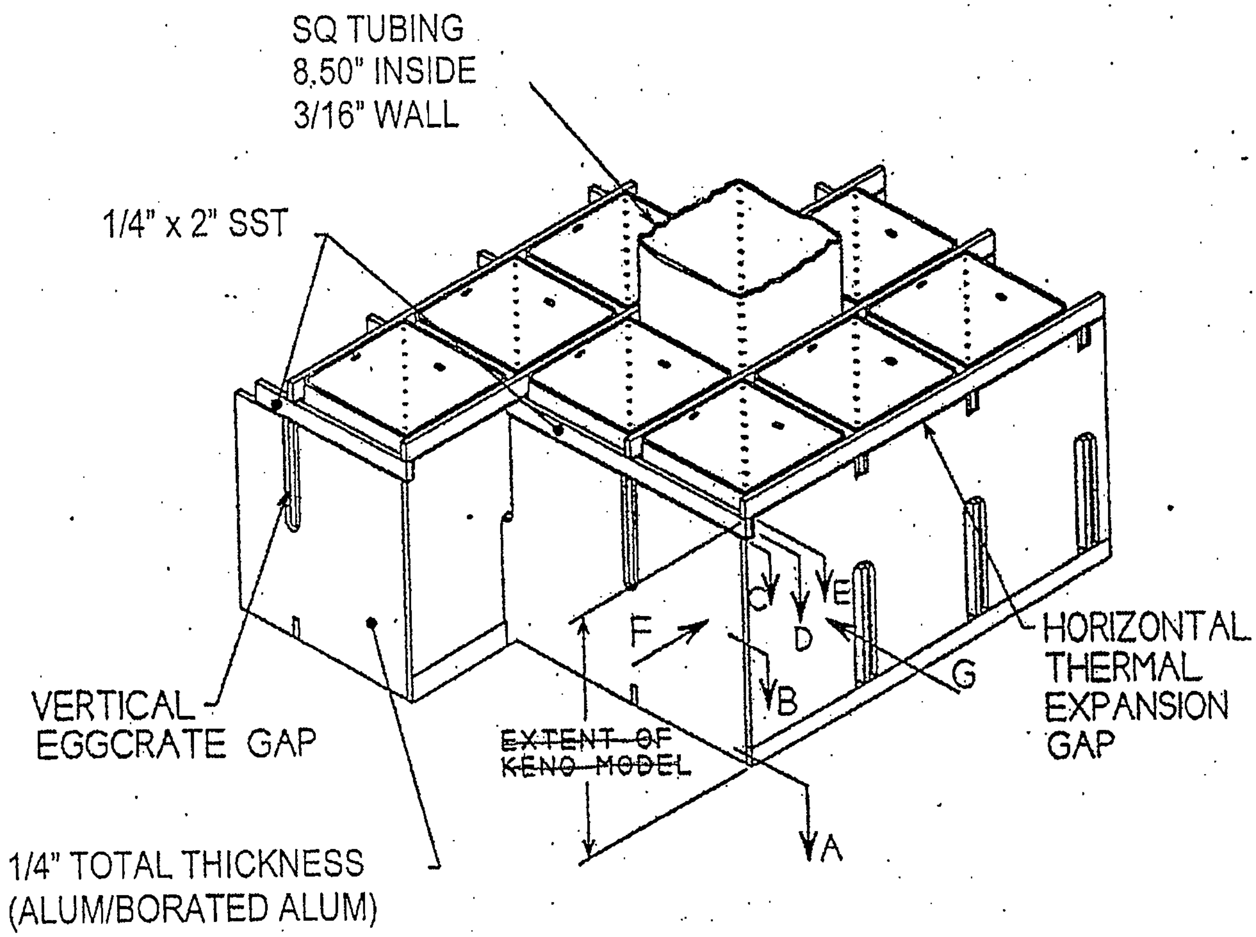
USAR
Figure 1.3-3
Revision 15



Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	NUHOMS CASK SUPPORT SKID	USAR Figure 1.3-4 Revision 15



Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	NUHOMS HYDRAULIC RAM SYSTEM	USAR Figure 1.3-5 Revision 15

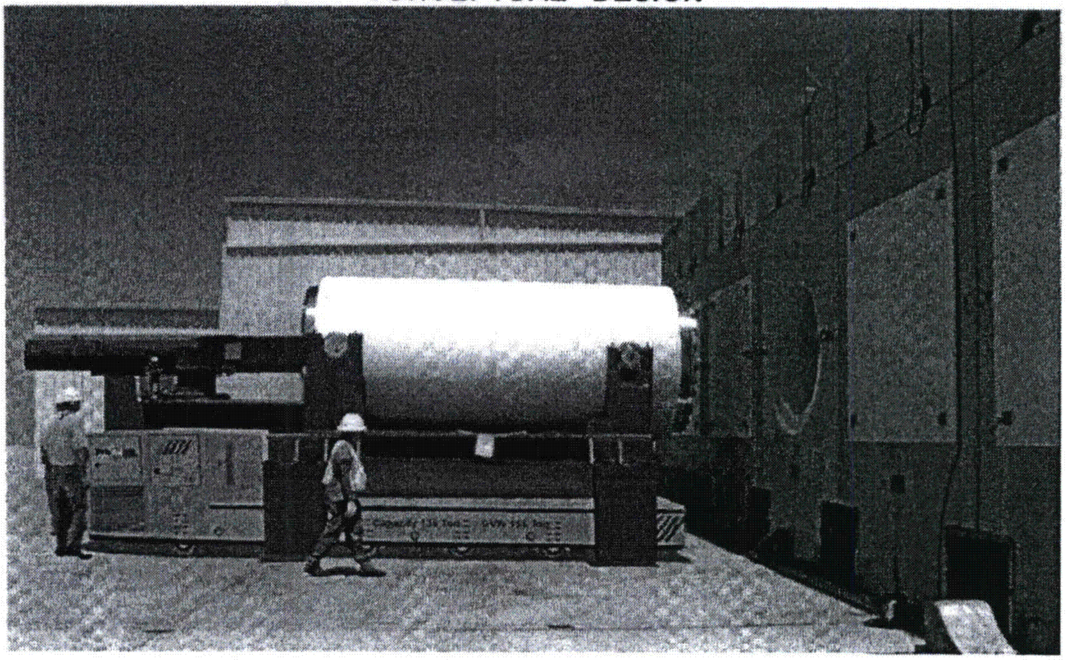


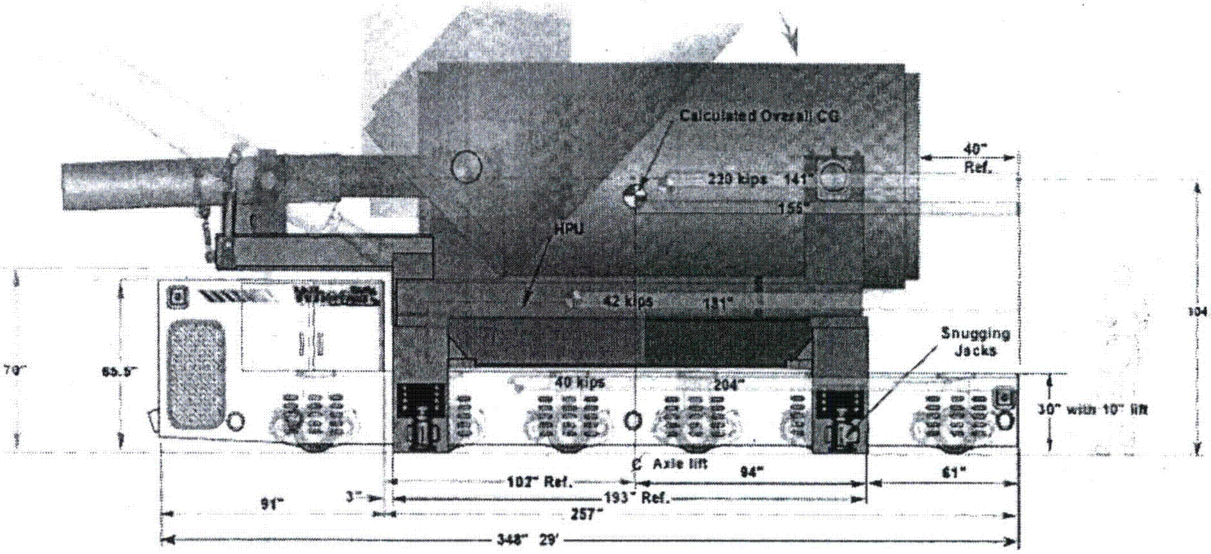
NOTE: All other Dry Shielded Canister components are the same as in Figure 1.3-1.

Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	NUHOMS-32P DRY SHIELDED CANISTER BASKET ASSEMBLY COMPONENTS	USAR Figure 1.3-7 Revision 15

DRAFT

CONCEPTUAL DESIGN





Calvert Cliffs Independent Spent Fuel Storage Installation	SELF-PROPELLED MODULAR TRANSPORTER WITH RAM AND SKID ASSEMBLY	USAR Figure 1.3-8 Rev. XX

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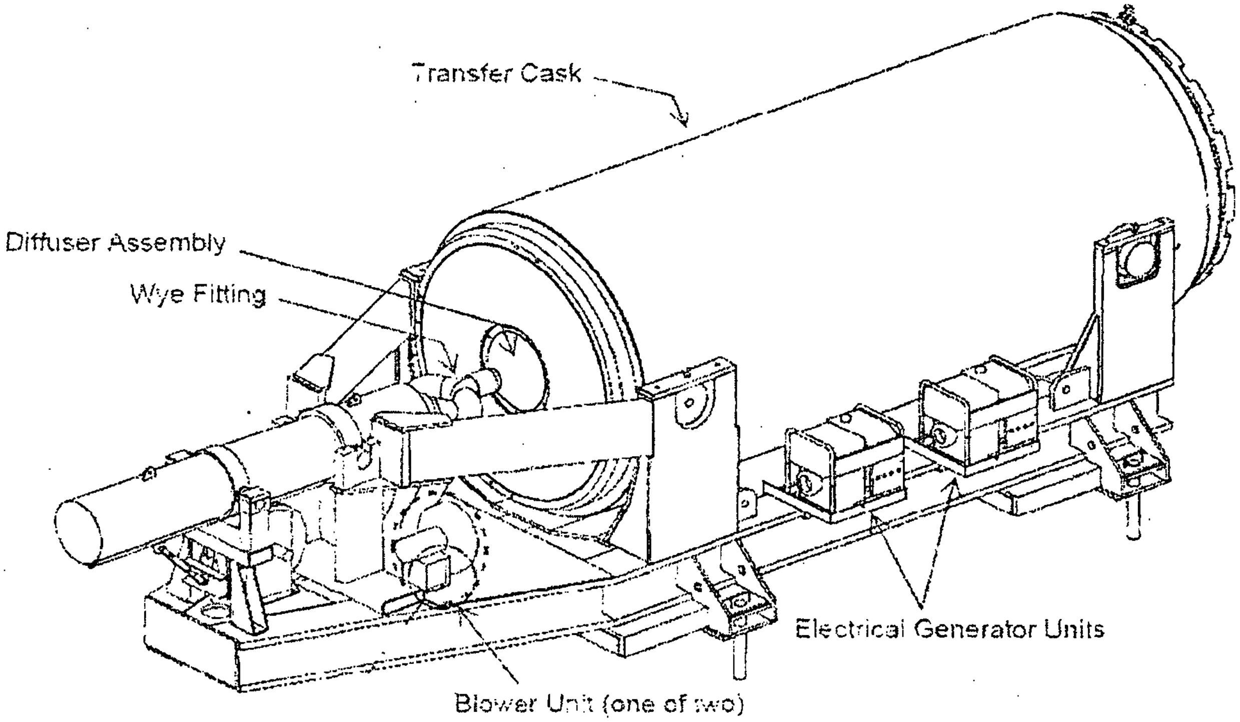
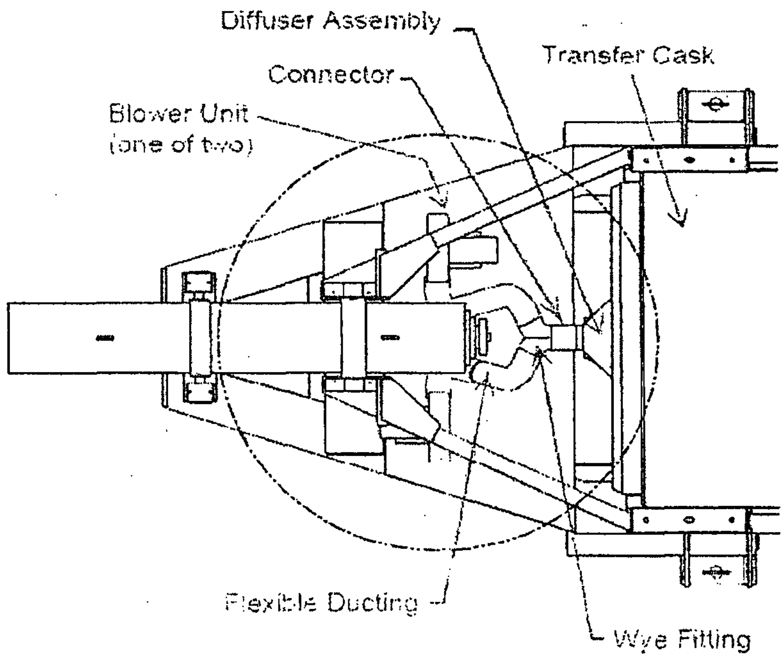
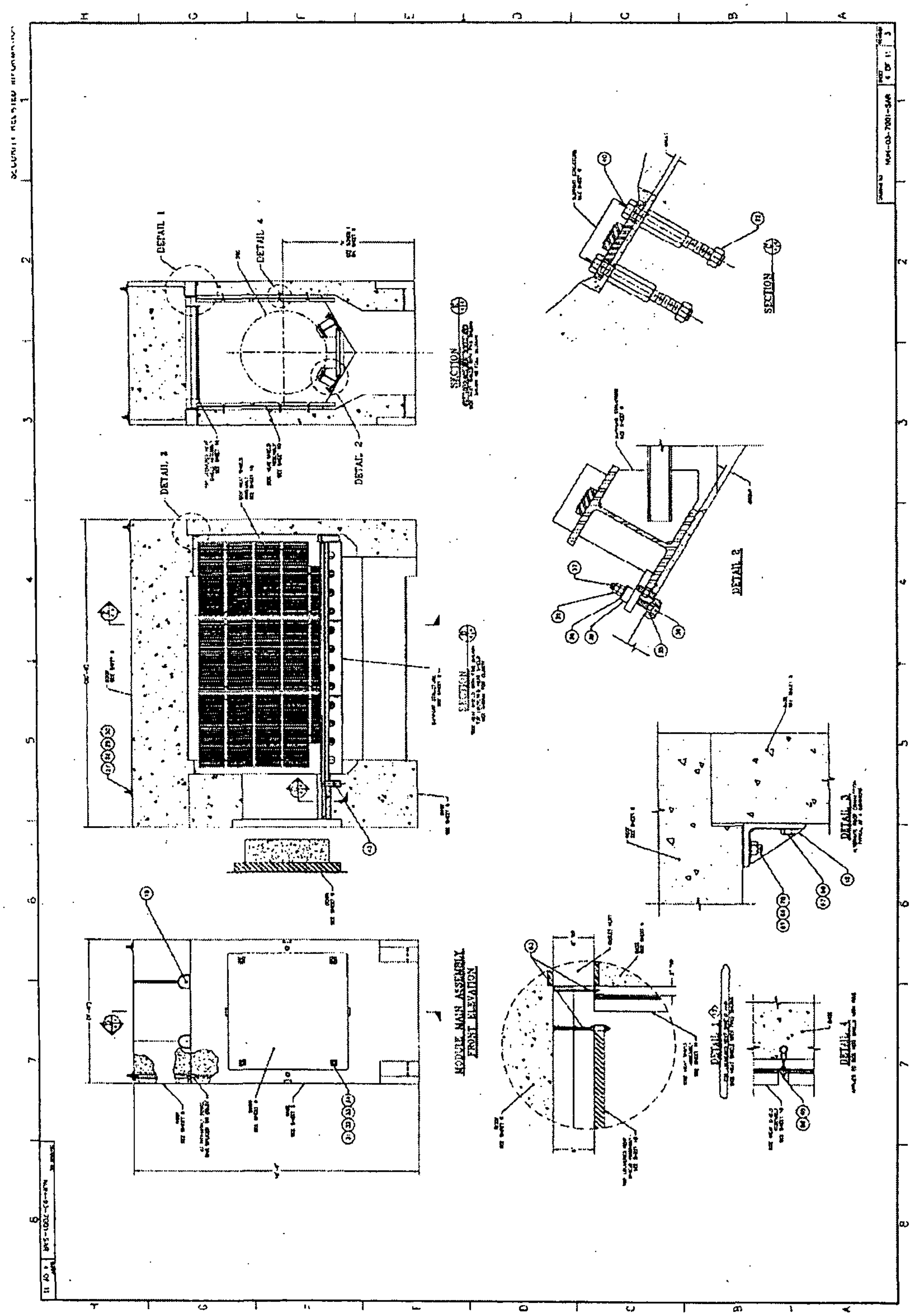




Calvert Cliffs Independent Spent Fuel Storage Installation	MODIFIED NUHOMS ON-SITE TRANSFER CASK	USAR Figure 1.3-9 Rev. XX

DRAFT



Calvert Cliffs Independent Spent Fuel Storage Installation	HSM-HB SCHEMATIC	USAR Figure 1.3-10 Rev. XX

CHAPTER 2

SITE CHARACTERISTICS

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2.3-4	9	Figure 2.4-6	0
2.3-5	22	Figure 2.5-1	0
2.3-6	22	Figure 2.5-2	0
2.3-7	15	Figure 2.5-3	0
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2.4-3	8	Figure 2.5-5C	0
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2.4-5	8	Figure 2.5-5E	0
2.4-6	8	Figure 2.5-5F	0
2.5-1	8	Figure 2.5-5G	0
2.5-2	8	Figure 2.5-5H	0
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SITE CHARACTERISTICS

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SITE CHARACTERISTICS

LIST OF ACRONYMS

ANSI	American National Standards Institute
BGE	Baltimore Gas and Electric Company
CAB	Controlled Area Boundary
CCNPP	Calvert Cliffs Nuclear Power Plant
HSM	Horizontal Storage Module
HSM-HB	High Burnup Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
LNG	Liquified Natural Gas
MLW	Mean Low Water
MRI	Meteorological Research Institute
MSL	Mean Sea Level
PMH	Probable Maximum Hurricane
SPT	Standard Penetration Test
SSI	Soil-Structure Interaction

2.0 SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 SITE LOCATION

The Independent Spent Fuel Storage Installation (ISFSI) is located on the Calvert Cliffs Nuclear Power Plant (CCNPP) site in Calvert County, MD at latitude 38°-25'-39.7" N and longitude 76°-26'-45" W. Calvert Cliffs Nuclear Power Plant, LLC owns and operates CCNPP. The power plant is approximately 10.5 miles southeast of Prince Frederick, MD, and is situated on the west bank of the Chesapeake Bay. The ISFSI is sited approximately 2300' southwest of the Power Plant at Elevation 114.0' above Mean Sea Level (MSL) and about 70' above the existing plant yard elevation. Figure 2.1-1 shows the ISFSI location with respect to neighboring states and counties within 50 miles. The metropolitan centers closest to the ISFSI are: Washington, DC, approximately 45 miles to the northwest; Baltimore, MD, approximately 60 miles to the north; Richmond, VA, approximately 80 miles to the southwest; and Norfolk, VA, approximately 110 miles to the south.

2.1.2 SITE DESCRIPTION

Figure 2.1-2 shows the ISFSI, property line, controlled area, power plant structures and general features of the area. The controlled area for the ISFSI is within the property boundary of the CCNPP, which covers 962 acres, and has a minimum radius of 3900' (1189 m). Calvert Cliffs Nuclear Power Plant, LLC owns all of the property within the controlled area boundary (CAB). Local and regional topography are shown on Figures 2.4-1 and 2.4-2, respectively.

2.1.2.1 Other Activities Within the Site Boundary

The ISFSI is located within the owner controlled area of the nuclear plant. The interaction between the ISFSI and the power plant is described in Chapter 5. Other non-plant related activities are limited to CCNPP's Visitors Center; Camp Conoy, a summer camp used by CCNPP for various recreational purposes; and a working farm of about 100 acres.

2.1.2.2 Boundaries for Establishing Effluent Release Limits

There are no effluent releases from the ISFSI. The CAB is shown in Figure 2.1-2. Access will be actively controlled at the ISFSI security perimeter fence. The minimum distance from the ISFSI to the CAB is 3900' (1189 m).

2.1.3 POPULATION DISTRIBUTION AND TRENDS

2.1.3.1 Population Within 10 Miles

Figure 2.1-3 shows the general locations of the ISFSI and the towns and other cultural features within 10 miles of the site. The population distribution is based on the 1980 census and the annual land use survey done for the Calvert Cliffs Radiological Environmental Monitoring Program. Table 2.1-1 gives the population distribution within 10 miles of Calvert Cliffs.

Between 1970 and 1980 the population of Calvert County increased by an average of 5.19% annually. The Tri-County Council of Southern Maryland projects an increase of almost 75% between 1980 and the year 2010. Solomons and the St. Leonard/Long Beach communities are major growth areas within the 10 mile radius of the ISFSI. Table 2.1-2 shows the projected population within 10 miles of the ISFSI for the year 2010 based upon the projections of the Tri-County Council.

2.1.3.2 Population Between 10 and 50 Miles

The population distribution shown in Table 2.1-3 is based on the 1980 census. Projections of population, Table 2.1-4, are based on the growth rates used in the CCNPP Updated Final Safety Analysis Report. Major population centers are Washington, DC, 45 miles northwest of the ISFSI with a 1980 population, including surrounding urbanized areas, of 2,763,105 and Annapolis, MD, population 31,740, located 40 miles to the north.

2.1.3.3 Transient Population

Winter daytime population variations are caused by schools, major employers, and facilities where a significant number of people gather regularly during the winter months. Summer daytime population variations are due to visitor attractions which include recreation areas, attractions such as a marine museum, and marinas. The summer night transient population adds 22.8% to the permanent population in Calvert County. St. Mary's County sees a 2% increase, Dorchester County a 21% increase. Table 2.1-5 identifies major public institutions within the 10 mile radius and their associated current populations.

2.1.4 USES OF NEARBY LAND AND WATERS

The Calvert Cliffs ISFSI is located approximately 3000' west of the western shore of the Chesapeake Bay. The Bay is used by the public for fishing, shellfish harvesting, boating, and swimming. The area within 10 miles of the ISFSI covers parts of Calvert, St. Mary's, and Dorchester counties. The counties are predominantly rural and are characterized by farmland and wetlands. The major crops are tobacco and corn.

Five miles to the southeast of the ISFSI is the Calvert Cliffs State Park. Ten miles south of the ISFSI is the Patuxent River Naval Air Station. Agricultural activities in the immediate area of the ISFSI have been addressed in the Environmental Report for CCNPP Units 1 and 2.

TABLE 2.1-1
1980 POPULATION DISTRIBUTION WITHIN 10 MILES

	0-1 Mile	1-2 Miles	2-3 Miles	3-4 Miles	4-5 Miles	5-10 Miles
N	0	0	0	0	0	0
NNE	0	0	0	0	0	0
NE	0	0	0	0	0	0
ENE	0	0	0	0	0	231
E	0	0	0	0	0	43
ESE	0	0	0	0	0	8
SE	0	2	0	2	2	0
SSE	0	10	0	183	150	327
S	0	24	29	77	217	2,207
SSW	0	32	70	55	162	2,755
SW	0	17	37	135	107	3,582
WSW	0	58	90	99	152	2,616
W	0	22	41	24	107	925
WNW	0	50	10	39	48	755
NW	0	0	660	226	210	650
NNW	0	0	0	0	0	409
TOTALS:	0	215	937	840	1,366	14,508

TABLE 2.1-2
PROJECTED POPULATION WITHIN 10 MILES FOR THE YEAR 2010

	0-1 Mile	1-2 Miles	2-3 Miles	3-4 Miles	4-5 Miles	5-10 Miles
N	0	0	0	0	0	0
NNE	0	0	0	0	0	0
NE	0	0	0	0	0	0
ENE	0	0	0	0	0	296
E	0	0	0	0	0	58
ESE	0	0	0	0	0	10
SE	0	4	0	4	442	0
SSE	0	18	0	385	312	515
S	0	43	42	649	449	12,548
SSW	0	42	76	113	338	5,299
SW	0	26	43	277	674	4,618
WSW	0	270	114	426	314	3,908
W	0	38	72	96	222	1,432
WNW	0	52	17	182	64	1,446
NW	0	0	1,086	462	2,651	852
NNW	0	0	0	0	0	624
TOTALS:	0	492	1,450	2,594	5,465	31,607

TABLE 2.1-3
1980 POPULATION DISTRIBUTION, 10 - 50 MILES

	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles
N	0	5,499	71,328	102,207
NNE	193	5,501	5,755	9,266
NE	833	6,105	12,617	9,817
ENE	2,295	16,226	8,724	20,296
E	522	1,204	4,230	41,637
ESE	742	781	3,852	19,305
SE	227	205	1,122	8,549
SSE	418	418	355	1,259
S	17,082	2,443	4,494	6,683
SSW	8,371	3,210	6,559	4,881
SW	5,317	1,919	4,228	6,087
WSW	4,896	4,093	7,690	6,487
W	5,518	5,836	5,821	19,038
WNW	8,047	24,283	39,497	163,167
NW	6,144	18,770	249,650	1,056,378
NNW	10,028	15,223	76,161	234,819
TOTALS:	70,633	111,716	502,083	1,709,876

TABLE 2.1-4
PROJECTED POPULATION, 10 - 50 MILES FOR THE YEAR 2010

	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles
N	0	9,137	105,460	147,511
NNE	209	8,384	6,781	12,695
NE	940	10,906	17,684	11,937
ENE	2,690	33,949	14,531	21,893
E	634	1,957	6,603	51,275
ESE	934	985	5,634	27,124
SE	337	313	1,776	12,865
SSE	732	772	608	2,029
S	29,516	4,112	6,659	10,166
SSW	14,263	4,924	8,407	7,008
SW	8,891	3,221	5,445	8,466
WSW	8,034	7,516	9,950	8,739
W	9,035	9,819	8,426	30,537
WNW	13,146	37,426	63,931	311,316
NW	9,490	31,363	432,300	1,799,508
NNW	14,644	27,570	141,067	356,982
TOTALS:	113,495	192,354	835,262	2,820,051

TABLE 2.1-5
MAJOR PUBLIC INSTITUTIONS WITHIN 10 MILES

FACILITY	SECTOR LOCATION	POPULATION
Governmental		
Naval Ordnance Lab and Recreation Facility	6-8 miles S & SSW	1,500
Calvert Cliffs State Park	2-4 miles S, SSE & SE	325
Chesapeake Biological Lab	8-9 miles S	125
Battle Creek Cypress Swamp Nature Area and Visitor Center	9-10 miles WNW	100
Flag Ponds Nature Park	1-2 miles NW	100
Calvert County Marine Museum	7-8 miles S	200
Schools		
Appeal Elementary	4-5 miles S	837
Mutual Elementary	6-7 miles WNW	782
Southern Middle	1-2 miles SSW	597
Our Lady Star of the Sea	7-8 miles S	120
Town Creek Elementary	9-10 miles SSW	320
St. John Elementary	9-10 miles SW	237
Hollywood Elementary	9-10 miles SW	268
Private Facilities		
Calvert Cliffs Nuclear Power Plant	2300' NW	1,100
Columbia Gas System	3-4 miles SSE	105

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

There are no other major nuclear facilities within a 50 mile radius of the ISFSI, other than the CCNPP. The Patuxent River Naval Air Station, located 10 miles south of the ISFSI, is the only nearby military installation. Nearby transportation routes are shown on Figure 2.1-3. The only industrial facility within approximately 10 miles of the ISFSI is the Cove Point Liquified Natural Gas (LNG) terminal and pipeline. There are no tall structures or discharge stacks on-site with the potential to impact the ISFSI upon collapse.

2.2.1 LOCAL AIRPORTS

Three airports operate within approximately 10 miles of the ISFSI and their location is shown on Figure 2.1-3. A helipad is located at the northern end of the CCNPP site, more than 1000' from the ISFSI. The helipad location is shown on Figure 2.1-2. The aircraft hazards analysis performed due to aircraft flights in the vicinity of CCNPP was found to be acceptable as documented in Reference 2.16.

2.2.1.1 Patuxent River Naval Air Station

Located approximately 10 miles south of the ISFSI is the only major aviation facility in the area. Known as the Patuxent River Naval Air Station, it operates all types of Naval Aircraft in test and development oriented missions. Most of the aircraft operate in specified restricted areas to the east and south; hence, their flight paths would be in these directions and away from the ISFSI. Aircraft involved in test and evaluation operations remaining in the local training pattern operate within 5 miles of the air station while involved in normal operations. Aircraft departing and arriving under Visual Flight Rules follow specific routes that go east and south and therefore their flight paths are away from the ISFSI, at altitudes of 1000' or 1500' depending on the type of aircraft.

Aircraft following Instrument Flight Rules follow preset arrival and departure procedures and none of these flight paths approach the ISFSI. Aircraft on a radar approach are vectored by a ground controller. However, in the event of loss of radar contact with the aircraft (and in training runs for such scenarios) some instrument approach and takeoff patterns pass at a ten nautical mile (11.5 mile) radius from the Naval Air Station, effectively flying aircraft directly over the ISFSI. Although these patterns pass over the ISFSI, it is unlikely that the aircraft come within three miles of the ISFSI because pilots should take a three mile bypass to avoid flyovers of the CCNPP site as directed.

Available information indicates about 100,000 takeoffs and landings per year with a peak of about 300 per day (Reference 2.17). The aircraft types and the maximum gross takeoff weights are listed in Table 2.2-1. The heaviest transient military aircraft visiting the base would be a Lockheed E-6A, which has a maximum gross takeoff weight of 240,000 lbs.

2.2.1.2 Chesapeake Ranch Airport

This private airport is located about 6 miles southeast of the ISFSI. It has a single 2,500' runway which limits the size of aircraft to single engine or light twin engine aircraft. No aircraft are permanently based there, and it is used mainly during summer months. A busy summer weekend would bring about six airplanes to the field. A representative sample is listed in Table 2.2-2.

The airport is located within the control zone of the Patuxent River Naval Air Station, which restricts free access to the field. All operations are under Visual Flight Rules only through a direct entry corridor provided by the Navy on a heading of 150° magnetic to the airport, at an altitude of 800'.

2.2.1.3 St. Mary's County Airport

This airport, located approximately 10 miles southwest of the plant, is owned by St. Mary's County. It has a single 3,250' runway with a full-length taxiway, a paint shop, four individual hangers and three ten-unit hangers. The airport is the base for about 100, mostly single engine, airplanes and can handle medium twin-engine, propeller-driven planes and small jet engine-driven planes. The Maryland State Police have a Dauphine rescue helicopter based at the airport (Tables 2.2-2 and 2.2-3).

The airport averages 3400 takeoffs and landings per month, with a peak daily activity of 300. Air traffic operates under Visual Flight Rules and Instrument Flight Rules with a pattern altitude of up to 2000'. The airport is situated on the edge of an airspace restricted for use by the Patuxent River Naval Air Station.

2.2.1.4 Corporate Helipad

A helipad is located at the northern end of the CCNPP site, more than 1000' from the ISFSI. Generally, this helipad is used for corporate flights from utility headquarters (normally less than 20 flights annually) and for an estimated six Medivac helicopter flights annually. There are no specific flight paths or exclusion areas for helicopter flights in the vicinity of the ISFSI; however, flight paths over the ISFSI are generally not used unless weather conditions warrant such a route to ensure a safe landing or takeoff. Helicopters using the corporate helipad weigh less than 12,000 pounds.

2.2.2 LIQUIFIED NATURAL GAS PLANT AND PIPELINE

The Cove Point LNG Terminal is located about 3.5 miles south-southeast of the Calvert Cliffs power plant. The Cove Point facility was designed as a receiving terminal for the importation of LNG with the capability of receiving LNG carriers at an average rate of once every 2.5 days. The LNG is unloaded and revaporized at the terminal and transported by a 36" pipeline to the Columbia transmission network at a connection in Loudoun County, VA.

The Cove Point Terminal, which received its first LNG shipment in March, 1978, has been idle since April 1980 but is scheduled to be reopened in 1994. The location of the terminal and pipeline relative to the ISFSI is shown on Figure 2.1-3. The effects of an LNG spill or explosion at the Cove Point Terminal or the pipeline are evaluated in Section 8.2.11.

TABLE 2.2-1
REPRESENTATIVE AIRCRAFT DATA AT THE PATUXENT RIVER NAVAL AIR STATION

AIRCRAFT	TYPE	MAXIMUM GROSS TAKE-OFF WEIGHT
A6	Attack bomber	60,626
A10	Attack bomber	50,000
AV8	V/STOL fighter/bomber	over 22,000
C2	Transport	54,830
C12	Transport	12,500
C130	Transport	175,000
C880	Transport	193,000
E2	Airborne early warning	49,638
E6	Transport	240,000
EA-6B	Electronic countermeasures	61,500
F14	Fighter	72,900
F18	Fighter	51,900
H1	Helicopter	16,800
H2	Helicopter	12,500
H3	Helicopter	21,000
H34	Helicopter	14,000
H46	Helicopter	23,000
H53	Helicopter	42,000
H60	Helicopter	23,000
H65	Helicopter	8,900
P3	Anti-Submarine Warfare patrol	139,760
S3	Anti-Submarine Warfare	52,539
T2	Trainer	13,284
T34	Trainer	4,425
T38	Trainer	12,500
T39	Trainer	18,650

TABLE 2.2-2
REPRESENTATIVE AIRCRAFT DATA, CHESAPEAKE RANCH AIRPARK

AIRCRAFT	TYPE	MAXIMUM GROSS TAKE-OFF WEIGHT
Aero Comm. Lark Comm.	Four place single engine	2,475
American Avia. AA1	Two place single engine	1,500
Beechcraft Musketeer	Four place single engine	2,750
Beechcraft Bonanza	Four/five place single engine	3,400
Beechcraft Baron E55	Four/six place twin engine	5,300
Cessna 150	Two place single engine	1,600
Cessna 172	Four place single engine	2,300
Cessna 177	Four place single engine	2,500
Cessna 182	Four place single engine	2,950
Cessna 206	Six place single engine	3,600
Cessna 210	Six place single engine	3,800
Cessna 310	Five place twin engine	5,300
Cessna 337	Four place twin engine	4,400
Champion Citabria	Two place single engine	1,650
Lake Amphibian	Four place single engine	2,400
Mooney Ranger	Four place single engine	2,575
Mooney Statesman	Four place single engine	2,532
Mooney Executive	Four place single engine	2,740
Mooney M22	Five place single engine	3,680
Piper Super Cub	Two place single engine	1,750
Piper Cherokee series	Two-four place single engine	2,900
Piper Cherokee Six	Six place single engine	3,400
Piper Comanche C	Four place single engine	3,200
Piper Twin Comanche	Four place twin engine	3,600
Piper Aztec	Six place twin engine	5,200

TABLE 2.2-3
REPRESENTATIVE AIRCRAFT DATA, ST. MARY'S COUNTY AIRPARK

St. Mary's County Airport would accommodate all of the aircraft listed in Table 2.2-2, as well as the following heavier types:

AIRCRAFT	TYPE	MAXIMUM GROSS TAKE-OFF WEIGHT
Aero Comm. Shrike Comm.	Four/six place twin engine	6,750
Aero Comm. Hawk Comm.	Eight place twin engine	9,400
Beechcraft Duke	Four/six place twin engine	6,775
Beechcraft Queen Air	Six/eleven place twin engine	8,800
Beechcraft King Air	Six/ten place twin engine	12,500
Cessna 401	Six/seven place twin engine	6,300
Cessna 414	Six/seven place twin engine	6,350
Cessna 421	Six/eight place twin engine	7,250
Citation I & II	Six/eight place jet engine	12,500
Piper Navajo	Six/ten place twin engine	6,200

2.3 METEOROLOGY

2.3.1 REGIONAL CLIMATOLOGY

2.3.1.1 Data Sources

Data acquired by the National Weather Service and summarized by the Environmental Data Services were used to determine the regional climatology as shown in Table 2.3-1. Local long-term weather station data were used from the Patuxent River Naval Air Station for periods of record from 1949 to 1980.

2.3.1.2 General Climate

The Chesapeake Bay area is marked by generally mild winters and summers. The Bay absorbs much of the sun's heat during the day and releases some of it during the night when the air above the water becomes cooler. This area is in the belt of "prevailing westerlies." Most of the weather comes from a westerly direction across the continental United States, or directly from Canada. Cold air masses, such as those which produce the colder days of winter and the cool days of summer, come generally from the northwest. Warm air masses originate either in the desert and plateau sections of the southwestern states and Mexico, or from over the Gulf of Mexico. Those from the Gulf generally produce considerable precipitation, while those from the desert regions generally produce warm or hot dry periods.

Northwesterly winds prevail from October to April, and southwesterly winds prevail in the warmer months. Ordinarily, the velocity of the wind varies directly with the intensity of the low pressure area and inversely with the distance from its center. The usual diurnal variations in wind speed occur with a minimum generally before dawn, increasing with the daily temperature to a maximum at the time of highest temperature. High winds of destructive velocity are rare.

2.3.1.3 Severe Weather

2.3.1.3.1 Maximum and Minimum Temperatures

As listed in Table 2.3-1, a maximum recorded local temperature of 103°F was recorded at Patuxent River Naval Air Station in July, 1980. The minimum temperature of -3°F was recorded at Patuxent River Naval Air Station in 1956 and 1977. In the region, a maximum temperature of 105°F during August, 1983 and a minimum temperature of -7°F in January, 1984 were recorded at Baltimore-Washington International Airport.

The Nuclear Regulatory Commission has not provided explicit guidance on selection of design temperatures for ISFSI analysis. The maximum and minimum temperatures presented in Table 2.3-1 are based on the historic extreme recorded temperatures at Patuxent River Naval Air Station. The CCNPP ISFSI site is located on the Chesapeake Bay, just 10 miles NNE of the Patuxent River Naval Air Station.

The station is an National Oceanic & Atmospheric Administration station which has been recording data continuously since 1945. It represents the most accurate record available for weather in the vicinity of CCNPP and is referenced as such in the CCNPP Final Safety Analysis Report. The data for Washington, DC and Baltimore, MD, was included in Table 2.3-1 to demonstrate that the Chesapeake Bay has a moderating influence on the temperature extremes. LaPlata and College Park measurement locations are located further inland from the Bay and have larger variations in temperature extremes.

Document NUREG/CR-1390 (Reference 2.14) is not included as a requirement in the criteria of 10 CFR Part 72, or by other documents providing guidance for the design and analysis of the CCNPP ISFSI. NUREG/CR-1390 itself also does not provide criteria for selection of design temperatures. If the probabilistic estimates of temperature extremes from Reference 2.14 are used for the location of the CCNPP ISFSI site, it will yield the following values. NUREG/CR-1390 presents three sets of isotherms for the U.S. representing the 2-year, 50-year, and 100-year return values for maximum and minimum temperatures corresponding to 0.5, 0.98, and 0.99 probability levels. The 2-year max/min return values for CCNPP are 98°F/8°F and the 50-year return values are 104°F/-8°F. The design values chosen for the CCNPP ISFSI, which has a 20-year license period, are 103°F/-3°F. These values are very close to the 50-year return values. The 100-year return values are 106°F/-16°F, not significantly different from the design values chosen.

Published data for the Patuxent River Naval Air Station shows the average daily maximum temperatures to be 85°F (July) and the average daily minimum temperature of 29°F. Also, Reference 2.15, page 24.8, shows that for Baltimore (latitude 39° 20' and longitude 76° 25') the 1% summer (July) and 99% winter (January) temperatures are 92°F and 14°F and the median of annual extremes are 97.9/7.2°F, respectively.

Based on these temperatures, the design temperatures of 103°F/-3°F are more closely applicable to the CCNPP ISFSI site than the suggested values of 116°F/-40°F. The 116°F/-40°F values were selected by other ISFSI site applicants applicable for their ISFSI site locations only.

Regulatory Guide 3.48 requires that the analysis consider off-normal and accident conditions for the ISFSI and refers to the four categories of design events defined in American National Standards Institute (ANSI) 57.9. Off-normal analyses, as defined in Regulatory Guide 3.48, include Design Events I and II. These events are expected to occur "frequently" (I) or "with moderate frequency or on the order of once during a calendar year of ISFSI operation" (II). Accident conditions defined in Regulatory Guide 3.48 include Design Events III and IV which are "infrequent events that could reasonably be expected to occur during the lifetime of the ISFSI" or postulated events.

The historical extreme temperatures of -3°F and 103°F recorded at Patuxent River Naval Air Station were used for all four types of design events. It is clear that the local historical extreme temperatures meet the requirements for analyses of Design Events I and II. Since the Patuxent River data includes 45 years of records and the ISFSI license period is 20 years, it also seems clear that the criterion of "events that could reasonably be expected to occur during the lifetime of the ISFSI," is enveloped by these temperatures. Only in the case of postulated events should the use of the historical extremes be questioned. While 10 CFR Part 71 specifically includes postulated temperatures of -40°F and 125°F for analysis, such definitions are not included in 10 CFR Part 72, Regulatory Guide 3.48 and ANSI 57.9. The events defined in the SAR Design Event IV category include tornado wind and tornado generated missiles, the cask drop accidents, HSM air flow blockages, etc. For accident analyses where weather may affect system performance, the historical extreme temperatures were used and appear to be appropriate and defensible based on existing regulatory guidance.

2.3.1.3.2 Extreme Winds

Records from Baltimore-Washington International Airport indicate that the fastest recorded wind speed was 80 mph in March of 1952. Available data from other stations indicate slower maximum winds. Winds can be expected to reach a fastest speed in excess of 50 mph in any month of the year as an estimate of maximum winds to be encountered. Destructive velocities are rare and occur mostly during summer thunderstorms.

2.3.1.3.3 Tornadoes

Five tornadoes were observed during the 10-year period 1953-1962 in the general vicinity of a single latitude-longitude square near the ISFSI. The mean annual frequency was 0.5 tornadoes per year and the probability of a tornado striking a single point within that area was calculated to be 3.75×10^{-4}. The recurrence frequency was calculated to be once about every 2,700 years.

2.3.1.3.4 Hurricanes and Tropical Storms

Approximately one hurricane per year poses a threat to the area, and about one hurricane every 10 years produces a significant effect. Northeasters, or extratropical storms, also can influence the area in terms of flooding of low-lying land. The ISFSI is, however, on high ground. The detrimental effects of northeasters are considerably less than those postulated for hurricanes.

2.3.1.3.5 Precipitation Extremes

Table 2.3-1 lists some extremes of meteorological measurements for selected National Weather Service stations in the Calvert Cliffs region. At Patuxent River Naval Air Station, the 24-hour maximum was 5.88" in August, 1969.

2.3.1.3.6 Thunderstorms

Fifteen years of records at Patuxent River Naval Air Station showed 814 observations of thunderstorm activity with an average duration of about 1 hour 20 minutes. Baltimore averages 27.6 thunderstorms per year. Ronald Reagan Washington National Airport reports 29.8 thunderstorms per year. June and July are the months of greatest frequency of thunderstorms.

2.3.1.3.7 Snow and Freezing Precipitation

The Patuxent River Naval Air Station records for 1949 through 1964 list 910 hours of snow and 264 hours of frozen or freezing precipitation, other than snow, for a total of 1,175 hours (70,500 minutes) in 15 years. Interpolating for a 10 year span yields 47,000 minutes. The regional maximum monthly snowfall occurred in February, 1979, when Baltimore received 33.1" and Washington received 30.6". The maximum 24-hour snowfall at Patuxent River Naval Air Station of 11.7" occurred in February, 1979.

2.3.2 LOCAL METEOROLOGY

2.3.2.1 Data Sources

The local meteorology is based upon on-site data collected since 1967 and off-site data from Patuxent River Naval Air Station. Additional data was taken from Andrews Air Force Base and Ronald Reagan Washington National Airport near Washington, DC, and Baltimore-Washington International Airport.

2.3.2.2 Topography

Detailed topographic features in the region surrounding the ISFSI are shown on Figure 2.4-2. A discussion of topographical effects on diffusion estimates is described in Section 2.3.4.1.

2.3.3 ON-SITE METEOROLOGICAL MEASUREMENT PROGRAM

On-site meteorological measurements include wind direction and speed, temperature, and vertical temperature gradient. The accident analysis meteorological data base is for the period January 1, 1984 — December 31, 1986. Joint frequency tables of wind direction, wind speed, and atmospheric stability are shown in Table 6.1-1A of the ISFSI Updated Environmental Report.

The meteorological data for Calvert Cliffs presented in this section were collected from three meteorological towers: the inner south tower, discontinued in 1975; the microwave tower (ST), discontinued in 1993; and the primary tower placed in service in 1982 (Figure 2.3-1). The joint frequency tables digitized from strip charts by Dames & Moore (1969-1980) and by Envirodata (1981-1984) were used.

The relative position of instruments with respect to the power plant and the ISFSI is noted in Figure 2.3-1. Relative elevations of both surface levels and instrument levels are depicted in Figure 2.3-2. The available meteorological instrumentation for each tower is described in Table 2.3-2.

Regular operability, maintenance and calibration checks of the meteorological instrumentation are performed. As required by the CCNPP Technical Requirements Manual, a daily channel check and semiannual calibration of the meteorological monitoring instrumentation channels are performed. The calibration is performed according to a surveillance procedure.

Data loggers located in the building at the meteorological tower site sample each meteorological data channel every second and average the data every 15 minutes. The 15 minute averages are transmitted to the plant computer. For long-term storage, the data is archived by Plant Information Technology. Data averages are displayed locally at plant computer workstations located in the Control Room, Technical Support Center, Technical Support Center Annex, Simulator Control Room, and Emergency Operations Facility. Data from these workstations is used to assess the impact of routine effluents and accidental effluents. The hardware components and display locations are shown in Figure 2.3-3.

2.3.4 DIFFUSION ESTIMATES

2.3.4.1 Basis

The design 2-hour relative concentration (χ/Q) at the CAB for an accidental release at the ISFSI is 3.0×10^{-4} sec/m^3. Meteorological conditions resulting in a higher value will occur less than 5% of the time annually.

The 3.0×10^{-4} sec/m^3 design χ/Q is a conservative estimate which is based on a Pasquill turbulence class of G and a transport speed of 1.0 m/sec. Meteorological data used in this diffusion estimate was measured by the Baltimore Gas and Electric Company (BGE) on-site measurements program during the period from January 1, 1984 through December 31, 1986. Frequency distribution statistics of stability type and wind speed class were compiled by Pickard, Lowe and Garrick for BGE and were used for these calculations (Reference 2.7). The Pasquill G type stability class is selected based on Delta T measurements between the 10-60 m interval at the BGE primary meteorological tower, which is approximately 1000' north of the ISFSI. During the 3-year period of record, the frequency of occurrence of a Pasquill G stability was 6%. A transport speed of 1.0 m/sec was selected during a G stability class. These meteorological conditions occurred less than 5% of the time annually during the measurement period. The distance to the nearest CAB used for this calculation is 3900' (1189 m).

Local topographic influences are considered in evaluating the diffusion characteristics of the ISFSI location. The ISFSI is located on a northeast-southwest ridge which descends in Elevation 40' to 60' northwest and south of the ISFSI out to 1000'. Because of this local terrain feature, an accidental release, during a strong inversion with near calm winds, would result in the plume becoming decoupled from the atmosphere and drain, due to gravity, on either side of the ridge. As a result, it is very likely that the plume will be contained within the Calvert Cliffs CAB until adequate mixing is available for dispersion. The table below is a list of χ/Q values as a function of distance from the storage facility:

Distance (m)*	(χ/Q) sec/m^3
1000	3.5×10^{-4}
1189 (CAB)	3.0×10^{-4}
1500	2.4×10^{-4}
2000	1.8×10^{-4}
2500	1.6×10^{-4}
3000	1.3×10^{-4}

* Distance, in this case, is the centerline distance from the storage facility for a plume under a Pasquill G class stability.

2.3.4.2 Calculations

The calculations of a 2-hour χ/Q value to estimate radiological doses for a potential accidental release from the ISFSI is based on a Pasquill G type stability with a wind speed of 1.0 m/sec. The following equation for calculating χ/Q is referenced in Nuclear Regulatory Commission Regulatory Guide 1.145 (Reference 2.8), and considers plume "meander" during low wind speeds and stable atmospheric conditions:

$$\chi/Q = \left[\bar{\mu}\pi\Sigma_y\sigma_z\right]^{-1} = 3.0 \times 10^{-4} \text{ sec/m}^3$$

Where:

χ/Q = relative concentration, sec/m^3

$\bar{\mu}$ = average wind speed in m/sec

π = 3.14159

σ_y = the lateral plume spread in meters as a function of atmospheric stability and distance

σ_z = the vertical plume spread in meters as a function of atmospheric stability and distance

Σ_z = (M-1) σ_{y800m} + σ_y, where M is obtained from Figure 3 in Reference 2.8

The σ_y and σ_z values obtained for the χ/Q calculations are estimated from the figures in Reference 2.9.

TABLE 2.3-1
METEOROLOGICAL EXTREMES IN THE CALVERT CLIFFS REGION

	Baltimore-Washington International Airport	Ronald Reagan Washington National Airport	Patuxent River Naval Air Station
Maximum temperature (°F)	105 (8/83)	103 (7/80)	103 (7/80)
Minimum temperature (°F)	-7 (1/84)	-5 (1/82)	-3 (1/65,77)
Max monthly rainfall (in)	18.35 (8/55)	14.31 (8/55)	15.51 (7/45)
Max monthly snowfall (in)	33.1 (2/79)	30.6 (2/79)	32.3 (2/79)
Max 24-hour rainfall (in)	7.82 (8/55)	7.19 (6/72)	5.88 (8/69)
Max 24-hour snowfall (in)	22.8 (2/83)	18.7 (2/79)	11.7 (2/79)
Fastest mile wind (mph)	80 (3/52)	78 (10/54)	Not Available

TABLE 2.3-2
ON-SITE METEOROLOGICAL STATIONS AND INSTRUMENTATION

DESIGNATION	ELEVATION	PERIOD	INSTRUMENTATION
Inner South Tower	48' MSL +12' Mast	1/9/69 to 1/12/70 2/11/69 to 2/20/69	MRI Mechanical Weather Station Model 1071 MRI Vector Vane Sigma Meter, Model 1053
	48' MSL +12' & 49.5'	5/15/69 to 9/4/74	Temperature Gradient System, Packard Bell Corp. (Beckman-Whitley), Model 327 Aspirated Radiation Shields
	48' MSL +12' Mast	6/1/69 to 8/7/69	MRI Wind Diffusion System, Model 2040
	48' MSL +33'	8/7/69 to 5/15/71	MRI Wind Diffusion System, Model 2040
	48' MSL +33' & 97'	5/15/71 to 8/14/75 9/29/71 to 8/14/75 7/17/71 to 8/14/75	MRI Wind Diffusion System, Model 2040 Temperature Gradient System Weather Measure Corp. Aspirated Radiation Shields w/Rosemount Sensors Beckman-Whitley Model WS-101 Quick Vane Wind System
		3/13/72 to 1/11/74	Gill Anemometer Bivane
Secondary Tower (Microwave Tower)	75' MSL +40' & 125' & 200'	8/8/73 to Fall 1993 8/8/73 to Fall 1993	125' & 200' MRI Wind Diffusion System, Model 2040 40', 125' & 200' Weather Measure Corp. Aspirated Radiation Shields w/Rosemount Sensors. Temperature Gradient System
		8/23/73 to Fall 1993	125' Weather Measure Corp. Dewpoint System
Meteorological Towers			
a. Primary Tower	110' MSL +33' & 197'	4/82 to Present 4/82 to Present 4/82 to Fall 1995 4/82 to Present	33' & 197' Wind Sensors 33' & 197' Temperature Sensors 33' Dewpoint Sensor 0' Rain Gauge
b. Backup Tower	110' MSL +33'	12/05 to Present 12/05 to Present	33' Wind Sensor 33' Temperature Sensor

MRI Meteorological Research Institute

2.4 SURFACE HYDROLOGY

2.4.1 HYDROLOGIC DESCRIPTION

2.4.1.1 Site and Facilities

The ISFSI is located approximately 2300' southwest of the existing CCNPP at Elevation 114.0' above MSL. The ISFSI yard is approximately 226' by 666' and was created by excavation on the north end of the yard and by earth fill on the south end of the yard. The horizontal storage module (HSM) foundations are supported near yard grade at Elevation 111.0'. The Dry Shielded Canisters are supported approximately 5.5' above yard grade and the air intakes for cooling of the spent fuel is 2' above yard grade. The topography and local drainage pattern in the immediate area of the ISFSI is shown on Figure 2.4-1. The ISFSI yard is well drained and not susceptible to flooding.

2.4.1.2 Hydrosphere

Calvert County is bounded on the east by the Chesapeake Bay and on the west by the Patuxent River. The area is characterized by gently rolling terrain. A drainage divide extends longitudinally across the county. The county is well drained by a relatively large number of streams, although most are less than 7 miles long. Many streams have moderately steep valley walls, while others form estuaries to the Patuxent River. Swampy areas and tidal flats are common along the coastal areas.

The topography of the CCNPP property around the Calvert Cliffs ISFSI is gently rolling with steeper slopes along stream courses. Local relief ranges up to about 130'. The area is well drained by short intermittent streams. A drainage divide, which is generally parallel to the coastline, extends across the area as shown on Figure 2.4-2. The area to the east of the divide includes the plant area and drains into the Chesapeake Bay. The western area which includes the ISFSI is drained by tributaries of Johns Creek which flow into St. Leonard Creek and subsequently into the Patuxent River. Grading performed during construction of the CCNPP has not substantially altered the drainage system shown on Figure 2.4-2.

The ISFSI is just west of the drainage divide shown on Figure 2.4-2 and is not subject to flooding.

The surface waters adjacent to the Calvert Cliffs ISFSI are used for navigation, recreation, and commercial fishing. Almost all potable water used in Calvert County is from subsurface sources. The major use of water is for domestic and agricultural purposes. A discussion of groundwater users is provided in Section 2.5.1.

2.4.2 FLOODS

2.4.2.1 Flood History

Historical storms and tides in the Chesapeake Bay, as well as maximum water levels resulting from a probable maximum hurricane (PMH) in the entrance to the Bay were established in order to evaluate the maximum flood level for design at the CCNPP. The resulting maximum water level including wind and wave runup is 28' above MSL which is about 86' below the ISFSI yard grade.

2.4.2.2 Flood Design Considerations

As discussed in Section 2.4.2.1, the ISFSI yard grade is 86' above the elevation of the probable maximum water level for the CCNPP and is not susceptible to flood. Accordingly, the ISFSI is not designed for flood. The design flood level for the ISFSI was established based on a review of historical flood levels near the ISFSI location and evaluation of the PMH postulated at the entrance to the Chesapeake Bay. The PMH and the resulting wave runup resulted in the maximum water level at the ISFSI.

2.4.3 PROBABLE MAXIMUM FLOOD ON STREAMS AND RIVERS

See Sections 2.4.2.1 and 2.4.2.2.

2.4.4 POTENTIAL DAM FAILURES, SEISMICALLY INDUCED

There are no upstream dams or river structures which could affect water levels at the ISFSI.

2.4.5 PROBABLE MAXIMUM SURGE AND SEICHE FLOODING

The Calvert Cliffs ISFSI yard grade is above the maximum water level for the CCNPP site. The PMH results in the highest water level for the ISFSI. The following discussion summarizes the historical maximum storms and tides and the basis of the maximum water level as shown in Reference 2.1.

Historic Storms and Tides

Historic accounts of early hurricanes affecting the Chesapeake Bay area date back to the 17th Century. Early chronologies of tidal flooding record extreme events which occurred in August 1667, October 1749, September 1769, and July 1788. United States Weather Bureau records show at least 80 tropical hurricanes or their remnants affected the Bay area in the 75 year period between 1889 and 1964. One of the most destructive hurricanes in recent years to affect the Chesapeake Bay region was the hurricane of August 23, 1933. Other notable storms include hurricanes "Hazel" in October 1954, "Connie" and "Diane" in August 1955 (only 5 days apart), "Donna" in September 1960, "Agnes" in 1972, "David" in 1979, and "Gloria" in 1985. The "Great Atlantic Hurricane" of September 1944, which passed some 50 miles offshore of the Chesapeake Bay, was also a storm of major size and intensity. As noted above, the relative frequency of hurricane occurrence for this area is slightly more than one hurricane per year. In general, record hurricanes passing over or near Chesapeake Bay have had central pressures of 27.8 to 28.5" of mercury; peak wind speeds over the ocean approaching 100 mph, and maximum winds over the bay area of up to 75 mph. The forward speed of these storms has ranged from 10 to 36 knots. Northeast storms also affect the Chesapeake Bay area; however, because of the general orientation of the bay, the magnitude of tides reached in the bay, is not as great as those generated by record hurricanes. The northeast storm of March 6-8, 1962, resulted in 4.9' mean low water (MLW) tide in the lower Potomac River.

Tides and Storm Surges

Normal tides in the bay area have two highs and lows roughly every 23.5 hours, with a higher high and lower low as a daily occurrence. The mean and spring tide ranges to be expected at the Calvert Cliffs shore are 1.2 and 1.4', respectively. The travel time of high and low tide occurrence from the Bay entrance to the local area is approximately 5 hours.

Storm surges and extreme high tides have been recorded at numerous locations in the Chesapeake Bay and in the various rivers entering the Bay. Peak tide elevations above MLW range from 8.2' at Solomons Island near the mouth of Patuxent River to 7.4' at Point Lookout at the mouth of the Potomac River. Tide levels of 4.1' and 5.1', respectively, were recorded at these locations in the October 1954 hurricane. In the August 23, 1933 hurricane, a peak tide of about 8.5' MLW occurred at Norfolk, VA. A generalized tide-frequency curve was developed for the Chesapeake Bay area utilizing observed hurricane surge elevations. A reproduction of that relation is shown on Figure 2.4-3. The tide elevations noted above include the cumulative effects of tidal surge, pressure effect, local wind effect, wave effect (in open bay areas) and the astronomical tidal component. The contribution of the latter can add as much as 3' to the total recorded hurricane tide height if the peak ocean surge entering the Bay coincides with a peak spring tide condition. The time of translation of tides up the Bay, noted above, is also an important consideration in determining the coincidence of both ocean and bay peak surge heights with normal and spring high and low tides.

Probable Maximum Hurricane

A comprehensive investigation of hurricane surge problems for the Chesapeake Bay area was made using parameters from Memorandum HUR 7-97, "Interim Report — Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coasts of the United States," Reference 2.10. Parameters describing the maximum probable hurricane were selected from Reference 2.10 at the approximate latitude of the Chesapeake Bay entrance (36.1°). Definition of each of those parameters for a maximum probable hurricane is given below.

Central Pressure (P_o) — Minimum central pressures in hurricanes passing over or near the Chesapeake Bay area have been as low as the 27.88" of mercury for the September 1944 hurricane. Except for hurricanes occurring within the last several decades, sufficient information on central pressures to establish a reliable pressure-frequency relationship for the area is not available. A central pressure of 26.94" was selected.

Asymptotic Pressure (P_n) — A value of 30.92" was selected to represent the peripherical pressure of the PMH.

Radius of Maximum Winds (R) — A value of 30 statute miles was used for this parameter, being considered representative of severe storm occurrences in the general area. Its use results in a storm of reasonable size for transposition purposes.

Forward Speed (T) — The value selected for forward speed is 23 mph, a moderate speed of translation. The forward speed of the storm affects not only the peak 30' overwater wind speed, but also the height of peak ocean tide at or near shore and the shape of the resulting hydrograph. In the case of the Chesapeake Bay, the forward speed is especially important in that it is related to the development of surge elevation within the Bay, the speed of the free wave up the Bay, and the resulting surge height at the ISFSI shore. A very slow moving storm would permit the Bay surge to crest at Calvert Cliffs before the maximum effect of crosswinds could reinforce and increase that height. A fast moving storm would result in the converse.

Maximum winds at radius R would be 124.7 mph; adding half the forward speed results in a peak isovel wind speed of 136.2 mph.

Path — The path selected for the PMH is shown in Figure 2.4-4. It would approach the coast from the east, curving northward on passing inland west of Chesapeake Bay.

Parametric relationships describing the wind speed profile, pressure profile, pressure effect profile and basic wind data used in constructing the isovel pattern for the PMH were derived using a computer program developed and employed by personnel of the Jacksonville District Corps of Engineers. Graphical representation of the overwater wind profile, the pressure and pressure effect profiles can be seen on Figure 2.4-5.

Tidal Surge Computations

Procedures used in the tidal surge analysis for the open ocean across the Continental Shelf are those described in the U.S. Army Coastal Engineering Research Center publication "Shore Protection — Planning and Design," Technical Report No. 4, Reference 2.11. Formula (1-65) shown on page 140 of that report was used for the computations. Peak tide at the Chesapeake Bay entrance will occur at time T_o. Peak winds in the zone of maximum winds were oriented over the shallow bay entrance channel area to obtain the maximum surge height. Based on the selected forward speed, the surge hydrograph at the coast would have about a 12 to 14 hour rise from slightly above normal tides to the peak surge at T_o. This is based on a comparison of storm features of the PMH with that of the August 1933 hurricane which affected the area.

Results — The peak tidal surge elevation that would occur at the Bay entrance was computed to be 18.67' MLW (17.32' MSL). It should be noted that wave effect was not considered to be applicable. Water depths in the channel entrance to the Bay would be on the order of 40' (22' depth + 18' surge). That depth would sustain a 30-32' wave which would move into the bay area to break farther inland. A reduction in ocean surge occurs as it passes into and up the Bay due to the comparative dimensions and hydraulic characteristics of the entrance channel and the various sections of the Bay between Hampton Roads and Baltimore. Movement of the surge up the Bay to the Calvert Cliffs area will occur at approximately the speed of the free wave in the Bay (about 24 to 27 mph depending on depth changes) and at a speed coincident with the speed of the hurricane. The presence of large rivers with added storage volume will result in a further minor reduction in surge height in its passage up the Bay. A factor of 0.96 x the surge elevation in the lower Bay will give the value of the surge elevation to be expected in the vicinity of the installation. Using that factor gives a surge elevation of 13.44' MLW (14.0x0.96). That elevation represents the height of the surge in the Bay as it moves northward past the Calvert Cliffs site. To that value must be added the additional effect of hurricane winds blowing from east to west across the Bay and the effect of coincident occurrence of normal high tide, plus any wave effect.

Surge Elevation at Plant Site — Movement of the PMH inland and overland will result in a reduction in intensity and wind speed. Wind directions slightly ahead of the zone of maximum winds will be oriented generally east to west over the Bay in the vicinity of the ISFSI at the time the peak surge reaches that area. An evaluation of wind speed and direction was made for that condition. Wind speeds of from 115 to 120 mph (117 mph average) were found to be applicable for the wind direction and fetch conditions shown on Figure 2.4-6. An effective crosswind of 94 mph was used to compute the additional height of Bay setup. The total fetch length is approximately 10 Statute miles. The computed setup elevation in the vicinity of the ISFSI was determined to be 15.21' MLW. A value of 1' was added to that elevation for estimated wave effect, giving a total peak surge elevation at the Calvert Cliffs shore of 16.21' MLW (15.6' MSL).

Wave Analysis — The significant wave height that can be expected to occur in the vicinity of the ISFSI during the PMH peak surge will be a function of wind speed, water depth, and length of available fetch. Evaluation of average water depth with fetch length in the Bay offshore indicated a 50' depth for about 7 miles; a 40' depth for about 9 miles. Using a wind speed of 94 mph results in a significant wave height of 11.4'; with a corresponding wave period of 9 sec. The wave will break in approximately 14.5' of water. The height of the wave above still water level would be 6.8' (11.4x0.6). Added to the peak surge elevation of 16.2', the elevation of the top of that wave, unbroken, would be 23.0' MLW.

The resulting maximum runup elevation of 27.5' MSL is well below the ISFSI yard grade elevation of 114.0'.

2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

The occurrence of tsunamis is infrequent in the Atlantic Ocean. Other than the tidal fluctuation recorded on the New Jersey shore during the Grand Banks earthquake of 1929, there was no record of tsunamis on the northeastern United States coast prior to about 1970. The earthquake of November 18, 1929, on the Grand Banks about 170 miles south of Newfoundland, resulted in a tsunami which struck the south end of Newfoundland, about 750 miles northeast of the Massachusetts Coast. This tsunami occurred at a time of abnormally high tide and resulted in some loss of life and destruction of property. The effect of this tsunami was recorded on tide gauges along the east coast of the United States as far south as Charleston, South Carolina. A tidal fluctuation of approximately nine-tenths of one foot was noted at Atlantic City, NJ and Ocean City, MD (Reference 2.12).

The Lisbon earthquake of November 1, 1755, produced great waves which contributed heavily to destruction on the coast of Portugal. These waves were noticeable in the West Indies. It has been reported that the Cape Ann, MA, earthquake of November 18, 1755, caused a tsunami in Saint Martin's Harbor in the West Indies. However, there is no record of tsunami occurrence along the east coast of the United States at this time and it appears that the Saint Martin's Harbor report actually refers to the tsunami caused by the Lisbon earthquake, which occurred less than 3-weeks before the Cape Ann shock. Some tsunami activity has occasionally followed earthquakes in the Caribbean, but none of these was reported in the United States.

The ISFSI is not susceptible to a tsunami effect. The maximum expected tsunami would result in only minor wave action, and the maximum expected storm wave effect, discussed in Section 2.4.5, is the basis of the ISFSI flood design.

2.4.7 ICE FLOODING

The ISFSI is not subject to floods resulting from ice on adjacent streams or the Chesapeake Bay. As discussed in Sections 2.4.1 and 2.4.2 the ISFSI is near the drainage divide between the Patuxent River and the Chesapeake Bay at Elevation 114.0' above MSL.

2.4.8 FLOODING PROTECTION REQUIREMENTS

The ISFSI is in a flood-free zone as described in Sections 2.4.1 and 2.4.2.

2.4.9 ENVIRONMENTAL ACCEPTANCE OF EFFLUENTS

There are no effluent releases from the ISFSI during normal operation as described in Section 2.4.1.2.

2.5 SUBSURFACE HYDROLOGY

A detailed discussion of the groundwater aquifers, flow directions, recharge points, and users is provided in Reference 2.1 and is summarized in the following sections.

2.5.1 REGIONAL CHARACTERISTICS

Ground water occurs in the surficial soils and is tapped by many shallow dug and driven wells. Ground water in deeper aquifers occurs under artesian conditions. These aquifers, the Piney Point, Nanjemoy, and Aquia formations, are separated from the surficial deposits by an aquiclude averaging about 270' in thickness. Recharge to these aquifers occurs in their outcrop areas about 15 to 30 miles west of the ISFSI. The geologic position of these aquifers relative to other formations in Calvert County is presented in Table 2.5-1 and Figure 2.6-4.

The hydrologic characteristics of the Piney Point, Nanjemoy, and Aquia formations are discussed in greater detail in the following subsections.

2.5.1.1 Piney Point Formation

The Piney Point formation consists of glauconitic sand interspersed with shell beds and a little clay. Well cuttings and particle-size analyses indicate that the aquifer is composed mainly of medium to fine sand. The formation occurs as a wedge-shaped geologic unit and is known only in southern Maryland. It is about 30' thick in the vicinity of the ISFSI and increases in thickness to the southeast.

The Piney Point formation is widely utilized as a source of ground water in Calvert County and adjoining St. Mary's County. Several hundred domestic wells in these two counties tap the Piney Point and the underlying Nanjemoy formations. These two aquifers are hydrologically connected. The yields of domestic wells generally range from about 3 to 20 gpm. The specific capacities of 25 selected wells in these formations range from 0.1 to 3.3 and average 1.2 gpm/ft of drawdown.

2.5.1.2 Nanjemoy Formation

The lower part of the Nanjemoy formation consists of an impermeable red clay known as the Marlboro Clay. The remainder of the formation consists chiefly of greensand, but contains some clayey greensand. A limited number of particle-size analyses indicate that the sand is predominantly medium to fine grained.

The Nanjemoy formation is an important aquifer in Calvert County where it is tapped by several hundred wells. Most wells are completed in the permeable water bearing sands occurring in the uppermost 80' of the formation and yield less than 10 gpm. The specific capacities of 11 wells in Calvert County tapping this aquifer range from 0.2 to 2.4 and average 0.8 gpm/ft of drawdown. On the basis of water level recovery measurements made during a pumping test, a coefficient of transmissibility of approximately 2,000 GPD/ft has been computed. The field coefficient of permeability was 66 GPD/ft^2. The results of a similar test in Prince George's County indicate coefficients of transmissibility ranging from 260 to 840 GPD/ft.

2.5.1.3 Aquia Formation

The Aquia formation is characterized by an abundance of glauconitic sand with some quartz sand and clay. The thickness of permeable sandy beds in the formation ranges up to slightly more than 40' in parts of Calvert County. Particle-size analyses of samples from the Aquia formation show that the sand is medium to fine-grained.

Yields of individual wells tapping this formation range from 125 to 350 gpm. The specific capacities of eight of these wells range from 0.8 to 4.2 and average 2.5 gpm/ft of drawdown. The results of six pumping tests indicate field coefficients of permeability ranging from 130 to 1,340 GPD/ft^2. Coefficients of transmissibility determined from the tests range from 5,500 to 33,000 GPD/ft.

Calvert Cliffs Nuclear Power Plant wells tap this formation.

2.5.1.4 Water Levels

The artesian head of the three principal aquifers in Calvert County is generally above sea level. The effect of tidal fluctuations on water levels was noticeable in observation wells completed in the Nanjemoy and Piney Point formations at Solomons Island about 7 miles south of the ISFSI. Recorder charts from these wells, which are 248 and 493' deep, respectively, show fluctuations of about 1/2'.

The approximate configurations of the piezometric surfaces of the Aquia and Nanjemoy formations are shown on Figure 2.5-1. The regional hydraulic gradient in the vicinity of the Calvert Cliffs ISFSI is to the southeast. However, the regional hydraulic gradient in the vicinity of the Calvert Cliffs ISFSI is to the southeast, local minor variations occur.

2.5.1.5 Public Water Supplies

Nearly all potable water used in Calvert County is from subsurface sources. The major use of water is for domestic and agricultural purposes. There are 15 towns and 8 private communities in Calvert County with public water supplies. The output from these systems is relatively small, but may increase substantially in the summer to accommodate the seasonal population increase. Data concerning the public water supplies is presented in Table 2.5-2. The locations of these supplies are shown on Figure 2.5-2.

2.5.1.6 Private Wells

Field surveys have determined that most domestic water supplies in Calvert County are obtained from private wells greater than 300' in depth. In some instances, other wells were less than 50' deep and of limited capacity. The yield of most private wells is less than 60 gpm. The locations of the deep wells, in the vicinity of the ISFSI, are shown on Figure 2.5-3.

There is one well between the ISFSI and Johns Creek. This well and all property between the ISFSI and Johns Creek is owned and controlled by CCNPP.

2.5.2 SITE CHARACTERISTICS

The depth of ground water at the Calvert Cliffs site was measured using piezometers installed in seven Dames and Moore exploratory borings in 1967. The piezometers consisted of small-diameter steel pipe equipped with a well point, or perforated polyvinyl chloride pipe. They were installed in borings DM-1, DM-2, DM-3, DM-5, DM-7, DM-8, and DM-9 immediately after completion of the drilling operations. Water levels ranged from 15' below the ground surface at DM-7 to 82' at DM-5. The locations of the borings are shown on Figure 2.5-4.

An in-situ soil percolation test was performed at the site in Miocene soils typical of those underlying the site. Results of this test indicate a permeability of less than 1 GPD/ft^2. The location of the test is shown on Figure 2.5-4.

Soil samples from borings at the ISFSI were tested in order to measure their grain-size distribution. The results of these analyses were used for soil classification purposes. The results are presented on Figures 2.5-5A through 2.5-5I. The location of borings and the subsurface profile are shown on Figures 2.6-7 and 2.6-7A.

The clay mineral content and the total cation exchange capacity of seven selected soil samples was analyzed by Dames and Moore in the late 1960's. The results of these tests are presented in Table 2.5-3.

Data obtained from the geologic exploratory borings indicate that a large portion of the ISFSI area is mantled by relatively permeable Pleistocene soils. These soils have been eroded from a portion of the area exposing the Chesapeake Group which includes the St. Mary's, Choptank, and Calvert formations. The Chesapeake Group consists of about 270' of impervious sandy and clayey silts of Miocene age. Underlying this material are the Piney Point, Nanjemoy, and Aquia formations of Eocene age.

The elevation of the phreatic surface changes with the surface topography and can be expected to fluctuate slightly as a result of climatic changes. The water table occurs generally within 30' of the ground surface. East of the topographic divide the direction of ground water movement is toward the Chesapeake Bay. The direction of ground water flow west of the divide is toward the existing stream valleys.

The underlying impervious sandy and clayey silts of the Chesapeake Group extend to about 200' below MSL. A percolation test conducted near the Calvert Cliffs ISFSI indicates a permeability of less than 1 GPD/ft^2. Particle-size analyses indicate that the permeability of the Chesapeake Group averages about 3 GPD/ft^2. The rate of ground water movement is extremely low (much less than 1" per day). The formation is an aquiclude which effectively confines the underlying artesian aquifers. Regional studies by the U.S. Geological Survey have shown that the head in the artesian aquifers is above sea level. The result is vertical upward leakage through the Chesapeake Group. The rate of leakage is extremely low because of the low permeability of the Miocene sediments.

At the ISFSI, the combined thickness of the aquifers within the Piney Point and Nanjemoy formations is about 80'. They occur at elevations ranging between 200 and 300' below MSL and are separated from the deeper Aquia formation by a layer of clay (Lower Nanjemoy) about 150' thick. The general direction of ground water movement in the Aquia formation is toward the southeast with a piezometric gradient of about 2' per mile.

Grain-size analyses of samples of the Piney Point formation collected at the site indicate a permeability of about 150 GPD/ft^2. This value is probably typical of both the Piney Point and Nanjemoy aquifers. It is estimated that the permeability coefficient of the Aquia formation may be on the order of 1,000 GPD/ft^2. The computed rate of flow of ground water through these aquifers ranges from about .07 to .004' per day. The possibility of accidental contamination of the Eocene aquifers beneath the ISFSI is remote because, (a) there are no effluent releases from the ISFSI, (b) the aquifers are covered by over 200' of relatively impervious soils, and (c) the vertical component of ground water movement is upward.

2.5.3 CONTAMINANT TRANSPORT ANALYSIS

As discussed in Section 2.5, there are no effluent releases from the ISFSI.

TABLE 2.5-1
GEOLOGIC UNITS IN CALVERT COUNTY

GEOLOGIC UNIT	APPROXIMATE RANGE IN THICKNESS (feet)	PHYSICAL CHARACTERISTICS	WATER BEARING PROPERTIES
Pleistocene surficial deposits	0-150	Silt and sand with some clay and gravel	Yields small quantities of water to relatively shallow dug or driven wells.
Chesapeake Group St. Mary's Fm. Choptank Fm. Calvert Fm.	30-325	Sandy and clayey silt with interbedded sand and fossiliferous layers	An aquiclude. Yields small supplies of water to a few dug wells.
Piney Point Fm.	0-60	Glauconitic sand	Yields up to 200 gpm are reported from drilled wells. An important aquifer in Calvert County.
Nanjemoy Fm.	40-240	Glauconitic sand with clayey layers. Basal part is red or gray clay	Yields of individual wells reported up to 60 gpm. An important aquifer in Calvert County.
Aquia Fm.	30-200	Green to brown glauconitic sand	Yields up to 300 gpm reported from wells. An important aquifer in Southern Maryland.
Brightseat Fm.	0-40	Gray to dark gray micaceous silty and sandy clay	Not known to be an aquifer in Southern Maryland.
Monmouth and Matawan Fm.	20-135	Sandy clay and sand, dark gray to black, with some glaucomite	Not a major aquifer in Southern Maryland, but yields up to 50 gpm have been reported.
Magothy Fm.	0-40	Light-gray to white sand and gravel with interbedded clay layers	A few wells reportedly yield up to 1,000 gpm but average yields considerably less. This aquifer is not used in Calvert County because of its depth.
Raritan Fm.	100	Interbedded sand and clay with ironstone modules	Yields up to a few hundred gpm reported. Not utilized in Calvert County due to depth.

TABLE 2.5-1
GEOLOGIC UNITS IN CALVERT COUNTY

GEOLOGIC UNIT	APPROXIMATE RANGE IN THICKNESS (feet)	PHYSICAL CHARACTERISTICS	WATER BEARING PROPERTIES
Patapsco Fm.	100-650	Interbedded sand, clay, and sandy clay	Large-diameter wells yield up to 1,000 gpm. Not used in Calvert County because of depth.
Arundel Clay Fm.	25-200	Red, brown, and gray clay	Not generally a water bearing formation.
Patuxent Fm.	100-450±	Chiefly gray and yellow sand with interbedded clay	Yields of several hundred gpm reported. Not utilized in Calvert County due to great depth.
Precambrian	Unknown	Gneiss, granite, gabbro, metagabbro, quartz diorite, and granitized schist	Yields moderate supplies of ground water, generally not more than 50 gpm. Not used in Calvert County because of its great depth.

TABLE 2.5-2
PUBLIC SUPPLY WELLS IN CALVERT COUNTY

TOWN	POPULATION SERVED	NUMBER OF CONNECTIONS	AVERAGE OUTPUT (mgd)	WELL	TOTAL DEPTH (ft)	DIA. (in)
Calvert Beach	222	74	0.017	1	475	5
Cavalier County	402	134	0.045	--	--	--
Chesapeake Beach	460	220	0.036	1[a]	400	8
				2	400	8
				3	400	6
				4[a]	400	2
				5	400	2
Chesapeake Heights	612	175	0.043	--	--	--
Chesapeake Ranch Estates	1995	--	0.132	1	400	4
				2	750	4
				3	750	4
Dares Beach	501	--	0.026	1	--	4
Hunting Hills	114	--	0.005	1	--	4
Kenwood Beach	255	--	0.021	1	--	4
				2	--	4
Lakewood	120	--	0.010	--	--	--
Long Beach	1275	--	0.080	1	525	3
				2	500	4
				3	475	4
				4	500	4
				5	357	4
Prince Frederick	372	269	0.091	1	--	6
Randle Cliffs	18	--	0.001	2	--	8
St. Leonard	75	--	0.012	1	550	5
Scientists Cliffs	500	230	0.045	1	240	6

TABLE 2.5-2
PUBLIC SUPPLY WELLS IN CALVERT COUNTY

TOWN	POPULATION SERVED	NUMBER OF CONNECTIONS	AVERAGE OUTPUT (mgd)	WELL	TOTAL DEPTH (ft)	DIA. (in)
Shores of Calvert	414	138	0.030	--	--	--
Solomons Island	400	--	0.020	1	--	--
Summit	80	9	0.002	--	--	--
Wallville Acres	21	7	0.002	--	--	--
Western Shores	175	49	0.022	1	325	3
White Sands	120	34	0.006	1	402	4
				2	315	2.5

NOTE: Information based on 1988 biennial update of the Calvert County Comprehensive Water and Sewerage Plant prepared by the Calvert County Department of Planning and Zoning.

[a] Wells not in use.

TABLE 2.5-3
CATION EXCHANGE AND X-RAY DIFFRACTION ANALYSES

BORING	DEPTH (feet)	SOIL TYPE	GRADATION IN % FINER[a] (in millimeters) .074	.048	.005	.002	% OF TOTAL CLAY MINERALS[b] Illite	Montmorillonite and Mixed Clay Minerals	Chlorite	TOTAL CATION EXCHANGE CAPACITY[c] Test 1	Test 2
DM-1	45	Gray Silty Clay	98	44	23	17	30	50	20	23.8	24.5
DM-6	43.5	Green Silty Sand	28	25	21	15	40	60	--	10.0	11.0
DM-6	115	Green Clayey Silt	98	91	44	31	30	50	20	24.3	27.3
DM-8	30	Green Silty Sand	28	25	22	15	35	65	--	13.8	12.8
DM-8	45	Green Clayey Silt	90	80	45	35	20	60	20	25.0	30.6
DM-9	5	Reddish-Brown Sandy Clay	80	65	38	34	--	--	100	10.0	10.3
DM-10	47	Gray Sandy Silt	59	22	16	11	20	80	--	14.4	10.0

[a] Soil samples soaked for 24 hours in 0.4% sodium hexametaphosphate before hydrometer analysis.
[b] X-ray diffraction analyses of minus 2 micron material.
[c] Because of the CaCO3 in some of the samples, the ammonium acetate method was used. Total Cation Exchange was determined on the minus 40 micron material.

2.6 GEOLOGY AND SEISMOLOGY

Specific soil testing has been performed at the designated location for the ISFSI. The data obtained from this testing is utilized in the foundation design of the ISFSI. It should be noted that foundation conditions at the ISFSI are generally typical of those encountered in the general area. The following sections discuss the Calvert Cliffs area geology and seismology. The initial geologic field work at the Calvert Cliffs ISFSI was performed prior to the licensing of the CCNPP and is summarized in the following sections. Additional evaluations were performed to confirm that the in-situ soil conditions are adequate to support the revised foundation loading due to a change in storage modules for Phases IV and V [use high burnup horizontal storage module (HSM-HB)]. A comparison of the in-situ soil conditions encountered during the subsurface investigation for the original ISFSI to the results of the recent subsurface investigation (2007) was made. Records of earthquake history in the eastern region since construction of Calvert Cliffs have been reviewed and indicate no recent seismic activity that would impact the design bases for the facility.

2.6.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

The Calvert Cliffs site is underlain by approximately 2,500' of southeasterly dipping sedimentary strata of Cretaceous and Tertiary age. Underlying these sediments are crystalline and metamorphic rocks of Precambrian and Early Paleozoic age.

Sediments of the Chesapeake Group of Pleistocene age underlie the ISFSI area to a depth of about 200'. The material in this group consists of essentially horizontally stratified sandy and clayey silt with occasional interbeds of sand and shells. It is relatively impervious and dense and provides adequate foundation support for the ISFSI. The Pleistocene sediments are underlain by dense, relatively pervious, glauconitic sand and silt of Eocene age.

No known or suspected faults are present in the sedimentary strata underlying the site. The closest known faults are located in the Piedmont Province in western Maryland, approximately 50 miles from the site.

The site is considered satisfactory, from a geologic standpoint, for construction and operation of the ISFSI.

2.6.1.1 Regional Geology

The site lies within the Coastal Plain Physiographic Province about 50 miles east of the Fall Zone. The Fall Zone separates the low-lying gently rolling terrain of the Coastal Plain from the higher relief of the Piedmont Physiographic Province. The provinces are shown on Figure 2.6-1.

The Coastal Plain in Maryland is a low plain rising from sea level to about Elevation +250' at the Fall Zone. Relief in the region ranges generally from about 20 to 100'. The regional slope of the Coastal Plain is to the east at approximately 1.5' per mile. The topography of the region is characterized by a series of broad, step-like terraces. The terraces are successively less dissected by stream erosion from west to east. The region is well drained by a large number of small streams.

The general geologic characteristics of the region are shown on Figure 2.6-2. The Piedmont Province consists of a complex of igneous and metamorphic rocks of Precambrian and Early Paleozoic age with areas of sedimentary and igneous rocks of Triassic age. Beneath the Coastal Plain Province these rocks are concealed by younger strata of Cretaceous and Tertiary age. The buried surface of the basement igneous and metamorphic rock slopes to the southeast at about 50' per mile. In the vicinity of the site, the surface of the basement complex is located approximately 2,500' below sea level. The Cretaceous and Tertiary strata consist of sedimentary deposits of silt, clay, sand, and gravel which exhibit considerable lateral and vertical variations in lithology and texture. The strata form a wedge shaped mass which thickens to the southeast and pinches out to the northwest toward the Fall Zone.

A generalized geologic cross-section of the Coastal Plain is presented on Figure 2.6-3. A detailed description of the Stratigraphy at the site is presented on Figure 2.6-4.

The thick sedimentary strata of the Coastal Plain in the vicinity of the site have remained essentially undeformed since they were deposited up to 135 million years ago. They are believed to have been affected only by slow regional crustal downwarping during their deposition. No known faults have been identified within the Cretaceous and Tertiary sedimentary deposits in the site area. Some local, very shallow folds have been recognized in the Coastal Plain sediments about 40 miles south of the site. These structures are possibly related to depositional conditions rather than to post-depositional tectonic activity. The strata exposed for many miles along the Chesapeake Bay shoreline show no visible signs of faulting or deformation.

There is no known fault or geologic evidence of faulting in the deep crystalline rocks in the area. The absence of deformation in the overlying sediments indicates that no major faults are present in the area. Significant tectonic features of the region are shown on Figure 2.6-5.

The closest known faults to the site are more than 50 miles to the west in the Precambrian and Early Paleozoic rocks of the Piedmont Physiographic Province. The rocks in the Piedmont are highly folded, and many zones of major faulting have been identified. Most earthquake activity in the region can be related to them. Some of these faults, the closest of which are located about 60 miles southwest of the site, theoretically could be projected beneath the Coastal Plain strata toward the general location of the site. However, such faults are local rather than regionally continuous and appear to be associated with individual fault troughs containing Triassic sediments.

The recognizable geologic history of the region begins with the deposition of Paleozoic sediments on a Precambrian granitic and metamorphic basement complex. Thick sequences of sedimentary rocks, which accumulated during the Cambrian and Ordovician Periods of geosynclinal deposition were subsequently uplifted, folded, faulted, and metamorphosed during the late Paleozoic Period of mountain building. This activity was followed by another period of uplift along the axis of the Appalachian Mountain chain at the end of the Triassic Period.

Slow regional downwarping of the Coastal Plain started during Early Cretaceous time and continued intermittently through Tertiary time. South and east of the Fall Zone the Piedmont was depressed below sea level providing a base on which the sediments were deposited. Several periods of submergence and emergence resulted in alternate deposition and erosion of continental and marine deposits throughout Cretaceous and Tertiary times.

Near the end of the Tertiary Period (Pliocene time) the area is believed to have been above sea level. This resulted in erosion of the sediments deposited previously during Early Pliocene and Late Miocene time, so that Miocene sediments are exposed in the area.

During Early Pleistocene time, the ocean advanced westward to the Fall Zone, completely covering the Coastal Plain. Fluctuating sea levels, occurring during Pleistocene time, resulted in alternating periods of erosion and deposition along what are now the major terraces and scarps of the region. A veneer of Pleistocene soils covers most of the Coastal Plain. At present, the land is again being submerged by a very slow rise of the sea level.

2.6.1.2 Site Geology

The site is located on the west shore of Chesapeake Bay in an area characterized by densely wooded, low, flat to gently rolling terrain of low to moderate relief. Ground surface elevations at the site range from sea level to about +130', with an average elevation of approximately +100'. Nearly vertical cliffs, over 100' high in places, are located along the shore of the Chesapeake Bay. The ISFSI is located in an area approximately 2800' from the Chesapeake Bay where the preexisting ground elevation is about 100-125'. The final yard grade elevation is about 114.0'.

Areas in the vicinity of the site above about Elevation +70' are underlain by sediments of Pleistocene age. These sediments consist primarily of silt and sand, and as encountered at the boring locations, range up to about 50' in thickness. The portions of the site below about Elevation +70', which includes the Calvert Cliffs plant area, are underlain by relatively impervious sediments of the Chesapeake Group of Miocene age. The contact between the Pleistocene and Miocene sediments is relatively even and slopes very gently toward the southeast. The surficial geology of the site is shown on Figure 2.6-6.

The details of the subsurface geology were originally investigated primarily by means of twelve exploratory borings (Reference 2.19) at the locations shown on Figure 2.6-7.

The initial borings near the HSMs ranged in depth from about 50' to 100' and were drilled with truck-mounted rotary drilling equipment. Data were obtained from the initial borings through continuous observation of drill cuttings and examination of undisturbed samples and bag samples collected by Law Engineering Company engineers.

Soil samples were obtained at intervals in each boring ranging between 2.5 and 5', utilizing a standard split-spoon sampler. The number of hammer blows required to drive the sampler a distance of 1' into undisturbed material is recorded in the column entitled "N" on each boring log.

All initial samples were examined and logged in the field and then shipped to Law Engineering's Baltimore office for further examination and appropriate laboratory testing. Detailed descriptions of the materials encountered in the borings are shown on Figures 2.6-7B through 2.6-7M. The depth of ground water after completion of drilling operations and the date on which the borings were completed are also presented on the logs.

The subsurface soils may be separated into four basic strata, as described in subsequent paragraphs. An estimated subsurface profile along the north-south axis of the site is shown on Figure 2.6-7A. Lines designating the interfaces between strata should be considered approximate and the actual transition may be gradual. Beneath a thin veneer of topsoil the borings encountered:

> Stratum I — consists of very loose or soft to very stiff brown clayey sands, silty sands, sandy clays, and sandy silts of the Pleistocene formation. These soils were encountered at each of the ten boring locations as the uppermost stratum. This stratum extended to depths of 12' and had an average thickness of about 8'. Standard penetration resistance values averaged about 10 bpf.
>
> Stratum II — comprised of Pleistocene deposits of firm to dense brown and tan silty fine to medium sand and fine to medium sand. Stratum II was encountered at each boring location beneath the Stratum I soils and appeared to have a generally uniform thickness of about 25'. The lower portion of this stratum often contained coarse sands. Standard penetration test resistances averaged about 23 bpf.
>
> Stratum III — consists of interbedded light gray to tan, brown, and dark gray sandy and silty clays, and clayey and silty sands of the Pleistocene formation. This stratum was encountered beneath the soils of Stratum II at each test location. Where penetrated this stratum had a thickness of about 20'. Standard penetration resistances varied widely in this stratum with values averaging about 7 bpf.
>
> Stratum IV — consists primarily of loose to very dense dark gray and green clayey and silty fine sands of the Miocene formation. These soils were encountered below the Stratum III soils in the deeper borings. Standard penetration test resistances for this stratum averaged about 13 bpf above a depth of 90' (Elevation +25' MSL) and increased to an average of 50 bpf below 90'. The substantial increase in blow count values generally coincided with an increase in shell fragments encountered below a depth of about 90'.

A laboratory testing program was initiated upon completion of the field testing. All testing was conducted in accordance with American Society for Testing and Materials recognized standards and currently accepted procedures.

Laboratory testing of representative soil samples was conducted to determine the soils physical characteristics and engineering properties for use in foundation design. Testing was conducted under static, dynamic, and remolded (fill) conditions in an attempt to simulate or model field conditions during the construction process and design life of the structures.

The testing program included: moisture contents, grain-size, Atterberg limits, and specific gravity tests to determine the soils gradation and plasticity characteristics; consolidated drained triaxial tests to determine the shear strength parameters of undisturbed near surface soils for bearing capacity analyses and slope stability evaluations; direct shear tests on remolded samples of proposed fill soils also for bearing capacity and slope stability evaluation; one-dimensional consolidation testing on undisturbed and remolded soils samples to determine the settlement characteristics of virgin fine-grained soils and proposed compacted fill, respectively; and dynamic triaxial shear tests to determine the dynamic soil properties used in the evaluation of liquefaction potential of virgin foundation soils. In addition, void ratio, natural density, and modified Proctor density tests were also evaluated.

Stratum I — Index properties testing of the upper near surface soils and proposed fill revealed natural moisture content values between 12 to 16%, percent fines varying from 13 to 57% with an average value of about 30%, and plasticity indices ranging from non-plastic to 15. Direct shear tests were conducted on remolded samples of proposed fill soils compacted to 95% of the modified Proctor maximum dry density.

Stratum II — Testing of the soils of this stratum revealed generally non-plastic sands with less than 14% fines. Consolidated drained triaxial tests on undisturbed samples were obtained at depths of 11 and 15' below the surface.

Stratum III — Laboratory testing of **stratum three** was conducted on both fine grained clay soils and silty and clayey sands. Classification tests conducted on the clay soils revealed generally uniform natural moisture contents between 31 and 33%, plasticity testing disclosed liquid limits ranging from 34 to 48 and corresponding plasticity indices from 18 to 30. Consolidation testing was conducted on two undisturbed clay samples obtained from Boring DE-3 at depths of 46 and 50'. The test results indicate soil compression indices (C_c) of 0.30 and 0.34 and associated initial void ratios (e_o) of 0.94 and 0.84, respectively. Maximum preconsolidation pressures of 6.5 and 8.0 were also recorded, resulting in estimated overconsolidation ratios of 1.3 and 1.9, respectively. Classification testing conducted on the granular soils of Stratum III disclosed non-plastic to slightly plastic soils with natural moisture contents between 14 and 30%. The samples contained between 16 and 49% fines (silt and clay fractions).

A subsurface investigation was performed by MACTEC (Reference 2.18) in 2007 to provide additional subsurface information for the expansion of the ISFSI facility. This included drilling four borings to a depth of 100 feet each along the west side and outside of the perimeter fence of the ISFSI facility. A seismic crosshole survey to obtain shear wave velocities for the materials beneath the site was also performed in one of the borings.

The generalized subsurface profile, as depicted by the four borings drilled by MACTEC, can be summarized as the upper 30 feet (layer one) consisting of silty sand and clayey sand with silt or clay having Standard Penetration Test (SPT) N-values between 6 and 20. Layer two is approximately 12 feet in thickness and consists primarily of a mixture of clay, silt, sandy silt, clayey sand, and silty sand with SPT N-values ranging from 2 to 8. Layer 3 is approximately 48 feet in thickness and consists of low plasticity clay, high plasticity clay and silt with sand seams having SPT N-values between 3 and 15. The lowest layer was encountered at a depth of 90 feet and consists of medium dense to very dense silty sand with shell fragments. All of the borings were terminated at a depth of 100 feet and no rock was encountered in any of the borings.

Groundwater was encountered at depths ranging from 12.5 feet to 50 feet during the drilling operations. The higher groundwater levels were most likely attributed to the introduction of water during the mud rotary drilling and possible cave-ins of the boreholes. For design, a groundwater level at a depth of 22.5 feet (Elevation 90 feet) is considered.

This subsurface soil profile is consistent with the soil profile encountered during the original subsurface investigation (Law report, Reference 2.19) performed for the initial ISFSI facility.

2.6.2 VIBRATORY GROUND MOTION

A seismological study for the Calvert Cliffs ISFSI has been performed to determine the Operating Basis and the Design Basis earthquakes for the site and the ground motion associated with them. This section summarizes the study that is detailed in Reference 2.1.

The site is located in a region which has experienced infrequent and minor earthquake activity. Most of the reported earthquakes are related to known faulting more than 50 miles west of the site in the Piedmont Physiographic Province. No known faults occur in the vicinity of the site. The closest earthquake (Intensity VII) which caused any structural damage occurred about 80 miles southwest of the site. Several minor shocks (no greater than Intensity V) have been reported within 50 miles of the site. Because of very limited data, it is not possible to determine whether or not these were of tectonic origin.

The foundations of the ISFSI are established in soil which will not undergo reduction in strength or increased settlement under design basis earthquake conditions.

Significant earthquake ground motion is not expected at the site during the life of the facility. On a conservative basis, the ISFSI is designed to respond elastically to horizontal ground accelerations as high as 15% of gravity and a simultaneous maximum vertical ground acceleration of 10% of gravity.

2.6.2.1 Tectonics

The site is located in the Coastal Plain Physiographic Province. This province is bounded on the east by the Atlantic Ocean, and on the west by the Fall Zone and the Piedmont Physiographic Province. The Coastal Plain consists of easterly dipping Cretaceous and Tertiary sediments which are about 2,500' thick at the site. Crystalline basement rock outcrops near the Fall Zone about 50 miles west of the site.

On the basis of regional data, the Cretaceous and Tertiary sediments are undeformed. The absence of folding and faulting in the sedimentary strata indicates that displacements along unknown faults which may be present in the basement have been negligible.

No known faults occur within the basement rock or sedimentary deposits in the vicinity of the site. The closest known fault systems are found in the rocks of the Piedmont, more than 50 miles west and northwest of the site. The Piedmont Province consists of igneous and metamorphic rocks of Precambrian and early Paleozoic Age, with areas of sedimentary and igneous rocks of Triassic Age. Major tectonic activity that has occurred in the Mid-Atlantic Region can be related to known faults in the Piedmont Province.

2.6.2.2 Seismicity

The site is situated in a region which has experienced only infrequent minor earthquake activity. No shock within 50 miles of the site has been large enough to cause significant structural damage. Since the region has been populated for over 300 years, it is probable that all earthquakes of moderate intensity, approximately VI[1] or greater, would have been reported during this period.

The first report of earthquake occurrence in the general area of the site dates back to the late 18th Century. Through 1989, 18 earthquakes with epicentral intensities of V or greater on the Modified Mercalli[1] Scale were reported within about 100 miles of the site. None of these shocks was greater than Intensity VII. Few were of high enough intensity to cause structural damage and only one of these shocks can be considered more than a minor disturbance. This was an Intensity VII shock near Wilmington, DE in 1871 about 100 miles from the site. A list of earthquakes of Intensity V or greater with epicenters located within a distance of about 100 miles of the site is presented in Table 2.6-1, Significant Earthquakes

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1 All intensity values in this report refer to the Modified Mercalli Scale as abridged in 1956 by Richter. The intensity scale is a means of indicating the relative size of an earthquake in terms of its perceptible effect.

within 100 miles of the site. Several smaller earthquakes, which are significant because of their proximity to the site, are also included in Table 2.6-1. The locations of these and other earthquakes in the region surrounding the site are shown on Figure 2.6-8, Epicentral Location Map. Several small shocks are shown on the Epicentral Location Map but not indicated in Table 2.6-1. Little information is available regarding these shocks.

Most of the reported earthquakes in the region have occurred in the Piedmont Physiographic Province west of the Fall Zone. The closest approach of the Fall Zone to the site is about 50 miles. These shocks were generally related to known faults in the Piedmont rocks.

There have been several large shocks with epicenters in the Coastal Plain, some of which were damaging. The largest of these is the Charleston, SC, earthquake of 1886, which had an epicentral intensity of about IX. This earthquake is probably related to faulting in the basement rock near Charleston.

Among the largest earthquakes in the Coastal Plain close enough to the site to be of significance was the 1927 earthquake near the northern New Jersey coast, about 180 miles northeast of the Calvert Cliffs site. The epicentral intensity of this earthquake was VII. Three shocks were felt over an area of about 3,000 square miles from Sandy Hook to Tom's River. Highest intensities were felt from Asbury Park to Long Branch where several chimneys fell, plaster cracked, and articles were thrown from shelves. This shock has not been related to any known geologic feature.

An earthquake which occurred near Wilmington, DE, in 1871 is the largest reported earthquake within 100 miles of the proposed ISFSI. It is not possible to accurately locate the epicenter of this shock with the limited data available, but it is probable that the shock occurred along the Fall Zone about 100 miles northeast of the site. The epicentral intensity of this shock is rated at VII. At Wilmington, chimneys toppled and windows broke. Damage was also reported at Newport, New Castle, and Oxford, DE. The earthquake was felt over a relatively small area of northern Delaware, southeastern Pennsylvania, and southwestern New Jersey.

Only one earthquake of Intensity V or greater has been reported within 50 miles of the ISFSI. This shock, which had a rated epicentral intensity of V, caused no structural damage. Its epicenter was located near Seaford, DE, about 45 miles northeast of the site.

2.6.2.3 Seismic Design

On the basis of the seismic history of the area, it does not appear likely that the ISFSI site will experience significant earthquake ground motion during the life of the installation. The ISFSI was conservatively designed to respond, with no loss of function, to horizontal earthquake ground accelerations of 15% of gravity, and vertical earthquake ground accelerations of 10% of gravity. It is not believed that this level of ground motion will be exceeded at the site during an earthquake similar to any historical event.

Ground motion response spectra for the Calvert Cliffs ISFSI are presented on Figures 2.6-9 and 2.6-10. The spectra have been normalized to a horizontal ground acceleration of 8% of gravity for the operating basis earthquake and 15% of gravity for the design basis earthquake.

A Soil-Structure Interaction (SSI) analysis specific to the CCNPP ISFSI HSM-HB pads was performed (Reference 2.22). Based on the results of the SSI analyses, the following is concluded:

- The maximum seismic acceleration at top of the HSM-HB concrete pad is 0.21g in the horizontal directions (in NS and EW directions) and 0.13g in the vertical direction,
- The maximum seismic acceleration at center of gravity of the HSM-HB is 0.23g in the horizontal directions (in NS and EW directions) and 0.13g in the vertical direction,
- The maximum seismic acceleration at the center of gravity of the DSC in the HSM-HB is 0.28g in the horizontal directions (in NS and EW directions) and 0.13g in the vertical direction, and
- The existing pads with fuel-loaded HSMs do not have an impact on the SSI results of the HSM-HB pads.

These values are enveloped by the 0.3g horizontal and 0.2g vertical design basis accelerations used for the HSM-HB.

2.6.3 SURFACE FAULTING

See Section 2.6.2.

2.6.4 STABILITY OF SUBSURFACE MATERIALS

Before the initial HSMs were constructed, twelve borings were drilled in the immediate vicinity of the ISFSI to evaluate the static and dynamic properties of materials underlying the site. The general site area is shown on Figure 2.4-1 and the location of all the initial borings is shown on Figure 2.6-7. The drilling, sampling, and testing were performed in accordance with methods specified by the American Society for Testing and Materials. Boring logs are shown on Figures 2.6-7B through 2.6-7M. A subsurface profile along the north-south axis of the site is shown on Figure 2.6-7A.

A discussion of subsurface conditions as determined for field and lab testing is presented in Section 2.6.1.2. Structural mat foundations bear on in-situ soils or structural fill compacted to 95% of modified proctor maximum dry density. The maximum allowable static bearing pressure used in analysis of the ISFSI structures is 3.0 kips/ft^2 which provides a safety factor of 4.0 for the ultimate bearing capacity of site soils. Calculated total settlements are estimated to be about 1.5". Consolidation characteristics of the site soils indicate that about 1" of the total settlement will occur during construction of the ISFSI structures. The remaining 1/2" of settlement will occur over the life of the structure.

The liquefaction potential of the sands and silty sands encountered in the borings were analyzed using both correlations of field and laboratory test data. A simplified procedure was used to compare standard penetration values (N) obtained from known liquefiable soils at various sites around the world with N values recorded for the ISFSI area soils.

The majority of the recorded N values fall within the zone considered to be unlikely to experience liquefaction. Eight of the 46 N values fall within the zone where prediction of liquefaction potential is difficult without some type of laboratory testing. Only two N values fall within the zone of high liquefaction potential.

Laboratory cyclic triaxial tests were performed on two undisturbed sand samples obtained from areas thought to be representative of the most potentially liquefiable soils encountered on site. The samples were obtained using a standard 3" diameter shelby tube sampler. The samples were tested at the geotechnical testing laboratories of Virginia Polytechnic Institute and State University in Blacksburg, VA.

Comparison of the cyclic stress ratio anticipated for this site under design conditions and that available based on the laboratory tests performed indicate liquifaction would not occur.

Another analysis of the liquefaction potential was performed using the cyclic strain method. This method incorporates the stiffness of the soil in terms of shear modulus value and the reduction of that value during the earthquake.

Within the soil profile encountered at the site, only the soils at or around a depth of 30' show a slight potential for a rise in pore pressure. However, the cyclic strain level is only slightly above the threshold value and given the low number of equivalent cycles for the design earthquake no significant potential for liquefaction is anticipated.

Liquefaction of site soils is not anticipated based on established water table depths, standard penetration testing and cyclic triaxial shear testing.

A calculation (Reference 2.20) was generated to determine the allowable bearing capacity and associated settlement for the layout of the HSM-HBs. Based on the properties of the soil materials encountered in the 2007 boring program, the allowable bearing capacity was determined to be 4000 psf considering a factor of safety of 3. For a design load of 2450 psf, the anticipated total settlement is 1.8 inches, comprised of approximately 1.1 inch of immediate settlement and 0.7 inches of long-term settlement. The anticipated differential settlement between the center of the modules and the corner is approximately 0.9 inches. Based on this analysis, approximately 60% of this settlement would occur immediately after application of the load. This settlement is generally within acceptable limits for the storage modules planned for this facility since tipover of the modules is not a concern (Reference 2.25).

The bearing capacity and settlement values determined from the new boring program are also in close agreement with the calculated settlements determined for the initial design of the facility.

Based on the soil data provided in the Geotechnical Exploration report (Reference 2.18), an evaluation of liquefaction potential of the soil profile at the ISFSI site has been performed. The soil profile at the site generally consists of silty sand and clayey sand with silt and clay. Screening method in combination with the empirical method, as recommended in Regulatory Guide 1.198 (Reference 2.23) has been used for the evaluation. Acceptance criteria are based on a factor of safety of 1.4 against liquefaction, as recommended in Regulatory Guide 1.198 for the method of evaluation. Based on the evaluation results, it is concluded that the factor of safety against liquefaction during a seismic event is 1.4 or greater. Hence, the site does not pose any soil liquefaction risk to the ISFSI pads during a seismic event.

Based on the results of the new (2007) subsurface exploration and comparison to the original design for the ISFSI, the following conclusions are made:

1. The 2007 subsurface exploration program confirmed that the soil conditions encountered are similar to the soil conditions encountered during the original subsurface exploration.
2. The factor of safety against liquefaction during a seismic event is 1.4 or greater. Hence, the site does not pose any soil liquefaction risk to the ISFSI pads during the seismic event (Reference 2.21)
3. The in-situ soils are capable of supporting a design load of 2450 psf from the HSM-HBs on shallow mat foundations.
4. The calculated bearing capacity of the soil and anticipated settlement for the HSM-HBs is very similar to that calculated for the original ISFSI foundations. This indicates close agreement between the soil conditions, as was expected.
5. Considering that the estimated total settlement of 1.8 inches and differential settlement of 0.9 inches are within acceptable ranges, no soil remediation is required.
6. Considering a frost depth of 24 inches and a mat thickness of 36 inches, no special subgrade treatment, i.e., the placement of additional frost free material, is required.
7. Since the HSM-HB are a horizontal storage system, tipover of the system is not a concern.

2.6.5 SLOPE STABILITY

As shown on Figure 2.4-1 the original ISFSI was created by a combination of cut and fill to minimize earthwork volumes. The resulting cuts, primarily on the north end of the site, are a maximum of about 16' high and the resulting fills on the south end of the yard are a maximum of 12' high. Since the original poured in place HSMs are located a minimum of 60' from any cut or fill slope and approximately 115' from the maximum cut slope and 95' from the maximum fill slope, there is no potential impact to the safe operation of the ISFSI by postulated site slope failures for the original poured in place HSMs.

Stability analyses were performed for the south slope of the southernmost ISFSI HSM-HB foundation pad to determine the factor of safety of the embankment fill against a mass instability. The soil profile analyzed included the fill placed for the ISFSI embankment, the in-situ soils, as well as the weight of the ISFSI pad and HSM-HBs placed on the ISFSI pad. The slope stability analyses were performed in two dimensions for both static and seismic conditions. The pseudo-static method was used for the seismic analyses. The analyses concluded that the ISFSI pad analyzed is stable under the static and seismic conditions (Reference 2.24).

TABLE 2.6-1
SIGNIFICANT EARTHQUAKES WITHIN 100 MILES OF THE SITE

YEAR	DATE	TIME	INTENSITY[b]	LOCATION	N. LAT.	W. LONG.	AREA FELT (sq. mi.)	DISTANCE FROM SITE (miles)
1733	Jun 14	--	[a]	Vicinity of Annapolis, MD	--	--	--	--
1758	Apr 24	--	[a]	Vicinity of Annapolis, MD	--	--	--	--
1774	Feb 21	14:00	VI	Richmond, VA	37.5	77.5	--	80
1833	Aug 27	06:00	VI	Central Virginia	37.75	78	52,000	90
1871	Oct 9	09:40	VII	Wilmington, DE	39.75	75.5	--	100
1875	Dec 22	23:45	VI	Near Richmond, VA	37	77.5	50,000	80
1876	Jun 19	--	[a]	Vicinity of Annapolis, MD	--	--	--	--
1879	Mar 25	19:30	IV-V	Northern Delaware	39.75	75.5	600	100
1883	Mar 11	18:57	IV-V	Harford County, MD	39.5	76.5	Local	80
	Mar 12	00:00	IV-V	Harford County, MD	39.5	76.5	Local	80
1885	Jan 2	21:16	V	Frederick County	39.5	77.5	3,500	80
1897	Dec 18	18:45	V	Ashland, VA	37.75	77.5	7,500	75
1906	May 8	12:41	V	Seaford, DE	38.75	75.75	400	45
1908	Aug 23	04:30	V	Powhatan, VA	37.5	78	450	95
1919	Sep 5	21:46	VI	Front Royal, VA	38.75	78.25	--	95
1930	Jan 18	--	IV[a]	Pines of the Severn, MD	--	--	--	--
1930	Nov 1	01:34	I-III[a]	Anne Arundel County, MD	39.0	76.5	Local	--
1949	May 8	06:01	IV-V	Richmond, VA	37.5	77.5	1,800	80
1966	May 31	06:19	IV-V	Central Virginia	37.5	78.0	--	100
1969	Dec 11	18:45	V	Richmond, VA	37.8	77.4	6,500	70
1971	Sep 12	00:06	V	Virginia	38.1	77.4	1,900	55
1972	Sep 5	16:00	V	Richmond, VA	37.6	77.7	2,300	85
1977	Feb 10	19:14	VI	Northern Delaware	39.5	75.3	Local	100

(a) Several small shocks in Maryland are included in this table. Little information is available regarding these reports, and the indicated epicenters are uncertain.

(b) ll intensity values refer to the Modified Mercalli Scale as abridged in 1956 by Richter. The intensity scale is a means of indicating the relative size of an earthquake in terms of its perceptible effect.

2.7 SUMMARY OF SITE CONDITIONS AFFECTING CONSTRUCTION AND OPERATING REQUIREMENTS

The site-specific phenomena and characteristics described in this Chapter have been used to define appropriate design criteria, as described in Chapter 3. Table 2.7-1 is a summary of site-specific information either newly established for the ISFSI or previously established for CCNPP.

TABLE 2.7-1
SITE CHARACTERISTICS SUMMARY

FACTOR	VALUE OR RANGE
Ambient temperature	-3°F to 103°F
Direct exposure to sunlight	82 Btu/hr/ft^2
Ambient humidity	0 to 100%
Tornado pressure drop	3 psi in 1.5 sec
Tornado winds:	
translational velocity	70 mph
rotational velocity	290 mph
Maximum flood level	Flood Free Zone
Snow and ice loadings	30 lbs/ft^2
Atmospheric dilution value (χ/Q)	3.0×10^{-4} sec/m^3
Design basis earthquake:	
max horizontal acceleration	0.15 g
max vertical acceleration	0.10 g
Operating basis earthquake:	
max horizontal acceleration	0.08 g
max vertical acceleration	0.053 g

2.8 REFERENCES

2.1 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

2.2 Calvert Cliffs Nuclear Power Plant Environmental Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company, November 16, 1970

2.3 Baltimore-Washington International Airport, 1987, Local Climatological Data, Annual Summary with Comparative Data, National Oceanic and Atmospheric Administration, Environmental Data Service, Asheville, NC

2.4 Washington, DC National Airport, 1984, Local Climatological Data, Annual Summary with Comparative Data, National Oceanic and Atmospheric Administration, Environmental Data Service, Asheville, NC

2.5 Patuxent River Naval Air Station, Revised Uniform Summary of Surface Weather Observations, U.S. Air Force Environmental Technical Applications Center Air Weather Service

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2.7 Comparison of 1984 Through 1986 Meteorological Data from the Calvert Cliffs Primary and Backup Meteorological Towers, Pickard, Lowe, and Garrick, Inc., Volume 1 and 2, Unpublished Study, January 1988

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2.9 Turner, D. B., Workbook of Atmospheric Dispersion Estimates, AP-26, Office of Air Programs, Cincinnati, OH, 1970

2.10 Interim Report — Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coasts of the United States, Memorandum HUR 7-97, Environmental Science Services Administration

2.11 Shore Protection Planning and Design, Technical Report No. 4, U.S. Army Coastal Engineering Research Center

2.12 United States Earthquakes (Serial Publications, 1928 through 1965), United States Department of Commerce, Coast and Geodetic Survey, Washington, DC

2.13 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)

2.14 "Probability Estimates of Temperature Extremes for the Contiguous United States," NUREG/CR-1390, U.S. NRC, May 1980

2.15 "1989 ASHRAE Handbook Fundamentals, I-P Edition," American Society of Heating and Air Conditioning Engineers, Inc., 1989

2.16 Letter from Ms. D. M. Skay (NRC) to Mr. C. H. Cruse (CCNPP), dated August 29, 2001, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Aircraft Hazards Analysis (TAC Nos. MA7229 and MA7230)

2.17 Final Environmental Impact Statement, Increased Flight and Related Operations in the Patuxent River Complex, Patuxent River, Maryland: Department of the Navy, Naval Air Warfare Center, Aircraft Division, December 1998

2.18 Report of Geotechnical Exploration, Independent Spent Fuel Storage Installation (ISFSI) Facility, Calvert Cliffs Nuclear Power Plant, MACTEC Engineering and Consulting (MACTEC), October 15, 2007, Project 3551-07-1070

2.19 Law Engineering Report of Geotechnical Exploration, Independent Spent Fuel Storage Facility, Calvert Cliffs Nuclear Power Plant, Law Engineering Report B8-962, March 10, 1989

2.20 CCNPP Calculation CA07261, Soil Design Parameters for ISFSI Mat Foundation, Sargent & Lundy Calculation 11562-019-CALGEO-1, Revision 1, September 17, 2009

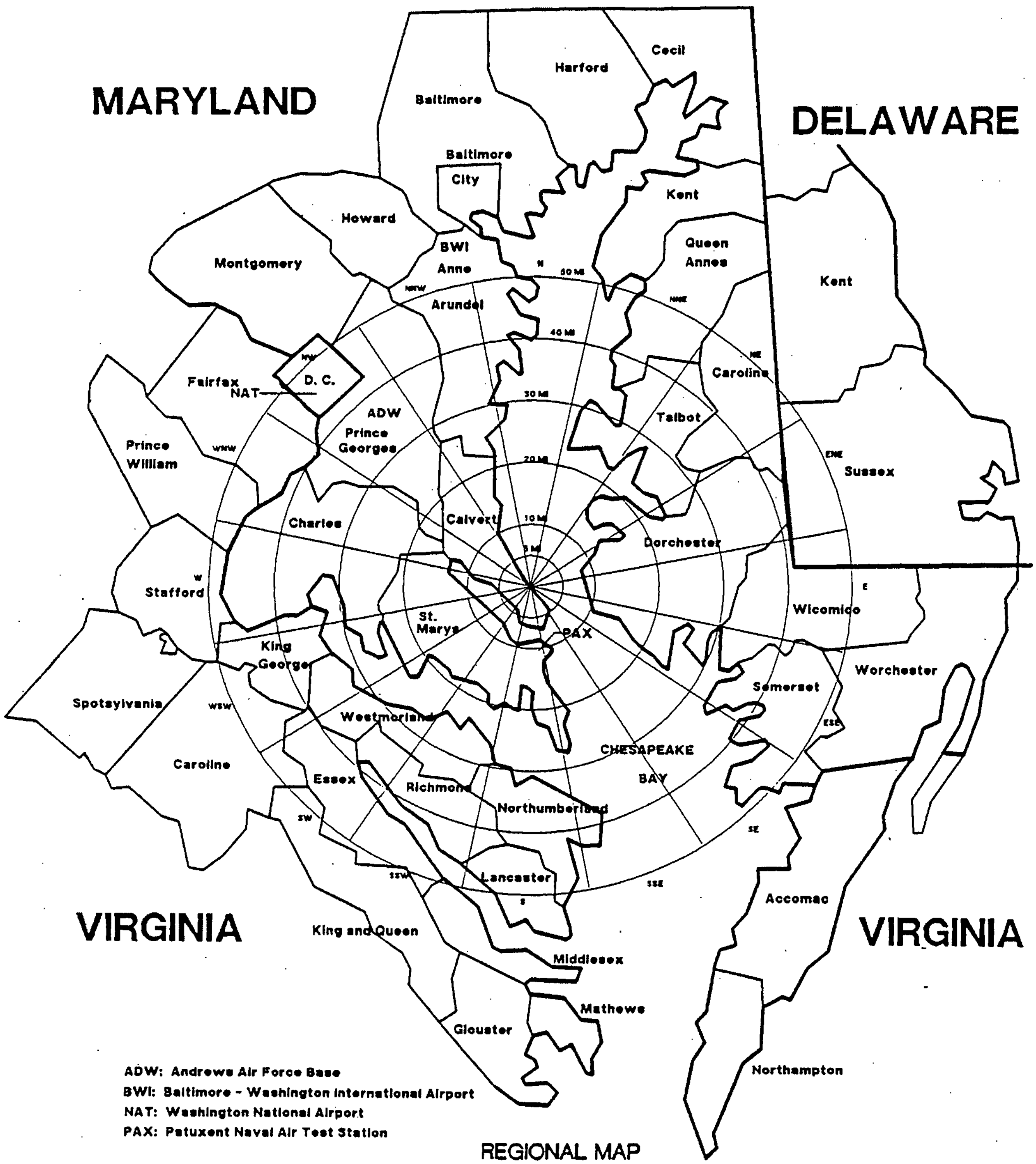
2.21 CCNPP Calculation, CA07266, Soil Liquefaction Evaluation, Sargent & Lundy Calculation 11562-019-ST-1, Revision 0, June 21, 2009

2.22 CCNPP Calculation CA07141, Seismic Soil-Structure Interaction Analysis of ISFSI Pad, Sargent & Lundy Calculation 11562-012-ST2, Revision 1, August 22, 2009

2.23 Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plants Sites, Regulatory Guide 1.198, U.S. Nuclear Regulatory Commission, November 2003

2.24 ISFSI Pad Embankment Slope Stability, Sargent & Lundy Calculation 11562-19-CALGEO-04, Revision 0, August 19, 2009

2.25 Constellation Generation Group, Calvert Cliffs Nuclear Power Plant Geotechnical Review and Evaluation, Sargent & Lundy Report 11562-012-CIVRPT-02, Revision 0, February 19, 2008



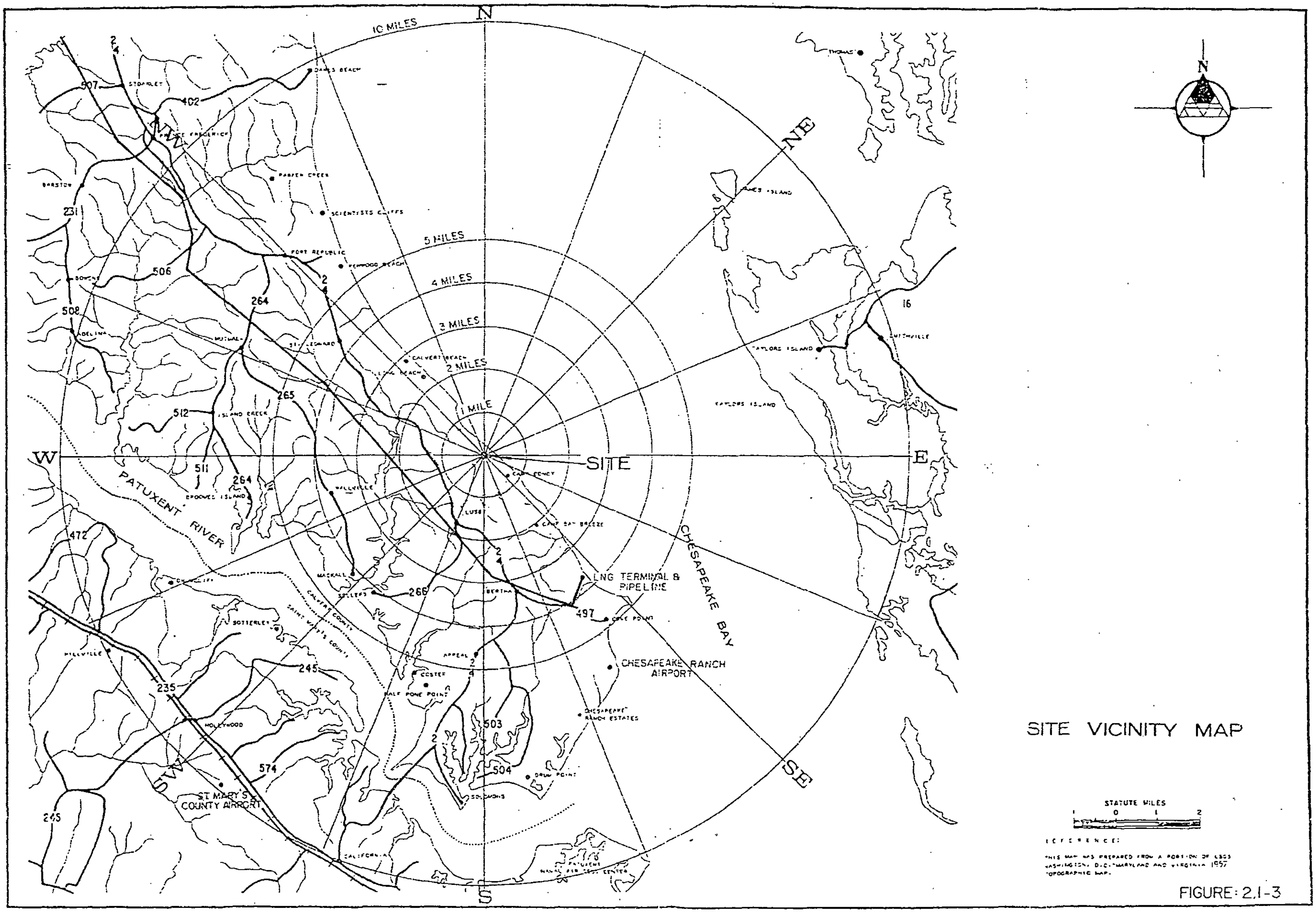
REGIONAL MAP
FIGURE 2.1-1

USAR Figure 2.1-2 CALVERT CLIFFS SITE PLAN

STOP, THINK, ACT AND REVIEW

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SITE PLAN

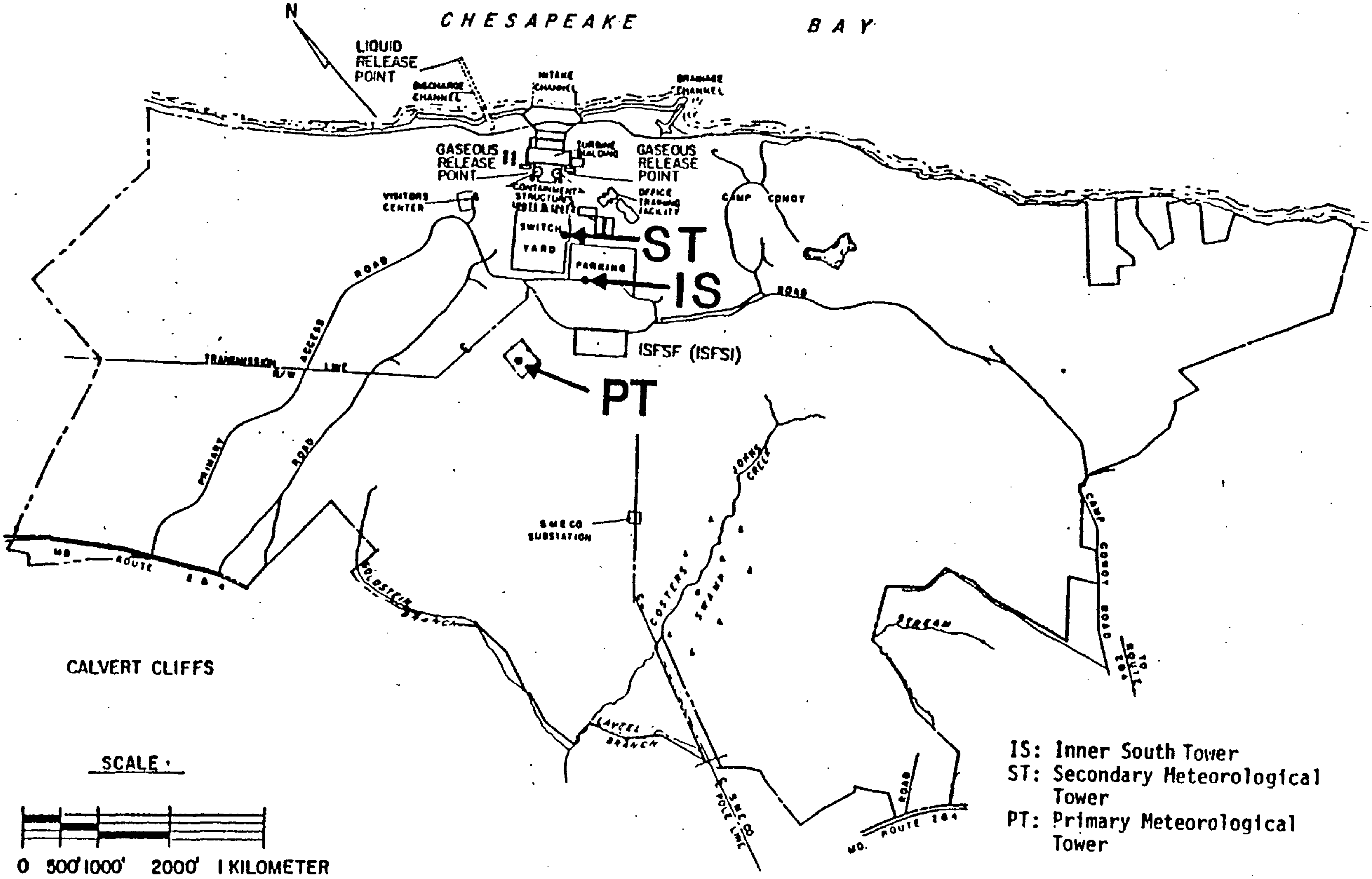


N
S
E
W
NE
NW
SE
SW
10 MILES
5 MILES
4 MILES
3 MILES
2 MILES
1 MILE
SITE
PATUXENT RIVER
CHESAPEAKE BAY
LNG TERMINAL & PIPELINE
CHESAPEAKE RANCH AIRPORT
ST MARY'S COUNTY AIRPORT
SITE VICINITY MAP
STATUTE MILES

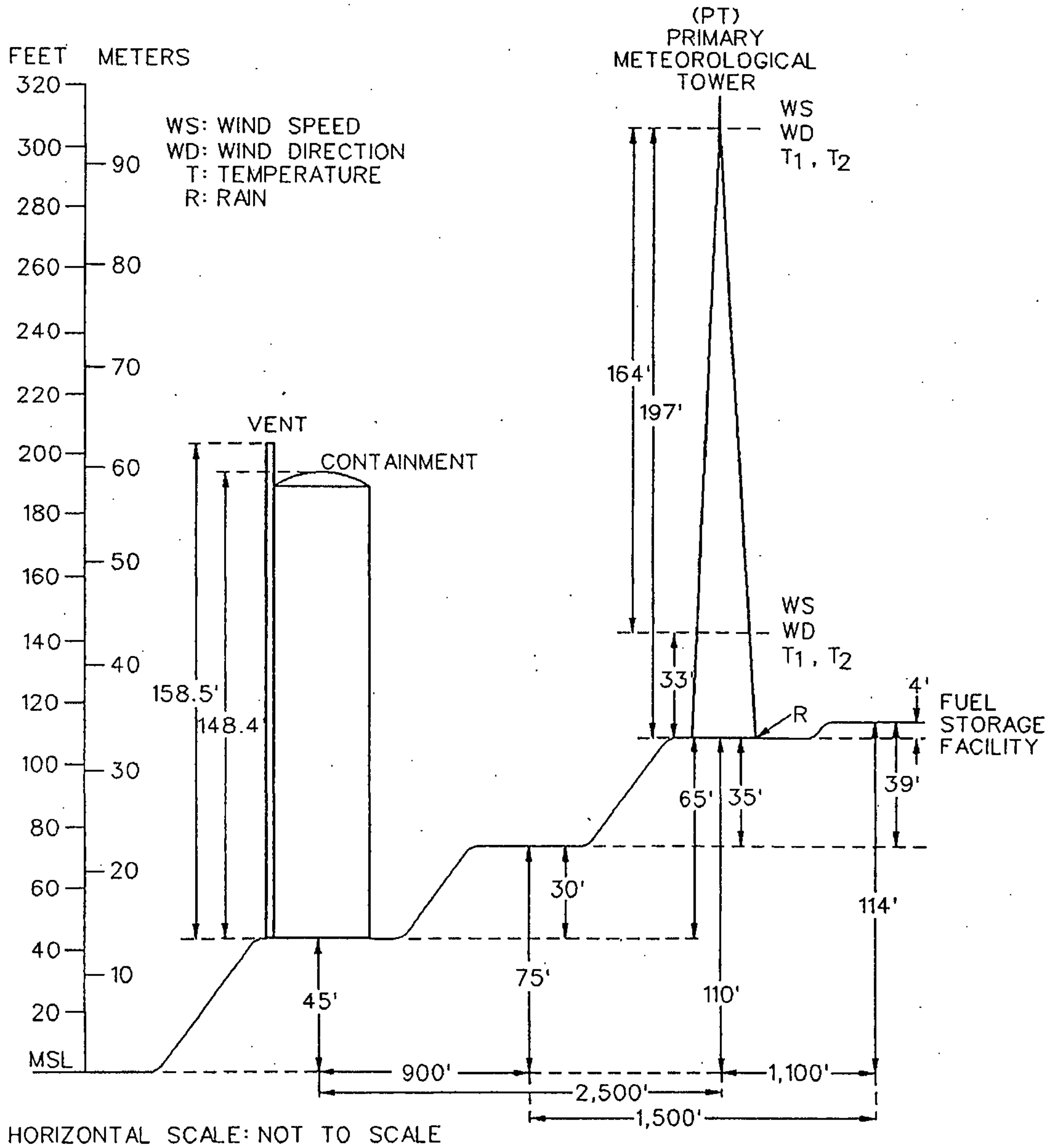
FIGURE: 2.1-3

FIGURE 2.3-1

METEOROLOGICAL INSTRUMENT LOCATIONS



IS: Inner South Tower
ST: Secondary Meteorological Tower
PT: Primary Meteorological Tower



Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	METEOROLOGICAL INSTRUMENT EVALUATIONS	USAR Figure 2.3-2 Revision 15

USAR Figure 2.3-3 METEOROLOGICAL DATA ACQUISITION SYSTEM

STOP, THINK, ACT AND REVIEW

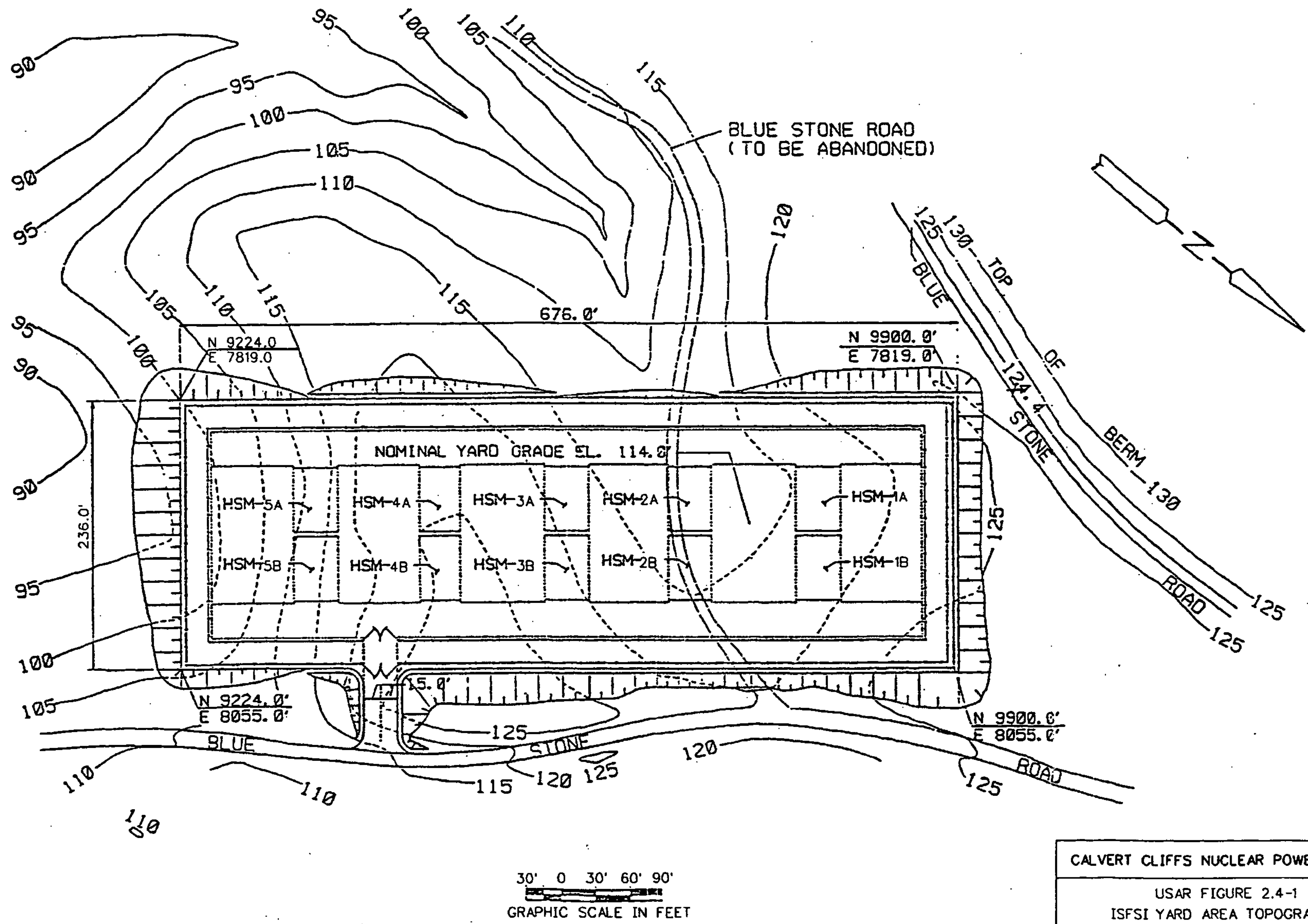
Security-Related Information

Figure Withheld Under 10 CFR 2.390

REV	DATE	DESCRIPTION	[illegible]	[illegible]	[illegible]	[illegible]	APPROVED
0	[illegible]	AS BUILT INFO ONLY. ESP ES199500051	[illegible]	—	—	—	KWJ
1	[illegible]	AS BUILT DCN 87113SH0012-2001SH0001 ES200500027-000	LAG CPM	—	—	—	CPM
2	[illegible]	AS BUILT ECN ECPCN-12-00642-CN-003 ECP-11-000347	[illegible]	—	—	—	KWJ
3	[illegible]	AS BUILT ECN ECP-11-000354-CN-174 REV.001	LAG KWJ	—	—	—	KWJ

PRIMARY METEOROLOGICAL INSTRUMENTATION SYSTEM INFORMATION USAR FIG. NO. 2.3-3

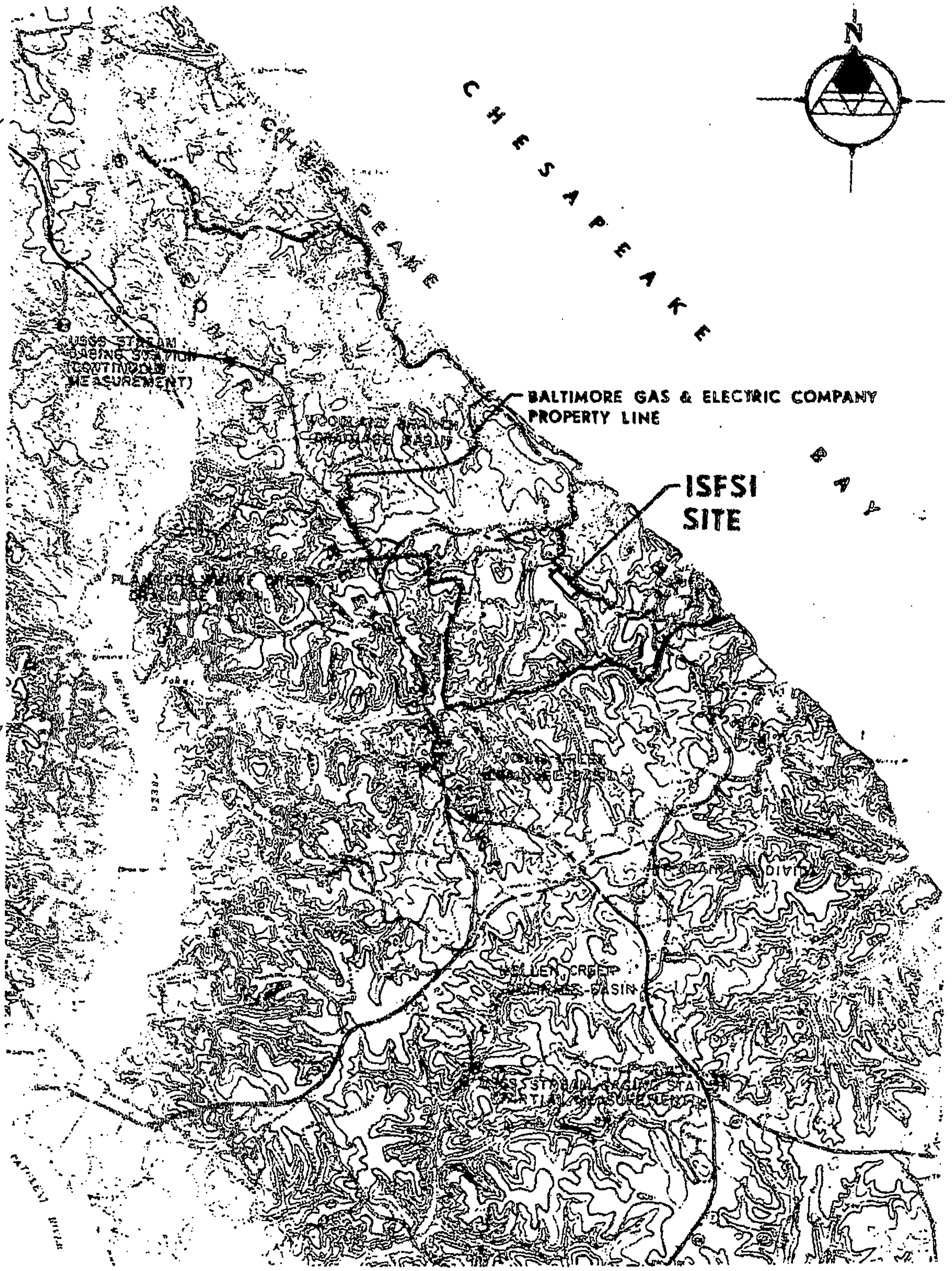
CALVERT CLIFFS NUCLEAR POWER PLANT
ENGINEERING SERVICES DEPARTMENT
CALVERT CLIFFS UNIT 1&2 | SIZE D | CAT. 1
SCALE: NONE | VENDOR DOC. NO. N/A | REV. 3
DWG. NO. 87113SH0012



CALVERT CLIFFS NUCLEAR POWER PLANT

USAR FIGURE 2.4-1
ISFSI YARD AREA TOPOGRAPHY

REV.10
69346.DGN



REGIONAL TOPOGRAPIC MAP

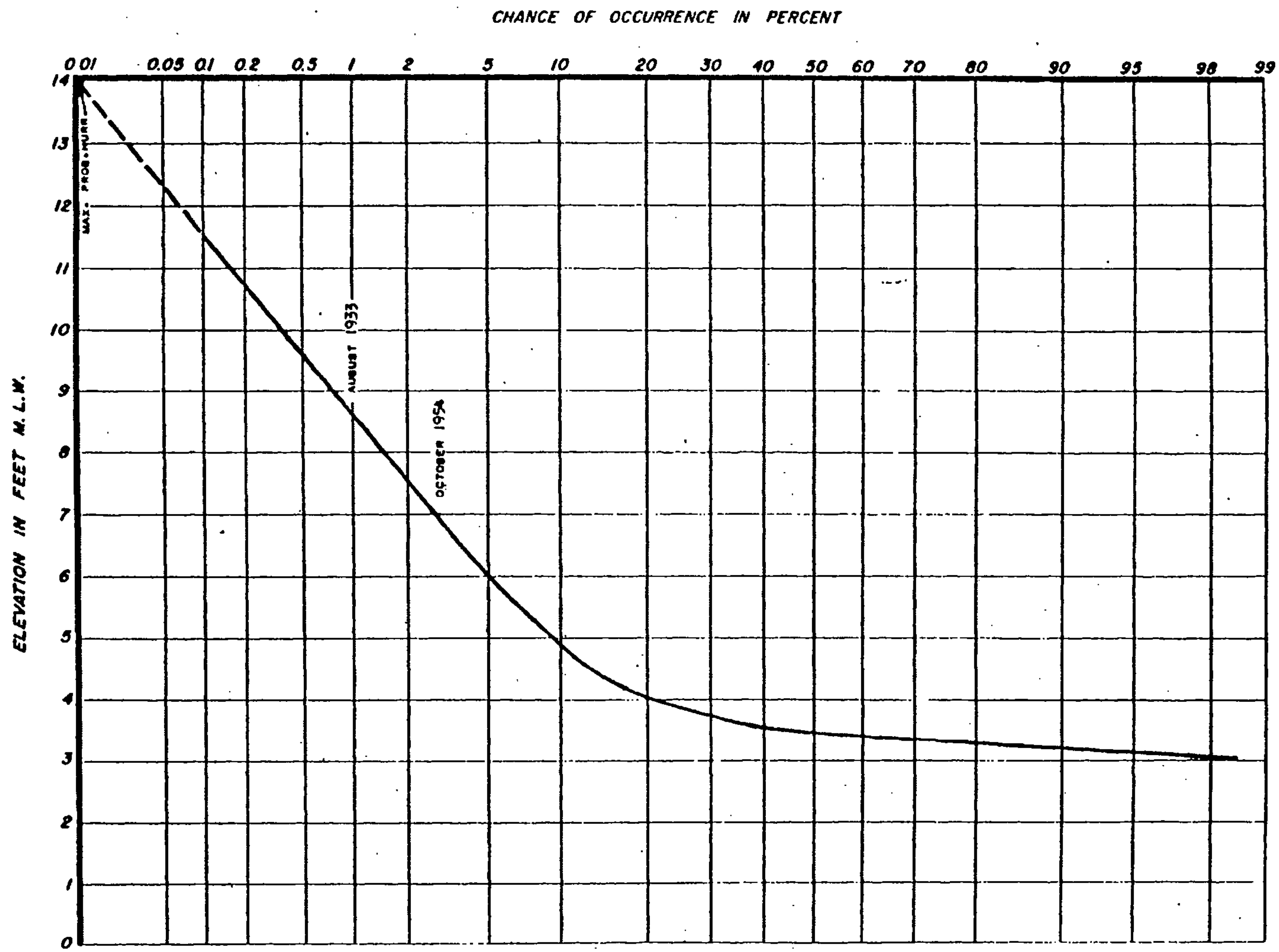
SHOWING SURFACE HYDROLOGY

REFERENCE:

THIS MAP WAS PREPARED FROM A PORTION OF USGS QUADRANGLE COVE POINT, MARYLAND.

FEET
1000 0 2000 4000

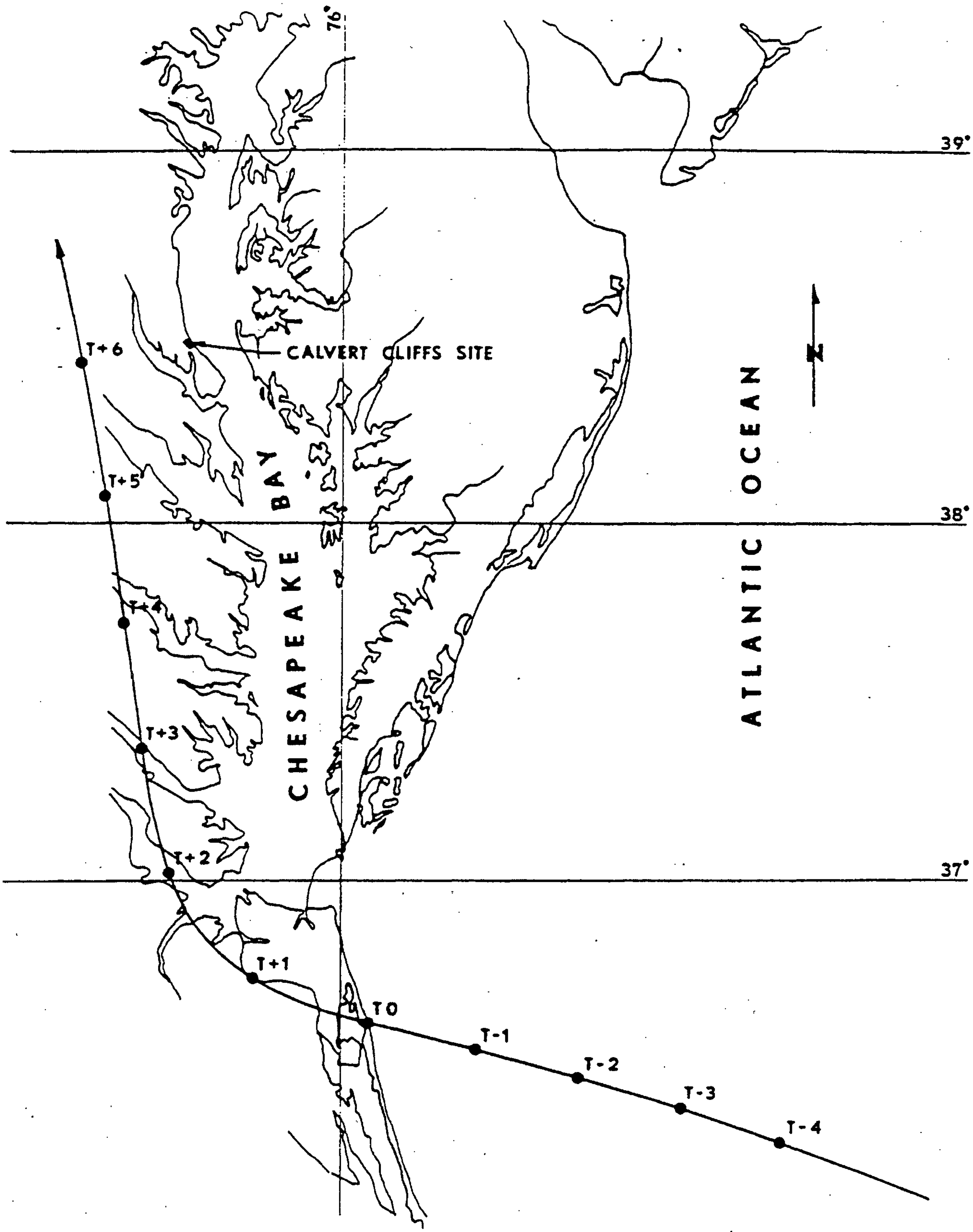
FIGURE 2.4-2



GENERALIZED TIME FREQUENCY CURVE

NOTE:
REPRODUCED FROM PLATE 3 OF H. DOC. NO. 350
88 CONGRESS, 2ND. SESS. "TIDEWATER PORTIONS
OF PATUXENT, POTOMAC AND RAPPAHANNOCK RIVERS,
INCLUDING ADJACENT CHESAPEAKE BAY SHORELINES"

FIGURE 2.4-3



PROBABLE MAXIMUM HURRICANE PATH
FIGURE 2.4-4

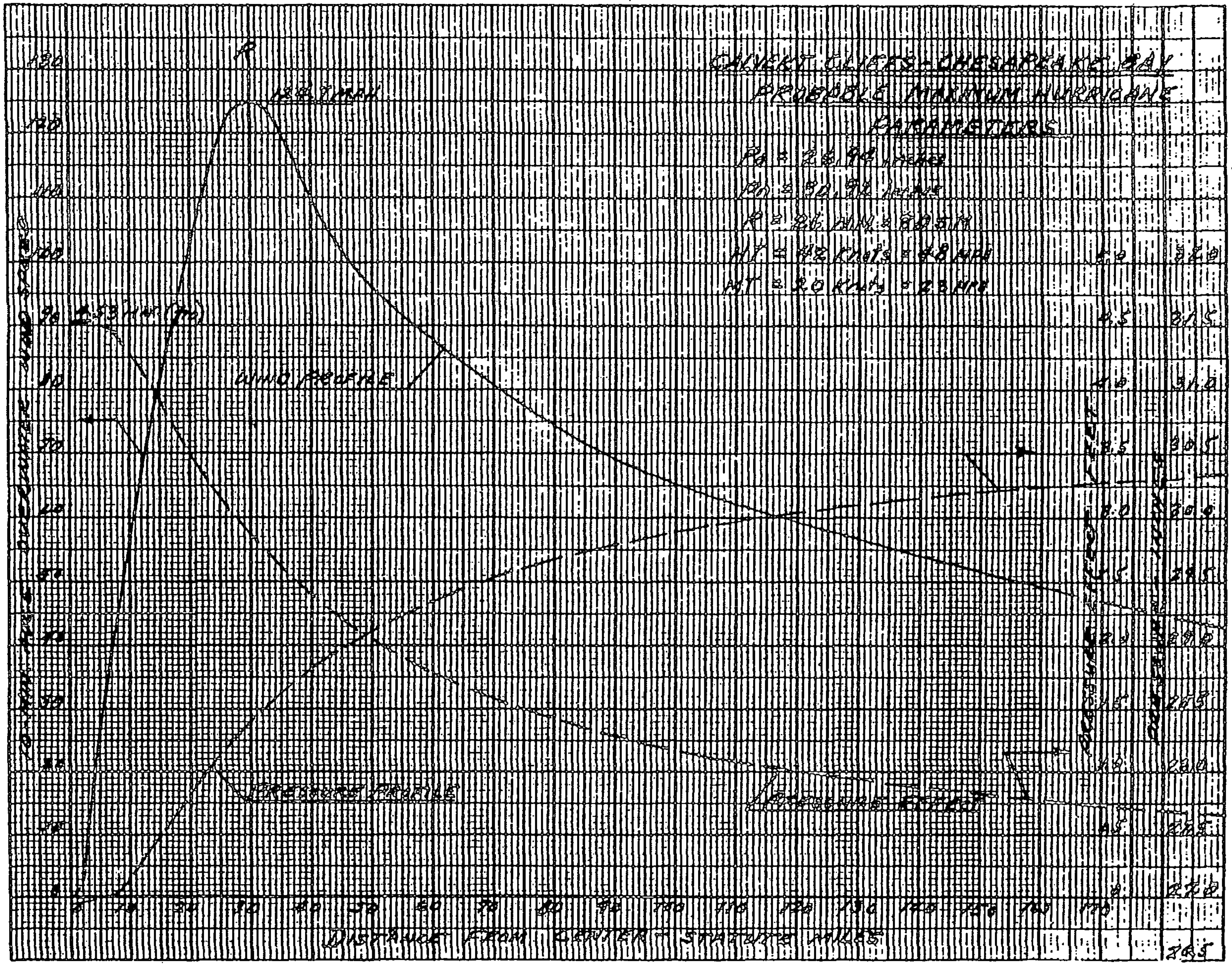
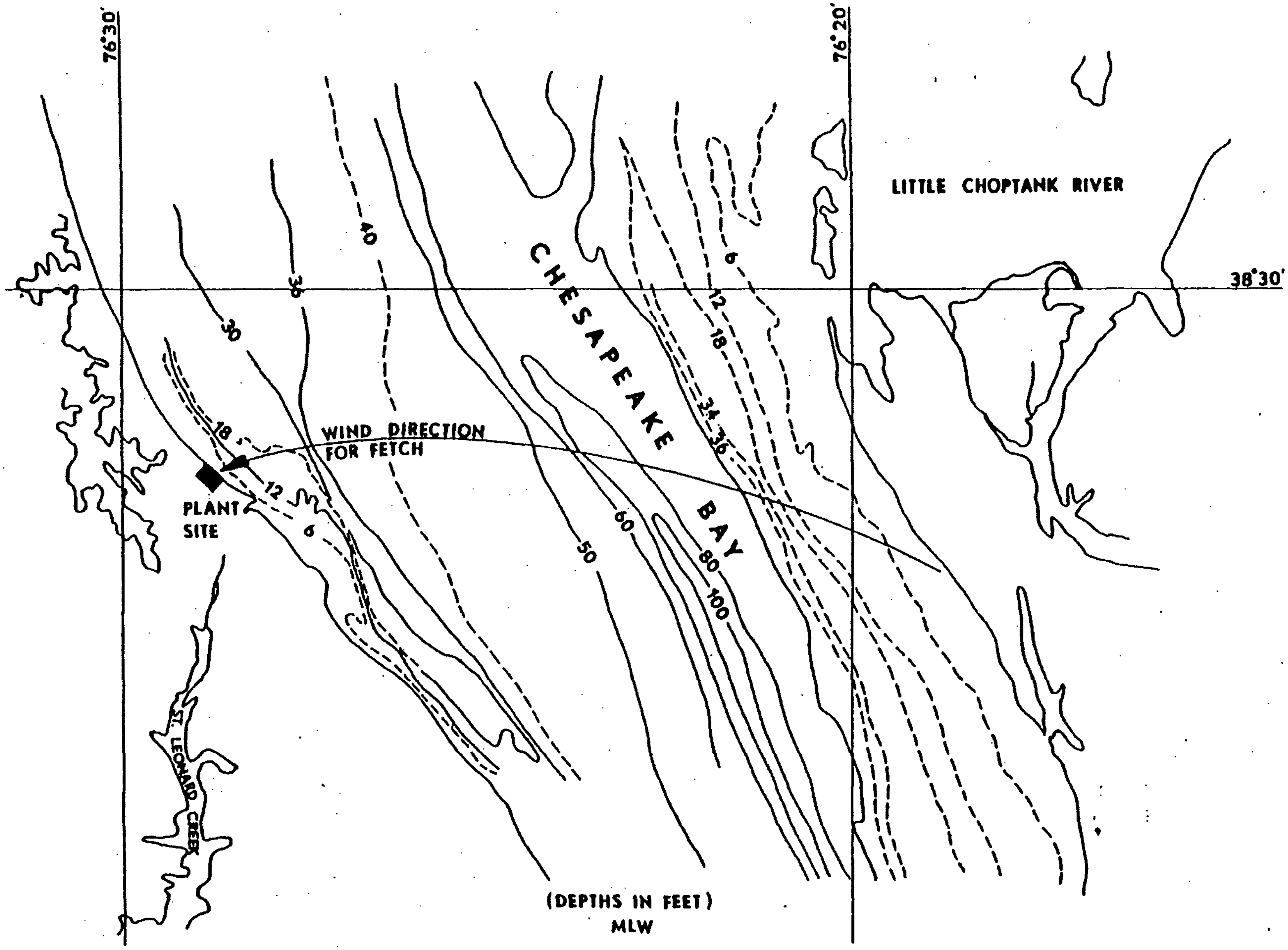
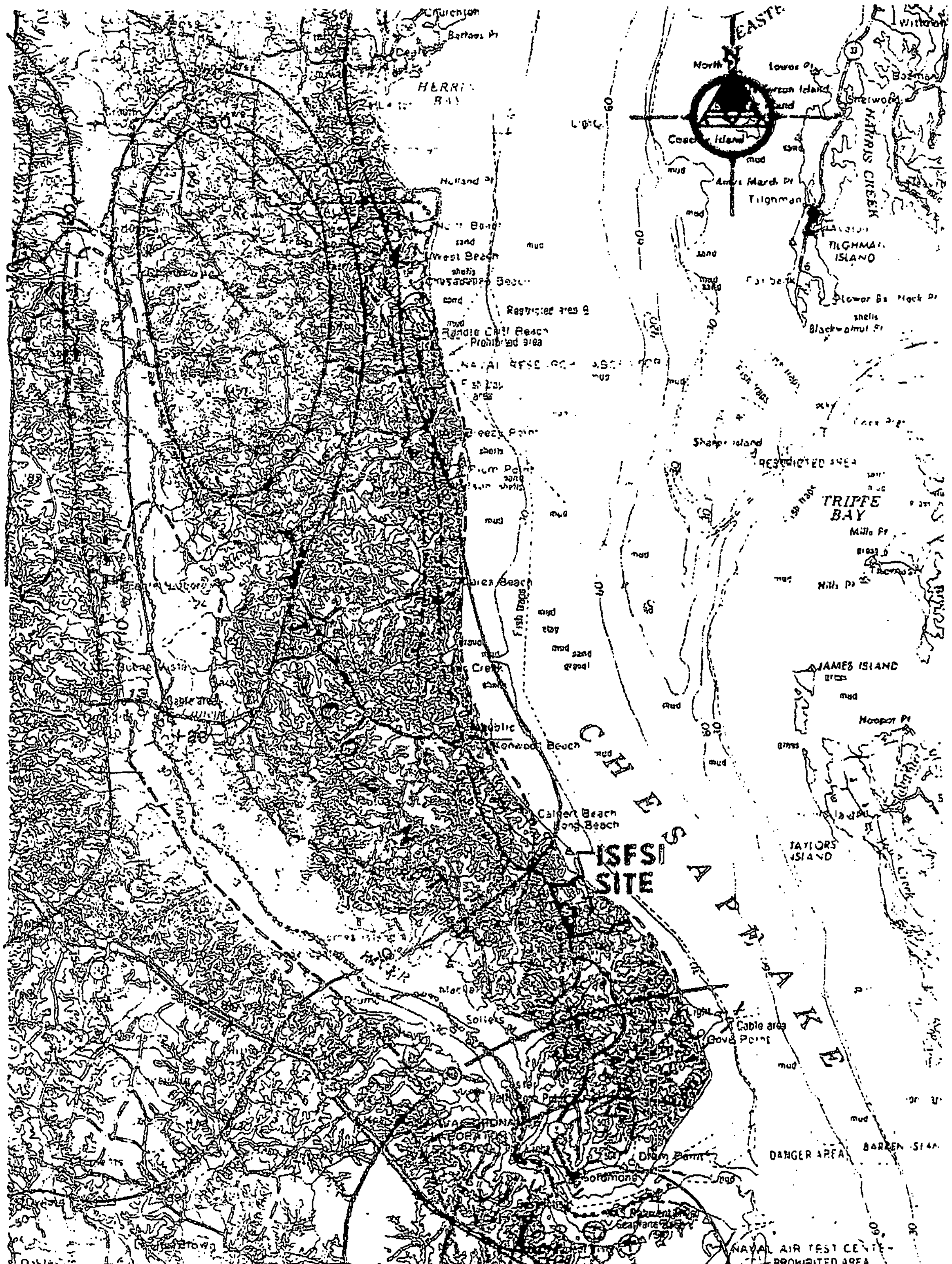

CALVERT CLIFFS - CHESAPEAKE BAY
PROBABLE MAXIMUM HURRICANE
PARAMETERS
WIND PROFILE
PRESSURE PROFILE
DISTANCE FROM CENTER - STATUTE MILES

FIGURE 2.4-5



WIND DIRECTION FOR FETCH

FIGURE 2.4-6



PIEZOMETRIC SURFACES IN CALVERT COUNTY

REFERENCE:

THIS MAP WAS PREPARED FROM A PORTION OF USGS. WASHINGTON, D.C.: MARYLAND: VIRGINIA 1957, SHEET NJ 18-4, REFERENCE FOR PIEZOMETRIC DATA IS MARYLAND DEPT. OF GEOLOGY MINES AND WATER RESOURCES BULLETIN NO. 15.

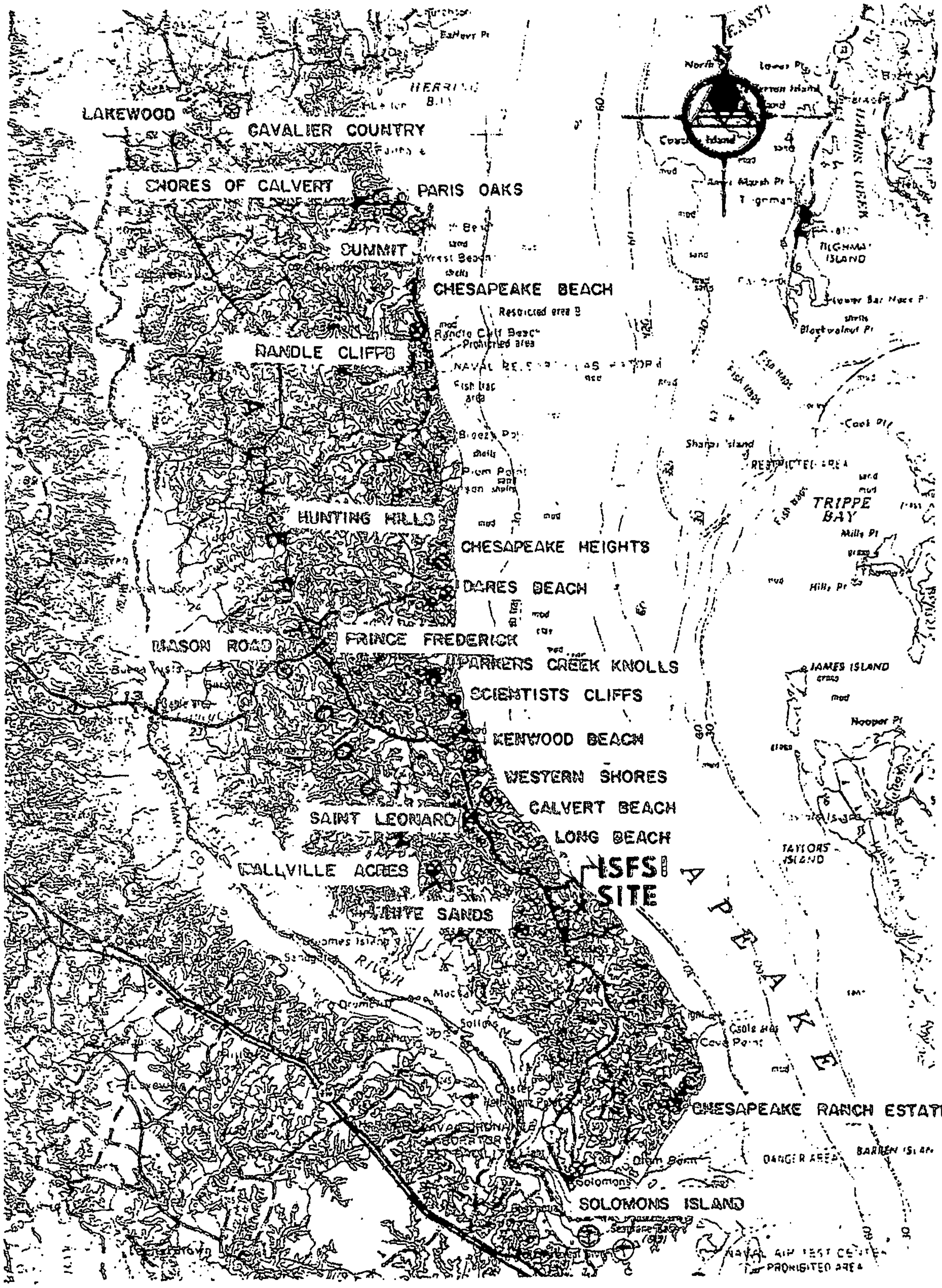
STATUTE MILES

1 0 1 2 3 4 5

KEY:

PIEZOMETRIC SURFACE OF AQUIA FORMATION ———
PIEZOMETRIC SURFACE OF NANJEMOY FORMATION — — —

FIGURE 2.5-1



PUBLIC WATER SUPPLIES IN CALVERT COUNTY

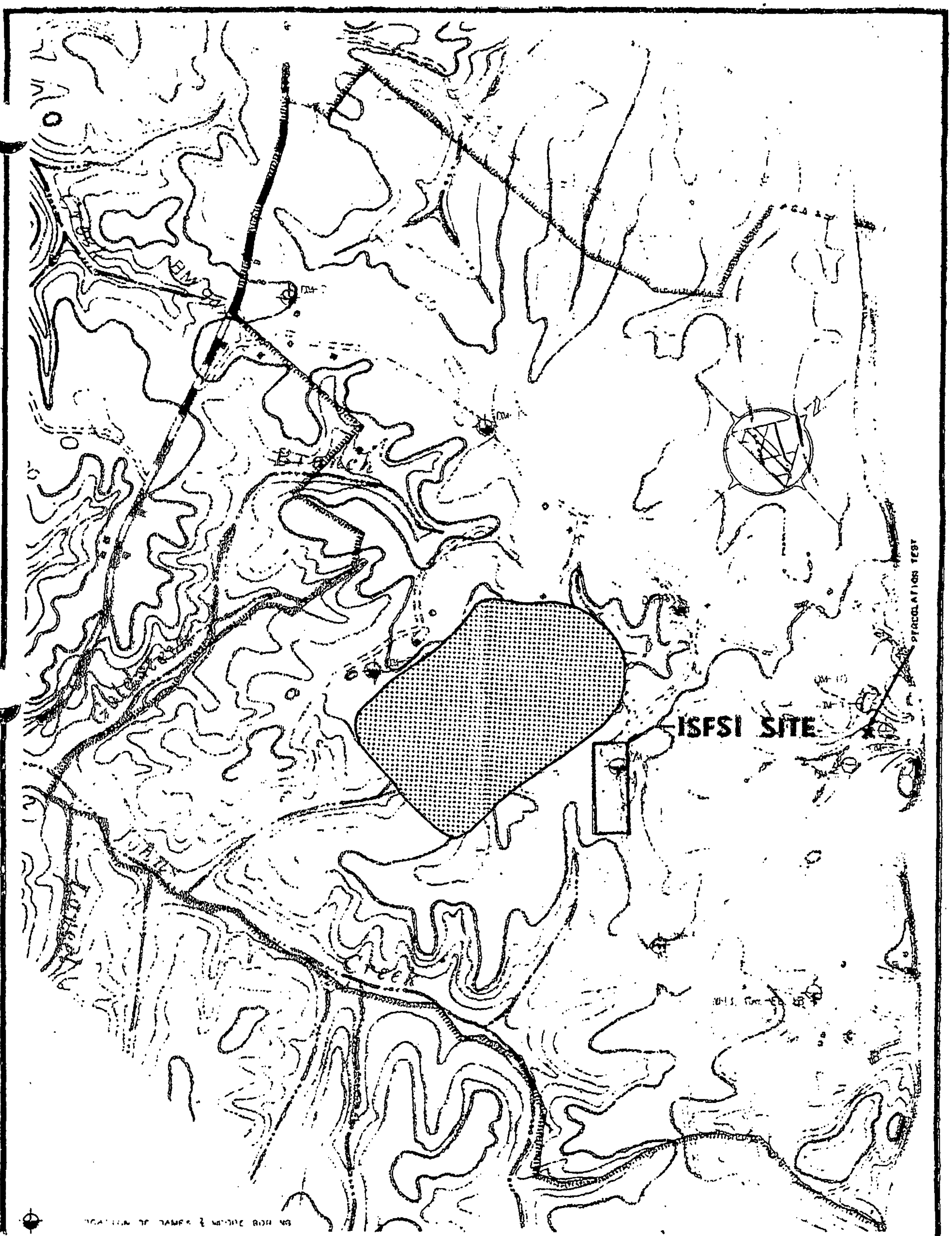
REFERENCE:

THIS MAP WAS PREPARED FROM A PORTION OF USGS WASHINGTON, D.C.; MARYLAND; VIRGINIA 1957, SHEET NJ 18-4, REFERENCE FOR PIEZOMETRIC DATA IS MARYLAND DEPT. OF GEOLOGY MINES AND WATER RESOURCES BULLETIN NO. 15.

STATUTE MILES

1 0 1 2 3 4 5

FIGURE 2.5-2



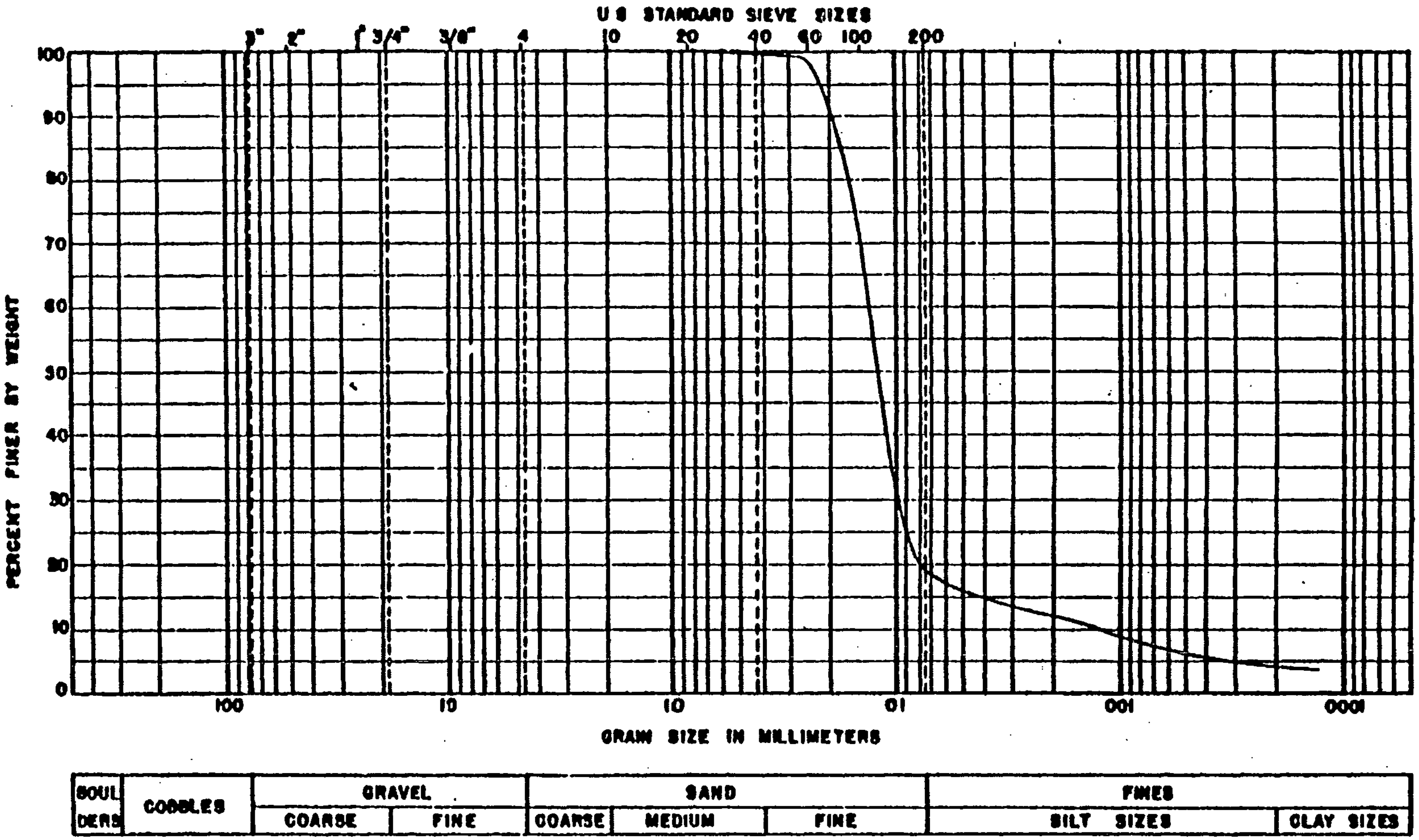
LOCATION OF DAMES & MOORE BORING

FEET
600 0 600 1200

OBSERVATION WELLS
PLOT PLAN

FIGURE 2.5-4

REFERENCE:
THIS MAP WAS PREPARED FROM A PORTION OF USGS., COVE POINT, MARYLAND "QUADRANGLE."



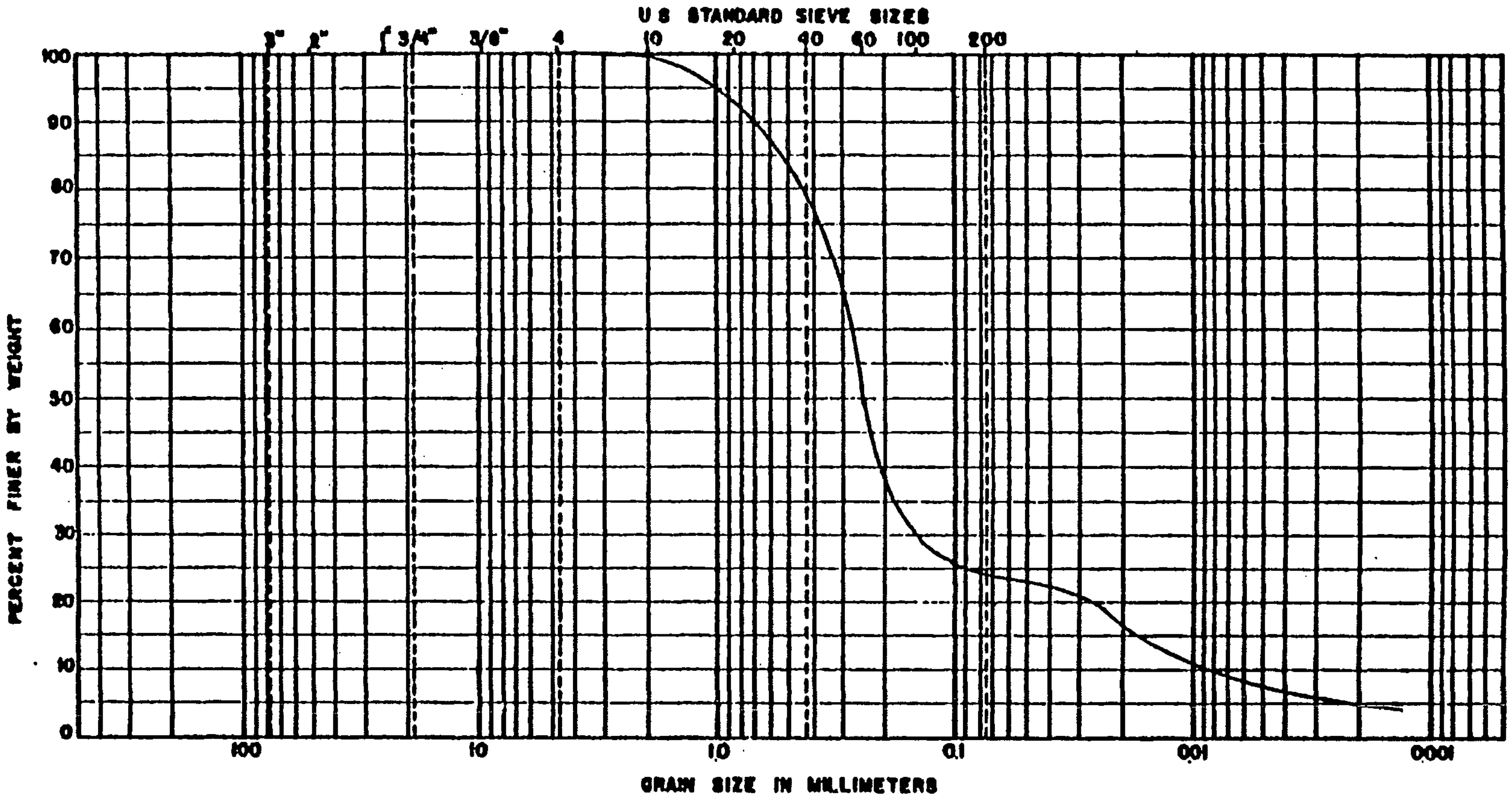
BORING NO	ELEV OR DEPTH	NAT WC	L L	P L	P I	DESCRIPTION OR CLASSIFICATION
DE-1	50 Ft.					Cu = 9.3 SM Cc = 4.1

GRAIN SIZE DISTRIBUTION

JOB NO. 88-692

LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-5A



BOULDERS	COBBLES	GRAVEL		SAND			FINES	
		COARSE	FINE	COARSE	MEDIUM	FINE	SILT SIZES	CLAY SIZES

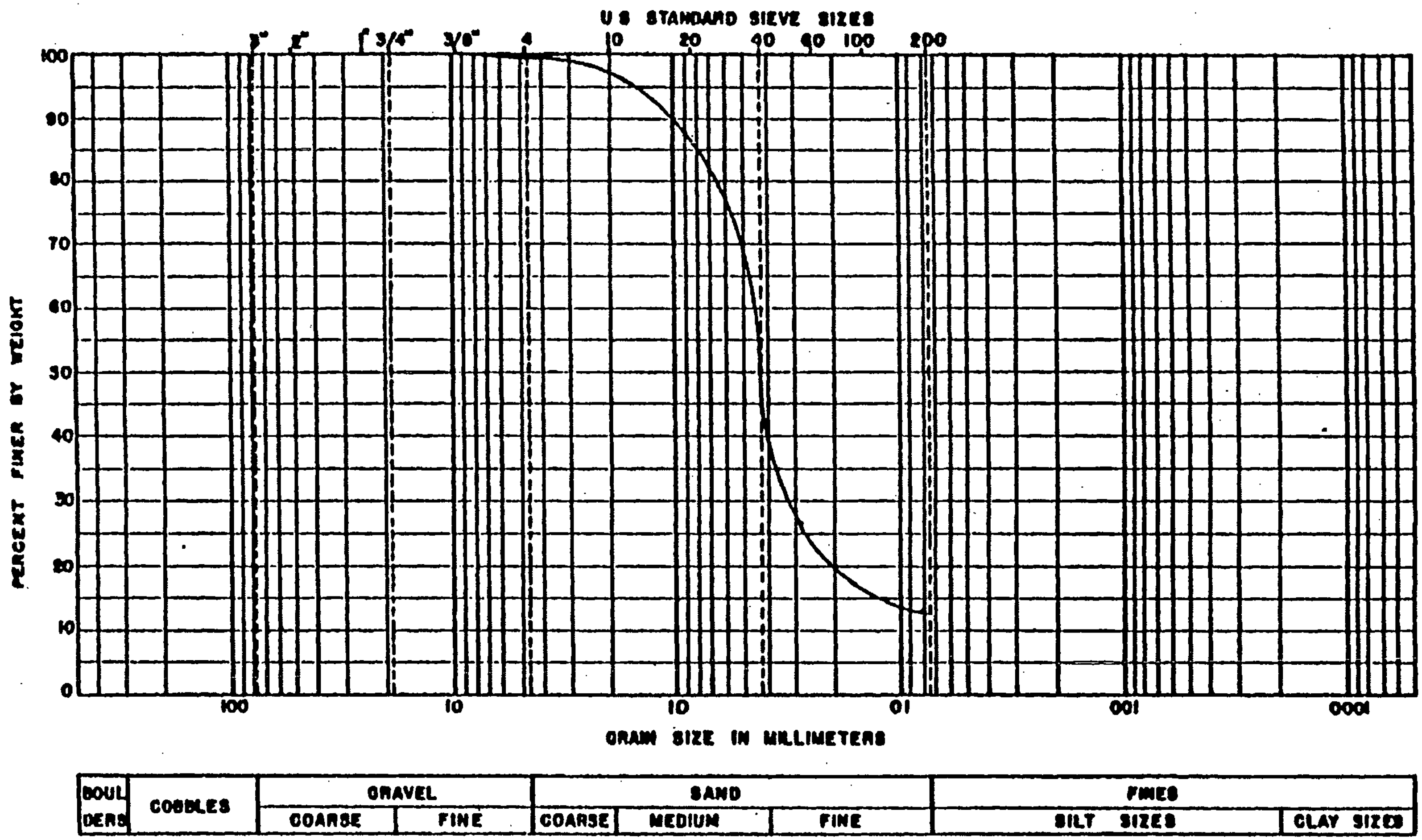
BORING NO	ELEV OR DEPTH	NAT WC	L L	P L	P I	DESCRIPTION OR CLASSIFICATION
DE-3	5 Ft.	11.5	15	NP	-	Cu = 37.9 Cc = 0.8 SM

GRAIN SIZE DISTRIBUTION

JOB NO. 88-692

LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-5B



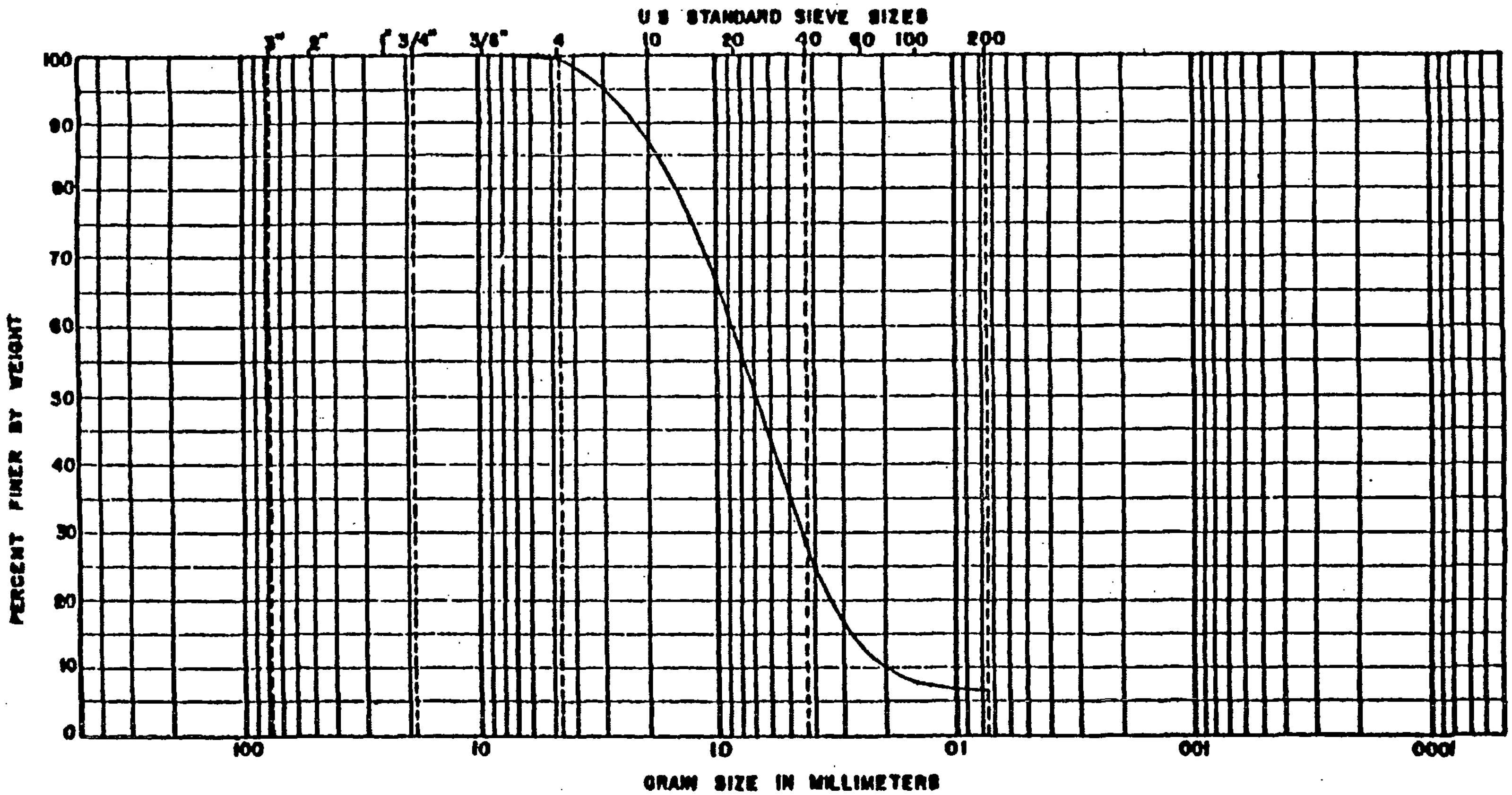
BORING NO	ELEV OR DEPTH	NAT WC	L L	PL	P I	DESCRIPTION OR CLASSIFICATION
DE-4	10 Ft.					Cu = 16.1 Cc = 8.1 SM

GRAIN SIZE DISTRIBUTION

JOB NO. 88-692

LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-5C



BOULDERS	COBBLES	GRAVEL		SAND			FINES	
		COARSE	FINE	COARSE	MEDIUM	FINE	SILT SIZES	CLAY SIZES

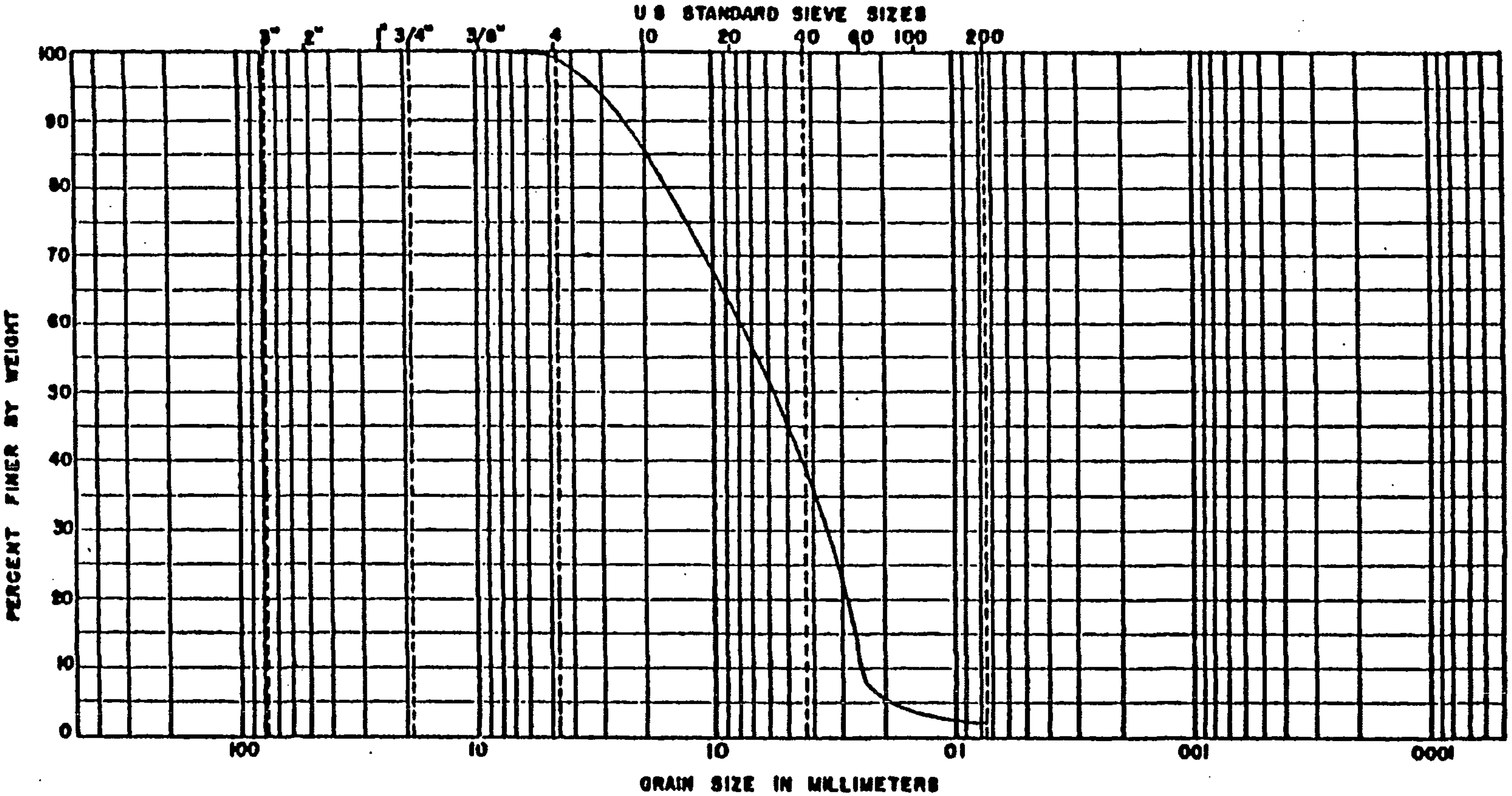
BORING NO	ELEV OR DEPTH	NAT WC	L.L	PL	PI	DESCRIPTION OR CLASSIFICATION
DE-4	15 Ft.					Cu = 4.4 Cc = 1.2 SP-SM

GRAIN SIZE DISTRIBUTION

JOB NO. B8-692

LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-5D



BOULDERS	COBBLES	GRAVEL		SAND			FINES	
		COARSE	FINE	COARSE	MEDIUM	FINE	SILT SIZES	CLAY SIZES

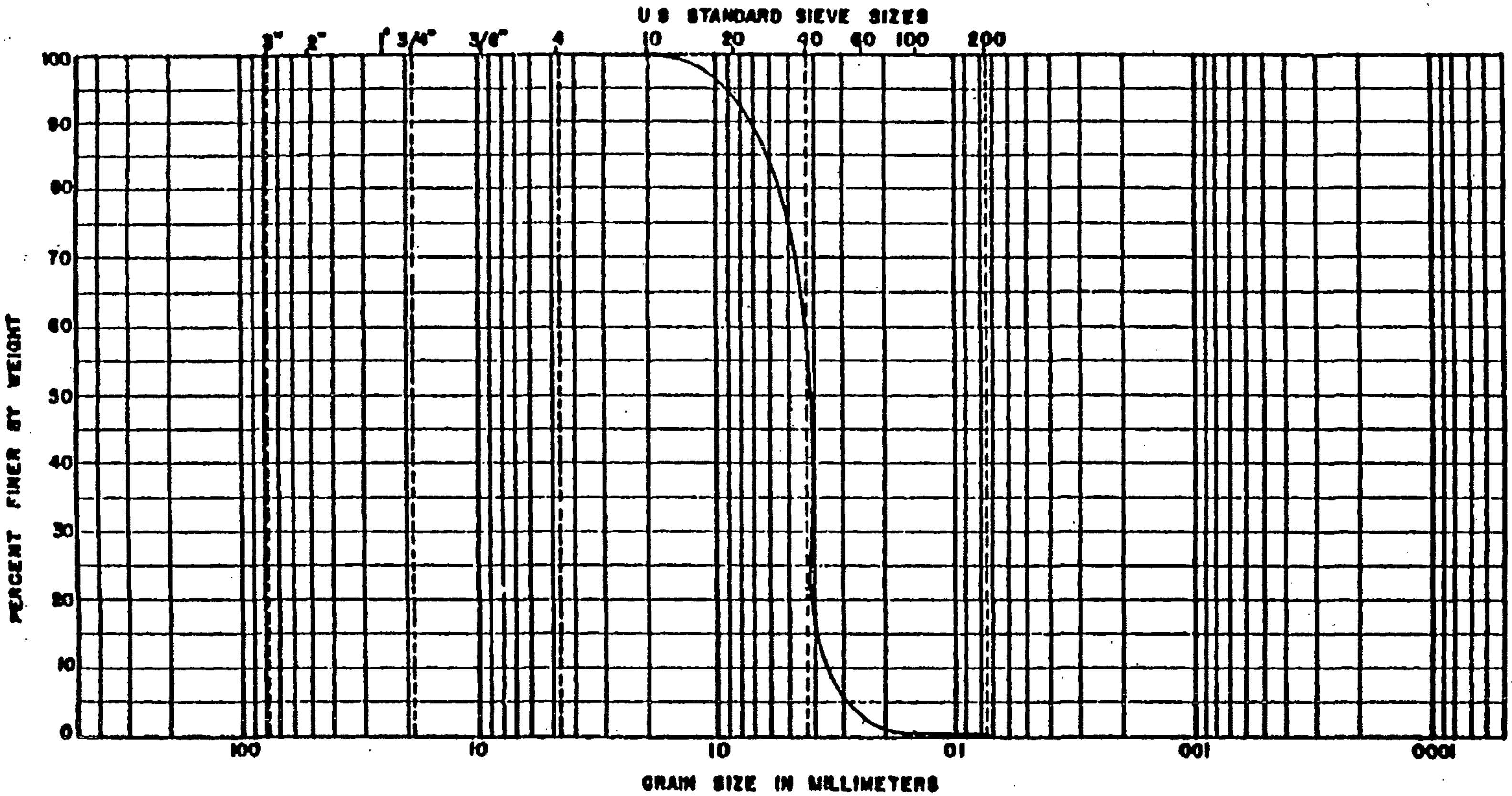
BORING NO	ELEV OR DEPTH	NAT WC	LL	PL	PI	DESCRIPTION OR CLASSIFICATION
DE-4	20 Ft.					Cu = 3.0 Cc = 0.6 SP

GRAIN SIZE DISTRIBUTION

JOB NO. 88-692

LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-5E



BOULDERS	COBBLES	GRAVEL		SAND			FINES	
		COARSE	FINE	COARSE	MEDIUM	FINE	SILT SIZES	CLAY SIZES

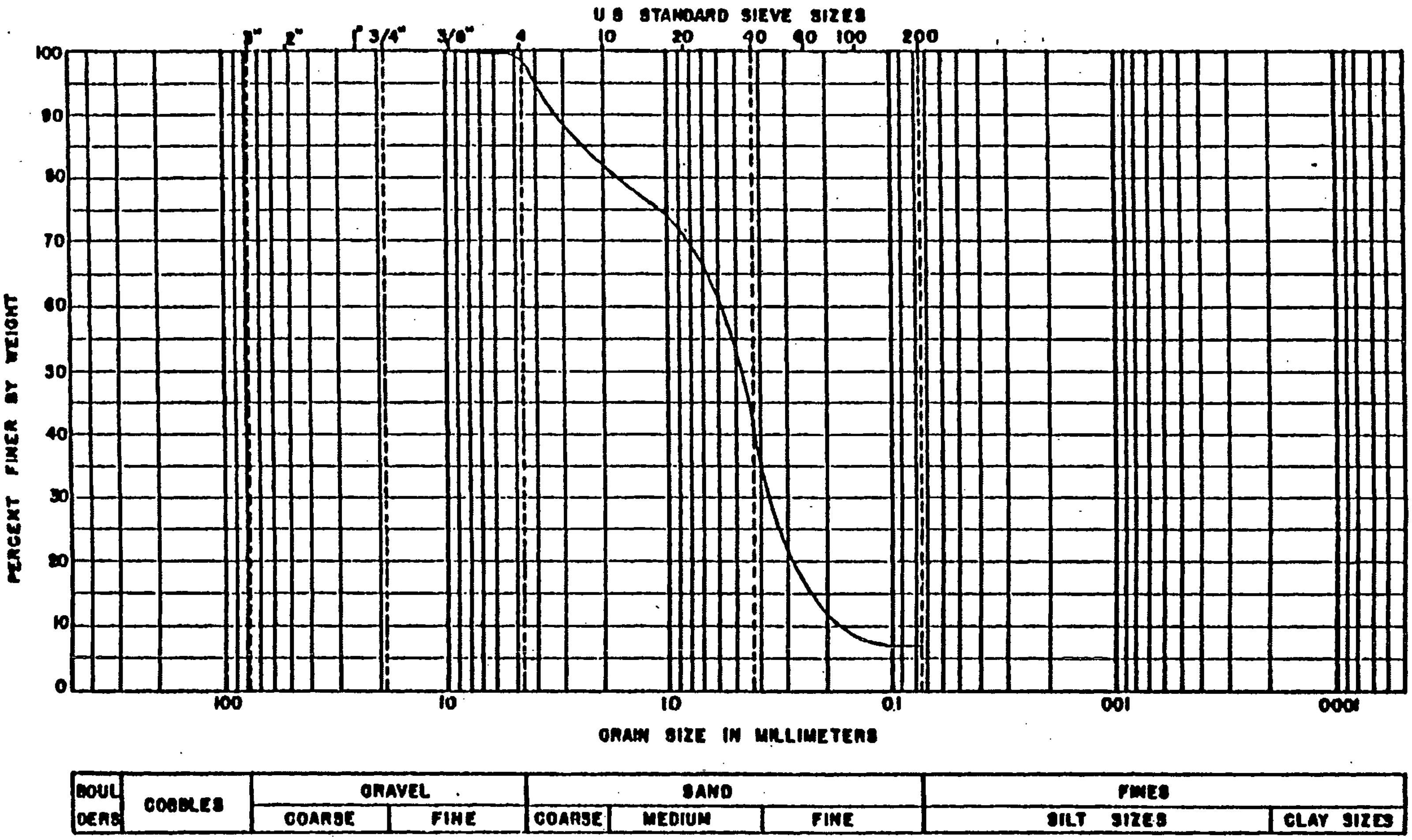
BORING NO	ELEV OR DEPTH	NAT WC	L L	PL	P I	DESCRIPTION OR CLASSIFICATION
DE-4	25 Ft.					Cu = 1.2 Cc = 1.1 SP

GRAIN SIZE DISTRIBUTION

JOB NO. 88-692

LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-5F



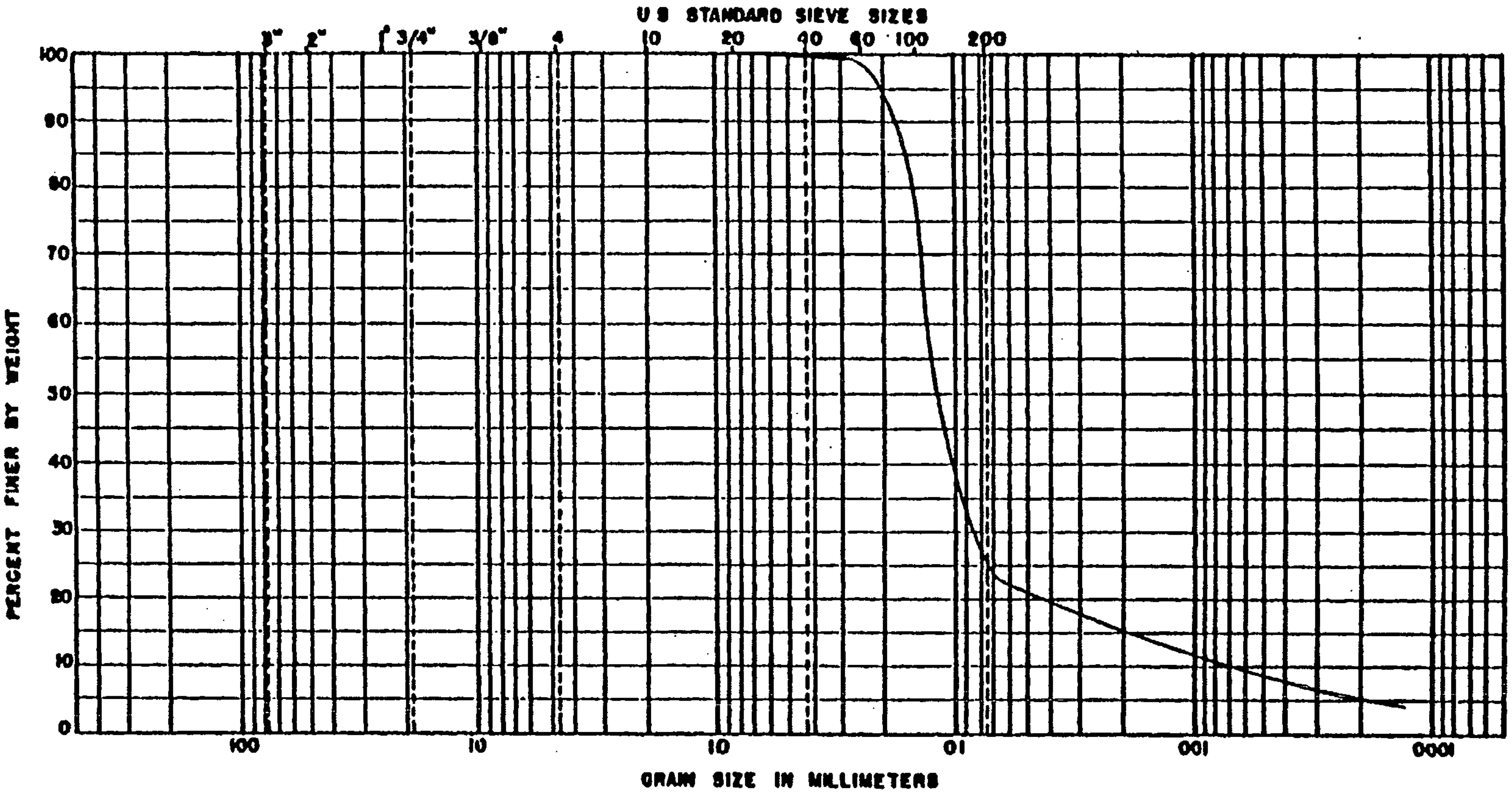
BORING NO	ELEV OR DEPTH	NAT WC	L L	P L	P I	DESCRIPTION OR CLASSIFICATION
DE-4	30 Ft.					Cu = 3.5 Cc = 1.3 SP-SM

GRAIN SIZE DISTRIBUTION

JOB NO. B8-692

LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-5G



BOULDERS	COBBLES	GRAVEL		SAND			FINES	
		COARSE	FINE	COARSE	MEDIUM	FINE	SILT SIZES	CLAY SIZES

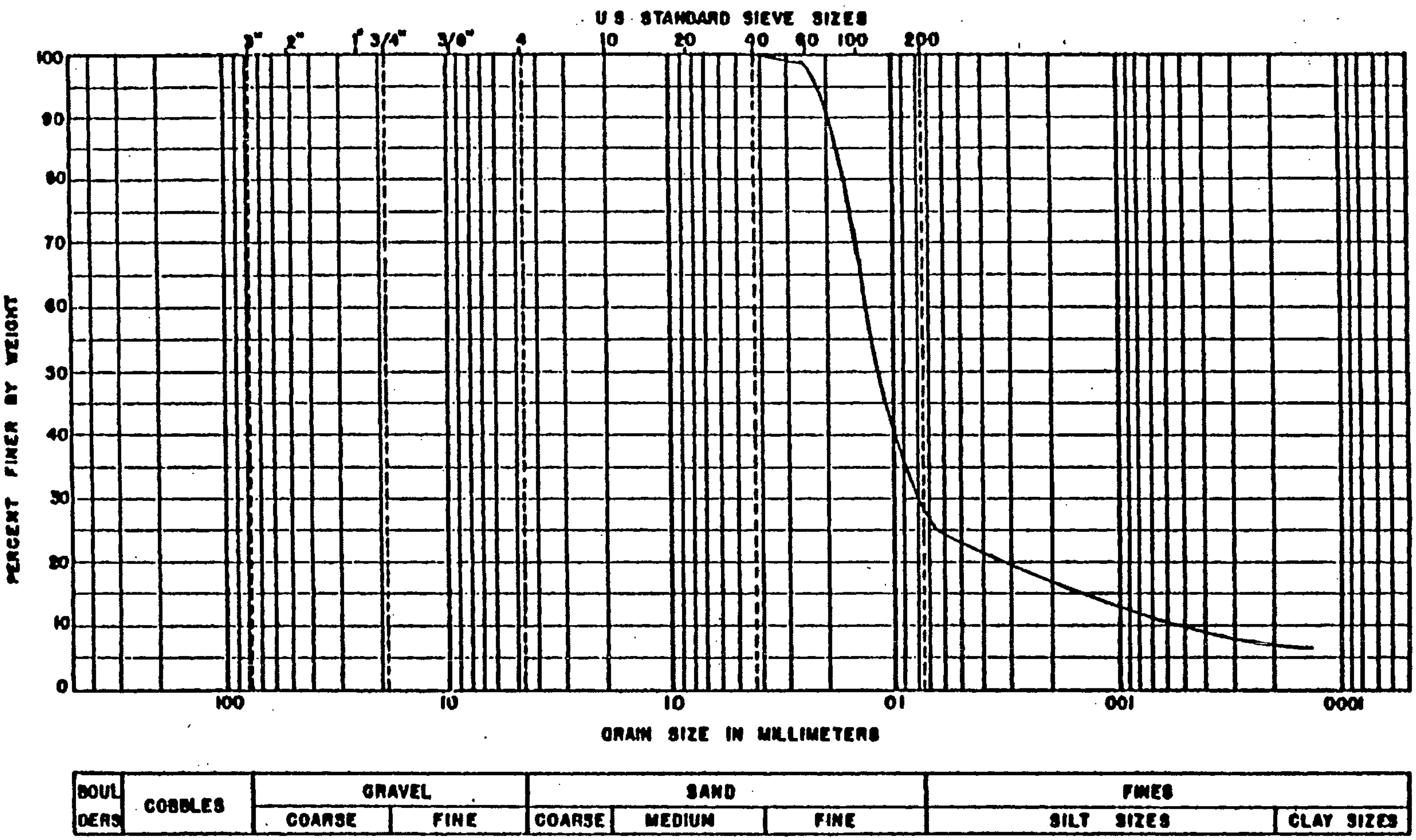
BORING NO	ELEV OR DEPTH	NAT WC	L L	P L	P I	DESCRIPTION OR CLASSIFICATION
DE-7	40 Ft.					Cu = 21.3 Cc = 8.8 SM

GRAIN SIZE DISTRIBUTION

JOB NO. 88-692

LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-5H



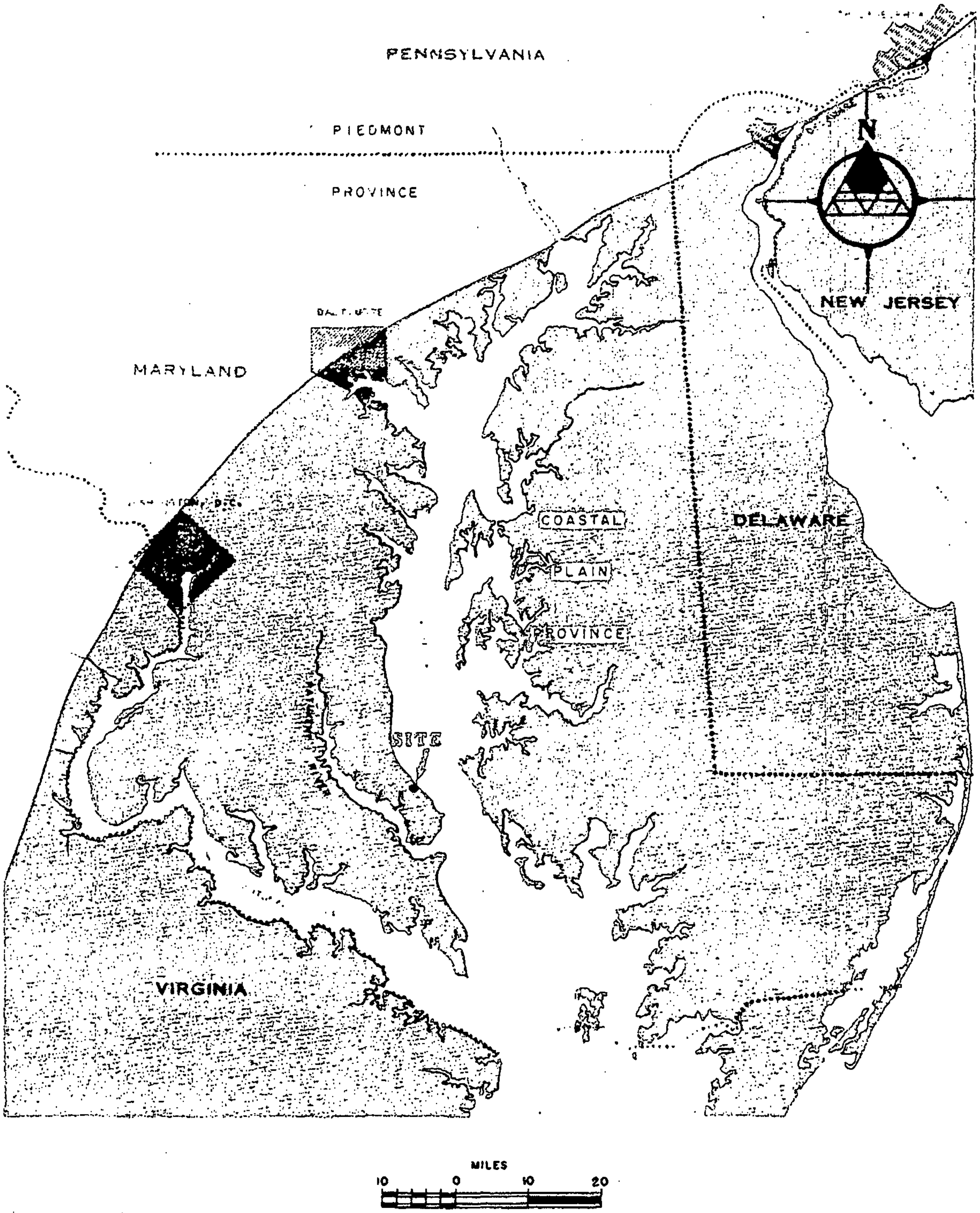
BORING NO	ELEV OR DEPTH	NAT WC	L L	P L	P I	DESCRIPTION OR CLASSIFICATION
DE-9	39 Ft.					Cu = 27.8 Cc = 11.2 SM

GRAIN SIZE DISTRIBUTION

JOB NO. 88-692

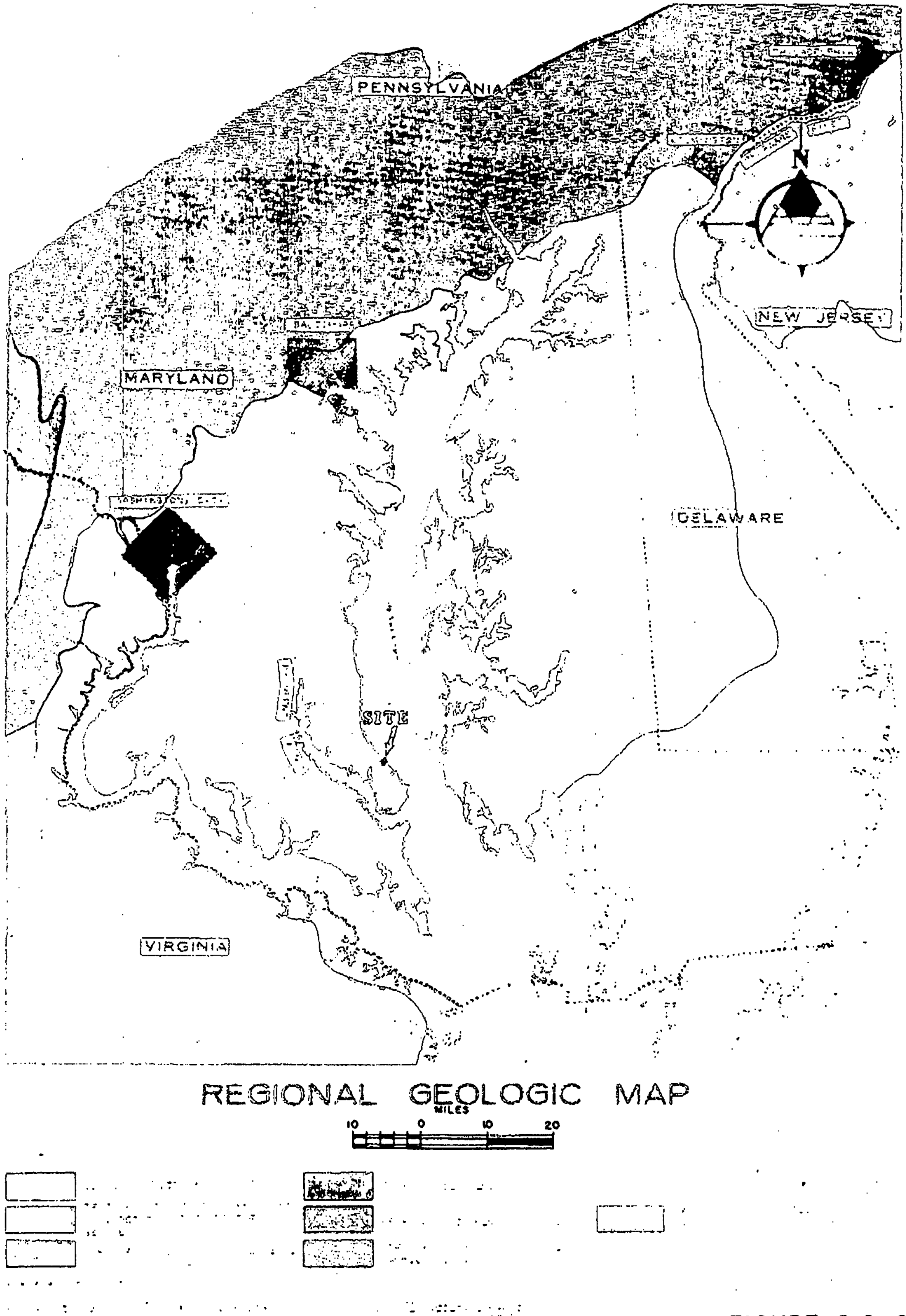
LAW ENGINEERING TESTING COMPANY

FIGURE 2.5-51



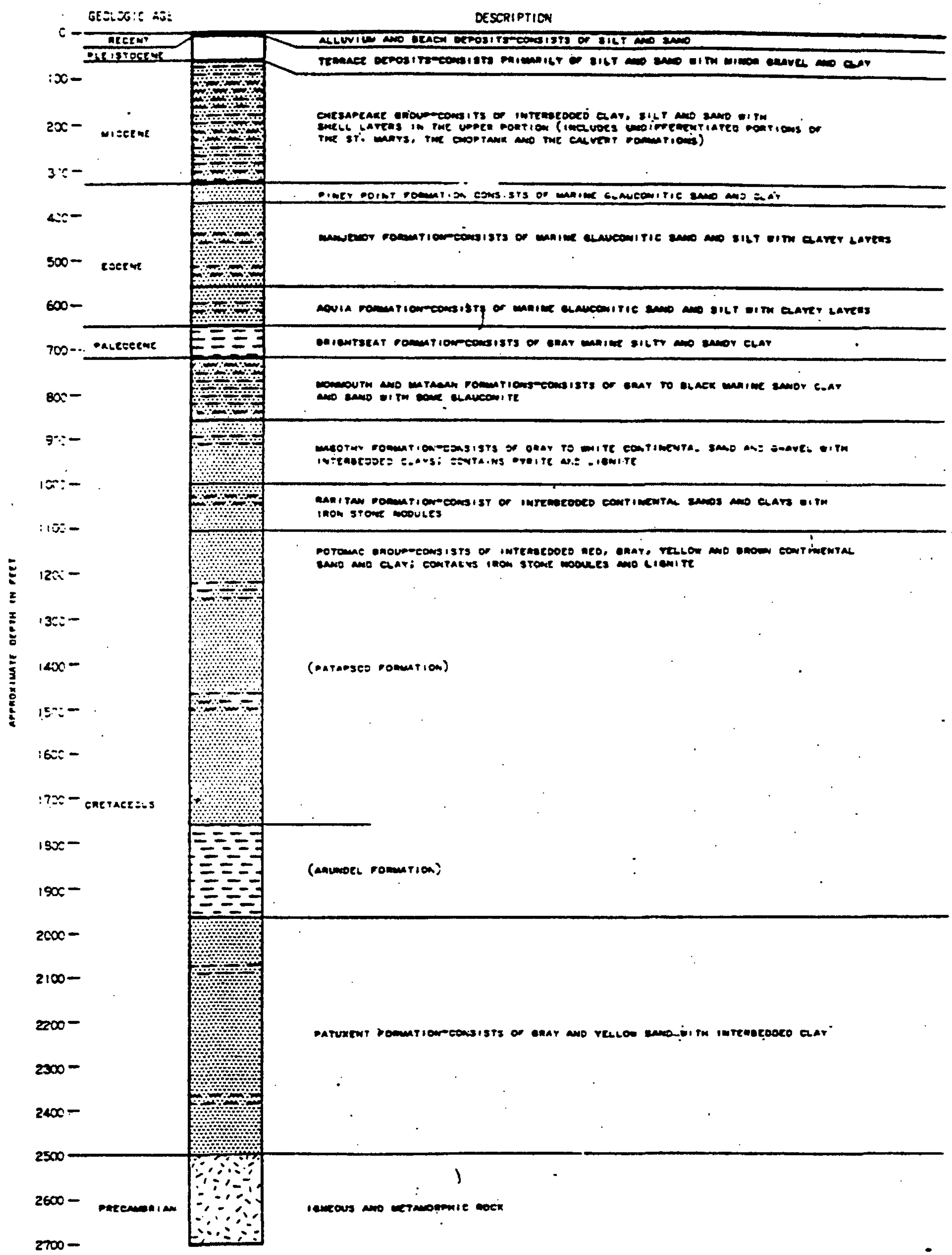
REGIONAL PHYSIOGRAPHIC MAP

FIGURE 2.6-1



REGIONAL GEOLOGIC MAP

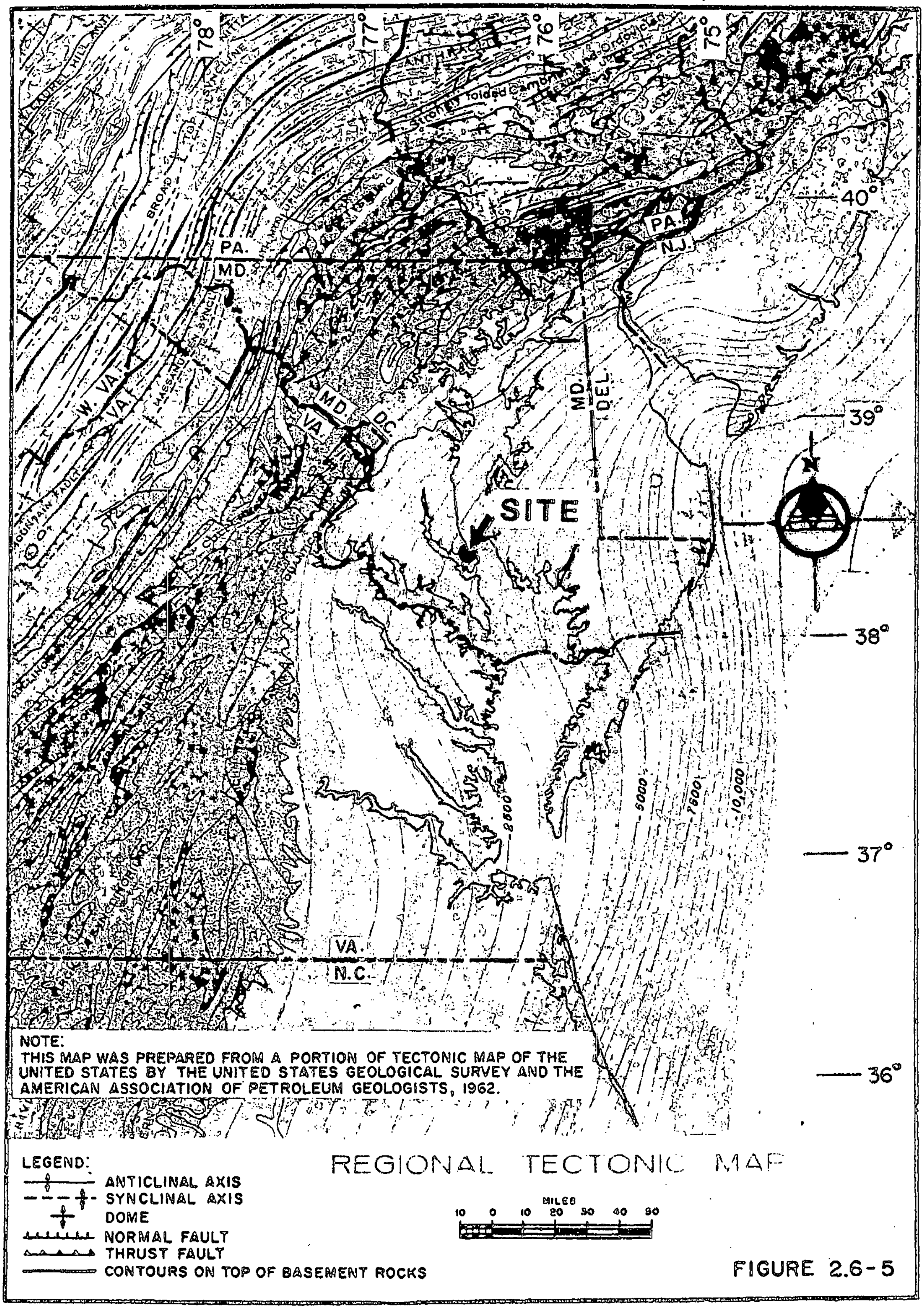
FIGURE 2.6-2



REFERENCE: GROUND-WATER RESOURCES OF THE SOUTHERN MARYLAND COASTAL PLAIN, MD. GEOLOGICAL SURVEY BULLETIN 15, 1955.

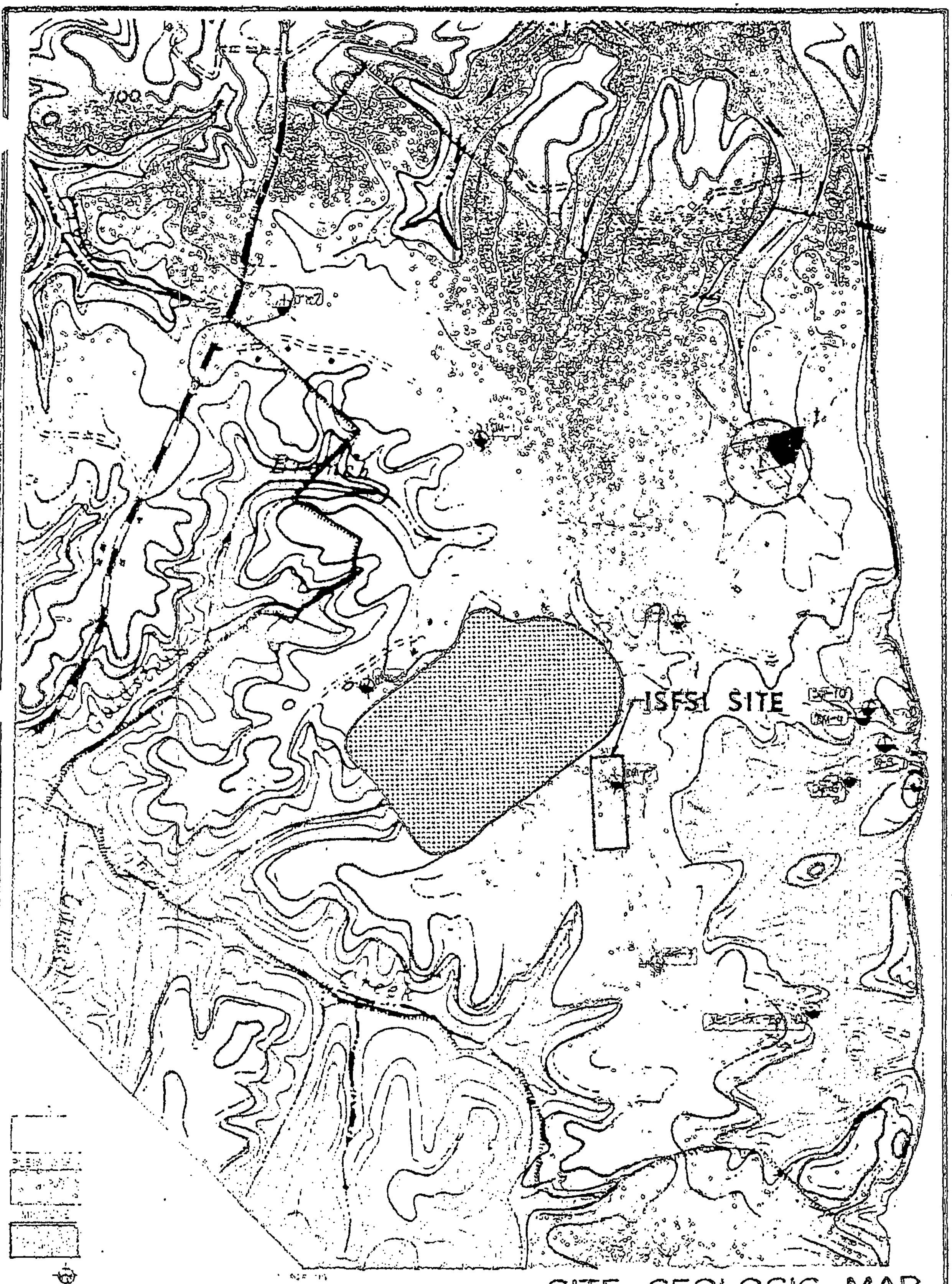
GEOLOGIC COLUMNAR SECTION - SITE AREA

FIGURE 2.6-4



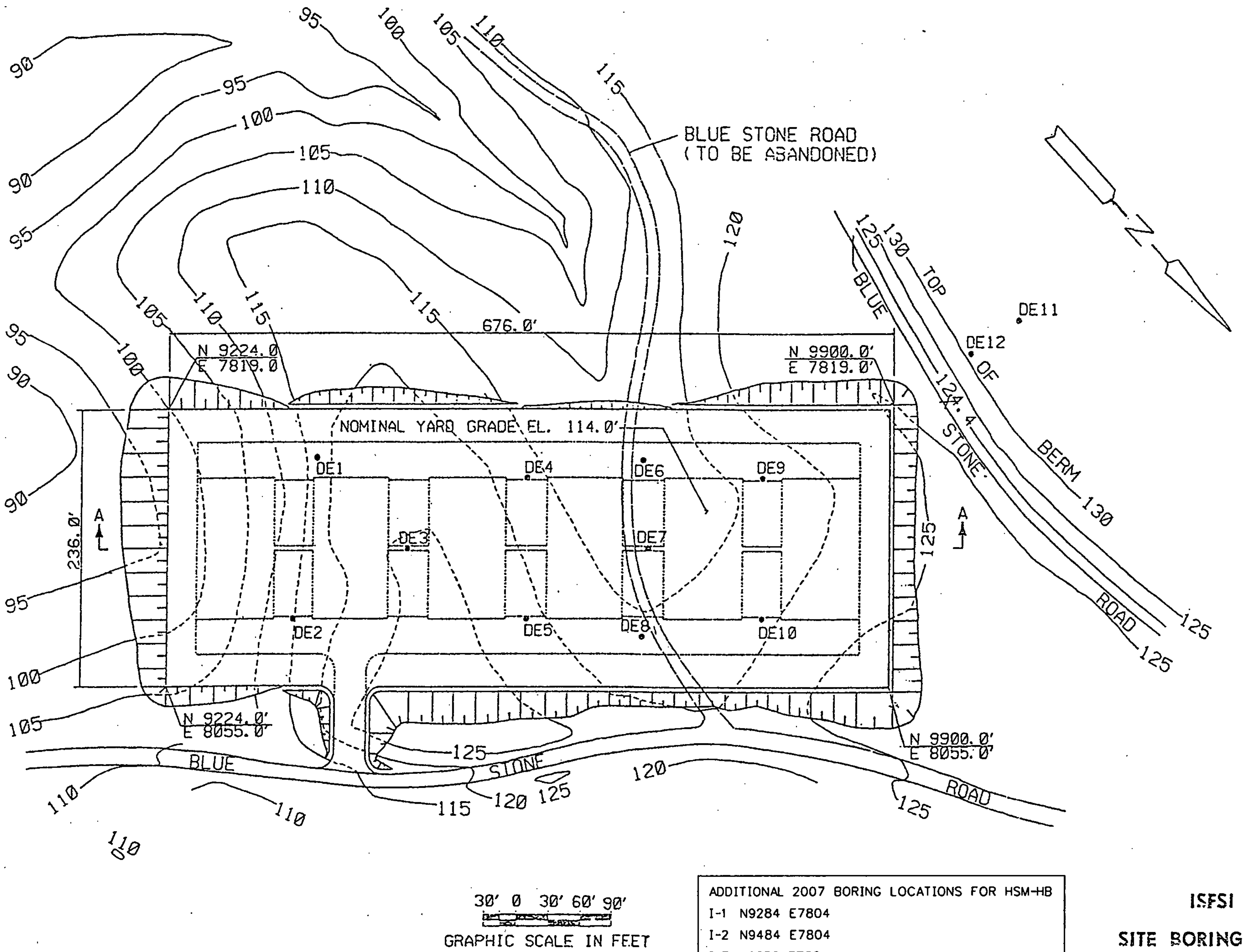
REGIONAL TECTONIC MAP

FIGURE 2.6-5



SITE GEOLOGIC MAP

FIGURE 2.6-6

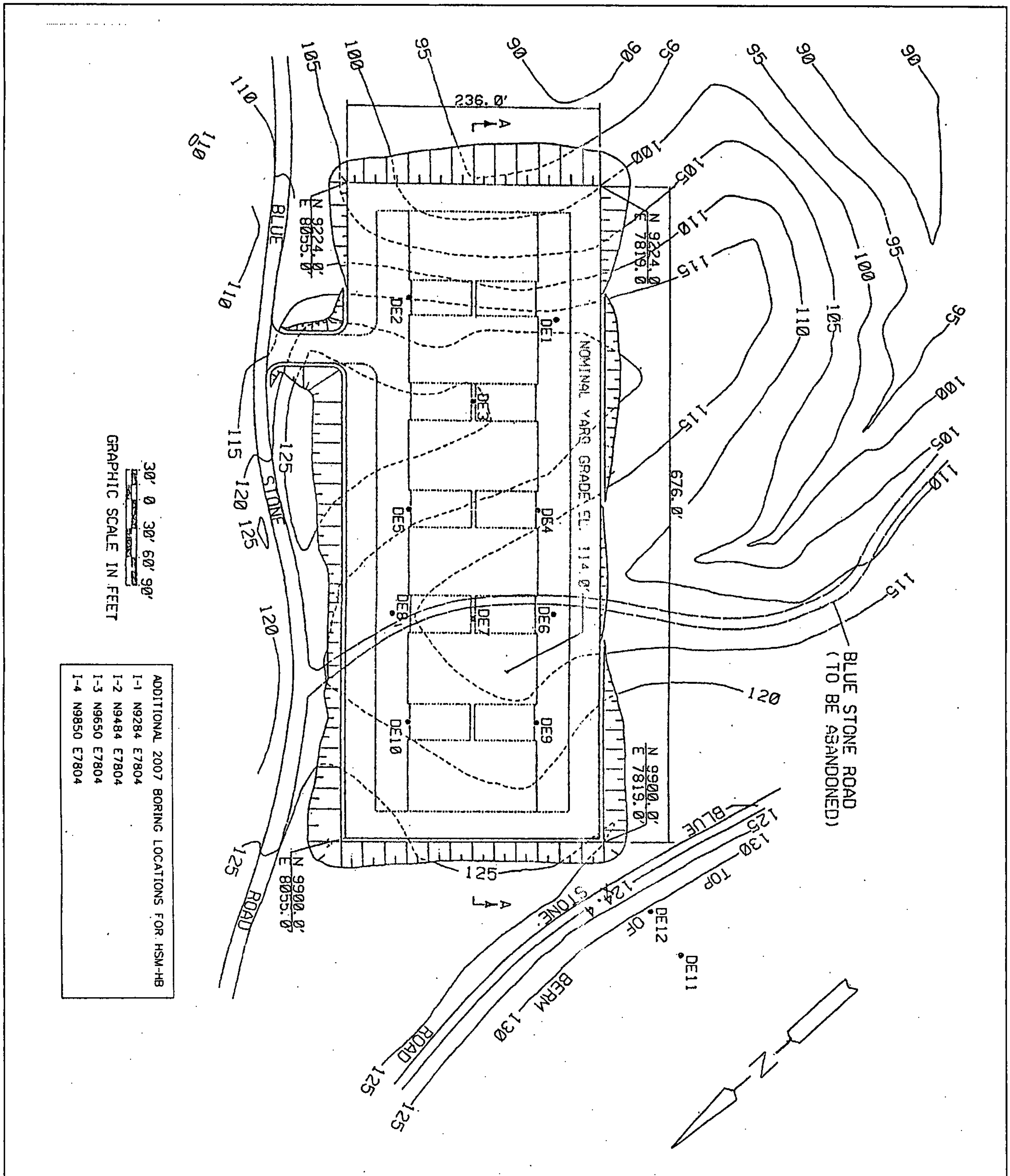


ADDITIONAL 2007 BORING LOCATIONS FOR HSM-HB
I-1 N9284 E7804
I-2 N9484 E7804
I-3 N9650 E7804
I-4 N9850 E7804

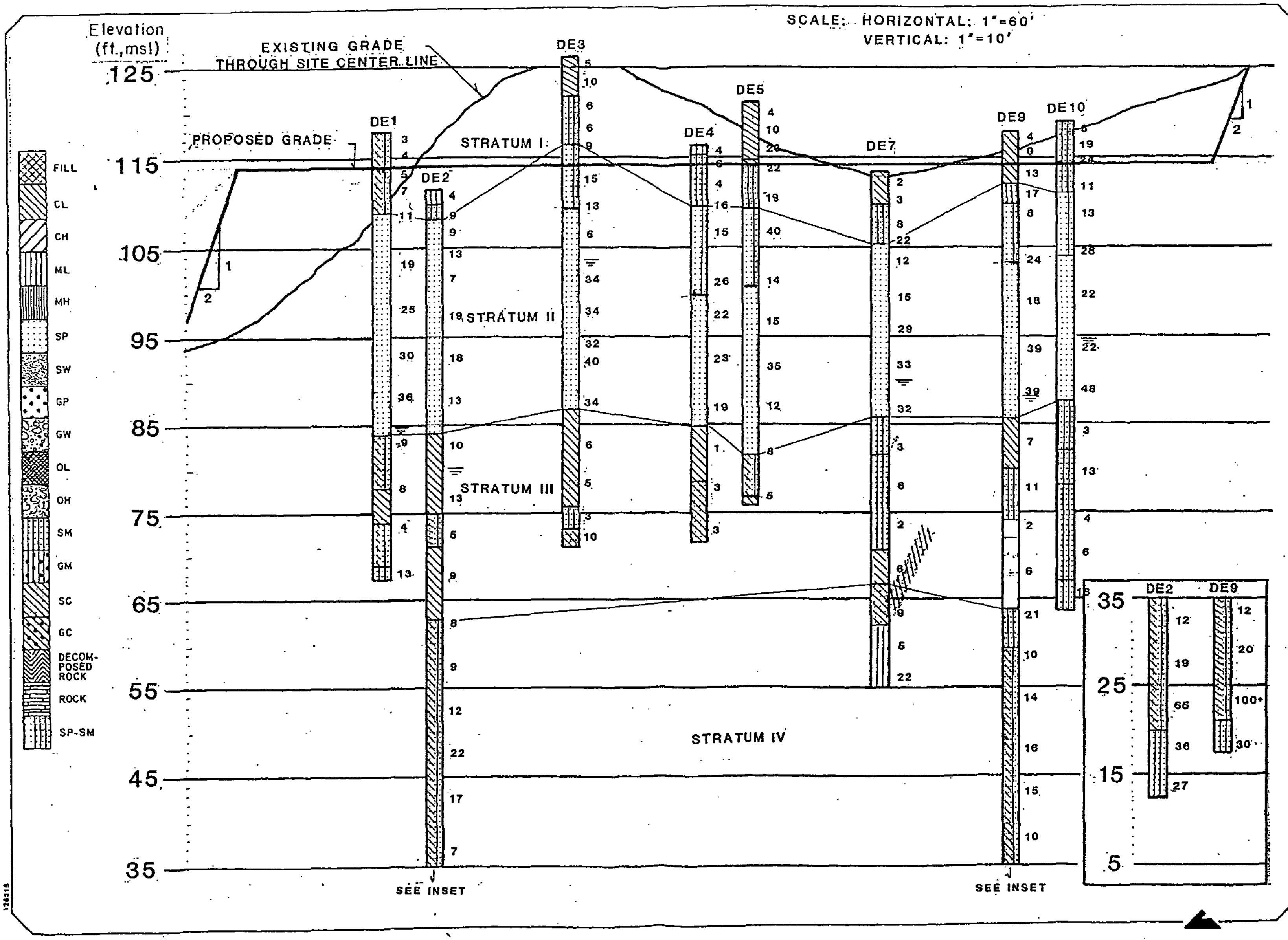
ISFSI

SITE BORING PLAN

FIGURE 2.6-7



Calvert Cliffs Independent Spent Fuel Storage Installation	ISFSI SITE BORING PLAN	USAR Figure 2.6-7 Rev. 22



CALVERT CLIFFS NUCLEAR POWER PLANT
ISFSF
CALVERT COUNTY, MARYLAND

ESTIMATED SUBSURFACE PROFILE A-A'

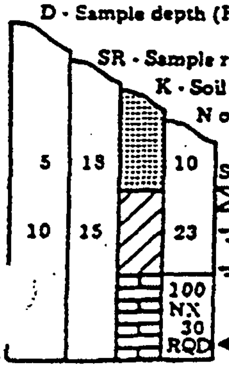
Job No. 88-692
Drawn: JHC
Checked: DDF
Date: 3-6-89

3

Figure 2.6-7A

LAW ENGINEERING TEST BORING RECORD

ELEV. 118.0'	STRATUM DEPTH 0.0'	VISUAL SOIL DESCRIPTION	D	SR	N or CR/RQD	REMARKS
	0.2	TOPSOIL				
		Very loose to loose brown clayey and silty fine to medium SAND (SC-SM), damp	1.0	18	3	
			2.5	14	4	
113			5.0	14	5	
			7.5	14	7	
	9.0	Firm to dense brown-tan fine to medium SAND (SP), moist				
108			10.0	12	11	
103			15.0	12	19	
98			20.0	12	25	
93		Some coarse sand below 25 feet	25.0	12	30	
88			30.0	10	36	
						@ 33', 2-13-89
	34.0	Loose light gray to tan clayey and silty fine SAND (SC-SM), moist				
83			35.0	10	9	



D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
Loss of Water

Page 1 of 2

BORING NUMBER DE-1

DATE DRILLED 1-31-89

JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7B (1 of 2)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
83'	35.0						
78	40.0	Firm light gray silty CLAY (CL), moist	40.0	18		8	
73	44.0	Very loose dark gray silty and clayey fine SAND (SC-SM), moist	45.0	18		4	C @ 44', 2-13-89
68	49.0	Firm tan silty fine SAND (SM), wet	50.0	18		13	
	50.5	Boring Terminated @ 50.5 Feet 2-1-89					

Page 2 of 2

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

5	18		10
10	15		23
			100 NX 30 RQD

SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
◄ Loss of Water

BORING NUMBER DE-1
DATE DRILLED 1-31-89
JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7B (2 of 2)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
111.9'	0.0'						
	0.2	TOPSOIL					
		Soft brown SILT (ML), little to some clay, moist	1.0	15		4	
	2.0	Loose brown silty fine to medium SAND (SM), damp	2.5	16		9	
	4.0	Loose to firm light brown to tan fine to medium SAND (SP), moist to wet					
106.9			5.0	18		9	
			7.5	18		13	
101.9			10.0	16		7	
96.9			15.0	12		19	
91.9		Some coarse sand below 20 feet	20.0	12		18	
86.9			25.0	14		13	
	28.0	Firm to stiff light gray silty CLAY (CL), moist					
81.9			30.0	18		10	
							@ 32', 2-13-89
76.9			35.0	18		13	

Page 1 of 3

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

D	SR	K	N or CR/RQD
5	15		10
10	15		23
			100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

SYMBOLS:
- Undisturbed Sample
- Water level, time of drilling
- Water level
- C Caved depth of boring
- Loss of Water

BORING NUMBER DE-2
DATE DRILLED 2-8-89
JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7C (1 of 3)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
76.9'	35.0'						
	37.0	Loose dark gray clayey and silty fine SAND (SC-SM), moist					
71.9			40.0	18		5	
	41.0	Stiff light gray silty CLAY (CL), moist					
		Occasional sand seams					
66.9			45.0	18		9	
	49.0	Loose to firm dark green clayey and silty fine SAND (SC-SM), moist					
61.9			50.0	18		8	
56.9			55.0	18		9	
51.9			60.0	18		12	C @ 60', 2-13-89
46.9			65.0	18		22	
41.9			70.0	18		17	

Page 2 of 3

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

5	18		10
10	15		23
			100 NX 30 RQD

SYMBOLS:
- Undisturbed Sample
- Water level, time of drilling
- Water level
- C Caved depth of boring
- Loss of Water

BORING NUMBER DE-2

DATE DRILLED 2-8-89

JOB NUMBER 88-592

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7C (2 of 3)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
41.9'	70.0'						
36.9			75.0	18		7	
31.9			80.0	18		12	
26.9			85.0	18		19	
21.9		Trace shells below 90 feet	90.0	18		65	
	92.0	Dense green silty fine SAND (SM), with shells, moist to wet					
16.9			95.0	18		36	
11.9	100.5		100.0	14		27	
		Boring Terminated @ 100.5 feet 2-9-89					

D - Sample depth (Ft.)

SR - Sample recovery (In.)

K - Soil Symbols

N or CR/RQD where:

D	SR	K	N or CR/RQD
5	18		10
10	15		23
			100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)

CR - % core recovery; NX or BX designates bit size (ASTM D-2113)

RQD - Rock Quality Designation

SYMBOLS:
- Undisturbed Sample
- Water level, time of drilling
- Water level
- C Caved depth of boring
- Loss of Water

BORING NUMBER DE-2

DATE DRILLED 2-8-89

JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7C (3 of 3)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
126.5'	0.0'						
	0.2	TOPSOIL					
		Firm to stiff brown fine sandy CLAY (CL), trace silt, damp	1.0	12		5	
			2.5	12		10	
	3.5	Loose to firm brown silty fine to medium SAND (SM), trace clay, damp					
121.5			5.0	18		6	
			7.5	18		6	
116.5	10.0	Firm light brown to tan silty fine to medium SAND (SP-SM), damp	10.0	16		9	10.5 – 12.5
			13.5	18		15	14.0 – 16.0
111.5							
			16.5	14		13	
	17.0	Loose to dense light brown to tan fine to medium SAND (SP), moist to wet					
106.5			20.5	10		6	
							@ 23', 2-13-89
101.5		More coarse sand between 25 feet and 35 feet	26.0	8		34	26.5 – 28.5
96.5			29.5	18		34	30.0 – 32.0
			33.0	10		32	
91.5			35.0	8		40	

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

5	18		10
10	15		23
			100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
Loss of Water

BORING NUMBER DE- 3
DATE DRILLED 2-1-89
JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

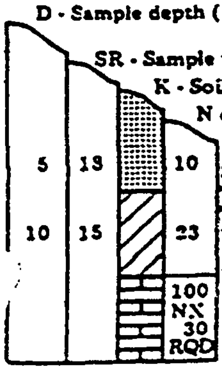
FIGURE 2.6-7D (1 of 2)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
91.5'	35.0'						
			40.0	8		34	
86.5	42.0	Firm light gray silty CLAY (CL), moist					
81.5			45.0	18		6	45.5 – 47.5 C @ 45', 2-13-89
			48.5	18		5	49.0 – 51.0
76.5							
	51.0	Very loose brown to tan silty fine to medium SAND (SM), wet	52.0	18		3	
	53.0	Stiff light gray to tan clayey SAND (SC), moist					
71.5	55.5	Boring Terminated @ 55.5 Feet 2-2-89	55.0	18		10	

Page 2 of 2

D - Sample depth (Ft.)



SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

SYMBOLS:
- Undisturbed Sample
- Water level, time of drilling
- Water level
- C Caved depth of boring
- ◀ Loss of Water

BORING NUMBER DE-3
DATE DRILLED 2-1-89
JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7D (2 of 2)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
116.7'	0.0'						
	0.2	TOPSOIL					
		Soft brown clayey SILT (ML), damp	1.0	14		4	
	1.5	Loose to very loose brown silty fine SAND (SM), damp	2.5	18		6	C @ 3', 2-13-89
111.7			5.0	18		4	
	7.0	Firm brown to tan silty fine to medium SAND (SP-SM), moist	7.5	18		16	
106.7			10.0	15		15	
101.7			15.0	10		26	
	17.0	Firm brown to tan fine to medium SAND (SP), moist to wet					
96.7		Some coarse sand below 20 feet	20.0	6		22	
91.7			25.0	12		23	
	29.0	Firm brown to tan silty fine to medium SAND (SP-SM), wet					
86.7			30.0	18		19	
	32.0	Very soft light gray silty CLAY (CL), with occasional sand seams, moist to wet					
81.7			35.0	18		1	

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

5 18 10
10 15 23
100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
Loss of Water

Page 1 of 2

BORING NUMBER DE-4
DATE DRILLED 1-31-89
JOB NUMBER B8-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7E (1 of 2)

LAW ENGINEERING TEST BORING RECORD

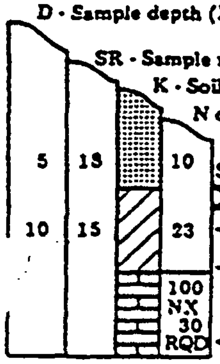
ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
81.7'	35.0'						
	38.0	Very loose light gray clayey fine SAND (SC), moist					
76.7			40.0	18		3	
71.7			45.0	18		3	
	45.5	Boring Terminated @ 45.5 Feet 1-31-89					

D - Sample depth (Ft.)

SR - Sample recovery (In.)

K - Soil Symbols

N or CR/RQD where:



SYMBOLS:
- Undisturbed Sample
- Water level, time of drilling
- Water level
- C Caved depth of boring
- Loss of Water

N - Penetration in blows per foot (ASTM D-1586)

CR - % core recovery; NX or BX designates bit size (ASTM D-2113)

RQD - Rock Quality Designation

Page 2 of 2

BORING NUMBER DE-4

DATE DRILLED 1-31-89

JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7E (2 of 2)

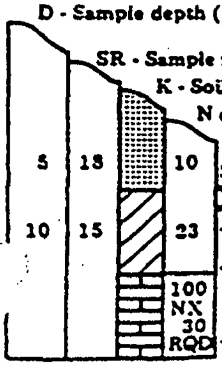
LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
121.5'	0.0'						
	0.2	TOPSOIL					
		Soft to very stiff brown sandy CLAY (CL), little silt, damp	1.0	10		4	
			2.5	18		10	
116.5			5.0	15		23	c @ 5.2', 2-13-89
	6.8	Firm brown-tan silty fine to medium SAND (SM), dry to damp	7.5	15		22	
111.5			10.0	12		19	
	12.0	Dense brown to tan silty fine to medium SAND (SP-SM), moist					
106.5			15.0	10		40	
101.5	20.0	Firm to dense brown to tan fine to medium SAND (SP), moist	20.0	6		14	
96.5		Some coarse sand below 25 feet	25.0	7		15	
91.5			30.0	10		35	
86.5			35.0	8		12	

Page 1 of 2

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation



SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
Loss of Water

BORING NUMBER DE- 5
DATE DRILLED 1-31-89
JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

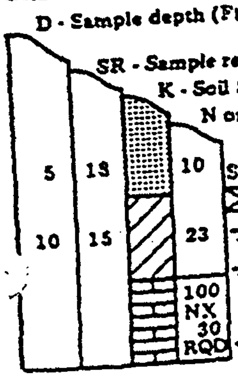
FIGURE 2.6-7F (1 of 2)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
86.5'	35.0'						
81.5	39.5	Loose dark gray silty and clayey fine SAND (SC-SM), moist	40.0	14		8	
	44.0	Loose light gray to tan clayey fine SAND (SC), moist	45.0	18		5	
76.5	45.5	Boring Terminated @ 45.5 Feet 1-31-89					

Page 2 of 2

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:



SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
◄ Loss of Water

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

BORING NUMBER DE-5
DATE DRILLED 1-31-89
JOB NUMBER 88-592

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7F (2 of 2)

AUGER PROBE BORING RECORD

ELEV. (ft.)	STRATUM DEPTH (ft.)	VISUAL SOIL DESCRIPTION	REMARKS
	0.0'		
	0.1	TOPSOIL Dark brown sandy CLAY	
	3.0	Dark brown silty CLAY with trace sand	
	8.0	Dark red brown silty SAND with trace clay	Cave @ 8.0', 2-1-89
	10.0	Auger Probe Terminated @ 10.0 Ft. 2-1-89	

NOTES: No Groundwater Encountered

Water level as noted

Water level at time of excavation

PROBE BORING NO. DE- 6

DATE EXCAVATED: 2-1-89

JOB NUMBER: 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7G

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
113.9'	0.0'						
	0.2	TOPSOIL					
		Very soft brown sandy CLAY (CL), some silt, damp	1.0	15		2	
			2.5	14		3	
	3.5	Firm to very stiff brown silty SAND (SM), dry					
108.9			5.0	14		8	
			7.5	18		22	
	8.5	Firm brown to tan fine to medium SAND (SP), moist					
103.9			10.0	14		12	
98.9			15.0	12		15	
							15.5
							17.5
			18.5	12		29	19.0
93.9		Wet below 20 feet					
							21.0
		Some coarse sand below 22 feet	22.0	6		33	
							24.0 @ 24.5', 2-6-89
88.9							26.0
			27.0	8		32	
	28.0	Very loose light gray to tan silty fine SAND (SM), little clay, wet					28.0
83.9							30.0
			31.0	18		3	
	32.0	Very loose tan silty fine SAND (SM), wet					32.0
							34.0 C @ 34.3', 2-6-89
78.9			35.0	18		6	

D - Sample depth (Ft.)

SR - Sample recovery (In.)

K - Soil Symbols

N or CR/RQD where:

D	SR	N or CR/RQD
5	18	10
10	15	23
		100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)

CR - % core recovery; NX or BX designates bit size (ASTM D-2113)

RQD - Rock Quality Designation

SYMBOLS:

Undisturbed Sample

Water level, time of drilling

Water level

C Caved depth of boring

Loss of Water

Page 1 of 2

BORING NUMBER DE-7

DATE DRILLED 2-2-89

JOB NUMBER B8-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

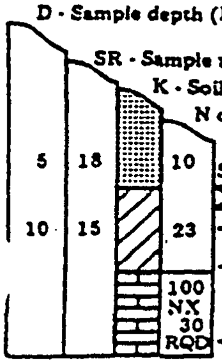
FIGURE 2.6-7H (1 of 2)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
78.9'	35.0'						
73.9			40.0	18		2	
	43.0	Firm light gray to tan fine sandy CLAY (CL), moist					
68.9			45.0	18		6	
	46.0	Loose dark gray clayey fine SAND (SC), moist					46.0 – 48.0
63.9			49.0	18		9	
	51.0	Firm to very stiff dark gray clayey and sandy SILT (ML), moist					
58.9			54.0	18		5	
			58.0	18		22	
	58.5	Boring Terminated @ 58.5 Feet 2-3-89					
53.9							

Page 2 of 2

D - Sample depth (Ft.)



SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
Loss of Water

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

BORING NUMBER DE-7
DATE DRILLED 2-2-89
JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7H (2 of 2)

AUGER PROBE BORING RECORD

ELEV. (ft.)	STRATUM DEPTH (ft.)	VISUAL SOIL DESCRIPTION	REMARKS
	0.0'		
	0.2	TOPSOIL	
		Dark brown sandy CLAY	
	3.5	Tan brown silty SAND	Cave @ 7.0', 2-1-89
	9.0	Fine to medium tan SAND and medium to coarse GRAVEL	
	10.0	Auger Probe Terminated @ 10.0 Ft 2-1-89	

NOTES: No Groundwater Encountered

Page 1 of 1

Water level as noted

Water level at time of excavation

PROBE BORING NO. DE-8

DATE EXCAVATED: 2-1-89

JOB NUMBER: 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-71

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
118.2'	0.0'						
	0.2	TOPSOIL					
		Soft to stiff brown silty CLAY (CL), damp	1.0	15		4	
			2.5	14		9	
113.2			5.0	16		13	
	6.0	Firm brown silty fine SAND (SM), damp	7.5	18		17	
	9.0	Loose to dense brown to tan silty fine to medium SAND (SP-SM), moist					
108.2			10.0	18		8	
103.2	15.0	Firm to dense brown to tan fine to medium SAND (SP), moist to wet	15.0	0		24	
98.2			20.0	12		18	
93.2			25.0	10		39	
		Wet below 28 feet					@ 28', 2-13-89
88.2			30.0	12		39	
	32.0	Firm light gray silty CLAY (CL), damp					
83.2			35.0	18		7	

Page 1 of 3

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

D	SR	K	N or CR/RQD
5	18		10
10	15		23
			100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

SYMBOLS:
- Undisturbed Sample
- Water level, time of drilling
- Water level
- C Caved depth of boring
- ◀ Loss of Water

BORING NUMBER DE-9
DATE DRILLED 2-6-89
JOB NUMBER B8-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7J (1 of 3)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
83.2'	35.0'						
	36.0	Firm tan silty fine SAND (SM), wet					36.0 38.0
			39.0	18		11	
78.2							
	44.0	Very loose light gray to tan clayey fine SAND (SC), moist					
73.2			45.0	18		2	
	46.0	Loose dark gray clayey fine SAND (SC), moist					
68.2			50.0	18		6	
	54.0	Firm tan silty fine SAND (SM), wet					C @ 54', 2-13-89
63.2			55.0	15		21	
	58.0	Stiff dark gray clayey and silty fine SAND (SC-SM), moist					
58.2			60.0	18		10	
							61.0 63.0
54.2			64.0	18		14	
48.2			70.0	18		16	

Page 2 of 3

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

D	SR	K	N or CR/RQD
5	18		10
10	15		23
			100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

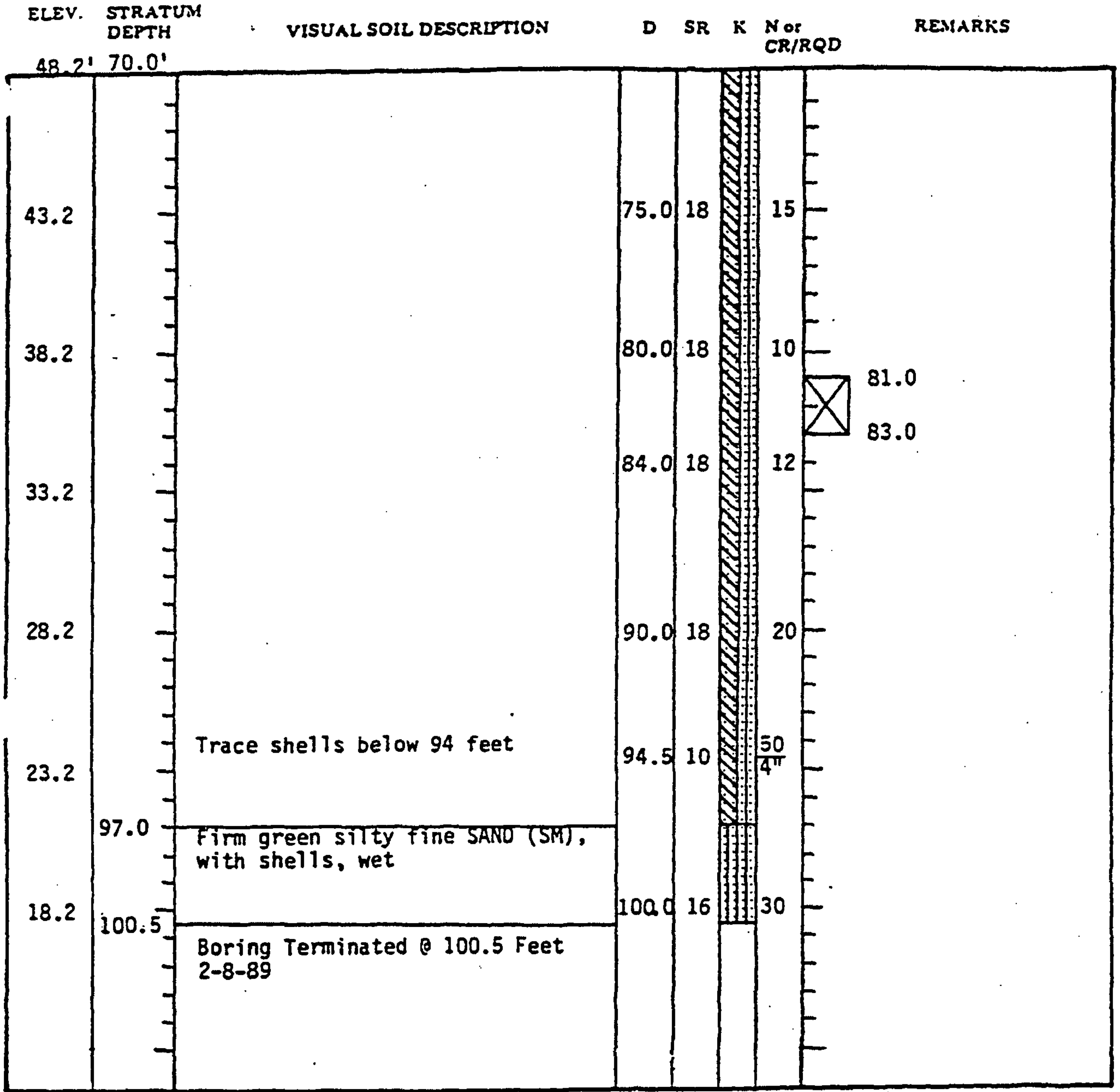
SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
◄ Loss of Water

BORING NUMBER DE- 9
DATE DRILLED 2-5-89
JOB NUMBER 88-592

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7J (2 of 3)

LAW ENGINEERING TEST BORING RECORD



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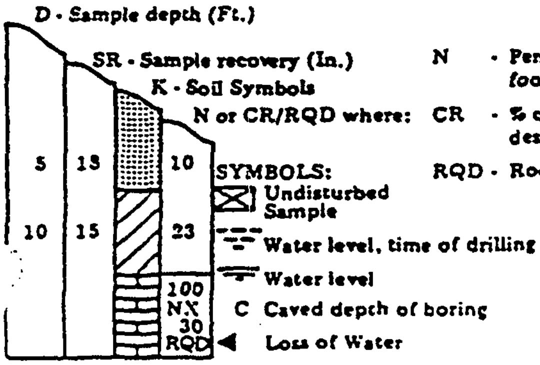


N - Penetration in blows per foot (ASTM D-1586)

CR - % core recovery; NX or BX designates bit size (ASTM D-2113)

RQD - Rock Quality Designation

BORING NUMBER DE- 9

DATE DRILLED 2-6-89

JOB NUMBER 88-592

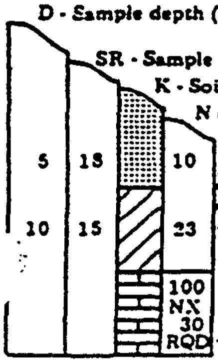
Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7J (3 of 3)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	K	N or CR/RQD	REMARKS
119.2'	0.0'						
	0.2	TOPSOIL					
		Loose to firm brown silty fine SAND (SM), trace clay, dry to damp	1.0	15		6	
			2.5	12		19	
114.2			5.0	16		24	
			7.5	16		11	
	8.0	Firm brown to tan silty fine to medium SAND (SP-SM), moist					
109.2			10.0	14		13	
104.2	15.0	Dense brown to tan fine to medium SAND (SP), damp to moist Wet below 16 feet, some coarse sand below 16 feet	15.0	18		28	
99.2			20.0	8		22	
							@ 24', 2-13-89
94.2			25.0	10		22	
89.2			30.0	10		48	
	32.0	Very loose light gray to tan clayey fine SAND (SC), wet					
84.2			35.0	12		3	

D - Sample depth (Ft.)



SYMBOLS:
- Undisturbed Sample
- Water level, time of drilling
- Water level
- C Caved depth of boring
- ◀ Loss of Water

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

BORING NUMBER DE-10
DATE DRILLED 2-6-89
JOB NUMBER B8-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7K (1 of 2)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	N or CR/RQD	REMARKS
84.2'	35.0'					
	37.0	Firm tan silty fine SAND (SM), moist				
79.2			40.0	10	13	
	41.0	Very loose to loose light gray to tan silty fine SAND (SM), little clay, wet				
74.2			45.0	18	4	
						46.0 – 48.0 (Undisturbed Sample); C @ 47', 2-13-89
69.2			50.0	18	6	
	52.0	Firm tan fine SAND (SM), moist				
64.2			55.0	18	16	
	55.5	Boring Terminated @ 55.5 Feet 2-6-89				

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

D	SR	N or CR/RQD
5	13	10
10	15	23
		100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
Loss of Water

BORING NUMBER DE-10
DATE DRILLED 2-6-89
JOB NUMBER B8-592

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7K (2 of 2)

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	N or CR/RQD	REMARKS
123.6'	0.0'					
	1.5	FILL - sampled as stiff brown fine sandy clay, trace organics, damp	1.0	15	10	
		FILL - sampled as brown to gray fine sand, damp	2.5	12	18	
118.6			5.0	18	8	
				24		6.0 – 8.0 (Undisturbed Sample)
			9.0	14	8	
113.6						Water level @ 11.5', 2-13-89
			12.0	14	5	
	14.0	FILL - sampled as very loose gray silty fine sand, moist				
108.6			15.0	14	1	
				18		16.0 – 18.0 (Undisturbed Sample)
	17.0	Firm brown fine to medium clayey and silty SAND (SC-SM), damp				
			19.0	16	12	
103.6	20.0	Firm tan silty fine to medium SAND (SM), damp				
			22.0	12	21	
98.6			25.0	10	21	
						C @ 27', 2-13-89
93.6			30.0	10	28	
	30.5	Boring Terminated @ 30.5 Feet 2-10-89				
88.6						

D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:

D	SR	K	N or CR/RQD
5	18		10
10	15		23
			100 NX 30 RQD

N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
Loss of Water

Page 1 of 1

BORING NUMBER DE-11
DATE DRILLED 2-10-89
JOB NUMBER 88-692

Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7L

LAW ENGINEERING TEST BORING RECORD

ELEV.	STRATUM DEPTH	VISUAL SOIL DESCRIPTION	D	SR	N or CR/RQD	REMARKS
130.9'	0.0'	FILL - sampled as loose brown silty sand, trace shells below 5 feet, some clay encountered at 8 feet, damp	1.0	14	5	
			2.5	14	9	
125.9			5.0	14	9	
			7.5	18	10	
120.9	9.0	FILL - sampled as stiff gray silt, trace shells, little organics, damp	10.0	15	7	
				0		11.0 – 13.0 (undisturbed sample); water level @ 12', 2-13-89
	12.0	Firm gray silty fine SAND (SM), trace shells, little organics, damp	14.0	18	13	
115.9	15.0	Firm brown clayey and silty SAND (SC-SM), damp		24		15.0 – 17.0 (undisturbed sample)
			18.0	18	21	
110.9						
105.9				24		24.0 – 26.0 (undisturbed sample); C @ 26', 2-13-89
	26.0	Firm to dense tan medium to coarse SAND (SP), damp	27.0	18	39	
100.9	30.5	Boring Terminated @ 30.5 Feet 2-10-89	30.0	12	29	
95.9						

Page 1 of 1

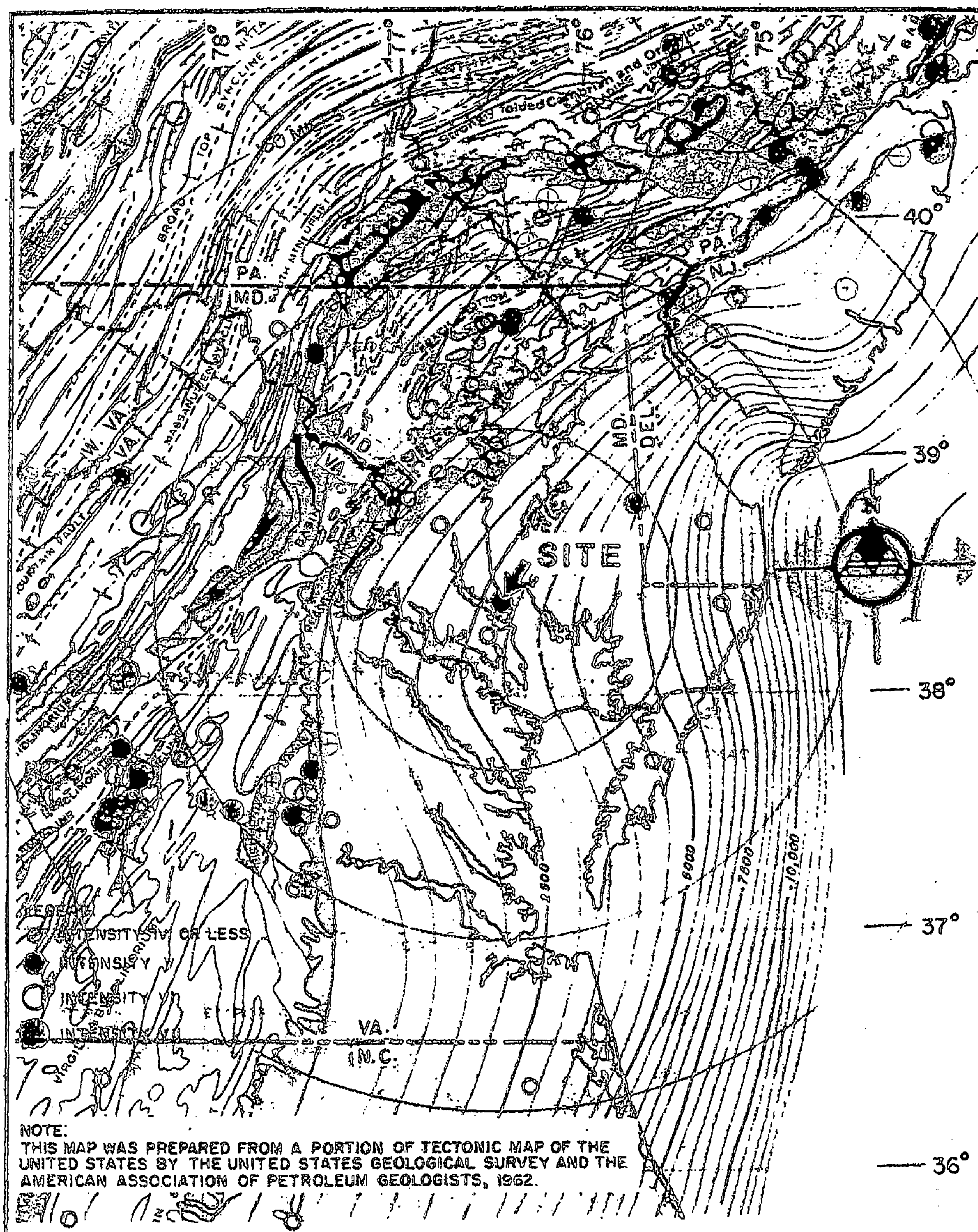
D - Sample depth (Ft.)
SR - Sample recovery (In.)
K - Soil Symbols
N or CR/RQD where:
N - Penetration in blows per foot (ASTM D-1586)
CR - % core recovery; NX or BX designates bit size (ASTM D-2113)
RQD - Rock Quality Designation

D	SR	K	N or CR/RQD
5	18		10
10	15		23
			100 NX 30 RQD

SYMBOLS:
Undisturbed Sample
Water level, time of drilling
Water level
C Caved depth of boring
Loss of Water

BORING NUMBER DE- 12
DATE DRILLED 2-10-89
JOB NUMBER 88-692
Calvert Cliffs Nuclear Power Plant
ISFSF
Calvert County, Maryland

FIGURE 2.6-7M



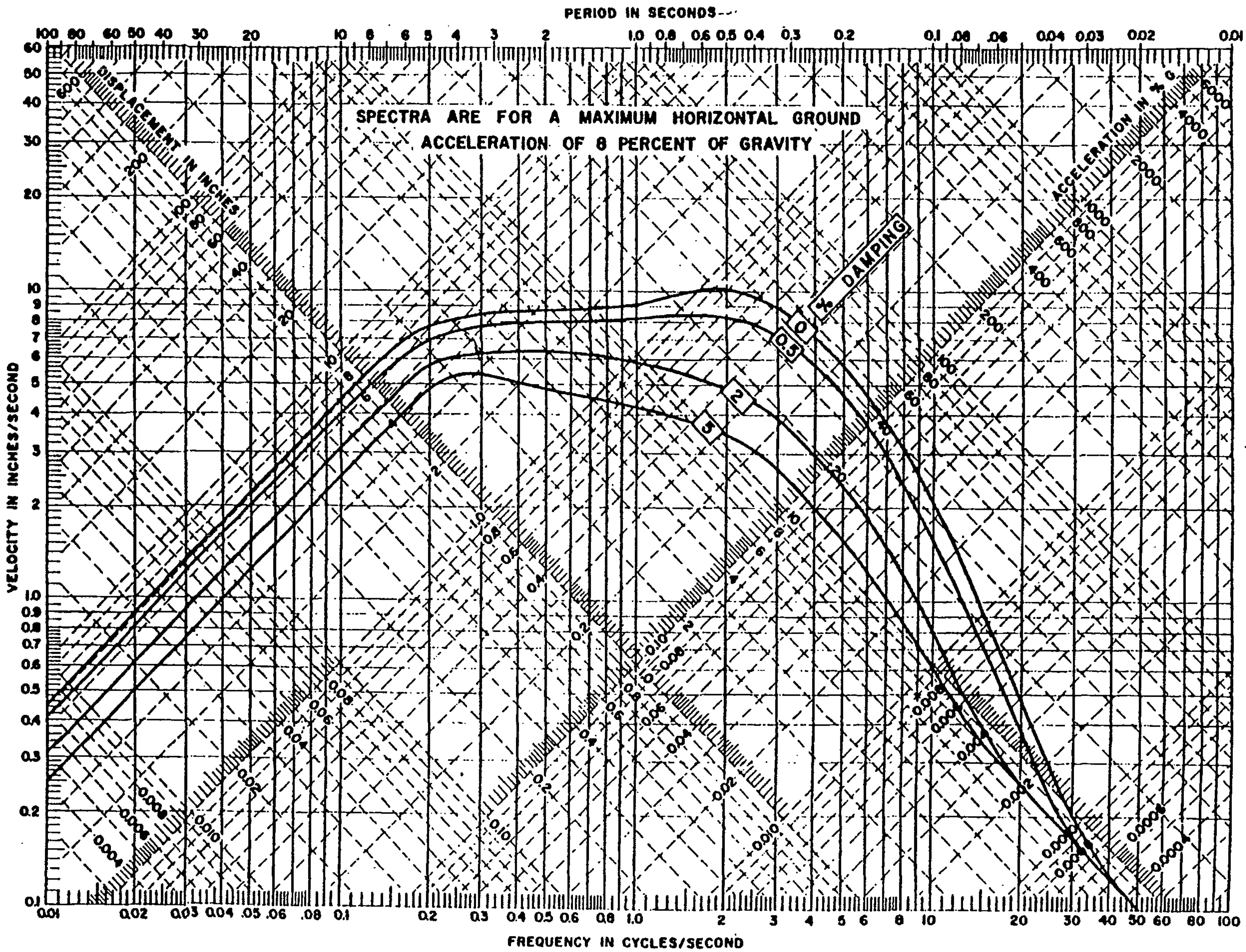
LEGEND:

- ANTICLINAL AXIS
- SYNCLINAL AXIS
- DOME
- NORMAL FAULT
- THRUST FAULT
- CONTOURS ON TOP OF BASEMENT ROCKS

EPICENTRAL LOCATION MAP

MILES
10 0 10 20 30 40 50

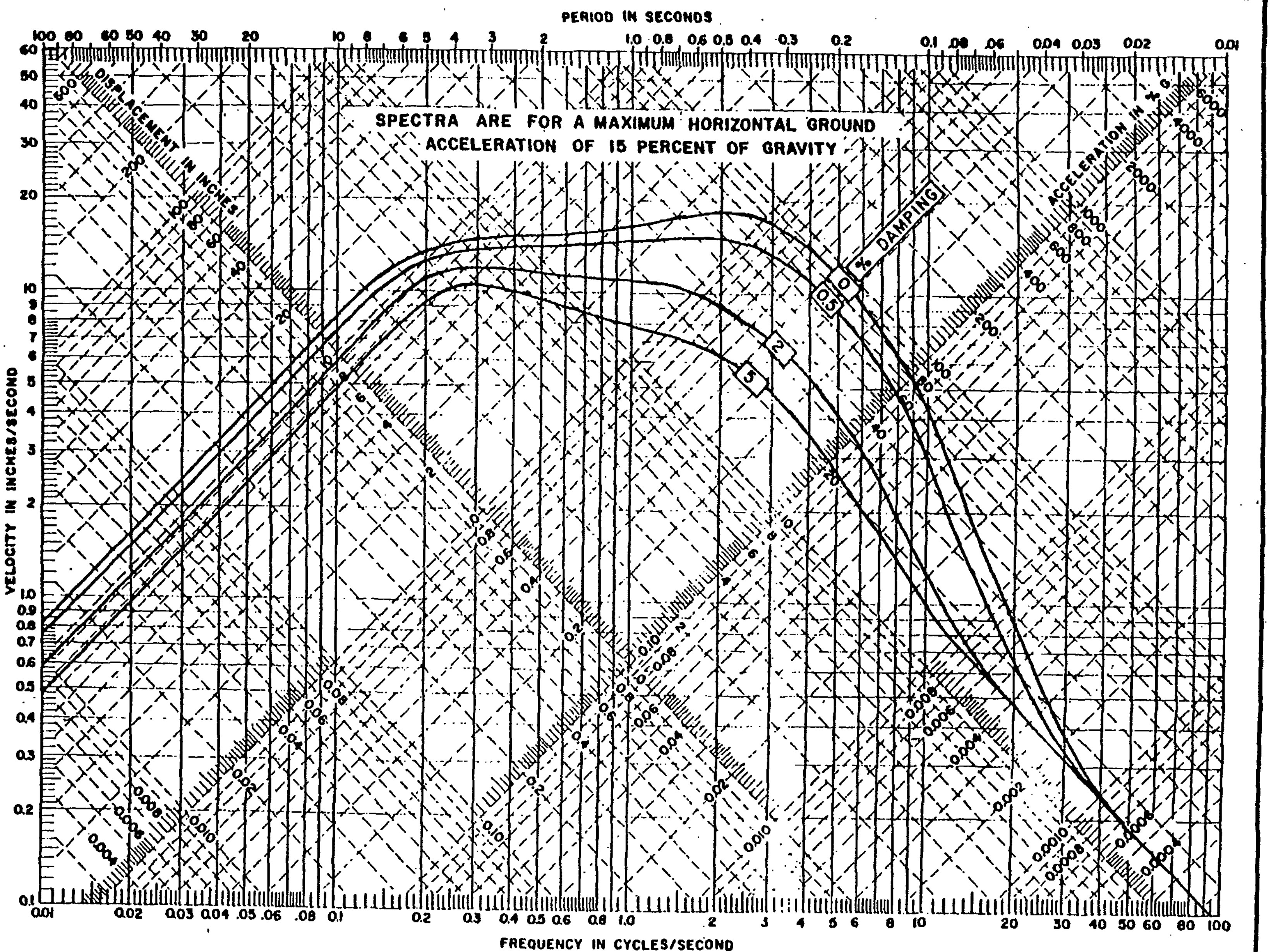
FIGURE 2.6-8



RESPONSE SPECTRA

OPERATING BASIS EARTHQUAKE

FIGURE 2.6-9



RESPONSE SPECTRA

DESIGN BASIS EARTHQUAKE

FIGURE 2.6-10

CHAPTER 3

PRINCIPAL DESIGN CRITERIA

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3.3-10	XX		
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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
HSM-HB	High Burnup Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
MW/MTHM	Megawatt/Metric Ton 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
SPMT	Self-Propelled Modular Transporter
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 with a NUHOMS-24P or NUHOMS-32P DSC. Table 3.1-6 provides the principal design parameters for the spent fuel assemblies to be stored in a Nuclear Horizontal Modular Storage (NUHOMS)-32PHB DSC in an HSM-HB at 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), Sections 12.8.1.3 and 13.8.1.3.

C. As reported in Sections 3.3.4, 12.3.3.4, and 13.3.3.4, 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 and Table 3.1-6. 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 or Table 3.1-6.

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 in Table 3.1-1 does account for changes after irradiation in the core.

The value for assembly length provided in Table 3.1-6 (<158 inches TBD) is based on information provided by the fuel vendors, which was that the maximum assembly length at 60,000 MWD/MTU for the Standard and VAP fuel and at 62,000 MWD/MTU for the AREVA fuel was predicted to be 157.44 inches TBD. The value provided on Table 3.1-6 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. This data is used for the loading of NUHOMS-24P and NUHOMS-32P DSCs. 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.

Tables 9.4-3 and 9.4-4 list the cooling times required to limit the total decay heat power per spent fuel assembly to 0.8 kW and 1.0 kW respectively, for typical ranges of discharge fuel assembly burnups and enrichments. This data is used for the loading of a NUHOMS-32PHB DSC.

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. Section 13.7 provides the corresponding information for a NUHOMS-32PHB DSC.

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. 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 or self-propelled modular transporter (SPMT), 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 and the transfer cask lid. The SPMT will utilize a different transfer cask skid and hydraulic ram system than the ones used with the transfer trailer. 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 IN A NUHOMS-24P OR NUHOMS-32P DSC

PARAMETER	VALUE
Physical Parameters:	
Maximum Assembly Length (with allowance for irradiation growth)	less than 158.0"
Nominal Cross-Sectional 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 Standard CE 14x14 Maximum)** 412 Kg/Assembly (NUHOMS-32P VAP CE 14x14 Maximum)**
Initial Fissile Content	≤ 4.5 w/o U^{235}
Total Gamma Source per Assembly	4.27×10^{15} photons/sec (NUHOMS-24P) 4.67×10^{15} photons/sec (NUHOMS-32P)
Total Neutron Source per Assembly	2.23×10^{8} neutrons/sec (NUHOMS-24P) 4.175×10^{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.
Principal design parameters for fuel to be stored in a NUHOMS-32PHB DSC are listed in Table 3.1-6.

* <u>Fuel Rods/Assembly (32P)</u>

Fuel assemblies with burnup < 47,000 MWD/MTU 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.

** Assembly specific source terms may use actual MTU.

TABLE 3.1-4
RADIOLOGICAL CRITERIA FOR STORAGE OF MATERIAL IN A NUHOMS-24P OR NUHOMS-32P DSC AT THE ISFSI

Gamma Source Spectrum NUHOMS-24P

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 NUHOMS-24P DSC

Energy Range (MeV)			Source Strength (N/sec assembly)
6.36	-	20.0	$6.100x10^{6}$
3.01	-	6.36	$4.486x10^{7}$
1.83	-	3.01	$5.007x10^{7}$
1.11	-	1.83	$4.658x10^{7}$
0.55	-	1.11	$4.287x10^{7}$
0.11	-	0.55	$2.959x10^{7}$
0.00335	-	0.11	$2.999x10^{6}$
< 0.00335			0
			$2.23x10^{8}$

TABLE 3.1-4
RADIOLOGICAL CRITERIA FOR STORAGE OF MATERIAL IN A NUHOMS-24P OR NUHOMS-32P DSC AT THE ISFSI

Gamma Source Spectrum for NUHOMS-32P DSC*

Emean (MeV)	Gamma/sec- Assembly	MeV/sec- Assembly
0.0100	1.12E+15	1.13E+13
0.0250	2.45E+14	6.12E+12
0.0375	3.04E+14	1.14E+13
0.0575	1.96E+14	1.13E+13
0.0850	1.31E+14	1.11E+13
0.1250	1.33E+14	1.67E+13
0.2250	1.15E+14	2.59E+13
0.3750	5.92E+13	2.22E+13
0.5750	2.00E+15	1.15E+15
0.8500	2.73E+14	2.32E+14
1.2500	8.94E+13	1.12E+14
1.7500	1.88E+12	3.28E+12
2.2500	4.62E+11	1.04E+12
2.7500	1.71E+10	4.70E+10
3.5000	2.13E+09	7.46E+09
5.0000	5.48E+06	2.74E+07
7.0000	6.31E+05	4.42E+06
9.5000	7.26E+04	6.90E+05
Total	**4.67E+15**	**1.61E+15**

44-group Neutron Source Spectrum for NUHOMS-32P DSC*

Group	Emin (MeV)	Emax (MeV)	Neutrons/ sec	Group	Emin (MeV)	Emax (MeV)	Neutrons/ sec
1	1.40E+01	2.00E+01	0.000E+00	20	1.66E+00	1.80E+00	1.640E+07
2	1.20E+01	1.40E+01	7.343E+04	21	1.57E+00	1.66E+00	1.091E+07
3	1.00E+01	1.20E+01	4.500E+05	22	1.50E+00	1.57E+00	9.098E+06
4	8.00E+00	1.00E+01	1.525E+06	23	1.44E+00	1.50E+00	7.771E+06
5	7.50E+00	8.00E+00	1.233E+06	24	1.33E+00	1.44E+00	1.556E+07
6	7.00E+00	7.50E+00	1.648E+06	25	1.20E+00	1.33E+00	1.924E+07
7	6.50E+00	7.00E+00	2.438E+06	26	1.00E+00	1.20E+00	2.992E+07
8	6.00E+00	6.50E+00	3.643E+06	27	8.00E-01	1.00E+00	2.910E+07
9	5.50E+00	6.00E+00	5.460E+06	28	7.00E-01	8.00E-01	1.680E+07
10	5.00E+00	5.50E+00	7.343E+06	29	6.00E-01	7.00E-01	1.678E+07
11	4.50E+00	5.00E+00	1.017E+07	30	5.12E-01	6.00E-01	1.443E+07
12	4.00E+00	4.50E+00	1.330E+07	31	5.10E-01	5.12E-01	3.280E+05
13	3.50E+00	4.00E+00	2.145E+07	32	4.50E-01	5.10E-01	9.839E+06
14	3.00E+00	3.50E+00	2.662E+07	33	4.00E-01	4.50E-01	8.198E+06
15	2.50E+00	3.00E+00	3.467E+07	34	3.00E-01	4.00E-01	1.583E+07
16	2.35E+00	2.50E+00	1.310E+07	35	2.00E-01	3.00E-01	3.838E+03
17	2.15E+00	2.35E+00	1.833E+07	36	1.50E-01	2.00E-01	1.919E+03
18	2.00E+00	2.15E+00	1.456E+07	37	1.00E-01	1.50E-01	1.919E+03
19	1.80E+00	2.00E+00	2.124E+07	38-44	1.00E-02	1.00E-01	0.000E+00
						Total	**4.175E+08**

* Reference 3.48.

TABLE 3.1-5
RADIOLOGICAL CRITERIA FOR STORAGE OF MATERIAL IN A NUHOMS-32PHB DSC AT THE ISFSI

Need Gamma Source Spectrum / Source Strength
And
Neutron Source Spectrum / Source Strength
For a fuel-loaded NUHOMS-32PHB DSC

See Table 3.1-4 of the Calvert Cliffs ISFSI USAR

TABLE 3.1-6
PRINCIPAL DESIGN PARAMETERS FOR FUEL TO BE STORED IN A NUHOMS-32PHB DSC

PARAMETER	VALUE
Physical Parameters:	
Maximum Assembly Length (with allowance for irradiation growth)	158.0"
Maximum Assembly Envelope	8.25" by 8.25"
Active Fuel Length	136.7"
Number of Fuel Rods/Assembly (including poison rods)	176
Number of Guide Spacers (including top and bottom end fittings)	9
Maximum Assembly Weight	1,450 lbs
Nominal Center-to-Center Distance Between Spacer Grids	18.86"
Thermal Characteristics:	
Decay Heat Power per Assembly	0.8 to 1.0 kW
Total Heat Load per DSC	29.6 kW
Radiological Characteristics:	
Maximum Burnup (Assembly Average for Standard and VAP Fuel)	60,000 MWD/MTU
Maximum Burnup (Assembly Average for AREVA Fuel)	62,000 MWD/MTU
Maximum Initial Enrichment	5.0 w/o U^{235}
Maximum Uranium Content	420 Kg/Assembly
Total Gamma Source per Assembly	TBD 2.52×10^{15} photons/sec
Total Neutron Source per Assembly	TBD 4.54×10^{8} neutrons/sec
Specific Power (core avg.)	32.2 MW/MTHM

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/HSM-HB 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 Sections 12.3 and 13.3 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 Sections 12.3 and 13.3, such as thermal loads and cask drop loads, are provided in Section 8 of Reference 3.1 and USAR Sections 12.8 and 13.8.

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 or SPMT, 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 Sections 12.3 and 13.3 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/HSM-HB 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

For the HSM and HSM-HB, 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 and using the method described in Reference 3.50 for the HSM-HB.

3.2.1.3 Ability of Structures to Perform

As detailed in Section 3.2.1.3 of Reference 3.1, the HSM/HSM-HB protects the DSC from adverse environmental effects and is the principal NUHOMS structure exposed to tornado wind and missile loads. The HSM/HSM-HB, 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/HSM-HB from adverse environmental effects such as tornado winds. The analyses of the HSM/HSM-HB and transfer cask for tornado effects is contained in Sections 8.2.1, 8.2.2, 12.8.2.1, and 12.8.2.2.

Since the Calvert Cliffs HSMs/HSM-HBs have been constructed outdoors in an open area, there is no possibility of an adjacent building collapsing on an HSM/HSM-HB. 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, 12.8.2.7, and 13.8.2.7.

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 or SPMT, 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 Sections 8.2.2 and 12.8.2.2.

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

3.2.3.1 HSM 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, 12.8.2.3 and 13.8.2.3.

3.2.3.2 HSM-HB Seismic Design

The seismic design criteria for the HSM-HB is consistent with the criteria set forth in Section 3.2.3, with the exception that the NRC RG 1.60 response spectra is anchored to a maximum ground acceleration of 0.30g for the horizontal components and 0.20g for the vertical component (Reference 3.51). The seismic design criteria of the HSM-HB bounds the design of the original poured in place HSMs. The seismic analyses of the HSM-HB and DSC are discussed further in Section 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 and high burnup horizontal storage module (HSM-HB) analyses to envelope all postulated live loadings, including snow and ice, as was done in Section 3.2.4 of Reference 3.1. Reference 3.49 provides the equivalent analyses for the HSM-HB. 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 and HSM-HB 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, 12.3-6, and 13.3-6.

3.2.5.3 NUHOMS Transfer Cask (Non-Forced-Cooling Configuration)

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.)

The forced-cooling configuration of the transfer cask is discussed in Section 13.TBD.

3.2.5.4 Transfer Equipment

The transfer equipment consists of the cask support skid, positioning system, trailer, tractor unit, SPMT 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.

3.2.6.3 NUHOMS-32PHB DSC

The weld requirements for the NUHOMS-32PHB DSC are described in Section 13.3.2.6.

TABLE 3.2-1
SUMMARY OF DESIGN CRITERIA FOR NON-IMPORTANT TO SAFETY COMPONENTS FOR THE NUHOMS-24P AND NUHOMS-32P DSC SYSTEMS[b]

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.

(b) These components are specific to the transfer trailer and the transfer cask in the non-forced cooling configuration.

TABLE 3.2-2
HSM/HSM-HB LOAD COMBINATION METHODOLOGY

HSM-HB CASE No.	HSM CASE No.	LOAD COMBINATION	LOADING NOTATION
C1C	1	1.4D + 1.7L	D = Dead Weight E = Earthquake Load F = Flood-Induced Loads H = Lateral Soil Pressure Load L = Live Load T = Normal Condition Thermal Load T_a = Off-normal or Accident Condition Thermal Load W = Wind Load
	2	1.4D + 1.7L + 1.7H	
C2C	3	0.75 (1.4D + 1.7L + 1.7H + 1.7T + 1.7W)	
C3C	4	0.75 (1.4D + 1.7L + 1.7H + 1.7T)	
C4C	5	D + L + H + T + E	
C5C	n/a	D + L + H + T + W	
C6C	6	D + L + H + T + F	
C7C	7	D + L + H + T_a	

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.

TABLE 3.2-5
SUMMARY OF DESIGN CRITERIA FOR NON-IMPORTANT TO SAFETY COMPONENTS FOR THE NUHOMS-32PHB DSC SYSTEM[b]

COMPONENT	DESIGN LOAD TYPE	DESIGN PARAMETERS	APPLICABLE CODE
Self-Propelled Modular Transporter	Dead Load	Weight of loaded DSC + Transfer Cask = 215,000 lbs Weight of Skid = 25,000 lbs Transporter Dead Load (Payload Capacity) = 30,000 lbs	N/A
	Operating Loads to Trailer Deck	a) Longitudinal load: TBD lb b) Transverse load: TBD lb	
Cask Support Skid	Dead Load	Weight of loaded DSC + Cask = TBD lb[a] enveloping	AISC Code – TBD
	Operating Loads	a) System loads: Longitudinal load: TBD lb Transverse load: TBD 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 - TBD
Hydraulic Ram System	Normal Operation	a) Operating force: 23,750 lb b) Maximum speed: TBD in/min	N/A
	Jammed Condition Handling	a) Operating force: 95,000 lb b) Maximum Speed: TBD in/min	
Cask Restraints	Normal Operation	Ram hydraulic cylinder normal operating force, TBD lb	ANSI/ANS 57.9 [TBD]
	Jammed Condition Handling	Ram hydraulic cylinder maximum operating force, TBD lb	ANSI/ANS 57.9 [TBD]
Vacuum Drying System	Normal Operation	DSC vacuum: 3.0 Torr	

(a) These components are specific to the self-propelled horizontal cask transporter and the transfer cask in the forced cooling configuration.

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, Section 12.3.3 for the NUHOMS-32P DSC and Section 13.3.3 for the NUHOMS-32PHB 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 $1x10^{-4}$ atm-cc/sec for the NUHOMS-24P and NUHOMS-32P DSCs. The maximum acceptable helium leak rate for a NUHOMS-32PHB DSC is less than $1x10^{-7}$ atm-cc/sec. The NUHOMS-32PHB DSCs are considered to be leaktight per ANSI N14.5-1997.

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 and in Section 13.3.3.4 for the NUHOMS-32PHB DSC. 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 in a NUHOMS-24P DSC

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 in a NUHOMS-24P DSC

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-24P DSC 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 ≤ 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/MTU 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 in a NUHOMS-24P DSC (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 multigroup 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":

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

Where:

ND_m = Effective boron number density of mixture (atoms/barn-cm)

D_m = Steam/liquid mixture effective density (gm/ml)
= $MF_s \times D_s + MF_1 \times D_1$

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 1.0×10^{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 for a NUHOMS-24P DSC

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 for a NUHOMS-24P DSC (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:

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

$$K_{eff} = 0.97968$$

Misloading of 24 High Enrichment Assemblies — With Boron Credit

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.

Optimum Moderation

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 for a NUHOMS-24P DSC

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 is 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 are employed to detect unauthorized access to the facility. In addition to the controlled access, the HSM/HSM-HB steel doors will be tack welded after insertion of a loaded DSC. The HSM doors weigh approximately 6 tons and the HSM-HB doors weigh approximately 8 tons. Both 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, 12, and 13.

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/HSM-HB 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, 12.8, and 13.8.

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.

Analyses performed for the HSM-HB (Reference 3.49) considered blast pressures exceeding the aforementioned overpressures (60 psi front, 24 psi side/roof).

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 are 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/HSM-HBs.

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/HSM-HBs.

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/HSM-HB surfaces

c. Interior inspection of HSMs/HSM-HBs 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, passive ventilation maintains the maximum normal operating fuel clad temperature to 620°F for HSM and 631°F for HSM-HB (assuming 103°F ambient temperature). During short-term conditions, such as DSC draining and drying, transfer of the DSC to/from the HSM/HSM-HB 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	Criticality Control, Fuel Support, Cover Gas Containment, Radioactive Material Confinement, Shielding (Lead Plugs)
NUHOMS-24P	
Guide Sleeves	
Spacer Disks	
Support Rods	
End Shield Plugs	
DSC Body	
End Cover Plates	
NUHOMS-32P and NUHOMS-32PHB	
Guide Sleeves	
Egg-Crate Plates	
Peripheral Steel Rails with Aluminum Inserts	
End Shield Plugs	
DSC Body	
End Cover Plates	
Horizontal Storage Module (HSM or HSM-HB)	Shielding, DSC Support, DSC Tornado Missile Protection, DSC Cooling
Concrete Shielding	
DSC Support Assembly	
HSM Passive Ventilation System	
Foundation	HSM Foundation Support
Transfer Components	Transfer Cask Movement, DSC Transfers
Cask Support Skid and Positioning System*	
Transfer Trailer*	
Hydraulic Ram	
Optical Alignment System	
Tractor*	

* The self-propelled horizontal cask transporter may be used instead of the transfer trailer, tractor and skid positioning system.
The self-propelled horizontal cask transporter will utilize a different transfer cask skid and hydraulic ram system than the ones used with the transfer trailer.

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 with burnup < 47,000 MWD/MTU 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.
See Table 13.3-1 for NUHOMS-32PHB 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.
See Table 13.3-2 for NUHOMS-32PHB 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 for the original poured in place HSM and in accordance with ACI 349-97 for the HSM-HB 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, SPMT) 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			
Helium			
Dry Shielded Canister (NUHOMS-32P and NUHOMS-32PHB)	Important to Safety	Safety-Related	SR
Basket			
Egg-Crate			
End Shield Plug/Support (top and bottom)			
DSC Body			
End Closure Plates			
Helium			
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			
Self-Propelled Modular Transporter (SPMT)			
Transfer Cask Skid used with the SPMT			
Ram Assembly used with the SPMT			
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), 12.3-3 (NUHOMS-32P DSC), and Table 13.3-3 (NUHOMS-32PHB DSC). Summaries of design criteria for off-normal and accident conditions are given in Tables 3.6-2 and 3.6-3 (NUHOMS-24P), Tables 12.3-4 and 12.3-5 (NUHOMS-32P DSC), and Tables 13.3-4 and 13.3-5 (NUHOMS-32PHB 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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3.3 A. G. Croff, C. W. Alexander, "Decay Characteristics of Once-Through LWR and LMFBR Spent Fuels, High-Level Wastes, and Fuel-Assembly Structural Material Wastes," ORNL/TM-7431, Oak Ridge National Laboratory, 1980

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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3.12 Combustion Engineering, Inc., "The ROCS and DIT Computer Codes for Nuclear Design," CE Power Systems, CENPD-266-NP, April, 1983

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)

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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

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

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3.36 Levy, I. S., et al. Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas, May 1987, PNL-6189 (UC-85), Pacific Northwest Laboratory, Richland, Washington

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

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

3.39 O. Ozer, "Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks - Overview of Fuel Management Analysis Methodology," pg. 203, SAND-0018, October 1989

3.40 D. G. Napolitano, "Proceedings of the International Topical Meeting on Safety Margins in Criticality Safety - Uncertainties in Criticality Analysis Which Affect the Storage and Transportation of LWR Fuel," pg. 297, November 1989

3.41 Calculation BGE001.0203

3.42 Calculation BGE001.0202

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

3.48 CCNPP Calculation No. CA06721, Source Terms for ISFSI 32P Burnup Extension, Revision 000

3.49 CCNPP Calculation No. CA07307, "HSM-HB Structural Analysis for NUHOMS-32PHB System," Revision 0000

3.50 American Society of Civil Engineers, ASCE 7-95, "Minimum Design Loads for Buildings and Other Structures"

3.51 Transnuclear, Inc., "Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1004, Transnuclear Document No. NUH-003, Revision 11

3.52 Transnuclear, Inc., "Updated Final Safety Analysis Report for the NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1030, Revision 2

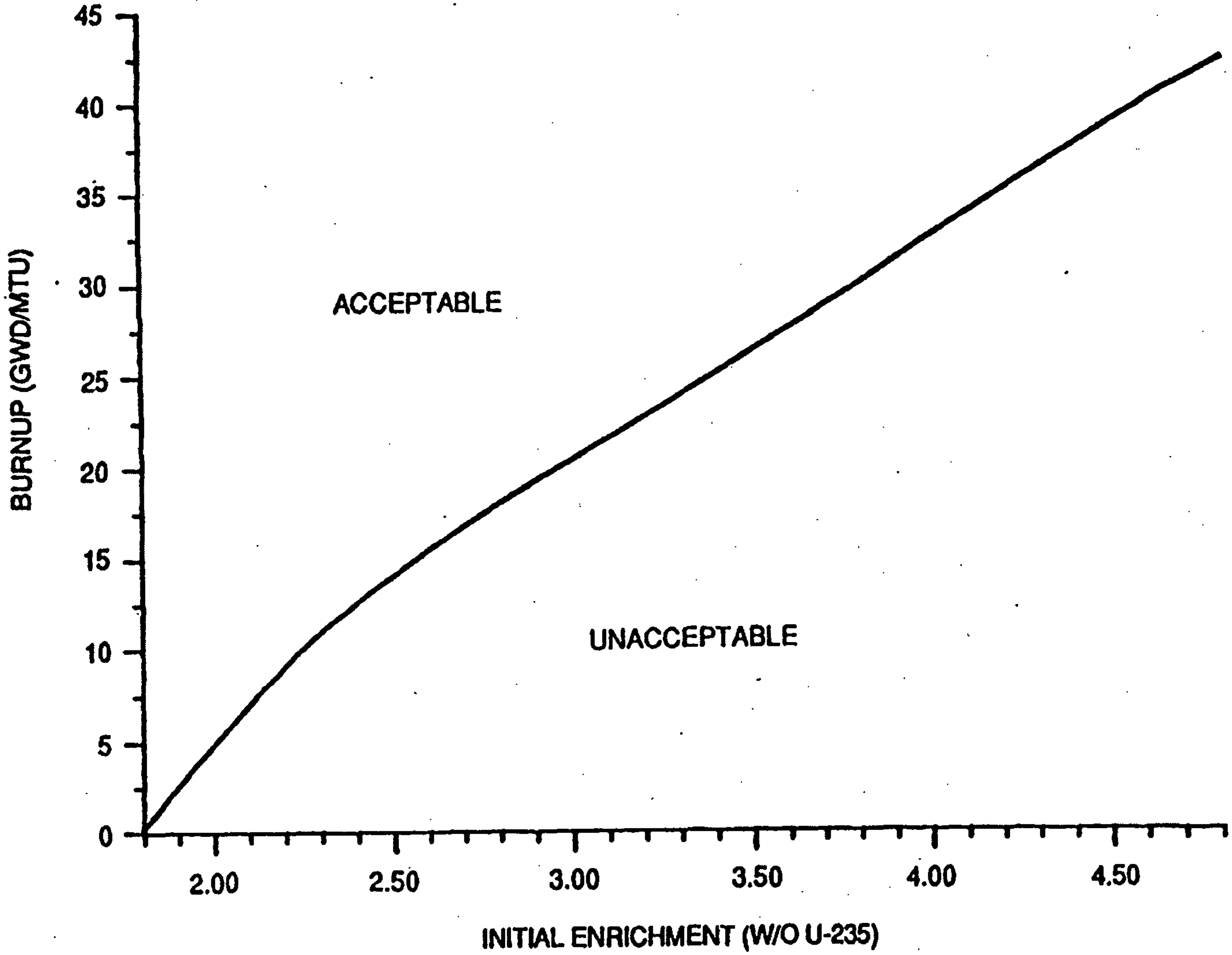
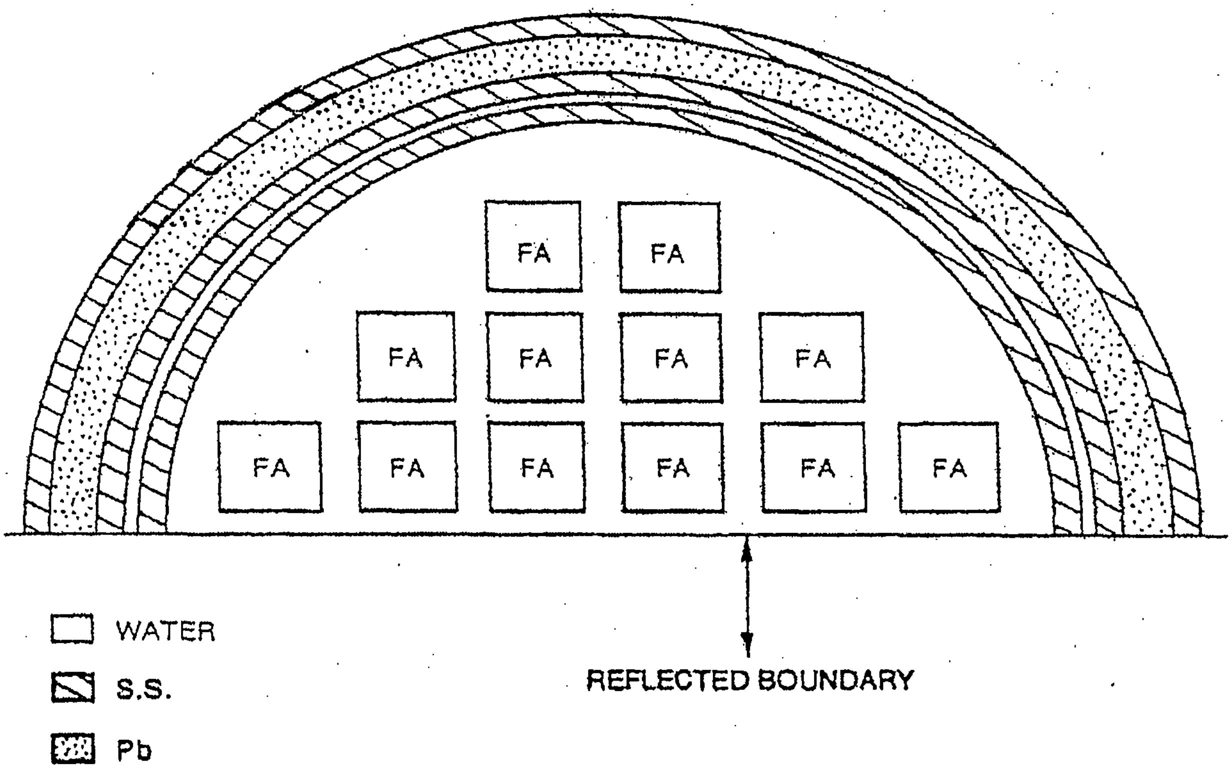


Figure 3.3-1

Calvert Cliffs NUHOMS®-24P Burnup Curve



Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	NUHOMS-24P DSC AND TRANSFER CASK KENO MODEL	USAR Figure 3.3-2 Revision 15

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LIST OF ACRONYMS

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
CFR	Code of Federal Regulations
DSC	Dry Shielded Canister
FPM	feet-per-minute
HSM	Horizontal Storage Module
HSM-HB	High Burnup Horizontal Storage Module
HRS	Hydraulic Ram System
ISFSI	Independent Spent Fuel Storage Installation
NFPA	National Fire Protection Association
NUHOMS	Nutech Horizontal Modular Storage®
SPMT	Self-Propelled Modular Transporter
SPS	Skid Positioning System
SRV	Safety Relief Valve

4.0 INSTALLATION DESIGN

4.1 SUMMARY DESCRIPTION

4.1.1 LOCATION AND LAYOUT OF THE INSTALLATION

The location and layout of the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) with respect to other plant site structures is shown in Figure 1.1-1. This figure also denotes the route for transport of the transfer cask carrying dry shielded canisters (DSCs) from the Auxiliary Building to the ISFSI.

The initial construction phase of the ISFSI included six poured in place 2x6 horizontal storage module (HSM) arrays which store up to 72 DSCs. Additional HSM storage capacity will be added incrementally by installing two 2x12 and one 1x12 modular high burnup horizontal storage module (HSM-HB) arrays as needed up to a total of 132 HSMs/HSM-HBs. Figure 4.1-2 shows the arrangement of the storage arrays.

The area around the ISFSI is sloped to direct surface drainage to collection ditches for channeling rain water away from the site. As noted in Section 2.4, the ISFSI is about 86' above the probable maximum flood elevation. Local intense rainfall is not a problem since adequate surface drainage exists at the ISFSI yard to assure that water will not collect to a depth of any concern (Reference 4.4).

The chosen transport route has been reviewed and is found to be in compliance with the design criteria of the transfer cask drop analysis discussed in Section 8.2 of the Nutech Horizontal Modular Storage® (NUHOMS)-24P Topical Report (Reference 4.1). Furthermore, the transport route has been reviewed to assure that no roadways, subgrade structures, buried pipes or trenches will be damaged by the transport trailer wheel loads. The approach slab has adequate space for turning the transport trailer and tow vehicle. No other turning areas are needed along the transport route. The roadway or ground surface elevation perpendicular to the route to or from the ISFSI within an 8.0' proximity of the transfer trailer is not more than 20" below the road surface centerline elevation. The paved portion of the road is a minimum of 16' wide and with the adjacent paved, gravel or soil shoulder the transfer route is at least 28' wide. The lowest point in the width of the transfer route is not lower than 20" below the road centerline. The transfer route contains typical roadside fixtures, including curbs, fences, guard rails, and light poles which do not constitute potential puncture mechanisms for the cask. The shoulders do not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture device. The components associated with the vehicle barrier system have been analyzed and do not represent a puncture risk to the transfer cask. The road is closed to other vehicles when transporting the spent fuel. The maximum drop height of the cask from the transfer trailer to the roadbed does not exceed 80".

The self-propelled horizontal cask transporter may be used in place of the transport trailer and tow vehicle. The self-propelled horizontal cask transporter has a smaller turning radius than the transport trailer and tow vehicle. This smaller turning radius allows the area between the HSMs to be smaller and therefore the ISFSI approach slabs for the HSM-Hs will be narrower than the approach slabs for the original poured in place HSMs. The ISFSI transport route from the auxiliary building to the Calvert Cliffs ISFSI approach slabs (including the approach slabs) was reviewed for the wheel loads of a fuel-loaded self-propelled horizontal cask transporter.

4.1.2 PRINCIPAL FEATURES

4.1.2.1 Site Boundary

The property owned by Calvert Cliffs Nuclear Power Plant, Inc. surrounding the Calvert Cliffs ISFSI is shown in Figure 2.1-2.

4.1.2.2 Controlled Area

The controlled area for the ISFSI, as defined by Title 10, Code of Federal Regulations (CFR) 72.106, is identified in Figure 2.1-2. Its border from the HSM array is a minimum of 3900' (1189 meters) as shown in Figure 2.1-2.

4.1.2.3 Site Utility Supplies and Systems

No utility systems are required for the storage phase of the ISFSI. Electrical power is provided to operate the hydraulic pumps used during DSC insertion or withdrawal operations at the HSM, and for lighting and security systems. No water or sewer systems are necessary. The existing plant page system is extended to provide telephone and paging communications.

4.1.2.4 Storage Facilities

There are no holding ponds, chemical or gas storage vessels, or other open-air tankage on or near the ISFSI. When empty DSCs are stored at the ISFSI site, they are placed horizontally on wooden cribbing with their ends facing north and south.

4.1.2.5 Stacks

The ISFSI has no stacks. Two HSM air outlet vents are located on the roof of each HSM. A concrete shielding cap is located over each outlet.

4.2 STORAGE STRUCTURES

4.2.1 STRUCTURAL SPECIFICATIONS

The principal storage structures in the ISFSI are the HSM/HSM-HB and the DSC. The original poured in place HSM and DSC design bases, materials of construction, codes and standards, etc., are in full compliance with the NUHOMS-24P Topical Report (Reference 4.1) and are listed below. Activities which are covered by the Quality Assurance Program are discussed in Chapter 11.

4.2.1.1 Horizontal Storage Modules – Original Poured in Place HSM and Modular HSM-HB

A. American Concrete Institute (ACI) 349-85 and 349B (HSM), 349-97 (HSM-HB), "Code Requirements for Nuclear Safety-Related Concrete Structures" (For design, not construction)

B. ACI 318-83 (HSM), 319-89 (HSM-HB), "Building Code Requirements for Reinforced Concrete" (For construction, not design)

C. American Welding Society D1.1-88 (HSM), D1.1-98 (HSM-HB), "Structural Welding Code-Steel"

D. American National Standards Institute (ANSI) 57.9-1984 (HSM and HSM-HB), "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)"

E. ANSI A58.1-1982 (HSM), American Society of Civil Engineers (ASCE) 7-95 (HSM-HB), "Minimum Design Loads for Building and Other Structures"

F. American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 8th Edition (HSM), 9th Edition (HSM-HB)

G. National Fire Protection Association, No. 78, "Lightning Protection Code," 1983 Edition (HSM), 1986 Edition (HSM-HB)

H. American Welding Society, AWS 01.6-1999, "Structural Welding Code – Stainless Steel" (HSM-HB)

4.2.1.2 Dry Shielded Canister – NUHOMS-24P

A. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III, Division 1, Subsection NB, "Class 1 Components," l983 Edition through Winter l985 Addenda used as a guide for design and fabrication. An N-stamp is not required.

B. ANSI/American Nuclear Society 57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)"

C. ANSI N14.5-1977, "Leakage Tests on Packages for Shipment of Radioactive Materials"

D. ANSI Y14.5M-1982 "Dimensioning and Tolerancing"

E. ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Power Plants"

F. PNL-6189, "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircalloy Clad Fuel Rods in Inert Gas"

G. PNL-4835, "Technical Basis for Storage of Zircalloy-Clad Spent Fuel in Inert Gases"

H. ANSI-57.2-1983, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants"

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

4.2.1.3 Dry Shielded Canister – NUHOMS-32P

Unless replaced below, the specification in Section 4.2.1.2 also applies.

A. ASME B&PV Code, Section III, Division 1, Subsection NB, NF, NG, and Appendix F, 1988 with 1999 Addenda. N – stamp is not required.

B. ANSI/ANS 57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)

C. ANSI N14.5-1987, Leakage Tests on Packages for Shipment for Radioactive Materials

4.2.1.4 Dry Shielded Canister – NUHOMS-32PHB

Unless replaced below, the specification in Section 4.2.1.2 also applies.

A. ASME B&PV Code, Section III, Division 1, Subsection NB, NF, NG, and Appendix F, 1988 with 1999 Addenda. N – stamp is not required.

B. ANSI/ANS 57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)

C. ANSI N14.5-1987, Leakage Tests on Packages for Shipment for Radioactive Materials

4.2.2 INSTALLATION LAYOUT

4.2.2.1 Building Plans

Engineering drawings of the storage structures (HSM/HSM-HB and DSC) have been developed and are maintained in accordance with the Quality Assurance Program described in Chapter 11.

4.2.2.2 Building Sections

Engineering drawings of the storage structures (HSM/HSM-HB and DSC) have been developed and are maintained in accordance with the Quality Assurance Program described in Chapter 11.

4.2.2.3 Confinement Features

Radioactive particulate matter and gaseous fission products are confined within the DSC. The containment features of the DSC are fully described in the NUHOMS-24P Topical Report (Reference 4.1). The integrity of the Calvert Cliffs ISFSI DSCs is tested using a helium leak test. The acceptance criterion for the test is specified in the Technical Specifications.

4.2.3 INDIVIDUAL UNIT DESCRIPTION

4.2.3.1 Horizontal Storage Module Description

The HSM/HSM-HB provides structural support for the DSC, protects the DSC against extreme natural hazards such as tornado missiles, and provides radiation shielding. The concrete walls form interconnected sub-units of modules. The poured inplace HSM units are six wide and two back-to-back forming 2x6 arrays. The HSM-HB units are 12 wide and two back-to-back, forming 2x12 arrays and one 1x12 array. The modules are designed to provide surface dose rates at or below the design criteria of 20 mrem/hr on the HSM side and 100 mrem/hr at the HSM door. The HSM array sizes selected are in compliance with the array size criteria discussed in Section 4.2.3.2 of Reference 4.1.

The HSM dissipates decay heat from the spent fuel by a combination of radiation, conduction, and convection. Natural convection air flow enters at the bottom of the HSM, circulates around the DSC, and exits through the flow channels in the HSM roof slab. A thermal radiation shield is used to reduce the HSM roof temperature to within acceptable limits for all conditions.

Two thermal considerations form the basis for the HSM design: maximum concrete temperatures and maximum fuel cladding temperatures. All concrete temperatures are within the limits set by ACI-349 except for the blocked vents case in which the concrete temperatures exceed the applicable ACI-349 limits but are the same as, or less than, the temperatures reported and accepted in Reference 4.1. The maximum fuel cladding temperatures are below the cladding temperature limit described in Chapter 3. The thermal analysis of the HSM, described in Sections 8.1.3 and 12.8, indicates that all temperatures will remain within acceptable limits under all conditions, including the blocked vent case, for at least 36 hours. The inspection interval for the HSM air inlets has therefore been specified in the Technical Specifications as every 24 hours.

The design of the HSM system includes consideration of both normal and off-normal operating conditions including a range of postulated and hypothetical accidents. The HSM design and analysis for the Calvert Cliffs ISFSI were performed in accordance with Chapters 3, 8, and 12 and Reference 4.1. The design calculations for the reinforced concrete are specific for the Calvert Cliffs ISFSI, and support the design as presented on the drawings. These calculations form the basis for reducing the amount of reinforcing steel in the HSM compared to the topical report design.

The approach slab in front of the HSMs is constructed separately from the HSM foundations. The HSM-HB units are installed on a load bearing foundation which consists of a reinforced concrete slab. The original HSM units, however, have individual foundations that are poured in place. Therefore, the transfer system is designed to accommodate any credible differential settling between the two slabs. The approach slab and HSM foundation have been designed to minimize differential settlement over the life of the facility.

4.2.3.2 Dry Shielded Canister Description

The DSC provides mechanical confinement of the stored fuel assemblies and all radioactive materials for two purposes: to prevent the dispersion of particulate or gaseous radionuclides from the fuel, and to maintain a barrier of helium around the fuel in order to mitigate corrosion of the fuel cladding and prevent expansive oxides from forming in the fuel itself.

Another function of the DSC is to provide for criticality safety during the wet loading operations and during the DSC drying operations. A detailed discussion of the criticality analyses is included in Sections 3.3.4, 12.3.3.4, and 13.3.3.4.

The DSC provides radiological shielding in both axial directions. The top shield plug serves to protect operating personnel during the DSC drying and sealing operations. The bottom end shielding reduces HSM door area dose rates during storage. The DSC shielding is designed for a maximum contact dose of 100 mrem/hr (flooded cavity).

The DSC is designed to slide from the transfer cask into the HSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces. This is accomplished by the addition of Nitronic 60 hard sliding rails to the transfer cask and HSM. The HSM and cask rails are coated with dry film lubricants.

The DSC provides physical protection and structural support of the spent fuel during loading operations and during storage. The DSC is designed so that the worst-case postulated accidents will not result in deformation of the basket or the DSC shell to such a degree that post-accident removal of intact fuel assemblies is prohibited.

Calvert Cliffs Nuclear Power Plant implemented use of a modified DSC design for DSC R025 through R048. Design changes were made to the internal basket assembly in order to better accommodate the effects of postulated cask drop accidents. The modified DSC internal basket assembly design closely resembles the original design. The notable difference is with the modified DSC, the guide sleeves are not attached to any spacer disc. Refer to the DSC analysis and internal basket analysis in Chapter 8 for further discussion.

4.3 AUXILIARY SYSTEMS

The ISFSI is a self-contained, passive storage facility which requires no auxiliary systems.

4.3.1 VENTILATION AND OFF-GAS

Spent fuel confined in storage at the ISFSI is cooled by conduction and radiation within the DSC, and conduction, convection, and radiation from the DSC surface. An air inlet near the bottom of the HSM/HSM-HB front wall and outlets in the HSM/HSM-HB roof allow convective cooling by natural circulation. The driving force for this ventilation system is described in Section 8.1.3. No auxiliary ventilation is used or required at the ISFSI. Fuel loading and DSC closure operations take place in the plant's Auxiliary Building and make use of the ventilation system in that facility. Auxiliary Building ventilation is discussed in Section 9.8.2.3 of Reference 4.2.

The Vacuum Drying System provides a means for removing water and water vapor from the DSC and for backfilling the DSC with helium. This function is required to ensure that fuel is stored in an inert atmosphere, to take advantage of the favorable heat transfer properties of helium, and to ensure the long-term maintenance of the fuel clad integrity.

The Vacuum Drying System is designed to operate in four modes: liquid removal by pump, liquid removal by a source of pressurized helium, nitrogen or air, vacuum drying, and helium backfill. The evacuation is performed in several stages to allow the DSC pressure to stabilize. When the pressure can be held at 3 torr for at least 30 minutes, this indicates that all liquid water has evaporated in the DSC cavity, and that the resulting inventory of oxidizing gases is less than 0.25% (Vol%). The cavity is then backfilled with helium. After again pumping the cavity down to 3 torr, a final helium backfill is made and the DSC is sealed. This process further reduces the partial pressure of any water vapor still present in the DSC.

4.3.2 ELECTRICAL

No electrical systems are required for the HSM/HSM-HB or DSC during long-term storage, other than for lighting and security system power. Electrical power is used during DSC closure operations in the plant's Auxiliary Building and during DSC transfer operations to the HSM/HSM-HB at the ISFSI. The required electrical power in the Auxiliary Building is obtained from the existing plant system. Power at the ISFSI is supplied from the site power distribution system which is powered from 11 and/or 21 13 kV plant busses.

4.3.3 AIR SUPPLY

Compressed helium or filtered plant air is used to force water from the DSC during closure operations.

4.3.4 STEAM SUPPLY AND DISTRIBUTION

There are no steam systems required.

4.3.5 WATER SUPPLY

Borated water is used to fill the DSC cavity prior to insertion into the spent fuel pool. The water source is compatible with the plant's existing spent fuel pool. The source of supply may be the pool itself. Demineralized water is needed for filling the DSC/cask annulus, and for washdown operations. This water is supplied by the existing Auxiliary Building demineralized water supply.

4.3.6 SEWAGE TREATMENT

No sewage treatment system is required for the ISFSI.

4.3.7 COMMUNICATIONS AND ALARM

No communication systems are required for the safe operation of the ISFSI. The existing plant page and telephone system has been extended to the ISFSI for convenience during transfer operations.

Security alarm systems are described in Reference 4.3.

4.3.8 FIRE PROTECTION

No fire detection or suppression system is required at the ISFSI, since no combustible materials are present within the ISFSI controlled area boundary. Response to a forest fire in the vicinity of the ISFSI is provided by Calvert Cliffs Nuclear Power Plant personnel, using portable fire suppression equipment. Off-site fire fighting equipment and personnel are also available, if needed.

4.3.9 MAINTENANCE

The NUHOMS system is designed to be totally passive with minimal maintenance requirements. During fuel storage, the system only requires visual inspection of the air inlets once every 24 hours to ensure that no blockage has occurred.

The transfer cask is designed to require only minimal maintenance. Transfer cask maintenance is limited to periodic inspection of critical components and replacement of damaged or non-functioning components. A detailed discussion of these requirements is provided in Section 4.5.

4.3.10 COLD CHEMICAL

There are no cold chemical systems at the ISFSI.

4.3.11 AIR SAMPLING

No air sampling systems are required at the ISFSI. Airborne activity during fuel loading and DSC closure operations is monitored by the existing Auxiliary Building ventilation and radiological detection systems.

4.4 DECONTAMINATION SYSTEMS

4.4.1 EQUIPMENT DECONTAMINATION

No equipment decontamination facilities will be needed at the ISFSI.

Within the Auxiliary Building, decontamination of equipment is required for the transfer cask and yoke exterior surfaces, the top surface of the DSC shield plug, and for tools which may become contaminated during DSC drying and sealing operations.

Decontamination of the transfer cask exterior after removal from the spent fuel pool is performed in the Auxiliary Building Cask Washdown Pit. The transfer cask is manually decontaminated using detergents and wiping cloths before removal from the Auxiliary Building. The DSC top shield plug is decontaminated in the same manner prior to being seal welded to the DSC body.

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

Contaminated tools will be cleaned using existing plant procedures and facilities.

4.4.2 PERSONNEL DECONTAMINATION

No personnel decontamination facilities are needed at the ISFSI. Personnel decontamination in the Auxiliary Building, if necessary, is accomplished using existing plant equipment and procedures.

4.5 TRANSFER CASK REPAIR AND MAINTENANCE

Since the transfer cask utilizes a solid neutron shield, no periodic maintenance other than a visual inspection is anticipated. The transfer cask and trailer will not be stored at the ISFSI when not in use, but will normally be stored in the Equipment Storage Building near the ISFSI. Repairs, if needed, will be performed in the Equipment Storage Building, a low-level radioactive waste processing and storage facility, or in the plant's Auxiliary Building, using existing plant equipment, personnel, and procedures.

4.7 FUEL HANDLING OPERATION SYSTEMS

Fuel handling at the Calvert Cliffs ISFSI is performed in two general locations. Individual fuel assemblies are loaded into the DSC in the plant's Auxiliary Building. Once in the DSC, the fuel is transported to the ISFSI and loaded into the HSM/HSM-HB. Fuel handling activities inside the Auxiliary Building are performed under the plant's 10 CFR Part 50 license. Fuel handling operations outside the Auxiliary Building are performed under the 10 CFR Part 72 license. Fuel handling operation systems in the Auxiliary Building include:

A. Spent Fuel Cask Handling Crane
B. Spent fuel handling machine
C. Transfer cask and lifting yoke

The fuel handling operation systems outside the Auxiliary Building include the equipment required to transport the DSC to the HSM and insert the DSC into the HSM. These are the:

A. Transfer cask
B. Transfer trailer, self-propelled modular transporter (SPMT), and skid
C. Skid Positioning System (SPS)
D. Hydraulic Ram System (HRS)

4.7.1 STRUCTURAL SPECIFICATIONS

The Spent Fuel Cask Handling Crane and spent fuel handling machine are described in Section 9.7 of the Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report. The bases and engineering design for the transfer cask are described in Section 4.7.4.1.

4.7.2 INSTALLATION LAYOUT

Fuel handling operations will occur within the existing Auxiliary Building. Figures 1-5 through 1-16 of Reference 4.2 show applicable plans and sections. Confinement features of the Auxiliary Building relate to ventilation systems and are described in Section 9.8 of Reference 4.2.

4.7.3 INDIVIDUAL UNIT DESCRIPTION

4.7.3.1 Cask Handling Crane

The Spent Fuel Cask Handling Crane will be used for all DSC and transfer cask movements within the Auxiliary Building. This crane meets the single-failure-proof criteria of NUREG-0554 and NUREG-0612.

4.7.3.2 Spent Fuel Handling Machine

The spent fuel handling machine will be used to load spent fuel assemblies into the DSC.

4.7.3.3 Transfer Cask

The transfer cask is a cylindrical vessel with a bottom end closure assembly and a bolted top cover plate. This cask is shown in Figure 4.7-1. The cask's cylindrical walls are formed from three concentric steel shells with lead poured between the inner liner and the structural shell to provide gamma shielding during DSC transfer operations. The outer shell forms an annular pressure vessel with a solid neutron absorbing material poured between the structural shell and outer shell to provide neutron shielding during DSC transfer operations.

The cask bottom end assembly is welded to the cylindrical shell assembly and includes two closure assemblies for the ram/grapple access penetration. A water-tight, bolted cover plate, with a core of solid neutron absorbing material, is used for transfer operation within the Auxiliary Building. The bolted ram access penetration cover plate assembly is replaced by a two-piece neutron shield plug assembly for transfer operations from/to the Auxiliary Building to/from the HSM/HSM-HB (not shown in Figure 4.7-1). At the HSM/HSM-HB site, the inner shield plug of the neutron shield plug assembly is removed to provide access for the ram/grapple to push/pull the DSC to/from the HSM/HSM-HB.

The top cover plate is bolted to the top flange of the cask during transport from/to the Auxiliary Building to/from the ISFSI. The top cover plate assembly consists of a thick structural plate with a thin shell encapsulating solid neutron shielding material. Two upper lifting trunnions are located near the top of the cask for downending/uprighting and lifting of the cask in the Auxiliary Building. Two lower trunnions, located near the base of the cask, serve as the axis of rotation during downending/uprighting operations and as supports during transport to/from the ISFSI.

The use of a solid neutron shield is a departure from the transfer cask design described in Reference 4.1. The material selected for use as a neutron shield is BISCo Products NS-3, a shop castable, fire resistant material with a high hydrogen content. It is designed for use in shielding doors, hatches, plugs, and other nuclear applications. The solid neutron shielding material used in the cask outer annulus, top and bottom covers, and temporary shield plug, produces water vapor and a small quantity of non-condensable gases when heated above 212°F. The off-gassing produces an internal pressure which increases with temperature. As the temperature is reduced, the off-gas products are reabsorbed into the matrix, and the pressure returns to atmospheric. The annular neutron shield containment is designed for an internal pressure of 95 psig and the bottom neutron shield containment is designed for an internal pressure of 45 psig.

On an extreme ambient day with a design basis heat load, the maximum NS-3 steady state temperature is conservatively calculated to be approximately 300°F. This temperature, assumed to exist throughout the entire shield, would result in an internal cavity pressure of 60 psig. Although this pressure is well within the design allowable value for the neutron shield, pre-set safety relief valves are included to protect the neutron shield cover in the event that its design pressure is exceeded. The relief valves are set at 95 psig and 40 psig, respectively, for the annular and bottom neutron shield covers. The release of off-gas products (water vapor) does not affect the predicted neutron doses since the hydrogen content assumed in the shielding analysis is 4.38 w/o. This compares with the manufacturer's guaranteed minimum hydrogen content of 4.85 w/o and actual test samples values of as-delivered product which average about 5.1 w/o (Reference 4.5).

The basis for the relief valve setpoint values is that the design pressures for the annulus and bottom shield are 95 psig and 45 psig, respectively. These cavities are tested during fabrication to pressures of 100 psig and 50 psig, respectively. Redundant relief valves are provided in each shield for additional safety margin. The transfer cask top lid neutron shield does not require safety relief valves (SRVs) as there are three 1/2" diameter holes open to atmosphere. The relief capacity of the transfer cask neutron shield SRVs is 80.55 scfm air @ 70°F for design pressure = 100 psig and 45.44 scfm air @ 70°F for design pressure = 50 psig.

Steady-state calculations for the extreme ambient day conditions of temperature and solar heat load assume continuous exposure to 103°F ambient. Lying horizontally, the top half of the transfer cask is assumed to receive a heat flux due to insolation of 127°Btu/hr-ft^2. The resulting mass average bulk temperature of the NS-3 in the neutron shield of the transfer cask is 277°F.

The design pressures are based on tests performed by Bisco on sealed NS-3 samples. These tests show that the maximum pressure in the neutron shield annulus will be 45 psig at an NS-3 bulk temperature of 278°F. Similarly, the maximum pressure of NS-3 in the transfer cask bottom neutron shield will be 15 psig, based on an NS-3 bulk temperature of 225°F. The setpoints of both SRVs have sufficient margin so that off-gassing of NS-3 is not expected to cause the SRVs to activate even under the worst design basis conditions. Test data further showed that the off-gassing of NS-3 causes very slow pressure buildup in the sealed NS-3 samples, so the relief capacity of the transfer cask neutron shield SRVs will not be exceeded in the unlikely event of SRV actuations.

The mixing and placement of NS-3 material is controlled by the cask fabricator under the direction of Bisco Products, Inc. The hydrogen content is assured by taking samples from each batch of material prior to shipment and by taking specific gravity samples of each batch after it is mixed and before it is poured. The absence of voids is controlled by geometry of the cask neutron shield design and by the orientation of the cask during the pour to ensure that no air pockets are trapped by the NS-3 as it fills each cavity into which it is poured. Specific procedures to prevent this are developed by the cask fabricator in conjunction with Bisco and Transnuclear West.

Although the loss of this solid shield is highly unlikely, each loss of neutron shield accident case presented in Section 8.2.5.3 of Reference 4.1 has been performed for the Calvert Cliffs transfer cask design. The transfer cask is designed to provide adequate shielding to maintain the maximum radiation surface dose to less than 5 R/hr combined gamma and neutron for a cask drop accident event with loss of neutron shielding.

The neutron shield cavity has 16 support angles which form a 45° angle with the transfer cask structural liner and the outer neutron shield panel. The 45° angle of the stand-offs minimize neutron streaming through the 16 support angles. The neutron and gamma dose rates on the outside transfer cask side surface are approximately the same. Any increase in neutron dose due to streaming through these support angles will be offset by a similar decrease in gamma dose thereby keeping approximately the same total dose on the transfer cask surface.

The effect of hot spots in the NS-3 region will not be a problem because maximum pressure in the NS-3 layer is 45 psig corresponding to the maximum average temperature of 280°F. The setpoint of the transfer cask SRVs is 95 psig in the neutron shield annulus. So there is considerable margin in the pressure relief capacity of the neutron shield cavity to accommodate any possible increase in the maximum pressure in the NS-3 region due to hot spots.

The transfer cask is designated important to safety since it provides shielding and protection of the DSC from impact loads. The codes and standards used to design and fabricate the transfer cask are presented in Section 4.7.4.

4.7.3.4 Transfer Cask Lifting Yoke

The lifting yoke is a special lifting device which provides the means for performing all cask handling operations within the plant's Auxiliary Building. It is designed to support a loaded transfer cask weighing up to 109.25 tons. A lifting pin connects the Spent Fuel Cask Handling Crane hook and the lifting yoke. The lifting yoke is shown in Figure 4.7-2.

The lifting yoke is designated safety-related since it is in the direct load path of the cask. The codes and standards used to design and fabricate the lifting yoke are presented in Section 4.7.4.

The lifting yoke is a passive, open hook design with two parallel lifting beams fabricated from thick, high-strength carbon steel plate material, with a decontaminable coating. It is designed to be compatible with the Spent Fuel Cask Handling Crane hook and load block. The lifting yoke engages the outer shoulder of the transfer cask lifting trunnions. To facilitate ease of shipment and maintenance, all yoke subcomponent structural connections are bolted.

4.7.3.5 Transfer Trailer/Self-Propelled Horizontal Cask Transporter

The transfer trailer and SPMT are designed for use with the ISFSI Transfer System. Their function is to move the transfer cask and cask skid from the Auxiliary Building to the ISFSI location. Once there, the trailer or SPMT remains passive during the transfer of the DSC into the HSM/HSM-HB.

The trailer/SPMT is a commercial grade item of the type commonly used to haul very heavy loads such as transformers, boilers, and construction equipment Figure 1.3-3 shows the transfer trailer and Figure 1.3-8 shows the SPMT. The codes and standards governing the design and construction of the trailer and SPMT are provided in Section 4.7.4.

The loading sequence for the transfer cask is shown in Figures 4.7-4 and 4.7-10 where it is illustrated that the cask is never lifted above the maximum drop height of 80" after it is loaded onto the cask skid. The transfer trailer and the SPMT and other transfer equipment are shown in their configuration at the HSM and HSM-HB in Figures 4.7-5 and 4.7-11. The trailer itself is considered not important to safety since its failure would not result in a cask drop exceeding the cases evaluated in Chapters 8, 12, and 13.

The trailer is configured as a 4x2 dolly. Eight hydraulic suspensions carry four pneumatic tires each and are located two wide, in four axle lines. There are a total of 32 tires. Hydraulic suspensions enable coupled steering of all axles around a common point, thus minimizing tire scuffing and the resulting damage to pavement and tires. The suspensions also allow other advantages, such as adjustable deck height, in-situ lockout or repair of failed suspensions or tires, and automatic compensation for road surface irregularities. The trailer has all-wheel braking using industrial grade air/spring brakes.

The trailer deck height is adjustable from 35" to 52" using the trailer suspension hydraulics (transport configuration). When the trailer is resting on its hydraulic jack feet (docking configuration), the deck height is adjustable from 35.11" to 44.80" using the hydraulic jack cylinders. The distance from the trailer deck elevation to the top of the skid pad bearing plates is fixed at 5.81".

The fully raised elevation of the skid pad bearing plates is therefore 57.81" in the transport mode and 50.61" in the docking mode.

The trailer is pulled by a drawbar steering unit which is connected to the wheel groups by mechanical linkages. A hydraulic cylinder provides the motive force for remote manual steering of the trailer. The trailer may also be steered manually using a remote steering control located on a pendant. This feature allows precise control as the trailer is backed up to the HSM/HSM-HB. The pendant allows the operator the freedom to observe the trailer from the side and also reduces the operational exposure by increasing operator distance and reducing operating time.

The SPMT is composed of four major systems: controls, power generation, hydraulic, and the drive system. Using these systems, an operator can manually position the vehicle to lift a maximum load of 150 tons. Operation is powered either by using building power through an umbilical cable ("Shore") or by relying on diesel to fuel the engine ("Gen") for electrical generation.

The transporter overall dimensions are 10' wide x 32' in length, with deck dimensions of 10' wide and 24' length. The vehicle has an unloaded weight of 48,500 lbs and a 270,000 lb rated load capacity. Minimum deck height is 26.5", and can be raised 10" to an overall height of 36.5". The vehicle is also capable of reaching unloaded speeds of 110 feet-per-minute (fpm). Loaded Fast speed is 60 fpm with a Medium loaded speed of 30 fpm. 10 fpm is Creep speed.

This transporter is designed to operate on smooth surfaces, such as concrete or asphalt, with a grade of no more that 6% (loaded or unloaded).

The transporter is equipped with twelve articulating axle Wheelift assemblies, each of which is comprised of a large-bore hydraulic cylinder, two servo motors and gearboxes, and two 18" diameter x 12" wide urethane drive wheels coupled by chains and sprockets to the gearboxes. The center of the assembly contains a custom lift column designed to provide on-center rotation as well as lift and suspension functions. The vehicle has 10" vertical stroke provided by the hydraulic cylinders to accommodate the clearance needed to drive under the load, and then lift and carry the load. Steering is accomplished by varying torques and speeds of the two servo motors on any given wheel set. Hard stops limit the module's rotation to approximately +140° to -50°, while software limits further reduce the rotational angle to preset limits depending on the mode of travel. The motors are designed to propel the module at speeds ranging from 0.0 to 110 fpm.

4.7.3.6 Skid Positioning System

The functions of the SPS are to hold the cask skid stationary (with respect to the transport trailer) during cask loading and transport, and to provide alignment between the transfer cask and the HSM prior to insertion or withdrawal of the DSC. It is composed of tie down or travel lock brackets, bolts, three hydraulically powered horizontal positioning modules, four hydraulic lifting jacks, and a remotely located hydraulic supply and control skid. The SPS hardware located on the transport trailer is illustrated in Figure 4.7-6.

The codes and standards governing the design and construction of the SPS are provided in Section 4.7.4. The SPS is considered not important to safety since its failure would not result in a cask drop as severe as the cases evaluated in Chapters 8 or 12.

The skid tie-down brackets are shown in Figure 4.7-7. The brackets were designed to withstand the design basis loads for the skid which are described in Chapters 8 or 12.

The hydraulic jacks are designed to support the transfer cask setdown load, and the loads applied to them during HSM loading and unloading. They are utilized at two locations: in the Auxiliary Building during cask downending, and at the ISFSI during cask alignment and DSC transfer. At both locations, their purpose is to provide a solid support for the trailer frame and skid. Three measures are taken to avoid accidental lowering of the trailer payload: the hydraulic pump will be de-energized after the skid has been aligned (the jacks are also hydraulically locked out during operation of the horizontal cylinders); there are mechanical locking collars on the cylinders; and pilot-operated check valves are located on each jack assembly to prevent fluid loss in the event of a broken hydraulic line.

Three positioning modules provide the motive force to horizontally align the skid and cask with the HSM/HSM-HB prior to insertion or retrieval of the DSC. The positioning modules controls are manually operated and hydraulically powered. The system is designed to provide the capability to align the cask to within the specified alignment tolerance. The alignment is verified using commercially available optical survey equipment.

Anti-friction pads constructed from woven teflon pads and steel are used to reduce the force required to align the cask. These pads are commonly used as bearings for bridges, tank supports, and hydro/electric gates. Four pads are mounted to the trailer frame. Steel boxes on the skid slide on the teflon surfaces and protect them from the weather. The travel of the skid is restricted by the stroke of the hydraulic positioning cylinders. In the event of cylinder failure, the boxes will protect the skid from excessive travel.

The hydraulic power supply and controls for the SPS are located on a skid which is normally stored on the hydraulic ram utility trailer. The hydraulic pump is powered by an electric motor. Directional metering valves are used to allow precise control of cylinder motion. The SPS is manually operated and has three operational modes: simultaneous actuation of the four vertical jacks or any pair of jacks, actuation of any single vertical jack, or actuation of any one of the three horizontal actuators. Simultaneous operation of the vertical jacks and the horizontal actuators is not possible. Fourteen small hydraulic quick connect lines provide power to the seven SPS hydraulic cylinders.

The vertical distance from the transfer cask upper trunnion centerline to the top of the skid pad bearing plates is 58.12".

For the transport trailer, the maximum cask centerline height therefore occurs in the transport configuration and is 57.81" + 58.12" = 115.93". The maximum potential drop height in this configuration is therefore 115.93" - 44.50" (the transfer cask outer radius) = 71.43".

For the transport trailer in the docking configuration, the transfer cask centerline can be raised to 50.61" (refer to question 3.0-20) + 58.12" = 108.73" which gives a design margin of about 6" over the HSM centerline height of 102.00" for concrete slope or irregularities.

The SPMT does not require an SPS.

4.7.3.7 Hydraulic Ram System for the Transfer Trailer and SPMT

The HRS provides the motive force for transferring the DSC between the HSM/HSM-HB and the transfer cask (Figure 1.3-5). Since operation of the HRS cannot result in damage to the fuel inside the DSC, it is considered not important to safety. The codes and standards used in the design of the HRS are listed in Section 4.7.4.

The HRS includes the following main subcomponents: one single-stage hydraulic cylinder; one grapple assembly; one hydraulic power unit; one ram/cask support frame assembly; one tripod support assembly; hydraulic hoses and fittings; one hose reel; all necessary appurtenances, pressure limiting devices and controls for the system operation; and, one light duty trailer (for transport and storage of all HRS equipment).

The HRS is designed to grapple, push or pull the DSC at any point in the extent of its horizontal travel between the cask and the HSM/HSM-HB. The HRS is designed to apply pushing and pulling forces of 20,000 pounds during normal operation. The HRS and all other components of the transfer system are conservatively designed for off-normal pushing and pulling loads of up to 80,000 pounds.

The ram hydraulic cylinder is provided with a support and alignment system which provides for the range of vertical and lateral motion necessary for alignment with the DSC, cask, and HSM/HSM-HB. The front of the ram hydraulic cylinder is aligned using a ram trunnion support assembly, and the rear of the ram is aligned using an adjustable tripod assembly.

The ram hydraulic power unit and controls are designed to provide the range of flows and pressures as required to push or pull the DSC under normal to maximum load conditions at safe design speeds. All controls are mounted in one trailer-mounted control panel. Safety features of the control system are included to prevent the inadvertent operation of the HRS, limit the speed and force of the ram cylinder, as well as to provide an emergency means of stopping the HRS.

The equipment safety concerns are addressed using a relatively simple control system and comprehensive operational procedures. All controls are manually operated. Pre-set pressure and flow control devices ensure that the maximum design forces and speeds of the hydraulic ram are not exceeded. System pressure gauges are provided to verify that design force limits are not exceeded.

Components defined in Reference 4.1 as important to safety are those which provide containment of the fuel or biological shielding to the public and plant personnel. The components are required to be constructed according to stringent codes under a quality assurance program commensurate with the importance of their functions. The hydraulic ram does not fall into this category, but has a secondary impact on system operational safety, similar in function to the tractor which tows the trailer from the fuel building to the ISFSI.

Nonetheless, safe DSC insertion and retrieval is the primary design objective of the HRS. The force applied to the ram is under the direct control of the ram operator and is regulated by a pressure-compensated flow control valve. The ram speed is similarly controlled. The ram force available to the operator is limited by two sets of factory set and sealed relief valves which control the maximum hydraulic pressure which can be applied to the ram in both extend and retract modes.

4.7.3.8 Ram Trunnion Support Frame

As shown in Figure 1.3-5, the ram trunnion support frame is a light weight AISC structural steel frame fabricated from tube steel and plate. The frame is bolted to tapped holes located in the transfer cask bottom flange, and the hydraulic ram front trunnions are mounted into pillow blocks located on the frame. All hydraulic ram push/pull loads are transmitted into this frame directly to the base of the transfer cask, through the cask to the cask restraint system and into the HSM/HSM-HB front wall.

4.7.3.9 Cask Support Skid

4.7.3.9.1 Transfer Trailer

The cask support skid is a structural steel frame fabricated from standard American Society for Testing and Materials A36 wide flange members, built up box beam cross-members and trunnion support towers. The cask support skid, shown in Figure 1.3-4, is designed according to the AISC code for its operating loads. The cask support skid is rigidly attached to the transfer trailer by four bolted brackets during transfer from the Auxiliary Building to the ISFSI. During cask alignment, the bolts are removed, and the alignment system is used to move the cask support skid into position. For this operation, the skid is supported by four Lubrite spherical bearing pads located on the trailer cross members. The transfer cask is supported on the front and rear trunnion support tower pillow blocks. For cask downending, the lower trunnions are engaged into the front pillow blocks, and the top section of the blocks installed. The cask lifting yoke and Spent Fuel Cask Handling Crane are then used to lower the upper trunnions into the rear trunnion supports. The yoke is then removed and the trunnion and capture plates are installed.

Several changes to the generic NUHOMS-24P design neutron shielding were required in order to accommodate the higher neutron source term. The changes to neutron shield thicknesses are summarized in Table 4.1-1. Other changes to the shielding design thicknesses were also made as indicated in the CCNPP ISFSI Updated Safety Analysis Report.

4.7.3.9.2 Self-Propelled Modular Transporter

The Doerfer Companies Wheelift SPMT, complete with Transfer Cask Support Skid, is designed to accept a Transfer Cask filled with a Dry Shield Canister that is lowered onto the Transfer Cask Support Skid in a vertical orientation. In the Down-ending Position (see Figure 1), the SPMT is lowered so that the Transfer Cask Support Skid rests on the pavement. Once in position, the Transfer Cask is lowered into the rear trunnion sleeves of the Transfer Cask Support Skid. It is then rotated down into a horizontal position so that the forward trunnions rest in the forward trunnion sleeves. The SPMT is designed with locator holes to accommodate different size Skids.

4.7.4 TRANSFER EQUIPMENT

Applicable sections of the following codes and standards are specified for the design, construction, and testing of the NUHOMS ISFSI transfer equipment components.

4.7.4.1 Transfer Cask and Lifting Yoke

A. ASME (B&PV) Code, Section III, Division 1, Subsection NC "Class 2 Components," 1983 Edition through Winter 1985 Addenda used as a guide for design and fabrication. An N-stamp is not required.

B. ASME (B&PV) Code, Section III, Division 1, Appendices

C. ANSI N14.6-1978 "Special Lifting Devices for Shipping Containers Weighing 10,000 lbs or more"

D. ANSI Y14.5-1982 "Dimensioning and Tolerancing"

E. ANSI 57.9-1984 "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)"

F. ANSI N45.2-1977 "Quality Assurance Requirements for Nuclear Power Plants"

G. American Welding Society D1.1-88, "Structural Welding Code - Steel"

H. Steel Structures Painting Council (Standards)

I. Electric Power Research Institute NP-4830, "Comparison of Pad Hardness Study with Drop Test Results"

J. Crane Manufacturers' Association of America Specification No. 70-1983, "Specifications for Electric Overhead Traveling Cranes"

4.7.4.2 Transfer System Equipment

A. AISC, "Manual of Steel Construction"

B. National Electrical Code

C. National Fluid Power Association (Standards)

D. National Electrical Manufacturer's Association (Standards)

E. American Society for Testing and Materials (Standards)

F. Steel Structures Painting Council (Standards)

G. ANSI (Standards)

H. American Welding Society, D1.1, "Structural Welding Code-Steel"

TABLE 4.1-1
GENERIC NUHOMS-24P DESIGN NEUTRON SHIELDING

Neutron Shield		NUHOMS-24P Generic Design	CCNPP NUHOMS ISFSI Design
Component	Direction		
Transfer Cask	Top Axial	2.00" NS-3	3.00" NS-3
Transfer Cask	Bottom Axial	2.50" NS-3	3.50" NS-3
Transfer Cask	Radial	3.00" Water	4.00" NS-3
HSM	Door (Axial)	2.00" NS-3	10.75" Concrete[(a)]
HSM	Radial	36" Concrete	36" Concrete
DSC	All	None	None

(a) This HSM door design is an improvement of the one originally presented in the initial submittal of the CCNPP ISFSI SAR. Calvert Cliffs Nuclear Power Plant will revise the SAR to incorporate this design.

4.8 REFERENCES

4.1 Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, Nutech Engineers, Inc., NUH-002, Revision 1A, July 1989

4.2 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

4.3 Calvert Cliffs Independent Spent Fuel Storage Installation Security Plan, Baltimore Gas and Electric Company

4.4 CCNPP ES200700037-000, CCNPP ISFSI Phase IV & V Expansion

4.5 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)

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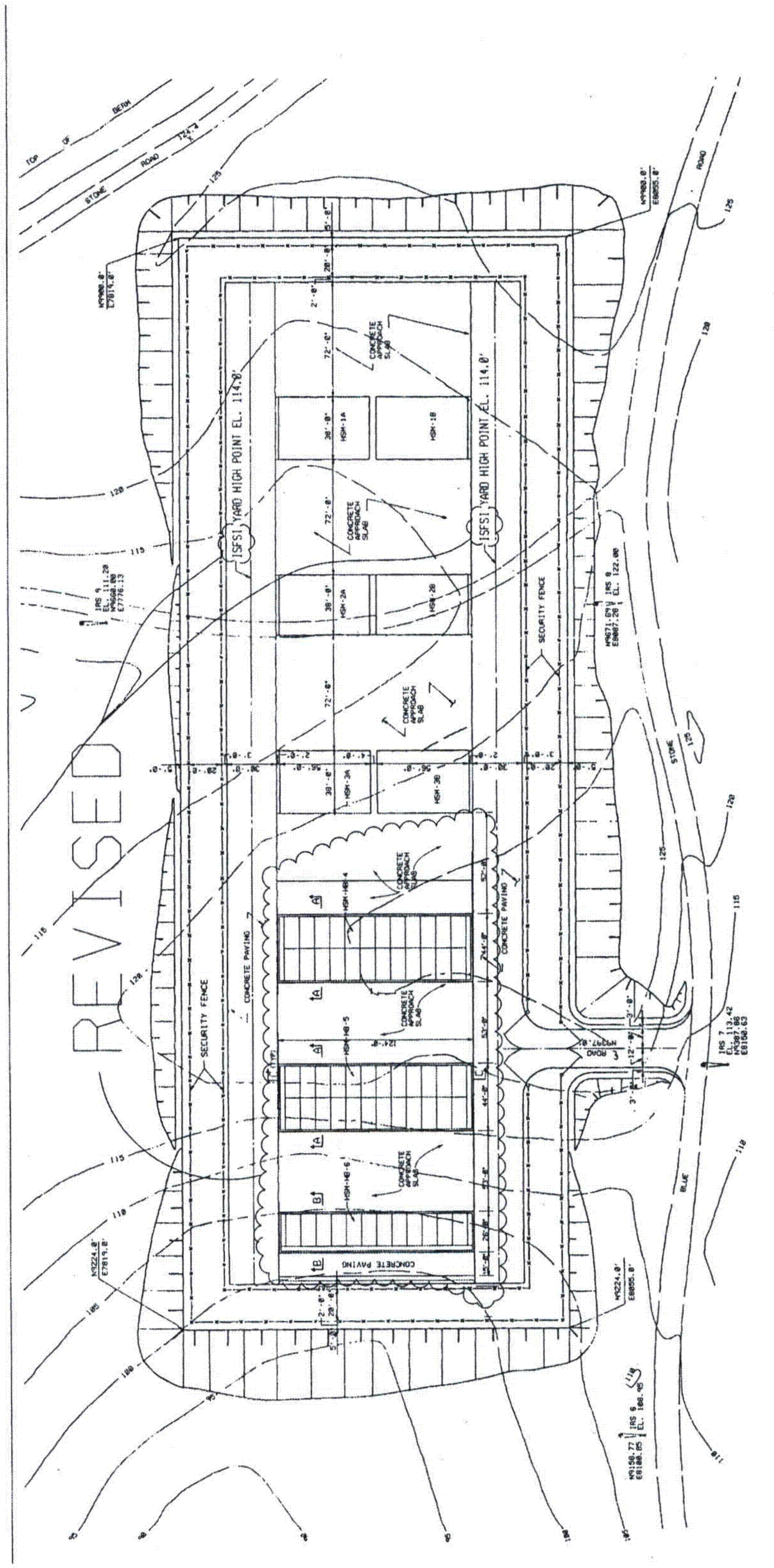
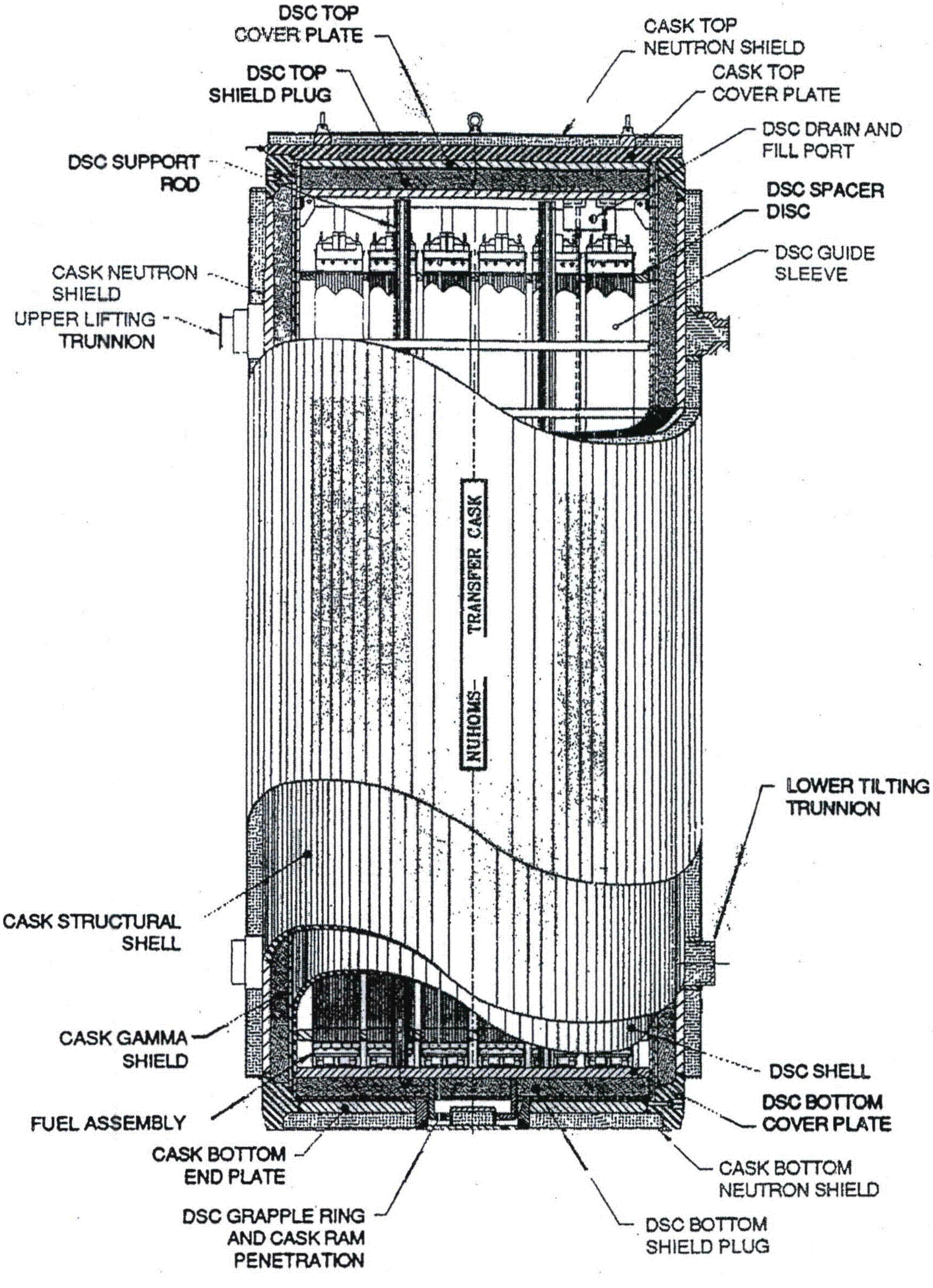
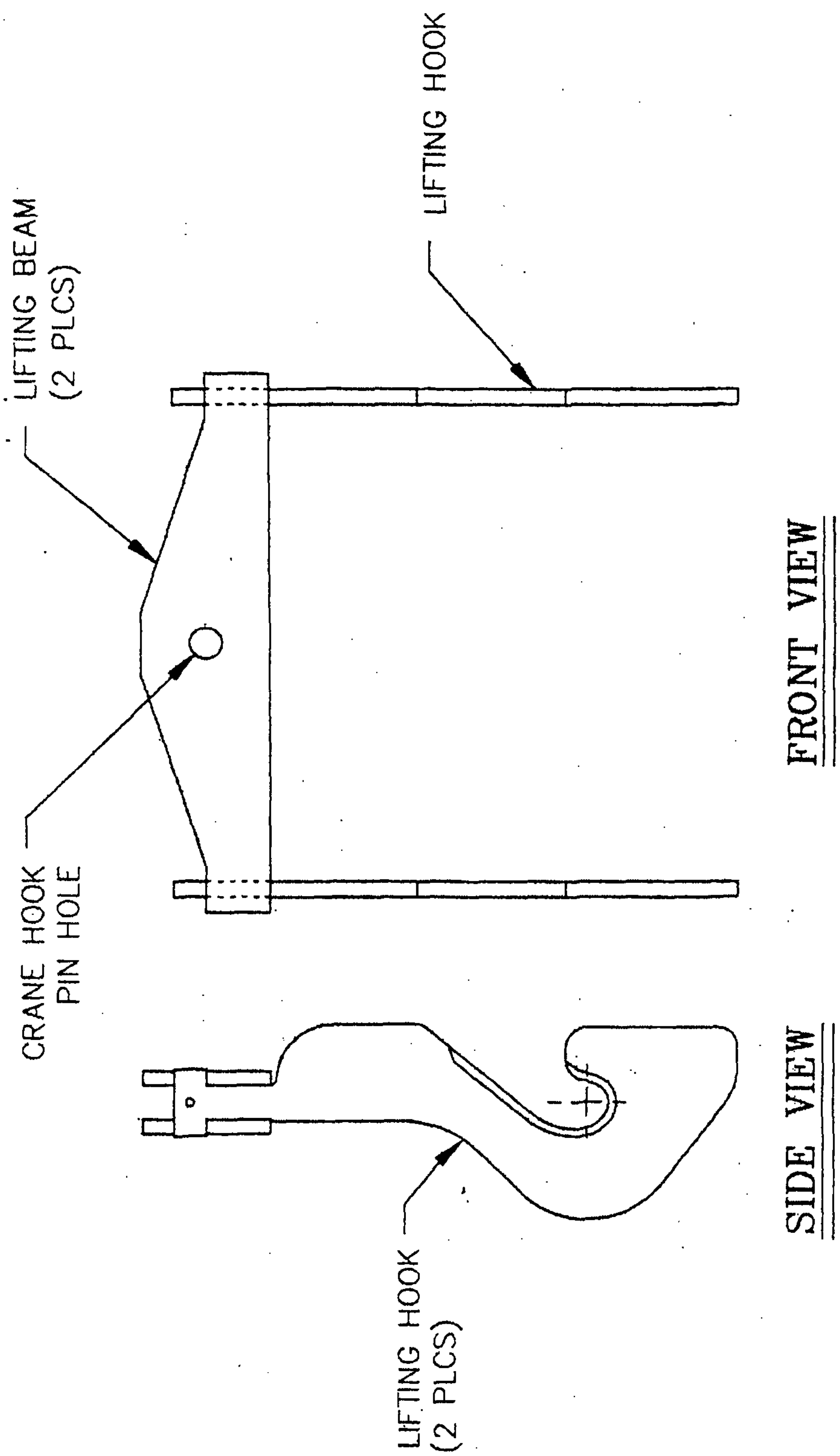


Calvert Cliffs Independent Spent Fuel Storage Installation	GENERAL LOCATION	USAR Figure 4.1-2 Rev. XX



Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	NUHOMS TRANSFER CASK (Shown with a NUHOMS-24P DSC)	USAR Figure 4.7-1 Revision 15



Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	NUHOMS LIFTING YOKE	USAR Figure 4.7-2 Revision 15

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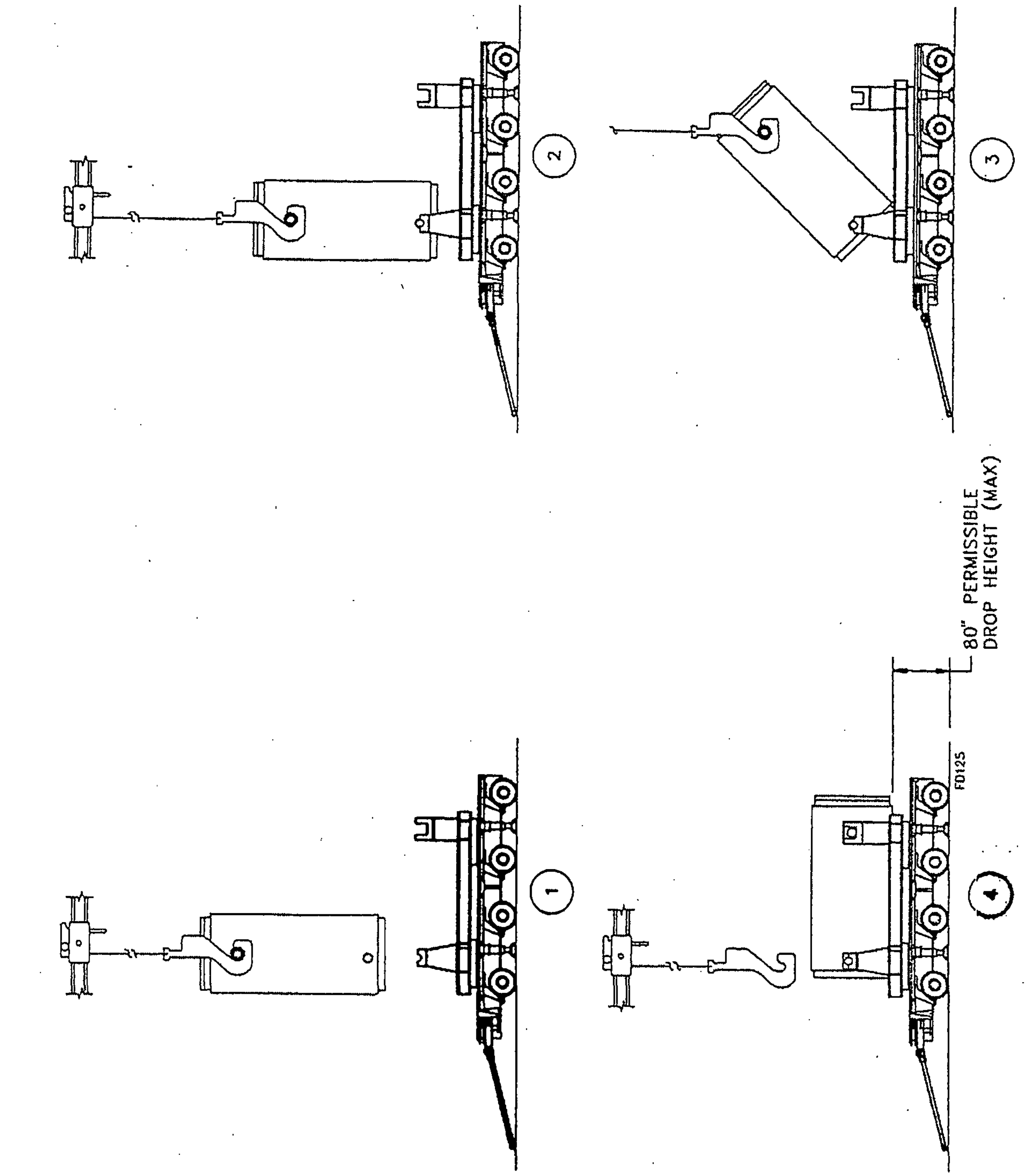
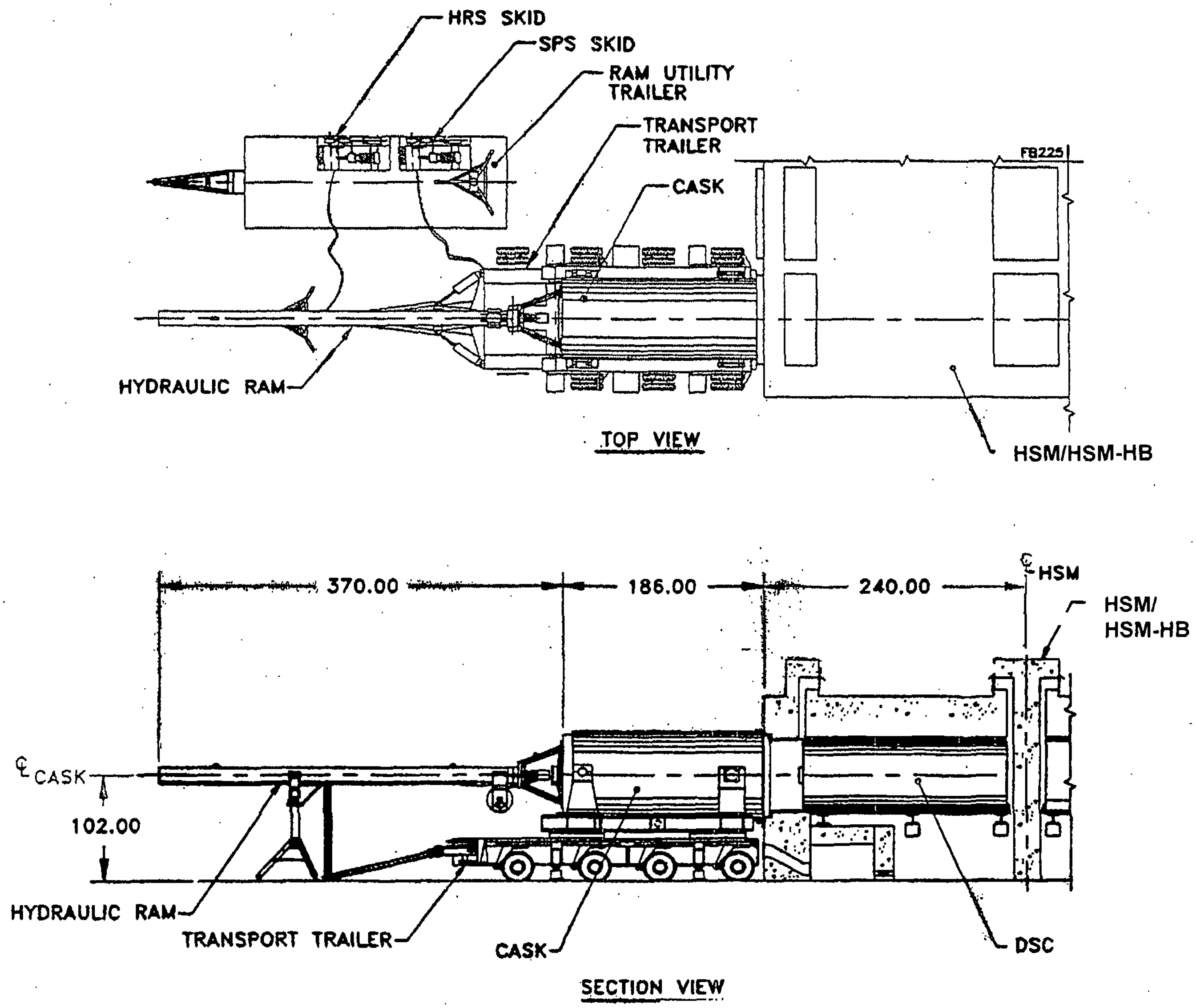
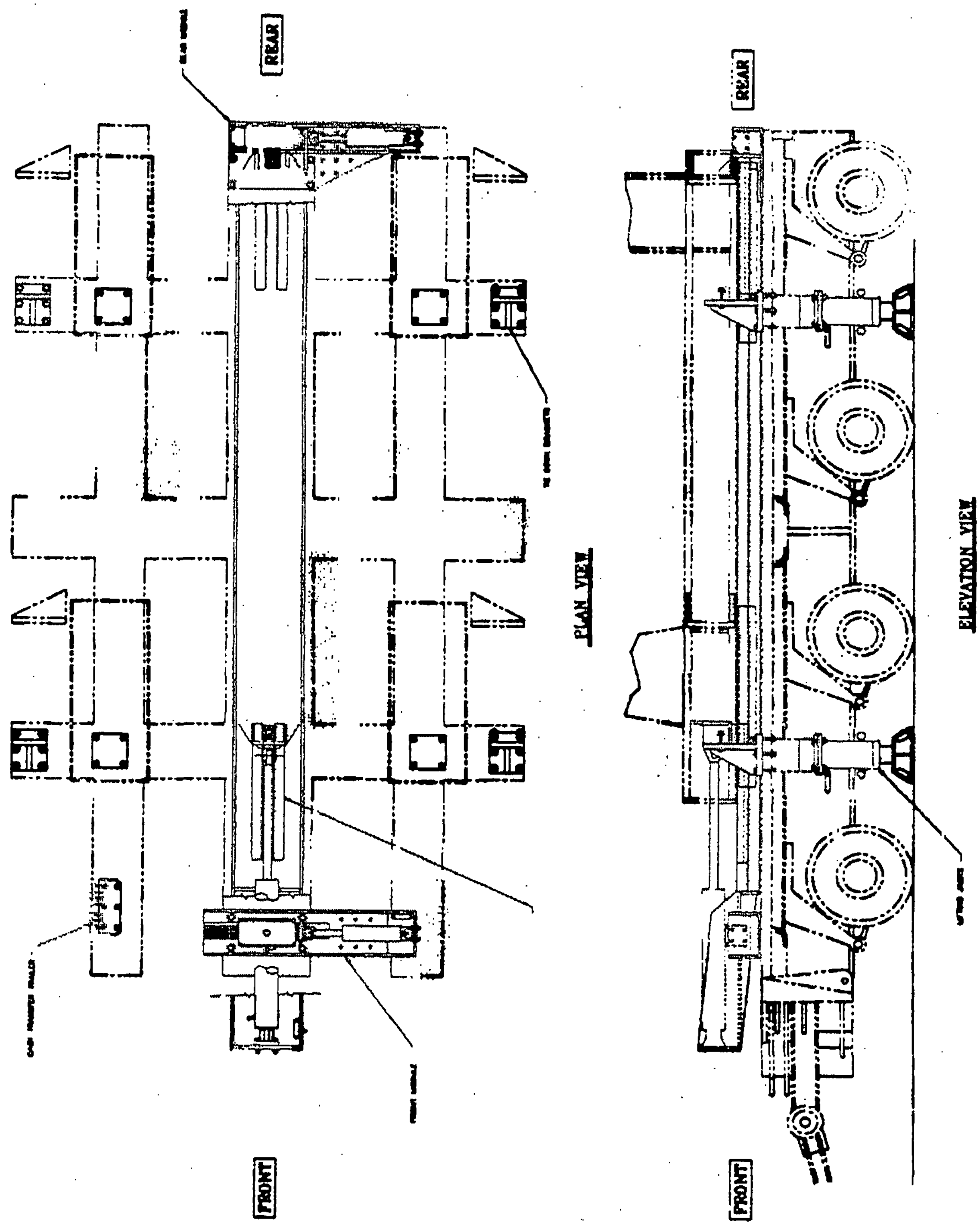


Calvert Cliffs Independent Spent Fuel Storage Installation	NUHOMS TRANSFER CASK LOADING SEQUENCE (TRANSFER TRAILER) [See Reference 4.5]	USAR Figure 4.7-4 Revision XX



Calvert Cliffs Independent Spent Fuel Storage Installation	NUHOMS TRANSFER SYSTEM CONFIGURATION (TRANSFER TRAILER) [See Reference 4.5]	USAR Figure 4.7-5 Revision 22



Calvert Cliffs Independent Spent Fuel Storage Installation	NUHOMS SKID POSITIONING SYSTEM (SPS) HARDWARE (TRANSFER TRAILER)	USAR Figure 4.7-6 Revision XX

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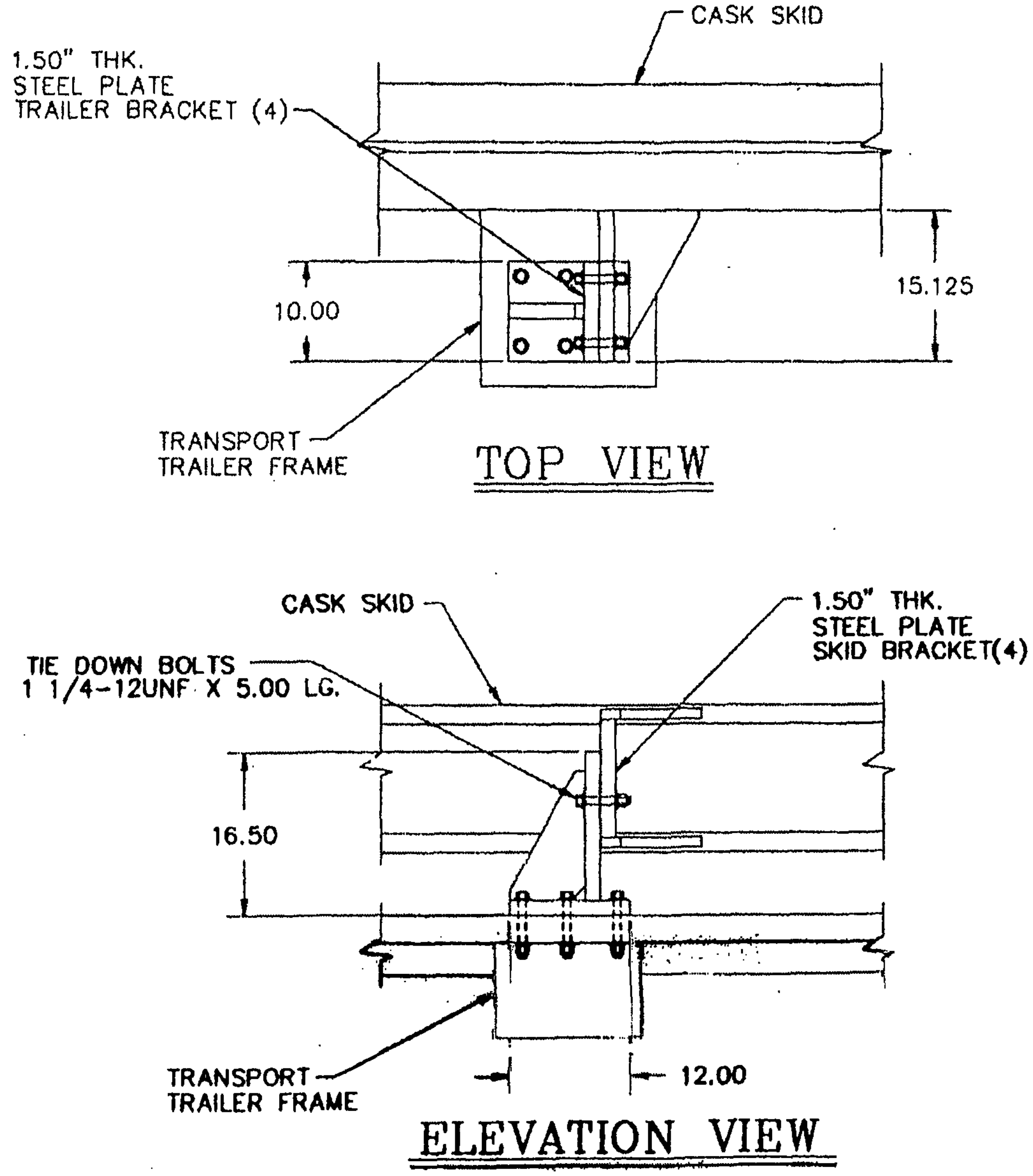

Calvert Cliffs Independent Spent Fuel Storage Installation	NUHOMS SKID TIE-DOWN BRACKET DETAIL (TRANSFER TRAILER)	USAR Figure 4.7-7 Revision XX

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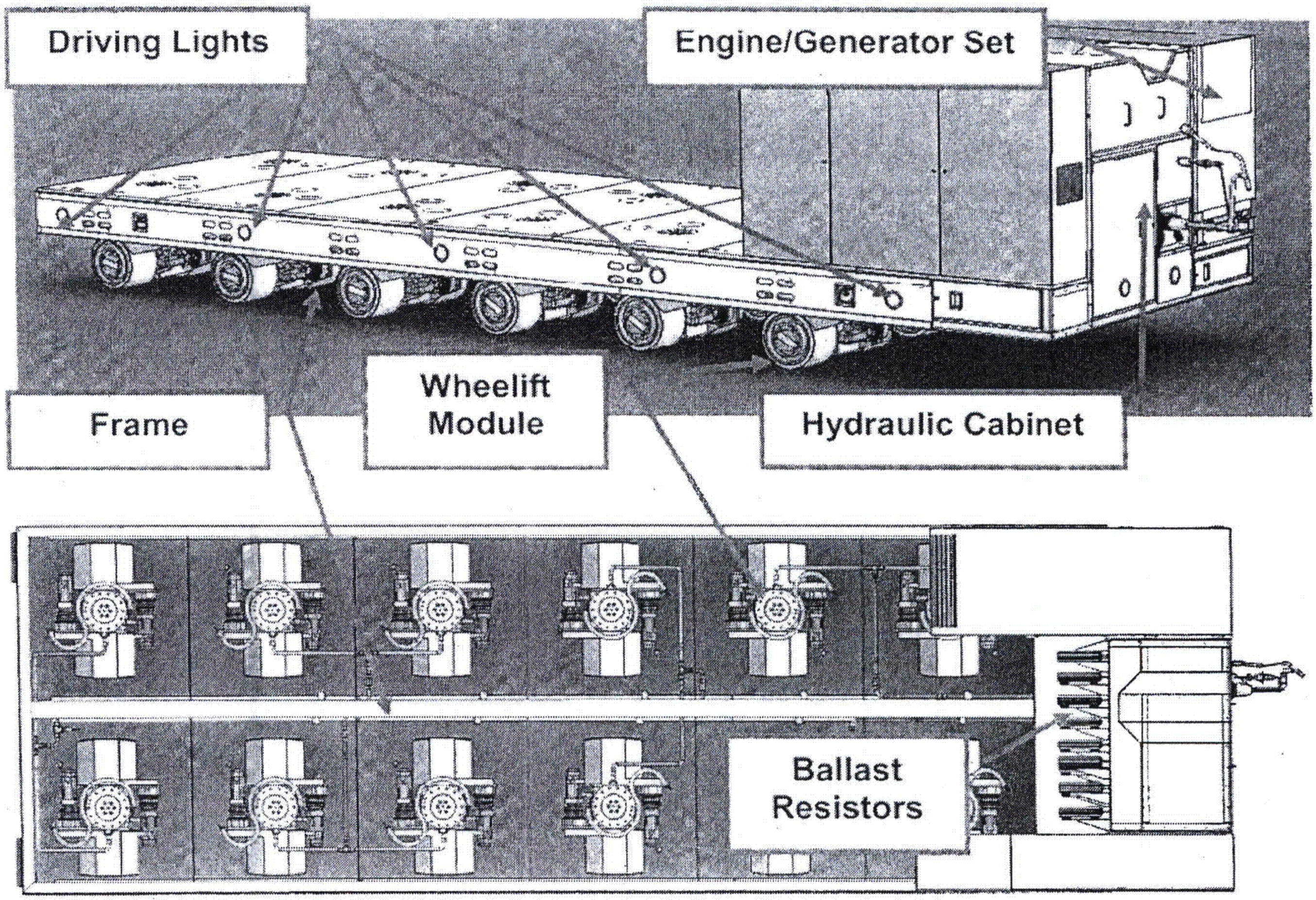

Calvert Cliffs Independent Spent Fuel Storage Installation	TRANSFER CASK LOADING SEQUENCE (SELF-PROPELLED HORIZONTAL CASK TRANSPORTER)	USAR Figure 4.7-10 Revision XX

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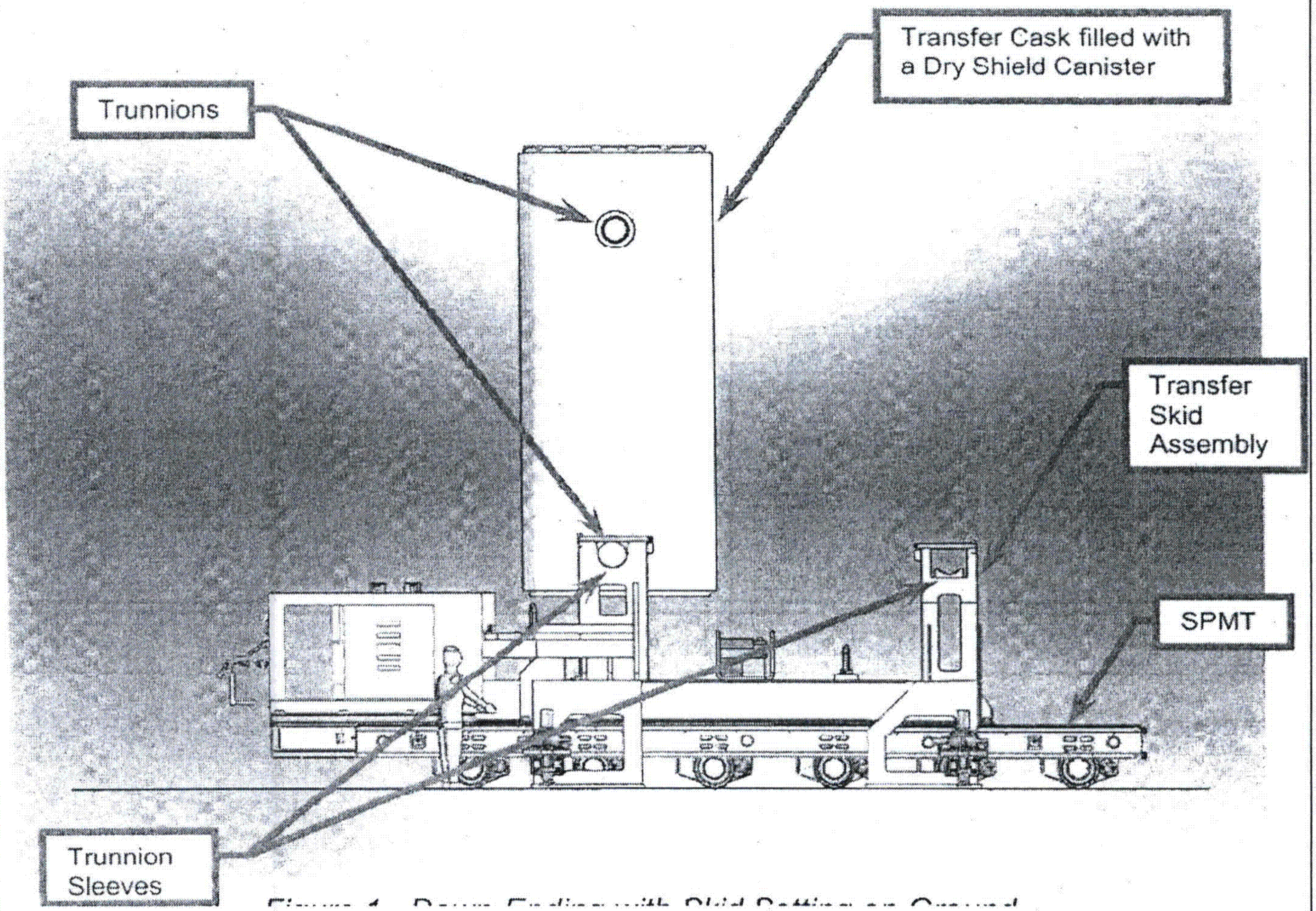

Calvert Cliffs Independent Spent Fuel Storage Installation	CASK TRANSFER CASK SYSTEM CONFIGURATION (SELF-PROPELLED HORIZONTAL CASK TRANSPORTER)	USAR Figure 4.7-11 Revision XX

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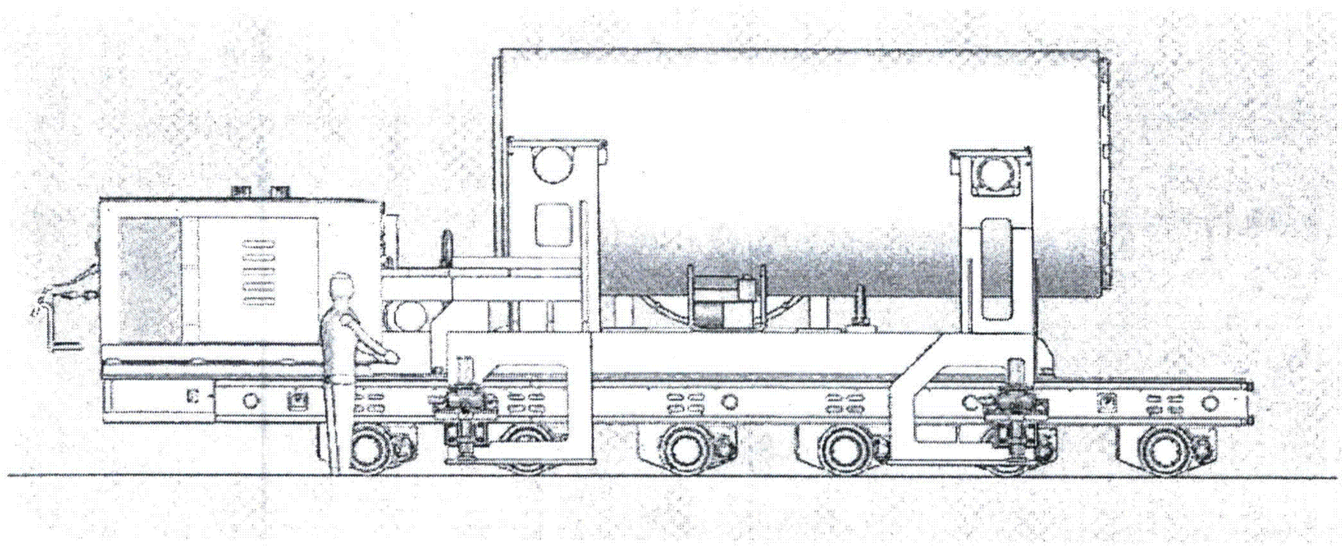
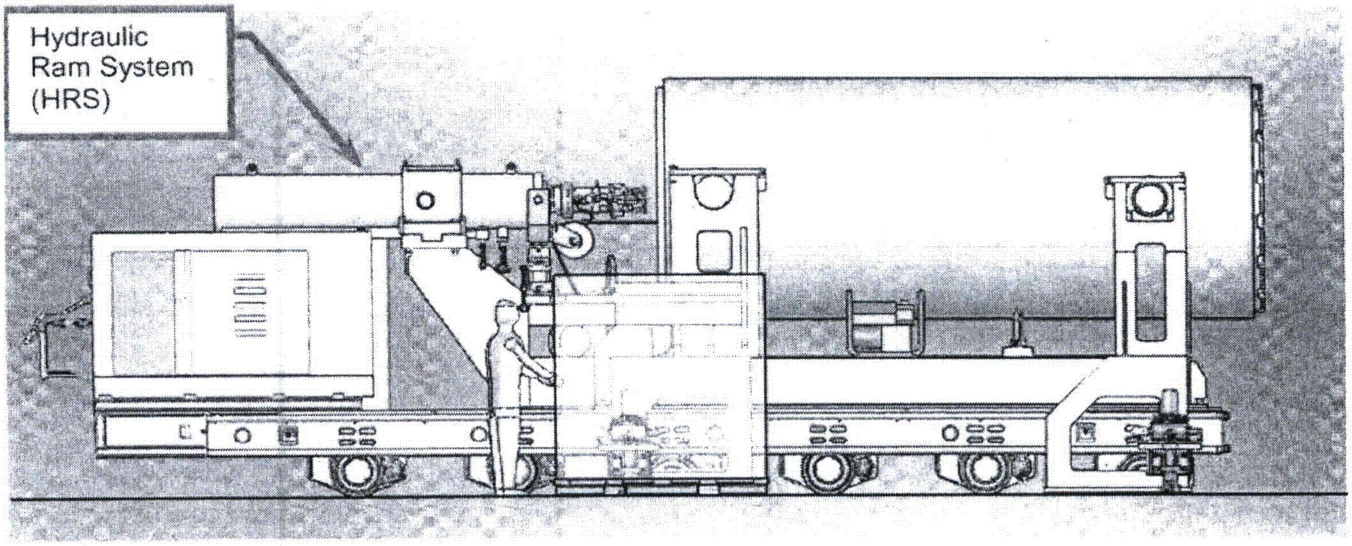


Calvert Cliffs Independent Spent Fuel Storage Installation	CASK SKID POSITIONING SYSTEM HARDWARE (SELF-PROPELLED HORIZONTAL CASK TRANSPORTER)	USAR Figure 4.7-12 Revision XX



Calvert Cliffs Independent Spent Fuel Storage Installation	CASK SKID TIE-DOWN BRACKET DETAIL (SELF-PROPELLED HORIZONTAL CASK TRANSPORTER)	USAR Figure 4.7-13 Revision XX

CHAPTER 5

OPERATION SYSTEMS

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LIST OF ACRONYMS

DSC	Dry Shielded Canister
HRC	Hydraulic Ram Cylinder
HSM	Horizontal Storage Module
HSM-HB	High Burnup Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
NUHOMS	Nutech Horizontal Modular Storage®
SPMT	Self-Propelled Modular Transporter
TCSS	Transfer Cask Support Skid
VDS	Vacuum Drying System

5.0 OPERATION SYSTEMS

5.1 OPERATION DESCRIPTION

This chapter describes the operation of the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI). The narrative describes operations unique to the Nutech Horizontal Modular Storage® (NUHOMS) systems, such as draining, drying, and closure of the dry shielded canister (DSC). Although some operational details are provided, the description is not intended to limit or restrict operation of the facility. Operational procedures may be revised according to the requirements of the plant, provided that the limiting conditions of operation are not exceeded. Standard fuel handling and cask handling operations performed under the plant's Title 10 Code of Federal Regulations Part 50 license are described in less detail.

5.1.1 NARRATIVE DESCRIPTION

The following steps are a description of the operating procedures for the NUHOMS system. They do not substitute for the procedure itself. A flowchart of these operations is provided in Figures 5.1-1 and 5.1-2.

5.1.1.1 Preparation of the Transfer Cask and the Dry Shielded Canister

Prior to storage, all candidate fuel assemblies are evaluated, using plant records or other means, to verify that they meet the physical, thermal, and environmental criteria specified in the Technical Specifications. Only fuel assemblies meeting the Technical Specification 1.0.j definition of undamaged fuel will be placed into storage in the ISFSI. Current emphasis on fuel reliability, related to operational concerns, results in identification of failed rods whenever it is believed that a significant number of failed rods could reside within the core.

When cladding defects are identified, the failed rods are removed from the fuel assemblies before reinserting them into the core. The assemblies are then no longer classified as "failed." (The failed rods are either stored in pin baskets, or encapsulation tubes, which will not be stored in the ISFSI). The fuel assemblies can then either be reinserted in the core, or after the proper cool-down time, put into storage in the ISFSI.

While a records search will confirm that those fuel assemblies designated for storage in the ISFSI are not among those that have been classified as failed, unless they have been repaired as described above, no special inspections to locate cladding defects are planned for the fuel assemblies prior to storage in the ISFSI. However, inert gas (helium or nitrogen) will be required for NUHOMS-32P DSC blow down if reactor records or fuel qualification tests results are not available to demonstrate that all fuel assemlbies to be loaded do not contain fuel rods with breached cladding (Technical Specification 1.0.i definition of intact fuel assembly).

The transfer cask is lifted from the trailer or SPMT to the 69' level of the Auxiliary Building and lowered into the cask washdown pit with the Spent Fuel Cask Handling Crane and the transfer cask lifting yoke. Scaffolding or a permanent support structure is provided to allow access to the top cover plate and the surface of the cask. The top cover plate is removed and the cask prepared for service.

The DSC is unpacked, cleaned, and examined for any physical damage which may have occurred since the receipt inspection. The DSC is lifted by the Spent Fuel Cask Handling Crane. The DSC is lowered into the cask and rotated as necessary to match the cask alignment marks.

The cask/DSC annulus is filled with clean, demineralized water. The inflatable seal is placed into the upper cask liner recess and pressurized with compressed air. The DSC cavity is filled with borated water from the fuel pool or an equivalent source. The top shield plug is placed on the DSC to ensure it fits properly and then the plug is removed.

The cask lifting yoke is positioned over the transfer cask and the cask lifting trunnions engaged. The yoke lifting hooks are properly positioned and engaged on the cask trunnions, and temporary scaffolding is moved away from the cask as necessary. The cask is lifted and positioned over the spent fuel pool. Prior to lifting the cask, the water level of the pool is adjusted as necessary to accommodate the cask/DSC volume. If the borated water in the DSC was obtained from the fuel pool, a level adjustment may not be required. The spent fuel pool boron concentration is verified prior to fuel loading.

5.1.1.2 Fuel Loading

The transfer cask is lowered into the spent fuel pool. The yoke is disengaged from the trunnions and moved clear of the cask. Using the fuel handling machine, one of the assemblies selected for storage is removed from the fuel rack and positioned over the DSC. The assembly is inserted into the basket guide sleeve according to the DSC loading plan. This process is repeated until all guide sleeves are filled. After the DSC has been fully loaded, the identity and location of each fuel assembly in the DSC is checked and recorded using an underwater TV camera or special optical equipment suitable for this purpose. When the identity of all fuel assemblies in the DSC has been verified, the shield plug assembly is positioned over the DSC, and lowered until it is properly seated.

The lifting yoke is engaged to the cask trunnions and visually verified that it is properly positioned and engaged. The transfer cask is raised, stopping vertical movement prior to breaking the surface of the pool. The top shield plug is inspected to verify that it is properly seated on the DSC. If it is not, the cask is lowered and the shield plug assembly repositioned. The cask is raised from the pool and the exposed portion sprayed with demineralized water. Any excess water is drained from the top of the DSC shield plug assembly back into the pool. The radiation levels at the center and perimeter of the top shield plug assembly and around the exposed surface of the cask are checked. The cask is lifted from the pool and moved to the cask washdown pit.

5.1.1.3 Cask/Dry Shielded Canister Drying Process

The rigging cables are disengaged from the top shield plug and the eyebolts are removed. The lifting yoke is disengaged from the trunnions and moved clear of the cask. The radiation levels along the surface of the cask are checked and it is decontaminated as necessary. Scaffolding is placed around the cask so that any point on its surface is easily accessible to personnel. The top shield plug surface and the exposed DSC shell are decontaminated and the inflatable cask/DSC annulus seal is removed. The cask drain line is connected to the cask, the cask cavity drain port is opened, and water drained from the annulus until the water level is approximately 12" below the top edge of the DSC shell. Swipes are taken around the outer surface of the DSC shell and checked for removable contamination. The top shield plug surface and exposed interior of the DSC shell above the top lead plug are dried. Radiation levels along the surface of the top shield plug are checked and temporary shielding is installed as necessary to minimize personnel exposure.

Calvert Cliffs performs a cost-benefit analysis to determine whether additional shielding is to be prescribed. The primary consideration in the analysis is the exposure incurred installing and removing the shielding vs. performing the task without shielding. While no specific radiation level is defined for implementing additional shielding, consideration is also given to the repetitiveness of the task in keeping the collective dose ALARA.

Exception is taken to a specification implied by Table 12-1 of the Nuclear Regulatory Commission Safety Evaluation Report included in Reference 1.2, limiting the time allowed to drain the DSC following removal from the fuel pool. The intent of the specification is not stated explicitly, but it can be inferred from the reference that its purpose is to provide additional margin for subcriticality by preventing reduced moderator density due to boiling. Optimum moderator density is analyzed explicitly and is assumed to occur for all design conditions. Since subcriticality is demonstrated by analysis for all conditions assuming that optimum moderator conditions apply, technical justification for such a limit does not appear to apply to the Calvert Cliffs fuel and DSC design.

The vacuum drying system (VDS) is connected to the DSC siphon and vent ports, and the liquid pump is used to pump approximately 40 to 60 gallons of water from the canister to the fuel pool in order to lower the water level in the DSC below the vent port opening. The VDS is disconnected from the DSC, and the vent port is opened to ensure that the DSC internal pressure remains atmospheric during the closure weld operation. Before welding or cutting of the top shield plug begins, a small tube is inserted into the DSC vent port. A hydrogen monitor is connected to the tube to continuously sample for hydrogen gas during the top shield plug welding/cutting process. If the hydrogen concentration reaches 60% of the lower flammability limit, welding/cutting activities will stop. The DSC air space will then be purged with filtered plant air. The top shield plug is tack welded to the DSC shell using the automatic welding machine. The shield plug seal weldment is placed and the automatic welding machine removed.

The VDS is connected to the DSC. The remaining water from the DSC cavity is removed by engaging the compressed helium supply, compressed nitrogen supply, or a compressed air source through the VDS and opening the valve to the DSC vent port, forcing the water from the DSC through the siphon port. When water stops flowing from the DSC, the siphon port valve is closed. The valve on the suction side of the vacuum pump is opened, the pump started, and a vacuum of 3 torr or less is drawn in the DSC cavity. The pressure in the DSC is reduced in steps to prevent the formation of ice in the DSC cavity or in the VDS. After pumping down to each level, the pump should be valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is reactivated and the pumpdown continued to the next step. It may be necessary to repeat some steps, depending on the rate and the extent of the pressure increase. After the DSC internal pressure is stabilized at 3 torr or less, the valve in the helium inlet is opened to allow helium to flow into the DSC. The DSC is pressurized with helium to 22 psia and the shield plug seal weld tested for leakage. After the seal welds' integrity is confirmed, the DSC is re-evacuated to 3 torr and backfilled with helium to a cavity pressure of 17.2 psia.

The maximum leakage rate is 10^{-4} atm-cc/sec. This is the lowest rate measurable for use with portable helium leak detectors. If a pressure of 1.5 atm developed within the DSC cavity for a period of 10 years, a leak rate of 10^{-4} atm-cc/sec would allow 47,300 cm^3 of helium to escape. This would be insignificant compared to the $6.75x10^6 cm^3$ of helium in the DSC initially. An alternate method of filling the DSC with a measured quantity of helium, 837 gm ± 120 gm, is acceptable.

5.1.1.4 Dry Shielded Canister Closure Operations

The VDS is disconnected from the DSC and the prefabricated plugs are seal welded over the DSC vent and siphon port openings. The top cover plate is placed over the shield plug. After proper fit-up between the plate and the DSC shell is verified, the top cover plate is tack welded to the shell using the automatic welding machine. The cover plate final closure weld is placed. The automatic welding machine is removed from the DSC. The cask drain port valve is opened and the remaining water removed from the cask/DSC annulus. The transfer cask cover plate is rigged, lowered onto the transfer cask, and bolted into place.

5.1.1.5 Transport of Cask to Transfer Trailer Skid and Horizontal Storage Module

When the transfer trailer or SPMT and cask support skid are prepared for service, the scaffolding is moved away from the cask as necessary, and the lifting yoke is engaged with the cask trunnions. The cask is lifted from the cask washdown pit, moved to a location over the transfer trailer or SPMT, and lowered until the lower trunnions rest in the cask support skid pillow blocks. The crane is moved forward while the cask is lowered until the cask upper trunnions are just above the skid upper trunnion pillow blocks. The positioning of the cask is inspected to ensure that the upper trunnions and pillow blocks are properly aligned, and the cask is lowered until the upper trunnions rest on the pillow blocks. The trunnions are inspected to ensure that they are properly seated and then the top halves of the pillow blocks are installed. The bottom ram access cover plate is removed from the cask and the temporary shield plug is installed. The ram support frame is installed on the bottom of the transfer cask.

5.1.1.6 Loading of Dry Shielded Canister into the Horizontal Storage Module

Prior to moving the cask from the Auxiliary Building, the horizontal storage module (HSM) door is removed from the HSM/HSM-HB (high burnup horizontal storage module) using a portable crane and the HSM/HSM-HB interior is inspected for debris. The doors on adjacent HSMs/HSM-HBs should remain in place. The air inlets and outlets are inspected to ensure that they are clear and free of debris, and the inlet and outlet screens are inspected for damage.

A suitable truck, tractor or SPMT is used to transport the cask to the ISFSI along the designated route. Once at the ISFSI, the trailer or SPMT is backed to within a few feet of the HSM/HSM-HB, and the position of the trailer checked to ensure that the centerlines of the cask and the HSM/HSM-HB approximately coincide. The trailer or SPMT is repositioned if necessary. The cask top cover plate is removed using a portable crane.

5.1.1.6.1 Dry Shielded Canister Loading When Utilizing Transfer Trailer

The trailer is backed to within a few inches of the HSM/HSM-HB, the trailer brakes set and the tractor disengaged. The towing vehicle is driven away from the transfer trailer to make room for the hydraulic ram. The hydraulic ram trailer is positioned close to the transfer trailer. The skid positioning system power unit is connected to the positioning system hydraulic panel on the trailer, and powered up. The skid tie-down bracket fasteners are removed and the skid positioning system is used to bring the cask into approximate vertical and horizontal alignment with the HSM/HSM-HB. Optical survey equipment and targets on the cask and the HSM/HSM-HB are used to adjust the position of the cask until it is properly aligned with the HSM/HSM-HB. The longitudinal actuator is used to fully insert the cask into the HSM/HSM-HB opening.

When the cask is aligned and docked at the HSM/HSM-HB, the skid positioning system power unit is powered down and the cask upper trunnions are secured to the front wall of the HSM/HSM-HB with the cask restraints. A portable crane is used to pick the hydraulic ram from its trailer and place it in position behind the cask, with its front trunnions in the ram support frame pillow blocks. The ram rear tripod is placed on the approach slab in approximate horizontal alignment with the cask and the ram leveled. The center section of the shield plug is removed from the bottom of the cask.

5.1.1.6.2 Dry Shielded Canister Loading When Utilizing SPMT

The SPMT System is driven to the HSM. The HSM cover is removed, exposing the final destination of the DSC. The SPMT is lowered so that the Transfer Cask Support Skid (TCSS) rests on the pavement. The TCSS is a large weldment with a capacity of 118 tons, designed to hold the Transfer Cask and DSC during transfer, loading, and unloading operations. The TCSS includes aluminum bronze trunnion inserts and a Gleason Reel for electrical cable storage. This framework also supports the Hydraulic Ram Cylinder (HRC) Assembly and a Hydraulic Tilt Cylinder Assembly. The HRC Assembly consists of the HRC and the Grapple Assembly. The HRC is a telescopic three stage double acting cylinder that can advance approximately 21 feet.

Each of these assemblies has chamfered locator blocks to guide each assembly into proper position when loading/connecting the hydraulic ram assembly to the TCSS. Mounted to the TCSS, is a hydraulic and electric line connection plate. This connection plate houses the quick disconnect fittings for the Ram and Tilt hydraulic cylinders and for the Grapple assembly cable.

The SPMT is then backed up and repositioned in relation to the TCSS. This allows the Skid to be positioned closer to the front of the SPMT for transfer of the Dry Shield Canister into the HSM. The SPMT must be positioned so that the two pins on the bottom of the Skid engage a hole and a slot on the top deck of the SPMT. The SPMT lifts the TCSS again and advances toward the HSM. The SPMT is manually aligned with the HSM. The Transfer Cask is then secured to the wall of the HSM with end user supplied equipment.

5.1.1.6.3 Insertion of Cask INTO HSM/HSM-HB from Transfer Trailer/SPMT

The ram hydraulic power supply is powered up and the ram extended through the bottom cask opening into the DSC grapple ring. The grapple hydraulic cylinder is activated to engage the grapple arms with the DSC grapple ring. The DSC is inserted into the HSM/HSM-HB by extending the hydraulic ram. The ram is stopped when the DSC reaches the support rail stops at the back of the module.

The ram grapple mechanism is disengaged and the hydraulic ram retracted and removed from the cask. The cask restraints are removed from the HSM/HSM-HB, and the cask skid retracted to its travel position. The DSC seismic restraint is installed, the HSM/HSM-HB door closed, and the door tack welded in place. The radiation dose rates are measured at key locations on the HSM/HSM-HB. The transfer cask top cover plate and ram access shield plug are replaced. The skid is secured onto the trailer/SPMT, the trailer jacks retracted, the skid positioning system disconnected and the trailer and cask towed or SPMT located to the designated equipment storage area.

5.1.1.7 Monitoring Operations

The HSM air inlet and outlet vents are inspected once every 24 hours to ensure that no debris is obstructing the air flow through the module.

5.1.1.8 Unloading the Dry Shielded Canister From the Horizontal Storage Module

All preliminary operations prerequisite to loading the DSC into the cask (cleaning, etc.) will be performed prior to transporting the cask to the HSM/HSM-HB. The tack welds securing the HSM/HSM-HB door will also be removed.

The cask is placed onto the support skid and trailer or SPMT and then moved to the HSM/HSM-HB. The ram support frame is installed if it has been removed from the cask and the temporary shield plug installed over the ram access penetration. The trailer or SPMT is backed to within a few feet of the HSM/HSM-HB and the cask top cover plate removed. The HSM/HSM-HB door and the DSC seismic restraint are removed. The trailer or SPMT is backed to within a few inches of the HSM/HSM-HB so that the cask is roughly aligned with the module. The skid tie-down brackets are removed and the cask brought into vertical and horizontal alignment with the HSM/HSM-HB. The cask is inserted into the HSM/HSM-HB recess and the cask restraint installed between the HSM/HSM-HB front wall and the cask trunnions. The hydraulic ram system is aligned with the cask and the center ram access penetration temporary shield plug removed. The ram is extended through the cask into the HSM/HSM-HB until it is inserted in the DSC grapple ring. The arms on the ram grapple mechanism are activated and the grapple ring engaged. The ram is retracted and the DSC pulled into the cask. The grapple arms are disengaged and the ram fully retracted from the cask. The ram access penetration temporary shield plug is replaced. The ram and ram mounting assembly are removed. The skid is restored to its travel lock position and secured with the skid tie-down brackets. The HSM/HSM-HB door is replaced and the cask cover plate placed onto the cask and bolted in place. When the transfer trailer is used for cask transport, the trailer jacks are raised and the trailer pulled away from the HSM/HSM-HB. When the SPMT is used for cask transport, the brakes are disengaged and the vehicle moves away from the HSM/HSM-HB.

5.1.1.9 Removal of Fuel from the Dry Shielded Canister

It is Calvert Cliffs Nuclear Power Plant, Inc.'s intent to ship intact DSCs to the repository. Since the issue of compatibility between DSCs and transport casks is not yet resolved, procedures are included here to show

how fuel can be returned to the spent fuel pool should that become necessary.

If a loaded DSC is dropped, then the DSC, transfer cask or fuel may have been damaged and an examination of these items shall be conducted.

The cask trailer is towed or the SPMT is moved from the ISFSI to the Auxiliary Building. The trailer or SPMT is positioned and readied for access by the Spent Fuel Cask Handling Crane. The cask bottom cover plate is installed and the upper halves of the skid trunnion support pillow blocks removed. The lifting yoke is attached to the crane hook, the yoke lowered and the cask upper trunnions engaged. The crane is moved toward the bottom of the cask while raising the hook, and the cask lifted from the trailer or SPMT. The cask is moved to the cask washdown pit and lowered. The cask is washed to remove any dirt which may have accumulated on it during the DSC loading and transfer operations. The cask lid is unbolted and removed and set aside. Temporary shielding is installed as required to reduce personnel exposure.

The process of DSC unloading is similar to that used for loading. Dry shielded canister opening operations described below are carefully controlled in accordance with current and approved plant procedures. This operation is performed under the Calvert Cliffs Nuclear Power Plant's standard health physics guidelines for welding, grinding, and handling of potentially highly contaminated equipment. This includes the use of prudent housekeeping measures and monitoring of airborne particulates.

The automatic welding machine, with the plasma torch attached, is placed on the top cover plate and aligned with the cover plate weldment. If necessary, an exhaust hood or tent is placed over the DSC, and the exhaust routed to the existing Auxiliary Building processing systems. The seal weld between the top cover plate and the DSC shell is removed using the plasma arc-gouging system. The Auxiliary Building's ventilation system should be operating at all times. Very little swarf is produced by the plasma arc torch, but all such material will be treated and handled in accordance with the plant's low level waste procedures. The top of the tent or exhaust hood is removed, if necessary, and the automatic welding machine is removed. The DSC top cover plate is removed. A portable drill press is placed on the top of the DSC shield plug and the drill positioned over the siphon port quick connect. A hole is drilled through the siphon port plug to expose the quick connect. The drill press is repositioned over the vent port and a second hole drilled to expose the vent port quick connect. Care should be taken to avoid drilling into the quick connects. The vent port is connected to the building processing system and the DSC filled with borated water from the fuel pool through the siphon port. The cask/DSC annulus is filled with demineralized water. The automatic welding machine, the tent, and the temporary shielding are reinstalled as required. Before welding or cutting of the top shield plug begins, a small tube is inserted into the DSC vent port. A hydrogen monitor is connected to the tube to continuously sample for hydrogen gas during the top shield plug welding/cutting process. If the hydrogen concentration reaches 60% of the lower flammability limit, welding/cutting activities will stop. The DSC air space will then be purged with filtered plant air. The seal weld between the top shield plug and the DSC shell is removed in the same manner as the top cover plate.

The cask surface is cleaned of any dirt and debris which may have accumulated during the weld removal operation. The cask/DSC annulus seal is installed. The yoke is positioned over the cask, the eyebolts installed into the shield plug assembly, and a sling cable connected to the eyebolts. The shield plug assembly is slowly lifted to verify that it is easily removable from the DSC. The shield plug assembly should be lifted no higher than the top of the DSC shell. The shield plug is replaced in the DSC and the yoke engaged onto the cask trunnions. The lifting hooks are visually inspected to ensure that they are properly positioned on the trunnions. The cask is moved to above the fuel pool. Prior to lowering the cask into the pool, the pool water level is adjusted as necessary to accommodate the volume which will be displaced by the cask during the operation. The cask is lowered into the spent fuel pool. The shield plug is lifted out and the fuel assemblies are placed in the pool storage racks.

5.1.2 PROCESS FLOW DIAGRAM

Process flow diagrams for the handling operations are presented in Figures 5.1-1 and 5.1-2. The location of the operations are shown at the top of the figures.

5.1.3 IDENTIFICATION OF SUBJECTS FOR SAFETY ANALYSIS

5.1.3.1 Criticality Prevention

Criticality in the DSC is prevented through a combination of geometrical separation of the fuel assemblies, the neutron absorption capability of the guide sleeves, and administrative controls on selection of fuel to be stored in the DSC. Administrative control of fuel selection is incorporated into plant procedures and is described in Chapter 9. The criticality analysis is described in Sections 3.3.4, 12.3.3.4, and 13.3.3.4.

5.1.3.2 Chemical Safety

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

5.1.3.3 Operation Shutdown Modes

The NUHOMS system during storage operations at the ISFSI is totally passive and therefore, this section is not applicable for storage.

The 32PHB DSC is designed for forced cooling during transport from the pool to ISFSI pad over certain heat loads. This operation is discussed in Section 13.x.x. Heidi to include paragraph here or in Chapter 13.

5.1.3.4 Instrumentation

The instruments used for DSC closure operations and for monitoring the DSC insertion operation at the HSM/HSM-HB are standard industry components. They are selected to meet the design requirements of the systems of which they are a part. No instrumentation is required for operation of the ISFSI.

5.1.3.5 Maintenance Techniques

The NUHOMS system is totally passive and therefore does not require maintenance. However, to insure that the ventilation airflow is not interrupted, the HSM/HSM-HB is inspected once every 24 hours for debris in the vent inlet or outlet openings.

Periodic inspection and maintenance of the transfer equipment is performed.

5.2 SPENT FUEL HANDLING SYSTEMS

5.2.1 SPENT FUEL RECEIPT, HANDLING, AND TRANSFER

Spent fuel receipt, handling, and transfer for the Calvert Cliffs ISFSI take place in the plant Auxiliary Building. Fuel handling equipment includes the Spent Fuel Cask Handling Crane, spent fuel handling machine, transfer cask, and lifting yoke. The Spent Fuel Cask Handling Crane and the spent fuel handling machine are described in Reference 5.1. The transfer cask is described in Section 4.7.3.3 and the lifting yoke is described in Section 4.7.3.4.

Section 5.1.1 gives a detailed description of the fuel retrieval process and Figure 1.3-6 depicts the fuel handling operations for the ISFSI. The DSC is placed into the transfer cask which is lowered into the spent fuel pool using the Spent Fuel Cask Handling Crane and the lifting yoke. The fuel is retrieved from the storage racks by the spent fuel handling machine, and placed in the DSC. The transfer cask is then removed from the pool and taken to the cask washdown pit where decontamination and DSC closure operations take place. The cask is then lifted to the transfer trailer or SPMT for transport to the ISFSI.

Shielding is maintained by the pool water and the transfer cask. Subcriticality is assured by the administrative procedures described in Section 9.4.1, and by the DSC design as discussed in Sections 3.3.4, 12.3.3.4, and 13.3.3.4.

5.2.2 SPENT FUEL STORAGE

The transfer of the spent fuel assemblies to the ISFSI is achieved using the transfer cask, transfer trailer or SPMT, cask support skid, skid positioning system, and the hydraulic ram system. The transfer cask is described in Section 4.7.3.3, the transfer trailer and SPMT are described in Section 4.7.3.5, the skid positioning system is described in Section 4.7.3.6, the hydraulic ram systems are described in Section 4.7.3.7, and the cask support skids are described in Section 4.7.3.9.

Section 5.1.1 gives a detailed description of the transfer of the fuel assemblies to the storage position. After the DSC is loaded with fuel, the transfer cask containing the DSC is secured to the cask support skid on the transfer trailer or SPMT. The cask support skid is fastened to the trailer or SPMT during transportation to the ISFSI using skid tie down brackets. A suitable tractor is used to transport the trailer to the ISFSI. When the tractor and trailer are used for DCS transport the transfer cask is aligned with the HSM/HSM-HB using the skid positioning system. The DSC is inserted into the HSM by the hydraulic ram when the transport trailer or SPMT are used.

The only surveillance required for the NUHOMS system is an inspection of the air inlets and outlets performed once every 24 hours as discussed in Section 5.1.1.7.

Retrieval of the DSC from the HSM/HSM-HB is performed in a manner similar to the storage operations, using the same equipment. Retrieval operations are described in Section 5.1.1.8.

5.3 OTHER OPERATING SYSTEMS

5.3.1 OPERATING SYSTEM

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

5.3.2 COMPONENT/EQUIPMENT SPARES

As stated in Section 8.2, the Calvert Cliffs ISFSI, including the air outlet shielding blocks, is designed to withstand all events including the effects of tornado missiles. Therefore, no component or equipment spares are required for the NUHOMS system.

5.4 OPERATION SUPPORT SYSTEMS

The NUHOMS system is a self-contained system and requires no effluent processing systems during operations.

5.4.1 INSTRUMENTATION AND CONTROL SYSTEMS

There are no instrumentation and control systems used during storage. The instrumentation and controls necessary for fuel handling and HSM/HSM-HB loading are described in Section 5.1.3.4.

5.4.2 SYSTEM AND COMPONENT SPARES

Since there are no instrumentation and control systems during storage, no system component spare parts are required.

5.5 CONTROL ROOM AND CONTROL AREAS

There is no Control Room or control areas for the NUHOMS system.

5.6 ANALYTICAL SAMPLING

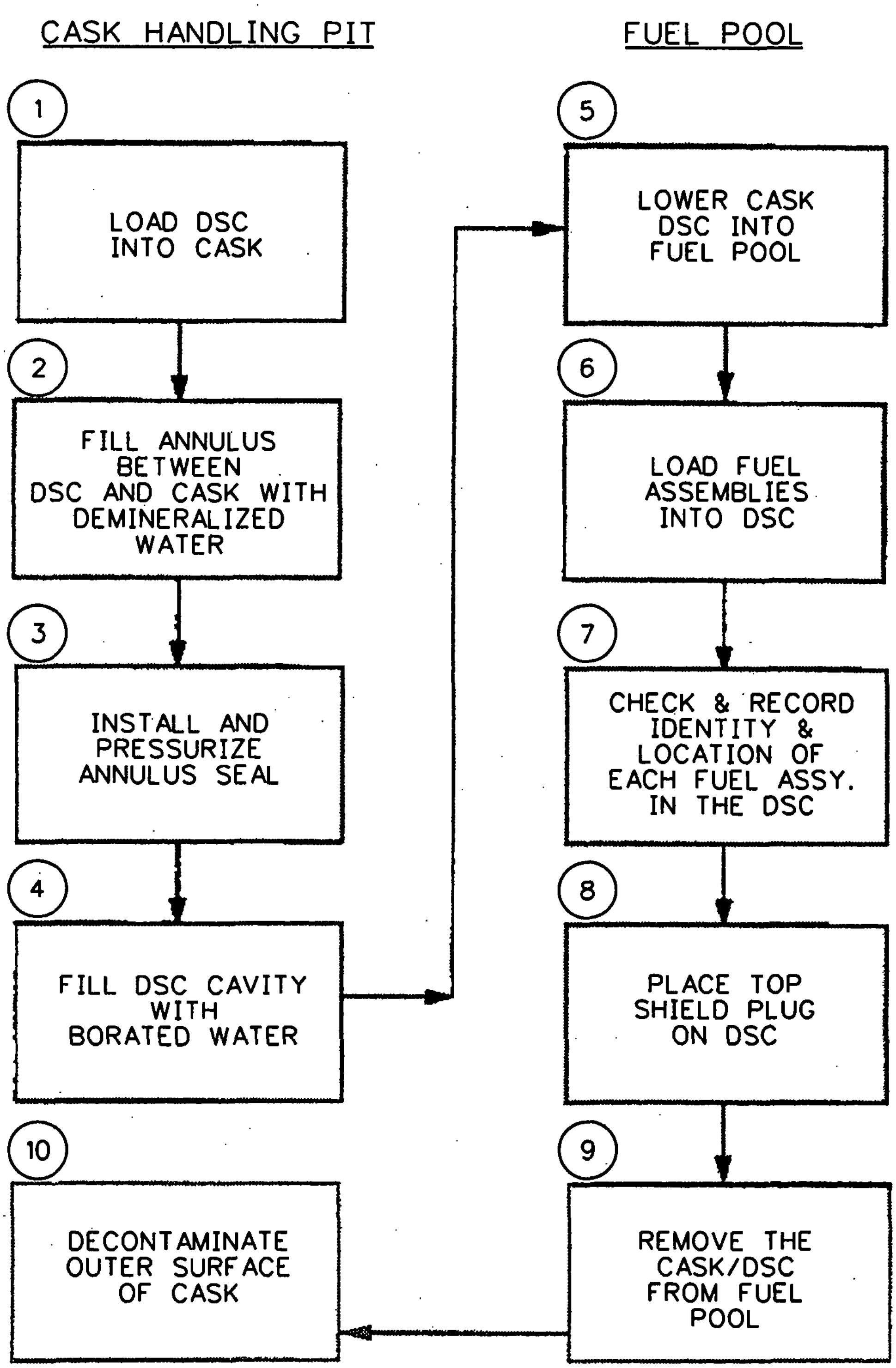
There is no analytical sampling used with the NUHOMS system.

5.7 REFERENCES

5.1 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

5.2 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)

DRAFT



Calvert Cliffs Independent Spent Fuel Storage Installation	ISFSI LOADING OPERATIONS FLOWCHART Page 1 of 3 [See Reference 5.2]	USAR Figure 5.1-1 Rev. XX

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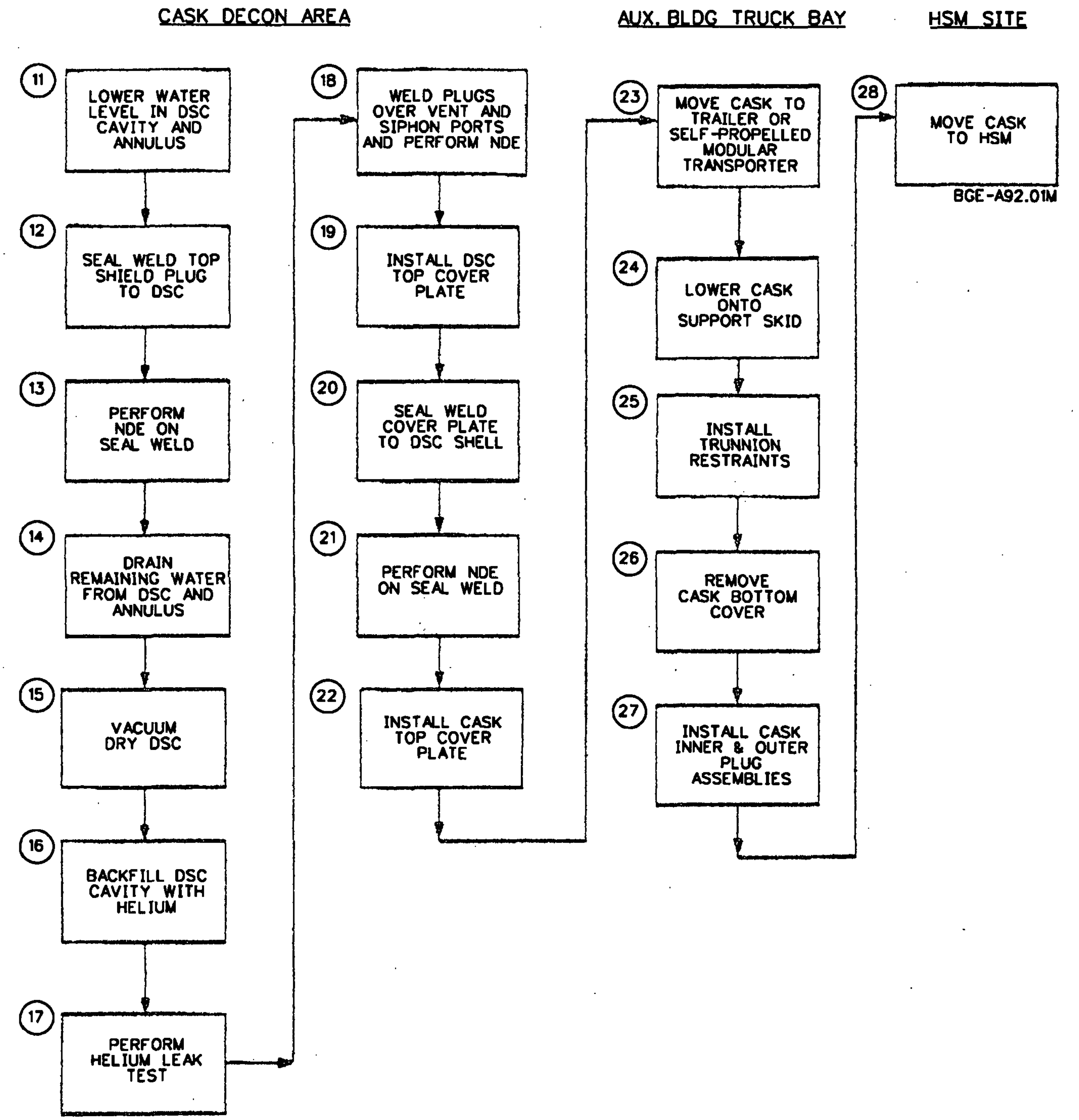

Calvert Cliffs Independent Spent Fuel Storage Installation	ISFSI LOADING OPERATIONS FLOWCHART Page 2 of 3 [See Reference 5.2]	USAR Figure 5.1-1 Rev. XX

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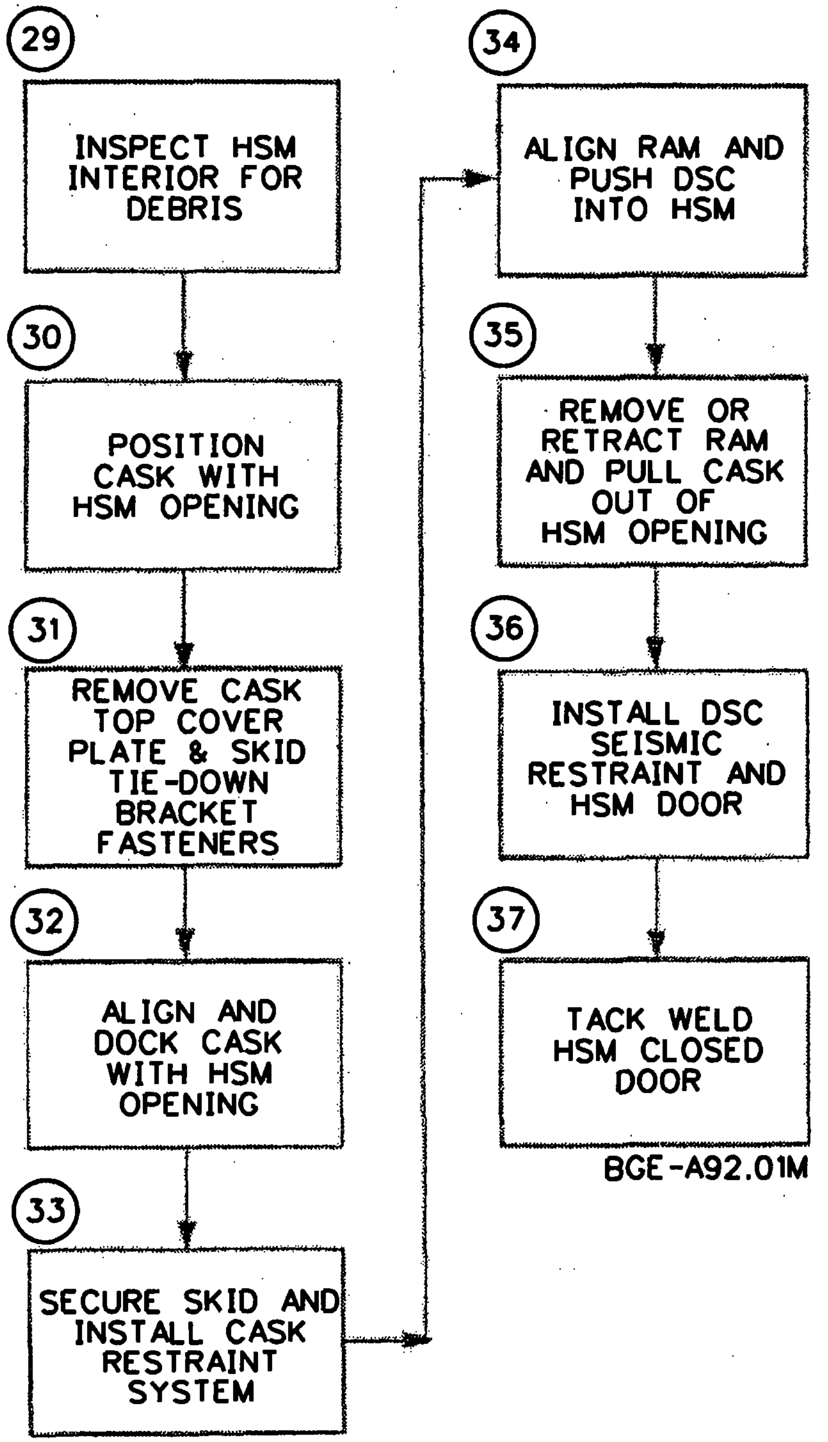
HSM SITE



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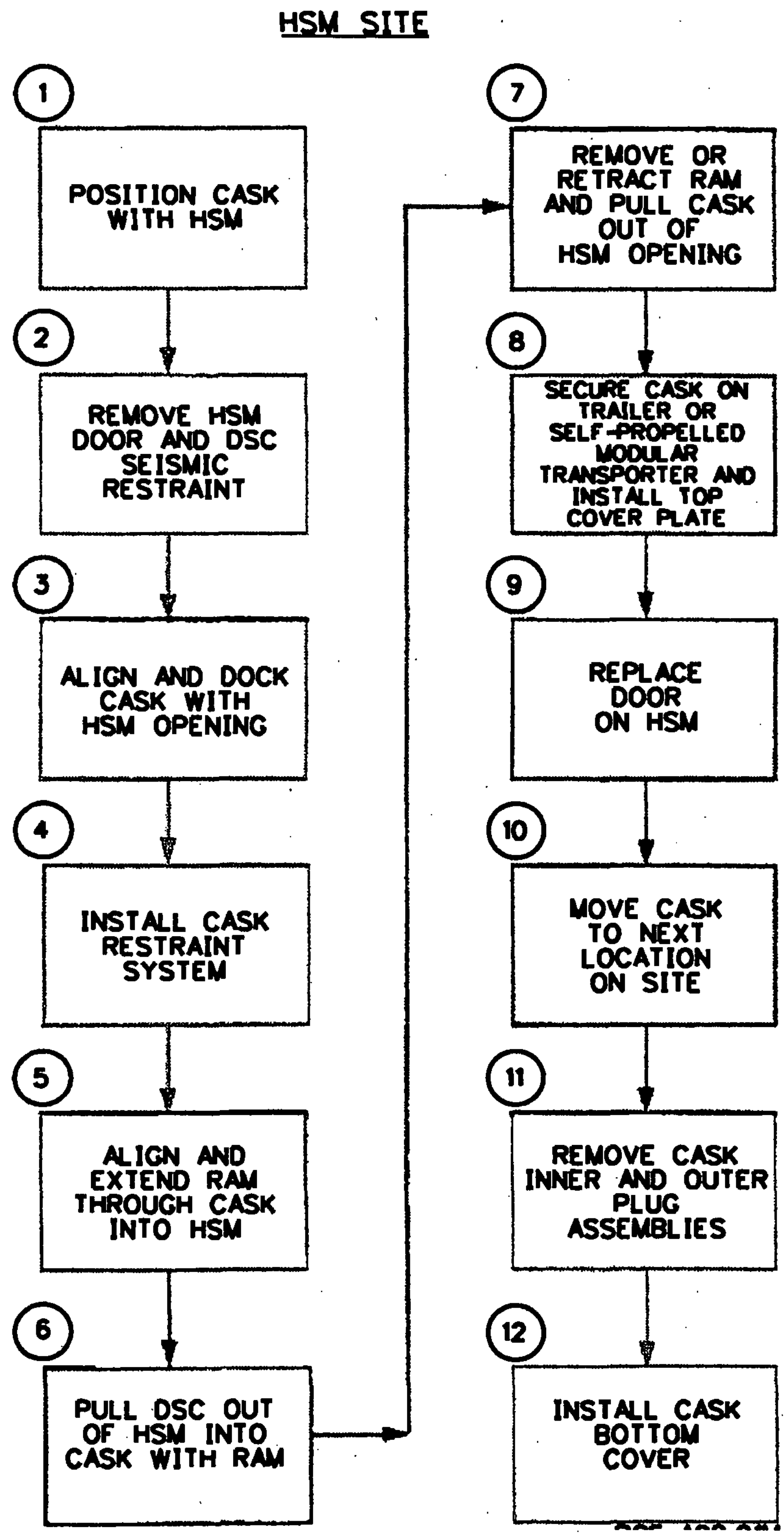

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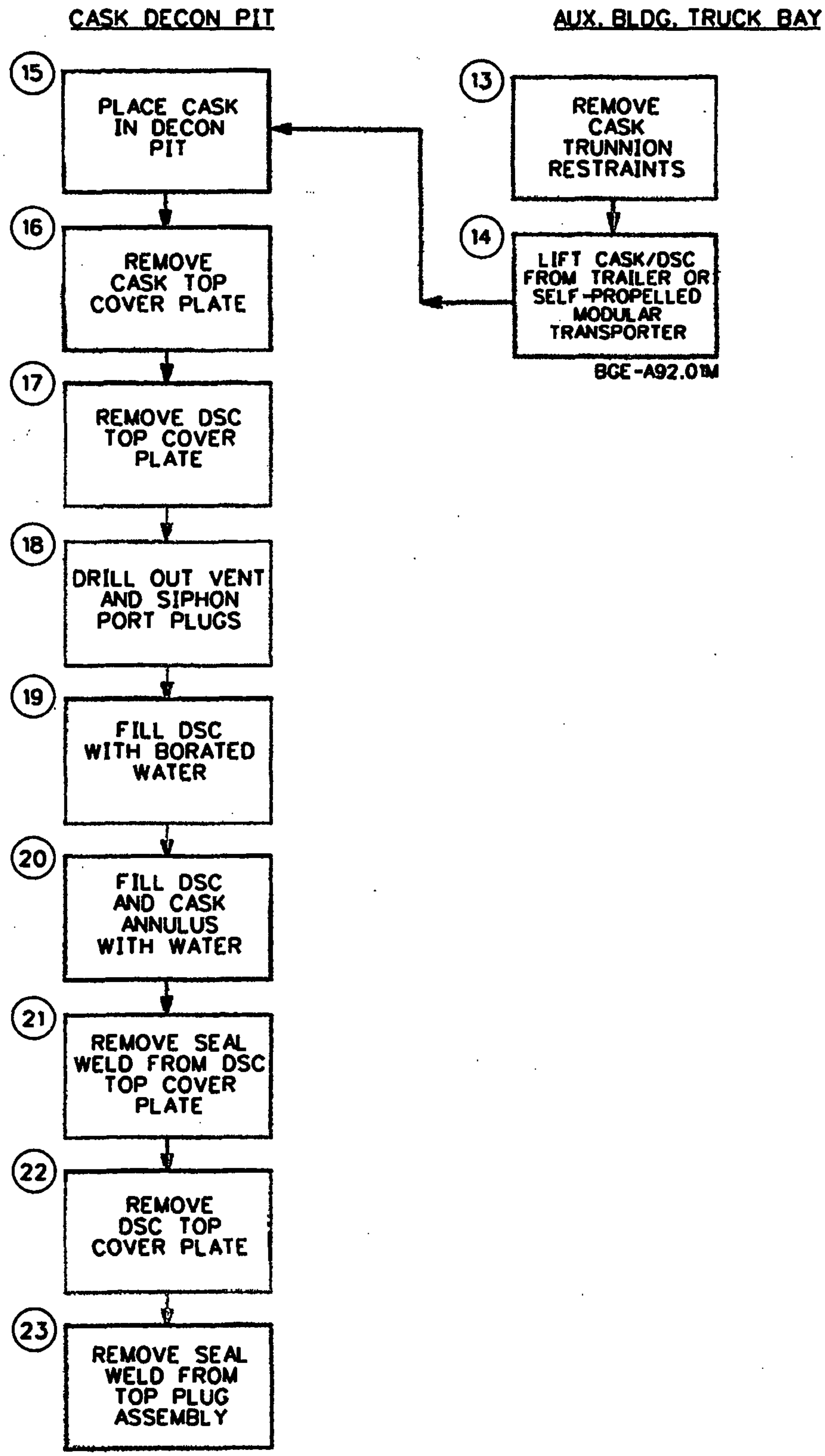

Calvert Cliffs Independent Spent Fuel Storage Installation	ISFSI LOADING OPERATIONS FLOWCHART Page 2 of 3 [See Reference 5.2]	USAR Figure 5.1-2 Rev. XX

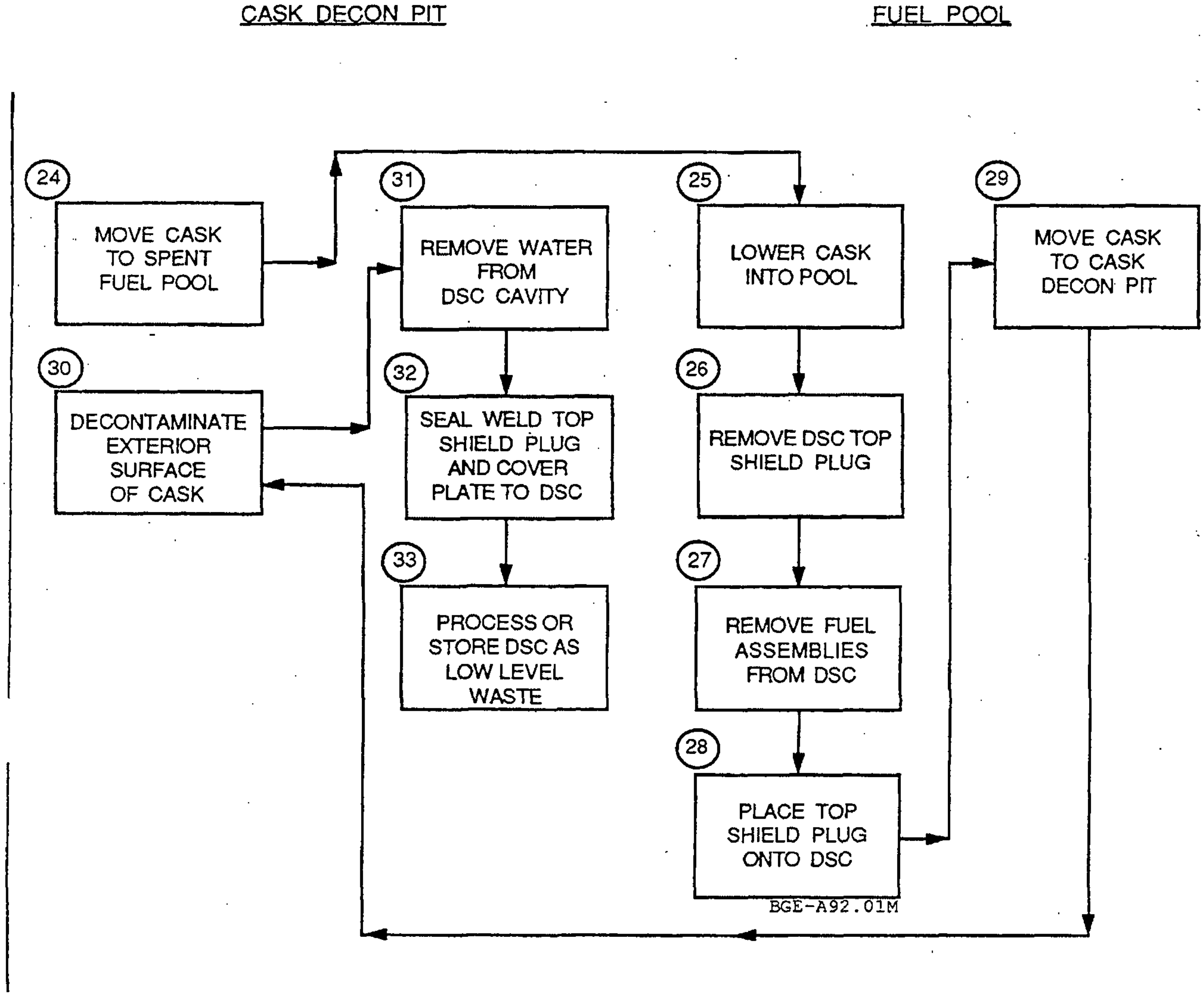

Figure 5.1-2

ISFSI Retrieval Operations Flowchart

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[See Reference 5.2]

CHAPTER 6

SITE GENERATED WASTE CONFINEMENT AND MANAGEMENT

LIST OF EFFECTIVE PAGES

CHAPTER 6

SITE GENERATED WASTE CONFINEMENT AND MANAGEMENT

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CHAPTER 6

SITE GENERATED WASTE CONFINEMENT AND MANAGEMENT

LIST OF ACRONYMS

DSC	Dry Shielded Canister
ISFSI	Independent Spent Fuel Storage Installation

6.0 SITE GENERATED WASTE CONFINEMENT AND MANAGEMENT

6.1 ON-SITE WASTE SOURCES

No radioactive wastes are generated during the storage life of the dry shielded canisters (DSCs). Radioactive wastes generated during loading operations are treated using existing Calvert Cliffs Nuclear Power Plant facilities and procedures.

6.2 OFF-GAS TREATMENT AND VENTILATION

No radioactive off-gas is produced at the Independent Spent Fuel Storage Installation (ISFSI). Potentially contaminated air, nitrogen, and helium purged from the DSC during evacuation are directed to the Auxiliary Building processing systems or the spent fuel pool.

6.3 LIQUID WASTE TREATMENT AND RETENTION

No liquid waste is produced at the ISFSI. Contaminated pool water removed from loaded DSCs will normally be drained back into the spent fuel pool with no additional processing. A small amount (<15 ft^3/DSC) of liquid waste results from transfer cask decontamination. The decontamination procedure results in a small amount of a detergent/demineralized water mixture being collected in the cask washdown pit. Liquid wastes collected in the cask washdown pit are directed to the power plant Liquid Waste Processing System described in Section 11.1.2.1 of Reference 6.1.

6.4 SOLID WASTES

No solid waste is produced at the ISFSI. A small quantity (<2 ft^3/DSC) of low level solid waste is generated as a result of DSC loading operations and transfer cask decontamination. The solid waste generated is processed by compaction using the Solid Waste Processing System as described in Section 11.1.2.3 of Reference 6.1. This low level waste consists of disposable Anti-C garments, tape, decon cloths, etc.

6.5 RADIOLOGICAL IMPACT OF NORMAL OPERATIONS — SUMMARY

No gaseous, liquid, or solid wastes are generated at the ISFSI. The small volumes of these wastes generated in the Auxiliary Building will have no significant impact on the ability of existing plant systems to process them.

6.6 REFERENCE

6.1 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

CHAPTER 7

RADIATION PROTECTION

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CHAPTER 7

RADIATION PROTECTION

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RADIATION PROTECTION

LIST OF ACRONYMS

ALARA	As Low As Reasonably Achievable
BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CFR	Code of Federal Regulations
DSC	Dry Shielded Canister
GS-RS	General Supervisor-Radiation Safety
HSM	Horizontal Storage Module
HSM-HB	High Burnup Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
NUHOMS	Nutech Horizontal Modular Storage®

7.0 RADIATION PROTECTION

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

7.1.1 POLICY CONSIDERATIONS

Baltimore Gas and Electric Company (BGE) Radiation Safety and As Low As Reasonably Achievable (ALARA) policies are described in the Quality Assurance Manual for Nuclear Power Plants, the Calvert Cliffs Radiation Safety Manual, and the ALARA Program. These policies are applied to the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI). Baltimore Gas and Electric Company is committed to a strong ALARA program in the design and operation of its nuclear facilities. The ALARA program follows the general guidelines of Regulatory Guides 1.8, 8.8, 8.10 and Title 10, Code of Federal Regulations (CFR) Part 20. Plant and design personnel are trained and updated on ALARA practices and dose reduction techniques. Design and implementation of systems and equipment are reviewed to ensure ALARA exposure on all new and modification projects. The basic ALARA program consists of:

A. The Calvert Cliffs Radiation Safety Manual, ALARA Program, and Radiation Safety (Implementation) Procedures.

B. Continued surveillance and evaluation of in-plant radiation and contamination conditions, as well as the monitoring and control of the exposure of personnel by radiation safety professionals and technical personnel.

C. The Radiation Safety-ALARA Unit comprised of radiation safety technical personnel, whose primary function is to perform ALARA reviews of operations, maintenance, and modifications.

D. Responsible Engineers who are responsible to assure ALARA considerations are accounted for in the design process.

Although upper level management is vested with the primary responsibility and authority for administering the Calvert Cliffs ALARA program, the responsibility for ALARA is extended through lower management to the individual employee and contractor.

The Vice President-Nuclear Energy Division has overall responsibility for all health and safety matters.

The Plant General Manager-Calvert Cliffs Nuclear Power Plant (CCNPP) Department is responsible for the protection of all persons against radiation and for compliance with Nuclear Regulatory Commission regulations and license conditions. This responsibility is, in turn, shared by all supervisors. Furthermore, all personnel are required to work safely and follow the regulations, rules, policies, and procedures that have been established for their protection.

The General Supervisor-Radiation Safety (GS-RS) is responsible for the administering and reviewing of the Calvert Cliffs ALARA program at the staff level. The GS-RS has direct access to the Vice President-Nuclear Energy Division on all matters vital to the radiation protection and ALARA programs.

The Supervisor of Radiation Safety-ALARA, who reports to the Assistant General Supervisor of Radiation Safety, is responsible for the ALARA program at the implementation level.

The Health Physics Consultant-Radiation Protection Manager establishes the Radiation Safety Program, including the program for handling and monitoring radioactive material for Calvert Cliffs, that is designed to assure compliance with applicable regulations, technical specifications, and regulatory guides. He also provides technical guidance and support for conducting this program, reviews the effectiveness and the results of the program and modifies it as required, based on experience and regulatory changes, to assure that occupational radiation exposures and exposure to the general public are maintained ALARA.

The GS-RS is responsible for conducting the radiation protection program for CCNPP, including the ISFSI. The GS-RS has the responsibility and authority to:

A. measure and control the radiation exposure of personnel
B. evaluate and review the radiological status of the plant
C. make recommendations for control or elimination of radiological hazards
D. assure that all personnel are trained in radiation protection
E. assist all personnel in carrying out their radiation responsibilities
F. protect the health and safety of personnel both on-site and in the surrounding area

To fulfill these responsibilities, radiological monitoring, survey, and personnel exposure control activities are performed on a continuing basis for plant operations and maintenance, including the ISFSI.

7.1.2 DESIGN CONSIDERATIONS – NUHOMS-24P

Section 12.7 contains a description of the radiation protection design features associated with the use of Nutech Horizontal Modular Storage® (NUHOMS)-32P dry shielded canisters (DSCs). The design considerations which ensure that occupational exposures for the NUHOMS-24P ISFSI are ALARA are discussed in Reference 7.1. The following paragraphs, which are numbered to correspond with Section 7.1.2 of Reference 7.1, discuss differences in the Calvert Cliffs implementation of the NUHOMS-24P generic design which affect the shielding design considerations.

1-7. Same as Reference 7.1.

8. The water used to fill the DSC cavity prior to immersion in the spent fuel pool will be borated. The shielding analyses were performed assuming that pure water was used to fill the cavity. The impact on the shielding calculation results is negligible (Reference 7.11).

   The cavity of the DSC will be submerged in the spent fuel pool for about 12 hours and, on removal from the pool, will contain borated water from the spent fuel pool for less than 50 hours. There is a substantial body of industry experience with exposure of austenitic stainless steels to borated water since that condition exists in most pressurized water reactor spent fuel pools. A literature search did not reveal any journal articles referring to corrosion of austenitic stainless in pools, from which one could infer that none has been observed since anecdotal experiences have all been very good. A Combustion Engineering, Inc. study on the effects of borated water on corrosion of low alloy steels reports a complete absence of corrosion in a very aggressive environment (dripping borated water and wet borated steam) for Type 304 and other corrosion resistant materials. The author concludes that "...corrosion resistant alloys such as ... Types 304 ... are not susceptible to borated water corrosion. Furthermore, stressed samples of these materials did not exhibit any localized forms of corrosion, such as stress corrosion cracking, hydrogen embrittlement, etc." (Reference 7.12)

After the DSC cavity has been drained, about 2 to 4 gallons of residual borated water will remain due to surface tension. As the borated water evaporates during the vacuum drying process, the ortho-boric acid crystals will precipitate out of the solution at concentrations substantially higher then 2000 ppm. After the free water has evaporated, a small amount of dehydrated B_2O_3 crystals will remain in the cavity. A literature search did not uncover any reference to corrosive or aggressive behavior of anhydrous boric acid and discussions with chemists at an industry supplier (U.S. Borax) revealed that it will only become corrosive in the molten state. The melting temperature for anhydrous boric acid is about 450°C, well above the peak DSC material temperatures and the peak fuel cladding temperatures. Even if corrosion of the DSC shell or basket is postulated, the extremely small quantity of borate and the expected corrosion mechanism of general surface oxidation is not likely to lead to degradation of the structural integrity of the DSC shell or basket.

9-13. Same as Reference 7.1.

14. Same as Reference 7.1 except that the shielding calculations were performed assuming that water would be present in the annular gap when the DSC is flooded, and that the annular gap would be drained right before the transfer cask cover plate is installed.

15. Same as Reference 7.1.

During the design phase of the Calvert Cliffs NUHOMS ISFSI, the NUHOMS-07P demonstration project was successfully completed at the H. B. Robinson plant. Comparison between predicted dose rates and those measured during the first fuel load at Robinson confirmed that the ALARA design considerations employed in the NUHOMS ISFSI design are sound and effective. The NUHOMS design incorporates certain improvements in the design and analysis of the radiation shielding as compared to the NUHOMS-07P system (References 7.1 and 7.2). Furthermore, lower design criteria for the horizontal storage module (HSM) average surface dose rates have been specified for the Calvert Cliffs ISFSI. Successful demonstration of the NUHOMS-07P system, design improvements in the generic NUHOMS-24P system, and lower design criteria for the HSM surface dose rates all ensure that the Calvert Cliffs ISFSI shielding design and occupational radiation exposures will be ALARA.

7.1.3 OPERATIONAL CONSIDERATIONS

Consistent with BGE's overall commitment to keep occupational radiation exposures ALARA, specific plans and procedures are followed by plant personnel to ensure that ALARA goals are achieved. Operational ALARA policy statements are formulated for the Nuclear Energy Division through the issuance of the Quality Assurance Manual for

Nuclear Power Plants, the Radiation Safety Manual, and the ALARA Program and are implemented by means of procedures. These statements and procedures are consistent with the intent of Section C.1 of Regulatory Guides 8.8 and 8.10.

Since the ISFSI is a passive system, no maintenance is expected on a normal basis in the facility. Maintenance operations on the transfer cask, transfer trailer, the SPMT and other ancillary equipment is performed in a very low-dose environment when fuel movement is not occurring. Planned maintenance activities are preventive in nature and include motor oil changes, hydraulic oil filter replacement and the like.

7.2 RADIATION SOURCES – NUHOMS-24P

Sections 12.7 and 13.7 discusses the radiation sources associated with the use of NUHOMS-32P and NUHOMS-32PHB DSCs, respectively.

7.2.1 CHARACTERIZATION OF SOURCES

The source terms used in the shielding analyses for the Calvert Cliffs ISFSI were developed in the same manner as for the NUHOMS-24P Topical Report (Reference 7.1). 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 a heat output of 0.66 kW. The fuel assembly with the largest source terms (both neutron and gamma) was found to be a 3.4 w/o initial enrichment, 42,000 MWD/MTU burnup, cooled for 8 years. These source strengths were used for shielding design throughout the ISFSI. The active fuel region neutron and gamma energy spectra for this reference fuel assembly is given in Tables 7.2-1 and 7.2-2 respectively.

The source modeling methodology is similar to the NUHOMS-24P generic design methodology and is fully described in References 7.14 (active fuel region) and 7.16 (upper and lower end fittings and plenum region). This source term methodology has also been demonstrated to produce conservative dose rate results compared to actual Calvert Cliffs ISFSI benchmarks, when coupled with the shielding methodology discussed in Section 7.3.2. No other significant radiological sources (such as storage containers or tanks) are located in the vicinity of the ISFSI. Figure 7.2-1 is a bounding radiological limit curve for assemblies at or below a thermal power of 660 watts and have cooled at least 9 years. Assemblies meeting the requirements of both Table 9.4-1 and Figure 7.2-1 may be loaded and stored in the ISFSI.

7.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

The release of airborne radioactive material is postulated for three phases of system operation: fuel handling in the spent fuel pool, drying and sealing of the DSC, and DSC transfer and storage.

Potential airborne releases from irradiated fuel assemblies in the pool are discussed in Reference 7.3.

Dry shielded canister drying and sealing operations are performed using procedures which prohibit airborne leakage. During these operations, all vent lines are routed to the Auxiliary Building's existing radwaste systems. Once the DSC is dried and sealed, there are no design basis accidents which could result in a breach of the DSC and the airborne release of radioactivity.

During transfer of the sealed DSC and subsequent storage in the HSM, the only postulated mechanism for the release of airborne radioactive material is the dispersion of non-fixed surface contamination on the DSC exterior. By filling the cask/DSC annulus with demineralized water, placing a mechanical seal over the annulus, and utilizing procedures which require examination of the annulus surfaces for smearable contamination, the contamination limits on the DSC can be kept below the permissible level for off-site shipments of fuel. Therefore, there is no possibility of significant radionuclide release from the DSC exterior surface during transfer or storage.

TABLE 7.2-1
NEUTRON ENERGY SPECTRUM

GROUP NUMBER	UPPER ENERGY (MeV)	GROUP FRACTION
1	1.50E+01	4.65E-04
2	1.22E+01	1.88E-03
3	1.00E+01	5.76E-03
4	8.18E+00	1.92E-02
5	6.36E+00	4.00E-02
6	4.96E+00	5.17E-02
7	4.06E+00	1.09E-01
8	3.01E+00	8.80E-02
9	2.46E+00	2.09E-02
10	2.35E+00	1.16E-01
11	1.83E+00	2.09E-01
12	1.11E+00	1.92E-01
13	5.50E-01	1.33E-01
14	1.11E-01	1.35E-02
15	3.35E-03	0.00E+00
16	5.33E-04	0.00E+00
17	1.01E-04	0.00E+00
18	2.90E-05	0.00E+00
19	1.07E-05	0.00E+00
20	3.06E-06	0.00E+00
21	1.12E-06	0.00E+00
22	4.14E-07	0.00E+00
		1.00E+00

For more information see Reference 7.11.

TABLE 7.2-2
GAMMA ENERGY SPECTRUM

GROUP NUMBER	UPPER ENERGY (MeV)	GROUP FRACTION
1	1.00E+01	2.99E-11
2	8.00E+00	2.60E-10
3	6.50E+00	2.26E-09
4	5.00E+00	0.00E+00
5	4.00E+00	2.41E-07
6	3.00E+00	1.88E-06
7	2.50E+00	3.28E-05
8	2.00E+00	5.16E-04
9	1.66E+00	0.00E+00
10	1.33E+00	2.93E-02
11	1.00E+00	5.94E-02
12	8.00E-01	0.00E+00
13	6.00E-01	4.29E-01
14	4.00E-01	1.20E-02
15	3.00E-01	2.29E-02
16	2.00E-01	2.91E-02
17	1.00E-01	7.39E-02
18	5.00E-02	3.44E-01
		1.00E+00

For more information see Reference 7.11.

7.3 RADIATION PROTECTION DESIGN FEATURES – NUHOMS-24P

7.3.1 INSTALLATION DESIGN FEATURES

The Calvert Cliffs ISFSI design features are discussed in detail in the NUHOMS-24P Topical Report (Reference 7.1).

7.3.2 SHIELDING

Reference 7.1 contains a complete description and illustration of the shielding design for the Calvert Cliffs ISFSI, with the exception of the shielding on the automatic welding machine, which is described in Reference 7.19. The operational shielding analyses performed in support of the Calvert Cliffs ISFSI design utilized the MCNP code. The storage term shielding analyses are identical in form to the calculations performed in support of the generic NUHOMS-24P design. References 7.1 and 7.15 through 7.17 contain a complete description of those shielding methodologies and models. References 7.4 through 7.9 and 7.18 are the shielding computer program packages and cross-section data used in the analyses.

As indicated above, the MCNP code was utilized to calculate operational dose rates at the locations of interest. The MCNP code is a general-purpose Monte-Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/ photon/electron transport, including the capability to calculate k_{eff} for critical systems. A benchmark of the MCNP models for the CCNPP ISFSI produced results that matched the actual ISFSI surveys or exceeded them by an average factor of 1.8.

The results of the shielding analyses are presented in Table 7.3-1. The CCNPP HSM doorway dose rate (3240 mrem/hr) is higher than the corresponding generic NUHOMS-24P dose rate (760 mrem/hr) due to design differences in the fuel to be stored, the DSC shield geometries, and the analysis methodology. The former value was calculated using the benchmarked MCNP model, while the latter value was calculated using the methodology presented in the NUHOMS-24P topical report. A number of different combinations of DSC, transfer cask, and HSM door shield thickness combinations were evaluated and the present design was determined to be acceptable in terms of the operational, regulatory, and ALARA objectives. Note that the only operation which takes place in the HSM doorway radiation field is the seismic restraint installation. The CCNPP NUHOMS seismic restraint design includes refinements which allow the restraint to be more easily handled and quickly placed than the generic design. Since the remainder of the operational and storage term radiological exposure rates are heavily influenced by the HSM door exterior dose rates, an improved HSM door design has been developed for the CCNPP ISFSI which reduces the HSM door centerline dose rate from 77 to 10 mrem/hr. The resulting dose rate is lower than that calculated for the topical report.

7.3.3 VENTILATION

The Calvert Cliffs ISFSI ventilation design is described in Section 4.3.1 and in the NUHOMS-24P Topical Report (Reference 7.1). Ventilation during DSC drying operations is described in Section 7.2.2.

7.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

The Calvert Cliffs ISFSI is expected to result in very low direct dose rates and zero radionuclide release during all credible phases of operation. In order to assure public and employee safety, the CCNPP environmental monitoring program has been expanded to include Dosimeters of Legal Record, air samplers, and vegetation and soil samples at the ISFSI site. Equipment at existing nearby monitoring sites are also upgraded as necessary to monitor the ISFSI.

TABLE 7.3-1
NUHOMS-24P SHIELDING ANALYSIS RESULTS NOMINAL DOES RATES (MREM/HR)

	LOCATION	NEUTRON	GAMMA (PRI + SEC)	TOTAL
	NUHOMS-24P DSC in HSM			
1.	HSM Wall or Roof	0.5	12	12.5
2.	HSM Air Outlet	1	81	82
3.	Center of Door	5	5	10
4.	Center of Doorway	621	2619	3240
5.	Air Inlet Vent	1	72	73
6.	1m from HSM Door	2	4	6
	NUHOMS-24P DSC in Cask			
1.	Centerline [(a)] DSC Shield Plug (Flooded DSC)	4	76	80
2.	DSC Cover Plate (Dry DSC)			
	2.1 Center	45	96	141
	2.2A Edge[(b)] (Wet Gap)	80	62	142
	2.2B Edge[(b)] (Dry Gap)	124	136	260
3.	Transfer Cask			
	3.1 Side	69	72	141
	3.2 Top	6.0	1.0	7.0
	3.3 Bottom	56	63	119

For more information see References 7.15 and 7.16.
Dose rates associated with the use of NUHOMS-32P DSCs are discussed in Section 12.7.

(a) The DSC/cask annular gap is filled with water. All but the top 6" of the DSC inner cavity is filled with water.

(b) Nominal at edge of cover plate. The total dose rate is approximately a factor of 3 lower at the top edge of the transfer cask, and several times higher inside the dry annulus.

7.4 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

7.4.1 OPERATIONAL EXPOSURE

This section establishes the expected cumulative dose delivered to operational personnel during the DSC fuel loading, closure, and transfer activities associated with placing one DSC into dry storage in a NUHOMS module. Chapter 5 describes the ISFSI operational procedures, a number of which involve radiation exposure to personnel.

The occupational dose received during fuel loading, closure, and transfer of the DSC at the HSM is estimated utilizing MCNP (Reference 7.18). Details on the modeling techniques used and results are in Reference 7.17.

The estimated occupational exposures for one HSM load are documented in Reference 7.17 and are based on the experience gained during the fuel loads at the Carolina Power and Light Company's H. B. Robinson NUHOMS-07P ISFSI and CCNPP. The estimated occupational exposures have been benchmarked (References 7.17 and 7.20) and demonstrated conservative to actual accumulated total occupational dose measured at fuel loads performed at CCNPP. The results are also consistent with subsequent experience gained during the fuel loads at the Duke Power Company's Oconee NUHOMS-24P ISFSI. The apparent differences between the NUTECH topical report and Calvert Cliffs ISFSI are due largely to the regrouping of the various fuel loading tasks and utilization of different calculational methodology.

The expected dose rates from transfer components are determined by the design of the permanent shielding. The design requirements for the shielding were based on a desire to attain the maximum permanent shielding that was economically practical. The system has been designed to meet the requirements of the ALARA program at Calvert Cliffs (Section 7.1). Since the shielding is fixed, the only credible cause for the operational dose rates to be significantly higher than expected is the misloading of fuel assemblies in the canister. Other operational controls are sufficient to preclude this condition such that additional operational controls on surface dose rates at specific locations during transfer operations are not necessary. Small variations above the expected operational dose rates at specific locations will be alleviated in accordance with normal plant ALARA procedures.

7.4.2 STORAGE TERM EXPOSURE

Figures 7.4-1 and 7.4-2 are graphs of the dose rate versus distance from the end and face, respectively, of an array of 132 NUHOMS HSMs (bounds NUHOMS-24P, NUHOMS-32P, and NUHOMS-32PHB DSC designs) configured as shown in Figure 1.2-1 and loaded with design basis fuel. The curve was constructed from the shielding analysis described in the previous sections. Direct neutron and gamma flux, as well as the air-scattered radiation from the module surfaces are included. No credit is taken for shielding by existing structures or terrain.

The radiation sources used for the direct and air-scattered dose calculations are in References 7.16 (NUHOMS-24P DSC), 7.23 (NUHOMS-32P DSC), and 7.24 (NUHOMS-32PHB). Direct and air-scattered dose rates are calculated using the computer code MCNP (Reference 7.18). The results depicted in Figures 7.4-1, 7.4-2, and 7.4-3 were generated using the bounding dose results at each location from the

MCNP models discussed in Sections 7.3 (Reference 7.21), 12.7.3 (Reference 7.23), and 13.7.3 (Reference 7.24).

The generic design depicted in Figure 7.4-1 of the NUHOMS-24P topical report cannot be compared directly to the Calvert Cliffs design since it only shows dose versus distance from a single 2x10 array of HSMs. Figures 7.4-1 and 7.4-2 also incorporate the effects of an improved HSM door design which substantially lowers area dose rates.

The ISFSI is surrounded by a large open area for operational and security purposes. Access to the storage modules is restricted such that during storage, no access closer than 50' is allowed except for security and surveillance inspection purposes. There are no work areas close to the ISFSI. Dose to workers at the power plant and other individuals in the unrestricted area due to exposure from the ISFSI is minimal and below regulatory limits (Reference 7.11).

If DSC transfers are performed during Camp Conoy hours, visitor access is restricted and controlled by Calvert Cliffs Nuclear Power Plant, Inc. Security. During the DSC seismic restraint installation, radiation dose rate measurements will be made by Radiation Safety Technicians at selected locations, to supplement environmental surveillance stations and Dosimeters of Legal Record and to validate calculated dose rates.

During the seismic restraint installation, a person standing outside the ISFSI security fence would be exposed to radiation from all closed/loaded HSMs as well as from the HSM where the seismic restraint is being installed. The dose contribution from each of these sources is as follows:

- Site: Figure 2.4-1 shows that the outer fence is 94 feet from the HSMs in the N/S direction, and 53 feet from the HSMs in the E/W direction. From Figures 7.4-1, 7.4-2, and 7.4-3, the maximum dose rates at these locations from a site consisting of 132 HSMs loaded with NUHOMS-32P, NUHOMS-24P, or NUHOMS-32PHB DSCs is 0.67 mrem/hr in the N/S direction, and 0.73 mrem/hr in the E/W direction.

  These site dose rates also bound storage of a NUHOMS-32P in an high burnup horizontal storage module (HSM-HB) since dose rates at the module surface are generally lower than the HSM (see Table 12.7-1).

- Seismic Restraint Installation: The minimum distance between the DSC surface and an individual at the closest approach to the outer ISFSI security fence (east/west portion of the fence) is estimated to be 66.7'. Note that at 66.7' from the DSC surface an individual at the outer security fence will not be directly in front of the HSM and can only see a fraction of the DSC end surface because it is partially shielded by the cask, HSM front wall, and the HSM door which is raised 2 feet from the closed position during seismic restraint installation. The dose rate at a distance of 66.7' directly in front of a loaded NUHOMS-24P HSM with the door fully open and unobstructed by the transfer cask is conservatively reported as 41 mrem/hr (Reference 7.16). At the same location, unobstructed by the transfer cask, the dose rate for the NUHOMS-32P HSM with the door open 2' for seismic restraint installation is 10.2 mrem/hr (Reference 7.23). The latter is used for this evaluation since it remains conservative relative to the actual configuration. NUHOMS-32P stored in the HSM is bounding for HSM-HB since dose rates in the open doorway are higher for the HSM than the HSM-HB (see Table 12.7-1).

Experience at Calvert Cliffs and Oconee indicates that seismic restraint installation requires a maximum of 5 minutes (0.08 hours) and can nominally be done in 1 minute (0.017 hours). Thus, a person just outside the fence would conservatively receive a maximum of 0.82 mrem (0.08 hr x 10.2 mrem/hr) during the seismic restraint installation process. If that person remained outside the fence for an entire hour during the seismic restraint installation, their dose would conservatively not exceed 1.55 mrem (0.82 mrem from seismic restraint installation + 0.73 mrem from the fully loaded ISFSI site). This is within the 10 CFR 20.1301 requirement that no individual (member of the public) may receive greater than 2 mrem in any 1 hour.

The air inlet and outlets of the HSM are localized areas compared to the overall HSM surfaces. Therefore, the surface dose rates at these locations are less representative than dose rates at the HSM walls and door for assessing the effect on the direct radiation levels associated with ISFSI operations to individuals located beyond the controlled area while the DSC is in storage. The HSM air inlet vent inspection dose is estimated to be less than 108 mrem/yr for both NUHOMS-24P and NUHOMS-32P DSC designs (Reference 7.22). Normally, remote cameras will be used for air inlet and outlet surveillance. Therefore, no exposure will result from their inspection. If the outlets were manually inspected, an additional exposure of 408 mrem/yr (bounds both NUHOMS-24P and NUHOMS-32P DSC designs) would be incurred (Reference 7.22). These values are derived by assuming that one inspector performs a daily inspection, walking at an average speed of 3 mph on a path passing directly between the north and south outlets on the HSM roof and far away (visual distance) from the front of the air inlet vents of the ISFSI. The distance between the inspector and each HSM front wall is assumed to be half-distance between each row of modules. It is further assumed that all five phases (132 HSMs) are filled with design basis fuel. No credit is taken for radioactive decay of the fuel during storage. Dose rates are based on those given in Figure 7.4-2 (front) and Tables 7.3-1 and 12.7-1 (roof) and Table 13.7-1.

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Calvert Cliffs Independent Spent Fuel Storage Installation	DOSE RATES VS N-S DISTANCE FROM HSM/MB (HSM FRONTS) [REFERENCE 7.23]	USAR Figure 7.4-3 Rev. XX

7.6 ESTIMATED OFF-SITE COLLECTIVE DOSE ASSESSMENT

7.6.1 EFFLUENT AND ENVIRONMENTAL MONITORING PROGRAM

No effluents are released from the ISFSI during operation. Effluents released during DSC loading are treated using existing power plant systems as described in Chapter 6. Since no effluents are released from the Calvert Cliffs ISFSI site, no monitoring program is required.

7.6.2 ANALYSIS OF MULTIPLE CONTRIBUTION

An analysis of multiple contributions was performed to determine the additional radiological impact that the ISFSI will impose on the population surrounding the CCNPP site. This impact, added to the contributions made by other uranium fuel cycle operations in the vicinity, was compared to the natural background radiation and the regulatory requirements of 10 CFR 72.104 and 40 CFR Part 190.

The maximally exposed member of the public is assumed to have continuous occupancy in the nearest residence to the ISFSI which is located 4705' from the facility. At that location, the dose rate from 120 HSMs filled to capacity with design basis fuel would be less than 2 mrem/yr from the ISFSI, and less than 13.5 mrem/year from the remaining fuel cycle operations in the vicinity (Reference 7.10). The collective dose due to the ISFSI for persons located within 0-2 miles is conservatively estimated as 28.25 person-mrem spread over 215 people. This is less than 1% of the collective dose from the remaining fuel cycle operations. It can be concluded that the radiation exposure due to the Calvert Cliffs ISFSI, combined with all other fuel cycle operations, will not exceed the regulatory requirements of 25 mrem/year in 10 CFR 72.104 and 40 CFR Part 190.

7.6.3 ESTIMATED DOSE EQUIVALENTS

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

7.6.4 LIQUID RELEASE

No liquids are released from the Calvert Cliffs ISFSI.

7.7 REFERENCES

7.1 Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, Nutech Engineers, Inc., NUH-002, Revision 1A, July 1989

7.2 Topical Report on the Nutech Horizontal Modular Storage System for Irradiated Fuel, NUHOMS-07P, Nutech Engineers, Inc., NUH-001, Revision 1A, June 1986

7.3 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

7.4 ANISN/PC, Multigroup One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering, CCC-514 Micro, Oak Ridge National Laboratory, January 1988

7.5 CASK, 40 Group Coupled Neutron and Gamma-Ray Cross-Section Data, DLC-23, Oak Ridge National Laboratory, June 1987

7.6 QAD-CGGP, A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor, CCC-493, Oak Ridge National Laboratory, July 1986

7.7 MicroSkyshine Manual, Version 2, Grove Engineering, Washington Grove, MD, July 1987

7.8 Microshield User's Manual, A Program for Analyzing Gamma Radiation and Shielding, Version 3, Grove Engineering, Inc., Washington Grove, MD

7.9 Croff, A. G., ORIGEN2 - A Revised and Updated Version of the Oakridge Generation and Depletion Code, ORNL-5621, Oak Ridge National Laboratory, 1980

7.10 Calvert Cliffs Nuclear Power Plant Environmental Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company, November 16, 1970

7.11 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated November 1, 1990, Response to NRC's Comments on Environmental Issues Regarding BGE's License Application for Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI)

7.12 J. F. Hall, "Corrosion of Low Alloy Steel Fastener Materials Exposed to Borated Water", Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Ed. by G. J. Theus and J. R. Weeks, The Metallurgical Society, 1988

7.13 Deleted

7.14 CCNPP Calculation CA05803, ISFSI 24P Assembly Insertion Requirements, December 31, 2001

7.15 CCNPP Calculation CA05924, Calvert Cliffs ISFSI/NUHOMS-24P Radiation Dose Rates for Cask Loading and Transfer, September 18, 2002

7.16 CCNPP Calculation CA05925, Calvert Cliffs ISFSI/NUHOMS-24P HSM Dose Rates, September 18, 2002

7.17 CCNPP Calculation CA05926, Calvert Cliffs ISFSI/NUHOMS-24P Occupational Doses, September 18, 2002

7.18 Los Alamos National Laboratory, LA-13709-M, MCNP-A General Monte Carlo N-Particle Transport Code, Version 4C, March 2000

7.19 CCNPP Drawing 84011SH0001, Revision 00, ISFSI Weld Machine Radiation Shield

7.20 CCNPP Engineering Package ES200101042, ISFSI 24P Design Basis Dose Calculations with MCNP, September 25, 2002

7.21 CCNPP Calculation CA06058, ISFSI 24P 5 Phase Site Dose Rates, September 25, 2002

7.22 CCNPP Calculation CA03904-002, HSM Inlet Vent Surveillance Dose

7.23 CCNPP Calculation CA06751, Horizontal Storage Module Dose Rates for ISFSI 32P Burnup Extension

7.24 CCNPP Calculation CA07332, Calvert Cliffs ISFSI/NUHOMS-32PHB Site Dose Analysis, July 1, 2010

7.7 REFERENCES

7.1 Topical Report for the Nutech Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P, Nutech Engineers, Inc., NUH-002, Revision 1A, July 1989

7.2 Topical Report on the Nutech Horizontal Modular Storage System for Irradiated Fuel, NUHOMS-07P, Nutech Engineers, Inc., NUH-001, Revision 1A, June 1986

7.3 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

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7.12 J. F. Hall, "Corrosion of Low Alloy Steel Fastener Materials Exposed to Borated Water", Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, Ed. by G. J. Theus and J. R. Weeks, The Metallurgical Society, 1988

7.13 Deleted

7.14 CCNPP Calculation CA05803, ISFSI 24P Assembly Insertion Requirements, December 31, 2001

7.15 CCNPP Calculation CA05924, Calvert Cliffs ISFSI/NUHOMS-24P Radiation Dose Rates for Cask Loading and Transfer, September 18, 2002

7.16 CCNPP Calculation CA05925, Calvert Cliffs ISFSI/NUHOMS-24P HSM Dose Rates, September 18, 2002

7.17 CCNPP Calculation CA05926, Calvert Cliffs ISFSI/NUHOMS-24P Occupational Doses, September 18, 2002

7.18 Los Alamos National Laboratory, LA-13709-M, MCNP-A General Monte Carlo N-Particle Transport Code, Version 4C, March 2000

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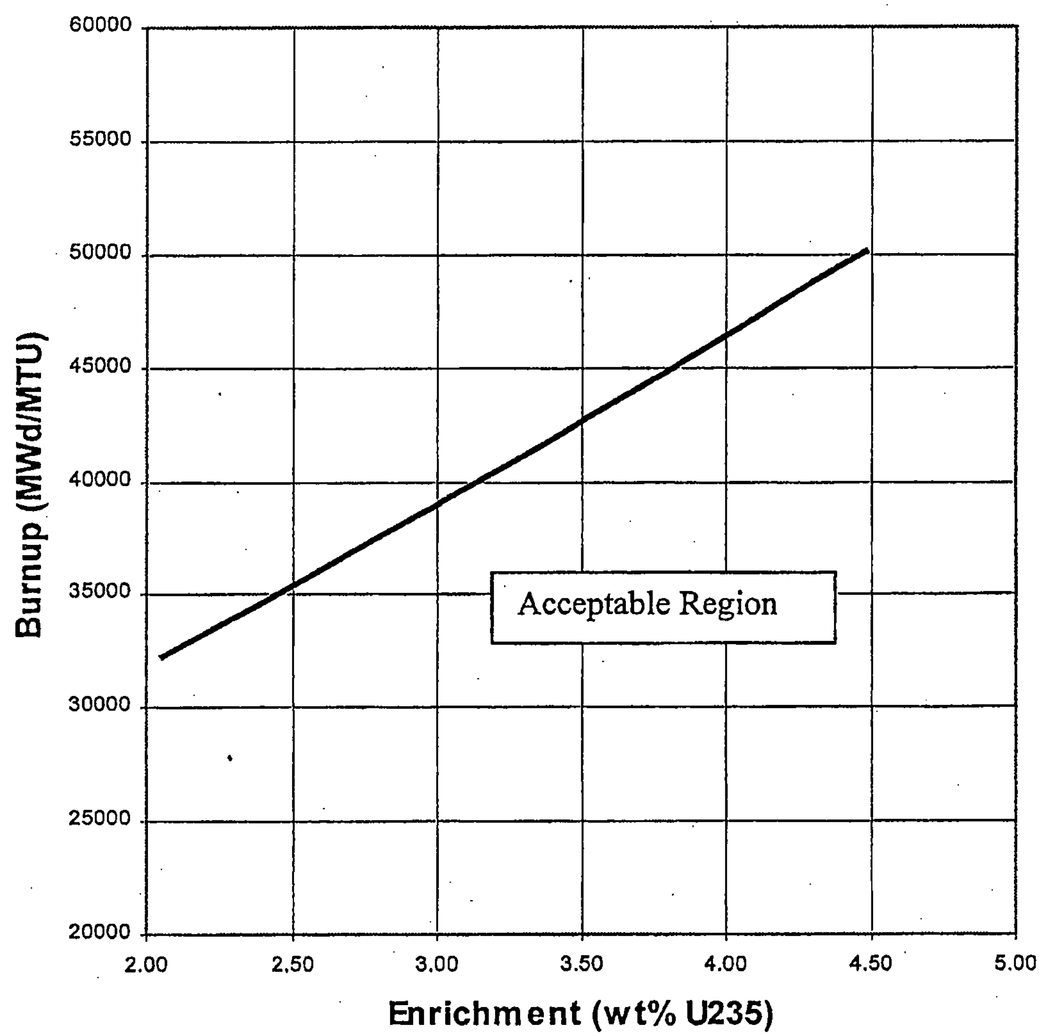
7.22 CCNPP Calculation CA03904-002, HSM Inlet Vent Surveillance Dose

7.23 CCNPP Calculation CA06751, Horizontal Storage Module Dose Rates for ISFSI 32P Burnup Extension

7.24 CCNPP Calculation CA07332, Calvert Cliffs ISFSI/NUHOMS-32PHB Site Dose Analysis, July 1, 2010



* This curve is valid for all assemblies that have cooled at least 9 years.

** All assemblies loaded into DSC must meet the source spectra requirements of Technical Specification 2.1.

*** This bounding radiological limit curve may be superceded with bundle specific source spectra calculations based on actual cooling times.

Calvert Cliffs Nuclear Power Plant, Independent Spent Fuel Storage Installation	24P DESIGN BASIS RADIOLOGICAL LIMIT CURVE FOR 660 WATTS/ASSEMBLY (Reference 7.20)	USAR Figure 7.2-1 Revision 20

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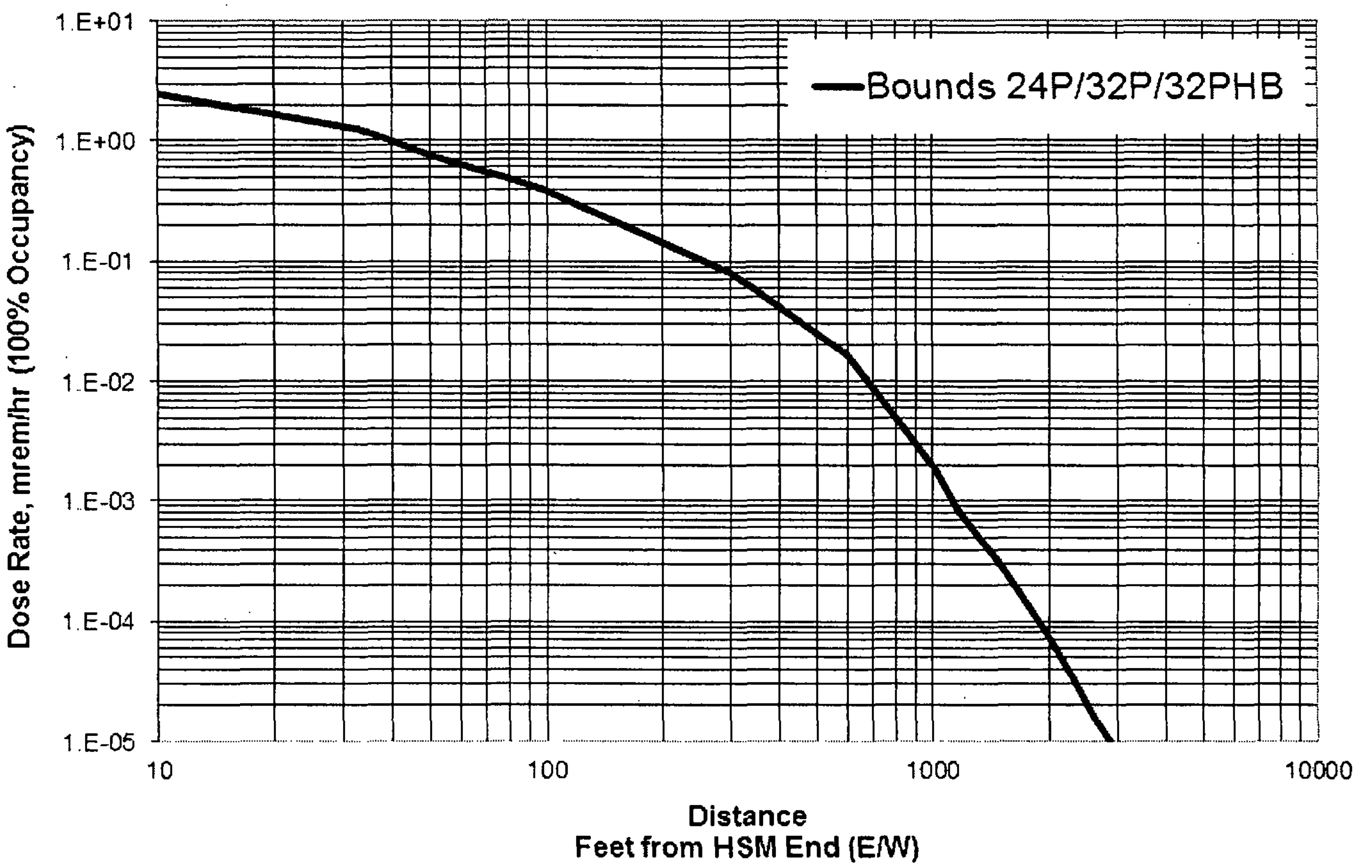
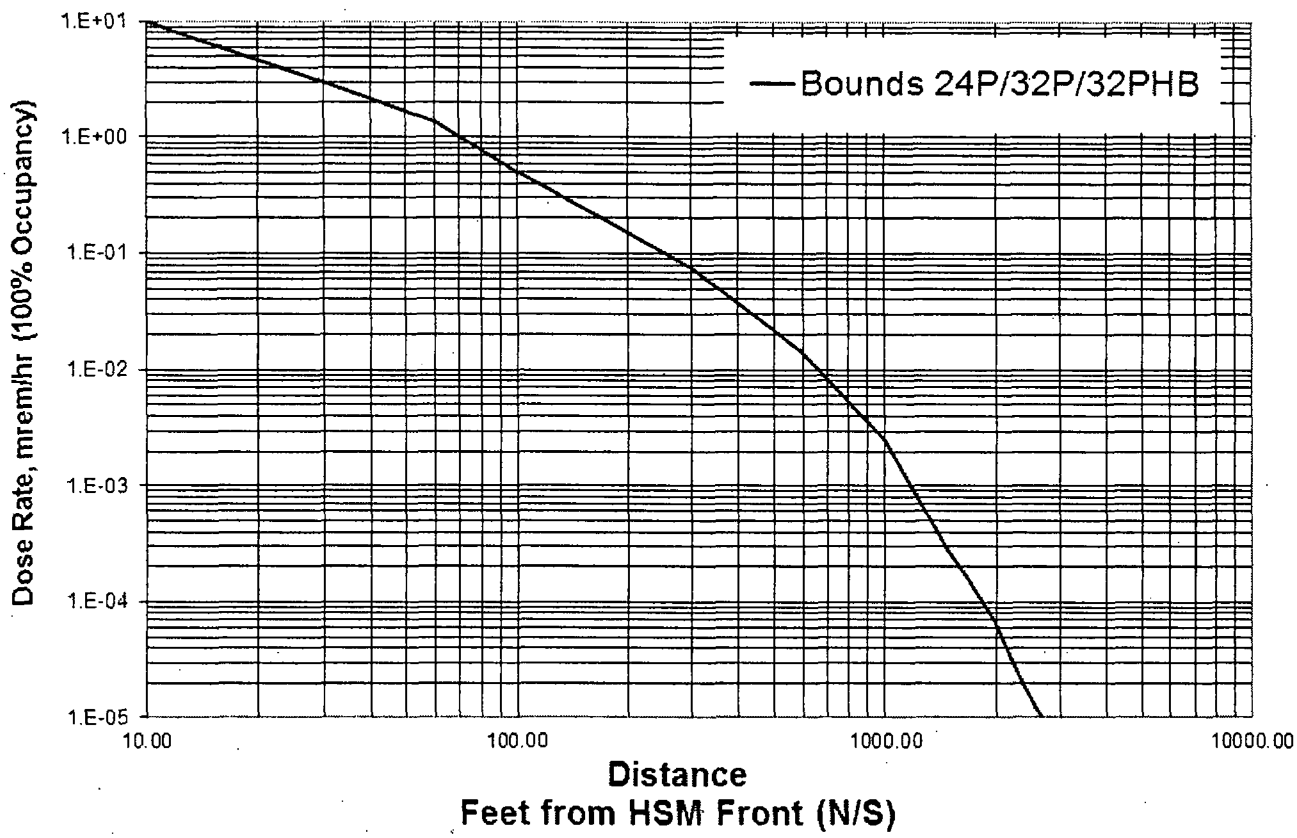


Calvert Cliffs Independent Spent Fuel Storage Installation	DOSE RATES VS E-W DISTANCE FROM HSM (HSM ENDS) [REFERENCE 7.23]	USAR Figure 7.4-1 Revision XX



Calvert Cliffs Independent Spent Fuel Storage Installation	DOSE RATES VS N-S DISTANCE FROM HSM (HSM FRONTS) [REFERENCE 7.23]	USAR Figure 7.4-2 Revision XX

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AISC	American Institute of Steel Construction
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CCNPP	Calvert Cliffs Nuclear Power Plant
CFR	Code of Federal Regulations
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DSC	Dry Shielded Canister
HSM	Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
LNG	Liquified Natural Gas
NRC	Nuclear Regulatory Commission
NUHOMS	Nutech Horizontal Modular Storage®
SER	Safety Evaluation Report

8.0 ACCIDENT ANALYSES – NUHOMS-24P

Analyses of all design events are reported in the Topical Report for the Nutech Horizontal Modular Storage® (NUHOMS)-24P System (Reference 8.1) for the generic NUHOMS-24P Independent Spent Fuel Storage Installation (ISFSI) design as required by American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.9-1984 (Reference 8.2). The analyses of these design events have been repeated for the Calvert Cliffs site-specific ISFSI design, and the results are reported in this section in the same format as in the Topical Report. The analytical assumptions, methodology, and computer codes used to generate the results in this section are identical to those used in the Topical Report, unless noted otherwise in the text.

This section and Section 12.8 discusses the accident analysis associated with the use of NUHOMS-32P dry shielded canisters (DSCs).

8.1 NORMAL AND OFF-NORMAL OPERATIONS

This section follows the format of the Topical Report (Reference 8.1) and includes the evaluation of Design Basis Type 1 Events (normal operating conditions) as defined in ANSI/ANS 57.9. These events, their bases, and analytical methodology are described in Section 8.1 of the Topical Report. The results of the analyses of these loads on the Calvert Cliffs ISFSI design are discussed in the following subsections.

8.1.1 NORMAL OPERATION STRUCTURAL ANALYSIS

The normal operating loads for the NUHOMS-24P important to safety components are shown in Table 8.1-1 of Reference 8.1. The method of analysis is described in Sections 8.1.1.2 through 8.1.1.9 of Reference 8.1. The material properties are shown in Table 8.1-2 of Reference 8.1. The results of the analyses for the Calvert Cliffs ISFSI components are given in Sections 8.1.1.2 through 8.1.1.9. An additional, comprehensive, structural reanalysis of the NUHOMS-24P DSC was performed and documented in References 8.23 and 8.24.

8.1.1.1 Normal Operation Structural Analysis

The loads applicable to the normal operation structural analysis are described in detail in Section 8.1.1.1 of Reference 8.1. The specific application of these loads to the Calvert Cliffs ISFSI design are described in the following paragraphs.

A. Dead Weight Loads

The Calvert Cliffs NUHOMS component weights (dead weight) are calculated based on the design dimensions and materials. Material densities are the same as those in Reference 8.1.

B. Design Basis Internal Pressure Loads

The DSC internal pressures for normal and off-normal conditions are calculated using maximum ambient temperatures and blowdown pressure conditions. As discussed in Reference 8.1, bounding accident condition pressures are conservatively based on cladding failure of all fuel rods in the DSC. The fission gas release fraction is assumed to be 30%, and all the fill gas is assumed to be released. The fuel rod average burnup is assumed to be 50,000 MWD/MTU. The effects of the postulated accident pressures are described in Section 8.2. Reflood pressure is included as a Service Level D condition, experienced during fuel retrieval, because the DSC will not be re-used after retrieval.

C. Design Basis Thermal Loads

The structural loads due to normal operating condition thermal expansion have been evaluated for a range of ambient temperatures from -3°F to 103°F. These temperatures represent the historical extremes recorded at Patuxent River Naval Air Station, some 10 miles from the ISFSI. The long-term average normal ambient temperature is conservatively assumed to be 70°F, as discussed in Section 8.1.1.C of Reference 8.1.

D. Operational Handling Loads

The significant operational handling load is the sliding transfer of the DSC from the cask to the DSC support rails in the horizontal storage module (HSM). Since the Calvert Cliffs DSC weighs less than that of the generic DSC design, these loads are consistent with those in Section 8.1.1.4.B of Reference 8.1.

E. Design Basis Live Loads

Live loads for the Calvert Cliffs ISFSI are enveloped by the generic live load of 200 lbf/ft^2 used in Section 8.1.1.5.A of Reference 8.1.

8.1.1.2 Dry Shielded Canister Analysis

Stresses were evaluated in the DSC due to:

A. Dead Weight Loads
B. Design Basis Normal Operating Internal Pressure Loads
C. Normal Operating Thermal Loads
D. Normal Operation Handling Loads

The analyses performed for DSCs R001 through R024 are similar to those presented in Section 8.1.1.2 of Reference 8.1 for the generic DSC. The ANSYS analytical model of the DSC shell assembly is used for the analysis of pressure, thermal, and handling loads. Stresses due to normal operating pressures are based on a bounding internal pressure of 10 psig, applied as a uniform load to the inner boundary of the analytical model. Also considered was the external hydrostatic pressure loading on the DSC shell, when the 3/8-inch annulus between the DSC shell and the transfer cask is filled with water. Circumferential shell temperature variations are analyzed using the ANSYS three-dimensional solid shell model.

Dry shielded canisters, starting with R025, employ a modified internal basket assembly design (refer to Section 4.2.3.2). The modified DSC was analyzed using analytical methods comparable to those described for DSCs R001 through R024 above. The modified DSC stresses remain within American Society of Mechanical Engineers (ASME) code allowable stresses (Reference 8.24).

8.1.1.3 Dry Shielded Canister Internal Basket Analysis

The DSC basket analysis was performed for:

A. Dead Weight Loads
B. Thermal Loads

The DSC basket dead weight and thermal analyses for DSCs R001 through R024 were performed according to the methodology described in Section 8.1.1.3 of Reference 8.1 with minor variations. The dead weight stresses for the spacer disk for out-of-plane loads (i.e., DSC in the vertical position) were obtained directly from the ANSYS spacer disk analytical model. The dead weight stresses in the plane of the spacer disk (i.e., DSC in the horizontal position) were obtained by conservatively evaluating the center spacer disk ligament using hand calculations for the combined dead load of 12 fuel assemblies. The thermal analysis of the spacer disk was performed by applying the calculated temperature distribution to the ANSYS analytical model.

Dry shielded canisters, starting with R025, employ a modified internal basket assembly design (refer to Section 4.2.3.2). The modified DSC was analyzed in a manner similar to that described for DSCs R001 through R024 above. The spacer disc maximum out-of-plane stress is much lower for the modified DSC because the guide sleeves are not attached to any spacer disc. The modified DSC spacer disc stresses remain within ASME code allowable stresses (Reference 8.24).

The maximum length of the irradiated spent fuel assemblies is 158.00" including thermal expansion. The minimum DSC cavity length is 158.24", including allowance for fabrication tolerances. Using the thermal expansion algorithms in Section 8.1.1.3.B of Reference 8.1, the hot cavity length is 158.47", allowing at least 0.47" of clearance between the fuel assembly and the DSC cavity ends.

8.1.1.4 Dry Shielded Canister Support Assembly Analysis

The Calvert Cliffs DSC support assembly consists of WF 8x40 support rail members with WF 8x48 cross members. A 3/4" x 6" rail cover plate runs the full length of the support rail members and extends into the HSM access opening. The support rail members are supported by structural embedments at the HSM front wall and by the cross members at the center and rear of the module.

The Calvert Cliffs DSC support assembly was analyzed using conservative hand calculations. The DSC support assembly is analyzed for the following loads as discussed in Section 8.1.1.4 of Reference 8.1.

A. Dead Weight Loads
B. Normal Operational Handling Loads
C. Thermal Loads

The DSC support assembly and its end connections were analyzed for the normal and off-normal loads. The calculated stresses were small, and the vertical deflections under the transfer cask loading were less than 0.1".

8.1.1.5 Horizontal Storage Module Analysis

The HSM configuration selected for the Calvert Cliffs ISFSI is a 2x6 array. This configuration is within the envelope of possible HSM configurations evaluated in Reference 8.1, which considers a range of HSM array sizes from a single module to a 2x10 array. Similar loads analyses to those described in Reference 8.1 for the generic HSM design have been performed for the Calvert Cliffs HSM design using site-specific parameters. The Calvert Cliffs HSM array was evaluated for the following normal operation loads.

A. Horizontal Storage Module Dead and Live Loads

The HSM dead and live loads were evaluated using the same methodology as discussed in Section 8.1.1.5 of Reference 8.1.

B. Concrete Creep and Shrinkage Loads

Loads due to creep and shrinkage of the concrete were determined by the same methodology described in Section 8.1.1.5 of Reference 8.1.

C. Horizontal Storage Module Thermal Loads

To evaluate the effects of thermal loads on the HSM, heat transfer analyses for a range of ambient temperatures were performed and the limiting thermal gradients and temperature values at various locations in the HSM determined. A description of the heat transfer analyses is provided in Section 8.1.3. Structural analyses of the HSM for the bounding temperatures were performed for the Calvert Cliffs 2x6 array. The analytical approach used is identical to that documented in Reference 8.1.

D. Radiation Effect on HSM Concrete

The effects of radiation on the compressive strength and modulus of elasticity of concrete were examined in Section 8.1.1.5 of Reference 8.1 and determined to be negligible. The integrated neutron and gamma fluxes are slightly greater for the Calvert Cliffs design, but the effect remains insignificant.

E. Horizontal Storage Module Design Analysis

The flexural and shear strength capacities of the HSM concrete sections were calculated using the ultimate strength method of American Concrete Institute (ACI) 349-85 as described in Section 8.1.1.5.E of Reference 8.1. The resulting capacities are reported in Table 8.1-8.

The calculations associated with the HSM walls, roof, and foundation are based on one-way slab design, and the concrete shear capacity used is that given in Equation 11-3 of ACI 349. The design as presented in the calculation has no mixed requirements from different portions of the ACI Code, but is a calculation performed on a consistent set of design criteria. In addition, careful reading of Section 11.8 of ACI 349 and the commentary to the Code reveal that the deep flexural member rules are intended for beams, which are distinct from slabs.

ACI 349, paragraph 11.5.5.1 specifically excludes slabs and footings from the requirement to have a minimum area of shear reinforcement. In addition, the special provisions for slabs and footings (paragraph 11.11) does not require shear reinforcement unless the factored shear force exceeds the shear strength ϕ Vc where Vc = $2\sqrt{f_{c}'}$ which is the basis for the shear strength determination in the reference calculation. Furthermore, the special provisions for walls (paragraph 11.10) refer to the rules for slabs for shear perpendicular to the plane of the wall, which are the only shear loads of concern for the walls. Shear loads in the plane of the wall are inconsequential as the walls are continuously supported by the foundation slab.

Therefore, the design is adequate and no shear reinforcing is required.

The calculated shear and moment forces, due to dead weight, live loads, creep effects, and normal and off-normal thermal conditions, were well within the capacities of HSM components.

8.1.1.6 Horizontal Storage Module Door Analyses

The HSM access opening for Calvert Cliffs is protected by a door which is similar to that shown in Figure 4.2-5 of Reference 8.1, except that the thickness and materials are different and the door support frame is welded to embedded plates with stud anchors. The Calvert Cliffs HSM access door consists of a 1-3/4" steel plate, 10-3/4" of concrete shielding material, and a 1/4" steel cover plate. Steel angle sections are attached to embedded plates in the HSM front wall to form guides to slide the HSM door vertically. The method of analysis for the Calvert Cliffs door and frame assembly is identical to that presented in Section 8.1.1.6 of Reference 8.1.

The maximum calculated bending and shear stresses in the angle are 23.9 ksi and 0.8 ksi, respectively. The maximum allowable bending and shear stresses, as reported in Section 8.1.1.6 of Reference 8.1 [i.e., American Institute of Steel Construction (AISC) 8th Edition], are 24.0 ksi and 14.40 ksi, respectively. The maximum tensile and shear stresses on the embedded anchor studs are 15.8 ksi and 8.8 ksi, respectively. The ACI 349-85 Appendix B allowables for the American Society for Testing and Materials A108 anchor studs are 45.0 ksi for tension and 38.3 ksi for shear. The combined calculated over allowable ratio for tension and shear is 0.58.

8.1.1.7 Heat Shield Analysis

The HSM heat shield and attachment details are identical to those of the generic design except the roof attachment bolts are smaller diameter. The heat shield analysis presented in Section 8.1.1.7 of Reference 8.1 is directly applicable to the Calvert Cliffs design except that maximum heat shield bending stress is 4.4 ksi and the maximum side wall bolt bending stress is 14.6 ksi.

8.1.1.8 Horizontal Storage Module Seismic Restraint for Dry Shielded Canister

Details of the analysis of the Calvert Cliffs DSC seismic restraint design are provided in Section 8.2.3.2. The seismic restraint weighs less than 30 pounds and is not equipped with a handle.

8.1.1.9 Transfer Cask Analysis

The transfer cask was evaluated for normal operating loads as follows:

A. Transfer Cask Dead Weight Loads

As in Section 8.1.1.9 of Reference 8.1, two cask dead weight cases are evaluated. The first is with a fully loaded cask hanging from its lifting trunnions; the second is the loaded cask supported horizontally on its skid.

B. Transfer Cask Normal Handling Loads

The transfer cask handling loads and their method of analysis are described in Section 8.1.1.9 of Reference 8.1. Transfer cask handling stresses are calculated for: (1) the cask supported at the trunnions vertically, horizontally, and tilted; (2) cask transfer operations; and (3) cask loading and unloading at the HSM.

C. Transfer Cask Normal Operation Thermal Loads

Transfer cask thermal loads are calculated using an axisymmetric cask model. A fuel assembly decay heat power of 15.8 kW was applied as a uniform heat flux to the transfer cask inner surfaces. Convection coefficients, applied as surface loads to the cask outer surfaces, are based on simplified equations for heat loss from various surfaces to air. They are: 0.0066 BTU/hr-in^2-°F, for the cylindrical shell; and 0.0051 BTU/hr-in^2-°F through the cask and plates. Two bounding ambient temperature cases are considered, consisting of -3°F and 103°F, representing the site-specific historical extremes. A bounding solar heat flux of 62 BTU/hr-ft^2, is also applied for the hot ambient case.

8.1.2 OFF-NORMAL LOAD STRUCTURAL ANALYSIS

Table 8.1-1a of Reference 8.1 lists the off-normal operating loads for the generic NUHOMS-24P system. No additional loads have been identified for the Calvert Cliffs ISFSI. The off-normal loads discussed include jammed DSC and off-normal thermal loads. The thermal stresses due to off-normal thermal loads were calculated in the same manner as described for the normal operating thermal loads. An additional, comprehensive, structural analysis of the DSC was performed and documented in Reference 8.23.

<u>8.1.2.1 Jammed Dry Shielded Canister During Transfer</u>

A. Postulated Cause of Jammed DSC

If the transfer cask is not accurately aligned with the HSM, the DSC might become bound or jammed during the transfer operation. The maximum tolerable misalignment for the DSC insertion operation is 0.25" as discussed in Section 8.1.2.1 of Reference 8.1.

B. Detection of Jammed DSC

When DSC jamming occurs, the hydraulic pressure in the ram will increase. When the hydraulic pressure corresponds to a force on the DSC of 20,000 lbf, the DSC will be presumed to be jammed. The normal pushing and pulling forces will be limited to 20,000 lbf with a ram system design capability of up to 80,000 lbf.

C. Analysis of Effects and Consequences

The analyses of the DSC under the assumed jamming and binding conditions are discussed in Section 8.1.2.1 of Reference 8.1. In both jammed DSC scenarios considered (axial sticking and binding), the stress on the DSC body is shown to be much less than the ASME code allowable stress at the ram system maximum force of 80,000 lbf. Therefore, plastic deformation of the DSC body will not occur and there is no potential for rupture.

D. Corrective Actions

Two courses of corrective action are open to the system operators. The operator may choose to apply a ram force of up to 80,000 lbf to the DSC without risk of damage to the DSC or other system components, as discussed in

Section 8.1.2.1 of Reference 8.1. This maximum force is limited by the design of the ram system. Such action could be warranted if the DSC encounters a higher than expected sliding friction coefficient, which could result from partial lubricant failure, debris in the cask annulus or on the support rails.

If the DSC is stuck or bound, then the corrective action will be to reverse the ram force and return the DSC to its initial position, either in the cask or in the HSM. The alignment problem then must be corrected and the transfer operation resumed.

8.1.2.2 Off-Normal Thermal Loads Analysis

The off-normal thermal condition for the Calvert Cliffs ISFSI is evaluated at the same historical temperature extremes of -3°F and 103°F as the normal thermal loads. A higher solar heat flux value is used for the evaluation of the loaded HSM and the DSC inside the HSM for the off-normal case, as discussed in Section 8.1.3.

A. Horizontal Storage Module Off-Normal Thermal Analysis

The analysis of the HSM for off-normal thermal loads is performed using the same methodology as for normal thermal loads, as described in Section 8.1.1.5.c. The DSC support assembly is designed with slotted holes as described in Section 8.1.1.4 of Reference 8.1, and therefore the increase in temperature has no effect on the DSC support structure.

B. Dry Shielded Canister Off-Normal Thermal Analysis

The analysis of the DSC and DSC basket, for the DSC inside the HSM, for off-normal thermal loads is performed using the same methodology as for normal thermal loads described in Sections 8.1.1.2 and 8.1.1.3. As previously stated, the off-normal thermal loads for the transfer cask are identical to the normal thermal loads. Therefore, the off-normal thermal loads for the DSC inside the transfer cask are identical to the normal thermal loads for the DSC inside the transfer cask, and are not considered further.

C. Transfer Cask Off-Normal Thermal Loads Analysis

As previously stated, the off-normal thermal loads for the transfer cask are identical to the normal thermal loads. Therefore, the off-normal thermal loads for the transfer cask and are not considered further.

8.1.3 THERMAL HYDRAULIC ANALYSES

For more information see Reference 8.16.

The following evaluations have been performed for the Calvert Cliffs ISFSI:

A. Thermal Analysis of the HSM
B. Thermal Analysis of the DSC in the HSM
C. Thermal Analysis of the DSC in the Transfer Cask

The analytical models of the HSM, the DSC, and the transfer cask are described in Section 8.1.3 of Reference 8.1.

The method described in Reference 8.1 for calculating the effective thermal conductivity of the fuel region was extended to include the Calvert Cliffs spent fuel assemblies. The experimental results at the E-MAD test facility (Reference 8.3) provide data for specific fuel assemblies with given heat generation and various boundary conditions. To determine the effective thermal conductivity of the fuel region, the maximum fuel clad temperature and a circumferentially uniform boundary temperature are required to utilize the analytical expression described in Reference 8.1. The key parameters in the expression are the temperature at the boundary of the fuel assembly and the peak temperature in the assembly that will simulate the parabolic temperature profile in a heat generating slab.

As in Reference 8.1, the temperature distribution in each fuel assembly in the DSC is different due to differing boundary conditions. Therefore, each fuel assembly in the DSC is considered individually. The resulting maximum temperature in each fuel assembly was located, and two perpendicular planes were passed through this point, as shown in Figure 8.1-9(a). Consistent with the basis for the Reference 8.3 results, the temperature on the sides were averaged to convert the non-symmetric temperature profile to an equivalent symmetric profile in each plane as depicted in Figure 8.1-9(b). Using this temperature value for the boundary temperature along with the maximum temperature in the fuel assembly, the effective thermal conductivity of the fuel region for that fuel assembly was determined, using the analytical expression and the method described in Reference 8.1.

Since the effective thermal conductivity is a function of temperature, an iterative analysis was performed. With the effective thermal conductivities for each fuel assembly in the DSC, the HEATING6 computer program (Reference 8.4) was used with the analytical model of the DSC to perform the thermal analysis of the DSC internal basket assembly and spent fuel assembly regions. The resulting calculated temperature profiles for the fuel regions were then used to calculate new values of effective thermal conductivities. This procedure was repeated until convergence was achieved.

The values at the lower end correspond to fuel assemblies located on the outer edge of the DSC. The values at the upper end correspond to fuel assemblies located toward the center of the DSC. The same procedure was used to calculate effective thermal conductivity of the fuel region for normal, off-normal, and accident conditions described in Section 8.1.3.2.

The differences in the effective fuel thermal conductivity (K_{eff}) and the generic NUHOMS-24P design described in Reference 8.1 are due to the extension of the method described in the NUTECH topical report to the Calvert Cliffs fuel assemblies.

The K_{eff} curve in Reference 8.1 is based on the experimental results provided in the spent fuel dry storage testing at the E-MAD facility (Reference 8.3). Experimental results for temperatures at the canister wall and center rod cladding temperatures from a single fuel assembly were used in Reference 8.1 to calculate the K_{eff}. The experimental data at various canister wall temperatures and the corresponding peak cladding temperature for a single fuel assembly was curve fitted to generate the K_{eff} plot in Reference 8.1.

The extension of this method to Calvert Cliffs fuel assemblies is described here. The analytical expression used to calculate the K_{eff} is the same as that described in Reference 8.1. The temperature distribution in each fuel assembly in the DSC is different due to differing boundary conditions. Therefore, K_{eff} for each fuel assembly in the DSC is considered individually.

The fuel assemblies located on the outer edge of the DSC have larger fuel clad temperature to boundary temperature gradients, as compared to fuel assemblies located towards the center of the DSC. The larger gradients result in lower effective thermal conductivities. Similar trends in the temperature gradient were also observed in the measured test results and analytically predicted by the COBRA-SFS computer code as documented in Reference 8.5.

The thermophysical properties of the materials of construction are given in Tables 8.1-5 and 8.1-6 of Reference 8.1. The thermal analyses were performed with the following ambient air temperatures:

A. Normal Conditions

Winter or summer conditions with an ambient temperature range from -3°F (minimum winter), 70°F (lifetime average), and 103°F (maximum summer) were considered. The -3°F and 103°F temperatures represent the historical extreme ambient temperatures recorded at Patuxent River Naval Air Station, some 10 miles from the Calvert Cliffs ISFSI. For the HSM, the vents were assumed to be open and a solar heat load of 82 BTUH/ft^2 is included for the 70°F and 103°F ambient temperature cases. The minimum (winter) and maximum (summer) temperature conditions were assumed to occur for a sufficient period of time such that steady-state conditions were achieved. The lifetime average ambient temperature of the HSM was assumed to be 70°F, as discussed in Section 8.1.1.C of Reference 8.1. For the transfer cask, winter or summer conditions with an ambient temperature range at -3°F (minimum winter) and 103°F (maximum summer) were considered. A solar heat load of 127 BTUH/ft^2 is included for the 103°F ambient case.

B. Off-Normal Condition

Extreme winter or summer conditions with an ambient temperature range of -3°F to 103°F were considered. For the HSM, the vents were assumed to be open. A solar heat flux of 127 BTUH/ft^2 is conservatively included for the 103°F ambient temperature to maximize the HSM roof concrete surface temperatures. This condition is assumed to occur for a sufficient period of time such that steady-state conditions are achieved. The off-normal thermal conditions for the transfer cask are identical to the normal thermal conditions previously described.

C. Accident Condition

An extreme summer condition with an ambient temperature of 103°F was considered. In addition, the HSM vents were assumed to be completely blocked for a period of 48 hours or less. A solar heat flux of 127 BTUH/ft^2 is conservatively included to maximize the HSM concrete temperatures.

8.1.3.1 Thermal Analysis of the Horizontal Storage Module

The HSM thermal analyses were performed for the design basis ambient air temperatures defined in Section 8.1.3. The model is described in Section 8.1.3.1.C of Reference 8.1.

Convection heat transfer from the DSC and HSM surfaces was modeled using a constant air temperature for the axial air gap regions between the DSC and HSM. These temperatures were also used to calculate the heat transfer coefficients for these gap regions.

Maximum temperatures on the DSC outer surfaces and the concrete inner and outer surfaces were calculated for the normal, extreme winter, and extreme summer ambient conditions, and the postulated accident conditions with blocked HSM vents.

8.1.3.2 Thermal Analysis of the Dry Shielded Canister in the Horizontal Storage Module

The DSC and fuel assembly heat transfer analyses with the DSC inside the HSM were performed for the design basis ambient air temperatures defined in Section 8.1.3. The analytical model is described in Section 8.1.3.2.A of Reference 8.1. The cases of interest are those which maximize fuel cladding temperature, so the evaluations are limited to summer ambient conditions as described in Section 8.1.3.1 of Reference 8.1. These temperatures were used to derive the DSC internal pressures. The spacer disk thermal analyses were performed using the analytical model described in Section 8.1.1 of Reference 8.1.

The peak fuel clad temperature limit of 335°C for the Calvert Cliffs NUHOMS design was calculated using fundamentally the same bounding conservative design criteria and analysis methods previously reviewed and approved by the NRC for the NUHOMS-07P and generic NUHOMS-24P designs (References 8.25 and 8.26).

The maximum allowable cladding temperature for long term storage of the spent fuel is based on Zircaloy behavior models developed to predict the behavior of irradiated materials. The primary applicable models are CSFSM models documented in Reference 8.27. This reference provides a simple relationship for calculating the cladding stress given the rod diameter, cladding thickness and internal pressure. The curves in the Reference are then used to determine the acceptable initial storage temperature for the given cladding stress and cooling time. Bounding end of life fission (plus fill) gas pressure was provided by the fuel manufacturer.

Using the above methodology the resulting maximum storage temperature for the design basis fuel to preclude damage of fuel cladding during long term storage was 335°C. The limiting fuel assembly has a maximum burnup of 50 gwd/mthm and cooling time of 12 years to reach 0.66 kW decay heat. The maximum allowable cladding temperature limit for lower burnups and shorter time resulting in 0.66 kW decay heat per assembly are higher than 335°C.

The acceptable peak clad temperature limit for accident conditions for ISFSI storage is the same as the generic NUHOMS-24P design (Reference 8.26). This limit is based on the empirical work presented in Reference 8.27.

8.1.3.3 Thermal Analysis of the Dry Shielded Canister in the Transfer Cask

The thermal analyses for the cases with the DSC inside the transfer cask were performed for the design basis ambient air temperatures defined in Section 8.1.3. The analyses were conducted using the models described in Section 8.1.3.3 of Reference 8.1.

Maximum fuel cladding and DSC shell temperatures and the average helium temperature were calculated for the normal ambient temperature of 70°F, abnormal summer ambient temperature of 103°F, and the postulated accident conditions with blocked HSM vents.

Prior to specifying NS-3 for the transfer cask neutron shield, the manufacturer (Bisco Products) was requested to provide test data to support the behavior of the NS-3 at elevated temperatures. A series of tests were performed on sealed samples of NS-3 to determine the maximum off gas pressure produced by the product at temperatures up to 280°F. The tests were conducted on sealed NS-3 samples to represent the closed neutron shield and the temperature sustained for sufficient time to produce equilibrium. The samples were cycled to room temperature and back to elevated temperature to ensure repeatability of data. At the conclusion of the tests, NS-3 samples were removed and the final hydrogen content measured. This was combined with the measured free water, produced as a result of NS-3 off gassing, to determine the total hydrogen content available for neutron shielding.

The results of these tests showed that the maximum system pressure due to NS-3 off gassing at 280°F was 45 psig. This is less than 50% of the neutron shield safety relief valve set point pressure of 95 psig.

The results of these tests also showed that the hydrogen loss from NS-3 was less than 10% at 280°F. Since the maximum pressure generated due to off gassing at 280°F was less than 50% of the neutron shield safety relief valve set point pressure of 95 psig, this hydrogen is not lost from the neutron shield cavity. The shielding analysis of the Calvert Cliffs NUHOMS design was performed assuming a conservative 10% hydrogen loss from the NS-3.

Differential thermal expansion of the NS-3 is not a problem as the published coefficient of thermal expansion (7.81×10^{-6}) is less than the 9.85×10^{-6} of the neutron shield stainless steel shell and unanticipated stresses will not occur.

The CCNPP cask liner and neutron shield temperatures are higher than those of the generic NUHOMS-24P design because:

1. The solar heat flux used in the generic NUHOMS-24P design for the 100°F ambient case was 62 Btu/hr-ft^2 compared to 127 Btu/hr-ft^2 used for the 103°F ambient case in the CCNPP ISFSI design.

2. The neutron shield in the generic NUHOMS-24P design is a 3.0" ethylene glycol (50/50 mixture) as compared to a 4.0" Bisco NS-3 for the CCNPP ISFSI design. The thermal conductivity of ethylene glycol is considerably higher than that of NS-3.

TABLE 8.1-8
MAXIMUM HORIZONTAL STORAGE MODULE REINFORCED CONCRETE BENDING MOMENTS AND SHEAR FORCES FOR NORMAL AND OFF-NORMAL LOADS

		HSM INTERNAL FORCES					
Structural Section	**Force[a] Component**	**Dead Weight**	**Creep Effects**	**Live Loads**	**Normal Thermal**	**Off-normal Thermal**	**Ultimate[b][c] Capacity**
Floor Slab	Shear						178
	Moment						1942
Inner Wall	Shear						26.6
	Moment						1027
End Wall	Shear						207
	Moment						1942
Roof Slab	Shear						178
	Moment						1942

NOTE: The calculated forces are less than the ultimate capacity.

For more information see Reference 8.16.

Ultimate Shear Capacity based on ACI 349.85 Section 11.8.3 Equation 11-27 for deep flexural members except for inner wall which is based on Equation 11-28.

(a) Shear values are in kips/ft. Moment values are in inch-kips/ft.

(b) Concrete and reinforcing steel properties were taken at 400°F to conservatively envelope all ambient cases.

(c) Ultimate capacities are reported for a 12" section of HSM using f'_c and f_y values at 400°F.

8.2 ACCIDENTS

This section addresses design events of the third and fourth types as defined by ANSI/ANS 57.9-1984 (Reference 8.2), and other credible accidents consistent with Title 10, Code of Federal Regulations (CFR) Part 72 which could impact the safe operation of the Calvert Cliffs ISFSI. The postulated events identified in Section 8.2 of Reference 8.1 and addressed herein for the Calvert Cliffs ISFSI are:

A. Loss of Air Outlet Shielding
B. Tornado Winds/Tornado Missile
C. Earthquake
D. Flood
E. Transfer Cask Drop
F. Lightning
G. Blockage of Air Inlets and Outlets
H. DSC Leakage
I. Accidental Pressurization of DSC

In addition, two additional Calvert Cliffs site-specific accidents have been identified and addressed. These are:

J. Forest Fire
K. Liquified Natural Gas (LNG) Plant or Pipeline Spill or Explosion

The accidents considered, and the associated components affected by each accident, are summarized in Table 8.2-1.

In the following sections, each accident condition is evaluated for applicability to the Calvert Cliffs ISFSI. For each applicable condition the accident cause, structural, thermal, radiological consequences, and recovery measures required to mitigate the accident are discussed. Where appropriate, resulting accident condition stresses were combined with those of normal operating loads in accordance with the load combination definitions of Section 3.2.5. Load combination results for the HSM, DSC, and transfer cask are discussed in Section 8.2.12. A reanalysis of the effects of all accidents on the DSC structure was performed and documented in Reference 8.23.

Reflood pressure is included as an ASME Service Level D activity but is not identified as an accident.

8.2.1 LOSS OF AIR OUTLET SHIELDING

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

The Calvert Cliffs air outlet shielding blocks are different from those described in Section 8.2.1 of Reference 8.1 in that they are designed to remain in place and withstand all design events including the effects of tornado missiles. Following the occurrence of such an event, the HSM air inlets and outlets will be inspected to confirm their condition. Therefore, this accident event is not applicable to the Calvert Cliffs ISFSI design and no further consideration is required.

8.2.2 TORNADO WINDS/TORNADO MISSILE

8.2.2.1 Cause of Accident

The tornado wind and missile loadings presented in Section 3.2.1 are used as the design basis for this accident condition.

8.2.2.2 Accident Analysis

The applicable parameters of the design basis tornado (DBT) are specified in Section 3.2.1. The DBT parameters specified in Section 3.2.1 are identical to those used in Reference 8.1 in the determination of forces on structures for this accident. The determination of tornado wind and tornado missile forces acting on the HSM was performed for Calvert Cliffs 2x6 module array using the same methodology as that documented in Section 3.2.1 of Reference 8.1. The HSM walls, roof, and foundation are connected by reinforcing steel. The transfer cask is designed for the tornado wind loads and tornado missiles defined in Section 3.2.1.

A. Effect of DBT Wind Pressure Loads on HSM

The effects of DBT wind loads on the HSM were evaluated using the STRUDL finite element model. An analysis was also performed to evaluate the potential for overturning and sliding of the 2x6 module array due to wind and seismic forces. The methodology employed in evaluating the applied and resisting forces and moments is the same as that used in Section 8.2.2.2 of Reference 8.1. The results showed that there was a sufficient margin of safety against the HSM overturning and sliding.

B. Effect of DBT Wind Pressure Loads on Transfer Cask

The transfer cask was evaluated for DBT wind pressures using the same methodology as Section 8.2.2.2 of Reference 8.1. The stabilizing moment of the Calvert Cliffs transfer cask, skid, and trailer, with a combined total weight of 255 kips and a trailer half wheel base of 72", is 1528 k-ft. The overturning moment for the combined projected area of 212 ft^2 and a total height of 147" is 515 k-ft. Therefore, the factor of safety against overturning of the Calvert Cliffs transfer cask is approximately 3.0.

As reported in Section 8.2.2.2 of Reference 8.1, the transfer cask stresses due to DBT wind pressures are small and were not considered further. Since the Calvert Cliffs transfer cask shell and cover plate thicknesses are equal to or greater than those used in the analysis of Reference 8.1, transfer cask stresses due to DBT wind loads are insignificant and do not need to be included in the load combinations.

C. Horizontal Storage Module Missile Impact Analysis

The effects of DBT missile loads on the HSM were evaluated using the same methodology as Section 8.2.2.2 of

Reference 8.1. The computed HSM forces and moments due to tornado missile impact were found to be well within the ultimate capacity.

As discussed in Section 8.2.1, the design of the Calvert Cliffs HSM air outlet shield blocks differs from that documented in Reference 8.1. The Calvert Cliffs air outlet shield blocks have been designed to withstand the effects of tornado generated missile loads, and therefore are not designed to be replaced. The Calvert Cliffs HSM air outlet shield blocks are integrally cast with the HSM concrete. The effects of DBT missile loads on the air outlet shield blocks were evaluated using similar methodology to that for the HSM walls documented in Section 8.2.2.2 of Reference 8.1. The maximum computed shear and moment due to tornado missile impact on the HSM air outlet shield blocks are 3.6 kips/ft and 293 inch-kips/ft, respectively. The corresponding air outlet shield block shear and moment capacities are 11.5 kips/ft and 364 inch-kips/ft, respectively.

D. Transfer Cask Missile Impact Analysis

The transfer cask was analyzed for the effects of the 3,967 lb automobile and 276 lb, 8" diameter, blunt-nosed shell tornado missiles specified for the HSM in Table 3.2-1 of Reference 8.1. The 1" diameter steel spherical missile was not evaluated as there are no critical openings, and the effects of the small sphere are enveloped by the 8" shell missile.

As described in Section 5.1.1, all cask handling outside the Auxiliary Building is performed with the cask secured horizontally on the transfer trailer. Therefore, the analysis was performed for the cask secured in the horizontal position on the support skid. The criteria used to evaluate the adequacy of the transfer cask for tornado generated missiles were stability, penetration resistance, and stress.

The components considered in this evaluation were the transfer cask structural shell and top and bottom cover plates. It was conservatively assumed that the solid neutron shield provides no structural strength or penetration resistance.

1. Stability Analysis

It was conservatively assumed that the missile impacts the uppermost part of the cask. The maximum angle of rotation of the cask/skid/trailer arrangement at impact was calculated as 1.9°, based on the conservation of angular momentum. Tip-over (i.e., instability of the cask/skid/trailer) occurs when the center of gravity of the cask is directly above the point of rotation, which is 35.2° from vertical. The maximum calculated rotation of 1.9° due to missile impact is approximately 5% of that necessary to cause overturning. Therefore, the stability of the cask/skid/trailer arrangement is maintained.

2. Penetration Analysis

Penetration of the cask structural shell by the 276 lb rigid missile was evaluated using two formulas obtained from the listed references. The energy absorbing capacity of the neutron shield material was conservatively ignored for this calculation. The first approach, suggested by Nelms (Reference 8.6), is for a leadbacked shell:

$$T = (KE/2.4\ S_u\ D^{1.6})^{0.71}$$
$$= 0.50"$$

Where:

T	=	Minimum required steel plate or shell thickness to resist penetration
KE	=	Kinetic energy = 1/2 mV^2
m	=	Mass of missile = 276 lb/g
	=	0.715 lb. sec^2/in
V	=	Velocity of missile
	=	2,218 in/sec
S_u	=	Ultimate strength of cask structural shell
	=	70,000 psi
D	=	Diameter of missile
	=	8.0"

The second formula used was developed by the Ballistic Research Laboratory (Reference 8.7):

$$T = (KE^{2/3}/672\ D)$$
$$= 0.52"$$

Where:

KE	=	Kinetic energy = 1/2 mV^2
m	=	Mass of missile
	=	8.57 lb sec^2/ft
V	=	Velocity of missile
	=	184.8 ft/sec
D	=	Diameter of missile
	=	8.0"

Both methods produce a consistent result which shows a predicted penetration of 0.5" compared to the minimum structural shell thickness of 1.5". Therefore, the DBT missile will not penetrate the cask structural shell and the DSC will not be affected by a DBT missile impacting the cask.

3. Stress Analysis

Conservative hand calculations were performed to determine the peak stresses in the cask shell, and the top and bottom cover plates due to DBT missile loads. The calculated stresses were lower than the allowable stresses. The analytical method for each of the load cases shown in this table are briefly described below.

a. Massive Missile Impact: Using the principle of conservation of angular momentum, the total impact forces were calculated at equilibrium. These forces were applied as a live load to the cask structural shell and as a pressure load to the top and bottom cover plates. The analytical method used was the same as that described above for the DBT wind pressure loads.

b. Penetration Resistance Missile: The impact force due to the 8" diameter, 276 lb missile was calculated using the principle of conservation of momentum to be 63.4 kips. Case 9a, Table 31 of Reference 8.8 was used to calculate the membrane and bending stress for the cask shell while Cases 16 and 17, Table 24 of Reference 8.8 were used to calculate the stresses in the top and bottom cover plates, respectively.

8.2.2.3 Accident Dose Consequences

All components of the ISFSI are capable of safely withstanding tornado wind loads and tornado generated missiles. Therefore, there is no accident dose associated with the DBT.

8.2.3 EARTHQUAKE

8.2.3.1 Cause of Accident

As specified in Section 3.2.3, an earthquake is postulated to occur at the Calvert Cliffs ISFSI with peak ground acceleration values of 0.15g horizontal and 0.10g vertical.

8.2.3.2 Accident Analysis

The analytical methods to evaluate earthquake loads for the Calvert Cliffs ISFSI are similar to those documented in Section 8.2.3.2 of Reference 8.1. The HSM and DSC support seismic analyses were performed using the Calvert Cliffs site-specific peak ground accelerations. The DSC and transfer cask are evaluated using the seismic accelerations discussed in Section 8.2.3.2 of Reference 8.1.

A. Dry Shielded Canister Seismic Analysis

1. Dry Shielded Canister Seismic Stress Analysis

The DSC seismic stress analysis for DSCs R001 through R024 was performed using the methodology of Section 8.2.3.2 of Reference 8.1. Seismic accelerations of 1.5g horizontal and 1.0g vertical are used in the evaluation per Reference 8.1.

Dry shielded canisters, starting with R025, employ a modified internal basket assembly design (refer to Section 4.2.3.2). The modified DSC was analyzed using analytical methods comparable to those described for DSCs R001 through R024 above. The modified DSC stresses remain within ASME code allowable stresses (Reference 8.24).

2. Dry Shielded Canister Seismic Stability Analysis

Dry shielded canister lift-off from the DSC support assembly rail during a seismic event was evaluated using the methodology of Section 8.2.3.2 of Reference 8.1 for horizontal and vertical accelerations of 0.41g and 0.26g, respectively. These accelerations correspond to the Regulatory Guide 1.60, Revision 1 peak response values for 7% damping using the Calvert Cliffs peak ground accelerations. The applied and stabilizing moments for the Calvert Cliffs DSC are calculated to be 951 inch-kips and 995 inch-kips, respectively, resulting in a factor of safety against lift-off of 1.05.

B. Horizontal Storage Module Seismic Analysis

1. Horizontal Storage Module Seismic Stress Analysis

The HSM seismic stress analysis was performed using the methodology described in Section 8.2.3.2 of Reference 8.1. A STRUDL finite element model was used to perform the analysis. Seismic accelerations of 0.51g horizontal and 0.33g vertical were used in the analysis. The resulting forces and moments in the HSM were found to be well within the ultimate capacity.

2. Horizontal Storage Module Seismic Stability Evaluation

The stability of the HSM during a seismic event was evaluated using the methodology of Section 8.2.3.2 of Reference 8.1. The Calvert Cliffs 2x6 module array was evaluated for sliding and overturning of the entire array, including the foundation. Accelerations of 0.41g horizontal and 0.26g vertical, which correspond to the peak structural acceleration values from Regulatory Guide 1.60, Revision 1 using the Calvert Cliffs peak ground accelerations, are used in this evaluation. The results showed that there was a sufficient margin of safety against the HSM seismic overturning and sliding.

C. Dry Shielded Canister Support Assembly Seismic Analysis

The DSC support assembly was analyzed for horizontal and vertical seismic accelerations of 0.61g and 0.39g, respectively, which correspond to the Regulatory Guide 1.60, Revision 1 peak structural accelerations for Calvert Cliffs peak ground accelerations, and indicates a multiple mode factor in accordance with NUREG-0800, Revision 1. The evaluation was performed using conservative hand calculations and included the dead weight of the DSC. The maximum calculated axial plus bending stress and shear stress in the wide flange support rails are 17.1 ksi and 11.5 ksi, respectively. The maximum calculated bending stress and shear stress in the wide flange cross member support beams are 22.3 ksi and 13.6 ksi, respectively. These compare with AISC Code allowables of 26.3 ksi for bending and 14.9 ksi for shear. The stresses due to the two horizontal and vertical seismic loads were combined absolutely and included in the load combination results reported in Section 8.2.12.

The use of the multi-mode factor is consistent with the approved NUHOMS topical report. As discussed in Section 8.2.3.2.A(ii) of the topical report, "The factor of 1.5 used in the DSC analysis to account for multi-mode behavior need not be included in the seismic accelerations for this analysis, as the potential for lift off is due to rigid body motion, and no frequency content effects are associated with this action." This analysis approach was accepted by the NRC for the topical report.

D. Transfer Cask Seismic Analysis

As discussed in Section 8.2.3.2 of Reference 8.1, the transfer cask may not be subjected to a horizontal acceleration in excess of 0.40g while standing vertically in the cask washdown pit. The maximum acceleration of the Calvert Cliffs cask washdown pit floor is 0.30g (zero period acceleration), which is within the Reference 8.1 requirements.

The seismic stress analysis for the Calvert Cliffs transfer cask supported by the transport trailer/skid was performed using the same methodology and seismic loads described in Section 8.2.3.2 of Reference 8.1.

8.2.3.3 Accident Dose Consequences

Major components of the Calvert Cliffs ISFSI have been designed and evaluated to withstand the forces generated by the Design Basis Earthquake (DBE). Hence, there are no dose consequences.

8.2.4 FLOOD

As discussed in Section 3.2.2, flood loads are not applicable to the Calvert Cliffs ISFSI.

8.2.5 CASK DROP

A transfer cask drop is any uncontrolled vertical movement resulting in an impact with a large horizontal surface. An uncontrolled vertical movement includes a cask tip-over and the cask falling off of the transfer trailer.

For more information see Reference 8.16.

8.2.5.1 Cause of Accident

This section addresses the structural integrity of the transfer cask, the DSC, and its internals under a postulated transfer cask accident condition. As discussed in Section 8.2.5.1 of Reference 8.1, an actual drop event is not considered credible. However, consistent with the criteria of Reference 8.1, it is postulated that the transfer cask described in Section 4.3 with the DSC inside will be subjected to an end, side, or oblique drop with a maximum height of 80" onto a thick concrete slab. A drop of greater than 80" is not considered because (a) transfer inside the Auxiliary Building will be performed using a single-failure-proof crane and (b) the transfer trailer and haul road are designed such that the transfer cask cannot be raised greater than 80" from the ground.

The transfer cask is transported along an asphalt or concrete paved road which is 16' wide and has 7 to 8' shoulders. The road is approximately 3,300 linear feet with slopes which range from 0% to 3% except for an approximate 50' length which carries a 5.7% slope. The roadbed is level except for a negligible 1% slope required to create a crown in the road for drainage and a transverse slope at any point along the transportation route of less than 10%. The shoulders are either level with the road or slope up from the road. In those locations where the paved road abuts up to existing blacktop, or concrete paving, the shoulder is discontinued. The shoulder may be paved, gravel or soil and contain typical roadside fixtures, including curbs, fences, guard rails and light poles which do not constitute potential puncture devices for the cask during a drop. The shoulders do not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture device during a cask drop. For the entire route that the transfer cask is transported there will exist a minimum 8' wide zone that is at or above the roadbed elevation.

The transfer trailer braking system is not operable independent of the prime mover. However, failure of the prime mover will cause the trailer braking system to fail-safe, that is "lock tight."

8.2.5.2 Accident Analysis

The Calvert Cliffs transfer cask and DSC were evaluated for the effects of a drop of 80" onto a hard concrete surface. The maximum computed surface hardness for the Calvert Cliffs Nuclear Power Plant (CCNPP) and the ISFSI was computed based on Reference 8.14, assuming a 3' thick reinforced concrete slab backed by well compacted sand and/or gravel. Based on the methodology and information provided in Reference 8.14, the target hardness was computed to be 112,000 for the end drop and 33,300 for the side drop. This resulted in maximum decelerations of 51g for the end drop case and 31g for the side drop. The DSC and transfer cask were conservatively designed for the 75g deceleration value discussed in Reference 8.1, which bounds the computed deceleration values for an 80" drop.

Dry Shielded Canister

The Calvert Cliffs NUHOMS-24P DSC shell was analyzed for an 80" drop accident using the analytical methods presented in Section 8.2.5.2 of Reference 8.1 for DSCs R001 through R024. An ANSYS model of the DSC shell was used to perform the vertical drop analysis.

The evaluation of the DSC shell for a postulated horizontal drop presented in Reference 8.1 bounds the Calvert Cliffs drop accident. The shell assembly, the top shield plug, and the basket assembly are heavier for the Calvert Cliffs design than the Oconee design, which may have been the basis for the Reference 8.1 analysis. The "end caps," or shield plugs, for the Calvert Cliffs DSC design are thicker than the Oconee design and, therefore, are heavier. However, for the side drop the load is carried in bearing on the inside surface of the transfer cask, i.e. there is no substantial bending mechanism at the end cap/shell juncture for the side drop event. Therefore there will not be significant bending stresses at the end cap/shell juncture as a result of the side drop. The weight of the lead shielding will have a substantial effect on the calculated end drop stresses for bending the shell, which has been evaluated specifically for Calvert Cliffs. The remaining DSC components were evaluated for the horizontal drop case using the same methodology as presented in Section 8.2.5.2 of Reference 8.1 for the Calvert Cliffs specific configuration. This included the consideration of misalignment of fuel assembly spacer grids and DSC spacer discs, where necessary (Reference 8.23).

The DSC basket assembly was analyzed for a vertical drop using an ANSYS model. In DSCs R001 and R024, the guide sleeves are attached to the bottom spacer disc with either clip angles or folded over the welded tabs cut from the guide sleeve wall. The attachments may fail at a deceleration rate of less than the design basis 75g, but in a bounding analysis they were conservatively assumed to remain intact up to 75g. With the attachments remaining intact, the guide sleeve loads are transferred to the spacer disc and support rods.

The integrity of the spacer disc during the postulated vertical drop was evaluated using a very detailed plastic analysis (Reference 8.23). The 1/4 symmetry ANSYS spacer disc model was incrementally loaded with and without guide sleeve masses until the 75g static equivalent drop load was reached. The calculated stresses were below the ASME Level D plastic analysis allowables. In addition, a very detailed plastic analysis was performed for the support rods, which determined that the rods would remain stable well beyond the 75g deceleration.

The expected failure of the guide sleeve to spacer disc welds during a drop accident event are not expected to have any effect on the retrievability of the fuel.

The guide sleeve to bottom spacer disc welds are provided for assembly of the DSC basket and to prevent the guide sleeves from moving during transportation. The lightweight guide sleeves are attached to the bottom spacer disc by four 0.25 in. plug welds or four fillet welds. For the design basis vertical drop load of 75g, the maximum calculated weld stress may exceed the Service Level D fillet weld allowable of 22.4 ksi.

Failure of the welds during a top end drop would result in the guide sleeves moving less than 4" until it impacts the inner cover plate. A bottom drop of the DSC would result in the guide sleeves moving less than .12" before impacting on the bottom cover plate. For the design basis 75g impact, the calculated guide sleeve compressive stress is below the AISC normal allowable compressive stress of 21.4 ksi for a column of these dimensions. Therefore, failure of the welds during a drop accident may result in minor deformation of the guide sleeves but would have no effect on the retrievability of the fuel.

The drop load stresses are all within the applicable Service Level D allowables. The fuel retrievability from the DSC will be assured for the postulated 80" drop accident.

Modified DSC Basket Assembly

Dry shielded canisters starting with R025 employ a modified internal basket assembly design (refer to Section 4.2.3.2). Structural analyses for the major DSC structural elements, including the shell, cover plates, spacer discs and support rods were performed using ANSYS finite element analysis. The modified DSC drop load stresses were determined to be within ASME code allowable stresses (Reference 8.24).

The guide sleeves in the modified DSC design are not attached to the spacer discs and the sleeves may move through the spacer discs openings until bearing occurs against an inner cover plate. In the case of a vertical drop, the only load acting on the guide sleeves in the longitudinal direction is that due to self weight. There is adequate margin against buckling of the guide sleeve. Similarly, for the spacer discs, the only load acting on the spacer discs during a vertical drop is the self weight of the disc. The discs remain elastic throughout the event. No fuel retrieval difficulties will occur due to guide sleeve or spacer disc elastic deflections.

Evaluation of the modified DSC guide sleeve for a horizontal drop conservatively considered a condition where a spent fuel assembly end plate is offset approximately 1" away from the face of the nearest spacer disc. The guide sleeve stresses remain below what is required to pierce the sleeve wall, and therefore, removal of an intact spent fuel assembly will not be prohibited.

The guide sleeves in the modified DSC have two extraction stops (metal tabs) that are mounted to the outside walls of each guide sleeve. The extraction stops are intended to prevent guide sleeve withdrawal in the event any incidental binding should occur during withdrawal of a spent fuel assembly. Extraction forces on the extraction stops have been analyzed. Extraction loads may cause minor dimpling of the guide sleeve, but will not affect the ability to retrieve an intact fuel assembly.

Transfer Cask

The Calvert Cliffs NUHOMS-24P transfer cask was analyzed for the 80" drop height accident (Reference 8.1) using the ANSYS 3-D transfer cask one-half model (Reference 8.33). For the vertical, horizontal, and corner drop orientations, the contacting surface was assumed to be rigid. A static equivalent load of 75g was applied. The internal loading of the DSC was represented as pressure loadings applied to the transfer cask inner surfaces.

The maximum computed decelerations for the transfer cask at the Calvert Cliffs ISFSI are bounded by the design deceleration of 75g. The integrity of fuel assemblies contained within a DSC, following a postulated 75g drop, was analyzed. The analyses consisted of an evaluation of the impact of the drop on all of the fuel assembly components; namely, the fuel rods, guide tubes, spacer grids, retention grid, and upper and lower end fittings. The objectives of evaluation were to determine the impact on safety issues, such as confinement, criticality, thermal response, and retrievability.

Stress intensities were calculated for the fuel rods, guide tubes, lower end fitting (including retention grid) (References 8.34 and 8.35). The ASME code allowables for these components were determined at temperatures that envelop the maximum cladding temperature of 635°F, and were 58.95, 57.96, 57.15 ksi, respectively. The calculated stress values were below the code allowables. Therefore, none of these components would fail following a drop.

The upper end fitting was analyzed for its limiting scenario of an upside-down vertical drop (Reference 8.36). The analysis was based on the consideration of the elastic-plastic material behavior. The acceptance criterion used was the lack of failure, rather than meeting the code minimum stress and strain values. It was determined that the upper end fitting ligaments would not fail, and that at least a factor of 2 margin would be available against ductile tearing. Any deformation of the upper end fitting ligaments would not impact fuel assembly retrievability.

The spacer grids were analyzed (Reference 8.37). The guide sleeve, in which the assembly resides, was assumed perfectly rigid for conservatism. An equivalent static impact force of 75g was used in the analysis to represent the worst case loading experienced by the spacer grids during a DSC accidental horizontal drop. The Zircaloy-4 spacer grid that is part of a spent fuel assembly is expected to be irradiated and thus exhibit a brittle behavior. Nonetheless, both brittle and ductile (un-irradiated) cases were considered in the calculation.

Based on the results of this evaluation, it was determined that major structural damage of the spacer grid would occur from a 75g accidental drop scenario for both ductile and brittle material behavior assumptions. In addition, the perimeter strip would likely fail at the lower levels of the spacer grid and thus allow fuel rods to be relocated from their original grid locations, and create the possibility of one of them getting wedged against the guide sleeve. It was further determined that, with this possibility, an additional pull force of about 220 lbs. would be required for retrieving the fuel assembly from the DSC. The additional required force is within the capacity of the fuel handling machine. Therefore, retrievability of the fuel assembly from the DSC would not be compromised.

The effect of impact of a broken spacer grid fragment, during a horizontal drop, on the fuel rod cladding was investigated (Reference 8.38). Various cladding vs. fragment orientations and edge conditions were considered. The maximum cladding wall stress was found to be less than the allowable stress of 80.5 ksi at 635°F.

The failure of spacer grids was determined also to cause a change in the fuel rod pitch from 0.58 inch to 0.465 inch. The impacts of this change on the criticality and cladding temperature were analyzed. The criticality calculation determined that the effective multiplication factor (k_{eff}) was still less than 0.95 (Reference 8.39). The cladding temperature evaluation determined that the temperature for the reduced rod pitch would be lower than that for the normal rod pitch because of better conductivity of the new arrangement (Reference 8.40).

8.2.5.3 Accident Dose Consequences

The cask drop analyses have shown that transfer cask, DSC, its internal basket assembly, and contained fuel will maintain their structural integrity through a cask drop. For the purpose of demonstrating the safety of the NUHOMS-24P system, it is conservatively assumed that the entire cask solid neutron shield is lost as a result of the design drop accident. This would result in an increase in the cask surface contact dose from 141 mrem/hr to 1126 mrem/hr (References 8.41, 8.42, and 8.43). An on-site worker at an average distance of 15' for the 8-hour recovery time, as discussed in Reference 8.1, would receive an additional dose of 776 mrem (97 mrem/hr x 8 hr).

8.2.6 LIGHTNING

8.2.6.1 Cause of Accident

Lightning striking an HSM and causing an off-normal operating condition is not considered a credible accident given the ISFSI lightning protection system provided. The lightning protection system for the ISFSI is designed in accordance with the Lightning Protection Code (Reference 8.15). This system precludes any damage to an HSM or its contained DSC due to lightning.

8.2.6.2 Accident Analysis

Should lightning strike an HSM, the normal operation of the HSM will not be affected since the current discharged by the lightning will follow the low impedance path to ground provided by the lightning protection system. Therefore, the DSC and HSM will not be damaged by heat or mechanical forces generated by current passing through the higher impedance concrete. Since the HSMs require no equipment for continued operation, the resulting current surge from a lightning strike will not affect the normal operation of the ISFSI.

8.2.6.3 Accident Dose Consequences

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

8.2.7 BLOCKAGE OF AIR INLETS AND OUTLETS

This accident involves the complete and total blockage of all HSM air inlets and outlets for a period of 48 hours.

8.2.7.1 Cause of Accident

Since the ISFSI is located outdoors, HSM air inlets and outlets could potentially be blocked by debris from such unlikely events as tornadoes. Independent Spent Fuel Storage Installation design features such as a perimeter fence and separation of air inlets and outlets reduce the potential for this accident.

8.2.7.2 Accident Analysis

The stresses caused by the additional weight of debris blocking the air inlets and outlets are bounded by the structural consequences of other accidents described in this section (i.e., tornado and earthquake analyses). The thermal consequences of this accident result from heating of the DSC and HSM due to the loss of natural convection cooling.

The thermal analyses to determine the temperature rise for the Calvert Cliffs HSM and DSC components due to blocked vents were performed using the methodology described in Section 8.2.7.2 of Reference 8.1. The calculated pressures in all cases were less than the design basis accident pressure of 50 psig discussed in Reference 8.1. The design basis pressure was considered in the DSC accident pressure evaluation presented in Section 8.2.9.

The thermally-induced stresses for the HSM for the blocked vent case were calculated using the STRUDL analytical model and the methodology discussed in Section 8.2.7.2 of Reference 8.1.

8.2.7.3 Accident Dose Consequences

There are no off-site dose consequences as a result of this accident. The only possible dose increase is related to a recovery operation where the on-site worker could receive an additional 584 mrem (73 mrem/hr x 8 hr) during an estimated 8 hour debris removal period (References 8.41, 8.42, and 8.43).

8.2.8 DRY SHIELDED CANISTER LEAKAGE

For more information see References 8.16 and 8.19.

As described in Section 3.3.2, the DSC is designed to ensure no leakage and the analysis for normal and accident conditions described in Reference 8.1 and this document have shown that there are no credible events which can breach the DSC pressure boundary or fail the double seal welds at each end of the DSC. However, to demonstrate the safety of the NUHOMS-24P system, a total and complete instantaneous leak of a single DSC is postulated, as described in Section 8.2.8 of Reference 8.1.

This postulated accident is the instantaneous release directly to the environment of fission gasses (mainly Kr-85) contained in all the fuel rods in all 24 fuel assemblies. This accident assumes that all fuel rods are ruptured and that concurrent DSC leakage occurs. Nuclide release fractions were obtained using the methodology of ANSI/ANS 5.4-1982. All other components of the storage system remain intact.

The following release mechanisms were evaluated in the manner of SAND80-2124 (Reference 8.28): impact rupture, burst rupture, diffusion, leaching, oxidation, and crud-release. It was concluded that noble gases may escape by diffusion and no other release mechanisms were probable.

Two systems for radionuclide confinement are provided by the NUHOMS-24P system: fuel rod cladding confines the fuel and fission products; and the DSC contains the contents of the fuel assemblies and the crud adherent to the fuel rods. Under normal conditions, both confinement features are intact. If any of the fuel rod cladding fails during storage, the DSC will become pressurized with a mixture of helium (the DSC fill gas) and fission gases. It is noted that the criteria for fuel cladding temperature in storage is established based on a probability of failure of the peak temperature fuel rods of less than 0.5% (PNL-6189) (Reference 8.29). In the postulated accident, the DSC is assumed to be pressurized by failure of 100% of all fuel rods, worst-case fission gas release fractions, and an elevated temperature equal to the worst-case thermal accident conditions. The DSC is designed and shown by structural analysis to withstand this pressure with a substantial margin of safety. Additionally, the DSC closure welds are fully redundant and both are welded and inspected to the standards of the ASME Code. While the release of 100% of the cladding gap fission gas is assumed for the purpose of accident assessment, no mechanistic release path exists.

Diffusion of the noble fission gas through the canister shell is assumed to occur. The release of particulate material would require gross breach of the DSC pressure boundary. Because of the ductile nature of the canister materials, the quality of construction inherent in the ASME Code, and the conservative nature of the design described above, a breach of sufficient size to cause release of radioactive particulate matter is incredible.

Therefore, complete particulate confinement is assumed in the referenced accident analyses.

The OCRWM Database (Reference 8.30) was used as a basis for the gross fuel radionuclide source term. A lengthy list of the nuclides present in one metric ton of heavy metal was extracted from the database and has been attached to this response. The data were obtained using ORIGEN calculations for 8 year cooled, 45,000 MWD/MTU burnup PWR fuel. The values in the "Curies" column should be multiplied by (0.386 MTHM * 24 assemblies/DSC) to obtain the total source term per DSC.

Argon, krypton, and radon are the only noble gases available for release. Kr-85 was determined to be the only significant contributor, as confirmed by Elias and Johnson (Reference 8.31). A release fraction of 2.1% was calculated using the methodology of ANSI/ANS-5.4-1982 (Reference 8.32). One-hundred percent of the "gap activity" was presumed to be available for release into the DSC cavity. Furthermore, the DSC release fraction was assumed to be 100%. The amount of Kr-85 used as a radiological source term for one ruptured DSC was therefore:

$$Q = 7.13E+03 \frac{Ci}{MTHM} \times 0.386 \frac{MTHM}{Assy} \times 24 \frac{Assy}{DSC} \times 0.021$$
$$= 1.39E+03\ Ci$$

Spent Fuel Repository Characteristics Data Base
Developed by: Oak Ridge National Laboratory, Oak Ridge, TN

Type of Reactor: PWR 45,000 Elapsed Decay: 8 years
All isotopes representing: All nuclides

Isotope	Curies	Percentage Of Total
H 3	6.25E+02	0.109 %
BE 10	6.66E-06	0.000 %
C 14	1.69E+00	0.000 %
SI 32	2.57E-07	0.000 %
P 32	2.57E-07	0.000 %
S 35	2.40E-09	0.000 %
CL 36	1.30E-02	0.000 %
AR 39	8.53E-05	0.000 %
AR 42	4.64E-13	0.000 %
K 40	5.90E-09	0.000 %
K 42	4.64E-13	0.000 %
CA 41	2.08E-04	0.000 %
CA 45	1.53E-06	0.000 %
SC 46	1.11E-10	0.000 %
V 50	1.11E-14	0.000 %
MN 54	1.24E+00	0.000 %
FE 55	6.18E+02	0.108 %
FE 59	9.34E-18	0.000 %
CO 58	2.55E-09	0.000 %
CO 60	2.87E+03	0.502 %
NI 59	4.76E+00	0.001 %
NI 63	7.08E+02	0.124 %

Isotope	Curies	Percentage Of Total
ZN 65	2.59E-02	0.000 %
SE 79	5.53E-01	0.000 %
KR 81	7.17E-07	0.000 %
KR 85	7.13E+03	1.247 %
RB 87	2.89E-05	0.000 %
SR 89	2.78E-12	0.000 %
SR 90	7.97E+04	13.934 %
Y 90	7.97E+04	13.934 %
Y 91	8.66E-10	0.000 %
ZR 93	2.61E+00	0.000 %
ZR 95	2.49E-08	0.000 %
NB 93M	1.02E+00	0.000 %
NB 94	1.62E+00	0.000 %
NB 95	5.72E-08	0.000 %
NB 95M	1.85E-10	0.000 %
MO 93	3.20E-02	0.000 %
TC 98	9.47E-06	0.000 %
TC 99	1.70E+01	0.003 %
RU103	6.00E-17	0.000 %
RU106	2.26E+03	0.395 %
RH102	2.75E-01	0.000 %
RH106	2.26E+03	0.395 %
PD107	1.54E-01	0.000 %
AG108	1.27E-03	0.000 %
AG108M	1.43E-02	0.000 %
AG109M	1.57E-02	0.000 %
AG110	2.12E-02	0.000 %
AG110M	1.60E+00	0.000 %
CD109	1.57E-02	0.000 %
CD113M	5.33E+01	0.009 %
CD115M	3.46E-17	0.000 %
IN113M	1.56E-05	0.000 %
IN114M	2.72E-16	0.000 %
IN115	1.57E-11	0.000 %
SN113	1.56E-05	0.000 %
SN119M	1.38E+00	0.000 %
SN121M	8.87E-01	0.000 %
SN123	5.12E-04	0.000 %
SN126	1.04E+00	0.000 %
SB124	4.37E-12	0.000 %
SB125	2.40E+03	0.420 %
SB126	1.45E-01	0.000 %
SB126M	1.04E+00	0.000 %
TE123	5.23E-12	0.000 %
TE123M	1.26E-06	0.000 %
TE125M	5.86E+02	0.102 %
TE127	1.05E-04	0.000 %
TE127M	1.07E-04	0.000 %
I129	4.23E-02	0.000 %

Isotope	Curies	Percentage Of Total
CS134	1.54E+04	2.692 %
CS135	5.66E-01	0.000 %
CS137	1.15E+05	20.105 %
BA137M	1.09E+05	19.056 %
LA138	2.24E-09	0.000 %
CE142	3.67E-05	0.000 %
CE144	7.94E+02	0.139 %
PR144	7.94E+02	0.139 %
PR144M	9.53E+00	0.002 %
ND144	2.18E-09	0.000 %
PM146	1.52E+00	0.000 %
PM147	1.58E+04	2.762 %
PM148M	1.55E-17	0.000 %
SM146	5.16E-07	0.000 %
SM147	5.15E-06	0.000 %
SM148	8.02E-11	0.000 %
SM149	9.82E-13	0.000 %
SM151	4.39E+02	0.077 %
EU150	2.95E-05	0.000 %
EU152	7.51E+00	0.001 %
EU154	8.45E+03	1.477 %
EU155	3.34E+03	0.584 %
GD152	6.06E-13	0.000 %
GD153	1.83E-02	0.000 %
TB160	9.08E-10	0.000 %
HO166M	4.87E-03	0.000 %
TM170	1.29E-08	0.000 %
TM171	8.78E-05	0.000 %
LU176	3.24E-11	0.000 %
LU177	2.83E-09	0.000 %
LU177M	1.23E-08	0.000 %
HF175	2.60E-12	0.000 %
HF181	9.76E-19	0.000 %
HF182	3.74E-07	0.000 %
TA182	1.43E-06	0.000 %
W181	5.24E-08	0.000 %
W185	5.58E-11	0.000 %
W188	3.39E-13	0.000 %
RE187	2.29E-08	0.000 %
RE188	3.42E-13	0.000 %
OS194	1.07E-10	0.000 %
IR192	1.16E-08	0.000 %
IR192M	1.16E-08	0.000 %
IR194	1.07E-10	0.000 %
PT193	1.11E-07	0.000 %
TL206	2.10E-08	0.000 %
TL207	7.99E-06	0.000 %
TL208	1.82E-02	0.000 %
TL209	9.85E-09	0.000 %

Isotope	Curies	Percentage Of Total
PB204	1.72E-16	0.000 %
PB205	2.18E-09	0.000 %
PB209	4.56E-07	0.000 %
PB210	5.05E-08	0.000 %
PB211	8.01E-06	0.000 %
PB212	5.05E-02	0.000 %
PB214	3.75E-07	0.000 %
BI208	3.42E-08	0.000 %
BI210	5.05E-08	0.000 %
BI210M	2.11E-08	0.000 %
BI211	8.01E-06	0.000 %
BI212	5.05E-02	0.000 %
BI213	4.56E-07	0.000 %
BI214	3.75E-07	0.000 %
PO210	5.99E-08	0.000 %
PO211	2.24E-08	0.000 %
PO212	3.24E-02	0.000 %
PO213	4.46E-07	0.000 %
PO214	3.75E-07	0.000 %
PO215	8.01E-06	0.000 %
PO216	5.05E-02	0.000 %
PO218	3.75E-07	0.000 %
AT217	4.56E-07	0.000 %
RN219	8.01E-06	0.000 %
RN220	5.05E-02	0.000 %
RN222	3.75E-07	0.000 %
FR221	4.56E-07	0.000 %
FR223	1.10E-07	0.000 %
RA223	8.01E-06	0.000 %
RA224	5.05E-02	0.000 %
RA225	4.56E-07	0.000 %
RA226	3.75E-07	0.000 %
RA228	7.46E-11	0.000 %
AC225	4.56E-07	0.000 %
AC227	7.99E-06	0.000 %
AC228	7.46E-11	0.000 %
TH227	7.90E-06	0.000 %
TH228	5.04E-02	0.000 %
TH229	4.56E-07	0.000 %
TH230	1.40E-04	0.000 %
TH231	1.85E-02	0.000 %
TH232	1.87E-10	0.000 %
TH234	3.12E-01	0.000 %
PA231	3.27E-05	0.000 %
PA233	4.76E-01	0.000 %
PA234	4.06E-04	0.000 %
PA234M	3.12E-01	0.000 %
U232	5.79E-02	0.000 %
U233	3.68E-05	0.000 %

Isotope	Curies	Percentage Of Total
U234	1.34E+00	0.000 %
U235	1.85E-02	0.000 %
U236	3.52E-01	0.000 %
U237	2.79E+00	0.000 %
U238	3.12E-01	0.000 %
U240	8.44E-07	0.000 %
NP235	6.42E-05	0.000 %
NP236	1.02E-05	0.000 %
NP237	4.76E-01	0.000 %
NP238	7.43E-02	0.000 %
NP239	3.08E+01	0.005 %
NP240M	8.44E-07	0.000 %
PU236	1.60E-01	0.000 %
PU238	4.40E+03	0.769 %
PU239	3.59E+02	0.063 %
PU240	5.80E+02	0.101 %
PU241	1.14E+05	19.930 %
PU242	2.64E+00	0.000 %
PU243	3.51E-07	0.000 %
PU244	8.45E-07	0.000 %
PU246	2.67E-14	0.000 %
AM241	2.01E+03	0.351 %
AM242	1.48E+01	0.003 %
AM242M	1.49E+01	0.003 %
AM243	3.08E+01	0.005 %
AM244	8.46E-11	0.000 %
AM245	1.04E-10	0.000 %
AM246	2.67E-14	0.000 %
CM242	1.26E+01	0.002 %
CM243	3.38E+01	0.006 %
CM244	3.30E+03	0.577 %
CM245	4.23E-01	0.000 %
CM246	1.02E-01	0.000 %
CM247	3.51E-07	0.000 %
CM248	9.75E-07	0.000 %
CM250	1.07E-13	0.000 %
BK249	7.21E-06	0.000 %
BK250	2.84E-11	0.000 %
CF249	1.16E-05	0.000 %
CF250	3.81E-05	0.000 %
CF251	3.50E-07	0.000 %
CF252	8.91E-06	0.000 %
ES254	2.83E-11	0.000 %
Subtotal Curies =	5.73E+05	100.000 %
Total all isotopes =	5.72E+05	

8.2.8.1 Cause of Accident

As described in Section 8.2.8.1 of Reference 8.1, the passive nature of the Calvert Cliffs NUHOMS system and the various design features preclude any credible event that could result in the rupture of all fuel rods concurrent with DSC leakage. However, to demonstrate the safety of the NUHOMS design, this accident assumes that the fuel rods and the DSC pressure boundary are ruptured due to an event of unspecified origin.

8.2.8.2 Accident Analysis

There are no structural or thermal consequences resulting from the DSC leakage accident described above. The radiological consequences of this accident are presented in Section 8.2.8.3.

8.2.8.3 Accident Dose Consequences

Whole body and maximum organ doses are calculated for a hypothetical individual assumed to be present at the nearest controlled area boundary location (with respect to the ISFSI, approximately 3900') for the duration of the event. A meteorological dispersion parameter (X/Q) of 3.0×10^{-4} sec/m^3 was used in calculating the maximum potential doses at the 3900' controlled area boundary. The resulting calculated doses are 0.1 mrem and 17.8 mrem for the maximum off-site total body and skin doses, respectively. These accident doses are well within the 10 CFR 72.106 limit of 5000 mrem.

8.2.9 ACCIDENTAL PRESSURIZATION OF DRY SHIELDED CANISTER

For more information see Reference 8.16.

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

8.2.9.1 Cause of Accident

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

8.2.9.2 Accident Analysis

The maximum DSC accident pressurization was calculated assuming that the fuel rod fission gas release fraction is 30%, and that the fuel rod fill gas pressure is 465 psia. The resulting internal DSC pressure was calculated at the Calvert Cliffs maximum ambient temperature of 103°F. The limiting accident for DSC pressurization is the HSM blocked vent case discussed in Section 8.2.7. Under these conditions, the gas temperatures in the DSC will rise to 548°F producing a DSC internal pressure of 49.9 psig (Reference 8.44). The maximum DSC shell local primary membrane stress intensity due to accident pressurization was calculated using 50 psig, the design basis accident pressure discussed in Reference 8.1, and was determined to be below the allowable stress.

The generic NUHOMS-24P Topical Report Section 10.3.1.1, "Spent Fuel Specifications" does not contain a fuel rod fill gas pressure limit. The assumed maximum initial fill gas pressure in the CCNPP fuel rods is 465 psia. This pressure was used only in calculating 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 end-of-life fill gas pressure in the CCNPP fuel rods is used as an input parameter for determining the maximum fuel clad temperature limit for long term dry storage of the spent fuel. The rod pressure used in this calculation was supplied by the fuel manufacturer. It includes the bounding fill gas pressure as well as bounding parameters for fission gas release. The peak fuel clad temperatures are not very sensitive to the end-of-life pressure in the typical range of interest. They are a strong function of burnup and cooling time. Since the analysis bounds all Calvert Cliffs fuels, it is unnecessary to specify all fuel fabrication parameters in this section.

The assumed maximum initial fill gas pressure in the CCNPP fuel rods is 465 psia. This value was provided by the fuel vendor as part of a transmittal of bounding fuel rod fill and fission gas pressures at various burnups. This pressure was used to calculate the quantity of helium gas available for release from fuel rods to the DSC cavity during accident conditions. The maximum partial pressure of fill gas is only 30% for a NUHOMS-24P DSC (Reference 8.44) of the total gas (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. Any comparison to fill gas pressures used for other purposes is moot.

The analysis of accidental pressurization of the DSC includes the effect of fuel burnup on internal fuel rod pressure by using the volume of fission gas generated in the fuel rod at the maximum burnup of 50 gwd/mthm. All fuel rods from all the spent fuel assemblies in the DSC are assumed to be ruptured and 100% of fuel rod fill gases and 30% of fission gases are assumed to be released to the DSC cavity. The results of the analysis show that the maximum DSC accident pressures are within the design basis limits.

8.2.9.3 Accident Dose Calculations

Since the maximum DSC accident pressure is within the design basis limits, there are no dose consequences.

8.2.10 FOREST FIRE

For more information see Reference 8.18.

This postulated event involves a forest fire occurring in the woods adjacent to the ISFSI.

8.2.10.1 Cause of Accident

A forest fire is postulated to occur due to a number of reasons, most likely natural causes (e.g., lightning) or man-made (accident or arson).

8.2.10.2 Accident Analysis

The Calvert Cliffs ISFSI was evaluated for a postulated forest fire assumed to occur at a distance of 130' from the nearest HSM. The flame front was assumed to be 200' long by 100' in height burning at an effective flame temperature of 1832°F for a period of 1 hour. The flame emissivity was assumed to be 0.9. Based on these parameters and an initial concrete temperature of 135°F, the maximum calculated HSM surface temperature is ~1475°F. THE resulting elevated temperatures at the surface of the HSM walls due to the postulated forest fire may cause cracking or spalling of the walls. The damage to the wall, based on the HSM wall temperature gradient resulting from the fire, will be limited to a thickness of 4.5" into the wall. The remainder of the wall thickness will remain within ACI 349 temperature limits. Fuel cladding temperature limits will be maintained within the fuel cladding short-term temperature limit. The effect of the surface cracking and spalling will be minimal with respect to the load capacity of the HSM walls. Dry shielded canister internal pressure limits (50 psig) will not be exceeded. The increase in HSM surface dose is from 7 mrem/hr to approximately 21 mrem/hr. This increase is not considered a "significant increase in occupational exposure" for the necessary repair activities. Actions to mitigate the fire and repair the HSMs will ensure that offsite dose consequences will be limited and of short duration and will remain within the limits of 10 CFR 72.106.

8.2.10.3 Accident Dose Consequences

There are no accident dose consequences associated with the postulated forest fire accident.

8.2.11 LIQUIFIED NATURAL GAS PLANT OR PIPELINE SPILL OR EXPLOSION

For more information see References 8.16 and 8.20.

This accident involves a possible LNG spill or explosion at the nearby Cove Point LNG terminal or an associated pipeline.

The Cove Point LNG Terminal is located approximately 4 miles south-southeast of the CCNPP site. The Terminal was built in the seventies and operated for 2 years between 1978 and 1980 before it ceased operation for commercial reasons. Columbia LNG applied for restart approval from the Federal Energy Regulatory Commission (Reference 8.22).

In the summer of 1989, upon learning of Columbia LNG's intention to restart the Cove Point Terminal, Baltimore Gas and Electric Company reviewed previous LNG hazards analyses and related commitments to the Nuclear Regulatory Commission (NRC). At that time, Baltimore Gas and Electric Company decided, as a conservative measure, to perform a new LNG hazards analysis to reflect current regulatory requirements using up-to-date statistical information and state-of-the-art analytical models. Baltimore Gas and Electric Company completed and submitted the new analysis (Reference 8.20).

The new analysis identified those hazards that might be present and that might have potential impact on the CCNPP site and then examined the probability of any of those hazards occurring and the consequences that might result. Finally, it determined the risk to the CCNPP site that could result from these scenarios by combining the probability and consequence of each scenario with the likelihood of various meteorological conditions and the likelihood of ignition. The results of the new analysis confirms the conclusions of the previous analyses that the operation of the Cove Point facility will not present any undue hazards to CCNPP or to the ISFSI located on the same site. The NRC concurred with the conclusions of the new analysis (Reference 8.21).

8.2.12 LOAD COMBINATIONS

The load categories associated with normal operating, off-normal, and accident conditions have been described and analyzed in previous chapters. Evaluation of the load combination for the NUHOMS important to safety components is addressed in this section.

The methodology used in combining normal operating, off-normal, and accident loads and their associated overload factors for various NUHOMS components is presented in Section 8.2.10 of Reference 8.1. The Reference 8.1 fatigue analysis envelopes the Calvert Cliffs NUHOMS system. The load combination analysis results showed that the calculated stresses are less than the code allowables for various load combinations shown in Tables 8.2-8, 8.2-9, 8.2-10, 8.2-12, 8.2-14, 8.2-15, and 8.2-16. The analyses demonstrate that the important to safety components of the Calvert Cliffs ISFSI are adequate to withstand all postulated loading combinations.

Horizontal storage module enveloping load combination results were obtained based on a conservative interpretation of the CCNPP calculation. The forces and moments are taken directly from the STRUDL output presented in the calculation, and the maximum values for each component tabulated. The reported accident thermal moments are equal to the maximum values from the load cases adjusted for the cracked section properties, as described in the calculation package.

To demonstrate the HSM design conservatism, the load combinations were derived by adding the absolute maximums for each member from each contributing load case regardless of sign. The dead loads were increased by 5% and the live load included at 100% as the absolute value is used. In all cases the required capacity was less than the calculated capacity for each member.

The front wall calculation presented in the HSM calculation package was prepared to demonstrate that, under the most conservative assumptions, adequate rebar is provided to carry the loads. A more realistic set of assumptions on the load distribution, span of the members and the applied loads results in calculated shears and moments which are about one-half of those reported in the calculation. Therefore, it is concluded that the front wall is not a critical member.

8.2.13 OTHER EVENT CONSIDERATIONS

8.2.13.1 Storage of Flammable Liquid Fuel

This section addresses the following three issues: (1) does the permanent storage arrangement of the liquid fuel represent a hazard; (2) does a fuel tanker truck, either transferring fuel into the storage tanks or in route to the tanks, create a hazard; and (3) does the standard transfer of liquid fuel into a vehicle, via the pumps, create a hazard? Each of these items will be addressed separately below.

Gasoline (4,000 gal), diesel fuel (4,000 gal), and waste oil (550 gal) are stored in underground storage tanks at the Transportation Facility. Underground storage of flammable and combustible liquids is considered the safest form of storage (National Fire Protection Association Fire Protection Handbook, 16th edition, page 11-36). National Fire Protection Association 30-1987, "Flammable and Combustible Liquid Code," does not require specific separation of underground tanks from structures except for the fill and vent connections. This distance is 5'. These underground tanks do not represent a credible hazard to the spent fuel shipments, especially when the distance from the road is considered.

Gasoline and diesel fuel are delivered to the Transportation Facility via tanker trucks. These tankers carry significant quantities of fuel. The shipments of gasoline and diesel fuel are relatively infrequent. The actual off-loading operation of the tankers takes place at the Transportation Facility. A distance of 25' between the off-loading facility and the nearest unrelated structure is required per NFPA 30-1987 (Section 5-4.4.1). The distance from the Transportation Facility to the normal path of travel of the spent fuel is in excess of 25'. Additionally, the slope of the land is such that a fuel spill or pool fire will flow away from the road. Also, there are plans to stop all tanker trucks, including those which are used to fill the fuel oil tanks for the auxiliary boilers and Emergency Diesel Generators, while spent fuel is being transported.

The normal operation of filling the fuel tanks of vehicles represents a less hazardous subset of the off-loading operation described above. The most important aspect is the slope of the land which will result in a spill flowing away from the road. Additionally, the distance provides adequate spatial separation in the event of a fire or explosion.

TABLE 8.2-1
NUHOMS ACCIDENT LOADING IDENTIFICATION

Accident Load Type	Component Load DSC Shell Assembly	DSC Internal Basket	DSC Support Assembly	HSM	Transfer Cask
Loss of HSM Air Outlet Shielding Blocks		(not applicable to Calvert Cliffs)			
Tornado Wind				X	X
Tornado Missiles				X	X
Earthquake	X	X	X	X	X
Flood		(not applicable to Calvert Cliffs)			
Accidental Cask Drop	X	X			X
Loss of Neutron Shield					X
Lightning				X	
Blockage of HSM Air Inlets and Outlets	X	X	X	X	
DSC Leakage		(radiological consequence only)			
DSC Accident Internal Pressure	X				
Forest Fire				X	
LNG Tank & Pipeline Explosion				X	
Load Combinations	X	X	X	X	X

TABLE 8.2-8
NUHOMS-24P DRY SHIELDED CANISTER ENVELOPING LOAD COMBINATION RESULTS FOR NORMAL AND OFF-NORMAL LOADS

(ASME Service Levels A and B)

DSC COMPONENTS	STRESS TYPE	CONTROLLING[a] LOAD COMBINATION	ALLOWABLE[b][c] STRESS (ksi)
DSC Shell	Primary Membrane	B2	18.7
	Membrane + Bending	B2	28.0
	Primary + Secondary	B2	56.1
Bottom Cover Plate	Primary Membrane	A4	18.7
	Membrane + Bending	B2	28.0
	Primary + Secondary	B2	56.1
Top Pressure Plate	Primary Membrane	A3	18.7
	Membrane + Bending	A3	28.0
	Primary + Secondary	A3	56.1
Top Structural Plate	Primary Membrane	A4	18.7
	Membrane + Bending	A4	28.0
	Primary + Secondary	A4	56.1
Spacer Disk	Primary Membrane	A3/A4	18.7
	Membrane + Bending	A3/A4	28.0
	Primary + Secondary	A3/A4	56.1
Support Rod	Primary Membrane	A3/A4	18.7
	Membrane + Bending	A3/A4	28.0
	Primary + Secondary	A3/A4	56.1

[a] See Table 3.2-3 for load combination nomenclature.
[b] See Table 3.2-6 of Reference 8.1 for allowable stress criteria. Material properties were obtained from Table 8.1-2 of Reference 8.1 at a design temperature.
[c] Allowables are for stainless steel material at 400°F. Carbon steel material allowables are higher.

TABLE 8.2-9
NUHOMS-24P DRY SHIELDED CANISTER ENVELOPING LOAD COMBINATION RESULTS FOR ACCIDENT LOADS

(ASME Service Level C)

DSC COMPONENTS	STRESS TYPE	CONTROLLING[a] LOAD COMBINATION	ALLOWABLE[b][c] STRESS (ksi)
DSC Shell	Primary Membrane	C2	21.6
	Membrane + Bending	C1	32.4
Bottom Cover Plate	Primary Membrane	C2	21.6
	Membrane + Bending	C5	32.4
Top Pressure Plate	Primary Membrane	C2	21.6
	Membrane + Bending	C2	32.4
Top Structural Plate	Primary Membrane	C2	21.6
	Membrane + Bending	C1	32.4
Spacer Disk	Primary Membrane	C1	21.6
	Membrane + Bending	C1	32.4
Support Rod	Primary Membrane	C1	21.6
	Membrane + Bending	C1	32.4

(a) See Table 3.2-3 for load combination nomenclature.

(b) See Table 3.2-6 of Reference 8.1 for allowable stress criteria. Material properties were obtained from Table 8.1-2 of Reference 8.1 at a design temperature.

(c) Allowables are for stainless steel material at 460°F. Carbon steel material allowables are higher.

TABLE 8.2-10
NUHOMS-24P DRY SHIELDED CANISTER ENVELOPING LOAD COMBINATION RESULTS FOR ACCIDENT LOADS

(ASME Service Level D)[c]

DSC COMPONENTS	STRESS TYPE	CONTROLLING[a] LOAD COMBINATION	ALLOWABLE[b] STRESS (ksi)
DSC Shell	Primary Membrane Membrane + Bending	D2 D2	43.2 64.0
Bottom Cover Plate	Primary Membrane Membrane + Bending	D2 D2	43.2 64.0
Top Pressure Plate	Primary Membrane Membrane + Bending	D2 D2	43.2 64.0
Top Structural Plate	Primary Membrane Membrane + Bending	D2 D2	43.2 64.0
Spacer Disk	Primary Membrane Membrane + Bending	D2 D2	44.8[e] 57.6[e]
Guide Sleeve	Primary Membrane	--	39.4[d]
Support Rods	Primary Membrane Membrane + Bending	D2 D2	43.2 64.0
Top End Structural Weld	Primary Membrane + Bending	--	21.6
Bottom End Structural Weld	Primary	--	44.9

For more information see Reference 8.16.

(a) See Table 3.2-3 for load combination nomenclature.
(b) See Table 3.2-6 of Reference 8.1 for allowable stress criteria. Material properties were obtained from Table 8.1-2 of Reference 8.1 at a design temperature.
(c) Allowables are for stainless steel material at 460°F. Carbon steel material allowables are higher.
(d) Allowable stress at 600°F.
(e) Allowable stresses based on plastic analysis.

TABLE 8.2-12
NUHOMS-24P DRY SHIELDED CANISTER SUPPORT ASSEMBLY ENVELOPING LOAD COMBINATION RESULTS

Component	Load Combination	AISC Allowable Stress Axial (ksi)	AISC Allowable Stress Bending (ksi)	AISC Allowable Stress Shear (ksi)
W10x68 Cross Beam	Normal Operation $DW_s + DW_c + HL_f$	14.8	17.6	10.6
	Off-Normal Operation $DW_s + HL_j$	14.8	17.6	10.6
	Accident $DW_s + DW_c + DBE$	22.3	26.3	14.9
WT6x115 Support Rail	Normal Operation $DW_s + DW_c + HL_f$	13.8	17.6	10.6
	Off-Normal Operation $DW_s + HL_j$	13.8	17.6	10.6
	Accident $DW_s + DW_c + DBE$	20.7	26.3	14.9

KEY: DW_s = Dead Weight Support Assembly, HL_j = Off-normal Handling Loads-Jammed, DW_c = Dead Weight Canister, HL_f = Normal Loads Friction, DBE = Seismic Loads

NOTES:
Allowable stresses taken at 600°F to conservatively envelope all ambient temperature cases.
Allowables for $DW_s + DW_c + DBE$ increased by 50% for axial and bending, and by 40% for shear.

TABLE 8.2-14
NUHOMS TRANSFER CASK ENVELOPING LOAD COMBINATION RESULTS FOR NORMAL AND OFF-NORMAL LOADS

(ASME Service Levels A and B)

TRANSFER CASK COMPONENT	STRESS TYPE	CONTROLLING[a] LOAD COMBINATION	ALLOWABLE[b] STRESS (ksi)
Structural Shell	Primary Membrane	A5/B2	21.7
	Membrane + Bending	A5/B2	32.6
	Primary + Secondary	A5/B2	65.1
Top Cover Plate	Primary Membrane	A5/B2	18.7
	Membrane + Bending	A5/B2	28.1
	Primary + Secondary	A5/B2	56.1
Inner Bottom 2" Cover Plate	Primary Membrane	A1	18.7
	Membrane + Bending	A1	28.1
	Primary + Secondary	A1	56.1

(a) See Tables 3.2-4 and 12.3-6 for load combination nomenclature.
(b) See Table 3.2-8 of Reference 8.1 for allowable stress criteria. Material properties were obtained from Table 8.1-2 of Reference 8.1 at a design temperature of 400°F.

TABLE 8.2-15
NUHOMS TRANSFER CASK ENVELOPING LOAD COMBINATION RESULTS FOR ACCIDENT LOADS

(ASME Service Level C)

TRANSFER CASK COMPONENT	STRESS TYPE	CONTROLLING[(a)] LOAD COMBINATION	STRESS (ksi) CALCULATED	STRESS (ksi) ALLOWABLE[(b)]
Structural Shell	Primary Membrane	C3	(c)	26.0
	Membrane + Bending	C3	(c)	39.1
Top Cover Plate	Primary Membrane	C2	(c)	22.4
	Membrane + Bending	C2	(c)	33.7
Inner Bottom 2" Cover Plate	Primary Membrane	C2	(c)	22.4
	Membrane + Bending	C2	(c)	33.7

(a) See Tables 3.2-3 and 12.3-6 for load combination nomenclature.
(b) See Table 3.2-8 of Reference 8.1 for allowable stress criteria. Material properties were obtained from Table 8.1-2 of Reference 8.1 at a design temperature of 400°F.
(c) Less than the allowable.

TABLE 8.2-16
NUHOMS TRANSFER CASK ENVELOPING LOAD COMBINATION RESULTS FOR ACCIDENT LOADS

(ASME Service Level D)

TRANSFER CASK COMPONENT	STRESS TYPE	CONTROLLING[a] LOAD COMBINATION	ALLOWABLE[b] STRESS (ksi)
Structural Shell	Primary Membrane	D3	49.0
	Membrane + Bending	D3	70.0
Top Cover Plate	Primary Membrane	D2/D3	44.9
	Membrane + Bending	D3	64.4
Inner Bottom 2" Cover Plate	Primary Membrane	D2/D3	44.9
	Membrane + Bending	D3	64.4

(a) See Tables 3.2-4 and 12.3-6 for load combination nomenclature.

(b) See Table 3.2-8 of Reference 8.1 for allowable stress criteria. Material properties were obtained from Table 8.1-2 of Reference 8.1 at a design temperature of 400°F.

8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

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

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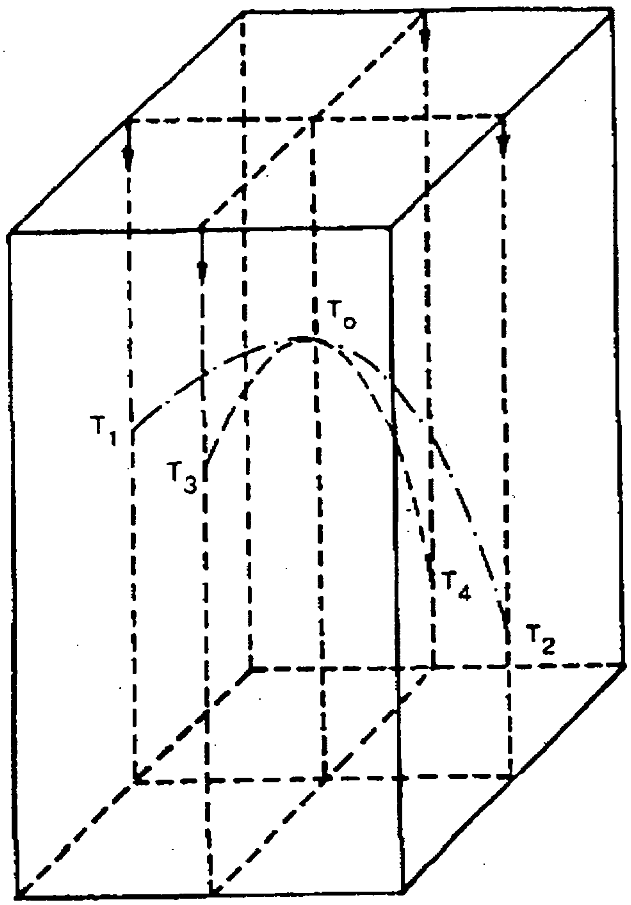
8.40 CCNPP Calculation CA05760, Effective Conductivity of the Reconfigured Fuel Assembly

8.41 CCNPP Calculation CA05924, Calvert Cliffs ISFSI/NUHOMS-24P Radiation Dose Rates for Cask Loading and Transfer

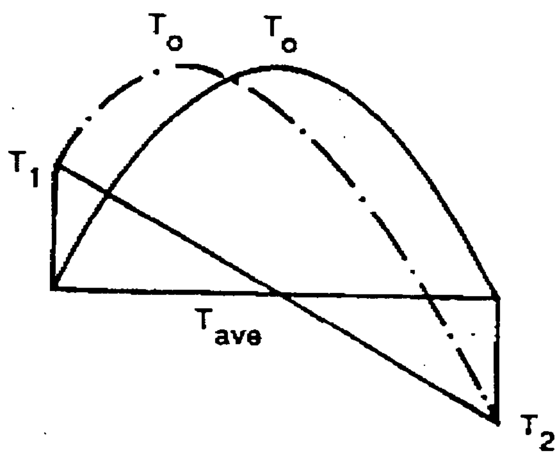
8.42 CCNPP Calculation CA05925, Calvert Cliffs ISFSI/NUHOMS-24P HSM Dose Rates

8.43 CCNPP Calculation CA05926, Calvert Cliffs ISFSI/NUHOMS-24P Occupational Doses

8.44 CCNPP Calculation CA-03947, Dry Storage DSC Internal Pressure



(a) Temperature distribution in a fuel assembly in two perpendicular planes.



(b) Average boundary temperature determination in each plane.

Calvert Cliffs Nuclear Power Plant	TYPICAL FUEL ASSEMBLY TEMPERATURE DISTRIBUTIONS	USAR Figure 8.1-9 Rev. 10

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LIST OF ACRONYMS

BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CFR	Code of Federal Regulations
DSC	Dry Shielded Canister
ERP	Emergency Response Plan
HSM	Horizontal Storage Module
HSM-HB	High Burnup Horizontal Storage Module
IFA	Irradiated Fuel Assembly
ISFSI	Independent Spent Fuel Storage Installation
NFSS	Nuclear Fuel Services Section
NUHOMS	Nutech Horizontal Modular Storage
SFSP	Spent Fuel Storage Project
SNM	Special Nuclear Material
SPMT	Self-Propelled Modular Transporter
UFSAR	Updated Final Safety Analysis Report

9.1.2 OPERATING ORGANIZATION, MANAGEMENT, AND ADMINISTRATIVE CONTROL SYSTEM

9.1.2.1 On-site Organization

The on-site organization of CCNPP is responsible for operation of the ISFSI. The Nuclear Fuels Management Unit of the NFSS maintains primary responsibility for spent fuel storage. The organization for CCNPP is fully described in UFSAR Section 12.1.

9.1.2.2 Personnel Functions, Responsibilities, and Authorities

The functions, responsibilities, and authorities of major personnel positions, including discussions of specific succession of responsibility for overall operation of CCNPP are described in UFSAR Section 12.1. These functions, responsibilities, and authorities extend to the Calvert Cliffs ISFSI.

9.1.3 PERSONNEL QUALIFICATION REQUIREMENTS

The minimum qualification requirements for major operating, technical and maintenance supervisory personnel, as well as the qualifications of persons assigned to managerial and technical positions, are as stated in UFSAR Section 12.1.

9.1.4 LIAISON WITH OTHER ORGANIZATIONS

Arrangements made with outside organizations are as described in Section 9.1.1.3 of this Chapter.

9.2 PREOPERATIONAL TESTING AND OPERATION

Prior to operation of the ISFSI, complete functional tests of the in-plant operations, transfer operations, and horizontal storage module (HSM) loading and retrieval were performed. These tests verified that the storage system components [e.g., dry shielded canister (DSC), transfer cask, transfer trailer, the SPMT, etc.] can be operated safely and effectively (Reference 9.12).

9.2.1 ADMINISTRATIVE PROCEDURES FOR CONDUCTING TEST PROGRAM

Preoperational testing was governed by Calvert Cliffs administrative procedures for conducting testing.

9.2.2 TEST PROGRAM DESCRIPTION

The testing program required use of a DSC, DSC mock-up, transfer cask and associated handling equipment, transfer trailer, the SPMT, and an HSM. The tests simulated, as nearly as possible, the actual operations involved in preparing a DSC for storage and demonstrated that they can be performed safely during actual emplacement of irradiated fuel assemblies (IFAs) in the ISFSI. Shielding verification, which is not completely achievable during dry runs, was accomplished during the initial IFA loadings.

9.2.2.1 Physical Facilities and Operations

9.2.2.1.1 Dry Shielded Canister and Associated Equipment

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

The part-length mock-up was configured exactly as the top end of the DSC with lead shield plug and covers. The mock-up was used for checkout of the automated welding equipment including actual welding of the simulated lead shield plug and top cover plate. Emphasis was placed on acceptability of the weld, as well as compliance with approved as low as reasonably achievable practices.

9.2.2.1.2 Transfer Cask and Handling Equipment

Functional testing was performed with the transfer cask and lifting yoke. These tests demonstrated that the transfer cask can be safely transported from the Auxiliary Building truck bay to the cask washdown pit. From there, it was placed into the spent fuel pool to verify clearances and travel path.

9.2.2.1.3 Off-Normal Testing of the DSC and Transfer Cask

In the unlikely event that a problem arises during actual loading of the IFAs into the DSC, seal welding of the DSC, or during emplacement of a loaded DSC into an HSM, no immediate action would be required since the fuel assemblies would be in a safe condition. The pre-operational testing program confirmed that the IFAs can be safely removed from the DSC by demonstrating that the DSC lids can be removed.

9.2.2.1.4 Transfer Trailer, SPMT, and HSMs

The transfer cask was placed on the transfer trailer, and then transported to the ISFSI and aligned with an HSM. Compatibility of the transfer trailer with the transfer cask, negotiation of the travel path to the ISFSI, and maneuverability within the confines of the ISFSI were verified.

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

The self-propelled horizontal cask transporter is used to move DSCs at various Constellation Energy Nuclear Group facilities. Therefore, the use of the self-propelled horizontal cask transporter for loading HSMs has been demonstrated at other Constellation Energy Nuclear Group ISFSIs.

9.2.2.1.5 Off-Normal Testing of the Transfer Trailer, SPMT and HSMs

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

The most severe condition would occur if a failure of the hydraulic ram, after partial insertion of a DSC into an HSM, were to prevent complete emplacement of the DSC. (Radiological shielding and decay heat removal are not compromised by this condition, but the transfer trailer may not be moved away until the DSC is completely within the confines of either the transfer cask or the HSM.) Pre-operational testing verified that reversal of DSC movement can be completed by the operator of the hydraulic ram.

9.2.3 TEST DISCUSSION

The purpose of the preoperational tests was to ensure that a DSC can be properly and safely placed in the spent fuel pool, loaded with IFAs, transported to the ISFSI, emplaced in the HSM, and removed from the HSM. Proper operation of the DSC, transfer cask, and transfer trailer, and the SPMT as well as the associated handling equipment (e.g., lifting yoke, welding equipment, vacuum drying equipment), provided such assurance.

Preoperational test requirements were specific. Detailed procedures were developed and implemented by Calvert Cliffs personnel who were responsible for ensuring that the test requirements were satisfied.

The expected results of the preoperational tests were the successful completion of the following: loading of a DSC into the transfer cask, seal welding of the mock-up DSC, placement of a DSC into the transfer cask into and out of the spent fuel pool, transporting the transfer cask loaded with a DSC and test weights to the ISFSI, and emplacement in an HSM and removal from an HSM. The tests were deemed successful since the expected results were achieved safely and without damage to any of the components or associated equipment.

Any equipment or components which required modification in order to achieve the expected results were retested to affirm that the modification was sufficient. If any pre-operational procedures were changed in order to achieve the expected results, the changes were incorporated into the appropriate operating procedures.

Power operation of CCNPP was not affected by testing of the storage system, and in-plant testing was conducted concurrently with plant operation. In-plant testing was conducted entirely within the Auxiliary Building, and was scheduled so that there was no conflict with refueling. All normal prerequisites for safe handling of components in, or near, the spent fuel pool were satisfied, and normal safety and radiological practices were employed.

9.3 TRAINING PROGRAM

All personnel working at the Calvert Cliffs ISFSI receive training and indoctrination geared toward providing and maintaining a well-qualified work force for safe and efficient operation of the ISFSI. The existing Calvert Cliffs training program, as described in CCNPP UFSAR Section 12.2, is used to provide this training and indoctrination. Additional sections have been added to this program to include information specific to the ISFSI.

9.3.1 PROGRAM DESCRIPTION

9.3.1.1 Training for ISFSI Operations Personnel

Generalized training is provided to operations personnel in the applicable regulations and standards and in the nuclear engineering principles of cooling, radiological shielding, and structural characteristics of the DSC/HSM.

Detailed operator training is provided for DSC preparation and handling, fuel loading, transfer cask preparation and handling, transfer trailer, and SPMT loading.

9.3.1.2 Training for Maintenance Personnel

Generalized training is provided to maintenance personnel on the applicable regulations and standards and on the nuclear engineering principles of cooling, radiological shielding, and structural characteristics of the DSC/HSM/HSM-HB (high burnup horizontal storage module).

Specific training is provided for use of the automated seal welding equipment for the top end shield plug and top cover plate, operation of the transfer trailer, the SPMT, alignment of the cask skid with the HSM/HSM-HB, alignment of the hydraulic ram assembly, and normal and off-normal operation of the hydraulic ram. Specific training is also provided for cleaning of the HSM/HSM-HB air inlets and outlets.

9.3.1.3 Training for Health Physics Personnel

Generalized training is provided to Health Physics personnel on the applicable regulations and standards and on the nuclear engineering principles of cooling, radiological shielding, and structural characteristics of the DSC/HSM/HSM-HB.

Specific training is provided in radiological shielding design of the system, particularly the top end shield plug, DSC/transfer cask and the DSC/HSM/HSM-HB.

9.3.1.4 Training for Security Personnel

Details of the training program for security personnel are provided in the Security Plan which is withheld from public disclosure in accordance with Title 10, Code of Federal Regulations (CFR) 2.790(d) and 10 CFR 73.21.

9.3.2 RETRAINING PROGRAM

Retraining is consistent with retraining requirements in effect at CCNPP for personnel involved in fuel handling operations.

9.3.3 ADMINISTRATION AND RECORDS

The organization responsible for training programs and for maintaining up-to-date records on the status of personnel training is the existing Nuclear Training Section at Calvert Cliffs.

9.4 NORMAL OPERATIONS

The Calvert Cliffs ISFSI utilizes the Nutech Horizontal Modular Storage® (NUHOMS) system, which is completely passive during storage. Therefore, no monitoring instruments or limiting control settings are utilized at the ISFSI. Other limits and controls that are applied to the system during fuel loading and DSC transfer to the ISFSI are the fuel selection criteria, DSC surface contamination limits and DSC vacuum and helium backfill pressures, DSC closure weld examination requirements, and cask height restrictions during transport.

The components of storage, the DSC, the HSM, and (during transfer) the transfer cask, have been analyzed for all credible equipment failure modes and extreme environmental conditions. No postulated event results in damage to fuel, release of radioactivity, or danger to the public health and safety. All operational equipment will be maintained, tested, and operated according to the implementing procedures developed for the ISFSI. The failure or unavailability of any operational component can result in delay in transfer of the DSC to the HSM, but will not result in an unsafe condition.

Under normal operations, the ISFSI provides for independent storage of spent fuel away from the CCNPP facilities. With the exception of some limited physical and continuous electronic security surveillance, the facility functions as a passive system once fuel has been loaded and stored at the ISFSI. Loading of fuel assemblies into the facility, which occurs periodically, requires specific procedures that are separate from those of normal plant operations.

9.4.1 ADMINISTRATIVE CONTROLS

Existing and proposed CCNPP organizational and administrative systems and procedures, record keeping, review, audit, and reporting requirements are used to ensure that the operations involved in the storage of spent fuel at the Calvert Cliffs ISFSI are performed in a safe manner. This includes both the selection of assemblies qualified for ISFSI storage and the verification of assembly identification numbers prior to and after placement into individual storage canisters.

9.4.1.1 Qualification of Spent Fuel

Fuel assembly qualification is based on the requirements for criticality control, decay heat removal, radiological protection, and structural integrity. Fuel assembly reactivity, radiological source strength, and decay heat removal capabilities are defined by three variables: (1) the initial enrichment of the unirradiated fuel assembly, (2) the final assembly burnup at discharge, and (3) the out-of-reactor cooling time. Table 9.4-1 (Reference 9.15) presents the minimum cooling time for each fuel batch to achieve the 0.66 kW decay heat limit. Tables 9.4-4 and 9.4-5 present the minimum cooling time for each specified fuel batch to achieve the 0.9 kW and 1.0 kW decay heat limits. Table 9.4-2 represents the acceptance criteria for minimum spent fuel burnup as a function of initial enrichment for the NUHOMS-24P DSC. The NUHOMS-32P DSC and NUHOMS-32PHB DSCs have no limit on minimum spent fuel burnup, but assemblies containing irradiated stainless steel replacement pins have additional acceptance criteria presented in Table 9.4-3. The administrative procedures controlling these variables are as described below.

Procedures currently in place for special nuclear materials (SNMs) accountability and record keeping are used to verify initial fuel assembly enrichment and burnup levels at discharge. New fuel enrichments and initial uranium isotopics are recorded from the Department of Energy/Nuclear Regulatory Commission Form 741s and stored in both a database file and on duplicate paper copies of the Form 741s. Individual fuel assembly burnups are also stored in the SNMs database. These values are generated by utilizing thermal energy production data determined by in-core flux mapping. Burnup and initial enrichment values from the SNM accountability records of each IFA are compared to the applicable limits to verify that the reactivity level is acceptable for DSC loading and storage. The enrichment vs. burnup method for reactivity verification will routinely be used, and required by procedures, for the NUHOMS-24P DSC. Calvert Cliffs Nuclear Power Plant reserves the right to rely on other Nuclear Regulatory Commission accepted analytical methods to qualify fuel assemblies in special cases.

Subcriticality in the NUHOMS-32P and NUHOMS 32PHB DSCs is assured by limiting the initial enrichment of unirradiated fuel assemblies to ≤ 4.5 wt% and 5.0 wt% U-235 respectively, by the presence of fixed neutron absorbing plates in the basket assembly, and by the presence of soluble boron in the spent fuel pool water.

For decay heat control, only those irradiated assemblies which do not exceed a decay heat level of 0.66 kW qualify for loading into the NUHOMS-24P and NUHOMS-32P DSCs. Irradiated assemblies which do not exceed sa decay heat level of 1.0 kW qualify for loading into the NUHOMS-32PHB DSC. Due to Co-60 production, assemblies with stainless steel replacement rods require additional cooling time beyond the time at which they reach 0.66 kW as shown in Table 9.4-3. Decay heat loadings at or below this level ensure that peak fuel rod cladding temperatures are maintained within acceptable levels. Since individual fuel assembly decay heat levels are a function of both the discharge burnup and the cooling time, procedural controls are used to verify these parameters prior to fuel assembly loading. For the Calvert Cliffs NUHOMS-24P and NUHOMS-32P DSC fuel design and operating histories, the cooling time necessary to achieve a 0.66 kW decay heat level is between 4 and 17 years. The cooling times of discharged fuel for NUHOMS-24P and NUHOMS-32P DSCs is presented in Table 9.4-1. For the Calvert Cliffs NUHOMS-32PHB DSC fuel designs and operating histories, the cooling time necessary to achieve 0.8 kW is between TBD and TBD years and the cooling time necessary to achieve 1.0 kW is between TBD and TBD years. The cooling times of discharged fuel for NUHOMS-32PHB DSC is presented in Tables 9.4-4 and 9.4-5. The variation in required cooling time is a very strong function of discharge burnup and a very weak function of initial enrichment.

Specific qualification of the fuel assembly radiological source term is not necessary prior to fuel loading. Analysis shows that the reference source term used to generate the surface dose rate values found in Chapters 7 and 12 is not exceeded by any fuel assembly meeting the limiting conditions for cooling times specified in Tables 9.4-1, 9.4-3, 9.4-4, and 9.4-5.

Assemblies that fall into the acceptance region of Table 9.4-2 qualify as candidates for NUHOMS-24P DSC ISFSI storage with the appropriate minimum cooling. Additional calculations relating reactivity (i.e., initial enrichment and discharge burnup) with decay heat and the required cooling time may be performed, as needed, to qualify future assemblies.

To ensure the structural integrity of the spent fuel to be loaded into the DSC, plant records of all known damaged assemblies are reviewed. A fuel assembly and component database has been compiled which incorporates previous sipping, ultrasonic testing, eddy current, and visual observation. This database is examined as a part of the dry storage qualification process to verify that assemblies with known cladding breaches are not included.

Fuel assemblies are also screened to ensure they conform to the requirements of Reference 9.14 to ensure the fuel assemblies with burnup < 47,000 MWD/MTU contain no more than two vacancies in any location within a column or row. This allows 28 vacancies per assembly with burnup < 47,000 MWD/MTU. The vacancies do not need to be adjacent to one another. The analysis finds fuel assemblies that conform to this configuration to be structurally sound under all anticipated conditions. Vacancies are restricted for fuel with burnup between 47,000 MWD/MTU and 52,000 MWD/MTU pending further structural analysis.

If the reactivity, decay heat, and structural integrity criteria are all met, then approval for dry storage for a given assembly is granted. This qualification is documented and subsequently referenced through ISFSI operating procedures prior to loading fuel into the DSC.

9.4.1.2 Spent Fuel Identification

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

9.4.2 RECORDS

The ISFSI records are maintained in accordance with the requirements of 10 CFR Part 72. Procedures have been developed for use by the Spent Fuel Storage Project (SFSP) which meet the requirements of 10 CFR Part 72 for records retention during the construction phase of the project (Reference 9.5). Additional procedures have been developed to encompass the fuel loading and storage phases of the project.

For SNM accountability, the management system in place for Calvert Cliffs Units 1 and 2 has been expanded to allow record-keeping relative to storage of fuel at the ISFSI. The requirements of 10 CFR 72.72, 10 CFR 72.74, 10 CFR 72.76, and 10 CFR 72.78

have been met by adding the ISFSI to our current system, which meets the equivalent requirements of 10 CFR 70.51, 10 CFR 70.52, 10 CFR 70.53, and 10 CFR 70.54, respectively. Horizontal storage module and DSC identification numbers, along with individual assembly locations within a DSC, are maintained in our SNM database consisting of core locations, spent fuel pool rack locations, etc. In this way, ISFSI SNM accountability requirements are met. Periodic physical inventory requirements are met by verifying that HSMs have not been tampered with since the previous inventory (References 9.9 and 9.11).

While 10 CFR 70.51 imposes a three year duration of records storage, by maintaining the ISFSI records for ISFSI lifetime plus five years the duration requirement of 10 CFR 72.72 is met. It is the intention of CCNPP to use the existing system for maintaining records, ensuring that the stricter of the requirements of the various Parts of 10 CFR are met.

TABLE 9.4-1
POST-DISCHARGE COOLING TIME *

NUHOMS-24P COOLING TIME TO MEET 0.66 kW DECAY HEAT LIMIT

INITIAL ENRICHMENT (W/O U^{235})	BURNUP (MWD/MTU)	COOLING TIME** (Years)
$4.00 < E \leq 4.50$	B.U. ≤ 47,000	10
$3.50 < E \leq 4.00$	45,000 < B.U. ≤ 47,000	11
	42,000 < B.U. ≤ 45,000	10
	B.U. ≤ 42,000	8
$3.00 < E \leq 3.50$	45,000 < B.U. ≤ 47,000	12
	42,000 < B.U. ≤ 45,000	11
	B.U. ≤ 42,000	9
$2.50 < E \leq 3.00$	45,000 < B.U. ≤ 47,000	13
	42,000 < B.U. ≤ 45,000	12
	39,000 < B.U. ≤ 42,000	10
	B.U. ≤ 39,000	8
$2.00 < E \leq 2.50$	45,000 < B.U. ≤ 47,000	15
	42,000 < B.U. ≤ 45,000	13
	39,000 < B.U. ≤ 42,000	11
	B.U. ≤ 39,000	9

* All assemblies loaded into DSC must meet the source spectra requirements of Technical Specification 2.1.

** These bounding cooling times may be superseded with bundle specific cooling times via explicit bundle specific decay heat calculations.

TABLE 9.4-1
POST-DISCHARGE COOLING TIME *

NUHOMS-32P COOLING TIMES** (Years)

Burnup (GWd/ MTU) / Enrichment	B ≤ 38	38 < B ≤ 39	39 < B ≤ 40	40 < B ≤ 41	41 < B ≤ 42	42 < B ≤ 43	43 < B ≤ 44	44 < B ≤ 45	45 < B ≤ 46	46 < B ≤ 47	47 < B ≤ 48	48 < B ≤ 49	49 < B ≤ 50	50 < B ≤ 51	51 < B ≤ 52
2.00≤E<2.10	8.4	8.9	9.4+	9.9+	10.5+	11.2+	11.9+	12.6+	13.4+	14.3+	15.2+	16.2+	17.3+	18.5+	19.7+
2.10≤E<2.20	8.3	8.7	9.2	9.8+	10.4+	11.0+	11.7+	12.5+	13.3+	14.1+	15.1+	16.0+	17.1+	18.2+	19.4+
2.20≤E<2.30	8.1	8.6	9.1	9.6	10.2+	10.8+	11.5+	12.3+	13.1+	13.9+	14.9+	15.9+	16.9+	18.0+	19.2+
2.30≤E<2.40	8.0	8.5	9.0	9.5	10.1+	10.7+	11.4+	12.1+	12.9+	13.8+	14.7+	15.7+	16.7+	17.8+	19.0+
2.40≤E<2.50	7.9	8.4	8.8	9.4	9.9	10.6+	11.2+	12.0+	12.8+	13.6+	14.5+	15.5+	16.5+	17.7+	18.8+
2.50≤E<2.60	7.8	8.2	8.7	9.3	9.8	10.4	11.1+	11.8+	12.6+	13.5+	14.4+	15.3+	16.4+	17.5+	18.7+
2.60≤E<2.70	7.7	8.2	8.6	9.1	9.7	10.3	11.0	11.7+	12.5+	13.3+	14.2+	15.2+	16.2+	17.3+	18.5+
2.70≤E<2.80	7.6	8.1	8.5	9.0	9.6	10.2	10.9	11.6	12.4+	13.2+	14.1+	15.0+	16.1+	17.2+	18.3+
2.80≤E<2.90	7.6	8.0	8.4	9.0	9.5	10.1	10.8	11.5	12.2	13.1+	13.9+	14.9+	15.9+	17.0+	18.2+
2.90≤E<3.00	7.5	7.9	8.4	8.9	9.4	10.0	10.7	11.4	12.1	12.9	13.8+	14.8+	15.8+	16.9+	18.0+
3.00≤E<3.10	7.4	7.9	8.3	8.8	9.3	9.9	10.6	11.3	12.0	12.8	13.7	14.6+	15.7+	16.7+	17.9+
3.10≤E<3.20	7.4	7.8	8.2	8.7	9.3	9.8	10.5	11.2	11.9	12.7	13.6	14.5	15.5+	16.6+	17.7+
3.20≤E<3.30	7.3	7.7	8.2	8.7	9.2	9.8	10.4	11.1	11.8	12.6	13.5	14.4	15.4	16.5+	17.6+
3.30≤E<3.40	7.3	7.7	8.1	8.6	9.1	9.7	10.3	11.0	11.7	12.5	13.4	14.3	15.3	16.4	17.5
3.40≤E<3.50	7.3	7.6	8.1	8.5	9.1	9.6	10.2	10.9	11.6	12.4	13.3	14.2	15.2	16.2	17.4
3.50≤E<3.60	7.2	7.6	8.0	8.5	9.0	9.6	10.2	10.8	11.6	12.3	13.2	14.1	15.1	16.1	17.2
3.60≤E<3.70	7.2	7.6	8.0	8.4	8.9	9.5	10.1	10.8	11.5	12.3	13.1	14.0	15.0	16.0	17.1
3.70≤E<3.80	7.2	7.5	7.9	8.4	8.9	9.4	10.0	10.7	11.4	12.2	13.0	13.9	14.9	15.9	17.0
3.80≤E<3.90	7.1	7.5	7.9	8.3	8.8	9.4	10.0	10.6	11.3	12.1	12.9	13.8	14.8	15.8	16.9
3.90≤E<4.00	7.1	7.5	7.9	8.3	8.8	9.3	9.9	10.5	11.2	12.0	12.8	13.7	14.7	15.7	16.8
4.00≤E<4.10	7.1	7.4	7.8	8.2	8.7	9.3	9.8	10.5	11.2	11.9	12.7	13.6	14.6	15.6	16.7
4.10≤E<4.20	7.0	7.4	7.8	8.2	8.7	9.2	9.8	10.4	11.1	11.8	12.7	13.5	14.5	15.5	16.6
4.20≤E<4.30	7.0	7.4	7.7	8.2	8.6	9.1	9.7	10.3	11.0	11.8	12.6	13.4	14.4	15.4	16.5
4.30≤E<4.40	7.0	7.3	7.7	8.1	8.6	9.1	9.7	10.3	10.9	11.7	12.5	13.3	14.3	15.3	16.4
4.40≤E<4.50	7.0	7.3	7.6	8.1	8.5	9.0	9.6	10.2	10.9	11.6	12.4	13.2	14.2	15.2	16.2

\+ indicates that additional cooling time beyond that shown must be determined through an assembly specific source term calculation to ensure compliance with Technical Specification 2.1.

* All assemblies loaded into DSC must meet the source spectra requirements of Technical Specification 2.1.

** These bounding cooling times may be superceded with bundle specific cooling times via explicit bundle specific decay heat calculations.

TABLE 9.4-1
POST-SHUTDOWN COOLING TIME TO MEET 800 WATTS DECAY HEAT LIMIT

Bu(GWd/MTU) / Enrichment	B <= 38	38 < B <= 39	39 < B <= 40	40 < B <= 41	41 < B <= 42	42 < B <= 43	43 < B <= 44	44 < B <= 45	45 < B <= 46	46 < B <= 47	47 < B <= 48	48 < B <= 49	49 < B <= 50	50 < B <= 51	51 < B <= 52	52 < B <= 53	53 < B <= 54	54 < B <= 55	55 < B <= 56	56 < B <= 57	57 < B <= 58	58 < B <= 59	59 < B <= 60	60 < B <= 61	61 < B <= 62
2.00 <= E < 2.10	6.9	7.1	7.4+	7.8+	8.1+	8.5+	8.9+	9.3+	9.9+	10.5+	11.2+	11.9+	12.6+	13.3+	14.1+	15.0+	15.9+	16.8+	17.8+	18.8+	19.9+	21.0+	22.2+	23.5+	24.8+
2.10 <= E < 2.20	6.7	7.0	7.3	7.6+	7.9+	8.3+	8.7+	9.2+	9.8+	10.4+	11.1+	11.7+	12.5+	13.2+	14.0+	14.8+	15.7+	16.7+	17.6+	18.7+	19.7+	20.9+	22.0+	23.3+	24.6+
2.20 <= E < 2.30	6.6	6.9	7.2	7.5	7.8+	8.2+	8.6+	9.1+	9.7+	10.3+	10.9+	11.6+	12.3+	13.1+	13.9+	14.7+	15.6+	16.5+	17.5+	18.5+	19.6+	20.7+	21.9+	23.1+	24.4+
2.30 <= E < 2.40	6.5	6.8	7.0	7.3	7.7	8.0+	8.5+	9.1+	9.6+	10.2+	10.8+	11.5+	12.2+	12.9+	13.7+	14.6+	15.4+	16.4+	17.3+	18.3+	19.4+	20.5+	21.7+	22.9+	24.2+
2.40 <= E < 2.50	6.4	6.7	6.9	7.2	7.5	8.0	8.4+	9.0+	9.5+	10.1+	10.7+	11.4+	12.1+	12.8+	13.6+	14.4+	15.3+	16.2+	17.2+	18.2+	19.2+	20.3+	21.5+	22.7+	24.0+
2.50 <= E < 2.60	6.3	6.6	6.8	7.1	7.4	7.9	8.3	8.9+	9.4+	10.0+	10.6+	11.3+	11.9+	12.7+	13.5+	14.3+	15.1+	16.0+	17.0+	18.0+	19.1+	20.2+	21.3+	22.5+	23.8+
2.60 <= E < 2.70	6.2	6.5	6.7	7.0	7.3	7.8	8.3	8.8	9.3+	9.9+	10.5+	11.1+	11.8+	12.5+	13.3+	14.1+	15.0+	15.9+	16.8+	17.8+	18.9+	20.0+	21.1+	22.3+	23.6+
2.70 <= E < 2.80	6.1	6.4	6.7	6.9	7.3	7.7	8.2	8.7	9.2+	9.8+	10.4+	11.0+	11.7+	12.4+	13.2+	14.0+	14.8+	15.7+	16.7+	17.7+	18.7+	19.8+	20.9+	22.1+	23.4+
2.80 <= E < 2.90	6.1	6.3	6.6	6.9	7.2	7.6	8.1	8.6	9.1	9.7+	10.3+	10.9+	11.6+	12.3+	13.0+	13.8+	14.7+	15.6+	16.5+	17.5+	18.5+	19.6+	20.8+	21.9+	23.2+
2.90 <= E < 3.00	6.0	6.3	6.5	6.8	7.1	7.5	8.0	8.5	9.0	9.6	10.1+	10.8+	11.4+	12.2+	12.9+	13.7+	14.5+	15.4+	16.4+	17.3+	18.4+	19.4+	20.6+	21.8+	23.0+
3.00 <= E < 3.10	6.0	6.2	6.5	6.7	7.0	7.5	7.9	8.4	8.9	9.4	10.0	10.7+	11.3+	12.0+	12.8+	13.6+	14.4+	15.3+	16.2+	17.2+	18.2+	19.3+	20.4+	21.6+	22.8+
3.10 <= E < 3.20	5.9	6.1	6.4	6.7	7.0	7.4	7.8	8.3	8.8	9.3	9.9	10.5	11.2+	11.9+	12.6+	13.4+	14.3+	15.1+	16.0+	17.0+	18.0+	19.1+	20.2+	21.4+	22.6+
3.20 <= E < 3.30	5.9	6.1	6.3	6.6	6.9	7.3	7.7	8.2	8.7	9.2	9.8	10.4	11.1+	11.8+	12.5+	13.3+	14.1+	15.0+	15.9+	16.9+	17.9+	18.9+	20.0+	21.2+	22.4+
3.30 <= E < 3.40	5.8	6.1	6.3	6.6	6.8	7.2	7.6	8.1	8.6	9.1	9.7	10.3	11.0	11.7+	12.4+	13.2+	14.0+	14.8+	15.7+	16.7+	17.7+	18.8+	19.9+	21.0+	22.2+
3.40 <= E < 3.50	5.8	6.0	6.2	6.5	6.8	7.1	7.6	8.0	8.5	9.0	9.6	10.2	10.8	11.5	12.3+	13.0+	13.8+	14.7+	15.6+	16.5+	17.5+	18.6+	19.7+	20.8+	22.0+
3.50 <= E < 3.60	5.8	6.0	6.2	6.5	6.7	7.1	7.5	7.9	8.4	8.9	9.5	10.1	10.7	11.4	12.1	12.9+	13.7+	14.5+	15.4+	16.4+	17.4+	18.4+	19.5+	20.7+	21.9+
3.60 <= E < 3.70	5.7	5.9	6.2	6.4	6.7	7.0	7.4	7.9	8.3	8.9	9.4	10.0	10.6	11.3	12.0	12.8	13.6+	14.4+	15.3+	16.2+	17.2+	18.3+	19.3+	20.5+	21.7+
3.70 <= E < 3.80	5.7	5.9	6.1	6.4	6.6	6.9	7.3	7.8	8.3	8.8	9.3	9.9	10.5	11.2	11.9	12.6	13.4	14.3+	15.1+	16.1+	17.1+	18.1+	19.2+	20.3+	21.5+
3.80 <= E < 3.90	5.6	5.8	6.1	6.3	6.6	6.9	7.3	7.7	8.2	8.7	9.2	9.8	10.4	11.1	11.8	12.5	13.3	14.1	15.0+	15.9+	16.9+	17.9+	19.0+	20.1+	21.3+
3.90 <= E < 4.00	5.6	5.8	6.0	6.3	6.5	6.8	7.2	7.6	8.1	8.6	9.1	9.7	10.3	11.0	11.6	12.4	13.2	14.0	14.9	15.8+	16.8+	17.8+	18.8+	20.0+	21.1+
4.00 <= E < 4.10	5.6	5.8	6.0	6.2	6.5	6.8	7.1	7.6	8.0	8.5	9.0	9.6	10.2	10.8	11.5	12.3	13.0	13.9	14.7	15.6	16.6	17.6+	18.7+	19.8+	21.0+
4.10 <= E < 4.20	5.5	5.7	5.9	6.2	6.4	6.7	7.1	7.5	7.9	8.4	8.9	9.5	10.1	10.7	11.4	12.1	12.9	13.7	14.6	15.5	16.5	17.5	18.5+	19.6+	20.8+
4.20 <= E < 4.30	5.5	5.7	5.9	6.1	6.3	6.6	7.0	7.4	7.9	8.3	8.9	9.4	10.0	10.6	11.3	12.0	12.8	13.6	14.5	15.4	16.3	17.3	18.4	19.5	20.6
4.30 <= E < 4.40	5.4	5.6	5.8	6.0	6.3	6.6	6.9	7.4	7.8	8.3	8.8	9.3	9.9	10.5	11.2	11.9	12.7	13.5	14.3	15.2	16.2	17.2	18.2	19.3	20.5
4.40 <= E < 4.50	5.4	5.5	5.7	6.0	6.2	6.5	6.9	7.3	7.7	8.2	8.7	9.2	9.8	10.4	11.1	11.8	12.6	13.4	14.2	15.1	16.0	17.0	18.1	19.2	20.3
4.50 <= E < 4.60	5.4	5.5	5.7	5.9	6.2	6.5	6.8	7.2	7.7	8.1	8.6	9.2	9.7	10.4	11.0	11.7	12.5	13.3	14.1	15.0	15.9	16.9	17.9	19.0	20.2
4.60 <= E < 4.70	5.4	5.5	5.7	5.8	6.1	6.4	6.8	7.2	7.6	8.1	8.6	9.1	9.7	10.3	10.9	11.6	12.4	13.1	14.0	14.9	15.8	16.8	17.8	18.9	20.0
4.70 <= E < 4.80	5.4	5.5	5.6	5.8	6.1	6.4	6.8	7.1	7.6	8.0	8.5	9.0	9.6	10.2	10.8	11.5	12.3	13.0	13.9	14.7	15.7	16.6	17.7	18.7	19.9
4.80 <= E < 4.90	5.4	5.5	5.6	5.8	6.1	6.4	6.7	7.1	7.5	8.0	8.4	9.0	9.5	10.1	10.8	11.4	12.2	12.9	13.8	14.6	15.5	16.5	17.5	18.6	19.7
4.90 <= E < 5.00	5.4	5.5	5.6	5.8	6.0	6.4	6.7	7.1	7.5	7.9	8.4	8.9	9.4	10.0	10.7	11.4	12.1	12.8	13.7	14.5	15.4	16.4	17.4	18.5	19.6

Note that + indicates that the neutron source term limits could not be met at the cooling time post-shutdown and that an assembly specific source term calculation is required to determine the cooling time for the assembly.

TABLE 9.4-1
POST-SHUTDOWN COOLING TIME TO MEET 1000 WATTS DECAY HEAT LIMIT

Bu(GWd/MTU) / Enrichment	B <= 38	38 < B <= 39	39 < B <= 40	40 < B <= 41	41 < B <= 42	42 < B <= 43	43 < B <= 44	44 < B <= 45	45 < B <= 46	46 < B <= 47	47 < B <= 48	48 < B <= 49	49 < B <= 50	50 < B <= 51	51 < B <= 52	52 < B <= 53	53 < B <= 54	54 < B <= 55	55 < B <= 56	56 < B <= 57	57 < B <= 58	58 < B <= 59	59 < B <= 60	60 < B <= 61	61 < B <= 62
2.00 <= E < 2.10	5.3	5.5	5.6	5.8	6.0+	6.2+	6.4+	6.6+	6.9+	7.2+	7.5+	7.9+	8.2+	8.6+	9.0+	9.5+	9.9+	10.4+	10.9+	11.5+	12.0+	12.6+	13.2+	13.9+	14.6+
2.10 <= E < 2.20	5.2	5.4	5.5	5.7	5.9	6.1+	6.3+	6.5+	6.8+	7.1+	7.4+	7.8+	8.2+	8.5+	9.0+	9.4+	9.8+	10.3+	10.8+	11.4+	11.9+	12.5+	13.1+	13.8+	14.4+
2.20 <= E < 2.30	5.2	5.3	5.5	5.6	5.8	6.0	6.2+	6.4+	6.7+	7.0+	7.3+	7.7+	8.1+	8.4+	8.9+	9.3+	9.7+	10.2+	10.7+	11.2+	11.8+	12.4+	13.0+	13.6+	14.3+
2.30 <= E < 2.40	5.1	5.2	5.4	5.6	5.7	5.9	6.1	6.4	6.6+	6.9+	7.3+	7.6+	8.0+	8.4+	8.8+	9.2+	9.6+	10.1+	10.6+	11.1+	11.7+	12.3+	12.9+	13.5+	14.2+
2.40 <= E < 2.50	5.0	5.2	5.3	5.5	5.7	5.9	6.1	6.3	6.6+	6.9+	7.2+	7.5+	7.9+	8.3+	8.7+	9.1+	9.5+	10.0+	10.5+	11.0+	11.6+	12.1+	12.7+	13.4+	14.0+
2.50 <= E < 2.60	5.0	5.1	5.3	5.4	5.6	5.8	6.0	6.2	6.5	6.8+	7.1+	7.4+	7.8+	8.2+	8.6+	9.0+	9.4+	9.9+	10.4+	10.9+	11.4+	12.0+	12.6+	13.2+	13.9+
2.60 <= E < 2.70	4.9	5.1	5.2	5.4	5.6	5.7	5.9	6.2	6.4	6.7	7.0+	7.4+	7.7+	8.1+	8.5+	8.9+	9.3+	9.8+	10.3+	10.8+	11.3+	11.9+	12.5+	13.1+	13.8+
2.70 <= E < 2.80	4.9	5.0	5.2	5.3	5.5	5.7	5.9	6.1	6.4	6.7	7.0	7.3+	7.6+	8.0+	8.4+	8.8+	9.2+	9.7+	10.2+	10.7+	11.2+	11.8+	12.4+	13.0+	13.6+
2.80 <= E < 2.90	4.8	5.0	5.1	5.3	5.5	5.6	5.8	6.0	6.3	6.6	6.9	7.2+	7.6+	7.9+	8.3+	8.7+	9.1+	9.6+	10.1+	10.6+	11.1+	11.7+	12.3+	12.9+	13.5+
2.90 <= E < 3.00	4.8	5.0	5.1	5.3	5.4	5.6	5.8	6.0	6.2	6.5	6.8	7.1	7.5+	7.8+	8.2+	8.6+	9.0+	9.5+	10.0+	10.5+	11.0+	11.6+	12.1+	12.7+	13.4+
3.00 <= E < 3.10	4.8	4.9	5.1	5.2	5.4	5.6	5.7	5.9	6.2	6.4	6.7	7.1	7.4	7.7+	8.1+	8.5+	8.9+	9.4+	9.9+	10.4+	10.9+	11.4+	12.0+	12.6+	13.3+
3.10 <= E < 3.20	4.8	4.9	5.0	5.2	5.3	5.5	5.7	5.9	6.1	6.4	6.7	7.0	7.3	7.7	8.0+	8.4+	8.9+	9.3+	9.8+	10.3+	10.8+	11.3+	11.9+	12.5+	13.1+
3.20 <= E < 3.30	4.7	4.9	5.0	5.2	5.3	5.5	5.7	5.9	6.1	6.3	6.6	6.9	7.2	7.6	8.0+	8.3+	8.8+	9.2+	9.7+	10.2+	10.7+	11.2+	11.8+	12.4+	13.0+
3.30 <= E < 3.40	4.7	4.8	5.0	5.1	5.3	5.4	5.6	5.8	6.0	6.2	6.5	6.8	7.2	7.5	7.9	8.3+	8.7+	9.1+	9.6+	10.1+	10.6+	11.1+	11.7+	12.3+	12.9+
3.40 <= E < 3.50	4.7	4.8	4.9	5.1	5.2	5.4	5.6	5.8	6.0	6.2	6.5	6.8	7.1	7.4	7.8	8.2	8.6+	9.0+	9.5+	10.0+	10.5+	11.0+	11.6+	12.2+	12.8+
3.50 <= E < 3.60	4.7	4.8	4.9	5.1	5.2	5.4	5.6	5.7	5.9	6.2	6.4	6.7	7.0	7.4	7.7	8.1	8.5	8.9+	9.4+	9.9+	10.4+	10.9+	11.4+	12.0+	12.7+
3.60 <= E < 3.70	4.6	4.8	4.9	5.0	5.2	5.4	5.5	5.7	5.9	6.1	6.3	6.6	6.9	7.3	7.6	8.0	8.4	8.8+	9.3+	9.8+	10.3+	10.8+	11.3+	11.9+	12.5+
3.70 <= E < 3.80	4.6	4.7	4.9	5.0	5.2	5.3	5.5	5.7	5.9	6.1	6.3	6.6	6.9	7.2	7.6	7.9	8.3	8.7	9.2+	9.7+	10.2+	10.7+	11.2+	11.8+	12.4+
3.80 <= E < 3.90	4.6	4.7	4.8	5.0	5.1	5.3	5.5	5.6	5.8	6.0	6.3	6.5	6.8	7.1	7.5	7.8	8.2	8.7	9.1	9.6+	10.1+	10.6+	11.1+	11.7+	12.3+
3.90 <= E < 4.00	4.6	4.7	4.8	5.0	5.1	5.3	5.4	5.6	5.8	6.0	6.2	6.5	6.7	7.1	7.4	7.8	8.2	8.6	9.0	9.5	10.0+	10.5+	11.0+	11.6+	12.2+
4.00 <= E < 4.10	4.5	4.7	4.8	4.9	5.1	5.2	5.4	5.6	5.8	6.0	6.2	6.4	6.7	7.0	7.3	7.7	8.1	8.5	8.9	9.4	9.9	10.4+	10.9+	11.5+	12.1+
4.10 <= E < 4.20	4.5	4.6	4.7	4.9	5.0	5.2	5.3	5.5	5.7	5.9	6.1	6.4	6.6	6.9	7.3	7.6	8.0	8.4	8.8	9.3	9.8	10.3	10.8+	11.4+	12.0+
4.20 <= E < 4.30	4.5	4.6	4.7	4.8	5.0	5.1	5.3	5.5	5.7	5.9	6.1	6.3	6.6	6.9	7.2	7.6	7.9	8.3	8.8	9.2	9.7	10.2	10.7	11.3+	11.9+
4.30 <= E < 4.40	4.4	4.6	4.7	4.8	4.9	5.1	5.3	5.4	5.6	5.8	6.0	6.3	6.5	6.8	7.1	7.5	7.9	8.3	8.7	9.1	9.6	10.1	10.6	11.2+	11.8+
4.40 <= E < 4.50	4.4	4.5	4.6	4.8	4.9	5.0	5.2	5.4	5.6	5.8	6.0	6.2	6.4	6.7	7.1	7.4	7.8	8.2	8.6	9.0	9.5	10.0	10.5	11.1	11.7+
4.50 <= E < 4.60	4.4	4.5	4.6	4.7	4.8	5.0	5.1	5.3	5.5	5.7	5.9	6.1	6.4	6.7	7.0	7.3	7.7	8.1	8.5	9.0	9.4	9.9	10.4	11.0	11.6
4.60 <= E < 4.70	4.4	4.5	4.6	4.7	4.8	4.9	5.1	5.3	5.4	5.6	5.8	6.1	6.3	6.6	6.9	7.3	7.6	8.0	8.4	8.9	9.3	9.8	10.4	10.9	11.5
4.70 <= E < 4.80	4.4	4.5	4.5	4.7	4.8	4.9	5.0	5.2	5.4	5.6	5.8	6.0	6.3	6.6	6.9	7.2	7.6	8.0	8.4	8.8	9.3	9.7	10.3	10.8	11.4
4.80 <= E < 4.90	4.4	4.4	4.5	4.6	4.7	4.9	5.0	5.1	5.3	5.5	5.7	6.0	6.2	6.5	6.8	7.2	7.5	7.9	8.3	8.7	9.2	9.7	10.2	10.7	11.3
4.90 <= E < 5.00	4.4	4.4	4.5	4.6	4.7	4.9	5.0	5.1	5.3	5.5	5.7	5.9	6.2	6.5	6.8	7.1	7.5	7.8	8.2	8.7	9.1	9.6	10.1	10.6	11.2

Note that + indicates that the neutron source term limits could not be met at the cooling time post-shutdown and that an assembly specific source term calculation is required to determine the cooling time for the assembly.

TABLE 9.4-2
NUHOMS-24P BURNUP CURVE DATA

INITIAL ENRICHMENT (W/O U^{235})	ACTUAL RESULTS (GWD/MTU)	4th ORDER CURVE FIT DATA[a] (GWD/MTU)
1.8	0.00	0.25
1.9	2.83	
2.	5.17	
2.1	7.28	
2.2	9.20	
2.3	10.73	10.95
2.4	12.56	
2.5	14.05	
2.6	15.45	
2.7	16.77	
2.8	17.65	18.03
2.9	19.25	
3.	20.44	
3.1	21.61	
3.2	22.78	
3.3	23.96	23.96
3.4	25.14	
3.5	26.33	
3.6	27.55	
3.7	28.79	
3.8	29.55	30.04
3.9	31.32	
4.	32.61	
4.1	33.91	
4.2	35.21	
4.3	36.37	36.50
4.4	37.77	
4.5	39.01	
4.6	40.20	
4.7	41.33	
4.8	42.11	42.38

NOTE:

Equation $BU = A*X^4 + B*X^3 + C*X^2 + D*X + E$

Where A = -0.7521212
B = 10.9435555
C = -58.678357
D = 149.626326
E = -134.88247

(a) Fuel burnup in excess of the curve fit data as shown in the table or as calculated by the fourth-order polynomial lie within the acceptance region illustrated in Figure 3.3-1.

This table is not used for assemblies loaded into a NUHOMS-32P or NUHOMS-32PHB DSC.

TABLE 9.4-3
ADDITIONAL COOLING TIME REQUIREMENTS FOR LOADING ASSEMBLIES WITH IRRADIATED STAINLESS STEEL INERT REPLACEMENT RODS IN A 32P DSC

Max SS Pin Exposure MWD/MTU	# SS Pins Allowable by Years after 660W			
	0 Year	1 Year	2 Years	3 Years
20000	0	12	21	30
30000	0	7	13	18
40000	0	5	9	13

TABLE 9.4-4
POST-DISCHARGE COOLING TIME TO MEET 0.8 kW DECAY HEAT LIMIT*

INITIAL ENRICHMENT (w/o U^{235})	BURNUP (MWD/MTU)	COOLING TIME** (Years)
4.50 < E ≤ 5.0	60,000 < B.U. ≤ 62,000	TBD
	58,000 < B.U. ≤ 60,000	TBD
	54,000 < B.U. ≤ 58,000	TBD
	50,000 < B.U. ≤ 54,000	TBD
	47,000 < B.U. ≤ 50,000	TBD
	45,000 < B.U. ≤ 47,000	TBD
	42,000 < B.U. ≤ 45,000	TBD
	B.U. ≤ 42,000	TBD
4.00 < E ≤ 4.50	60,000 < B.U. ≤ 62,000	TBD
	58,000 < B.U. ≤ 60,000	TBD
	54,000 < B.U. ≤ 58,000	TBD
	50,000 < B.U. ≤ 54,000	TBD
	47,000 < B.U. ≤ 50,000	TBD
	45,000 < B.U. ≤ 47,000	TBD
	42,000 < B.U. ≤ 45,000	TBD
	B.U. ≤ 42,000	TBD
3.50 < E ≤ 4.00	60,000 < B.U. ≤ 62,000	TBD
	58,000 < B.U. ≤ 60,000	TBD
	54,000 < B.U. ≤ 58,000	TBD
	50,000 < B.U. ≤ 54,000	TBD
	47,000 < B.U. ≤ 50,000	TBD
	45,000 < B.U. ≤ 47,000	TBD
	42,000 < B.U. ≤ 45,000	TBD
	B.U. ≤ 42,000	TBD
3.00 < E ≤ 3.50	60,000 < B.U. ≤ 62,000	TBD
	58,000 < B.U. ≤ 60,000	TBD
	54,000 < B.U. ≤ 58,000	TBD
	50,000 < B.U. ≤ 54,000	TBD
	47,000 < B.U. ≤ 50,000	TBD
	45,000 < B.U. ≤ 47,000	TBD
	42,000 < B.U. ≤ 45,000	TBD
	B.U. ≤ 42,000	TBD
2.50 < E ≤ 3.00	B.U. ≤ 39,000	TBD
2.00 < E ≤ 2.50	B.U. ≤ 39,000	TBD

* All assemblies loaded into DSC must meet the source spectra requirements of Technical Specification 2.1 (Appendix "A" to SNM-2505).

** These bounding cooling times may be superseded with bundle specific cooling times via explicit bundle specific decay heat calculations.

TABLE 9.4-5
POST-DISCHARGE COOLING TIME TO MEET 1.0 KW DECAY HEAT LIMIT*

INITIAL ENRICHMENT (w/o U^{235})	BURNUP (MWD/MTU)	COOLING TIME** (Years)
4.50 < E ≤ 5.0	60,000 < B.U. ≤ 62,000	TBD
	58,000 < B.U. ≤ 60,000	TBD
	54,000 < B.U. ≤ 58,000	TBD
	50,000 < B.U. ≤ 54,000	TBD
	47,000 < B.U. ≤ 50,000	TBD
	45,000 < B.U. ≤ 47,000	TBD
	42,000 < B.U. ≤ 45,000	TBD
	B.U. ≤ 42,000	TBD
4.00 < E ≤ 4.50	60,000 < B.U. ≤ 62,000	TBD
	58,000 < B.U. ≤ 60,000	TBD
	54,000 < B.U. ≤ 58,000	TBD
	50,000 < B.U. ≤ 54,000	TBD
	47,000 < B.U. ≤ 50,000	TBD
	45,000 < B.U. ≤ 47,000	TBD
	42,000 < B.U. ≤ 45,000	TBD
	B.U. ≤ 42,000	TBD
3.50 < E ≤ 4.00	60,000 < B.U. ≤ 62,000	TBD
	58,000 < B.U. ≤ 60,000	TBD
	54,000 < B.U. ≤ 58,000	TBD
	50,000 < B.U. ≤ 54,000	TBD
	47,000 < B.U. ≤ 50,000	TBD
	45,000 < B.U. ≤ 47,000	TBD
	42,000 < B.U. ≤ 45,000	TBD
	B.U. ≤ 42,000	TBD
3.00 < E ≤ 3.50	60,000 < B.U. ≤ 62,000	TBD
	58,000 < B.U. ≤ 60,000	TBD
	54,000 < B.U. ≤ 58,000	TBD
	50,000 < B.U. ≤ 54,000	TBD
	47,000 < B.U. ≤ 50,000	TBD
	45,000 < B.U. ≤ 47,000	TBD
	42,000 < B.U. ≤ 45,000	TBD
	B.U. ≤ 42,000	TBD
2.50 < E ≤ 3.00	B.U. ≤ 39,000	TBD
2.00 < E ≤ 2.50	B.U. ≤ 39,000	TBD

* All assemblies loaded into DSC must meet the source spectra requirements of Technical Specification 2.1 (Appendix "A" to SNM-2505).

** These bounding cooling times may be superseded with bundle specific cooling times via explicit bundle specific decay heat calculations.

9.5 EMERGENCY PLANNING

The Emergency Response Plan (ERP) for CCNPP has been determined to be adequate for events which might occur involving the ISFSI. The ERP has been prepared in accordance with the requirements of 10 CFR 50.47, and therefore, satisfies the requirements of 10 CFR 72.32.

The CCNPP ERP has been developed to protect the general public and site personnel from possible consequences of an emergency condition. This plan, combined with its implementation procedures and the Radiological Emergency Plans of the state and local agencies, allows for (a) early recognition and classification of a possible emergency condition; (b) prompt notification, via reliable communication channels, of agencies and personnel to augment the normal operating personnel; (c) planned actions to be taken to protect the population-at-risk.

The CCNPP staff is trained to cope with emergencies. Written agreements with Federal agencies, private contractors, and coordinated state and local agency emergency plans (required by law) provide assistance to ensure resources can be readily available in as short a time as possible to cope with emergencies and protect the population-at-risk. The agencies, and the resources they will provide, are described in the ERP and the "Maryland Disaster Assistance Plan, Annex Q, Radiological Emergency Plan." Both plans describe the roles of the various state and local agencies and their interfaces for carrying out protective and parallel actions in a 10-mile-radius plume zone and a 50-mile-radius ingestion zone.

The ERP describes (1) the emergency classification system used at the plant; (2) the organizational control of emergencies, including on-site, off-site, and augmentation organizations; (3) the emergency measures to be taken; and (4) available emergency facilities and equipment.

Procedures for implementation of the CCNPP ERP are contained in the Emergency Response Plan Implementation Procedures.. These procedures are distributed to those individuals, and/or facilities where immediate availability of such procedures would be required during an emergency. The Emergency Response Plan Implementation Procedures provide the following information:

A. Means of classifying emergencies;

B. Lists of available equipment;

C. Directions for meeting notification requirements;

D. Directions for seeking emergency assistance;

E. Detailed instructions to individuals responsible for (a) assessing emergency conditions and (b) providing steps to be taken to mitigate the consequences of the accident.

The Emergency Response Plan Implementation Procedures are used in conjunction with applicable plant operating, radiological control, and security procedures to correct the emergency condition and to mitigate the consequences of the accident. Further details of the CCNPP ERP are contained in UFSAR Section 12.6.

9.6 DECOMMISSIONING PLAN

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

All components of the NUHOMS system are manufactured of materials similar to those found at the existing CCNPP (e.g., reinforced concrete, stainless steel, lead). These components will be decommissioned by the same methods in place to handle those materials within the plant. Any of the components that may be contaminated will be cleaned and/or disposed of using the decommissioning technology available at the time of decommissioning.

Removal of fuel assemblies from the DSC can be done in the plant's spent fuel pool, as described in Chapter 5, or the DSC could also be qualified for off-site shipment in a suitable transportation cask licensed to 10 CFR Part 71. If such transport is made, the DSC could be disposed of as-is at the final spent fuel repository. If the DSC is not compatible with the final repository handling systems, fuel transfer to a suitable container can be performed in any suitable large hot cell or off-site fuel pool.

The detailed decommissioning plan for Calvert Cliffs ISFSI is provided in Reference 9.13.

9.7 REFERENCES

9.1 Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Docket Nos. 50-317 and 50-318, Baltimore Gas and Electric Company

9.2 SFSP Procedure SFSPP-1, Preparation and Control of Spent Fuel Storage Procedures

9.3 SFSP Procedure SFSPP-2, Control of Changes and Deviations Found During Construction

9.4 SFSP Procedure SFSPP-4, Procurement

9.5 SFSP Procedure SFSPP-13, Records Retention

9.6 SFSP Procedure SFSPP-14, Nuclear Related Indoctrination, Training and Qualification

9.7 QAP-36, Independent Spent Fuel Storage Installation

9.8 10 CFR Part 72 Quality Assurance Program for the Spent Fuel Storage Project

9.9 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)

9.10 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated September 30, 1991, Response to NRC's Follow Up Comments on the Safety Analysis Report (SAR) for BGE's License Application for Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI)

9.11 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated December 27, 1991, Response to Requests for Additional Information (RAI), Dated December 12 and 19, 1991, on the Safety Analysis Report (SAR) for BGE's License Application for Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI)

9.12 Letter from Charles H. Cruse (BGE) to Region I, Regional Administrator (NRC), dated October 19, 1993, Calvert Cliffs Nuclear Power Plant Independent Spent Fuel Storage Installation; Docket No. 72-8150-317/318, Preoperational Test Acceptance Criteria and Test Results

9.13 Letter from Mr. G. C. Creel (BGE) to Director, Office of Nuclear Material Safety and Safeguards (NRC), dated August 18, 1992, Revision to the ISFSI Decommissioning Plan

9.14 CCNPP Calculation No. CA06354, "Accidental Drop Loading Evaluation of 14x14 Fuel Assembly with Missing Fuel Rods"

9.15 CCNPP ES200600043, Implementation of ISFSI License Amendment No. 9 Prior to 2010 Loadings

CHAPTER 10

OPERATING CONTROLS AND LIMITS

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CHAPTER 10

OPERATING CONTROLS AND LIMITS

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10.0 OPERATING CONTROLS AND LIMITS

Some of the information formerly contained in this chapter has been moved to other, more appropriate parts of the Updated Safety Analysis Report. Other information was duplicated in the Independent Spent Fuel Storage Installation Technical Specifications and so was deleted from this chapter. Still other information in the chapter was submitted to the Nuclear Regulatory Commission as part of the proposed Technical Specifications, but was not approved. This last information was historical in nature and was also deleted from this chapter.

Table 10.3-2, referenced in the Nuclear Regulatory Commission Safety Evaluation Report dated November 25, 1992, is now Table 9.4-2.

CHAPTER 11

QUALITY ASSURANCE

LIST OF EFFECTIVE PAGES

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QUALITY ASSURANCE

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CHAPTER 11

QUALITY ASSURANCE

LIST OF ACRONYMS

CFR	Code of Federal Regulations
CCNPP	Calvert Cliffs Nuclear Power Plant
HSM	Horizontal Storage Module
HSM-HB	High Burnup Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
NPMD	Nuclear Project Management Department
NSR	Non-Safety-Related
QA	Quality Assurance

11.0 QUALITY ASSURANCE

The quality assurance (QA) program for the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) covers the construction phase, the operational phase, and the decommissioning phase of structures, systems, and components of the ISFSI important to safety. The construction phase includes design, fabrication, construction, and start-up testing. The operational phase includes operation, maintenance, and modification.

The QA program consists of a QA policy, Nuclear Program Directive EN-6, and a QA manual for the horizontal storage modules (HSMs).

The program is applicable to those structures, systems, components, and activities associated with the ISFSI design, construction, maintenance, and operation allowed per the requirements of Subpart G of 10 Code of Federal Regulations (CFR) Part 72, "Quality Assurance." The QA manual for HSM construction phase contains the implementing procedure. Operation phase implementing procedures have also been developed.

11.1 10 CFR PART 50, APPENDIX B, QUALITY ASSURANCE PROGRAM

Activities associated with the operational phase and the decommissioning phase are controlled by existing programs and policies under the Calvert Cliffs 10 CFR Part 50 Appendix B QA program. The construction phase of components identified as safety related in Table 3.4-1 of the ISFSI Updated Safety Analysis Report are also controlled by the 10 CFR Part 50 Appendix B QA program. Activities associated with the construction phase of those components identified as designated non-safety-related (NSR) in Table 3.4-1 are controlled by the 10 CFR Part 72 Subpart G QA program described in Section 11.2 of the ISFSI Updated Safety Analysis Report.

Table 3.4-1 identifies the transfer cask and dry shielded canisters as important to safety; these items, along with the cask lifting yoke, have also been identified as safety-related under 10 CFR Part 50, Appendix B. The Calvert Cliffs Nuclear Power Plant 10 CFR Part 50, Appendix B, Quality Assurance Program is established, maintained, and executed with regard to these components of the ISFSI. This QA program is described in the Quality Assurance Topical Report, as referenced in Appendix 1B of the Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report. This QA program was submitted to the Nuclear Regulatory Commission on December 5, 2005, and was approved by the Nuclear Regulatory Commission on December 21, 2006.

Additionally, activities associated with the operational phase and the decommissioning phase of the ISFSI are controlled by existing programs and policies under the Calvert Cliffs Nuclear Power Plant 10 CFR Part 50, Appendix B, QA Program.

11.2 10 CFR PART 72, SUBPART G, QUALITY ASSURANCE PROGRAM

Table 3.4-1 identifies the Horizontal Storage Modules (HSMs) as important to safety. For construction of additional HSMs, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP) will execute the applicable criteria of 10 CFR Part 72, Subpart G in a graded approach to an extent that is commensurate with the HSMs importance to safety. This graded approach is also known as "augmented quality," or "designated non-safety-related (NSR)," and meets the requirements of 10 CFR Part 50 and 10 CFR Part 72. "Designated NSR" is defined as the quality program which involves the application of certain QA requirements to components to provide reasonable assurance that the structures, systems, components, and activities meet regulatory requirements and commitments. Horizontal Storage Module construction will be controlled in accordance with the following criteria.

11.2.1 QUALITY ASSURANCE ORGANIZATION (10 CFR 72.142, 72.144, 72.160, 72.174, 72.176)

Construction activities will be administered by the Nuclear Project Management Department (NPMD) organization. This organization will use experienced individuals to perform inspection and audit activities.

The QA program for construction will be documented in the QA Manual for the HSM Construction Phase of the ISFSI. This Manual will be prepared by the NPMD organization for "designated NSR" items and services, and approved by the Manager-NPMD.

For "designated NSR" components, inspection of activities affecting quality during construction will be performed as described in the HSM construction guidelines and design specifications. This program establishes inspection requirements to be performed by individuals other than those who performed the activity being inspected, to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Conformance with procurement documents and inspection of items and systems following installation are included in the program.

The records associated with "designated NSR" items are subject to administrative controls and maintained until the Nuclear Regulatory Commission terminates the ISFSI license, in accordance with CCNPP's Nuclear Program.

Planned and periodic audits are carried out to verify compliance with the QA program and to determine the effectiveness of the program in accordance with CCNPP's Nuclear Program.

11.2.2 DESIGN CONTROL (10 CFR 72.146)

The original design of the HSM, its design basis, and any changes to the design not resulting from HSM construction are subject to the existing controls of CCNPP's 10 CFR Part 50, Appendix B, QA Program or the architect engineer's QA program. The architect engineer's QA program has been approved through CCNPP's 10 CFR Part 50, Appendix B, QA Program. If HSM design changes are required solely as a result of the HSM construction activities, the design changes will be administered in accordance with CCNPP's Nuclear Program.

The QA manual for the HSM construction phase of the ISFSI is subject to audit by Nuclear Performance Assessment Department. This manual is an implementing document for the ISFSI EN-6. Proposed changes to EN-6 are reviewed by Nuclear Performance Assessment Department for compliance with applicable standards and licensing basis.

11.2.3 DOCUMENT CONTROL (10 CFR 72.152, 72.150)

Measures to control the issuance of "designated NSR" documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality, are established by various responsible organizations. The measures assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the activity is performed. These measures also ensure that changes to documents are reviewed and approved in accordance with CCNPP's Nuclear Program.

Activities such as HSM/HSM-HB (high burnup horizontal storage module) design modifications, procurement, construction, test, inspection, maintenance, and modification of "designated NSR" components are prescribed and accomplished in accordance with CCNPP's Nuclear Program.

11.2.4 PROCUREMENT DOCUMENT CONTROL (10 CFR 72.148)

Procurement documents for HSM/HSM-HB construction materials and components classified "designated NSR" will be processed in accordance with CCNPP's Nuclear Program.

11.2.5 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES (10 CFR 72.154)

Materials and services will be controlled in accordance with CCNPP's Nuclear Program.

Thorough receipt inspection is performed on the HSM/HSM-HB materials to ensure they conform to the construction specifications and the standards referenced therein. For example, concrete and rebar materials are inspected to verify conformance with the American Concrete Institute's requirements. These controls are similar to, but not as proceduralized as, the procurement and dedication of commercial grade items for use in safety-related applications.

11.2.6 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS (10 CFR 72.156)

Materials, parts, and components of those items determined to be "designated NSR" will be identified and controlled in accordance with CCNPP's Nuclear Program.

11.2.7 HANDLING, STORAGE, AND SHIPPING (10 CFR 72.166, 72.170)

For "designated NSR" items, the handling, storage, shipping, and preservation of materials will be controlled in accordance with CCNPP's Nuclear Program.

Controls for identifying, documenting, segregating, reviewing, reporting and the tagging of non-conforming materials, parts, components, or services will be applied to the "designated NSR" activities in accordance with CCNPP's Nuclear Program.

11.2.8 CONTROL OF SPECIAL PROCESSES (10 CFR 72.158)

Special processes, including welding, heat treating, and non-destructive testing for "designated NSR" components, will be controlled in accordance with CCNPP's Nuclear Program.

Calvert Cliffs Nuclear Power Plant ensures that contractors and their subcontractors acceptably control special processes through surveillance and CCNPP qualification training of contractor personnel.

11.2.9 TEST CONTROL (10 CFR 72.162, 72.164, 72.168)

Testing will be performed as described in the HSM design specifications, construction program guidelines, and procurement documents to demonstrate those structures, systems, and components will perform satisfactorily in service upon completion of the construction phase.

For "designated NSR" items, CCNPP and its approved suppliers shall implement measures to ensure that tools, gauges, instruments, and other measuring and testing devices used in the activities affecting quality are properly controlled, calibrated, and adjusted at regular intervals against certified or recognized standards, and maintained traceable to the National Institute of Standards and Technology.

Measures will be used to indicate the status of inspections and tests performed upon individual items.

11.2.10 CORRECTIVE ACTION (10 CFR 72.172)

For the construction phase of "designated NSR" components, any discrepancies discovered by the inspector or contractor of CCNPP will be resolved by reworking the item to comply with drawing or specification requirements; or by documenting and resolving the discrepancy in accordance with the construction specification; or by accepting the discrepancy "as-is" and documenting the resolution by an engineering evaluation; or by scrapping the item.

In the case of a significant condition adverse to quality, measures are established to ensure that the cause of the condition is determined and corrective action is taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to the appropriate levels of management in accordance with CCNPP's Nuclear Program.

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LIST OF ACRONYMS

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
CCNPP	Calvert Cliffs Nuclear Power Plant
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
CSAS25	Criticality Safety Analysis Sequence No. 25
DSC	Dry Shielded Canister
HSM	Horizontal Storage Module
HSM-HB	High Burnup Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
NUHOMS	Nutech Horizontal Modular Storage®
TR	Topical Report
USAR	Updated Safety Analysis Report
USL	Upper Subcriticality Limit
VAP	Value Added Pellet

12.0 NUHOMS-32P DRY SHIELDED CANISTER

An evaluation of the Nutech Horizontal Modular Storage (NUHOMS)-32P Dry Shielded Canister (DSC) used in the NUHOMS dry storage system is presented in this Chapter. Chapter 1 is revised to include information for both the NUHOMS-24P and NUHOMS-32P DSCs. Chapters 2 through 11 primarily apply to the NUHOMS-24P DSC, whereas Sections 12.2 through 12.11 provide the same information as it applies to the NUHOMS-32P DSC.

General references are identified throughout the body of this chapter, and are listed in Section 12.12. General references are intended to provide background information or additional detail that the reader may refer to in order to learn more about a particular topic presented in this document, but are not considered part of the Updated Safety Analysis Report (USAR). A referenced document shall be considered to be a part of the USAR only if it is clearly annotated as being "incorporated by reference" in Chapter 12 of this report. Documents that are incorporated by reference are subject to the same administrative controls and regulatory requirements as the USAR.

12.1 INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

The introduction and general description of the NUHOMS-32P is integrated into Chapter 1.

12.2 SITE CHARACTERISTICS

The Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) site characteristics are discussed in Chapter 2. The evaluation presented in Chapter 2 is not affected by the addition of the NUHOMS-32P DSC as part of the NUHOMS System.

12.3 PRINCIPAL DESIGN CRITERIA

12.3.1 PURPOSE OF THE CALVERT CLIFFS INDEPENDENT SPENT FUEL STORAGE INSTALLATION

Information contained in Section 3.1 is applicable to the NUHOMS-32P DSC design as well.

12.3.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

Compared to the existing NUHOMS-24P DSC, the main difference in the principal design parameters for the NUHOMS-32P DSC consists of an increase in canister weight, radiological source, and decay heat due to the addition of eight more fuel assemblies plus modifications to the DSC basket assembly. The NUHOMS-32P DSC is handled and stored in the HSM in the same manner as the NUHOMS-24P DSC. The NUHOMS-32P DSC is handled and stored in the HSM-HB in a similar manner to the HSM with slight differences during HSM-HB loading due to design differences. The environmental conditions and natural phenomena for a NUHOMS-32P DSC are the same as those described in Section 3.2 except as discussed below.

12.3.2.1 Tornado Wind and Tornado-Generated Missile Loadings

For the HSM, the principal tornado wind and tornado-generated missile loading criteria are not dependent on the type of DSC employed and, therefore, are the same for the NUHOMS-32P and NUHOMS-24P DSCs. The tornado wind and tornado-generated missile loading criteria are described in Section 3.2.1 for the HSM. They are based on Reference 3.1, which applies to the NUHOMS-24P DSC, but are applicable to the NUHOMS-32P DSC as well.

For the HSM-HB, the principal tornado wind is the same as the HSM. The tornado generated missiles are those corresponding to the HSM-H design (Reference 12.77) and are either more conservative or the same as the HSM. The HSM-HB tornado missile criteria are derived using the same basis as the HSM (Reference 3.8). The tornado wind and tornado-generated missile loading criteria are described in Section 3.2.1.

Evaluation of the NUHOMS-32P DSC for tornado wind and tornado generated missiles is presented in Section 12.8.2.2.

12.3.2.2 Water Level (Flood) Design

As stated in Section 3.2.2, the Calvert Cliffs ISFSI is not subject to flooding. Therefore, the type of DSC and horizontal storage module HSM/HSM-HB used is independent of the flood water level.

12.3.2.3 Seismic Design

For the HSM, the principal seismic loading criteria are not dependent on the type of DSC employed and, therefore, are the same for the NUHOMS-32P and NUHOMS-24P DSCs. The seismic loading criteria are described in Section 3.2.3.

For the HSM-HB, the principal seismic loading criteria are those corresponding to the HSM-H design (Reference 12.77) and are more conservative than those described in Section 3.2.3 for the HSM. The HSM-HB seismic criteria derived using the same basis as the HSM (NRC RG 1.60 and 1.61). The seismic loading criteria are described in Section 3.2.3.

Evaluation for the use of the NUHOMS-32P DSCs for seismic loading is presented in Section 12.8.2.3.

12.3.2.4 Snow and Ice Loadings

The snow and ice loads are the same for the NUHOMS-32P and NUHOMS-24P DSCs and are described in Section 3.2.4.

12.3.2.5 Combined Load Criteria

12.3.2.5.1 HSM and HSM-HB

The Calvert Cliffs site-specific load combinations matrix and design criteria for the HSM storing the NUHOMS-32P and NUHOMS-24P DSCs are the same and are presented in Section 3.2.5.1 and Table 3.2-2. These load combinations have also been evaluated for storage of a NUHOMS-32P DSC in the high burnup horizontal storage module (HSM-HB).

Structural evaluation of the HSM for the NUHOMS-32P DSC is presented in Section 12.8.

12.3.2.5.2 NUHOMS-32P DSC

The Calvert Cliffs site-specific load combinations for the NUHOMS-32P DSC are presented in Table 12.3-6.

Structural evaluation of the NUHOMS-32P DSC basket assembly is presented in Section 12.8.

12.3.2.5.3 NUHOMS Transfer Cask

The same transfer cask is used for the NUHOMS-32P and NUHOMS-24P DSCs. The Calvert Cliffs site-specific load combinations and allowable stress criteria for the transfer cask are the same as those presented in Section 3.2.5.3.

Structural evaluation of the transfer cask for the NUHOMS-32P DSC is presented in Section 12.8.

12.3.2.5.4 NUHOMS System Transfer Equipment

The load combinations and acceptance criteria for the transfer equipment are not affected by the use of the NUHOMS-32P DSC and are the same as those in Section 3.2.5.4.

12.3.2.6 Weld Requirements

The NUHOMS-32P DSC shell assembly welded joint details are the same as for the NUHOMS-24P DSC shell assembly (References 12.35 through 12.46).

The NUHOMS-32P DSC basket structure welded joint details are shown in References 12.35 through 12.46. Full penetration welds in the DSC basket assembly are examined by progressive penetrant test examination in accordance with the requirements of paragraph NG-5231 of the American Society of Mechanical Engineers (ASME) Code Section III, Subsection NG (References 12.26 and 12.41).

Weld analysis results for the NUHOMS-32P DSC and the transfer cask are presented below.

<u>DSC Shell and Basket Assembly Welds</u> (Reference 12.22)

- Shell Welds

The DSC Shell can be manufactured from pieces of cylindrical plates. These shell pieces are welded together by full penetration welds, both longitudinally and circumferentially. The weld is also 100% radiographed.

- Support Ring Welds

The Support Ring consists of a 1.75" high tapered ring and a 0.25" high plate that are welded to the DSC shell. When the Support Ring is loaded, the 1/4" fillet weld between the ring and plate is in compression, and the upper ring will bear against the lower plate. The welds that require analysis are the two 1/4" groove welds at the top and bottom of the Support Ring weldment. The Support Ring supports the weight of the top lead shield plug assembly, conservatively assumed to weigh 8,000 lbs. The Support Ring self-weight is an additional 300 lbs. The critical loading condition occurs during the 75g bottom end vertical drop. The Service Level D allowable weld stress at the DSC design temperature of 380°F is 22.55 ksi. The calculated weld shear and tensile stress is within allowable limits.

- Lead Shield Plug Assembly Welds

Each of the four top shield plug assembly round bars are welded to the inner cover plate by a 5/16" fillet weld. The 1.5" diameter bars are loaded only during preliminary handling of the top shield plug assembly. In all other cases, the weld is in compression. The weight of the top shield plug assembly is conservatively 8,000 lbs. The allowable stress at room temperature is 10.0 ksi. Conservatively assuming that two of the four plugs carry the load, the calculated stress on the weld is within the allowable limits.

- Top Shield Plug Pressure Boundary Weld

The inner cover plate on the top shield plug is welded to the DSC shell by a 3/16" groove weld. The highest load on the weld occurs during the 100 psig accident pressure load case. The pressure is applied to the top inner cover plate and loads the weld in pure shear. The Service Level D allowable weld stress at 380°F is 22.55 ksi. The calculated stress is within allowable limits.

- Top Cover Plate Weld

The top cover plate is welded to the DSC by a groove weld, which has a ½" minimum throat. The maximum load occurs during the bottom end drop, where the weld is loaded by the weight of the outer cover plate at 75g. The Service Level D allowable weld stress at 380°F is 22.55 ksi. The calculated shear stress on the weld is within allowable limits.

- Bottom Cover Plate Weld

The bottom end structural weld is a full penetration weld and is 100% radiographed.

- Bottom Lead Casing Plate Welds

The bottom lead casing plate is welded to the lead casing shell plate by a 1/4" groove weld. During a lifting of the DSC into the transfer cask, the shearing load on this weld is equal to the dead weight of the bottom shield plug and the bottom lead liner plate (conservatively 7,500 lbs). The allowable stress at room temperature is 10.0 ksi. The calculated weld shear stress is within allowable limits.

The lead casing shell is welded to the bottom cover plate by a 1/4" groove weld on the outside, and a 1/8" fillet weld on the inside. These welds carry the same loads as the 1/4" groove weld between the lead liner plate and the lead casing shell and are acceptable.

The 5/16" bottom weld applied to the plug post and bottom cover plate interface is a non-structural weld.

- Grapple Ring Assembly Welds

The grapple ring is welded to the grapple ring shell by a full penetration groove weld.

The grapple ring assembly plate is welded to the grapple ring shell on the inside by a 3/8" groove weld with a 1/8" fillet weld, and on the outside by a 1/4" fillet weld. These welds carry some of the lead shield during a vertical lift. The 1/4" groove weld is adequate under this loading condition.

The grapple ring shell is welded to the inner cover plate by a full penetration weld with a 1/8" fillet cover weld.

A 1/8" cover fillet weld over a 3/8" groove weld is applied to the outside of the grapple ring shell and the bottom cover plate interface. Conservatively ignoring the cover fillet weld, the 3/8" groove weld is loaded by a maximum of 95,000 lbs when the DSC is pushed by the ram assembly. The Service Level A allowable stress at 380°F is 9.48 ksi. The calculated total shear stress on the weld is within the allowable limit.

- Vent and Siphon Port Welds

The plates welded to the vent and siphon ports form part of the pressure boundary. The plates are 3.40" in diameter, and are welded to the drain and fill ports by 3/16" groove welds. Under worst case conditions, the plates carry a pressure of 100 psig. The Service Level D allowable weld stress at 350°F is 22.55 ksi. The calculated weld stress is within the minimum allowable stress.

- Guide Sleeve Fusion Weld

The weld strength of each fusion weld nugget connecting the fuel compartment guide sleeves of the DSC basket structure is required to be determined by shear tests using test specimens made from production material and shall have a minimum capacity of 16.0 kips at 70°F (Reference 12.41). The allowable design strength of the fusion weld is established by applying a safety factor of 2 and correction for the 600°F design temperature (Reference 12.28).

Correcting for reduction in the weld allowable stress from 75 ksi at room temperature to 63.4 ksi at 600°F, the minimum provided weld capacity at 600°F is 15.5 kips. The calculated load in the fusion weld connection of the fuel compartment guide sleeves is within the minimum allowable stress.

<u>Transfer Cask Welds</u> (Reference 12.25)

- Weld At Inner Bottom Cover Plate/Ram Access Penetration Ring

This weld is analyzed for the critical lift handling condition and the vertical bottom drop accident condition.

  - Critical Lift Handling Condition

  The inner bottom cover plate supports the weight of the DSC, conservatively taken to be 95 kips (Reference 12.34), during the critical lift condition. The load is increased by 15% to account for motion loads. The fuel and DSC are considered to load the cask base plate as a uniform pressure load. The remaining weight of the DSC, which includes the basket, shell, and top plate are conservatively represented as a pressure load that decreases toward the center of the plate. The calculated fillet weld size required is 0.35" (Reference 12.25). The existing 3/8" minimum weld size is adequate.

  - Vertical Drop Condition (Level D)

  The weld stress for this condition is based on the results of the calculation for the NUHOMS-24P DSC (Reference 12.50) for the transfer cask vertical bottom drop accident and scaling up the stress by a factor of 1.1 to reflect the weight increase of the NUHOMS-32P DSC. The weld stress (using stainless steel allowables) is within the minimum allowable stress of 22.4 ksi.

- Weld at Cask Inner Shell Plate/ Top Flange Ring

The weld stress is based on the results of calculation for the NUHOMS-24P DSC (Reference 12.50) for transfer cask vertical bottom drop accident and scaling up by a factor of 1.1 the stress to reflect the weight increase of the

NUHOMS-32P DSC. The calculated weld stress is less than the allowable stress of 22.4 ksi.

The analyses described above demonstrate that the important-to-safety components of the Calvert Cliffs ISFSI with NUHOMS-32P DSC are adequate to withstand all postulated loads and loading combinations.

12.3.3 SAFETY PROTECTION SYSTEMS

12.3.3.1 General

Section 3.3 discusses the Calvert Cliffs ISFSI design for the safe and secure long-term containment and storage of spent fuel. The NUHOMS-32P DSC is designed for storage of spent nuclear fuel as described in Section 3.3.1 and in the following subsections.

12.3.3.2 Protection by Multiple Confinement Barriers and Systems

The NUHOMS-32P DSC provides confinement of the spent fuel similar to the NUHOMS-24P DSC. Sealing of the NUHOMS-32P DSC is leak tested in accordance with American National Standards Institute (ANSI) N14.5 after loading and sealing the canister, as described in Section 3.3.2.

Containment of radioactive material associated with spent fuel assemblies is provided by fuel cladding, the DSC stainless steel shell and double seal welded primary and secondary closures. As described in Section 3.3.2, there are no credible events that will breach a DSC to provide a possible leakage path to the environment.

12.3.3.3 Protection by Equipment and Instrumentation Selection

The protection by equipment and instrumentation is not impacted by the use of the NUHOMS-32P DSC, and remains the same as presented in Section 3.3.3.

12.3.3.4 Nuclear Criticality Safety

The NUHOMS-32P DSC internals are designed to provide nuclear criticality safety during all phases of NUHOMS system operations and storage, including wet loading operations and postulated accident conditions. The Calvert Cliffs site-specific NUHOMS-32P DSC design satisfies the requirements of 10 Code of Federal Regulations (CFR) 72.124 for normal, off-normal, and accident conditions.

12.3.3.4.1 Control Methods for Prevention of Criticality

Criticality control is provided during the transfer 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 of the fuel, fixed neutron absorbers in the NUHOMS-32P basket, 2,450 ppm soluble boron in the spent fuel pool water, and Calvert Cliffs' administrative controls for fuel identification, verification, and handling.

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 is 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, there is no credible accident scenario which would result in the possibility of the entrance of a moderator into the DSC; nor is there a credible accident scenario which would prohibit the canister from being opened and re-flooded.

12.3.3.4.2 Design Parameters for Criticality Model

The design basis criticality analysis uses design parameters for Combusting Engineering, Inc. (CE) design 14x14 standard and Value Added Pellet (VAP) fuel assemblies containing UO_2 enriched up to 4.5 wt% U^{235} with geometry and fuel characteristics as shown in Table 3.3-3. Only VAP Batches 1NT, 1T, 2S, 1V, 1W, and 2T (excluding 2TF and 2TW lead fuel assemblies) are analyzed for loading in the NUHOMS-32P DSC. The nominal dimensions of the NUHOMS-32P DSC are provided in Table 12.3-1. The geometry is illustrated in Figure 12.3-1. A summary of the design parameters for the criticality analysis is presented in Table 12.3-2.

Additional analyses are performed for fuel misloads and accidents. The design parameters for misloads and accidents are also presented in Table 12.3-2.

12.3.3.4.3 Criticality Analysis Methods

Effective neutron multiplication factors, k_{eff}, are calculated using the Criticality Safety Analysis Sequence No. 25 (CSAS25), of the SCALE-4.4 package of codes, and the 44 Group ENDF-V cross-section library. The CSAS25 control module allows simplified data input to the functional modules BONAMI-S, NITAWL-S, and KENO V.a. These modules process the required cross-sections and calculate the k_{eff} of the system. BONAMI-S performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL-S applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0010 for all calculations.

The final k_{eff} that is calculated represents the maximum value of the effective multiplication factor with a 95% probability at a 95% confidence level (95/95). The "worst case" k_{eff} values from the CSAS25 output are adjusted for uncertainty, such that:

$$k_{eff} = k_{keno} + 2\sigma_{keno}$$

A series of 121 benchmark criticality calculations are documented in Reference 12.6. These calculations assume unirradiated fuel in the criticality analysis and use the SCALE-4.4 computer code package. The upper subcriticality limit (USL), as described in Section 4 of NUREG/CR-6361 (Reference 12.5), is determined using the results of these 121 benchmark calculations. The benchmark problems used to perform this verification are representative of benchmark arrays of commercial light water reactor fuels with the following characteristics:

- water moderation
- boron neutron absorbers
- unirradiated light water reactor type fuel (no fission products or "burnup credit")
- near room temperature (vs. reactor operating temperature)
- close reflection
- uranium oxide fuel

The 121 uranium oxide experiments are chosen to model a wide range of uranium enrichments, fuel pin pitches, assembly separation, water/fuel ratio, concentration of soluble boron and control elements in order to test the codes ability to accurately calculate k_{eff}. The minimum value of the USL from Reference 12.6 over the parameter range (in this case, the assembly separation distance) is 0.9422. This USL value (0.9422) is based on a methodology bias and an administrative 5% margin on criticality. That is, k_{eff} < USL, ensures that k_{eff} is less than 0.95 (with 95% probability and 95% confidence) when bias and uncertainty are taken into account.

For the criticality analyses, the criticality limits are shown in the following equation:

$$k_{eff} = (k_{keno} + 2\sigma_{keno}) \leq 0.9422$$

12.3.3.4.4 Normal Conditions (References 12.7 and 12.71)

The calculated normal condition, "worst-case," reactivity (maximum k_{eff}) of a fully loaded Calvert Cliffs NUHOMS-32P DSC is 0.9412. This is below the USL (0.9422), thus confirming that the "worst case" k_{eff} is $\leq$ 0.95. It conservatively includes allowances for uncertainties due to fuel positioning, basket rail modeling, compartment tube dimensions, poison plate thickness, and optimum moderator density. The "worst case" configuration includes the following:

- no credit for burnable absorbers in the fuel rods (e.g., erbia, etc.),
- fuel is unirradiated,
- the maximum uniform enrichment, 4.5 wt% U^{235}, for all 32 assemblies,
- an "inward" loading of all the 32 CE 14x14 standard or VAP fuel assemblies (i.e., all fuel assemblies are shifted toward the center of the DSC),
- credit for 90% of the absorber material (B^{10}) in the fixed neutron absorbers in the NUHOMS-32P basket assembly,
- a minimum compartment tube dimension of 8.47",
- an internal moderator (soluble boron at 2,450 ppm) density of 70%,
- an external (to the DSC and internal to the transfer cask) moderator (pure water) density of 10%,
- AREVA fuel was not analyzed.

To evaluate assemblies containing vacancies and inert stainless steel rods, Reference 12.58 utilizes the same methodology as Reference 12.7 to analyze criticality under normal conditions. No changes were made to the computer code software and version or to the characteristics of the KENO V.a models other than the changing of the fuel array input to include vacancies and inert stainless steel pins. The material specification used for the stainless steel rods is consistent with the methodology in Reference 12.7. The calculated 95/95 (95% probability and 95% confidence) k_{eff} for all cases analyzed are below the USL of 0.9422.

12.3.3.4.5 Off-Normal Conditions (References 12.8 and 12.71)

Four postulated off-normal conditions are analyzed:

- The misloading of AREVA or VAP fuel assemblies into the DSC with an initial enrichment of 5.0 wt% U^{235},
- Cask Drop Accidents,
- B^{10} Absorber Plates at Minimum Thickness, and
- Optimum Moderator Density.

These analyses confirm that the off-normal conditions will not result in a DSC storage array with a reactivity higher than the USL of 0.9422.

Misloading of AREVA or VAP Assemblies (5 wt% U^{235})

AREVA or VAP fuel assemblies with enrichments exceeding 4.5 wt% U^{235} are not currently licensed for storage in

NUHOMS-32P canisters. However because it resides in the spent fuel pool, a misload of higher enriched AREVA or VAP fuel into a canister was analyzed. The criticality analysis for the fuel misloads demonstrates that a maximum of one AREVA or VAP fuel assembly at an enrichment of 5.0 wt% U^{235} can be misloaded and transferred under optimum moderator density conditions. The k_{eff} for this case is 0.9418.

Reference 12.58 shows a clear and expected trend of lower k_{eff} with increasing number of fuel rods removed. This trend applies to the case of accidental misloading of VAP or AREVA assemblies as well. Therefore, there is no impact on the misload criticality analysis results as presented above.

Cask Drop Accidents

The criticality analysis for the cask drop accidents demonstrates that the most reactive configuration is the triple contingency accident involving fuel damage, optimum pitch (due to grid deformation), and optimum moderator density. For the helium-moderated system, the k_{eff} is 0.5737 which is below the USL (0.9422). For the borated water moderated system, the maximum k_{eff} is 0.9413, which is also below the USL (0.9422).

Reference 12.59 determines the structural adequacy of a standard fuel assembly with burnup < 47,000 MWD/MTU containing vacancies following a cask drop event. The results show that the bending stresses are acceptable provided that no more than two vacancies are in any one column or row of the fuel assembly. It is not required that the vacancies be adjacent. Vacancies are restricted for fuel with burnup between 47,000 MWD/MTU and 52,000 MWD/MTU pending further structural analysis. The criticality analysis of Reference 12.58 has postulated that an assembly containing numerous vacancies can be postulated to be rearranged into a 13x13 array if grid spacer integrity is compromised. Reference 12.58 has analyzed a 13x13 array with maximum pitch separation and has determined that the k_{eff} of such a scenario is 0.9012. The most reactive case, with vacancies, was found to be when one vacancy was present in each of the four center assemblies. The assemblies were modeled in the standard 14x14 array and at optimum fuel rod pitch; the resulting k_{eff} is 0.9390. Both of these scenarios results in a k_{eff} less than the bounding analysis which finds the k_{eff} to be 0.9413. All of these values are less than the USL of 0.9422.

B10 Poison Plate Thickness Variation

The criticality analysis for sensitivity to B^{10} absorber plate thickness demonstrates that there is enough conservatism in the plate loading of 10.0 mg B^{10}/cm^2 to offset changes in reactivity due to a reduction in thickness. Credit is taken for

90% of the B^{10} loading in the analysis. For the normal case, with an absorber plate thickness of 0.04", the maximum k_{eff} is calculated to be 0.9412 (Reference 12.7). For the "worst case," with a B^{10} loading of 8.964 mg/cm^2, a thickness of 0.035", and optimum moderator density, the k_{eff}, is calculated to be 0.9357 (Reference 12.8).

Section 12.3.3.4.7 has a detailed discussion on poison plate acceptance testing.

<u>Optimum Moderation</u>

Since all reported reactivities include an allowance for optimum moderator density, and all reported reactivities are less than the USL, a criticality event due to moderator density alone is not credible. For a misload of a 5% enriched VAP or AREVA assembly, optimum moderator density will not result in criticality, provided that at least 2,450 ppm of boron is present in the water inside the DSC. Therefore subcriticality is assured, even in the event that a flooded DSC remains out of the pool long enough for boiling to occur.

12.3.3.4.6 Criticality Analysis Method Verification

The analysis method which ensures a subcriticality margin of greater then 5% under all normal conditions uses the CSAS25, of the SCALE-4.4 package of codes, and the 44 Group ENDF-V cross-section library.

A series of 121 benchmark criticality calculations are documented in Reference 12.6. These calculations assume unirradiated fuel in the criticality analysis and use the SCALE-4.4 computer code package to demonstrate its applicability and to establish methods bias and variability.

12.3.3.4.7 B^{10} Poison Plate Testing

<u>Description</u>

The poison plates consist of wrought aluminum containing boron, which is isotopically enriched to approximately 95 wt% B^{10}. Because of the negligibly low solubility of boron in solid aluminum, the boron appears entirely as discrete second phase particles of AlB_2 in the aluminum matrix. The effect on the properties of the matrix aluminum alloy are those typically associated with a uniform fine (1-10 micron) dispersion of an inert equiaxed second phase.

The nominal plate thickness is 0.04".

The nominal boron concentration is 3.9 wt%. The design minimum B^{10} areal density is 0.0100g B^{10}/cm^2.

Functional Requirements of Poison Plates

The poison plates serve as a neutron absorber for criticality control and as a heat conduction path. The NUHOMS-32P DSC safety analysis does not rely upon their mechanical strength. The radiation and temperature environment in the cask is not severe enough to damage the aluminum matrix that retains the boron-containing particles. To assure performance of the plates' important-to-safety functions, the critical variables that need to be verified are thermal conductivity and B^{10} areal density.

Borated Aluminum Test Coupon and Lot Definitions

Test coupons will be taken so that there is at least one coupon contiguous with each plate. These coupons will be used for neutron transmission and thermal conductivity testing.

A lot is defined as all the plates produced from a single cast ingot, or all the plates produced from a single heat.

Thermal Conductivity Testing of Poison Plates

The poison plate material is qualification-tested to verify that the thermal conductivity equals or exceeds the design requirements.

Testing may be by American Society for Testing and Materials (ASTM) E1225 (Reference 12.51), ASTM E1461 (Reference 12.52), or equivalent method.

B10 Areal Density Testing of Poison Plates

The testing program for the NUHOMS-32P DSC poison plates meet the requirements of NUREG/CR-5661 (Reference 12.53).

The effective B^{10} content is verified by neutron transmission testing of the coupons. The transmission through the coupons is compared with transmission through calibrated standards. The neutron transmission testing measurements are taken using a collimated neutron beam. The neutron transmission test procedure includes provisions to vary the selected measurement location along the coupon length.

The acceptance criterion for neutron transmission testing is that the B^{10} areal density, minus 3σ based on the number of neutrons counted for that measurement, must be greater than or equal to the minimum value 0.0100g B^{10}/cm^2.

Macroscopic uniformity of B^{10} distribution is verified by neutron radioscopy/radiography of the coupons. The acceptance criterion is that there is uniform luminance across the coupon. This inspection shall cover the entire coupon. If a coupon fails this test, the associated plate is rejected.

In addition, a statistical analysis of the neutron transmission results for all plates in a lot is performed. This analysis shall demonstrate, using a one-sided tolerance limit factor for a normal distribution with at least 95% probability, the areal density is greater than or equal to the specified minimum value of 0.0100g B^{10}/cm^2 with 95% confidence level.

12.3.3.5 Radiological Protection

The discussion presented in Section 3.3.5 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System. See Section 12.7 for additional discussion on radiation protection design considerations for the NUHOMS-32P DSC.

12.3.3.6 Fire and Explosions Protection

The discussion presented in Section 3.3.6 is applicable to NUHOMS-32P DSC. The effects of a forest fire around the facility are discussed in Section 12.8.2.10.

12.3.3.7 Materials Handling and Storage

The evaluation presented in Section 3.3.7 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS system, with the exception of peak cladding temperatures which are higher than those of the NUHOMS-24P DSC. For long-term storage, HSM passive ventilation maintains the maximum normal operating fuel clad temperature at 620°F or less (assuming 103°F ambient temperature) as documented in Reference 12.1. 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 (References 12.1, 12.2, 12.3, and 12.4), the fuel cladding temperature maximum value is 838°F (Reference 12.9), which is significantly less than the maximum allowable value of 1,058°F.

12.3.3.8 Industrial and Chemical Safety

The discussion presented in Section 3.3.8 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System.

12.3.4 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

The discussion presented in Section 3.4 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System.

12.3.5 DECOMMISSIONING CONSIDERATIONS

The discussion presented in Section 3.5 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System.

12.3.6 SUMMARY OF DESIGN CRITERIA

Tables 12.3-3 to 12.3-5 provided a summary of the design criteria information for the normal, off-normal, and accident conditions for the NUHOMS-32P DSCs, respectively.

TABLE 12.3-1
NUHOMS-32P DRY SHIELDED CANISTER DIMENSIONS

GEOMETRY DESCRIPTION	NOMINAL DIMENSIONS (inches)
Guide Sleeve Inside Diameter	8.50
Guide Sleeve Thickness	0.1874
Center to Center Spacing	9.125
Stainless Steel Strip Thickness	0.25
Aluminum + Poison Plate Thickness	0.25
Basket Assembly Length	158.0 max.
DSC Shell Outside Diameter	67.25
DSC Shell Inside Diameter	66.0
DSC Shell Length (with grapple ring)	176.5
DSC Shell Thickness	0.625
Top Shield Plug Thickness	6.25 [(a)]
Top Cover Plate Thickness	1.25
DSC Lead Shielding Thickness	
• Top Shield Plug	4.0 min.
• Bottom Shield Plug	4.25 min.
Vent / Siphon Port Tube Inside Diameter	1.05

(a) The top casing plate which is part of the top shielding plug has been determined, using ultrasonic testing inspections, to have a reduced plate thickness. All of the 32Ps top casing plates have a reduced plate thickness from 0.75" to a new potential minimum thickness to as low as 0.27". Reference 12.66 and 12.67 qualified a new top casing plate thickness of 0.25" for all of the 32P DSCs. The nominal thickness of 6.25" for the top shielding plug remains unchanged. The details of the 32Ps casing plate thicknesses have been evaluated in ECP-11-000793.

TABLE 12.3-2
DESIGN PARAMETERS FOR CRITICALITY ANALYSIS OF THE NUHOMS-32P DSC

PARAMETERS	DESIGN VALUE
FUEL ASSEMBLIES	
Number/Type	32/CE design 14x14
Rod Array	14x14[d]
Number of Fuel Rods	176[c]
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	Not Applicable for NUHOMS-32P DSC
FISSILE CONTENT	
wt% U^{235}	4.5 max.
wt% U^{235} (misload)	5.0 max.
FUEL PELLETS	
Density (standard)	96.0% Theoretical
Density (VAP)	95.0% Theoretical (max Batches 1NT, 1T, 2S, 1V, 1W & 2T)
Density (AREVA or VAP misload)	96.66% Theoretical
Diameter (inches) (standard)	0.3765[a]
Diameter (inches) (AREVA or VAP)	0.3810
FUEL ROD CLADDING Material	Zircaloy-4 or Zirlo (VAP only), M5 (AREVA misload)
Thickness (inches) (standard)	0.028[a]
Thickness (inches) (VAP or AREVA)	0.026
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 COMPARTMENTS	
Material	Stainless Steel
Thickness (inches)	0.1874
Inside Diameter (inches)	8.5
DSC POISON PLATES	
Number	150
Material	Borated Aluminum Alloy or Boron Aluminum Metal Matrix Composite
Density (g/cm^3)	2.693
Thickness (inches)	0.04
B^{10} Areal density (mg/cm^2)	10
Location	See Figure 12.3-1

TABLE 12.3-2
DESIGN PARAMETERS FOR CRITICALITY ANALYSIS OF THE NUHOMS-32P DSC

PARAMETERS		DESIGN VALUE
DSC FILL MATERIAL		
Material	(wet)	Borated Water (2,450 ppm min)
Moderator Density	(wet)	0.01% to 100%
Material	(dry)	helium
Moderator Density	(dry)	1.785E-04 g/cm^3 /atm
DSC SHELL		
Material		Stainless Steel
Thickness (inches)		0.625
Outside Diameter (inches)		67.25
CASK		
Material		Stainless Steel/Lead
Thickness (inches)		6.25[(b)]
Outside Diameter (inches)		80.5[(b)]

(a) The fuel pellet outside diameter 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 the design basis analysis.

(b) Exclusive of the cask neutron shield.

(c) <u>Fuel Rods/Assembly (32P)</u>

Standard fuel assemblies with burnup < 47,000 MWD/MTU 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.

(d) Accident analyses for fuel assemblies containing vacancies considers multiple array configurations. See Reference 12.58 for details.

TABLE 12.3-3
NUHOMS-32P SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
HSM/HSM-HB	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	USAR Table 3.2-2	Load combination methodology	ANSI 57.9-1984 Sec 6.17.1.1
	Design Basis Operating Temperature		DSC with spent fuel rejecting 21.12 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*	ANSI 57.9-1984
	Snow and Ice Loads	USAR 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	USAR 4.2.3.1	Contact dose rate on HSM exterior surface ≤ 20 mrem/hr. HSM door ≤ 100 mrem/hr.	ANSI 57.9-1984
DSC	Dead Loads		Weight of loaded DSC: 91,000 lb nominal, 95,000 lb enveloping	ANSI 57.9-1984
	Design Basis Internal Pressure Load		DSC internal pressure 10.1 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		DSC decay heat 21.12 kW. Ambient air temperature -3°F to 103°F	ANSI 57.9-1984
	Operational Handling	USAR Table 3.2-1	Hydraulic ram load: 20,000 lb	ANSI 57.9-1984
	Criticality	USAR 12.3.3.4	K_{eff} less than 0.95	ANSI 57.9-1984

TABLE 12.3-3
NUHOMS-32P SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
DSC Support Assembly	Operational Handling	USAR 12.8.1.1.4	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	USAR 12.8.1.1.9	a) Vertical orientation, self weight + loaded DSC + water in cavity: 220,000 lb enveloping	ANSI 57.9-1984
			b) Horizontal orientation, self weight + loaded DSC on transfer skid: 220,000 lb enveloping	ANSI 57.9-1984
	Snow and Ice Loads	USAR 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		Loaded DSC rejecting 21.12 kW decay heat. Ambient air temperature range -3°F to 103°F	ANSI 57.9-1984
	Shielding	USAR 12.7.1.2	Contact dose rate ≤ 200 mrem/hr.	ANSI 57.9-1984
Transfer Cask Upper Trunnions	Operational Handling	USAR 12.8.1.1.9	a) Upper lifting trunnions while in Auxiliary Building: i) Stress must be less than yield stress for 6 times critical load of 126,500 lb/trunnion nominal	ANSI N14.6-1978
		USAR 12.8.1.1.9	ii) Stress must be less than ultimate stress for 10 times critical load	

TABLE 12.3-3
NUHOMS-32P SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
		USAR Table 3.2-1	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	USAR 12.8.1.1.9	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	USAR 12.8.1.1.9	Hydraulic ram load due to friction of extracting loaded DSC: 20,000 lb	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

(ASME B&PV Code-1983, with Addenda up to 1985 for HSM and Transfer Cask)
(ASME B&PV Code-1998, with Addenda up to 1999 for DSC)

* Value shown for HSM. HSM-HB design parameters envelope the HSM.

TABLE 12.3-4
NUHOMS-32P SUMMARY OF DESIGN PARAMETERS FOR OFF-NORMAL OPERATING CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
HSM/HSM-HB	Off-Normal Temperature		-3°F and 103°F ambient temperature*	ANSI 57.9-1984
	Jammed Condition Handling	USAR 12.8.1.2.1	Hydraulic ram load equal to 80,000 lb	ANSI 57.9-1984
	Load Combination	USAR Table 3.2-2	Load combination methodology	ANSI 57.9-1984 Sec 6.17.1.1
DSC	Off-normal Temperature		-3°F and 103°F ambient temperature	ANSI 57.9-1984
	Off-normal Pressure		DSC internal pressure 10.8 psig	ANSI 57.9-1984
	Blowdown Pressure		DSC internal pressure: 40.0 psig	10 CFR 72.122(b)
	Jammed Condition Handling	USAR 12.8.1.2.1	Hydraulic ram load equal to 80,000 lb	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	USAR 12.8.1.2.1	Hydraulic ram load: 80,000 lb	ANSI 57.9-1984
	Load Combination	TR Table 8.2-11	Load combination methodology	ANSI 57.9-1984
Transfer Cask	Off-normal Temperature		-3°F and 103°F ambient temperature	ANSI 57.9-1984
	Jammed Condition Handling	USAR 12.8.1.2.1	Hydraulic ram load: 80,000 lb	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
	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

* HSM-HB design parameters envelope the HSM.

TABLE 12.3-5
NUHOMS-32P SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
HSM	Design Basis Tornado	USAR 3.2.1	Max velocity 360 mph Max wind pressure 304 psf	RG 1.76 ANSI 58.1 1982
	Load Combination	USAR Table 3.2-2	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	USAR 2.4.2	Dry Site	
	Seismic	USAR 12.3.2.3	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	USAR 12.8.2.7	HSM vents (inlet/outlet) blocked for 36 hrs or less. HSM inside surface temp: 387°F	ANSI 57.9-1984
	Fire	USAR 12.8.2.10	1 hour forest fire 65' from HSM	
	Explosions	USAR 8.2.11	Probability of liquefied natural gas spill affecting HSM < 10^{-7}	NUREG-0800 Section 2.2.3
HSM-HB	Design Basis Tornado	USAR 3.2.1	Max velocity 360 mph Max wind pressure 344 psf	RG 1.76 ASCE 7-95
	Load Combination	USAR Table 3.2-2	Load Combination Methodology	ANSI 57.9-1984
	Design Basis Tornado Missiles	USAR 12.3.2.1	Automobile, 4,000 lb, 195 fps 8" diam shell, 276 lb, 185 fps 12" steel pipe, 1500 lb, 205 fps 6" steel pipe, 285 lbs, 230 fps Wood plank, 200 lbs, 440 fps (300mph) 3" steel pipe, 115 lbs, 268 fps 1" steel rod, 8 lbs, 317 fps	NUREG-0800 Sec 3.5.1.4

TABLE 12.3-5
NUHOMS-32P SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
	Flood	USAR 12.3.2.2	Dry Site	
	Seismic	USAR 12.3.2.3	Horizontal ground acceleration 0.3g (both directions) Vertical ground acceleration 0.2g 7% critical damping	NRC RGs 1.60 and 1.61
	Accident Condition Temperature		HSM-HB vents (inlet/outlet) blocked for 36 hours or less. HSM-HB inside surface temp: 333°F	ANSI 57.9-1984
	Fire	USAR 12.8.2.10	1 hour forest fire 65' from HSM-HB	
	Explosions	USAR 8.2.11	Probability of liquefied natural gas spill affecting HSM-HB $< 10^{-7}$	NUREG-0800 Section 2.2.3
DSC	Accident Drop	USAR 12.8.2.5	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	USAR 12.8.2.3.2	Horizontal acceleration: 1.5g Vertical acceleration: 1.0g 3% critical damping	NRC RGs 1.60 and 1.61
	Accident Internal Pressure (HSM/HSM-HB vents blocked)	USAR 12.8.2.7	DSC internal pressure: 100 psig based on 100% fuel clad rupture and fill gas release, and ambient air temp. = 103°F. DSC shell temperature: 571°F (HSM) 473°F(HSM-HB) Blocked vent time = 36 hrs	10 CFR 72.122(b)

TABLE 12.3-5
NUHOMS-32P SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
	Accident Conditions	TR Table 3.2-6	Service Level D	ASME B&PV Code Sec III, Div 1, NB, Class 1
	Reflood Pressure	USAR 12.8.2	DSC internal pressure: 40.0 psig	10 CFR 72.122(i)
DSC Support Assembly	Seismic	USAR 12.8.2.3.2	DSC reaction loads: Horizontal acceleration: 0.61g Vertical acceleration: 0.39g 7% critical damping	NRC RGs 1.60 and 1.61
	Load Combination	USAR Table 12.8-4	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	USAR 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 12.3-5
NUHOMS-32P SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

COMPONENT	DESIGN LOAD TYPE	REFERENCE	DESIGN PARAMETERS	APPLICABLE CODE
	Accident Drop	USAR 12.8.2.5	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.

TABLE 12.3-6
NUHOMS-32P 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	3	4	5
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	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		X		
Internal Pressure	Normal Operating	P_n			X		X				X	X	X		X	X					
	Hydrostatic	P_h		X																	
	Off-normal (blowdown)	P_b				X		X	X	X											
	Accident (inner boundary)	P_{a1}																	X	X	X
	Accident (outer boundary)	P_{a2}															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 B&PV Code Service Level			A	A	A	A	B	B	B	B	C	C	C	C	C	C	D	D	D	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	D_3	D_4	D_5

(1) The Table has been modified to include hydrostatic and blowdown pressure 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 12.3-6
NUHOMS-32P 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	3	4	5
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	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		X		
Internal Pressure	Normal Operating	P_n			X		X				X	X	X		X	X					
	Hydrostatic	P_h		X																	
	Off-normal (blowdown)	P_b				X		X	X	X											
	Accident (inner boundary)	P_{a1}																	X	X	X
	Accident (outer boundary)	P_{a2}															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 B&PV Code Service Level			A	A	A	A	B	B	B	B	C	C	C	C	C	C	D	D	D	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	D_3	D_4	D_5

(1) The Table has been modified to include hydrostatic and blowdown pressure 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.

12.4 INSTALLATION DESIGN

The discussion presented in Chapter 4 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System. Chapter 4 describes the installation design associated with the Calvert Cliffs ISFSI and related systems. The narrative describes the installation design unique to the NUHOMS systems, such as the storage structures, auxiliary systems, decontamination systems, transfer cask repair and maintenance, and the fuel handling operation systems. The Calvert Cliffs ISFSI is a self-contained, passive storage facility, which requires no auxiliary systems.

12.5 OPERATION SYSTEMS

The discussion presented in Chapter 5 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System. Chapter 5 describes the operation of the Calvert Cliffs ISFSI. The narrative describes operations unique to the NUHOMS systems, such as draining, drying, and closure of the DSC. Although some operational details are provided, the description is not intended to limit or restrict operation of the facility. Operational procedures may be revised according to the requirements of the plant, provided that the limiting conditions of operation are not exceeded.

12.6 SITE GENERATED WASTE CONFINEMENT AND MANAGEMENT

The discussion presented in Chapter 6 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System. Chapter 6 describes the on-site waste sources, off-gas treatment and ventilation, liquid waste treatment and retention, solid wastes and radiological impact of normal operations of the Calvert Cliffs ISFSI.

12.7 RADIATION PROTECTION

The NUHOMS-32P DSC provides enhanced shielding which helps to compensate for the additional spent fuel elements. Dose rates for the NUHOMS-32P DSC are presented in Table 12.7-1. The Calvert Cliffs site-specific NUHOMS-32P DSC design meets the requirements of 10 CFR 72.104 and 10 CFR 72.106 for normal, off-normal, and accident conditions.

The principal design features of the NUHOMS-32P DSC are listed in Table 1.3-1 and shown in Figure 12.3-1. Tables 1.2-1 and 12.3-1 lists the capacity, dimensions, and design parameters for the NUHOMS-32P DSC.

The differences between the NUHOMS-32P DSC and NUHOMS-24P DSC that affect shielding and radiation protection are:

- an increase from 24 to 32 spent fuel assemblies,
- an addition of full-length aluminum plates (borated and non-borated) between the guide sleeves,
- an addition of full-length stainless steel rails and aluminum rail inserts between the DSC stainless steel cylindrical shell and the outside guide sleeves,
- a redesign of the top shield plug (including vent and siphon ports), and
- increasing the assembly neutron source term from 2.23E+08 n/sec/assy to 4.175E+08 n/sec/assy (Reference 12.47).

The radiation protection and shielding aspects of the NUHOMS-32P DSC, and the effects of these differences, are addressed in detail below.

12.7.1 ENSURING THAT THE OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

12.7.1.1 Policy Considerations

The discussion in Section 7.1.1 is unchanged and equally applicable to the NUHOMS-32P DSC.

12.7.1.2 Design Considerations – NUHOMS-32P DSC

NUHOMS-32P DSCs can store 32 spent fuel assemblies as opposed to NUHOMS-24P DSCs that store 24. Thus the radiation source is significantly increased in the NUHOMS-32P DSC. The design considerations which ensure that occupational exposures for the NUHOMS ISFSI are as low as reasonably achievable, are discussed in Section 7.1.2. The following paragraphs, which are numbered to correspond with Section 7.1.2, discuss differences in the NUHOMS-24P and NUHOMS-32P designs which affect the shielding design considerations.

1-7. Same as Section 7.1.2.

8. Section 7.1.2 states that the cavity of the NUHOMS-24P DSC will be submerged in the spent fuel pool for about 12 hours and, on removal from the pool, will contain borated water from the spent fuel pool for less than 50 hours. Because of the larger number of fuel assemblies (32 opposed to 24) to be stored in the NUHOMS-32P DSC, it is expected that the DSC will be submerged in the pool for a longer

period of time than the NUHOMS-24P DSC. This additional submersion time does not affect the performance of the austenitic stainless steel as discussed in Section 7.1.2. The NUHOMS-32P DSC also contains aluminum plates (borated and un-borated). There is a substantial body of industry experience with exposure of aluminum to borated and unborated water and its performance is not affected by the pool conditions.

9-15. Same as Section 7.1.2.

12.7.1.3 Operational Considerations

The discussion in Section 7.1.3 is unchanged and equally applicable to the NUHOMS-32P DSC.

12.7.2 RADIATION SOURCES – NUHOMS-32P

12.7.2.1 Characterization of Sources

The neutron and gamma source-terms used to analyze the NUHOMS-32P DSC are based on the following spent fuel assembly:

1. The bounding neutron source was based on the spent fuel assembly with an initial enrichment of 3.4% U^{235}, a 52,000 MWD/MTU discharge burnup, and a cooling time of 16 years;

2. The bounding gamma source was based on the spent fuel assembly with an enrichment of 4.3% U^{235}, a burnup of 40,000 MWD/MTU, and a cooling time of 7 years.

The design basis neutron source per fuel assembly that is used to calculate dose rates from the NUHOMS-32P DSC is increased from 2.23E+08 n/sec/assy to 4.175E+08 n/sec/assy. This increase accounts for the fact that some Calvert Cliffs spent fuel assemblies with a thermal output $\leq$ 660 watts have a neutron source $\geq$ 2.23E+08 n/sec/assy. This NUHOMS-32P DSC design basis source, 4.175E+08 n/sec/assy, bounds all assemblies that have an initial U^{235} enrichment $\leq$ 4.5%, burnup $\leq$ 52,000 MWD/MTU, and a thermal output $\leq$ 660 W. The neutron source-term is further increased by a factor 1.107 to account for axial peaking.

The design basis gamma source per fuel assembly for the NUHOMS-32P DSC is 1.61E+15 Mev/sec/assembly that is slightly higher than the source used for the NUHOMS-24P DSC that is 1.53E+15 Mev/sec/assembly.

The neutron and gamma source spectra calculated using the SAS2H sequence of the SCALE 4.4 computer code (Reference 12.63) are provided in Table 3.1.4. SAS2H is a control module that implements an analytic sequence of which the functional modules are BONAMI-S, NITAWL-II, XSDRNPM-S, COUPLE, and ORIGEN-S. ORIGEN-S, which is essentially an updated version of ORIGEN2.1, performs point depletion calculations to provide updated isotopic concentrations. Following the SAS2H depletion, a decay-only ORIGEN-S run is utilized to obtain the thermal and radiological assembly source terms and spectra in the desired energy group structure at 60 different times following shutdown. The SAS2H/ORIGEN-S code system was verified, validated and used in accordance with the CCNPP software qualification program.

The addition of stainless steel rods to the fuel assembly increases Co-60 activity due to impurities resulting in a harder spectrum and the need for extended cooling times. Details of the changed gamma source are presented in Reference 12.58. The additional loading criteria for assemblies with irradiated stainless steel replacement rods are presented in Table 9.4-3.

The ORIGEN-ARP sequence of the SCALE 6.0 code system (Reference 12.76) may also be used in place of the SAS2H sequence of SCALE 4.4 for NUHOMS-32P DSC fuel characterization. ORIGEN-ARP (Automatic Rapid Processing) uses an algorithm that allows the generation of cross-section libraries for the ORIGEN-S code by interpolation over cross-section libraries pre-generated by the SAS2H or TRITON sequences of SCALE. To ensure that the level of conservatism is essentially the same or greater than that of the SAS2H sequence of SCALE 4.4, the calculated minimum cooling times to reach the NUHOMS-32P design basis neutron, gamma, and thermal source terms must be biased upwards by 3.1% when using SCALE 6 ORIGEN-ARP with the included CE 14x14 cross-section libraries (Reference 12.75).

12.7.2.2 Airborne Radioactive Material Sources

The discussion in Section 7.2.2 is unchanged and equally applicable to the NUHOMS-24P DSC and NUHOMS-32P DSC.

12.7.3 RADIATION PROTECTION DESIGN FEATURES – NUHOMS-32P

12.7.3.1 Installation Design Features

The discussion in Section 7.3.1 is applicable to the NUHOMS-32P DSC.

12.7.3.2 Shielding

The shielding analyses that support the NUHOMS-32P DSC are identical in form and methodology to the design basis analyses that support the NUHOMS-24P DSC. The methodology and model are described in detail in References 12.29 through 12.33. A comparison to actual NUHOMS-24P DSC data demonstrates that the methodology and model are conservative (Reference 12.33).

The results of the shielding analyses are presented in Table 12.7-1. In most cases, the calculated dose rates outside the NUHOMS-32P DSC are higher than the NUHOMS-24P DSC due to eight additional spent fuel assemblies and increased neutron source-term. The shielding provided by the NUHOMS-32P basket assembly helps to compensate for the increased radiation from the additional fuel assemblies.

12.7.3.3 Ventilation

The discussion in Section 7.3.3 is applicable to the NUHOMS-32P DSC.

12.7.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The discussion in Section 7.3.4 is applicable to the NUHOMS-32P DSC.

12.7.4 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

12.7.4.1 Operational Exposure

The discussion of Section 7.4.1 is applicable to NUHOMS-32P DSC.

12.7.4.2 Storage Term Exposure

The discussion of Section 7.4.2 is applicable to NUHOMS-32P DSC.

12.7.5 HEALTH PHYSICS PROGRAM

The discussion in Section 7.5 is unchanged and equally applicable to the NUHOMS-24P DSC and NUHOMS-32P DSC.

12.7.6 ESTIMATED OFF-SITE COLLECTIVE DOSE ASSESSMENT

12.7.6.1 Effluent and Environmental Monitoring Program

The discussion in Section 7.6.1 is applicable to the NUHOMS-32P DSC.

12.7.6.2 Analysis of Multiple Contribution

The discussion in Section 7.6.2 is applicable to the NUHOMS-32P DSC.

12.7.6.3 Estimated Dose Equivalents

The discussion in Section 7.6.3 is applicable to the NUHOMS-32P DSC.

12.7.6.4 Liquid Release

The discussion in Section 7.6.4 is applicable to the NUHOMS-32P DSC.

TABLE 12.7-1
NUHOMS-32P SHIELDING ANALYSIS RESULTS - NOMINAL DOSE RATES (mrem/hr)

LOCATION	NEUTRON[c]	GAMMA[c] (PRI + SEC)	TOTAL[c]
NUHOMS-32P DSC in HSM			
1. HSM Wall or Roof	0.5	13	13.5
2. HSM Air Outlet	1.3	74.1	75.4
3. Center of Door	5.9	8	13.9
4. Doorway (Maximum, 1 ft. into opening)	1247	3412	4659
5. Air Inlet Vent	1	60	61
6. 1m from HSM Door	3	5.9	8.9
NUHOMS-32P DSC in HSM-HB			
1. HSM Wall or Roof	0.7	4.7	5.4
2. HSM Air Outlet	6.2	41.9	48.1
3. Center of Door	0.2	0.4	0.6
4. Doorway (Maximum, 1 ft. into opening)	668	3206	3874
5. Air Inlet Vent	3.7	85.2	88.9
6. 1m from HSM Door	0.2	0.5	0.7
NUHOMS-32P DSC in Cask			
1. Centerline DSC Shield Plug (Flooded DSC)[a]	1.4	178.2	180
2. DSC Cover Plate (Dry DSC)			
2.1 Center	58	148	206
2.2A Edge[b] (Wet Gap)	112	114	226
2.2B Edge[b] (Dry Gap)	158	192	350
3. Transfer Cask			
3.1 Side	90	74	164
3.2 Top	8	4.8	12.8
3.3 Bottom	110	105	215

(a) The DSC/cask annular gap is filled with water. All but the top 6" of the DSC inner cavity is filled with water.

(b) Nominal at edge of cover plate. The total dose rate is approximately a factor of 3 lower at the top edge of the transfer cask, and several times higher inside the dry annulus.

(c) From References 12.29, 12.30, and 12.31.

12.8 ACCIDENT ANALYSIS – NUHOMS-32P

Analyses of all design events for the NUHOMS-24P DSC have been reanalyzed for the NUHOMS-32P DSC. The results are reported in this section in the same format as in Chapter 8. The analytical assumptions, methodology, and computer codes used to generate the results in this section are identical to those used in Chapter 8 unless otherwise noted in the text.

12.8.1 NORMAL AND OFF-NORMAL OPERATIONS

This section follows the format of Section 8.1 and includes the evaluation of the normal and off-normal events for the NUHOMS-32P DSC.

12.8.1.1 Normal Operation Structural Analysis

The normal operating loads for the NUHOMS-32P important to safety components are shown in Table 8.1-1 of Reference 12.21. A comprehensive structural analysis of the NUHOMS-32P DSC was performed and documented in Reference 12.22.

12.8.1.1.1 Normal Operation Structural Analysis

The loads applicable to the normal operation structural analysis such as the dead weight load, internal pressure load, thermal load, handling loads, and live loads are calculated on the same bases as described in detail in Section 8.1.1.1.

12.8.1.1.2 DSC Analysis

Stresses were evaluated in the DSC due to:

A. Dead Weight Loads
B. Design Basis Normal Operating Internal Pressure Loads
C. Normal Operating Thermal Loads
D. Normal Operation Handling Loads

The NUHOMS-32P DSC is analyzed using analytical methods comparable to those described for the NUHOMS-24P DSC in Section 8.1.1.2. The ANSYS analytical model used for the analysis of dead weight, pressure, thermal, and handling loads is similar to that used in the evaluation of the NUHOMS-24P DSC, modified as necessary to reflect the NUHOMS-32P shell assembly. The ANSYS analysis model is described in Reference 12.22. Stresses due to normal operating pressures are based on a bounding internal pressure of 30 psig, applied as a uniform load to the inner boundary of the analytical model. Also considered was the external hydrostatic pressure loading on the DSC shell, when the 3/8" annulus between the DSC shell and the transfer cask is filled with water. Circumferential shell temperature variations are analyzed using the ANSYS three-dimensional solid shell model. The NUHOMS-32P DSC stresses remain within ASME code allowable stresses (Reference 12.24).

12.8.1.1.3 DSC Internal Basket Analysis

The DSC basket analysis was performed for:

A. Dead Weight Loads
B. Thermal Loads

The fuel assembly weight of 1,450 lbs (Reference 12.34) and 158" length (Reference 12.28) are used in the analysis. The basket temperature is taken as 650°F uniform. The peripheral rail temperature is taken as 500°F uniform.

The three-dimensional finite element analysis ANSYS model used in the evaluation is described in Reference 12.28. The analysis model consists of a 10.28" slice of the NUHOMS-32P DSC basket, rails, and canister using SHELL 43 elements, with appropriate boundary conditions applied at the cut faces of the model. The fuel assembly and aluminum plates are not included in the analysis model. The fuel assembly weight is applied as pressure on the basket plates. The weight of the aluminum plates is accounted for by increasing the density of stainless steel tubes. The aluminum plate stiffness is conservatively neglected in the analysis. The fusion welds connecting the fuels compartment guide sleeves and the bolts connecting the rails are modeled by three-dimensional PIPE20 elements. Gap elements (CONTACT 52) are used to simulate the interface between the basket rails and inner side of canister as well as between outer side of canister and inside of transfer cask.

The results of the analysis show that the stresses in the DSC are within the allowable stress limits (Reference 12.27).

12.8.1.1.4 DSC Support Assembly Analysis

The Calvert Cliffs HSM DSC support assembly components are the same as described in Section 8.1.1.4.

The HSM DSC support assembly was reevaluated for the following loads by scaling the results of the NUHOMS-24P DSC evaluation to reflect the weight increase of the NUHOMS-32P DSC. The allowable stresses are taken at a bounding temperature of 600°F for all conditions, including normal operation (Reference 12.23).

A. Dead Weight Loads
B. Normal Operational Handling Loads
C. Thermal Loads

The calculated stresses were small, and the vertical deflections under the transfer cask loading were less than 0.1".

The HSM-HB support structure consists of two rail assemblies, each at 30 degrees from the vertical center line of the DSC. Two cross members connect the two rail assemblies by two gusset plates welded to the rail web and the flanges. The steel support structure supports the DSC stored inside the module. Each rail assembly of the DSC support structure consists of the following components:

1. A 12x96 Rail Section 187" long made up of ASTM A992 material.
2. A 1" thick slotted plate made up of A572 Grade 50 material.
3. A 3/16" thick support plate made up of nitronic 60 (RC 29-35) material which provides a smooth support for the DSC to slide.
4. A rail extension flange which consists of 1" thick flange plate (A572, Grade 50 material), and 3/16" thick rail support extension plate (nitronic 60 material).

The rail extension flange is attached to a 1" thick embedded base plate (A36 material) by two 1-1/2" bolts. For all normal operating conditions the DSC support steel temperature is assumed to be 300°F (556°F for accident blocked vent condition).

The HSM-HB DSC support was evaluated for the same types of loads as listed for the HSM above. The calculated stresses were small and meet all code allowables. (Reference 12.72)

12.8.1.1.5 HSM/HSM-HB Analysis

The same HSM array size (2x6) used in Section 8.1.1.5 was used for the NUHOMS-32P DSC. The HSM-HB arrangement used is that of two 1x12 back-to-back arrays. A DSC mass enveloping that of the NUHOMS-32P DSC is used for HSM-HB. The following loads are considered in the structural analysis for normal operation loads.

A. HSM/HSM-HB Dead and Live Loads

The dead and live loads were evaluated using the ANSYS methodology as discussed in Reference 12.23 (HSM) and Reference 12.72 (HSM-HB) for the NUHOMS-32P DSC.

B. Concrete Creep and Shrinkage Loads

Loads due to creep and shrinkage of the concrete are determined by the same methodology described in Section 8.1.1.5.B.

C. HSM Thermal Loads

The thermal loads, temperature-dependent material properties, analysis approach, and analysis results are documented in Reference 12.23 (HSM) and Reference 12.73 (HSM-HB).

Conservatively, an enveloping design temperature of 400°F is used for all conditions for HSM, including normal operation (Reference 12.23).

D. Radiation Effect on HSM Concrete

The effects of radiation on the HSM concrete were determined to be negligible for the NUHOMS-24P DSC in Section 8.1.1.5.D. The effect of radiation on the HSM/HSM-HB concrete remains negligible for the NUHOMS-32P DSC. The neutron fluence from the NUHOMS-32P DSC remains below the threshold for neutron induced degradation of concrete and the gamma flux is less than the NUHOMS-24P DSC value.

E. HSM/HSM-HB Design Analysis

Structural re-evaluation of the concrete structure, DSC support structure, and miscellaneous components for the effects of the increased weight and thermal load of the NUHOMS-32P DSC are documented in Reference 12.23 (HSM) and Reference 12.72 (HSM-HB).

For the HSM the effects of the weight increase are addressed either by scaling the existing stress results of the NUHOMS-24P DSC analyses or by reanalysis of the affected components.

The thermal load evaluation of the HSM concrete structure is performed with the ANSYS computer program using a representative 2-D analytical model of the HSM.

The thermal load evaluation of the HSM-HB concrete structure is performed with the ANSYS computer program using a half-symmetric 3-D finite element model of the HSM-HB containing the NUHOMS-32P DSC.

The results of the evaluation confirm that the normal operation moment and shear in the HSM concrete structure are less than the ultimate moment and shear capacity shown in Table 8.1-8 of Chapter 8.

12.8.1.1.6 HSM Door Analysis

The discussion in Section 8.1.1.6 is applicable to the NUHOMS-32P DSC.

The HSM-HB front shield door is a composite door, which consists of a rectangular steel face plate at the front attached to a circular reinforced concrete block at the rear. The circular concrete block is inset into the HSM-HB doorway and the rectangular steel face plate of the door is attached to the front wall concrete using four bolts anchored through four embedments.

The shield door is free to grow in the radial direction when subjected to thermal loads. Therefore, there will be no stresses in the door due to thermal growth. The dead weight, tornado wind, differential pressure, and flood loads cause insignificant stresses in the door compared to stresses due to missile impact load. Therefore, the door is evaluated only for the missile impact load. The computed maximum ductility ratio for the door is less than 1 (compared to the allowable ductility of 20) (Reference 12.72, Section 10.2.5 Part A).

For the door anchorage, the controlling load is tornado generated differential pressure drop load. The maximum tensile force per bolt (there are four bolts that attach the door assembly to the front concrete wall of the HSM-HB) is 4.5 kips. This is less than the allowable load per bolt of 44.3 kips (Reference 12.72, Section C6.2). The concrete pull-out strength is conservatively estimated as 24 kips which is greater than the ultimate capacity of the four bolts, thus satisfying the ductility requirements of the ACI Code.

12.8.1.1.7 Heat Shield Analysis

The discussion in Section 8.1.1.7 is applicable to the NUHOMS-32P DSC in the HSM. Similar to the HSM, the HSM-HB provides flat stain less steel heat shields on the side walls of the base unit and under the roof.

The HSM-HB top heat shield consists of two heat shield panels. Each panel has a 12 gauge 304 stainless steel sheet which is 0.1054" thick. Both the panels of the roof heat shield are suspended from the roof by fifteen rods of ½" diameter ASTM A193, Grade B7 in three rows, bolted to the sheets. The combined axial and bending stress in the rods is 59.5 ksi. The allowable stress is 70.2 ksi (Reference 12.72, B5.1). The HSM-HB side heat shield consists of four 12 gauge 304 stainless steel supported off the base unit side wall by thirty four rod stand-offs threaded into concrete embedments. The maximum axial and bending stress in the rods is about 1.4 ksi and 79.3 ksi, respectively. The axial and bending stress allowable for the rods is 67.9 ksi and 112.3 ksi, respectively

(Reference 12.72, B5.2). The maximum temperature used in the stress analysis of the heat shields is 270°F (Reference 12.72, B3.1), which bounds the temperatures determined for storage of the NUHOMS-32P DSC for normal and off-normal conditions (Reference 12.73).

12.8.1.1.8 HSM/HSM-HB Seismic Restraint for DSC

The HSM seismic restraint is described in Section 8.1.1.8. The HSM-HB seismic restraint consists of a tube steel embedment located within the bottom center of the round access opening of the HSM-HB, and a tube steel retainer assembly that drops into the embedment cavity after the NUHOMS-32P DSC transfer is complete. The drop-in retainer extends approximately 4 inches above the rail to provide axial restraint of the NUHOMS-32P DSC. Details of the analysis of the Calvert Cliffs DSC seismic restraint is provided in Reference 12.23 (HSM) and Reference 12.72 (HSM-HB).

12.8.1.1.9 Transfer Cask Analysis

The same transfer cask is used to transport the NUHOMS-24P and NUHOMS-32P DSCs and was re-evaluated for the same normal operation loads identified in Section 8.1.1.9, but considering the increased weight of the NUHOMS-32P DSC payload. A bounding design temperature of 400°F is used for both normal and off-normal operating conditions.

Re-evaluation of the transfer cask for the increased weight of the NUHOMS-32P DSC payload, based on the stress results of the ANSYS analyses of the transfer cask, is presented in Reference 12.25.

The resulting maximum dead weight, thermal, and handling stresses in the transfer cask and its components are within the allowable stress limits.

12.8.1.2 <u>Off-Normal Load Structural Analysis</u>

The off-normal loads for the NUHOMS-32P DSC are the same as those identified in Section 8.1.2.

12.8.1.2.1 Jammed DSC During Transfer

This off-normal condition results from the DSC becoming jammed in the transfer cask or the HSM/HSM-HB during the transfer operation.

A. Postulated Cause of Jammed DSC

The discussion in Section 8.1.2.1.A is applicable to the NUHOMS-32P DSC.

B. Detection of Jammed NUHOMS-32P DSC

The discussion in Section 8.1.2.1.B is applicable to the NUHOMS-32P DSC.

C. Analysis of Effects and Consequences

The analyses of the NUHOMS-32P DSC under the assumed jamming and binding conditions are documented in Reference 12.22.

The stresses on the NUHOMS-32P DSC body have been analyzed for a maximum ram force of 95,000 lbf. The calculated stresses were much less than the ASME code allowable stress criteria. Therefore, plastic deformation of the NUHOMS-32P DSC body will not occur and there is no potential for rupture.

D. Corrective Actions

The courses of action open to the system operators to correct a jammed NUHOMS-32P DSC are described in Section 8.1.2.1.D.

12.8.1.2.2 Off-Normal Thermal Loads Analysis

Structural analyses of the ISFSI components for off-normal thermal loads are discussed below using the same temperature extremes of -3°F and 103°F as the NUHOMS-24P design.

A. HSM/HSM-HB Off-Normal Thermal Analysis

The methodology used for the off-normal thermal loads structural analysis of the HSM/HSM-HB concrete structure storing the loaded NUHOMS-32P DSC is the same as for the normal thermal loads structural analysis of the structure described in Section 12.8.1.1.5.C. The DSC support assembly is designed with slotted holes as described in Section 8.1.2.2.A and, therefore, the increase in temperature has no effect on the DSC support structure.

B. DSC Off-Normal Thermal Analysis

Off-normal thermal loads structural analysis of the NUHOMS-32P DSC shell assembly and the DSC fuel basket assembly for the DSC inside the HSM/ HSM-HB are performed using the same methodology as for the normal thermal loads structural analyses of these components.

As discussed in Section 12.8.1.1.9, the off-normal thermal loads for the transfer cask are identical to the normal thermal loads. Therefore, the off-normal thermal loads for the DSC inside the transfer cask are identical to the normal thermal loads for the DSC inside the transfer cask, and are not considered further.

C. Transfer Cask Off-Normal Thermal Analysis

As previously stated, the off-normal thermal loads for the transfer cask are identical to the normal thermal loads. Therefore, the off-normal thermal loads for the transfer cask are not considered further.

12.8.1.3 Thermal Hydraulic Analyses

The following evaluations have been performed for the Calvert Cliffs ISFSI:

A. Thermal Analysis of the HSM
B. Thermal Analysis of the DSC in the HSM
C. Thermal Analysis of the DSC in the Transfer Cask

The analytical models of the HSM, the NUHOMS-32P DSC, and the transfer cask are described in References 12.11 (HSM), Reference 12.12 (DSC in the HSM), and 12.10 (DSC in the transfer cask). The analytical models of the NUHOMS-32P DSC in the HSM-HB are described in References 12.73 and 12.74.

The method described in References 12.13 and 12.18 is used for calculating the effective thermal conductivity of the spent fuel assemblies.

The primary portion of the thermal evaluation of the NUHOMS-32P design uses a methodology that differs from the thermal analysis methodology utilized for the NUHOMS-24P. The NUHOMS-32P thermal methodology has three major new features:

- Solution Method – ANSYS finite element
- Model Geometry – 3D
- Treatment of Effective Transverse Thermal Conductivity of Fuel – A detailed finite element model of the spent fuel according to method of the TRW Spent Nuclear Fuel Effective Conductivity Report (Reference 12.14)

The new methodology for the NUHOMS-32P DSC in the HSM and transfer cask has been compared in detail with the methodology used for the thermal analysis of the Transnuclear NUHOMS-32PT design (Reference 12.54). The use of the NUHOMS-32PT methodology is appropriate for the Calvert Cliffs ISFSI with the NUHOMS-32P DSC per Reference 12.55.

The radial effective thermal conductivity for helium backfill conditions is determined by creating a two-dimensional finite element ANSYS model of the fuel assembly centered within a basket compartment. The outer surfaces, representing the fuel compartment walls, are held at a constant temperature, and decay heat is applied to the fuel pellets within the model. A maximum fuel assembly temperature is then determined. From the heat load, maximum fuel temperature, and outer surface temperature, the effective fuel conductivity can be determined via an equation given in Reference 12.14.

The axial effective conductivity (K_{axl}) is determined directly from the geometry and conductivities of the fuel components.

Mass-weighted averages are used in the determination of the effective density (ρ_{eff}) and specific heat values ($C_{p,eff}$).

The effective properties of the fuel are shown in Reference 12.13.

The HSM-HB with NUHOMS-32P DSC thermal analyses are performed using methods, assumptions, and conservatism identical to those for the NUHOMS-24PTH DSC design in the HSM-H (Reference 12.77). The use of the NUHOMS-24PTH/HSM-H methodology is appropriate for the Calvert Cliffs ISFSI with the NUHOMS-32P/HSM-HB per Reference 12.78.

The thermal analyses are performed with the following ambient air temperatures:

A. Normal Conditions

The discussion in Section 8.1.3.A is applicable to the NUHOMS-32P DSC.

B. Off-Normal Condition

The discussion in Section 8.1.3.B is applicable to the NUHOMS-32P DSC.

C. Accident Condition

An extreme summer condition with an ambient temperature of 103°F was considered in Reference 12.2. In addition, the HSM vents are assumed to be completely blocked for a period of 36 hours or less. A solar heat flux of 127 Btu/hr-ft^2 is conservatively included to maximize the HSM concrete temperatures.

No thermal accident conditions are considered for the transfer cask.

12.8.1.3.1 Thermal Analysis of the HSM

The HSM/HSM-HB thermal analyses are performed for the ambient air temperatures defined in Section 12.8.1.3.

The decay heat load is transferred from the DSC to the HSM/HSM-HB air space by convection and then is removed from the HSM/HSM-HB by natural convection air flow. Heat is also radiated from the DSC surface to the heat shield and HSM/HSM-HB walls where the natural convection air flow and conduction through the walls removes the heat. The solar heat flux is applied only to the HSM/HSM-HB roof. Heat transfer from the outer surface of the HSM/HSM-HB roof is by natural convection and radiation to the ambient air. Heat transfer from the HSM/HSM-HB floor slab is by conduction to the soil below.

Maximum temperatures on the DSC outer surfaces and the concrete inner and outer surfaces are calculated for the normal, off-normal winter, and off-normal summer ambient conditions, and the postulated accident conditions with blocked HSM/HSM-HB vents.

12.8.1.3.2 Thermal Analysis of the DSC in the HSM

The DSC and fuel assembly heat transfer analysis with the DSC inside the HSM/HSM-HB was performed for the ambient air temperatures defined in Section 12.8.1.3. The analytical model is described in Reference 12.12 for HSM and Reference 12.73 for HSM-HB. The cases of interest are those that maximize fuel cladding temperature (summer ambient conditions) as described in References 12.1 and 12.2. Reference 12.9 evaluates the impact of aluminum/ poison material plates in the DSC basket assembly on the maximum component temperatures. In the analysis, an effective conductivity is determined for the aluminum/poison plates and is substituted for that of the all aluminum interior basket plates in the model from Reference 12.12. The modified model is run for normal, off-normal, and blocked vent conditions. The analysis shows that the modified plates have a negligible impact on maximum component temperatures. The temperatures from these cases are used to derive the DSC internal pressures in Reference 12.17.

Reference 12.79 demonstrates that the effective thermal conductivity of a metal matrix composite poison plate paired with an aluminum plate also meets or exceeds the effective thermal conductivity of the basket in the design basis model.

The maximum allowable cladding temperature for long-term storage is 335°C as discussed in Section 8.1.3.2.

The acceptable peak clad temperature limit for accident conditions for ISFSI storage is the same as the generic NUHOMS-24P DSC design (Reference 8.26). This limit is based on the empirical work presented in Reference 8.27. The peak fuel clad temperature limit (short-term) of 570°C (1,058°F) is specified in the off-normal DSC thermal calculation (Reference 12.1 for HSM and Reference 12.73 for HSM-HB).

12.8.1.3.3 Thermal Analysis of the DSC in the Transfer Cask

The thermal analyses for the cases with the DSC inside the transfer cask are performed for the ambient air temperatures defined in Section 12.8.1.3. The analyses are conducted using the model described in Reference 12.12.

Maximum fuel cladding and DSC shell temperatures are calculated for the normal ambient temperature of 70°F; abnormal summer ambient temperature of 103°F and the postulated accident conditions with blocked HSM vents.

In the models presented in Reference 12.10, the gap size between the inner shell of the transfer cask and the outer shell of the DSC is assumed uniform in all directions. Reference 12.19 performs a sensitivity analysis of gap symmetry between the DSC and transfer cask. In addition, Reference 12.20 investigates the sensitivity of maximum temperatures to axial gap size. For the asymmetries and gap sizes investigated, there is little impact on the maximum component temperatures.

Reference 12.3 evaluates the impact of aluminum/poison material plates in the basket on the maximum component temperatures. In the analysis, an effective conductivity is determined for the aluminum/poison plates and is substituted for that of the all aluminum interior basket plates in the models from References 12.15 and 12.16. The analysis shows that the modified plates have a negligible impact on maximum component temperatures.

This also remains true for paired aluminum/metal matrix composite poison plates (Reference 12.79).

Use of NS-3 for the transfer cask neutron shield is discussed in Section 8.1.3.3.

12.8.2 ACCIDENTS

This section addresses design events of the third and fourth types as defined by ANSI/American Nuclear Society 57.9-1984 (Reference 8.2), and other credible accidents consistent with 10 CFR Part 72 which could impact the safe operation of the Calvert Cliffs ISFSI. The postulated events identified in Section 8.2 of Reference 8.1 and addressed therein for the Calvert Cliffs ISFSI are:

A. Loss of Air Outlet Shielding
B. Tornado Winds/Tornado Missile
C. Earthquake
D. Flood
E. Transfer Cask Drop
F. Lightning
G. Blockage of Air Inlets and Outlets

H. DSC Leakage
I. Accidental Pressurization of DSC

In addition, two additional Calvert Cliffs site-specific accidents have been identified and addressed. These are:

A. Forest Fire
B. Liquified Natural Gas Plant or Pipeline Spill or Explosion

The accidents considered, and the associated components affected by each accident, are summarized in Chapter 8, Table 8.2-1.

In the following sections, each accident condition is evaluated for applicability to the Calvert Cliffs ISFSI. For each applicable condition the accident cause, structural, thermal, radiological consequences, and recovery measures required to mitigate the accident are discussed. Where appropriate, resulting accident condition stresses were combined with those of normal operating loads in accordance with the load combination definitions of Section 12.3.2.5. Load combination results for the HSM/HSM-HB, NUHOMS-32P DSC, and transfer cask are discussed in Section 12.8.2.12.

Reflood pressure is included as an ASME Service Level D activity but is not identified as an accident.

12.8.2.1 Loss of Air Outlet Shielding

The discussion provided in Section 8.2.1 is applicable to NUHOMS-32P DSC in the HSM. As with the HSM, this accident is also not credible for the HSM-HB. The air outlet vent concrete covers of the HSM-HB are also designed to remain in place and withstand all design events including the effects of tornado missiles (Reference 12.72, Section 12.3).

12.8.2.2 Tornado Winds/Tornado Missile

The discussion provided in Section 8.2.2 is applicable to NUHOMS-32P DSC. Structural analysis of the HSM-HB concrete structure, DSC supports, and miscellaneous structural steel components of the HSM-HB also confirm integrity when subjected to tornado winds and missiles (Reference 12.72).

12.8.2.3 Earthquake

12.8.2.3.1 Cause of Accident

As specified in Section 3.2.3, a Design Basis Earthquake with peak ground acceleration values of 0.15g horizontal and 0.10g vertical is postulated to occur at the Calvert Cliffs ISFSI.

12.8.2.3.2 Accident Analysis

Earthquake loads for the evaluation of the HSM concrete structure, DSC support assembly in the HSM, and the transfer cask are determined by scaling the results of the NUHOMS-24P DSC seismic analysis reflecting the weight increase (References 12.23 and 12.25).

The methodology used for structural evaluation of the HSM-HB is identical to the methodology used for the evaluation of HSM-H for 24PTH DSC, 61BTH DSC, and 32PTH1 DSC in CoC 1004 and 32PTH DSC in CoC 1030 (References 12.77 and 12.80). A bounding calculation assuming a DSC weight of 110,000 lbs is used in this analysis. In addition, the analysis performed for the HSM-HB uses seismic accelerations of 0.3g horizontally and 0.2g vertically, with a 7% critical damping factor, which are twice the Calvert Cliffs ISFSI design basis seismic requirement. The use of this methodology is appropriate for the Calvert Cliffs ISFSI with the NUHOMS-32P DSC in the HSM-HB per Reference 12.78.

The NUHOMS-32P DSC and the DSC basket assembly earthquake loads were determined by a computer analysis (Reference 12.22).

A. DSC Seismic Analysis

1. DSC Seismic Analysis

Inside the HSM the combined earthquake load of 1.5g transverse, 1.5g axial, and 1.0g vertical is applied to the finite element model depicting the horizontal orientation of the DSC in storage. In addition to the seismic loads, 1.0g vertical acceleration is added to account for the self-weight effects.

A three-dimensional finite element model of the basket, rails, and the DSC canister was constructed by using the ANSYS computer program. Since the seismic loading is non-symmetric, a 360° model is used. Details of the analysis model and boundary conditions used are described in Reference 12.28. The finite element model used in the seismic analysis is shown in Reference 12.22.

A nonlinear stress analysis is conducted for computing the elastic stresses in basket and canister shell models using ANSYS computer program. The nonlinearity of analysis results from the gap elements used in the analysis model. Details of the analysis are documented in Reference 12.22 for the DSC canister and Reference 12.28 for the DSC basket assembly.

The resulting maximum stresses in the DSC canister and in the DSC basket assembly remain within the specified ASME code allowable stress criteria.

2. DSC Seismic Stability Analysis

An evaluation for the potential of the NUHOMS-32P DSC to lift-off from the DSC support assembly rail during a seismic event is documented in Reference 12.22. The seismic loadings applied to the DSC that would cause instability are based on conservative rigid range seismic acceleration inputs to the HSM of 0.25g horizontal in both transverse and longitudinal directions, and 0.17g vertical. The stability analysis is based on showing that the overturning moment of the DSC on the HSM support structure is smaller than the restoring force moment due to gravity. The non-rigid body modes of the DSC do not contribute to overturning so that the use of the rigid body accelerations is appropriate.

The resultant horizontal acceleration used to calculate the overturning moment is based on the square root of sum of the square of 0.25g in both the transverse and longitudinal directions, equal to 0.35g. The vertical acceleration used to calculate the minimum restoring force is gravity less the vertical acceleration. The restoring moment is determined to be 1.37 times greater than the overturning moment (Reference 12.22). Therefore, the DSC canister is stable during a seismic accident.

Reference 12.72 (Section 5.9) demonstrates that the DSC also remains stable in the HSM-HB during a bounding seismic event. A margin of safety of 1.23 against DSC liftoff from the HSM-HB rails is determined.

B. HSM/HSM-HB Seismic Analysis

1. HSM/HSM-HB Seismic Stress Analysis

Seismic evaluation of the HSM concrete structure is performed using the same methodology as described in Section 8.2.3.2.B.1 for the NUHOMS-24P DSC, with scaled seismic forces applied to the STRUDL finite element analysis model to reflect the weight increase of the NUHOMS-32P DSC.

The resulting forces and moments in the HSM/ HSM-HB are found to be within the ultimate capacity (Reference 12.23 for HSM and Reference 12.72 for HSM-HB).

2. HSM/HSM-HB Seismic Stability Evaluation

There is no change in the dimensions of the HSM. As such, the seismic stability of the HSM is not affected by storing the NUHOMS-32P DSC.

Reference 12.72 demonstrates that overturning and sliding will not occur the HSM-HB during a bounding seismic event.

C. DSC Support Assembly Seismic Analysis

The NUHOMS-32P DSC support assembly geometry for the HSM and, therefore, the DSC support structure computer analysis model is the same as for the NUHOMS-24P DSC. The earthquake loads due to the effects of the weight increase of the NUHOMS-32P DSC are evaluated by scaling the NUHOMS-24P DSC seismic analysis results.

The stress evaluation of the HSM DSC support structure components is performed by hand calculations and the results are within acceptable limits (Reference 12.23).

Stress analyses for the HSM-HB support structure using a bounding DSC weight also demonstrates results within acceptable limits for a bounding seismic event (Reference 12.72, Section 9).

D. Transfer Cask Seismic Analysis

Seismic stresses for the transfer cask with a NUHOMS-32P DSC are determined by conservatively scaling the seismic analysis results of the transfer cask - NUHOMS-24P DSC assembly to account for the increased payload of the NUHOMS-32P (Reference 12.25). This conservative method assumes the weight of the entire transfer cask-DSC system increases by the DSC weight change of 36%, instead of the more accurate 10%. This is done because some components, such as the inner liner, are loaded primarily by the DSC and thus the stresses would be underestimated using the 10% assembly weight increase. The resulting stress intensities are significantly below the ASME Level C allowable stress limits at the transfer cask design temperature of 400°F.

Seismic stability of the transfer cask is not a function of the DSC weight and, therefore, remains unaffected by the use of the NUHOMS-32P DSC.

12.8.2.3.3 Accident Dose Consequences

Major components of the Calvert Cliffs ISFSI have been designed and evaluated to withstand the forces generated by the Design Basis Earthquake. Hence, there are no dose consequences.

12.8.2.4 Flood

As discussed in Section 3.2.2, flood loads are not applicable to the Calvert Cliffs ISFSI.

12.8.2.5 Cask Drop

This section addresses the structural integrity of the transfer cask, the NUHOMS-32P DSC, and its internals under a postulated transfer cask accident condition.

12.8.2.5.1 Cause of Accident

As discussed in Section 8.2.5.1 of Reference 12.21, an actual drop event is not considered credible. However, consistent with the criteria of Reference 12.21, it is postulated that the transfer cask with the DSC inside will be subjected to an end, side, or oblique drop with a maximum height of 80" onto a thick concrete slab. A drop of greater than 80" is not considered because (a) transfer inside the Auxiliary Building will be performed using a single-failure-proof crane and (b) the transfer trailer and haul road are designed such that the transfer cask cannot be raised greater than 80" from the ground.

The transfer cask is transported along an asphalt or concrete paved road which is 16' wide and has 7 to 8' shoulders. The road is approximately 3,300 linear feet with slopes which range from 0% to 3% except for an approximate 50' length which carries a 5.7% slope. The roadbed is level except for a negligible 1% slope required to create a crown in the road for drainage and a transverse slope at any point along the transportation route of less than 10%. The shoulders are either level with the road or slope up from the road. In those locations where the paved road abuts up to existing blacktop, or concrete paving, the shoulder is discontinued. The shoulder may be paved, gravel, or soil and contain typical roadside fixtures, including curbs, fences, guard rails, and light poles which do not constitute potential puncture devices for the cask during a drop. The shoulders do not contain items such as light pole pedestals which protrude above the shoulder surface and could represent a potential cask puncture device during a cask drop. For the entire route that the transfer cask is transported there will exist a minimum 8' wide zone that is at or above the roadbed elevation.

The transfer trailer braking system is not operable independent of the prime mover. However, failure of the prime mover will cause the trailer braking system to fail-safe, that is "lock tight."

12.8.2.5.2 Accident Analysis

The drop height (80"), drop orientations, the properties of the target concrete surface, and the methodology used for the evaluation of the transfer with the NUHOMS-32P DSC as payload are the same as described in Section 8.2.5.2.

The design basis cask drop decelerations are specified in Table 12.3-5.

<u>NUHOMS-32P DSC</u>

Four accidental drop orientations are postulated for analysis.

A. Top end vertical drop
B. Bottom end vertical drop
C. Horizontal side drop
D. Corner Drop

A. Top End Vertical Drop

A 360° three-dimensional finite element model of the basket, rails, and the NUHOMS-32P DSC was constructed with the ANSYS computer program. Gap elements are used to simulate the interface between the basket rails and inner side of the canister. Details of the analysis model and boundary conditions used are described in Reference 12.22.

Two loadings are applied to the top end drop case:

1. 75g drop load only.
2. 75g drop load with accident pressure (100 psig).

The weight of the basket and fuel assemblies is idealized as an equivalent pressure load against the inner cover plate of the top shield plug. The effect of the combined basket and fuel weight at a maximum 75g acceleration is simulated by equivalent pressure. The effect of the self-weight of the NUHOMS-32P DSC shell assembly at 75g is also applied to the analysis model.

The resulting maximum stress intensities for the "75g drop only" load case and for the "75g drop with accident pressure" show that the case without internal pressure is the bounding analysis.

The maximum stress intensity for the "75g drop only" load case is located in the bottom shield plug side casing. Since these maximum stress intensities are due to the moments at the junction of the respective plates and casings which resist the bending of the flat plates, these are classified as Q stresses in accordance with Note (2) of Table 3217-1 131 of the ASME Code (Reference 12.19) and are ignored for accident condition evaluation.

B. Bottom End Vertical Drop

The DSC shell assembly is analyzed for the bottom end vertical drop using the same three-dimensional ANSYS model described above and the same two loadings. The annular area on the bottom end of the model is fixed in the axial direction. This is the area on the bottom lead liner, which contacts the transfer cask.

As in the Top End Vertical Drop analysis, the case without internal pressure governs.

The maximum stress intensity for the "75g drop only" load case is located in the top shield plug side casing. Since this maximum stress intensity is due to the moments at the junction of the respective plates and casings which resist the bending of the top inner cover plate, it is classified as Q stress in accordance with Note (2) of Table 3217-1 131 of the ASME Code (Reference 12.19) and is ignored for accident condition evaluation.

C. Horizontal Side Drop

A NUHOMS-24P DSC shell assembly consisting of a spacer disk and guide sleeves has been analyzed in Reference 12.49. Because the spacer disks apply a series of concentrated loads, and the NUHOMS-32P basket design will apply a distributed load, the NUHOMS-24P DSC analysis results in Reference 12.49 are conservatively scaled by a factor of 1.37 to reflect the weight increase of the NUHOMS-32P DSC. In order to account for accident pressure, the side drop stress intensities are added directly to the accident pressure stress intensities. The net stress intensities are, therefore, very conservative.

D. Corner Drop

No evaluation is required for the comer drop since the stresses are bounded by the vertical drop stresses (Reference 12.49).

Summary of Results

The maximum membrane plus bending stress enveloping all the cask drop scenarios evaluated above are below the Level D allowable stresses for the NUHOMS-32P DSC during a cask drop accident (Reference 12.22).

The maximum shear load on the fusion weld connecting the guide sleeves is 6,378 lb (Reference 12.22). The required strength of the fusion weld by testing is 16 kips, (Reference 12.41) which allows a safety factor of 2.

NUHOMS-32P DSC Basket Assembly

A. Horizontal Side Drop

The basket and canister are analyzed for two modes of side drops. Firstly, the cask is assumed to drop away from the transfer support rails. Under this condition, 0°, 45°, and 60° orientation of side drops are evaluated to bound the possible maximum stress cases. Secondly, the side drop occurs on transfer cask support rails at 180° orientation. The load resulting from the fuel assembly weight, for 1g and 75g accelerations, is applied as equivalent pressure on the plates. At 0° and 180° orientations, the pressure acts only on the horizontal plates while at other orientations, it is divided in components to act on both horizontal and vertical plates of the basket.

A nonlinear static stress analysis of the DSC basket structural assembly is conducted for computing the stresses for the 0°, 45°, 60°, and 180° drop orientations. A three-dimensional ANSYS model was used for this evaluation. The maximum load of 75g was applied in each analysis. Details of the analysis are described in Reference 12.28.

Buckling is analyzed using a nonlinear finite element analysis of the basket by applying load at the 0°, 30°, and 45° orientations relative to the basket plates. A maximum load of 150g was applied in each analysis and automatic time stepping was initiated allowing the program to determine the actual size of the load sub-step for a converged solution. The last converged solution represents the buckling load. In all orientations analyzed, the buckling load exceeded the

USAR defined minimum applied load of 75g. Details of the analysis are described in Reference 12.65.

B. Vertical End Drop

During an end drop, the fuel assemblies and fuel compartments are forced against the bottom of the canister/cask. For any vertical or near vertical loading, the fuel assemblies react directly against the bottom or top end of the canister/cask and not through the basket structure as in lateral loading. It is only the dead weight of the basket that causes axial compressive stress during an end drop. Axial compressive stresses are conservatively computed by assuming that all load acts on the fuel compartment guide sleeves during an end drop.

Summary of Results

The enveloping maximum membrane plus bending stresses in the basket assembly structural components are below the ASME Code Level D allowable stresses for the NUHOMS-32P DSC basket assembly during a transfer cask drop accident.

Transfer Cask

The evaluation of the transfer cask drop accident with a NUHOMS-32P DSC is based on the results of the drop evaluation for NUHOMS-24P DSC using the ANSYS computer model. The maximum stress intensities for individual components of the transfer cask are obtained by scaling to reflect the increased weight of the loaded NUHOMS-32P DSC, and are compared to Level D elastic allowables.

No evaluation is required for the corner drop since the stresses are bounded by the vertical drop stresses (Reference 12.50).

A. Top and Bottom End Vertical Drops

In the end drops, nearly all of the DSC load is taken by the underlying end components of the transfer cask. Therefore, the stresses in these components are increased by a scaling factor of 1.51 to reflect the ratio of the weights of the NUHOMS-32P and NUHOMS-24P DSCs. The stresses in the remaining components of the transfer cask are the same with either DSC since the transfer cask weight remains unchanged.

The structural integrity of a VAP CE 14x14 fuel assembly with Zircaloy clad fuel rods loaded in a

NUHOMS-32P DSC when subjected to an 80-inch transfer cask end drop accident was evaluated using a two part method. VAP fuel dimensions are bounding for standard fuel in an end drop. First, the transfer cask is analyzed for the end drop with a drop height of 80 inches and a dynamic finite element analysis program (LS-DYNA) is used to determine the rigid body acceleration time history of the NUHOMS-32P cask during impact on a concrete pad with subgrade soil (Reference 12.64). The model consists of the cask, the simplified DSC structure, a concrete impact pad, and the subgrade soil. Only ½ of the cask, DSC structure, concrete, and soil are modeled as the entire arrangement is symmetric about the X-Y plane. The resulting rigid body acceleration time histories are computed by LS-DYNA for the bottom plates + resin and the circumferential shell. These time histories are then used as input to the second part of the analysis (Reference 12.60), which also used the LS-DYNA code, to determine the maximum principal strain in the fuel pin. To determine the strain ductility demand on the fuel cladding, the inelastic buckling capacity of the fuel rods needs to be taken into account. The approach is taken to analyze the fuel rod as an elastic-plastic beam-column with initial bowing under dynamic impact condition. The finite element model of a single fuel rod with lateral displacement constraints is used to study the structural adequacy of a fuel rod under an 80" end drop condition. An effective Zircaloy cladding thickness that is reduced by the maximum 125 micron oxide thickness recommended (Reference 12.70) for burnups between 50 and 60 GWd/MTU was used for this analysis. This oxide thickness is bounding for Zirlo clad fuel, which has a maximum oxide thickness that is less than 100 microns at a burnup of 52 GWd/MTU (Reference 12.68, Figure 4.5.2-1). In Reference 12.60, two different models are used to determine the maximum principal strain the fuel cladding, however, only the second model (i.e., "Model II"), where the unfiltered deceleration time history of the transfer cask end plate from Reference 12.64 is used, represents the approved design basis. The results show the maximum principal strain of the fuel assembly is 0.341% which is still far less than the yield strain of 0.926% at 750°F for the fuel assembly. This conclusion may also be applied to Zirlo clad fuel, which has strain capability and yield strength exceeding that of Zircaloy (Reference 12.68, Sections 5.3.5 and 5.3.7). Thus, cladding integrity is confirmed for transfer cask end drop accident conditions for the NUHOMS-32P.

B. Horizontal Side Drop

The effect of the added weight of the transfer cask with NUHOMS-32P DSC payload is accounted for by increasing the stresses in the transfer cask structural shell and the inner liner by a factor of 1.1. This increase factor reflects the increased combined weight of the transfer cask and the NUHOMS-32P DSC. This is conservative in that the NUHOMS-32P DSC basket assembly spreads the weight evenly along the length of the transfer cask, while the NUHOMS-24P DSC basket design uses spacer plates which apply concentrated pressure loads at the spacer plate locations. The stresses in the end components of the transfer cask are not changed since they do not see the effect of the increased payload.

The structural integrity of a CE 14x14 fuel assembly with Zircaloy clad fuel rods loaded in a NUHOMS-32P DSC when subjected to an 80-inch (75g) transfer cask side drop accident at 750°F is also evaluated (References 12.61 and 12.69). The cladding tube is analyzed by ANSYS 10.0 as a continuous beam by using the actual span lengths and spacer widths. The beam model is subjected to lateral loads due to cladding tube and fuel pellet mass inertia. An effective cladding thickness that is reduced by the maximum 125 micron oxide thickness recommended for burnups between 50 and 60 GWd/MTU was used for this analysis. Fuel rod stress with this level of oxidation was determined to be 38.5 ksi, for Standard CE 14x14 fuel and 45.4 ksi for VAP CE 14x14 fuel, which is well below the ASME code allowable cladding yield strength of 92 ksi for the cask side drop anaylysis. As discussed for end drop, this conclusion is also applicable to Zirlo clad CE 14x14 fuel, which has a lower oxidation rate and superior mechanical properties compared to Zircaloy (Reference 12.68). Thus, cladding integrity is confirmed for the transfer cask side drop accident condition for the NUHOMS-32P.

<u>Summary of Results</u>

The enveloping maximum membrane plus bending stresses are below allowable ASME Code Level D limits for the dropped transfer cask accident.

For the NUHOMS-32P, fuel rod cladding stress analyses were performed for the side and end drops using an effective cladding thickness that is reduced by the maximum 125 micron oxide thickness recommended for burnups between 50 and 60 GWd/MTU. The results indicate that drop accident induced stresses in fuel rods with this level of oxidation will also remain below the fuel cladding yield strength. Thus, the structural integrity of the fuel cladding is confirmed for accident conditions (the cask drop event) for the extended burnup NUHOMS-32P.

12.8.2.5.3 Accident Dose Consequences

Dose calculations for the transfer cask drop accident with a NUHOMS-32P DSC use the same methodology and model as the calculations for the NUHOMS-24P DSC, and continue to assume that the neutron shielding is lost. The doses for the NUHOMS-32P DSC, with the increased neutron source term, are 1,251.2 mrem/hr on contact, and 145.5 mrem/hr at 15' (Reference 12.29). The contact dose rate, 1,251.2 mrem/hr, remains below the limit of 5 rem/hr for this accident (Section 4.7.3.3). The recovery dose to an on-site worker, at an average distance of 15', increases from 776 mrem to 1,164 mrem (145.5 mrem/hr x 8 hr = 1,164 mrem). The recovery dose remains below the limit of 5 rem at the site boundary since the total dose at 15' is much less than 5 rem.

12.8.2.6 Lightning

The lightning evaluation presented in Section 8.2.6 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System.

12.8.2.7 Blockage of Air Inlets and Outlets

This accident is postulated to consist of the complete and total blockage of all HSM air inlets and outlets for a period of 36 hours.

12.8.2.7.1 Cause of Accident

See Section 8.2.7.1.

12.8.2.7.2 Accident Analysis

The stresses caused by the additional weight of debris blocking the air inlets and outlets are bounded by the structural consequences of other accidents described in this section (i.e., tornado and earthquake analyses). The thermal consequences of this accident result from heating of the DSC and HSM/HSM-HB due to the loss of natural convection cooling.

The thermal analyses to determine the temperature rise for the Calvert Cliffs HSM/HSM-HB and DSC components due to blocked vents are performed using the ANSYS finite element methodology (Reference 12.2 for HSM and Reference 12.74 for HSM-HB). The design basis pressure is considered in the DSC accident pressure evaluation presented in Section 12.8.2.9.

The thermally induced stresses for the HSM/HSM-HB for the blocked vent case are calculated using an ANSYS finite element model and the methodology discussed in Reference 12.23 for HSM and Reference 12.72 for HSM-HB. The thermally induced stresses for the DSC during accident conditions are addressed in Reference 12.22.

12.8.2.7.3 Accident Dose Consequences

The discussion in Section 8.2.7.3 is unchanged and is conservative for the NUHOMS-32P DSC in the HSM because the possible recovery dose is lower. The dose rate at the HSM air inlet vent for the NUHOMS-32P DSC is 61 mrem/hr (Reference 12.30), which is less than the 73 mrem/hr for the NUHOMS-24P DSC. The possible recovery dose incurred by an onsite worker for debris removal is 488 mrem (61 mrem/hr x 8 hr) for the NUHOMS-32P DSC, in the HSM which is less than the 584 mrem for the NUHOMS-24P DSC (Section 8.2.7.3).

The dose rate at the HSM-HB air inlet vent for the NUHOMS-32P DSC is 88.9 mrem/hr (Reference 12.30), not including credit for additional gamma shielding provided by the alternate bird screen design. This would increase the on-site worker 8-hour debris removal dose to 711 mrem for a NUHOMS-32P DSC in an HSM-HB.

<u>12.8.2.8 Dry Shielded Canister Leakage</u>

As described in Section 12.3.3.2, the DSC is designed to ensure no leakage and the analyses for normal and accident conditions have shown that there are no credible events which can breach the DSC pressure boundary or fail the double seal welds at each end of the DSC. However, a total and instantaneous leak of a single NUHOMS-32P DSC is postulated using the same methodology as in Section 8.2.8 except using a 24 month operating cycle (resulting in a release fraction of 9.13%).

The resulting calculated doses for the NUHOMS-32P DSC are:

Off-site total body dose:	0.71 mrem,
Off-site skin dose:	119.3 mrem.

These doses remain within the 10 CFR 72.106 limit of 5,000 mrem.

<u>12.8.2.9 Accidental Pressurization of Dry Shielded Canister</u>

This accident addresses the consequences of accidental pressurization of the NUHOMS-32P DSC.

12.8.2.9.1 Cause of Accident

See Section 8.2.9.1.

12.8.2.9.2 Accident Analysis

The maximum NUHOMS-32P DSC pressurization is calculated assuming that 100% of the fuel rods in a DSC rupture and release the fission and fuel rod fill gasses to the DSC cavity. The fuel rod fission gas release fraction is assumed to be 30% and the fuel rod fill gas release fraction is assumed to be 100%. The maximum fuel rod fill gas pressure is assumed to be 465 psia and is used to calculate the quantity of fill gas released from fuel rods to the DSC cavity during fuel rod rupture conditions. The internal DSC pressure is calculated at the maximum ambient temperature of 103°F and a solar heat flux of 127.0 Btu/hr-ft^2.

The limiting accident for DSC pressurization is the HSM blocked vent case as discussed in Section 12.8.2.7. Under these conditions, the gas temperatures in the DSC will rise to 735°F with a DSC internal pressure of 99.4 psig (Reference 12.17). This is based on a 48 hour vent blockage timeframe, which is conservative compared to the HSM blocked vent accident time of 36 hours in Section 12.8.2.7.

The maximum DSC pressure boundary stress intensities due to accident pressurization are calculated using 100 psig which bounds the 99.4 psig maximum DSC internal pressure. The calculated component stress intensities at 100 psig were determined to be below the allowable stress limits.

For NUHOMS-32P DSC, the maximum partial pressure of fill gas is 35% of the total gas pressure in the DSC, which is still not a major contributor to the accident DSC internal pressures.

The analysis of accidental pressurization of the DSC includes the effect of fuel burnup on internal fuel rod pressure by using the volume of fission gas generated in the fuel rod at the maximum burnup of 52,000 MWD/MTU. The results of the analysis show that the maximum DSC accident pressures are within the allowable design bases limits (Reference 12.62).

12.8.2.9.3 Accident Dose Calculations

Since the maximum NUHOMS-32P DSC accident pressure is within the design basis limits, there are no dose consequences.

12.8.2.10 Forest Fire

This postulated event involves a forest fire occurring in the woods adjacent to the Calvert Cliffs ISFSI.

12.8.2.10.1 Cause of Accident

See Section 8.2.10.1.

12.8.2.10.2 Accident Analysis

The initial parameters used in Section 8.2.10.2 for the forest fire evaluation remain unchanged for the NUHOMS-32P DSC with the exception of the initial HSM/HSM-HB concrete temperature. The NUHOMS-32P forest fire evaluation is based on an initial peak fuel region side wall concrete temperature of 183°F, which represents the average between the maximum inside and outside 3-foot concrete wall temperatures at 103°F ambient (Reference 12.56). This concrete temperature bounds that for storage of the NUHOMS-32P DSC in the HSM-HB (Reference 12.73, Figure 6-2). The damage to the wall, based on the HSM wall temperature gradient resulting from the fire, will be limited to a thickness of 6" into the wall (Reference 12.56). The remainder of the wall thickness will remain within American Concrete Institute 349 temperature limits. Fuel cladding temperature limits will be maintained within the fuel cladding short-term temperature limit. The effect of the surface cracking and spalling will be minimal with respect to the load capacity of the HSM/HSM-HB walls. The NUHOMS-32P DSC internal pressure limit (100 psig) will not be exceeded. The increase in HSM surface dose is from 13.5 mrem/hr to approximately 60.4 mrem/hr (Reference 12.30). This dose rate bounds storage of the NUHOMS-32P in the HSM-HB since the thickness of the roof and walls is 8 inches greater than the HSM. This increase is not considered a "significant increase in occupational exposure" for the necessary repair activities. Actions to mitigate the fire and repair the HSMs/HSM-HBs will ensure that offsite dose consequences will be limited and of short duration and will remain within the limits of 10 CFR 72.106.

12.8.2.10.3 Accident Dose Consequences

There are no accident dose consequences associated with the postulated forest fire accident.

12.8.2.11 Liquified Natural Gas Plant or Pipeline Spill or Explosion

Use of the NUHOMS-32P DSC design does not change the analysis described in Section 8.2.11.

12.8.2.12 Load Combinations

The load categories associated with normal, off-normal, and accident conditions have been described and analyzed in previous chapters. Evaluation of the load combination for the NUHOMS-32P important to safety components is addressed in this section.

The methodology used in combining normal, off-normal, and accident loads and their associated overload factors for various NUHOMS-32P components is presented in Reference 12.22. The load combination analysis results showed that the calculated stresses are less than the code allowables for various load combinations shown in Tables 12.8-1, 12.8-2, 12.8-3, 12.8-4, 8.2-14, 8.2-15, and 8.2-16.

When compared to the NUHOMS-24P DSC, the confinement boundary stress allowables (Table 12.8-3) have been altered to an elastic/plastic analysis for all accident conditions except for the 100 psig applied to the inner pressure boundary combinations (D_3, D_4, & D_5)

- Horizontal storage module enveloping load combination results were obtained based on a conservative interpretation of the Calvert Cliffs Nuclear Power Plant (CCNPP) calculation. The forces and moments, including thermal loads, are taken from the ANSYS output presented in Reference 12.23 for the poured in place HSM and Reference 12.72 for the HSM-HB.

The load combination analysis results show that the calculated stresses are less than the code allowable stresses for all the specified normal, off-normal, and accident condition load combinations.

12.8.2.13 Other Event Considerations

Use of the NUHOMS-32P DSC design does not change the analysis described in Section 8.2.13.

12.8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

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

TABLE 12.8-1
NUHOMS-32P DRY SHIELDED CANISTER ENVELOPING LOAD COMBINATION RESULTS FOR NORMAL AND OFF-NORMAL LOADS

(ASME Service Levels A and B)

DSC COMPONENTS	STRESS TYPE	CONTROLLING[a] LOAD COMBINATION	ALLOWABLE[b][c] STRESS (ksi)
DSC Shell	Primary Membrane	B2	18.96
	Membrane + Bending	B2	28.44
	Primary + Secondary	B2	56.88
Bottom Cover Plate	Primary Membrane	A4	18.96
	Membrane + Bending	B2	28.44
	Primary + Secondary	B2	56.88
Top Pressure Plate	Primary Membrane	A3	18.96
	Membrane + Bending	A3	28.44
	Primary + Secondary	A3	56.88
Top Structural Plate	Primary Membrane	A4	18.96
	Membrane + Bending	A4	28.44
	Primary + Secondary	A4	56.88

(a) See Table 12.3-6 for load combination nomenclature.
(b) See Table 3-4 of Reference 12.22 for allowable stress criteria. Material properties were obtained from Table 3.5 of Reference 12.22 at a design temperature.
(c) Allowables are for stainless steel material at 380°F.

TABLE 12.8-2
NUHOMS-32P DRY SHIELDED CANISTER ENVELOPING LOAD COMBINATION RESULTS FOR ACCIDENT LOADS

(ASME Service Level C)

DSC COMPONENTS	STRESS TYPE	CONTROLLING[a] LOAD COMBINATION	ALLOWABLE[b][c] STRESS (ksi)
DSC Shell	Primary Membrane	C2	21.6
	Membrane + Bending	C1	32.4
Bottom Cover Plate	Primary Membrane	C2	21.6
	Membrane + Bending	C5	32.4
Top Pressure Plate	Primary Membrane	C2	21.6
	Membrane + Bending	C2	32.4
Top Structural Plate	Primary Membrane	C2	21.6
	Membrane + Bending	C1	32.4

(a) See Table 12.3-6 for load combination nomenclature.

(b) See Table 3-4 of Reference 12.22 for allowable stress criteria. Material properties were obtained from Table 3-5 of Reference 12.22 at a design temperature.

(c) Allowables are for stainless steel material at 460°F.

TABLE 12.8-3
NUHOMS-32P DRY SHIELDED CANISTER ENVELOPING LOAD COMBINATION RESULTS FOR ACCIDENT LOADS

(ASME Service Level D)[c]

DSC COMPONENTS	STRESS TYPE	CONTROLLING[a] LOAD COMBINATION	ELASTIC ALLOWABLE[b] STRESS (ksi)	ELASTIC-PLASTIC ALLOWABLE[d] STRESS (ksi)
DSC Shell	Primary Membrane	D2	43.2	44.6
	Membrane + Bending	D2	63.6	57.3
Bottom Cover Plate	Primary Membrane	D2	43.2	44.6
	Membrane + Bending	D2	63.6	57.3
Top Pressure Plate	Primary Membrane	D2	43.2	44.6
	Membrane + Bending	D2	63.6	57.3
Top Structural Plate	Primary Membrane	D2	43.2	44.6
	Membrane + Bending	D2	63.6	57.3
Basket Assembly	Primary Membrane	D2	44.4	
	Membrane + Bending	D2	57.0	
Top End Structural Weld	Primary	--	21.6	22.3
	Membrane + Bending		N/A	
Bottom End Structural Weld	Primary	--	45.1	45.1

For more information see Reference 12.22.

(a) See Table 12.3-6 for load combination nomenclature.
(b) See Table 3-4 of Reference 12.22 for allowable stress criteria. Material properties were obtained from Table 3-5 of Reference 12.22 at a design temperature.
(c) Allowables are for stainless steel material at 460°F.
(d) Used on 100 psi pressure applied at outer boundary case (D_5).

TABLE 12.8-4
NUHOMS-32P DRY SHIELDED CANISTER SUPPORT ASSEMBLY ENVELOPING LOAD COMBINATION RESULTS

Component	Load Combination	AISC Allowable Stress for HSM Axial (ksi)	Bending (ksi)	Shear (ksi)
W8x48 Cross Beam	Normal Operation $DW_s + DW_c + HL_f$	15.5	17.6	10.6
	Off-Normal Operation $DW_s + HL_j$	26.3	29.9	18.0
	Accident $DW_s + DW_c + DBE$	24.8	28.1	16.7
W8x40 Support Rail	Normal Operation $DW_s + DW_c + HL_f$	14.6	17.6	10.6
	Off-Normal Operation $DW_s + HL_j$	24.8	29.9	18.0
	Accident $DW_s + DW_c + DBE$	23.4	28.1	16.7

KEY: DW_s = Dead Weight Support Assembly, HL_j = Off-normal Handling Loads-Jammed, DW_c = Dead Weight Canister, HL_f = Normal Loads Friction, DBE = Seismic Loads

NOTES:
Allowable stresses taken at 600°F to conservatively envelope all ambient temperature cases.
Allowables for $DW_s + DW_c + DBE$ increased by a factor of 1.6.

TABLE 12.8-4
NUHOMS-32P DRY SHIELDED CANISTER SUPPORT ASSEMBLY ENVELOPING LOAD COMBINATION RESULTS

AISC Allowable Stress Radios for HSM-HB

Rail Component Results

Load Comb.	Interaction Ratio	Shear Stress Ratio	Stiffener Plate Stress Ratio
C1S	0.35	0.67	0.19
C2S	0.58	0.84	0.00
C3S	0.58	0.93	0.22
C4S	0.51	0.96	0.18
C5S	0.40	0.63	0.55

Extension Plates and Cross Members Results

Load Comb.	Extension Plates Interaction Ratio	Cross Members Stress Ratio
C1S	0.77	0.25
C2S	0.77	0.32
C3S	0.71	0.21
C4S	0.60	0.25
C5S	0.71	0.33

12.9 CONDUCT OF OPERATIONS

The Calvert Cliffs ISFSI is operated under the same corporate management organization responsible for operation of the CCNPP. The conduct of operations for the Calvert Cliffs ISFSI are described in Chapter 9.0. The discussion presented in Chapter 9.0 is not affected by the addition of the NUHOMS-32P DSC to the NUHOMS System.

12.10 OPERATING CONTROLS AND LIMITS

The discussion presented in Chapter 10 is not affected by the addition of the NUHOMS-32P DSC to the CCNPP NUHOMS System.

12.11 QUALITY ASSURANCE

The quality assurance program for the Calvert Cliffs ISFSI covers the construction phase, the operational phase, and the decommissioning phase of structures, systems, and components of the Calvert Cliffs ISFSI important to safety. The Calvert Cliffs ISFSI quality assurance program is discussed in Chapter 11. The discussion presented in Chapter 11 is not affected by the addition of the NUHOMS-32P DSC to the CCNPP NUHOMS System.

12.12 REFERENCES

12.1 CCNPP Calculation No. CA06306, "DSC Thermal Analysis - Off Normal Conditions (Max. Summer Temperature)"

12.2 CCNPP Calculation No. CA06304, "HSM Thermal Analysis - Accident Conditions (Blocked Vents)"

12.3 CCNPP Calculation No. CA06313, "Thermal Analysis of Transfer Cask with Poison Material in Basket"

12.4 CCNPP Calculation No. CA06314, "Thermal Analysis of Vacuum Drying"

12.5 NUREG/6361, dated March 01, 1997, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," March 01, 1997

12.6 CCNPP Calculation CA05895, "Criticality Benchmarks"

12.7 CCNPP Calculation CA06227, "Criticality Analysis of the NUHOMS-32P for Calvert Cliffs ISFSI"

12.8 CCNPP Calculation CA05896 "Criticality Analysis for Fuel Misloads and Accidents"

12.9 CCNPP Calculation CA06312, "Thermal Analysis of Storage Cases Poison Plates in Basket"

12.10 CCNPP Calculation No. CA06296, "Finite Element Models, Thermal Analysis"

12.11 CCNPP Calculation No. CA06301, "HSM Thermal Analysis - Normal Storage Conditions"

12.12 CCNPP Calculation No. CA06305, "DSC Thermal Analysis - Normal Storage Conditions"

12.13 CCNPP Calculation No. CA06295, "Effective Fuel Properties"

12.14 "Spent Nuclear Fuel Effective Conductivity Report," TRW Environmental Safety Systems, Inc., Document Identifier BBA000000-01717-5705-00010 Revision 00

12.15 CCNPP Calculation No. CA06297, "Transfer Thermal Analysis, 103°F"

12.16 CCNPP Calculation No. CA06298, "Transfer Thermal Analysis, -3°F Ambient"

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

12.18 CCNPP Calculation No. CA06308, "Effective Fuel Properties for Vacuum Drying"

12.19 CCNPP Calculation No. CA06309, "Sensitivity of the Component Temperatures to the Position of the DSC in the Transfer Cask"

12.20 CCNPP Calculation No. CA06310, "Sensitivity of the Transfer Cask Thermal Analysis to the Axial Gaps Between DSC and Transfer Cask"

12.21 "The Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel NUHOMS-24P," Revision 1A

12.22 CCNPP Calculation No. CA06359, "NUHOMS 32P - DSC Structural Analysis"

12.23 CCNPP Calculation No. CA06364, "NUHOMS 32P - CCNP ISFS HSM Facility Evaluation"

12.24 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB and NC and Appendix F, 1998 Edition with 1999 Addenda

12.25 CCNPP Calculation No. CA06329, "NUHOMS-32P - Transfer Cask Structural Analysis"

12.26 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG and Appendix F, 1998 including 1999 Addenda

12.27 CCNPP Calculation No. CA06326, "NUHOMS 32P Basket Stress Analysis for Storage Loads (Normal and Accident)"

12.28 CCNPP Calculation No. CA06335, "Basket Stress Analysis Due to Accident Transfer Drop"

12.29 CCNPP Calculation No. CA06750, "Loading and Transfer Dose Rate for ISFSI 32P Burnup Extension"

12.30 CCNPP Calculation No. CA06751, " Horizontal Storage Module Dose Rates for ISFSI 32P Burnup Extension

12.31 CCNPP Calculation No. CA06327, "Shielding Evaluation with the New Top Shield Plug for NUHOMS-32P"

12.32 CCNPP Calculation No. CA05924, "Calvert Cliffs ISFSI/NUHOMS-24P Radiation Dose Rates for Cask Loading and Transfer"

12.33 CCNPP Calculation No. CA05925, "Calvert Cliffs ISFSI/NUHOMS-24P HSM Dose Rates"

12.34 CCNPP Calculation No. CA06319, "NUHOMS-32P – Weight Calculation of DSC/TC"

12.35 CCNPP Drawing 84218SH0001, "NUHOMS-32P DSC Shell & Bottom Plug Assembly"

12.36 CCNPP Drawing 84219SH0001, "NUHOMS-32P DSC Shell & Siphon Pipe Assembly Details"

12.37 CCNPP Drawing 84220SH0001, "NUHOMS-32P DSC Siphon Pipe/Adapter & Lifting Block Details"

12.38 CCNPP Drawing 84221SH0001, "NUHOMS-32P DSC Top Shield Plug Details"

12.39 CCNPP Drawing 84222SH0001, "NUHOMS-32P Top Cover Plate & Siphon/Vent Port Covers"

12.40 CCNPP Drawing 84223SH0001, "NUHOMS-32P DSC Basket Assembly"

12.41 CCNPP Drawing 84224SH0001, "NUHOMS 32P DSC Basket Details"

12.42 CCNPP Drawing 84225SH0001, "NUHOMS-32P DSC Basket Rails & Shims"

12.43 CCNPP Drawing 84226SH0001, "NUHOMS-32P DSC Basket Plate & Rail Inserts"

12.44 CCNPP Drawing 84227SH0001, "NUHOMS-32P DSC Parts List"

12.45 CCNPP Drawing 84234SH0001, "NUHOMS-32P DSC Final Assembly, Field Welding & Testing"

12.46 CCNPP Drawing 84235SH0001, "NUHOMS-32P DSC ASME Code Exceptions"

12.47 CCNPP Calculation No. CA06721, "Source Terms for ISFSI 32P Burnup Extension," Revision 0

12.48 CCNPP Calculation No. CA06637, "Sensitivity Analysis of Homogenized Fuel Region"

12.49 CCNPP Calculation No. CA04977, "NUTECH Horizontal Module System (NUHOMS) 24P ISFSI Shielded Canister Structural Analysis for Sixteen New Assemblies"

12.50 CCNPP Calculation No. CA04141, "ISFSI Transfer Cask Structural Analysis"

12.51 ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"

12.52 ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"

12.53 NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," 1997

12.54 Nuclear Regulatory Commission Safety Evaluation Report, Docket No. 72-1004, Standardized NUHOMS Modular Storage System for Irradiated Nuclear Fuel, Certificate of Compliance No. 1004, Amendment 5 (NUHOMS-32PT System)

12.55 CCNPP 10 CFR 72.48 Safety Evaluation No. SE00163, Revision 1, "Use of NUHOMS-32P Dry Shielded Canister"

12.56 CCNPP Calculation No. CA06653, "Forest Fire Evaluation for the NUHOMS-32P for CCNPP"

12.57 CCNPP Calculation No. CA06629, "Forest Fire Radiological Evaluation for the NUHOMS-32P HSM for CCNPP"

12.58 CCNPP Calculation No. CA06367, "Comparison of the Radiological, Thermal, and Reactivity Characteristics of Assemblies with Missing or Inert Fuel Rods with the 32P ISFSI DSC Design Basis"

12.59 CCNPP Calculation No. CA06354, "Accidental Drop Loading Evaluation of 14x14 Fuel Assembly with Missing Fuel Rods"

12.60 CCNPP Calculation No. CA07295, Transnuclear Calculation No. NUH32P+.0204, "Fuel End Drop Analysis for NUH32P+ Using LS-DYNA," Revision 1, August 2010

12.61 CCNPP Calculation No. CA06822-0001, Transnuclear Calculation No. NUH32P+.0201, "NUHOMS 32P CE14x14 Fuel Cladding Strength Under Accident Side Drop Conditions," Revision 1, December 2008

12.62 CCNPP Calculation No. CA06771, Transnuclear Calculation No. NUH32P+.0402, "Effect of Updated Fuel Performance Data on NUHOMS 32P+ Internal Pressure," Revision 0, August 2007

12.63 SCALE 4.4, "A Modular Code System for Performing Standarized Computer Analysis for Licensing Evaluations," NUREG/CR-0200, Revision 6, September 1998

12.64 CCNPP Calculation No. CA07294, Transnuclear Calculation No. NUH32P+.0203, "32P+ Transfer Cask Impact onto the Concrete Pad LS-DYNA Analysis (80 inch End Drop)," Revision 0, March 2010

12.65 CCNPP Calculation No. CA07098, NUHOMS® 32P Basket Buckling Analysis Using Full ANSYS Model

12.66 CCNPP Calculation No. CA07660, NUHOMS 32P-DSC Side Drop Evaluation for Reduced Lead Plug Top Casing Plate Thickness

12.67 CCNPP Calculation No. CA07661, NUHOMS 32P-DSC Structural Analysis Due to Under Thickness of lead Plug Top Casing Plate

12.68 CENPD-404-P, "Implementation of Zirlo Cladding Material in CE Nuclear Power Fuel Assembly Designs", Revision 0000

12.69 CCNPP Calculation CA07302, "PWR Fuel Rod Drop Accident Side Drop Loading Stress Analysis for NUHOMS 32PHB System"

12.70 CCNPP Calculation CA06758, "Fuel Performance Data for Calvert Cliffs Dry Storage (ISFSI) for Fuel Batches C1N through C1T and C2M through C2S"

12.71 CCNPP Calculation CA08032, "Criticality Analysis of the NUHOMS 32P Loaded with VAP Fuel for the Calvert Cliffs ISFSI"

12.72 CCNPP Calculation CA07307, "HSM-HB Structural Analysis for NUHOMS-32PHB System"

12.73 CCNPP Calculation CA08096, "Thermal Performance of 32P DSC Inside HSM-HB for Normal and Off-Normal Conditions"

12.74 CCNPP Calculation CA08107, "Thermal Performance of 32P DSC In HSM-HB for Blocked Vent Accident Condition"

12.75 CCNPP Calculation Change Notice CA06721-0000, Source Terms for ISFSI 32P Burnup Extension

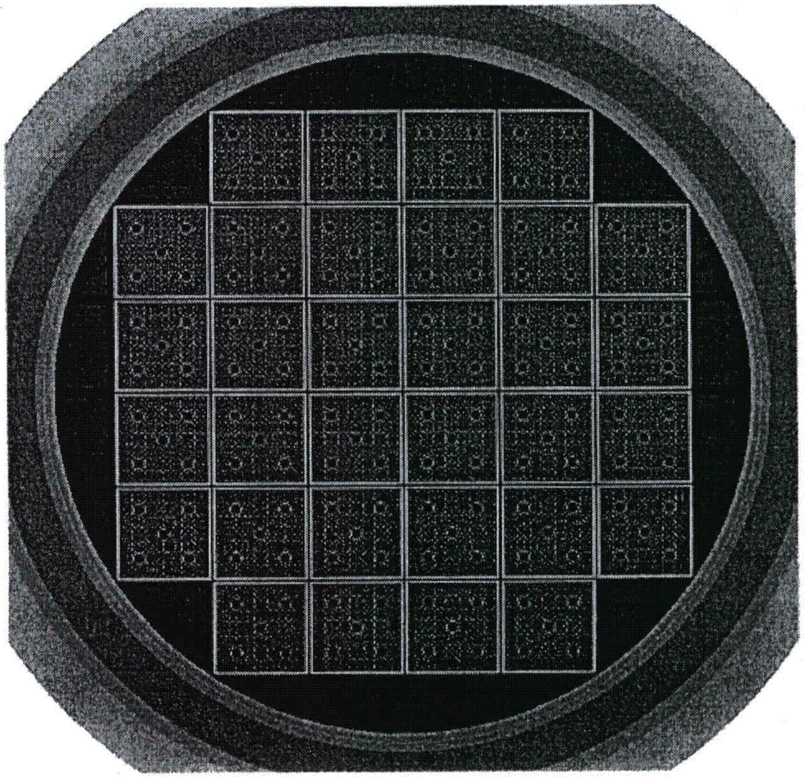
12.76 ORNL/TM-2005/39 Version 6, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," Oak Ridge National Laboratory, January 2009

12.77 Transnuclear, Inc., "Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1004, Transnuclear Document No. NUH-003, Revision 11

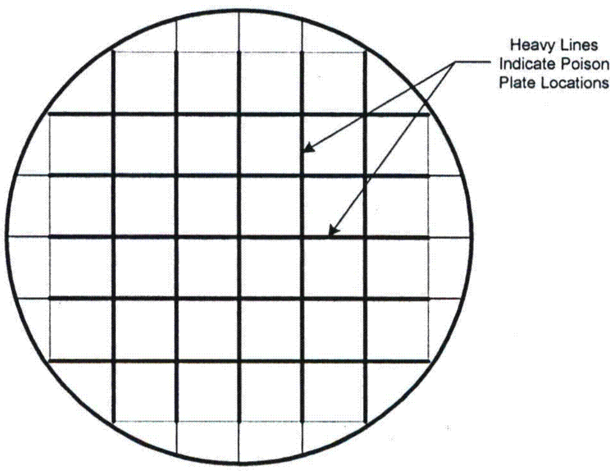
12.78 SE00528, 72.48 Evaluation for ECP-13-00637: ECP to Justify Storing 32P DSCs Inside New HSM-HB Modules

12.79 CCNPP Calculation CA08084, "NUHOMS 32P DSC Criticality and Thermal Performance Using MMC Poison Material"

12.80 Transnuclear, Inc., "Updated Final Safety Analysis Report for the NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1030, Revision 2



NUHOMS-32P DSC and Transfer Cask KENO V.a Criticality Model



Poison Plate Locations in DSC and Transfer Cask KENO V.a Criticality Model

ENCLOSURE 3

ANSYS FILE LIST

The hard drive with these files has been provided to J. M. Goshen (NRC, NMSS)

ENCLOSURE 3
ANSYS FILE LIST

1-403
32PHB_HLZC2.MAC
32PHB_Mat1.inp
32PHB_Model.db
32PHB_TC_ACC_NS_Map.cbdo
32PHB_TC_OFN_TRANS_20hr_Map.cbdo
32PHB_TC_VERT_TRANS_20hr_Map.cbdo
32PHB_TC2M.db
32PHB_TC2M.inp
32PHB_TC2M.mntr
32PHB_TC2M.out
32PHB_TC2M.rth
32PHB_TC2M.s01
32PHB_TC2M.s02
32PHB_TC2M.s03
GetHL
Macro.mac
MaxMinN
Results.mac
Summary_32PHB_TC2M.txt

2-403
32PHB_HLZC2.MAC
32PHB_Mat1.inp
32PHB_Model.db
32PHB_STB1M.db
32PHB_STB1M.inp
32PHB_STB1M.mntr
32PHB_STB1M.out
32PHB_STB1M.rth
32PHB_STB1M.s01
32PHB_STB1M.s02
32PHB_STB1M.s03
Get HL
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MaxMinN
Results.mac
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TS31kw_40.txt
TS31kw_100.txt
TS31kw_105.txt
TS31kw_bv40.txt

3-407
32PHB_ir_max.txt
32PHB_ir1.db
32PHB_ir1.out
32PHB_ir1.rth
32PHB_ir1.txt
fuel.db
fuel.sub
matpr-ir1_32P.txt
mtr_ir.sub

4-409
CCNPP-HSM-FF.db
CCNPP-HSM-FF.inp
CCNPP-HSM-FF.mntr
CCNPP-HSM-FF.out
CCNPP-HSM-FF.rth
CCNPP-HSM-FF.s01
CCNPP-HSM-FF.s02
CCNPP-HSM-FF.s03
CCNPP-HSM-FF.s04
CCNPP-HSM-FF.s05
CCNPP-HSM-FF.s06
CCNPP-HSM-FF.s07
CCNPP-HSM-FF.s08
CCNPP-HSM-FF.s09
CCNPP-HSM-FF.s10
CCNPP-HSM-FF.s11
CCNPP-HSM-FF.s12
CCNPP-HSM-FF.s13
CCNPP-HSM-FF.s14
CCNPP-HSM-FF.s15
CCNPP-HSM-FF.s16
CCNPP-HSM-FF.s17
CCNPP-HSM-FF.s18
CCNPP-HSM-FF.s19
CCNPP-HSM-FF.s20
CCNPP-HSM-FF.s21
CCNPP-HSM-FF.s22
CCNPP-HSM-FF.s23
CCNPP-HSM-FF.s24
CCNPP-HSM-FF.s25
CCNPP-HSM-FF.s26
CCNPP-HSM-FF.s27
CCNPP-HSM-FF.s28
CCNPP-HSM-FF.s29
CCNPP-HSM-FF.s30
Concrete_PEAK.txt
forest_fire.sub

ENCLOSURE 3
ANSYS FILE LIST

5-410
BCss.mac
conv_link.mac
HC_HCL.mac
HC_HPLD.mac
HC_HPLU.mac
HC_IPLD.mac
HC_IPLU.mac
HC_ROOF.mac
HC_VPL.mac
HCIPLUm.mac
HSMH61BTH.cdb
HSMHm.node
HTCS61BTH.txt
Mat61BTH.mac
nodegrps.txt
NUH61BTH.sub
runST105-31.inp
runST105-31.out
St105-31.db
St105-31.rth
St105-31.s01
supelem.txt

0402
32PHB_TC_ACC_NS
32PHB_TC_ACC_NS_map
32PHB_TC_ACC_PF
32PHB_TC_initial
32PHB_TC_OFN_TRANS
32PHB_TC_OFN_TRANS_20hr_Map
32PHB_TC_OFN_TRANS_SENS
32PHB_TC_VERT_TRANS
32PHB_TC_VERT_TRANS20hr_mAP
CCNPP-FC-TC

21.12kW Horizontal
32PHB_DSC_OFN_SS_21kW
32PHB_TC_OFN_SS_21kW
32PHB_TC_OFN_SS_21kW_Map

21.12kW Vertical
32PHB_DSC_VERT_SS
32PHB_TC_VERT_SS
32PHB_TC_VERT_SS_Map
PP_32PHB_TC_VERT_SS

32PHB_VDN1MH
32PHB_HLZC2.MAC
32PHB_Mat1H.inp
32PHB_Mat1N. inp
32PHB_Model.db
32PHB_VDN1MH.BCS
32PHB_VDN1MH.db
32PHB_VDN1MH.err
32PHB_VDN1MH.inp
32PHB_VDN1MH.mntr
32PHB_VDN1MH.out
32PHB_VDN1MH.rth
32PHB_VDN1MH.s01
32PHB_VDN1MH.s02
32PHB_VDN1MH.s03
32PHB_VDN1MH.s04
32PHB_VDN1MH.s05
32PHB_VDN1MH.s06
32PHB_VDN1MH.s07
32PHB_VDN1MH.s08
32PHB_VDN1MH.s09
32PHB_VDN1MH.s10
eccp-ansys-job.txt
GetHL
Macro.mac
MaxMinN
Results.mac
Summary_32PHB_VDN1MH.txt

ENCLOSURE 3
ANSYS FILE LIST

<u>32PHB_VDY4</u>
32PHB_HLZC2.MAC
32PHB_Mat1.inp
32PHB_Model.db
32PHB_VDY4.BSC
32PHB_VDY4.db
32PHB_VDY4.err
32PHB_VDY4.inp
32PHB_VDY4.mntr
32PHB_VDY4.out
32PHB_VDY4.rth
32PHB_VDY4.s01
32PHB_VDY4.s02
32PHB_VDY4.s03
32PHB_VDY4.s04
32PHB_VDY4.s05
32PHB_VDY4.s06
32PHB_VDY4.s07
32PHB_VDY4.s0832PHB_VDY4.s09
32PHB_VDY4.s10
eccp-ansys-job.txt
GetHL
Macro.mac
MaxMinN
Results.mac
Summary_32PHB_VDY4.txt