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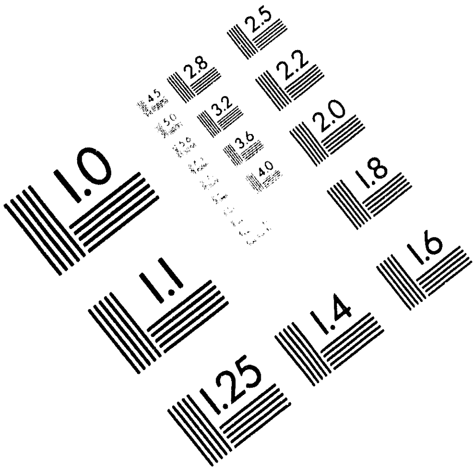

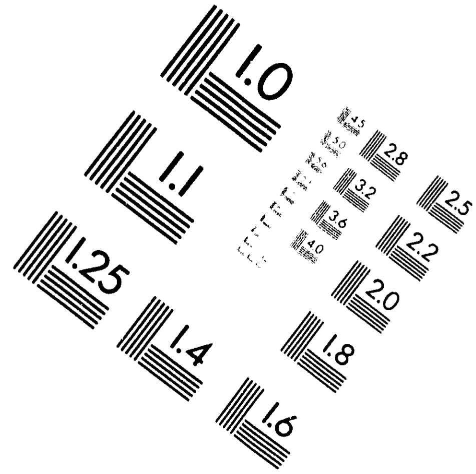

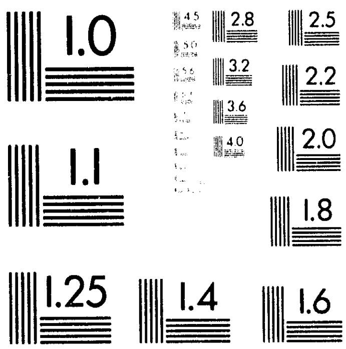

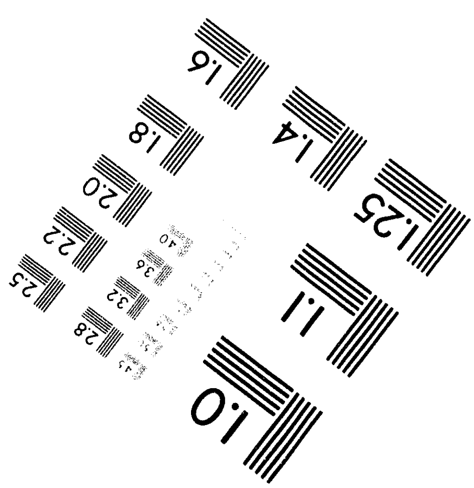

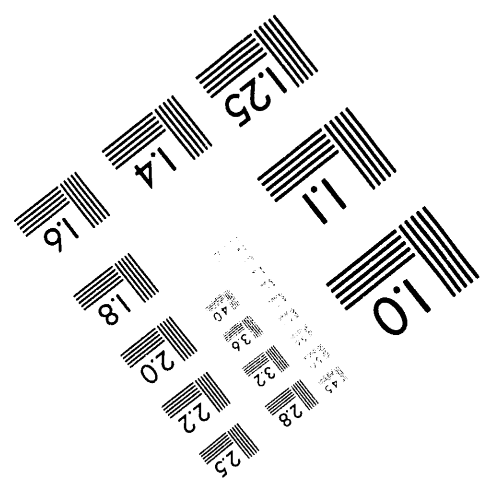

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EGG-WM-11011

Feasibility Study for a DOE Research and Production Fuel Multipurpose Canister

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Published February 1994

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MASTER

Prepared for the
U.S. Department of Energy
Assistant Secretary for Environmental Restoration and Waste Management
Under DOE Idaho Operations Office
Contract DE-AC07-76ID01570

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ABSTRACT

This is a report of the feasibility of multipurpose canisters for transporting, storing, and disposing of Department of Energy research and production spent nuclear fuel. Six representative Department of Energy fuel assemblies were selected, and preconceptual canister designs were developed to accommodate these assemblies. The study considered physical interface, structural adequacy, criticality safety, shielding capability, thermal performance of the canisters, and fuel storage site infrastructure. The external envelope of the canisters was designed to fit within the overpack casks for commercial canisters being developed for the Department of Energy Office of Civilian Radioactive Waste Management. The budgetary cost of canisters to handle all fuel considered is estimated at $170.8M. One large conceptual boiling water reactor canister design, developed for the Office of Civilian Radioactive Waste Management, and two new canister designs can accommodate at least 85% of the volume of the Department of Energy fuel considered. Canister use minimizes public radiation exposure and is cost effective compared with bare fuel handling. Results suggest the need for additional study of issues affecting canister use and for conceptual design development of the three canisters.

EXECUTIVE SUMMARY

The Office of Civilian Radioactive Waste Management (OCRWM) of the U.S. Department of Energy (DOE) is responsible for the transportation, storage, and disposal of U.S. commercial yspent nuclear fuel (SNF). The OCRWM is heading the design of a multipurpose canister (MPC) system to manage the commercial SNF. This MPC system improves safety and reduces the cost of SNF management when compared to handling individual commercial fuel assemblies for each phase. This MPC program involves placing up to 40 individual commercial fuel assemblies in large, sealed, inerted (helium gas backfilled) MPCs so that individual nuclear fuel assemblies can be handled only once, instead of six or seven times as in current methods of fuel transportation, storage, and disposal. The term *multipurpose* results from using the same canister for all three operation phases: transportation, storage, and disposal. It is assumed in the OCRWM MPC program that the robust MPC design requirements will be stringent enough to meet future Waste Acceptance Criteria (WAC) for the future Mined Geologic Disposal System (MGDS) whenever it is promulgated. The conceptual OCRWM MPCs have been designed to meet 10 CFR 71 transportation requirements and will be transported in Department of Transportation (DOT) Type-B transportation casks.

The same cost and safety advantages that the OCRWM MPC affords commercial fuel (as compared to individual fuel-assembly handling) should also be achievable by use of an MPC system for noncommercial, DOE research and production SNF (including foreign fuel to be reclaimed by the U.S.). This limited study was therefore undertaken to determine the feasibility of using an MPC system for DOE fuel.

Six DOE fuel-assembly types, which make up at least 90% of DOE research and production SNF including foreign fuel by volume, were selected for this study [DOE fuel at the Savannah River Site (SRS), which may still be reprocessed, was excluded from consideration]. These six DOE fuel-assembly types are N Reactor Mark IV assemblies, N Reactor Mark 1A assemblies, Advanced Test Reactor (ATR) assemblies, Three Mile Island Unit 2 (TMI-2) canisters, Fort St. Vrain (FSV) graphite-block assemblies, and Materials Test Reactor-type (MTR-type) assemblies from foreign reactors. Using actual DOE fuel-parameter values (and assumed values where actual values were not obtained), this study found no impediments to preclude using the MPC system for DOE fuel storage, transportation, and disposal of all six fuel types, although open issues have been identified. The study concludes that the six fuel types can be transported, stored, and disposed of by two new preconceptual MPC designs and by one existing conceptual OCRWM MPC design.

The OCRWM assumption that the conceptual commercial-fuel OCRWM MPC design will meet the eventual MGDS WAC is also assumed for the DOE-fuel MPC, since it will be designed to meet the same criteria. Some of the DOE fuel considered in this study is clad with aluminum. Although maximum fuel storage temperatures have been established for zircaloy-clad fuels, similar storage and disposal temperatures have yet to be established for aluminum-clad fuels. Reference 1 proposes and assumes that the temperature limits for zircaloy-clad fuel (380°C @ 5 years cooling time and 340°C @ 10 years cooling time) can be applied to the ATR and MTR-type aluminum-clad fuels if the aluminum-clad fuel assemblies are sealed and inerted in stainless steel (SS) cans before being placed in the MPC. Unlike the commercial fuel, ATR and

MTR-type fuels contain highly enriched uranium (HEU). The MGDS WAC will probably require that the MPCs be analyzed for criticality effects of severe MPC degradation during long-term disposal.

In this limited feasibility study, the three DOE-fuel MPCs mentioned above were constrained to (a) interface with all conceptual OCRWM MPC overpack cask designs and (b) meet the same configuration, criticality, shielding, structural, and thermal requirements as the conceptual OCRWM MPCs. Commercial conceptual overpack casks have been designed for transportation, storage, and disposal.

All six DOE fuel assembly types can be accommodated by using either the large commercial BWR MPC, the new FSV/TMI-2 MPC, or the new N Reactor MPC. An amendment to the future OCRWM MPC license will be required to license these MPC designs.

One aspect of using MPCs for DOE fuel that is different from using MPCs for commercial fuel is the DOE fuel-storage site infrastructure and 125-ton[a] MPC handling and loading capabilities. Waste management planning for the DOE fuel is ongoing. As a result of the DOE National Program Interim Storage Plan, the storage location and condition of much of this fuel may change in the near future. Therefore, if the MGDS WAC has to be in place prior to the design and implementation of the MPC system, the current DOE fuel storage locations and facility infrastructure could greatly change. The Idaho National Engineering Laboratory (INEL) facilities that store the TMI-2 canisters and the ATR cut fuel assemblies can handle and transport the 125-ton MPCs without upgrades. However, a 150-ton trailer may have to be leased or purchased to transport the 125-ton MPCs at the INEL. The FSV fuel blocks are stored at the INEL's Idaho Chemical Processing Plant (ICPP) Irradiated Fuels Storage Facility (IFSF). To handle the FSV/TMI-2 125-ton MPC, the IFSF Graphite Storage Facility (GSF) crane needs to be upgraded from 60 to 101 tons, and the handling cave transfer car needs to be modified and upgraded from 100 to 101 tons. An alternative is available, but probably would result in higher overall costs. The N Reactor fuel is stored in the two Hanford Area 100, 105-K basins. Each basin is served by its own 30-ton crane. These two cranes must be replaced with 114-ton cranes to handle the N Reactor 125-ton MPC. The MTR-type foreign fuel, when received, will probably be stored at the SRS receiving basin for offsite fuels (RBOF). The MTR-type 125-ton MPC can be shipped directly to the RBOF by rail car. The RBOF contains two 50-ton cranes, which can be combined to give a 100-ton rating. However, these cranes do not meet current regulatory standards and may need to be upgraded for lifting the MTR-type 125-ton MPC, which weighs 93 tons.

A total of 420 MPCs are required to accept all of the DOE fuel assemblies studied in this report. Sixty of these MPCs are the large commercial BWR MPC design, 113 are the new FSV/TMI-2 MPCs (the hexagonal-grid basket MPC design), and 247 are the N Reactor MPCs (new rectangular-grid basket MPC design). The budgetary equipment costs of the three MPC

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a. The term *125-ton MPC* refers to a package consisting of an MPC, its spent fuel payload, MPC shield plug, transportation overpack-cask body and lifting yoke, and water in the MPC when lifted from a pool. The 125-ton designation is the nominal-crane-hook capacity required to lift the 125-ton MPC package. The actual required crane-hook capacity depends on MPC design and fuel weight.

designs are $432K for the large commercial BWR MPC, $327K for the FSV/TMI-2 MPC, and $437K for the N Reactor MPC. The total budgetary equipment cost for the 420 DOE-fuel MPCs is $170.8M.

Although this study assumes that all six DOE candidate fuel-assembly types will eventually be accepted for disposal at the MGDS, there are unresolved disposal issues with some of the fuel. The N Reactor fuel resides in water-filled canisters with no water drain or drying provisions. The ATR and MTR-type fuels are clad with aluminum and, unlike zircaloy-clad fuels, are not yet approved for disposal at the MGDS due to the HEU content and the lack of aluminum-clad fuel allowable storage and disposal temperatures. For this reason, and the need to provide structural support for the inertial loads of the stacked fuel assemblies inside the MPC during transportation, this study recommends that the ATR and MTR-type fuels be placed in special SS confinement cans. FSV fuel and TMI-2 canisters also may be unacceptable for MGDS disposal. The TMI-2 canisters may continue radiolytic gas generation (from water entrapped in the licon) following drying. Therefore, TMI canisters may have the potential to overpressure the sealed MPCs. This study includes a thermal evaluation of the ATR and MTR-type fuels. Since aluminum-clad fuel storage and disposal temperature limits have not yet been established, these limits had to be assumed. When the actual limits are established, this analysis should be verified or revised.

CONTENTS

FIGURES

TABLES

ACRONYMS

ALARA	as low as reasonably achievable
ATR	Advanced Test Reactor
BWR	boiling water reactor
CFA	Central Facilities Area
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
FSV	Fort St. Vrain
GSF	Graphite Storage Facility
HEU	highly enriched uranium
IAEA	International Atomic Energy Agency
ICPP	Idaho Chemical Processing Plant
IFSF	Irradiated Fuel Storage Facility
INEL	Idaho National Engineering Laboratory
MGDS	Mined Geologic Disposal System
MPC	multipurpose canister
MPU	multipurpose unit
MRS	monitored retrievable storage
MTR	Materials Test Reactor
MTU	metric ton of uranium
MTHM	metric ton of heavy metal
MW·d	megawatt-day
NEPA	National Environmental Policy Act

NRC	U.S. Nuclear Regulatory Commission
NuPac	Nuclear Packaging, Inc.
OCRWM	Office of Civilian Radioactive Waste Management
ORNL	Oak Ridge National Laboratory
PWR	pressurized water reactor
RBOF	receiving basin for offsite fuel
SRS	Savannah River Site
SS	stainless steel
TAN	Test Area North
TMI-2	Three Mile Island Unit 2
TRA	Test Reactor Area
WAC	Waste Acceptance Criteria

Feasibility Study for a DOE Research and Production Fuel Multipurpose Canister

1. INTRODUCTION

The U.S. Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) is developing a multipurpose canister (MPC) system for receiving, storing, and disposing of commercial reactor spent nuclear fuel (SNF). Figure 1 is a sectional view of a large conceptual OCRWM MPC (throughout the remainder of this report, the conceptual OCRWM MPCs are referred to as OCRWM MPCs). The large OCRWM MPC consists of a cylindrical shell, bottom plate, and top closure components, is 193.0 in. high, and has an outside diameter (OD) of 60.3 in. The internal cavity contains a "basket," which is designed to accept commercial nuclear fuel assemblies. The basket can incorporate neutron absorbing material to reduce criticality coefficients. A more detailed description of the MPC is found in Reference 1.

The commercial fuel assemblies will be loaded into the MPC, which will then be seal welded and backfilled with helium. The fuel will remain in the MPC through transportation, storage in a monitored retrievable storage (MRS) facility, and disposal in a Mined Geologic Disposal System (MGDS). The MPC shown in Figure 1 provides little shielding and is therefore placed in shielded overpack casks for each handling phase: a transportation overpack cask for shipping, a storage overpack cask or concrete module for storage, and a disposal overpack cask for placement in the MGDS. The advantage of the MPC system is that individual spent fuel assemblies can be handled only once, and subsequently, only the sealed MPCs will be handled. This greatly reduces the amount of individual fuel-assembly handling and decreases potential spread of contamination. Figure 2 shows a conceptual exploded view of the MPC and its transportation overpack cask.

Reference 1 is a preliminary evaluation report on the OCRWM MPC. It includes studies of (a) the large MPC that holds 21 PWR or 40 BWR assemblies and can be used by commercial reactor pools that can handle 125-ton loads and ship 180 tons by rail and (b) a small MPC (which holds 2 PWR or 4 BWR assemblies and has a legal-weight truckload of 25 tons within its shipping cask) for reactor pools that cannot handle large loads or must ship by truck since the sites do not have rail access. Reference 1 showed that the cost and radiation exposure to workers and public from using the small MPCs is high compared to the large MPCs. Reference 3, which is the OCRWM MPC conceptual design summary report generated by the DOE management and operations (M&O) contractor, does not consider using the small MPCs for this reason. Reference 3 has included a new medium-sized rail MPC that holds 12 PWR or 24 BWR assemblies and has a 75-ton under-the-hook load with its shipping cask. This is primarily for use at reactor pools that cannot handle the large 125-ton MPC or to ensure criticality safety if burnup credit cannot be taken for the PWR fuel.

References 1 and 3 also address potential use of a multipurpose unit (MPU) for commercial fuel. The MPU is an integral combination of an MPC and permanent external overpack cask. Since Reference 1 indicates that use of the MPU is less economical and no safer than the

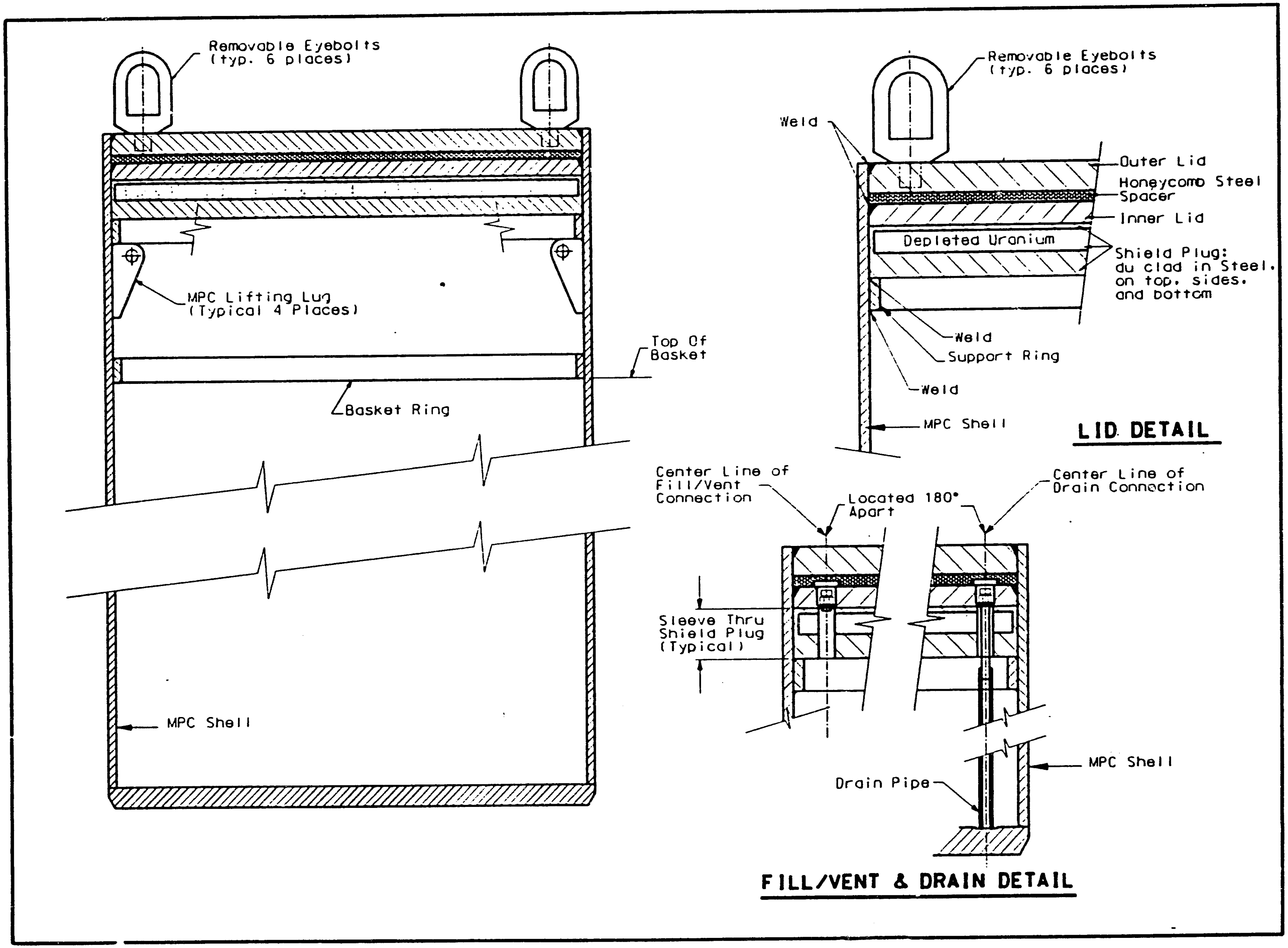

Figure 1. OCRWM MPC shell design.

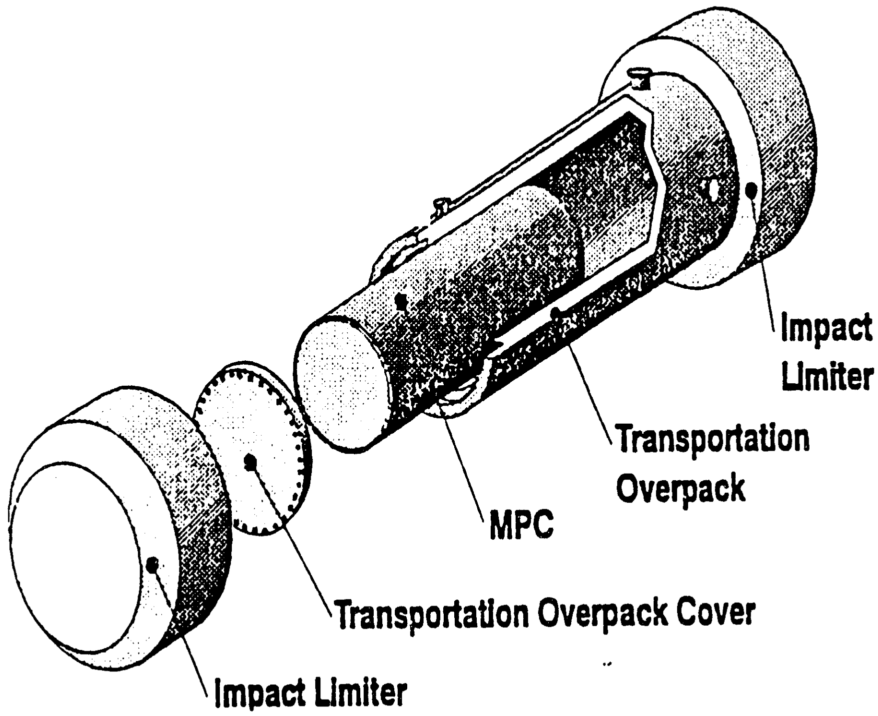

Figure 2. MPC and transportation overpack-cask system.

MPC/overpack-cask system, MPUs were not considered herein for use with DOE fuel.

Reference 1 indicates that the large OCRWM MPCs should be used exclusively, if possible. This is because they offer the minimum radiation exposure to the public and workers (by reducing individual fuel assembly handling) and offer the minimum life-cycle cost, when compared to all other MPCs studied.

This limited report only considered using the large 125-ton MPC design for DOE fuel. In future studies, it is recommended that the small and intermediate MPC sizes also be considered for use with DOE fuel, where difficult handling, criticality, thermal, or shielding problems arise.

According to Reference 1, the MGDS will include a facility for dry-cask fuel transfer, a crane with a minimum hook capacity of 125 tons, and provisions to handle, load, and seal the large OCRWM MPCs.

The large MPC, together with its fuel payload, shield plug, transportation overpack-cask body and lifting yoke, and water load as lifted from a pool, has a nominal under-the-crane-hook weight of 125 tons. Although the actual under-the-hook weight varies depending on the basket design and fuel weight and is somewhat below 125 tons, the nominal designation of 125 tons is still used. The large OCRWM MPC design takes credit for fissile material burnup in the criticality calculations and can accommodate 21 commercial pressurized water reactor (PWR) or 40 commercial boiling water reactor (BWR) fuel assemblies depending on the basket design. Figure 3 is a cross-section of the BWR 40-cell square-grid OCRWM MPC basket.

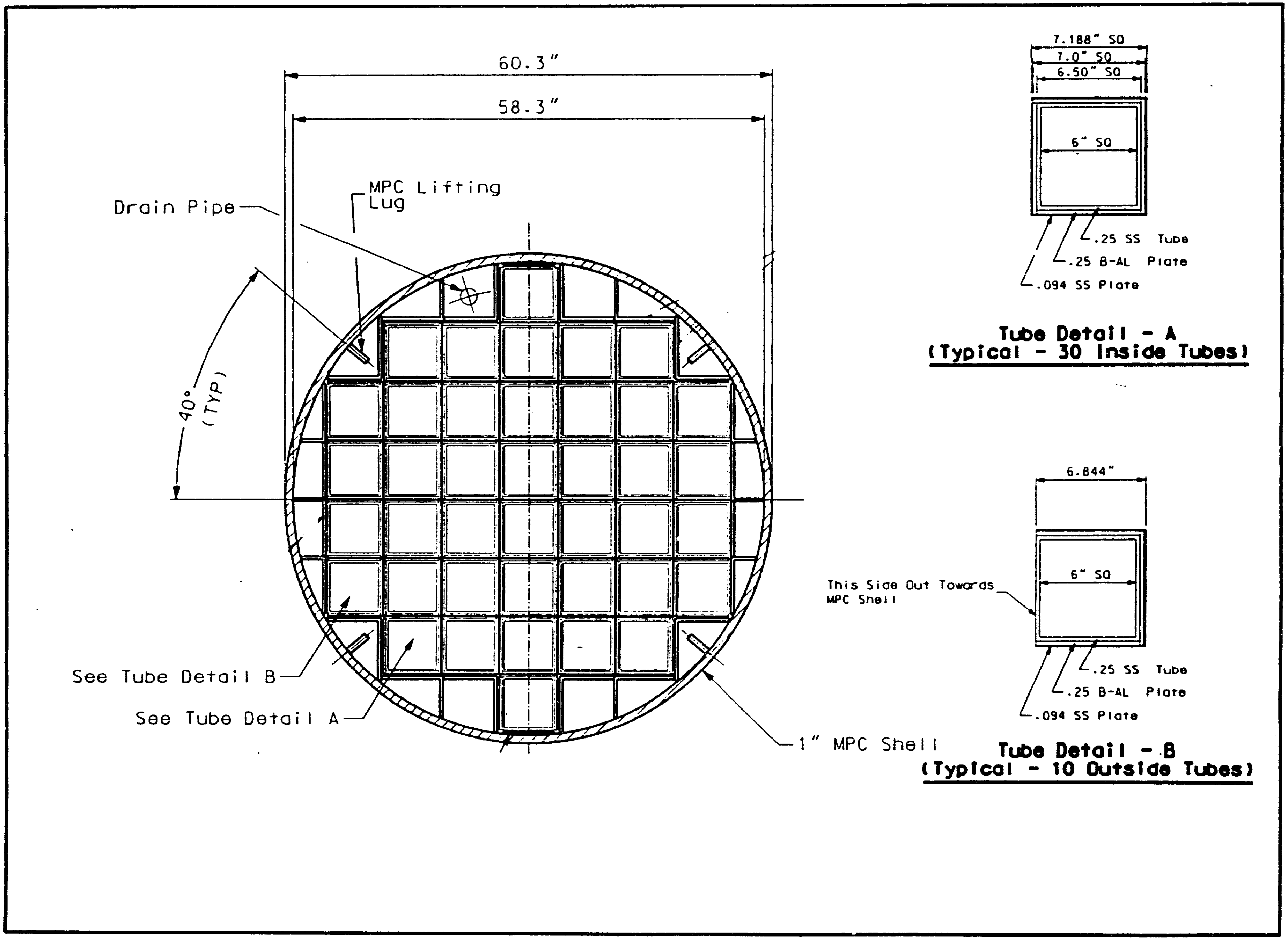

Figure 3. Commercial BWR MPC basket design.

2. PURPOSE

This report studies the feasibility of using the large OCRWM MPC, or modified versions, for transporting, storing, and disposing of DOE and foreign spent nuclear fuel. The reduced radiation exposure and economic advantages of the large OCRWM MPC, which are indicated in Reference 1, should also be achievable for DOE fuel.

Reference 1 addresses all aspects of OCRWM MPC system operations from fuel loading at reactor spent fuel pools to final placement of the MPC in the MGDS. The study contained herein, however, is based on MPCs designed for specific DOE fuel. Once loaded and placed in the transportation overpack cask, this DOE-fuel MPC package must be indistinguishable from the 125-ton OCRWM MPCs with regard to subsequent transportation and final disposal in the MGDS. Therefore, only the MPC loading and handling operations at the DOE sites are studied, and all other aspects of the DOE-fuel MPC operations are assumed to be identical to corresponding operations for the OCRWM MPC. Therefore, details of these identical operations are not repeated here.

3. METHOD OF FEASIBILITY EVALUATION

Six representative DOE fuel types (which constitute at least 90% of the applicable DOE fuel by volume) were selected, and preconceptual DOE-fuel MPC designs were developed to accommodate these assemblies. The external envelope of the DOE-fuel MPC was designed to fit within the same transport, storage, and disposal overpack casks used for the OCRWM MPCs. The total number of MPCs required to accommodate each of the six fuel assembly types was determined, and budgetary costs for these DOE-fuel MPCs were estimated. Issues that may impede using MPCs for these six DOE fuel types were also identified. The following is a summary of the methods used to evaluate the MPC for DOE fuel:

- Six DOE fuel types that represent the majority of the DOE fuel volume and geometries were selected, and their parameters were determined.

- The feasibility of using a large OCRWM MPC (or modified large OCRWM MPC) for disposing of the above DOE fuels was investigated. In determining feasibility, the following factors were considered: physical interface, structural adequacy, criticality safety, shielding adequacy, and thermal performance.

- The DOE 125-ton MPC handling, loading, and transporting scenarios at the DOE locations that store the six fuel types were investigated. Handling deficiencies, required facility modifications, and alternative handling or loading methods were determined.

- Total quantity and budgetary cost of the DOE-fuel MPCs required to dispose of all six fuel types studied were estimated.

- Unresolved issues, problems, and disadvantages of using MPCs for these six DOE fuel types were identified.

4. DOE SPENT NUCLEAR FUEL

The OCRWM MPCs are designed to interface with two basic types of commercial SNF assemblies: PWRs and BWRs. DOE fuel, on the other hand, comprises numerous types and sizes of assemblies with varying fuel forms and enrichment. DOE fuel is generally categorized as special research reactor and other miscellaneous fuel, production reactor fuel, and naval reactor SNF. Other DOE fuels include those from U.S. research reactors not at DOE sites, and U.S.-origin fuel returned from foreign research reactors. Some SNF generated at civilian power plants has been transferred to DOE ownership, generally as the result of cooperative research and development activities conducted at DOE laboratories. Most of the DOE fuel can be placed in the following basic categories:

- Production fuel: N Reactor, Savannah River Site (SRS) K Reactor, etc.
- Graphite fuel: Fort St. Vrain (FSV), Peach Bottom, etc.
- Aluminum plate fuel: Advanced Test Reactor (ATR) type
- Miscellaneous commercial fuel
- Three Mile Island Unit 2 (TMI-2) core debris
- U.S.-supplied foreign fuel [Primarily Materials Test Reactor (MTR) type].

Table 1 shows the six types of DOE fuel selected for this study. The six types were selected because of their large volume, and diversity of nuclear material, and geometry. The Mark 1A and

Table 1. Major DOE fuel assembly types and quantities.

Type of fuel assembly	Quantity of fuel assembly
N Reactor fuel (Mark IV)	65,000
N Reactor fuel (Mark 1A)	38,640
TMI-2 core canisters	342
Fort St. Vrain fuel blocks[a]	2,208
ATR fuel	759
Foreign MTR fuel	5,600

a. This is the total quantity of FSV graphite fuel block assemblies. Approximately 726 assemblies are now in storage at the INEL, and 1,482 assemblies are currently held by Public Service Company of Colorado.

Mark IV N Reactor production fuels were selected because they constitute at least 90% of the DOE-fuel volume currently at DOE facilities [not including the fuel at the Savannah River Site (SRS) currently being considered for reprocessing]. The Advanced Test Reactor (ATR) and Materials Test Reactor-type (MTR-type) foreign fuel assemblies were selected because they represent a large volume of fuel and may be difficult to transport and dispose of. This is because they contain highly enriched uranium (HEU) and aluminum cladding for which peak storage and disposal temperatures have not yet been established. FSV fuel and TMI-2 core debris storage canisters were selected because of their diverse geometry, unique fuel composition, and large volume. Over 95% of the total volume of DOE fuel currently at U.S. DOE facilities [with the exception of that which may be reprocessed at the Savannah River Site (SRS)] is of these six fuel types.

The foreign fuel, which consists of approximately 5,600 MTR-type HEU assemblies containing fuel of U.S. origin, will be returned from foreign research reactors. When received by DOE, these foreign fuel assemblies will probably be stored in the receiving basin for offsite fuels (RBOF) at the SRS.

Pertinent data for these six assemblies are given in Table 2. These data were gathered from existing documents and from verbal input from DOE site personnel knowledgeable of the specific fuel assemblies. These data, along with design information for existing casks used to transport these fuel assemblies, were used to perform the DOE-fuel MPC feasibility studies. The six DOE fuel assemblies are described in more detail below.

DOE fuel also includes a quantity of Naval SNF from Navy vessel reactors. This fuel was not considered in this report due to its sensitive security classification.

4.1 N Reactor Fuel

A typical N Reactor fuel assembly is shown in Figure 4. N Reactor fuel assemblies consist of two concentric annular uranium metal tubes clad with zircaloy on the inside and outside. Tube ends are sealed, the OD of the assemblies is 2.4 in., and assembly lengths vary from 15.0 to 26.1 in. Each fresh fuel assembly contains a maximum of 0.22 kg of fissile material. There are approximately 103,640 N Reactor assemblies having a total weight of approximately 2,096 metric tons of heavy metal (MTHM). The N Reactor fuel is currently stored in over 7,000 16-gage double-tube SS canisters. Each canister holds 14 assemblies.

4.2 TMI-2 Debris Canisters

A typical TMI-2 canister is shown in Figure 5. TMI-2 canisters are basically 14.0-in. outside diameter (OD) x 149.75-in. long SS cylinders. They contain fuel and core debris, miscellaneous decontamination waste, and filters. There are 342 canisters having a total of 81.6 MTHM. The enrichment of the original TMI-2 fuel was approximately 3%.

Table 2. Major DOE fuel assembly types and parameters.

Physical Parameters	N-Reactor Mark IV 65,000 Assemblies	N R 38,6
Total weight (metric tons)	1,477	634
Total weight of heavy elements (U+Th+Pu) (metric tons)	1468.5	627.
Maximum assembly unit weight (kg)	23.6 (fresh fuel wt)	16.8 (fres
Maximum assembly length (in.) (include control components, as applicable)	17.4 to 26.1	15 to
Maximum fuel assembly width or equivalent assembly cross-section data; alternatively, minimum fuel cell clearance (in.)	2.4 O.D.	2.4
Active fuel length (in.)	14.5 to 26.7	14.6
Distance of active fuel from bottom of fuel assembly support base (in.)	0.220	0.21
Fuel pellet diameter (in.) (or equivalent fuel dimension)	Not applicable	Not
Fuel clad thickness (in.) (or equivalent fuel dimension)	0.015 to 0.025[a], 0.020 to 0.040	
Fuel-pin pitch (in.) (center-line to center-line pin or plate spacing)	Not applicable	Not
Number and location of nonfuel tubes and holes within fuel assembly array	Not applicable	Not
Fuel cladding material	Zircaloy-2	
Initial fuel composition	Metallic uranium 0.947% ^{235}U enrichment	Met 1.15 enri
Maximum initial fissile material content per assembly[c] (kg)	0.216	0.18
Initial fuel density (gm/cm³) (or fuel mass loading range of fuel assembly)	Not available	Not

	Fuel Description			
actor Mark 1A 0 Assemblies	TMI-2 Core Debris 342 canisters	Ft. Saint Vrain 2,208 Blocks	ATR Plate Fuel 1,534 Assemblies[h]	Foreign: MTR, 5,600 Assemblies
	342.1 gross wt 139 net wt	110.0	15.3 Max.	41.4
	81.6	8.6	2.5	1.2 (Approx.)
fuel wt)	1,330 gross wt 785 net wt	165	9.988	7.4
20.9	149.75	32.7	51	48.63
O.D.	14.0 O.D., with 9 x 9 in. sq. inside cavity	Hexagonal, 14.2 Across Flats	4.34	2.8 x 3.17
to 20.5	less than 138	30.2	48	23.5
	6	0.5	0.75	12.5
pplicable	Not applicable	0.5	Not applicable	Not applicable
	Not applicable	Not applicable	0.015	0.015
pplicable	Not applicable	See drawing	TBD (plate pitch)	0.2 (plate pitch)
pplicable	Not applicable	114 (see drawing)	Not applicable	Not applicable
	originally zirc.	Not applicable	6061 Alum.	1100 Alum.
llic uranium % ^{235}U avg. hment	Originally commercial B&W fuel (15 x 15)	93% ^{235}U enriched	UAL_x HEU, 93% ^{235}U enriched	UAL_x (HEU metal and 1100 Al)
	Not available	1.26	1.075	0.13 to 0.73
vailable	Originally commercial B&W fuel (15 x 15)	Particle-fuel packets	6.8	TBD

Table 2. (continued).

Fuel Cycle Parameters	N-Reactor Mark IV 65,000 Assemblies
Reference burnup maximum source terms[c,d]	45,300 MWd/MTHM (average)
Reference burnup minimum criticality[c,e]	0
Maximum sustained power level (MW)	4,000
Minimum cooling time since discharge (years)[c]	6
Reference gamma source term (mev/sec/fuel assembly and spectrum)[f]	1.632E+14
Reference neutron source term (n/sec/fuel assembly and spectrum)[f]	2.689E+04
Reference decay heat (watts/fuel assembly)[f]	1.764

a. Inner annular fuel component.
b. Outer annular fuel component.
c. A statistical data base or tabular report that further describes the range and distribution of fuel assem
is preferred for assessing the degree of fuel compatibility/acceptance achievable within the conceptua
d. Specify design maximum burnup corresponding to thermal and radiological source terms and cooling
e. Design minimum burnup goal for maximum fissile material content specified, i.e., burnup credit acc
f. Corresponds to specified maximum burnup and cooling time.
g. TMI fuel burnup was estimated at 3,176 MWd/MTHM at the time of the accident, which is approxi
normal spent fuel. The maximum weight (of fuel debris and/or other debris) within a canister is 785
rubble, the effective fuel density may be higher (by maybe two or three times) than that within the st
h. 759 are in storage now and 775 will be added by the year 2001 (75 elements will be added in FY-94
and 125 elements will be added each year for FY-95 through FY-99.
i. @ 150 days after removal from the ATR reactor.

Fuel Description

Reactor Mark 1A 8,640 Assemblies	TMI-2 Core Debris 342 Canisters	Ft. Saint Vrain 2,208 Blocks	ATR Plate Fuel 759 Assemblies[h]	Foreign: MTR, 5,600 Assemblies
,174 MWd/MTHM average)	(g)	26,000 MWd/MTIBM	2.3E+21 fissions/cm^3	65%
	(g)	6,000 MWd/MTIBM	1.8E+21 fissions/cm^3	0
,000	(g)	900	8/assembly	70
	14	10	0	0
.952E+14	(g)	1,500 to 2,000 Rem @ 3 ft	3.0+15 Avg.[i]	Not available
.912E+04	(g)	5.0E+05	Not applicable	Not available
.523	100	40.0	1,758[i]	Not available

ly characteristics
MPC.
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ptance.

ately 10% of the burnup for
kg. Since this may be
ucture of an intact fuel assembly.
and FY-00,

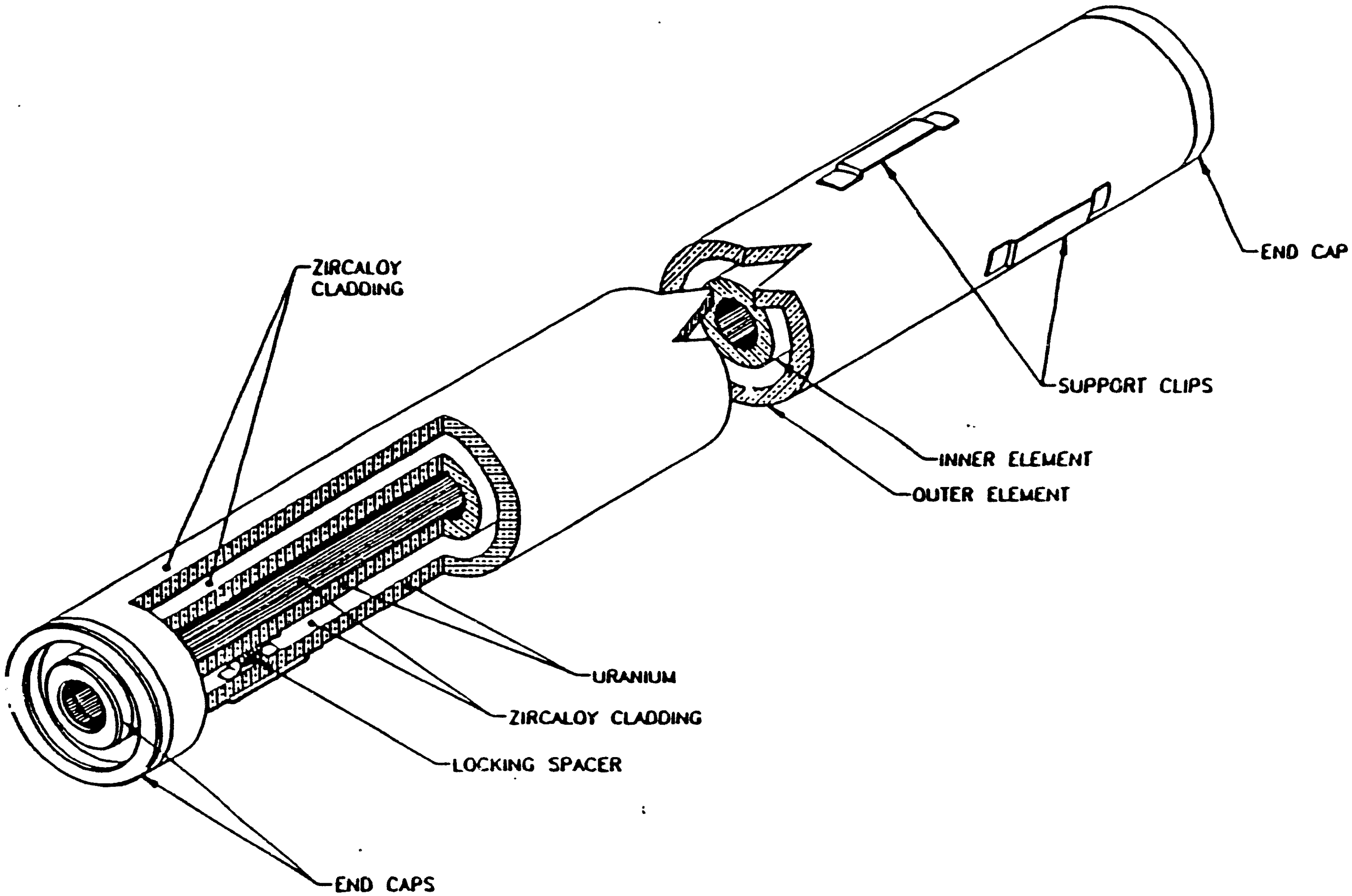


Figure 4. N Reactor Fuel Assembly.

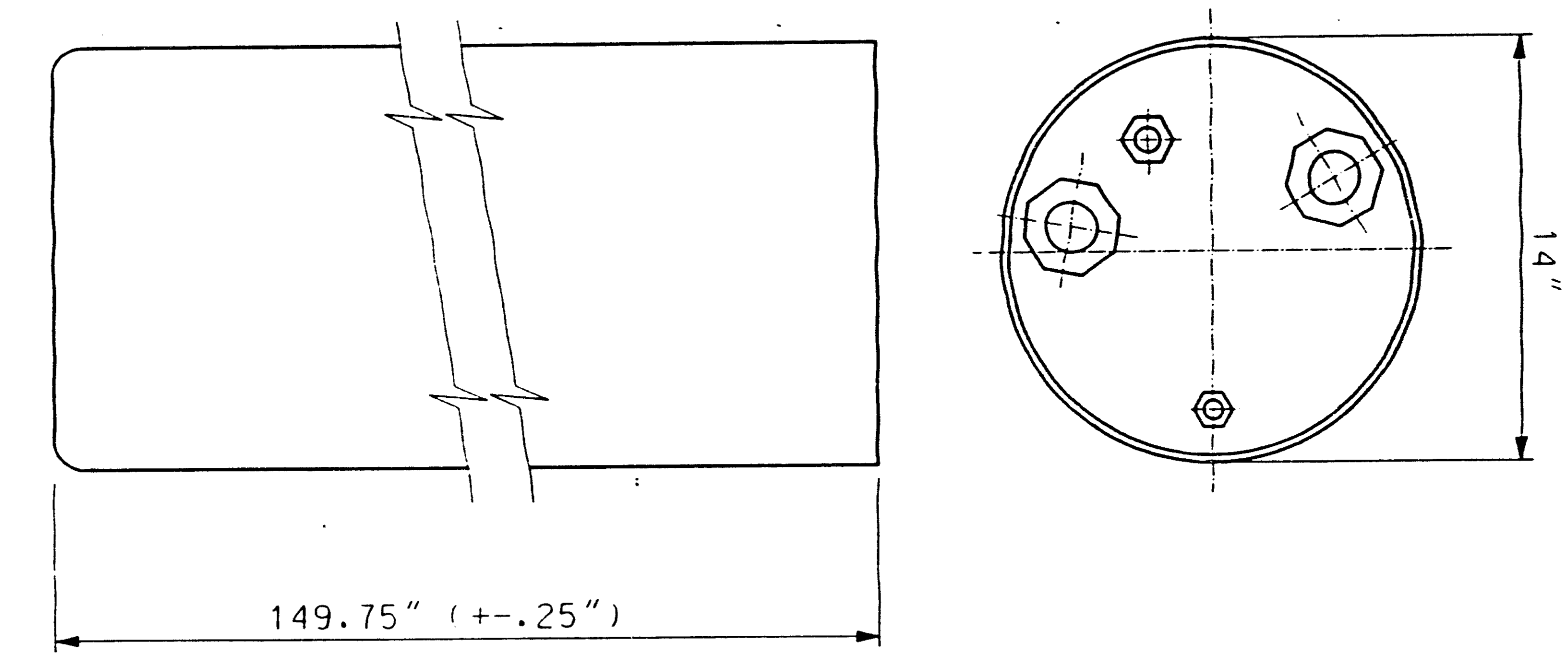


Figure 5. TMI-2 fuel canister envelope.

4.3 FSV Fuel

A typical FSV fuel assembly is shown in Figure 6. FSV fuel assemblies consist of hexagonal graphite blocks 14.17 in. across the flats by 32.1 in. long. The assemblies contain TRISO-coated uranium carbide fuel kernels within fuel packets placed in numerous 0.5-in. diameter longitudinal holes within the graphite blocks. There are 2,208 fuel assemblies having a total of 8.6 MTHM. The enrichment of the original FSV fuel was approximately 93%.

4.4 ATR Fuel

A typical ATR fuel assembly is shown in Figure 7. Each ATR fuel assembly consists of 19 curved aluminum fuel plates with highly enriched aluminum-clad uranium-aluminum fuel compound, separated by thin coolant passages. The fuel assemblies are 4.2 in. wide and 51 in. long in their storage condition. Each fresh fuel assembly has a fissile material content of up to 1.075 kg. There will be a total of 1,534 ATR spent fuel assemblies in storage by the year 2002. These assemblies will have a total of approximately 2.5 MTHM. The enrichment of the original ATR fuel was approximately 93%.

4.5 MTR-Type Foreign Fuel

A typical MTR-type fuel assembly is shown in Figure 8. MTR-type assemblies having fuel of U.S. origin will be returned from foreign research reactors. The assemblies are similar to the ATR fuel and are composed of curved fuel plates. They are 3.2 in. wide and 51 in. long, and each fresh fuel assembly contains up to 0.73 kg of fissile material. There may be approximately 5,600 MTR-type foreign fuel assemblies returned from foreign research reactors, having a total of approximately 1.2 MTHM. The enrichment of the original MTR-type fuel was approximately 93%.

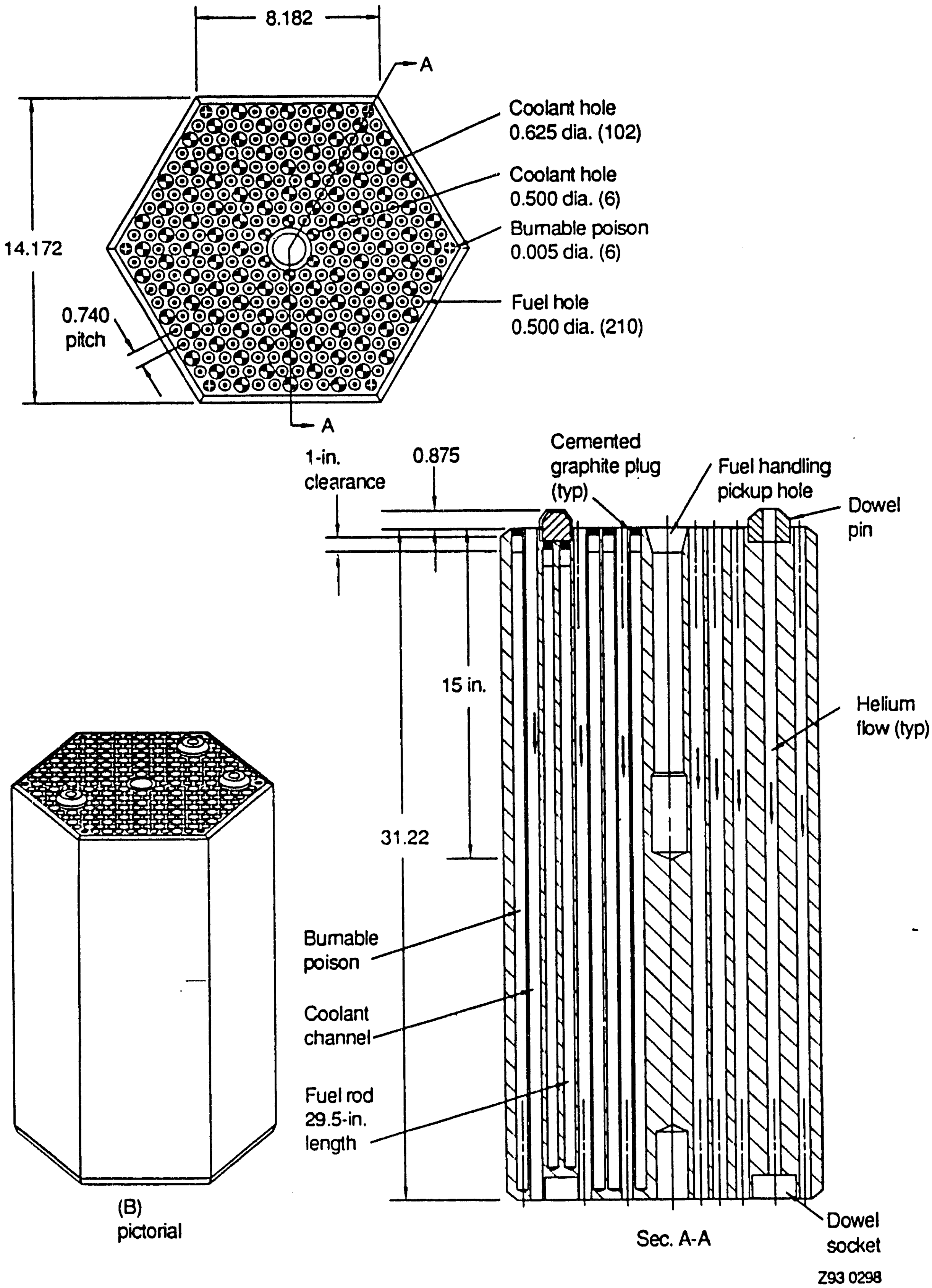


Figure 6. Typical FSV graphite fuel block.

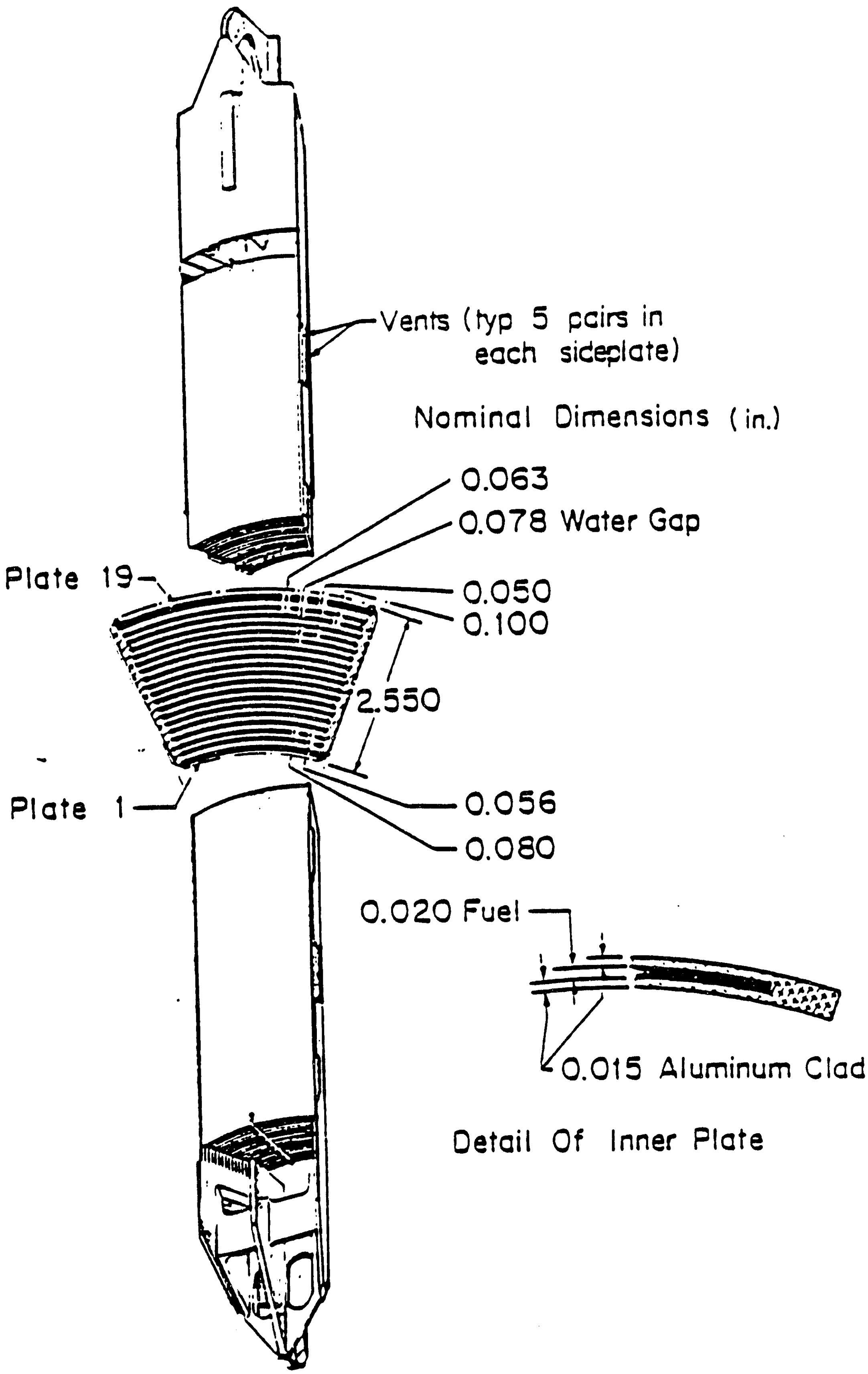


Figure 7. ATR fuel assembly.

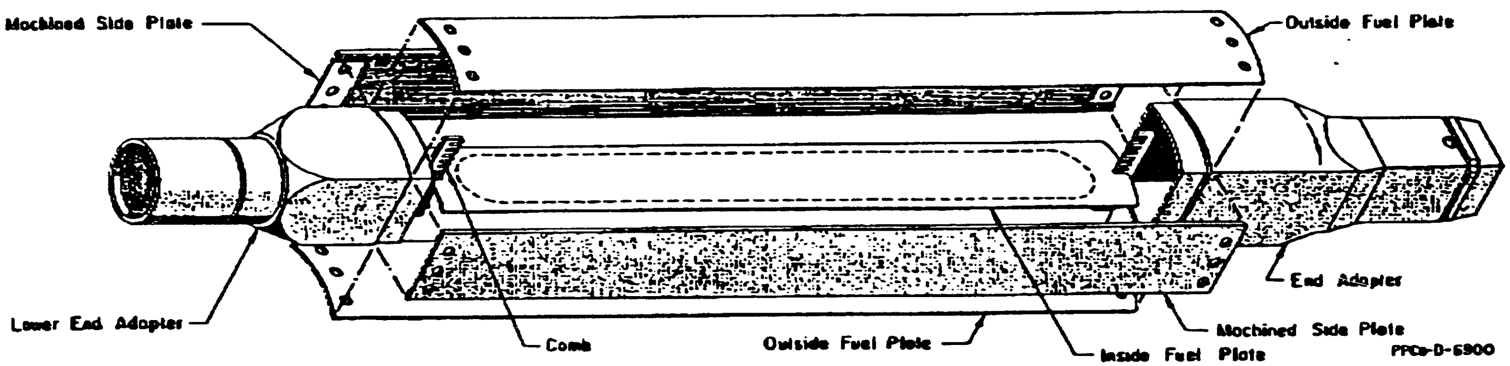


Figure 8. MTR fuel assembly.

5. DOE-FUEL MPC DESIGN FEASIBILITY EVALUATION SUMMARY

Reference 2 is the limited feasibility evaluation. which resulted in the two new preconceptual DOE-fuel MPC designs for handling the six selected DOE fuel assemblies. In addition to these two new designs, an existing conceptual OCRWM BWR MPC is used for a portion of the DOE fuel. These three designs were based initially on the physical arrangement of fuel in the OCRWM MPC. Table 3 is a comparison of the main parameters of these three DOE-fuel MPCs to those of the large commercial BWR MPC. Reference 2 was performed by the contractor assigned to the preconceptual and conceptual design of the OCRWM MPCs. The three DOE-fuel MPC designs that resulted from the Reference 2 study will collectively handle all six of the DOE fuel-assembly types listed in Table 2. This section is a summary of the Reference 2 results.

The fuel data in Table 2 were compiled to perform the feasibility study, which was performed in Reference 2. Where values in Table 2 are listed as Not Available, values were assumed as necessary. These assumptions are specifically listed in the following sections.

The feasibility study resulted in evaluations of the six DOE-fuel MPCs for structural, criticality, thermal, and shielding adequacy. The thermal and shielding evaluations were performed on a package consisting of the DOE-fuel MPC placed within the commercial OCRWM MPC transportation overpack cask (which would be used to transport the DOE-fuel MPCs). In some cases, actual analysis was done; while in others, known commercial-fuel parameter values were compared to the DOE fuel parameters, and estimates of adequacy, or potential problems were determined. During any future work on the DOE-fuel MPCs, further analysis in all four areas is recommended. The structural comparison of the commercial and DOE-fuel MPCs ensured that the latter will meet the necessary 10 CFR 71 transportation package requirements.

5.1 N Reactor MPC

5.1.1 Assumed Fuel Values

Since all required information was not available for the N Reactor fuel assemblies, the following assumptions were made to complete the feasibility study. The N Reactor fuel thermal source term was estimated, based on commercial PWR fuel cycle information, to be 67 W per assembly at 5 years cooling time. More detailed analysis using ORIGEN computer code models for specific N Reactor fuel irradiation conditions is recommended prior to making final conclusions regarding MPC system adequacy for storing N Reactor fuel (i.e., heat transfer characteristics and fuel acceptance limitations). A comparison of expected N Reactor fuel source terms to commercial PWR fuel source terms can be made based on the N Reactor initial enrichment and burnup information from Table 2. The radiological assessment assumes that N Reactor fuel with 45,000 MWd/MTIHM burnup will exhibit radiological source terms through time compared to commercial PWR fuel. Due to the substantially lower enrichment and lower neutron energy spectrum condition present within a heavy-water reactor during irradiation, it is anticipated that PWR radiological source terms corresponding to 45,000 MWd/MTU burnup would significantly over estimate the actual N Reactor source term. It is assumed that the maximum peak fuel clad temperature limit for the N Reactor fuel assemblies is 340°C for

Table 3. Comparison of large OCRWM MPC and DOE-fuel MPC parameters.

Fuel assembly type	Length (in.)	Diameter/ thickness (in./in.)	Basket type	125-ton MPC crane hook weight[d] (tons)
Commercial BWR	193.0	60.30/1.0	Commercial BWR	109[e]
DOE ATR and MTR[a]	193.0	60.30/1.0	Commercial BWR	93[e]
DOE FSV	170.38[b]	60.30/1.5	New hexagonal-grid	101[f]
DOE TMI-2 canisters	170.38[b,c]	60.30/1.5	New hexagonal-grid	105[f]
DOE N Reactor Mark IV or 1A	173.75[b]	60.30/1.5	New rect.-grid	114

a. This MPC is identical to the large commercial BWR MPC.

b. This MPC requires length spacers when placed in the overpack casks.

c. The TMI-2 MPC is identical to DOE FSV MPC.

d. This weight is the under-the-hook crane weight and consists of the MPC loaded with the respective type and quantity of fuel, MPC shield plug, transportation overpack-cask body and its lifting yoke, and water in the MPC when lifted from a pool.

e. The difference in total weight is caused by the difference in weight between the commercial BWR fuel and the ATR or MTR fuel.

f. The difference in total weight is caused by the difference in weight between the FSV graphite fuel block assemblies and the TMI-2 canisters.

10-year-old fuel and 380°C for 5-year-old fuel. It was assumed that the thin-wall N Reactor fuel canisters (16 gauge) will not withstand the hypothetical end-drop accident required for transportation by 10 CFR 71, if they are stacked upon each other without individual supports.

5.1.2 N Reactor MPC Design

A new MPC design is required for the N Reactor fuel canisters, since they will not fit within an existing OCRWM MPC basket. The new N Reactor MPC shell design is shown in Figure 9. The only difference between it and the large OCRWM MPC shell is that the N Reactor shell is 170.38 in. high and 1.5 in. thick. All N Reactor fuel is currently stored in water-filled storage canisters, each holding 14 fuel assemblies. A typical N Reactor storage canister is shown in Figure 10 and comprises two 8-in. diameter SS cylinders (each of which holds seven assemblies) joined with cross members. There are 3,815 N Reactor canisters that have sealed lids and are filled with water except for a 2.5 in. nitrogen gas blanket at the top. There are also 3,666 canisters that do not have lids. They are open, water filled, and are scheduled to have lids installed in 1996.

The final fuel-confinement barrier is required to withstand immersion in water to a depth of 200 m, as required by the International Atomic Energy Agency (IAEA) criterion SS6 (IAEA SS6), which is invoked by DOT. The reference case PWR- and BWR-sealed MPC cylinder shell designs, which are 1.0-in thick, are not required to withstand immersion in 200 m of water because the fuel cladding is considered to be the final fuel confinement barrier. However, if failed fuel is placed in the MPC, the MPC is the final confinement barrier and it must withstand the 200-m immersion. Since the condition of the N Reactor fuel storage canisters is unknown, and since some of the N Reactor fuel assemblies are breached, the MPC should be considered to be the final fuel-confinement barrier. Therefore, the N Reactor MPC cylinder shell thickness was increased from the 1.0-in. OCRWM MPC thickness to 1.5 in. so it could withstand the immersion pressure.

Figure 11 shows the new 6-cell rectangular-grid MPC basket designed specifically for holding the N Reactor canisters. The canisters will be stacked five high in each of the six MPC cells, giving a total of 30 canisters within each MPC. Since each canister holds 14 N Reactor fuel assemblies, each MPC will hold a total of 420 assemblies. The N Reactor canisters may not be able to withstand the inertial transportation loads when stacked five high in the MPC. Therefore, a special "bucket" (shown in Figure 12) was designed to individually support each N Reactor canister within the MPC. After an N Reactor canister is placed in a bucket, the bucket and canister are lowered into one of the six MPC basket cells shown in Figure 11. Table 3 shows parameters of the N Reactor MPC design. More design details may be found in Reference 2.

5.1.3 Feasibility Results

5.1.3.1 Structural Evaluation. The structural condition of the N Reactor canister walls is unknown. Therefore, no structural credit was taken for the canister in this evaluation. The canister buckets will be stacked in the rectangular array shown in Figure 11. The bucket design allows uniform distribution of fuel inertial loads resulting from a 10 CFR 71 hypothetical 9-m side drop accident. The notched bucket design accommodates the canister lid locking mechanism. This notched opening allows the entire length of the canister wall to bear evenly against the

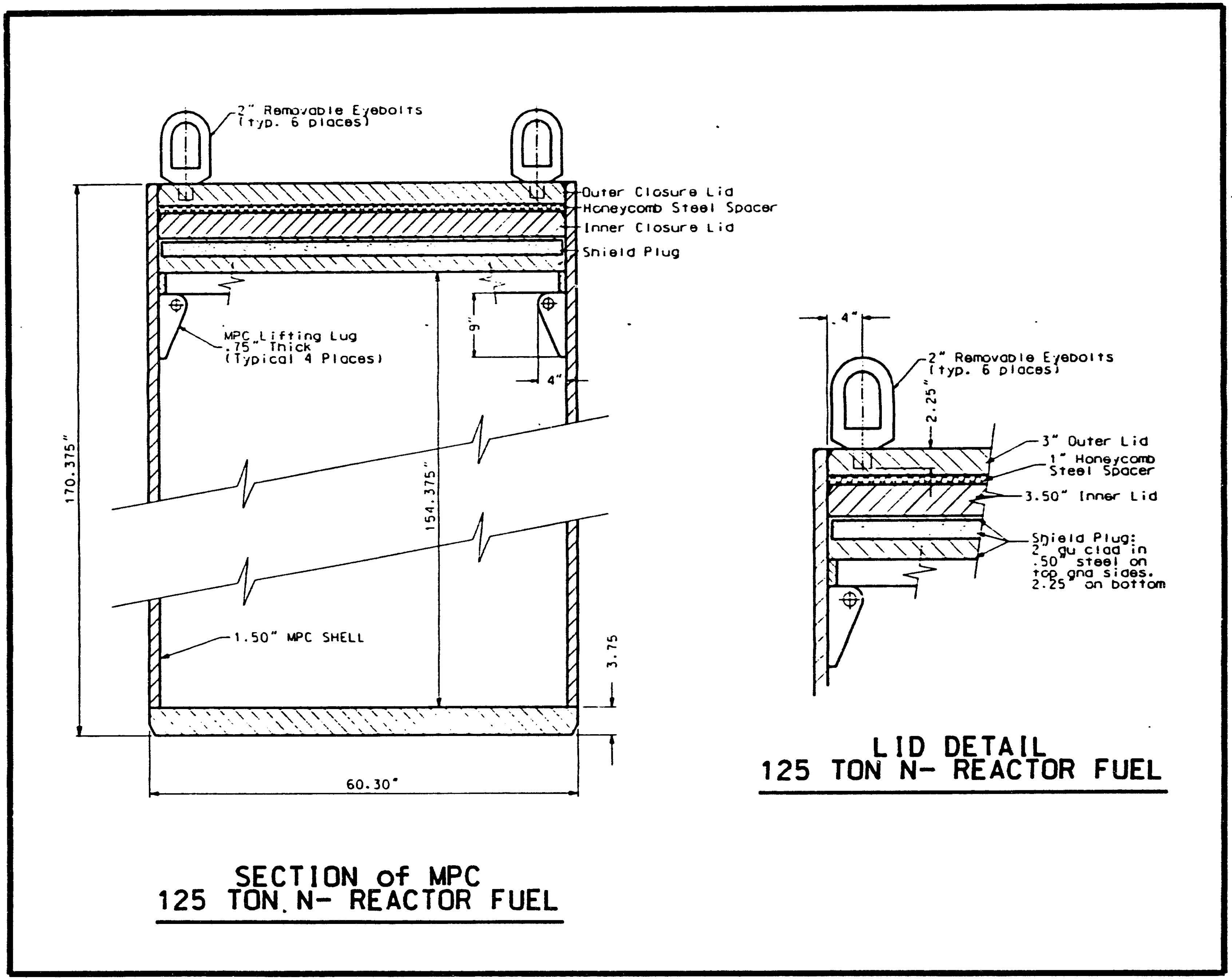


Figure 9. N Reactor MPC shell design.

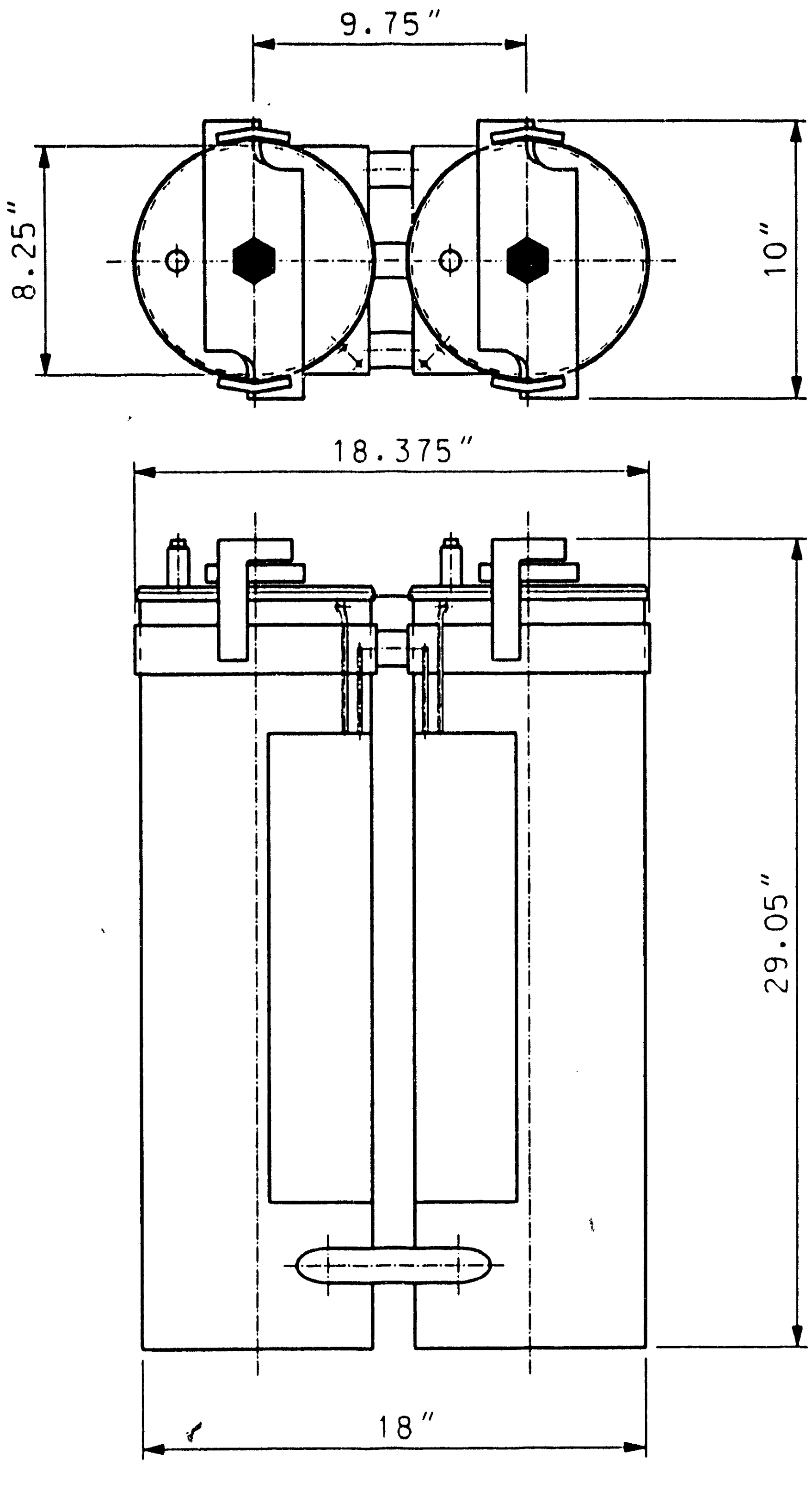


Figure 10. N Reactor storage canister.

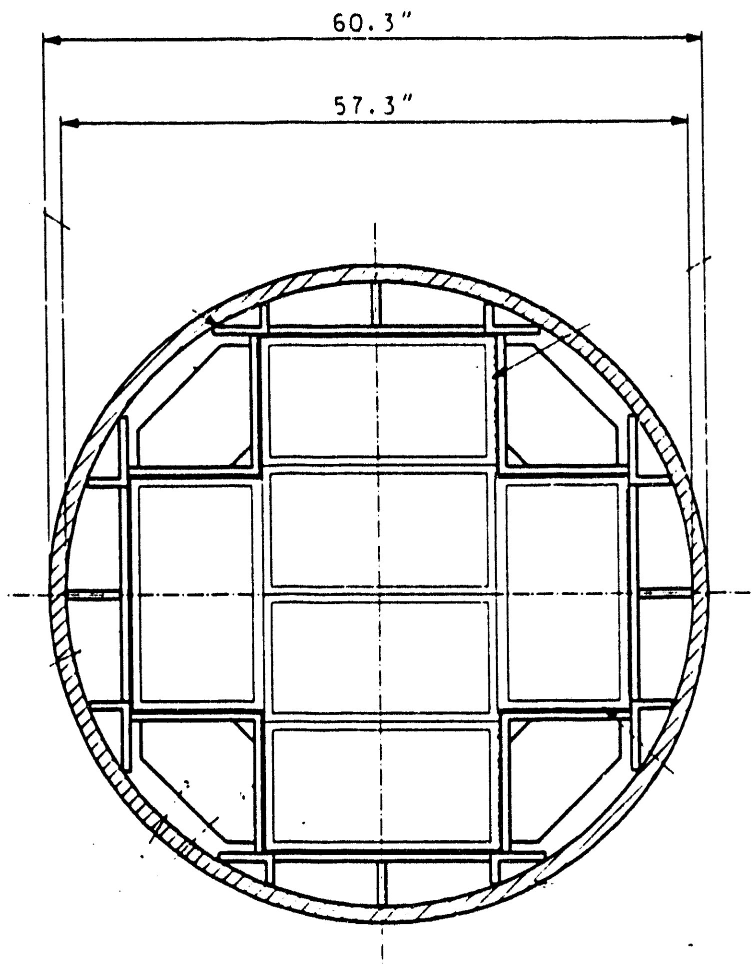


Figure 11. N Reactor MPC basket design.

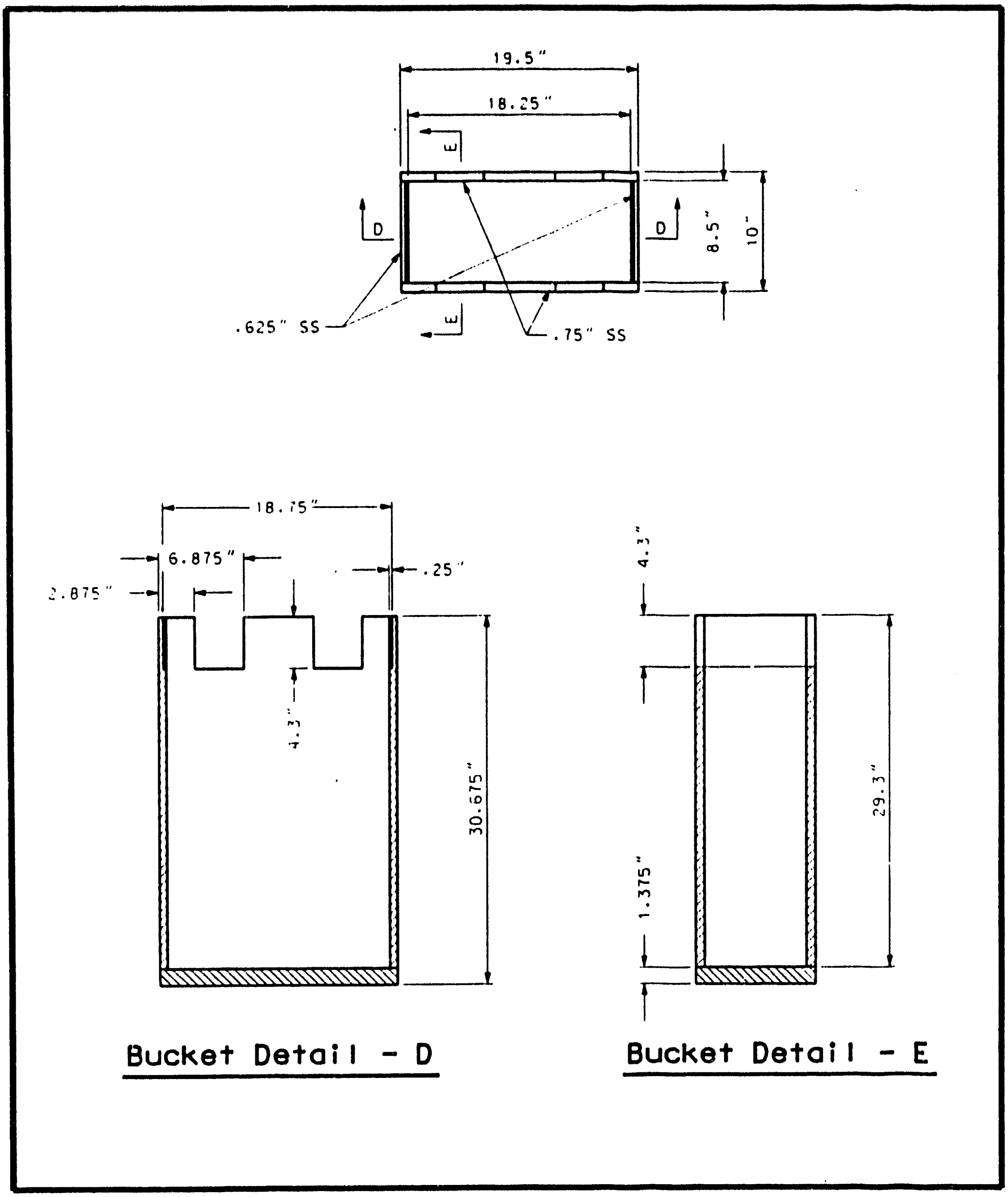


Figure 12. N Reactor canister bucket design.

inside wall of the bucket in a side drop. The result is an even distribution of load into the bucket and into the MPC basket wall. The walls of the bucket also serve as a compressive load path for fuel inertial loads resulting from a 10 CFR 71 hypothetical 9-m end or corner drop accident. Since the buckets are stacked five high, the bottom of one bucket transfers the fuel/canister load into the walls of the bucket below. Therefore, the heaviest loaded bucket is at the bottom. Its walls are sized to provide axial support for the fuel and buckets above it. Its 1.38-in. bottom plate is designed to accommodate the concentrated footprint inertial loading of the fuel canister assembly that it supports.

The N Reactor MPC basket consists of six rectangular cells with 0.75-in. walls. Two of the basket cells are perpendicular to the other four; these two cells have a l-in. wall that faces toward the MPC center. In a 10 CFR 71 hypothetical 9-m (30-ft.) side drop accident, the inertial load of the fuel canisters is transferred through the bucket wall and into the 0.75-in. cell walls. The thicker walls are required to support concentrated compressive loads at their midspan resulting from the staggered cell arrangement. These forces result from inertial loads of adjacent cells. The structural integrity of the individual N Reactor canisters to withstand the inertial load of the fuel they contain should be evaluated during the next phase of design.

The MPC cylindrical shell wall is 1.5-in. It has a 3.5-in. thick inner top lid and 3.75-in. thick bottom plate. These walls form a boundary capable of withstanding hydrostatic pressure resulting from water immersion at a depth of 200 m. The thickness requirements due to the 200-m immersion requirements are the limiting case for the design of these components. ASME Code Case N-284, "Metal Containment Shell Buckling Design Methods Section III, Division 1, Class MC," was used to derive the 1.5-in. MPC thickness. Other load cases considered for the shell are a required internal pressure of 100 psi, and a side drop from a 9-m height in accordance with 10 CFR 71 test requirements. The shell ends will experience local bending stresses due to the transfer of a bending moment from the flat pressure boundaries (i.e., the MPC lid and bottom). The 1.5-in. shell thickness is sufficient to accommodate these stresses. An allowable plate and shell stress intensity of $1.5 \cdot S_m$, as defined in NRC Regulatory Guide 7.6 and in ASME Code Section III, Subsection NB, was considered in this evaluation.

Details of the upper MPC components are shown in Figure 9. The 3-in. outer lid, weld, 316 L SS internal threads and eyebolts are sized to withstand the stresses of a redundant three-point lift or robust 6-point lift. Six 2-in. diameter, high-strength eyebolts having a minimum yield strength of 70 ksi and minimum tensile strength of 95 ksi were evaluated. The thread engagement length was 2.5 in. Stresses in all components evaluated are below the limits defined by NUREG-0612, ANSI N14.6, and 10 CFR 71.

The precrushed steel honeycomb shown in Figure 9 allows a more uniform distribution of the MPC contents inertial load against the end wall of the transportation cask in a 10 CFR 71 hypothetical 9-m end drop accident.

All evaluations were based on quasi-static structural analyses applying a 60 g acceleration to the mass of the MPC contents and s..ucture. This acceleration is approximately twice that predicted from a 9-m drop of a rigid 125-ton transportation package equipped with the isotropic crush strength aluminum honeycomb impact limiters used in the MPC transportation cask conceptual design.

5.1.3.2 Criticality Evaluation. The preliminary N Reactor MPC design was evaluated for criticality safety. The results of this evaluation indicate that criticality control considerations should not be a significant driver of N Reactor fuel MPC designs.

The N Reactor MPC basket cells are formed from 0.75-in. thick SS plates (peripheral cells use l-in. plates on the side facing the center of the MPC, which was modeled as 0.75-in. in the criticality analysis). The cell opening is 10.25 x 19.75 in. The N Reactor canister buckets are formed from 0.75-in. thick SS plates. The bucket opening is 8.5 x 18.25 in. The existing N Reactor fuel canisters were not explicitly modeled in the criticality evaluation. This configuration was evaluated using a 2-D Monte-Carlo criticality analysis model (i.e., infinite fuel assembly axial length). The analysis method uses the Criticality Safety Analysis Sequence No. 4 (CSAS4) included in the SCALE-3 package of codes developed by Oak Ridge National Laboratory (ORNL). CSAS4 and the 123GROUPGMTH master cross-section library included in the SCALE-3 package was used to calculate K_{eff} values for the N Reactor MPC design. The CSAS4 criticality analysis sequence uses a cross-section processing code (NITAWL) and a three-dimensional Monte-Carlo code (KENO-Va) for calculating K_{eff}.

Credit was taken for the inherent neutron absorbing capability of fixed structural components within the MPC basket. Each N Reactor fuel assembly was treated as a heterogeneous system with the annular fuel regions modeled explicitly. The criticality model assumes that groups of seven N Reactor fuel assemblies are packaged in their existing sealed storage canisters with a triangular pitch of 2.8 in. The N Reactor canisters are modeled with water even though they would have to be dried to be disposed of in the MGDS. The water as well as the MPC shell are explicitly modeled to ensure conservative reflector effects.

The preliminary MPC design and N Reactor fuel assembly loading described above were evaluated and determined to be critically safe. The calculated K_{eff} value for the 420 N Reactor fuel assembly MPC is well below the 0.95 acceptance value (including uncertainties) for fuel storage and disposal generally established for commercial reactor fuels.

In general, variations in N Reactor fuel assembly designs or MPC geometry and material uncertainties were not specifically evaluated in this study, but the calculated K_{eff} result for this design indicates sufficient margin to accommodate expected impacts of such uncertainties. Moderator density effects on storage array reactivity were not considered, but are not expected to result in any increase in K_{eff} above that calculated for the full density water condition.

5.1.3.3 Thermal Evaluation. A preliminary thermal evaluation of the N Reactor fuel was performed using the ORNL SCOPE computer code. This analysis determined the total heat output of the MPC package produced by the fuel at the maximum allowable temperature of 380°C. This predicted heat output was 24.4 kW per MPC, or 58 W per assembly for the 420 N Reactor assemblies in the MPC.

After this analysis was completed, it was determined that the maximum N Reactor fuel thermal output is actually 1.76 W per assembly. The thermal performance of the N Reactor MPC is therefore considered to be qualified since this is well below the calculated allowable maximum of 58 W per assembly.

5.1.3.4 Shielding Evaluation. A source term shielding analysis was performed on the Mark IV fuel assemblies after Reference 2 was issued. This analysis indicates that within 5 years after removal from the reactor the radiation on the exterior of the transportation cask, emanating from the 420 N Reactor assemblies in the MPC, will be below the required limits.

After Reference 2 was issued, an actual N Reactor fuel source-term shielding analysis was performed with the N Reactor MPC in the transportation cask. This analysis indicated that within 5 years of the fuel being removed from the reactor, the radiation on the exterior of the transportation cask will be below the required 10 CFR 71 transportation limits.

5.2 FSV MPC

5.2.1 Assumed Fuel Values

Since all required information was not available for the FSV fuel assemblies, the following assumptions were made to complete the feasibility study. Based on a comparison of thermal source terms for FSV fuel (40 W per assembly) and reference PWR fuel for the MPC system (675 W per assembly), it is assumed that the radiological source term for FSV fuel will be significantly less than the design basis for the MPC system. Due to the high-temperature operating levels for FSV fuel, it is assumed that the maximum fuel temperature limit for storage and transport of the FSV fuel assemblies will not be a limiting consideration. The FSV fuel consists of highly enriched TRISO-coated uranium carbide fuel kernels. The experience with this fuel type has demonstrated the integrity of this coating system and its ability to retain fission products at very high temperatures. It is therefore assumed that this fuel does not have to be placed in canisters prior to placing it in the MPC.

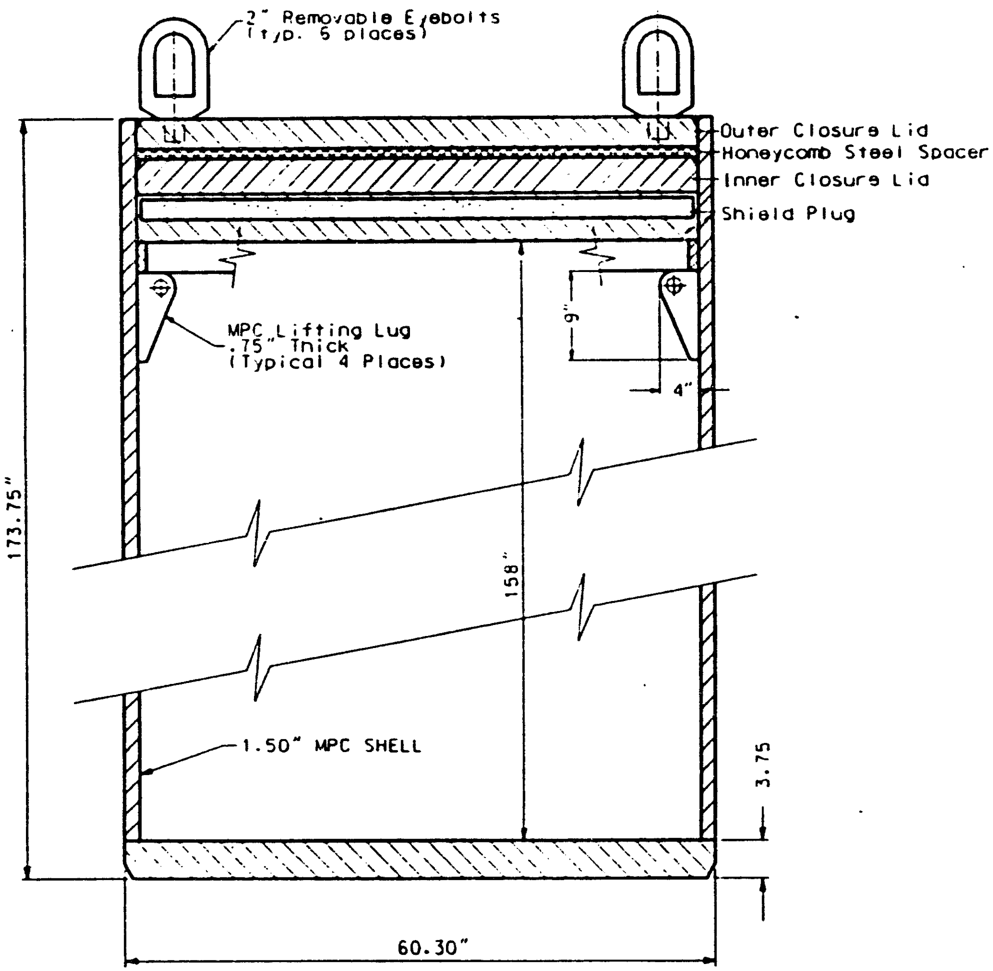
5.2.2 FSV MPC Design

The FSV fuel will require a new MPC design. The FSV MPC shell is shown in Figure 13. The only difference in the FSV MPC and OCRWM MPC shells is that the FSV shell is 173.75 in. high and 1.5 in. thick.[b] Figure 14 shows the new 7-cell hexagonal-grid MPC basket designed for holding the FSV fuel blocks. Table 3 includes parameters of the FSV MPC design. More FSV MPC design details can be found in Reference 2.

5.2.3 Feasibility Results

5.2.3.1 Structural Evaluation. The FSV MPC shell, bottom, and lids were sized in accordance with loadings of the N Reactor MPC described above. The FSV MPC hexagonal basket design was based on a structural analysis using the inertial load of 7 TMI-2 core debris canisters (see section 5.3.3.1) since the latter loads envelop those of the 35 FSV fuel blocks. The density of the TMI-2 fuel in an intact assembly was doubled to account for the random mass

b. This thickness increase is not needed for FSV fuel, but it will be needed for the TMI-2 fuel for the same reasons as explained for the N Reactor in Section 5.1.2. The number of different DOE MPC designs is minimized by using one MPC design for both the FSV fuel and TMI-2 canisters, since both designs can use identical baskets.



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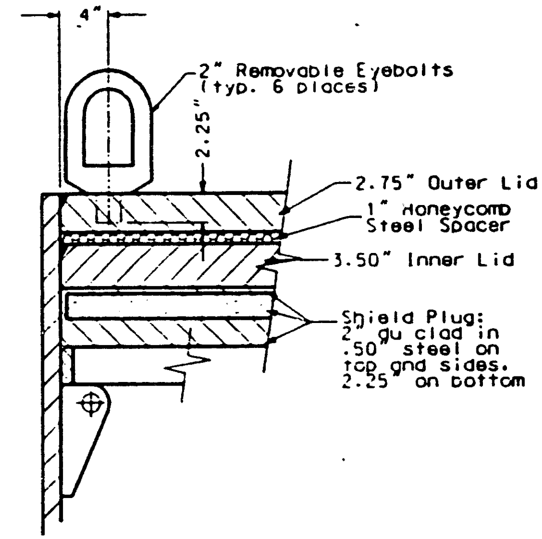

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Figure 13. FSV/TMI-2 MPC shell design.

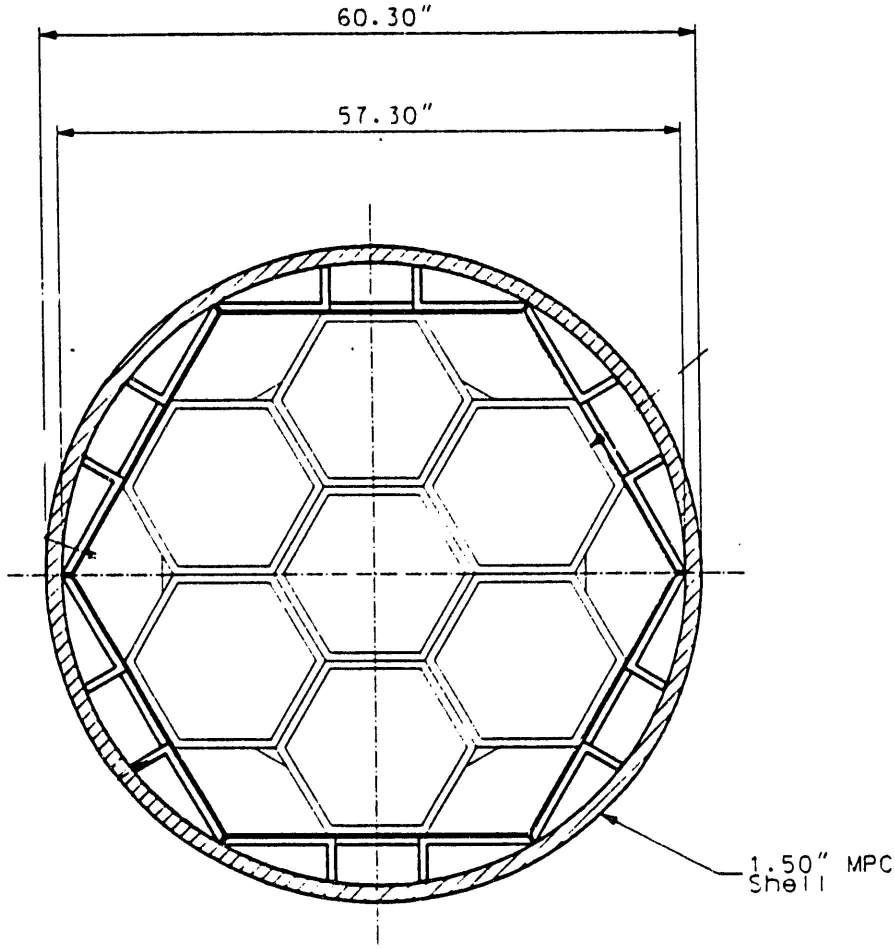

Figure 14. FSV/TMI-2 MPC basket design.

distribution of core debris. The MPC basket for the FSV fuel consists of seven 0.75-in. thick hexagonal plates. The walls have been sized for membrane compressive loading and local bending stresses caused by a 10 CFR 71 hypothetical 9-m side drop accident which is the limiting structural case for the basket design. Gusset plates and support angles will protect the basket as it deforms under side drop loadings. The FSV MPC basket structure was analyzed for a 60 g quasi-static loading. Stresses in the basket satisfy the ASME Section III, 1.5 · S_m allowable stress intensity.

5.2.3.2 Criticality Evaluation. The FSV MPC design was not specifically analyzed for criticality since the criticality characteristics of the relatively small number of 35 FSV fuel assemblies within the MPC are not expected to impose any significant design constraints for criticality.

The basket array is formed by seven hexagonal cells spaced in a triangular array on a nominal 15.922-in. center-to-center spacing. The hexagonal basket cell opening is 14.422 in. across the flats. The basket cells are fabricated from 0.75-in. thick SS plates and do not contain supplementary neutron absorbing panels.

The relatively large mass required to cause criticality in typical graphite moderated systems such as FSV, significantly reduce MPC criticality safety design concerns. For example, the FSV reactor consisted of 1,482 fuel assemblies compared to only 35 in the FSV MPC, and the FSV MPC system is limited to seven columns. Therefore, the neutron leakage from this system will sufficiently limit K_{eff} and preclude the need for additional spacing between basket cells or use of supplementary neutron absorbing panels. The 0.75-in. thick hexagonal SS basket cell walls will also result in significant internal neutron absorption.

The assumption that the preliminary FSV MPC system will not be constrained by significant criticality considerations is supported by criticality calculations[2] performed previously by General Atomics (GA) in support of FSV fuel storage. These calculations confirm that separating FSV fuel assemblies with cylinders of SS is sufficient to control criticality under normal and accident conditions. GA considered infinite arrays of storage cylinders over a range of center-to-center spacings (including a collapsed case with cylinders touching) and considered both dry and flooded conditions. Since the GA calculations include cylindrical cells instead of the FSV MPC close-packed hexagonal cells, the calculations are not directly applicable to the FSV MPC. However, since the worst-case GA K_{eff} was less than 0.95 and the calculations were performed on an infinite array of cells, the results provide confidence that a limited number of FSV fuel columns can be safely stored within close-packed SS cells.

Since detailed FSV MPC criticality analyses have not been performed, it is possible that neutron absorbing panels should be added to the basket to ensure criticality safety. Additional space margin exists within the MPC basket to allow the placement of neutron absorbing panels within the array, if necessary. It may also be possible to place neutron absorbing rods in the existing FSV fuel block burnable poison rod holes, if needed.

5.2.3.3 Thermal Evaluation. The FSV fuel canisters were evaluated using a SCOPE computer model having circular cell opening to approximate an FSV MPC hexagonal close-pack basket cells. The FSV fuel decay heat of 40 W per assembly results in a total heat load of only 1.4 kW for all 35 assemblies in the MPC. The results of the FSV MPC thermal evaluation indicate that the transportation cask outer surface temperature will be 87°C (188°F), the cask lead shielding temperature will be 93°C (200°F), and the fuel temperature will be 108°C (226°F).

Preliminary results indicate that FSV MPC package temperatures for storage and transportation are well within acceptable lead shielding temperature limits. Calculated fuel temperatures are low, which indicate that the FSV MPC concept is feasible from a thermal analysis standpoint.

5.2.3.4 Shielding Evaluation. Radiological source strengths for the FSV fuel assemblies were reviewed and compared to the LWR fuel characteristics, but rigorous shielding calculations were not performed. The decay heat for the FSV fuel is 40 W per assembly at 26,000 MWd/MTIBM and 10 years decay time. The total heat load of 1.4 kW per MPC is about a factor of 10 less than that of the 21 PWR fuel assemblies contained in the large commercial fuel PWR MPC (which is the design-basis fuel for the large MPC). However, the FSV fuel density is a factor of three less than the PWR reference fuel, and therefore, the FSV MPC package self-shielding will be significantly less. Considering these somewhat offsetting conditions, the dose rates from the FSV transportation cask are not anticipated to exceed those for the OCRWM PWR MPC design.

5.3 TMI-2 MPC

5.3.1 Assumed Fuel Values

Since all required information was not available for the TMI-2 core debris, the following assumptions were made to complete the feasibility study. The TMI-2 core debris canister thermal source terms were estimated from the ORNL data base using B&W 15 x 15 fuel assemblies with burnup of 10,000 MWd/MTU and 15-year decay time which resulted in 292 W per assembly. Since the TMI-2 core debris is canisterized, it is assumed that the maximum fuel temperature limit for storage and transport of the TMI-2 core debris canisters will not be a limiting consideration.

5.3.2 TMI-2 MPC Design

The TMI-2 MPC is identical to the FSV MPC design described above. Since the TMI-2 canisters, which contain spent fuel debris, may not withstand immersion to 200 m, the TMI-2 MPC shell thickness was increased 1.5 in., similar to the N Reactor MPC shell, in order to pass this analysis. Table 3 includes parameters of the TMI-2 MPC design. More TMI-2 MPC design details may be found in Reference 2.

5.3.3 Feasibility Results

5.3.3.1 Structural Evaluation. The TMI-2 MPC is identical to the FSV MPC and will accommodate one TMI-2 fuel canister in each of its seven basket cells. As mentioned above in the FSV MPC structural section, the inertial load of the TMI-2 core debris canisters was used as the basis for the structural analysis of the seven-cell hexagonal basket design for this MPC. The density of fuel in the TMI-2 canister was assumed to be double the density of intact TMI-2 fuel assemblies to account for the random mass distribution of the core debris. The evaluation indicated that the TMI-2 MPC is structurally adequate.

5.3.3.2 Criticality Evaluation. The TMI-2 MPC design was not specifically analyzed for criticality in this study. Although the criticality characteristics of B&W 15 x 15 fuel assemblies and fuel debris packaged within the cylindrical TMI failed fuel canisters require significant evaluation, such analysis was beyond the scope of this study. The criticality evaluation for the TMI-2 MPC was limited to the review of References 4, 5, and 6 as called out in Reference 2 of this document. Based on that review, it is anticipated that the TMI-2 MPC basket design could be demonstrated acceptable without significant modification.

However, since a detailed criticality analysis was not performed on the TMI-2 MPC design, it is possible that additional neutron absorbing materials should be added to ensure criticality safety. Additional space exists within the MPC basket structure for adding neutron absorbing panels, if necessary. Also, reducing the TMI-2 MPC capacity from seven to six canisters, by leaving the center basket cell empty, is an option to address criticality concerns.

5.3.3.3 Thermal Evaluation. The TMI-2 core debris canisters contain B&W 15 x 15 fuel material which has a burnup of about 3,176 MWd/MTHM. The decay heat value was based on the ORNL data for the B&W fuel: 10,000 MWd/MTHM burnup, 15-year decay time, 1.69 weight percent ^{235}U enrichment, and 292 W of decay heat per assembly.

The TMI-2 fuel canisters were evaluated using a SCOPE computer model having circular cell opening to approximate a TMI-2 MPC hexagonal basket cell. Each canister was assumed to have an effective fuel density of three times normal, resulting in a decay heat value of 876 W per canister, for a total heat load for the seven canisters within the TMI-2 MPC of 6.1 kW. The results of the TMI-2 MPC thermal evaluation indicate that the transportation cask outer surface temperature will be 102°C (215°F), the cask lead shielding temperature will be 124°C (255°F), the canister temperature will be 177°C (350°F), and fuel temperature will be 192°C (378°F). These preliminary results indicate that the TMI-2 canisterized fuel debris does not exceed any of the MPC package thermal limitations for transportation or storage.

Subsequent to this analysis it was determined that the maximum thermal output for a TMI-2 canister was not allowed to exceed 100 W at the time of its shipment to the INEL. This was due to a restriction in the certificate of compliance, USA/9200/B(M)F, for the NuPac 125-B casks. These results show that the thermal performance of the TMI-2 MPC and transportation cask is well within acceptable lead shielding and fuel temperature limits.

5.3.3.4 Shielding Evaluation. Radiological source strengths for the TMI-2 core debris canisters were reviewed and compared to the LWR fuel characteristics, but rigorous shielding calculations were not performed. This fuel has a low burnup of 3,176 MWd/MTU and a long decay time of 14 years. Since the radiological source term is much less than the reference source term for the commercial PWR MPC, there should be no shielding limitations on the TMI-2 canisters within the TMI-2 MPC package.

5.4 ATR MPC

5.4.1 Assumed Fuel Values

Since not all the required information was available for the ATR fuel assemblies, the following assumptions were made to complete the feasibility study. The ATR thermal source term datum of 1,758 W per assembly at 120 days decay has been verified as being in the approximate range for an assembly operating at 6.25 MW (250 MW per 40 assemblies) for 60 days (from ORIGEN printout provided for the ATR fuel). The verification was performed using NRC Branch Technical Position ASTB 9-2 methodology developed for use on commercial LWR fuel designs and conservative fuel-cycle assumptions (also includes a 20% factor of conservatism). Since fuel and package temperatures exceed acceptable levels at this high heat output, it was determined that a minimum decay time in excess of 120 days is necessary prior to dry storage of ATR fuel assembly. A minimum delay time was established by first estimating the fuel-assembly decay heat source term corresponding to the assumed peak clad temperature limit of 380°C and then determining the decay time necessary to reduce the source term to this level. The decay time estimate was calculated using delay time scaling factors based on PWR decay heat source term characteristics provided by NRC Regulatory Guide 3.54. Based on a comparison of thermal source terms for ATR fuel (295 W per assembly) and reference PWR fuel for the MPC system (675 W per assembly), it was assumed that the radiological source term for ATR fuel was significantly higher than the design basis for the MPC system. This conclusion was based on a 2.5-year decay time and assumed that 120 ATR assemblies would be loaded in an MPC. It is assumed that the maximum peak fuel clad temperature limits for the ATR fuel assemblies are 340°C for 10-year-old fuel and 380°C for 5-year-old fuel. This is predicated on sealing each fuel assembly in an inerted SS can.

5.4.2 ATR MPC Design

The ATR MPC design is identical to the large OCRWM BWR MPC. Its shell, which is 193 in. high, 60.3 in. OD, and 1.0 in. thick, is shown in Figure 1. It uses the commercial 40-cell BWR basket shown in Figure 3. Table 3 includes parameters of the ATR MPC design. More ATR MPC design details may be found in Reference 2.

5.4.3 Feasibility Results

5.4.3.1 Structural Evaluation. As explained below, in Section 5.4.3.3, each ATR fuel assembly within the ATR MPC will be contained within a sealed can shown in Figure 15. The existing 40 BWR MPC design will hold three such cans per cell, or 120 cans total.

The ATR fuel assemblies are relatively light weight, approximately 22 lb each, or 66 lb per basket cell. The fuel cans are estimated at 50 lb each or 150 lb per basket cell. The cans and

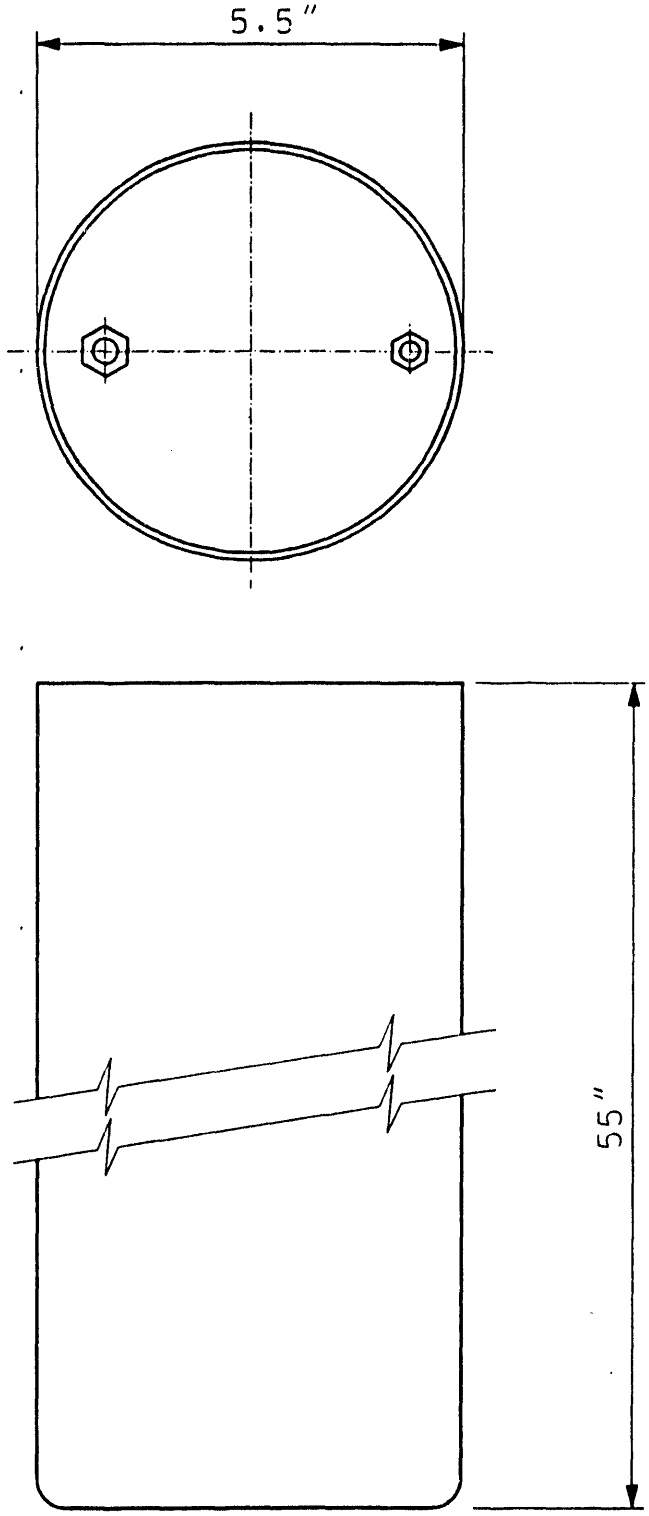

Figure 15. ATR/MTR fuel assembly can.

fuel assemblies therefore weigh 216 lb per basket cell. The 40 BWR MPC basket cells are each designed to accommodate a 730-lb BWR fuel assembly. The existing structural analysis of the 40 BWR MPC load envelops the ATR fuel load, which is therefore acceptable without further structural review. The 0.19-in. wall of the ATR fuel can is adequate to support longitudinal inertial loads when stacked three high.

5.4.3.2 Criticality Evaluation. The ATR MPC design was not specifically analyzed for ATR fuel assembly criticality as part of this study, but a criticality analysis for the MTR type fuel, below, was performed for the same MPC basket design. The ATR and MTR-type fuels are similar in that both are aluminum clad, high enrichment, uranium metal plate type fuel assemblies of similar size and uranium loading. Additionally, the same 40 BWR MPC fuel basket design is proposed for use with both ATR and MTR-type fuels. These and other similarities provide a high degree of assurance that the MTR analysis results will be reasonably applicable to the ATR fuel. Based on the results of the MTR criticality analysis, it is anticipated that the 40 BWR MPC basket design proposed for use with ATR fuel will be able to be demonstrated acceptable with a worst-case K_{eff} of less than 0.80.

The above conclusion is valid during transportation and storage, but not necessarily during long-term disposal in the MGDS. Since the ATR contains HEU, potential long-term degradation of the MPC package and subsequent theoretical fuel geometry redistribution during disposal could result in relatively close packing of some of the 0.129 metric tons of fissile material. Under this condition, and assuming a worst-case geometry, the ATR HEU fuel may fail the MGDS criticality criteria for severe degradation. In that case, each MPC may be required to contain so few ATR fuel assemblies that it will be impractical to use MPCs for this fuel. An alternative solution may be immobilizing the fuel within the MPC by injecting a stable medium into the MPC cavity after the fuel cans have been inserted.

5.4.3.3 Thermal Evaluation. The performance characteristics of HEU fuels with irradiated aluminum cladding and the lack of regulatory criteria for storage, transportation, and disposal make it difficult to assess the probability of this fuel being accepted at the MGDS. Additional study of MGDS acceptance and licensing assessment of this material are recommended during the next phase of design. It is assumed here that the temperature limits for zircaloy-clad fuel (380°C at 5 years decay time and 340°C at 10 years decay time) will be applicable to the ATR fuel if each assembly is sealed within an inerted SS can before being placed in the MPC. This thermal evaluation is therefore based on the assumption that existing limits for zircaloy-clad fuel are also allowed for the ATR aluminum-clad fuel contained within the can. The proposed design for this can is shown in Figure 15.

Table 2 indicates an ATR fuel decay heat value of 1,758 W per assembly after 120 days of cooling. A SCOPE computer code evaluation was conducted to estimate the maximum allowable decay heat that would be acceptable in a 40 BWR fuel assembly design for the zircaloy-clad fuel temperature limits. The limiting total thermal heat load allowed for the MPC was determined to be 35.4 kW at which the transport cask lead temperature will approach the melting point. For 120 ATR fuel assemblies in the MPC, this corresponds to a maximum allowable thermal load of 295 W per fuel assembly. At 35.4 kW, the transportation cask outer surface temperature will be 171°C (339°F), the cask lead shielding temperature will be 299°C (570°F), the canister temperature will be 363°C (685°F), and fuel temperature will be 378°C (712°F).

Based on a thermal load of 1,738 W per assembly at 120 days cooling time, it is estimated that 2.5 years decay time is necessary to reduce the thermal output to below the 295 W per assembly maximum allowable. Therefore, if the ATR fuel is allowed to cool for longer than 2.5 years, it should be able to be safely placed in the ATR MPC from a thermal standpoint. More detailed analysis using ORIGEN computer code models for specific ATR fuel irradiation conditions is recommended before making final conclusions regarding MPC system adequacy for storing or transporting ATR fuel.

5.4.3.4 Shielding Evaluation. After Reference 2 was issued, source term and shielding calculations were performed for ATR fuel to show the shielding performance of the ATR MPC within the transportation cask. These analyses showed that, after approximately 2.5 years of fuel cooling time, the radiation on the exterior of the transportation cask will be below the required 10 CFR 71 limits.

5.5 MTR MPC

5.5.1 Assumed Fuel Values

Since all required information was not available for the MTR-type fuel assemblies, the following assumptions were made to complete the feasibility study. Conservative decay heat source terms were derived for MTR-type fuel from a listing of typical MTR-type fuel parameters for various foreign facilities. A maximum MTR burnup of 283 gm ^{235}U is indicated by the listing. By correlating decay heat source term to the mass of uranium burned, PWR decay heat listings provided in NRC Regulatory Guide 3.54 were used to estimate a thermal source term for MTR-type fuel. MTR-type fuel decay heat source terms for 5 and 10 years decay were estimated to be 18 and 11 W per assembly, respectively. Based on a comparison of thermal source terms for MTR-type fuel (18 W per assembly) and reference PWR fuel for the MPC system (675 W per assembly), it is assumed that the radiological source term for MTR-type fuel will be significantly less than the design basis for the MPC system. This conclusion assumes at least 5 years cooling of MTR-type fuel prior to MPC loading or transport. It was assumed that the maximum peak fuel clad temperature limits for the MTR-type fuel assemblies are 340°C for 10-year-old fuel and 380°C for 5-year-old fuel. This is predicated on sealing each fuel assembly in an inerted SS can.

5.5.2 MTR MPC Design

The MTR MPC design is identical to the ATR MPC design discussed above in Section 5.4.2.

5.5.3 Feasibility Results

5.5.3.1 Structural Evaluation. Each fuel canister shown in Figure 15 will contain one MTR-type fuel assembly. The existing 40 BWR MPC design will hold three canisters per cell, or 120 canisters total. The MTR-type fuel assemblies are relatively light weight, approximately 16 lb each, or 48 lb per basket cell. The fuel canisters are estimated at 50 lb each or 150 lb per basket cell. The canisters and fuel assemblies therefore weigh 198 lb per cell. The 40 BWR MPC basket cells are each designed to accommodate a 730-lb BWR fuel assembly. Therefore, the existing structural analysis of the 40 BWR MPC envelops the MTR-type fuel loads and is acceptable without further structural review. The 0.19-in. wall of the ATR fuel can is adequate to support longitudinal inertial loads when stacked three high.

5.5.3.2 Criticality Evaluation. The MTR MPC design has been evaluated with respect to criticality safety. The MTR MPC and basket configuration is that of the 125-ton OCRWM 40 BWR MPC. This basket configuration has been determined to be capable of accepting up to 4.4 weight percent ^{235}U for 40 unirradiated BWR fuel assemblies from a criticality standpoint. The criticality analysis performed for the MTR MPC assumed that borated aluminum alloy was used in the basket.

The basket was evaluated using a 2-D Monte-Carlo criticality analysis similar to that used for the N Reactor fuel in Section 5.1.3.2. Credit was taken for the inherent neutron absorbing capability of fixed structural components within the MPC basket. Each MTR-type fuel assembly was treated as a heterogeneous system with the plate fuel regions modeled explicitly. The criticality model assumed that a single MTR-type fuel assembly is packaged in a sealed inerted SS can (Figure 15) and placed in an MPC basket cell. The MTR MPC spent fuel storage array was modeled with infinite axial length. Water as well as the MPC shell and metallic structural support members surrounding the periphery of the MTR-type fuel assemblies were explicitly modeled to ensure reflector effects were simulated.

The calculated K_{eff} value of the MTR MPC was determined to be less than 0.80. This calculated result is well below the acceptance criteria for fuel handling, storage, transport, and disposal generally established for commercial reactor fuels (i.e., K_{eff} < 0.95 including consideration of uncertainties).

In general, variations in MTR-type fuel assembly designs and fissile material loading or geometrical and material uncertainties were not specifically evaluated for this study, but the calculated nominal K_{eff} for this design indicates sufficient margin to accommodate expected impacts of such uncertainties. Moderator density effects on storage array reactivity have not been considered as part of this preliminary evaluation.

The MTR-type fuel assemblies also contain HEU, similar to ATR fuel, and are therefore potentially subject to the same long-term severe degradation disposal criticality concerns as the ATR fuel. Therefore, the MTR-type fuel assemblies may be subject to the same criticality limitations and alternatives as discussed above for the ATR fuel.

5.5.3.3 Thermal Evaluation. A listing of typical MTR-type fuel parameters for foreign facilities was used to derive conservative decay-heat source terms for MTR-type fuel assemblies. The listing shows that the maximum MTR burnup is 283 gm of ^{235}U. By correlating the decay-heat source term to the mass of uranium burned, MTR-type fuel decay-heat source terms for 5 and 10 years decay were estimated. These values were 18 and 11 W per assembly, respectively. The MTR-type fuel was thermally evaluated using a SCOPE computer model with 18 W per assembly decay heat, which produced 2.16 kW total for the MPC. Results of the MTR-type fuel thermal evaluation indicate a transportation cask outer surface temperature of 89°C (193°F), a cask lead shielding temperature of 97°C (207°F), a canister temperature of 103°C (218°F), and a fuel temperature of 104°C (220°F). These results show that the thermal performance of the FSV MPC and transportation cask is well within acceptable lead shielding and fuel temperature limits.

5.5.3.4 Shielding Evaluation. Radiological source strengths for the MTR-type fuel assemblies were reviewed and compared to the LWR fuel characteristics, but rigorous shielding calculations were not performed. Shielding required for the MTR-type fuel is considered to be enveloped by that required by the analyses performed for the ATR fuel.

6. STORAGE FACILITY INTERFACE

To determine the method of handling and loading the DOE-fuel MPCs, the infrastructure for cask MPC and fuel handling at the DOE fuel storage locations were investigated. If a fuel storage location has a crane lifting capacity greater than the under-the-hook weight of the DOE 125-ton MPCs, cask transport provisions, high bays, and provisions for dry cask fuel transfer or underwater cask fuel transfer, then the DOE 125-ton MPCs may be directly loaded, seal welded, and inert-gas-backfilled at the DOE fuel storage location. Otherwise, alternative or interim fuel loading and transport methods may be required before final loading of the fuel into the DOE 125-ton MPCs.

MPC fuel loading scenarios for commercial BWR and PWR fuel at commercial reactor storage pools are discussed for reference, and the fuel handling infrastructure at the DOE facilities, which store the six candidate DOE fuel assembly types, was investigated. Methods are discussed for loading the DOE-fuel 125-ton MPCs at each DOE storage location, and limitations and deficiencies are listed.

6.1 OCRWM 125-Ton MPC/Reactor SNF Pool Interface

Reference 1 indicates that, where possible, OCRWM reactor fuel will be loaded into the commercial 125-ton MPCs at the reactor spent fuel pools. These commercial 125-ton MPCs will be transported by rail car to the MRS for storage, and subsequently to the MGDS for disposal. However, some of the reactor spent fuel pool facilities do not have the capacity to handle the 125-ton MPCs, and therefore a number of alternate MPC loading and handling scenarios were proposed in Reference 1. The following two alternative scenarios are similar to two listed in Reference 1, but have been modified to suit DOE-fuel MPCs.

Scenario 1 DOE fuel storage facilities that are not capable of handling 125-ton MPCs in their existing spent fuel storage areas or pools, would load the fuel into small onsite transfer casks in the storage area and transfer the fuel to 125-ton MPCs in an onsite dry-cask transfer facility. The 125-ton MPC would then be placed in shielded storage at a DOE facility. After the MGDS is opened, the MPCs would be placed in a transportation overpack cask and transported there by rail. At the MGDS, the MPC would be removed from the transportation overpack cask and placed in a disposal overpack cask, which would then be placed in disposal in the MGDS.

Scenario 2 DOE fuel storage facilities that are incapable of handling 125-ton MPCs in their spent fuel storage areas would load the fuel assemblies into standard light-weight transportation casks. These casks would then be transported by legal-weight truck to the MGDS, where the fuel assemblies would be transferred to a 125-ton MPC in the MGDS dry-cask fuel transfer facility. At the MGDS, the MPC would be placed in a disposal overpack-cask, which would then be placed in disposal in the MGDS.

Scenario 2 will only be able to be implemented after the MGDS is opened in approximately 30 years. Current DOE National Program SNF interim storage planning indicates that in the near future, long before opening the MGDS, much of the DOE-fuel will be removed from existing storage areas, canned, and placed in interim shielded dry storage at various DOE facilities. Since the facility infrastructure at those future storage locations is unknown, no infrastructure evaluation for MPC use is possible for Scenario 2.

6.2 DOE-Fuel MPC Interface with DOE Fuel Storage Facilities

To fully determine the feasibility of using the MPC concept for DOE fuel, the cask handling infrastructure was examined. This established whether the sites can load fuel directly into 125-ton MPCs at the storage locations, or if alternate fuel handling methods must be used. Scenario 1 above may be able to be used at DOE fuel storage locations where the DOE 125-ton MPCs cannot be directly handled or loaded.

Since the maximum legal-weight truckload is 25 tons, rail cars are required for normal transport of DOE 125-ton MPCs. Rail car access exists at all DOE sites, but not at all fuel storage locations at the sites.

For the fuel types shown in Table 2, the N Reactor fuel is stored at the Hanford site in Washington State; the TMI-2 fuel canisters, ATR fuel assemblies, and FSV graphite fuel blocks are stored at the INEL; and the MTR-type foreign fuel will probably be stored at the SRS in South Carolina. The future receipt and storage location(s) of foreign MTR-type fuel assemblies will not be established until National Environmental Policy Act (NEPA) documentation is in place. However, it is assumed that the SRS RBOF will be selected for storing this fuel. Potential fuel handling and loading methods for the DOE-fuel MPCs at Hanford, INEL, and SRS are discussed below.

6.2.1 Hanford

6.2.1.1 Preferred Method. The N Reactor Mark 1A and Mark IV fuel assemblies listed in Table 2 are stored in the 105-K East (105-KE) and 105-K West (105-KW) basins in the 100 Area of the Hanford site. Figure 4 shows a typical N Reactor fuel assembly. Table 2 indicates that in excess of 103,000 total Mark 1A and Mark IV fuel assemblies are stored in the basins. All N Reactor fuel is stored in canisters that hold 14 fuel assemblies each. A typical N Reactor canister is shown in Figure 10. The 3,815 canisters in the 105-KW basin have sealed lids and are filled with water except for a 2.5 in. nitrogen gas blanket. The 3,666 canisters in the 105-KE basin are not scheduled to have lids installed until 1996. Since the fuel assemblies are in varying states of deterioration, removing them from the canisters is not recommended and may not be possible in some cases. There are no provisions in the fuel canister design for removing the water, and it may be difficult to obtain permits for offsite transportation, storage, or disposal of these canisters if the standing water is not removed. This could be a limiting restriction, independent of the MPC system, in disposing of the N Reactor fuel.

The 105-KE and 105-KW basins are each served by separate 30-ton cranes. To load the N Reactor canisters directly into the N Reactor 125-ton MPCs at the basins, these cranes would have to be replaced with cranes and rails rated at a minimum of 114-tons hook load. In addition, equipment would have to be designed and built to breach the canisters, dewater them, dry the fuel within, and reseal them. This is assuming that there are no insurmountable safety concerns associated with storing the fuel in the dry condition. These modifications are probably the least expensive method which will allow use of the N Reactor 125-ton MPC. The MPCs could then be directly loaded underwater in the basins with dry fuel, removed, dewatered, dried, seal welded, backfilled, and closed. The 125-ton MPC transportation overpack cask would then be used to transfer the MPC to a shielded interim storage area at a DOE site. The MPCs would eventually be removed from storage, placed into the transportation casks, and shipped to the operational MGDS. Once at the MGDS, the MPC would be transferred to a disposal overpack cask for permanent disposal.

Each 125-ton N Reactor MPC would hold 420 N Reactor fuel assemblies. Therefore, 247 N Reactor MPCs would be needed to accommodate the total of 103,640 N Reactor fuel assemblies. The N Reactor fuel constitutes the majority of the DOE fuel, and therefore this fuel is paramount to the success of a project for MPC disposal of DOE fuel.

6.2.1.2 Alternative Methods. Regarding Scenario 1 above, no operational hot shop, cave, or canyon exists at Hanford that is capable of handling the N Reactor 125-ton MPC or performing the dry-cask fuel transfer option. Therefore, no viable alternatives to the preferred method exist.

6.2.2 INEL

6.2.2.1 TMI-2 Canisters.

6.2.2.1.1 Preferred Method—The preferred method for the TMI-2 canisters is similar to Scenario 1 above. The envelope dimensions of a typical TMI-2 canister are shown in Figure 5. The canisters primarily contain fuel debris, reactor components, and filters from the damaged TMI-2 reactor. They are stored in the Test Area North (TAN), Building 607 storage pool, adjacent to the 607 hot shop. The 607 hot shop is a dry-cask fuel transfer facility similar to that discussed in Scenario 1 above.

Provided that after the TMI-2 canisters are dried, they do not generate enough gas to overpressure the MPC, the following method would be used to place them into the 125-ton MPCs. The empty TMI-2 125-ton MPC would be delivered by rail car to the Central Facilities Area (CFA) 200-ton gantry crane where it (and its cradle) would be transferred to a leased or DOE-owned 150-ton trailer (this trailer does not exist at the INEL at this time). From there, it would be transported on the trailer to the TAN 607 hot shop. Special provisions or appeals for transporting this heavy load from CFA to TAN may be required.

The TMI-2 125-ton MPC would be moved into the TAN hot shop and placed upright on the hot shop floor using the 110-ton bridge crane (the maximum under-the-hook weight of the TMI-2 125-ton MPC is 105 tons). The TMI-2 canisters would then be brought into the hot shop from the 607 storage pool. After being purged of water (by an existing TMI-2 canister water purge system) and internally dried (by a TMI-2 canister drying system that is planned but not yet designed), the canisters would be loaded into the 125-ton MPC with the canister grapple and hot shop crane. The MPC would be seal welded, backfilled, closed, and loaded back onto its cradle on the MPC trailer. The 125-ton MPC transportation overpack cask would be used to transfer the MPC to a shielded interim storage area at a DOE site. The MPCs would eventually be removed from storage, placed into the transportation casks, transported by trailer back to the

CFA gantry crane where the MPC, cask, and its cradle would be loaded onto the rail car and transported to the operational MGDS facility. Once at the MGDS, the MPC would be transferred to the disposal overpack cask for placement in permanent disposal.

Each TMI-2 MPC would hold seven fuel canisters. Therefore, 49 TMI-2 MPCs would be needed to accommodate the total of 342 canisters.

6.2.2.1.2 Alternative Method—No viable alternatives for the TMI-2 canister exist.

6.2.2.2 FSV Fuel.

6.2.2.2.1 Preferred Method—FSV fuel was obtained from Public Service of Colorado. Figure 6 shows a typical FSV graphite fuel block. The fuel blocks were transported by trailer to the INEL in the 30-ton FSV-1A cask, which holds a single column of six FSV fuel blocks in end-to-end fashion. The FSV-1A cask was transported by trailer to the ICPP Irradiated Fuels Storage Facility (IFSF) Building 603 Graphite Storage Facility (GSF) where it was off-loaded using the outdoor 60-ton crane (the 001 crane) and placed on the GSF handling cave transfer car. This car, rated at 100 tons, transferred the cask to the GSF handling cave. Once in the cave, the cask was opened and the individual FSV fuel blocks were removed and transferred into IFSF fuel assembly storage cans, which hold a single column four blocks high. These cans were then transferred by shuttle car from the handling cave into the GSF storage room and placed in the GSF storage racks.

The 001 crane is rated at 60 tons, but its rails are rated at 170 tons. The crane would have to be replaced or upgraded to handle the FSV 125-ton MPC. The other limiting weight restriction is the 100-ton handling cave transfer car, which should be able handle the FSV 125-ton MPC (which has a maximum under-the-hook weight of 101 tons). Minor transfer-car modifications may be required to interface with the MPC transportation overpack cask.

With these two facility items upgraded, the FSV 125-ton MPC would be transported by trailer from the CFA gantry crane to the IFSF and placed in the handling cave as described above for the FSV-1A cask. It would then be loaded with FSV fuel blocks, which would be removed from the fuel assembly storage cans. The MPC would then be seal welded, backfilled, closed, and subsequently transported out of the IFSF. The 125-ton MPC transportation overpack cask would be used to transfer the MPC to a shielded interim storage area at a DOE site. The MPC would eventually be removed from storage, placed in the transportation casks, and transported by trailer back to the CFA gantry crane, where it and its cradle would be loaded onto the rail car and transported to the operational MGDS facility. Once at the MGDS, the MPC would be transferred to the disposal overpack cask for placement in permanent disposal.

Each FSV MPC would hold 35 FSV graphite fuel blocks. Therefore, 64 FSV MPCs would be needed to accommodate all 2,208 FSV graphite fuel blocks.

6.2.2.2.2 Alternate Method—Scenario 1 is the alternate for the FSV fuel. If the 001 crane and the handling cave transfer car are not upgraded and modified, the 30-ton FSV-1A cask could be used to transport six graphite fuel blocks at a time to the TAN hot shop dry-cask fuel transfer area where they could be transferred into an FSV 125-ton MPC. Three hundred sixty-eight round trips of the FSV-1A cask would be required to transport all 2,208 fuel blocks to the hot shop. The 125-ton MPC transportation overpack cask would be used to transport the MPC to a shielded interim storage area at a DOE site. The 125-ton MPC would eventually be transported by trailer back to the CFA gantry crane, where it and its cradle would be loaded onto the rail car and transported to the MGDS facility. Once at the MGDS, the MPC would be transferred to the disposal overpack cask for placement in permanent disposal.

6.2.2.3 ATR Spent Fuel.

6.2.2.3.1 Preferred Method—A typical ATR fuel assembly is shown in Figure 7. After removal from the reactor, the upper and lower end fittings of these assemblies are removed. The assemblies are then transported from the ATR reactor canal at the Test Reactor Area (TRA) to the ICPP in the ATR Spent Fuel Element Transfer Cask. Until 1989, ATR fuel assemblies were dissolved and processed at the ICPP. Since that time the ATR fuel assemblies have been temporarily stored in the ICPP Building 666 pool. The pool holds 759 assemblies; and over the next 7 years, an additional 775 assemblies will be added, resulting in a total of 1,534 fuel assemblies.

To load this fuel into the ATR 125-ton MPC, the MPC would be transported by trailer the short distance from the CFA gantry crane to the ICPP Building 666 storage pool. The 130-ton ICPP 666 bridge crane would be used to stand upright the 125-ton MPC, remove it from the trailer, and submerge it to the bottom of the storage pool. As indicated in Reference 2, the ATR fuel assemblies should be sealed in dry cans before placing them in the MPC.[c] The 10-ton Building 666 crane would be used to move the ATR cut fuel assemblies to the canning area. After placing a fuel assembly in the underwater can, the can would be dried, inerted, and sealed. The 10-ton crane would then place the cans into the ATR 125-ton MPC. Once loaded with 120 canned inerted fuel assemblies, the MPC would be removed from the pool, purged of water, dried, seal welded, backfilled, and closed. The 125-ton MPC transportation overpack cask would be used to transport the MPC to a shielded interim storage area at a DOE site. The MPC would eventually be removed from storage, placed in the transportation casks, transported by trailer back to the CFA gantry crane, where it and its cradle would be loaded onto the rail car and transported to the operational MGDS facility. Once at the MGDS, the MPC would be transferred to the disposal overpack cask for placement in permanent disposal.

Each ATR MPC would hold 120 ATR fuel assemblies. Therefore, 13 ATR MPCs would be needed to accommodate the total of 1,534 ATR fuel assemblies.

6.2.3 MTR-Type Foreign Fuel

Figure 8 shows a typical MTR-type fuel assembly. The MTR-type fuel assemblies are 51 in. long including a 12-in.-long fitting at each end. This report assumes that these end fittings will remain intact. As indicated above, it is assumed that the foreign fuel will be stored in the RBOF basins at the SRS. The MTR-type fuel would need to be canned; and therefore, a new canning

c. The ICPP is currently considering a facility for dry-canning ATR and other fuels. Such a facility could be used for canning the ATR fuel before placing it in the ATR MPCs.

facility will be required at the RBOF for use with the MPC system. Each BWR MPC would hold 120 canned MTR-type fuel assemblies. Therefore, 47 MTR MPCs are required to accommodate the total of 5,600 MTR-type foreign fuel assemblies.

The RBOF Receiving and Storage Area can accommodate spent fuel casks transported by either trailer or rail car. The facility has two 50-ton bridge cranes, which can be combined to provide a 100-ton lift rating, and a 3-ton overhead monorail hoist-transfer system for moving individual fuel assemblies and fuel bundles. The RBOF water basin consists of a cask unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin, all of which are interconnected by transfer canals. Although the RBOF bridge cranes have a combined rating of 100 tons, they do not meet present standards for lifts near that rating. Therefore, upgrades may be required to bring them into current compliance for lifting the MTR-type 125-ton MPC, which weighs 93 tons (see Table 3).

To load MTR-type fuel into the MTR 125-ton MPC, the MPC would be transported by rail car to the RBOF Receiving and Storage Area. The two upgraded 50-ton RBOF bridge cranes would be used to stand upright the MTR 125-ton MPC, remove it from the rail car, and submerge it to the bottom of the cask unloading basin. As indicated in Reference 2, the MTR-type fuel assemblies would be sealed in dry cans before placing them in the MPC. Equipment for this canning operation does not exist at SRS at this time and is not in planning. The 3-ton monorail hoist would be used to move the MTR-type fuel assemblies to the canning area. After placing a fuel assembly in the underwater can, the can would be purged of water, dried, inerted, and sealed. The monorail hoist would then be used to place the can into the MTR 125-ton MPC. Once loaded with canned fuel assemblies, the MPC would be removed from the cask unloading basin, purged of water, dried, seal welded, backfilled, and closed. The 125-ton MPC transportation overpack cask would be used to transport the MPC to a shielded interim storage area at a DOE site. The MPCs would eventually be removed from storage, placed into the transportation casks, and transported by rail to the operational MGDS facility. Once at the MGDS, the MPC would be transferred to the disposal overpack cask for placement in permanent disposal.

7. COST ESTIMATE

Table 4 is a preliminary cost estimate of the DOE-fuel MPCs required to accommodate all the fuel represented by the six fuel-assembly types considered in this study. The table includes the six fuel-assembly types, fuel-assembly quantities, number of fuel assemblies per MPC, total MPC quantities, and MPC costs.

Four hundred twenty MPCs are required to handle all DOE fuel assemblies of the six types studied in this report: 60 are commercial BWR MPCs for ATR and MTR-type fuels, 113 are FSV/TMI-2 MPCs, and 247 are N Reactor MPCs. The budgetary costs of the three MPCs are $432K for each commercial BWR MPC, $327K for each FSV/TMI-2 MPC, and $437K for each N Reactor MPC. The total budgetary cost of all 420 DOE-fuel MPCs is $170.8M.

Table 4. Cost estimate for DOE fuel MPCs.

Fuel assembly type	Total number of assemblies	Fuel assemblies per MPC	Total MPCs required	Budgetary cost per MPC ($K)	Total cost ($M)
N Reactor Mark IV	65,000	420	155	$437	$67.8
N Reactor Mark 1A	38,640	420	92	$437	$40.2
FSV	2,208	35	64	$327	$20.9
TMI-2	342	7	49	$327	$16.0
ATR	1,534	120	13	$432	$5.6
MTR-Type (Foreign)	5,600	120	47	$432	$20.3
Totals			420		$170.8

8. DOE-FUEL MPC ISSUES

The following are problem areas and issues, which may impede use of the MPC system for DOE spent fuel, found during this feasibility study. Some areas may not actually materialize as problems or may not be problem areas when further investigated. Some of these problem areas are unique to the fuel itself, and independent of which handling system, MPC or other, are used to transport, store, and dispose of SNF. At this time, there is no evidence that any of them preclude use of the MPC for any of the DOE fuels studied in this report.

DOE fuels have not been accepted for disposal at an MGDS. This acceptance would have to be gained before the fuels could be disposed of. The new DOE-fuel MPC designs will require a license amendment (to the anticipated future OCRWM MPC license) before they can be used for DOE fuel transportation.

The N Reactor fuel is stored in water-filled canisters with no drain provisions. These canisters may not be accepted for offsite transportation or disposal in the MGDS, with standing water. Reference 2 recommends that the N Reactor canisters be purged of water and opened to the MPC internal environment to facilitate vacuum drying and helium backfill of the N Reactor canisters. Technical problems may exist, however, which require that N Reactor fuel be wet-stored only, and not dried. The structural condition of the N Reactor canisters is uncertain. Therefore, structural performance of the canisters in a 10 CFR 71 hypothetical accident scenario could not be evaluated.

The performance characteristics of irradiated ATR and MTR-type SNFs with irradiated aluminum cladding, and the lack of regulatory criteria for storage, transportation, and disposal, make it difficult to assess the probability of this fuel being accepted at the MGDS. This problem is not unique to use of the MPC system for fuel handling. The assumption made by Reference 2 that the existing temperature limits for zircaloy-clad fuel are applicable to the aluminum-clad fuel, if it is sealed in a SS inerted can, has not been verified. The SRS RBOF would require new canning and drying equipment to can the MTR-type foreign fuel prior to placing it in the MPC. There is no such canning equipment planned at RBOF. This canning equipment may be required for any storage and disposal system, however, not just the MPC system. The ATR and MTR-type HEU fuels, when placed in the MPC, may not meet MGDS disposal criticality requirements due to potential long-term, severe degradation of the MPC system. In this case, a less-than-critical mass of fuel could be placed within each MPC, or a small MPC could be used to keep the mass low. Problems could also arise with licensing the ATR and MTR-type spent fuels for transportation, but this may not be unique to using the MPC system.

The TMI-2 canisters will only be able to be placed in a sealed MPC if, after internal drying, they do not generate enough gas from radiolytic decomposition of residual water to overpressure the MPC. The structural condition of the TMI-2 core debris canisters shown in Figure 5 is uncertain, and therefore the 10 CFR 71 structural performance of the canisters in a hypothetical accident scenario could not be evaluated in Reference 2. None of the five DOE facilities storing the six fuel types studied will be able to handle and load the MPCs without some upgrades or new equipment.

Current planning indicates that the OCRWM MPCs will not be available to receive fuel until at least 1998 (this date should also apply to any DOE-fuel MPCs). Current DOE National Program plans indicate that much of the SNF now in temporary storage at DOE facilities will soon be placed in canisterized dry interim storage. If the DOE fuel is placed in interim dry storage prior to placement in MPCs, much cost and storage duplication may result, and the infrastructure study herein will no longer apply.

The DOE-fuel MPCs and OCRWM MPCs share a potential disadvantage because they are designed to be permanently sealed with large, full-penetration groove welds. These welds would have to be cut out and the lids discarded if future content inspection or characterization is required. This would add much rework, increase labor and material costs, and increase personnel radiation exposure. This problem is currently being considered in the OCRWM MPC program.

9. CONCLUSIONS

9.1 MPC and DOE-Fuel Interface

More than 95% of the volume of DOE fuel (not including the fuel at SRS currently being considered for reprocessing) can be accommodated by use of one large commercial BWR MPC design and two new MPC designs. The commercial BWR MPC can be used for the ATR and MTR-type fuel assemblies. Therefore, the ATR and MTR-type fuel assemblies can be handled by the OCRWM MPC design for which licensing is currently planned. Each large commercial BWR MPC can hold 120 ATR fuel assemblies, or 120 MTR-type fuel assemblies.

The N Reactor fuel canisters, FSV fuel assemblies, and TMI-2 canisters require new MPC designs. The FSV/TMI-2 MPC will be 19.25 in. shorter and 0.5 in. thicker than the large OCRWM MPC and will be fitted with a new hexagonal-grid basket. Each of these FSV/TMI-2 MPCs holds 35 FSV graphite blocks or seven TMI-2 canisters. The N Reactor MPC will be 22.62 in. shorter and 0.5 in. thicker than a large OCRWM MPC and will be fitted with a new rectangular-grid basket. Each of these N Reactor MPCs holds 30 N Reactor canisters, which equates to 420 N Reactor fuel assemblies. To license the two new MPC designs, the original OCRWM MPC license would have to be amended.

It is feasible to use the MPC concept for storage and transportation of all six DOE fuel-assembly types studied in this report. However, MPCs containing ATR and MTR-type fuels may not meet the MGDS criticality disposal criteria unless long-term immobilization techniques are employed. If the ATR and MTR-type fuels are not suitable for the MPC system because of this long-term disposal criteria, the volume of DOE fuel that can be handled by MPCs may be reduced from 95% to approximately 85% of the total inventory.

This report recommends that the ATR and MTR-type aluminum fuels be sealed in SS cans shown in Figure 15 to provide an additional level of containment. This is assumed to allow the aluminum-clad fuel to operate at the same storage and disposal temperatures as have been established for zircaloy-clad fuel. The cans will also protect the fuel from inertial transportation loads since the assemblies are stacked three high in the MPCs.

All of the DOE-fuel MPCs can be placed in the same shielded transportation, storage, or disposal overpack casks that will be used for large OCRWM MPCs, and therefore, transportation and MGDS operations would be the same for the DOE-fuel MPCs as for the large OCRWM MPCs. Since the two new DOE-fuel MPC designs are shorter than the OCRWM MPCs by approximately 20 in., the former will require that inexpensive spacers be placed inside the overpack casks after the MPCs are inserted.

The fuel data in Table 2 were compiled to perform the Reference 2 feasibility study. As mentioned above, values in Table 2, listed as "Not Available," were assumed. These assumptions and the issues in Section 8 should be further investigated during the next phase of DOE-fuel MPC design.

The conclusion of this MPC/DOE fuel interface evaluation is that, for most of the DOE fuel, no impediments were found that preclude using the MPC system for DOE fuel storage, transportation, and disposal. However, open issues have been identified. The cost and safety advantage that the OCRWM MPC affords to commercial spent nuclear fuel disposition should also be able to be realized in the disposition of most of the DOE fuel. A more detailed, conceptual design study should be initiated to further evaluate this concept and its open issues. This would also allow the DOE Spent Nuclear Fuels program to correlate its activities with the Civilian Radioactive Waste Management program.

9.2 Storage Facility Interface

The storage-site infrastructure for handling and loading the 125-ton MPCs was studied. Table 5 summarizes the results of this infrastructure study. The current DOE storage facilities for the TMI-2 canisters and the ATR cut fuel assemblies can handle and transport the 125-ton MPCs with minor upgrades and a purchased or leased 150-ton cask transporter trailer. To handle the FSV 125-ton MPC, the IFSF Storage Facility 001 crane must be upgraded from 60 to 101 tons, and the handling cave transfer car must be modified and upgraded from 100 to 101 tons. Other options are available, but probably would result in higher overall costs. The N Reactor fuel storage facility must have two 30-ton cranes replaced with two 114-ton cranes to handle the N Reactor 125-ton MPC. Other options are available but probably would result in higher overall costs. The two SRS RBOF 50-ton cranes would probably have to be brought up to current standards to handle the MTR-type 125-ton MPC with its under-the-hook weight of 93 tons. New dry-canning systems would be required at the ICPP IFSF Building 666 storage pool for canning the ATR fuel, and at the SRS RBOF for canning the MTR-type foreign fuel. According to Reference 1, the OCRWM MPC program will have portable automatic welding systems available for seal welding the MPCs. The DOE-fuel MPCs could either use this commercial equipment or purchase a dedicated welding system, which could serve all the DOE-fuel MPC sites.

To put these required facility upgrades in perspective, the most expensive of them should cost no more than that of several MPCs. Therefore, they are not considered prohibitive to using the 125-ton MPCs for disposition of DOE fuel.

9.3 Quantities and Cost Estimate

A total of 420 MPCs are required to handle all DOE fuel assemblies of the six types studied in this report. These 420 MPCs include 60 large commercial BWR MPCs, 113 FSV/TMI-2 MPCs, and 247 N Reactor MPCs at a total cost of $170.8M.

Table 5. DOE facility requirements for loading and handling the DOE 125-ton MPC.

		125-Ton MPC Loading Method	
Fuel Assembly Type	Fuel Assembly Location	Direct MPC Loading at Fuel Storage Pool	MPC Loading at Onsite, Dry-Cask Fuel Transfer Facility
N Reactor	Hanford 105-k Basins	P	-
FSV	ICPP IFSF Graphite Storage Facility	P	A
TMI-2	TAN 607 Pool	-	P
ATR	ICPP Building 666 Pool	P	A
MTR	SRS RBOF Basins	P	A

Key: P = Preferred option
A = Alternate option

10. RECOMMENDATIONS

The following is a summary of future tasks recommended in this report for the next phase of this work:

- Initiate conceptual design on the three DOE-fuel MPC preconceptual designs proposed in the report.
- Perform further analysis of structural, criticality safety, shielding, and thermal performance of the DOE-fuel MPC system, especially areas based on comparisons of DOE fuel with commercial LWR fuel, but for which no specific analysis was done.
- Investigate possibility of obtaining canning equipment for MTR-type foreign fuel at the SRS RBOF.
- Further investigate allowable storage and disposal temperature limits for ATR and MTR-type aluminum-clad fuels.
- Investigate using the MPC system for DOE fuels other than the six types investigated in this report.
- Study the potential for MGDS acceptance of HEU fuels, aluminum-clad fuels, N Reactor fuel, TMI-2 core debris, and FSV fuel.
- Study transportation licensing issues associated with transporting HEU fuel, aluminum-clad fuel, and N Reactor fuel off the DOE sites.
- Investigate potential for N Reactor fuel canister water removal and drying.
- Investigate the potential TMI-2 core debris radiolytic gas generation from water entrapped in the licon following canister drying, as it applies to overpressuring the MPC.
- Evaluate the structural integrity of the individual TMI-2 and N reactor canisters to withstand 10 CFR 71 accident inertial transportation loads.
- Expand the study of the DOE storage facility infrastructure for handling and loading the DOE-fuel MPCs, and the required modifications and upgrades to those facilities.
- Investigate use of small 25-ton MPCs and medium 75-ton MPCs for facilities where the large 125-ton MPCs cannot meet the requirements for DOE fuel.
- Investigate immobilization of the ATR and MTR-type fuels by injection of stable media into MPCs loaded with fuel for reducing the long-term MPC degradation criticality concerns.

11. REFERENCES

1. *A Preliminary Evaluation of Using Multipurpose Canisters Within the Civilian Radioactive Waste Management System*, Revision 0, Contract DE-AC01-91RW00134, March 24, 1993, Doc. A00000000-AA-07-00002, TRW Environmental Safety Systems, Inc.

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3. *Multipurpose Canister (MPC) Implementation Program Conceptual Design Phase Report, Volume I - MPC Conceptual Design Summary Report*, Final Draft, Contract DE-AC01-91RW00134, September 30, 1993, Doc. A20000000-00811-5705-00001, WBS: 3.1.07, TRW Environmental Safety Systems, Inc.