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BSC	DESIGN CALCULATION OR ANALYSIS COVER SHEET	1. QA: QA 2. Page 1

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A. Listing of Computer Files	7
B. One Compact Disk Containing All Files Listed in Attachment A	1
C. Drawing of Aging Area General Arrangement Plan and Sections	2

RECORD OF REVISIONS

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00A	Initial issue	44	III-1	C.E. Sanders	D.S. Kimball	S. Su
00B	Update and revise the criticality safety evaluation of the Aging Facility. Revision includes inclusion of all waste types, update Category 1 and 2 event sequences, and further evaluation of the MSC. Extensive Revision. All pages are affected.	74	III-1	C.E. Sanders	G. Radulescu	S. Su
00C	Update the document to reflect the design changes and include more supporting MCNP calculations. Extensive revisions, all pages affected.	85	C-2	Cheng T. Hsu Cheng Tsang Hsu 8/1/2006	Charlotta E. Sanders [signature] 08/01/2006	Abdelhalim A. Alsaed [signature] 8/2/2006

DISCLAIMER

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LIST OF ACRONYMS AND ABBREVIATIONS

ATR	Advanced Test Reactor
BSC	Bechtel SAIC Company, LLC
BWR	Boiling Water Reactor
CD	Compact Disc
CFR	Code of Federal Regulations
cm	centimeters
CRWMS M&O	Civilian Radioactive Waste Management System Management & Operating Contractor
CSNF	Commercial Spent Nuclear Fuel
DBGM	design basis ground motion
DOE	U.S. Department of Energy
DPC	Dual-Purpose Canister
FFTF	Fast Flux Test Facility
FSAR	Final Safety Analysis Report
HLW	High Level Waste
i.d.	inner diameter
in.	inch
k_{eff}	effective neutron multiplication factor
LWBR	Light Water Breeder Reactor
MCNP	Monte Carlo N-Particle transport code
MGR	Monitored Geological Repository
MPC	Multi Purpose Canister
NRC	U.S. Nuclear Regulatory Commission
NSNFP	National Spent Nuclear Fuel Program
o.d.	outer diameter
OFA	Optimized Fuel Assembly
PDC	Project Design Criteria
PRD	Project Requirements Document
PWR	Pressurized Water Reactor
σ	standard deviation
SFA	Spent Fuel Assembly

SS	Stainless Steel
SNF	Spent Nuclear Fuel
TD	theoretical density
TMI	Three Mile Island
TRIGA	Training Research Isotopes General Atomics
USL	upper sub-critical limit
WP	Waste Package
wt%	weight percent

1. PURPOSE

This design calculation is a revision of the previous criticality evaluation of the operations and processes that are performed in the Aging Facility. It will also demonstrate and assure that the storage and aging operations to be performed in the Aging Facility meet the criticality safety design criteria in the *Project Design Criteria Document* (BSC 2005i, Section 4.9.2.2), and the nuclear criticality safety requirements described in the *SNF Aging System Description Document* (BSC 2005c, p. 3-24). The scope of this design calculation covers the systems and processes for aging commercial spent nuclear fuel (CSNF) and Department of Energy (DOE) SNF/High-Level Waste (HLW) prior to its placement in the final waste package (WP) (BSC 2005c, Section 1.1). The SNF aging system addresses thermal management for the repository and also provides flexibility for the repository by aging CSNF, and for management of DOE SNF, and DOE HLW until it can be processed by the available facilities (BSC 2005c, p. 1-2). The criticality evaluation of the Aging Facility features the following:

- Evaluate normal operations and Category 1 and 2 event sequences important to criticality, which are applicable to the Aging Facility. Selections of such sequences are guided by those from the *Categorization of Event Sequences for License Application* (BSC 2005e, Section 7.2).
- Further evaluate the design and criticality controls required for a storage/aging cask, referred to as site-specific cask, to accommodate commercial fuel outside the content specification in the Certificate of Compliance for the existing NRC-certified storage casks. In addition, evaluate the design requirements for the site-specific cask that will accommodate DOE SNF/HLW.
- Modify the document to conform to the *Desktop Information for Format of Calculations and Analyses and Treatment of Inputs and Assumptions* (EG-DSK-3003).
- Update calculations for DOE canisters inside the site-specific cask since the current design includes a Ni-Gd basket.
- Include a criticality evaluation for site specific canister in the site-specific cask.

The objective of this design calculation is to provide criticality safety results to support the preliminary design of the Aging Facility. The results presented in this document are limited to the current design and are consistent with the *SNF Aging System Description Document* (BSC 2005c).

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2.3 DESIGN OUTPUTS

This calculation/evaluation will be used as input for other calculations.

3. ASSUMPTIONS

3.1 ASSUMPTIONS REQUIRING VERIFICATION

None

3.2 ASSUMPTIONS NOT REQUIRING VERIFICATION

3.2.1 It is assumed that omitting the grid plates, spacers, and hardware in the fuel assembly tend to produce higher reactivity values for Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) fuel cask.

Rationale: The omission of grid plates, spacers, and hardware in the fuel assembly is conservative (General Atomics 1993, p. 6.4-1) from the perspective of increasing reactivity, which is shown in *Transportation Cask Receipt/Return Facility Criticality Safety Evaluations* (BSC 2005f, Table 5.2-1). It reduces the quantity of absorbers and increases the amount of moderator. For under-moderated systems such as light water reactor fuel assemblies, this increases the moderation, and therefore, the reactivity, provided the point of optimum moderation is not exceeded.

3.2.2 The site-specific cask for commercial fuel is assumed to be similar in design, other than the neutron poison loading/configuration, to the Holtec Multi Purpose Canister (MPC)-24 for PWR fuel and the MPC-68 for BWR fuel.

Rationale: Since the site-specific cask is still being developed, the criticality control features will be similar to the existing NRC-certified storage casks. The most reactive configurations with different fuel enrichments and various storage casks for the PWR and BWR fuels are shown in Tables 3.2-1 and 3.2-2, respectively. The PWR 15x15 fuel assembly in the MPC-24 provides the highest k_{eff}. The BWR 8x8C04 (identified generally as a GE8x8R) fuel assembly in the MPC-68 provides the highest k_{eff} except for the Siemens 11x11 fuel assembly in the FuelSolutions W74 Canister. The difference of reactivity is insignificant but GE fuel is the majority of BWR CSNF now stored at reactor sites (BSC 2005c, Table 4-3).

The aging cask subsystem shall accommodate metal site-specific casks for aging uncanistered CSNF with performance criteria to accept 32 PWR fuel assemblies and 68 BWR fuel assemblies (BSC 2005c, Section 3.1.2.3.3). The TN-32 is designed to store 32 Westinghouse 14x14 (standard or OFA), 15x15, and 17x17 (standard or OFA) or B&W 17x17 Mark BW spent fuel assemblies (Hunter 2004, p. 5.1-1). The TN-32 configuration is less reactive than that of MPC-24 as shown in Table 3.2-1. Similarly, the TN-68 configuration is less reactive than that of MPC-68 as shown in Table 3.2-2. Therefore, the calculations performed with the MPC-24 and MPC-68 are considered bounding of the TN-32 and TN-68 configurations.

Table 3.2-1 Most Reactive Configurations of PWR Fuels in Various Storage Casks

Cask/Canister Type	Fuel Type/Description	Enrichment, (wt% ^{235}U)	k_{eff}	σ	$k_{eff}+2\sigma$	Reference Source
FuelSolutions W21 Canister	Westinghouse 15x15 (bounding)	4.7	0.93931	0.00097	0.94125	Sisley, S.E. 2003a, Table 6.4-6
FuelSolutions W21 Canister	Westinghouse 17x17 (bounding)	4.7	0.93598	0.00097	0.93792	Sisley, S.E. 2003a, Table 6.4-6
FuelSolutions W21 Canister	Westinghouse 17x17 B (bounding)	4.6	0.93769	0.001	0.93969	Sisley, S.E. 2003a, Table 6.4-6
FuelSolutions W21 Canister	Westinghouse 17x17 A, Multiple Package Array, Normal Operating Condition	4.6	0.93795	0.00104	0.94003	Sisley, S.E. 2003a, Table 6.4-9
FuelSolutions W21 Canister	Westinghouse 17x17 OFA, Single Package, Normal Operating Condition	4.6	0.93734	0.00099	0.93932	Sisley, S.E. 2003a, Table 6.4-11
MPC-24	PWR 15x15[a] Fuel Assembly without Soluble Boron	4.1	0.9395	N/A	N/A	Holtec International 2002, Section 6.2.2.1
TN-32	Westinghouse and B&W Fuel Assembly	4.05	0.9315	0.0009	0.9333	Hunter 2004, p. 6.4-3
NAC-MPC 26-Assembly Basket	Westinghouse Vantage 5H	3.93	0.9219	N/A	N/A	Thompson, T.C. 2003, Section 6.4.2.2.2
NAC-MPC 24-Assembly Basket	Westinghouse Vantage 5H	4.61	0.9313	N/A	N/A	Thompson, T.C. 2003, Section 6.4.2.2.2

[a]The 17x17A01 assembly (otherwise known as a Westinghouse 17x17 OFA) has a similar reactivity and was selected for the criticality evaluation as a reference PWR assembly.

Table 3.2-2 Most Reactive Configurations of BWR Fuels in Various Storage Casks

Cask/Canister Type	Fuel Type/Description	Enrichment, (wt% ^{235}U)	k_{eff}	σ	$k_{eff}+2\sigma$	Reference Source
FuelSolutions W74 Canister	Siemens 11x11, Pattern 3 Base Case	4.1	0.93831	0.00088	0.94007	Sisley, S.E. 2003b, Table 6.4-2
MPC-68	BWR 8x8C04 Fuel Assembly	4.2	0.9334	0.0007	0.9348	Holtec International 2002, Table 6.2.3
TN-68	GE9 8x8 Fuel Assembly	3.7	0.9235	0.0016	0.9267	Hunter 2002, Table 6.4-1
TN-68	GE12 10x10 Fuel Assembly	3.7	0.927	0.0015	0.9300	Hunter 2002, Table 6.4-1

3.2.3 The site-specific cask for DOE canisters is assumed to have an inside diameter similar to storage casks already NRC-certified for commercial fuel.

Rationale: Since both commercial and DOE site-specific casks will be stored on the aging pad, it would be appropriate if each site-specific cask was similar in size for uniformity, ease in design, and ease in handling the storage casks. The internal configuration of vertical overpacks will accommodate the dimensions and characteristics of disposable canisters now being developed by DOE (BSC 2005c, Section 4.1.1.3).

3.2.4 It is assumed the overpack thickness of the site-specific cask containing DOE canisters is 15 inches and is made out of concrete.

Rationale: Since the site-specific cask for DOE canisters is still being developed, the overpack thickness and material will be similar to the existing NRC-certified storage casks. Site-specific concrete overpacks for DPCs, DOE SNF canisters, or DOE HLW canisters are considered among five types of vertical site-specific casks for the aging cask subsystem (BSC 2005c, Section 2.3.3). It is expected that a slight variation in the thickness will not affect the reactivity of the system.

3.2.5 It is assumed that the isotopic concentrations generated with the Babcock & Wilcox (B&W) 15x15 assembly type for PWR fuel (BSC 2003b) and the General Electric (GE) 7x7 assembly type for BWR fuel (Wimmer 2004) used in the burnup-credit calculation are conservative for the Westinghouse 17x17 OFA PWR and GE 8x8 BWR spent fuel.

Rationale: The B&W 15x15 fuel assembly has a large initial fuel loading of approximately 464 kgU/assembly (DOE 1987, p. 2A-31). The initial loading of Westinghouse 17x17 OFA is around 426 kgU/assembly (DOE 1987, p. 2A-349), while the fuel loading per unit height is about the same for both fuel assembly types (the active fuel length is 144 in. for the Westinghouse 17x17 OFA (DOE 1987, p. 2A-351) and 141.8 in. for the B&W 15x15 fuel assembly (DOE 1987, p. 2A-33)). The Westinghouse 17x17 OFA contains 264 fuel rods (DOE 1987, p. 2A-351) and the B&W 15x15 fuel assembly contains 208 fuel rods (DOE 1987, p. 2A-33). It indicates that the fuel loading per fuel rod is larger for the B&W 15x15 fuel assembly. Further, the total surface area of the fuel rods for the B&W 15x15 fuel assembly (based on fuel pellet diameter per DOE 1987, p. 2A-33) is approximately 10% less than that of the fuel rods for the Westinghouse 17x17 OFA (DOE 1987, p. 2A-351). The smaller surface-to-volume ratio results in greater self-shielding and higher fissile isotope content with burnup. Consequently, the isotopic concentrations generated with the B&W 15x15 fuel assembly are conservative relative to the Westinghouse 17x17 OFA for given fuel enrichment and burnup. The same reasoning applies to the GE 7x7 fuel assembly versus the GE 8x8 fuel assembly (Wimmer 2004 pp. 8-9).

3.2.6 The internal basket structure and configuration of the MPC-24 and MPC-68 is assumed to be the same (for the purpose of the burnup-credit evaluation) when loading B&W

15x15 and GE 7x7 fuel assemblies, respectively, as compared to the Westinghouse 17x17 OFA and GE 8x8 fuel assembly.

Rationale: This assumption was used for the burnup-credit evaluation where the intent was to evaluate other fuel assemblies to compare reactivity to Westinghouse 17x17 OFA and GE 8x8 fuel assembly (for fresh fuel) when applying burnup-credit. For a one-to-one comparison of the PWR and BWR fuel assemblies, it is reasonable to maintain the same basket structure of the MPC-24 and MPC-68.

4. METHODOLOGY

4.1 QUALITY ASSURANCE

This calculation is prepared in accordance with EG-PRO-3DP-G04B-00037, *Calculations and Analyses.* The cask transfer, aging pad (except support structures), and aging cask subsystems of the SNF Aging System have been classified as important to safety in the *Q-list* (BSC 2005a, Table A-1). This calculation provides the criticality safety results to support the design of the Aging Facility. Therefore, this design calculation is subject to the requirements of the *Quality Assurance Requirements and Description* (DOE 2006).

Electronic management of information generated from these calculations is controlled in accordance with IT-PRO-0009, *Control of the Electronic Management of Information.* The computer input and output files generated from this calculation are stored on a Compact Disc (CD), and included as an attachment to this document (Attachment B).

4.2 USE OF SOFTWARE

4.2.1 MCNP

The MCNP code (CRWMS M&O 1998a) was used to calculate the effective neutron multiplication factor, k_{eff}, for all systems presented in this report. The software specifications are as follows:

- Program Name: MCNP (CRWMS M&O 1998a)
- Version/Revision Number: Version 4B2LV
- Status/Operating System: Qualified/HP-UX B.10.20
- Software Tracking Number: 30033 V4B2LV
- Computer Type: HP 9000 Series Workstations
- CPU Number: 700887.

The input and output files for the various MCNP calculations are contained on a CD (Attachment B) and the files are listed in Attachment A.

The MCNP software used was: (1) appropriate for the criticality (k_{eff}) calculations (Briesmeister 1997), (2) used only within the range of validation as documented in CRWMS M&O (1998b, Section 3.1), and (3) obtained from Software Configuration Management in accordance with appropriate procedures.

4.2.2 MICROSOFT EXCEL 97 SR-2

- Title: Excel
- Version/Revision Number: Microsoft® Excel 97 SR-2
- This version is installed on a PC running Microsoft Windows 2000 with CPU number 152382.

The files for the various Excel calculations are contained on a CD (Attachment B) and the files are listed in Attachment A.

The Excel software was used to calculate weight percent of each component (i.e., ^{235}U, ^{238}U and O) in fresh UO_2 as a function of initial enrichment and to determine Boral loading and thicknesses. Further, the Excel software was also used to calculate weight fractions as well as to illustrate results in Sections 6.2 and 7. The calculations performed with Excel can be reproduced and checked by hand. Excel is exempt from qualification per Section 2.1.6 of LP-SI.11Q-BSC, *Software Management*.

4.3 CRITICALITY SAFETY ANALYSIS

The criticality safety calculations presented in this document evaluate the array configuration of the storage/aging casks on the aging pads in the Aging Facility to assure it meets the criticality safety requirements under normal conditions as well as for Category 1 and 2 events. Moderator conditions are varied to find the most reactive configuration. The poison (Boral) areal density used in this calculation for the commercial SNF aging casks is varied to accommodate a fuel in a sensitivity analysis to evaluate the capability of the casks to accommodate SNF with up to 5.0 wt% initial enrichment. The process and methodology for criticality safety analysis given in the *Preclosure Criticality Analysis Process Report* (BSC 2004e, Section 2.2.7) will be implemented in these calculations. The following method will be pursued for each waste form and cask/canister configuration (BSC 2004e, Section 2.2.7):

- The design basis for the Aging Facility relies on the most reactive fuel assemblies.
- The effective neutron multiplication factor (k_{eff}), including all biases and uncertainties at a 95 percent confidence level, will not exceed 0.95 under all credible normal, off-normal, and accident conditions. The latter two conditions are inclusive of Category 1 and 2 event sequences.
- Conservative dimensional variables will be used (e.g., assembly pitch, manufacturing tolerances for assemblies, etc.) in order to maximize reactivity.
- Conservative assumptions will also be used regarding fuel materials including not taking credit for burnable poisons, or ^{234}U and ^{236}U, fission product or transuranic absorbers in fuel, and use of the most reactive fuel stack density.
- Credit can only be taken for up to 75 % of the neutron absorbing material for criticality control (e.g., grid plates).
- Consideration will be given to extensions of important data variables such as quantity and density of moderator.
- Consideration will be given to potential reflectors during transfer and placement.

These calculations use the qualified software MCNP (Briesmeister 1997 and CRWMS M&O 1998a). MCNP is a three-dimensional Monte Carlo particle transport code with the capability to calculate eigenvalues for neutron systems. The Nuclear Regulatory Commission (NRC) accepts MCNP in NUREG-1567 (NRC 2000, p. 8-10) for criticality calculations.

5. LIST OF ATTACHMENTS

		Number of Pages
Attachment A	Listing of Computer Files	7
Attachment B	One Compact Disk Containing All Files Listed in Attachment A	1
Attachment C	Drawing of Aging Area General Arrangement Plan and Sections	2

6. CALCULATION

All technical product inputs and sources of the inputs used in the development of this calculation are documented in this section. Attachment C features a preliminary drawing (BSC 2005l) of the Aging Facility as the date of this calculation. The purpose of this drawing is to show the aging area general arrangement plan and sections.

6.1 CALCULATIONAL INPUTS

6.1.1 Design Requirements and Criteria

The design criteria for criticality safety analysis provided in Section 4.9.2.2 of the *Project Design Criteria Document* (BSC 2005i) are used in these calculations. The pertinent criteria for Aging Facility criticality include the following (BSC 2005i, Section 4.9.2.2):

- Under all normal conditions and Category 1 and Category 2 event sequences, the calculated multiplication factor, k_{eff}, at the upper limit of a two-sided 95 percent confidence interval, shall not exceed the upper subcritical limit (a limiting value of k_{eff} that accounts for biases and uncertainties, and an administrative margin to ensure subcriticality). This criterion is a consequence of 10 CFR 63.112(e)(6), which requires that the repository provide the means to prevent and control criticality.

- For fixed-neutron absorbers used for criticality control such as grid plates or inserts, no more than 75 percent credit of the neutron absorber content shall be used for preclosure criticality analyses, unless comprehensive fabrication acceptance tests verify that the presence and uniformity of the neutron absorber are more effective.

The pertinent safety requirements applicable to Aging Facility criticality as given in the *SNF Aging System Description Document* (BSC 2005c, Section 3.1.1.1) are as follows:

- The site-specific canister shall be designed to ensure nuclear criticality safety with optimum moderation and the most reactive waste forms. Criticality safety will be maintained despite geometric rearrangements due to a drop or other handling incidents.

- Site-specific casks shall be designed to ensure nuclear criticality safety with optimum moderation and the most reactive waste forms. Criticality safety will be maintained despite any geometric rearrangements due to a drop or other handling incident (BSC 2005c, Section 3.1.1.1.33).

The sealed site-specific canister in conjunction with the overpack shall provide criticality control with the following Performance Acceptance Criteria (BSC 2005c, Section 3.1.1.3.6):

1. The methodology defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003) shall be used to demonstrate acceptable criticality control for the overpacks in conjunction with the site-specific canister. [PRD-013/T-016, PRD-013/T-038, PRD-002/T-012)].

2. The site-specific disposable canister in conjunction with its overpack or waste package shall meet the following criteria inferred from PDC 4.9.2.2:

   a. Shall be designed to meet nuclear criticality safety during preclosure such that the calculated effective neutron multiplication factor, k_{eff}, including all biases and uncertainty at a 95 percent confidence level, shall not exceed 0.95 under all normal conditions, and Category 1 and Category 2 event sequences. [PDC 4.9.2.2.1]

   b. Shall be redesigned if the total probability of criticality is greater than or equal to one over the 10,000-year regulatory period. [PDC 4.9.2.2.2]

   c. Shall be sealed to prevent moderator from entering the site-specific canister internals. [PDC 4.9.2.2.3(2)]

   d. Shall be designed such that adequate controls and procedures can be effectively implemented to ensure that neutron absorber materials are inserted into the site-specific canister basket as required. [PDC 4.9.2.2.5]

3. Criticality control analysis shall consider the effects of credible preclosure event sequences including, but not necessarily limited to:

   a. Drops of the site-specific disposable canister by itself or while contained within an overpack.

   b. Design basis ground motion DBGM-2.

   c. Credible Fires.

   d. Credible missile impacts.

The functional requirement 3.2.3.1 of the *SNF Aging System Description Document* (BSC 2005c, p. 3-24) states that "the aging system shall be designed and operated to prevent any credible criticality event from occurring [10 CFR 63.112(e)(6)]". More specifically, the aging pads will be equipped with cask anchoring systems, if required, for the vertical casks to comply with seismic and missile requirements at the pads. Flood drainage channels will surround the aging pads that are sized to carry away water from a design basis flood. The aging pad will require only minimal fire protection features to provide a defense in depth approach to the fire protection of the facility. In addition, these casks must also remain stable against tipover, drop, or slapdown from imposed seismic loads resulting from DBGMs (BSC 2005c, Section 4.1.1.1).

No Category 1 and Category 2 event sequences applicable to the Aging Facility have been identified in the *Categorization of Event Sequences for License Application* document (BSC 2005e, Section 7). However, for defense-in-depth, a drop or slap down scenario causing rearrangement of the fuel assemblies in the aging casks has been evaluated in Section 6.2.4.

6.1.2 Storage/Aging Cask Selection

The aging system includes vertical and horizontal cask systems that contain SNF and the heavy equipment components needed to transfer the cask configurations (BSC 2005c, p. 4-1). Vertical commercially available NRC-licensed storage systems include TN-32, TN-68, BNFL FuelSolutions Storage System, Holtec MPC-24, MPC-68, NAC-MPC 24 and 26 assembly baskets (Tables 3.2-1 and 3.2-2). As background information, the horizontal systems available for SNF aging include different NUHOMS designs. Only the NUHOMS-24PT1 and NUHOMS-61BT are currently licensed for both transport and storage (BSC 2005c, Table 4-6). Both vertical and horizontal storage types use a dual-purpose canister (DPC) to contain fuel assemblies in a basket. Criticality control features for the storage systems typically use Boral to provide fixed poison for neutron absorption.

The fuel basket in the DPC has met the criticality safety requirements in 10 CFR 72 for storage (10 CFR 72.2) as well as 10 CFR 71 for transportation (10 CFR 71.0). The storage systems listed above have previously been certified to this standard. For storage, 10 CFR 72 requires a detailed safety analysis that addresses criticality safety in particular (10 CFR 72.124). Credit for criticality analyses performed for the storage and transportation conditions should cover all repository conditions including normal operations, and Category 1 and 2 event sequences (10 CFR 72.122, 10 CFR 72.236(c)). With this credit, additional criticality evaluations are required only for site-specific conditions that may not be covered under 10 CFR 72 (such as taking credit for only 75% of fixed neutron absorbers) or for conditions outside those listed in the Certificate of Compliance (such as a higher fuel enrichment).

The vertical and horizontal systems use nearly identical casks and overpacks. The results in Sections 7.1 (PWR fuel) and 7.2 (BWR fuel) consistently demonstrate that the conditions outside the overpack (e.g., spacing, moderation, reflection) have no discernable impact on the reactivity of the cask. This indicates that the casks are neutronically isolated and consequently the cask orientation (vertical versus horizontal) will not matter. An evaluation of one type of storage/aging rack will be sufficient to demonstrate the effect of site-specific conditions such as mist. This criticality evaluation focuses on the vertical cask system only.

A representative vertical cask per Assumption 3.2.2 (Table 3.2-1 for PWR and Table 3.2-2 for BWR) is selected here for criticality calculations to demonstrate compliance with the criticality safety requirements. The selected cask is HI-STORM 100, as this system is currently qualified for high seismic requirements (similar to those of the YMP) to ensure that the YMP seismic spectrum will be enveloped (Cogema 2004, p.5). In addition, the HI-STORM 100 cask is the bounding configuration among the PWR and BWR casks in accordance with Tables 3.2-1 and 3.2-2.

The fuel basket designs used for this criticality evaluation were a 24 PWR assembly basket and a 68 BWR assembly basket as specified in the *Final Safety and Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System*) (Holtec International 2002).

6.1.3 Most Reactive Fuel Selection

In accordance with the requirements given in *Preclosure Criticality Analysis Process Report* (BSC 2004e, Section 2.2.7), the criticality safety evaluation is based on the most reactive fuel assemblies. An evaluation to determine the most reactive commercial fuel assemblies was performed in the *Final Safety and Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System*). Based upon this analysis, the Westinghouse 17x17 OFA was selected for PWR fuel (Holtec International 2002, Section 6.2-2) and the GE 8x8 array was selected for the BWR fuel (Holtec International 2002, Section 6.2-3).

The DOE fuel types have been categorized into nine fuel groups (Mecham, D.C. 2004, Section 4.2.4.1) for evaluation, i.e.,

1. Uranium Metal fuels (N-Reactor)
2. Uranium-Zirconium/Uranium-Molybdenum fuels (Enrico Fermi Liquid Metal Reactor)
3. Uranium Oxide fuels (high enriched uranium - Shippingport PWR)
4. Uranium Oxide fuels (low enriched uranium - Three Mile Island (TMI)-2 PWR)
5. Uranium-Aluminum fuels (Advanced Test Reactor)
6. Uranium/Thorium/Plutonium Carbide fuels (Ft. St. Vrain Gas Cooled Reactor)
7. Mixed Oxide fuels (Fast Flux Test Facility (FFTF) Reactor)
8. Uranium/Thorium Oxide fuels (Shippingport Light Water Breeder Reactor (LWBR))
9. Uranium-Zirconium-Hydride fuels (Training Research Isotopes General Atomics (TRIGA)).

Only Fermi, Fort St. Vrain and FFTF are chosen for overpack calculations as shown in Section 7.4, since they are among the most reactive fuel types (BSC 2005d, Table 6.2-1).

6.1.4 Upper Subcritical Limit

The definition of upper subcritical limit (USL) from (BSC 2004e, Section 3.5) is:

$$k_S + \Delta k_S \leq USL \qquad (1)$$

where k_S is the MCNP calculated value for the system, Δk_S is an allowance for (a) statistical or convergence uncertainties, or both in the computation of k_S, (b) material and fabrication tolerances, and (c) uncertainties due to the geometric or material representations used in the computational method.

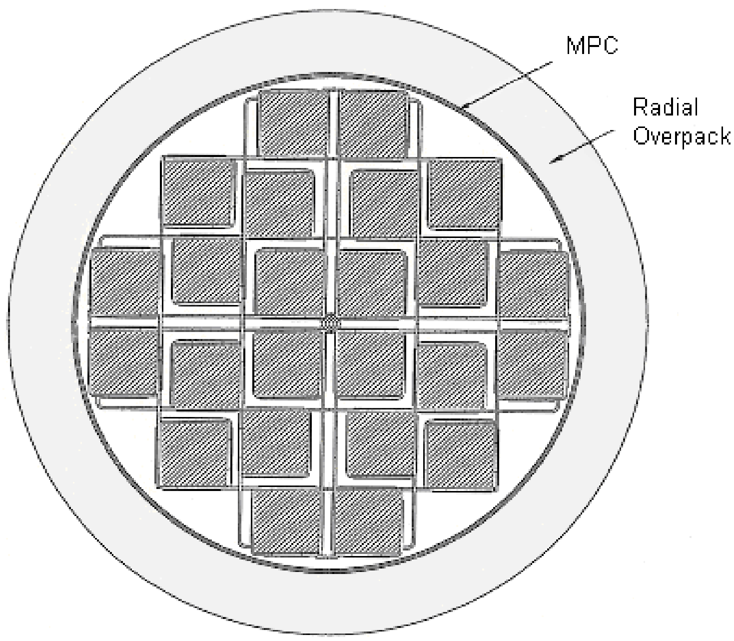
For commercial spent nuclear fuel, the USL is calculated to be 0.9472, which provides a margin of 0.0028 (0.0021 + 0.0007) to the criticality safety criterion (0.95). Applicable code bias for similar fuel type and enrichment range of this analysis has been estimated to be 0.0021 (value increased by truncation) with a standard deviation of $\pm$ 0.0007 (Holtec International 2002, Appendix 6 A-2). Holtec design is used for CSNF evaluation. For site-specific canisters, the USL is calculated to be 0.9394 (BSC 2005g, p. 31), which provides a 0.05 administrative margin from the value (Framatome ANP 2003, Table11) based on laboratory critical experiments. For DOE fuel canisters, the USL is calculated to be 0.9131, which provides a 0.05 administrative margin from the established critical limit of 0.9631 as described in the *Canister Handling Facility Criticality Safety Calculations* (BSC 2005d, Section 5.1.3)

6.1.5 Storage/Aging Cask Calculation Inputs

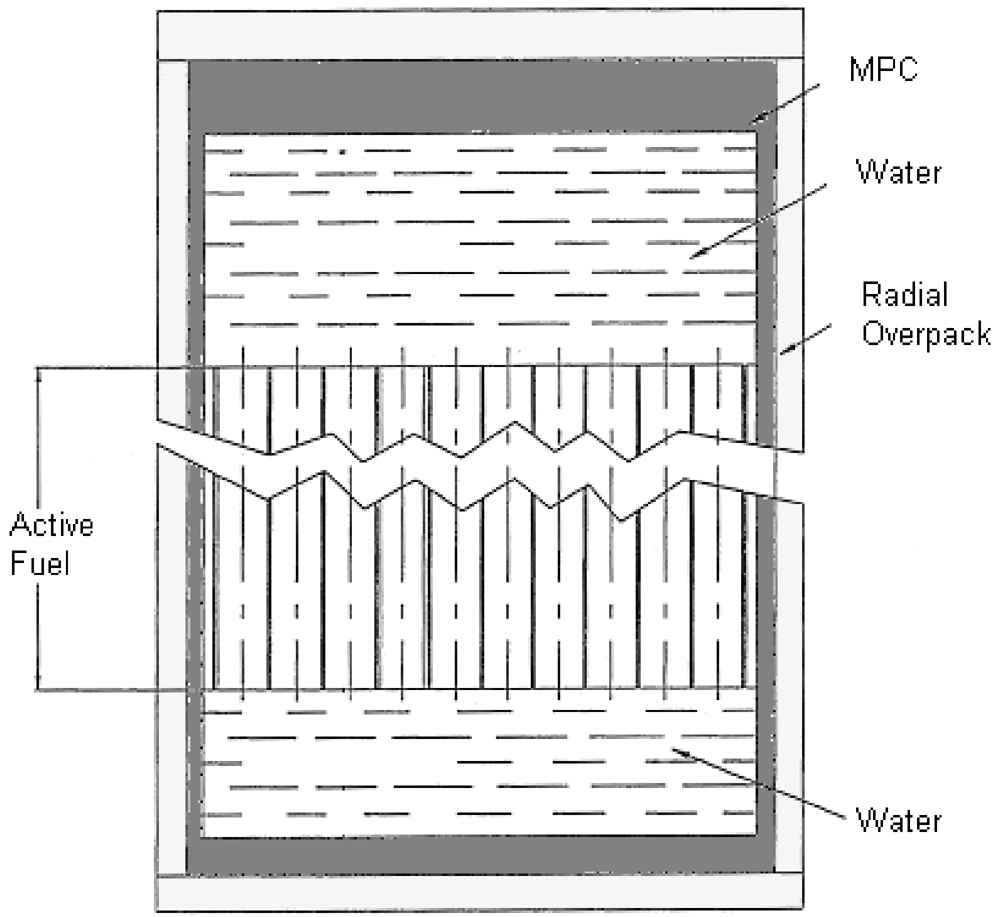
The HI-STORM 100 storage casks in the Aging Facility were simulated in accordance with the method used in *Final Safety and Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System)* (Holtec International 2002, Section 6.3). Radially reflective boundaries were added to simulate an infinite array of storage/aging casks. Physical inputs for the storage/aging casks are described in the following subsections.

6.1.5.1 PWR Configuration and Physical Dimensions

The MPC-24 for PWR fuel consists of a concrete cask with steel shells and an interior 24 PWR assembly basket. Figure 6.1-1 displays the planar cross-section of the MPC-24 inside the overpack and Figure 6.1-2 presents the axial view. Note that the axial reflection is considered by adding a 30 cm water region above and below the storage/aging cask. However, it will be shown later that the conditions outside the overpack have no discernable impact on the reactivity of the cask.



NOTE: Not to scale. (Source: Holtec International 2002, Figure 6.3-4)
Figure 6.1-1 Radial View of the MPC-24 PWR Fuel Storage Cask



NOTE: Not to scale. Also, details of the overpack geometry are not shown in this figure.
(Source: Holtec International 2002, Figure 6.3-7)

Figure 6.1-2 Axial View of the MPC-24 PWR Fuel Storage Cask

The PWR storage rack basket cells were simulated featuring SS walls with a Boral panel situated on each side (Holtec International 2002, Figure 6.3.1). In the MCNP calculation, the Boral panel features various ^{10}B loading and panel thicknesses. The Boral thickness, T, is related to the areal density by the expression:

$$T = \frac{M}{S_a} \times \frac{N_A}{M_a} \times \frac{1}{A} \tag{2}$$

where

M = weight (g) of ^{10}B
S_a = surface area (Boral areal densities ranges from 0.020 g $^{10}B/cm^2$ to 0.080 g $^{10}B/cm^2$)
M/S_a = areal density
N_A = Avogadro's constant (6.023E+23 atoms/mole (Parrington et. al. 1996))
M_a = ^{10}B atomic weight (10.0129371 g/mole (Parrington et. al. 1996))
A = ^{10}B atom density
T = thickness (cm)

It should also be mentioned that equation 2 is derived from the definition of atom density, A, as described below:

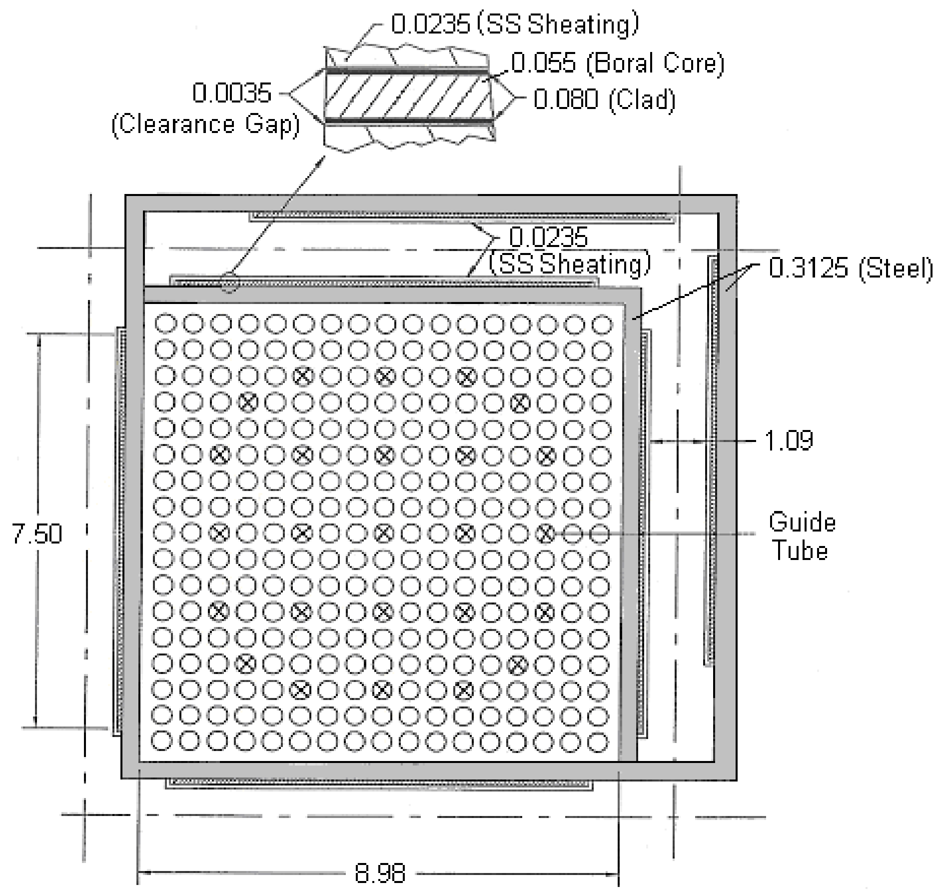
$$A = \frac{N_a}{V} = \frac{N_m \times N_A}{V} = \frac{M}{M_a} \times \frac{N_A}{V} = \frac{M}{S_a \times T} \times \frac{N_A}{M_a} \tag{3}$$

where

N_a = number of atoms
N_m = number of moles
V = volume

The selection of Boral thicknesses and ^{10}B loadings can be found in Excel file *boral.xls*. Note that the calculations of the content of ^{10}B in B are based on its atomic weight rather than the weight fraction. This has no impact on k_{eff} as demonstrated in Section 7.3. It should be emphasized that selection of Boral panels is based on a specific weight percent of the materials, B_4C and Al and a desired thickness. These two parameters ultimately govern the ^{10}B loading (see Section 7.3 for further discussion).

The storage rack basket cells contain Westinghouse 17x17 Optimized Fuel Assembly (OFA) assemblies, since this is the most reactive PWR fuel (Section 6.1.3). Figure 6.1-3 displays the storage rack basket cell with the Westinghouse 17x17 OFA. Table 6.1-1 features the radial dimensions of the storage rack and cell geometry while Table 6.1-2 shows the axial dimensions. Table 6.1-3 displays the specifications of the PWR fuel assembly. Note that only the active fuel region was included in the simulation (Assumption 3.2.1).



NOTE: Dimensions are in inches.
(Source: Holtec International 2002, Figure 6.3-1)

Figure 6.1-3 PWR Storage Rack Basket Cell Containing W 17x17 OFA

Table 6.1-1 Radial Dimensions of the MPC-24, Overpack, and Cell Geometry

Component	cm	in	Reference
SS overpack outer shell thickness	1.905	0.75	Holtec International 2002, Figure 5.3.10
Concrete overpack thickness	67.945	26.75	Holtec International 2002, Figure 6.3.4
Concrete overpack, o.d.	332.74	131	Holtec International 2002, Figure 5.3.10
SS overpack inner shell	196.85	77.5	Holtec International 2002, Figure 5.3.10
Cavity (water), o.d.	190.5	75	Holtec International 2002, Figure 5.3.10
MPC storage basket, o.d.	173.6725	68.375	Holtec International 2002, Figure 6.3.4
MPC storage basket, i.d.	171.1325	67.375	Holtec International 2002, Figure 6.3.4
Center column	6.985	2.75	Holtec International 2002, Drawing 3926 (Sheet 2)
Assembly inside dimension	22.8092	8.98	Holtec International 2002, Figure 6.3.1 & Table 6.3.3
Cell pitch	27.7012	10.906	Holtec International 2002, Table 6.3.3 & Drawing 3926 (Sheet 3)
Flux trap	2.7686	1.09	Holtec International 2002, Figure 6.3.1 & Table 6.3.3
Cell wall thickness (SS)	0.79375	0.3125	Holtec International 2002, Figure 6.3.1
SS sheathing	0.05969	0.0235	Holtec International 2002, Figure 6.3.1
Boral thickness [a]	0.1397	0.055	Holtec International 2002, Figure 6.3.1
Al thickness (Clad)	0.0254	0.01	Holtec International 2002, Figure 6.3.1
Boral width - wide	19.05	7.5	Holtec International 2002, Figure 6.3.1
Boral width – narrow [b]	15.875	6.25	Holtec International 2002, Drawing 3926 (Sheet 2)
Boral clearance gap	0.00889	0.0035	Holtec International 2002, Figure 6.3.1

[a] Boral thicknesses (e.g., 0.2057 cm) for variations in ^{10}B loading can be found in Excel file *boral.xls*
[b] The periphery Boral panels have reduced width.

Table 6.1-2 Axial Dimensions of the MPC-24, Overpack, and Cell Geometry

Component	cm	in	Reference
Lower water thickness (below active fuel region)	10.16	4	Holtec International 2002, Figure 6.3.7
Upper water thickness (above active fuel region)	15.24	6	Holtec International 2002, Figure 6.3.7
MPC baseplate	6.35	2.5	Holtec International 2002, Drawing 3923 (Sheet 2)
MPC lid	24.13	9.5	Holtec International 2002, Drawing 3923 (Sheet 2)
Bottom overpack SS plate thickness (top layer)	12.7	5	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Bottom overpack concrete plate thickness	43.18	17	Holtec International 2002, Figure 6.3.7
Bottom overpack SS plate thickness (bottom layer)	5.08	2	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top overpack SS plate thickness (top layer)	10.16	4	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top overpack concrete plate thickness	26.67	10.5	Holtec International 2002, Figure 6.3.7
Top overpack SS plate thickness (bottom layer)	3.175	1.25	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top gap (between MPC and overpack)	3.81	1.5	Approximated from Holtec International 2002, Drawings 1495 (Sheet 2) and 3923 (Sheet 3)

Table 6.1-3 Specifications of the PWR W 17 x17 OFA

Parameter	cm	in	Reference [b]
Rod pitch	1.2598	0.496	Holtec International 2002, p. 2.1-11
Active fuel length	381	150.000	Holtec International 2002, p. 2.1-11
Cladding outside diameter	0.9144	0.360	Holtec International 2002, p. 2.1-11
Cladding inside diameter	0.8002	0.315	Holtec International 2002, p. 2.1-11
Pellet outside diameter	0.784352	0.309	Holtec International 2002, p. 2.1-11
Guide/instrument tube outside diameter	1.204	0.474	Sanders and Wagner 2002, p.8
Guide/instrument tube thickness	0.04064	0.016	Holtec International 2002, p. 2.1-11
Array size	17x17		Sanders and Wagner 2002, p.8
Number of fuel rods	264		Sanders and Wagner 2002, p.8
Number of guide/instrument tubes [a]	25		Sanders and Wagner 2002, p.8

[a] Locations of guide tubes shown in Figure 6.1-3 can be seen in Wagner and Parks 2000, p. 8
[b] Holtec International 2002, p. 6.2-37 demonstrates that the dimensions cited are conservative

6.1.5.2 PWR Material Compositions

The calculations were performed with either the isotopic compositions given in weight percent (wt%) or atom densities (atoms/barn-cm), depending on the source of the input. Table 6.1-4 displays the relevant materials used for the storage/aging cask and the Westinghouse 17x17 OFA PWR fuel.

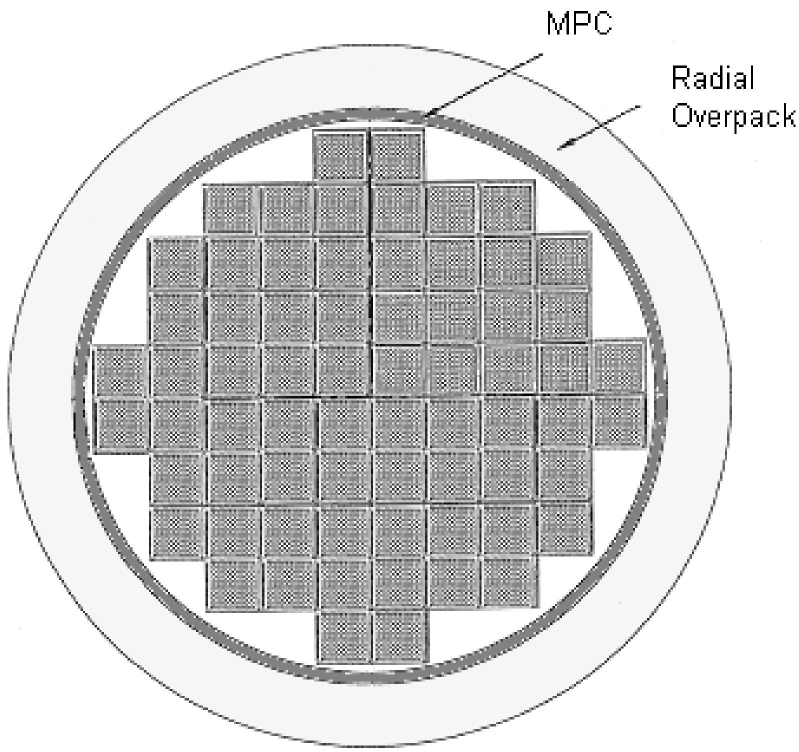
Table 6.1-4 Material Properties for the Storage Cask and PWR Fuel

Material	Density (g/cm^3)	Element	Weight Fraction or Weight Percent (wt%)	Atom Fraction or Atom Density (atoms/barn-cm)	Reference/ Remark
H_2O	1.0 [a]	H O	N/A	fraction - 0.6667 fraction - 0.3333	Holtec International 2002, p. 6.3-12
SS304 (vessel & cell wall)	7.84	Cr Mn Fe Ni	N/A	1.761E-02 1.761E-03 5.977E-02 8.239E-03	Holtec International 2002, p. 6.3-13
Concrete	2.35	H O Na Al Si K Ca Fe	fraction-6.00E-03 fraction-5.00E-01 fraction-1.70E-02 fraction-4.80E-03 fraction-3.15E-01 fraction-1.90E-02 fraction-8.30E-02 fraction-1.20E-02	N/A	Holtec International 2002, p. 6.3-14
Carbon Steel	7.8212	Fe C	N/A	8.35E-02 3.93E-03	Hunter 2004, Table 5.3-1
Air	1.204E-03	O N Ar C	fraction-2.32E-01 fraction-7.55E-01 fraction-1.29E-02 fraction-1.00E-04	N/A	Weast 1985, p. F-156; at_dens.xls in Attachment B
Al (Boral panel)	2.7	Al	N/A	0.06026	Holtec International 2002, p. 6.3-13
Boral (0.02 g $^{10}B/cm^2$) [b, c]	2.66	B-10 B-11 C Al	5.443E-02 2.414E-01 8.210E-02 6.222E-01	N/A	Holtec International 2002, p. 6.3-9
UO_2 – (fuel) 4.00 % enriched	10.522	U-235 U-238 O-16	3.526 84.62 11.85	N/A	Holtec International 2002, p. 6.3-9
UO_2 – (fuel) 4.50 % enriched	10.522	U-235 U-238 O-16	3.9667 [d] 84.1831 [d] 11.8502 [d]	N/A	-----------
UO_2 – (fuel) 5.00 % enriched	10.522	U-235 U-238 O-16	4.408 83.74 11.85	N/A	Holtec International 2002, p. 6.3-9
Zr (Cladding)	6.55	Zr	100	N/A	Holtec International 2002, p. 6.3-12

[a] The moderator density was varied between 0.0 – 1.0 g/cm^3 to study moderator density variations in Section 7
[b] Calculations for varied Boral loading can be found in Excel file *boral.xls*
[c] The ^{10}B loading of 0.020 g/cm^2 is 75 % of the minimum loading 0.0267 g/cm^2 (Holtec International 2002, p. 6.2-3)
[d] Calculations can be found in Excel file *fuelcomp.xls* (source for the atomic weight: Parrington et. al., 1996)

6.1.5.3 BWR MPC-68 Configuration and Physical Dimensions

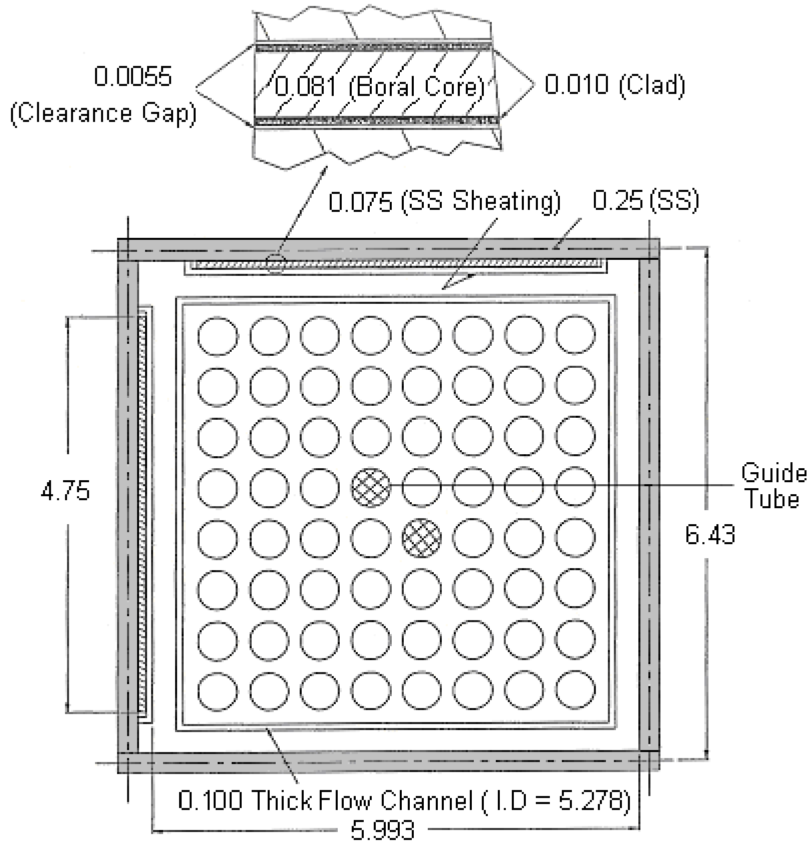
The MPC-68 for BWR fuel consists of a concrete cask with steel shells and an interior 68 BWR assembly basket. Figure 6.1-4 displays the planar cross-section of the MPC-68 cask and Figure 6.1-2 presents the axial view (it is the same as for the MPC-24).



NOTE: Not to scale.
(Source: Holtec International 2002, Figure 6.3-6)

Figure 6.1-4 Radial View of the MPC-68 BWR Fuel Storage Cask

The storage rack basket cells contain GE 8x8 standard assemblies, since this is the most reactive BWR fuel (Section 6.1.3). Figure 6.1-5 displays the storage rack basket cell with the GE 8x8 assembly. Table 6.1-5 features the radial dimensions of the storage rack and cell geometry while Table 6.1-6 shows the axial dimensions. Table 6.1-7 displays the specifications of the BWR fuel assembly. Note that only the active fuel region was included in the simulation (Assumption 3.2.1).



NOTE: Dimensions are in inches.
(Source: Holtec International 2002, Figure 6.3-3)

Figure 6.1-5 BWR Storage Rack Basket Cell Containing GE 8x8 Assembly

Table 6.1-5 Radial Dimensions of the MPC-68, Overpack, and Cell Geometry

Component	cm	in	Reference
SS overpack outer shell thickness	1.905	0.750	Holtec International 2002, Figure 5.3.10
Concrete overpack thickness	67.945	26.750	Holtec International 2002, Figure 6.3.4
Concrete overpack, o.d.	332.74	131.000	Holtec International 2002, Figure 5.3.10
SS overpack inner shell, o.d.	196.85	77.500	Holtec International 2002, Figure 5.3.10
Cavity (water), o.d.	190.5	75.000	Holtec International 2002, Figure 5.3.10
MPC storage basket, o.d.	173.6725	68.375	Holtec International 2002, Figure 6.3.6
MPC storage basket, i.d.	171.1325	67.375	Holtec International 2002, Figure 6.3.6
Cell box inside dimension	15.2222	5.993	Holtec International 2002, Figure 6.3.3 & Table 6.3.3
Cell pitch	16.3322	6.430	Holtec International 2002, Table 6.3.3 & Figure 6.3.3
Cell plate thickness	0.635	0.250	Holtec International 2002, Figure 6.3.3 & Table 6.3.3
SS sheathing	0.1905	0.075	Holtec International 2002, Figure 6.3.3
Boral thickness	0.2057	0.081	Holtec International 2002, Figure 6.3.3
Al thickness (Clad)	0.0254	0.010	Holtec International 2002, Figure 6.3.3
Boral width	12.065	4.750	Holtec International 2002, Figure 6.3.3
Boral clearance gap	0.01397	0.0055	Holtec International 2002, Figure 6.3.3

Table 6.1-6 Axial Dimensions of the MPC-68, Overpack, and Cell Geometry

Component	cm	in	Reference
Lower water thickness (below active fuel region)	18.542	7.300	Holtec International 2002, Figure 6.3.7
Upper water thickness (above active fuel region)	21.4884	8.460	Holtec International 2002, Figure 6.3.7
MPC baseplate	6.35	2.500	Holtec International 2002, Drawing 3923 (Sheet 2)
MPC lid	24.13	9.500	Holtec International 2002, Drawing 3923 (Sheet 2)
Bottom overpack SS plate thickness (top layer)	12.7	5.000	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Bottom overpack concrete plate thickness	43.18	17.000	Holtec International 2002, Figure 6.3.7
Bottom overpack SS plate thickness (bottom layer)	5.08	2.000	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top overpack SS plate thickness (top layer)	10.16	4.000	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top overpack concrete plate thickness	26.67	10.500	Holtec International 2002, Figure 6.3.7
Top overpack SS plate thickness (bottom layer)	3.175	1.250	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top gap (between MPC and overpack)	3.81	1.500	Approximated from Holtec International 2002, Drawings 1495 (Sheet 2) and 3923 (Sheet 3)

Table 6.1-7 Specifications of the BWR GE 8x8 Standard Assembly

Parameter	cm	in	Reference
Rod pitch	1.6256[a]	0.640	Holtec International 2002, p. 2.1-14
Active fuel length	381	150	Holtec International 2002, p. 2.1-14
Cladding outside diameter	1.2268	0.483	Holtec International 2002, p. 2.1-14
Cladding inside diameter	1.0796	0.425	Holtec International 2002, p. 2.1-14
Pellet outside diameter	1.0566	0.416	Holtec International 2002, p. 2.1-14
Guide/instrument tube outside diameter	1.0566	0.416	Holtec International 2002, Figure 6.3-3
Guide/instrument tube thickness	0	0	Holtec International 2002, p. 2.1-14
Array size	8x8		Holtec International 2002, p. 6.2-42
Number of fuel rods	62		Holtec International 2002, p. 6.2-42
Number of guide/instrument tubes	2		Holtec International 2002, p. 2.1-14

[a] Holtec International 2002, p. 6.2-42 demonstrates that using a rod pitch of either 1.62814 cm or 1.6256 cm is acceptable.

6.1.5.4 BWR Material Compositions

The BWR material compositions are identical to those of the PWR material specifications, except for those listed in Table 6.1-8.

Table 6.1-8 Material Properties for the Storage Cask and BWR Fuel

Material	Density (g/cm^3)	Element	Weight Percent (wt%)	Atom Fraction or Atom Density (atoms/barn-cm)	Reference/ Remark
Boral [a, b] (0.0279 g $^{10}B/cm^2$)	2.66	Al B-10 B-11 C	N/A	3.805E-02 8.071E-03 3.255E-02 1.015E-02	Holtec International 2002, p. 6.3-12
UO_2 – (fuel) 4.20 % enriched	10.522	U-235 U-238 O-16	3.702 84.45 11.85	N/A	Holtec International 2002, p. 6.3-11

[a] Calculations for varied Boral loading can be found in Excel file *boral.xls*
[b] The ^{10}B loading of 0.0279 g/cm^2 is 75 % of the minimum loading 0.0372 g/cm^2 (Holtec International 2002, p. 6.2-5)

6.1.6 PWR Site-Specific Canister Calculation Input

A conceptual design of a 21-PWR site-specific canister with a 210.0 in. length and 66.0 in. diameter weighing up to 47 tons has been initiated (BSC 2005c, Section 3.1.2.3.4 and Table 4-1). These dimensions are also shown in *21-PWR Site-Specific Canister/Basket Canister Shell Detail Drawing* (BSC 2005k).

The geometric descriptions for B&W 15x15 fuel used in the 21-PWR Site-Specific Canister can be found in the referenced document, *21-PWR Site-Specific Canister Loading Curve Evaluation*

(BSC 2005g). The physical dimensions for the canister and fuel assembly are shown in Sections 5.4 and 5.4.1 of the document (BSC 2005g), respectively. The fuel assembly specifications are shown in Table 6.1-9 (BSC 2005g, Table 16).

Table 6.1-9 B&W 15x15 Fuel Assembly Specifications

Assembly Component	Specification
Fuel Pellet Outer Diameter	0.93980 cm
Fuel Rod Cladding Inner Diameter	0.95758 cm
Fuel Rod Cladding Outer Diameter	1.09220 cm
Guide Tube Inner Diameter	1.26492 cm
Guide Tube Outer Diameter	1.34620 cm
Instrument Tube Inner Diameter	1.12014 cm
Instrument Tube Outer Diameter	1.38193 cm

Source: Punatar 2001, p. 2-5

The material compositions for B&W 15x15 fuel used in the 21-PWR Site-Specific Canister can be found in the referenced document, *21-PWR Site-Specific Canister Loading Curve Evaluation* (BSC 2005g). Materials for the canister are shown in Tables 6.1-23 and 6.1-19 through 6.1-22 (BSC 2005g, Section 5.3.2). Materials for the fuel assembly are shown in Tables 6.1-10 through 6.1-16 (BSC 2005g, Section 5.3.3). Materials for the uranium fuel are shown in Table 6.1-17 (BSC 2005g, Section 5.3.4).

Table 6.1-10 Zircaloy-4 Material Composition

Element/Isotope	Wt%	Element/Isotope	Wt%
Cr-50	0.0042	Fe-57	0.0045
Cr-52	0.0837	Fe-58	0.0006
Cr-53	0.0097	O-16	0.1250
Cr-54	0.0024	Zr-nat	98.1150
Fe-54	0.0119	Sn-nat	1.4500
Fe-56	0.1930	Density[a] = 6.56 g/cm^3	

Source: ASTM B 811-97 2000, p. 2, Table 2

NOTES: [a]From ASM International 1990, p. 666, Table 6.

Table 6.1-11 SS304 Material Composition

Element	Wt%	Element	Wt%
Carbon	0.080 (0.030[a])	Chromium	19.000
Nitrogen	0.100	Manganese	2.000
Silicon	0.750	Iron	Balance 68.745 (68.045[a])
Phosphorous	0.045	Nickel	9.250 (10.000[a])
Sulfur	0.030	Density = 7.94 g/cm^3	

Source: ASME (2001, Section II,Part A, SA-240, Table 1); Density from ASTM G 1-90 (1999, p. 7, Table X1)
NOTE: [a]Carbon and Nickel composition corresponds to SS304L which yields a different iron balance

Table 6.1-12 Inconel 718 Material Composition

Element	Wt%	Element	Wt%	Element	Wt%
Nickel	52.500	Molybdenum	3.050	Silicon	0.180
Chromium	19.000	Titanium	0.900	Carbon	0.040
Iron	18.500	Aluminum	0.500	Sulfur	0.008
Niobium[a]	5.130	Manganese	0.180	Density = 8.19 g/cm^3	

Source: Lynch 1989, p. 496

NOTE: [a] Reference identifies this material as "columbium," which is actually the element niobium.

Table 6.1-13 End-Fitting Component Material Volume Fractions

Assembly Design	Stainless Steel Type 304	Inconel	Zircaloy-4	Moderator
Upper End-Fitting	0.2756	0.0441	0.0081	0.6722
Lower End-Fitting	0.1656	0.0306	0.0125	0.7913

Source: Punatar 2001, Table 2-3

Table 6.1-14 End-Fitting Homogenized Material Compositions

Element/ Isotope	Upper End-Fitting Wt%[a]	Lower End-Fitting Wt%[a]
C-nat	0.0245	0.0203
N-14	0.0668	0.0539
Si-nat	0.5210	0.4229
P-31	0.0301	0.0243
S-32	0.0209	0.0170
Cr-50	0.6181	0.5098
Cr-52	12.3822	10.2114
Cr-53	1.4309	1.1800
Cr-54	0.3622	0.2987
Mn-55	1.3563	1.0968
Fe-54	2.6847	2.1808
Fe-56	43.6633	35.4677
Fe-57	1.0269	0.8342
Fe-58	0.1380	0.1121
Ni-58	8.3820	7.2490
Ni-60	3.3394	2.8880
Ni-61	0.1476	0.1277
Ni-62	0.4777	0.4132
Ni-64	0.1263	0.1093
H-1	2.2972	3.6312
O-16	18.2314	28.8196
Al-27	0.0552	0.0514
Ti-nat	0.0993	0.0925
Nb-93	0.5659	0.5272
Mo-nat	0.3364	0.3135
Zr-nat	1.5920	3.2990
Sn-nat	0.0235	0.0488
Density (g/cm^3)	3.2748	2.4388

[a] Homogenization used stainless steel 304L values for carbon, nickel, and iron

Table 6.1-15 Fuel Rod Plenum Material Volume Fractions

Assembly Design	**Plenum Location**	**Type 304 Stainless Steel**	**Gas (modeled as void)**	**Zircaloy-4**
Babcock & Wilcox 15×15	Upper	0.0811	0.7793	0.1396
	Lower	0.1569	0.5973	0.2458

Source: Punatar 2001, Table 2-9 and Figures 2-3 and 2-7

NOTE: Volume fractions are renormalized to exclude the cladding, which is modeled explicitly in the input.

Table 6.1-16 Fuel Rod Plenum Homogenized Material Compositions

Element/Isotope	Wt% of Element/Isotope in Material Composition	
	Upper Fuel Rod Plenum[a]	Lower Fuel Rod Plenum[a]
C-nat	0.0124	0.0131
N-14	0.0413	0.0436
Si-nat	0.3096	0.3270
P-31	0.0186	0.0196
S-32	0.0124	0.0131
Cr-50	0.3302	0.3485
Cr-52	6.6148	6.9806
Cr-53	0.7644	0.8067
Cr-54	0.1935	0.2042
Mn-55	0.8257	0.8720
Fe-54	1.5943	1.6829
Fe-56	25.9299	27.3712
Fe-57	0.6099	0.6438
Fe-58	0.0820	0.0865
Ni-58	2.7744	2.9298
Ni-60	1.1053	1.1672
Ni-61	0.0489	0.0516
Ni-62	0.1581	0.1670
Ni-64	0.0418	0.0442
O-16	0.0734	0.0705
Zr-nat	57.6077	55.3392
Sn-nat	0.8514	0.8178
Density (g/cm^3)	1.5597	2.8583

[a] Homogenization used stainless steel 304L values for carbon and nickel

Table 6.1-17 Fresh Fuel Compositions

Enrichment (wt% ^{235}U)	Wt% ^{234}U	Wt% ^{235}U	Wt% ^{236}U	Wt% ^{238}U	Wt% Oxygen
1.5	0.0106	1.3222	0.0061	86.8098	11.8513
2.0	0.0144	1.7630	0.0081	86.3625	11.8519
2.5	0.0184	2.2037	0.0101	85.9152	11.8526
3.0	0.0224	2.6444	0.0122	85.4677	11.8533
3.5	0.0265	3.0851	0.0142	85.0202	11.8540
4.0	0.0306	3.5258	0.0162	84.5727	11.8547
4.5	0.0348	3.9665	0.0182	84.1251	11.8553
5.0	0.0390	4.4072	0.0203	83.6775	11.8560

The input file for MCNP calculation is modified from the file named 5.0 (BSC 2005g, Attachment III) with the following changes:

- Modify surface cards for the canister lid top and bottom surfaces to adjust the total length from 214 in. to 210.5 in. as shown in the *21-PWR Site-Specific Canister/Basket Shield & Inner Seal Plug Detail Drawing* (BSC 2005j) and *21-PWR Site-Specific Canister/Basket Canister Shell Detail Drawing* (BSC 2005k)
- Retain the canister geometry but replace the waste package by the concrete overpack, maintain the same MPC-24 geometry in radial direction, but expand the axial length in order to fit the canister.
- Replace the surrounding tuff by water or air depending on wet or dry conditions outside the cask
- Add the material compositions of SS304 steel, concrete, and air, which are shown in Table 6.1-4.

The thickness of cask body wall, lid, bottom, and protective cover for metal overpack, are 24.13, 26.67, 26.035, and 0.9652 cm, respectively (Hunter 2004, Table 5.1-1). The material for the cask body is carbon steel for which the density, element and atom densities are shown in Table 6.1-4.

Note that the 21-PWR site-specific canister calculations include ^{234}U and ^{236}U in the fuel composition (Table 6.1-17), which is counter to the conservative statements regarding the use of fuel materials in Section 4.3. However, the quantity of the isotopes is so small, and the fuel density used is higher than the typical value so the presence of these isotopes should not have any significant impact on the result. The fuel composition was not changed to be consistent with the previously performed calculation of the 21-PWR site-specific canister.

6.1.7 BWR Site-Specific Canister Calculation Input

The canister shell was composed of SA-240 S31603 as described in Table 6.1-18. The fuel basket plates were composed of Ni-Gd Alloy (Unified Numbering System [UNS] designation is UNS N06464) with 1.5 wt% Gd as described in Table 6.1-19, and the thermal shunts were composed of SB-209 A96061 T4 (aluminum 6061) as described in Table 6.1-20. The basket side and corner guides, and the basket stiffeners were composed of Grade 70 A 516 carbon steel as described in Table 6.1-21. Site-specific canister/basket material thicknesses were taken from the drawings in Attachment B. The tube to plate separation distance was 0.3170 cm, which was derived in BSC (2005g, Attachment IV, file *misc.xls*).

The overpack was described in Section 6.1.5.3 with the overall length adjusted to 635.635 cm in order to accommodate the site-specific canister/basket. The value for the length of the overpack was based on maintaining the axial dimensions and clearance between the top of the site-specific canister/basket and the bottom of the overpack lid. The waste package consists of two barriers - an inner vessel, and an outer corrosion barrier. The inner vessel was composed of SA-240 S31600 as described in Table 6.1-22, and the outer corrosion barrier was composed of Alloy 22 (UNS N06022) as described in Table 6.1-23.

The chromium, nickel, and iron elemental weight percents obtained from the references were expanded into their constituent natural isotopic weight percents for use in MCNP. This expansion was performed by: (1) calculating a natural weight fraction of each isotope in the elemental state, and (2) multiplying the elemental weight percent in the material of interest by the natural weight fraction of the isotope in the elemental state to obtain the weight percent of the isotope in the material of interest. This process is described mathematically in Equations 4 and 5. The atomic mass values and atom percent of natural element values for these calculations are from Parrington et al. (1996, pp. 18 to 63).

$$WF_i = \frac{A_i(At\%_i)}{\sum_{i=1}^{I} A_i\left(At\%_i\right)} \tag{4}$$

where

WF_i = the weight fraction of isotope$_i$ in the natural element
A_i = the atomic mass of isotope$_i$
$At\%_i$ = the atom percent of isotope$_i$ in the natural element
I = the total number of isotopes in the natural element

$$Wt\%_i = WF_i(E_{wt\%}) \tag{5}$$

where

$Wt\%_i$ = the weight percent of isotope$_i$ in the material composition
WF_i = the weight fraction of isotope$_i$ from Equation 4
$E_{wt\%}$ = the referenced weight percent of the element in the material composition

Table 6.1-18 Material Specifications for SA-240 S31603

Element/Isotope	Wt%	Element/Isotope	Wt%
C-nat	0.0300	Fe-54	3.7036
N-14	0.1000	Fe-56	60.2343
Si-nat	0.7500	Fe-57	1.4167
P-31	0.0450	Fe-58	0.1904
S-32	0.0300	Ni-58	8.0641
Cr-50	0.7103	Ni-60	3.2127
Cr-52	14.2291	Ni-61	0.1420
Cr-53	1.6443	Ni-62	0.4596
Cr-54	0.4162	Ni-64	0.1216
Mn-55	2.0000	Mo-nat	2.5000
Density[a] = 7.98 g/cm^3			

Source: ASME 2001, Section II, Part A, SA-240, Table 1

NOTES: [a]Density from ASTM G 1-90 (1999, p. 7, Table X1)

Table 6.1-19 Material Specifications for Ni-Gd Alloy (UNS N06464) with 1.5 wt% Gd[a]

Element/Isotope	Wt%	Element/Isotope	Wt%
C-nat	0.0100	Gd-152	0.0029
Mn-55	0.5000	Gd-154	0.0320
Si-nat	0.0800	Gd-155	0.2187
Cr-50	0.6602	Gd-156	0.3045
Cr-52	13.2247	Gd-157	0.2343
Cr-53	1.5283	Gd-158	0.3742
Cr-54	0.3868	Gd-160	0.3335
Ni-58	43.3679	Fe-54	0.0565
Ni-60	17.2778	Fe-56	0.9190
Ni-61	0.7637	Fe-57	0.0216
Ni-62	2.4717	Fe-58	0.0029
Ni-64	0.6537	S-32	0.0050
Mo-nat	14.5500	P-31	0.0050
Co-59	2.0000	O-16	0.0050
Density = 8.76 g/cm^3			

Source: ASTM B 932-04 2004, Table 1 and Section 8

NOTE: [a]1.5wt% Gd is based on typical value of 75% credit (NRC 2000, p. 8-4) allowed for fixed neutron absorbers and a nominal Gd loading of 2.0 wt% for Ni-Gd Alloy

Table 6.1-20 Material Specifications for SB-209 A96061 T4

Element/Isotope	Wt%	Element/Isotope	Wt%
Si-nat	0.6000	Mg-nat	1.0000
Fe-54	0.0396	Cr-50	0.0081
Fe-56	0.6433	Cr-52	0.1632
Fe-57	0.0151	Cr-53	0.0189
Fe-58	0.0020	Cr-54	0.0048
Cu-63	0.1884	Ti-nat	0.1500
Cu-65	0.0866	Al-27[a]	96.9300
Mn-55	0.1500	Density[b] = 2.713 g/cm^3	

Source: ASM International 1990, p. 102

NOTES: [a]Zn cross-section data unavailable; therefore, it was substituted as Al-27
[b]ASME 2001, Section II, Table NF-2 indicates a converted value from 0.098 lb/in^3 to 2.713 g/cm^3

Table 6.1-21 Grade 70 A516 Carbon Steel Composition

Element/Isotope	Wt%	Element/Isotope	Wt%
C-nat	0.2700	Fe-54	5.5558
Mn-55	1.0450	Fe-56	90.3584
P-31	0.0350	Fe-57	2.1252
S-32	0.0350	Fe-58	0.2856
Si-nat	0.2900	Density = 7.850 g/cm^3	

Source: ASME 2001, Section II, Part A, SA-516/SA-516M, Table 1; Density from ASME 2001, Section II, Part A, SA-20, Section 14.1

Table 6.1-22 Material Specifications for SA-240 S31600

Element/Isotope	Wt%	Element/Isotope	Wt%
C-nat	0.0800	Fe-54	3.7007
N-14	0.1000	Fe-56	60.1884
Si-nat	0.7500	Fe-57	1.4156
P-31	0.0450	Fe-58	0.1902
S-32	0.0300	Ni-58	8.0641
Cr-50	0.7103	Ni-60	3.2127
Cr-52	14.2291	Ni-61	0.1420
Cr-53	1.6443	Ni-62	0.4596
Cr-54	0.4162	Ni-64	0.1216
Mn-55	2.0000	Mo-nat	2.5000
Density[a] = 7.98 g/cm^3			

Source: ASME 2001, Section II, Part A, SA-240, Table 1
NOTES: [a]Density from ASTM G 1-90 (1999, p. 7, Table X1)

Table 6.1-23 SB-575 N06022 Material Composition

Element/ Isotope	Wt%	Element/ Isotope	Wt%
C-nat	0.0150	Co-59	2.5000
Mn-55	0.5000	W-182[a]	0.7877
Si-nat	0.0800	W-183[a]	0.4278
Cr-50	0.8879	W-184[a]	0.9209
Cr-52	17.7863	W-186[a]	0.8636
Cr-53	2.0554	V	0.3500
Cr-54	0.5202	Fe-54	0.2260
Ni-58	36.8024	Fe-56	3.6759
Ni-60	14.6621	Fe-57	0.0865
Ni-61	0.6481	Fe-58	0.0116

Element/ Isotope	Wt%	Element/ Isotope	Wt%
Ni-62	2.0975	S-32	0.0200
Ni-64	0.5547	P-31	0.0200
Mo-nat	13.5000	Density = 8.69 g/cm^3	

Source: DTN: MO0003RIB00071.000
NOTES: [a]W-180 cross section libraries are not available so the atom percents of the remaining isotopes were used to renormalize the elemental weight and derive isotopic weight percents excluding the negligible 0.120 atom percent in nature contribution from W-180.

The design basis fuel assembly design selected for the site-specific canister evaluations corresponds to that of the GE 8x8 with two water rods described in Section 6.1.5.3. This assembly was selected for use in order to remain consistent with the MPC-68 evaluations.

6.1.8 Site-Specific Aging Cask Calculation Input

Calculations were performed to determine additional criticality controls required for the site-specific cask to accommodate commercial fuel outside the content specification for the MPC-24 and MPC-68. It was assumed that the site-specific cask is similar in design to the MPC-24 and MPC-68 (Assumption 3.2.2). In addition to varying the ^{10}B loading in the neutron poison of the internal basket (i.e., Boral), as discussed in Section 6.1.5, B_4C was also investigated as an alternative neutron poison. Further, additional criticality controls were investigated including increased fuel assembly spacing, reduction of number of assemblies in the site-specific cask, and inclusion of burnup-credit nuclides in the fuel. The latter evaluation features fuel burnups of 10 GWd/MTU, 20 GWd/MTU, and 30 GWd/MTU with an initial fuel enrichment of 5 wt% and 5 year cooling time for both PWR and BWR fuel. The burnup ranges are conservatively chosen based on PWR and BWR SNF discharge data shown in Figures 6.1-6 and 6.1-7. Table 6.1-24 displays the neutron poison properties utilized in the site-specific cask evaluations for commercial fuel and Table 6.1-25 shows the fuel properties for the burnup-credit evaluations. The selection of the isotopes for inclusion for the burnup-credit calculations are taken from the *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003, Table 3-1). Note that fuel properties are for B&W 15x15 and GE 7x7 fuel assembly types, which are also included in the evaluations of burnup-credit. Previous studies have been made identifying the bounding isotopic concentrations in burnup-credit applications for B&W 15x15 PWR fuel (BSC 2003b) and GE 7x7 BWR fuel (Wimmer 2004) as a function of initial enrichment and burnup. Per Assumption 3.2.5, these PWR and BWR isotopic concentrations will be bounding for the W 17x17 OFA and GE 8x8 fuel assembly, respectively. These bounding isotopic concentrations were utilized in the MCNP calculation for consistency with burnup-credit criticality calculations on the Yucca Mountain Project and to produce a bounding k_{eff} for the MPC-24 and MPC-68, respectively. The calculations were performed with the entire selection of the principal isotopes for commercial SNF burnup credit (YMP 2003, Table 3-1). The fuel density was increased to 10.741 g/cm^3 to be consistent with the density used in the bounding isotopic concentration calculations (BSC 2003b, p. 55 & Wimmer 2004, p. 103). The isotopic concentrations were utilized for 10, 20, and 30 GWd/MTU and taken from Table 18 (BSC 2003b) for PWR fuel and Table 25 (Wimmer 2004) for BWR fuel. The basket structures in the MPC-24 and MPC-68 for inclusion of the B&W 15x15 and GE 7x7 fuel assemblies, respectively, are identical to that of the W 17x17 OFA and GE 8x8 fuel assembly arrangement (Assumption 3.2.6).

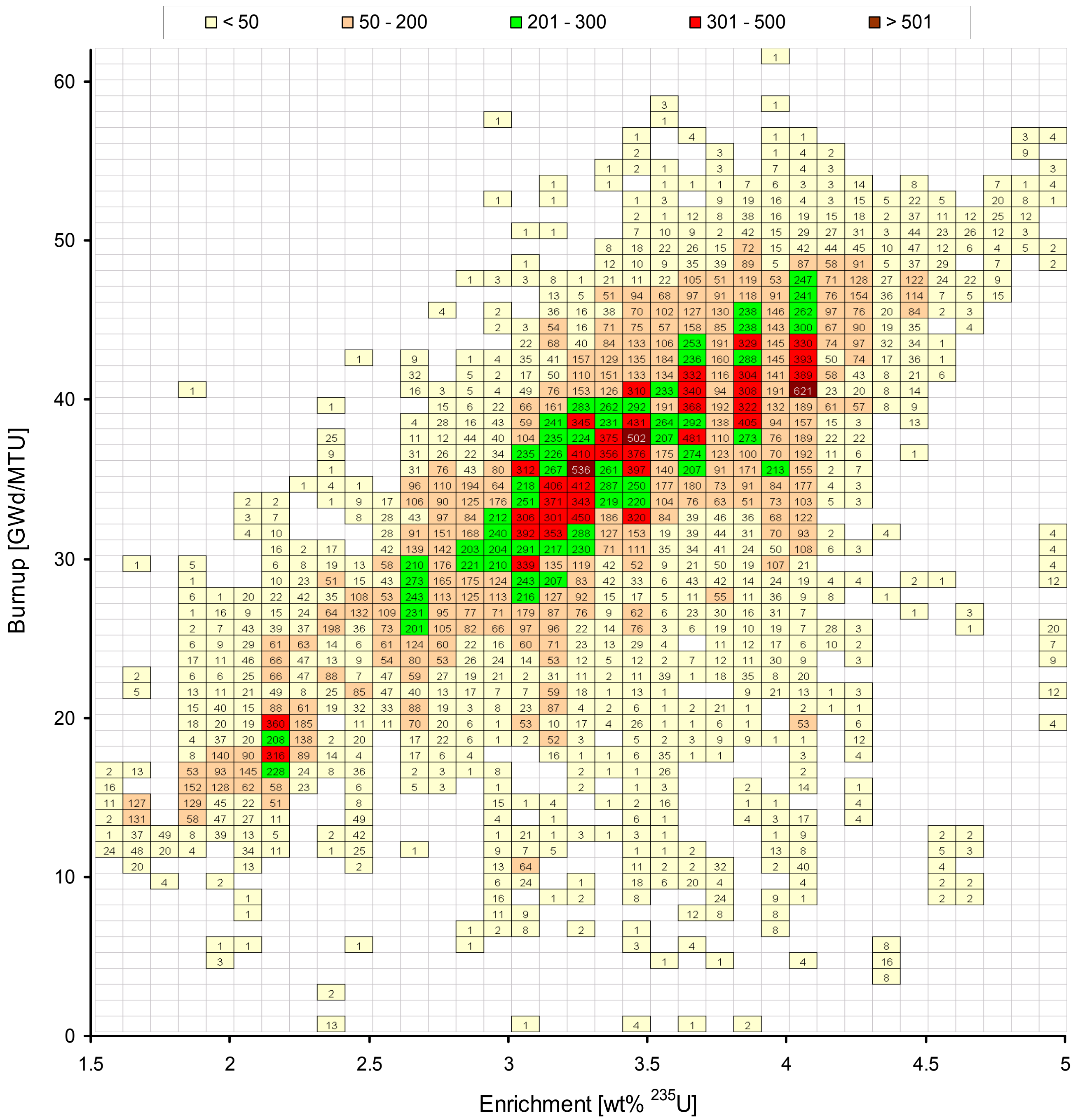


Figure 6.1-6 PWR SNF Discharge Data as of December 31, 1998
(Extracted from BSC 2004j, Figure 7 & PWR_Assembly.xls)

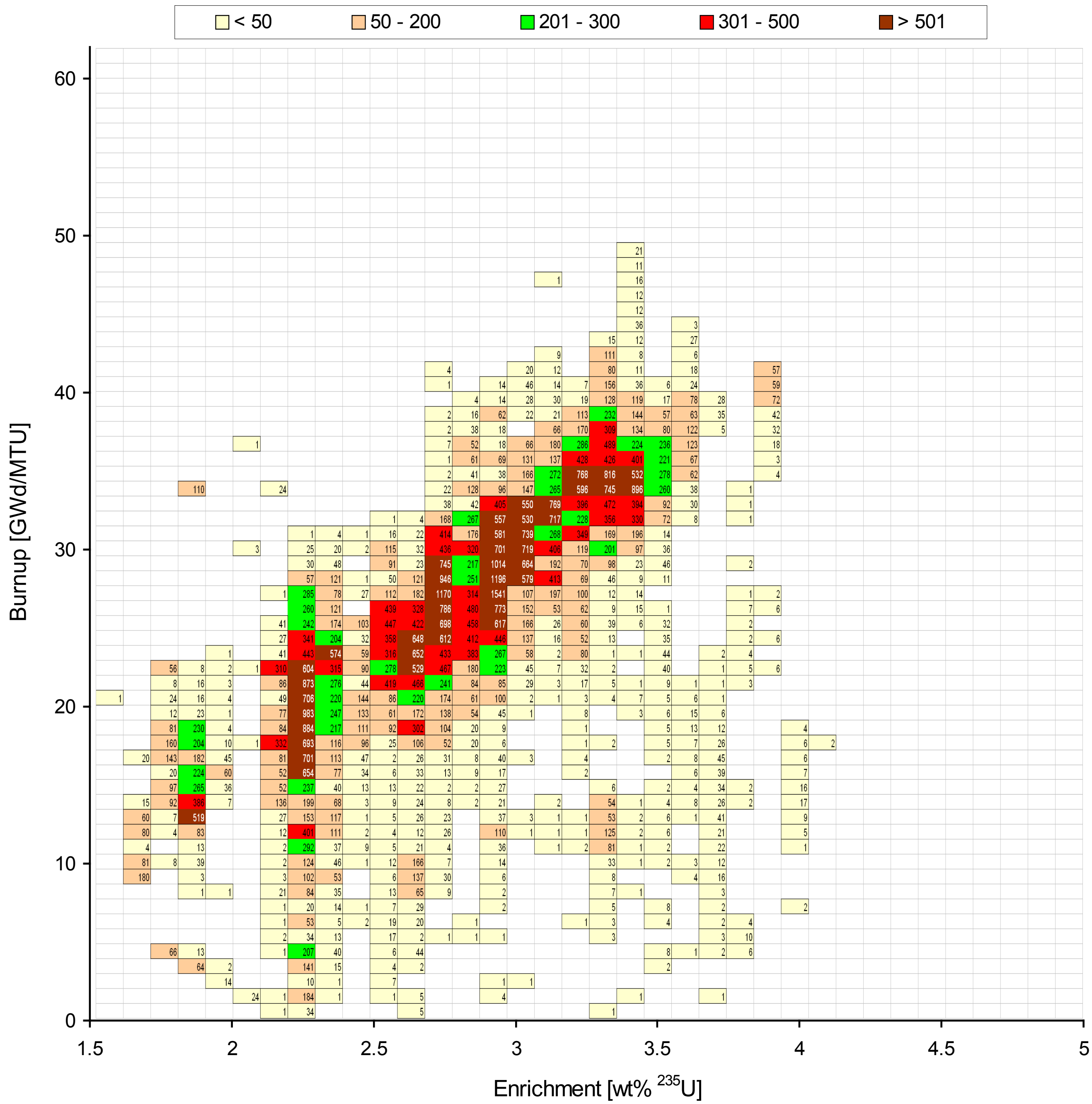


Figure 6.1-7 BWR SNF Discharge Data as of December 31, 1998
(Extracted from BSC 2004j, Figure 9 & BWR_Assembly.xls)

Table 6.1-24 Material Properties Utilized for Site-Specific Cask Evaluations

Material	Density (g/cm^3)	Element	Atom Density (atoms/barn-cm)	Reference
B_4C	2.346	B-10 [a]	2.022E-02	General Atomics 1993, p. 6.3-4
		B-11	8.207E-02	
		C	2.557E-02	

[a] Equivalent of 14.3 physical wt%, which is approximately 75 % of the weight fraction of B-10 in B.

Table 6.1-25 Fuel Properties for Burnup-Credit Evaluation

	Isotopic Concentrations (atoms/barn-cm) [a, b]					
Isotope	10 GWd/MTU PWR Fuel	10 GWd/MTU BWR Fuel	20 GWd/MTU PWR Fuel	20 GWd/MTU BWR Fuel	30 GWd/MTU PWR Fuel	30 GWd/MTU BWR Fuel
U-235	9.48E-04	9.70E-04	7.35E-04	7.66E-04	5.59E-04	5.87E-04
U-234	9.38E-06	8.92E-06	8.22E-06	7.67E-06	7.23E-06	6.76E-06
U-238	2.26E-02	2.25E-02	2.24E-02	2.22E-02	2.22E-02	2.20E-02
Pu-238	1.90E-07	4.60E-07	1.05E-06	2.04E-06	2.83E-06	4.57E-06
Pu-239	1.01E-04	1.94E-04	1.57E-04	3.05E-04	1.87E-04	3.59E-04
Pu-240	9.41E-06	1.32E-05	2.45E-05	3.08E-05	4.00E-05	4.65E-05
Pu-241	3.11E-06	4.63E-06	1.18E-05	1.39E-05	2.15E-05	2.40E-05
Pu-242	2.00E-07	2.24E-07	1.66E-06	1.27E-06	4.88E-06	3.47E-06
Am-241	8.92E-07	1.36E-06	3.56E-06	4.34E-06	6.78E-06	7.93E-06
O-16	4.76E-02	4.76E-02	4.73E-02	4.73E-02	4.69E-02	4.69E-02
Mo-95	1.58E-05	1.50E-05	3.03E-05	2.88E-05	4.38E-05	4.17E-05
Tc-99	1.54E-05	1.50E-05	2.98E-05	2.88E-05	4.32E-05	4.16E-05
Ru-101	1.32E-05	1.31E-05	2.62E-05	2.58E-05	3.91E-05	3.80E-05
Rh-103	8.58E-06	9.23E-06	1.67E-05	1.75E-05	2.42E-05	2.44E-05
Ag-109	5.58E-07	7.22E-07	1.66E-06	1.80E-06	3.07E-06	3.04E-06
Nd-143	1.38E-05	1.33E-05	2.54E-05	2.53E-05	3.50E-05	3.56E-05
Nd-145	9.44E-06	9.06E-06	1.79E-05	1.71E-05	2.56E-05	2.44E-05
Sm-147	3.66E-06	3.39E-06	6.35E-06	5.63E-06	8.36E-06	7.26E-06
Sm-149	1.97E-07	5.34E-07	2.15E-07	6.77E-07	2.18E-07	7.31E-07
Sm-150	2.92E-06	2.81E-06	6.40E-06	6.31E-06	9.95E-06	9.74E-06
Sm-151	4.65E-07	7.79E-07	6.05E-07	1.26E-06	6.99E-07	1.60E-06
Sm-152	1.36E-06	1.15E-06	2.87E-06	2.35E-06	4.21E-06	3.36E-06
Eu-151	1.91E-08	3.27E-08	2.49E-08	5.39E-08	2.87E-08	6.87E-08
Eu-153	7.07E-07	7.80E-07	1.90E-06	1.95E-06	3.39E-06	3.31E-06
Gd-155	1.85E-08	2.84E-08	4.21E-08	6.73E-08	8.03E-08	1.29E-07
U-233	4.80E-11	8.24E-11	8.36E-11	1.23E-10	1.09E-10	1.42E-10
U-236	5.79E-05	6.70E-05	9.77E-05	1.08E-04	1.28E-04	1.35E-04
Np-237	2.48E-06	4.59E-06	6.62E-06	1.09E-05	1.14E-05	1.68E-05
Am-242m	5.79E-10	1.41E-09	6.51E-09	1.31E-08	1.98E-08	4.16E-08
Am-243	1.09E-08	1.87E-08	1.92E-07	2.13E-07	8.60E-07	9.15E-07

[a] BSC 2003b, Table 18 (PWR fuel)
[b] Wimmer 2004, Table 25 (BWR fuel)

Two types of aging casks are being considered. The first is a site-specific cask placed in a vertical orientation on the aging pad. Aging of uncanistered CSNF in a vertical orientation takes place within all vertical metal (bolted) casks. Metal/concrete casks are used to house canistered CSNF or DOE SNF and HLW canisters. The internal configuration of vertical overpacks will accommodate the dimensions and characteristics of disposable canisters now being developed by DOE (BSC 2005c, Section 4.1.1.3).

The calculations for DOE fuel contained in a site-specific cask feature similar overpack dimensions to the site-specific cask utilized for commercial SNF (Assumption 3.2.3). The evaluated designs feature an inside diameter of 69.5 in. and 77.5 in., respectively. These dimensions are consistent with the TN-68 (Hunter 2002, Figure 5.1-1) and HI-STORM cask systems (see Table 6.1-1). Per Assumption 3.2.4, the overpack consists of 15 in. concrete. DOE canisters containing FFTF, Fort St. Vrain, and Enrico Fermi, were placed inside the overpack (inside diameter of 69.5 in.) in a 3x3 square pitch configuration as illustrated in Figure 6.1-8. A larger overpack inside diameter (77.5 in.) was utilized for the Enrico Fermi canisters to place 10 and 12 canisters in a close-packed triangular pitch configuration as shown in Figure 6.1-9. To analyze different configuration, the Enrico Fermi canisters were also placed in a circular pitch configuration (overpack inside diameter of 69.5 in.). In addition, Savannah River Site (SRS) HLW glass composition canisters were also placed in a circular pitch configuration to study the impact of HLW on k_{eff}. The SRS HLW glass canisters inside diameter is 24 in. and its chemical composition is shown in Table 6.1-26. Note that the SRS HLW glass configuration case was included for completeness only and the effect of this configuration on k_{eff} is expected to be minor due to the fissile-diluted composition of HLW glass. The two circular pitch configurations are illustrated in Figure 6.1-10.

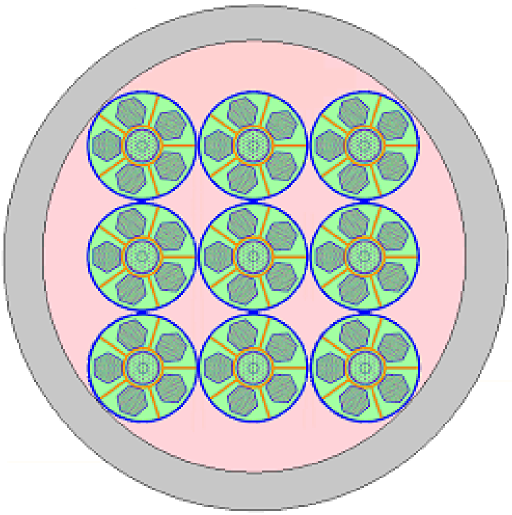
Only Fermi, Fort St. Vrain and FFTF are chosen for overpack calculations since they are among the most reactive fuel types. These most reactive configurations are demonstrated in the *Canister Handling Facility Criticality Safety Calculations* (BSC 2005d, Table 6.2-1) for the nominal case, which is dry inside. However, for defense in depth the canisters and the aging casks are flooded with water to show comparison of the most reactive DOE fuels. The input files are modified from fftP5a0A, fermPAa0, and fsvAa0 (BSC2005d, Attachment II) for the FFTF, Fermi, and Fort St. Vrain canisters, respectively.

The external environment surrounding overpack features either water or air acting as a reflector. Air as a reflector is the normal condition. Water as a reflector is also considered since this is a possibility that water presents on the aging pad in case of flooded scenario.

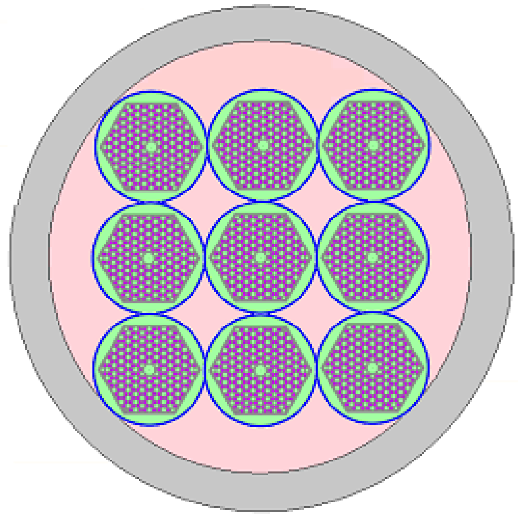
Table 6.1-26 Chemical Composition of SRS DHLW Glass

Element/Isotope	Composition [a] (wt%)	Element/Isotope	Composition [a] (wt%)
O	4.4770E+01	Ni	7.3490E-01
U-234	3.2794E-04	Pb	6.0961E-02
U-235	4.3514E-03	Si	2.1888E+01
U-236	1.0415E-03	Th	1.8559E-01
U-238	1.8666E+00	Ti	5.9676E-01
Pu-238	5.1819E-03	Zn	6.4636E-02
Pu-239	1.2412E-02	B-10	5.9176E-01
Pu-240	2.2773E-03	B-11	2.6189E+00
Pu-241	9.6857E-04	Li-6	9.5955E-02
Pu-242	1.9168E-04	Li-7	1.3804E+00
Cs-133	4.0948E-02	F	3.1852E-02
Cs-135	5.1615E-03	Cu	1.5264E-01
Ba-137	1.1267E-01	Fe	7.3907E+00
Al	2.3318E+00	K	2.9887E+00
S	1.2945E-01	Mg	8.2475E-01
Ca	6.6188E-01	Mn	1.5577E+00
P	1.4059E-02	Na	8.6284E+00
Cr	8.2567E-02	Cl	1.1591E-01
Ag	5.0282E-02		
Density [b] at 25 °C = 2.85 g/cm^3			

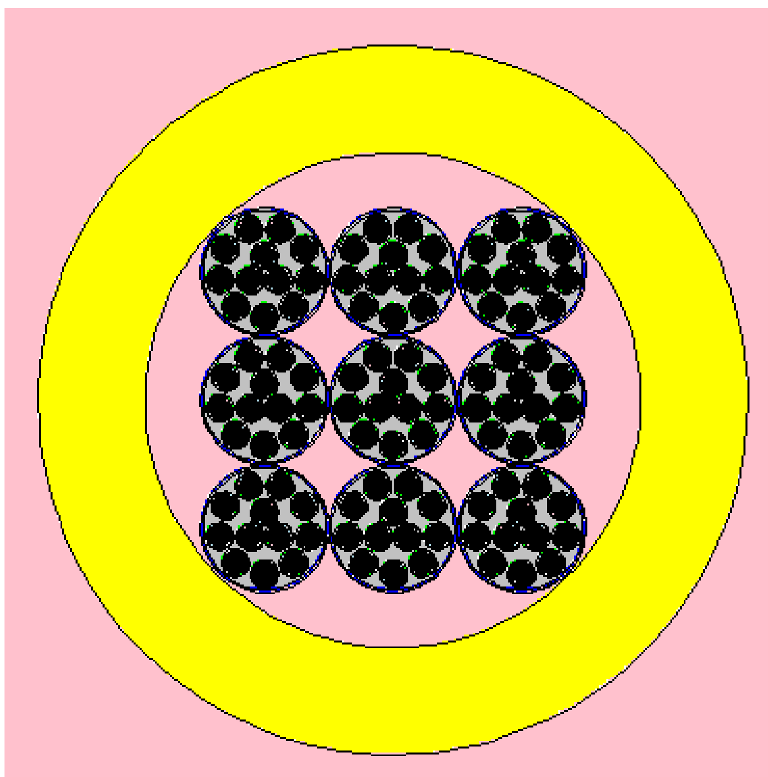
[a] CRWMS 1999, p. 7.
[b] Stout and Leider 1991, p. 2.2.1.1-4 (upper limit)



FFTF Canisters

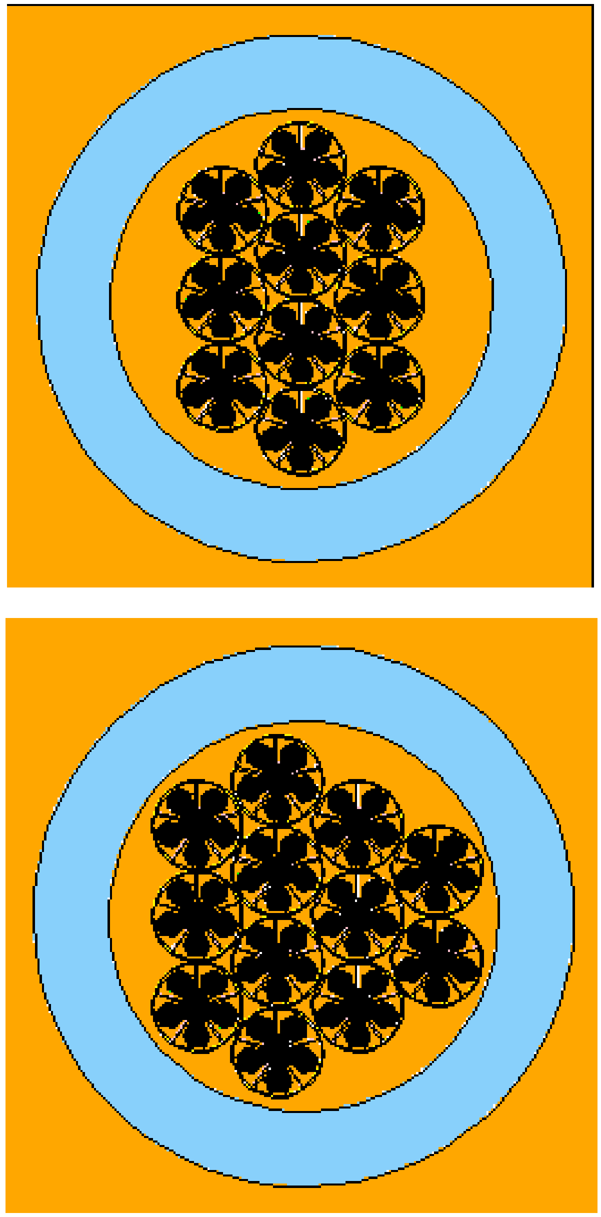


Fort St. Vrain Canisters



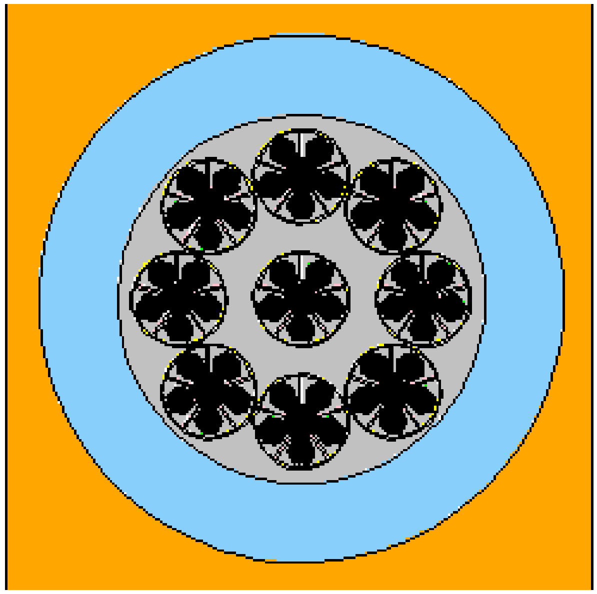
Enrico Fermi Canisters

Figure 6.1-8 Illustration of Site-Specific Cask Containing DOE Fuel Canisters (Overpack I.D.=69.5 in)

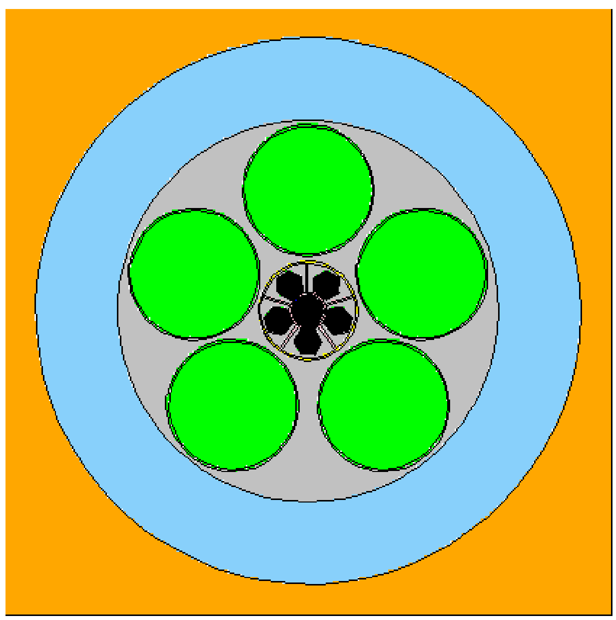


Overpack Inside Diameter = 77.5 in

Figure 6.1-9 Illustration of Site-Specific Cask Containing FFTF Canisters



FFTF Canisters



FFTF Canister with SRS HLW Glass Canisters

Figure 6.1-10 Illustration of Site-Specific Cask Containing FFTF and SRS HLW Glass Canisters

6.1.9 Inputs from Outside Sources

Table 6.1-27 lists the direct inputs to this calculation that are obtained from outside sources along with the justification of suitability for their intended use within this product.

Table 6.1-27 Justifications for Outside Sources

Source of Input	Description	Justification
HOLTEC International. 2002. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (Hi-Storm 100 Cask System).* HOLTEC Report HI-2002444. Two volumes. NRC Docket No. 72-1014. Marlton, New Jersey: HOLTEC International. TIC: 255899.	Physical dimensions, material specification, fuel contents relating to the HI-STORM cask system	This source document is a Final Safety Analysis Report (FSAR) that was submitted to the NRC by Holtec for a transportation cask. It is used in this document to demonstrate that a NRC licensed cask can be handled safely, from a criticality standpoint. The conclusions of this product do not directly rely upon the exact parameters of the transportation cask described in this source document.

6.2 CRITICALITY CALCULATIONS

The process and methodology for criticality safety analysis given in the *Preclosure Criticality Analysis Process Report* (BSC 2004e, Sections 2.2.6 and 2.2.7) were implemented. Dry and flooded conditions were featured in the MCNP calculations. Note that for in-package operations burnup credit is allowed, which was explored as an option to criticality control in Section 6.2.2. In addition, reflective boundary conditions are applied to all calculations.

6.2.1 Moderator Density Variations

Moderator density, which could vary from dry to fully moderated conditions under accident scenarios, were varied over the range of 0.0 to 1.0 g/cm^3 both inside and outside of the storage/aging cask for PWR and BWR fuel assemblies. The results are presented in Sections 7.1 and 7.2.

6.2.2 Evaluation of Criticality Controls for Site-Specific Cask (Commercial Fuel)

The MPC-24 and MPC-68 are licensed to only hold up to 4.0 wt% PWR (Holtec International 2002, p. 6.2-37) and 4.2 wt% BWR (Holtec International 2002, p. 6.2-42) enriched fuel, respectively. For the purpose of storing enriched fuel of up to 5.0 wt% in the site-specific cask (per Assumption 3.2.2, the site-specific cask is designed to be similar to the MPC-24 and MPC-68), the following scenarios were evaluated to meet the USL:

- Increase the Boral loading. An alternate neutron poison, B_4C, was also studied for both the MPC-24 and MPC-68.

- Reduce the number of assemblies contained in the site-specific cask.
- Increase fuel assembly spacing.
- Include burnup-credit nuclides in the fuel.

Two additional assembly types were studied for the burnup-credit calculation to investigate if the Westinghouse 17x17 OFA and the GE 8x8 fuel assembly are the most reactive fuel types when applying burnup-credit. The additional fuel assemblies are B&W 15x15 and GE 7x7 and their physical description is documented in BSC 2004h (pp. 28 and 37). The fuel rod pitch of B&W 15x15 is 1.44272 cm, the fuel pellet diameter is 0.93624 cm, and the clad outer diameter is 1.0922 cm with a clad thickness of 0.06731 cm. The fuel rod pitch of GE 7x7 is 1.8745 cm, the fuel pellet diameter is 1.21158 cm, and the clad outer diameter is 1.43 cm with an inner diameter of 1.2421 cm.

Previous studies indicate that the B&W 15x15 fuel assembly requires a higher burnup for initial enrichments up to 4 wt% than the Westinghouse 17x17 fuel design to fit the loading curve (Wagner and Sanders 2003, p. 64). The MCNP calculations utilizing burnup credit simulate the fuel region as one node, as opposed to applying an axial burnup profile. This simulation approach is slightly conservative in lower burnup cases for PWR fuel with initial enrichments ranging between 2.0 – 5.0 wt% (BSC 2004k, Table 20). The same one node representation of the fuel region is applied to the BWR fuel. Also, note that the MCNP calculations for the GE 7x7 fuel assembly do not include the fuel rods containing Gd_2O_3 for conservatism (these rods are simulated as regular fuel rods with the same initial enrichment).

As described in Section 6.1.8, the burnup-credit evaluations were performed with previously evaluated bounding isotopic concentrations for B&W 15x15 PWR fuel (BSC 2003b) and GE 7x7 BWR fuel (Wimmer 2004). This was done to ensure a bounding k_{eff} value for the MPC-24 and MPC-68, respectively, and to be consistent with previously performed burnup-credit criticality evaluations on the Yucca Mountain Project. Per Assumption 3.2.5, the same bounding isotopic concentrations for B&W 15x15 and GE 7x7 are also used in the burnup-credit calculation for the Westinghouse 17x17 OFA and GE 8x8 assembly types.

6.2.3 Evaluation of Site-Specific Cask for DOE Canisters

Various loading scenarios were evaluated for the most reactive DOE fuel types as described in Section 6.1.8. These include square pitch loading, triangular pitch loading, and circular pitch loading to ensure the most reactive configuration. The overpack inside diameter was varied to increase the number of DOE canister inside the site-specific cask to ensure a criticality safe configuration.

6.2.4 Category 1 and 2 Event Sequences

No Category 1 and Category 2 event sequences applicable to the Aging Facility have been identified in the *Categorization of Event Sequences for License Application* document (BSC 2005e, Section 7). Design basis accidents have been evaluated for the HI-STORM 100 cask system (Holtec International 2002, Chapter 11). It was concluded that the design basis accidents have no effect on the design parameters important to criticality safety (e.g., flux trap, neutron poison, spacing), and consequently, there is no increase in reactivity due to a credible accident condition (Holtec International 2002, p. 6.4-6).

For defense-in-depth, a drop or slap down scenario causing rearrangement of the fuel assemblies was evaluated for the MPC-24 and MPC-68. Studies show that an increase in fuel pin pitch (flooded conditions) increases k_{eff} and the peak value for a W 17x17 OFA occurs at 1.45 cm (BSC 2005h, Table 6.3-6). The peak k_{eff} value for a GE 8x8 fuel assembly occurs at a pin pitch of 1.90 cm per Table 6.2-1. Note that a simplified MCNP calculation was used for this study only simulating a single fuel pin cell with reflective boundary conditions and 5.0 wt% fresh fuel enrichment. The results in Table 6.2-1 are only intended to show the trend in k_{eff} and not provide an absolute value.

Table 6.2-1 k_{eff} of Pin Pitch Increase of GE 8x8 Fuel

Pin Pitch (cm)	k_{eff}	σ	MCNP files
1.6256 (regular)	1.50017	0.00026	bwr8x85, bwr8x85.out
1.70	1.51652	0.00026	bwr085, bwr085.out
1.80	1.52820	0.00023	bwr090, bwr090.out
1.90	1.53107	0.00022	bwr095, bwr095.out
1.95	1.52934	0.00023	bwr0975, bwr0975.out
2.00	1.52623	0.00024	bwr100, bwr100.out

Calculations were performed for the MPC-24 and MPC-68 (flooded conditions and 4.0 and 4.2 wt% enrichment, respectively) featuring the bottom 15 cm and 30 cm reconfigured to a pin pitch of 1.45 cm and 1.90 cm, respectively. The scenarios could include when all of the fuel assemblies in the MPC-24 and MPC-68 reconfigured and when only the fuel assemblies located in the center reconfigured. Spacers that prevents the fuel assembly from bowing out, or bending, are located near the ends, as well as approximately 30 cm from the ends of the fuel assemblies (DOE 1987, p. 2A-353). Complete damage of the bottom spacer allows the fuel assembly below the next intact spacer to bend during a drop event. Bending results in a greater assembly separation that affects the lower section of assembly length.

Table 6.2-2 shows the results from the calculations including the damage heights of 15 and 30 cm. It indicates an increase in reactivity compared to that in Tables 7.1-1 and 7.2-1 with the corresponding fuel enrichment. For the 30 cm damage fuel height, k_{eff} exceeds the USL (except MPC68C30) while the 15 cm damage fuel height, k_{eff} remains below the USL for both BWR and PWR accident conditions. It should also be pointed out that in order for these most reactive pin pitches to occur, the internal basket structure must completely fail. If the spacers only were to fail due to a drop and the internal basket structures remain intact, the maximum possible pin pitches will be less than those considered in Table 6.2-2 for both PWR and BWR fuel. Calculations show that a change in reactivity due to this latter scenario is very minor for both fuel types (MCNP files: MPC24b2c & MPC24b2c.out, and MPC68B30 & MPC68B30.out with k_{eff} equating to 0.93199 ± 0.00030 and 0.93973 ± 0.00026, respectively). More analysis of fuel assembly bowing in different shapes, damaged lengths, and flooded water heights including full length bowing and fully flooded conditions are shown in *Fuel Handling Facility Criticality Safety Calculations* (BSC 2005b, Section 5.2.3.3). Even if the internal basket structure remains intact, the water level still needs to be controlled less than 130 cm for a full length bowing scenario (BSC 2005b, Table 5.2-7).

To further defend the high k_{eff}'s for a 30 cm damage fuel height, it should be explained that the canisters will be dry inside with a proper sealed lid. Procedures require the canister be seal-welded and a dryness test be performed (Holtec International 2002, p. 1.2-19). Fuel reconfiguration in a dry condition is not expected to change k_{eff} significantly (BSC 2005h, Section 6.3.2) and the results in Section 7 of this document indicates that k_{eff} of a dry storage/aging configuration is well below the USL.

Table 6.2-2 Fuel Reconfiguration Evaluation for PWR and BWR Fuel

Damaged Height (cm)	k_{eff}	σ	MCNP files	k_{eff}	σ	MCNP files
	Only Center Fuel Assemblies Bowed Out (12 PWR & 36 BWR Assemblies)					
	W 17x17 OFA			**GE 8x8**		
15	0.93380	0.00028	MPC2415C	0.93981	0.00028	MPC68C15
30	0.97632	0.00030	MPC2430C	0.94659	0.00028	MPC68C30
	All Fuel Assemblies Bowed Out					
	W 17x17 OFA			**GE 8x8**		
15	0.93408	0.00031	MPC24E15	0.93940	0.00026	MPC68M15
30	0.98517	0.00028	MPC24b2E	0.95147	0.00027	MPC68M30

[a] The output files to each run have the same name as the corresponding input file but with a .out extension (e.g., the output file matching input file MPC2415C is MPC2415C.out).

7. RESULTS AND CONCLUSIONS

This section presents the results of the criticality calculations and makes recommendations for additional criticality safety design features as appropriate. The outputs presented in this document are all reasonable compared to the inputs and the results are suitable for the intended use. The uncertainties are taken into account by consistently using a conservative approach, which is the result of the assumptions and methods described in Sections 3 and 4, respectively.

7.1 MPC-24 (PWR FUEL)

Table 7.1-1 shows the k_{eff} values of the MPC-24 (PWR fuel) with varied initial enrichment. The calculation features an infinite array of casks fully flooded inside and 30 cm of water reflection outside. It can be seen that in order for the resulting k_{eff} to remain below the USL, the maximum fuel loading is 4.0 wt% enriched fuel. This is consistent with the recommendations in the Certificate of Compliance (Holtec International 2002, p. 6.2-37). If a higher enrichment will be considered (i.e., 4.5 or 5.0 wt%), a higher ^{10}B loading in the Boral panel needs to be implemented or an alternate neutron poison needs to be used for the internal basket. Section 7.3 presents calculations in which the Boral loading has been increased along with an alternative neutron poison.

Table 7.1-1 MPC-24 with Varied Fuel Enrichment

Enrichment (wt%)	k_{eff}	σ	MCNP files
4.0	0.93265	0.00030	MPC24-2c, MPC24-2c.out
4.5	0.95442	0.00030	MPC24-2d, MPC24-2d.out
5.0	0.97211	0.00026	MPC24-2, MPC24-2.out

The internal and external moderator conditions of the MPC-24 were altered in order to find the most reactive configuration for the Aging Facility. The scenarios considered include dry or flooded inside of the cask (i.e., inside the MPC and the gap between the MPC and the overpack) with a dry or flooded outside cask environment (i.e., outside the overpack). The calculations feature 5 wt% fuel enrichment and reflective boundaries with 30 cm radial separation. The results from the calculations are presented in Table 7.1-2. The calculations for the dry-dry, dry-wet, and wet-dry conditions are revised from the previous version. The reasons are changing air density, compositions, and three cells (2, 381, and 391) material from water to air in the dry condition. Note that the k_{eff} values for flooded inside cask conditions exceed the USL because 5 wt% enriched fuel was used in the calculations. The highest k_{eff} value is shown when the cask is fully flooded inside. This observation is further supported by the HI-STORM Final Safety Analysis Report (FSAR) where calculations proved that fully flooded condition corresponds to the highest k_{eff} (Holtec International 2002, p. 6.4-3). Calculations were also performed in the HI-STORM FSAR; it was shown that reducing the internal moderation results in a monotonic reduction in reactivity (Holtec International 2002, Table 6.4.1). It should also be mentioned that partial flooding was evaluated in the HI-STORM FSAR and it was demonstrated that the fully flooded condition is the most reactive (Holtec International 2002, Table 6.4.2).

Table 7.1-2 MPC-24 with Varied Moderator Condition

Moderation conditions	k_{eff}	σ	MCNP files
dry inside cask, dry outside cask	0.34594	0.00011	mpc24dd, mpc24dd.out
dry inside cask, flooded outside cask	0.34594	0.00010	mpc24dw, mpc24dw.out
flooded inside cask, dry outside cask	0.97261	0.00029	mpc24wd, mpc24wd.out
flooded inside cask, flooded outside cask	0.97211	0.00026	MPC24-2, MPC24-2.out

To ensure neutronic decoupling between the casks, the radial distance of the casks was altered. The most reactive configuration, based on Table 7.1-2, was used and the radial distances were changed from an infinite array of casks virtually touching each other (0.2 cm separation) to a 60 cm separation distance (cask surface to surface). The results displayed in Table 7.1-3 are within 2 sigma, which indicate that the MPC-24 cask ensures no neutronic interaction between casks. The results further indicate that the 30 cm flooded separation as simulated in MCNP is enough to ensure the most reactive configuration.

Table 7.1-3 MPC-24 with Varied Separation Distance

Distance between casks (cm)	k_{eff}	σ	MCNP files
0.2	0.97238	0.00027	MPC24-2b, MPC24-2b.out
30	0.97211	0.00026	MPC24-2, MPC24-2.out
60	0.97211	0.00026	MPC24-2a, MPC24-2a.out

The external environment of the cask could be somewhere between dry and fully flooded conditions. The condition can be referred to as mist and represents a range of 0.02 to 0.1 g/cm^3. Table 7.1-4 displays k_{eff} as a function of outside cask moderator density for 4.0 wt% enrichment, flooded inside cask conditions, and an infinite cask array (30 cm separation). It can be seen that the k_{eff} value for a fully flooded cask is independent of the external moderator (the small variations in the listed values are due to statistical uncertainties that are inherent to MCNP). The same observations were made in the HI-STORM FSAR (Holtec International 2002, p.6.4-3).

Table 7.1-4 MPC-24 with Varied Outside Moderator Densities

Outside moderator density (g/cm^3)	k_{eff}	σ	MCNP files
0.0	0.93342	0.00028	MPC24m0, MPC24m0.out
0.02	0.93233	0.00028	MPC24m2, MPC24m2.out
0.035	0.93300	0.00029	MPC24m3, MPC24m3.out
0.05	0.93344	0.00027	MPC24m5, MPC24m5.out
0.07	0.93262	0.00030	MPC24m7, MPC24m7.out
0.085	0.93264	0.00029	MPC24m8, MPC24m8.out
0.1	0.93264	0.00029	MPC24m1, MPC24m1.out
0.5	0.93263	0.00029	MPC24m50, MPC24m50.out
1.0	0.93265	0.00030	MPC24-2c, MPC24-2c.out

Mist conditions were also simulated in the region between the overpack and MPC (this is not a sealed space due to a built in ventilation system). As before, the mist condition represents a moderator density range of 0.02 to 0.1 g/cm^3. Table 7.1-5 displays k_{eff} as a function of moderator density between the overpack and MPC for 4.0 wt% enrichment, flooded inside and outside cask conditions, and an infinite cask array (30 cm separation). The results are within 2 sigma and below the USL.

Table 7.1-5 MPC-24 with Varied Moderator Density between Overpack and MPC

Moderator density between overpack and MPC (g/cm^3)	k_{eff}	σ	MCNP files
0.02	0.93340	0.00028	MPC24m2a, MPC24m2a.out
0.1	0.93270	0.00029	MPC24m1a, MPC24m1a.out
1.0	0.93265	0.00030	MPC24-2c, MPC24-2c.out

In summary, the results consistently demonstrate that the conditions outside the overpack (e.g., spacing, moderation, reflection) have no discernable impact on the reactivity of the cask. This indicates that the casks are neutronically isolated and consequently the cask orientation (e.g., vertical versus horizontal) will not matter.

7.1.1 21-PWR Site-Specific Canister/Basket Evaluation

The aging cask subsystem shall provide site-specific metal and/or concrete ventilated overpacks for vertical aging of disposable canisters, which are loaded with CSNF.

The internal and external moderator conditions of the site-specific cask with concrete overpack were altered in order to find the most reactive configuration for the Aging Facility. The scenarios considered include dry or flooded inside the cask (i.e., inside the canister and the gap between the canister and overpack) with a dry or flooded outside cask environment. The calculations feature 5 wt% enrichment fresh fuel and reflective boundaries with 30 cm radial separation.

The results from the calculations are presented in Table 7.1-6. Note that the k_{eff} values for flooded inside cask conditions exceed the USL because 5 wt% enriched fuel was used in the calculations. These results are only intended to show the most reactive configuration and not to produce an absolute k_{eff} value. It can be seen from the results that the highest k_{eff} value is for fully flooded inside cask conditions

Table 7.1-6 PWR Canister inside Concrete Overpack with Various Moderator Conditions

Moderation conditions	k_{eff}	σ	MCNP files
dry inside cask, dry outside cask	0.41229	0.00021	scpcdd2, scpcdd2.out
dry inside cask, flooded outside cask	0.41209	0.00023	scpcdw2, scpcdw2.out
flooded inside cask, dry outside cask	1.12942	0.00054	scpcwd2, scpcwd2.out
flooded inside cask, flooded outside cask	1.12942	0.00054	scpcww2, scpcww2.out

To ensure neutronic decoupling between the casks, the radial distance of the casks was altered. The most reactive configuration, based on Table 7.1-6, was used and the radial distances were varied for an infinite array of casks virtually touching each other (0.2 cm separation) to a 60 cm separation distance (cask surface to surface). The results displayed in Table 7.1-7 indicated that the concrete overpack ensures no neutronic interaction between the casks.

Table 7.1-7 PWR Canister inside Concrete Overpack with Various Separation Distances

Distance between casks (cm)	k_{eff}	σ	MCNP files
0.2	1.12942	0.00054	scpcww4, scpcww4.out
30	1.12942	0.00054	scpcww2, scpcww2.out
60	1.12942	0.00054	scpcww3, scpcww3.out

For defense in depth the PWR canister (5 wt% enrichment fuel without burnup credit) inside metal overpack is flooded. The criticality calculation with various moderator conditions is presented in Table 7.1-8. The results are not much different from that of concrete overpack in Table 7.1-6.

Table 7.1-8 PWR Canister inside Metal Overpack with Various Moderator Conditions

Moderation conditions	k_{eff}	σ	MCNP files
dry inside cask, dry outside cask	0.43580	0.00024	scpmdd2, scpmdd2.out
dry inside cask, flooded outside cask	0.42691	0.00023	scpmdw2, scpmdw2.out
flooded inside cask, dry outside cask	1.12827	0.00058	scpmwd2, scpmwd2.out
flooded inside cask, flooded outside cask	1.12827	0.00058	scpmww2, scpmww2.out

7.2 MPC-68 (BWR FUEL)

Table 7.2-1 shows the k_{eff} values of the MPC-68 (BWR fuel) with varied initial enrichment. The calculation features an infinite array of casks fully flooded inside and 30 cm of water reflection outside. It can be seen that in order for the resulting k_{eff} to remain below the USL, the maximum fuel loading is 4.2 wt% enriched fuel. This is consistent with the recommendations in the Certificate of Compliance (Holtec International 2002, p. 6.2-42). If a higher enrichment will be considered (i.e., 4.5 and 5.0 wt%), a higher ^{10}B loading in the Boral panel needs to be implemented or an alternate neutron poison needs to be used for the internal basket. Section 7.3 presents calculations in which the Boral loading has been increased along with an alternative neutron poison.

Table 7.2-1 MPC-68 with Varied Fuel Enrichment

Enrichment (wt%)	k_{eff}	σ	MCNP files
4.2	0.93697	0.00028	MPC68-2, MPC68-2.out
4.5	0.95145	0.00026	MPC68-45, MPC68-45.out
5.0	0.97380	0.00032	MPC68-5, MPC68-5.out

The inside and outside moderator conditions of the MPC-68 were altered in order to find the most reactive configuration for the casks. The scenarios considered include flooded inside of the cask (i.e., inside the MPC and between the MPC and overpack) with a dry and flooded outside cask environment (i.e., outside overpack), respectively. The calculations feature 4.2 wt% fuel enrichment and reflective boundaries with 30 cm radial separation. The results from the calculations are presented in Table 7.2-2. It can be seen that the most reactive configuration is for a fully flooded cask, which is also noted in the HI-STORM FSAR (Holtec International, p. 6.4-3). As with the PWR case, partial flooding was evaluated in the HI-STORM FSAR for BWR fuel and it was demonstrated that the fully flooded condition is the most reactive (Holtec International 2002, Table 6.4.2).

Table 7.2-2 MPC-68 with Varied Moderator Condition

Moderation conditions	k_{eff}	σ	MCNP files
dry inside cask, dry outside cask	0.39362	0.00010	MPC68-1, MPC68-1.out
dry inside cask, flooded outside cask	0.39333	0.00012	MPC68-1a, MPC68-1a.out
flooded inside cask, dry outside cask	0.93697	0.00028	MPC68-2a, MPC68-2a.out
flooded inside cask, flooded outside cask	0.93697	0.00028	MPC68-2, MPC68-2.out

The insensitivity of the outside environment of the MPC-68 can further be confirmed by calculating mist outside conditions, i.e., an outside moderator range of 0.02 to 0.1 g/cm^3. Table 7.2-3 displays k_{eff} as a function of outside cask moderator density for 4.2 wt% enrichment, flooded inside cask conditions, and an infinite cask array (30 cm separation). It can be seen that the k_{eff} value for a fully flooded cask is independent of the external moderator. The same observations were made in the HI-STORM FSAR (Holtec International 2002, Table 6.4.1).

Table 7.2-3 MPC-68 with Varied Outside Moderator Densities

Outside moderator density (g/cm^3)	k_{eff}	σ	MCNP files
0.0	0.93697	0.00028	MPC68-2a, MPC68-2a.out
0.02	0.93697	0.00028	MPC68m2, MPC68m2.out
0.05	0.93697	0.00028	MPC68m5, MPC68m5.out
0.07	0.93697	0.00028	MPC68m7, MPC68m7.out
0.1	0.93697	0.00028	MPC68m1, MPC68m1.out
0.5	0.93697	0.00028	MPC68m50, MPC68m50.out
1.0	0.93697	0.00028	MPC68-2, MPC68-2.out

Mist conditions were also simulated in the region between the overpack and MPC (this is not a sealed space due to a built in ventilation system). As before, the mist condition represents a moderator density range of 0.02 to 0.1 g/cm^3. Table 7.2-4 displays k_{eff} as a function of moderator density between the overpack and MPC for 4.2 wt% enrichment, flooded inside and outside cask conditions, and an infinite cask array (30 cm separation). The results are within two times of sigma.

Table 7.2-4 MPC-68 with Varied Moderator Density between Overpack and MPC

Moderator density between overpack and MPC (g/cm^3)	k_{eff}	σ	MCNP files
0.02	0.93713	0.00028	MPC68m2a, MPC68m2a.out
0.1	0.93733	0.00028	MPC68m1a, MPC68m1a.out
1.0	0.93697	0.00028	MPC68-2, MPC68-2.out

As for the PWR evaluation, the BWR results consistently demonstrate that the conditions outside the overpack (e.g., spacing, moderation, reflection) have no discernable impact on the reactivity of the cask. Again, this indicates that the casks are neutronically isolated and consequently the cask orientation (e.g., vertical versus horizontal) will not matter.

7.2.1 44-BWR Site-Specific Canister/Basket Evaluation

Sensitivity studies were performed to observe the site-specific canister/basket performance for storage and aging operations in the Aging Facility. A brief description of the sensitivity studies performed and their results are provided.

In each of the sensitivity cases, the site-specific canister/basket dimensions correspond to those from BSC 2005m and BSC 2005n, which are provided in Attachment B. Each configuration is represented with the site-specific canister/basket emplaced inside a waste package or overpack with reflective surfaces consistent with the cases discussed in Section 6.1.5. The design information for the overpack is provided in Section 6.1.5.3, and the waste package dimensions come from BSC 2003d and BSC 2003e which are provided in Attachment B.

7.2.2 Case Descriptions and Results

A search for optimum water moderator density within the site-specific canister/basket was performed. This set of cases was used to show that the fuel assemblies placed into a site-specific canister/basket configuration is an under-moderated system. Moderator density values were varied from 0.0 g/cm^3 through 1.0 g/cm^3. The water was represented inside the site-specific canister/basket with void in the external regions. Base case values correspond to a fresh fuel assembly with 5.0 wt% U^{235} initial enrichment. The results of this set of cases are presented in Table 7.2-5 and are illustrated in Figure 7.2-1.

Table 7.2-5 Moderator Density Sensitivity Results

Moderator Density (g/cm^3)	Overpack		Filename
	k_{eff}	σ	
0.0	0.34583	0.00020	5.0vv
0.1	0.52719	0.00036	5.0a
0.2	0.63388	0.00046	5.0b
0.3	0.71283	0.00046	5.0c
0.4	0.77313	0.00050	5.0d
0.5	0.81786	0.00057	5.0e
0.6	0.85400	0.00049	5.0f
0.7	0.88269	0.00056	5.0g
0.8	0.90722	0.00054	5.0h
0.9	0.92636	0.00055	5.0i
1.0	0.94154	0.00057	5.0wvv

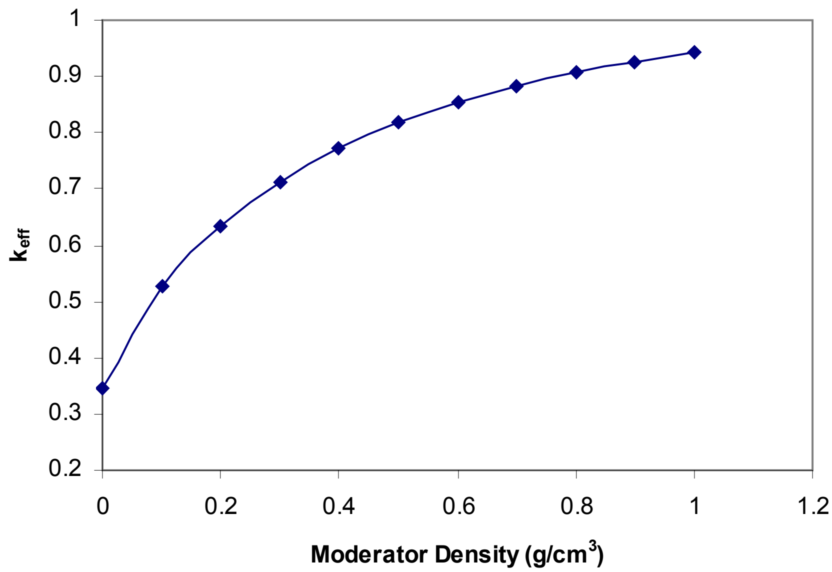


Figure 7.2-1 Moderator Density Sensitivity Results

Additional cases observing k_{eff} under different reflector conditions was also performed. The cases evaluated are for water, void, or both in different regions within the configuration. The results of this set of cases are presented in Table 7.2-6.

Table 7.2-6 Reflector Sensitivity Results

Case Description	Overpack		Filename
	k_{eff}	σ	
dry inside and dry outside of the site-specific canister	0.34583	0.0002	5.0vv
dry inside and flooded outside of the site-specific canister	0.32965	0.00018	5.0vw
flooded inside and dry outside of the site-specific canister	0.94154	0.00057	5.0wvv
flooded inside and flooded outside of the site-specific canister	0.94210	0.00052	5.0ww

The results show that the cases with or without moderator present are slightly above or way below the USL value of 0.9394. Under normal operations, it is expected that there will not be any moderator present within the site-specific canister/basket. The fully flooded cases are only evaluated to demonstrate the defense in depth features of the design. The corresponding MCNP input and output files for the cases used in this evaluation are provided electronically in Attachment B in ASCII format. Filenames with an "O" at the end are output files.

7.3 SITE-SPECIFIC CASK FOR COMMERCIAL FUEL

It was shown in Sections 7.1 and 7.2 that when loading the MPC-24 and MPC-68 with 5.0 wt% enriched fuel, k_{eff} exceeds the USL. The site-specific cask must be able to accommodate 5.0 wt% enriched fuel. One way to accomplish this is to increase the neutron poison in the storage/aging casks. Table 7.3-1 displays k_{eff} as function of Boral loading for both the MPC-24 and MPC-68. Complete data was not available regarding the possible Boral configurations, but Achudume (2004) indicates that there are limitations to g $^{10}B/cm^2$ loading (due to B_4C-to-Al ratio) as well as Boral plate thickness that can be manufactured. The upper limit of the ^{10}B loading is currently approximate 0.04 g $^{10}B/cm^2$. Increases in ^{10}B loading above this limit would lead to diminishing returns in neutron absorption capabilities since ^{10}B in the Boral plates would have reached saturation point (Achudume 2004). The HI-STORM FSAR material specifications for Boral imply that the B_4C-to-Al ratio is approximately between 35/65 to 40/60 (see Excel file *boral.xls* for B_4C-to-Al ratio calculations), which was implemented for the calculations presented in the table. There are also some calculations featuring a higher B_4C-to-Al ratio (80/20). While this composition might be unrealistic to manufacture, the results from these calculations were included to demonstrate the diminishing returns in neutron absorption capabilities of the Boral plates above a certain B_4C-to-Al ratio. Table 7.3-1 also shows that neither the MPC-24 nor the MPC-68 can hold 5 wt% enriched fuel, even with an increased ^{10}B loading, and still be below the USL. Utilizing Boral in the MPC as a fixed neutron absorber to accommodate 5 wt% enriched fuel, more information regarding possible ^{10}B loading and panel thickness from the manufacturer is needed so that a safe loading can be identified.

Table 7.3-1 k_{eff} as a Function of Boral Loading in Storage Casks

Boral loading (g $^{10}B/cm^2$)	Boral thickness (cm)	B_4C/Al ratio (%)	k_{eff}	σ	MCNP files
MPC-24 Cask – PWR Fuel (5 wt% enriched fuel)					
0.02	0.1397	39/61	0.97211	0.00026	MPC24-2, MPC24-2.out
0.04	0.1397	78/22	0.95857 [a]	0.00029	MPC24B4, MPC24B4.out
0.027	0.2057	36/64	0.96975	0.00030	MPC24B63, MPC24B63.out
0.031	0.2057	41/59	0.96751	0.00029	MPC24B64, MPC24B64.out
0.06	0.2057	80/20	0.95429	0.00028	MPC24B6, MPC24B6.out
0.035	0.2717	35/65	0.96920	0.00029	MPC24B83, MPC24B83.out
0.04	0.2717	40/60	0.96602	0.00029	MPC24B84, MPC24B84.out
0.08	0.2717	80/20	0.95202	0.00027	MPC24B8, MPC24B8.out
MPC-68 Cask – BWR Fuel (5 wt% enriched fuel)					
0.0279	0.2057	37/63	0.97380	0.00032	MPC68-5, MPC68-5.out
0.031	0.2057	41/59	0.96785	0.00029	MPC68B3, MPC68B3.out
0.04	0.2057	53/47	0.95367	0.00026	MPC68B4, MPC68B4.out

[a] This case was also computed with a calculated ^{10}B content based on atom fraction (see Section 6.1.5.1) and produced a k_{eff} of 0.95937 ± 0.00030 (MCNP files: MPC24B4t & MPC24B4t.out). Note that the two k_{eff} values are within the statistical uncertainty.

Even though the site-specific cask is similar in design to an existing storage cask design (Assumption 3.2.2), a solution to be able to store 5.0 wt% enriched fuel is to exchange the internal basket. Instead of utilizing Boral panels, the internal basket could consist of B_4C aligned by SS similar to the GA-9 cask design (General Atomics 1993, p. 6.3-2). Calculations were performed for the MPC-24 and MPC-68 with 0.2717 cm and 0.2057 cm, respectively, thick Boral panels exchanged to B_4C. This is a conservative approximation since the GA-9 cask consists of B_4C for the full width of the fuel assembly while the Boral panels only covers partial width of the fuel assembly. Table 7.3-2 presents the results and it can be seen that the k_{eff} is below the USL for both casks. Consequently, B_4C could be used as a neutron poison for the internal basket to accommodate 5.0 wt% enriched PWR fuel. Further studies, however, would need to be performed to determine the internal basket layout and dimensions before implementing B_4C into the site-specific cask design.

Table 7.3-2 MPC-24 and MPC-68 with B_4C Neutron Poison

Neutron poison material	k_{eff}	σ	MCNP files
MPC-24 – PWR Fuel (5 wt% enriched fuel)			
B_4C	0.94647	0.00028	MPC24b4c, MPC24b4c.out
MPC-68 – BWR Fuel (5 wt% enriched fuel)			
B_4C	0.90997	0.00029	MPC68b4c, MPC68b4c.out

Additional criticality control mechanisms exist, in addition to increasing the neutron poison, that can be varied to ensure that the site-specific cask can accommodate 5.0 wt% enriched fuel. As stated in Section 6.2.2, the number of assemblies contained in the site-specific cask can be reduced, fuel spacing can be increased and burnup-credit nuclides can be included in the fuel composition. Table 7.3-3 shows the results from the variations in the criticality control mechanisms, including reduction of number of fuel assemblies and increased fuel assembly spacing. It can be seen that reducing the number of assemblies is not very efficient to reduce k_{eff}. Increasing the fuel assembly spacing is a lot more efficient reducing the k_{eff} to below the USL. Note that a 1 cm increase in the fuel spacing requires a slightly larger inside overpack diameter to properly accommodate the fuel and fuel baskets. Table 7.3-4 shows the impact on k_{eff} by including burnup-credit nuclides in the fuel composition. It can be seen that including the actinides in the fuel composition for low burnups (conservative approximation) also proves to be effective in reducing k_{eff} to an acceptable value for both PWR and BWR fuel. Note that the Westinghouse 17x17 OFA fuel assembly is slightly more reactive when applying burnup-credit (5 wt% initial enrichment) than the B&W 15x15 as presented in Table 7.3-4. The Westinghouse 17x17 OFA fuel assembly is also more reactive in the MPC-24 for fresh fuel evaluations (see footnote 'a' of Table 7.3-4). The GE 8x8 fuel assembly is the more reactive BWR fuel assembly when applying burnup-credit (5 wt% initial enrichment). In addition, it is more reactive in the MPC-68 for fresh fuel calculations as well (see footnote 'b' of Table 7.3-4). All calculations presented in the tables below includes a Boral loading of 0.04 g $^{10}B/cm^2$ (0.1397 cm Boral panel thickness) for PWR fuel and 0.031 g $^{10}B/cm^2$ (0.2057 cm Boral panel thickness) for BWR fuel.

Table 7.3-3 Criticality Control Variations for MPC-24 and MPC-68

Scenario Description	MPC-24 (PWR Fuel)			MPC-68 (BWR Fuel)		
	k_{eff}	σ	MCNP files	k_{eff}	σ	MCNP files
	Reduced Number of Assemblies					
20 PWR/ 60 BWR [c]	0.95955	0.00028	MPC24B84 MPC24B84.out	0.96377	0.00028	MPC68B3W MPC68B3W.out
12 PWR/ 48 BWR [c]	0.95466	0.00027	MPC24B12 MPC24B12.out	0.95487	0.00028	MPC68BW1 MPC68BW1.out
	Increased Fuel Spacing					
+ 0.5 cm [a]	0.96146	0.00029	MP24B84S MP24B84S.out	0.94208	0.00028	MP68B3S5 MP68B3S5.out
+ 1.0 cm [b]	0.93944	0.00029	MP24B1S MP24B1S.out	0.91545	0.00026	MPC68B3S MPC68B3S.out

[a] Increased PWR assembly pitch is 28.20124 cm (11.1 in) and 8.1111 cm (3.2 in) for the BWR assembly.
[b] Increased PWR assembly pitch is 28.70124 cm (11.3 in) and 8.6111 cm (3.4 in) for the BWR assembly.
[c] Removed assemblies from the peripheral locations.

Table 7.3-4 Burnup-Credit Evaluations for PWR and BWR Fuel

Burnup (GWd/MTU)	k_{eff}	σ	MCNP files	k_{eff}	σ	MCNP files
	Use of Burnup Credit – PWR fuel					
	W 17x17 OFA			**B&W 15x15 [a]**		
10	0.90453	0.00026	M24B10BU M24B10BU.out	0.90065	0.00030	M15B10pi M15B10pi.out
20	0.85609	0.00027	M24B20BU M24B20BU.out	0.85216	0.00028	M15B20pi M15B20pi.out
30	0.81069	0.00025	M24B30BU M24B30BU.out	0.80743	0.00029	M15B30pi M15B30pi.out
	Use of Burnup Credit – BWR fuel					
	GE 8x8			**GE 7x7 [b]**		
10	0.91778	0.00026	M68B10BU M68B10BU.out	0.90747	0.00027	M7B10pi M7B10pi.out
20	0.88582	0.00024	M68B20BU M68B20BU.out	0.87429	0.00026	M7B20pi M7B20pi.out
30	0.85095	0.00026	M68B30BU M68B30BU.out	0.83938	0.00026	M7B30pi M7B30pi.out

[a] Note that k_{eff} is 0.92589 ± 0.00028 (MCNP files: MPCbw15 & MPCbw15.out) for fresh B&W 15x15 fuel (4.0 wt% enrichment & 0.02 g ^{10}B/cm^2 Boral loading) in the MPC-24, which is less than k_{eff} of W 17x17 OFA (see Table 7.1-1 for comparison).

[b] Note that k_{eff} is 0.92935 ± 0.00027 (MCNP files: M7x7-2 & M7x7-2.out) for fresh GE 7x7 fuel (4.2 wt% enrichment & 0.0279 g ^{10}B/cm^2 Boral loading) in the MPC-68, which is less than k_{eff} of GE 8x8 fuel (see Table 7.2-1 for comparison).

It can be seen that a slightly higher burnup is needed to safely include BWR fuel with 5 wt% initial enrichment in the MPC-68 than required for the PWR fuel for storage in the MPC-24. Also note that k_{eff} is significantly reduced by taking credit for all principal isotopes associated with commercial SNF burnup.

7.4 SITE-SPECIFIC CASK FOR DOE FUEL CANISTERS

Criticality calculations for Enrico Fermi, Fort St. Vrain, and FFTF canisters inside the site-specific cask are performed in this section. The moderator condition inside the site-specific cask is varied between wet and dry. The external of the site-specific cask is varied between water and air as the reflector and with the reflective boundary to simulate the infinite array of the aging casks. Table 7.4-1 presents the DOE fuel types placed inside the site-specific cask (15 in thick concrete overpack with an inside diameter of 69.5 in) in a 3x3 square pitch canister array (canisters are touching each other). It can be seen that the k_{eff}'s are below the USL for all three DOE fuel types. In addition, the Enrio Fermi calculations show that the k_{eff} of the site-specific cask is independent on distance to the next site-specific cask and outside conditions, which means the site-specific cask is neutronically isolated.

Table 7.4-1 k_{eff} of Various DOE Canisters Inside a Site-Specific Cask

Distance (cm)	k_{eff} (Air)	σ	MCNP files	k_{eff} (Water)	σ	MCNP files
Enrico Fermi (3x3 square pitch array), wet internally						
0.2	0.64873	0.00079	fmwcsa2 fmwcsa2.out	0.64839	0.00077	fmwcsw2 fmwcsw2.out
30	0.64811	0.00077	fmwcsa1 fmwcsa1.out	0.64839	0.00077	fmwcsw1 fmwcsw1.out
Enrico Fermi (3x3 square pitch array), dry internally						
0.2	0.69818	0.00060	fmdcsa2 fmdcsa2.out	0.69861	0.00059	fmdcsw2 fmdcsw2.out
30	0.69921	0.00060	fmdcsa1 fmdcsa1.out	0.69853	0.00055	fmdcsw1 fmdcsw1.out
FFTF (3x3 square pitch array), wet internally						
0.2	0.78127	0.00087	fftpsw2 fftpsw2.out	0.78178	0.00088	fftpsw1 fftpsw1.out
FFTF (3x3 square pitch array), dry internally						
0.2	0.72176	0.00061	fftpsd2 fftpsd2.out	0.71949	0.00059	fftpsd1 fftpsd1.out
Fort St. Vrain (3x3 square pitch array), wet internally						
0.1	0.77897	0.00083	fsvasw2 fsvasw2.out	0.77929	0.00087	fsvasw1 fsvasw1.out
Fort St. Vrain (3x3 square pitch array), dry internally						
0.1	0.42819	0.00061	fsvasd2 fsvasd2.out	0.42508	0.00064	fsvasd1 fsvasd1.out

Table 7.4-2 shows the FFTF fuel in a larger overpack (inside diameter of 77.5 in) with a higher number of canisters placed in a close-packed triangular pitch array (see Figure 6.1-9). While k_{eff} increases somewhat, there is still no criticality concern. The smaller overpack (inside diameter of 69.5 in) was also used to calculate k_{eff} of FFTF fuel surrounded by 5 SRS HLW glass canisters (diameter of 24 in). As expected, and shown in Table 7.4-2, this configuration is subcritical. In addition, the 3x3 square pitch arrangement for FFTF fuel is compared to that of a circular pitch (eight canisters in a circle and one in the center per Figure 6.1-10); the former shows a little higher k_{eff}.

Table 7.4-2 k_{eff} of Various FFTFCanister Configurations

Distance (cm)	Number of Canisters	k_{eff}	σ	MCNP files
Overpack Inside Diameter = 77.5 in (Triangular Pitch), wet internally				
30 (water)	10	0.79069	0.00083	ffwctw1, ffwctw1.out
30 (water)	12	0.79521	0.00090	ffwctw2, ffwctw2.out
Overpack Inside Diameter = 77.5 in (Triangular Pitch), dry internally				
30 (water)	10	0.74986	0.00062	ffdctw1, ffdctw1.out
30 (water)	12	0.80125	0.00065	ffdctw2, ffdctw2.out
Overpack Inside Diameter = 69.5 in (Circular Pitch), wet internally				
30 (air)	9	0.77003	0.00088	ffwcca1, ffwcca1.out
30 (air)	1 [a]	0.75480	0.00084	ffwcca2, ffwcca2.out
Overpack Inside Diameter = 69.5 in (Circular Pitch), dry internally				
30 (air)	9	0.68989	0.00060	ffdcca1, ffdcca1.out
30 (air)	1 [a]	0.42027	0.00039	ffdcca2, ffdcca2.out

[a] Canister placed in the center of site-specific cask surrounded by 5 SRS HLW glass canisters (Figure 6.1-10)

FFTF and Fort St. Vrain cases except one (ffdctw2, FFTF Triangular 12 canisters) indicate a lower k_{eff} when it is dry inside the site-specific cask. All Enrico Fermi cases indicate a higher k_{eff} when it is dry inside the site-specific cask. For the Enrico Fermi cases, when flooded inside, water is introduced in the gaps where iron shot containing $GdPO_4$ are filled. The neutron absorption of the gadolinium poison is more effective in a wet condition. The result depends on both neutron moderation and absorption of the configuration.

7.5 EVALUATIONS OF CATEGORY 1 AND 2 EVENT SEQUENCES

No Category 1 and 2 event sequences applicable to Aging Facility have been identified (BSC 2005e, Section 7). Per the discussion presented in Section 6.2.4, design basis accidents have been evaluated for the HI-STORM 100 cask system and were found to be within the criticality safety design limits. In addition, defense-in-depth calculations were performed for potential drop or slap down scenarios. The nominal representation of the event proved to be within the criticality safety design limits.

If a drop or collision of a site-specific cask would cause the internal basket structure to fail completely, the pin pitch of the assembly could expand. Table 6.2-2 shows that with only 30 cm damaged fuel height and fully flooded conditions, the k_{eff} could exceed the USL. Therefore, during drop or slap down scenarios, the internal basket structure of site-specific cask must remain intact, or moderator control is required in order to maintain criticality safety.

7.6 CONCLUSIONS AND RECOMMENDATIONS

The Aging Facility and its processes have been evaluated for criticality safety for normal operations, Category 1 and 2 event sequences. The results presented in this document lead to the following conclusions and recommendations:

- The MPC-24 with overpack used as a representative aging cask for PWR fuel assemblies can be stored safely on the aging pads with fuel content per the Certificate of Compliance (maximum 4.0 wt% fuel enrichment). The MPC-68 with overpack used as a representative aging cask for BWR fuel assemblies can also be stored safely on the aging pads with fuel contents up to 4.2 wt% enrichment while remaining below USL.

- Mist conditions (i.e., moderator densities between 0.02 to 0.1 g/cm^3) surrounding the outside of the casks do not vary the k_{eff} much from that of fully flooded outside surroundings (i.e., moderator density of 1.0 g/cm^3). Results show that a fully flooded internal cask is independent of the external moderator.

- 21-PWR site-specific canister was evaluated containing 5 wt% enriched fresh fuel inside both concrete and metal overpacks under dry condition, the k_{eff} is well below the USL (Section 7.1.1). However, when fully flooded, the k_{eff} exceeds the USL. The maximum fresh fuel enrichment that would meet the loading curve criteria is 2.137 wt% U235 for the preclosure bounding configuration (BSC 2005g, Section 6.1). Any higher fuel enrichment requires burnup credit. The minimum required burnup for 5 wt% initial enrichment fuel is 36.38 GWd/MTU, which includes 5% burnup record uncertainty (BSC 2005g, Table 21). The condition outside overpacks and the separation distances have no discernable impact on the results. Note that the site-specific canister shall be sealed to prevent moderator from entering (Section 6.1.1). No Category 1 and Category 2 event sequences applicable to the Aging Facility have been identified, therefore the possibility of a breached canister being flooded is beyond Category 2. In addition, moderator control is required at the aging pads. Flood drainage channels will surround the aging pads that are sized to carry away water from a design basis flood. This precludes the possibility that site-specific casks could be subject to the probable maximum flood (BSC2005c, Section 4.1.1.1).

- 44-BWR site-specific canister featuring 5 wt% enrichment fresh fuel inside the concrete overpack, the k_{eff} is below the USL when dry, and with various moderator densities less than one inside (Section 7.2.2). However, when flooded inside and with moderator density equal to one, the k_{eff} slightly exceeds the USL. The same moderator control as the above item is required here.

- The results for PWR, BWR, and DOE fuel canisters calculations when placed inside the aging cask consistently demonstrate that the conditions outside the overpack (e.g., spacing, moderation, reflection) have no discernable impact on the reactivity of the cask. This indicates that the site-specific casks are neutronically isolated and consequently the cask orientation (e.g., vertical versus horizontal) will not matter.

- In order to accommodate 5.0 wt% enriched commercial fuel in the site-specific cask, another neutron poison besides Boral might be included in the internal basket. To utilize Boral in the site-specific cask as a fixed neutron absorber, more information from the manufacturer needs to be obtained regarding possible ^{10}B loading options as well as panel thickness so that a safe loading can be identified. This analysis shows that B_4C would be acceptable as an internal basket material to meet the USL. If B_4C is chosen as a neutron poison for the site-specific cask, the exact dimensions and B_4C contents need to be evaluated during the detailed design phase. Other alternatives include larger assembly separation (1 cm addition) and taking credit for low burnups.

ATTACHMENT A

LISTING OF COMPUTER FILES

This attachment lists the input and output file names for the MCNP and Excel calculations. All files are stored on an electronic medium (compact disc, attachment B).

```
Volume in drive D is 060606_1530
 Volume Serial Number is 0259-43EC

 Directory of D:\

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05/22/2006  10:12a           991,213 scpcdw2.out
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05/22/2006  10:12a           867,200 scpcwd2.out
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08/17/2005  07:56a             467,678 5.0bO
08/17/2005  07:56a             111,699 5.0c
08/17/2005  07:56a             466,842 5.0cO
08/17/2005  07:56a             111,723 5.0d
08/17/2005  07:56a             463,922 5.0dO
08/17/2005  07:56a             111,723 5.0e
08/17/2005  07:56a             466,286 5.0eO
08/17/2005  07:56a             111,723 5.0f
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08/17/2005  07:56a             111,723 5.0g
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08/17/2005  07:56a             111,723 5.0i
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08/17/2005  07:56a             111,596 5.0vw
08/17/2005  07:56a             111,742 5.0ww
08/17/2005  07:56a             474,474 5.0vvO
08/17/2005  07:56a             474,370 5.0vwO
08/18/2005  11:50a             111,727 5.0wvv
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08/13/2004  04:05p               2,972 bwr0975
08/13/2004  04:05p             279,360 bwr0975.out
08/13/2004  04:05p               2,972 bwr100
08/13/2004  04:05p             279,566 bwr100.out
08/13/2004  04:05p               2,976 bwr8x85
08/13/2004  04:05p             279,360 bwr8x85.out
              12 File(s)      1,694,408 bytes

 Directory of D:\PWR

08/29/2005  03:11p      <DIR>          .
08/29/2005  03:11p      <DIR>          ..
07/20/2005  12:03p      <DIR>          BORAL
07/20/2005  12:03p      <DIR>          MIST
```

```
04/05/2004  09:58a              18,990 MPC24-2
04/05/2004  09:58a             527,686 MPC24-2.out
04/05/2004  09:58a              18,990 MPC24-2a
04/05/2004  09:58a             527,686 MPC24-2a.out
04/05/2004  09:58a              18,992 MPC24-2b
04/05/2004  09:58a             527,784 MPC24-2b.out
04/05/2004  09:58a              18,982 MPC24-2c
04/05/2004  09:58a             527,990 MPC24-2c.out
04/05/2004  09:58a              18,994 MPC24-2d
04/05/2004  09:58a             527,990 MPC24-2d.out
              10 File(s)      2,734,084 bytes

 Directory of D:\PWR\BORAL

07/20/2005  12:03p      <DIR>          .
07/20/2005  12:03p      <DIR>          ..
04/07/2004  09:55a              19,018 MPC24B4
04/07/2004  09:55a             527,686 MPC24B4.out
04/05/2004  09:58a              19,122 MPC24b4c
04/05/2004  09:58a             528,502 MPC24b4c.out
04/07/2004  09:55a              19,173 MPC24B6
04/07/2004  09:55a             528,495 MPC24B6.out
04/07/2004  09:55a              19,165 MPC24B63
04/07/2004  09:55a             528,544 MPC24B63.out
04/07/2004  09:55a              19,164 MPC24B64
04/07/2004  09:55a             528,495 MPC24B64.out
04/07/2004  09:55a              19,181 MPC24B8
04/07/2004  09:55a             528,544 MPC24B8.out
04/07/2004  09:55a              19,196 MPC24B83
04/07/2004  09:55a             528,603 MPC24B83.out
04/07/2004  09:55a              19,182 MPC24B84
04/07/2004  09:55a             528,446 MPC24B84.out
              16 File(s)      4,380,516 bytes

 Directory of D:\PWR\MIST

07/20/2005  12:03p      <DIR>          .
07/20/2005  12:03p      <DIR>          ..
04/05/2004  09:58a              18,995 MPC24m0
04/05/2004  09:58a             527,888 MPC24m0.out
04/05/2004  09:58a              18,910 MPC24m1
04/05/2004  09:58a             528,412 MPC24m1.out
04/07/2004  09:00a              18,930 MPC24m1a
04/07/2004  09:01a             528,010 MPC24m1a.out
04/05/2004  09:58a              18,912 MPC24m2
04/05/2004  09:58a             528,206 MPC24m2.out
04/05/2004  09:58a              18,930 MPC24m2a
04/05/2004  09:58a             528,010 MPC24m2a.out
04/05/2004  09:58a              18,912 MPC24m3
04/05/2004  09:58a             528,363 MPC24m3.out
04/05/2004  09:58a              18,911 MPC24m5
04/05/2004  09:58a             528,412 MPC24m5.out
04/05/2004  09:58a              18,910 MPC24m50
04/05/2004  09:58a             528,059 MPC24m50.out
04/05/2004  09:58a              18,912 MPC24m7
04/05/2004  09:58a             528,412 MPC24m7.out
04/05/2004  09:58a              18,912 MPC24m8
```

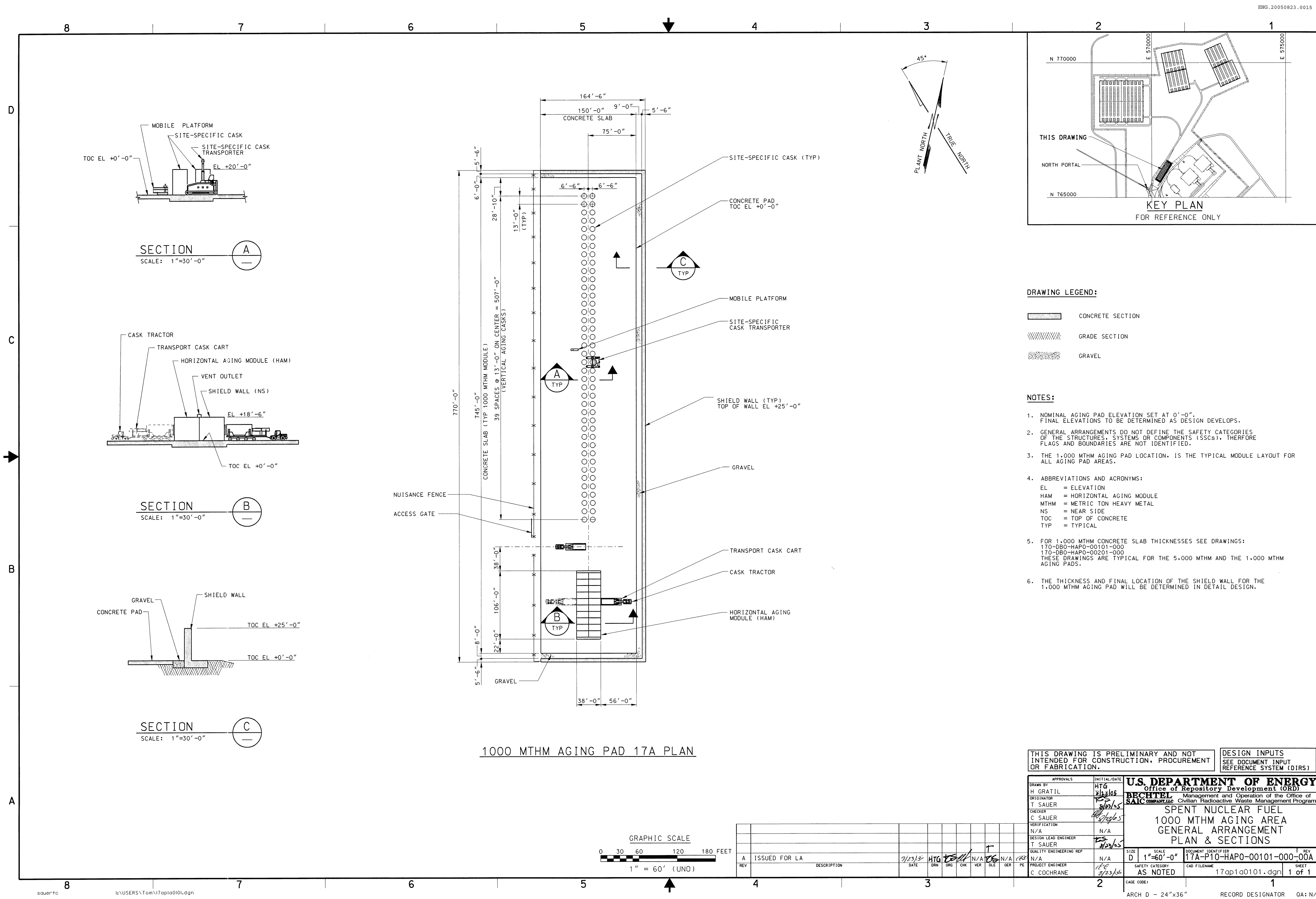
```
04/07/2004  09:00a             528,412 MPC24m8.out
              20 File(s)      5,471,418 bytes

     Total Files Listed:
             265 File(s)    196,537,899 bytes
              27 Dir(s)               0 bytes free
```

ATTACHMENT B

ONE COMPACT DISK CONTAINING ALL FILES LISTED IN ATTACHMENT A

ATTACHMENT C

DRAWING OF AGING AREA GENERAL ARRANGEMENT PLAN AND SECTIONS


ENG.20050823.0015
SECTION A
SCALE: 1"=30'-0"
MOBILE PLATFORM
SITE-SPECIFIC CASK
SITE-SPECIFIC CASK TRANSPORTER
TOC EL +0'-0"
EL +20'-0"
SECTION B
SCALE: 1"=30'-0"
CASK TRACTOR
TRANSPORT CASK CART
HORIZONTAL AGING MODULE (HAM)
VENT OUTLET
SHIELD WALL (NS)
EL +18'-6"
TOC EL +0'-0"
SECTION C
SCALE: 1"=30'-0"
GRAVEL
CONCRETE PAD
SHIELD WALL
TOC EL +25'-0"
TOC EL +0'-0"
164'-6"
150'-0"
CONCRETE SLAB
9'-0"
5'-6"
75'-0"
6'-6"
6'-6"
770'-0"
745'-0" CONCRETE SLAB (TYP 1000 MTHM MODULE)
39 SPACES @ 13'-0" ON CENTER = 507'-0" (VERTICAL AGING CASKS)
28'-10"
13'-0" (TYP)
38'-0"
106'-0"
8'-0"
22'-0"
5'-6"
38'-0"
56'-0"
NUISANCE FENCE
ACCESS GATE
GRAVEL
SITE-SPECIFIC CASK (TYP)
CONCRETE PAD TOC EL +0'-0"
MOBILE PLATFORM
SITE-SPECIFIC CASK TRANSPORTER
SHIELD WALL (TYP) TOP OF WALL EL +25'-0"
GRAVEL
TRANSPORT CASK CART
CASK TRACTOR
HORIZONTAL AGING MODULE (HAM)
1000 MTHM AGING PAD 17A PLAN
45°
PLANT NORTH
TRUE NORTH
N 770000
E 570000
E 575000
THIS DRAWING
NORTH PORTAL
N 765000
KEY PLAN
FOR REFERENCE ONLY
DRAWING LEGEND:
CONCRETE SECTION
GRADE SECTION
GRAVEL
NOTES:
1. NOMINAL AGING PAD ELEVATION SET AT 0'-0". FINAL ELEVATIONS TO BE DETERMINED AS DESIGN DEVELOPS.
2. GENERAL ARRANGEMENTS DO NOT DEFINE THE SAFETY CATEGORIES OF THE STRUCTURES, SYSTEMS OR COMPONENTS (SSCs), THERFORE FLAGS AND BOUNDARIES ARE NOT IDENTIFIED.
3. THE 1,000 MTHM AGING PAD LOCATION, IS THE TYPICAL MODULE LAYOUT FOR ALL AGING PAD AREAS.
4. ABBREVIATIONS AND ACRONYMS:
EL = ELEVATION
HAM = HORIZONTAL AGING MODULE
MTHM = METRIC TON HEAVY METAL
NS = NEAR SIDE
TOC = TOP OF CONCRETE
TYP = TYPICAL
5. FOR 1,000 MTHM CONCRETE SLAB THICKNESSES SEE DRAWINGS: 170-DB0-HAP0-00101-000 170-DB0-HAP0-00201-000 THESE DRAWINGS ARE TYPICAL FOR THE 5,000 MTHM AND THE 1,000 MTHM AGING PADS.
6. THE THICKNESS AND FINAL LOCATION OF THE SHIELD WALL FOR THE 1,000 MTHM AGING PAD WILL BE DETERMINED IN DETAIL DESIGN.
THIS DRAWING IS PRELIMINARY AND NOT INTENDED FOR CONSTRUCTION, PROCUREMENT OR FABRICATION.
DESIGN INPUTS
SEE DOCUMENT INPUT REFERENCE SYSTEM (DIRS)
U.S. DEPARTMENT OF ENERGY
Office of Repository Development (ORD)
BECHTEL SAIC COMPANY, LLC Management and Operation of the Office of Civilian Radioactive Waste Management Program
SPENT NUCLEAR FUEL
1000 MTHM AGING AREA
GENERAL ARRANGEMENT
PLAN & SECTIONS
DRAWN BY H GRATIL
ORIGINATOR T SAUER
CHECKER C SAUER
VERIFICATION N/A
DESIGN LEAD ENGINEER T SAUER
QUALITY ENGINEERING REP N/A
PROJECT ENGINEER C COCHRANE
SIZE D
SCALE 1"=60'-0"
DOCUMENT IDENTIFIER 17A-P10-HAP0-00101-000-00A
SAFETY CATEGORY AS NOTED
CAD FILENAME 17ap1a0101.dgn
SHEET 1 of 1
ARCH D - 24"x36"
RECORD DESIGNATOR
QA: N/A
A ISSUED FOR LA
REV DESCRIPTION DATE DRN ORG CHK VER QLE DER PE
GRAPHIC SCALE
0 30 60 120 180 FEET
1" = 60' (UNO)
sauertc
I:\USERS\Tom\17ap1a0101.dgn