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

3. System SNF Aging System	4. Document Identifier 170-00C-HA00-00100-000-00B
5. Title Aging Facility Criticality Safety Calculations	
6. Group Nuclear Analysis	
7. Document Status Designation ☒ Preliminary ☐ Final ☐ Cancelled	
8. Notes/Comments	

NOTICE OF OPEN CHANGE DOCUMENTS - THIS DOCUMENT IS IMPACTED BY THE LISTED CHANGE DOCUMENT AND CANNOT BE USED WITHOUT IT.

---

1) ECN-001, DATED 07/27/2005
2) ECN-002, DATED 08/18/2005

Attachments	Total Number of Pages
I. Listing of Computer Files	8
II. One Compact Disc Containing All Files Listed in Attachment I (1 of 1)	0
III. Sketch of the aging pad in the Aging Facility	1

RECORD OF REVISIONS

9. No.	10. Reason For Revision	11. Total # of Pgs.	12. Last Pg. #	13. Originator (Print/Sign/Date)	14. Checker (Print/Sign/Date)	15. QER (Print/Sign/Date)	16. Approved/Accepted (Print/Sign)	17. Date
00A	Initial issue	44	III-1	C.E. Sanders	D.S. Kimball	D.J. Tunney	S. Su	04/12/04
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 [signature] 09/09/2004	G. Radulescu Georgeta Radulescu 9/9/2004	W.N. Dockery W.N. Dockery 10 SEPT 2004	S. Su [signature]	9/10/04

CONTENTS

LIST OF FIGURES

Page

LIST OF TABLES

Page

LIST OF ACRONYMS AND ABBREVIATIONS

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
DOE	U.S. Department of Energy
DPC	Dual-Purpose Canister
FFTF	Fast Flux Test Facility
HLW	High Level Waste
i.d.	inner diameter
in.	inch
k_{eff}	neutron effective multiplication factor
LWBR	Light Water Breeder Reactor
MCNP	Monte Carlo N-Particle transport code
MGR	Monitored Geological Repository
MPC	Multi Purpose Canister
MSC	MGR Site specific Cask
NRC	U.S. Nuclear Regulatory Commission
o.d.	outer diameter
OFA	Optimized Fuel Assembly
PDC	Project Design Criteria
PWR	Pressurized Water Reactor
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

The purpose of this design calculation is to revise and update the previous criticality calculation for the Aging Facility (documented in BSC 2004a). This design calculation will also demonstrate and ensure 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* (Doraswamy 2004, Section 4.9.2.2), and the functional nuclear criticality safety requirement described in the *SNF Aging System Description Document* (BSC [Bechtel SAIC Company] 2004f, p. 3-12). The scope of this design calculation covers the systems and processes for aging commercial spent nuclear fuel (SNF) and staging Department of Energy (DOE) SNF/High-Level Waste (HLW) prior to its placement in the final waste package (WP) (BSC 2004f, p. 1-1). Aging commercial SNF is a thermal management strategy, while staging DOE SNF/HLW will make loading of WPs more efficient (note that aging DOE SNF/HLW is not needed since these wastes are not expected to exceed the thermal limits form emplacement) (BSC 2004f, p. 1-2). The description of the changes in this revised document is as follows:

- Include DOE SNF/HLW in addition to commercial SNF per the current *SNF Aging System Description Document* (BSC 2004f).
- Update the evaluation of Category 1 and 2 event sequences for the Aging Facility as identified in the *Categorization of Event Sequences for License Application* (BSC 2004c, Section 7).
- Further evaluate the design and criticality controls required for a storage/aging cask, referred to as MGR Site-specific Cask (MSC), 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 required for the MSC that will accommodate DOE SNF/HLW.

This design calculation will achieve the objective of providing the criticality safety results to support the preliminary design of the Aging Facility. As the ongoing design evolution remains fluid, the results from this design calculation should be evaluated for applicability to any new or modified design. Consequently, the results presented in this document are limited to the current design. The information contained in this document was developed by Environmental and Nuclear Engineering and is intended for the use of Design and Engineering in its work regarding the various criticality related activities performed in the Aging Facility. Yucca Mountain Project personnel from Environmental and Nuclear Engineering should be consulted before the use of the information for purposes other than those stated herein or use by individuals other than authorized personnel in Design and Engineering.

The SNF Aging System has been classified as safety category in the *Q-list* (BSC 2004i, p. A-7). 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 2004). Performance of the work scope as described and development of the associated technical product conform to the procedure AP-3.12Q, *Design Calculations and Analyses*.

2. METHOD

2.1 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 ensure 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 enrichment of 5.0 wt%. 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. Note that the terms "model(s)" and "modeling" as used in this calculation document refer to the geometric configurations of the criticality cases analyzed. 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 is predicated upon the most reactive fuel assemblies
- The multiplication factor (k_{eff}) will not exceed 0.95, including all biases and uncertainties in the data and method of the analysis, under all normal, and Category 1 and 2 event sequences
- A range of modeling dimensional variables will be used (e.g., assembly pitch, manufacturing tolerances for assemblies, etc.) that should provide limiting values
- Conservative modeling assumptions will also be used regarding materials in fuel including not accounting for burnable poisons in fuel, no credit for ^{234}U and ^{236}U or fission products in fuel, and use of the most reactive fuel stack density
- Credit can only be taken for up to 75 % (NRC 2000, Section 8.4.1.1) of the neutron absorbing material in criticality controls (e.g., grid plates or inserts).
- Moderator density will be varied over the range of 0.0 through 1.0 in order to evaluate for optimum moderation conditions.

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

2.2 ELECTRONIC MANAGEMENT OF INFORMATION

Electronic management of information generated from these calculations is controlled in accordance with AP-3.13Q, *Design Control*. The computer input and output files generated from this calculation are stored on a Compact Disc (CD), and submitted as an attachment to this document (Attachment II).

3. ASSUMPTIONS

3.1 The current facility layout of the Aging Facility and its process design is used for these calculations. The aging pad consists of a 2 x 40 array of vertical casks.

Rationale: The facility (Attachment III) and its process design are in the preliminary stage of design development. However, the process functions are expected to remain unchanged. It is assumed that design changes to the facility layout will have little or no impact on the criticality results or conclusions presented in this document. A range of design variations were evaluated that are expected to cover potential design changes of the Aging Facility.

Usage: This assumption is used throughout this design calculation.

3.2 The MCNP models include axial reflection by modeling a water region above and below the storage/aging cask with an assumed height of 30 cm.

Rationale: The specified water thickness simulates infinite water reflection. The actual structure of the fuel assembly and storage racks will provide reduced reflection due to axial leakage via the fuel pin plenums and neutron absorption in the fuel assembly end fittings and the rack structure.

Usage: This assumption is used in Section 5.1.

3.3 It is assumed that omitting the grid plates, spacers, and hardware in the fuel assembly tend to produce higher reactivity values for PWR and BWR fuel cask.

Rationale: The calculated eigenvalue of the system model increases by excluding those materials beyond the active fuel region and replacing them with water (General Atomics 1993b, p. 6.4-1). Under-moderated lattices will have less moderator displacement by not modeling the spacer grids, for example, and thereby increasing the moderator effectiveness.

Usage: This assumption is used in Section 5.1.

3.4 The MGR Site Specific Cask (MSC) for commercial fuel is assumed to be identical in design, other than the neutron poison loading/configuration, to the Multi Purpose Canister (MPC)-24 for PWR fuel and the MPC-68 for BWR fuel.

Rationale: Since the MSC is still being developed, the criticality control features will be similar to the existing NRC-certified storage casks.

Usage: This assumption is used in Sections 5.1 and 5.2.

3.5 The MSC for DOE canisters is assumed to have a similar inside diameter to already NRC-certified storage casks for commercial fuel.

Rationale: Since both commercial and DOE MSCs will be stored on the aging pad, it would be appropriate if each MSC were similar in size for uniformity, ease in design, and ease in handling the storage casks.

Usage: This assumption is used in Sections 5.1 and 5.2.

3.6 It is assumed the overpack thickness of the MSC containing DOE canisters is 15 inches and is made out of concrete.

Rationale: Since the MSC for DOE canisters is still being developed, the overpack thickness and material will be similar to the existing NRC-certified storage casks.

Usage: This assumption is used in Sections 5.1 and 5.2.

3.7 The Fort St. Vrain fuel is assumed to have a U-235 enrichment of 100 %.

Rationale: This assumption was used to introduce conservatism into the calculation.

Usage: Section 5.1.

3.8 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 is 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), indicating 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 the surface area of the fuel rods for the Westinghouse 17x17 OFA (DOE 1987, p. 2A-351). The smaller surface area results in greater self-shielding and higher fissile isotope content with burnup. Consequently, the isotopic concentrations generated with the B&W 15x15 fuel assembly is 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).

Usage: Sections 5.1.7 and 5.2.3.

3.9 It is assumed for the burnup-credit evaluation presented in this document that a one node representation in MCNP of the fuel region (as opposed to applying an axial burnup profile) is slightly conservative for both PWR and BWR fuel.

Rationale: Studies show that a one node axial fuel region representation versus a multi-node axial fuel region representation is slightly conservative in most cases for PWR fuel with initial enrichments ranging between 2.0 – 5.0 wt% and a burnup range of 10-45 GWd/MTU (BSC 2003a, p. 36). It is therefore reasonable to assume that the BWR fuel would display similar trends to the PWR fuel.

Usage: This assumption is used in Section 5.2.3.

3.10 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 is 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.

Usage: This assumption is used in Section 5.1.7.

3.11 It is assumed that for commercial spent nuclear fuel, the upper subcritial limit (USL) is 0.9472 as a limit in order to meet the design criteria that k_{eff} can not exceed 0.95 including uncertainties and bias at 95% confidence level (Doraswamy 2004, Section 4.9.2.2). In other words, the USL provides a margin of 0.0028 (0.95 - 0.9472) to account for code bias and uncertainties at 95% confidence level. A more conservative USL of 0.925 is assumed for DOE fuel canisters.

Rationale: Uncertainties and bias that need to be considered in this analysis pertain to statistical uncertainties, dimensional uncertainties, code bias, and tolerance uncertainties. For commercial spent nuclear fuel, 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 ± 0.0007 (Holtec International 2002, Appendix 6 A-2). Note that the uncertainties associated with the MCNP calculated k_{eff} values are not included in the USL (see discussion in Section 5.1.4).

The assumption of USL=0.925 for DOE fuel canisters provides a total allowance of 0.025 to account for calculational bias and all uncertainties including statistical, dimensional and tolerance uncertainties. This USL is consistent with the critical limit minus the administrative margin of 0.05 for representative intact-moderated DOE fuel (BSC 2003c, p. 41).

Usage: This assumption is used throughout this document.

4. USE OF COMPUTER SOFTWARE

4.1 BASELINED SOFTWARE

4.1.1 MCNP

The MCNP code (CRWMS M&O 1998a) was used to calculate the 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 II) and the files are listed in Attachment I.

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

4.2 COMMERCIAL OFF-THE-SHELF SOFTWARE

4.2.1 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 503009

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

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 5.2 and 6. 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, *Software Management*.

5. CALCULATION

All technical product inputs and sources of the inputs used in the development of this calculation are documented in this section. Attachment III features a sketch of the Aging Facility as of the date of this calculation, and may not reflect the ongoing design evolution. The purpose of this sketch is to show the functional areas where the SNF will be stored in the storage/aging casks.

5.1 CALCULATIONAL INPUTS

5.1.1 Design Requirements and Criteria

The design criteria for criticality safety analysis provided in Section 4.9.2.2 of the *Project Design Criteria Document* (Doraswamy 2004) are used in these calculations. The pertinent criteria for Aging Facility criticality include the following (Doraswamy 2004, Section 4.9.2.2):

- Burnup credit is used for in-package criticality evaluations. Also, ensure that there is no credible criticality event under normal conditions and Category 1 and 2 event sequences.
- The multiplication factor (k_{eff}) will not exceed 0.95, including all biases and uncertainties in the data and method of the analysis, under all normal and off-normal event sequences.
- The facility design will utilize a favorable geometry and/or fixed neutron absorbers for criticality control.

The functional requirement 3.2.3.1 of the *SNF Aging System Description Document* (BSC 2004f, p. 3-12) states that the "aging system shall be designed and operated to prevent any credible criticality event from occurring". The basis for this requirement is to meet 10 CFR 63.112(e)(6), which states that the aging system shall be designed to "prevent and control criticality". This also requires that "fissile materials shall be properly packed to prevent contact with moderators (e.g. snow, rainfall, floodwater, etc.)" (BSC 2004f, p. 3-12).

5.1.2 Storage/Aging Cask Selection

The aging facility can accommodate both horizontal and vertical storage/aging systems (BSC 2004f, p. 4-1). As indicated in Assumption 3.1, only vertical storage/aging systems will be considered in this calculation, which is justified later in this section. Vertical commercially available NRC-licensed storage systems include TN-32, TN-68, BNFL FuelSolutions Storage System, Holtec HI-STAR 100, Holtec HI-STORM, NAC MPC and NAC UMS (Cogema 2004, Table 1-1). As background information, the horizontal systems available for SNF aging include NUHOMS-24PT1 for PWR SNF (Cogema 2004, p. 5). 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). License applicants are required to design criticality safety controls according to the double contingency principle and include margins of safety (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 evaluation is required only for site-specific conditions which 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 overpack. 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, as mentioned earlier. The results in Sections 6.1 (PWR fuel) and 6.2 (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.

A representative vertical cask 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).

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).

5.1.3 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 should be 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*). 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 that were evaluated, to determine the most reactive fuel, has been categorized into nine fuel groups (Mecham, D.C. 2004, Section 4.2.4.1):

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 (foreign research reactor – Melt & Dilute)
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)).

Note that both Mark 1A and Mark IV type fuel are considered for N Reactor and type "D" and type "K" canister are evaluated for TMI-2 fuel. Section 5.2.1 presents the most reactive DOE fuel evaluation demonstrating that the Enrico Fermi, Fort St. Vrain and FFTF were the most reactive DOE fuel types.

5.1.4 Upper Subcritical Limit

In accordance with the requirements given in *Preclosure Criticality Analysis Process Report* (BSC 2004e, Section 2.2.7), k_{eff} should not exceed 0.95, including all biases and uncertainties in the data and method of the analysis. All evaluations utilizing the HISTORM-100 cask system are performed for the worst case combination of manufacturing tolerances with respect to criticality (Holtec International 2002, p.6.3-2). Evaluations were performed to determine the effects of tolerances (Holtec International 2002, Tables 6.3-1 & 6.3-2). It was determined that design parameters important to criticality safety are fuel enrichment, the inherent geometry of the fuel basket structure and the fixed neutron absorbing panels (Boral) (Holtec International 2002, p. 6.3-3). Further, the results presented in Section 6 of this report are within the bounds of the k_{eff} values demonstrated in the *Final Safety and Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System)* to cover uncertainties and bias.

Per Assumption 3.11, a system is considered acceptably subcritical if the calculated k_{eff} value plus calculation uncertainties (i.e., 2 times the standard deviation associated with the MCNP calculated value) lies at or below 0.9472 for commercial spent nuclear fuel or 0.925 for DOE fuel canisters. The definition of upper subcritial limit (USL) is (BSC 2004e, Section 3.5):

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

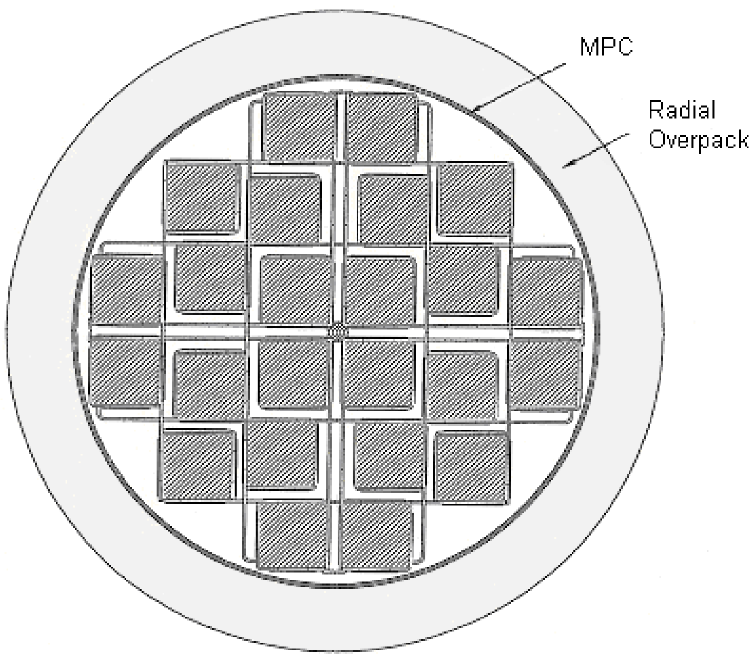
where k_S is the MCNP calculated value for the system, Δk_S is an allowance for (a) statistical or convergence uncertainties, or both in the computation of k_s, (b) material and fabrication tolerances, and (c) uncertainties due to the geometric or material representations used in the computational method [Note: allowance for items (b) and (c) can be obviated by using bounding representations]. As an example, if the standard deviation associated with the MCNP calculated value for commerical spent nuclear fuel is 0.00028 (see Section 6), the MCNP calculated k_S value can't exceed 0.94664 (0.9472 – 2x0.00028), per expression 1, in order to meet the USL. For a more detailed description of USL determination and criterion, see BSC 2004e (Sections 3.4.1, 3.4.2, and 3.5). For commercial spent nuclear fuel, the criticality evaluation was performed for the worst-case configuration and condition, which already accounted for all uncertainites other than the MCNP statistical uncertainty (Holtec International 2002, p.6.3-2). Based on this bounding representation, items (b) and (c) mentioned above were eliminated.

5.1.5 Storage/Aging Cask Calculation Inputs

The HI-STORM 100 storage casks in the Aging Facility were modeled in accordance with 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, Section 6.3). The cask was modeled with radially reflective boundaries to simulate an infinite array of storage/aging casks. This is bounding of the 2 x 40 array featured in the current design (Assumption 3.1) of the Aging Facility (Attachment III). Physical inputs for the storage/aging casks are described in the following subsections.

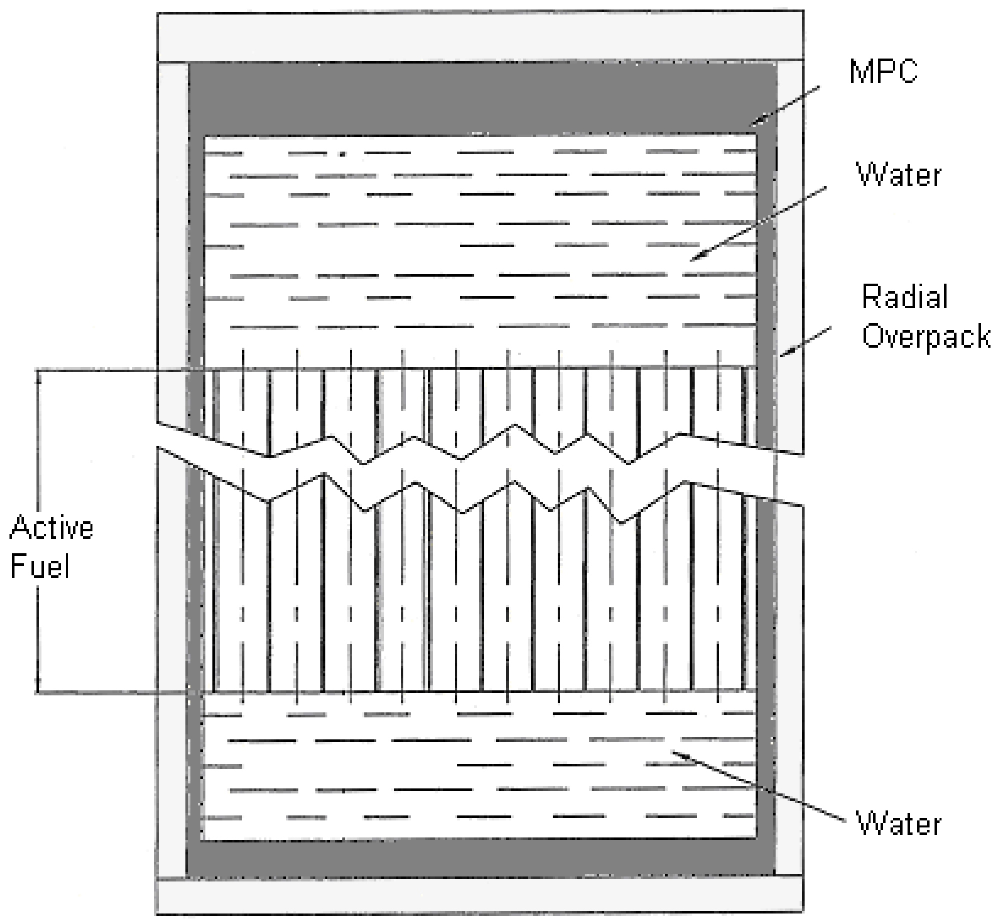
5.1.5.1 PWR MPC-24 Configuration and Physical Dimensions

The MPC-24 for PWR fuel consists of a concrete cask with steel shells and an interior 24 PWR assembly basket. Figure 5.1-1 displays the planar cross-section of the MPC-24 calculational model inside the overpack and Figure 5.1-2 presents the axial view. Note that the model also includes axial reflection by modeling a 30 cm water region above and below the storage/aging cask (Assumption 3.2).



NOTE: Not to scale.

Figure 5.1-1 Radial View of the MPC-24 PWR Fuel Storage Cask
(Source: Holtec International 2002, Figure 6.3-4)



NOTE: Not to scale. Also, details of the overpack geometry are not shown in this figure.

Figure 5.1-2 Axial View of the MPC-24 PWR Fuel Storage Cask
(Source: Holtec International 2002, Figure 6.3-7)

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

$$T = \frac{M}{S_a} \times \frac{N_A}{M_a} \times \frac{1}{A} \qquad \text{(equation 2)}$$

where

M = weight (g) of ^{10}B
S_a = surface area (Boral areal densities are ranging 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:

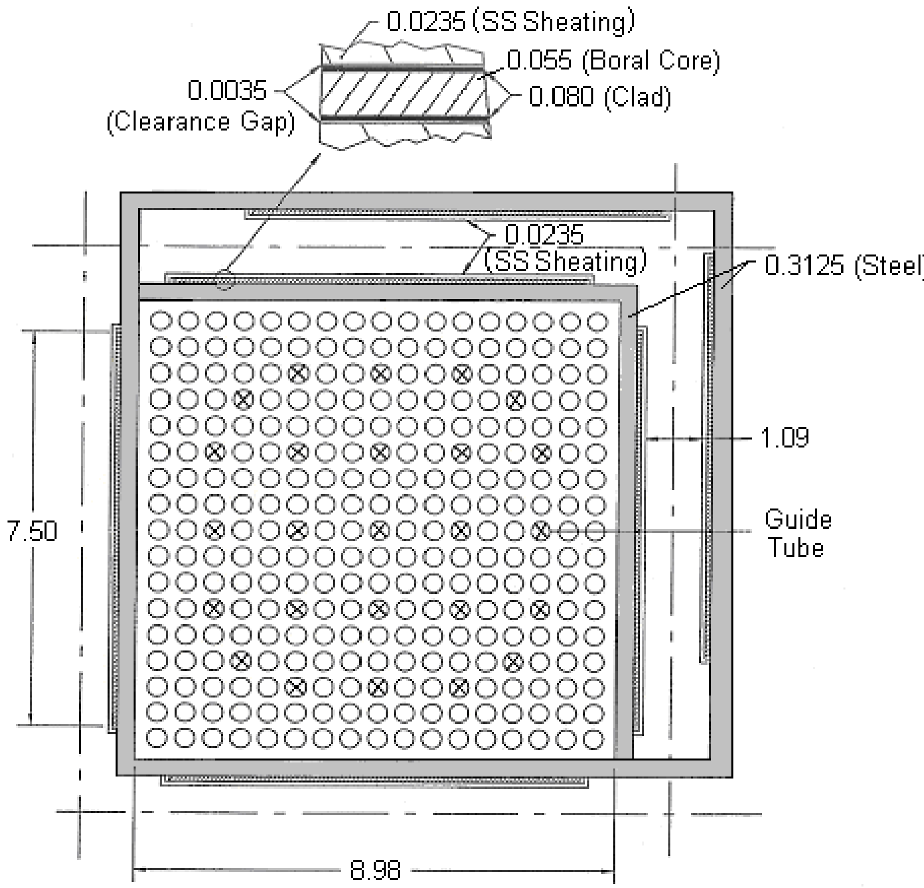
$$A = \frac{N_a}{V} = \frac{N_m \times N_A}{V} = \frac{M}{M_a} \times \frac{N_A}{V} = \frac{M}{S_a \times T} \times \frac{N_A}{M_a} \qquad \text{(equation 3)}$$

where

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

The selections of Boral thicknesses and ^{10}B loading 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 6.3. It should be emphasized that Boral panels are selected based on a specific weight percent of B_4C and Al and a desired thickness. These two parameters ultimately govern the ^{10}B loading (see Section 6.3 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 5.1.3). Figure 5.1-3 displays the storage rack basket cell with the Westinghouse 17x17 OFA. Table 5.1-1 features the radial dimensions of the storage rack and cell geometry while Table 5.1-2 shows the axial dimensions. Table 5.1-3 displays the specifications of the PWR fuel assembly. Note that only the active fuel region was included in the model (Assumption 3.3).



NOTE: Dimensions are in inches.

Figure 5.1-3 PWR Storage Rack Basket Cell Containing W 17 x 17 OFA
(Source: Holtec International 2002, Figure 6.3-1)

Table 5.1-1 Radial Dimensions of the MPC-24, Overpack, and Cell Geometry

Component	Dimension (cm)	Reference
SS overpack outer shell thickness	1.905	Holtec International 2002, Figure 5.3.10
Concrete overpack thickness	67.945	Holtec International 2002, Figure 6.3.4
Concrete overpack, o.d.	332.74	Holtec International 2002, Figure 5.3.10
SS overpack inner shell	196.85	Holtec International 2002, Figure 5.3.10
Cavity (water), o.d.	190.50	Holtec International 2002, Figure 5.3.10
MPC storage basket, o.d.	173.6725	Holtec International 2002, Figure 6.3.4
MPC storage basket, i.d.	171.1325	Holtec International 2002, Figure 6.3.4
Center column	6.985	Holtec International 2002, Drawing 3926 (Sheet 2)
Assembly inside dimension	22.8092	Holtec International 2002, Figure 6.3.1 & Table 6.3.3
Cell pitch	27.7012	Holtec International 2002, Table 6.3.3 & Drawing 3926 (Sheet 3)
Flux trap	2.7686	Holtec International 2002, Figure 6.3.1 & Table 6.3.3
Cell wall thickness (SS)	0.79375	Holtec International 2002, Figure 6.3.1
SS sheathing	0.05969	Holtec International 2002, Figure 6.3.1
Boral thickness [a]	0.1397	Holtec International 2002, Figure 6.3.1
Al thickness (Clad)	0.0254	Holtec International 2002, Figure 6.3.1
Boral width - wide	19.05	Holtec International 2002, Figure 6.3.1
Boral width – narrow [b]	15.875	Holtec International 2002, Drawing 3926 (Sheet 2)
Boral clearance gap	0.00889	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 5.1-2 Axial Dimensions of the MPC-24, Overpack, and Cell Geometry

Component	Dimension (cm)	Reference
Lower water thickness (below active fuel region)	10.16	Holtec International 2002, Figure 6.3.7
Upper water thickness (above active fuel region)	15.24	Holtec International 2002, Figure 6.3.7
MPC baseplate	6.35	Holtec International 2002, Drawing 3923 (Sheet 2)
MPC lid	24.13	Holtec International 2002, Drawing 3923 (Sheet 2)
Bottom overpack SS plate thickness (top layer)	12.70	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Bottom overpack concrete plate thickness	43.18	Holtec International 2002, Figure 6.3.7
Bottom overpack SS plate thickness (bottom layer)	5.08	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top overpack SS plate thickness (top layer)	10.16	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top overpack concrete plate thickness	26.67	Holtec International 2002, Figure 6.3.7
Top overpack SS plate thickness (bottom layer)	3.175	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top gap (between MPC and overpack)	3.81	Approximated from Holtec International 2002, Drawings 1495 (Sheet 2) and 3923 (Sheet 3)

Table 5.1-3 Specifications of the PWR W 17 x17 OFA

Parameter	Dimension (cm)	Reference [b]
Rod pitch	1.2598	Holtec International 2002, p. 2.1-11
Active fuel length	381.0	Holtec International 2002, p. 2.1-11
Cladding outside diameter	0.9144	Holtec International 2002, p. 2.1-11
Cladding inside diameter	0.8002	Holtec International 2002, p. 2.1-11
Pellet outside diameter	0.784352	Holtec International 2002, p. 2.1-11
Guide/instrument tube outside diameter	1.204	Sanders and Wagner 2002, p.8
Guide/instrument tube thickness	0.04064	Holtec International 2002, p. 2.1-11
Array size	17 x 17	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 5.1-3 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

5.1.5.2 PWR Material Compositions

The calculations were performed with either the isotopic compositions given in weight density (wt%) or atom densities (atoms/barn-cm), depending on the source of the input. Table 5.1-4 displays the relevant materials used for the storage/aging cask and the PWR fuel.

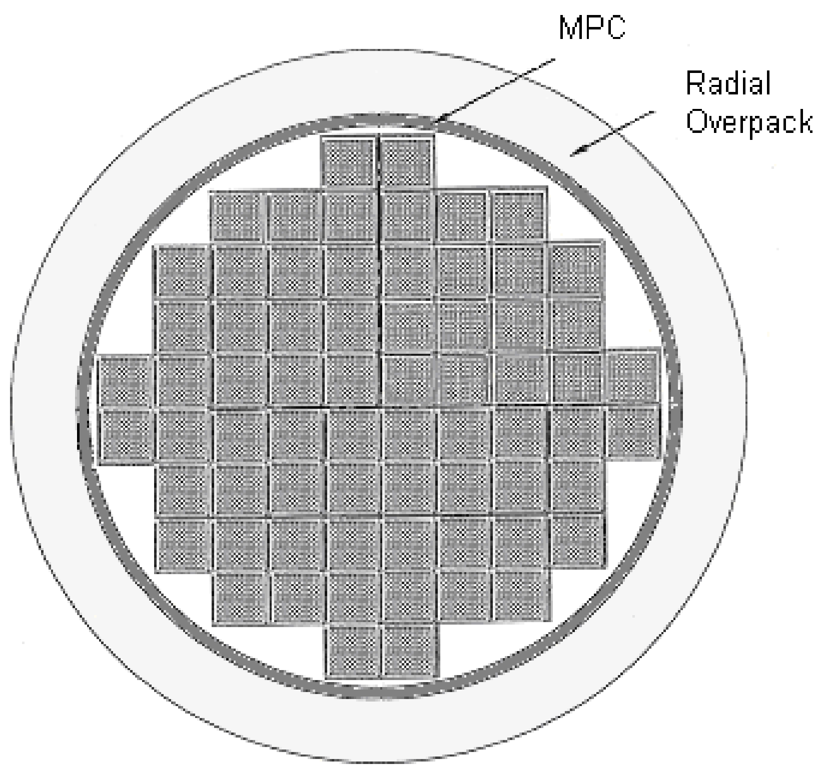
Table 5.1-4 Material Properties for the Storage Cask and PWR Fuel

Material	Density (g/cm^3)	Element	Weight Fraction or Weight Percent (wt %)	Atom Fraction or Atom Density (atoms/barn-cm)	Reference/ Remark
H_2O (throughout model)	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
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 6
[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)

5.1.5.3 BWR MPC-68 Configuration and Physical Dimensions

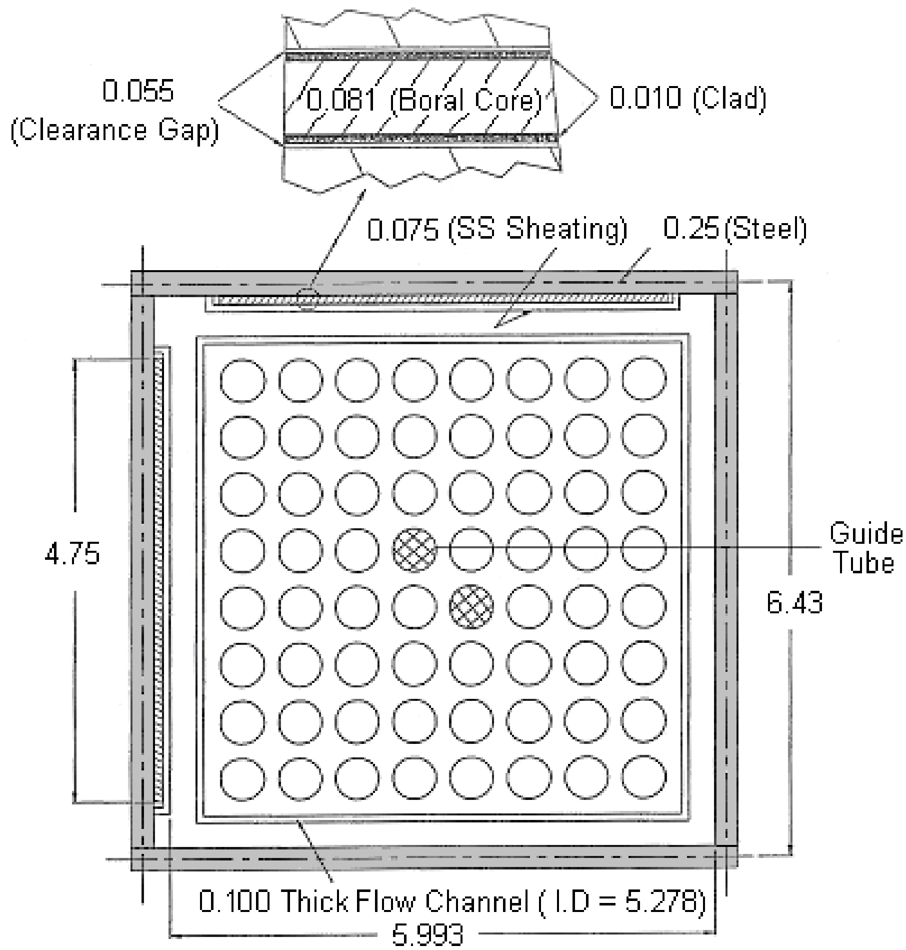
The MPC-68 for BWR fuel consists of a concrete cask with steel shells and an interior 68 BWR assembly basket. Figure 5.1-4 displays the planar cross-section of the MPC-68 cask calculational model and Figure 5.1-2 presents the axial view (it is the same as for the MPC-24). Note that the model also includes axial reflection by modeling a 30 cm water region above and below the storage/aging cask (Assumption 3.2).



NOTE: Not to scale.

Figure 5.1-4 Radial View of the MPC-68 BWR Fuel Storage Cask
(Source: Holtec International 2002, Figure 6.3-6)

The storage rack basket cells contain GE 8 x 8 standard assemblies, since this is the most reactive BWR fuel (Section 5.1.3). Figure 5.1-5 displays the storage rack basket cell with the GE 8 x 8 assembly. Table 5.1-5 features the radial dimensions of the storage rack and cell geometry while Table 5.1-6 shows the axial dimensions. Table 5.1-7 displays the specifications of the BWR fuel assembly. Note that only the active fuel region was included in the model (Assumption 3.3).



NOTE: Dimensions are in inches.

Figure 5.1-5 BWR Storage Rack Basket Cell Containing GE 8 x 8 Assembly
(Source: Holtec International 2002, Figure 6.3-3)

Table 5.1-5 Radial Dimensions of the MPC-68, Overpack, and Cell Geometry

Component	**Dimension (cm)**	**Reference**
SS overpack outer shell thickness	1.905	Holtec International 2002, Figure 5.3.10
Concrete overpack thickness	67.945	Holtec International 2002, Figure 6.3.4
Concrete overpack, o.d.	332.74	Holtec International 2002, Figure 5.3.10
SS overpack inner shell, o.d.	196.85	Holtec International 2002, Figure 5.3.10
Cavity (water), o.d.	190.50	Holtec International 2002, Figure 5.3.10
MPC storage basket, o.d.	173.6725	Holtec International 2002, Figure 6.3.6
MPC storage basket, i.d.	171.1325	Holtec International 2002, Figure 6.3.6
Cell box inside dimension	15.2222	Holtec International 2002, Figure 6.3.3 & Table 6.3.3
Cell pitch	16.3322	Holtec International 2002, Table 6.3.3 & Figure 6.3.3
Cell plate thickness	0.635	Holtec International 2002, Figure 6.3.3 & Table 6.3.3
SS sheathing	0.1905	Holtec International 2002, Figure 6.3.3
Boral thickness	0.2057	Holtec International 2002, Figure 6.3.3
Al thickness (Clad)	0.0254	Holtec International 2002, Figure 6.3.3
Boral width	12.065	Holtec International 2002, Figure 6.3.3
Boral clearance gap	0.01397	Holtec International 2002, Figure 6.3.3

Table 5.1-6 Axial Dimensions of the MPC-68, Overpack, and Cell Geometry

Component	**Dimension (cm)**	**Reference**
Lower water thickness (below active fuel region)	18.542	Holtec International 2002, Figure 6.3.7
Upper water thickness (above active fuel region)	21.4884	Holtec International 2002, Figure 6.3.7
MPC baseplate	6.35	Holtec International 2002, Drawing 3923 (Sheet 2)
MPC lid	24.13	Holtec International 2002, Drawing 3923 (Sheet 2)
Bottom overpack SS plate thickness (top layer)	12.70	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Bottom overpack concrete plate thickness	43.18	Holtec International 2002, Figure 6.3.7
Bottom overpack SS plate thickness (bottom layer)	5.08	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top overpack SS plate thickness (top layer)	10.16	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top overpack concrete plate thickness	26.67	Holtec International 2002, Figure 6.3.7
Top overpack SS plate thickness (bottom layer)	3.175	Holtec International 2002, Figure 6.3.7 & Drawing 1495 (Sheet 2)
Top gap (between MPC and overpack)	3.81	Approximated from Holtec International 2002, Drawings 1495 (Sheet 2) and 3923 (Sheet 3)

Table 5.1-7 Specifications of the BWR GE 8 x 8 Standard Assembly

Parameter	Dimension (cm)	Reference
Rod pitch	1.6256[a]	Holtec International 2002, p. 2.1-14
Active fuel length	381.0	Holtec International 2002, p. 2.1-14
Cladding outside diameter	1.2268	Holtec International 2002, p. 2.1-14
Cladding inside diameter	1.0796	Holtec International 2002, p. 2.1-14
Pellet outside diameter	1.0566	Holtec International 2002, p. 2.1-14
Guide/instrument tube outside diameter	1.0566	Holtec International 2002, Figure 6.3-3
Guide/instrument tube thickness	0.0	Holtec International 2002, p. 2.1-14
Array size	8 x 8	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.

5.1.5.4 BWR Material Compositions

The BWR material compositions are identical to those of the PWR material specifications, except for those listed in Table 5.1-8.

Table 5.1-8 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)

5.1.6 DOE Fuel Canisters

Calculations were performed to determine the most reactive/bounding DOE fuel canister for ultimate placement in the MGR Site specific Cask (MSC) designed for DOE fuel. Table 5.1-9 presents the physical dimensions of the canisters and Table 5.1-10 shows the DOE fuel parameters. Figure 5.1-6 displays the DOE canisters considered in this evaluation, as described in Section 5.1.3, in the radial view. An axial representation of the DOE SNF canisters is also included in Figure 5.1-6. Table 5.1-11 displays the relevant material properties for DOE non-fuel materials used in the MCNP models. Table 5.1-12 presents the isotopic content of the fuel materials for each DOE type fuel considered in this calculation. It should be mentioned that the MCNP input files from the *Canister Handling Facility Criticality Safety Calculations* document (BSC 2004b) were used in the calculations presented in this document. Changes to the MCNP input files include varied boundary conditions as well as various placement and loading scenarios inside the MSC. For more details regarding canister physical dimensions, see Section 5.1.4 (BSC 2004b) and Section 5.1.2 (BSC 2004b) for more specifics regarding DOE fuel parameters.

Table 5.1-9 Physical Dimensions of DOE Canisters

DOE Fuel Type	Canister o.d. (cm)	Canister length (cm)	Canister Capacity	Reference
Enrico Fermi	45.72	–	3360 fuel pins (2 sets of 12 tubes each containing 140 pins)	CRWMS M&O 2000a, p. 12
FFTF	45. 72 (0.95 cm wall thickness)	456.90 (414.50 cm internal length)	1302 fuel pins (6 assemblies with each 217 fuel pins)	CRWMS M&O 1999a, Figures 5-3 & 5-4
Fort St. Vrain	45. 72 (0.95 cm wall thickness)	457.0 (411.71 cm internal length)	5 fuel elements stacked vertically	BSC 2001a, p. 15
Melt & Dilute	45. 72 (0.95 cm wall thickness)	299.90 (254.0 cm internal length)	3-6 ingots (depending on the dimensions of the individual ingots) stacked vertically	BSC 2001b, p.11
N Reactor	64.29	419.84	270 fuel elements MARK IV (54 fuel elements stacked 5 high) [a]	CRWMS M&O 2001, p. 14 DOE 2000, pp. 23-25 (canister capacity)
Shippingport LWBR	45. 72 (0.95 cm wall thickness)	457.0 (411.71 cm internal length)	7428 fuel rods (12 assemblies with each 619 fuel rods)	CRWMS M&O 2000b, p. 18 DOE 1999b, p. 16 (canister capacity)
Shippingport PWR	45. 72 (0.95 cm wall thickness)	268.09 (internal length)	1 fuel cluster	CRWMS M&O 2000c, p. 15
TMI-2 (D canister) [b]	35. 56 (0.64 cm wall thickness)	380.37 (346.55 cm internal length)	1 fuel assembly (15x15 array having 204 fuel rods)	DOE 2003, pp. 21 (canister capacity), 25 & 26
TRIGA	45. 72 (0.95 cm wall thickness)	254.70 (internal length)	111 fuel elements (37 fuel elements stacked 3 high)	CRWMS M&O 1999d, p. 13

[a] Mark 1A contains 48 fuel elements stacked 5 high, comprising a total of 240 fuel elements (DOE 2000,Fig. 4-2).
[b] The K canister has a large internal diameter over which fuel matrix material is not constrained (see Fig. 5.1-3)

Table 5.1-10 DOE Fuel Parameters

DOE Fuel Type	Max. fissile enrichment (%)[a]	Fuel o.d. (cm) [b]	Clad i.d. (cm)	Clad o.d. (cm)	Pin Pitch (cm) [c]	Fuel length (cm)	Reference
Enrico Fermi	25.69	0.376	0.376	0.401	0.52 [h]	77.47	DOE 1999a, p.8 CRWMS M&O 2000a, p. 12 (clad)
FFTF	25.95	0.495	0.508	0.584	0.726	237.24	INEEL 2002, p.15, 17 (pin pitch) & Fig. 3 (fuel o.d.)
Fort St. Vrain	100.0 (Assumption 3.7)	1.245	–	–	1.880	–	Taylor 2001, p. 21 & Fig. 2-3 (pin pitch)
Melt & Dilute	20.0	41.91	–	–	–	76.2	BSC 2001c, p.3
N Reactor – outer fuel tube [d]	1.25 [e]	6.096 4.496 [f]	6.096 4.496	6.223 4.607	7.80 [h]	53.0	DOE 2000, Tables 3-1 & 3-2 (clad)
N Reactor – inner fuel tube [d]	1.25 [e]	3.175 1.118 [f]	3.175 1.118	3.378 1.245	7.80 [h]	53.0	DOE 2000, Tables 3-1 & 3-2 (clad)
Shippingport LWBR	4.90	0.640	0.734	0.778	0.937 [h]		DOE 1999b, p. 16 (enr.), Fig. 3-3 (pin pitch), Table 3-5 (fuel o.d.) & Table 3-8 (clad)
Shippingport PWR	93.2	–	–	–	–	–	DOE 1999c, Table 3-1
TMI-2	2.96	0.936	0.958	1.092	1.5 (TMI-2D) [h] 1.9 (TMI-2K)	360.12	DOE 2003, p. 19 (enr.), p. 21, p. 22 (fuel length) & p.23 (fuel o.d.)
TRIGA	70.0	3.480 0.635 [g]	3.490	3.592	6.03 [h]	38.10	DOE 1999d, p. 19

[a] This is the total fissile content divided by the total heavy metal mass x 100.
[b] For fuel in the form of cylindrical rods, this is the fuel outside diameter
[c] For fuel in the form of cylindrical rods, this is the nominal pin pitch in the canister
[d] See Figure 5.1-3 for locations of outer and inner fuel tubes
[e] The enrichment for Mark IV (case B) is 0.95 %
[f] Inside diameters of fuel tubes
[g] Inside diameters of fuel tube
[h] Pitch resulting in the largest value of k_{eff} for a single canister (BSC 2004b, Table 6-1 & Attachment 3)

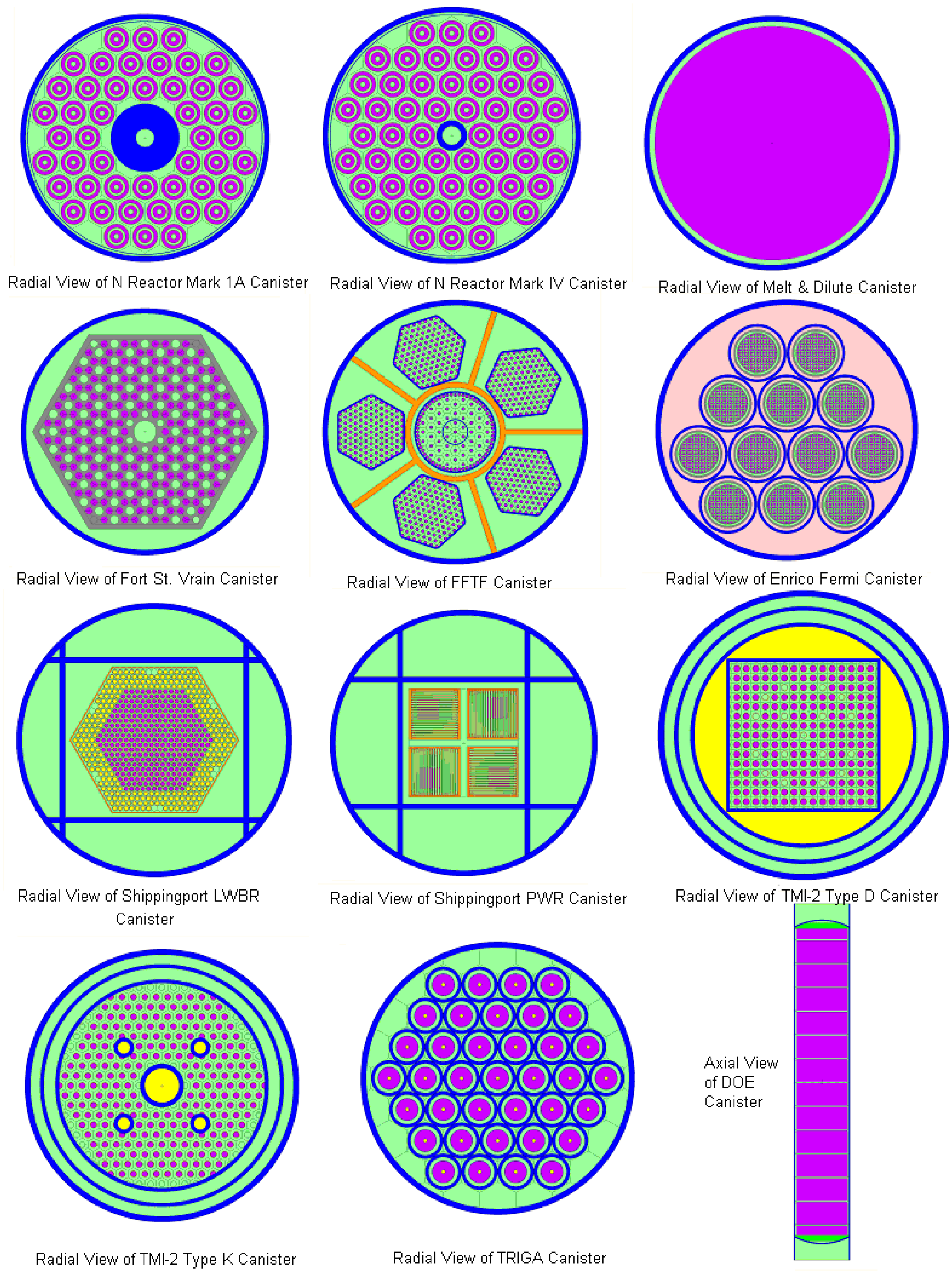


Figure 5.1-6 Radial and Axial View of the DOE Fuel Canisters

Table 5.1-11 Material Properties for DOE Non-Fuel Materials

Material	Density (g/cm^3)	Weight Percent (wt%)	Reference/ Remark
H_2O (throughout model)	1.0	H - 0.6666667 [a] O - 0.3333333 [a]	–
Magnuson Concrete	2.147	O:49.94 Ca:22.63 C:10.53 Mg:9.42 Si:4.21 K:0.9445 Al:0.7859 Fe:0.5595 Ti:0.148 Na:0.1411 H:0.3319 S:0.2483 Cl:0.0523 Mn 0.0512	NRC 1997, Volume 3, p. M8.2.4
Type 304L Stainless Steel	7.94	Fe:68.045 Cr:19.0 Ni:10.0 Mn:2.0 Si:0.75 N:0.1 P:0.045 S:0.03 C:0.03	ASTM A 276-91a 1991, p. 2 ASTM G1-90 1999, Table X1
Type 316L Stainless Steel	7.98	Fe:65.295 Cr:17.0 Ni:12.0 Mn:2.0 Mo:2.5 Si:1.0 N:0.1 P:0.045 S:0.03 C:0.03	ASTM A 276-91a 1991, p. 2 ASTM G1-90 1999, Table X1
Type 516 Carbon Steel	7.85	Fe:98.33 Mn:1.025 Si:0.275 P:0.035 S:0.035 C:0.3	ASME 2001, Sec IIA, SA-516/SA-516M & Sec IIA, SA-20/SA-20M, item 14

[a] Values given in atom fraction and not wt %

Table 5.1-12 Material Properties for Each DOE Fuel Type [a]

DOE Fuel Type	Density (g/cm^3)	Weight Percent (wt%)	Neutron Absorber (kg) [e]
Enrico Fermi	17.424	U-235:22.96 U-238:66.41 Mo:10.63	3.0 [b]
FFTF	10.02	O:11.63 U-235:0.13 U-238:62.37 Pu-239:22.54 Pu-240:3.01 Pu-241:0.26 Pu-242:0.06	19.26 [c]
Fort St. Vrain	1.991	Th-232:25.69 C:64.81 U-235:3.54 Si:5.96	------
Melt and Dilute	3.00	U-235:3.64 U-238:14.56 Al:77.97 Gd:0.50 H:0.37 O:2.96	4.73 [d]
N Reactor	18.39	U-235:1.25 U-238:98.75	------
Shippingport LWBR	9.71	O:12.12 U-233:4.57 U-234:0.06 U-238:0.02 Th-232:83.23	------
Shippingport PWR – zone 1	6.36	U-235:45.04 U-238:3.29 Ca:3.72 Zr:29.54 O:18.41	------
Shippingport PWR – zone 2	6.36	U-235:32.98 U-238:2.41 Ca:4.15 Zr:39.98 O:20.48	------
Shippingport PWR – zone 3	6.36	U-235:21.74 U-238:1.59 Ca:4.57 Zr:49.67 O:22.43	------
TMI-2	10.42	U-235:2.61 U-238:85.53 O:11.86	------
TRIGA	6.58	U-235:5.94 U-238:2.56 Zr:89.91 H:1.59	------

[a] BSC 2004b, Table 5-3. Also, see BSC 2004b, Section 5.1.2 for fuel description.
[b] Neutron absorber (Gd) contents in canister were varied. 1 vol% corresponds to 3 kg (CRWMS M&O 2000a, p.12)
[c] Neutron absorber (Gd) contents in canister were varied. 5 wt% corresponds to 19.26 kg, which is the maximum amount of gadolinium (CRWMS M&O 1999a, p.21)
[d] Neutron absorber (Gd) contents in ingots were varied. 0.5 wt% corresponds to 4.73 kg (BSC 2001c, p.3)
[e] The present calculation uses 0.75 of neutron absorber percentages listed.

5.1.7 MGR Site Specific Cask

Calculations were performed to determine additional criticality controls required for the MSC to accommodate commercial fuel outside the content specification for the MPC-24 and MPC-68. It was assumed that the MSC is similar in design to the MPC-24 and MPC-68 (Assumption 3.4). In addition to varying the ^{10}B loading in the neutron poison of the internal basket (i.e., Boral), as discussed in Section 5.1.5, 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 MSC, 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 5.1-7 and 5.1-8. Table 5.1-13 displays the neutron poison properties utilized in the MSC evaluations for commercial fuel and Table 5.1-14 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 3.8, 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 model 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 structure 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 3.10).

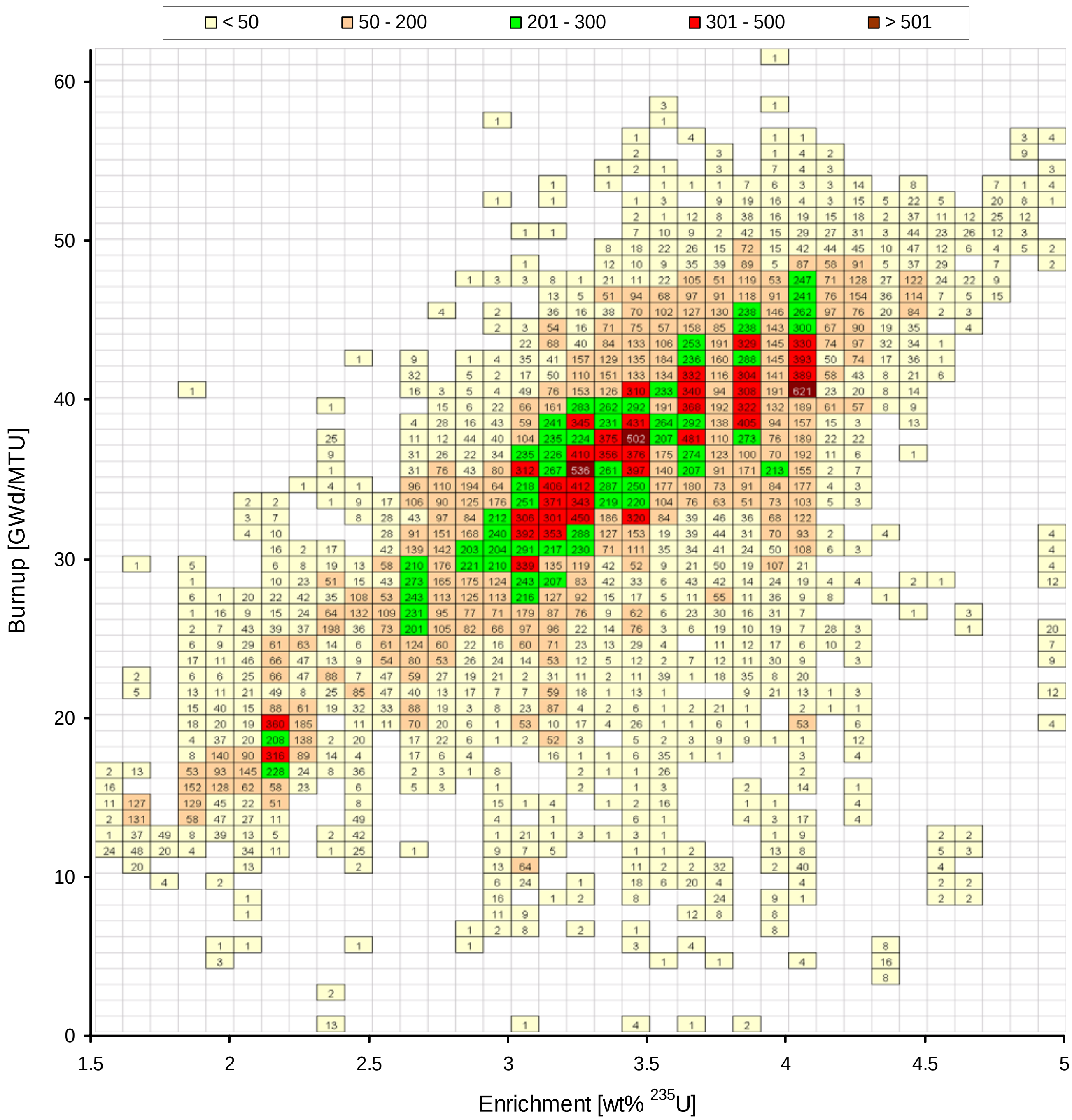


Figure 5.1-7 PWR SNF Discharge Data as of December 31, 1998
(Extracted from BSC 2004j, Figure 7 & PWR_Assembly.xls)

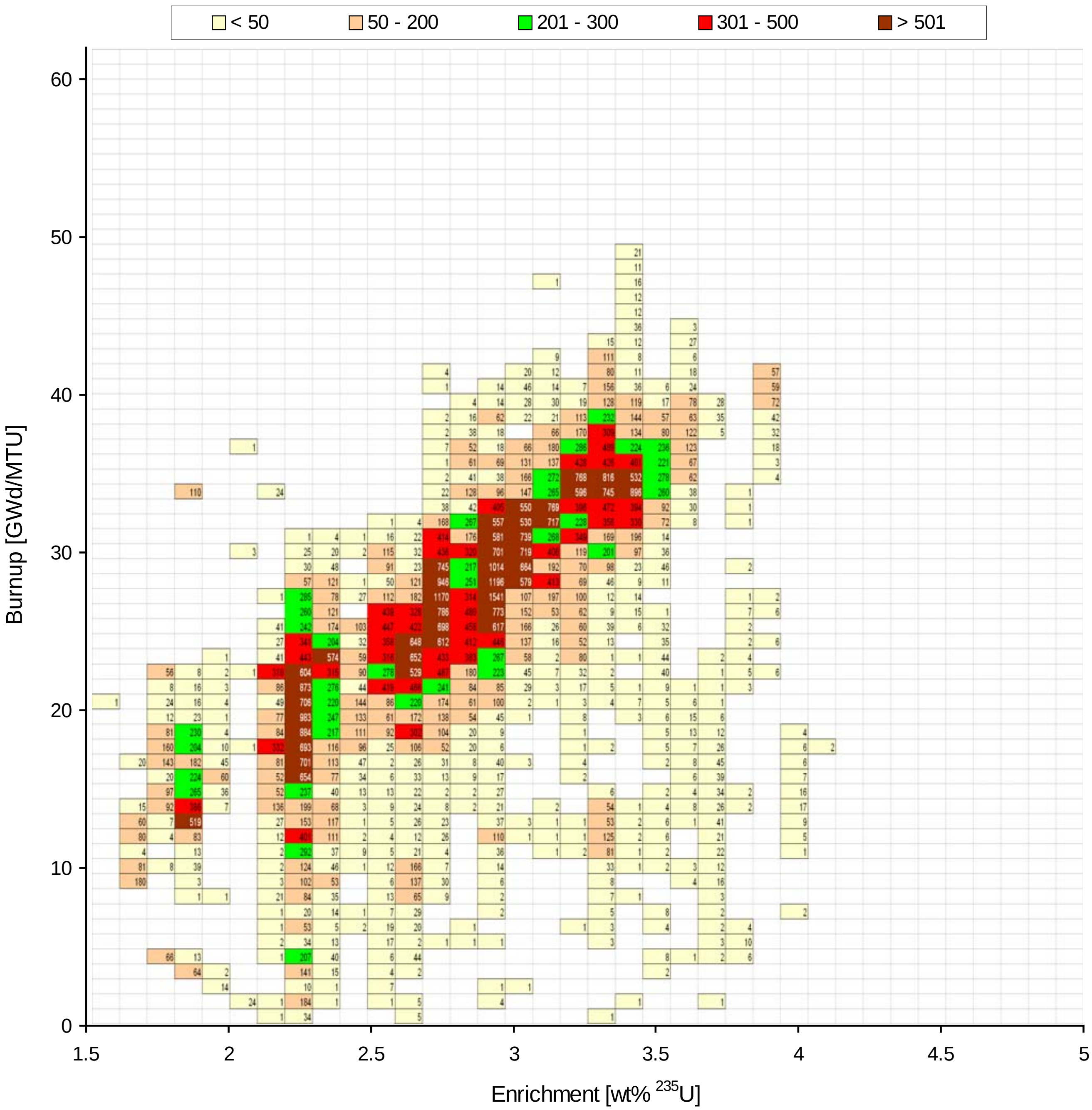


Figure 5.1-8 BWR SNF Discharge Data as of December 31, 1998
(Extracted from BSC 2004j, Figure 9 & BWR_Assembly.xls)

Table 5.1-13 Material Properties Utilized for MSC Evaluations

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

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

Table 5.1-14 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)

The calculations for DOE fuel contained in a MSC feature similar overpack dimensions to the MSC utilized for commercial SNF (Assumption 3.5). The MSC designs evaluated 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 5.1-1). Per Assumption 3.6, the overpack consists of 15 in. concrete. The calculations presented in Section 5.2.1 of this document shows that the Enrico Fermi, Fort St. Vrain and FFTF are the most reactive DOE fuel types. DOE canisters containing these fuel types were placed inside the overpack (inside diameter of 69.5 in.) in a 3x3 square pitch configuration as illustrated in Figure 5.1-9. 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 5.1-10. To ensure the most reactive 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 5.1-15. 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 5.1-11.

Table 5.1-15 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 1999b, p. 7.
[b] Stout and Leider 1991, p. 2.2.1.1-4 (upper limit)

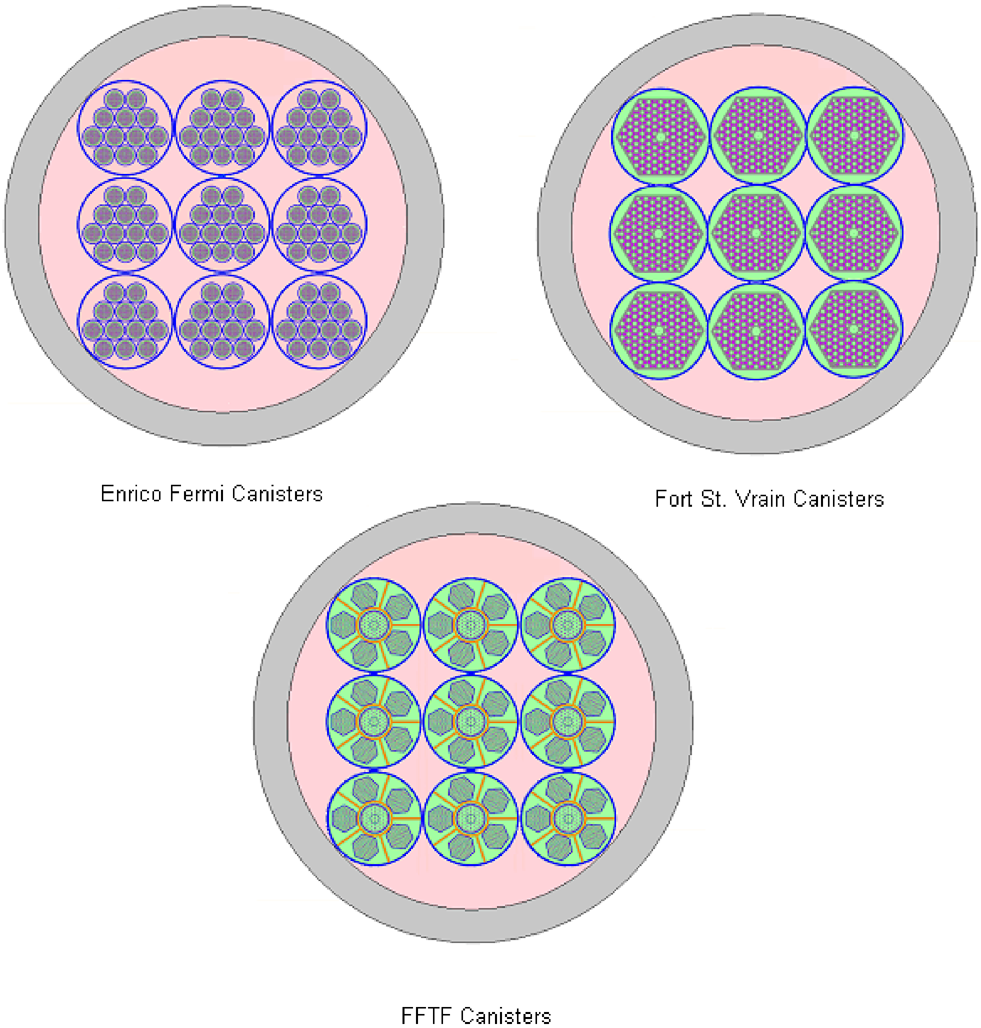


Figure 5.1-9 Illustration of MSC Containing DOE Fuel Canisters (Overpack I.D.=69.5 in)

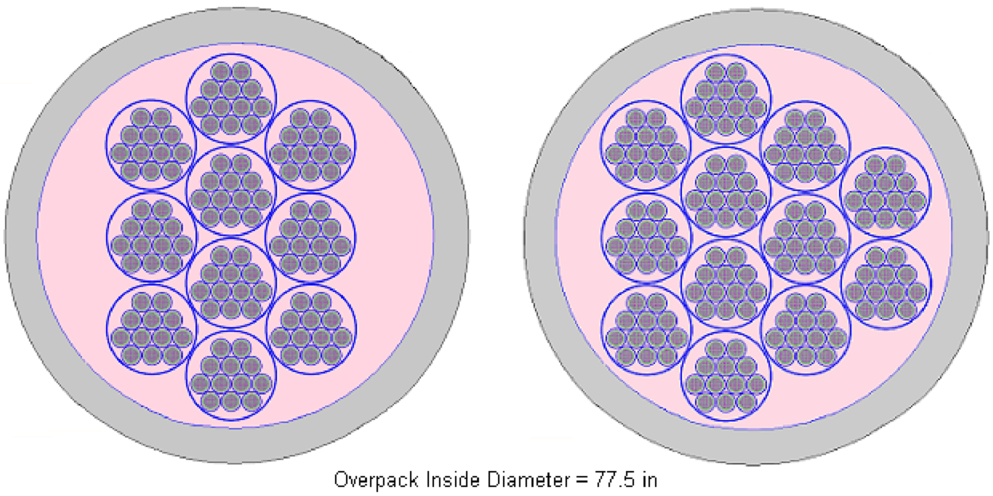


Figure 5.1-10 Illustration of MSC Containing Enrico Fermi Canisters

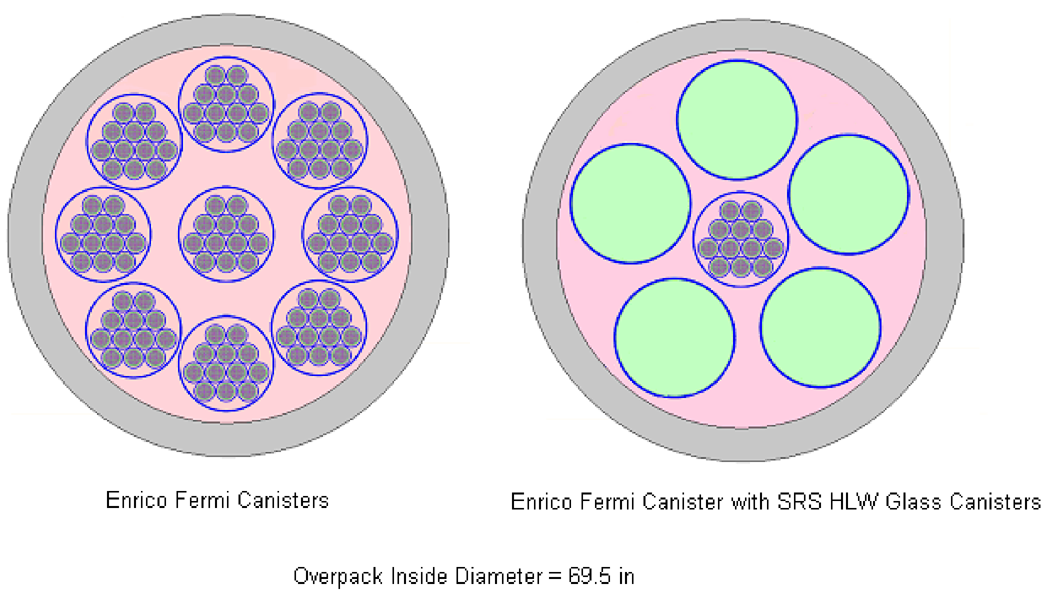


Figure 5.1-11 Illustration of MSC Containing Enrico Fermi and SRS HLW Glass Canisters

5.1.8 Category 1 and 2 Event Sequences

This design calculation considered Category 1 and Category 2 event sequences as identified in the *Categorization of Event Sequences for License Application* (BSC 2004c, Section 7). However, no event sequences have been identified for the Aging Facility. In addition, Section 7 of BSC 2004c does not identify any criticality events as Category 1 or Category 2 because it takes credit for criticality controls and design features. Consequently, all potential events in the Aging Facility that were listed under the category of "Fissile" (BSC 2004c, Section 6.3) have instead been considered in the evaluation presented in Section 5.2.5.

5.2 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 in these calculations. This process and methodology require for out-of-package operations, as stated earlier in Section 2, consideration of the most reactive fuel assembly, the multiplication factor will not exceed 0.95 including all uncertainties and bias, no burnup credit, and no credit for ^{234}U and ^{236}U. Further, all calculations were performed with MCNP and feature flooded fuel pin gaps and only 75 % credit for the fixed neutron absorber. Note that for in-package operations burnup credit is allowed, which was explored as one option to criticality control in Section 5.2.3. In addition, reflective boundary conditions are applied to all models.

5.2.1 Selection of Most Reactive DOE Fuel

The various DOE fuel types introduced in Section 5.1.3 were evaluated as an infinite array of single canisters with varying distances of separation and reflector materials. Each canister was fully flooded on the inside, which previous studies have shown is the most reactive scenario (BSC 2004d, Table 6.2-1). The reflector materials used on the outside of the canisters are concrete, water and air. Table 5.2-1 presents the k_{eff} values for each DOE fuel type as a function of distance between the canisters. It can be seen that the Enrico Fermi, Fort St. Vrain and FFTF fuel types are the most reactive. Further, the results shown in the table indicates that the canisters are not neutronically isolated since the closer they are placed together, the higher the k_{eff} values. This is illustrated in Figure 5.2-1 for the Enrico Fermi fuel where k_{eff} is plotted against canister separation distance in air, water, and concrete surroundings. It is also interesting to note that in almost all cases the highest k_{eff} value is produced when the canister is surrounded by air. This is because the water and concrete tends to moderate the system since there is virtually no leakage of neutrons (the MCNP model features reflective boundary conditions to simulate infinite arrays of canisters).

Table 5.2-1 k_{eff} of Various DOE Canisters

Distance (cm)	k_{eff} (Air)	St. Dev	MCNP files [a]	k_{eff} (Water)	St. Dev	MCNP files [a]	k_{eff} (Concr-ete)	St. Dev	MCNP files [a]
	Enrico Fermi								
0.2	0.96924	0.00020	efwds6a7	0.90215	0.00021	efwds6h7	0.95449	0.00021	efwds6c7
30	0.91813	0.00019	efwds6a1	0.86894	0.00022	efwds6h1	0.89595	0.00020	efwds6c1
60	0.89604	0.00023	efwds6a3	0.86918	0.00022	efwds6h3	0.89341	0.00023	efwds6c3
120	0.87667	0.00022	efwds6a6	0.86877	0.00022	efwds6h6	0.89206	0.00022	efwds6c6
180	0.86949	0.00022	efwds6a9	0.86877	0.00022	efwds6h9	0.89274	0.00022	efwds6c9
	FFTF								
0.2	0.92604	0.00076	ffwds15a	0.89085	0.00077	ffwds15h	0.91917	0.00077	ffwds15c
60	0.87117	0.00078	ffwds30a	0.86196	0.00080	ffwds30h	0.87618	0.00082	ffwds30c
	Fort St. Vrain								
0.2	0.94428	0.00075	fswds00a	0.86709	0.00079	fswds00h	0.92196	0.00081	fswds00
60	0.88793	0.00082	fswds30a	0.81628	0.00088	fswds30h	0.84817	0.00080	fswds30c
	TRIGA								
0.2	0.87028	0.00105	trwds60a	0.84673	0.00100	trwds60h	0.86800	0.00095	trwds60c
60	0.83425	0.00110	trwds30a	0.82533	0.00105	trwds30h	0.83639	0.00108	trwds30c
	Melt & Dilute								
0.2	0.57691	0.00112	mdwds00a	0.41097	0.00117	mdwds00h	0.53849	0.00122	mdwds00
60	0.41264	0.00123	mdwds30a	0.32187	0.00127	mdwds30h	0.39096	0.00132	mdwds30c
	Shippingport LWBR								
0.2	0.88019	0.00105	slwds94a	0.86996	0.00108	slwds94h	0.87660	0.00110	slwds94
60	0.87174	0.00108	slwds30a	0.86589	0.00103	slwds30h	0.86728	0.00111	slwds30c
	Shippingport PWR								
0.2	0.88772	0.00097	spwds00a	0.87980	0.00106	spwds00h	0.88364	0.00103	spwds00
60	0.88096	0.00103	spwds30a	0.87739	0.00097	spwds30h	0.87834	0.00106	spwds30c
	N Reactor (A & B)								
0.2 (A)	0.91161	0.00055	nrwdsAa	0.86487	0.00060	nrwdsHa	0.91387	0.00066	nrwds78a
0.2 (B)	0.89428	0.00059	nrwdsAb	0.86186	0.00058	nrwdsHb	0.89433	0.00056	nrwds78b
60 (A)	0.82581	0.00067	nrwds3Aa	0.82306	0.00060	nrwds3Ha	0.85048	0.00061	nrwds3Ca
60 (B)	0.83953	0.00064	nrwds3Ab	0.83629	0.00057	nrwds3Hb	0.85321	0.00061	nrwds3Cb
	TMI-2 (“D” & “K”)								
0.2 (D)	0.87841	0.00097	tmwdsDa	0.86165	0.0009	tmwdsDh	0.87138	0.00101	tmwds15d
0.2 (K)	0.84168	0.00095	tmwdsKa	0.81426	0.00097	tmwdsKh	0.83179	0.00093	tmwds19k
60 (D)	0.86583	0.00095	tmwds3Da	0.84878	0.00097	tmwds3Dh	0.85728	0.00099	tmwds30d
60 (K)	0.82515	0.00087	tmwds3Ka	0.79387	0.00094	tmwds3Kh	0.80553	0.00099	tmwds30k

[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 efwds6a3 is efwds6a3.out).

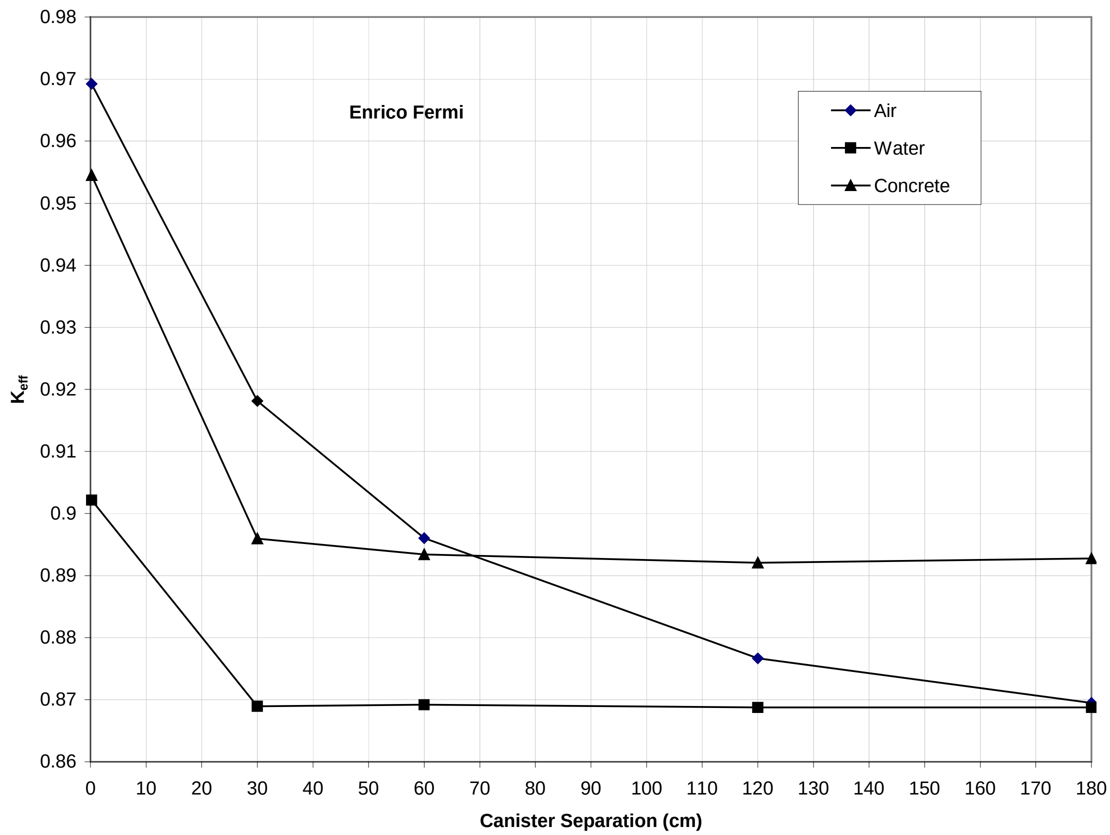


Figure 5.2-1 Illustration of k_{eff} versus Enrico Fermi Canister Separation

5.2.2 Moderator Density Variations

Moderator density, which could vary from dry to fully moderated conditions under accident conditions, were varied over the range of 0.0 to 1.0 g/cm^3 both on the inside and outside of the storage/aging cask for PWR and BWR fuel assemblies. The results are presented in Sections 6.1 and 6.2.

5.2.3 Evaluation of Criticality Controls for MSC (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 MSC (per Assumption 3.4, the MSC 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 MSC
- 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 model the fuel region as one node, as opposed to applying an axial burnup profile. This modeling approach is slightly conservative for PWR fuel (BSC 2003a, p. 36) and is assumed to be the same for BWR fuel (Assumption 3.9). Also note that the MCNP calculations for the GE 7x7 fuel assembly does not include the fuel rods containing Gd_2O_3 for conservatism (these rods are modeled as regular fuel rods with the same initial enrichment).

As described in Section 5.1.7, 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 3.8, 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.

5.2.4 Evaluation of MSC for DOE Canisters

Various loading scenarios were evaluated for the most reactive DOE fuel types as described in Section 5.1.7. 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 MSC to ensure a criticality safe configuration.

5.2.5 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 2004c, Section 7). As mentioned earlier in Section 5.1.8, Section 7 of BSC 2004c also does not identify any criticality events as Category 1 or Category 2 because it takes credit for criticality controls and design features such as those identified in the present document. Consequently, all potential events in the Aging Facility that were listed under the category of "Fissile" (BSC 2004c, Section 6.3) have instead been considered in this evaluation and are presented in Table 5.2-2.

Table 5.2-2 Criticality Related Events for the Aging Facility

Section [a]	Criticality Event Description	Criticality Safety Evaluation
6.3.7.6.1	Drop or collision of a DPC and a rearrangement of the container internals	Regulatory compliance with 10 CFR 50, 71 and 72 provides assurance of criticality safety for this event. In addition, see drop/slap down scenario evaluated below.
6.3.7.6.2	Drop or collision of an MSC and a rearrangement of the container internals	Per Assumption 3.4, the MSC is similar in design to a NRC-certified cask. There is no effect on the criticality control features of the system as a result of this event shown by the cask handling accident evaluation in Holtec International 2002, Chapter 11. Furthermore, there is no moderator intrusion to make the configuration more reactive. In addition, see drop/slap down scenario evaluated below.

[a] BSC 2004c

In addition to the evaluations presented in Table 5.2-2, 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 2004g, Section 5.2.3.2). The peak k_{eff} value for a GE 8x8 fuel assembly occurs at a pin pitch of 1.90 cm per Table 5.2-3. Note that a simplified MCNP model was used for this study only modeling a single fuel pin cell with reflective boundary conditions and 5.0 wt% fresh fuel enrichment. The results in Table 5.2-3 are only intended to show the <u>trend</u> in k_{eff} and not provide an absolute value.

Table 5.2-3 k_{eff} of Pin Pitch Increase of GE 8x8 Fuel

Pin Pitch (cm)	k_{eff}	St. Dev.	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, respectively, each reconfigured to a pin pitch of 1.45 cm and 1.90 cm. The calculations were preformed for scenarios when all of the fuel assemblies in the MPC-24 and MPC-68 were reconfigured and when only the fuel assemblies located in the center were reconfigured. Spacers that prevents the fuel from bowing out, or bending, are located near the assembly 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 spacer below the next intact spacer to bend during a drop event. Bending results in a greater assembly separation that affects only approximately the last 30 cm of assembly length. Modeling a larger assembly pitch for the last 30 cm of assembly length results in the most conservative critical configuration. However, this configuration is less realistic since assembly separation increases continuously from the intact spacer to the assembly bottom end. Therefore, a more realistic reconfiguration was also modeled, which consists of a greater assembly pitch for the last 15 cm of assembly height.

Table 5.2-4 shows the results from the calculations described above. It can be seen from the table that the increase in reactivity is fairly substantial when fuel reconfiguration occurs (compare k_{eff} values to Tables 6.1-1 and 6.2-1). The 30 cm damage fuel height cases exceed the design criteria while the 15 cm damage fuel height cases meet the design criteria for both the BWR and PWR accident conditions. Since the 15 cm damage fuel height cases is an adequate modeling height, a fuel reconfiguration will not pose a criticality concern. 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 the results presented in Table 5.2-4 for both PWR and BWR fuel. Calculations show that an increase in reactivity due to this latter scenario is very minor for both fuel types (MCNP files: MPC24b2c & MPC24b2c.out, MPC68B30 & MPC68B30.out).

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 of a dry fuel is not expected to increase k_{eff} significantly (BSC 2004g, Section 5.2.3.2) and the results in Section 6 of this document indicates that k_{eff} of a dry storage/aging configuration is below 0.4.

Table 5.2-4 Fuel Reconfiguration Evaluation for PWR and BWR Fuel

Damaged Height (cm)	k_{eff}	St. Dev.	MCNP files	k_{eff}	St. Dev.	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).

6. 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 methods and assumptions described in Sections 2 and 3, respectively.

6.1 MPC-24 (PWR FUEL)

Table 6.1-1 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 0.95 (including all bias and uncertainties), 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 6.3 presents calculations in which the Boral loading has been increased along with an alternative neutron poison.

Table 6.1-1 MPC-24 with Varied Fuel Enrichment

Enrichment (wt %)	k_{eff}	St. Dev.	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 flooded inside of the cask (i.e., inside the MPC and between the MPC and the overpack) with a dry and flooded outside cask environment (i.e., outside the overpack), respectively. The calculations feature 5 wt% fuel enrichment and reflective boundaries with 30 cm radial separation. The results from the calculations are presented in Table 6.1-2. Note that the k_{eff} values for flooded inside cask conditions exceed the upper subcritical limit (USL) due to the fact that 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. This observation is further supported by the HI-STORM FSAR where calculations also 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 in where 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 6.1-2 MPC-24 with Varied Moderator Condition

Moderation conditions	k_{eff}	St. Dev.	MCNP files
dry inside cask, dry outside cask	0.35658	0.00013	MPC24-4, MPC24-4.out
dry inside cask, flooded outside cask	0.35646	0.00013	MPC24-1a, MPC24-1a.out
flooded inside cask, dry outside cask	0.97261	0.00029	MPC24-3, MPC24-3.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 6.1-2, was used and the radial distances were changed from an infinite array of casks virtually touching each other (0.1 cm separation) to a 60 cm separation distance. The results displayed in Table 6.1-3 indicated that the MPC-24 cask ensures no neutronic interaction between casks. The results further indicate that the 30 cm flooded separation as modeled in MCNP is enough to ensure the most reactive configuration. Note that when k_{eff} values calculated by MCNP are within 2 sigma, they are the same number at the 95% confidence limit (this due to statistical uncertainties that are inherent to MCNP). Also, as mentioned earlier, these results are only intended to show the trends and not to produce an absolute k_{eff} value.

Table 6.1-3 MPC-24 with Varied Separation Distance

Distance between casks (cm)	k_{eff}	St. Dev.	MCNP files
0.1	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 6.1-4 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 6.1-4 MPC-24 with Varied Outside Moderator Densities

Outside moderator density (g/cm^3)	k_{eff}	St. Dev.	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 modeled 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 6.1-5 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). It can be seen that k_{eff} somewhat increased but is still below the USL. Also, this small increase in k_{eff} is most likely due to statistical uncertainties that are inherent to MCNP.

Table 6.1-5 MPC-24 with Varied Moderator Density between Overpack and MPC

Moderator density between overpack and MPC (g/cm^3)	k_{eff}	St. Dev.	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.

6.2 MPC-68 (BWR FUEL)

Table 6.2-1 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 0.95 (including all bias and uncertainties), 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 6.3 presents calculations in which the Boral loading has been increased along with an alternative neutron poison.

Table 6.2-1 MPC-68 with Varied Fuel Enrichment

Enrichment (wt %)	k_{eff}	St. Dev.	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 the overpack) with a dry and flooded outside cask environment (i.e., outside the 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 6.2-2. 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 6.2-2 MPC-68 with Varied Moderator Condition

Moderation conditions	k_{eff}	St. Dev.	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 6.2-3 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 6.2-3 MPC-68 with Varied Outside Moderator Densities

Outside moderator density (g/cm^3)	k_{eff}	St. Dev.	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 modeled 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 6.2-4 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). It can be seen that k_{eff} somewhat increased but is still below the USL. Also note, as mentioned earlier, that when k_{eff} values calculated by MCNP are within 2 sigma, they are the same number at the 95% confidence limit (this due to statistical uncertainties that are inherent to MCNP).

Table 6.2-4 MPC-68 with Varied Moderator Density between Overpack and MPC

Moderator density between overpack and MPC (g/cm^3)	k_{eff}	St. Dev.	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.

6.3 MSC FOR COMMERCIAL FUEL

It was shown in Sections 6.1 and 6.2 that when loading the MPC-24 and MPC-68 with 5.0 wt% enriched fuel, k_{eff} exceeds the USL. The MSC 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 6.3-1 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 thicknesses that can be manufactured. The upper limit of the ^{10}B loading is currently approximately 0.04 g $^{10}B/cm^2$ (Achudume 2004). This limit is because a ^{10}B loading above this limit would lead to diminishing returns in neutron absorption capabilities since the Boral plates would have reached saturation point. This can also be seen from the results for the PWR fuel calculations presented below in Table 6.3-1. 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 below. There are also some calculations featuring a higher B_4C-to-Al ratio (80/20) presented in Table 6.3-1 below. 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 (as stated earlier). Table 6.3-1 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. In order to utilize Boral in the MPC as a fixed neutron absorber to accommodate 5 wt% enriched fuel, more information from the manufacturer needs to be obtained regarding possible ^{10}B loading options as well as panel thickness options so that a safe loading can be identified.

Table 6.3-1 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}	St. Dev.	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 5.1.5.1) and produced a k_{eff} of 0.95937 $\pm$ 0.00030 (MCNP files: MPC24B4t & MPC24B4t.out). Note that the two k_{eff} values are within the statistical uncertainty

Even though the MSC is similar in design to an existing storage cask design (Assumption 3.4), 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-4 cask design (General Atomics 1993b, p. 6.3-2). Calculations were performed for the MPC-24 and MPC-68 with 0.2717 cm thick Boral panels exchanged for B_4C for the MPC-24 and 0.2057 cm thick Boral panel for the MPC-68. This is a conservative approximation since the GA-4 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 6.3-2 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 MSC design.

Table 6.3-2 MPC-24 and MPC-68 with B_4C Neutron Poison

Neutron poison material	k_{eff}	St. Dev.	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 MSC can accommodate 5.0 wt% enriched fuel. As stated in Section 5.2.3, the number of assemblies contained in the MSC can be reduced, fuel spacing can be increased and burnup-credit nuclides can be included in the fuel composition. Table 6.3-3 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 6.3-4 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 B&W 15x15 fuel assembly is slightly more reactive when applying burnup-credit (5 wt% initial enrichment) than the Westinghouse 17x17 OFA as presented in Table 6.3-4. The Westinghouse 17x17 OFA is, however, the more reactive fuel assembly in the MPC-24 for fresh fuel evaluations (see footnote ‘a’ of Table 6.3-4). 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 6.3-4). 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 6.3-3 Criticality Control Variations for MPC-24 and MPC-68

Scenario Description	MPC-24 (PWR Fuel)			MPC-68 (BWR Fuel)		
	k_{eff}	St. Dev.	MCNP files	k_{eff}	St. Dev.	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 6.3-4 Burnup-Credit Evaluations for PWR and BWR Fuel

Burnup (GWd/MTU)	k_{eff}	St. Dev.	MCNP files	k_{eff}	St. Dev.	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 6.1-1 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 6.2-1 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.

6.4 MSC FOR DOE FUEL CANISTERS

It was demonstrated in Section 5.2.1 that Enrico Fermi is the most reactive DOE fuel followed by Fort St. Vrain and FFTF. Table 6.4-1 presents the DOE fuel types placed inside the MSC (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). The MSC feature reflective boundary conditions. It can be seen that the k_{eff}'s are below the USL for all three DOE fuel types. Also, the Enrio Fermi calculations show that the k_{eff} of the MSC is independent on distance to the next MSC and outside conditions, which means the MSC is neutronically isolated.

Table 6.4-1 k_{eff} of Various DOE Canisters Inside an MSC

Distance (cm)	k_{eff} (Air)	St. Dev	MCNP files	k_{eff} (Water)	St. Dev	MCNP files
	Enrico Fermi (3x3 square pitch array)					
0.2	0.89411	0.00077	efa3x3a efa3x3a.out	0.89397	0.00075	efa3x3 efa3x3.out
30	0.89480	0.00078	efa3x30a efa3x30a.out	0.89425	0.00074	efa3x30 efa3x30.out
	FFTF (3x3 square pitch array)					
0.2	0.88254	0.00103	ffa0g5a ffa0g5a.out	0.88275	0.00100	ffa0g5 ffa0g5.out
	Fort St. Vrain (3x3 square pitch array)					
0.1	0.85363	0.00103	fswwa00a fswwa00a.out	0.85473	0.00098	fswwa00 fswwa00.out

Table 6.4-2 shows the Enrico Fermi 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 5.1-10). 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 Enrico Fermi fuel surrounded by 5 SRS HLW glass canisters (diameter of 24 in). As expected, and shown in Table 6.4-2, this configuration is subcritical. In addition, a comparison was made of the 3x3 square pitch arrangement for Enrico Fermi fuel presented in Table 6.4-1 to that of a circular pitch (8 canisters in a circle and 1 in the center per Figure 5.1-11). Comparing the results in Table 6.4-1 to that in Table 6.4-2, a square pitch produces a higher k_{eff}.

Table 6.4-2 k_{eff} of Various Enrico Fermi Canister Configurations

Distance (cm)	Number of Canisters	k_{eff}	St. Dev	MCNP files
Overpack Inside Diameter = 77.5 in (Triangular Pitch)				
30 (water)	10	0.90121	0.00077	efa10T, efa10T.out
30 (water)	12	0.90238	0.00079	efa12T, efa12T.out
Overpack Inside Diameter = 69.5 in (Circular Pitch)				
30 (air)	9	0.88365	0.00073	efa330aR, efa330aR.out
30 (air)	1 [a]	0.87029	0.00078	efaHLWc, efaHLWc.out

[a] Canister placed in the center of MSC surrounded by 5 SRS HLW glass canisters (Figure 5.1-11)

6.5 CATEGORY 1 AND 2 EVENT SEQUENCES

No Category 1 and 2 event sequences applicable to Aging Facility have been identified (BSC 2004c). Per the discussion presented in Section 5.2.5, potential events in the Aging Facility were evaluated 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, while the bounding representation exceeded the limits. However, the bounding scenario is considered beyond Category 2 (see Section 5.2.5).

6.6 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, designed to hold 24 PWR assemblies, can safely be stored on the aging pads with fuel content per the Certificate of Compliance (maximum 4.0 wt% fuel enrichment). The MPC-68, designed to hold 68 PWR assemblies, can also safely be stored on the aging pads with fuel contents up to 4.2 wt% enrichment while remaining below USL.

- Reactivity of the loaded casks decreases with reduction in moderator density.

- Maximum reactivity is reached when the fuel storage/aging casks are fully flooded with water at full density (1.0 g/cm^3).

- Mist conditions (i.e., moderator densities between 0.02 to 0.1 g/cm^3) surrounding the outside of the casks do not cause the k_{eff} to go beyond that for 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.

- The PWR, BWR, and DOE fuel canisters results when placed inside the MSC 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 MSCs 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 MSC, another neutron poison besides Boral might need to be included in the internal basket. In order to utilize Boral in the MPC as a fixed neutron absorber to accommodate 5 wt% enriched fuel, more information from the manufacturer needs to be obtained regarding possible ^{10}B loading options as well as panel thickness options 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. Should B_4C be chosen as a neutron poison for the MSC, exact dimensions and B_4C contents need to be evaluated during the detailed design phase.

- To accommodate 5.0 wt% enriched commercial fuel in the MSC, a larger assembly separation (1 cm addition) or taking credit for low burnups (conservative) is needed to allow for a criticality safe configuration.

- Category 1 and 2 event sequences potentially occurring in the Aging Facility do not compromise criticality safety.

7. REFERENCES

10 CFR 63. 2002. Energy: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada. Readily available.

10 CFR 71. 1999. Energy: Packaging and Transportation of Radioactive Material. Readily available.

10 CFR 72. 1999. Energy: Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste. Readily available.

AP-3.12Q, Rev. 2, ICN 2. *Design Calculations and Analyses.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Managment. ACC: DOC.20040318.0002.

AP-3.13Q, Rev. 3, ICN 3. *Design Control.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20040202.0006.

ASME (American Society of Mechanical Engineers) 2001. *2001 ASME Boiler and Pressure Vessel Code (includes 2002 addenda).* New York, New York: American Society of Mechanical Engineers. TIC: 251425.

ASTM A 276-91a. 1991. *Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes.* Philadelphia, Pennsylvania: American Society for Testing and Materials. TIC: 240022.

ASTM G 1-90 (Reapproved 1999). 1999. *Standard Practice for Preparing, Cleaning, and Evaluating Corrosion Test Specimens.* West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 238771.

Achudume A. 2004. "Phone Conversation Regarding B-10 Loading in Boral Plates for Spent Nuclear Fuel Storage/Transportation Casks." Memo from A. Achudume to File, March 29, 2004. ACC: MOL.20040330.0243.

Briesmeister, J.F., ed. 1997. *MCNP-A General Monte Carlo N-Particle Transport Code.* LA-12625-M, Version 4B. Los Alamos, New Mexico: Los Alamos National Laboratory. ACC: MOL.19980624.0328.

BSC (Bechtel SAIC Company) 2001a. *Intact and Degraded Mode Criticality Calculations for the Codisposal of Fort Saint Vrain HTGR Spent Nuclear Fuel in a Waste Package.* CAL-EDC-NU-000007 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20011015.0020.

BSC (Bechtel SAIC Company) 2001b. *Intact and Degraded Mode Criticality Calculations for the Codisposal of Melt and Dilute Ingots in a Waste Package.* CAL-EDC-NU-000006 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010730.0062.

BSC (Bechtel SAIC Company) 2001c. *Statement of Work for DOE—Office of Civilian Radioactive Waste Management, Technical Assistance on Melt-Dilute Criticality and Shielding Analyses, Revision 2, May 30, 2001*. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010619.0626.

BSC (Bechtel SAIC Company) 2003a. *21-PWR Waste Package with Absorber Plates Loading Curve Evaluation.* CAL-DSU-NU-000006 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20031222.0007.

BSC (Bechtel SAIC Company) 2003b. *Isotopic Generation and Confirmation of the PWR Application Model.* CAL-DSU-NU-000004 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20031110.0003.

BSC (Bechtel SAIC Company) 2003c. *Criticality Model Report.* MDL-EBS-NU-000003 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20031013.0002.

BSC (Bechtel SAIC Company) 2004a. *Aging Facility Criticality Safety Calculations.* 170-00C-HA00-00100-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040415.0012.

BSC (Bechtel SAIC Company) 2004b. *Canister Handling Facility Criticality Safety Calculations.* 190-00C-CH00-00100-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040302.0017.

BSC (Bechtel SAIC Company) 2004c. *Categorization of Event Sequences for License Application.* 000-00C-MGR0-00800-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040722.0001.

BSC (Bechtel SAIC Company) 2004d. *Fuel Handling Facility Criticality Safety Calculations.* 210-00C-FH00-00400-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040630.0012.

BSC (Bechtel SAIC Company) 2004e. *Preclosure Criticality Analysis Process Report.* TDR-EBS-NU-000004 REV 03. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040524.0001.

BSC (Bechtel SAIC Company) 2004f. *SNF Aging System Description Document.* 170-3YD-HA00-00100-000-002. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040709.0018.

BSC (Bechtel SAIC Company) 2004g. *Surface Facility Criticality Safety Calculations.* 100-00C-WHS0-00100-000-00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040309.0001.

BSC (Bechtel SAIC Company) 2004h. *PWR and BWR Source Term Sensitivity Study*. 000-00C-MGR0-00300-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040114.0003.

BSC (Bechtel SAIC Company) 2004i. *Q-List.* 000-30R-MGR0-00500-000-000 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040721.0007.

BSC (Bechtel SAIC Company) 2004j. *Inventory and Characteristics of Potential Repository Wastes.* 000-00C-MGR0-01200-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040517.0003.

Cogema. 2004. *Evaluation of 10CFR72 - Licensed Casks for Use in Aging Spent Nuclear Fuel.* COGEMA-C0115-RP-04-001, Rev. 0. Las Vegas, Nevada: Cogema. ACC: ENG.20040405.0025.

CRWMS M&O 1998a. *Software Code: MCNP*. V4B2LV. HP, HPUX 9.07 and 10.20; PC, Windows 95; Sun, Solaris 2.6. 30033 V4B2LV.

CRWMS M&O 1998b. *Software Qualification Report for MCNP Version 4B2, A General Monte Carlo N-Particle Transport Code.* CSCI: 30033 V4B2LV. DI: 30033-2003, Rev. 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980622.0637.

CRWMS M&O 1999a. *Fast Flux Test Facility (FFTF) Reactor Fuel Criticality Calculations.* BBA000000-01717-0210-00016 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990426.0142.

CRWMS M&O 1999b. *DOE SRS HLW Glass Chemical Composition.* BBA000000-01717-0210-00038 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990215.0397.

CRWMS M&O 1999c. Reserved.

CRWMS M&O 1999d. *TRIGA Fuel Phase I and II Criticality Calculation.* CAL-MGR-NU-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991209.0195.

CRWMS M&O 2000a. *Enrico Fermi Fast Reactor Spent Nuclear Fuel Criticality Calculations: Degraded Mode.* CAL-EDC-NU-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000802.0002.

CRWMS M&O 2000b. *Intact and Degraded Criticality Calculations for the Codisposal of Shippingport LWBR Spent Nuclear Fuel in a Waste Package.* CAL-EDC-NU-000004 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000922.0093.

CRWMS M&O 2000c. *Intact and Degraded Criticality Calculations for the Codisposal of Shippingport PWR Fuel in a Waste Package.* CAL-EDC-NU-000002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000209.0233.

CRWMS M&O 2001. *Intact and Degraded Component Criticality Calculations of N Reactor Spent Nuclear Fuel.* CAL-EDC-NU-000003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20010223.0060.

DOE (U.S. Department of Energy) 1987. *Appendix 2A. Physical Descriptions of LWR Fuel Assemblies.* Volume 3 of *Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation.* DOE/RW-0184. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: HQX.19880405.0024.

DOE (U.S. Department of Energy) 1999a. *Fermi (U-Mo) Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-035, Rev. 0. Washington, D.C.: U.S. Department of Energy. TIC: 242461.

DOE (U.S. Department of Energy) 1999b. *Shippingport LWBR (Th/U Oxide) Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-051, Rev. 0. Washington, D.C.: U.S. Department of Energy, Office of Environmental Management. TIC: 245631.

DOE (U.S. Department of Energy) 1999c. *Shippingport PWR (HEU Oxide) Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-040, Rev. 0. Washington, D.C.: U.S. Department of Energy. TIC: 243528.

DOE (U.S. Department of Energy) 1999d. *TRIGA (UZrH) Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-048, Rev. 0. Washington, D.C.: U.S. Department of Energy. TIC: 244162.

DOE (U.S. Department of Energy) 2000. *N Reactor (U-Metal) Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-056, Rev. 0. [Washington, D.C.]: U.S. Department of Energy, Office of Environmental Management. TIC: 247956.

DOE (U.S. Department of Energy) 2003. *TMI Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-084, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. ACC: MOL.20031013.0388.

DOE (U.S. Department of Energy) 2004. *Quality Assurance Requirements and Description.* DOE/RW-0333P, Rev. 16. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20040907.0002.

Doraswamy, N. 2004. *Project Design Criteria Document.* 000-3DR-MGR0-00100-000-002. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040721.0003.

General Atomics. 1993a. Reserved.

General Atomics. 1993b. *GA-9 Legal Weight Truck From-Reactor Spent Fuel Shipping Cask, Final Design Report.* 910354/0. San Diego, California: General Atomics. ACC: NNA.19940513.0214.

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.

Hunter, I. 2002. "TN-68 FSAR Update per 10CFR72.248 Docket 72-1027." Letter from I. Hunter (Transnuclear) to M. Ross-Lee (NRC), May 31, 2002, E-19584, with enclosure. TIC: 255259.

INEEL (Idaho National Engineering and Environmental Laboratory) 2002. *FFTF (MOX) Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-032, Rev. 1. Idaho Falls, Idaho: U.S. Department of Energy, Idaho National Operations Office. TIC: 252933.

LP-SI.11Q-BSC, Rev. 0, ICN 0. *Software Management.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20040225.0007.

Mecham, D.C., ed. 2004. *Waste Package Component Design Methodology Report.* 000-30R-WIS0-00100-000-002. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20040713.0003.

NRC (U.S. Nuclear Regulatory Commission) 1997. *SCALE, RSIC Computer Code Collection (CCC-545).* NUREG/CR-0200, Rev. 5. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 235920.

NRC (U.S. Nuclear Regulatory Commission) 2000. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG-1567. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 247929.

Parrington, J.R.; Knox, H.D.; Breneman, S.L.; Baum, E.M.; and Feiner, F. 1996. *Nuclides and Isotopes, Chart of the Nuclides.* 15th Edition. San Jose, California: General Electric Company and KAPL, Inc. TIC: 233705.

Sanders, C.E. and Wagner, J.C. 2002. *Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit.* NUREG/CR-6760. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 254260.

Stout, R.B. and Leider, H.R., eds. 1991. *Preliminary Waste Form Characteristics Report.* Version 1.0. Livermore, California: Lawrence Livermore National Laboratory. ACC: MOL.19940726.0118.

Taylor, L.L. 2001. *Fort Saint Vrain HTGR (Th/U Carbide) Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-060, Rev. 0. [Washington, D.C.]: U.S. Department of Energy, Office of Environmental Management. TIC: 249783.

Wagner, J.C. and Parks, C.V. 2000. *A Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analyses for PWR Spent Fuel*

Pool Storage. NUREG/CR-6683. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 254257.

Wagner, J.C. and Sanders, C.E. 2003. *Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs.* NUREG/CR-6800. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20040720.0144.

Wimmer, L.B. 2004. *Isotopic Generation and Confirmation of the BWR Appl. Model.* 32-5035847-01. Las Vegas, Nevada: Areva. ACC: DOC.20040630.0007.

YMP (Yucca Mountain Site Characterization Project) 2003. *Disposal Criticality Analysis Methodology Topical Report.* YMP/TR-004Q, Rev. 02. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: DOC.20031110.0005.

8. ATTACHMENTS

This calculation document includes three attachments:

ATTACHMENT I	Listing of Computer Files (8 pages)
ATTACHMENT II	One Compact Disk Containing All Files Listed in Attachment I (1 of 1) (0 pages)
ATTACHMENT III	Sketch of the Aging Pad in the Aging Facility (1 page)

ATTACHMENT I
LISTING OF COMPUTER FILES

This attachment lists the input and output file names for the MCNP and Excel calculations. All input and output are stored on an electronic medium (compact disc) in ASCII format as part of this attachment.

Date	Time	File Size	File Name
03/31/2004	01:25p	17,408	fuelcomp.xls
07/23/2004	11:43a	57,344	boral.xls
04/05/2004	09:58a	19,227	PWR/MPC24-1a
04/05/2004	09:58a	527,535	PWR/MPC24-1a.out
04/05/2004	09:58a	18,990	PWR/MPC24-2
04/05/2004	09:58a	527,686	PWR/MPC24-2.out
04/05/2004	09:58a	18,990	PWR/MPC24-2a
04/05/2004	09:58a	527,686	PWR/MPC24-2a.out
04/05/2004	09:58a	18,992	PWR/MPC24-2b
04/05/2004	09:58a	527,784	PWR/MPC24-2b.out
04/05/2004	09:58a	18,982	PWR/MPC24-2c
04/05/2004	09:58a	527,990	PWR/MPC24-2c.out
04/05/2004	09:58a	18,994	PWR/MPC24-2d
04/05/2004	09:58a	527,990	PWR/MPC24-2d.out
04/05/2004	09:58a	19,003	PWR/MPC24-3
04/05/2004	09:58a	527,731	PWR/MPC24-3.out
04/05/2004	09:58a	19,225	PWR/MPC24-4
04/05/2004	09:58a	527,535	PWR/MPC24-4.out
04/05/2004	09:58a	18,995	PWR/MIST/MPC24m0
04/05/2004	09:58a	527,888	PWR/MIST/MPC24m0.out
04/05/2004	09:58a	18,910	PWR/MIST/MPC24m1
04/05/2004	09:58a	528,412	PWR/MIST/MPC24m1.out
04/07/2004	09:01a	18,930	PWR/MIST/MPC24m1a
04/07/2004	09:01a	528,010	PWR/MIST/MPC24m1a.out
04/05/2004	09:58a	18,912	PWR/MIST/MPC24m2
04/05/2004	09:58a	528,206	PWR/MIST/MPC24m2.out
04/05/2004	09:58a	18,930	PWR/MIST/MPC24m2a
04/05/2004	09:58a	528,010	PWR/MIST/MPC24m2a.out
04/05/2004	09:58a	18,912	PWR/MIST/MPC24m3
04/05/2004	09:58a	528,363	PWR/MIST/MPC24m3.out
04/05/2004	09:58a	18,911	PWR/MIST/MPC24m5
04/05/2004	09:58a	528,412	PWR/MIST/MPC24m5.out
04/05/2004	09:58a	18,910	PWR/MIST/MPC24m50
04/05/2004	09:58a	528,059	PWR/MIST/MPC24m50.out
04/05/2004	09:58a	18,912	PWR/MIST/MPC24m7
04/05/2004	09:58a	528,412	PWR/MIST/MPC24m7.out
04/05/2004	09:58a	18,912	PWR/MIST/MPC24m8

Date	Time	File Size	File Name
04/07/2004	09:00a	528,412	PWR/MIST/MPC24m8.out
04/07/2004	09:55a	19,018	PWR/BORAL/MPC24B4
04/07/2004	09:55a	527,686	PWR/BORAL/MPC24B4.out
04/05/2004	09:58a	19,122	PWR/BORAL/MPC24b4c
04/05/2004	09:58a	528,502	PWR/BORAL/MPC24b4c.out
04/07/2004	09:55a	19,173	PWR/BORAL/MPC24B6
04/07/2004	09:55a	528,495	PWR/BORAL/MPC24B6.out
04/07/2004	09:55a	19,165	PWR/BORAL/MPC24B63
04/07/2004	09:55a	528,544	PWR/BORAL/MPC24B63.out
04/07/2004	09:55a	19,164	PWR/BORAL/MPC24B64
04/07/2004	09:55a	528,495	PWR/BORAL/MPC24B64.out
04/07/2004	09:55a	19,181	PWR/BORAL/MPC24B8
04/07/2004	09:55a	528,544	PWR/BORAL/MPC24B8.out
04/07/2004	09:55a	19,196	PWR/BORAL/MPC24B83
04/07/2004	09:55a	528,603	PWR/BORAL/MPC24B83.out
04/07/2004	09:55a	19,182	PWR/BORAL/MPC24B84
04/07/2004	09:55a	528,446	PWR/BORAL/MPC24B84.out
04/05/2004	09:59a	10,247	BWR/MPC68-1
04/05/2004	09:59a	467,928	BWR/MPC68-1.out
04/05/2004	09:59a	10,249	BWR/MPC68-1a
04/05/2004	09:59a	467,722	BWR/MPC68-1a.out
04/05/2004	09:59a	10,159	BWR/MPC68-2
04/05/2004	09:59a	468,372	BWR/MPC68-2.out
04/05/2004	09:59a	10,161	BWR/MPC68-2a
04/05/2004	09:59a	468,319	BWR/MPC68-2a.out
04/05/2004	09:59a	10,167	BWR/MPC68-45
04/05/2004	09:59a	468,215	BWR/MPC68-45.out
04/05/2004	09:59a	10,167	BWR/MPC68-5
04/05/2004	09:59a	466,931	BWR/MPC68-5.out
04/05/2004	09:59a	10,252	BWR/MIST/MPC68m1
04/05/2004	09:59a	468,794	BWR/MIST/MPC68m1.out
04/07/2004	09:56a	10,283	BWR/MIST/MPC68m1a
04/07/2004	09:56a	468,588	BWR/MIST/MPC68m1a.out
04/05/2004	09:59a	10,252	BWR/MIST/MPC68m2
04/05/2004	09:59a	468,794	BWR/MIST/MPC68m2.out
04/07/2004	09:56a	10,283	BWR/MIST/MPC68m2a
04/07/2004	09:56a	468,197	BWR/MIST/MPC68m2a.out
04/05/2004	09:59a	10,249	BWR/MIST/MPC68m5
04/05/2004	09:59a	468,794	BWR/MIST/MPC68m5.out
04/07/2004	09:56a	10,249	BWR/MIST/MPC68m50
04/07/2004	09:56a	468,794	BWR/MIST/MPC68m50.out
04/05/2004	09:59a	10,249	BWR/MIST/MPC68m7
04/05/2004	09:59a	468,794	BWR/MIST/MPC68m7.out
04/07/2004	09:57a	10,174	BWR/BORAL/MPC68B3

Date	Time	File Size	File Name
04/07/2004	09:57a	468,264	BWR/BORAL/MPC68B3.out
04/07/2004	09:57a	10,174	BWR/BORAL/MPC68B4
04/07/2004	09:57a	468,166	BWR/BORAL/MPC68B4.out
04/05/2004	09:59a	10,142	BWR/BORAL/MPC68b4c
04/05/2004	09:59a	468,271	BWR/BORAL/MPC68b4c.out
07/23/2004	03:07p	12,987	MSC/efa10T
07/23/2004	03:07p	674,725	MSC/efa10T.out
07/23/2004	03:07p	13,190	MSC/efa12T
07/23/2004	03:07p	677,304	MSC/efa12T.out
07/23/2004	03:07p	12,908	MSC/efa330aR
07/23/2004	03:07p	673,494	MSC/efa330aR.out
07/23/2004	03:07p	12,918	MSC/efa3x3
07/23/2004	03:07p	671,521	MSC/efa3x3.out
07/23/2004	03:07p	12,918	MSC/efa3x30
07/23/2004	03:07p	671,577	MSC/efa3x30.out
07/23/2004	03:07p	12,912	MSC/efa3x30a
07/23/2004	03:07p	670,808	MSC/efa3x30a.out
07/23/2004	03:07p	12,918	MSC/efa3x37
07/23/2004	03:07p	671,521	MSC/efa3x37.out
07/23/2004	03:07p	12,912	MSC/efa3x3a
07/23/2004	03:07p	671,199	MSC/efa3x3a.out
07/23/2004	03:07p	12,912	MSC/efa3x3a7
07/23/2004	03:07p	671,297	MSC/efa3x3a7.out
07/23/2004	03:07p	16,200	MSC/ffa0g5
07/23/2004	03:07p	489,223	MSC/ffa0g5.out
07/23/2004	03:07p	16,194	MSC/ffa0g5a
07/23/2004	03:07p	488,582	MSC/ffa0g5a.out
07/23/2004	03:10p	12,139	MSC/fswwa00
07/23/2004	03:10p	534,969	MSC/fswwa00.out
07/23/2004	03:10p	12,133	MSC/fswwa00a
07/23/2004	03:10p	534,861	MSC/fswwa00a.out
09/02/2004	12:43p	20,013	MSC/M24B10BU
09/02/2004	12:43p	534,900	MSC/M24B10BU.out
09/02/2004	12:43p	19,996	MSC/M24B20BU
09/02/2004	12:43p	534,965	MSC/M24B20BU.out
09/02/2004	12:44p	20,009	MSC/M24B30BU
09/02/2004	12:43p	534,867	MSC/M24B30BU.out
08/06/2004	02:52p	19,736	MSC/MP24B1S
08/06/2004	02:52p	528,967	MSC/MP24B1S.out
08/06/2004	02:52p	19,636	MSC/MP24B84S
08/06/2004	02:52p	528,875	MSC/MP24B84S.out
07/23/2004	03:24p	10,385	MSC/MP68B3S5
07/23/2004	03:24p	468,519	MSC/MP68B3S5.out
07/23/2004	03:12p	19,218	MSC/MPC24B12

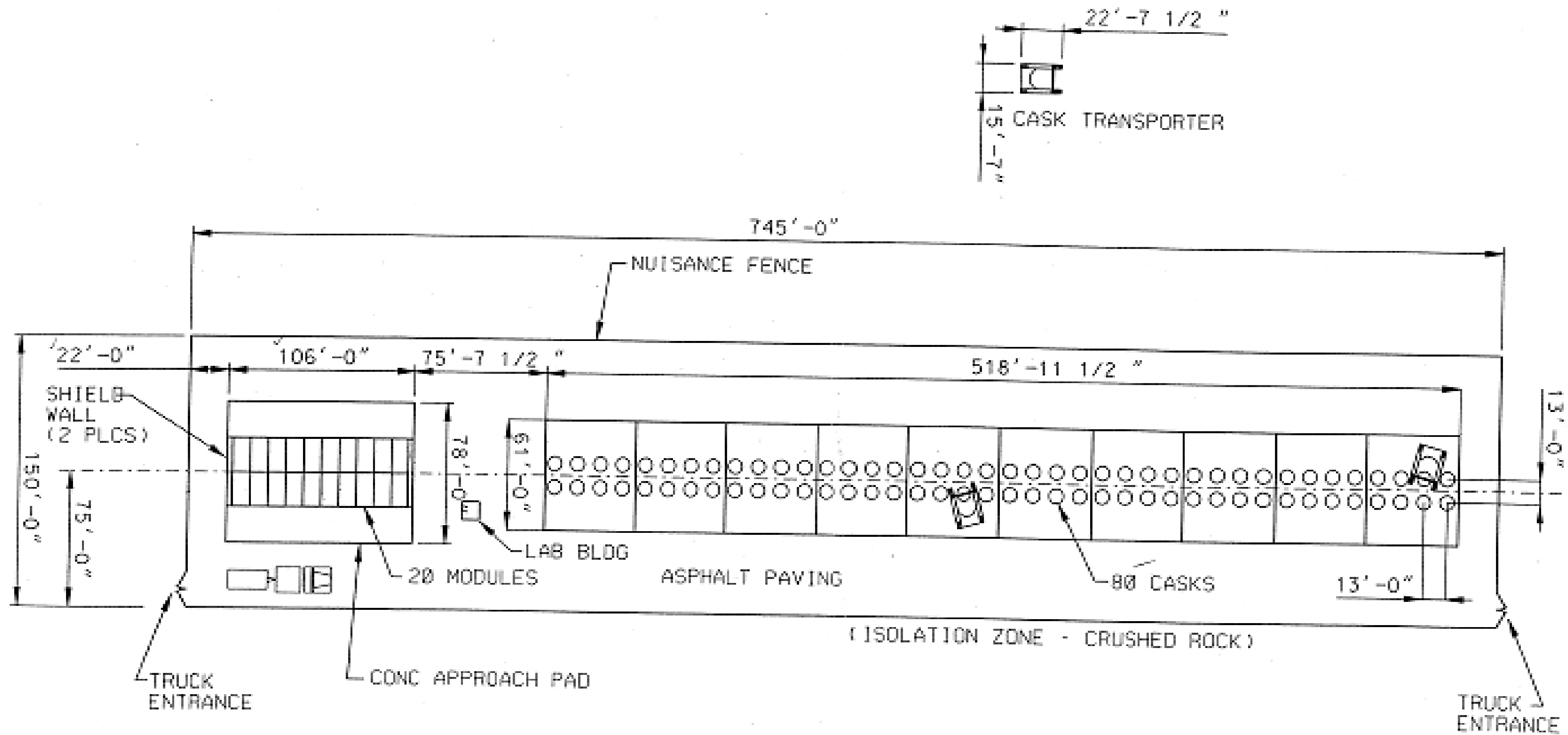
Date	Time	File Size	File Name
07/23/2004	03:12p	528,685	MSC/MPC24B12.out
07/23/2004	03:12p	19,253	MSC/MPC24B84
07/23/2004	03:12p	528,793	MSC/MPC24B84.out
07/23/2004	03:24p	10,385	MSC/MPC68B3S
07/23/2004	03:24p	468,166	MSC/MPC68B3S.out
07/23/2004	03:23p	10,174	MSC/MPC68B3W
07/23/2004	03:23p	468,215	MSC/MPC68B3W.out
07/23/2004	03:23p	10,174	MSC/MPC68BW1
07/23/2004	03:23p	468,166	MSC/MPC68BW1.out
09/02/2004	12:43p	10,992	MSC/M68B10BU
09/02/2004	12:43p	474,734	MSC/M68B10BU.out
09/02/2004	12:43p	10,992	MSC/M68B20BU
09/02/2004	12:43p	474,685	MSC/M68B20BU.out
09/02/2004	12:43p	10,992	MSC/M68B30BU
09/02/2004	12:43p	474,685	MSC/M68B30BU.out
08/09/2004	08:27a	10,122	MSC/M7x7-2
08/09/2004	08:27a	467,946	MSC/M7x7-2.out
08/09/2004	08:34a	35,197	MSC/MPC2415C
08/09/2004	08:34a	605,341	MSC/MPC2415C.out
08/09/2004	08:34a	35,196	MSC/MPC2430C
08/09/2004	08:34a	605,390	MSC/MPC2430C.out
08/09/2004	08:34a	35,200	MSC/MPC24b2E
08/09/2004	08:34a	601,198	MSC/MPC24b2E.out
08/09/2004	08:34a	35,200	MSC/MPC24E15
08/09/2004	08:34a	601,198	MSC/MPC24E15.out
08/09/2004	08:32a	15,460	MSC/MPC68C15
08/09/2004	08:32a	497,965	MSC/MPC68C15.out
08/09/2004	08:32a	15,462	MSC/MPC68C30
08/09/2004	08:32a	497,965	MSC/MPC68C30.out
08/09/2004	08:31a	15,492	MSC/MPC68M15
08/09/2004	08:31a	496,201	MSC/MPC68M15.out
08/09/2004	08:32a	15,492	MSC/MPC68M30
08/09/2004	08:32a	496,299	MSC/MPC68M30.out
08/09/2004	04:13p	19,018	MSC/MPC24B4t
08/09/2004	04:13p	527,784	MSC/MPC24B4t.out
08/12/2004	11:02a	18,860	MSC/MPCbw15
08/12/2004	11:02a	527,542	MSC/MPCbw15.out
08/13/2004	10:43a	19,702	MSC/M15B10pi
08/13/2004	10:43a	534,012	MSC /M15B10pi.out
08/13/2004	10:43a	19,685	MSC /M15B20pi
08/13/2004	10:43a	533,963	MSC /M15B20pi.out
08/13/2004	10:43a	19,698	MSC /M15B30pi
08/13/2004	10:43a	534,110	MSC /M15B30pi.out
08/13/2004	10:44a	10,960	MSC /M7B10pi

Date	Time	File Size	File Name
08/13/2004	10:44a	474,014	MSC /M7B10pi.out
08/13/2004	10:44a	10,947	MSC /M7B20pi
08/13/2004	10:44a	474,269	MSC /M7B20pi.out
08/13/2004	10:44a	10,947	MSC /M7B30pi
08/13/2004	10:44a	474,112	MSC /M7B30pi.out
09/09/2004	11:37a	14,227	DOE/efaHLWc
09/09/2004	11:37a	693,869	DOE/efaHLWc.out
07/23/2004	02:56p	18,068	DOE/efwds6a1
07/23/2004	02:56p	710,214	DOE/efwds6a1.out
07/23/2004	02:56p	18,068	DOE/efwds6a3
07/23/2004	02:56p	720,292	DOE/efwds6a3.out
07/23/2004	02:56p	18,068	DOE/efwds6a6
07/23/2004	02:56p	720,292	DOE/efwds6a6.out
07/23/2004	02:56p	18,069	DOE/efwds6a7
07/23/2004	02:56p	721,236	DOE/efwds6a7.out
07/23/2004	02:56p	18,072	DOE/efwds6a9
07/23/2004	02:56p	727,148	DOE/efwds6a9.out
07/23/2004	02:56p	18,076	DOE/efwds6c1
07/23/2004	02:56p	715,371	DOE/efwds6c1.out
07/23/2004	02:56p	18,076	DOE/efwds6c3
07/23/2004	02:56p	728,948	DOE/efwds6c3.out
07/23/2004	02:56p	18,076	DOE/efwds6c6
07/23/2004	02:56p	727,751	DOE/efwds6c6.out
07/23/2004	02:56p	18,077	DOE/efwds6c7
07/23/2004	02:56p	725,973	DOE/efwds6c7.out
07/23/2004	02:56p	18,080	DOE/efwds6c9
07/23/2004	02:56p	747,495	DOE/efwds6c9.out
07/23/2004	02:56p	18,075	DOE/efwds6h1
07/23/2004	02:56p	726,295	DOE/efwds6h1.out
07/23/2004	02:56p	18,075	DOE/efwds6h3
07/23/2004	02:56p	726,173	DOE/efwds6h3.out
07/23/2004	02:56p	18,075	DOE/efwds6h6
07/23/2004	02:56p	725,988	DOE/efwds6h6.out
07/23/2004	02:56p	18,137	DOE/efwds6h7
07/23/2004	02:56p	723,158	DOE/efwds6h7.out
07/23/2004	02:57p	18,079	DOE/efwds6h9
07/23/2004	02:57p	725,988	DOE/efwds6h9.out
07/23/2004	02:57p	15,536	DOE/ffwds15a
07/23/2004	02:57p	683,164	DOE/ffwds15a.out
07/23/2004	02:57p	15,544	DOE/ffwds15c
07/23/2004	02:57p	687,288	DOE/ffwds15c.out
07/23/2004	02:57p	15,542	DOE/ffwds15h
07/23/2004	02:57p	683,080	DOE/ffwds15h.out
07/23/2004	02:57p	15,536	DOE/ffwds30a

Date	Time	File Size	File Name
07/23/2004	02:57p	682,846	DOE/ffwds30a.out
07/23/2004	02:57p	15,544	DOE/ffwds30c
07/23/2004	02:57p	686,963	DOE/ffwds30c.out
07/23/2004	02:57p	15,542	DOE/ffwds30h
07/23/2004	02:57p	683,073	DOE/ffwds30h.out
07/23/2004	02:58p	11,180	DOE/fswds00
07/23/2004	02:58p	609,614	DOE/swds00.out
07/23/2004	02:58p	11,170	DOE/fswds00a
07/23/2004	02:58p	605,691	DOE/fswds00a.out
07/23/2004	02:58p	11,176	DOE/fswds00h
07/23/2004	02:58p	606,132	DOE/fswds00h.out
07/23/2004	02:58p	11,171	DOE/fswds30a
07/23/2004	02:58p	605,590	DOE/fswds30a.out
07/23/2004	02:58p	11,180	DOE/fswds30c
07/23/2004	02:58p	610,024	DOE/fswds30c.out
07/23/2004	02:58p	11,177	DOE/fswds30h
07/23/2004	02:58p	606,019	DOE/fswds30h.out
07/23/2004	02:59p	5,678	DOE/mdwds00
07/23/2004	02:59p	342,992	DOE/mdwds00.out
07/23/2004	02:59p	5,672	DOE/mdwds00a
07/23/2004	02:59p	339,417	DOE/mdwds00a.out
07/23/2004	02:59p	5,676	DOE/mdwds00h
07/23/2004	02:59p	339,641	DOE/mdwds00h.out
08/06/2004	02:50p	5,672	DOE/mdwds30a
08/06/2004	02:50p	339,417	DOE/mdwds30a.out
08/06/2004	02:50p	5,679	DOE/mdwds30c
08/06/2004	02:50p	342,992	DOE/mdwds30c.out
07/23/2004	02:59p	5,676	DOE/mdwds30h
07/23/2004	02:59p	339,433	DOE/mdwds30h.out
07/23/2004	03:04p	7,431	DOE/nrwds3Aa
07/23/2004	03:04p	412,419	DOE/nrwds3Aa.out
07/23/2004	03:03p	7,454	DOE/nrwds3Ab
07/23/2004	03:03p	413,166	DOE/nrwds3Ab.out
07/23/2004	03:04p	7,439	DOE/nrwds3Ca
07/23/2004	03:04p	416,948	DOE/nrwds3Ca.out
07/23/2004	03:03p	7,462	DOE/nrwds3Cb
07/23/2004	03:03p	415,333	DOE/nrwds3Cb.out
07/23/2004	03:04p	7,437	DOE/nrwds3Ha
07/23/2004	03:04p	412,954	DOE/nrwds3Ha.out
07/23/2004	03:03p	7,460	DOE/nrwds3Hb
07/23/2004	03:03p	412,520	DOE/nrwds3Hb.out
07/23/2004	03:04p	7,438	DOE/nrwds78a
07/23/2004	03:04p	416,646	DOE/nrwds78a.out
07/23/2004	03:03p	7,461	DOE/nrwds78b

Date	Time	File Size	File Name
07/23/2004	03:03p	416,965	DOE/nrwds78b.out
07/23/2004	03:04p	7,430	DOE/nrwdsAa
07/23/2004	03:04p	412,419	DOE/nrwdsAa.out
07/23/2004	03:03p	7,453	DOE/nrwdsAb
07/23/2004	03:03p	412,953	DOE/nrwdsAb.out
07/23/2004	03:04p	7,436	DOE/nrwdsHa
07/23/2004	03:04p	411,773	DOE/nrwdsHa.out
07/23/2004	03:03p	7,459	DOE/nrwdsHb
07/23/2004	03:03p	411,996	DOE/nrwdsHb.out
07/23/2004	03:05p	11,855	DOE/slwds30a
07/23/2004	03:05p	489,184	DOE/slwds30a.out
07/23/2004	03:05p	11,863	DOE/slwds30c
07/23/2004	03:05p	493,888	DOE/slwds30c.out
07/23/2004	03:05p	11,861	DOE/slwds30h
07/23/2004	03:05p	490,309	DOE/slwds30h.out
07/23/2004	03:05p	11,863	DOE/slwds94
07/23/2004	03:05p	493,937	DOE/slwds94.out
07/23/2004	03:05p	11,855	DOE/slwds94a
07/23/2004	03:05p	489,712	DOE/slwds94a.out
07/23/2004	03:05p	11,861	DOE/slwds94h
07/23/2004	03:05p	490,140	DOE/slwds94h.out
07/23/2004	03:05p	27,739	DOE/spwds00
07/23/2004	03:05p	799,627	DOE/spwds00.out
07/23/2004	03:05p	27,731	DOE/spwds00a
07/23/2004	03:05p	795,321	DOE/spwds00a.out
07/23/2004	03:05p	27,737	DOE/spwds00h
07/23/2004	03:05p	795,866	DOE/spwds00h.out
07/23/2004	03:05p	27,731	DOE/spwds30a
07/23/2004	03:05p	795,003	DOE/spwds30a.out
07/23/2004	03:05p	27,739	DOE/spwds30c
07/23/2004	03:05p	799,425	DOE/spwds30c.out
07/23/2004	03:05p	27,737	DOE/spwds30h
07/23/2004	03:05p	795,988	DOE/spwds30h.out
07/23/2004	03:05p	6,599	DOE/tmwds15d
07/23/2004	03:05p	447,720	DOE/tmwds15d.out
07/23/2004	03:06p	7,742	DOE/tmwds19k
07/23/2004	03:06p	470,547	DOE/tmwds19k.out
07/23/2004	03:06p	6,599	DOE/tmwds30d
07/23/2004	03:06p	447,928	DOE/tmwds30d.out
07/23/2004	03:06p	7,742	DOE/tmwds30k
07/23/2004	03:06p	470,535	DOE/tmwds30k.out
07/23/2004	03:06p	6,591	DOE/tmwds3Da
07/23/2004	03:06p	446,164	DOE/tmwds3Da.out
07/23/2004	03:06p	6,597	DOE/tmwds3Dh

Date	Time	File Size	File Name
07/23/2004	03:06p	446,596	DOE/tmwds3Dh.out
07/23/2004	03:06p	7,734	DOE/tmwds3Ka
07/23/2004	03:06p	466,113	DOE/tmwds3Ka.out
07/23/2004	03:06p	7,740	DOE/tmwds3Kh
07/23/2004	03:06p	466,695	DOE/tmwds3Kh.out
07/23/2004	03:06p	6,591	DOE/tmwdsDa
07/23/2004	03:06p	445,960	DOE/tmwdsDa.out
07/23/2004	03:06p	6,597	DOE/tmwdsDh
07/23/2004	03:06p	446,285	DOE/tmwdsDh.out
07/23/2004	03:06p	7,734	DOE/tmwdsKa
07/23/2004	03:06p	466,207	DOE/tmwdsKa.out
07/23/2004	03:06p	7,740	DOE/tmwdsKh
07/23/2004	03:06p	466,480	DOE/tmwdsKh.out
07/23/2004	03:06p	6,120	DOE/trwds30a
07/23/2004	03:06p	461,099	DOE/trwds30a.out
07/23/2004	03:06p	6,127	DOE/trwds30c
07/23/2004	03:06p	465,114	DOE/trwds30c.out
07/23/2004	03:06p	6,125	DOE/trwds30h
07/23/2004	03:06p	461,568	DOE/trwds30h.out
07/23/2004	03:06p	6,119	DOE/trwds60a
07/23/2004	03:06p	460,895	DOE/trwds60a.out
07/23/2004	03:06p	6,127	DOE/trwds60c
07/23/2004	03:06p	464,976	DOE/trwds60c.out
07/23/2004	03:06p	6,125	DOE/trwds60h
07/23/2004	03:06p	461,421	DOE/trwds60h.out
08/13/2004	04:05p	2,972	PinCell/bwr085
08/13/2004	04:05p	279,360	PinCell/bwr085.out
08/13/2004	04:05p	2,972	PinCell/bwr090
08/13/2004	04:05p	279,566	PinCell/bwr090.out
08/13/2004	04:05p	2,972	PinCell/bwr095
08/13/2004	04:05p	279,360	PinCell/bwr095.out
08/13/2004	04:05p	2,972	PinCell/bwr0975
08/13/2004	04:05p	279,360	PinCell/bwr0975.out
08/13/2004	04:05p	2,972	PinCell/bwr100
08/13/2004	04:05p	279,566	PinCell/bwr100.out
08/13/2004	04:05p	2,976	PinCell/bwr8x85
08/13/2004	04:05p	279,360	PinCell/bwr8x85.out



SKETCH - LAYOUT FOR 1000 MTHM