Task Order 17 – Cask Design Study Final Report

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DISCLAIMER

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LIST OF ACRONYMS

AAR	Association of American Railroads
ALARA	As Low as Reasonably Achievable
ANSI	American National Standards Institute
ANSYS	Analysis System
APSRA	Axial Power Shaping Rod Assemblies
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BPRA	Burnable Poison Rod Assembly
BWR	Boiling Water Reactor
C	Celsius
CE	Combustion Engineering
CFR	Code of Federal Regulations
cg	center of gravity
Ci	Curie
CRA	Control Rod Assemblies
DFC	Damaged Fuel Canister
DOE	Department of Energy
DOT	Department of Transportation
DSC	Dry Shielded Canisters
F	Fahrenheit
FA	Fuel Assembly
FANP	Framatome Advanced Nuclear Power
FQT	Fuel Qualification Table
GWd/MTU	gigawatt days per metric ton of uranium
HAC	Hypothetical Accident Conditions
HB	High burnup
Hr	Hour
ID	Internal Diameter
ISG	Interim Staff Guidance
J	Joule
K	Kelvin (degree)

kPa	kilopascals
kW	kilowatt
m	meter
MCNP	Monte Carlo N-Particle
MMC	Metal Matrix Composite
MNOP	Maximum Normal Operating Pressure
mol	mole
NAS	nickel-alloy steel
NCT	Normal Conditions of Transport
NFAH	Non Fuel Assembly Hardware
NRC	Nuclear Regulatory Commission
NSA	Neutron Source Assemblies
OD	Outside Diameter
OM	Operations Manual
ORA	Orifice Rod Assemblies
psi	pounds per square inch
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCCA	Rod Cluster Control Assemblies
SAR	Safety Analysis Report
SFP	Spent Fuel Pool
SNF	Spent Nuclear Fuel
SOW	Statement of Work
SRP	Standard Review Plan
TBD	To Be Determined
TBV	To Be Verfied
Tndt	nil-ductility transition temperature
TO	Task Order
TPA	Thimble Plug Assemblies
UNF	Used Nuclear Fuel
U.S.	United States
USL	Upper Subcritical Limit
VDS	Vacuum Drying System

VSI	Vibration Suppression Insert
WE	Westinghouse
wt%	weight percent

EXECUTIVE SUMMARY

Under Task Order 17 of the industry Advisory and Assistance Contract to the Department of Energy (DOE) DE-NE0000291, the AREVA Team has provided a conceptual design for a reusable transportation cask (the 6625B-HB) capable of transporting BWR and PWR used nuclear fuel (UNF) assemblies, including high burnup UNF. These assemblies can be shipped either as bare fuel or fuel loaded into damaged fuel canisters (DFCs). The 6625B-HB cask has been designed with reasonable assurance it can be licensed by the Nuclear Regulatory Commission (NRC) under 10 CFR 71, fabricated within existing facilities, used by most utilities, and transported by rail. The level of detail is intended to provide support to the analyses and planning activities DOE is performing to support the waste management system. These details include cask characteristics (e.g., capacity, dimensions, and masses), characteristics of the UNF that can be shipped in this cask system, estimates of costs of this cask system, an assessment of operational activity durations and associated cumulative doses for loading and unloading this cask system, and identification of potential limitations or anticipated licensing considerations of relevance for this cask system.

The 6625B-HB is capable of handling a large range and size of UNF assemblies, including the restrictive short cooled, high burnup UNF assemblies (> 45 GWd/MTU and at least 5 years cooled). The 6625B-HB contains up to 24 PWR or 61 BWR UNF assemblies placed into baskets designed for holding either bare fuel or fuel packaged into DFCs. The designed transportation package of the 6625B-HB consists of the payload baskets, a lead-shielded cask body, an inner closure lid with lead and steel gamma shielding, an outer steel lid, upper end structure, lower end structure with lead and steel gamma shielding, and upper and lower impact limiters. To ensure reusability, the lids are bolted to the cask body, and the BWR and PWR baskets are designed to be interchangeable.

The design of the 6625B-HB took into consideration data collected from the spent fuel pools (SFPs) at the six operating Duke Energy reactor sites. This collected data includes characteristics of the SFP inventory (e.g., burnup, years cooled), duration of activities associated with loading of dry storage casks/canisters, weight capacity of cranes and floors, design information for yokes, and estimation of damaged and failed UNF quantities. This data was used to inform several of the activities performed for this task order, including the evaluation of: (1) the ability to efficiently perform loading operations of the 6625B-HB; (2) the effectiveness of the 6625B-HB for off-loading the contents of SFPs for both PWR and BWR UNF; (3) alternative loading patterns for UNF into the existing basket structure of the 6625B-HB; and (4) the quantity of DFCs containing damaged UNF that the 6625B-HB should be conservatively designed to load.

The analyses performed to model the 6625B-HB included: shielding, criticality (including some burnup credit studies), structural, and thermal. An iterative process between these analyses occurred to ensure the conceptual design met specified regulatory and task-specific criteria related to weight limitations (gross hook weight ≤ 125 tons), size limitations (diameter ≤ 128 inches), thermal limits (accessible surface ≤ 185°F), and dose rate limits (normal condition ≤ 10 mrem/hr at 2 m), while maintaining structural integrity, operational flexibility, processing throughput rates, reasonable assurance for licensing, reasonable cost structures, and the ability to fabricate in existing facilities. The principle iterations in this process involved the most limiting of these criteria, which included (from most restrictive), the weight, the thermal limits, and the shielding limits.

This report presents the design details of the 6625B-HB, which includes dimensional and component and overall weight information. The first portion of the report is aligned with the chapters of a Safety Analysis Report for transportation packages to ensure information required for a license application is identified, and to provide a basis for a reasonable assurance the 6625B-HB could be licensed with the NRC. The following portions of the report include: (1) a cost basis for the 6625B-HB cask, DFCs, and auxiliary loading equipment; (2) the concept of operations that includes operational steps and associated estimated person hours and doses for loading and unloading activities of the 6625B-HB; (3) the planning and performing of NRC required maintenance and maintenance required to support operations; (4) a listing of key information of the 6625B-HB cask system; and (5) a listing of how the cask meets certain task order requirements. Three trade studies are also included in the report and examine the impact of: (1) using DFCs to package UNF over packing bare UNF into the basket on the 6625B-HB; (2) increasing the length of the 6625B-HB to allow for the inclusion of all UNF existing in the U.S.; and (3) loading damaged and failed UNF into the 6625B-HB. The report includes a special features section that identifies items that could be introduced into the design of the 6625B-HB and/or into its operations to allow for system optimization. These features have the potential to increase cask capacity, reduce costs, reduce maintenance activities, increase operational efficiency, and reduce cumulative doses, and depending on future objectives of the waste management system may merit consideration for future work.

Based on the design of the 6625B-HB and the collected data from the Duke Energy SFPs, the 6625B-HB could be an ideal cask to transport UNF from these SFPs with minimal impact to existing operations at the operating reactor sites. However, there is one significant limitation to utilizing the 6625B-HB at the Duke Energy SFPs, many of the SFPs are currently not capable of handling a 125 ton cask due to crane and/or floor capacity limits. Thus, a thorough investigation into establishing crane and floor loading capacities (actual and administratively controlled limits), as well as door sizes that may pose limitations for these cask systems, for U.S. SFPs is recommended.

In summary, the AREVA team has developed a conceptual design of a reusable, rail transportation cask package (the 6625B-HB) capable of shipping high burnup PWR and BWR UNF that can be placed either bare into the basket or packaged into a DFC, which is then placed into the basket. Considering the AREVA team's significant experience in developing NRC licensed transportation cask packages and considering the elements of those packages included in the design the 6625B-HB, there is a reasonable assurance the 6625B-HB can be licensed with the NRC for the transport of the range of BWR and PWR UNF specified in this report, including short-cooled, high burnup UNF.

1.0 INTRODUCTION & OVERVIEW

This report, the Cask Design Study Draft Final Report, summarizes the work performed on Task Order 17 "Spent Nuclear Fuel Transportation Cask Design Study" as defined under Subtask 6 of the Statement of Work (SOW). Results for cask concepts under study and results of supporting analyses are presented in this report and the cumulative work presented in this report represents the 90-percent-completion point of the task.

This report provides details of a conceptual cask system designed to be used for rail transport of used nuclear fuel (UNF), including high burnup UNF. This conceptual cask system utilizes bolted inner and outer lids that allow for it to be reused and credited for moderator exclusion under specific conditions. By use of four different basket designs, this cask system is designed and optimized to be capable of handling the following UNF arrangements:

- Bare pressurized water reactor (PWR) UNF
- Bare boiling water reactor (BWR) UNF
- PWR UNF placed in damaged fuel canisters (DFCs)
- BWR UNF placed in DFCs

In this study, DFCs are analyzed primarily for intact fuel; however, in the trade studies in *Section 8*, the packaging of some specific forms of damaged fuel (e.g., fuel with an expanded pitch) in the DFCs is assessed.

Table 1-1 maps the items/activities requested in *Section 2* of the SOW to the sections of this report where this material can be found.

Table 1-1: SOW Item/activity Map to report sections

Item/ Activity	Description	Report Section(s)
1	Develop a reusable SNF rail-type transportation cask system design concept optimized for transport of intact bare (not canistered) SNF.	2.1.1, 2.1.2
2	Develop a reusable SNF rail-type transportation cask system design concept optimized for transport assuming all SNF assemblies are in DFCs.	2.1.1, 2.1.2
3	For each design concept described in items 1 and 2 above, develop estimates of the up-front costs associated with design, analysis, testing, and licensing the cask and of the cost to fabricate the entire transport cask system, including cask internals and impact limiters.	3.0
4	For each design concept described in items 1 and 2 above, develop a concept of operations, including assessments of the time and motion required for loading the fuel at the reactor pools and unloading from the transport casks.	2.7, 4.0
5	Identify equipment maintenance requirements including testing, maintenance, and performance requirements for structures, systems and components (SSCs) important to safety.	2.8, 5.0
6	Provide additional key information associated with each of the SNF transportation cask system design concepts, including information on dimensions, component masses, total mass for both fully loaded and unloaded conditions. maximum thermal loading, and estimated dose rates during normal conditions of transport.	2.1.2 (dimensions & masses) 2.3 (thermal loading) 2.5 (dose rates) 6.0 (general)

Item/ Activity	Description	Report Section(s)
7	Cask System Requirements: In addition to providing reasonable assurance that the cask concepts would be capable of meeting 10 CFR 71 requirements, the casks system must be able to meet the following requirements:	7.0 (general)
	a) The system design concept, including impact limiters, will have a maximum width of 128 inches. The cask design concept, including impact limiters, shall not be wider than this maximum railcar width.	2.1.1, 2.1.2, 2.2.3
	b) The system must allow for the transportation of high-burnup fuel (>45GWd/MTU) with a target of transporting fuel with an average assembly burnup of up to 62.5GWd/MT with up to 5.0 wt% enrichment and out-of-reactor cooling time of 5 years.	2.1.2, 2.5.2
	c) Reasonable assurance that the design concepts can accommodate essentially the entire existing and future inventory of commercial light-water reactor SNF must be provided, without undue penalty (e.g. reduced cask capacity resulting in sub-optimization for the majority of anticipated shipments). Specific fuel designs or attributes (e.g. fuel length, assembly decay heat limits, or burnup limits) not allowed by the cask design concepts must be identified.	8.0
	d) In addition to the NRC's regulations, design activities shall also consider applicable regulatory guides and recent licensing experience and actions related to transportation cask design, fabrication, and operations.	2.1.1, 2.7
	e) The cask system for DFCs in all positions will place constraints on capacity due to the size of the DFCs. The design concepts should satisfy all appropriate regulatory and operational limits, while maximizing capacity.	2.2.1, 8.1
	f) The transportation casks shall be capable of being closed and reopened multiple times, so the cask can be reused for many shipments. The method for closing and reopening shall be described. Factors limiting the possible number of times that the cask can be reused shall be identified, along with possible means for extending life and reusability of the casks.	2.1.2, 2.2.2, 2.4.1, 2.7, 2.8.1
	g) The loaded and closed DFCs shall also be capable of being reopened, to allow assembly repackaging, and the method for reopening shall be described.	2.1.2
	h) Consistent with current industry designs, the DFCs shall be vented at the top and bottom.	2.1.2
8	To cover the trade space between the two design concepts described in items 1 and 2 above, a study to assess how important cask attributes, such as capacity and cost, are expected to vary as the number of assemblies in DFCs which the cask must be able to accommodate is varied.	8.0
9	Consideration of any special features which could be introduced into the cask design concepts which would allow for optimization, such as increased capacity, reduced cost, and/or reduced maintenance shall be explored by the Contractor, the results of which are to be made available in the final report.	9.0

This report is divided into eight primary sections plus an introduction, conclusion, reference list, and two appendices. *Section 2.0* of this report is aligned with the layout of a transportation cask Safety Analysis Report (SAR) as established in the Standard Review Plan (SRP) for transportation packages (NUREG-1617). This ensures information addressed in a SAR by an applicant to the Nuclear Regulatory Commission (NRC) is included in this report and, combined with the knowledge that this conceptual cask design utilized materials and analyses methods with a previous licensing precedent, will provide reasonable assurance the cask system designed under this Task Order (TO) (the 6625B-HB) can be licensed by the NRC. Actual licensing of this conceptual cask system by the NRC would require completion of detailed design calculations, fabrication drawings, testing, etc. that are outside the scope of this TO.

Sections 3.0 through *9.0* are aligned directly with items/activities identified in the SOW. *Section 3.0* provides information on the estimated cost ranges for the 6625B-HB. *Section 4* takes the operations described in *Section 2.7* and performs a time-dose study to estimate total doses, total person-hours, total duration/clock time, and total number of shifts for preparation, loading, and unloading operations for PWR and BWR UNF in both bare and in DFC configurations. *Sections 5, 6,* and *7* cover material specifically requested in the SOW for material that is mostly covered in *Section 2*. *Section 8* contains three trade studies on: (1) the impacts DFCs have on the 6625B-HB design; (2) the impact fuel length has on the 6625B-HB design; and (3) the ability of the 6625B-HB to load damaged and potentially failed UNF (as defined in *Section 2.1*).

Appendix A documents exercises performed with Duke Energy using the transportation cask system designed in this TO, the 6625B-HB. These exercises involved examining the ability of the 6625B-HB transportation cask system to unload Duke Energy’s spent fuel pools (SFPs) at a rate sufficient to eliminate the need to move any additional UNF into onsite dry storage, while at the same time trying to minimize any increase in the duration necessary to load cask systems (storage or transportation) at the SFP, and to characterize the fuel in the SFPs to support elements of the design of the 6625B-HB (e.g., fuel heat loading patterns). The results from these exercises informed several of the items/activities performed for this TO including the evaluation of: (1) the ability to efficiently perform loading operations of the 6625B-HB; (2) the effectiveness of the 6625B-HB for off-loading the contents of SFPs for both PWR and BWR fuel; (3) alternative loading patterns for UNF into the existing basket structure of the 6625B-HB; and (4) the quantity of DFCs containing damaged UNF that the 6625B-HB should be conservatively designed to load.

Appendix B contains a cask data template supplied to the AREVA team by the DOE used to support waste management system analysis studies. The AREVA team filled out the template for the conceptual cask systems designed in this Study.

The conclusion to this report provides a summary of some of the important attributes of the four cask designs established by this TO and also contrasts the bare UNF cask designs against the cask designs that place the UNF into DFCs.

2.0 DEVELOPMENT OF A CONCEPTUAL REUSABLE UNF RAIL-TYPE TRANSPORTATION CASK SYSTEM

2.1 General Information

This section presents a general introduction and description of the 6625B-HB[1] package conceptual design for DOE TO17, *Spent Nuclear Fuel Transportation Cask Design Study*. The 6625B-HB package is a reusable rail cask designed to transport UNF. This report demonstrates reasonable assurance the 6625B-HB cask design has the capability to meet the fundamental licensing requirements as a Type B(U)F–96 shipping container in accordance with the provisions of Title 10, Part 71 of the Code of Federal Regulations (10 CFR 71) [1].

The major components comprising the package are discussed in *Section 2.1.2.1* and are illustrated in **Figure 2.1-1, Figure 2.1-2, Figure 2.1-3, Figure 2.1-4, Figure 2.1-5,** and **Figure 2.1-7**.

2.1.1 Package Design Information

The 6625B-HB packaging has been developed as a rail cask to transport bare undamaged UNF or UNF contained within DFCs. The cask is designed to accommodate essentially the entire existing and future inventory of commercial light-water reactor UNF (see *Section 8.2*). Both BWR and PWR UNF are considered. Within the packaging, bare fuel or fuel in DFCs is contained in basket structures, specifically designed for each fuel type, which provide for heat rejection and criticality control.

The packaging consists of a payload basket, a lead-shielded cask body, an inner closure lid with lead and steel gamma shielding, an outer steel lid, upper end structure, lower end structure with lead and steel gamma shielding, and upper and lower impact limiters. The packaging is of conventional design and utilizes American Society of Mechanical Engineers (ASME) alloy steel as its primary structural material. The packaging is designed to provide leaktight containment of the radioactive contents under all Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC).

The 6625B-HB packaging is designed for direct loading in a commercial nuclear power plant's fuel pool. The package is designed to be transported singly, with its longitudinal axis horizontal, by rail or highway truck as an exclusive use shipment. When loaded and prepared for transport, the 6625B-HB package can contain up to 24 PWR or 61 BWR UNF assemblies, is 261.5 inches long, 126 inches in diameter (over the impact limiters), and has a nominal weight of 151.1 tons.

2.1.1.1 Applicable Regulatory Requirements

The 6625B-HB packaging is designed as a Type B(U)F–96 shipping container in accordance with the provisions of 10 CFR 71 [1]. Some key regulations are listed in the following sections. In addition, a discussion of the planned licensing approach is included.

[1] The naming convention applied to this cask is consistent with that performed by AREVA for other projects (mainly for DOE): "6625" is for the cask cavity inner diameter (66.25 inches), "B" to identify this as a Type B(U)F-96 shipping container per 10 CFR 71, and "HB" for high burnup. Thus, this cask is a 66.25-inch shipping container meeting Type B requirements of 10 CFR 71, and designed to handle high burnup UNF.

2.1.1.1.1 Thermal Regulations

10 CFR §71.71 requires that the package components shall remain within their respective temperature limits under the NCT.

10 CFR §71.73 requires that the package shall be shown to retain sufficient thermal protection following the HAC free and puncture drop scenarios to maintain all package component temperatures within their respective short term limits during the regulatory fire event and subsequent package cool-down.

10 CFR §71.43(g) requires the maximum temperature of the accessible package surfaces shall be less than 185°F for the maximum decay heat loading, with an ambient temperature of 100°F, and no insolation.

2.1.1.1.2 Containment Regulations

The release of radioactive material from a Type B package shall not exceed the values specified in 10 CFR 71, as follows:
Under NCT, there would be no loss or dispersal of radioactive contents as demonstrated to a sensitivity of 10^{-6} A_2 per hour.

Under HAC, there would be no escape of Kr exceeding 10 A_2 in 1 week, and no escape of other radioactive material exceeding a total amount A_2 in 1 week.

2.1.1.1.3 Shielding Regulations

10 CFR 71.47(b)(3) limits the NCT dose rate to 10 mrem/hr at a distance of 2 m from the vehicle. This is typically the limiting NCT dose rate location.

10 CFR 71.47(b)(1) limits the NCT dose rate to 200 mrem/hr on the surface of the package. A personnel barrier will be used at the approximate radius of the impact limiter so that the outer surface of the package is not close to the cask body.

10 CFR 71.47(b)(2) limits the NCT dose rate to 200 mrem/hr on the surface of the vehicle. Because the package is large (approximately the same size as the vehicle), the vehicle surface dose rate is essentially the same as the package surface dose rate.

10 CFR 71.51(a)(2) limits the HAC dose rate to 1000 mrem/hr at a distance of 1 m from the surface of the package.

2.1.1.1.4 Criticality Regulations

10 CFR 71.55(b): The contents of the package must be subcritical assuming optimum moderation with fresh water with the contents in their as-loaded condition.

10 CFR 71.55(d): For NCT, the package may be assumed to be dry as it is leaktight under normal conditions.

10 CFR 71.55(e): For HAC, the condition of the fuel is unknown due to the limited properties of the cladding for high burnup UNF.

2.1.1.1.5 Package Licensing Approach

The 6625B-HB packaging is designed to transport both low and high burnup fuel assemblies. To address concerns with demonstrating compliance with the requirements of 10 CFR 71, the licensing approach employed is described below. This licensing approach and analytical evaluations presented in this report ensure adequacy of the structural, thermal, containment, shielding, and criticality design features of the 6625B-HB transport package.

2.1.1.1.6 Burnup Credit and Moderator Exclusion

To meet the requirements of 10 CFR §71.55(b) (as-loaded fuel condition with fresh water intrusion), burnup credit will be required for PWR UNF. Burnup credit is not required for BWR UNF. Taking credit for burnup reduces the system reactivity due to depletion of fissile material and growth of fission product poisons. To meet the requirements of 10 CFR §71.55(d) (packages under NCT), burnup credit is not required because the package is assumed to be dry, as it is leaktight under NCT. To meet the requirements of 10 CFR §71.55(e) (packages under HAC), moderator exclusion (not burnup credit) is used as the licensing basis because the condition of the fuel is unknown. Burnup credit is not required because moderator exclusion results in a low reactivity. To ensure a robust design, 'defense-in-depth' cases are also performed in support of 10 CFR §71.55(e). In the defense-in-depth cases, reasonable fuel damage is assumed with fresh water moderation and burnup credit is applied. For the defense-in-depth cases, the upper subcritical limit (USL) may be based upon an administrative margin of 0.02 (USL ~ 0.98; an administrative margin of 0.05 is used under all other conditions).

Moderator exclusion is employed as a licensing basis to demonstrate compliance with the sub-criticality requirements of 10 CFR §71.55(e). The guidance and criteria provided in Interim Staff Guidance 19 (ISG-19) [2] are employed for this purpose. Specifically, 10 CFR §71.55(e)(2) states that to demonstrate sub-criticality under HAC, it must be assumed that "water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents." ISG-19 establishes criteria under which it is possible to demonstrate that the worst-case damaged condition of the package does not result in water in-leakage. This allows the HAC criticality calculations that form the licensing basis to be performed assuming there is no water in-leakage.

2.1.1.2 Transport Index

The 6625B-HB cask will be transported by exclusive use and the transport index is not applicable.

2.1.1.3 Criticality Safety Index

Based on the criticality assessment provided in *Section 2.6, Criticality Review*, the criticality safety index for the 6625B-HB package is zero.

2.1.2 Packaging Description

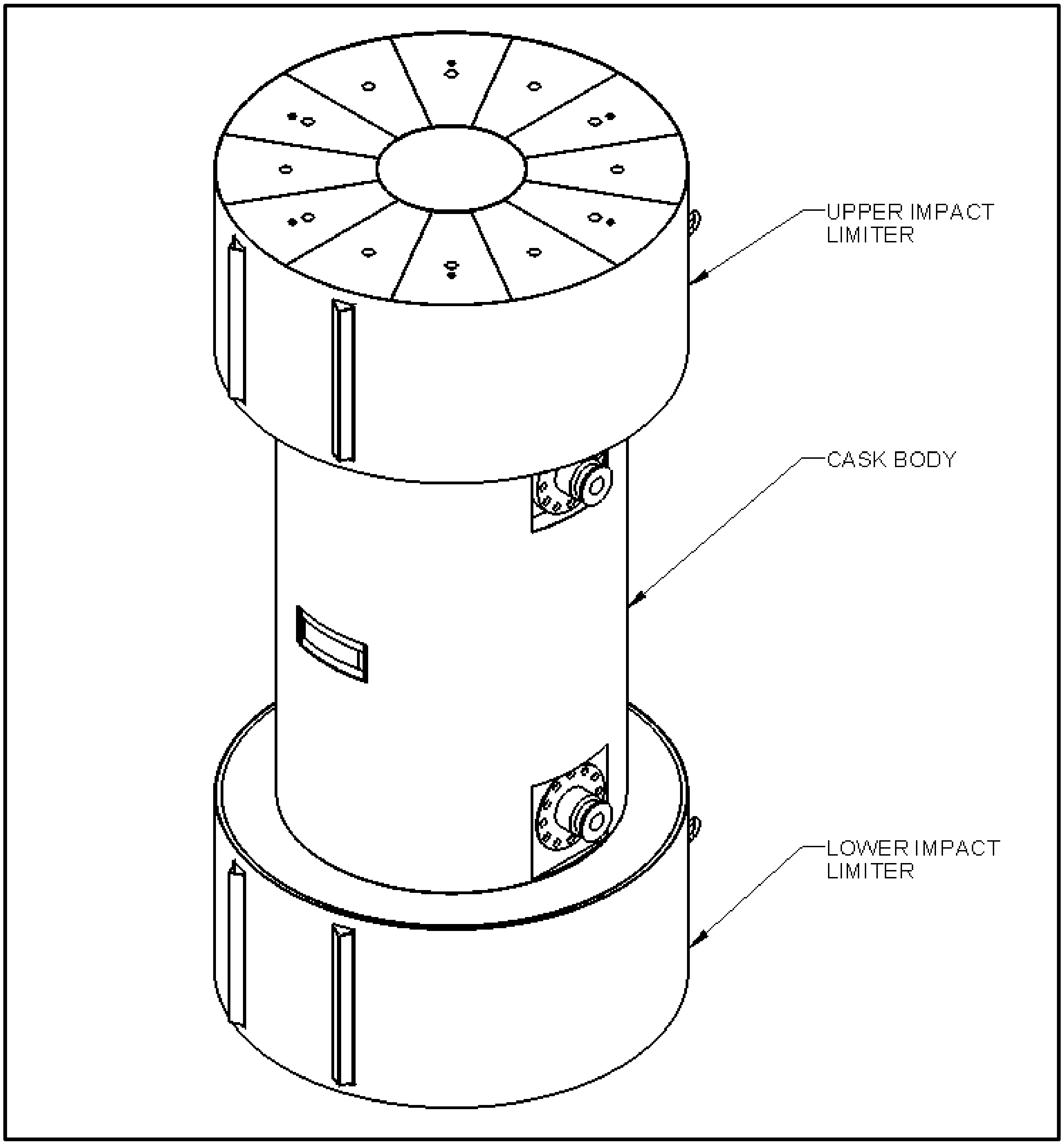
2.1.2.1 Packaging

The AREVA 6625B-HB packaging can be used to transport several types of BWR fuel assemblies with or without fuel channels or PWR fuel assemblies with or without non-fuel assembly hardware. The fuel assemblies are contained in a single BWR or PWR basket. The packaging can accommodate bare fuel assemblies or fuel contained within damaged fuel cans. The 6625B-HB packaging is designed for a maximum decay heat load of 30.4 kW. The actual decay head load depends on the contents being transported and the loading configuration. The UNFs that may be transported in the 6625B-HB packaging are presented in *Section 2.1.2.2*. The 6625B-HB packaging, shown in **Figure 2.1-1**, consists of the following components:

- A 6625B-HB cask consists of a containment boundary, structural shell, gamma shielding material, and solid neutron shield. The containment boundary consists of a cylindrical inner shell, a bottom closure plate, an upper end structure, a top inner lid with innermost seals and closure bolts, vent and drain ports with closure bolts and seals, and containment welds. A redundant mechanical closure is provided by the outer lid with outermost seals and closure bolts, and the vent port with closure bolts and seals. The transport cask cavity also contains an inert gas atmosphere (helium).

- Sets of removable front and rear trunnions, which are bolted to the outer shell of the cask, provide support, lifting, and rotation capability for the 6625B-HB cask.
- Impact limiters consisting of balsa and redwood encased in stainless steel shells are attached to each end of the 6625B-HB cask during shipment. A thermal shield is provided between each impact limiter and the cask to minimize heat transfer to the impact limiters. Each impact limiter is held in place by 12 attachment bolts.
- A personnel barrier is mounted to the transport frame to prevent unauthorized access to the cask body.
- The 6625B-HB cask includes a fuel basket assembly, located inside the cask cavity. The basket assembly locates and supports the fuel assemblies (or damaged fuel cans), transfers heat to the cask wall, and provides neutron absorption to satisfy nuclear criticality requirements. The PWR fuel basket assembly will contain 24 bare fuel assemblies or DFCs containing damaged fuel assemblies, and the BWR fuel basket assembly will hold 61 bare fuel assemblies or DFCs containing damaged fuel assemblies.

FIGURE 2.1-1: 6625B-HB PACKAGING COMPONENTS



2.1.2.1.1 6625B-HB Transport Cask

The 6625B-HB cask is fabricated primarily of nickel-alloy steel (NAS). Other materials include the cast lead shielding between the containment boundary and inner shell, the elastomer O-ring seals (Viton O-rings), the borated resin neutron shield, and the carbon steel closure bolts. Socket-headed cap screws (bolts) are used to secure the inner and outer lids to the cask body. The body of the cask consists of a 1.25-inch thick, 66.25-inch inside diameter, NAS inner (containment) shell and a 2.75-inch thick, 80.25-inch outside diameter, NAS structural shell which sandwich the 3.00-inch thick, cast lead shielding material. Lead shielding is also located in the lower end structure and in the inner lid.

The overall dimensions of the 6625B-HB packaging are 261.50 inches long and 126.00 inches in diameter with both impact limiters installed. The transport cask body is 200.50 inches long and 80.25 inches in diameter. The cask diameter including the radial neutron shield is 93.25 inches. The length of cask cavity is 182.00 inches and 66.25 inches in diameter. Detailed design drawings for the 6625B-HB packaging are provided in *Section 2.1.5*. **Table 2.1-1** summarizes the nominal dimensions of the 6625B-HB packaging components.

The maximum gross weight of the loaded package is 150.1 tons including a maximum payload of 20.6 tons. A summary of overall component weights is shown in **Table 2.1-2**. The yoke weight was chosen to be 7,500 lbs. This is reasonable based on past 125 ton adjustable hook yoke designs. Heavier yokes have been used in the past but newer designs have proven that lighter options are feasible. Some designs use Kevlar straps in the place of solid arms to drastically reduce weight depending on the site requirements. See Appendix A2 for a table showing the current yoke weights utilized by Duke Energy reactors. Trunnions, attached to the cask body, are provided for lifting and handling operations, including rotation of the packaging between the horizontal and vertical orientations. The 6625B-HB packaging is transported in the horizontal orientation, on a specially designed shipping frame, with the lid end facing the direction of travel. The basket, loaded with a used fuel payload, is shipped dry in a helium atmosphere. Both the transport cask inner cavity and the secondary containment cavity are filled with helium. The heat generated by the UNF assemblies is rejected to the environment by conduction, convection, and radiation. No active cooling is required.

TABLE 2.1-1: NOMINAL DIMENSIONS OF THE 6625B-HB PACKAGING

Nominal Dimensions	Inches
6625B-HB packaging overall length with impact limiters	261.50
6625B-HB packaging overall length without impact limiters	200.50
6625B-HB cask impact limiter outside diameter	126.00
6625B-HB cask outside diameter (without impact limiters)	93.25
6625B-HB cask outside diameter (without impact limiters and neutron shield)	80.25
6625B-HB cask cavity inner diameter	66.25
6625B-HB cask cavity length	182.00
6625B-HB cask inner shell radial thickness	1.25
6625B-HB cask lead gamma shielding radial thickness	3.00
6625B-HB cask body outer shell radial thickness	2.75
6625B-HB cask inner lid thickness	3.00
6625B-HB cask inner lid lead gamma shielding and steel box thickness	3.50
6625B-HB cask outer lid thickness	2.50
6625B-HB cask bottom closure plate thickness	1.25
6625B-HB cask bottom lead gamma shielding thickness	4.50
6625B-HB cask bottom lower end structure thickness	2.75
6625B-HB cask resin and copper box neutron shield height (center)	6.25
6625B-HB cask resin and copper box neutron shield height (ends)	5.25

TABLE 2.1-2: OVERALL WEIGHTS OF THE 6625B-HB PACKAGING

Nominal Weight (lbs.)	PWR	BWR
Weight of Fuel Assembly (maximum)	1,715	705
Weight of DFC	55	31
Total fuel and DFC weight (24 PWR, 61 BWR)	42,480	44,896
Cask body weight	147,682	
Cask inner lid	8,021	
Cask outer lid	3,563	
Empty basket	39,222	30,755
Loaded basket (total contents)	81,702	75,651
Empty cask (cask body, inner lid)	155,703	
Loaded cask - without outer lid	237,405	231,355
Water in cask cavity	6,452	11,304
Lifting yoke weight	7,500	
Hook weight 1 - filled with water and inner lid installed	251,357	250,159
Hook weight 2 - water removed and inner lid installed	244,905	238,855
Hook weight 3 - water removed and inner and outer lid installed	248,468	242,417
Weight of Impact limiters	24,234	
Weight of Package - loaded cask with outer lid and impact limiters	265,202 (132.6 ton)	259,152 (129.6 ton)
Personnel Barrier	5,000	
Skid	30,000	
Total Loaded Weight of Package for transport - Weight of package, personnel barrier and skid	300,202 (150.1 ton)	294,152 (147.1 ton)

Note: For sites with a crance capacity limited to 125 tons, water would be removed from the cask prior to removing the cask from the pool. Therefore, Hook Weight 2 would be applicable.

2.1.2.1.2 Containment Vessel

The 6625B-HB cask containment boundary, as shown in **Figure 2.1-2**, consists of the 66.25-inch diameter inner shell, a 1.25-inch thick bottom plate, an upper end structure, a 3.00-inch thick inner lid with a 3.50-inch thick shield plug with innermost seals and closure bolts, vent, and drain ports with closure bolts and seals, and containment welds. A 66.25-inch diameter, 182.00-inch long cavity is provided within the containment boundary as shown in **Figure 2.1-3**.

A redundant mechanical closure is provided by the 2.50-inch thick outer lid with outermost seals and closure bolts, and the vent port with closure bolts and seals. The outer closure lid along with the space between the lids meets the design and manufacturing criteria such that it can be merged with the inner containment space to define an extended containment boundary. The extended containment boundary will be used only in the unlikely event that the boundary defined by the inner lid ceases to meet leaktight criteria.

Both closure lids have been designed to perform the containment function with final qualification by leakage rate testing according to ANSI N14.5 [34] as specified in *Section 2.8*.

The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the 6625B-HB cask cavity. Helium within the cavity assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation. To preclude air in-leakage, the cask cavity is pressurized with helium to above atmospheric pressure.

The containment shell material is SA-203, Grade E, and the upper and lower end structure and bottom closure plate materials are SA-350-LF3. The inner and outer lids are constructed from SA-350-LF3 or SA-203, Grade E material. The seals used for penetrations are fluorocarbon elastomer (Viton) O-ring seals. The AREVA 6625B-HB packaging containment vessel is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB [3] of the ASME Code to the maximum practical extent. In addition, the design meets the requirements of Regulatory Guides 7.6 [4] and 7.8 [5]. The containment boundary is shown in **Figure 2.1-2**. The fabrication requirements (including examination and testing) of the containment boundary are discussed in *Section 2.4*.

FIGURE 2.1-2: 6625B-HB PACKAGE CONTAINMENT BOUNDARY

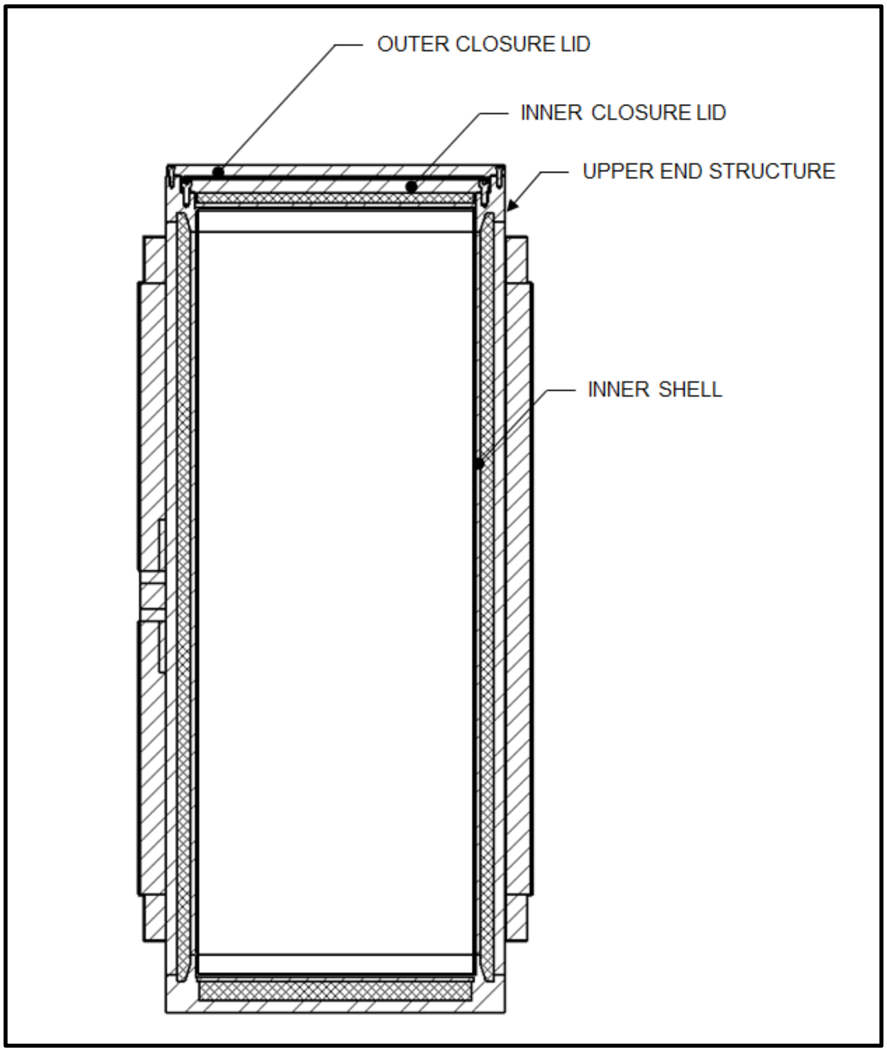
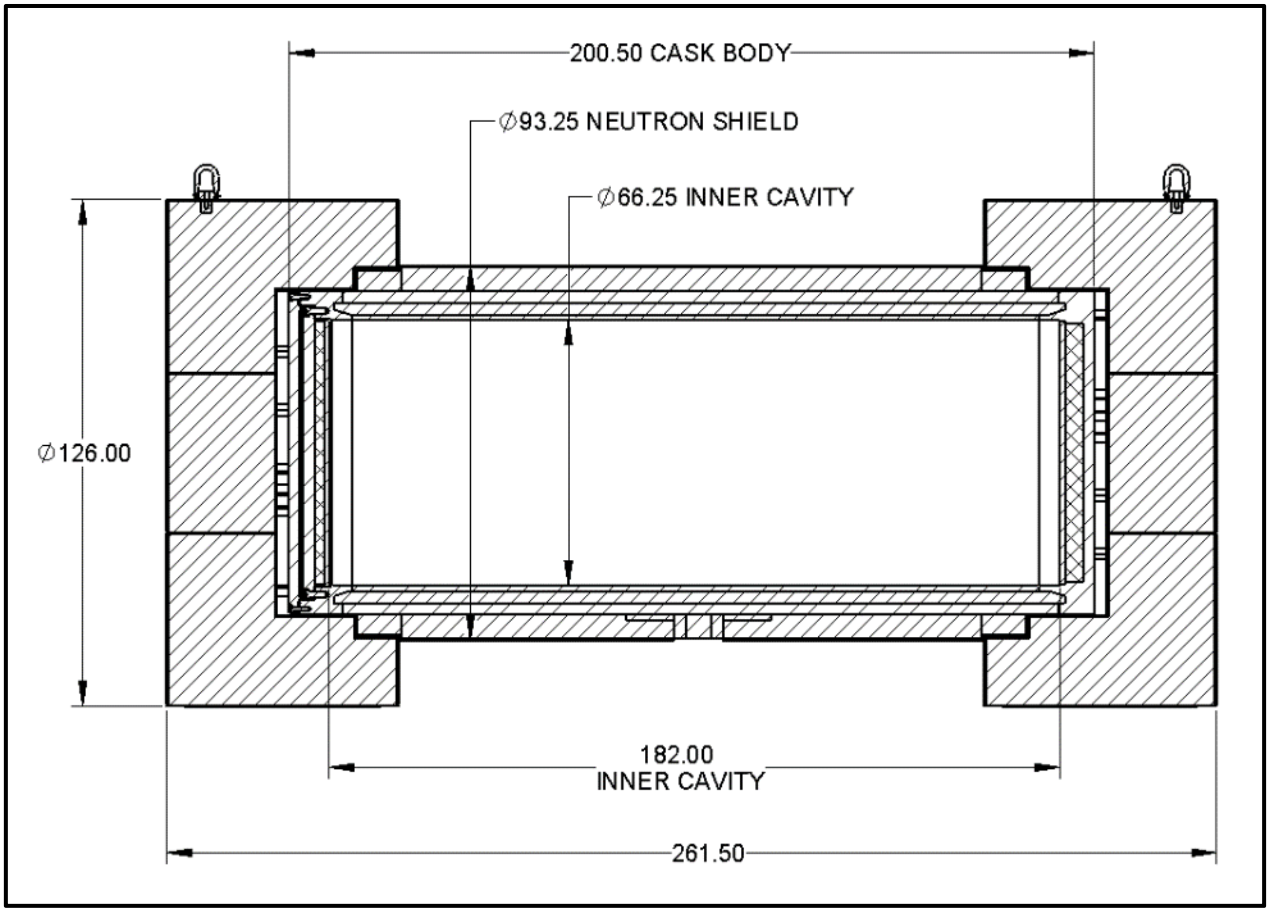


FIGURE 2.1-3: 6625B-HB PACKAGE DIMENSIONS

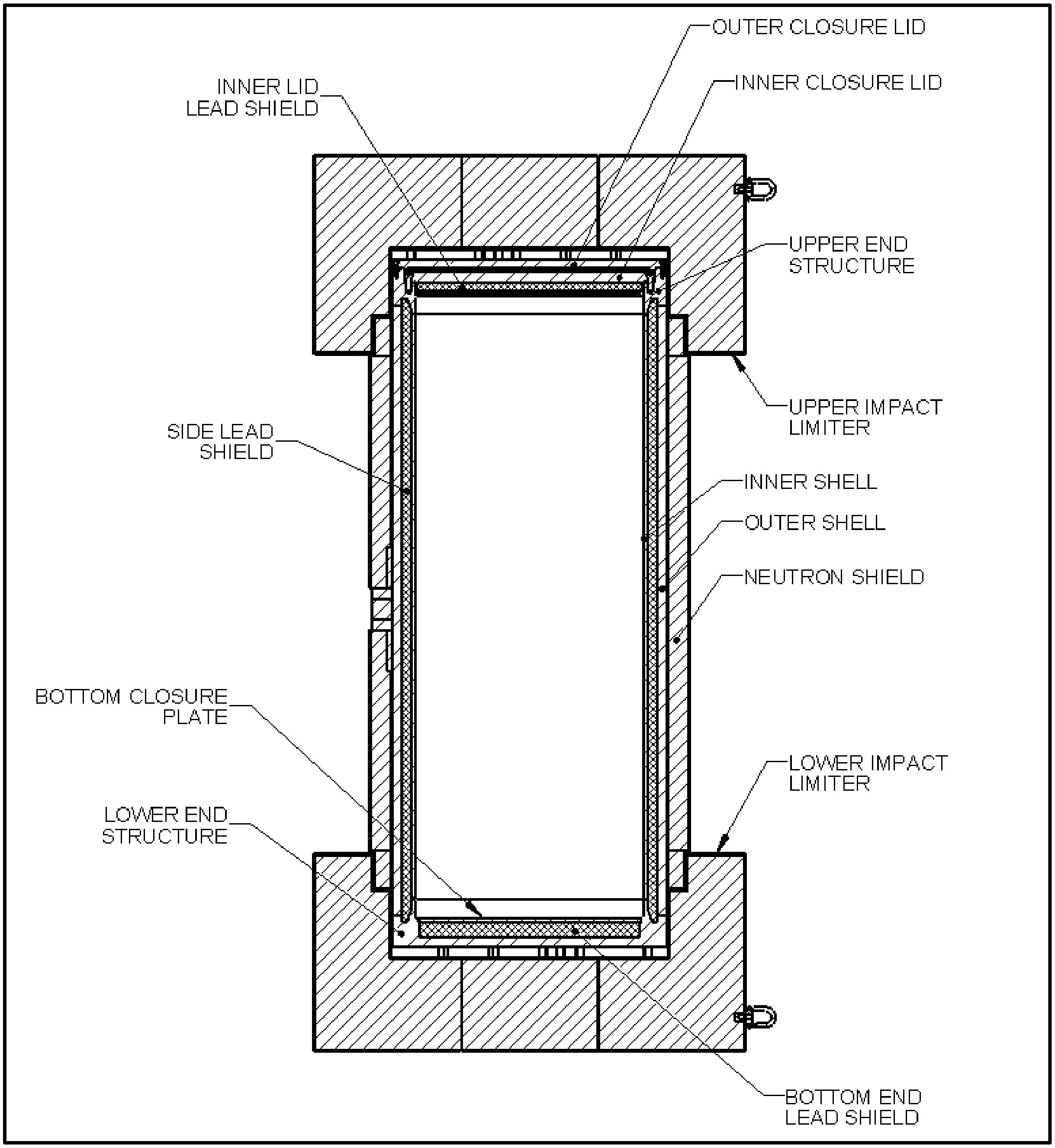


2.1.2.1.3 Gamma and Radial Neutron Shielding

The lead and steel shells of the 6625B-HB cask provide shielding between the fuel and the exterior surface of the package for the attenuation of gamma radiation (**Figure 2.1-4**). The 3.00-inch thick annular lead shield between the inner and outer shell provides radial gamma shielding. Axial gamma shielding is provided by the 2.50-inch thick lead shield in the inner lid and a 4.50-inch thick lead shield enclosed in the lower end structure with a 1.25-inch bottom closure plate. A borated resin compound surrounding the outer shell provides neutron shielding. The resin compound is cast into long, slender copper containers. The total thickness of the resin and copper is 6.25 inches between the impact limiters. The resin and copper extend under the impact limiters at a thickness of 5.25 inches for a distance of 11.00 inches on each end. The array of resin-filled containers is enclosed within a 0.25-inch thick outer stainless steel neutron shield outer shell. For installation, the copper boxes are held in place with temporary straps and brazed together. The neutron shield outer shell is installed in two halves and welded together, shrinking to sandwich the copper boxes between the outer shell and the neutron shield outer shell. In addition to serving as resin containers, the copper tubes provide a heat conduction path from the cask body to the neutron shield shell.

The structural evaluation of the 6625B-HB cask body is presented in *Section 2.2*.

FIGURE 2.1-4: 6625B-HB PACKAGE CROSS SECTION



2.1.2.1.4 Tiedown and Lifting Devices

There are four trunnion sockets on the cask; two front trunnion sockets, and two rear trunnion sockets. The sockets accommodate removable trunnions for handling, lifting, and rotating the cask. These trunnion sockets are attached to the structural shell. Each trunnion has a single shoulder and is designed as non-redundant lifting device. The trunnions are fabricated and load tested in accordance with ANSI N14.6 [6]. During transport, the trunnions are removed, and four trunnion plugs, containing neutron-shielding material, are bolted to the four trunnion sockets. When the cask is in the horizontal position, a shear key receptacle on the bottom of the cask accepts a shear block, which reacts to the longitudinal transportation loads. The shear key receptacle is welded to the structural shell and protrudes through the neutron shield. During transport the receptacle interfaces with the shear block attached to the transport skid. During site operations, a shear key plug, containing neutron-shielding material, is bolted to the shear key receptacle.

2.1.2.1.5 Impact Limiters

The upper and lower impact limiters, shown in **Figure 2.1-1**, absorb energy during impact events by crushing the balsa and redwood materials. The two impact limiters are identical and have an outside diameter of 126 inches and a height of 58 inches. The inner and outer shells are Type 304 stainless steel material joined by radial gussets of the same steel material. The gussets limit the stresses in the 0.25-inch thick stainless steel outer cylinder and end plates due to pressure differentials caused by elevation and temperature changes during normal transport. The metal structure locates, supports, confines, and protects the wood energy absorption material.

Each impact limiter is attached to the 6625B-HB cask by 12 attachment bolts. The attachment bolts are designed to secure the impact limiters to the cask body during all NCT and HAC.

Each impact limiter is provided with fusible plugs that are designed to melt during a fire accident event, thereby relieving any excessive internal pressure. Each impact limiter incorporates three hoist rings for handling, and two support angles for supporting the impact limiter in a vertical position during storage. The hoist rings are threaded into the impact limiter steel shell, while the support angles are welded to the shell. Prior to transport, the impact limiter hoist rings are removed and replaced with bolts to prevent the threaded holes from being utilized as a tie-down device.

An aluminum thermal shield is added to each impact limiter to reduce the temperature of the impact limiter wood. The details of the thermal shield are included in *Section 2.3.1*.

The functional description as well as the performance analysis of the impact limiters is provided in *Section 2.2*. The description and results of the predicted impact limiter dynamic response are also provided in that section.

2.1.2.2 Contents

2.1.2.2.1 24 PWR Basket

The basket structure is designed, fabricated and inspected in accordance with ASME Boiler and Pressure Vessel (B&PV) Code Subsection NG [7]. The nominal dimensions and component weights for the basket are provided in **Table 2.1-3**. The 24 PWR basket is heavier than past designs due to its increased length and solid rails. The 24 PWR fuel basket is shown in **Figure 2.1-5**. The 24 PWR basket is designed to accommodate (24) intact PWR UNF assemblies with or without Non Fuel Assembly Hardware (NFAH) or (24) DFCs loaded with intact PWR UNF assemblies with or without

NFAH. The basket can accommodate 8 DFCs loaded with damaged fuel assemblies. Damaged fuel is defined in **Table 2.1-4**. See *Section 8.0* for more information on damaged fuel.

The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) with the space between adjacent tubes filled with aluminum and neutron absorbing plates, which are surrounded by support rails. The basket structure is open at each end; therefore, longitudinal fuel assembly loads are applied directly to the cask body and not the fuel basket structure. The stainless steel tube assembly laterally supports the fuel assemblies. The basket rails and the cask inner shell laterally support the basket. The stainless steel and aluminum basket rails are oriented parallel to the longitudinal axis of the cask. The rails are attached to the periphery of the basket to provide support and to establish and maintain basket orientation. Shear keys are welded to the inner wall of the cask and mate with notches in the basket support rails to prevent the basket from rotating during normal operations.

Aluminum and neutron-absorbing poison plates are sandwiched between the fuel compartments. The neutron absorber plates are constructed of borated aluminum. The neutron absorber plates provide criticality control and, together with the aluminum plates, provide a heat conduction path from the fuel assemblies to the cask wall. The minimum natural or enriched boron aluminum alloy B-10 areal density is 40.6 mg/cm^2 for the 24 PWR basket. The maximum allowable decay heat load for the various 24 PWR basket loading configurations is 30.4 kW.

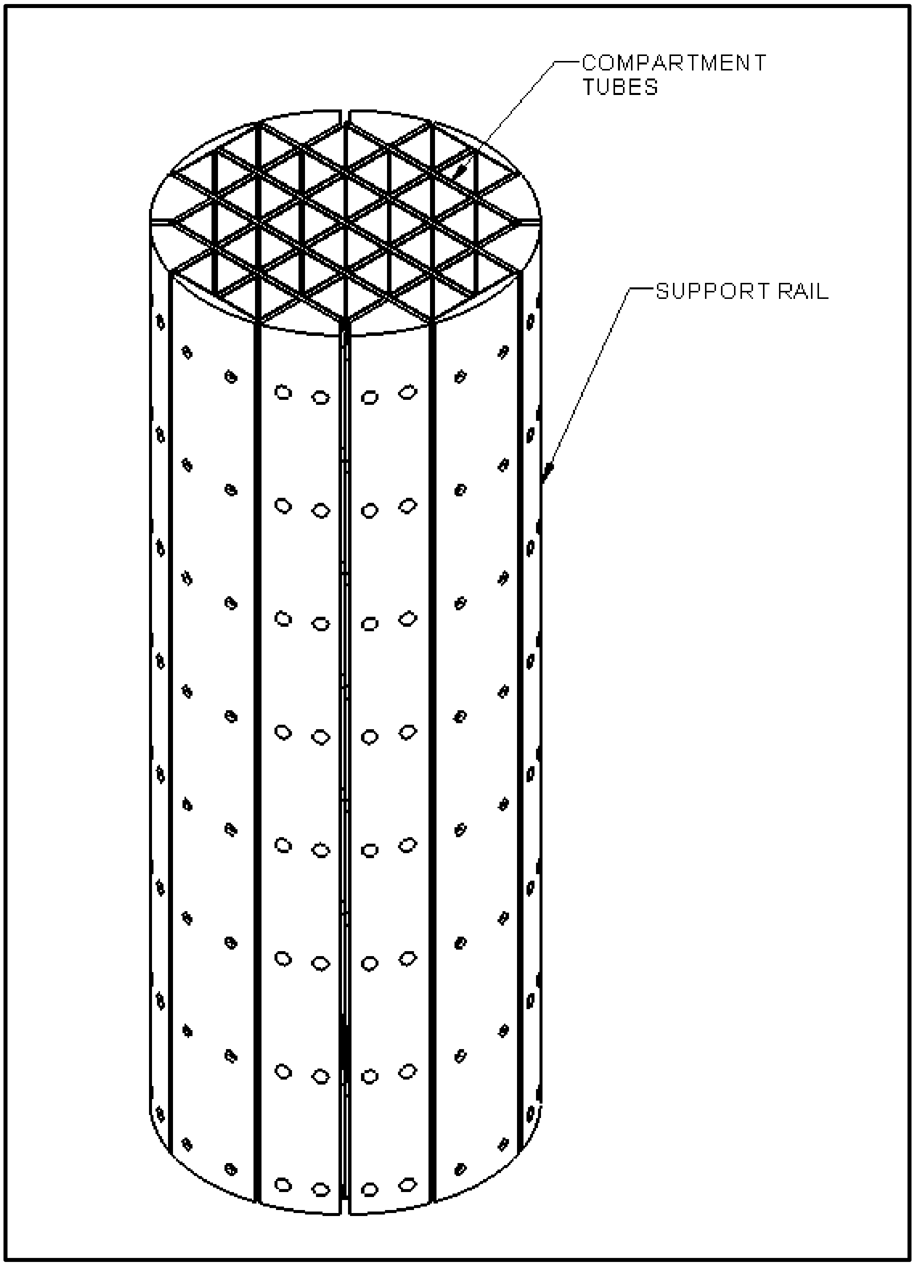
DFCs are provided for transport of high burnup intact fuel assemblies or damaged fuel. Each DFC is constructed of sheet metal and is provided with a welded bottom closure and a locking top closure, which allows lifting of the DFC with the enclosed assembly. The DFC is provided with screens at the bottom and top to contain the UNF assembly and allow fill/drainage of water from the DFC during loading operations. The DFC is protected by the fuel compartment tubes and its only function is to confine the fuel assembly. The fuel compartment tubes are capped with a removable sleeve that, when uninstalled, facilitates DFC insertion. Refer to *Section 2.1.5* for more information.

TABLE 2.1-3: KEY DESIGN PARAMETERS OF THE 24 PWR BASKET

Nominal Dimensions	Inches
Cask Cavity Length (in)	182.00
Basket Length (in)	181.50
Basket Diameter (in)	66.00
Basket Component Weights	**Lbs.**
Solid Aluminum Support Rail – 45 (8)	11,361
Solid Aluminum Support Rail – 90 (4)	2,996
Basket Tubes (24)	11,204
Aluminum Plates	7,609
Steeel Basket Plates	2,952

Notes :
1. Unless stated otherwise, nominal values are provided.
2. The total basket weight, including additional misc. components, hardware, and welds is listed in **Table 2.1-2**.

FIGURE 2.1-5: 6625B-HB 24 PWR BASKET



2.1.2.2.2 24 PWR Basket Contents

The 24 PWR basket is designed to transport PWR UNF as specified in **Table 2.1-4**. The UNF to be transported is limited to the following:

- maximum irradiated assembly length of 180 inches (does not include thermal growth). Fuel spacers are used to accommodate shorter UNF assemblies.
- maximum assembly average initial enrichment of 5.0 wt.% U-235
- maximum allowable assembly average burnup is limited to 62.5 GWd/MTU

The minimum cooling time requirements are given in **Table 2.1-7**. The 24 PWR basket is also designed to transport NFAH with radiological characteristics as listed in **Table 2.1-6**. The NFAH include burnable poison rod assemblies (BPRAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), neutron source assemblies (NSAs), and Neutron Sources.

The basket is divided into inner and peripheral zones, as shown in **Figure 2.1-6**. Fuel in each zone is governed by the cooling times presented in **Table 2.1-7**. These cooling times are determined to meet temperature and dose rate limits. The PWR basket can accommodate 8 damaged and 16 undamaged fuel assemblies with the damaged fuel assemblies loaded in Zone 4.

TABLE 2.1-4: PWR FUEL SPECIFICATION FOR THE FUEL TO BE TRANSPORTED IN THE 24 PWR BASKET

Physical Parameter	
Fuel Class	Intact unconsolidated B&W 15x15, WE 17x17, CE 16x16 and CE System 80 (without NFAH), CE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies (with or without non fuel assembly hardware) that are enveloped by the design characteristics in *Section 2.1.2.2.2*. Equivalent reload fuel manufactured by same or other vendors but enveloped by the design characteristics listed in *Section 2.1.2.2.2* is also acceptable.
Damaged Fuel	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel assembly shall be handled by normal means. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
Failed Fuel	Failed PWR fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.
Non Fuel Assembly Hardware (NFAH)	• Up to 24 NFAH are authorized for transportation in the 24 PWR basket • Authorized NFAH include BPRAs, TPAs, CRAs, RCCAs, APSRAs, ORAs, VSIs, NSAs, and neutron sources. • Design basis radiological characteristics for the NFAH are listed in **Table 2.1-6**.
Maximum Assembly Width for Intact Fuel Assemblies Only	8.54 inches
Number of Intact Assemblies	≤24
Maximum Assembly plus NFAH weight (includes weight of DFC)	1,770 lbs
Thermal/Criticality Parameter	
Fuel Assembly Burnup and minimum Cooling Time	Per Table 2.1-7.
Maximum Decay Heat Limits for Zones 1, 2, 3, and 4 Fuel	Per Figure 2.1-6
Decay heat per Basket	≤ 30.4 kW for the 24 PWR Basket with decay heat limit for Zones 1, 2, 3 and 4 as specified in **Figure 2.1-6**
Burnup Credit Restrictions	Per Table 2.1-5

Note: The definition of damaged fuel is not intended to be consistent with ISG-1.

TABLE 2.1-5: MINIMUM REQUIRED BURNUP VS ENRICHMENT FOR BURNUP CREDIT– INTACT ASSEMBLIES

Enrichment (%U235)	Assembly Burnup (GWD/MTU)
2.0	10
3.0	24
3.5	31
3.75	35
4.0	38
4.6	46
5.0	52

TABLE 2.1-6: RADIOLOGICAL CHARACTERISTICS FOR NFAH TRANSPORTED IN THE 6625B-HB 24 PWR BASKET

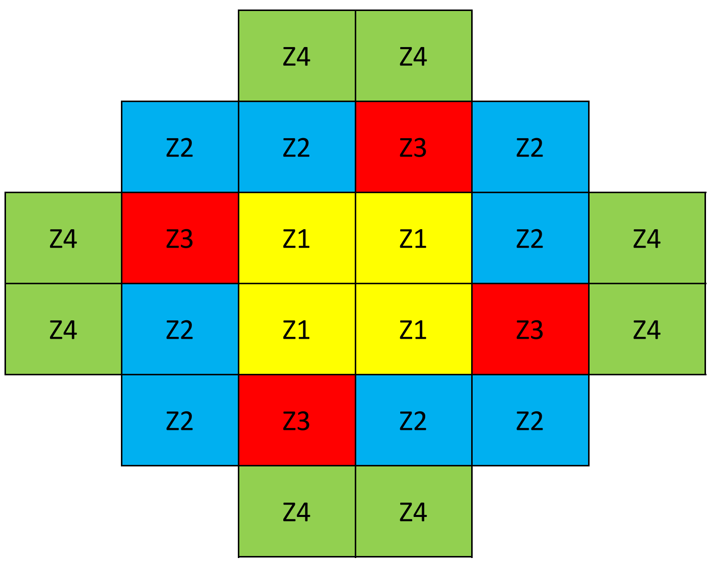
Fuel Assembly Region	Co-60 (Ci) per Fuel Assembly
Active Fuel	600
Plenum	30
Top Nozzle	20

TABLE 2.1-7: PWR FUEL QUALIFICATIONS TABLE FOR THE 6625B-HB 24 PWR BASKET

		Minimum Cooling Time (years)		
		Zone 1/4	Zone 2	Zone 3
Maximum Burnup (GWd/MTU)	Minimum Enrichment (%)	Heat ≤ 0.9 kW	Heat ≤ 1.4 kW	Heat ≤ 2.1 kW
≤ 30	≥ 1.8	≥ 5	≥ 5	≥ 5
≤ 37	≥ 2.3	≥ 6.5	≥ 5	≥ 5
≤ 45	≥ 2.8	≥ 10	≥ 5	≥ 5
≤ 53	≥ 3.3	≥ 16	≥ 6.5	≥ 5
≤ 62.5	≥ 3.8	≥ 26	≥ 9	≥ 5

Note: The numbers in blue are determined by inspection because the upper bound decay heat cannot be achieved for this burnup/cooling time combination.

FIGURE 2.1-6: HEAT LOAD ZONING CONFIGURATION FOR 6625B-HB 24 PWR BASKET



	Zone 1/4	Zone 2	Zone 3
Maximum Decay Heat (kW/FA)	0.9	1.4	2.1
No. of Fuel Assemblies	12	8	4
Maximum Decay Heat per Zone (kW)	3.6/7.2	11.2	8.4
Maximum Decay Heat per Basket (kW)	30.4		

2.1.2.2.3 61 BWR Basket

The basket structure is designed, fabricated, and inspected in accordance with ASME B&PV Code Subsection NG [7]. The overall length and diameter of the basket is provided in **Table 2.1-8**. The 61 BWR fuel basket is shown in **Figure 2.1-7**. The 61 BWR basket is designed to accommodate 61 intact BWR fuel assemblies with or without fuel channels or 61 DFCs loaded with intact BWR fuel assemblies with or without fuel channels. The basket can accommodate 12 DFCs loaded with damaged fuel assemblies. Damaged fuel is defined in **Table 2.1-9**. Refer to *Section 8.0* for more information on damaged fuel.

TABLE 2.1-8: KEY DESIGN PARAMETERS OF THE 61BWR BASKET

Parameter	61BWR Basket
Cask Cavity Length (in)	182.00
Basket Length (in)	181.50
Basket Diameter (in)	66.00

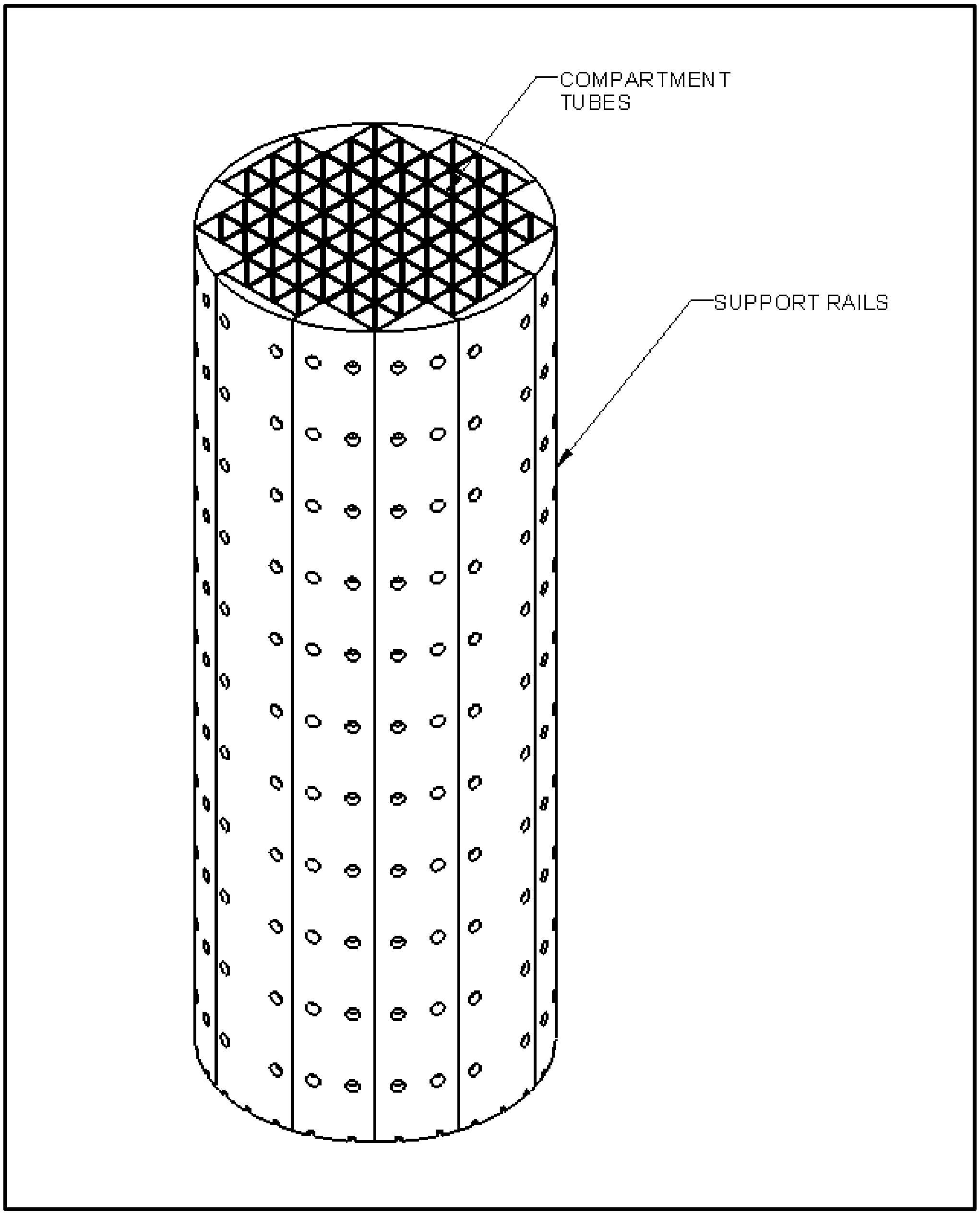
Note : Unless stated otherwise, nominal values are provided.

The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by neutron absorber plates and surrounded by larger stainless steel boxes and support rails. The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the cask body and not the fuel basket structure. The stainless steel structural boxes laterally support the fuel assemblies. The basket rails and the cask shell laterally support the basket. The stainless steel basket rails are oriented parallel to the longitudinal axis of the cask. The rails are attached to the periphery of the basket to provide support and to establish and maintain basket orientation. Shear keys are welded to the inner wall of the cask and mate with notches in the basket support rails to prevent the basket from rotating during normal operations.

Neutron absorbing poison plates are sandwiched between the fuel compartments. The neutron absorber plates are constructed of borated aluminum. The neutron absorber plates provide criticality control and provide a heat conduction path from the fuel assemblies to the cask wall. The minimum natural or enriched boron aluminum alloy B-10 areal density is 85.3 mg/cm^2 for the 61 BWR basket. The maximum allowable decay heat load for the various 61 BWR basket loading configurations is 30.3 kW.

DFCs are provided for transport of high burnup intact UNF assemblies or damaged fuel. Each DFC is constructed of sheet metal and is provided with a welded bottom closure and a locking top closure, which allows lifting of the DFC with the enclosed assembly. The DFC is provided with screens at the bottom and top to contain the assembly and allow fill/drainage of water from the DFC during loading operations. The DFC is protected by the fuel compartment tubes and its only function is to confine the fuel assembly. The fuel compartment tubes are capped with a removable sleeve that, when uninstalled, facilitates DFC insertion. Refer to *Section 2.1.5* for more information.

FIGURE 2.1-7: 6625B-HB 61 BWR BASKET



2.1.2.2.4 61 BWR Basket Contents

Each of the 61 BWR basket configurations is designed to transport BWR UNF as specified in **Table 2.1-9**. The UNF to be transported is limited to the following:

- maximum irradiated assembly length of 180 inches (does not include thermal growth). Fuel spacers are used to accommodate shorter UNF assemblies.
- maximum assembly average initial enrichment of 5.0 wt. % U-235
- maximum allowable assembly average burnup is limited to 62.5 GWd/MTU

The minimum cooling time requirements are given in **Table 2.1-10**.

The basket is divided into inner and peripheral zones, as shown in **Figure 2.1-8**. Fuel in each zone is governed by the cooling times presented in **Table 2.1-10**. These cooling times are determined to meet temperature and dose rate limits. The BWR basket can accommodate 12 damaged and 49 undamaged fuel assemblies with the damaged fuel assemblies loaded in Zone 4.

TABLE 2.1-9: BWR FUEL SPECIFICATION FOR THE FUEL TO BE TRANSPORTED IN THE 61 BWR BASKET

Physical Parameter	
Fuel Class	Intact 7x7, 8x8, 9x9, or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or ABB or other vendors that are enveloped by the fuel assembly design characteristics in *Section 2.1.2.2.4*. Equivalent reload fuel manufactured by the same or other vendors but enveloped by the design characteristics listed in *Section 2.1.2.2.4* is also acceptable.
Damaged Fuel	Damaged BWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel assembly shall be handled by normal means. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
Failed Fuel	Failed BWR fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.
Non Fuel Assembly Hardware (NFAHs)	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Maximum Assembly Width for Intact Fuel Assemblies Only	5.62 inches
Number of Intact Assemblies	≤ 61
Maximum Assembly plus CC weight (includes weight of DFC)	705 lbs
Thermal/Criticality Parameter	
Fuel Assembly Average Burnup and minimum Cooling Time	Per Table 2.1-10
Maximum Decay Heat Limits for Zones 1, 2, 3, and 4 Fuel	Per Figure 2.1-8
Decay heat per Basket	≤ 30.3 kW for the 61BWR Basket with decay heat limit for Zones 1, 2, 3 and 4 as specified in **Figure 2.1-8**.

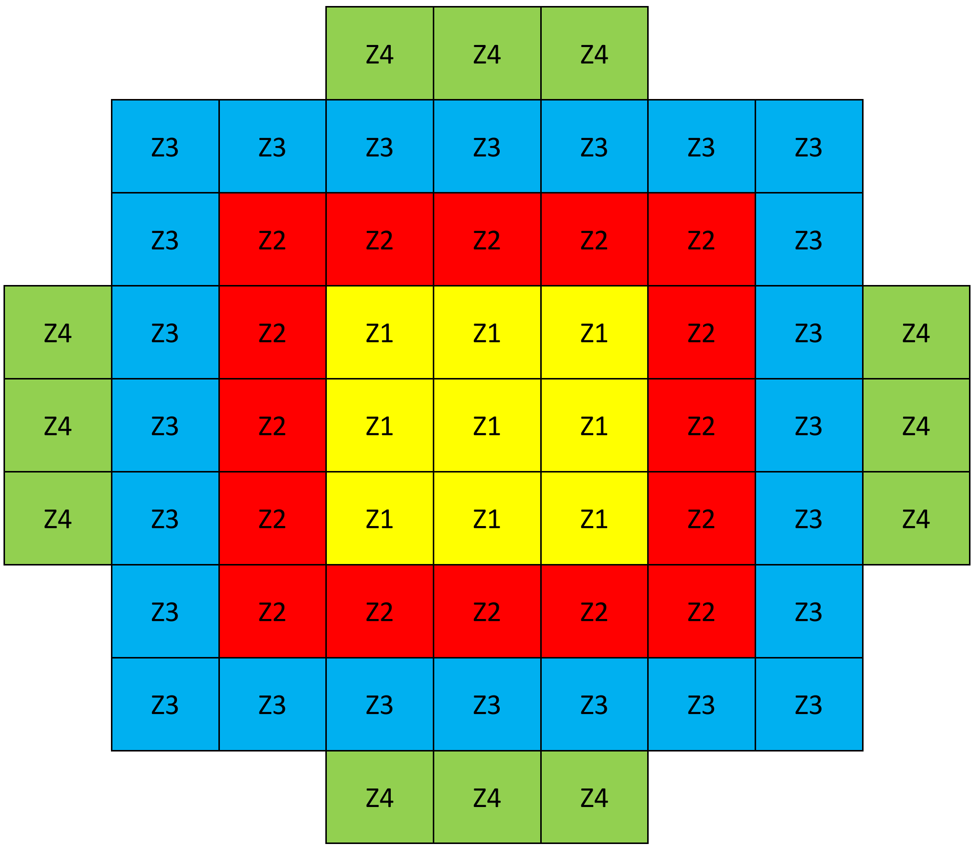
Note: The definition of damaged fuel is not intended to be consistent with ISG-1

TABLE 2.1-10: BWR FUEL QUALIFICATION TABLE FOR THE 61 BWR BASKET

		Minimum Cooling Time (years)		
		Zone 1/4	Zone 2	Zone 3
Maximum Burnup (GWd/MTU)	Minimum Enrichment (%)	Heat ≤ 0.33 kW	Heat ≤ 0.78 kW	Heat ≤ 0.45 kW
≤ 29	≥ 1.5	≥ 5	≥ 5	≥ 5
≤ 35	≥ 2.2	≥ 6	≥ 5	≥ 5
≤ 39	≥ 2.4	≥ 7.2	≥ 5	≥ 5
≤ 45	≥ 2.8	≥ 10	≥ 5	≥ 6
≤ 53	≥ 3.3	≥ 16	≥ 5	≥ 8
≤ 62.5	≥ 3.8	≥ 25	≥ 5	≥ 12.5

Note: The numbers in blue are determined by inspection because the upper bound decay heat cannot be achieved for this burnup/cooling time combination.

FIGURE 2.1-8: HEAT LOAD ZONING CONFIGURATION FOR 6625B-HB 61 BWR BASKET



	Zone 1/4	Zone 2	Zone 3
Maximum Decay Heat (kW/FA)	0.33	0.78	0.45
No. of Fuel Assemblies	21	16	24
Maximum Decay Heat per Zone (kW)	3.0/4.0	12.5	10.8
Maximum Decay Heat per Basket (kW)	30.3		

2.1.3 Operational Features

The 6625B-HB packaging is not considered to be operationally complex and is designed to be compatible with standard SFP loading/unloading methods. All operational features are readily apparent and described in *Section 2.7*. The sequential operational steps to be followed for cask loading, testing, and unloading operations are also provided in *Section 2.7*.

2.1.4 Summary

The 6625B-HB conceptual design is described in *Section 2.1.2* and evaluated in *Section 2.2* through *Section 2.6*. The conceptual design is based on current state-of-the-art industry designs that have been licensed by the NRC. By using proven methodologies that have been utilized in previously licensed casks, a high degree of confidence is obtained for this design. This approach provides assurance that the design will functionally work, as well as meet the regulatory requirements. The proven methodologies ensure that the demonstration of compliance can be made to the current regulatory organization in an acceptable manner. Although detailed analysis was not performed in all areas within the restraints of this conceptual design program, the configurations used in the design are known to be proven acceptable based on very detailed analysis and testing over the years. The design was developed using currently accepted industry practices with the goal of maximizing the package contents within the design basis restrictions. These include a weight limit, a required cavity length, and the thermal, shielding, and criticality restraints of the contents. A detailed discussion of the package sizing is included in *Section 2.2*. The selected design was optimized for transporting high burnup fuel as defined in *Section 2.1.2.2*.

2.1.5 Packaging General Arrangement Drawings

The packaging general arrangement drawings consist of:

- Concept 6625B-HB Package, Task Order 17 (**Figure 2.1-9 thru Figure 2.1-16**)
- 24 PWR Basket, Concept 6625B-HB Package, Task Order 17 (**Figure 2.1-17 thru Figure 2.1-21**)
- 61 BWR Basket, Concept 6625B-HB Package, Task Order 17 (**Figure 2.1-22 thru 2.1-26**)

FIGURE 2.1-9: CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (PART 1 OF 8)

LIST OF MATERIALS

QTY A4	QTY A3	QTY A2	QTY A1	ITEM NO	PART NO	DESCRIPTION	SPECIFICATION
			-	A1		TOP ASSEMBLY	
		-	1	A2		CASK ASSEMBLY	
	-	1	-	A3		CASK WELDMENT	
-			1	A4		INNER LID WELDMENT	
-	-	-	2	1		IMPACT LIMITER ASSEMBLY	
-	-	AR	-	2		RESIN BOX, 1/8 THK	COPPER
-	-	AR	-	3		RESIN	VYAL B
-	-	4	-	4		TRUNNION	ASME SA-182-F316N
-	-	8	-	5		TRUNNION SPACER PLATE, 1/2 THK	ASME SA-240, TYPE 304
-	-	1	-	6		OUTER LID, 2-1/2 THK	ASME SA-350-LF3 OR SA-203 GRADE E
-	-	24	-	7		IL ATTACHMENT BLOCK	ASME SA-182-F316N
-	-	AR	-	8		PLATE, 1/4 THK	ASME SA-240, TYPE 304
-	-	1	-	9		OUTER SHEET, 1/4 THK	ASME SA-240, TYPE 304
-	-	96	-	10		LID BOLT, 1-1/2-12 UNF-2A	ASME SA-540 OR B23, CL 1
-	-	48	-	11		TRUNNION BOLT	ASME SA-540 GR B23, CL 1
-	-	-	24	12		IL ATTACHMENT BOLT, 1-1/2-12 UNF-2A	ASME SA-540 GR B23, CL 1
-	-	-	24	13		WASHER	STAINLESS STEEL
-	-	-	1	14		SHEAR KEY PLUG	POLYPROPYLENE
-	-	4	-	15		O-RING	FLUOROCARBON
-	1	-	-	16		PLATE KEYWAY ASSEMBLY	
-	1	-	-	17		PLATE, BOTTOM CLOSURE, 1-1/4 THK	ASME SA-203, GRADE E
-	2	-	-	18		TRUNNION PLATE LOWER, 7.0 THK	ASME SA-82-F316N
-	2	-	-	19		TRUNNION PLATE UPPER, 7.0 THK	ASME SA-82-F316N
1	-	-	-	20		INNER LID, 3.0 THK	ASME SA-350-LF3 OR SA-203 GRADE E
1	-	-	-	21		INNER LID BOTTOM, 1/2 THK	ASME SA-350-LF3 OR SA-203 GRADE E
1	-	-	-	22		PLATE, 1 THK	ASME SA-350-LF3 OR SA-203 GRADE E
-	1	-	-	23		PLATE, INNER SHELL 1-1/4 THK	ASME SA-203, GRADE E
AR	AR	-	-	24		LEAD	ASTM B29
-	1	-	-	25		PLATE, OUTER SHELL, 2-3/4 THK	ASME SA-203, GRADE E
-	1	-	-	26		FORGING, UPPER END STRUCTURE	ASME SA-350-LF3
-	1	-	-	27		FORGING, LOWER END STRUCTURE	ASME SA-350-LF3
-	-	96	-	28		WASHER	STAINLESS STEEL

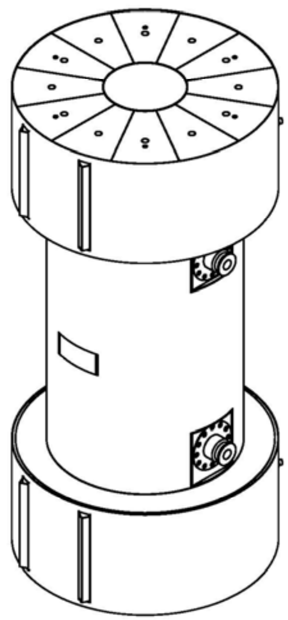


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FIGURE 2.1-10: CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (PART 2 OF 8)

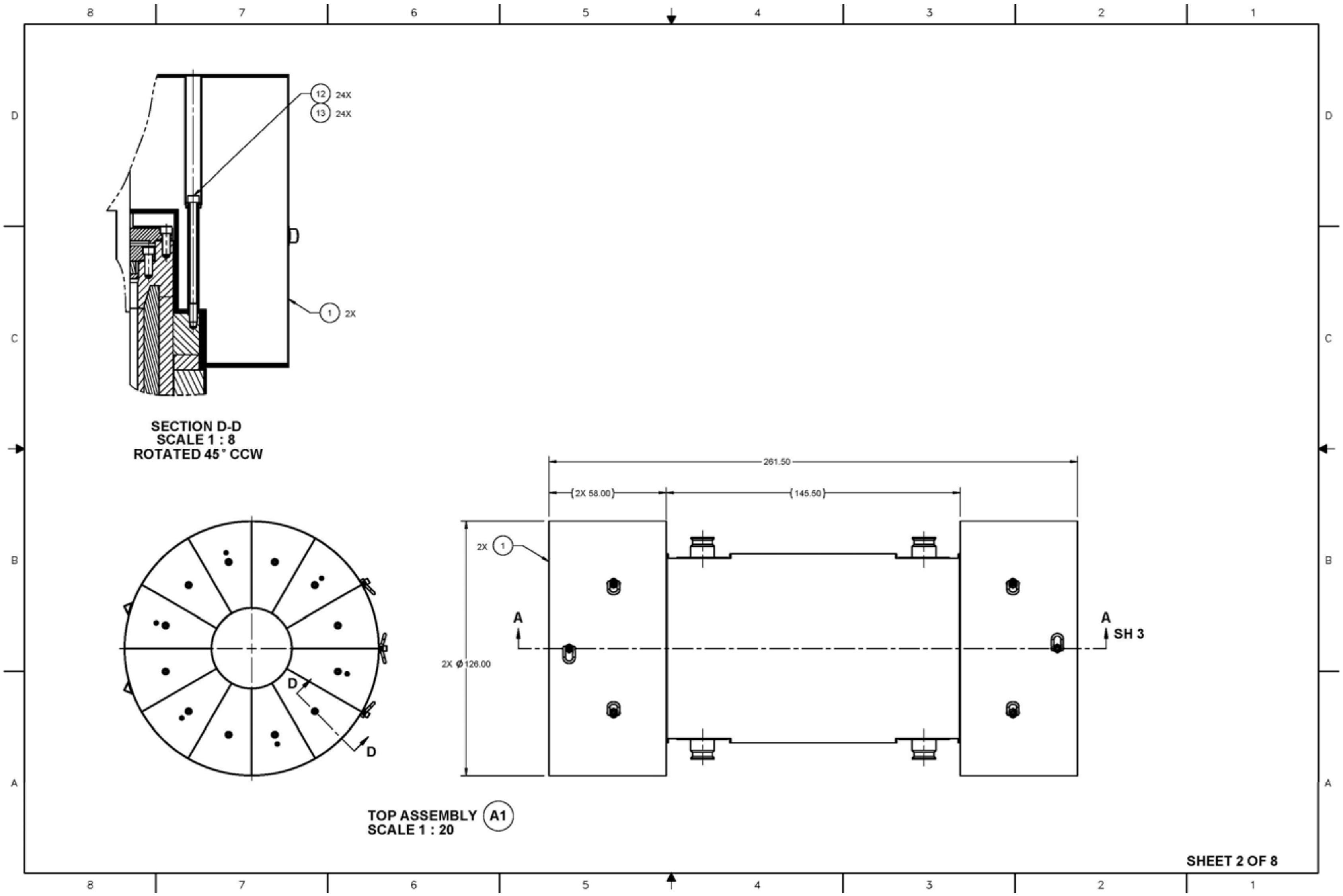

FIGURE 2.1-11: CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (PART 3 OF 8)

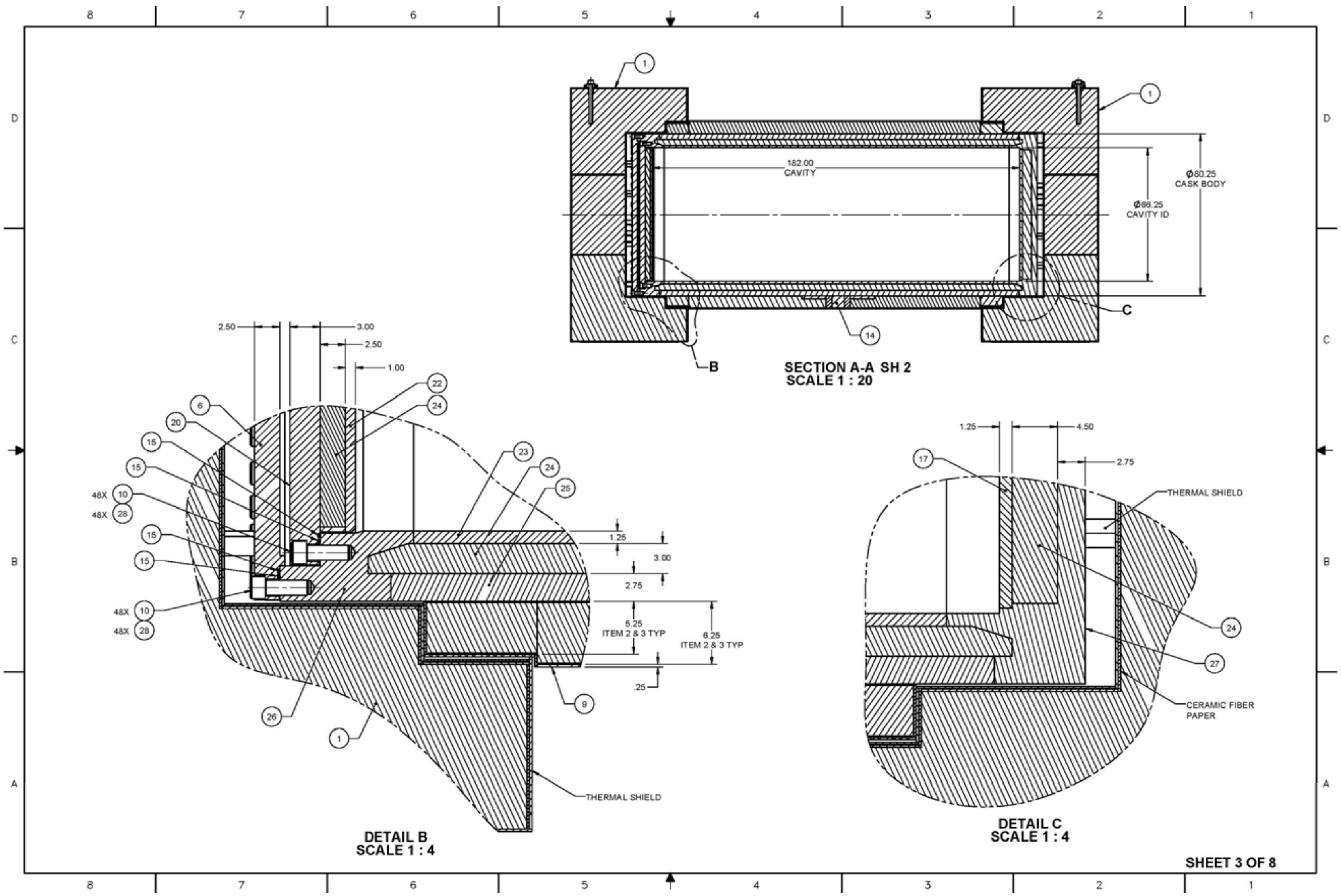

FIGURE 2.1-12: CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (PART 4 OF 8)

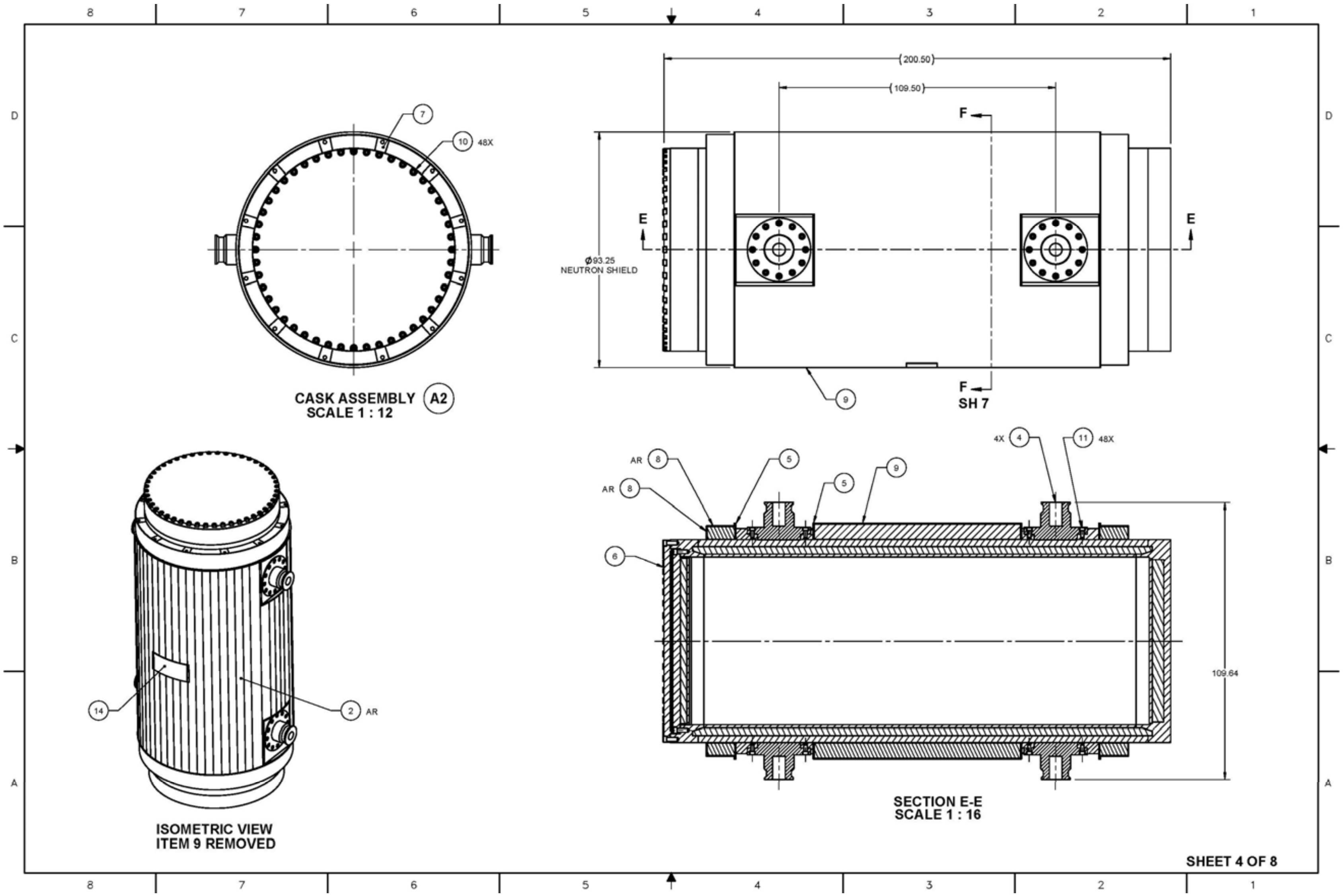

FIGURE 2.1-13: CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (PART 5 OF 8)

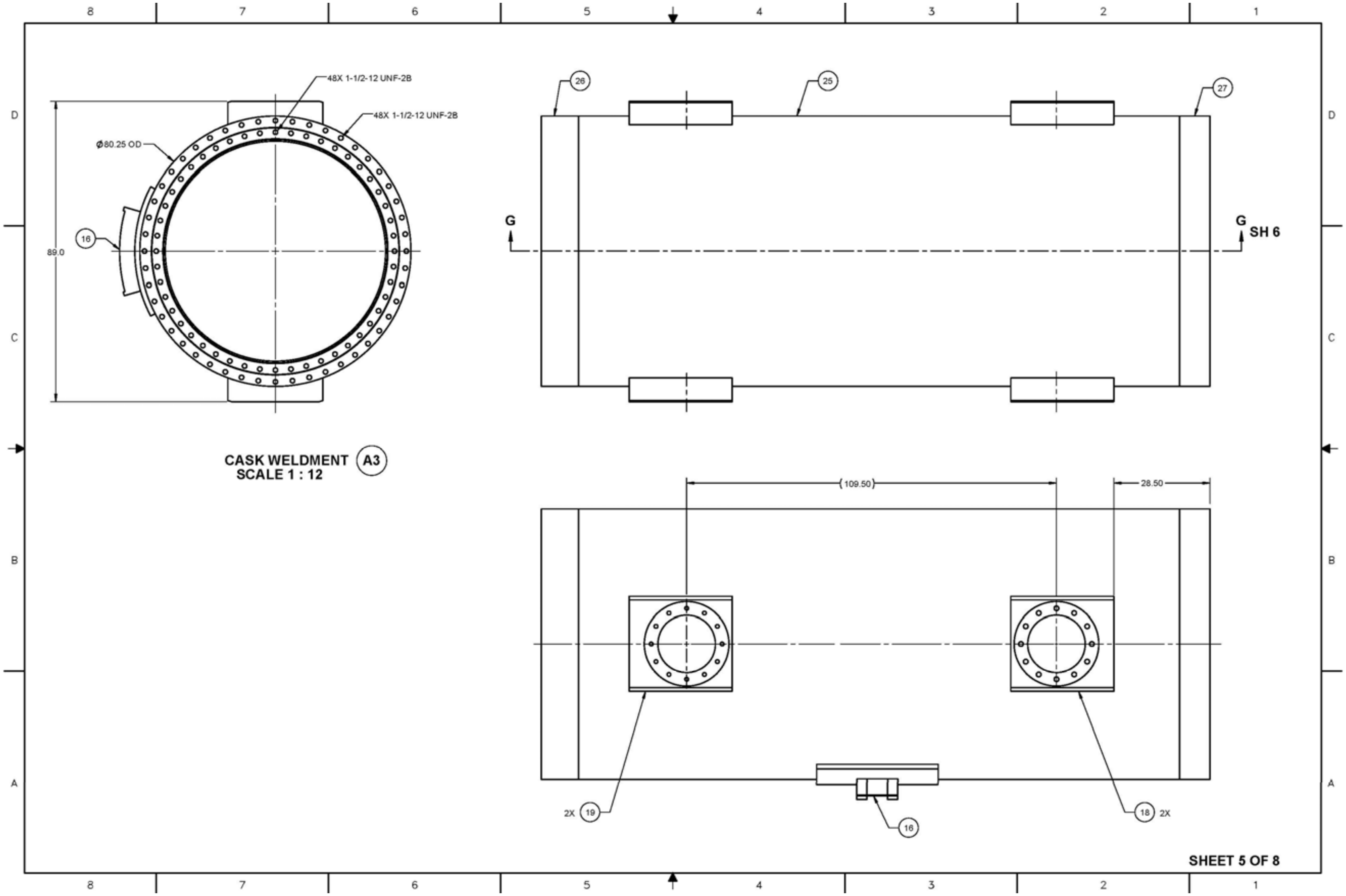

FIGURE 2.1-14: CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (PART 6 OF 8)

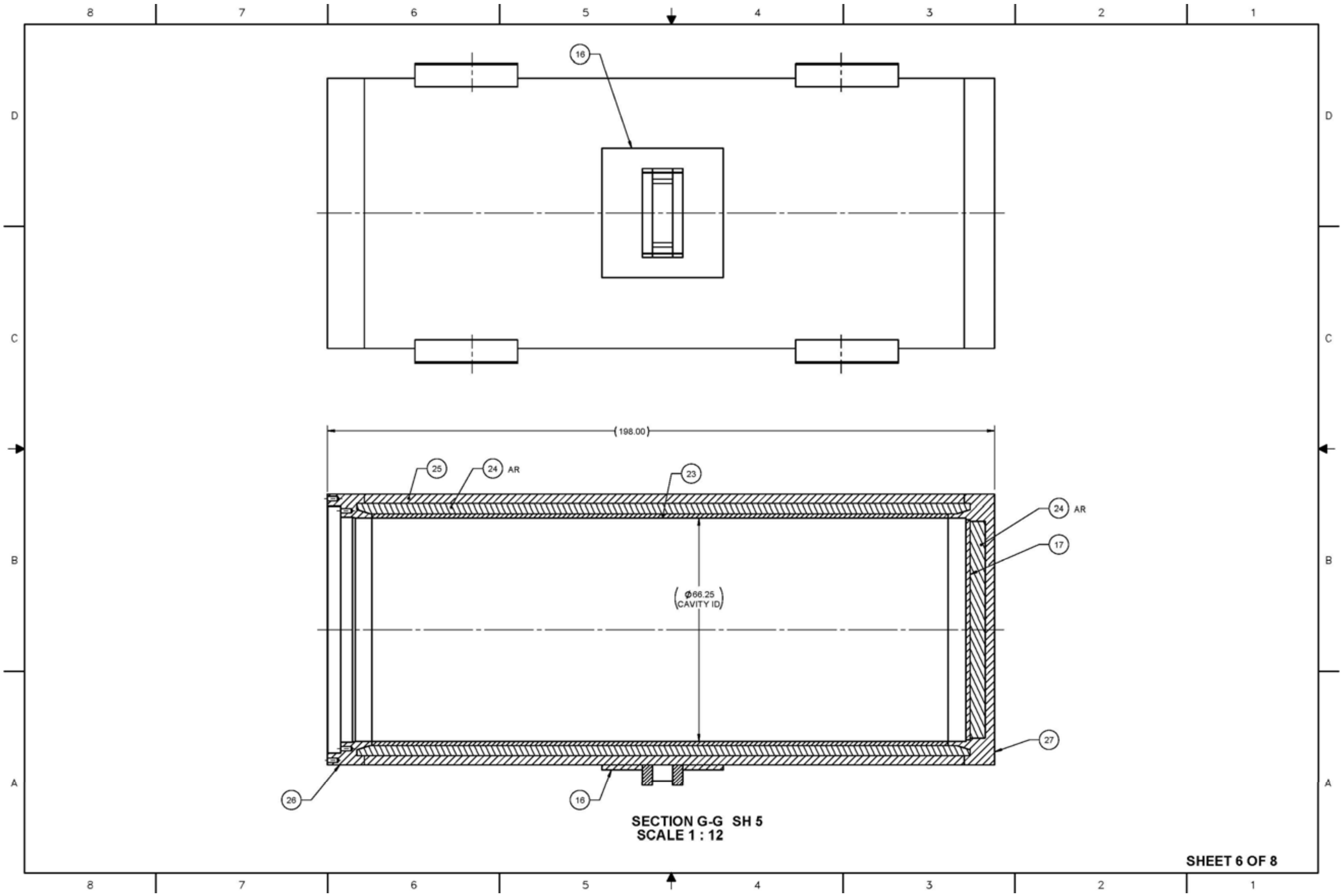

FIGURE 2.1-15: CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (PART 7 OF 8)

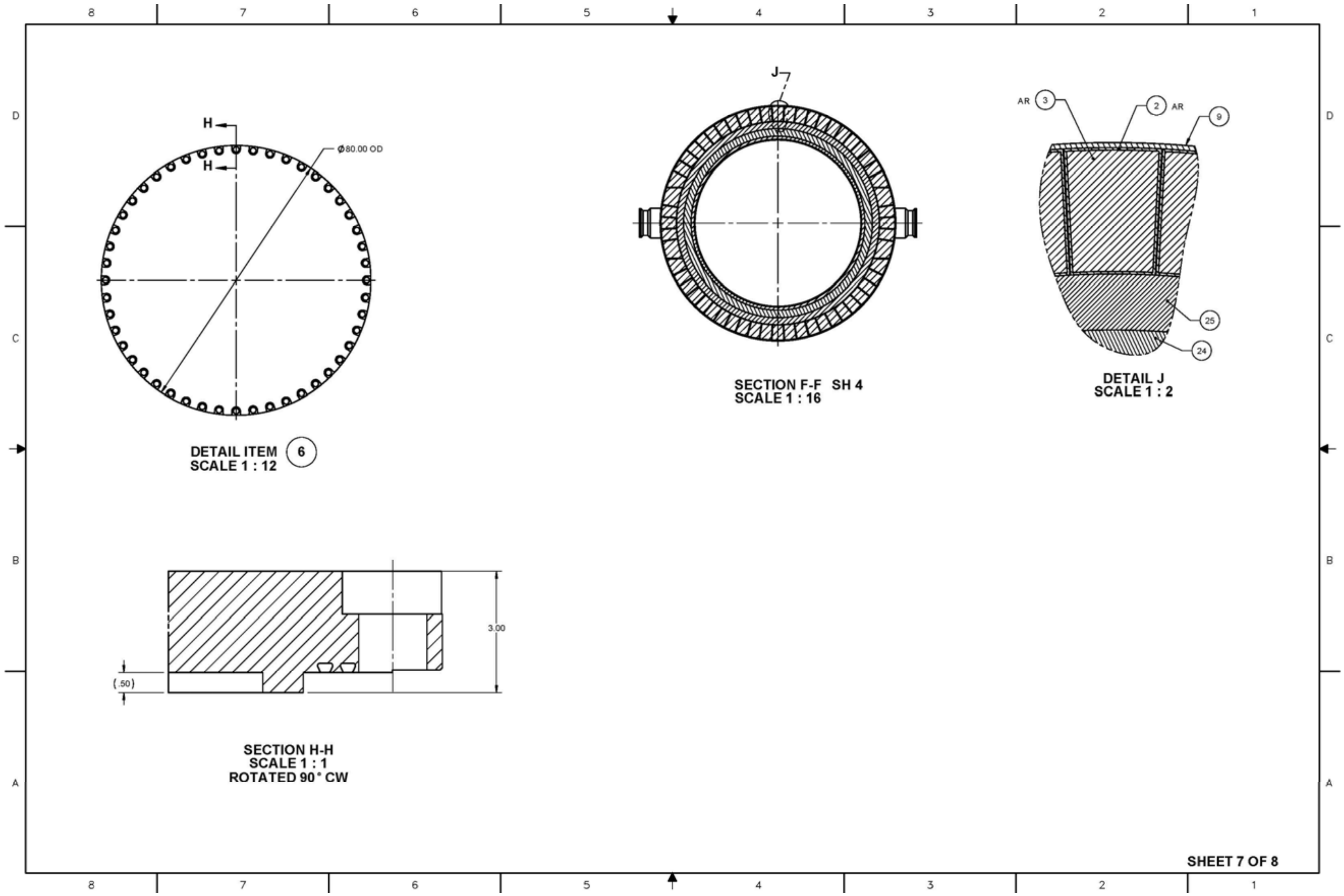

FIGURE 2.1-16: CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (PART 8 OF 8)

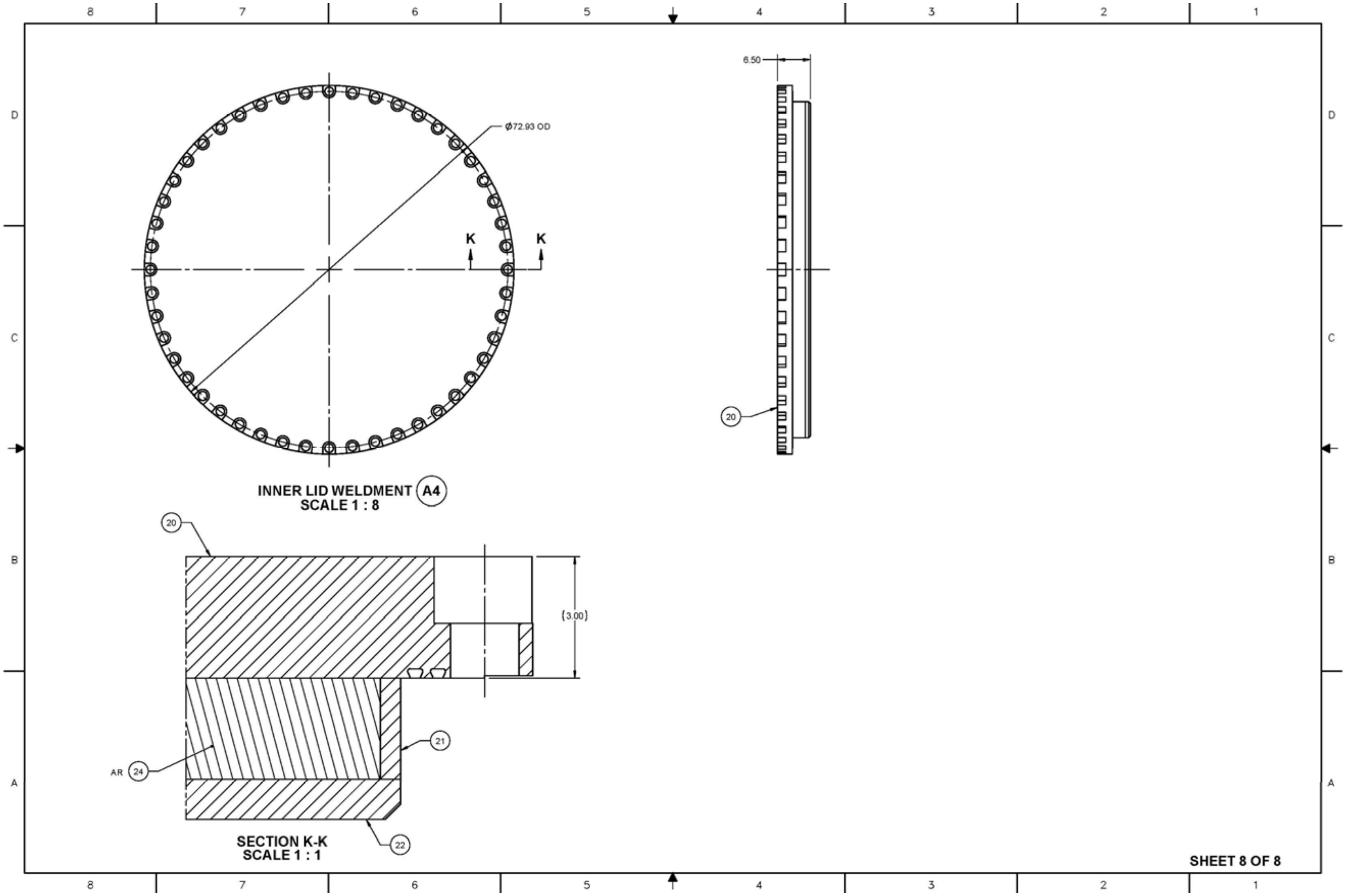

FIGURE 2.1-17: 24 PWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (1 OF 5)

LIST OF MATERIALS

QTY A2	QTY A1	ITEM NO	PART NO	DESCRIPTION	SPECIFICATION
	-	A1		24 PWR FUEL BASKET	
-	AR	A2		DAMAGED FUEL CAN PWR	
-	10	1		PLATE, 3/4 THK	ASTM B209, 6061
-	24	2		TUBE, 1/4 THK	ASTM A554, GRADE MT 304
-	24	3		REMOVABLE SLEEVE, 1/4 THK	ASTM A554, GRADE MT 304
-	AR	4		NEUTRON ABSORBING SHEET, 1/8 THK	ENRICHED BORON ALUMINUM ALLOY
-	4	5		SUPPORT RAIL, 90	ASTM B209, 6061
-	8	6		SUPPORT RAIL, 45	ASTM B209, 6061
-	AR	7		STUD	ASME SA-276, TYPE 304
-	AR	8		HEX NUT	STAINLESS STEEL
-	AR	9		WASHER	STAINLESS STEEL
1	-	10		ROUND TOOL SOCKET, Ø1.5	ASME SA-276, TYPE 304
1	-	11		DFC LINER	ASTM A554, GRADE MT 304
1	-	12		LID LINER, 1/8 THK	ASTM A554, GRADE MT 304
12	-	13		PLATE	ASME SA-240, TYPE 304
1	-	14		PLATE RING	ASME SA-240, TYPE 304
4	-	15		PLATE	ASME SA-240, TYPE 304
1	-	16		LOCKING PLATE	ASME SA-240, TYPE 304
1	-	17		BOTTOM PLATE	ASME SA-240, TYPE 304
1	-	18		SHEET PERFORATED	ASME SA-240, TYPE 304
1	-	19		SHEET	ASME SA-240, TYPE 304

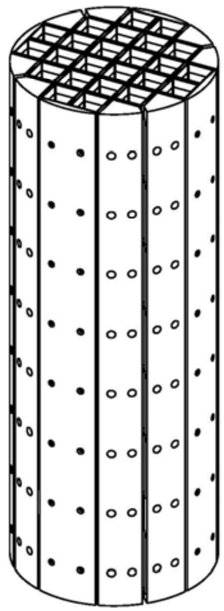

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FIGURE 2.1-18: 24 PWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (2 OF 5)

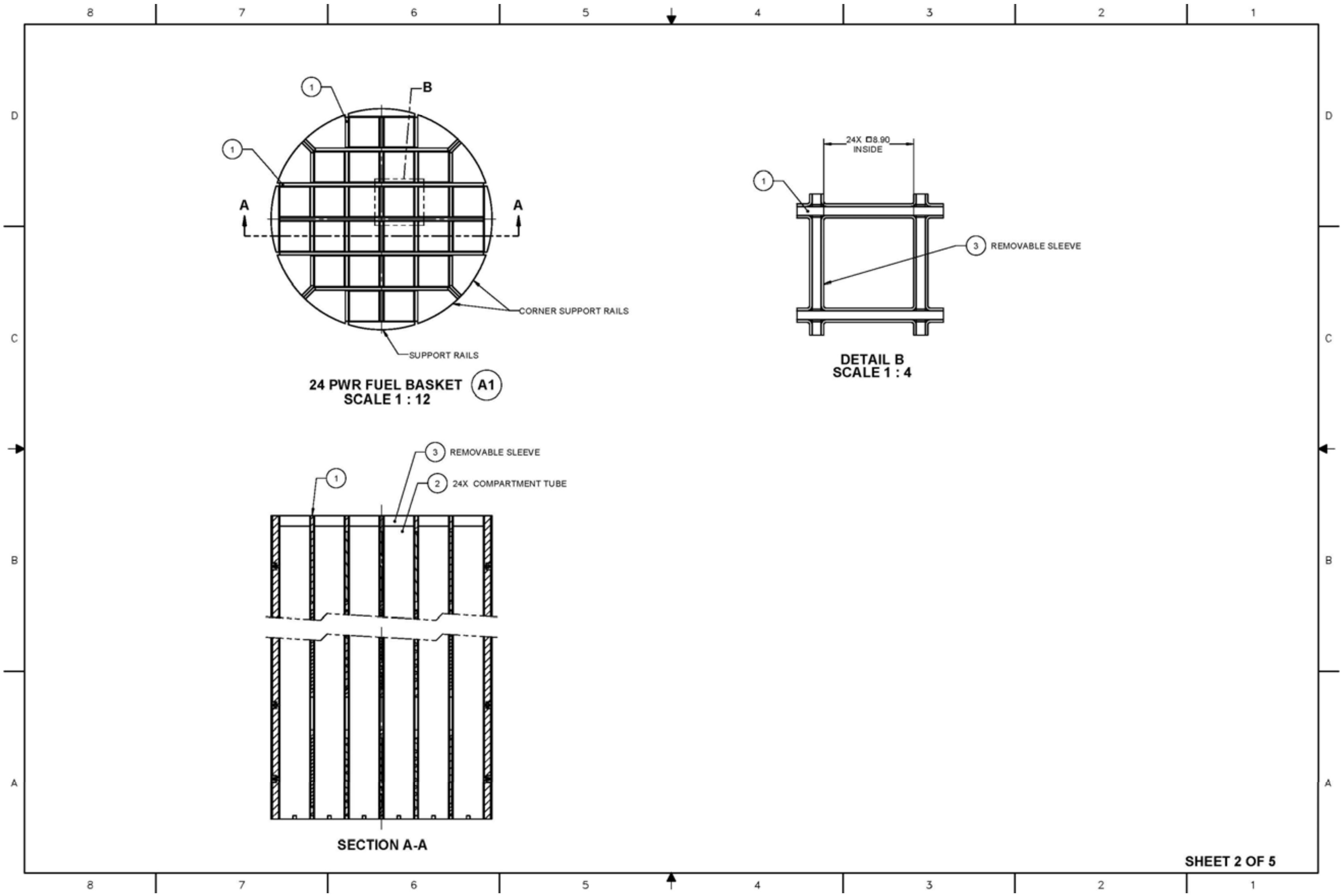

FIGURE 2.1-19: 24 PWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (3 OF 5)

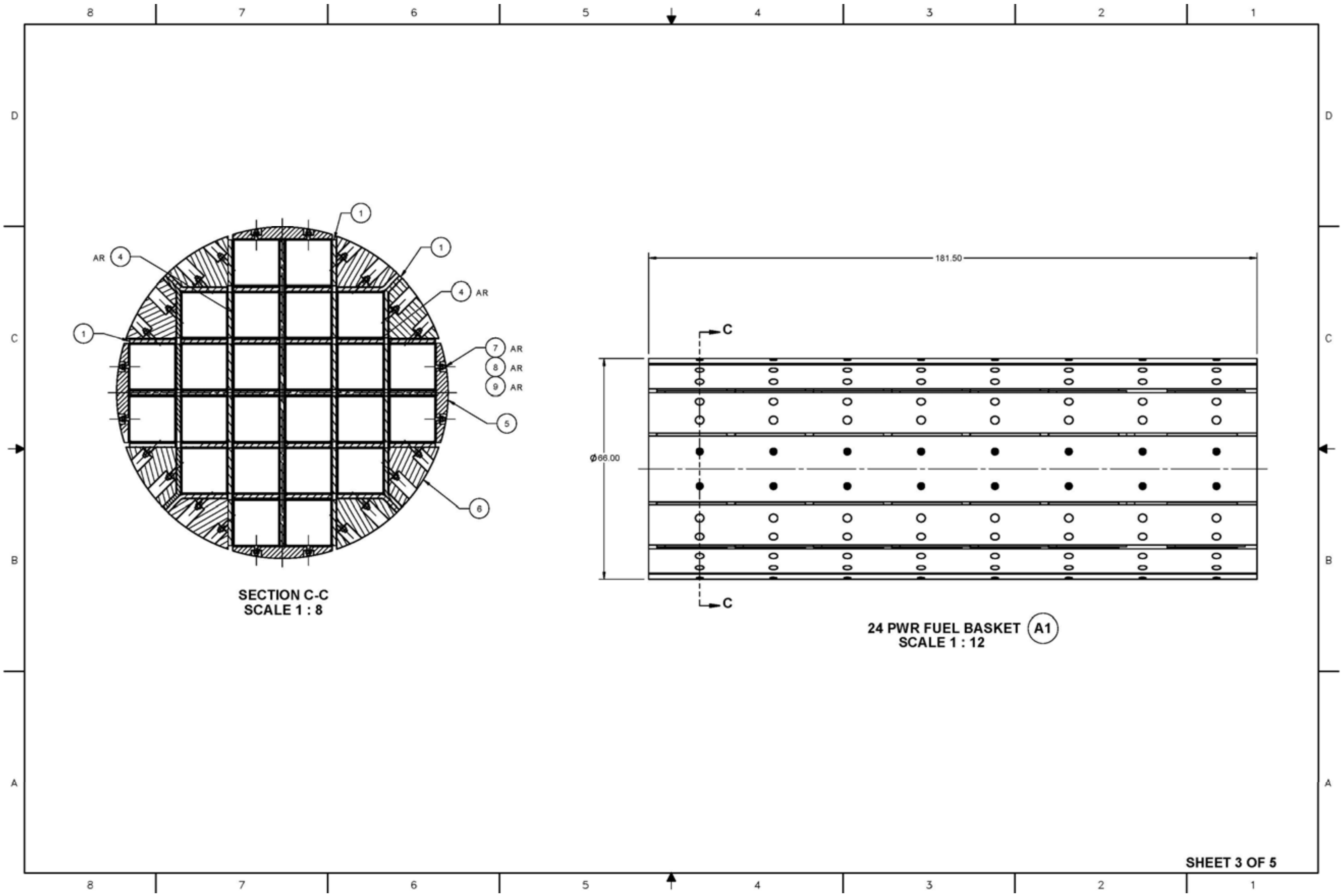

FIGURE 2.1-20: 24 PWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (4 OF 5)

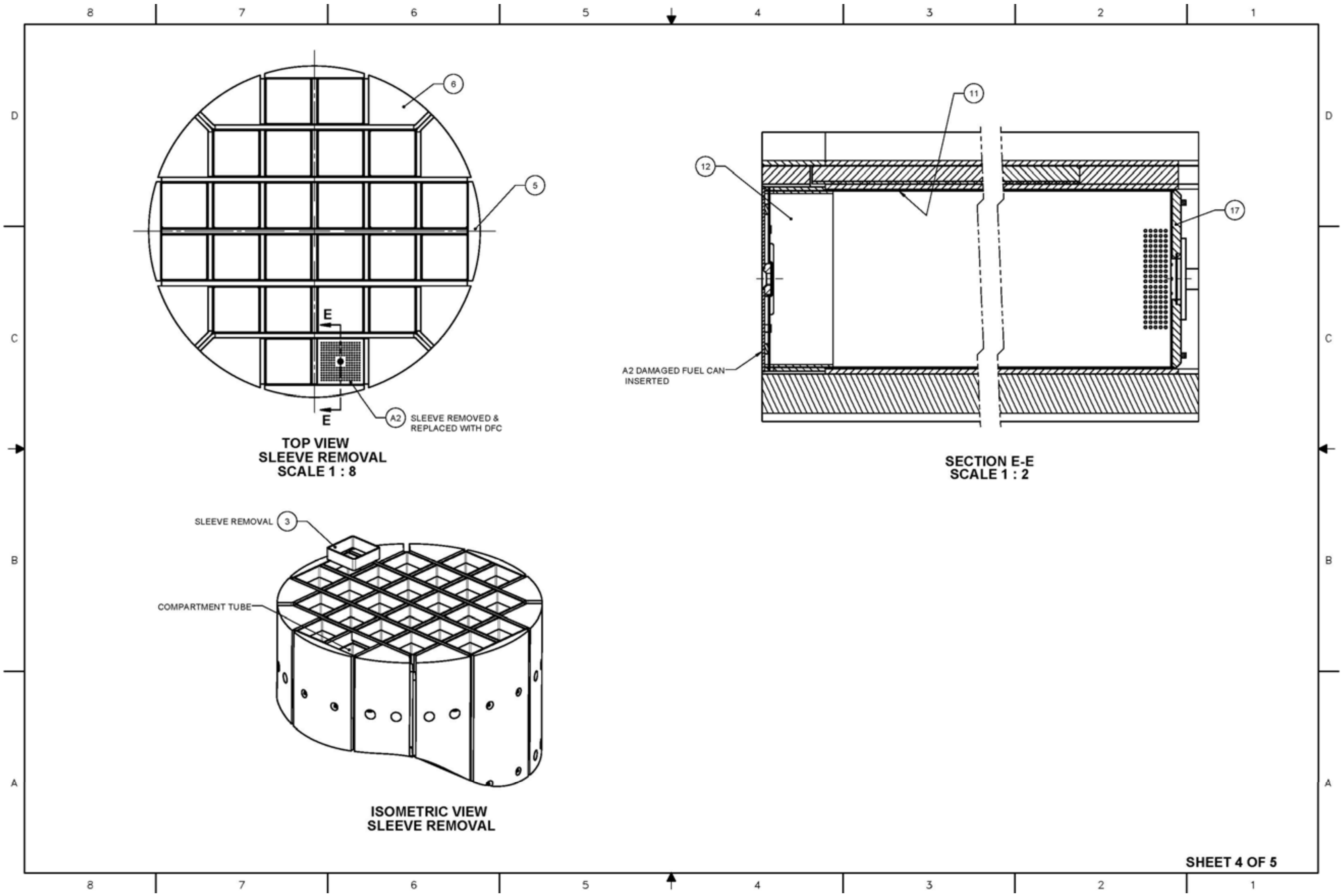

FIGURE 2.1-21: 24 PWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (5 OF 5)

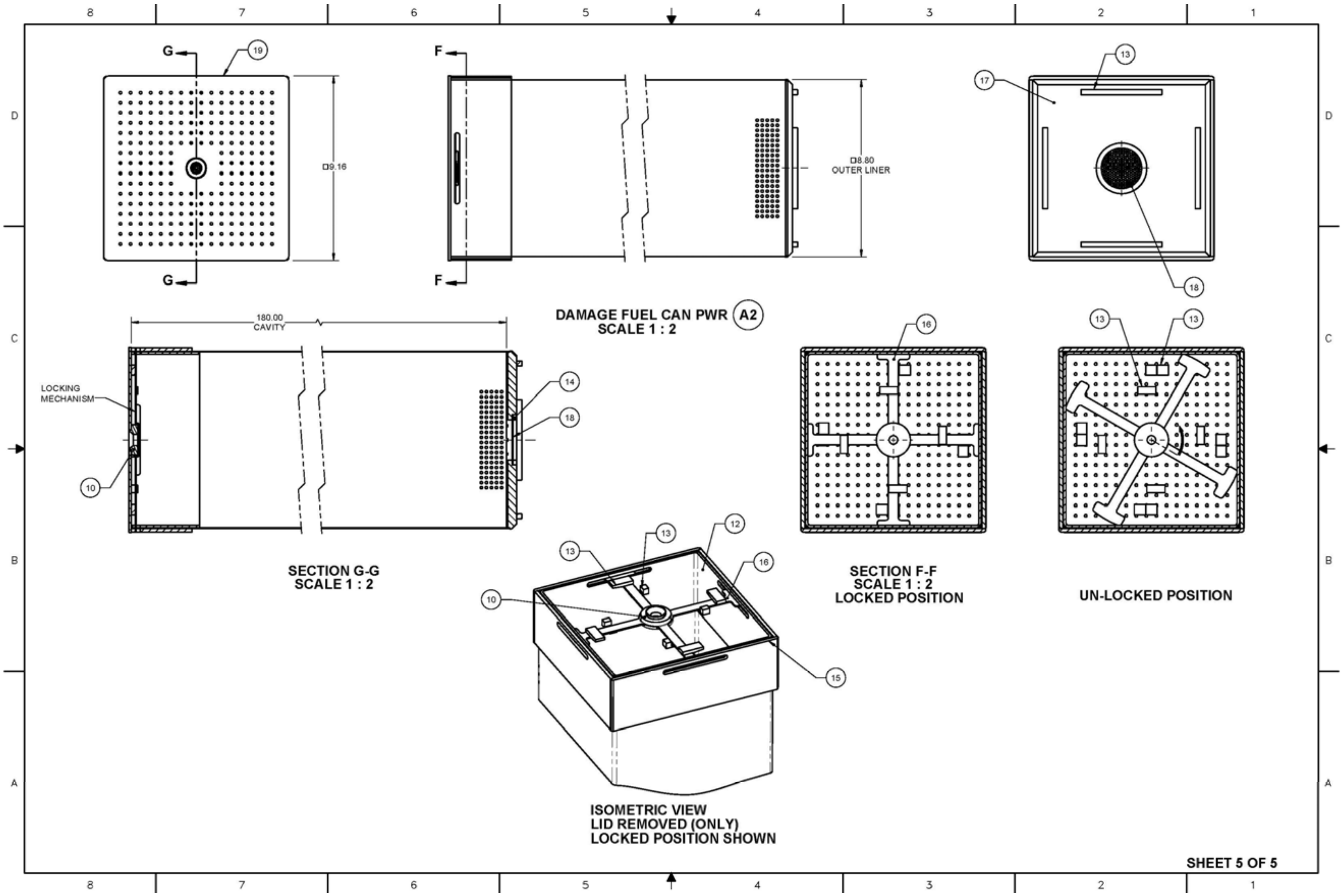

FIGURE 2.1-22: 61 BWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (1 OF 5)

LIST OF MATERIALS

QTY A2	QTY A1	ITEM NO	PART NO	DESCRIPTION	SPECIFICATION
	-	A1		61 BWR FUEL BASKET	
-	AR	A2		DAMAGE FUEL CAN BWR	
-	61	1		TUBE, .135 THK	ASTM A554, GRADE MT 304
-	18	2		SHEET, FUEL WRAP, .105 THK	ASME SA-240, TYPE 304
-	61	3		REMOVABLE SLEEVE, .135 THK	ASTM A554, GRADE MT 304
-	AR	4		NEUTRON ABSORBING SHEET, .31 THK	ENRICHED BORON ALUMINUM ALLOY
-	4	5		SUPPORT RAIL, 90	ASTM B209, 6061
-	8	6		SUPPORT RAIL, 45	ASTM B209, 6061
-	AR	7		STUD	ASME SA-276, TYPE 304
-	AR	8		HEX NUT	STAINLESS STEEL
-	AR	9		WASHER	STAINLESS STEEL
1	-	10		ROUND TOOL SOCKET	ASME SA-276, TYPE 304
1	-	11		DFC LINER	ASTM A554, GRADE MT 304
1	-	12		LID LINER, 1/8 THK	ASTM A554, GRADE MT 304
12	-	13		PLATE	ASME SA-240, TYPE 304
1	-	14		PLATE, RING	ASME SA-240, TYPE 304
4	-	15		PLATE	ASTM SA-240, TYPE 304
1	-	16		LOCKING PLATE	ASME SA-240, TYPE 304
1	-	17		BOTTOM PLATE	ASME SA-240, TYPE 304
1	-	18		SHEET PERFORATED	ASME SA-240, TYPE 304
1	-	19		SHEET	ASME SA-240, TYPE 304

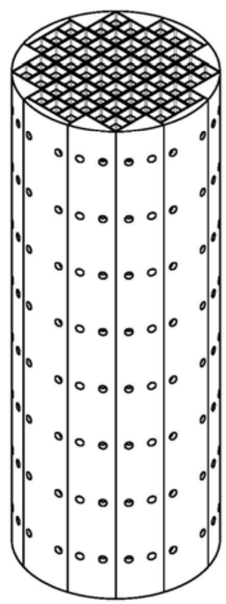

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FIGURE 2.1-23: 61 BWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (2 OF 5)

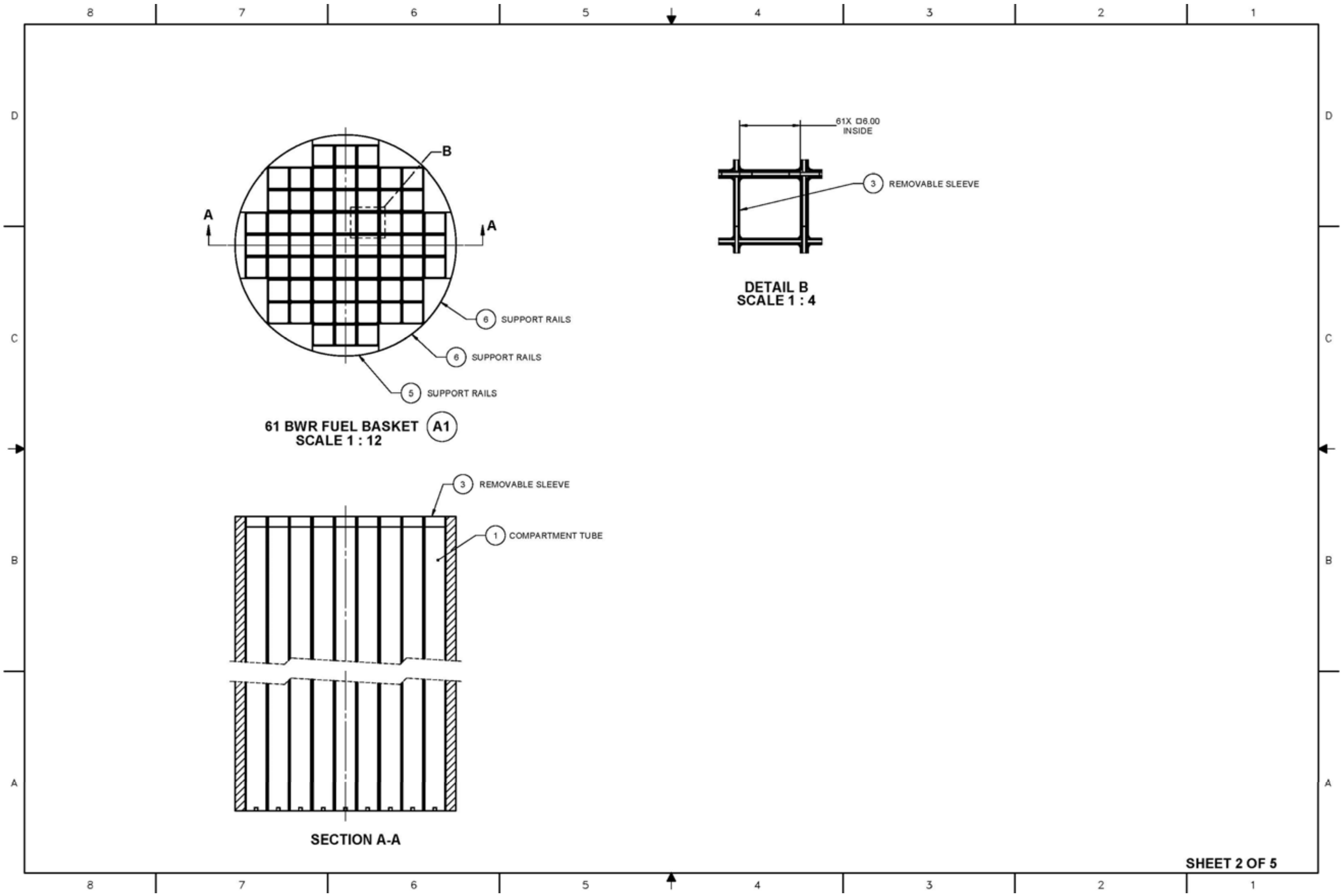


FIGURE 2.1-24: 61 BWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (3 OF 5)

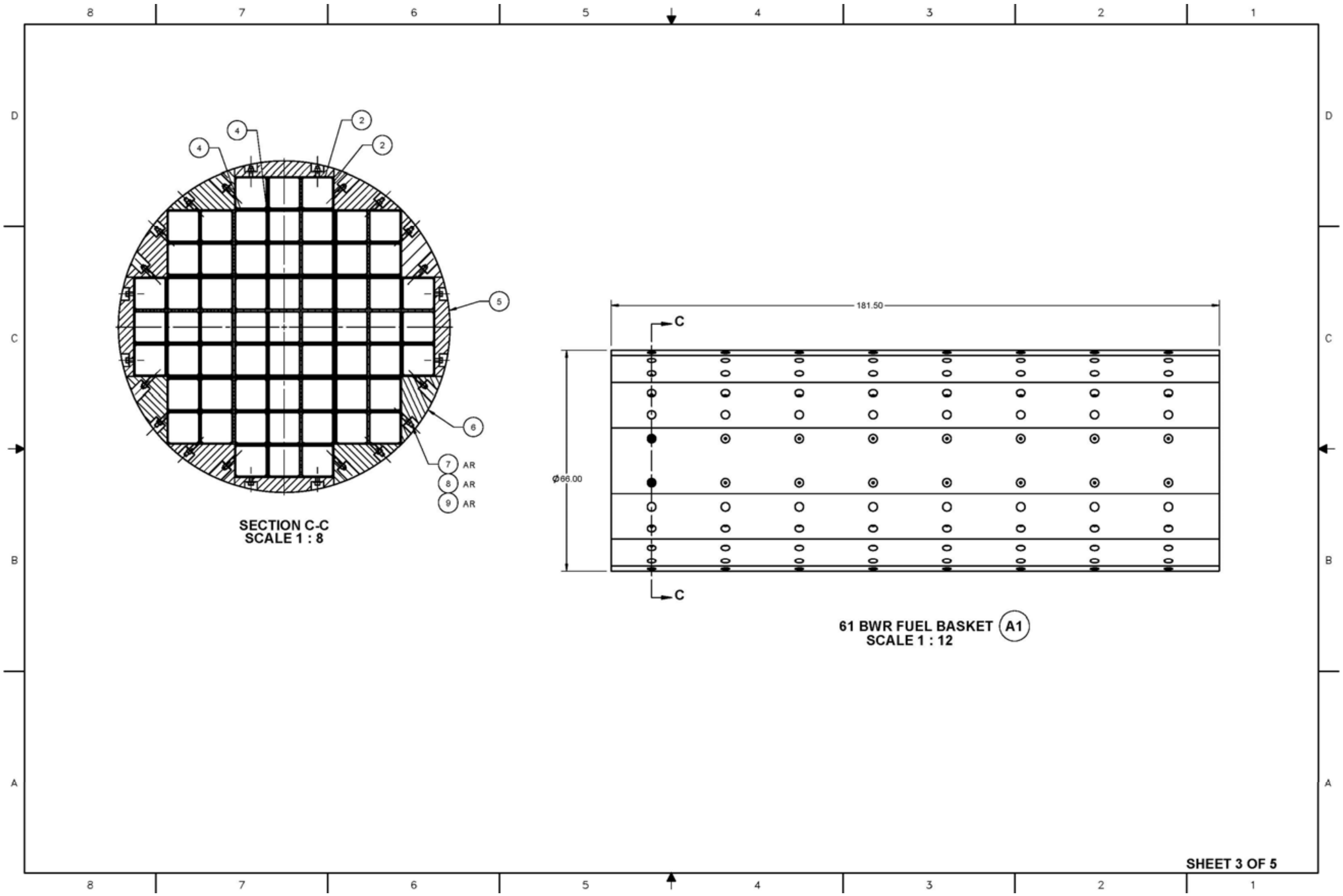


FIGURE 2.1-25: 61 BWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (4 OF 5)

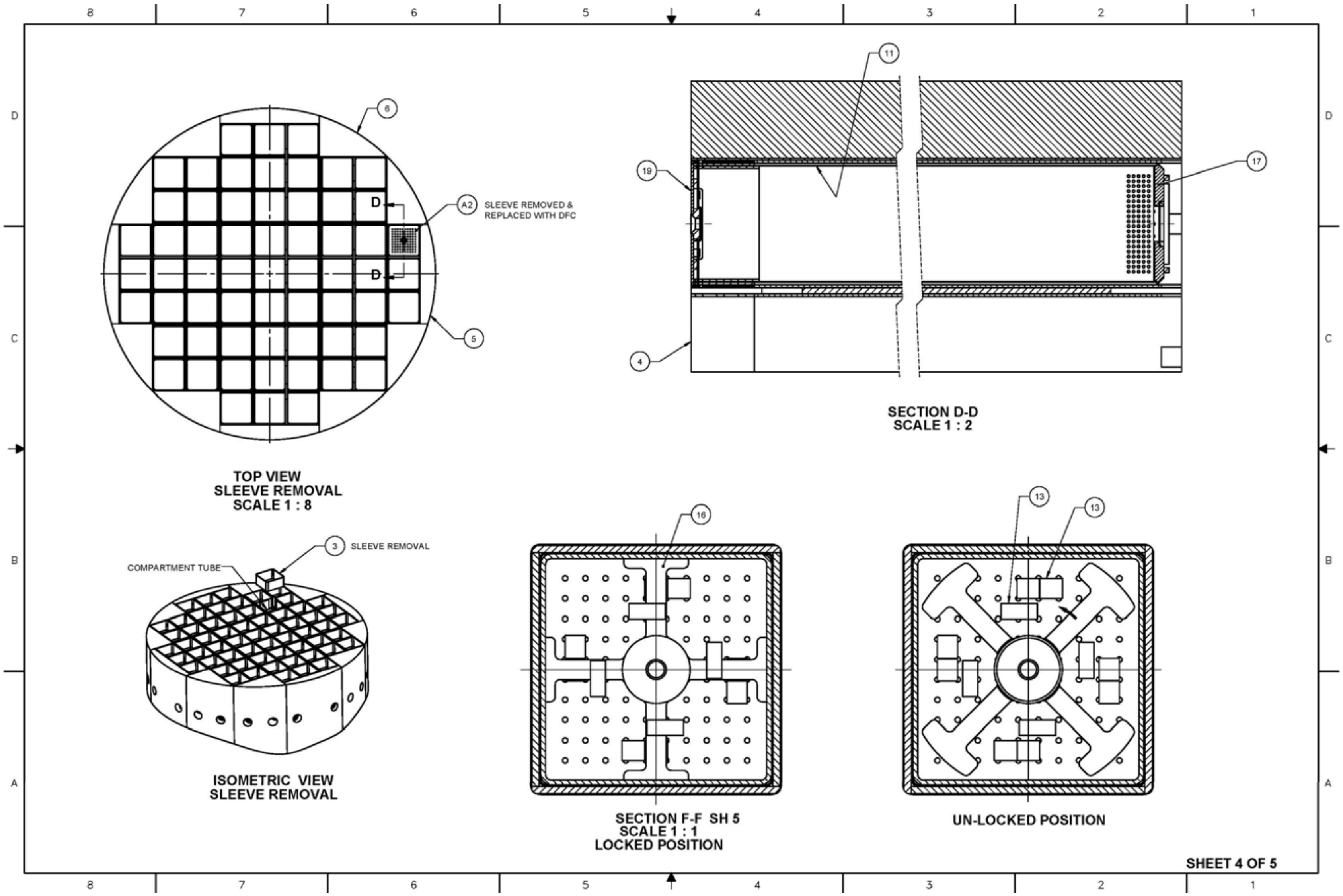
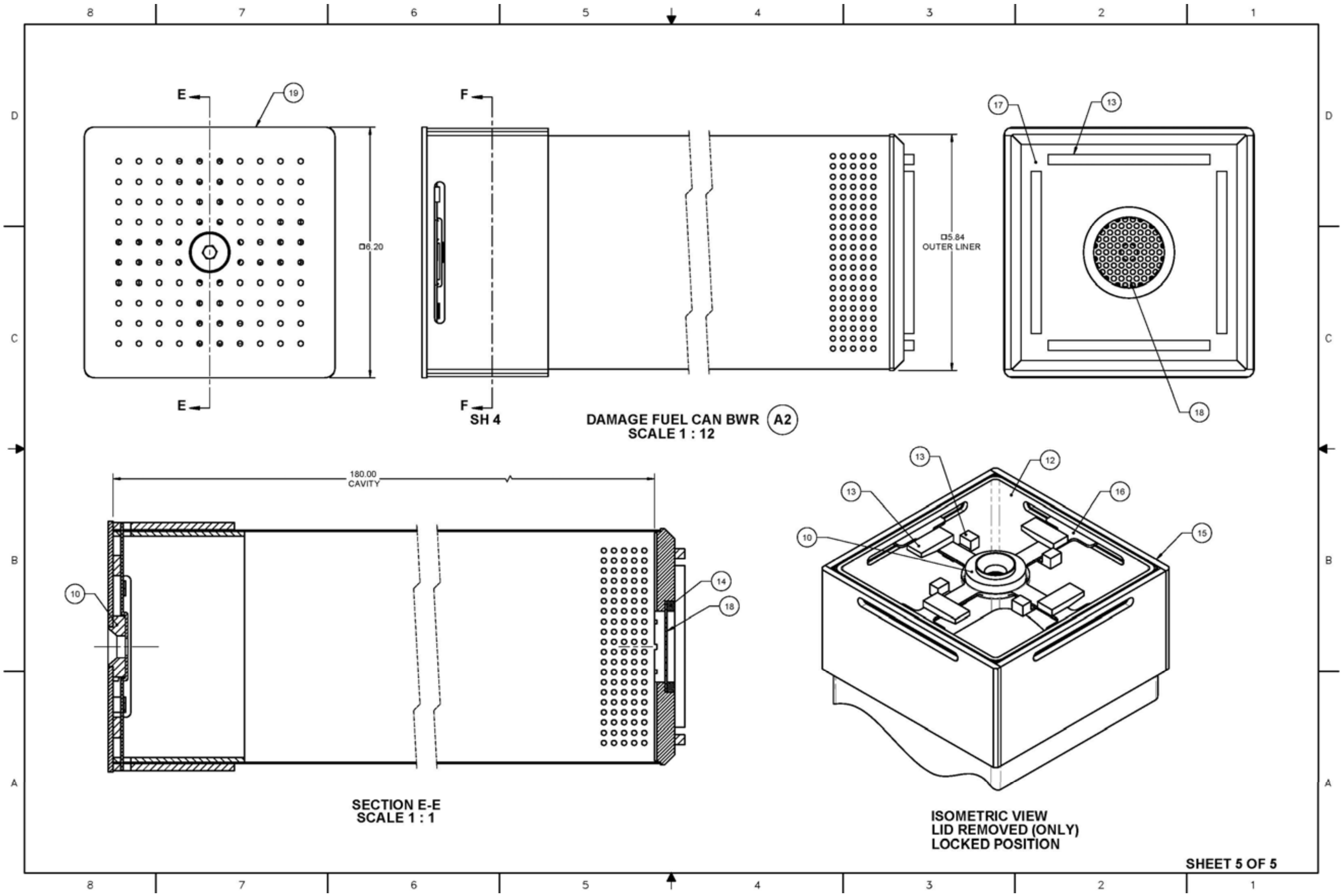


FIGURE 2.1-26: 61 BWR BASKET, CONCEPT 6625B-HB PACKAGE, TASK ORDER 17 (5 OF 5)



2.2 Structural Qualification for the 6625B-HB Cask

The structural evaluation of the 6625B-HB cask is shown in this section. *Section 2.2.1* discusses the cask and UNF capacity sizing method used in the initial stages of the design. This is followed by *Section 2.2.2*, which compares and evaluates the proposed design with previously licensed similar features. Finally, *Section 2.2.3* discusses the impact limiter design features supporting the structural evaluation of *Section 2.2.2*.

2.2.1 Initial Sizing of the 6625B-HB Cask

The 6625B-HB cask is designed to contain 24 PWR fuel assemblies or 61 BWR fuel assemblies. The capacity of the cask was determined as described below:

- One of the requirements was to propose a cask with a reasonable expectation it would be licensed by the NRC. This was accomplished by basing the design of the 6625B-HB cask features on existing (NRC-licensed) designs.
- AREVA has designed many transfer and several transport casks for high burnup fuel. One of the recently licensed (and currently starting production) casks is the MP197HB cask. This design was recently licensed by the NRC and thus has high relevance for this task.

The UNF capacity and cask-sizing effort proceeded as follows:

- A spreadsheet was created using the weight of the two ends of the MP197HB cask (plus the dry shielded canister (DSC) end shielding) and the weight of the central section per inch of length was determined. The central section included a 0.50-inch thick DSC wall plus 1.25-inch thick inner shell plus 3.00-inch lead, and 2.75-inch thick outer shell.
- The weight generated by the spreadsheet was verified by comparing it to the weight of the MP197HB cask as documented in the MP197HB weight calculation.
- The weight of the lifting yoke (based on previous designs) was estimated as 5 tons (since revised to 7,500 lbs). The weight of the lifting yoke was deducted from the maximum below the hook weight (125 tons) and the design weight of the cask was found.
- The weights for the fixed ends were then deducted from the design cask weight. This remainder was used to determine the maximum internal diameter (ID) for the cask that would meet the weight requirement.

These initial results used two different internal cavity lengths of 199.25 (from the MP197HB cask) and 188.5 from the MP187 cask. These initial results predicted either 16 or 18 fuel assemblies could be carried in the cask by weight. These estimates included water inside the cask when lifting from the pool.

Three cask IDs resulted from this weight estimate, Ø 68”, Ø 56”, and Ø 42”. Fuel layouts were prepared for each ID. It was demonstrated that for the 42-inch ID, 12 fuel assemblies would almost fit and that a minimum ID of 58 inches was required for 16 fuel assemblies.

Fuel basket layouts for 16, 18, and 19 fuel assemblies were generated. The layouts for 18 and 19 fuel assemblies were non-symmetric, which would make the thermal and shielding evaluations unnecessarily complicated. Symmetric fuel layouts may only be achieved with a limited number of fuel assemblies and in the area of interest for this cask, these are 16 or 24 fuel assemblies. Therefore, the goal was to fit 24 fuel assemblies in the cask.

The first step in trying to accommodate 24 fuel assemblies was to reduce the weight by reducing the inner wall shielding by the 0.50-inch DSC wall thickness. The ends of the cask were also refined such that the weight of the ends was a product of the cask ID. Before this point, the cask ends were fixed at the original MP197HB weight. It was found that the weight limit of 120 tons could only be met for 24 fuel assemblies with this design when the packaging cavity did not include water. Removal of the water (before the cask was lifted from the pool) allowed the cask to fit 24 fuel assemblies with both the upper and lower (removable) trunnions installed.

A cask with 24 fuel assemblies was deemed to be the new baseline.

The next step in the weight and cavity evaluation was to include an inner and outer lid, which supports a claim allowing for moderator exclusion in the accident case. The cavity length was standardized at 180.50 inches. A correlation was then made to existing (licensed) designs where 24 PWR assemblies were carried to establish the corresponding number of BWR assemblies, which was 61.

The inclusion of 61 BWR assemblies necessitated a slightly larger cask ID (increasing weight). The dual lid configuration also caused a weight increase. These combined changes increased the total weight such that the 120-ton limit was again exceeded.

At this point, the shielding in the ends was modified from that used in the MP197HB to more closely resemble the TN-40 cask. Additionally, lead was substituted for steel shielding in the lid and bottom end.

These changes allowed the cask to meet the 120-ton limit while containing either 24 PWR or 61 BWR assemblies with water in the cavity during lifting. This design is at the weight limit and proposed changes to the neutron shield required that water be removed from the cavity during lifting. The change to the neutron shield entailed extending a reduced height section of the neutron shield under an overlapping section of the impact limiter shell by approximately 22 inches. This weight increase necessitated the removal of water from the cavity.

The 180.5-inch length discussed above was based on direction that a 180-inch long (post irradiation) PWR fuel assembly was to be accommodated. The final design includes a cavity length of 182 inches, which also accommodates the thermal expansion of the DFCs.

The structural design features of the 6625B-HB cask are largely based on the MP197HB cask. There are also features adopted from other licensed packagings. The structural evaluation will thus largely rely on comparisons between the features of the 6625B-HB cask and the MP197HB cask.

Some of the structurally important differences between the 6625B-HB and the MP197HB are shown in **Table 2.2-1**.

TABLE 2.2-1: CONTRASTING CASK FEATURES

6625B-HB	MP197HB
Radial shielding provided by 1.25-inch thick inner shell, 3-inch thick lead gamma shield and a 2.75-inch thick outer (structural) shell.	In addition to the radial shielding described for the 6625B-HB, 0.50 inches of additional shielding is provided by the DSC shell.
Bottom is formed from a 2.75-inch thick forging with a 65.25-inch diameter by 4.50-inch thick lead gamma shield enclosed by a 1.25-inch thick inner shell. The RAM access opening and cover have been removed. This feature is not required since the fuel is not contained in a canister.	Bottom is formed from a 6.50-inch thick forging (additional shielding was provided by the various DSCs licensed for transport. The bottom of the cask includes an approximate 28-inch opening for manipulation of the DSC.
Dual lid Inner lid – 3 inches thick protected lid with 2.50-inch thick lead gamma shield enclosed in a 0.50-inch thick cylinder and a 1.00-inch thick bottom (closure) plate Outer Lid – 2.50-inch thick steel unprotected lid	Single 4-inch thick steel protected lid
Cavity Length – 182 inches to accommodate a 180-inch (post irradiated) intact fuel assembly enclosed in a DFC. The DFC is constructed of stainless steel. The additional length will accommodate the differential thermal expansion between the DFC and the cask.	Cavity Length – 199.25 inches
Cavity Diameter – 66.25 inches	Cavity Diameter – 70.50 inches
Loaded weight – 240,968 lb with 24 CE 16 X 16, System 80 fuel assemblies in DFCs. (42,480 lb payload)	Loaded weight – 275,500 lb with maximum (61,605 lb) payload

2.2.2 Structural Qualification / Comparison with Previously Licensed Cask Features

Common features between the MP197HB and the 6625B-HB include the materials of construction. Both the MP197HB and 6625B-HB casks are largely constructed from SA-203, Grade E carbon steel (a Nickel-alloy steel) and both casks use lead for additional gamma shielding.

There is a high degree of confidence that the proposed 6625B-HB cask design will be licensable. This confidence is based on the reliance on previously licensed features along with the smaller overall diameter, length, and lower weight of the 6625B-HB compared to the MP197HB.

The following sections will review the relevant design features for the 6625B-HB cask and provide justification for the acceptance of the structural design of these features.

The bounding stresses all result from a drop event and, except for the puncture drops, the accelerations experienced as a result of the event are dependent on the design of the impact limiters. The impact limiters (*Section 2.2.3*) are designed such that the accelerations experienced by the cask will, in general, be equal to or less than the accelerations experienced by the MP197HB cask. Where the predicted accelerations exceed those for the example design (generally the MP197HB) additional justification will be provided.

2.2.2.1 Inner and Outer Shells

The inner and outer shells of the 6625B-HB cask are 1.25 inches and 2.75 inches thick respectively and fabricated from SA-203, Grade E steel—the same thicknesses and material used in the MP197HB cask. The inner and outer shell will experience the largest stresses for side drop, slap down, and puncture accidents. The neutron shield shell (the 0.25-inch thick shell surrounding the neutron shield) is neglected in these analyses because its major purpose is to maintain / protect the neutron shield. It is not evaluated as a structural component of the packaging.

As discussed previously, the 6625B-HB cask has smaller inner and outer diameters (OD), a shorter length, and a lower weight than the MP197HB cask and the thickness of the outer shell is maintained. The result of these changes is a lower section modulus (due to the smaller diameter), and a lower bending load for both the side drop and slap down. The reduced bending is a result of both the lower weight and the shorter length.

The bounding condition for the NCT is a 1-foot side drop, and for the HAC, a 30-foot slap down drop. The predicted acceleration for the NCT side drop is greater than the acceleration used to evaluate the MP197HB. Ignoring the bending reduction due to the shorter length and lower weight, a positive margin is maintained for the NCT side drop.

The minimum margin is for the HAC drop and is 10 percent of the allowable stress. The shorter length and lower weight reduce the applied moment on the cask for a similar HAC drop, and will tend to reduce the resulting stress; therefore, the analysis performed in support of the MP197HB license is bounding for the 6625B-HB cask.

2.2.2.2 Upper End Structure

The upper end structure of the 6625B-HB cask is fabricated from the same SA-350, Grade LF3 material used for the MP197HB cask. Its thickness has been increased from about 7 inches thick to 7.75 inches thick to accommodate the dual lid configuration of this cask.

The bounding stress for the MP197HB was from an NCT load combination for a 1-foot side drop at a high environmental temperature with internal pressure. The bounding accident (HAC) load combination included a 30-foot 'slapdown' on the lid end with internal pressure (slapdown defined as a secondary impact). The NCT case has a 15 percent margin and the HAC has a 20 percent margin. The increased flange thickness with a lower overall weight and smaller cask diameter indicates the margin for the 6625B-HB cask will be higher than for the MP197HB cask.

2.2.2.3 Cask Bottom

The bottom of the 6625B-HB cask differs significantly from the MP197HB cask in that the 6625B-HB cask bottom consists of a 2.75-inch thick forged outer shell with a 4.50-inch thick lead gamma shield and a 1.25-inch thick closure plate, while the MP197HB cask bottom was a 6.50-inch thick steel forging. The lead gamma shield in the bottom of the 6625B-HB has a diameter smaller than the cask inside diameter and thus provides an improved load path to the cask sidewalls because the lead is not in the direct load path. The bottom forgings are fabricated from SA-350, Grade LF3 steel for both the MP197HB and 6625B-HB casks.

The bottom forging of the MP197HB cask was validated through analysis using ANSYS®. The design of the 6625B-HB cask will be validated using manual calculations. The two cask bottoms will be modeled as circular flat plates of constant thickness, fixed at the edge and with the load applied as a uniform pressure. This corresponds to Case 10b from Table 24 of Roark's formulas for Stress and Strain, Sixth Edition. The manual calculation performed using the load inputs and geometry of the MP197HB will be compared to the results of the Finite Element Analysis (FEA) and a correlation developed.

The analysis inputs for the finite element analysis of the MP197HB, with the exception of the edge restraints, are the same as described above (a flat plate of constant thickness with the load applied as a uniform pressure). The loads applied to the bottom of the MP197HB cask include both an internal pressure from the payload and an external pressure from the impact limiter support. In lieu of performing two manual calculations and subtracting the results, the net pressure (the difference between the pressure due to the payload and the pressure due to the impact limiter reaction) will be used in the calculation.

The two pressures used in the analysis of the MP197HB cask are 1849.1 psi, downward, due to the payload and 2126.5 psi, upward, due to the support provided by the impact limiter. The difference in these pressures is 277.4 psi and would be applied in the upward direction. These pressure loads are developed based on the 55g acceleration for the MP197HB end drop.

Using the method of Roark's, the uniform 277.4 psi pressure is applied to the 6.5 inch thick, 70.5 inch diameter bottom plate of the MP197HB. The resulting bending stress is combined with the shear stress determined for the same load on the perimeter of the plate. These stresses are combined to a stress intensity of 6.8 ksi.

The stress result reported for the bottom plate of the MP197HB cask is 10.5 ksi. The result of the FEA is approximately 55% higher than the manually calculated result. This correlation will be used to qualifiy the design of the bottom of the 6625B-HB cask.

The pressure loads developed for the MP197HB are based largely on the ratio of the component weights and are used to develop a load distribution. Using the same method employed for the MP197HB to develop the distributed load for the 6625B-HB cask, the internal pressure (due to the payload) is determined to be 948 psi and the external support pressure is 1027 psi. The nominal loads used to develop these pressures are multiplied by the 40 g end drop acceleration predicted for the 6625B-HB cask. The resulting differential pressure for the 6625B-HB cask is 79 psi and as for the case of the MP197HB cask, the net pressure is upward.

The configuration of the bottom of the 6625B-HB cask, as discussed earlier, includes three sections. First is the inner bottom plate (1.25 inches thick), the second is the lead gamma shield (4.5 inches thick) and third is the outer bottom plate (2.75 inches thick). As determined above, the net load on the bottom of the 6625B-HB cask is upward.

Calculating the stress in the outer bottom plate, using the same Roark's method, the calculated stress intensity (SI) is 8.6 ksi. Applying the correlation factor determined above for the MP197HB, the resulting stress is 13.3 ksi. The resulting margin is approximately 3.7.

The construction of the bottom necessitates an additional consideration. The multi-layer bottom allows for a gap between the inner or outer bottom plate and the lead gamma shield. For a cask fabrication, this gap is assumed to be limited to 0.125 inch (in an actual fabrication, this value would be specified). The significance of the gap is that while the net pressure would cause the thicker outer bottom plate to support the inner bottom plate, this support would not occur until the gap is closed through deflection of the inner or outer bottom plates (or both).

Performing the same Roark calculation on the 1.25 inch thick inner bottom plate determines that a pressure of 33 psi is required to close the 0.125 inch gap. This pressure results in a stress of 17 ksi is much less than the allowable stress for the inner bottom plate. Following the closure of the gap, the inner bottom plate is supported by the outer bottom plate through the lead gamm shield.

In conclusion, the design of the bottom of the 6625B-HB cask is sufficient to sustain the expected loads.

2.2.2.4 Cask Lid(s)

The configuration of the lids of the 6625B-HB and the MP197HB is different in that the 6625B-HB cask has two lids, an inner 3.00-inch thick lid and an outer 2.50-inch thick lid, while the MP197HB cask has a single 4.50-inch thick lid. The lids for both casks are fabricated from SA-203, Grade E steel or SA-350, Grade LF3. Spacer blocks are attached to the top of the inner lid such that contact with the outer lid occurs following a small initial deflection.

An additional difference is that the inner lid of the 6625B-HB cask also has a 2.50-inch thick lead gamma shield enclosed and attached by a 0.50-inch thick cylinder and a bottom 1.00-inch thick closure plate.

2.2.2.4.1 Inner Lid

The construction of the inner lid (neglecting the lead gamma shield but including the steel gamma shield enclosure) results in a section modulus more than 16 percent larger than that of the lid of the MP197HB cask. The inner lid would not be subjected to puncture. The loads on the inner lid include the payload weight and internal pressure. In the case of a lid down end drop (HAC), the inner lid is supported, following a small initial deflection, by the outer lid and the impact limiter as is the case for the MP197HB cask. The inner lid is not subjected to the puncture loads or external pressure loads present for the lid on the MP197HB cask.

2.2.2.4.2 Outer Lid

The outer lid is 2.50 inches thick and is described as an unprotected lid for purposes of lid bolt analysis. The lid is described as unprotected since it is not located within a recess in the upper flange of the cask. Unprotected lids can cause shear to be experienced by the lid bolts; however, the outer lid for the 6625B-HB cask has an inner shear lip. This shear lip will react such that shear loads are not experienced by the lid bolts.

Nominally, there is a gap between the inner and outer lids. The size of this gap is minimized and controlled by attaching spacer blocks to the face of the inner lid such that the deflection of the inner lid is minimized and support is provided through the outer lid and ultimately the impact limiter, as is the case for the MP197HB cask.

The lid will be subjected to puncture as discussed in *Section 2.2.2.4*. Calculating the required lid thickness, using Nelm's equation and conservatively neglecting the protection offered by the lid end impact limiter, the minimum required lid thickness is determined to be 2.492 inches. The thickness of the outer lid is 2.5 inches and is therefore acceptable. This evaluation conservatively assumes an unyielding puncture bar.

2.2.2.4.3 Conclusion

The stresses reported for the lid of the MP197HB are local stresses with the minimum margin reported as 48 percent for an accident (HAC) load combination including a 30-foot slap down with external pressure.

The stiffer construction of the inner lid for the 6625B-HB combined with the lower load on the inner lid (smaller payload, smaller diameter, greater section modulus and the same internal pressure) support the expectation the margins for the inner lid will be larger than those shown for the lid on the MP197HB. The inner lid is also supported, after a small initial deflection, by the outer lid and impact limiter.

The outer lid, while thinner than the lid of the MP197HB cask, is not required to react the same load as the lid on the MP197HB cask. The outer lid, by itself, is of sufficient thickness to withstand the HAC puncture. The margin for the 6625B-HB cask lids is not expected to be lower than the margin reported for the MP197HB cask.

2.2.2.5 Lid Bolts

The lid bolts used for both the inner and outer lids of the 6625B-HB cask are the same material, size, and quantity (for each lid), as are used on the MP197HB cask.

The outer lid is supported by the impact limiter in a manner similar to the MP197HB cask. The inner lid is protected as is the lid on the MP197HB cask. The outer lid, while unprotected, incorporates a shear lip. The shear lip will transfer the shear load directly to the cask upper flange thus serving the same function as a protected lid in eliminating bolt shear loads.

The payload for the 6625B-HB cask is less than the MP197HB cask payload and the diameter for the inner lid is less that for the MP197HB. The diameter of the outer lid is larger than the lid on the MP197HB, however, the supported diameter is smaller (the inside diameter of the inner O-Ring) and the external pressure is the same supported by the MP197HB lid. Each lid (inner and outer) on the 6625B-HB cask is fastened with 48 bolts. Each lid supports a lower load than the lid on the MP197HB. Therefore, the minimum margin for the lid bolts is expected to be significantly higher than the 1.11 margin for the MP197HB lid bolts.

2.2.2.6 Trunnions

The front and rear trunnions on the 6625B-HB cask are employed only at the loading and unloading facilities. The trunnions are removed after the cask is placed on the skid to are replaced with shielding plugs. The shielding plugs provide additional neutron shielding and removal of the trunnions prevents their use as tiedowns during transport.

The upper and lower trunnions are of identical design to those used on the MP197HB cask and perform the same functions (the lower trunnions are used to support and guide the cask during rotations to/from the horizontal orientation on the transport skid and the upper trunnions are used to lift and transport the loaded and unloaded cask).

The trunnions are of the same design as the MP197HB trunnions and support a lower weight, therefore, the margins are expected to be higher than those determined for the MP197HB cask.

2.2.2.7 Fuel Baskets

6625B-HB PWR Basket

- Designed using the licensed NUH24PTH Basket structural components
- Transition rail construction revised to solid rails
- Length increased

6625B-HB BWR Basket

- Designed using the licensed NUH61PTH Basket structural components
- Transition rail construction revised to solid rails

The baskets are designed using the same components used in the design of previously licensed fuel support baskets except the support rails were changed to solid sections.

The overall cask length of the 6625B-HB cask is shorter than the MP197HB cask. This shorter length will result in lower accelerations for the slap down accident drops. Additionally, the impact limiters are designed, in general, so that the accelerations experienced by the lower weight 6625B-HB cask will be the same or lower than for the MP197HB cask. Where the predicted accelerations exceed those for the example design (generally the MP197HB) additional justification will be provided.

The baseline fuel baskets are structurally acceptable. The baskets designed for the 6625B-HB cask utilize the same or better components (solid rails); therefore, the 6625B-HB baskets will be acceptable.

2.2.3 Impact Limiter Evaluation

2.2.3.1 Introduction

This section contains the evaluations of the 6625B-HB cask impact limiters to ensure they mitigate the worst case free fall conditions for both the NCT and the HAC requirements prescribed by 10 CFR 71 [1]. This section also contains the consideration of different energy absorbing materials used in licensed used fuel cask designs and the evaluation of the impact limiter attachment bolts.

2.2.3.2 Impact Limiter Design Basis

To ensure the impact limiters have reasonable assurance of being licensed by the NRC, the impact limiter design will use the licensed MP197HB UNF rail cask impact limiters as a design basis. This design basis was chosen for the following reasons:

- The licensed weight of the MP197HB is 152.0 tons (see Section A.1.2.1.1 of [13]), which is 12 percent greater than the maximum transport weight of the 6625B-HB at 132.6 tons (cask weight with the impact limiters from **Table 2.1-2**).
- The outer diameter of the MP197HB cask is 84.50 inches with an impact limiter diameter of 126 inches. The outer diameter of the 6625B-HB cask is nearly identical at 80.25 inches, with the same 126-inch impact limiter outer diameter. This impact limiter outer diameter design is within the allotted profile requirements.
- The maximum thermal loading of the MP197HB is 32 kW. This heat load bounds the heat load of the 6625B-HB which is 30.4 kW. The MP197HB impact limiter design is able to meet the 10 CFR 71 requirements for both NCT and HAC.

2.2.3.3 Energy Absorption Material Comparison

The energy absorption materials in the MP197HB are primarily redwood and balsa. These materials are well understood and have been used in past licensed UNF transport cask impact limiter designs. The other two impact mitigating materials that are more commonly used in NRC licensed UNF transport cask designs are polyurethane foam and aluminum honeycomb.

Polyurethane foam is a well-characterized material that is used in many radioactive material packaging designs. Its material crush strength does, however, vary significantly with temperature and has an NCT temperature limit approximately equal to its glass transition temperature of 279°F [29]. Polyurethane foam has a thermal conductivity of 0.002875 Btu/hr-in-°F (for 25 lb per cubic foot [29]). Redwood in the NCT thermal model has thermal conductivity of 0.0378 Btu/hr-in-°F (see **Table 2.3-5**) and the model reports a peak temperature of approximately 285°F in the wood impact limiter (see **Table 2.3-2**). This means that the peak NCT temperature in a foam impact limiter would be higher than this value. Thus, it is expected to exceed the recommended material temperature limit for the region near the cask and is therefore not a recommended material for this design.

Biaxial aluminum honeycomb has also been used in licensed impact limiter designs such as the NUHOMS-MP187 Multipurpose Cask (referred to herein as MP187) [30]. It can be specified with material crush strengths similar to that of redwood and has a similar linear crush profile up to a crush strain of about 55 to 80 percent of the original height [31]. This material also has a recommended long-term service temperature limit of 350°F [32] and has a bulk thermal conductivity of more than 0.715 Btu/hr-in-°F when using densities above 8 lb/ft^3 [32]. This is more than 10 times the thermal conductivity of wood.

Aluminum honeycomb, however, does have two drawbacks that must be considered. The first drawback is that that required crush strength of the aluminum honeycomb in the 6625B-HB is significantly different than that of the aluminum honeycomb licensed MP187. This difference in required crush strength is driven by the difference in cask geometry and weight. Therefore, the aluminum honeycomb in the 6625B-HB would require a testing program to a level that is able to support the package licensing by the NRC. The second drawback is that, in past packaging experience, the cost of using aluminum honeycomb in impact limiters has been significantly higher compared to other energy absorption materials and techniques.

Therefore, the 6625B-HB cask impact limiters will be constructed entirely out of balsa and redwood like the MP197 and MP197-HB UNF rail casks.

2.2.3.4 6625B-HB Impact Limiter Geometry

The impact limiter profile geometry evaluated is very similar to the MP197HB in that it has an air gap at the end of the cask and thermal shields between the cask and impact limiters. One significant difference between the 6625B-HB impact limiter and the MP197HB impact limiters is that a slightly thinner neutron shield continues underneath the impact limiter as shown in **Figure 2.1-3** and **Figure 2.1-4**.

The internal wood configuration evaluated is based on the impact limiter design of the MP197HB due to the similarity in weight and size of the cask to the 6625B-HB.

To maximize the stroke, and minimize the acceleration forces, the outer diameter of the impact limiter was selected to be 126 inches. This outer diameter of the impact limiter is the same as specified on the MP197HB [13], and will allow the impact limiter to remain under the 128-inch profile. The length of the impact limiter was maintained from the MP197HB.

The impact limiter outer stainless steel shell thickness and steel gusset type geometry was also maintained from the MP197HB SAR design basis with minimal changes.

2.2.3.5 Free Drop Orientations and Response Criteria

For NCT, the free drop orientations considered will be a 1.00-foot end drop on both the lid and bottom end of the cask, and a 1.00-foot side drop. In past licensing of similar UNF packages, the worst case 30 foot HAC drop orientations considered in safety analysis reports are as follows: a package side drop, end drop, center of gravity (cg) of over cask cover drop, and an angled drop with a slapdown. Since the 6625B-HB is symmetric, no difference is made between the lid and bottom end of the package for this evaluation.

The acceleration design criteria experienced at the center of gravity of the package during both these conditions are shown in **Table 2.2-2**:

TABLE 2.2-2: HAC IMPACT ACCELERATION DESIGN CRITERIA

Drop Orientation		Targeted g-loads at center of gravity of the cask
30 ft End Drop		55
30 ft Side Drop		55
30 CG Over Corner Drop		40
30 ft Slapdown	1st Impact	Translation: 25, Rotation: 132 rad/s
	2nd Impact	Translation: 32, Rotation: 166 rad/s
1 ft end drop		18
1 ft side drop		19

These accelerations are based on the MP197HB impact response (see A.2.13.12.10 of [13]) and are cited as being bounding for the 6625B-HB structural evaluation in *Section 2.2.2*.

2.2.3.6 Free Drop Response Evaluation

Side HAC and NCT Free Drop

In the side orientation, the 6625B-HB has less stroke available than the MP197-HB due to the neutron shield extending underneath the impact limiter. This orientation will be analyzed using an AREVA proprietary limiter scoping method to give reasonable assurance that the impact limiter response is bounded the MP197-HB.

End HAC and NCT Free Drop

Due to the reduced weight and diameter of the cask, this orientation is evaluated using an AREVA proprietary limiter scoping evaluation method to give reasonable assurance that the impact limiter response is bounded the MP197-HB.

Center of Gravity (CG) over Corner HAC Free Drop

The 6625B-HB impact limiter corner geometry is not changed from the MP197-HB design. With lower impact responses confirmed for both the end and side drops, the cg-over corner free drop orientation is justified in being assumed less severe than the MP197-HB results in **Table 2.2-2**.

Slapdown HAC Free Drop

The 6625B-HB is shorter and lighter than the MP197-HB. The 6625B-HB also has a similar HAC side drop response. Therefore, the 6625B-HB slapdown drop response (both primary and secondary) is justified in being assumed less severe than the MP197-HB results in **Table 2.2-2**.

Scoping Evaluation Comparison

The AREVA proprietary impact limiter scoping evaluation is based on conservation of energy principles. This evaluation method was compared against the MP197 1/3 scale test results published in Section 2.10.9 of [13] to establish a material bias factor. The evaluation was also compared to the MP197 NCT free drop predictions documented in published in Table 2.10.8-14 of [13]. The results of this comparison are documented in **Table 2.2-3**. The predicted accelerations are within ±5 g and the predicted crush depth is within 0.5 inches of the MP197 test results.

TABLE 2.2-3: EVALUATION COMPARISON

Drop Orientation	MP197 Data	Evaluation Results	Difference
30 ft Side Drop	61 g 8.25 inches	57.4 g 7.97 inches	-3.6 g -0.3 inches
1 ft side drop	24 g Not Specified	24.5 g 0.7 inches	0.5 g NA
30 ft End Drop (-20°F)	65 g 7.50 inches	62.6 g 7.73 inches	-2.4 g 0.23 inches
1 ft end drop	10 g Not Specified	12.2 g 1.1 inches	2.2 g NA

Scoping Evaluation Results

Using the material bias factor established in the comparison, the 6625B-HB scoping analysis results in **Table 2.2-4**, show the anticipated accelerations for both configurations are within ±5 g's of the MP197HB targeted accelerations in **Table 2.2-2**. The predicted acceleration for the NCT side drop is greater than the acceleration used to evaluate the MP197HB. However, due to the shorter length and lower weight of the 6625B-HB, a positive margin is maintained for the NCT side drop.

TABLE 2.2-4: 6625B-HB IMPACT EVALUATION

Drop Orientation	6625B-HB g-loads at center of gravity of the cask	Targeted g-loads at center of gravity of the cask
30 ft Side Drop	49	55
1 ft side drop	22	19
30 ft End Drop (-20°F)	40	55
1 ft end drop	20	18

The results of the scoping evaluation are only for assurance of the ability to license the 6625B-HB. A formal analysis, using finite element techniques, of the impact response of the 6625B-HB would need to be performed to satisfy NRC licensing basis.

2.2.3.7 Impact Limiter Bolt Evaluation

The 6625B-HB impact limiters are fastened onto the cask with 12 custom socket-head cap screws, referred to herein as impact limiter bolts. These impact limiter bolts enter through the outer end of the impact limiter through 12 holes spaced between the impact limiter gussets (shown in *Section 2.1.5* in **Figure 2.1-10**) and fasten to the cask.

The design of the impact limiter bolts also uses the MP197HB as a licensed design basis. Since the HAC impact acceleration shown in **Table 2.2-4** are nearly identical to those in **Table 2.2-2**, and the 6625B-HB is lighter, there will be significantly lower loads on the impact limiter bolts. Therefore, it can be concluded that using a similar bolt design to the MP197HB for the 6625B-HB will be sufficient.

2.3 Thermal Review

This section is used to describe the thermal design aspects and limitations of the 6625B-HB high burnup fuel transportation package. The evaluations presented within this section are intended to show that the package will comply with the thermal requirements of 10 CFR 71 [1] as a Type B(U)F-96 shipping container when used to ship irradiated PWR and BWR fuel assemblies.

Specifically, the package is shown to remain within the temperature limits under NCT (10 CFR §71.71), retain sufficient thermal protection following the HAC free and puncture drop scenarios to maintain all package component temperatures within their respective short-term limits during the regulatory fire event and subsequent package cool-down (10 CFR §71.73), and that the maximum temperature of the accessible package surfaces remains less than 185°F for the maximum decay heat loading, and an ambient temperature of 100°F, in the shade (10 CFR §71.43(g)).

Applicable design features are identified and described as they pertain to this section. Materials, properties, and component specifications used in construction of the package are presented to ensure compliance with thermal limitations. The maximum decay heat loads are identified for the various payload configurations. Component temperatures and the maximum internal pressures based on the maximum loading are shown for NCT and HAC.

2.3.1 Description of Thermal Design

2.3.1.1 Design Features

The package is designed to transport 24 PWR or 61 BWR fuel assemblies; either bare or in DFCs. A full description of the design is found in *Section 2.1* of this report.

The package is designed to reject heat passively primarily through conduction and radiation internally and through convection and radiation externally. The package has sufficient thermal mass to mitigate the effects of thermal transients. Component materials are selected to meet long-term NCT and short-term HAC thermal loadings. The impact limiters provide thermal protection to each end of the package. Pressure relief is not incorporated into the package containment boundary.

2.3.1.1.1 Cask Body

The cask body consists of an inner and outer right circular cylinder shell of plate or forged steel. The two shells are concentrically aligned and connected at both ends with an upper and lower steel forging. The space formed between the two shells is filled with a gamma shield of commercial quality lead.

A neutron shield, formed out of a polymer resin, encased in copper walled tubes, surrounds the outer shell. The resin-filled tubes extend along the length of the cask and have a trapezoidal cross section. The sidewalls of the tubes allow for heat transfer from the outer cask shell to an outer skin of stainless steel.

2.3.1.1.2 Cask Closures

The primary cask closures consist of an inner and outer closure lid. Multiple penetrations through both the inner and outer lids are provided to allow for testing, draining, and purging of the cask. An O-ring seal in the confinement boundary is provided for each closure lid. A second O-ring is located on the lid flange of each lid to allow for testing of the containment boundary. Both O-rings are laid out in concentric channels on the lower flange surface of each lid.

2.3.1.1.3 Impact Limiters

Two impact limiters are used to reduce the impact forces on the cask found in the HAC 30-foot drop and puncture bar drop damage. The impact limiters also provide thermal insulation to the ends of the cask during the 30-minute HAC fire event. Each impact limiter completely encloses either the top or bottom end of the cask providing thermal insulation.

Each impact limiter consists of a closed-end cylindrical shell of Type 304 stainless steel enclosing a composite wood core of redwood and balsa wood. The cylindrical stainless steel shell lines the recess incorporated in one end of each impact limiter to accept the cask. Twelve Type 304 stainless steel gussets travel radially through the impact limiter to provide rigidity and to assist in the rejection of excess heat accumulating in the impact limiter. The wood core of the impact limiter is isolated from the hotter portions of the steel shell with a layer of ceramic fiber paper.

2.3.1.1.4 Thermal Shield

A thermal shield is incorporated into the design between each end of the cask and its respective impact limiter. The heat shield is fabricated from 6061-T6 aluminum sheets and creates a 3.00-inch thick air layer between the end of the cask and its inner surface. The heat shield extends through a cylindrical portion along the inner recess of the impact limiter and extends a flat edge outside of the cask boundary to reject excess heat.

2.3.1.1.5 Fuel Baskets

Two fuel baskets used to accommodate BWR and PWR fuel assemblies, and BWR and PWR DFCs are considered for this design. The fuel basket configurations include:

- 24 PWR Fuel Assembly Basket
- 61 BWR Fuel Assembly Basket

Each fuel basket cell is constructed with Type 304 stainless steel sheet or tube. The cells are imbedded in a lattice of 6061-T6/T651 aluminum plate used to reduce the weight and to improve heat transfer of the basket. 6061-T6/T651 extruded aluminum rails are used to transition the basket lattice to the cylindrical surface of the cask inner shell.

2.3.1.2 Decay Heat

The package is designed to safely contain either 24 PWR or 61 BWR fuel assemblies with a combined thermal load of no more than 30.4 kW. To effectively distribute heat and maintain criticality and shield requirements, the fuel assemblies must be loaded based on a multiple zone layout in each basket. There are four zones in each fuel basket. The zone distributions for both fuel baskets are provided in *Section 2.1.2.2*.

Two loading configurations are analyzed for each payload type: a primary configuration that is selected to maximize the number of fuel assemblies transported and an alternate configuration to maximize the individual fuel assembly heat load. The allowed PWR and BWR fuel thermal loading configurations per zone are listed in **Table 2.3-1**. While the primary configuration for each basket (config. 1 for both PWR and BWR) is shown the alternate configurations may require additional elaboration.

The alternate configuration for the PWR uses the same loading zones with the fuel assemblies loaded into zones 1 through 3 limited to a maximum decay heat of 1.9 kW. The eight outer cells of zone 4 are loaded with dummy fuel assemblies with no internal heat load. This limits the combined thermal load of the package to the 30.4 kW limit. The alternate PWR heat load layout is provided in Figure 2.5-9.

The alternate configuration for the BWR uses the same loading zones with all fuel assemblies loaded into the basket limited to 600W of decay heat. Sixteen dummy fuel assemblies are loaded into all cells of zone 4 and the four corner locations of zone 3. This limits the combined thermal load of the package to 27.0 kW. The alternate BWR heat load layout is provided in Figure 2.5-10.

TABLE 2.3-1: SUMMARY OF PERMISSIBLE FUEL BASKET LOADING

Payload	Zone	Number of Cells Per Zone	Max Decay Heat Per Assembly (kW)	Max Decay Heat Per Zone (kW)	Max Decay Heat Per Payload (kW)
PWR Fuel	1	4	0.9	3.6	30.4
	2	8	1.4	11.2	
	3	4	2.1	8.4	
	4	8	0.9	7.2	
PWR Fuel (Alt. Config.)	1	4	1.9	7.6	30.4
	2	8	1.9	15.2	
	3	4	1.9	7.6	
	4	0(1)	0	0	
BWR Fuel	1	9	0.33	3.0	30.3
	2	16	0.78	12.5	
	3	24	0.45	10.8	
	4	12	.33	4.0	
BWR Fuel (At. Config.)	1	9	0.6	5.4	27.0
	2	16	0.6	9.6	
	3	20(2)	0.6	12.0	
	4	0(3)	0	0	

Notes: (1) In the alternate PWR configuration the zone 4 fuel assemblies are replaced with aluminum spacers.
(2) In the alternate BWR configuration zone 3 is short-loaded by four assemblies with dummy fuel assemblies inserted as spacers.
(3) In the alternate BWR configuration the zone 4 fuel assemblies are replaced with aluminum spacers.

2.3.1.3 Summary Tables of Temperatures

A summary of the maximum predicted temperatures under NCT and HAC for the package is presented in **Table 2.3-2**. The NCT temperatures are based on a steady-state analytical model of the package with an ambient temperature of 100°F and the 10 CFR §71.71(c)(1) [1] prescribed insolation averaged over 24 hours.

The maximum predicted temperatures under the HAC fire event for the package are determined using a transient simulation of the package. Deformation of the package is conservatively selected to bound predicted damage to the impact limiters. Damage to the other packaging components is assumed to be negligible.

TABLE 2.3-2: SUMMARY OF TEMPERATURES FOR NCT AND HAC (°F)

Location / Component	Temperature		Maximum Allowable		Margin	
	NCT	HAC	NCT	HAC	NCT	HAC
Fuel Cladding[1]	582	860	752	1,058	170	198
Basket Steel Plates	528	773	800	800	272	27
Basket Aluminum Plates[1]	528	773	550	N/A	22	N/A
Basket Aluminum Rails[1]	401	530	550	N/A	149	N/A
Helium Fill Gas (average) [1]	409	471				
Inner Shell	348	518	700	2,600	352	2,082
Gamma Shield (Lead)	346	515	620	620	274	105
Outer Structural Shell	314	511	700	2,600	386	2,089
Neutron Shield Resin	296		320		24	
Neutron Shield Box	298	882	1,983	1,983	1,685	1,101
External Skin	299	1,151	800	2,600	501	1,449
Bottom Forging	335	231	700	2,600	365	2,369
Upper Forging	212	447	700	2,600	488	2,153
Inner Lid	208	417	700	2,600	492	2,183
Outer Lid	207	412	700	2,600	493	2,188
Impact Limiter Shell	291	1,473	800	2,600	509	1,127
Impact Limiter Gusset	288	1,459	800	2,600	512	1,141
Impact Limiter Insulation	290	1,319	2,300	2,300	2,010	981
Impact Limiter Wood - Maximum - Average	 285 145	 1,472 427	 320 212	 N/A N/A	 35 67	
Thermal Shields	292	1,409	550	N/A	258	
Maximum Surface Temperature[2]	269/126		185		59	
Inner Lid Seal	208	417	400	482	192	65
Outer Lid Seal	207	412	400	482	193	70

Notes:
(1) Values are bounding for both fuel baskets
(2) Surface temperature shown with and without personnel barrier. The temperature with the personnel barrier installed is taken at Impact Limiter exterior surfaces.

The NCT results demonstrate that a significant thermal margin exists for all of the containment boundary components. The aluminum basket components, basket steel plates, neutron shield resin, and the impact limiter wood margins are also positive but small enough to limit the maximum payload heat to 30.4 kW. The temperature margins could be improved by the addition of fins on the outer surface of the package.

The maximum temperatures of the neutron shield resin remains below the maximum allowable of 320°F. The maximum surface temperature for accessible surfaces, 185°F, as permitted by 10 CFR 71.43(g) for NCT exclusive use, is 185°F and is exceeded. However, the maximum accessible surface temperature drops well below the limit with the use of a personnel protective barrier.

The HAC simulation results demonstrate that the containment boundary components remain within their accident condition allowable temperatures. Note that the components that exceed their allowable temperatures are not required to meet containment during and after the fire.

2.3.1.4 Summary Table of Maximum Pressures

Table 2.3-3 presents the maximum cavity pressures for both NCT and HAC. As a comparison, the analysis of the MP197HB package requires that the maximum NCT and HAC pressures are 15 psig and 140 psig respectively. Based on a gas fill temperature of 70°F and a fill pressure of 14.7 psi, the maximum NCT cavity pressure will be less than 15 psig. Similarly, based on initial results, the maximum HAC cavity pressure is predicted to remain less than 140 psig.

TABLE 2.3-3: MAXIMUM PRESSURES

Load Condition	Cask Cavity Pressure (psig)
NCT Hot	11.0
HAC Hot	67.7

2.3.2 Material Properties and Component Specifications

2.3.2.1 Material Properties

The thermal material properties used in this evaluation may be found in Table 2.3-4 and Table 2.3-5. Structural steels used in fabricating the packaging include ASME SA-350, Type LF3 and SA-203 Grade E carbon alloy steel, and ASME SA-240, Type 304 stainless steel. ASME SA-516, Grade 70 carbon steel thermal properties were used in place of SA-350, Type LF3 and SA-203, Grade E in the model. The thermal properties are similar at room temperature and converge at higher temperatures. Therefore the effect is negligible.

Both ASME SB-209 6061-T6/651 and 6063-T6/651 aluminum are utilized in the packaging and the fuel basket plates. The thermal properties for both the steels and aluminum alloys are extracted from ASME B&PV Code, Section II, Part D [8]. The thermal properties of electrolytic copper and commercial grade lead used in this analysis are extracted from the Handbook of Heat Transfer [10].

The thermal properties of resin are based on a proprietary report from AREVA TN [11]. All wood in this model is considered to be redwood (Sequoia) with properties from the Wood Handbook [12].

Void spaces within the package are filled with either helium or air. Thermal properties for these two gases are from The Handbook of Heat Transfer [10]. The material properties of the gasses used in the models are found in **Table 2.3-6**.

The thermal properties for the PWR fuel assemblies are determined in the NUHOMS®-MP197 Transportation Package SAR [13]. Thermal properties of the BWR assemblies are determined using the methodology of the MP197HB thermal analysis.

The PWR basket uses thin sheets of boron enriched aluminum as a neutron absorbing sheet. The BWR basket uses boron enriched aluminum plates between the steel fuel cells for neutron absorption. In both cases the properties of 6061 aluminum are used in this analysis to represent the neutron absorber sheet.

Note that he material properties used in this thermal analysis are not identical to those specified in the final conceptual design. However, the properties are sufficiently similar to support the conceptual capability of the HB-6625B-HB design for the specified thermal loads.

TABLE 2.3-4: THERMAL PROPERTIES OF MODEL MATERIALS

Material	Temperature (°F)	Thermal Conductivity (Btu/(h-in-°F))	Specific Heat (Btu/(lb$_m$-°F)	Density (lb$_m$/in^3)
SA-240 Type 304	70	0.717	0.116	0.289
	100	0.725	0.117	
	200	0.775	0.122	
	300	0.817	0.125	
	400	0.867	0.129	
	500	0.908	0.131	
	600	0.942	0.133	
	700	0.983	0.135	
	800	1.025	0.136	
	900	1.058	0.137	
	1000	1.092	0.138	
SA-516 Type 70, or SA-36	70	2.275	0.105	0.284
	100	2.300	0.108	
	200	2.317	0.116	
	300	2.275	0.122	
	400	2.208	0.127	
	500	2.142	0.131	
	600	2.075	0.136	
	700	2.008	0.142	
	800	1.933	0.148	
	900	1.858	0.156	
	1000	1.758	0.163	
SB-6061 Aluminum	70	8.008	0.213	0.098
	100	8.075	0.215	
	150	8.167	0.218	
	200	8.250	0.221	
	250	8.317	0.223	
	300	8.383	0.226	
	350	8.442	0.228	
	400	8.492	0.230	
SB-6063 Aluminum	70	10.07	0.213	0.098
	100	10.03	0.215	
	150	9.975	0.218	
	200	9.917	0.221	
	250	9.875	0.223	
	300	9.842	0.226	
	350	9.833	0.228	
	400	9.800	0.230	

TABLE 2.3-5: THERMAL PROPERTIES OF MODEL MATERIALS CONTINUED

Material	Temperature (°F)	Thermal Conductivity (Btu/(h-in-°F))	Specific Heat (Btu/(lb$_m$-°F)	Density (lb$_m$/in^3)
Copper	-10	19.62	0.0901	0.323
	80	19.00	0.0922	
	260	18.81	0.0946	
	440	18.55	0.0970	
	620	18.36	0.1029	
	980	18.33	0.1070	
	1340	18.29	0.1065	
	1700	18.25	0.1147	
Redwood	100	0.0378 (0.019 after fire)	0.312	0.007
	200		0.363	0.006
	300		0.414	0.005
	400		0.466	0.005
	500		0.517	0.004
	600		0.568	0.004
Vyal B (or equivalent) Resin	100	0.039	0.256	0.0632
	104		0.256	
	140		0.260	
	176		0.282	
	212		0.301	
	284		0.358	
	320		0.380	
Lead ASTM B29, chemical lead	-10	1.733	0.030	0.411
	80	1.700	0.031	0.409
	260	1.637	0.032	0.406
	440	1.579	0.033	0.402
	620	1.512	0.034	0.398
Ceramic Fiber Paper (Lytherm® Grade 1530-L/1535-L)	-40	0.0030	0.194	0.00434
	500	0.0030		
	800	0.0040		
	1300	0.0060		
	1600	0.0070		
	2000	0.0110		

TABLE 2.3-6: THERMAL CONDUCTIVITY OF GASSES (ATMOSPHERIC PRESSURE)

Temperature (°F)	Thermal Conductivity of Helium (Btu/h-in-°F)	Thermal Conductivity of Air (Btu/h-in-°F)
-10	--	0.0011
80	0.0072	0.0013
260	0.0086	0.0016
440	0.0102	0.0019
620	0.0119	0.0022
980	0.0148	0.0027
1340	0.0174	0.0032
1430	0.0181	--

Note: Values from the Handbook of Heat Transfer, 3rd Edition

2.3.2.2 Technical Specifications of Components

The packaging components of concern in the thermal analysis include those which when exposed to higher than normal temperature will fail to perform their intended function. The components of concern include O-ring lid seals, neutron shield resins, impact limiter crushable material (redwood and balsa wood), and any additional organic material that may be included in the cask design. All components are selected so that they retain their key physical characteristics down to -40°F.

Both Type 304 stainless and the carbon steel are limited to 800°F and 700°F respectively during NCT due to structural limitations. During HAC, the limit is 2,600°F for both steels.
Long term exposure charts shown that the yield and ultimate strength of 6061-T6 aluminum retain approximately 10 percent of their room temperature values at 550°F [37]. In addition, the elastic modulus of 6061-T6 aluminum is shown to remain at 76 percent of its room temperature value when exposed to 550°F for an extended length of time. Therefore, the NCT and HAC temperature limit for aluminum is considered 550°F where structural support is required. The HAC temperature limit for non-structural aluminum is 1,080°F. The temperature limit of copper is its melting point at 1,983°F.

The O-ring seals form both containment and test boundaries within the package. A failure of the O-ring seal may allow for a breach of the containment boundary. The long term 400°F thermal limit of the O-ring seals is based on the material properties of the fluorocarbon seals (Viton O-rings) in the containment vessel for NCT and HAC [14]. The short term limit for use in the hypothetical accident condition analysis is 482°F [14]. The cold ambient temperature of -40°F will not have an adverse effect on the sealing function of the O-rings.

The maximum allowable neutron shield resin temperature is 320°F during NCT to ensure its function [11]. The gamma shield temperature is limited to a maximum temperature of 620°F to prevent melting of the lead [10].

A peak temperature limit of 320°F and an average bulk temperature limit of 212°F are considered for the impact limiter wood to prevent excessive reduction of structural properties at elevated temperatures [12].

The fuel assembly cladding material has a temperature limit of 752°F for NCT and 1,058°F for HAC [15].

2.3.3 Evaluation of Accessible Surface Temperatures

The accessible surface temperature of the cask shall be less than 185°F to comply with 10 CFR §71.43(g) for NCT exclusive use [1]. The initial evaluation shows that the outer surface of the cask reaches a peak temperature of 272°F under the prescribed loading conditions. A personnel barrier is required to prevent access to the accessible hot surface of the cask. The personnel barrier reduces the accessible surface temperature to 142°F which is located on the external surface of the lower impact limiter.

The barrier consists of a stainless steel mesh supported by a stainless steel frame connected to the package so that it encloses the cask body between the impact limiters. The steel mesh has an open area fraction of of approximately 80 percent, which prevents interference of cask surface convection.

2.3.4 Normal Conditions of Transport

The package is evaluated for NCT for both hot and cold conditions. The cask is oriented horizontally for transportation. These evaluations are considered steady state and therefore apply for both the immediate and long-term temperature maximums. The hot and cold NCT cases are described below.

2.3.4.1 Heat and Cold

2.3.4.1.1 Maximum Temperatures

To meet the heat requirements of 10 CFR 71 [1], the cask is exposed to solar insolation at an ambient temperature of 100°F. The maximum heat load from the fuel is distributed on the inner surface of the model cavity along the active length of the fuel. Since there is a slight variation in the start of the active fuel length between the PWR and BWR packages both configurations are analyzed. The maximum of the predicted common package component temperatures for NCT are listed in **Table 2.3-7** (both PWR and BWR) and are shown in **Figure 2.3-1** (PWR) and **Figure 2.3-2** (BWR).

The maximum predicted PWR basket component temperatures are listed in **Table 2.3-8**. The steady state temperature in the PWR basket reaches a maximum value in the near the longitudinal center of the basket heat load zone, as shown in **Figures 2.3-3** through **2.3-6**.

The maximum predicted BWR basket component temperatures are listed in **Table 2.3-9**. The steady state temperature in the BWR basket reaches a maximum value in the Zone 2 fuel region (780W fuel region), as shown in **Figures 2.3-7** and **2.3-8**.

TABLE 2.3-7: CASK NCT TEMPERATURE RESULTS (°F)

Location / Component	Allowable	Temperature	Minimum Margin
Inner Wall - Maximum - Average	700	348 306	352 394
Gamma Shield (Lead)	620	346	274
Outer Wall - Maximum - Average	700	314 254	386 446
Neutron Shield Resin - Maximum - Average	320	296 245	24 75
Neutron Shield Box	1,983	298	1,685
Outer Shell		299	401
Bottom Structure		335	365
Upper Structure	700	212	488
Inner Lid		208	492
Outer Lid		207	493
Impact Limiter Shell	800	291	509
Impact Limiter Gusset		288	512
Impact Limiter Insulation	2,300	290	2,010
Impact Limiter Wood	320	285	35

- Maximum - Average	212	145	67
Thermal Shields	550	292	258
Maximum Surface Temperature - Without Personnel Barrier - With Personnel Barrier	185	274 142	-89 43
Inner Lid Seal	400	208	192
Outer Lid Seal		207	193

TABLE 2.3-8: PWR BASKET NCT TEMPERATURE RESULTS (°F)

Location / Component	Temperature (°F)						
	Allow.	Config. 1	Config. 2	DFC Config. 1	DFC Config. 2	Max.	Margin (Max)
Fuel Cladding	752	565	582	561	578	582	170
Basket Steel Plates	800	426	444	425	443	444	356
Basket Aluminum Plates/ Poison	N/A	425	444	425	443	444	N/A
Basket Aluminum Rails	N/A	387	386	387	386	387	N/A
DFC	800	N/A	N/A	429	451	451	349
Helium Fill Gas (Average)	N/A	354	355	353	354	355	N/A

TABLE 2.3-9: BWR BASKET NCT TEMPERATURE RESULTS

Location / Component	Temperature (°F)				
	Allow.	Config. 1	Config. 2	Max.	Margin
Fuel Cladding	752	552	493	552	200
Basket Steel Plates	800	528	482	528	272
Basket Aluminum Plates	N/A	528	482	528	N/A
Basket Aluminum Rails	N/A	401	400	401	N/A
Helium Fill Gas (Average)	N/A	409	387	409	N/A

FIGURE 2.3-1: NCT PWR STEADY STATE TEMPERATURE PROFILE

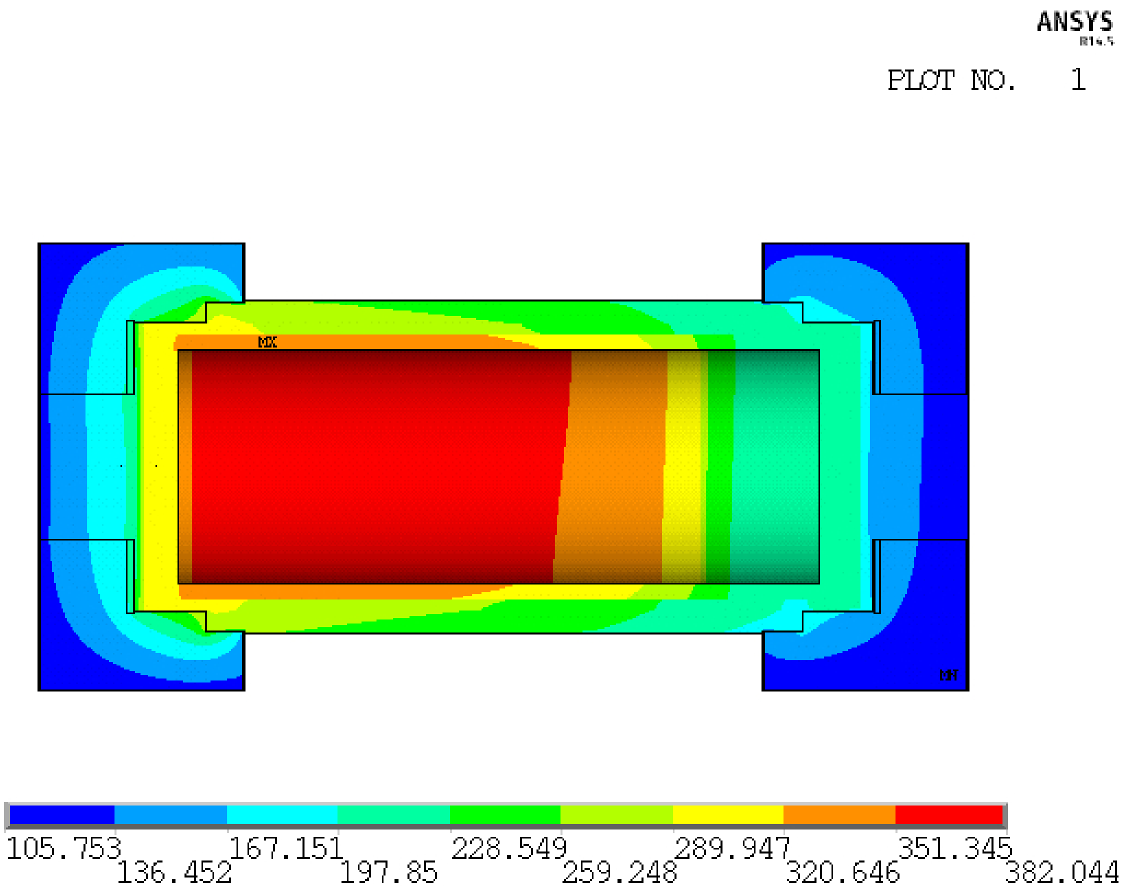


FIGURE 2.3-2: NCT BWR STEADY STATE TEMPERATURE PROFILE

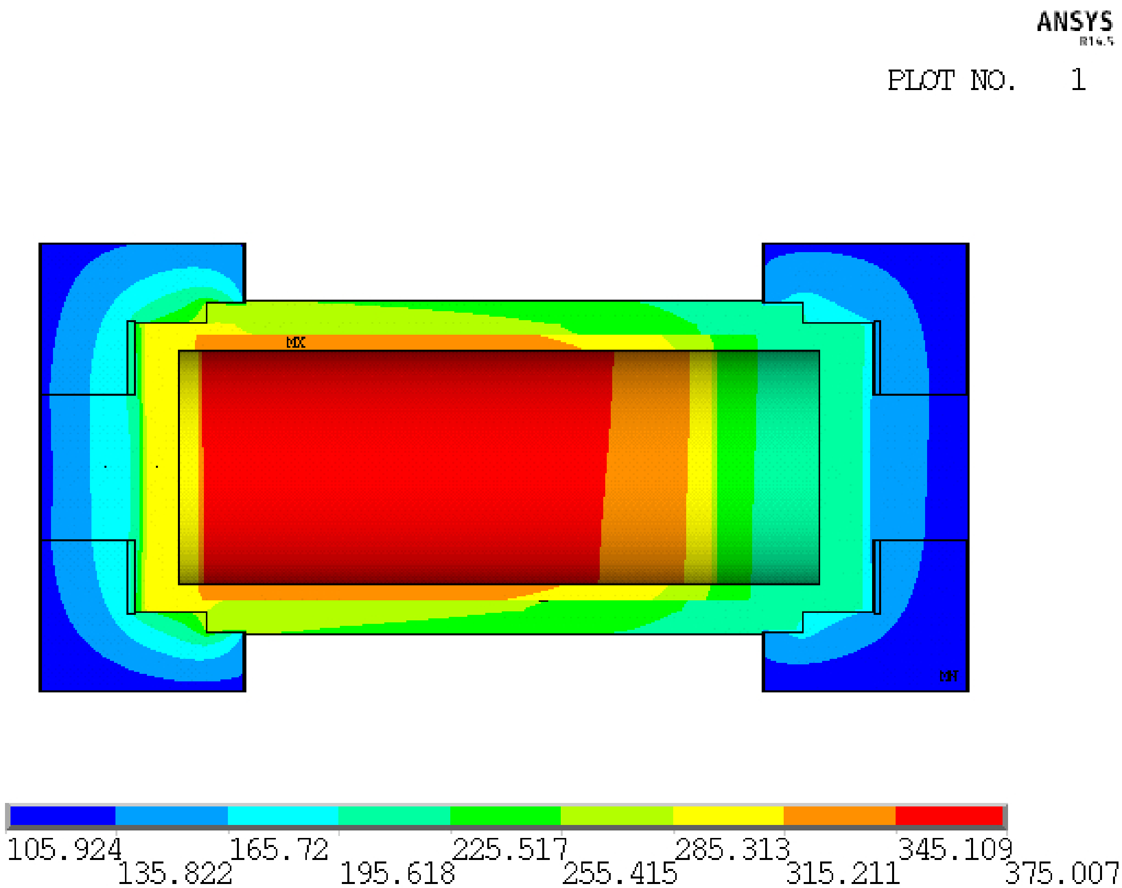
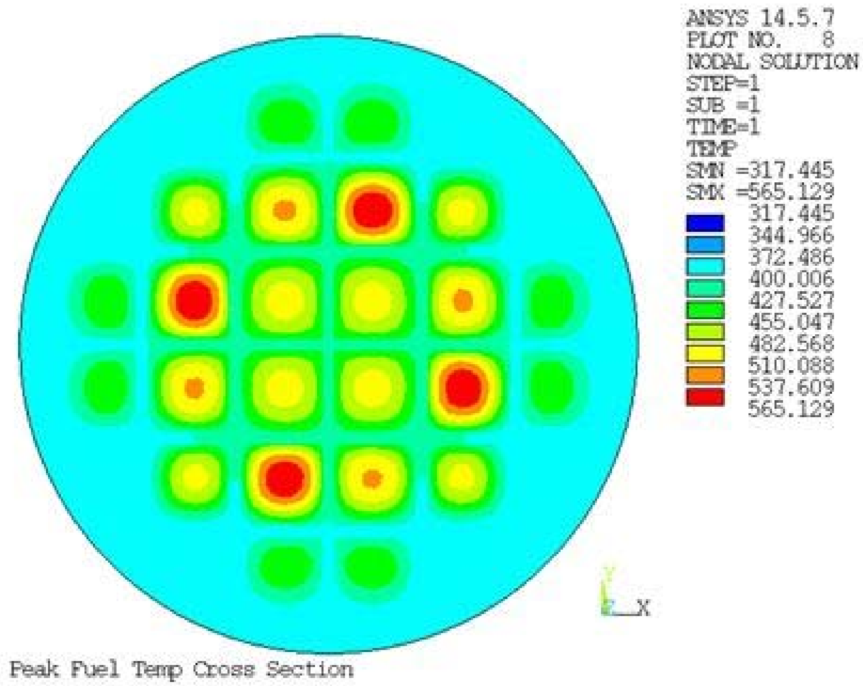


FIGURE 2.3-3: PWR BASKET TEMPERATURE PROFILE (CONFIG. 1)



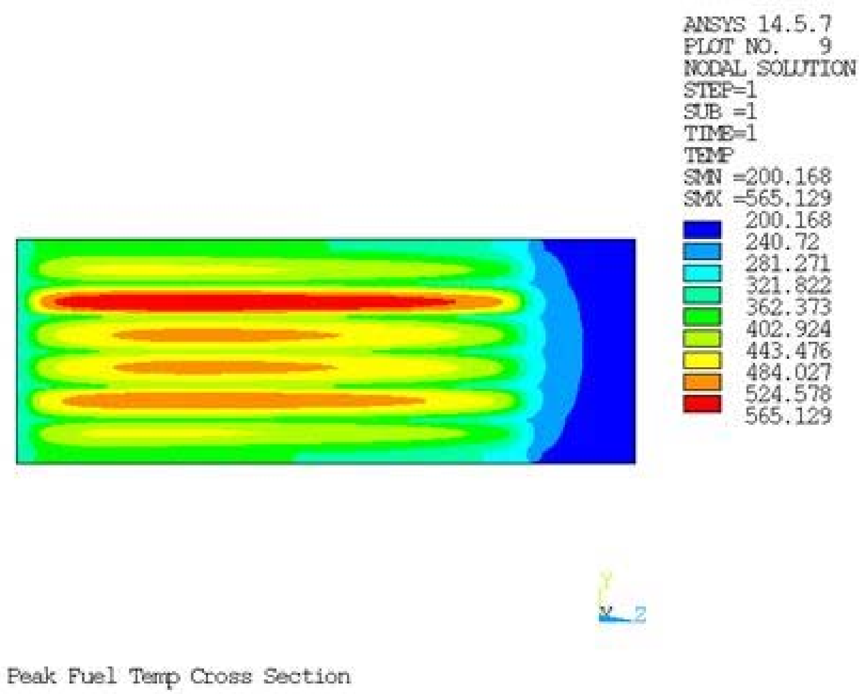
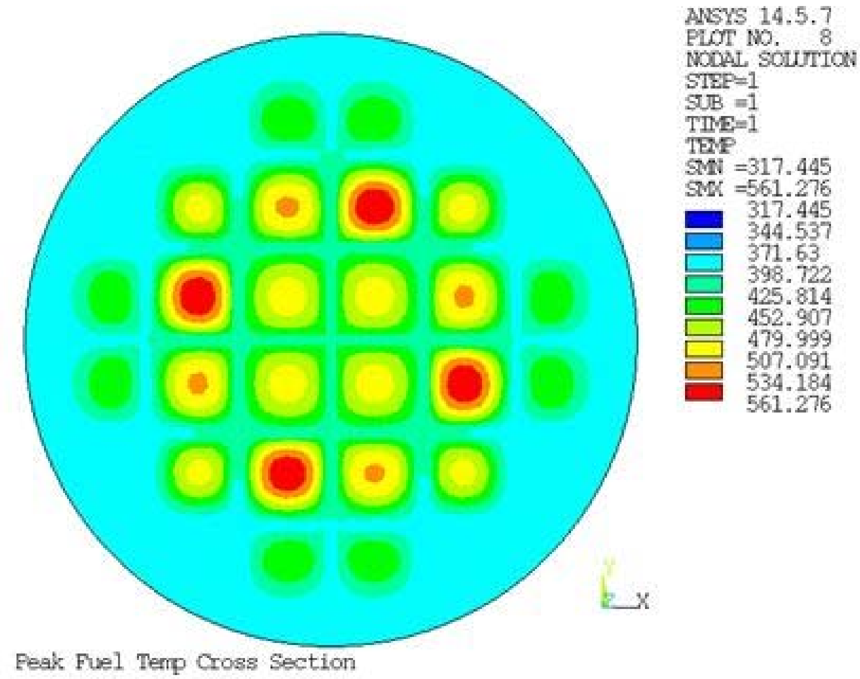


FIGURE 2.3-4: PWR BASKET TEMPERATURE PROFILE (DFC CONFIG. 1)



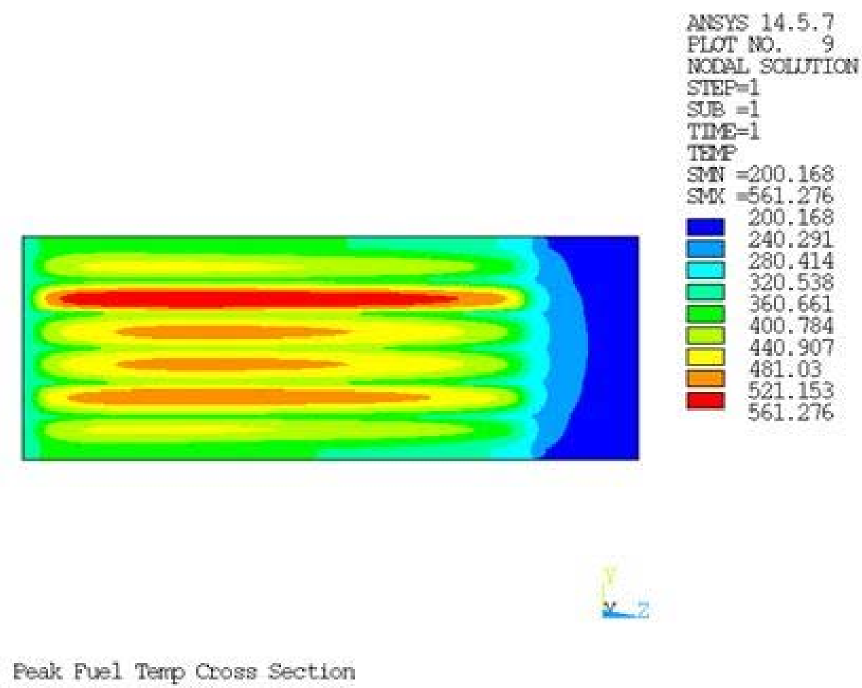
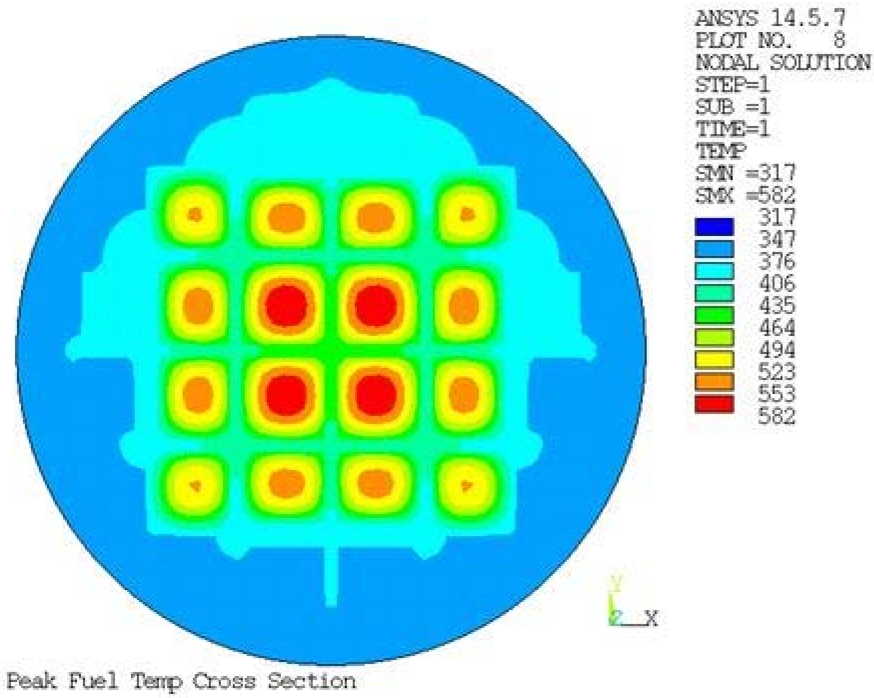


FIGURE 2.3-5: PWR BASKET TEMPERATURE PROFILE (CONFIG. 2)



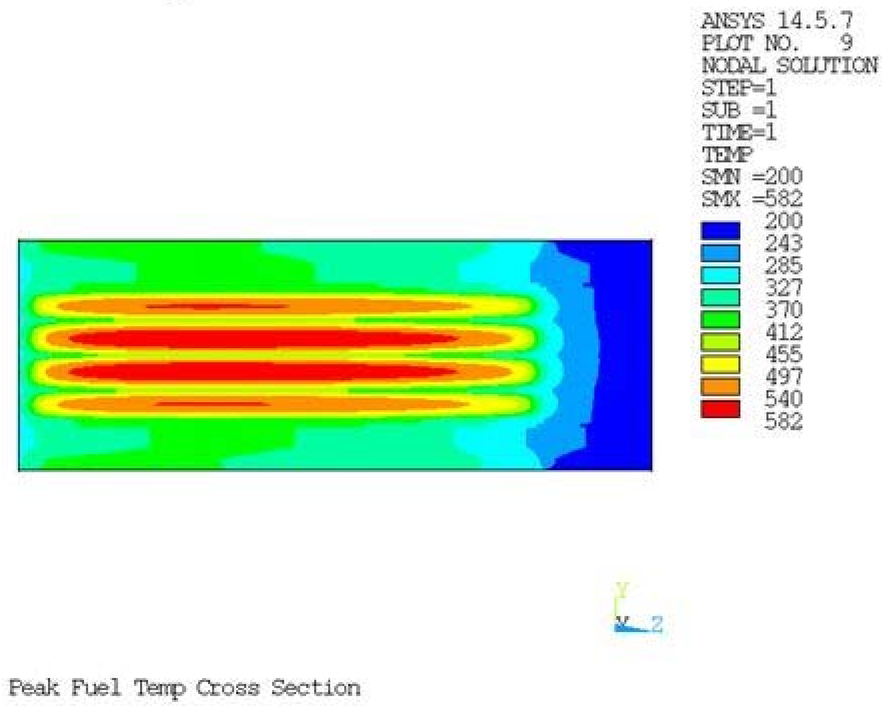
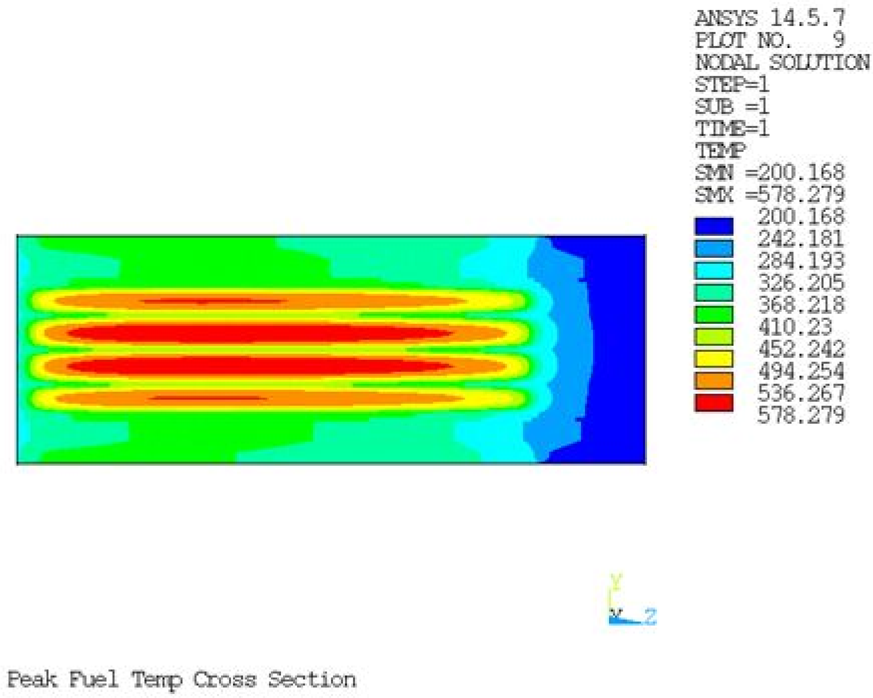


FIGURE 2.3-6: PWR BASKET WITH DFCS TEMPERATURE PROFILE, DFC CONFIG. 2)



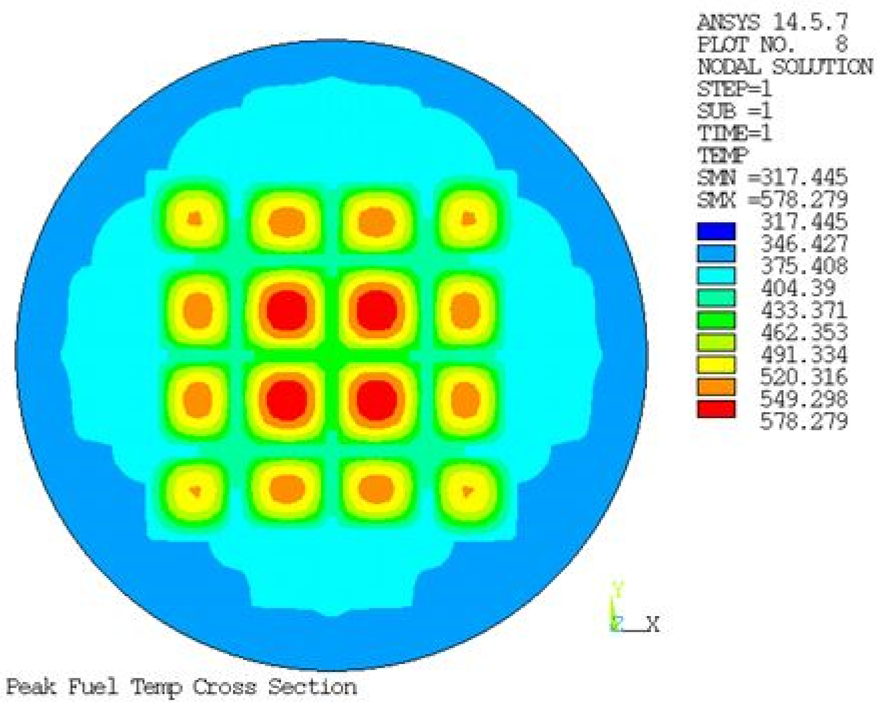


FIGURE 2.3-7: BWR BASKET STEADY STATE TEMPERATURE PROFILE

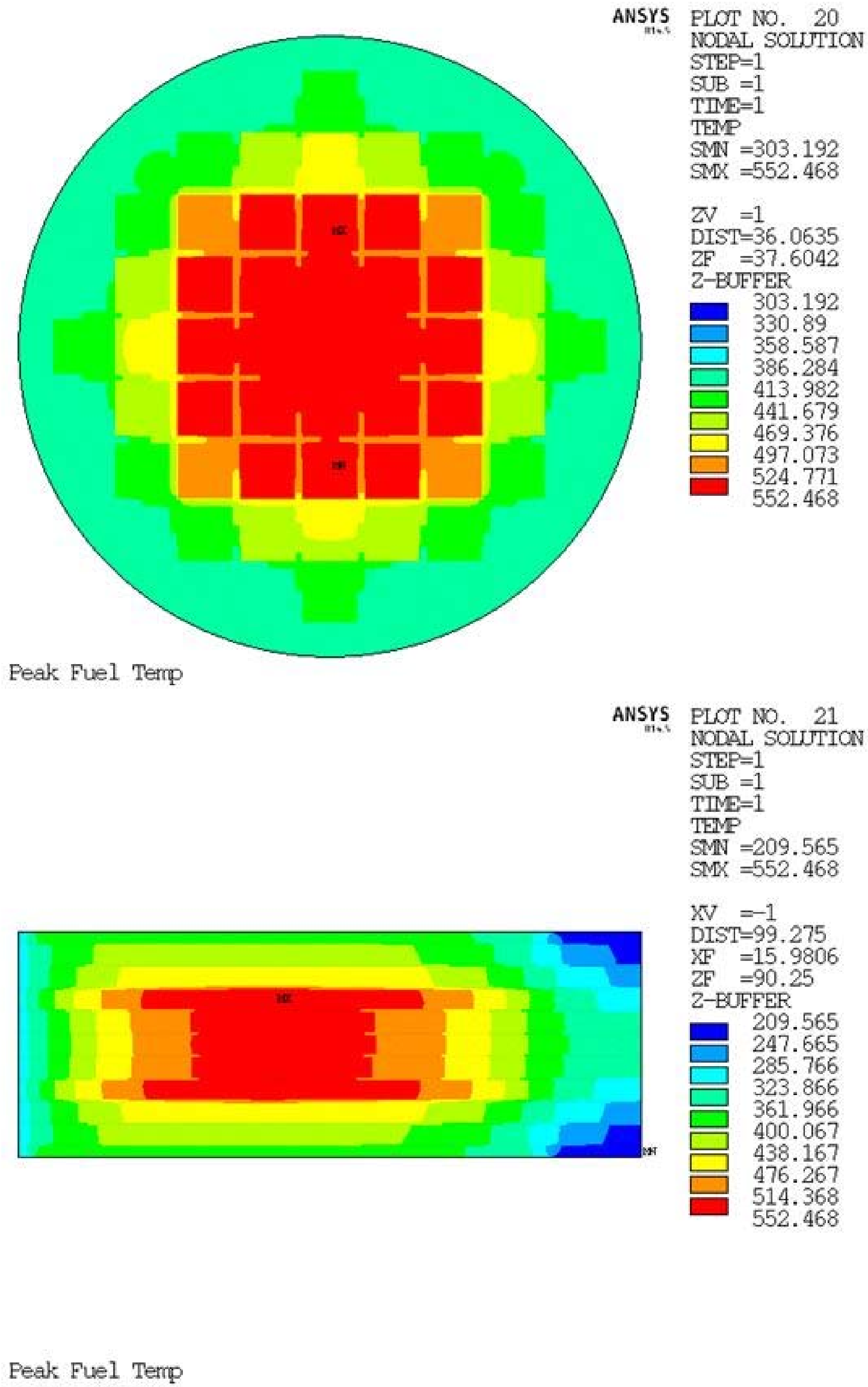
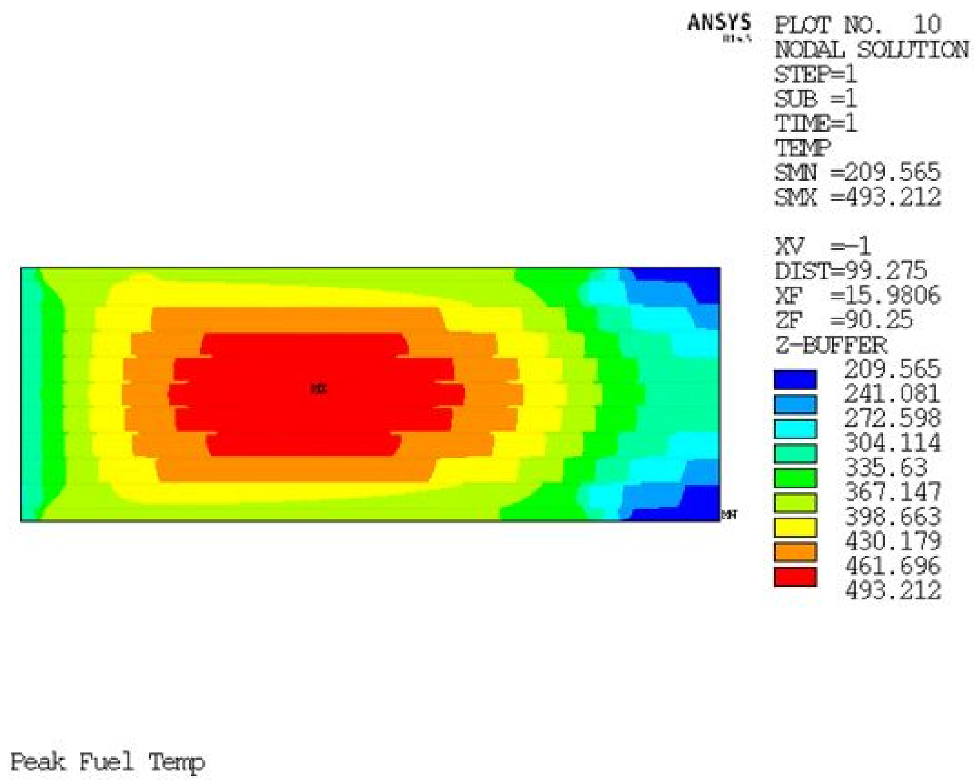


FIGURE 2.3-8: PWR BASKET WITH DFCS TEMPERATURE PROFILE, ALT. CONFIG.



2.3.4.1.2 Minimum Temperatures

The minimum temperatures of the package will be found at the lowest environmental temperature of -40°F. This occurs when there is no insulation and no internal heat load. Conservatively, the internal heat of the fuel is neglected. In this case, the components of the basket reach a steady state temperature of -40°F. All materials used in construction of the package are capable of retaining their structural properties down to this temperature. The packaging materials and components are shown to meet the minimum temperature requirements of 10 CFR 71 [1].

2.3.4.2 Maximum Normal Operating Pressure

The maximum normal operating pressure (MNOP) for the transportation cask is estimated as part of the thermal evaluation. The MNOP of the cask is calculated using the average steady state temperature of the cask fill gas. The total quantity of the fill gas is calculated based on the original quantity of fill gas in the cask payload cavity, the fill gas released from the fuel rods, and the gasses produced through irradiation.

The MNOP pressures are shown in **Table 2.3-10**. In both cases the pressures due to the thermal load are below the 30 psig value used in the structural evaluation.

TABLE 2.3-10: MAXIMUM NCT

Load Case	Cask Cavity Pressure (psig)
NCT BWR Fuel	11.0
NCT PWR Fuel	9.3

2.3.4.3 Maximum Thermal Stress

Analysis of the MP197HB cask included thermal loads in excess of those seen in the 6625B-HB cask. The maximum differential temperature included for the MP197HB is 65°F for a 69BTH DSC, cold NCT with no insolation[13]. The differential temperature for the 6625B-HB is 52°F. The stresses due to differential thermal expansion for the 6625B HB are thus expected to be lower than those analyzed for the MP197HB and will result in a higher positive margin.

2.3.4.4 Vacuum Drying Operations

Vacuum drying operations occur after fuel loading and dewatering has been completed, however prior to the pressurized helium backfill that is required for transport. The cask cavity is evacuted using a vacuum pump to remove residual moisture. A thermal analysis of vacuum drying is normally preformed to determine the maximum allowable time for this operation. A detailed analysis of vacuum drying has not been completed. However, a survey of NUHOMS® System DFCs with similar cavity volumes, basket designs, and fuel payloads demonstrate that vacuum drying of the cask may be performed within a reasonable period of time without exceeding the maximum fuel cladding temperature.

2.3.5 Hypothetical Accident Condition

The thermal HAC consists of a fully engulfing fire at a flame temperature of 1,475°F for 30 minutes. Damage to the package due to the drop and puncture cases is considered. The bounding thermal case from the NCT analysis was selected for the HAC fire case. Given the similarity of the thermal profiles of the two packaging loading profiles the basket models were used to determine the bounding case. In this case, the higher temperatures of the fuel cladding found in PWR basket loading configuration 1 (without DFCs) was used.

2.3.5.1 Initial Conditions

The initial conditions of the HAC are similar to the NCT conditions found in *Section 2.3.4*. To simulate damage to the package the diameter and length of the impact limiter have been reduced. The reduction of diameter and length was chosen to present approximately 70 percent crush damage to the impact limiters. The reduction in volume reduces the effective insulation of the impact limiters resulting in conservative values for peak fire induced temperatures. The volume reduction is larger than that used in the MOP197HB Analysis, but results in a more conservative solution due to the loss of insulation capacity protecting the ends of the cask.

2.3.5.2 Fire Test Conditions

The package is analytically analyzed to determine the effect of a regulatory fire in accordance with 10 CFR 71. The model of the package is exposed to a 1,475°F fire for a period of 30 minutes. The fire is followed by a 30-minute smoldering period in which portions of the impact limiter wood are assumed to continue to thermally load the package. The package is then allowed to passively cool for an additional 20 hours.

2.3.5.3 Maximum Temperature

The maximum HAC fire temperatures are found in **Table 2.3-11**. A temperature versus time plot is shown in **Figure 2.3-9** with the sidewall temperature plot in **Figure 2.3-10**. PWR basket component temperatures over time are shown in **Figure 2.3-11**. The PWR cask loading conditions described in *Section 2.3.4* are used as the initial starting conditions for the HAC case, as shown in **Figure 2.3-12**.

The transient simulation starts with the 30-minute fire loading. The external loads on the package are replaced with a convective forced convection boundary layer at a flame temperature of 1,475°F (802°C). The temperature distribution at the end of the 30-minute fire is shown in **Figure 2.3-13**.

The next 30-minutes of the transient simulation consist of a smoldering phase in which the surface temperature of the wood inside of the impact limiters is held at an elevated temperature. This phase is used to represent the combustion of the outer layer of the impact limiter wood, which is exposed to temperatures high enough to char during the fire. The temperature distribution at the end of the 30-minute smoldering phase is shown in **Figure 2.3-14**.

The final phase is the cool down where the package is allowed to cool without the external fire load. The temperature distribution at the end of the 20-hour cool-down phase is shown in **Figure 2.3-15**.

The thermal shield of both impact limiters is exposed to direct contact with the flame. This results in a peak temperature on its exposed surfaces above both the NCT and HAC limits. However, this component is not required for maintaining containment but is used to protect the impact limiter wood during NCT. Therefore the allowable temperature limit is not applicable to this analysis.

TABLE 2.3-11: HAC TEMPERATURE RESULTS

Location / Component	Temperature	Allowable	Margin
Inner Wall - Maximum - Average	 518 411	 2,600 2,600	 2,082 2,189
Gamma Shield (Lead)	515	620	105
Outer Wall	511	2,600	2,089
Neutron Shield Box	882	1,983	1,101
Outer Shell	1,151	2,600	1,449
Bottom Structure	231	2,600	2,369
Upper Structure	447	2,600	2,153
Inner Lid	417	2,600	2,183
Outer Lid	412	2,600	2,188
Impact Limiter Shell	1,473	2,600	1,127
Impact Limiter Gusset	1,459	2,600	1,141
Impact Limiter Wood - Maximum - Average	 1,472 427	 N/A N/A	 N/A N/A
Thermal Shields	1,409	N/A	N/A
Inner Lid Seal	417	482	65
Outer Lid Seal	412	482	70
Fuel Cladding	860	1,058	198
Steel Plates	773	2,600	1,827
Aluminum Rail	530	550	20
Cask Fill Gas (Average)	471	N/A	N/A

FIGURE 2.3-9: FIRE TEMPERATURE RESULTS

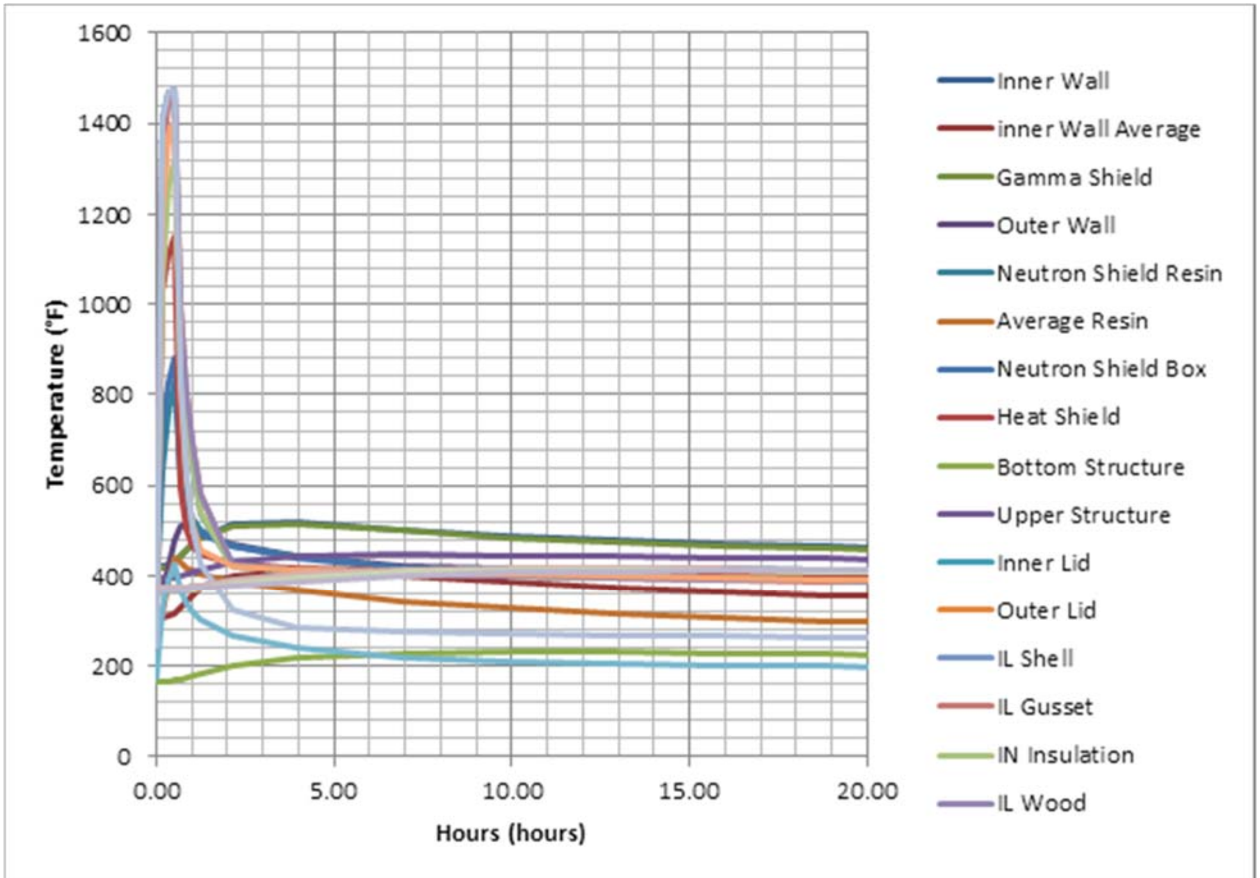


FIGURE 2.3-10: SIDEWALL FIRE TEMPERATURE RESULTS

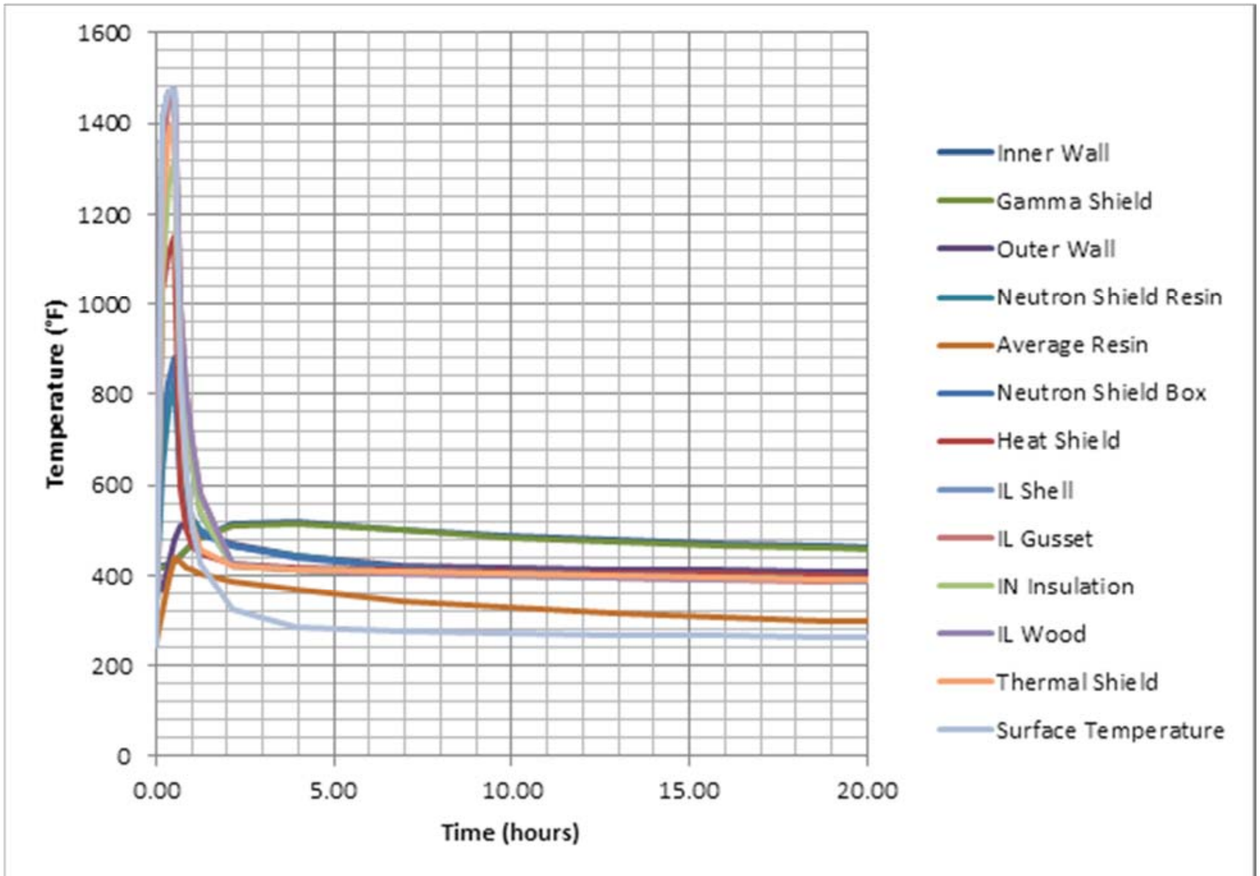


FIGURE 2.3-11: PWR BASKET FIRE TEMPERATURE RESULTS

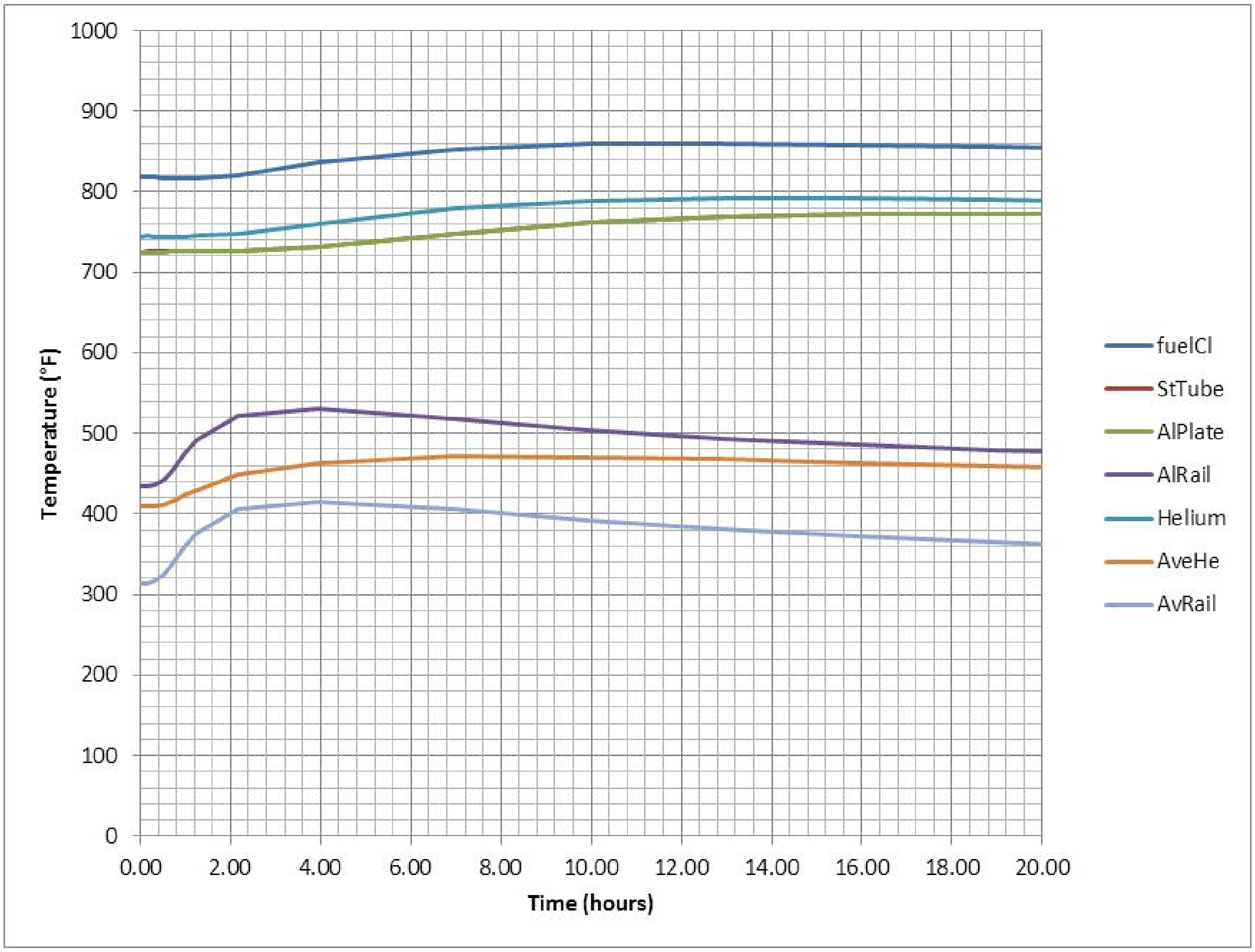


FIGURE 2.3-12: INITIAL TEMPERATURES

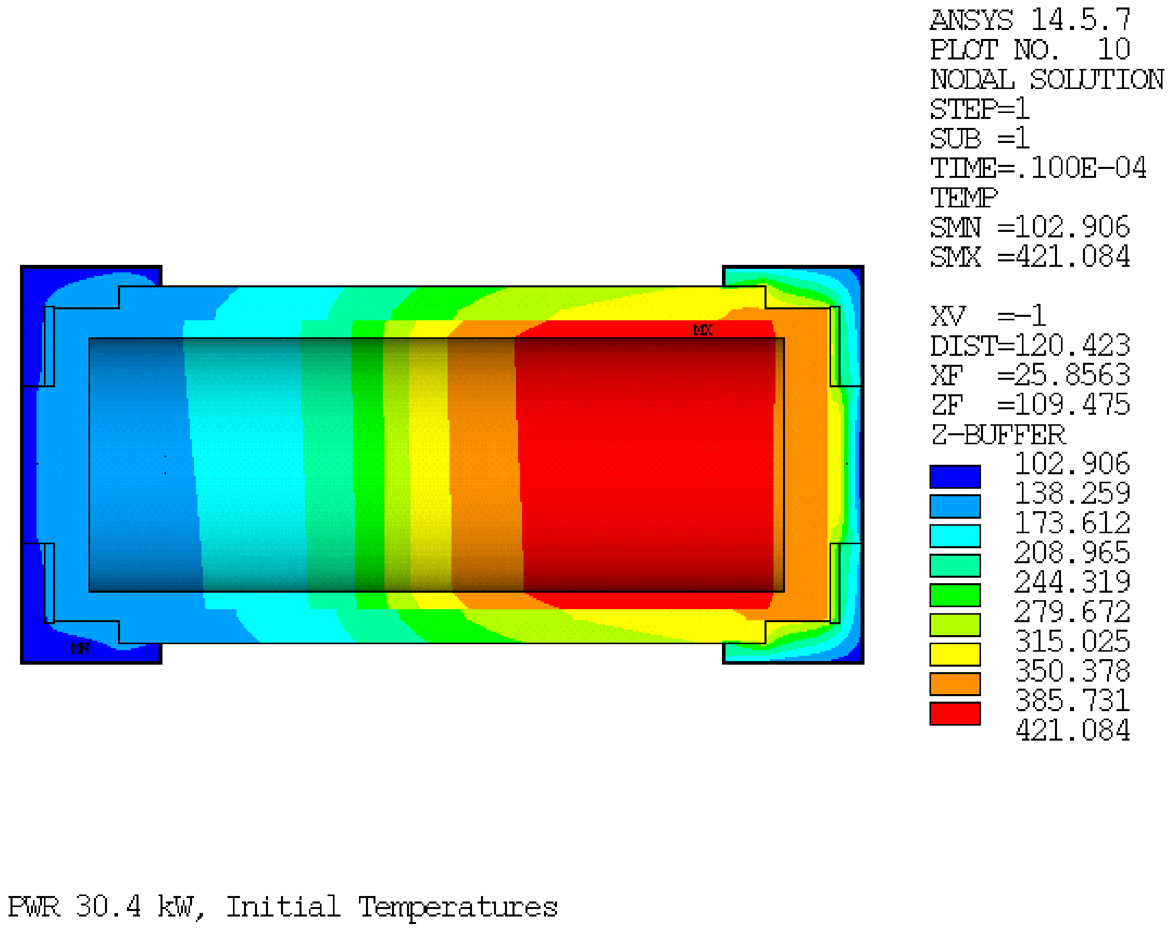


FIGURE 2.3-13: CASK TEMPERATURES AFTER 30-MINUTE FIRE

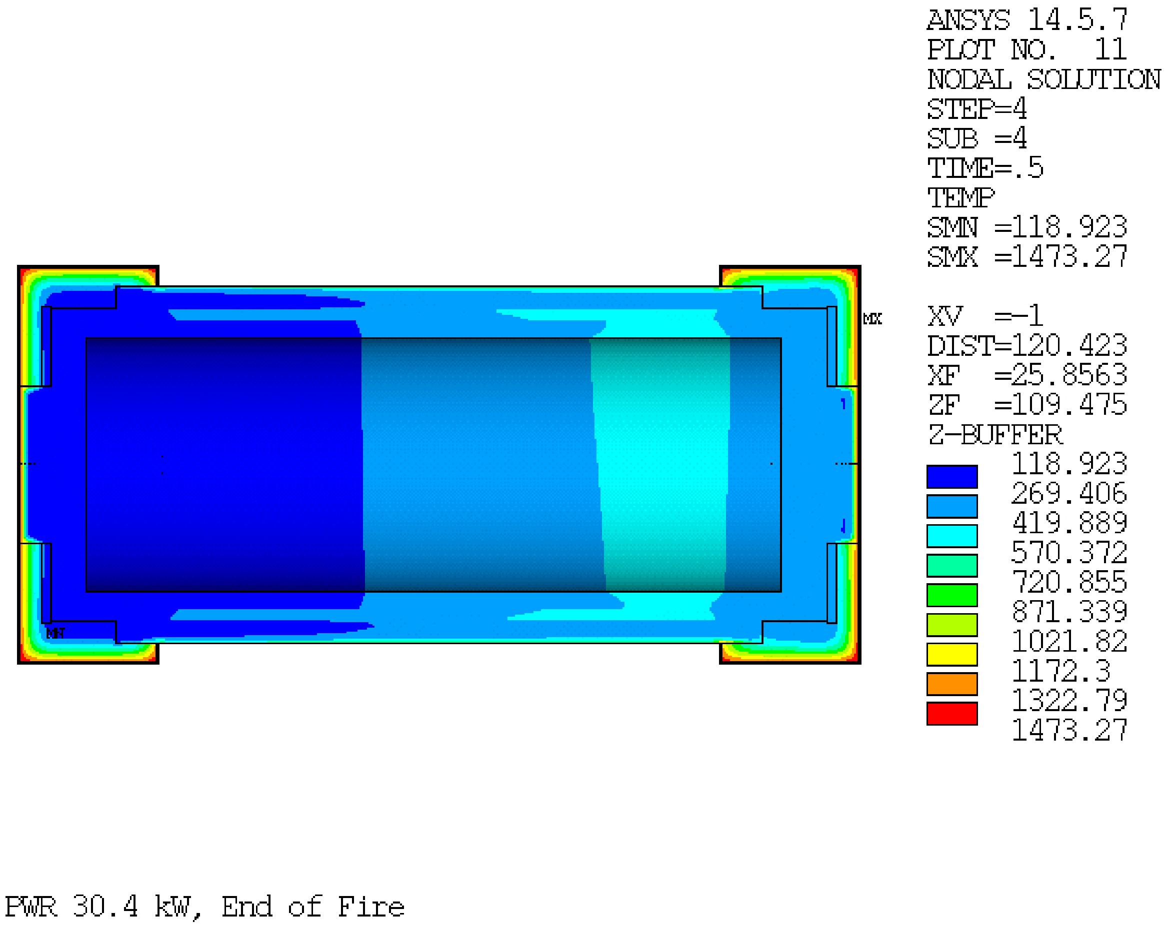


FIGURE 2.3-14: CASK TEMPERATURES AFTER 30 MINUTE WOOD SMOLDERING

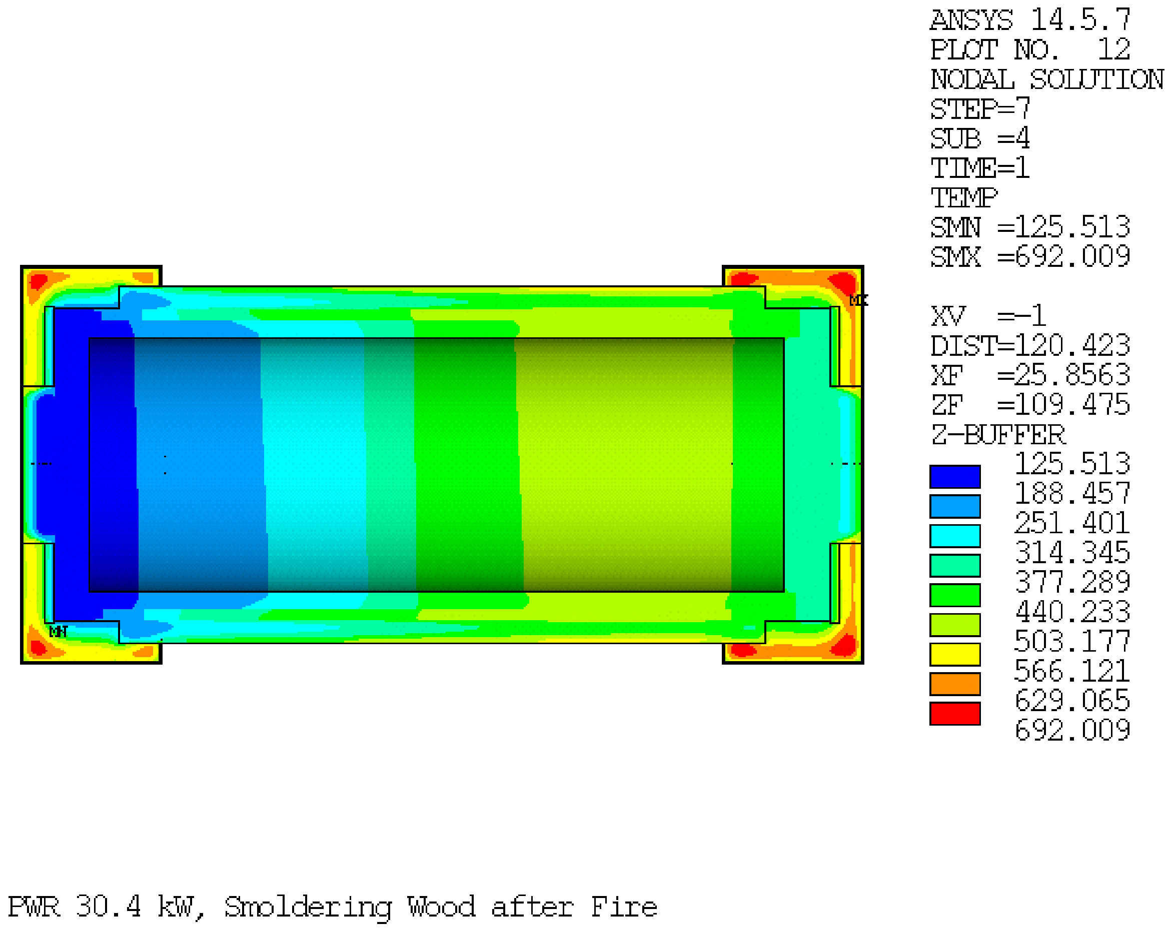
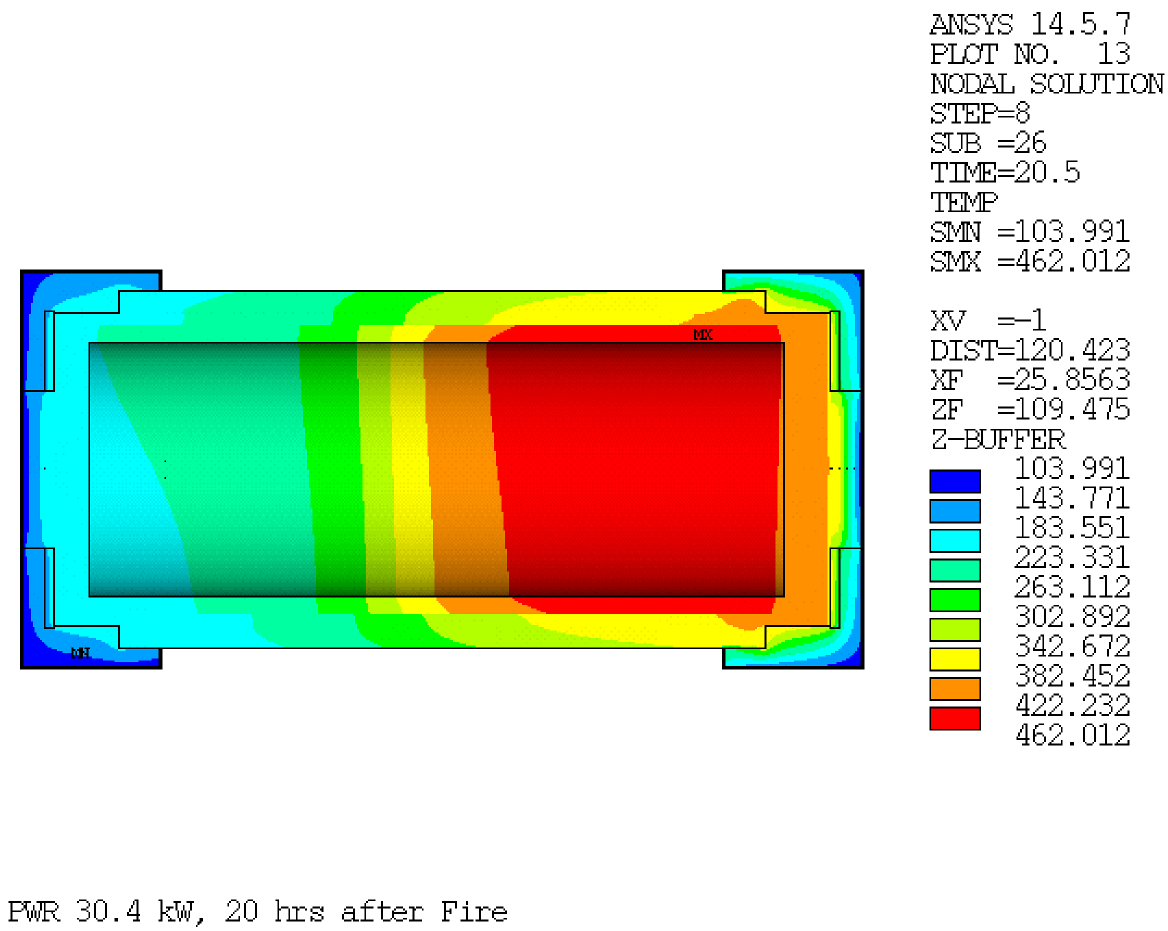


FIGURE 2.3-15: CASK TEMPERATURES AFTER COOL DOWN



2.3.5.4 Maximum Pressure

The maximum HAC pressure of the cask is calculated using the average steady state temperature of the cask fill gas. The total quantity of the fill gas is calculated based on the original quantity of fill gas in the cask payload cavity, the fill gas released from the fuel rods, and the gasses produced through irradiation.

The maximum pressures are shown in **Table 2.3-12**.

TABLE 2.3-12: MAXIMUM HAC PRESSURES

Load Case	Cask Cavity Pressure (psig)
HAC BWR Fuel	67.7
HAC PWR Fuel	63.6

2.3.5.5 Maximum Thermal Stress

The temperature differential between the cask components described in *Section 2.3.4.3* for the NCT maximum thermal stress is similar to that found during the fire. This implies that the stresses during the fire analysis will be similar in magnitude. The allowable stress used in accident stress analysis is generally higher than that used in normal condition analysis. Therefore, the NCT analysis is considered bounding.

2.3.6 Computer Program Description

This analysis was performed using a combination of classical hand calculation methods and computer software. The primary analysis was completed using ANSYS® Mechanical 14.5 (ANSYS). ANSYS® is an implicit finite element analysis code developed for structural analysis and extended to provide tools for thermal analysis of models.

2.3.7 Model Description

The thermal simulation used ANSYS® to implement finite element models of the packaging, fuel baskets, and fuel. The models were used to determine the bounding temperatures expected during NCT and HAC load cases. Separate, but similar models were used for the NCT and HAC cases. This section provides a general description of the finite element models used in this analysis.

The thermal model was based on the model used to license the MP197HB package. The cask model was intended to maximize the internal component temperatures to provide a conservative representation of the fuel and basket temperatures. Hotspots due to contact of the basket with the cask may be underrepresented in the results. However, given the large margin in the cask inner wall temperatures (i.e. a peak temperature of 348°F in the inner wall material with an allowable of 700°F) the results are still considered valid for the maximum fuel load of 30.4 kW.

2.3.7.1 NCT Thermal Model

The NCT analysis of the package used three separate finite element models. The three models were used in eight NCT load cases to provide reasonable assurance that the packaging meets the requirements specified in 10 CFR 71 [1]. The three models included:

1. A half-symmetric, three-dimensional model of the cask based on that used in the analysis of the MP197HB Transportation Cask. This model captured the full length of the packaging, allowed for application of the interior and exterior heat fluxes, and reduces computation resources to the minimum required to achieve a solution.
2. A fully symmetric, three-dimensional model of the PWR fuel basket. This model captured the full length of the basket, allowed for non-symmetrical loading pattern of PWR fuel assemblies, and captured the variation of temperatures along the length of the basket.
3. A fully symmetric, three-dimensional model of the BWR fuel basket. This model also captured the full length of the BWR basket, allowed for non-symmetrical loading pattern of fuel assemblies, and captured the variation of temperatures along the length of the basket.

The cask model was run with four separate load cases. Two cases were run for both PWR and BWR fuel loadings; an NCT insolation load to account for maximum component temperatures and a no-insolation to recover the maximum surface temperatures as required in 10CFR71 §71.43(g) [1].

All models utilized 3-D solid and shell elements to construct the model. Each package was assigned temperature dependent material properties as described in *Section 2.3.2-1*. **Figure 2.3-16** shows an overview of the cask model. **Figure 2.3-17** through **2.3-19** show the PWR basket model. **Figure 2.3-20** through **Figure 2.3-22** show the BWR basket model.

As the bounding temperature case the PWR basket model were run with 4 load cases; two separate load patterns with and without DFC. The BWR basket model was run with two load cases; two different load patterns, both without a DFC. Given that the results of the PWR basket model showed a slight decrease in the fuel cladding and basket component temperatures with the use of a DFC no DFC load cases were run for the BWR baskets.

2.3.7.2 HAC Thermal Model

The HAC analysis of the package used two separate finite element models. A transient model of the package consisted of exposure to a 30-minute fire with a 30-minute smoldering stage and a 20-hour cool-down period. The bounding thermal case from the NCT analysis (PWR basket loading pattern 1 without DFCs) was used to recover the maximum fuel cladding and packaging component temperatures.

The bounding thermal case from the NCT analysis was selected for the HAC fire case. Given the similarity of the thermal profiles of the two packaging loading profiles the basket models were used to determine the bounding case. In this case, the higher temperatures of the fuel cladding found in PWR basket loading configuration 1 (without DFCs) was used.

The HAC cask FEA model is based on the NCT FEA model with the following geometry adjustments:

1. The length of the impact limiters was decreased to simulate the maximum end drop damage (70% decrease in length to account for crush of the impact limiter),
2. The gap between the thermal shield and the end of the cask was decreased to account for shifting of the impact limiter during the end drop.
3. Additionally, the diameter of the impact limiters was decreased to bound the predicted damage due to a side drop corresponding to an expected 70% crush of the impact limiter shell and foam.

It is expected that this reduction in the insulation value of the impact limiters will result in conservative temperatures in the fire accident case. Puncture damage to the impact limiters is assumed to be bounded by the results of this method as well.

FIGURE 2.3-16: PACKAGE MODEL WITH IMPACT LIMITERS

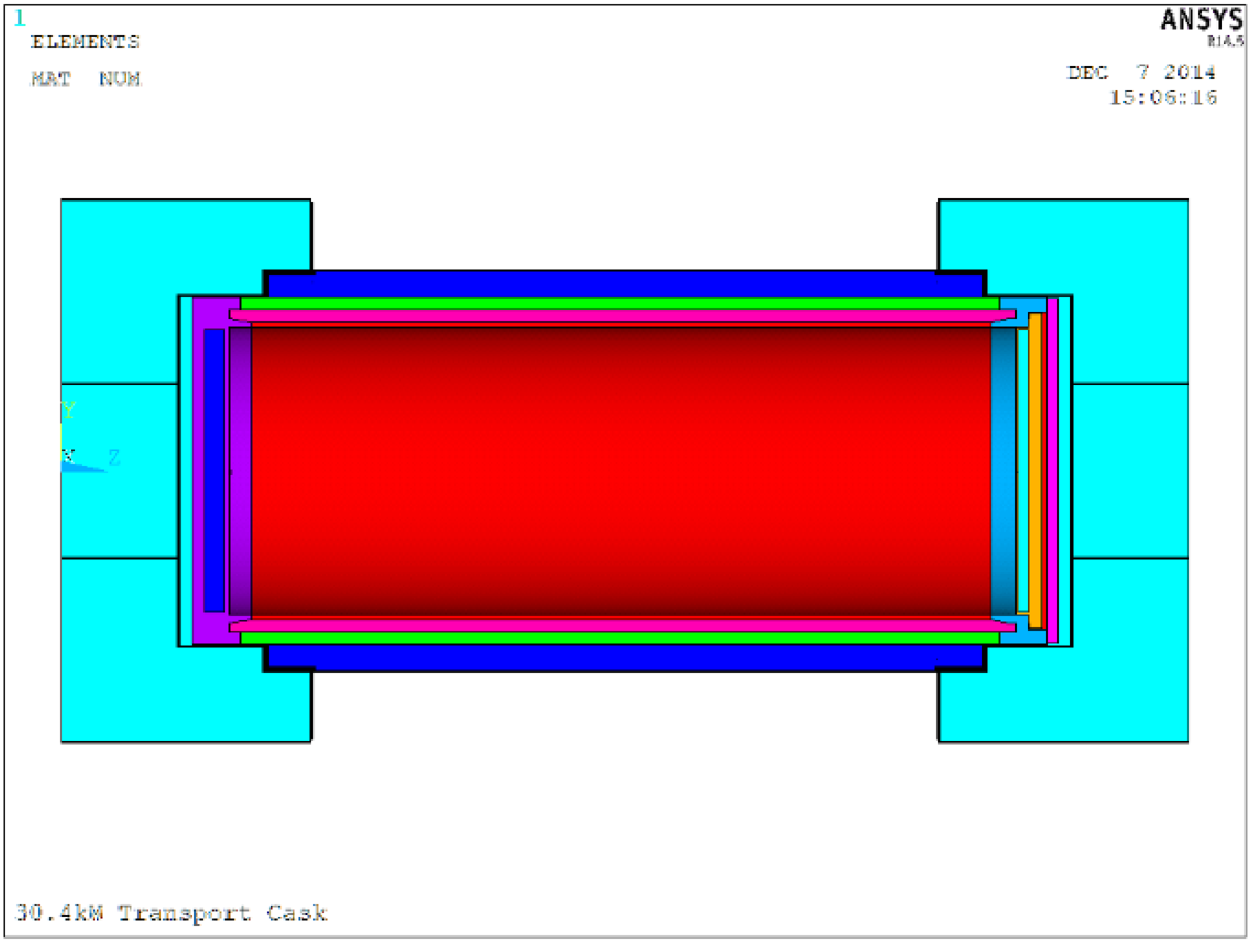


FIGURE 2.3-17: PWR BASKET MODEL

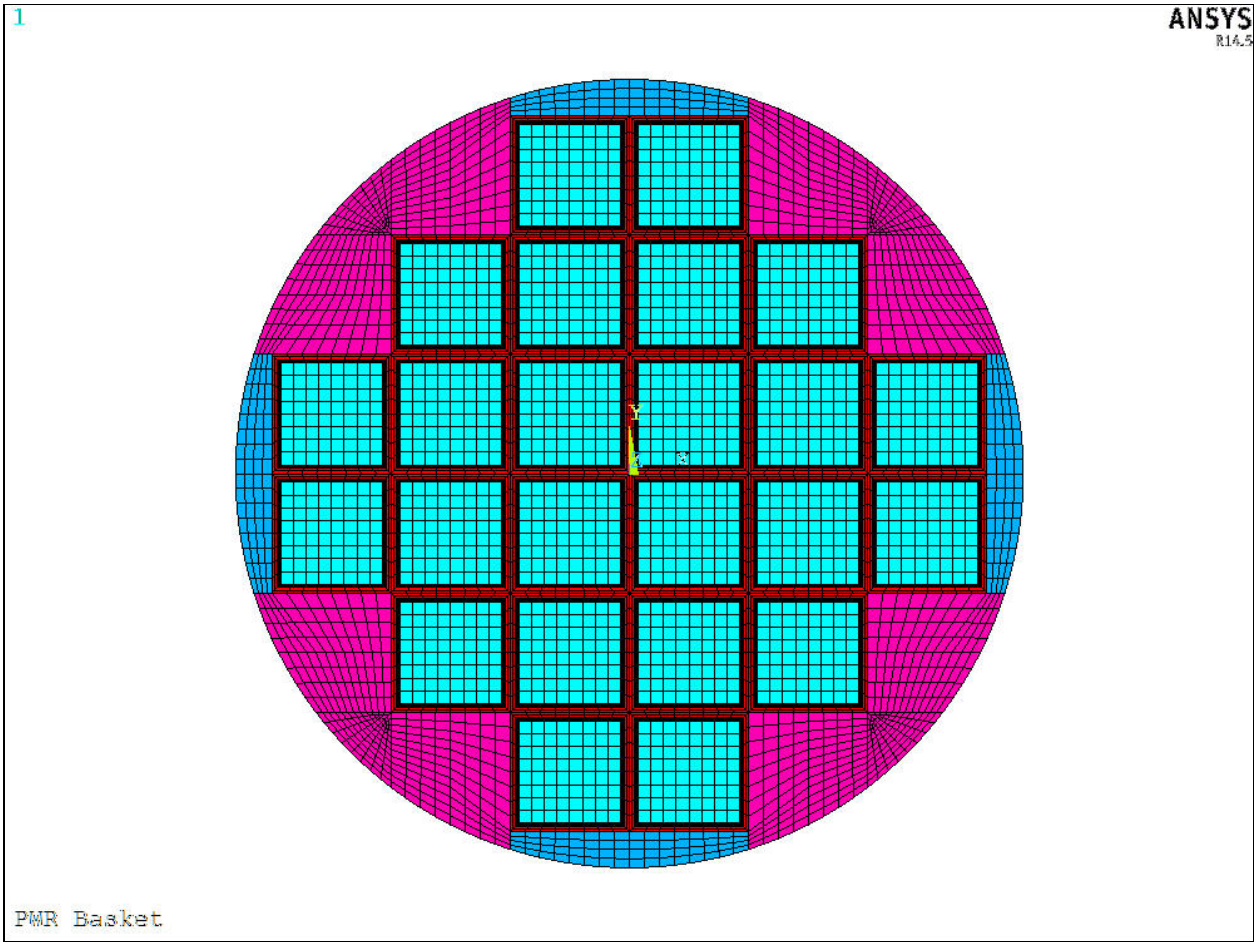


FIGURE 2.3-18: PWR BASKET MODEL CELL DETAIL

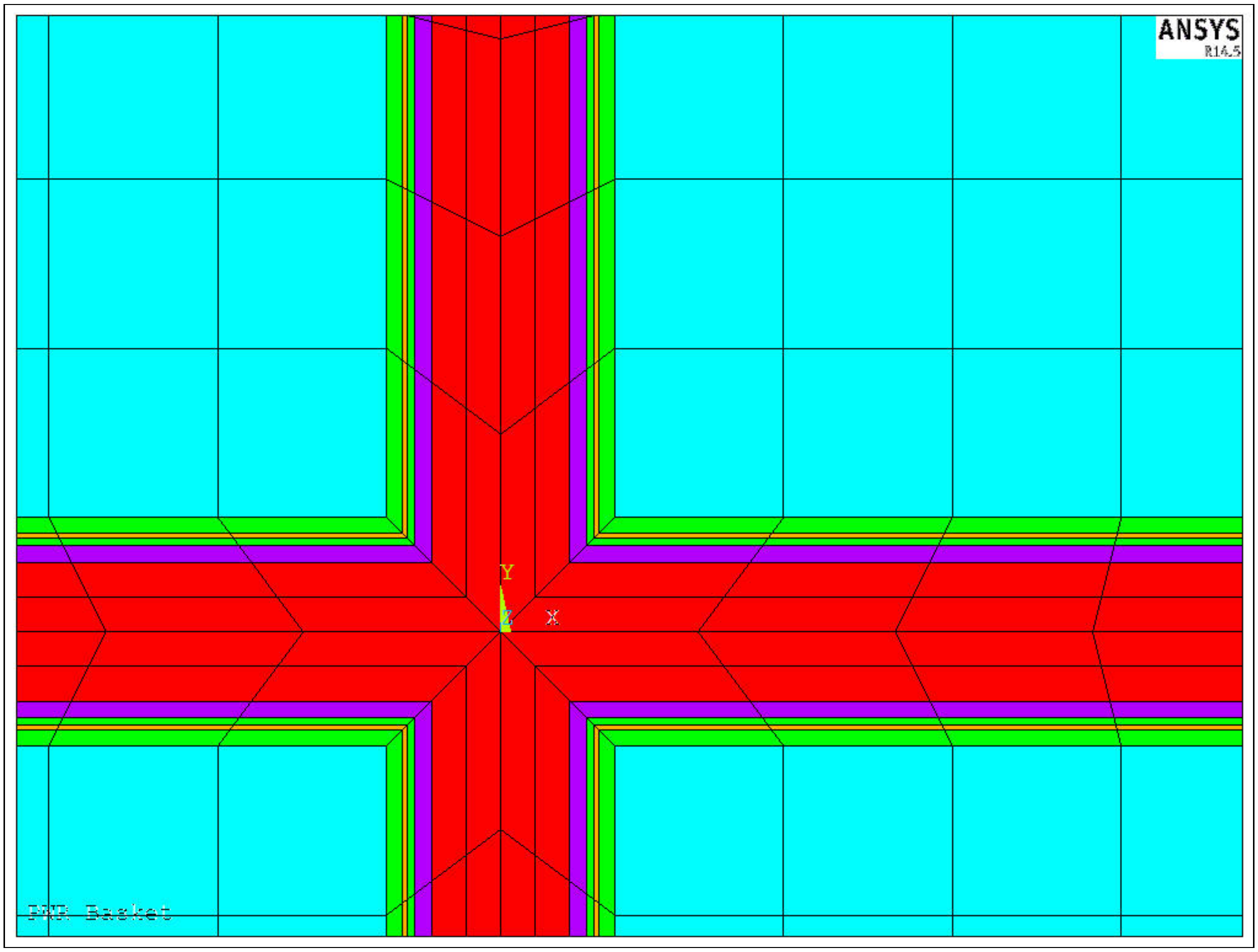


FIGURE 2.3-19: PWR BASKET MODEL ISOMETRIC VIEW

FIGURE 2.3-20: BWR BASKET MODEL

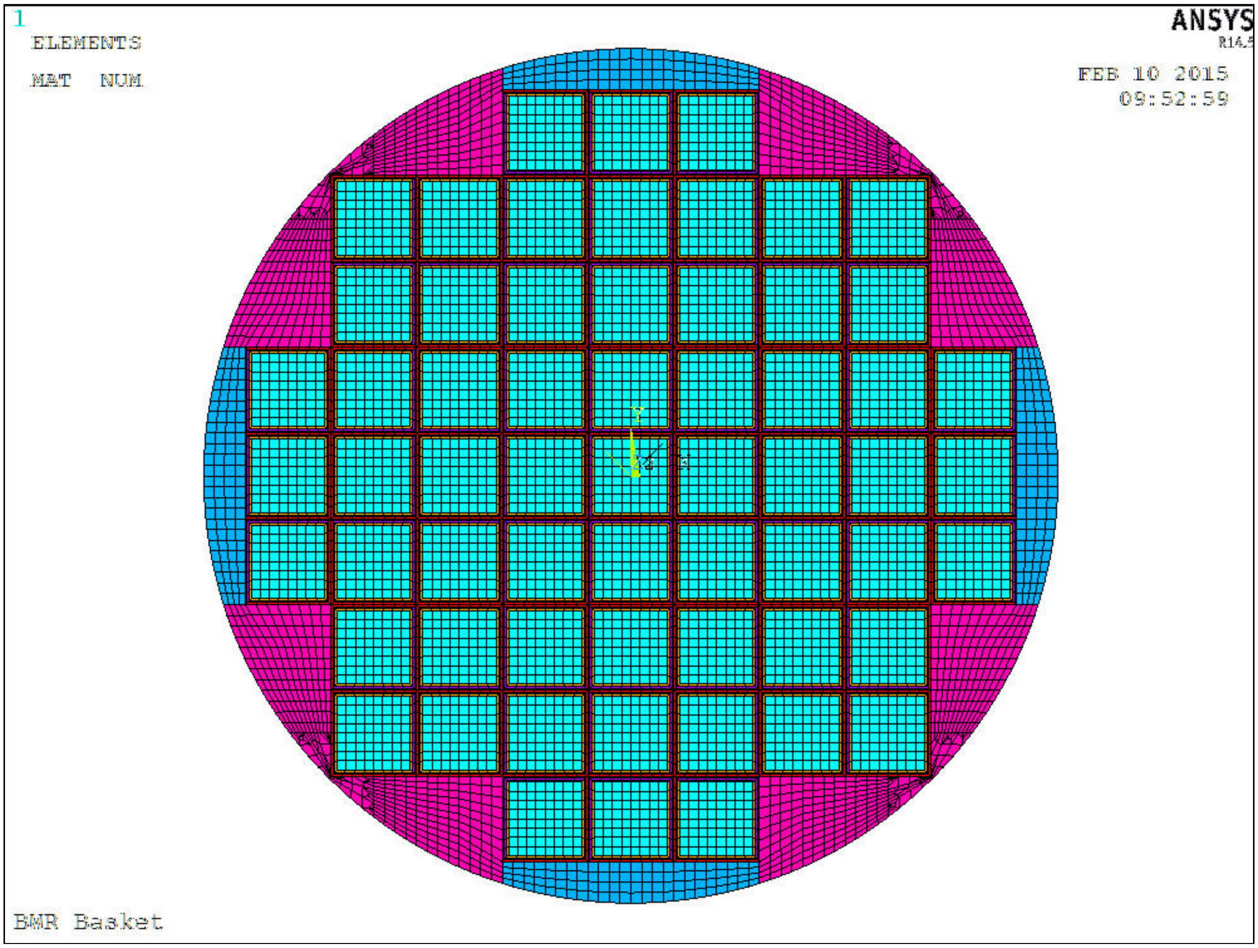


FIGURE 2.3-21: BWR BASKET MODEL CELL DETAIL

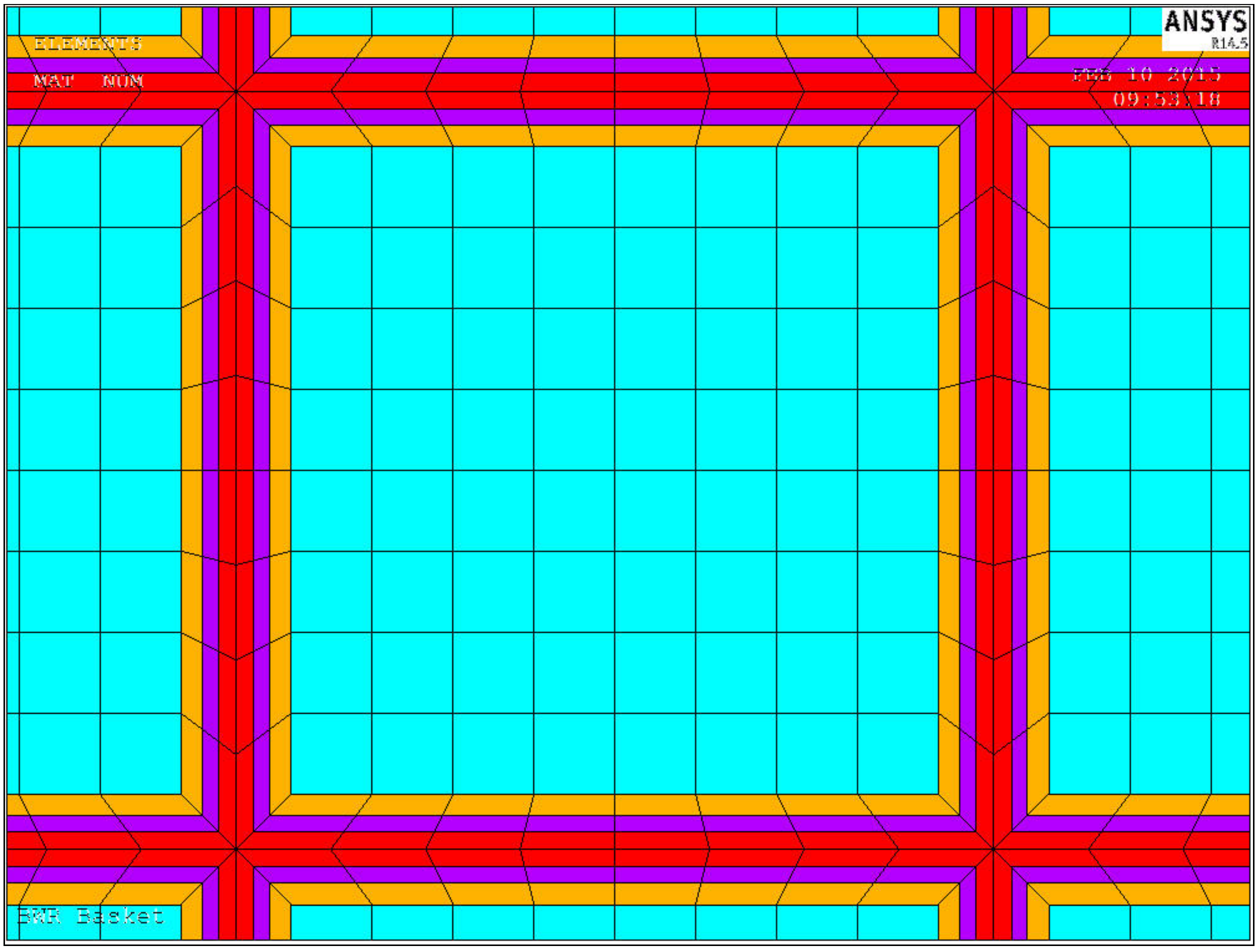
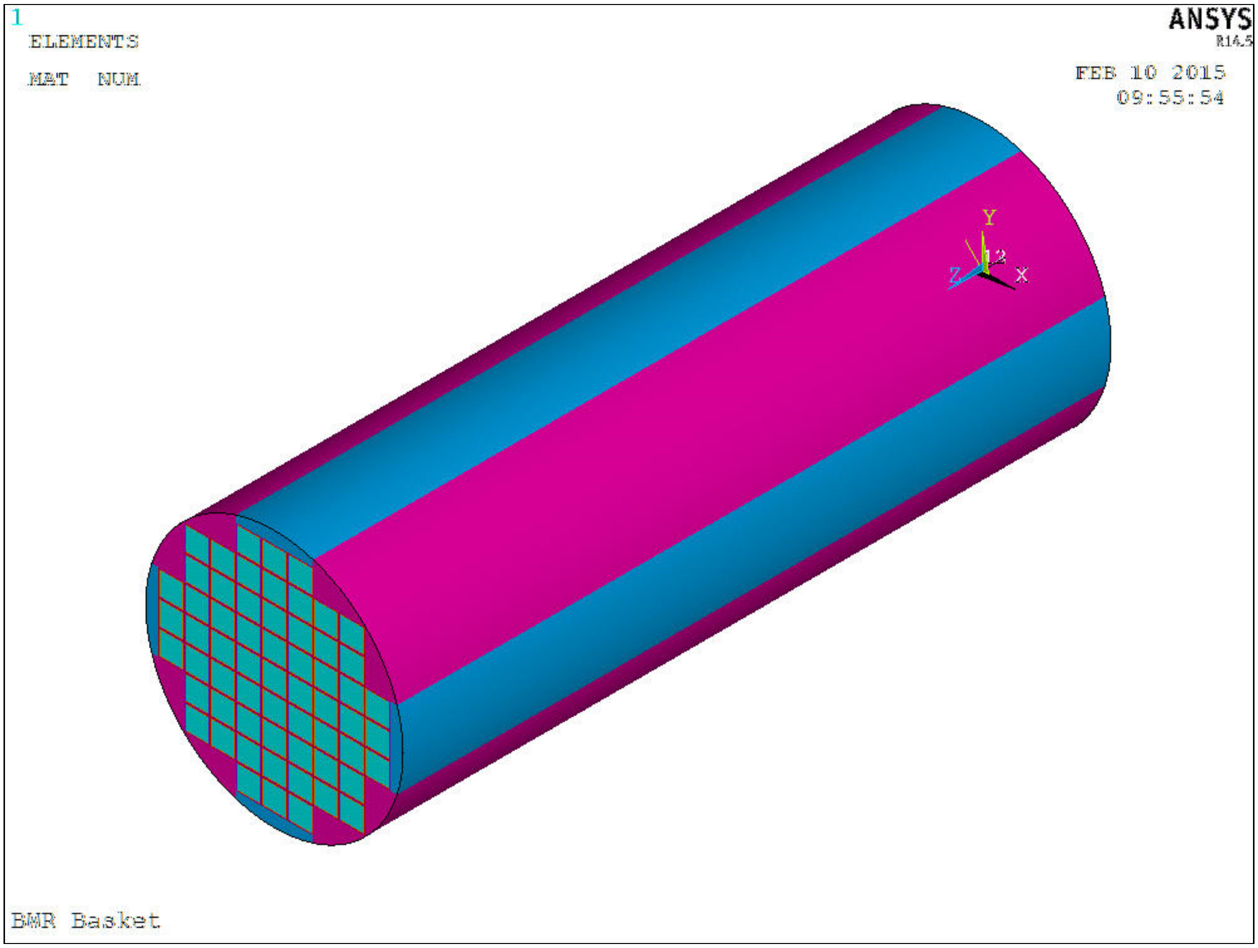


FIGURE 2.3-22: BWR BASKET MODEL ISOMETRIC VIEW



2.4 Containment Review

2.4.1 Description of Containment System

2.4.1.1 Containment Boundary

The 6625B-HB package provides a single level of leaktight containment, defined as a leakage rate of less than 1×10^{-7} reference cubic centimeters per second (ref-cm^3/s), air per ANSI N14.5 [34]. The containment boundary for the 6625B-HB cask consists of the following elements:

- The lower end structure bottom plate
- The inner cylindrical shell
- The upper end structure
- The inner closure lid
- The inner lid containment O-ring seal (the inner seal in the inner closure lid face-type seal)
- The vent port in the inner closure lid (closed with a brass port plug, sealed with a sealing washer)
- The drain port (closed using a brass port plug, sealed with a sealing washer)
- The welds connecting the containment boundary

The extended containment boundary consists of the following elements:

- The space between the inner and outer lids including the upper end structure
- The outer closure lid
- The outer lid containment O-ring seal (the inner seal in the outer closure lid face-type seal)
- The vent port in the outer closure lid (closed with a brass port plug, sealed with a sealing washer)

The containment boundary is shown in **Figure 2.1-2**. The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity. Helium assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation that might otherwise lead to gross failure.

2.4.1.2 Containment Vessel

The cask containment boundary consists of the 1.25-inch thick inner shell, a 1.25-inch thick bottom plate with a 66.25-inch diameter, an upper end structure, a 3.00-inch thick inner lid with a 3.50-inch thick shield plug (lead gamma shielding and steel box with innermost seals and closure bolts, vent and drain ports with closure bolts and seals, and containment boundary welds. A 66.25-inch diameter, 182.00-inch long cavity is provided.

The 2.50-inch thick outer closure lid with outermost seals and closure bolts, and vent port with closure bolts and seals along with the space between the lids meets the design and manufacturing criteria such that it can be merged with the inner containment space to define an extended containment boundary. The extended containment boundary will be utilized only in the unlikely event that the boundary defined by the inner lid fails.

The containment shell material is SA-203, Grade E, and the upper and lower end structures and bottom closure plate materials are SA-350-LF3. The inner and outer lids are constructed from SA-350-LF3 or SA-203, Grade E material. The 6625B-HB packaging containment vessel is designed, fabricated, examined, and tested in accordance with the requirements of Subsection NB

[3] of the ASME B&PV Code to the maximum practical extent. In addition, the design meets the requirements of Regulatory Guides 7.6 [4] and 7.8 [5]. The containment boundary is shown in **Figure 2.1-2**.

The materials of construction meet the requirements of ASME B&PV Section III, Subsection NB-2000 and Section II, material specifications or the corresponding ASTM specifications. The containment vessel is designed to the ASME B&PV Code, Section III, Subsection NB, Article 3200.

The containment vessel is fabricated and examined in accordance with NB-2500, NB-4000, and NB-5000. Also, weld filler materials conform to NB-2400 and the material specification requirements of Section II, Part C of the ASME B&PV.

The materials of the 6625B-HB packaging will not result in any significant chemical, galvanic, or other reaction.

2.4.1.3 Containment Penetrations

Besides the bolted closure lids, the only penetrations into the containment boundary are the drain and vent ports in the inner and outer lids. Each penetration is designed and tested to ensure leaktight-sealing integrity (i.e., a leakage rate not exceeding 1×10^{-7} ref–cm^3/s) per ANSI N14.5.

2.4.1.4 Seals

Containment seals are located at the inner lid, outer lid, the drain plug and the vent plugs. The inner seal in all cases is the primary containment seal. The outer, secondary seals facilitate leakage rate testing of the inner containment seal for both the inner and the outer lids. There are also test ports provided for these two closures. The test ports are not part of the containment boundary.

All the seals used in the 6625B-HB cask containment boundary are static face seals. The seal areas are designed for no significant plastic deformation under normal and accident loads as discussed in *Section 2.2*. The lid bolts are tightened to maintain a seal load during all load conditions as discussed in *Section 2.2*. The seals used for penetrations are fluorocarbon elastomer O-ring seals. All sealing surfaces are stainless steel and are machined to a 63 RMS or finer surface finish.

A fluorocarbon elastomeric seal was selected for use on the 6625B-HB package because it has acceptable characteristics over a wide range of parameters. The fluorocarbon compound specified is VM835-75 or equivalent, which meets the military rubber specification MIL-R-83485; (note that this specification has been superseded by AMS-R-83485). Fluorocarbon O-rings are used in applications where temperatures are between -15°F and 400°F. The VM835-75 compound, as listed on page 8-4 of the Parker O-ring Handbook [14], is specially formulated for use at temperatures as low as -40 °F while maintaining the upper temperature limit of 400°F.

2.4.1.5 Welds

All welds used in the containment boundary are full penetration and volumetrically nondestructively inspected to ensure structural and containment integrity. The welds joining the inner shell to either end structure are ultrasonically inspected in accordance with the ASME B&PV Code, Subsection NB, Article NB–5000, and Section V, Article 4 [18]. The weld joining the inner shell and the lower end structure may be optionally radiograph inspected in accordance with the ASME B&PV Code, Subsection NB, Article NB–5000, and Section V, Article 2 [19]. All containment boundary welds are inspected by liquid penetrant inspection on the final pass in accordance with the ASME B&PV Code, Subsection NB, Article NB–5000, and Section V, Article 6 [20].

2.4.1.6 Closure

The inner lid completes the containment boundary, and is attached to the cask body with (48), SA-540, Grade B23, Class 1, 1.50-inch diameter bolts. The outer lid is attached to the cask body with (48) SA-540, Grade B23, Class 1, 1.50-inch diameter bolts. The bolt tightening torque required for both lid bots is 950 – 1,040 lb-ft. The closure bolt evaluation is discussed in *Section 2.2*. Closure of the vent and drain ports is accomplished by a single 0.75-inch brass or ASTM A193 B8 bolt with a sealing washer under the head of the bolt.

The inner and outer closure lids cannot become detached by any internal pressure, NCT, or HAC events. The inner closure lid, including the vent port is completely covered by the outer closure lid and the outer closure lid is completely covered by the upper impact limiter, which is attached to the cask with 12, SA-540 Grade B23, Class 1, 1.50-inch diameter bolts. Thus, the containment openings cannot be inadvertently opened.

2.4.2 Containment Under Normal Conditions of Transport

The results of the NCT structural and thermal evaluations presented in *Section 2.2* and *Section 2.3* respectively demonstrate that there is no release of radioactive materials per the ‘leaktight’ definition of ANSI N14.5 under any of the NCT tests described in 10 CFR §71.71 [1].

2.4.3 Containment Under Hypothetical Accident Conditions

The results of the HAC structural and thermal evaluations performed in *Section 2.2* and *Section 2.3* respectively, demonstrate that there is no release of radioactive materials per the ‘leaktight’ definition of ANSI N14.5 under any of the HAC tests described in 10 CFR §71.73.

2.4.4 Leakage Rate Tests for Type B Packages

2.4.4.1 Fabrication Leakage Rate Test

During fabrication, the containment boundary is leakage rate tested as described in *Section 2.8*. The fabrication leakage rate tests are consistent with the guidelines of ANSI N14.5. This leakage rate test verifies the containment integrity of the 6625B-HB packaging to a leakage rate not to exceed 1×10^{-7} ref–cm^3/s, air.

2.4.4.2 Maintenance/Periodic Leakage Rate Tests

Annually, or at the time of damaged containment seal replacement or sealing surface repair, the containment O–ring seal and the vent port and drain port sealing washers are leakage rate tested as described in *Section 2.8*. The maintenance/periodic leakage rate tests are consistent with the guidelines of ANSI N14.5. This test verifies the sealing integrity of the containment seals to a leakage rate not to exceed 1×10^{-7} ref– cm^3/s, air.

2.4.4.3 Preshipment Leakage Rate Tests

Prior to shipment of the loaded 6625B-HB cask, the containment O–ring seal and the vent port and drain port sealing washers are leakage rate tested per *Section 2.8*. The preshipment leakage rate tests are consistent with the guidelines of ANSI N14.5. This test verifies the sealing integrity of the containment seals to a leakage rate sensitivity of 1×10^{-3} ref–cm^3/s, air.

2.5 Shielding

The purpose of this section is to describe the shielding evaluations performed for the 6625B-HB transportation package. This package shall meet the dose rate requirements for exclusive use transportation in an open vehicle per 10 CFR 71. The normal condition dose rate limits are:

1. 10 CFR 71.47(b)(1): 200 mrem/hr on the accessible surfaces of the package
2. 10 CFR 71.47(b)(2): 200 mrem/hr on the surfaces of the vehicle, including the top and underside
3. 10 CFR 71.47(b)(3): 10 mrem/hr 2 m from the side surfaces of the vehicle
4. 10 CFR 71.47(b)(4): 2 mrem/hr in any occupied location. This limit does not apply to private carries if exposed personnel wear radiation dosimetry

For accident conditions, the dose rate is limited to 1,000 mrem/hr at a distance of 1 m from the surfaces of the package (10 CFR 71.51(a)(2)).

2.5.1 Description of the Shielding Design

Gamma shielding is primarily provided by the steel-lead-steel design of the cask body. The inner shell of the cask is 1.25-inch thick carbon steel, a lead core is 3.00 inches thick, and an outer carbon steel shell is 2.75 inches thick. The bottom of the package features 1.25-inch thick carbon steel (inner), 4.50-inch thick lead, and 2.75-inch thick carbon steel (outer). The cask has a double-lid design. The inner lid features 1.00-inch thick carbon steel (inner), 2.50-inch thick lead, and 3.00-inch thick carbon steel (outer). The outer lid features 2.50-inch thick carbon steel.

Neutron shielding is provided by 6 inches of VYAL-B neutron shielding resin at the side of the cask between the impact limiters. The neutron shield extends approximately 11 inches under the impact limiters, although in this region, the resin is only 5 inches thick.

2.5.2 Source Specification

The gamma and neutron source terms are developed using the ORIGEN-ARP module of the SCALE 6.0 code package [21]. For design purposes, the B&W 15x15 fuel assembly is utilized as the bounding PWR fuel assembly, and the GE 7x7 is utilized as the bounding BWR fuel assembly. These fuel assemblies are selected because the heavy metal loading (0.492 MTU for PWR and 0.198 MTU for BWR) bounds most fuel assembly types and the source term is proportional to the heavy metal loading. The minimum cooling times developed for the design basis assemblies are applied to assemblies with a lower fuel loading.

ORIGEN-ARP libraries *TN_BW15x15_NX2_R0* (for PWR) and *GE7x7-0* (for BWR) are used in the analysis. The PWR library is developed by AREVA TN using conservative assumptions for fuel temperature and water density. The BWR library is provided with ORIGEN-ARP. For simplicity, the source terms are generated using continuous power operation (i.e., no down time between cycles). The specific power is fixed at 40 MW/MTU for all inputs (assembly power of 19.68 MW for PWR and 7.92 MW for BWR) and the irradiation time is selected to give the target burnup using the simple equation Time (days) = Burnup (MWd/MTU)/40 (MW/MTU).

The proposed PWR heat load zone configuration is shown in **Figure 2.1-6**. The basket holds 24 PWR fuel assemblies in 4 zones:

- Zone 1: 0.9 kW/FA (4 fuel assemblies)
- Zone 2: 1.4 kW/FA (8 fuel assemblies)
- Zone 3: 2.1 kW/FA (4 fuel assemblies)

- Zone 4: 0.9 kW/FA (8 fuel assemblies)

The proposed BWR heat load zone configuration is shown in **Figure 2.1-8**. The basket holds 61 BWR fuel assemblies in 4 zones:

- Zone 1: 0.33 kW/FA (9 fuel assemblies)
- Zone 2: 0.78 kW/FA (16 fuel assemblies)
- Zone 3: 0.45 kW/FA (24 fuel assemblies)
- Zone 4: 0.33 kW/FA (12 fuel assemblies)

For both the PWR and BWR baskets, Zones 1 and 4 have the same maximum assembly decay heat but are treated separately because the bounding source in Zone 1 may be different than in Zone 4.

The package shall accept fuel with a maximum burnup of 62.5 GWd/MTU. The minimum cooling time for all fuel assemblies is 5 years. A limited number of burnup, enrichment, and cooling time combinations are examined to develop fuel qualification tables (FQTs) to be used for fuel loading. FQTs for PWR and BWR fuel are provided in **Table 2.5-1** and **2.5-2,** respectively. Reasonable minimum enrichments are selected, as lower enrichments cause the source term to increase. These FQTs are highly simplified compared to what might be used in an actual transportation cask license. A detailed FQT for a transportation license may have a large matrix of burnup/enrichment/cooling time combinations for each heat load of interest. These FQTs are based solely on heat load.

The decay heat in the active fuel region is calculated using ORIGEN-ARP. The decay heat from the end hardware (and NFAH for PWR fuel) is primarily due to Co-60. While the decay heat from irradiated metal must be considered, this decay heat is a small fraction of the decay heat produced by the fission products and actinides and is neglected when developing the FQTs.

An active fuel source term is developed for each of the burnup and enrichment combinations shown in the FQTs. Only two light elements are included in the source term input files—cobalt and nickel—because Co-60 is generated from activation of Co-59 and Ni-60.

A design basis source term is selected for each zone. The limiting dose rate for a transportation package is typically the dose rate at a distance of 2 m from the side of the vehicle. Therefore, the design basis source terms are selected to maximize the dose rate at this location. Dose rates at this location are computed using Monte Carlo N-Particle (MCNP)5 v1.51 and a detailed radial model of the package. In the MCNP models, the basket is modeled explicitly and the fuel is homogenized within the basket compartments. The dose rate is tallied 2 m from the impact limiter surface and is explicitly computed for each zone and source term. The details of the MCNP modeling are described in the next section.

Based on the dose rate 2 m from the side of the vehicle computed using MCNP, the following PWR source terms are bounding:

- Zone 1: 62.5 GWd/MTU, 26 year cooled
- Zone 2: 62.5 GWd/MTU, 9 year cooled
- Zone 3: 62.5 GWd/MTU, 5 years cooled
- Zone 4: 30 GWd/MTU, 5 years cooled

Based on the dose rate 2 m from the side of the vehicle computed using MCNP, the following BWR source terms are bounding:

- Zone 1: 62.5 GWd/MTU, 25 year cooled
- Zone 2: 62.5 GWd/MTU, 5 year cooled
- Zone 3: 62.5 GWd/MTU, 12.5 years cooled
- Zone 4: 29 GWd/MTU, 5 years cooled

Note that the highest burnup source (maximum neutron source) is bounding in Zones 1, 2, and 3, although a lower burnup, shorter cooling time source is bounding in Zone 4. The reason is that Zones 1, 2, and 3 are highly self-shielded by fuel assemblies so that neutrons primarily contribute to the tally. Zone 4 is the outermost zone and gammas are more dominant because Zone 4 fuel is the least shielded.

The design basis source terms are selected based upon the active fuel source only (no fuel hardware). However, there is a Co-60 source in the end hardware regions. The fuel assembly hardware regions are divided into bottom nozzle, plenum, and top nozzle regions. Because these regions are outside of the active core region, these regions will experience a reduced flux. For PWR fuel, flux scaling factors of 0.1 are used in the top nozzle, and 0.2 in the plenum and bottom nozzle [22]. For BWR fuel, flux scaling factors of 0.1 are used in the top nozzle, 0.2 in the plenum, and 0.15 bottom nozzle [22]. The light element masses in the end regions are scaled by the flux factors prior to input to ORIGEN-ARP. The only light elements of interest are cobalt and nickel, as these elements activate to Co-60. The light element masses used as input to ORIGEN-ARP are summarized in **Table 2.5-3**.

NFAH, such as BPRAs, is also authorized for transportation with PWR fuel. NFAH is a source of Co-60 activation as well as a small quantity of decay heat. The NFAH source is accounted for by adding a Co-60 source to the PWR fuel source terms. The NFAH source is the same for each basket location. The assumed NFAH source per basket location is as follows:

- NFAH Bottom Nozzle: none
- NFAH Active Fuel: 600 Ci Co-60 per fuel assembly
- NFAH Plenum: 30 Ci Co-60 per fuel assembly
- NFAH Top Nozzle: 20 Ci Co-60 per fuel assembly

Based on the above considerations, a total source term for NCT is determined. The PWR NCT source term in each of the four zones is provided in **Tables 2.5-4** through **2.5-7**. The sources presented are the sum of the fuel and NFAH sources. The BWR NCT source term in each of the four zones is provided in **Tables 2.5-9** through **2.5-12**.

For HAC, the neutron shield is assumed completely lost while the gamma shield receives little damage. In the absence of a neutron shield, the neutron component dominates the dose rate. Therefore, the source in each region is selected to maximize the neutron source. Zones 1, 2, and 3 have the same source terms for NCT and HAC because the neutron component is already maximized. However, the source in Zone 4 is changed to a high-burnup source for HAC, as shown in **Table 2.5-8** and **2.5-13** for PWR and BWR fuel, respectively.

The neutron source is presented as a total source magnitude in **Tables 2.5-4** through **2.5-13** because the source is almost entirely from decay of Cm-244. Because Cm-244 typically results in 95 percent of the neutron source term, the Cm-244 spectrum is used for the neutron source in MCNP. The 'raw' neutron source computed by ORIGEN-ARP is reported in the source term summary tables. However, the neutron sources computed by ORIGEN-ARP do not account for the axial burnup profile of the fuel, because ORIGEN-ARP considers the fuel only in two dimensions. The increase of the neutron source magnitude due to the axial burnup profile is estimated to be 1.152 and 1.210 for PWR and BWR fuel, respectively. (These values are derived later in **Tables 2.5-14** and **2.5-15**).

Subcritical neutron multiplication is suppressed in MCNP by using the NONU card. Subcritical neutron multiplication is accounted for by multiplying the raw neutron source by 1/(1-k), where k is the effective multiplication factor for the system. Because the neutron source is primarily from fuel with an initial enrichment of 3.8 percent and the fuel is burned to a high value of 62.5 GWd/MTU, the reactivity of the high-burnup fuel is low. A reasonable value for subcritical neutron multiplication is 0.26 for both PWR and BWR fuel. Therefore, to combine the effects of the axial burnup profile and subcritical neutron multiplication, the raw neutron sources are scaled by 1.152/(1-0.26) = 1.557 for PWR fuel and 1.210/(1-0.26) = 1.635 for BWR fuel. The scaled neutron sources used in the MCNP input files are also provided in the source term summary tables.

All source terms are provided on a per-assembly basis. However, MCNP requires a total normalization factor, which is the total particles/s in the model. The total particles/s for the various models is as follows.

PWR Cases:

- NCT gamma: 1.402E+17 γ/s per cask
- NCT neutron: 3.236E+10 n/s per cask
- HAC gamma: 1.242E+17 γ/s per cask
- HAC neutron: 3.698E+10 n/s per cask

BWR Cases:

- NCT gamma: 1.384E+17 γ/s per cask
- NCT neutron: 3.363E+10 n/s per cask
- HAC gamma: 1.307E+17 γ/s per cask
- HAC neutron: 3.628E+10 n/s per cask

Fuel is modeled in MCNP with an axial burnup profile. The gamma source term is proportional to the burnup. The PWR gamma axial profile is obtained from [23] for fuel with burnups >46 GWd/MTU. This profile is for 18 axial nodes of equal size and is provided in **Table 2.5-14**. The BWR gamma axial profile is for fuel with a burnup of 40.2 GWd/MTU. This distribution is highly peaked and is conservative. This profile is for 25 axial nodes of equal size and is provided in **Table 2.5-15**. The neutron source is approximately proportional to the 4th power of the burnup and the neutron axial source distribution is developed by raising the gamma profile to the 4th power. The average value of the neutron axial source distribution is 1.152 and 1.210 for PWR and BWR fuel, respectively, which represents the increase of the neutron source due to the applied axial burnup profile.

TABLE 2.5-1: PWR FUEL QUALIFICATION TABLE

Maximum Burnup (GWd/MTU)	Minimum Enrichment (%)	Minimum Cooling Time (years)		
		Zone 1/4	Zone 2	Zone 3
		Heat ≤ 0.9 kW	Heat ≤ 1.4 kW	Heat ≤ 2.1 kW
≤ 30	≥ 1.8	≥ 5	≥ 5	≥ 5
≤ 37	≥ 2.3	≥ 6.5	≥ 5	≥ 5
≤ 45	≥ 2.8	≥ 10	≥ 5	≥ 5
≤ 53	≥ 3.3	≥ 16	≥ 6.5	≥ 5
≤ 62.5	≥ 3.8	≥ 26	≥ 9	≥ 5

Note: The numbers in blue are determined by inspection because the upper bound decay heat cannot be achieved for this burnup/cooling time combination.

TABLE 2.5-2: BWR FUEL QUALIFICATION TABLE

Maximum Burnup (GWd/MTU)	Minimum Enrichment (%)	Minimum Cooling Time (years)		
		Zone 1/4	Zone 2	Zone 3
		Heat ≤ 0.33 kW	Heat ≤ 0.78 kW	Heat ≤ 0.45 kW
≤ 29	≥ 1.5	≥ 5	≥ 5	≥ 5
≤ 35	≥ 2.2	≥ 6	≥ 5	≥ 5
≤ 39	≥ 2.4	≥ 7.2	≥ 5	≥ 5
≤ 45	≥ 2.8	≥ 10	≥ 5	≥ 6
≤ 53	≥ 3.3	≥ 16	≥ 5	≥ 8
≤ 62.5	≥ 3.8	≥ 25	≥ 5	≥ 12.5

Note: The numbers in blue are determined by inspection because the upper bound decay heat cannot be achieved for this burnup/cooling time combination.

TABLE 2.5-3: LIGHT ELEMENT INPUT TO ORIGEN-ARP

PWR				
Element	Bottom Nozzle (g)	Active Fuel (g)	Plenum (g)	Top Nozzle (g)
Cobalt	0.945	4.065	0.555	0.549
Nickel	256.53	2604	562.36	175.31
BWR				
Element	Bottom Nozzle (g)	Active Fuel (g)	Plenum (g)	Top Nozzle (g)
Cobalt	0.358	1.62	0.115	0.129
Nickel	68.22	412.8	18.7	50.0

Note: The masses in the bottom nozzle, plenum, and top nozzle provided in this table have been reduced by flux scaling factors.

TABLE 2.5-4: PWR NCT AND HAC SOURCE TERM PER FUEL ASSEMBLY, ZONE 1 (INCLUDES NFAH)

Burnup: 62.5 GWd/MTU Enrichment: 3.8% Cooling Time: 26 years Decay Heat: 0.9 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	7.459E+09	8.326E+14	3.648E+10	2.452E+10
1.00E-01	1.329E+09	2.422E+14	6.925E+09	4.780E+09
2.00E-01	3.211E+08	1.509E+14	1.671E+09	1.153E+09
3.00E-01	1.673E+07	4.596E+13	8.470E+07	5.779E+07
4.00E-01	2.184E+07	3.035E+13	1.109E+08	7.577E+07
6.00E-01	3.242E+06	2.468E+13	1.109E+07	6.062E+06
8.00E-01	2.138E+06	1.603E+15	6.070E+06	2.796E+06
1.00E+00	1.796E+07	1.594E+13	9.154E+07	6.263E+07
1.33E+00	3.878E+11	5.991E+13	2.021E+12	1.395E+12
1.66E+00	1.095E+11	1.237E+13	5.707E+11	3.940E+11
2.00E+00	4.614E-20	7.799E+10	1.011E-19	3.153E-20
2.50E+00	2.620E+06	4.284E+09	1.366E+07	9.428E+06
3.00E+00	2.239E+03	8.816E+08	1.167E+04	8.054E+03
4.00E+00	2.267E-22	6.721E+07	4.969E-22	1.549E-22
5.00E+00	0.000E+00	2.269E+07	0.000E+00	0.000E+00
6.50E+00	0.000E+00	9.104E+06	0.000E+00	0.000E+00
8.00E+00	0.000E+00	1.786E+06	0.000E+00	0.000E+00
1.00E+01	0.000E+00	3.792E+05	0.000E+00	0.000E+00
Total	5.065E+11	3.018E+15	2.637E+12	1.820E+12
Neutron Source				
Raw source: 6.624E+08 n/s Including peaking and subcritical neutron multiplication: 1.031E+09 n/s				

TABLE 2.5-5: PWR NCT AND HAC SOURCE TERM PER FUEL ASSEMBLY, ZONE 2 (INCLUDES NFAH)

Burnup: 62.5 GWd/MTU Enrichment: 3.8% Cooling Time: 9 years Decay Heat: 1.4 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	6.324E+10	1.371E+15	7.549E+10	5.733E+10
1.00E-01	1.243E+10	3.701E+14	1.466E+10	1.131E+10
2.00E-01	3.001E+09	2.823E+14	3.540E+09	2.728E+09
3.00E-01	1.497E+08	7.994E+13	1.773E+08	1.360E+08
4.00E-01	1.963E+08	4.969E+13	2.325E+08	1.783E+08
6.00E-01	1.427E+07	4.210E+14	1.878E+07	1.255E+07
8.00E-01	5.968E+06	2.799E+15	8.738E+06	5.048E+06
1.00E+00	1.624E+08	2.209E+14	1.921E+08	1.475E+08
1.33E+00	3.629E+12	1.495E+14	4.279E+12	3.301E+12
1.66E+00	1.025E+12	3.277E+13	1.208E+12	9.321E+11
2.00E+00	3.935E-04	2.267E+11	8.625E-04	2.689E-04
2.50E+00	2.452E+07	1.310E+11	2.891E+07	2.231E+07
3.00E+00	2.095E+04	9.607E+09	2.470E+04	1.906E+04
4.00E+00	3.244E-22	9.425E+08	7.110E-22	2.217E-22
5.00E+00	0.000E+00	4.279E+07	0.000E+00	0.000E+00
6.50E+00	0.000E+00	1.717E+07	0.000E+00	0.000E+00
8.00E+00	0.000E+00	3.369E+06	0.000E+00	0.000E+00
1.00E+01	0.000E+00	7.154E+05	0.000E+00	0.000E+00
Total	4.733E+12	5.776E+15	5.581E+12	4.305E+12
Neutron Source				
Raw source: 1.247E+09 n/s Including peaking and subcritical neutron multiplication: 1.941E+09 n/s				

TABLE 2.5-6: PWR NCT AND HAC SOURCE TERM PER FUEL ASSEMBLY, ZONE 3 (INCLUDES NFAH)

Burnup: 62.5 GWd/MTU Enrichment: 3.8% Cooling Time: 5 years Decay Heat: 2.1 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	1.064E+11	2.139E+15	1.056E+11	8.273E+10
1.00E-01	2.104E+10	6.065E+14	2.066E+10	1.637E+10
2.00E-01	5.079E+09	5.023E+14	4.992E+09	3.951E+09
3.00E-01	2.529E+08	1.402E+14	2.492E+08	1.966E+08
4.00E-01	3.316E+08	9.433E+13	3.267E+08	2.578E+08
6.00E-01	2.286E+07	1.605E+15	2.482E+07	1.760E+07
8.00E-01	8.937E+06	4.230E+15	1.081E+07	6.794E+06
1.00E+00	2.745E+08	7.290E+14	2.704E+08	2.134E+08
1.33E+00	6.142E+12	2.406E+14	6.029E+12	4.778E+12
1.66E+00	1.734E+12	7.460E+13	1.702E+12	1.350E+12
2.00E+00	6.271E+02	1.816E+12	1.375E+03	4.285E+02
2.50E+00	4.150E+07	3.057E+12	4.073E+07	3.229E+07
3.00E+00	3.546E+04	1.350E+11	3.481E+04	2.759E+04
4.00E+00	3.529E-22	1.261E+10	7.736E-22	2.412E-22
5.00E+00	0.000E+00	4.997E+07	0.000E+00	0.000E+00
6.50E+00	0.000E+00	2.005E+07	0.000E+00	0.000E+00
8.00E+00	0.000E+00	3.934E+06	0.000E+00	0.000E+00
1.00E+01	0.000E+00	8.354E+05	0.000E+00	0.000E+00
Total	8.009E+12	1.037E+16	7.863E+12	6.232E+12
Neutron Source				
Raw source: 1.458E+09 n/s Including peaking and subcritical neutron multiplication: 2.270E+09 n/s				

TABLE 2.5-7: PWR NCT SOURCE TERM PER FUEL ASSEMBLY, ZONE 4 (INCLUDES NFAH)

Burnup: 30 GWd/MTU Enrichment: 1.8% Cooling Time: 5 years Decay Heat: 0.9 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	8.022E+10	1.185E+15	8.315E+10	6.705E+10
1.00E-01	1.585E+10	3.483E+14	1.627E+10	1.327E+10
2.00E-01	3.827E+09	2.902E+14	3.929E+09	3.202E+09
3.00E-01	1.906E+08	8.130E+13	1.962E+08	1.594E+08
4.00E-01	2.499E+08	5.758E+13	2.572E+08	2.090E+08
6.00E-01	1.735E+07	6.713E+14	1.966E+07	1.427E+07
8.00E-01	6.835E+06	1.925E+15	8.606E+06	5.511E+06
1.00E+00	2.068E+08	2.785E+14	2.129E+08	1.730E+08
1.33E+00	4.628E+12	1.337E+14	4.746E+12	3.873E+12
1.66E+00	1.307E+12	3.898E+13	1.340E+12	1.094E+12
2.00E+00	5.580E+02	1.350E+12	1.223E+03	3.813E+02
2.50E+00	3.127E+07	2.554E+12	3.207E+07	2.618E+07
3.00E+00	2.672E+04	1.016E+11	2.740E+04	2.236E+04
4.00E+00	1.662E-23	9.423E+09	3.642E-23	1.136E-23
5.00E+00	0.000E+00	1.015E+07	0.000E+00	0.000E+00
6.50E+00	0.000E+00	4.075E+06	0.000E+00	0.000E+00
8.00E+00	0.000E+00	7.993E+05	0.000E+00	0.000E+00
1.00E+01	0.000E+00	1.697E+05	0.000E+00	0.000E+00
Total	6.036E+12	5.013E+15	6.190E+12	5.051E+12
Neutron Source				
Raw source: 2.914E+08 n/s Including peaking and subcritical neutron multiplication: 4.536E+08 n/s				

TABLE 2.5-8: PWR HAC SOURCE TERM PER FUEL ASSEMBLY, ZONE 4 (INCLUDES NFAH)

Burnup: 62.5 GWd/MTU Enrichment: 3.8% Cooling Time: 26 years Decay Heat: 0.9 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	7.459E+09	8.326E+14	3.648E+10	2.452E+10
1.00E-01	1.329E+09	2.422E+14	6.925E+09	4.780E+09
2.00E-01	3.211E+08	1.509E+14	1.671E+09	1.153E+09
3.00E-01	1.673E+07	4.596E+13	8.470E+07	5.779E+07
4.00E-01	2.184E+07	3.035E+13	1.109E+08	7.577E+07
6.00E-01	3.242E+06	2.468E+13	1.109E+07	6.062E+06
8.00E-01	2.138E+06	1.603E+15	6.070E+06	2.796E+06
1.00E+00	1.796E+07	1.594E+13	9.154E+07	6.263E+07
1.33E+00	3.878E+11	5.991E+13	2.021E+12	1.395E+12
1.66E+00	1.095E+11	1.237E+13	5.707E+11	3.940E+11
2.00E+00	4.614E-20	7.799E+10	1.011E-19	3.153E-20
2.50E+00	2.620E+06	4.284E+09	1.366E+07	9.428E+06
3.00E+00	2.239E+03	8.816E+08	1.167E+04	8.054E+03
4.00E+00	2.267E-22	6.721E+07	4.969E-22	1.549E-22
5.00E+00	0.000E+00	2.269E+07	0.000E+00	0.000E+00
6.50E+00	0.000E+00	9.104E+06	0.000E+00	0.000E+00
8.00E+00	0.000E+00	1.786E+06	0.000E+00	0.000E+00
1.00E+01	0.000E+00	3.792E+05	0.000E+00	0.000E+00
Total	5.065E+11	3.018E+15	2.637E+12	1.820E+12
Neutron Source				
Raw source: 6.624E+08 n/s Including peaking and subcritical neutron multiplication: 1.031E+09 n/s				

TABLE 2.5-9: BWR NCT AND HAC SOURCE TERM PER FUEL ASSEMBLY, ZONE 1

Burnup: 62.5 GWd/MTU Enrichment: 3.8% Cooling Time: 25 years Decay Heat: 0.33 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	3.803E+09	3.429E+14	1.583E+09	1.568E+09
1.00E-01	6.817E+08	9.649E+13	2.194E+08	2.575E+08
2.00E-01	1.729E+08	6.283E+13	9.457E+07	6.766E+07
3.00E-01	9.040E+06	1.931E+13	5.687E+06	3.687E+06
4.00E-01	1.297E+07	1.297E+13	1.342E+07	5.575E+06
6.00E-01	4.459E+07	1.031E+13	2.236E+08	2.958E+07
8.00E-01	2.339E+07	6.595E+14	1.167E+08	1.557E+07
1.00E+00	9.108E+06	5.582E+12	2.888E+06	3.526E+06
1.33E+00	1.986E+11	8.502E+12	6.319E+10	7.494E+10
1.66E+00	5.609E+10	9.337E+11	1.785E+10	2.116E+10
2.00E+00	6.101E+00	3.324E+10	3.158E+01	4.069E+00
2.50E+00	1.342E+06	1.728E+09	4.270E+05	5.064E+05
3.00E+00	1.147E+03	1.436E+08	3.648E+02	4.327E+02
4.00E+00	1.371E-11	2.445E+07	4.173E-12	4.773E-12
5.00E+00	0.000E+00	8.251E+06	0.000E+00	0.000E+00
6.50E+00	0.000E+00	3.312E+06	0.000E+00	0.000E+00
8.00E+00	0.000E+00	6.496E+05	0.000E+00	0.000E+00
1.00E+01	0.000E+00	1.379E+05	0.000E+00	0.000E+00
Total	2.595E+11	1.219E+15	8.330E+10	9.806E+10
Neutron Source				
Raw source: 2.382E+08 n/s Including peaking and subcritical neutron multiplication: 3.894E+08 n/s				

TABLE 2.5-10: BWR NCT AND HAC SOURCE TERM PER FUEL ASSEMBLY, ZONE 2

Burnup: 62.5 GWd/MTU Enrichment: 3.8% Cooling Time: 5 years Decay Heat: 0.78 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	6.245E+10	8.529E+14	8.989E+10	2.785E+10
1.00E-01	9.595E+09	2.397E+14	3.455E+09	3.640E+09
2.00E-01	3.640E+09	1.949E+14	7.505E+09	1.749E+09
3.00E-01	2.073E+08	5.551E+13	5.171E+08	1.052E+08
4.00E-01	4.576E+08	3.852E+13	1.642E+09	2.622E+08
6.00E-01	6.945E+09	5.574E+14	3.583E+10	4.628E+09
8.00E-01	3.622E+09	1.600E+15	1.869E+10	2.414E+09
1.00E+00	4.690E+10	2.442E+14	1.416E+10	1.449E+10
1.33E+00	2.758E+12	6.859E+13	8.775E+11	1.041E+12
1.66E+00	7.789E+11	2.258E+13	2.478E+11	2.939E+11
2.00E+00	1.460E+02	7.235E+11	9.344E+01	1.063E+02
2.50E+00	1.864E+07	1.226E+12	5.929E+06	7.032E+06
3.00E+00	1.592E+04	5.345E+10	5.066E+03	6.008E+03
4.00E+00	2.090E-11	4.998E+09	6.363E-12	7.276E-12
5.00E+00	0.000E+00	1.755E+07	0.000E+00	0.000E+00
6.50E+00	0.000E+00	7.042E+06	0.000E+00	0.000E+00
8.00E+00	0.000E+00	1.382E+06	0.000E+00	0.000E+00
1.00E+01	0.000E+00	2.933E+05	0.000E+00	0.000E+00
Total	3.671E+12	3.876E+15	1.297E+12	1.390E+12
Neutron Source				
Raw source: 5.053e+08 n/s Including peaking and subcritical neutron multiplication: 8.260E+08 n/s				

TABLE 2.5-11: BWR NCT AND HAC SOURCE TERM PER FUEL ASSEMBLY, ZONE 3

Burnup: 62.5 GWd/MTU Enrichment: 3.8% Cooling Time: 12.5 years Decay Heat: 0.45 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	1.983E+10	4.793E+14	1.496E+10	8.117E+09
1.00E-01	3.544E+09	1.297E+14	1.187E+09	1.340E+09
2.00E-01	1.053E+09	9.338E+13	1.280E+09	4.532E+08
3.00E-01	5.646E+07	2.764E+13	8.512E+07	2.542E+07
4.00E-01	1.016E+08	1.796E+13	2.546E+08	5.172E+07
6.00E-01	1.037E+09	5.436E+13	5.336E+09	6.904E+08
8.00E-01	5.406E+08	9.262E+14	2.783E+09	3.601E+08
1.00E+00	1.528E+08	2.989E+13	4.685E+07	5.040E+07
1.33E+00	1.028E+12	2.531E+13	3.272E+11	3.880E+11
1.66E+00	2.904E+11	4.056E+12	9.239E+10	1.096E+11
2.00E+00	8.301E+00	4.971E+10	4.297E+01	5.536E+00
2.50E+00	6.949E+06	5.849E+09	2.211E+06	2.622E+06
3.00E+00	5.937E+03	4.814E+08	1.889E+03	2.240E+03
4.00E+00	1.785E-11	6.883E+07	5.431E-12	6.212E-12
5.00E+00	0.000E+00	1.318E+07	0.000E+00	0.000E+00
6.50E+00	0.000E+00	5.290E+06	0.000E+00	0.000E+00
8.00E+00	0.000E+00	1.038E+06	0.000E+00	0.000E+00
1.00E+01	0.000E+00	2.203E+05	0.000E+00	0.000E+00
Total	1.345E+12	1.788E+15	4.455E+11	5.087E+11
Neutron Source				
Raw source: 3.795E+08 n/s Including peaking and subcritical neutron multiplication: 6.204E+08 n/s				

TABLE 2.5-12: BWR NCT SOURCE TERM PER FUEL ASSEMBLY, ZONE 4

Burnup: 29 GWd/MTU Enrichment: 1.5% Cooling Time: 5 years Decay Heat: 0.33 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	4.740E+10	4.601E+14	6.329E+10	2.056E+10
1.00E-01	7.479E+09	1.334E+14	2.654E+09	2.779E+09
2.00E-01	2.685E+09	1.106E+14	5.036E+09	1.247E+09
3.00E-01	1.504E+08	3.140E+13	3.451E+08	7.388E+07
4.00E-01	3.197E+08	2.276E+13	1.089E+09	1.788E+08
6.00E-01	4.574E+09	2.379E+14	2.359E+10	3.048E+09
8.00E-01	2.385E+09	7.202E+14	1.231E+10	1.590E+09
1.00E+00	3.788E+10	9.335E+13	1.143E+10	1.170E+10
1.33E+00	2.151E+12	3.489E+13	6.866E+11	7.953E+11
1.66E+00	6.074E+11	1.065E+13	1.939E+11	2.246E+11
2.00E+00	1.198E+02	5.443E+11	5.145E+01	8.758E+01
2.50E+00	1.453E+07	1.016E+12	4.639E+06	5.374E+06
3.00E+00	1.242E+04	4.114E+10	3.964E+03	4.592E+03
4.00E+00	2.730E-12	3.819E+09	8.310E-13	9.505E-13
5.00E+00	0.000E+00	3.590E+06	0.000E+00	0.000E+00
6.50E+00	0.000E+00	1.441E+06	0.000E+00	0.000E+00
8.00E+00	0.000E+00	2.827E+05	0.000E+00	0.000E+00
1.00E+01	0.000E+00	6.001E+04	0.000E+00	0.000E+00
Total	2.861E+12	1.857E+15	1.000E+12	1.061E+12
Neutron Source				
Raw source: 1.028E+08 n/s Including peaking and subcritical neutron multiplication: 1.680E+08 n/s				

TABLE 2.5-13: BWR HAC SOURCE TERM PER FUEL ASSEMBLY, ZONE 4

Burnup: 62.5 GWd/MTU Enrichment: 3.8% Cooling Time: 25 years Decay Heat: 0.33 kW				
Gamma Source				
Emax (MeV)	Bottom Nozzle (γ/s)	Active Fuel (γ/s)	Plenum (γ/s)	Top Nozzle (γ/s)
5.00E-02	3.803E+09	3.429E+14	1.583E+09	1.568E+09
1.00E-01	6.817E+08	9.649E+13	2.194E+08	2.575E+08
2.00E-01	1.729E+08	6.283E+13	9.457E+07	6.766E+07
3.00E-01	9.040E+06	1.931E+13	5.687E+06	3.687E+06
4.00E-01	1.297E+07	1.297E+13	1.342E+07	5.575E+06
6.00E-01	4.459E+07	1.031E+13	2.236E+08	2.958E+07
8.00E-01	2.339E+07	6.595E+14	1.167E+08	1.557E+07
1.00E+00	9.108E+06	5.582E+12	2.888E+06	3.526E+06
1.33E+00	1.986E+11	8.502E+12	6.319E+10	7.494E+10
1.66E+00	5.609E+10	9.337E+11	1.785E+10	2.116E+10
2.00E+00	6.101E+00	3.324E+10	3.158E+01	4.069E+00
2.50E+00	1.342E+06	1.728E+09	4.270E+05	5.064E+05
3.00E+00	1.147E+03	1.436E+08	3.648E+02	4.327E+02
4.00E+00	1.371E-11	2.445E+07	4.173E-12	4.773E-12
5.00E+00	0.000E+00	8.251E+06	0.000E+00	0.000E+00
6.50E+00	0.000E+00	3.312E+06	0.000E+00	0.000E+00
8.00E+00	0.000E+00	6.496E+05	0.000E+00	0.000E+00
1.00E+01	0.000E+00	1.379E+05	0.000E+00	0.000E+00
Total	2.595E+11	1.219E+15	8.330E+10	9.806E+10
Neutron Source				
Raw source: 2.382E+08 n/s Including peaking and subcritical neutron multiplication: 3.894E+08 n/s				

TABLE 2.5-14: PWR AXIAL SOURCE DISTRIBUTIONS

Node	Gamma Axial Source Distribution	Neutron Axial Source Distribution
1 (bottom)	0.573	0.108
2	0.917	0.707
3	1.066	1.291
4	1.106	1.496
5	1.114	1.540
6	1.111	1.524
7	1.106	1.496
8	1.101	1.469
9	1.097	1.448
10	1.093	1.427
11	1.089	1.406
12	1.086	1.391
13	1.081	1.366
14	1.073	1.326
15	1.051	1.220
16	0.993	0.972
17	0.832	0.479
18 (top)	0.512	0.069
Average	1.000	1.152

TABLE 2.5-15: BWR AXIAL SOURCE DISTRIBUTIONS

Node	Gamma Axial Source Distribution	Neutron Axial Source Distribution
1 (bottom)	0.714	0.260
2	1.099	1.460
3	1.219	2.204
4	1.231	2.298
5	1.220	2.215
6	1.180	1.938
7	1.153	1.766
8	1.131	1.634
9	1.114	1.540
10	1.119	1.569
11	1.107	1.502
12	1.081	1.364
13	1.072	1.322
14	1.058	1.253
15	1.031	1.131
16	1.018	1.074
17	1.018	1.074
18	1.021	1.088
19	1.014	1.057
20	0.979	0.918
21	0.917	0.707
22	0.843	0.505
23	0.719	0.267
24	0.554	0.094
25 (top)	0.203	0.002
Average	1.000	1.210

2.5.3 Model Specification

A detailed three-dimensional shielding model of the package and its contents is developed using the MCNP5 v1.51 computer program. All relevant features are modeled. The PWR MCNP model geometry for NCT is shown in **Figures 2.5-1** through **2.5-4**. The BWR MCNP model geometry for NCT is shown in **Figures 2.5-5** through **2.5-6**. Features not modeled that displace neutron shielding include the trunnions, trunnion attachment blocks, and shear key. The trunnions are removed prior to transport and replaced with plugs of neutron shielding material. It is assumed that the trunnion shield plugs could be designed to provide equivalent neutron shielding compared to the primary neutron shield. Twelve trunnion attachment blocks are used on each end to bolt the impact limiters to the cask. The attachment blocks would not result in significant neutron streaming.

The shear key secures the cask to the transport vehicle and is also a neutron-streaming path. When the cask is not in the transportation configuration the shear key is filled with neutron shielding material to mitigate the neutron streaming. However, in the transport configuration, the shear key mates with the transport skid. In the transport configuration the shear key is in a downward orientation so that the neutron radiation is directed under the vehicle. The dose rate on the underside of the vehicle is limited to 200 mrem/hr. High dose rates in this region could be mitigated by modifications to the shear key design or eliminating the shear key and restraining the package using other means.

The inner diameter of the cask is 66.25 inches. The inner shell of the cask is 1.25-inch thick carbon steel, a lead core is 3 inches thick, and an outer carbon steel shell is 2.75 inches thick. A neutron shield is attached to the side of the cask. The neutron shield is comprised of 60 copper 'boxes' filled with VYAL-B neutron shielding resin. The copper thickness is 0.125 inches and the VYAL-B resin thickness is 6.00 inches. A stainless steel shell 0.25 inches thick covers the neutron shield and is modeled as carbon steel (modeling stainless steel as carbon steel for this thin shell has a negligible effect on the results).

The neutron shield features 6.00-inch thick VYAL-B resin between the impact limiters. The neutron shield extends approximately 11.00 inches under the impact limiters, although in this region it is only 5.00 inches thick.

The bottom of the package features 1.25-inch thick carbon steel (inner), 4.50-inch thick lead, and 2.75-inch thick carbon steel (outer). The cask has a double-lid design. The inner lid features 1.00-inch thick carbon steel (inner), 2.50-inch thick lead, and 3.00-inch carbon steel (outer). The outer lid features 2.50 inches of carbon steel.

The impact limiters are modeled explicitly with an outer carbon steel shell 0.25 inches thick (the actual impact limiter shells are stainless steel, although modeling stainless steel as carbon steel for thin shells has a negligible effect on the results). The impact limiters are conservatively filled with low-density balsa wood, although the actual impact limiters are filled with a mix of balsa wood and higher density redwood. The impact limiters have an outer diameter of 126 inches and an overall height of ~58 inches. The impact limiter end surfaces and impact limiter radius form the boundary of the package, as the personnel barrier is located at the approximate radius of the impact limiters.

Each PWR fuel assembly is modeled as four axial regions: bottom nozzle (8.375-inch length), active fuel (142.29-inch length), plenum (8.73-inch length), and top nozzle (6.23-inch length). For simplicity, the mass of fuel assembly material is homogenized within each region. The PWR fuel is modeled with a cross section of 8.9 inches x 8.9 inches.

Each BWR fuel assembly is modeled as four axial regions: bottom nozzle (6.65-inch length), active fuel (144-inch length), plenum (12.93-inch length), and top nozzle (12.62-inch length). For simplicity, the mass of fuel assembly material is homogenized within each region. The BWR fuel is modeled with a cross section of 5.52 inches x 5.52 inches.

DFCs, if present, are not modeled because DFCs would provide additional shielding.

The design basis PWR and BWR fuel assemblies have total as-modeled lengths of 165.625 inches and 176.2 inches, respectively, while the cask cavity length is 182.0 inches. Therefore, the extra space will be filled with a spacer. Rather than modeling this extra space, the cask is axially shortened so that the fuel just fits inside the cask cavity. This places the source at the closest location to each end and maximizes the dose rates on the cask centerline through the impact limiters.

In the PWR basket model, each basket location is modeled with a 0.25-inch thick box of stainless steel that forms the main structure of the basket. Between the basket compartments is a borated aluminum neutron absorbing sheet 0.12 inches thick and an aluminum plate 0.75 inches thick. Solid aluminum transition rails form the interface between the basket and the cask inner diameter.

In the BWR basket model, each basket location is modeled with a 0.17-inch thick stainless steel box, which forms the main structure of the basket. This thickness is an average value that preserves the mass of the basket structural material, which is not constant for each basket location. This approximation is sufficient for shielding purposes. Between each box is 0.3-inch thick borated aluminum poison.

For HAC, the neutron shield and impact limiters are modeled as void—see **Figure 2.5-7** for PWR (BWR is similar). The fuel also may become damaged in the accident and is assumed to form rubble that collects at the bottom of the cask. Cases are examined in which the active fuel region is axially compressed 25 percent and 50 percent—see **Figure 2.5-8** for PWR (BWR is similar). The density of the active fuel is allowed to increase to conserve mass. In the compressed fuel cases the axial source distribution is modeled as flat because for such a large degree of damage mixing of various fuel segments would occur.

FIGURE 2.5-1: PWR MCNP NCT RADIAL GEOMETRY

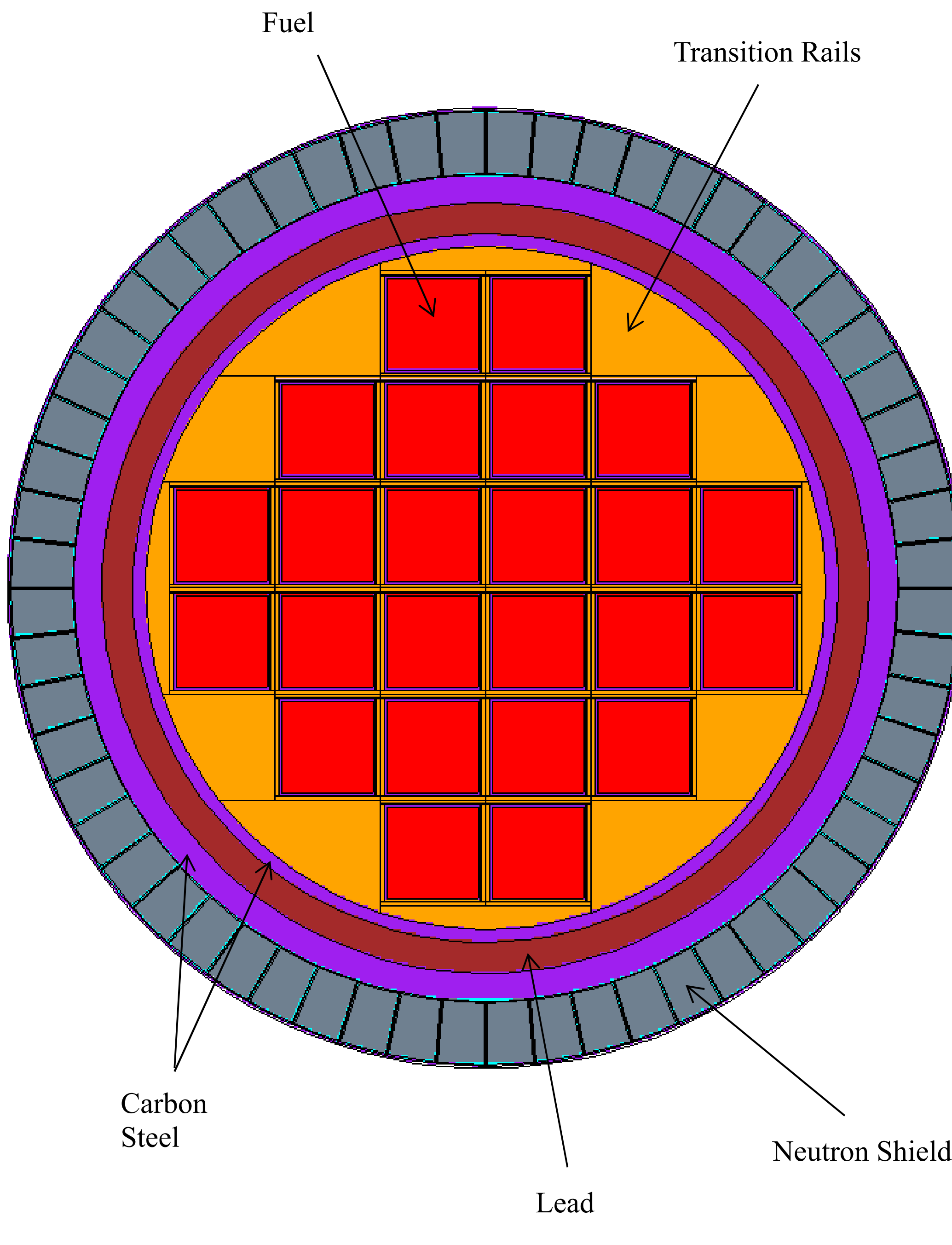


FIGURE 2.5-2: PWR MCNP NCT GEOMETRY, BASKET CLOSE-UP

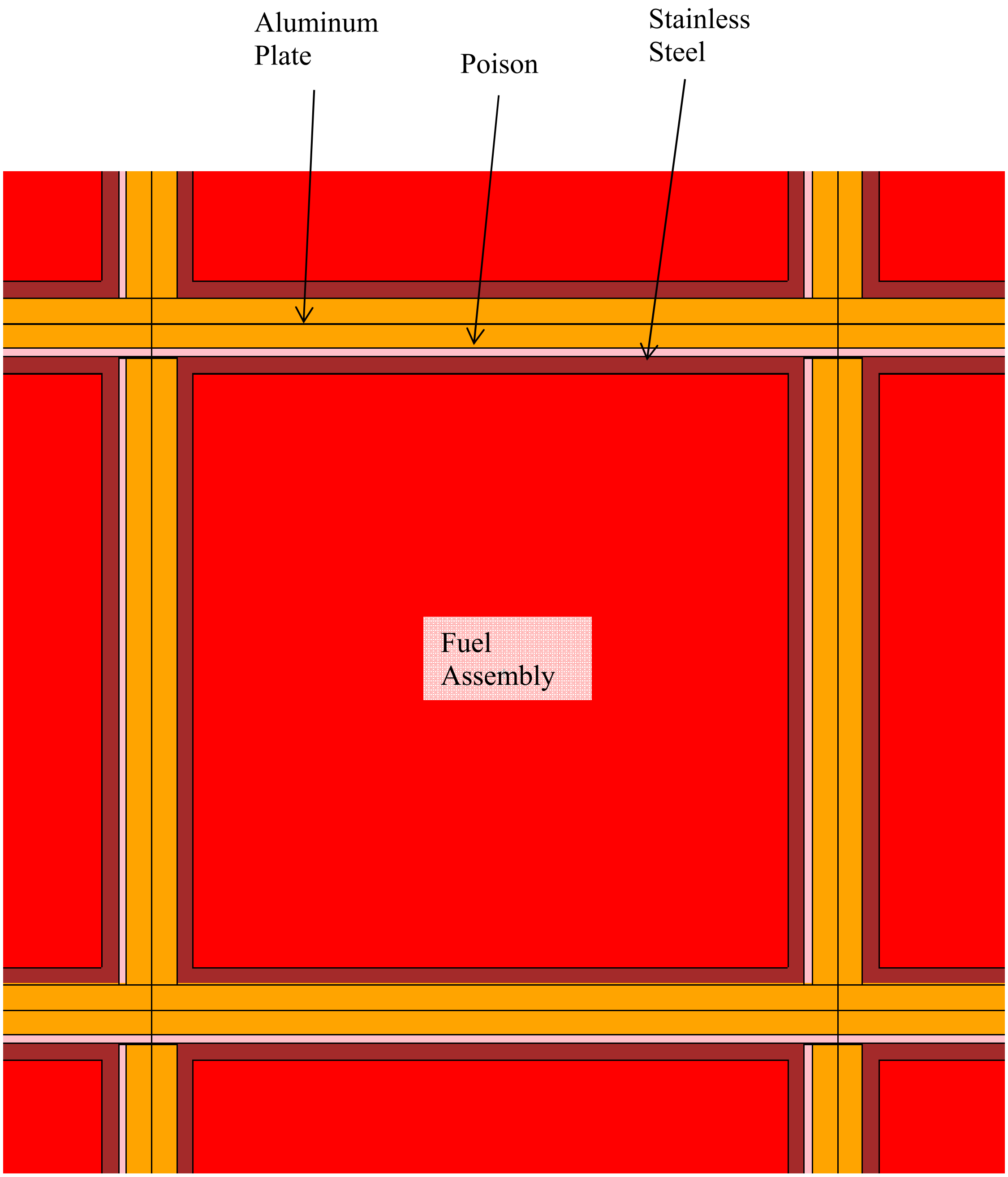


FIGURE 2.5-3: MCNP NCT GEOMETRY, NEUTRON SHIELD CLOSE-UP

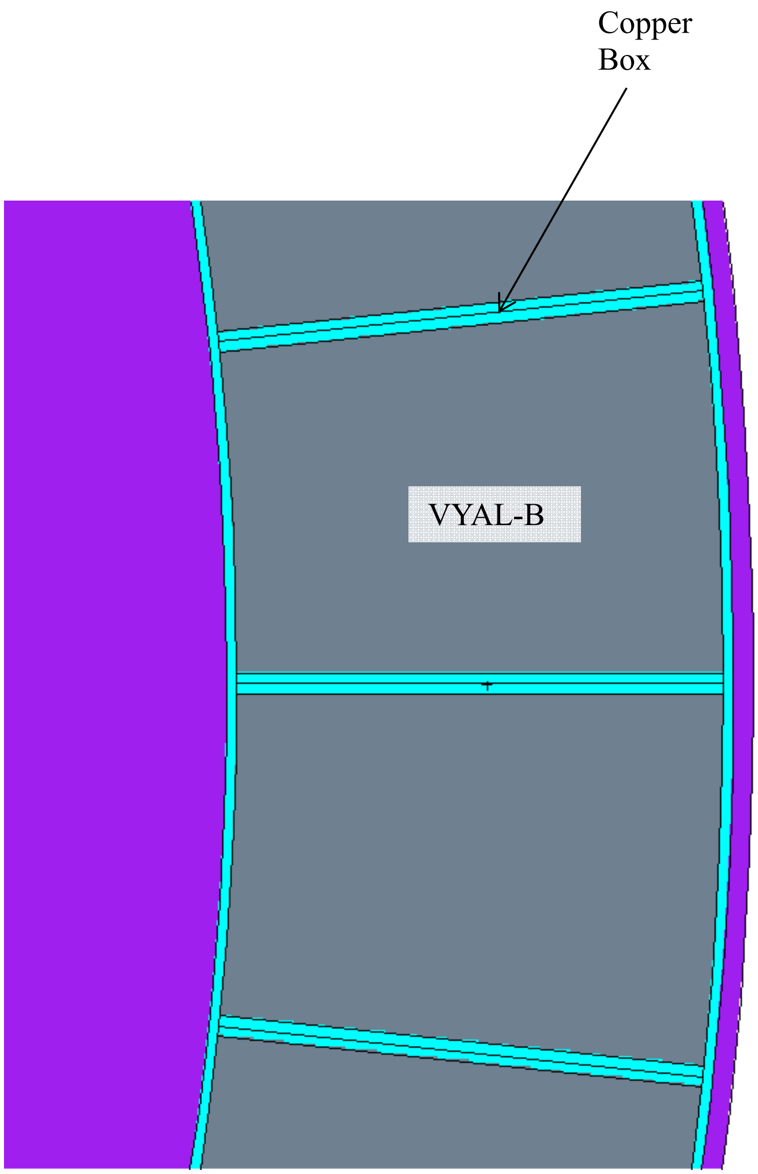


FIGURE 2.5-4: PWR MCNP NCT GEOMETRY, AXIAL VIEW

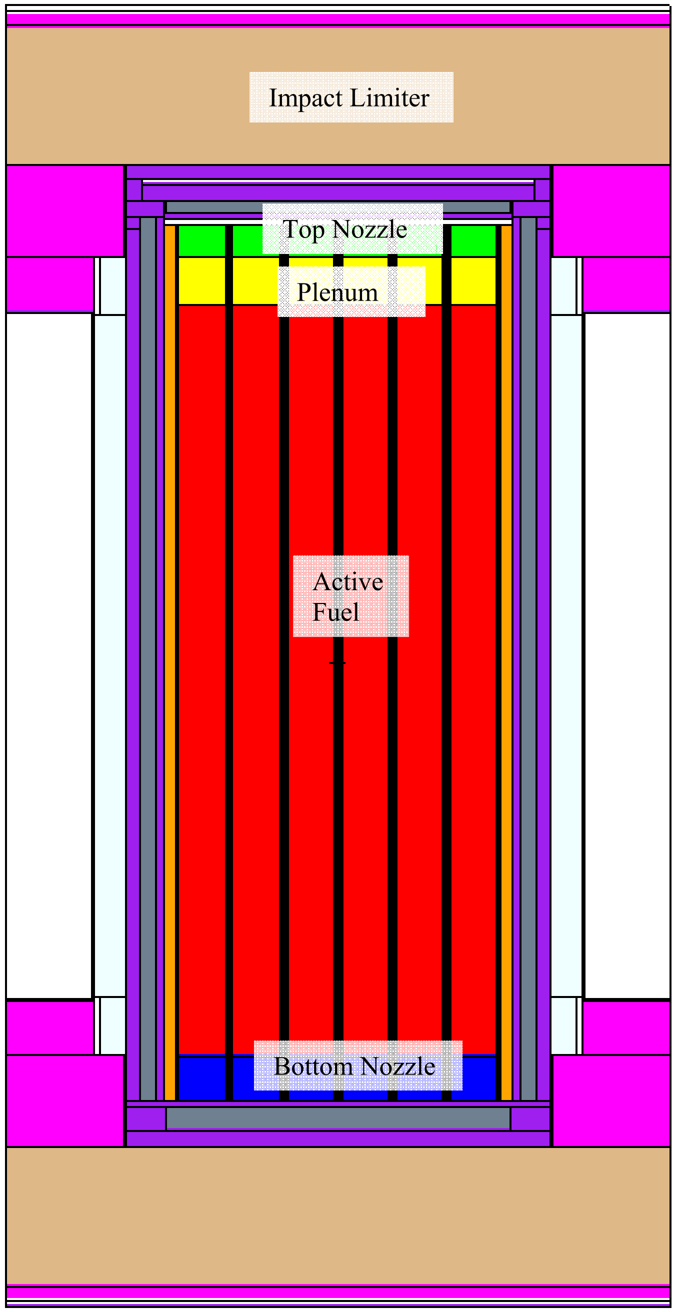


FIGURE 2.5-5: BWR MCNP RADIAL GEOMETRY

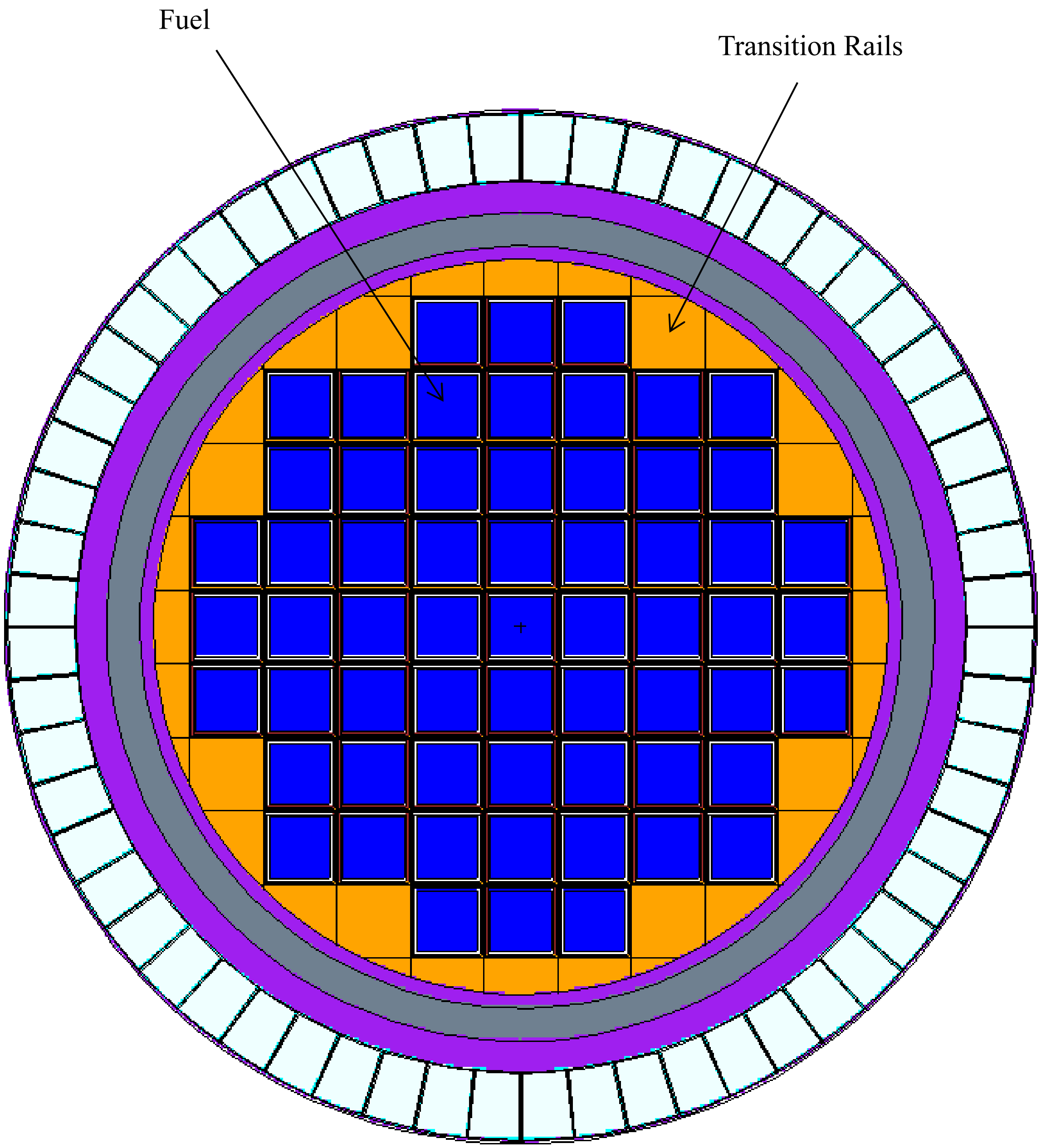


FIGURE 2.5-6: BWR MCNP NCT GEOMETRY, BASKET CLOSE-UP

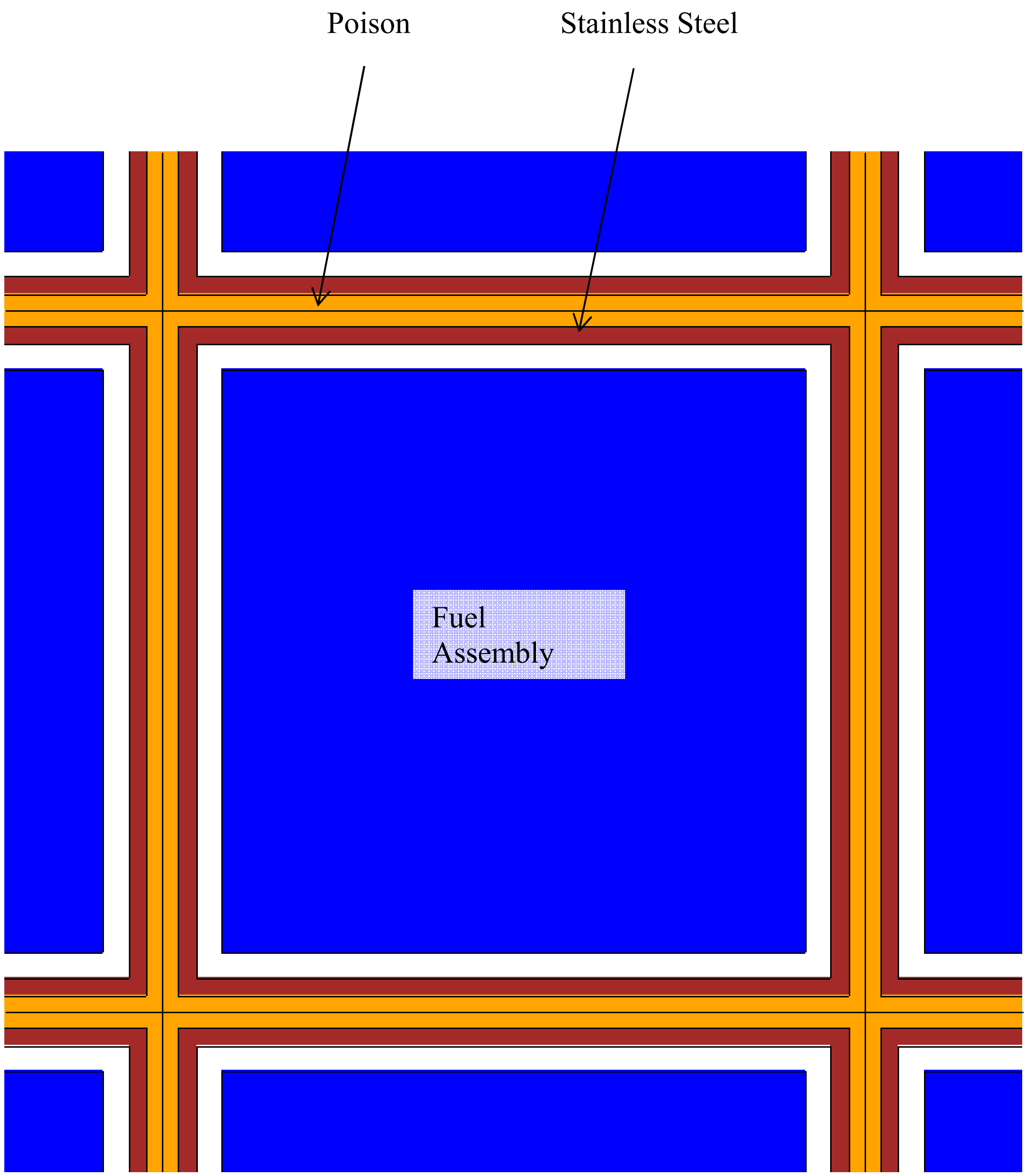


FIGURE 2.5-7: PWR MCNP HAC GEOMETRY, AXIAL VIEW

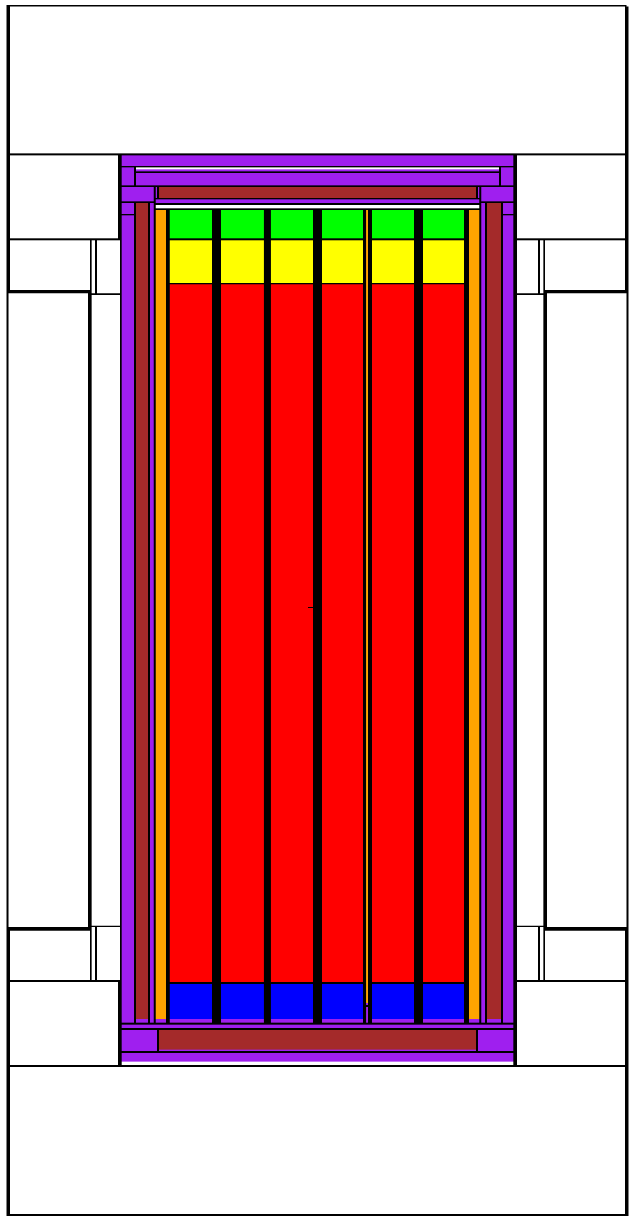
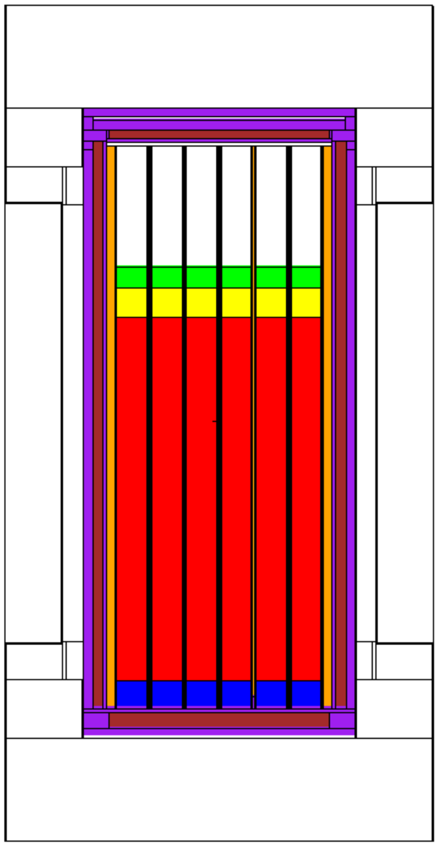
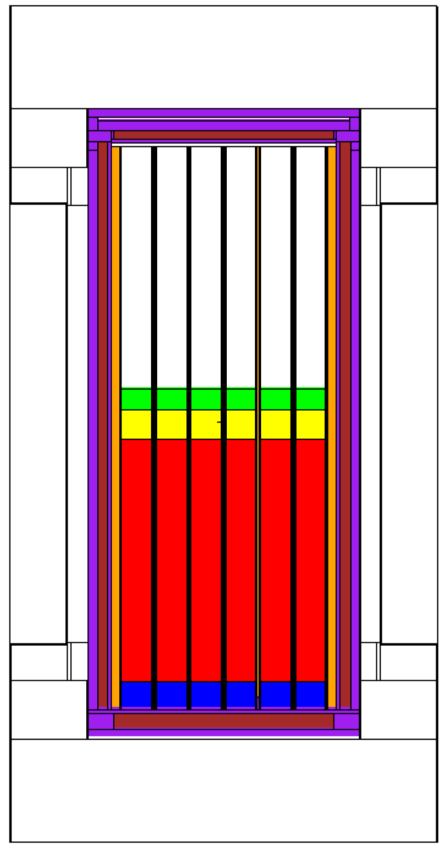


FIGURE 2.5-8: PWR MCNP HAC GEOMETRY, COMPRESSED CASES, AXIAL VIEW



25% Compression



50% Compression

2.5.4 Material Specification

Various materials are used in the MCNP models. Lead is modeled as pure lead with a reduced density of 11.18 g/cm^3. The aluminum plates of the PWR basket are modeled as pure aluminum with a density of 2.7 g/cm^3. The aluminum transition rails are modeled as pure aluminum with a reduced density of 2.43 g/cm^3 (90 percent theoretical density) to account for holes and other gaps in the transition rails. Copper in the resin boxes is modeled as pure copper with a density of 8.96 g/cm^3.

Stainless steel is modeled with a density of 8.0 g/cm^3 with the composition provided in **Table 2.5-16** [26]. Carbon steel is modeled with a density of 7.8212 g/cm^3 with the composition provided in **Table 2.5-17** [21]. Balsa wood and redwood are modeled with the chemical formula $C_6H_{10}O_5$ [21] with a density of 0.112 g/cm^3 for balsa 0.299 g/cm^3 for redwood.

The borated aluminum neutron absorber is modeled as a mixture of B-10 and aluminum with B-10 areal densities of 20 mg/cm^2 and 50 mg/cm^2 in the PWR and BWR models, respectively. This is less than the actual areal densities of 40.6 mg/cm^2 and 85.3 mg/cm^2, respectively. The composition is provided in **Table 2.5-18**; however, the boron is not a strong neutron absorber in the models because the spectrum is fast. Note that the BWR poison plate thickness is 2.5 times thicker than the PWR poison plate thickness, resulting in a larger B-10 areal density for the BWR basket compared to the PWR basket, although the as-modeled poison composition is the same for both.

VYAL-B is used as the neutron shielding material. VYAL-B has a density of 1.75 g/cm^3. The composition of VYAL-B is proprietary.

The fuel assemblies are homogenized for simplicity. The PWR and BWR homogenized fuel assembly representations are provided in **Table 2.5-19** and **2.5-20**, respectively. The uranium is modeled with an enrichment of 5 percent in the PWR models and 4 percent in the BWR models, although the enrichment selected has little effect on the results because fission is suppressed using the NONU card.

TABLE 2.5-16: STAINLESS STEEL 304 COMPOSITION

MCNP Material ID	Weight Fraction
6000	4.0000E-04
14000	5.0000E-03
15031	2.3000E-04
16000	1.5000E-04
24000	1.9000E-01
25055	1.0000E-02
26000	7.0173E-01
28000	9.2500E-02
Density = 8.0 g/cm^3	

TABLE 2.5-17: CARBON STEEL COMPOSITION

MCNP Material ID	Weight Fraction
6000	0.01
26000	0.99
Density = 7.8212 g/cm^3	

TABLE 2.5-18: BORATED ALUMINUM NEUTRON ABSORBER COMPOSITION

MCNP Material ID	Weight Fraction
5010	2.4304E-02
13027	9.7570E-01
Density = 2.7 g/cm^3	

TABLE 2.5-19: PWR HOMOGENIZED FUEL COMPOSITIONS

MCNP Material ID	Bottom Nozzle (atom/b-cm)	Active Fuel (atom/b-cm)	Plenum (atom/b-cm)	Top Nozzle (atom/b-cm)
8016	-	1.348E-02	-	-
13027	1.314E-05	3.614E-06	6.394E-05	2.981E-05
22000	9.875E-06	2.716E-06	4.804E-05	2.240E-05
24000	1.881E-03	6.617E-05	1.058E-03	2.988E-03
25055	1.648E-04	-	-	2.489E-04
26000	5.964E-03	8.451E-05	1.285E-03	9.174E-03
28000	1.209E-03	1.437E-04	2.543E-03	2.222E-03
40000	6.228E-03	3.790E-03	3.890E-03	-
42000	1.848E-05	5.082E-06	8.991E-05	4.191E-05
50000	7.806E-05	4.750E-05	4.875E-05	-
92235	-	3.393E-04	-	-
92238	-	6.373E-03	-	-
Total	1.557E-02	2.434E-02	9.025E-03	1.473E-02

TABLE 2.5-20: BWR HOMOGENIZED FUEL COMPOSITIONS

MCNP Material ID	Bottom Nozzle (wt. fraction)	Active Fuel (wt. fraction)	Plenum (wt. fraction)	Top Nozzle (wt. fraction)
6012	6.300E-04	2.000E-06	1.400E-04	4.500E-04
8016	-	9.613E-02	-	-
14000	8.030E-03	6.600E-05	1.770E-03	8.390E-03
15031	3.500E-04	1.000E-06	8.000E-05	2.500E-04
22000	2.100E-04	4.400E-05	-	2.810E-03
24000	1.500E-01	8.560E-04	3.441E-02	1.232E-01
25055	1.564E-02	4.300E-05	3.540E-03	1.115E-02
26000	5.357E-01	1.971E-03	1.226E-01	3.898E-01
28000	8.037E-02	1.496E-03	1.679E-02	1.351E-01
40000	2.059E-01	1.817E-01	8.087E-01	3.240E-01
50000	3.040E-03	2.682E-03	1.194E-02	4.780E-03
72000	2.000E-05	1.800E-05	8.000E-05	3.000E-05
92234	-	2.550E-04	-	-
92235	-	2.860E-02	-	-
92236	-	1.320E-04	-	-
92238	-	6.860E-01	-	-
Total	1.0	1.0	1.0	1.0
Density (g/cm^3)	1.864	3.960	0.947	0.624

2.5.5 Evaluation

2.5.5.1 Flux to Dose Rate Conversion

MCNP is used to compute the flux at locations of interest, and the flux is converted to dose rates using ANSI/ANS-6.1.1-1977 flux to dose rate conversion factors [25].

2.5.5.2 Dose Rate Tallies

The following dose rates are needed:

- The dose rate on the accessible surfaces of the package is limited to 200 mrem/hr. The accessible surfaces of the package are defined as the flat outer surfaces of the impact limiters and the cylindrical surface at the impact limiter radius. A personnel barrier is located at the approximate radius of the impact limiters and prevents access to the cask surface. The dose rate on the cask surface is limited to 1,000 mrem/hr and the cask surface dose rates are significantly less.

- The dose rate on the surface of the transport vehicle (rail car) is limited to 200 mrem/hr. The transport vehicle surface dose rates are bounded by the dose rates computed on the surfaces of the package. The vehicle surface would have six faces. The end faces correspond to the flat outer surfaces of the impact limiters, while the top, underside, and side surfaces of the vehicle are bounded by the package side dose rates.
- The dose rates 2 m from the vertical surfaces of the vehicle are limited to 10 mrem/hr. The vehicle is assumed to be the same length as the package, which conservatively reduces the distance to the end tally locations. Credit could be taken for this distance if a minimum rail car length were defined. The package is approximately 6.2 m long and a 40-foot long rail car is approximately 12.2 m long. Therefore, the true distance to the end tally may be (12.2-6.2)/2+2 = 5 m if the package is axially centered on the rail car.
- The dose rate in any occupied location is limited to 2 mrem/hr. The dose rate in the occupied location is not specifically computed because the distance from the package to the occupied location is not defined. However, because the dose rate is less than 10 mrem/hr 2 m from the end of the impact limiter and the distance to the occupied location is likely large, it may be inferred that the dose rate in any occupied location would be below the 2 mrem/hr limit. Also, this dose rate limit may be waived if dosimetry is worn during transportation.

Dose rates on the side of the package are computed using radial mesh tallies. Side dose rates are tallied at the impact limiter radius and 2 m from the impact limiter radius over 25 axial segments. To accelerate model convergence, these tallies are circumferential averages around the package. However, the dose rate around the circumference of the package will not be uniform because the aluminum transition rails vary in thickness and the source varies in magnitude by zone. A separate radial mesh tally is used with one bin along the length of the fuel assembly but with 24 angular bins. The angular tally is used only to compute an angular peaking factor, and this peaking factor is applied to the circumferential average tally to compute a maximum radial dose rate. The dose rate is computed by combining two tallies because a single mesh tally with 25 axial bins and 24 angular bins would likely have large Monte Carlo fluctuations and require long run times. The ‘two-step’ tally approach converges quickly.

Dose rates are calculated at the end impact limiter surfaces and 2 m from these surfaces using point detectors on the cask centerline.

In the HAC models, the dose rate is calculated 1 m from the surface of the cask because the neutron shield and impact limiters have been removed. The side dose rates are computed using the same two-step method used to compute the side NCT dose rates, and end dose rates are computed using point detectors on the cask centerline.

2.5.5.3 Radiation Levels

NCT dose rate results for PWR and BWR fuel are summarized in **Table 2.5-21**. The maximum package surface dose rate is 63.2 mrem/hr and occurs on the side at the impact limiter radius. This dose rate is lower than the limit of 200 mrem/hr. The dose rate limit on the vehicle surfaces is also met because the package surface is assumed to be the same as the vehicle surface. The maximum dose rate 2 m from the vertical surfaces of the vehicle is 9.3 mrem/hr and occurs at the side of the vehicle. This dose rate is below the limit of 10 mrem/hr.

The maximum dose rate on the package side at the impact limiter radius occurs next to the bottom nozzle for both the PWR and BWR baskets. At this location the neutron shield is absent (see **Figure 2.5-4**), leading to a large neutron component of the total dose rate, as well as Co-60 from the bottom nozzle. The maximum side dose rate for the BWR basket is almost twice as large as the PWR basket (63.2 mrem/hr vs. 33.8 mrem/hr) because the neutron source peaks closer to the bottom for BWR fuel. However, these peak dose rates are highly localized. The average dose rate along the side of the package at the impact limiter radius is very similar for the PWR and BWR baskets, and the peak dose rates 2 m from the side of the vehicle is almost identical for both baskets.

HAC dose rate results for PWR and BWR fuel are summarized in **Table 2.5-22**. The limiting dose rate occurs when the fuel is undamaged. The maximum dose rate 1 m from the package surface is 921 mrem/hr and occurs at the side of the package. Note that the dose rate is almost entirely due to neutron radiation. As the fuel is axially compressed towards the bottom end of the cask, the side dose rate decreases (due to increased self-shielding of the fuel) while the dose rate at the bottom of the cask increases (because the source is closer to the tally location). However, fuel reconfiguration causes a decrease in the maximum dose rate.

For simplicity, subcritical neutron multiplication is assumed to be the same for the HAC compressed fuel cases, although the system k_{eff} may increase by ~20 percent as the fuel compresses. A ~20 percent increase in k_{eff} would cause an increase in the neutron dose rate by ~8% percent ([1-0.26]/[1-0.26*1.2]) in the 50 percent compression case and ~4 percent in the 25 percent compression case. However, the intact fuel case remains bounding even if these effects are considered.

TABLE 2.5-21: NCT MAXIMUM DOSE RATE RESULTS

PWR								
Location	Gamma (mrem/ hr)	Neutron (mrem/ hr)	(n,γ) (mrem/ hr)	Total (mrem/ hr)	Angular Peaking	Max. (mrem/ hr)	Un.	Limit (mrem/ hr)
Impact Limiter radius	12.0	9.8	3.5	25.3	1.3	33.8	1%	200
Impact Limiter bottom end	4.8	3.9	7.2	15.9	1.0	15.9	3%	200
Impact Limiter top end	18.6	1.4	3.3	23.3	1.0	23.3	5%	200
2 m side	4.5	1.9	1.3	7.7	1.2	9.3	1%	10
2 m bottom end	1.4	0.5	0.9	2.7	1.0	2.7	2%	10
2 m top end	4.4	0.2	0.4	4.9	1.0	4.9	6%	10
BWR								
Location	Gamma (mrem/ hr)	Neutron (mrem/ hr)	(n,γ) (mrem/ hr)	Total (mrem/ hr)	Angular Peaking	Max. (mrem/ hr)	Un.	Limit (mrem/ hr)
Impact Limiter radius	16.0	31.0	7.2	54.2	1.2	63.2	2%	200
Impact Limiter bottom end	7.4	11.0	18.1	36.5	1.0	36.5	4%	200
Impact Limiter top end	10.1	0.4	1.0	11.6	1.0	11.6	12%	200
2 m side	3.2	3.7	1.5	8.4	1.1	9.3	2%	10
2 m bottom end	1.9	1.4	2.0	5.4	1.0	5.4	3%	10
2 m top end	2.1	0.1	0.1	2.2	1.0	2.2	2%	10

TABLE 2.5-22: HAC MAXIMUM DOSE RATE RESULTS

PWR								
Location	**Gamma (mrem/ hr)**	**Neutron (mrem/ hr)**	**(n,γ) (mrem/ hr)**	**Total (mrem/ hr)**	**Angular Peaking**	**Max. (mrem/ hr)**	**Un.**	**Limit (mrem/ hr)**
Intact Fuel								
1 m side	22.1	793.8	2.2	818.0	1.1	875.7	1%	1000
1 m bottom end	13.4	132.7	0.6	146.7	1.0	146.7	1%	1000
1 m top end	52.8	52.7	0.4	105.8	1.0	105.8	1%	1000
25% Axial Compression of the Fuel								
1 m side	20.1	662.3	1.8	684.1	1.1	734.6	1%	1000
1 m bottom end	13.5	503.9	1.3	518.7	1.0	518.7	1%	1000
50% Axial Compression of the Fuel								
1 m side	19.3	600.1	1.6	621.0	1.1	656.8	1%	1000
1 m bottom end	13.6	571.6	1.5	586.7	1.0	586.7	1%	1000
BWR								
Location	**Gamma (mrem/ hr)**	**Neutron (mrem/ hr)**	**(n,γ) (mrem/ hr)**	**Total (mrem/ hr)**	**Angular Peaking**	**Max. (mrem/ hr)**	**Un.**	**Limit (mrem/ hr)**
Intact Fuel								
1 m side	13.6	876.5	1.5	891.6	1.03	920.9	1%	1000
1 m bottom end	21.9	426.6	0.9	449.3	1.00	449.3	1%	1000
1 m top end	25.3	17.0	0.1	42.4	1.00	42.4	2%	1000
25% Axial Compression of the Fuel								
1 m side	12.3	730.7	1.5	744.5	1.05	781.2	1%	1000
1 m bottom end	20.2	617.2	0.9	638.4	1.00	638.4	1%	1000
50% Axial Compression of the Fuel								
1 m side	12.3	654.7	1.4	668.4	1.05	703.2	1%	1000
1 m bottom end	21.1	694.0	1.1	716.3	1.00	716.3	1%	1000

2.5.6 Alternate Heat Load Zoning Configurations

If sufficient colder fuel is present in an SFP, the heat load zone configurations shown in **Figures 2.1-6** and **2.1-8** are adequate. However, if all available fuel assemblies are thermally hot, the number of fuel assemblies that may be shipped could be severely limited. Therefore, additional heat load zoning configurations are developed.

For PWR fuel, a uniform heat load is explored, and the lowest uniform heat load that meets the dose rate requirements is 1.1 kW per basket location. This configuration has little practical value, as most of the higher burnup fuels have heat loads that exceed 1.1 kW. Also, 1.1 kW is only slightly hotter than the 0.9 kW used in Zones 1 and 4 in the baseline heat load zone configuration. Therefore, the uniform heat load is not considered further.

Likewise, a uniform heat load of 0.5 kW is examined for BWR fuel. The NCT dose rate limit is not met for this heat load, and the decay time for the 62.5 GWd/MTU fuel is 10 years for this decay heat. Therefore, a uniform heat load for BWR fuel would have a heat load per FA < 0.5 kW and cooling times > 10 years for the highest burned fuel. As with the PWR fuel, a uniform heat load for BWR fuel is of little practical value and is not considered further.

To ship a large number of thermally hot fuel assemblies, short-loading the package will likely be required. If the PWR basket is limited to 16 fuel assemblies in the inner locations, the heat load is 1.9 kW per fuel assembly to maintain the same overall heat load. This configuration is shown in **Figure 2.5-9**. The outer basket locations are modeled as empty (no dummy assemblies). A separate FQT is developed for this configuration, as shown in **Table 2.5-23**.

In a similar fashion, a short-loaded BWR configuration is shown in **Figure 2.5-10**, and the corresponding FQT is shown in **Table 2.5-24**. In the alternate BWR configuration, 45 fuel assemblies are allowed. The outer basket locations are modeled as empty (no dummy assemblies).

The dose rate results for these configurations are summarized in **Table 2.5-25** and **Table 2.5-26** for NCT and HAC, respectively. Based on these simple FQTs, all NCT and HAC dose rate limits are met.

Note that the outer basket assemblies are modeled as empty, which is conservative for shielding. If dummy assemblies are required in these locations to facilitate heat removal, the dose rates will decrease due to the increased shielding.

FIGURE 2.5-9: ALTERNATE PWR HEAT LOAD ZONE CONFIGURATION (KW)

		Empty	Empty		
	1.9	1.9	1.9	1.9	
Empty	1.9	1.9	1.9	1.9	Empty
Empty	1.9	1.9	1.9	1.9	Empty
	1.9	1.9	1.9	1.9	
		Empty	Empty		

Note that the outer basket assemblies are modeled as empty, which is conservative for shielding. If dummy assemblies are required in these locations to facilitate heat removal, the dose rates will decrease due to the increased shielding.

FIGURE 2.5-10: ALTERNATE BWR HEAT LOAD ZONE CONFIGURATION (KW)

			Empty	Empty	Empty			
	Empty	0.6	0.6	0.6	0.6	0.6	Empty	
	0.6	0.6	0.6	0.6	0.6	0.6	0.6	
Empty	0.6	0.6	0.6	0.6	0.6	0.6	0.6	Empty
Empty	0.6	0.6	0.6	0.6	0.6	0.6	0.6	Empty
Empty	0.6	0.6	0.6	0.6	0.6	0.6	0.6	Empty
	0.6	0.6	0.6	0.6	0.6	0.6	0.6	
	Empty	0.6	0.6	0.6	0.6	0.6	Empty	
			Empty	Empty	Empty			

Note that the outer basket assemblies are modeled as empty, which is conservative for shielding. If dummy assemblies are required in these locations to facilitate heat removal, the dose rates will decrease due to the increased shielding.

TABLE 2.5-23: ALTERNATE PWR FQT

Maximum Burnup (GWd/MTU)	Minimum Enrichment (%)	Minimum Cooling Time (years) (Heat ≤ 1.9 kW)
≤ 58	≥ 3.6	≥ 5
≤ 62.5	≥ 3.8	≥ 5.6

TABLE 2.5-24: ALTERNATE BWR FQT

Maximum Burnup (GWd/MTU)	Minimum Enrichment (%)	Minimum Cooling Time (years) (Heat ≤ 0.6 kW)
≤ 50	≥ 3.1	≥ 5
≤ 56	≥ 3.5	≥ 5.8
≤ 62.5	≥ 3.8	≥ 7

TABLE 2.5-25: NCT MAXIMUM DOSE RATE RESULTS, ALTERNATE CONFIGURATIONS

PWR								
Location	Gamma (mrem/ hr)	Neutron (mrem/ hr)	(n,γ) (mrem/ hr)	Total (mrem/ hr)	Angular Peaking	Max. (mrem/ hr)	Un.	Limit (mrem/ hr)
Impact Limiter radius	6.3	17.9	6.1	30.3	1.2	35.5	1%	200
Impact Limiter bottom end	5.5	6.0	11.4	22.9	1.0	22.9	4%	200
Impact Limiter top end	19.4	2.1	5.5	27.0	1.0	27.0	6%	200
2 m side	2.7	2.8	2.0	7.5	1.1	8.2	1%	10
2 m bottom end	1.1	0.7	1.5	3.4	1.0	3.4	3%	10
2 m top end	3.8	0.3	0.7	4.8	1.0	4.8	4%	10
BWR								
Location	Gamma (mrem/ hr)	Neutron (mrem/ hr)	(n,γ) (mrem/ hr)	Total (mrem/ hr)	Angular Peaking	Max. (mrem/ hr)	Un.	Limit (mrem/ hr)
Impact Limiter radius	8.8	44.1	10.0	62.9	1.1	69.7	1%	200
Impact Limiter bottom end	7.6	12.9	22.9	43.4	1.0	43.4	4%	200
Impact Limiter top end	11.6	0.6	1.7	14.0	1.0	14.0	7%	200
2 m side	2.1	4.9	1.8	8.9	1.04	9.3	1%	10
2 m bottom end	1.6	1.8	2.6	6.0	1.0	6.0	2%	10
2 m top end	2.2	0.1	0.2	2.5	1.0	2.5	4%	10

TABLE 2.5-26: HAC MAXIMUM DOSE RATE RESULTS, ALTERNATE CONFIGURATIONS

PWR								
Location	Gamma (mrem/ hr)	Neutron (mrem/ hr)	(n,γ) (mrem/ hr)	Total (mrem/ hr)	Angular Peaking	Max. (mrem/ hr)	Un.	Limit (mrem/ hr)
Intact Fuel								
1 m side	37.5	836.7	2.3	876.5	1.1	956.1	1%	1000
BWR								
Location	**Gamma (mrem/ hr)**	**Neutron (mrem/ hr)**	**(n,γ) (mrem/ hr)**	**Total (mrem/ hr)**	**Angular Peaking**	**Max. (mrem/ hr)**	**Un.**	**Limit (mrem/ hr)**
Intact Fuel								
1 m side	31.9	938.8	1.7	972.4	1.02	991.3	1%	1000

2.6 Criticality Evaluations

The applicable regulations are as follows:

10 CFR 71.55(b): The contents of the package must be subcritical assuming optimum moderation with fresh water with the contents in their as-loaded condition. This regulation applies to a single package reflected with 12 inches of water. To meet the requirements of 10 CFR 71.55(b), burnup credit is required for PWR fuel. Burnup credit is not required for BWR fuel.

10 CFR 71.55(d): For NCT, the package may be assumed to be dry as it is leaktight under normal conditions. Evaluations are performed for an infinite array of packages.

10 CFR 71.55(e): For HAC, the condition of the fuel is unknown due to the unknown properties of the cladding at high burnup. To meet the criticality requirements for unknown fuel conditions, moderator exclusion is needed. Evaluations are performed for an infinite array of packages.

Because the requirements of 10 CFR 71.55(d) and (e) are performed for an infinite array of packages, a criticality safety index of zero is justified. These cases are performed unmoderated and burnup credit is not needed for the unmoderated PWR cases. Also, the regulatory evaluations described above are performed for an upper subcritical limit (USL) with an administrative margin of 0.05.

In addition to the cases described above, "defense in depth" HAC cases are performed. In the defense in depth cases, reasonable fuel damage is assumed with full water moderation. For these cases, it is sufficient to model only a single package reflected with water. Due to the high reactivities encountered for damaged and moderated fuel, a USL with an administrative margin of 0.02 is acceptable. Burnup credit is required for PWR fuel.

For the PWR analysis in which burnup credit is taken, the effect of a single fuel assembly mis-load is evaluated. For these cases, it is sufficient to model only a single package reflected with water. For this evaluation, a USL with an administrative margin of 0.02 is acceptable. A BWR mis-load analysis is not required because all BWR fuel is modeled as fresh.

2.6.1 Description of the Criticality Design

Fixed neutron absorbers and favorable geometry ensure the criticality safety of the system. The neutron absorber is present in the form of borated aluminum plates. This material is ideal for long-term use in the radiation and thermal environments of the package. The B-10 areal density is 40.6 mg B-10/cm^2 for the PWR basket (modeled in KENO as 0.125 inches thick). The poison plates in the BWR basket are 0.31 inches thick (modeled in KENO as 0.200 inches thick), with an areal density of 85.3 mg/cm^2. For both baskets, a 90 percent credit is taken in the analysis for the B-10 loading, resulting in as-modeled B-10 areal densities of 36.6 mg B-10/cm^2 for the PWR basket and 76.8 mg B-10/cm^2 for the BWR basket.

Moderator exclusion is also credited in the §71.55(e) analysis due to the double lid design of the cask.

2.6.2 PWR Criticality Analysis

2.6.2.1 Modeling Approach

Figure 2.6-1 shows the cross section of the 6625B-HB PWR fuel basket assembly. It consists of an egg-crate plate design. The fuel assemblies are housed in 24 stainless steel fuel compartments. The basket structure, including the fuel compartments, is held together with stainless steel insert plates and the neutron absorber and aluminum plates that form the egg-crate structure. The basket compartment structure is connected to perimeter rail assembly, portions of it comprising a solid aluminum interface. The perimeter rail interface provides the circular perimeter geometry that fits the basket inside the canister shell. The absorber/aluminum plates are located between the fuel compartments as shown in **Figure 2.6-1**. There are no plates on outward faces that do not connect to adjacent assemblies.

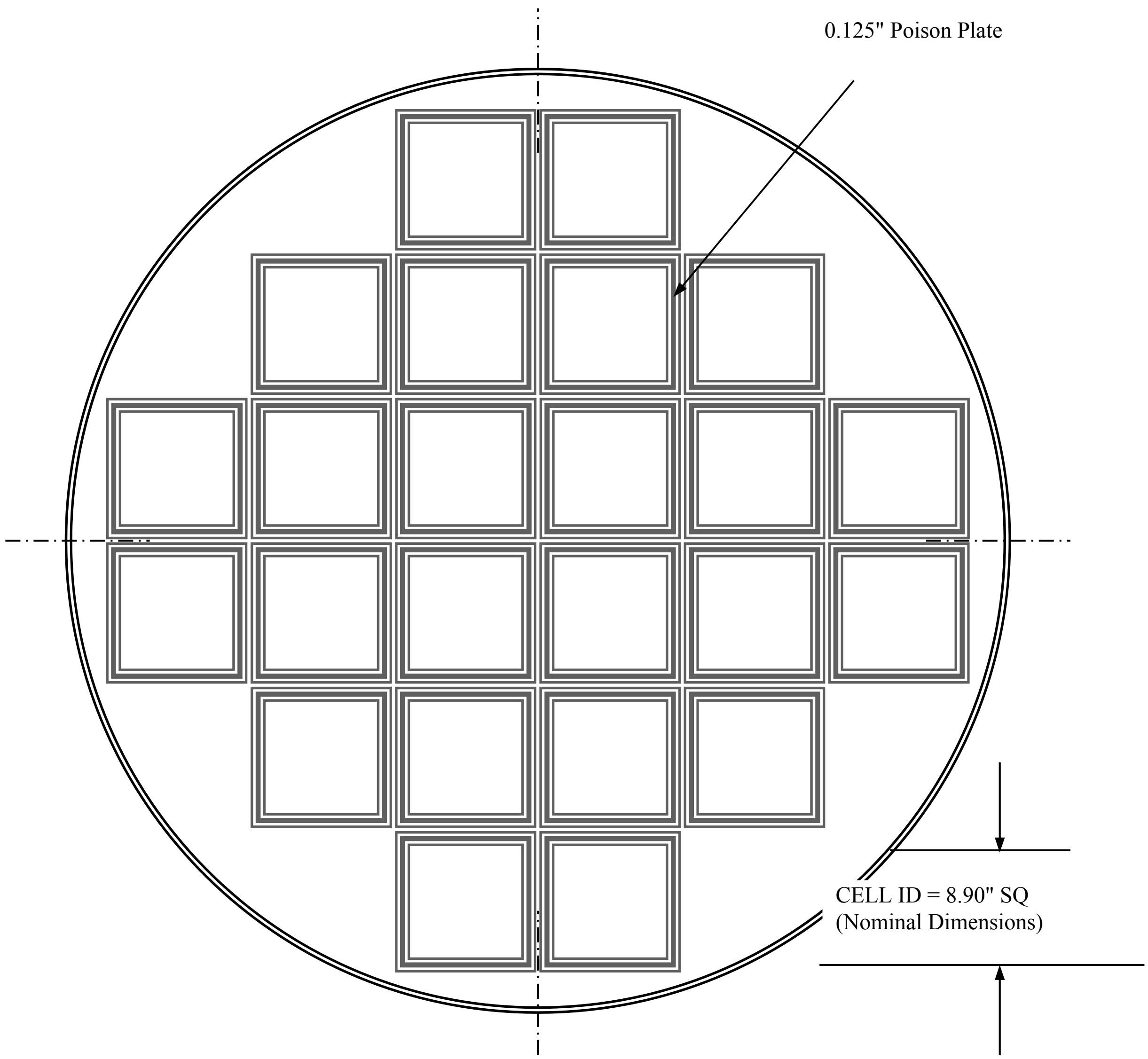


FIGURE 2.6-1: 24 PWR FUEL BASKET CROSS-SECTION

The fuel basket assembly is placed in the 6625B-HB shipping cask. The criticality safety model assumed the cask to be composed of three inches of lead encased in carbon steel. The model further did not include the neutron shield as a conservatism.

Burnup credit is taken in the §71.55(b) criticality evaluation for the PWR fuel. The PWR model includes the effects of uneven axial distribution of the burnup profile (i.e., the ends of the assemblies have lower burnup than does the middle, causing the end sections to be more reactive than the middle). Short-lived isotopes, particularly Xe-135, are not included in the model. The evaluations are performed crediting that the fuel assemblies had cooled for 5 years following irradiation. The length of the cooling time is significant because reactivity decreases with time due to the decay of Pu-241. Assemblies with longer cooling times may also be shipped, but the analysis conservatively assumes a cooling time of 5 years.

The geometry for the shipping cask is very similar to that of the MP197HB shipping cask. In both designs, 24 PWR fuel assemblies are placed in the borated aluminum-lined cells with an internal dimension of 8.90 inches. The analysis for the MP197HB is given in [13]. This analysis showed that the most reactive fuel assembly type is the Westinghouse 17x17 fuel assembly. This type of fuel assembly is used in the analysis here. The model also used a minimal inside dimension of 8.765 inches (based on manufacturing tolerances), which placed the assemblies closer together and hence increased assembly-to-assembly interaction.

2.6.2.2 PWR Basket Evaluation for 10 CFR 71.55(b)(d)(e)

The cask cavity is flooded to meet the requirements of 10 CFR 71.55(b). The evaluations determine k_{eff} with the CSAS25 control module of SCALE-4.4 [28] for each initial enrichment, including all uncertainties to assure criticality safety under all credible conditions. The results of the evaluation demonstrate that the maximum expected k_{eff}, including statistical uncertainty, will be less than the Upper Subcritical Limit (USL) determined from a statistical analysis of benchmark criticality experiments. The USL from Section A.6.5.10 of [13] is 0.9412. The statistical analysis procedure includes a confidence band with an administrative safety margin of 0.05. To include the uncertainties in isotopic production and distribution, an additional bias of 0.0225 is included. This bias is based on the actinide bias of 0.0175 given in Section A.6.5.14.1.3 and a horizontal distribution bias of 0.005 given in Section A.6.5.14.3 of [13]. (Note that uncertainties in the fission product concentrations are directly included in the KENO material inputs.) This gives a total USL of 0.9412 – 0.0175 – 0.005 = 0.9187.

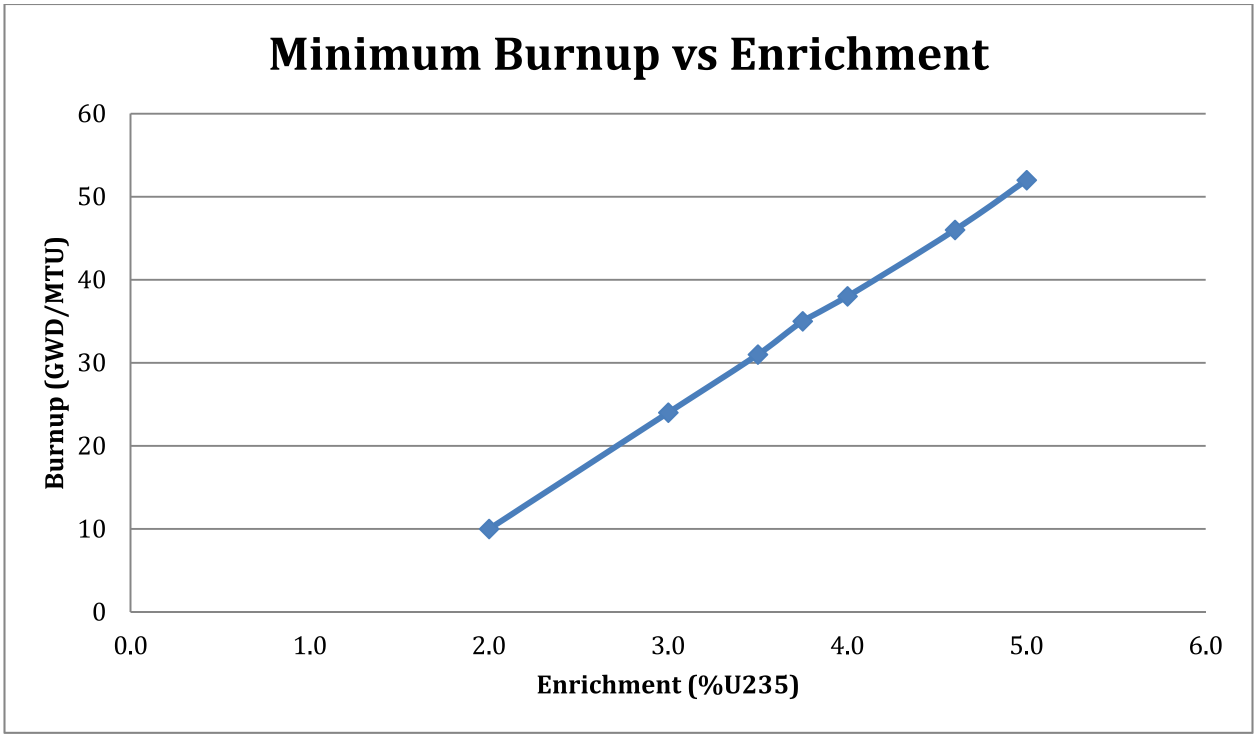
The minimum required fuel assembly burnup as a function of enrichment is shown in **Table 2.6-1** and **Figure 2.6-2** for intact fuel assemblies as determined by the criticality safety analysis. Those assemblies with the same or higher burnups than shown are acceptable to ship.

TABLE 2.6-1: MINIMUM ASSEMBLY BURNUP VS ENRICHMENT –UNDAMAGED PWR ASSEMBLIES

Enrichment (%U235)	Assembly Burnup (GWD/MTU)	k_{eff}	σ	k_{eff} +2σ
2.0	10	0.9036	0.0008	0.9052
3.0	24	0.9158	0.0009	0.9176
3.5	31	0.9138	0.0007	0.9152
3.75	35	0.9119	0.0007	0.9133
4.0	38	0.9149	0.0008	0.9165
4.6	46	0.9065	0.0007	0.9079

5.0	52	0.9036	0.0008	0.9052
USL = 0.9187				

FIGURE 2.6-2: MINIMUM PWR ASSEMBLY BURNUP VS ENRICHMENT



If NFAH (such as BPRAs) are shipped with the assemblies in the guide tubes, the value of k_{eff} decreases. This is due to the lower amount of moderator in the fuel region. As such, criticality safety criteria will be met for the same burnup and enrichment as shown in **Figure 2.6-2**.

The KENO-Va models used to develop **Table 2.6-1** do not include DFCs. Intact assemblies may be shipped in DFCs, and the DFCs may impact the reactivity of the system. If all 24 positions have intact PWR fuel in DFCs, there may be a slight increase in reactivity. This is due to compensating effects. The first is that the DFCs displace moderator, which can decrease reactivity. However, since the DFC is on the periphery of the fuel assembly, it also increases interaction between adjacent assemblies, which increases reactivity. This is seen from results shown in **Tables 2.6-2** through **2.6-4**.

For this part of the analysis, the KENO-Va input was modified to place the assemblies in the center of the shipping cask cell to allow room for modeling the DFCs. (The base criticality analysis assumes fuel assemblies are shifted into corners of the cell, which increases neutron interaction.) **Table 2.6-2** shows the values of k_{eff} for centered assemblies with no DFCs. DFCs 0.031 inches thick were then placed around the assemblies. These results are shown in **Table 2.6-3**. After modeling the DFCs, it is seen in these tables that reactivity may either decrease or increase, as seen in **Table 2.6-4**, with the change dependent on both the initial enrichment and modeled burnup. There is no firm amount of change. Therefore, the licensing analysis for this shipping cask will address this issue to ensure that the burnup-enrichment curve contains sufficient margin to allow the use of cans.

To meet the requirements of 10 CFR 71.55(d)(e), the cask is modeled with dry conditions within the cask cavity, although water continues to fill the pellet-clad gap and the cask is modeled in an infinite array. A void was placed around the cask and full reflection assumed to simulate an infinite array. A void was chosen as this maximizes neutron interaction between casks. Burnup credit is not required. A model with fresh 5 percent enriched fuel, under dry conditions, results in a value of $k_{eff}+2\sigma$ of 0.5777, which is well below the USL.

TABLE 2.6-2: UNDAMAGED PWR ASSEMBLIES – NO DFCS – CENTERED IN CELLS

Enrichment (%U235)	Assembly Burnup (GWD/MTU)	k_{eff}	σ	k_{eff} +2σ
2.0	10	0.9039	0.0008	0.9055
3.0	24	0.9130	0.0007	0.9144
3.5	31	0.9105	0.0007	0.9119
3.75	35	0.9094	0.0007	0.9108
4.0	38	0.9104	0.0008	0.9120
4.6	46	0.9038	0.0009	0.9056
5.0	52	0.9011	0.0008	0.9027

TABLE 2.6-3: UNDAMAGED PWR ASSEMBLIES – ENCASED IN DFCS – CENTERED IN CELLS

Enrichment (%U235)	Assembly Burnup (GWD/MTU)	k_{eff}	σ	k_{eff} +2σ
2.0	10	0.8986	0.0008	0.9002
3.0	24	0.9129	0.0008	0.9145
3.5	31	0.9081	0.0009	0.9099
3.75	35	0.9101	0.0008	0.9117
4.0	38	0.9119	0.0008	0.9135
4.6	46	0.9017	0.0008	0.9033
5.0	52	0.8985	0.0010	0.9005

TABLE 2.6-4: REACTIVITY DIFFERENCE – NO DFC VS WITH DFC PWR FUEL

Enrichment (%U235)	Assembly Burnup (GWD/MTU)	Δk_{eff}
2.0	10	-0.0053
3.0	24	0.0001
3.5	31	-0.0020
3.75	35	0.0009
4.0	38	0.0015
4.6	46	-0.0023
5.0	52	-0.0022

2.6.2.3 PWR Defense in Depth Analysis

Following the approach used for the MP197HB, additional ‘defense-in-depth’ models were run for reasonable damaged fuel geometry under HAC and allowing fresh water moderation. For these cases, it was assumed the fuel rod pitch expanded to the maximum allowable by the container geometry (no DFCs). Since the interior of the fuel-shipping cell is 8.765 inches, and the nominal fuel assembly width is 8.432 inches, this corresponds to an increase in rod pitch of 3.9

percent. The accident condition also had a lower administrative margin of 2 percent, which would place the USL at 0.9487. The 'defense-in-depth' results are shown in **Table 2.6-5** and show that the same burnup curve is acceptable.

TABLE 2.6-5: ASSEMBLY BURNUP VS ENRICHMENT – DAMAGED PWR ASSEMBLIES, DEFENSE IN DEPTH (HAC)

Enrichment (%U235)	Assembly Burnup (GWD/MTU)	k_{eff}	σ	$k_{eff}+2\sigma$
2.0	10	0.9211	0.0008	0.9227
3.0	24	0.9360	0.0007	0.9374
3.5	31	0.9354	0.0007	0.9368
3.75	35	0.9336	0.0008	0.9352
4.0	38	0.9370	0.0008	0.9386
4.6	46	0.9311	0.0007	0.9325
5.0	52	0.9292	0.0008	0.9308
USL = 0.9487				

2.6.2.4 PWR Mis-loaded Fuel Assembly Analysis

A possible hazard to be analyzed is the inadvertent loading of a high reactivity fuel assembly into the PWR shipping cask. A criticality event is not considered credible should such an assembly be placed into the cask during loading operations. This is because PWR spent fuel pools are borated, normally at about 3000 ppm boron concentrations, whenever fuel movement is being performed. This amount of boron is sufficient to preclude criticality. The scenario being considered here is that the high reactivity assembly is not noticed during cask loading operations and is still loaded inside the cask during shipment.

The most reactive fuel assembly that could be mis-loaded is a fresh 5% enriched fuel assembly. A case was run with such an assembly located in the center of the basket, which would be the most reactive position. The criterion to be met is that the presence of this assembly would still result in a value of $k_{eff}+2\sigma$ this less than the USL for Defense in Depth conditions of 0.9487. The result of that case is shown in **Table 2.6-6**. It is seen that this criteria is not met. This indicates that strong administrative controls will be required during fuel loading if such a fuel assembly is also located in the spent fuel pool.

Additional cases were also run examining the effects of having a lower enriched assembly being mis-loaded. The purpose is to show at what enrichment it would be necessary for additional administrative controls. These results are also shown in **Table 2.6-6**. It is seen that any fuel assembly with an enrichment greater than ~3.6%U-235 would not meet the USL criteria. This will call for tight administrative controls that check that only allowed assemblies are loaded into the cask.

TABLE 2.6-6: MISLOADED HIGH REACTIVITY PWR ASSEMBLY

Enrichment (%U235)	k_{eff}	σ	$k_{eff}+2\sigma$
5.0	0.9743	0.0008	0.9759
4.6	0.9635	0.0009	0.9653
4.2	0.9566	0.0009	0.9584
4.0	0.9533	0.0011	0.9555
3.8	0.9481	0.0009	0.9499

3.6	0.9433	0.0009	0.9451
USL = 0.9487			

2.6.2.5 PWR Criticality Analysis Conclusion

The design of the 24 UNF PWR basket meets the requirements set in 10 CFR 71 for normal and accident conditions. Strict controls during fuel loading will be required to prevent a high-reactivity assembly mis-load.

2.6.3 BWR Criticality Analysis

2.6.3.1 Modeling Approach

A BWR UNF fuel basket is designed to hold 61 BWR fuel assemblies. **Figure 2.6-3** shows the cross section of the 6625B-HB fuel basket assembly for 61 BWR fuel assemblies in the cask. It consists of an egg-crate plate design. The fuel assemblies are housed in 61 stainless steel fuel compartments. The basket structure, including the fuel compartments, is held together with stainless steel insert plates and the poison plates that form the egg-crate structure, similar in form to that for the PWR fuel assemblies. The borated plates are conservatively modeled as 0.200 inches thick with an areal density of 76.8 mg B-10/cm^2 after 90 percent credit.

The BWR fuel analysis did not apply burnup credit, instead conservatively assuming all fuel was fresh. As all fuel is fresh, there is no bias due to actinide uncertainty or radial burnup gradient. As such, the USL is 0.9412.

The criticality safety model for the cask is the same used for the PWR assemblies, which assumes the cask to be composed of three inches of lead encased in carbon steel. The model also did not include the neutron shield as a conservatism. Full water reflection was used around the cask.

2.6.3.2 BWR Basket Evaluation for 10 CFR 71.55(b)(d)(e)

For the 10 CFR 71.55(b) analysis, a case with fresh 5 percent enriched GE10x10 fuel assemblies centered in the fuel compartments, without zircaloy channels, and with fully flooded conditions is modeled in KENO-Va. The result is $k_{eff}+2\sigma = 0.9250$, which meets the USL of 0.9412. A similar model with zircaloy fuel channels is also developed. The fuel channel model is shown in **Figure 2.6-4**, with the detail shown in **Figure 2.6-5**. This model gives $k_{eff}+2\sigma = 0.9252$, which is statistically equal. This shows the presence of the fuel channels has a minimal impact on reactivity. When the non-channeled assemblies are modeled being placed toward the centerline of the basket to increase assembly interaction (see **Figure 2.6-3**), the model gives $k_{eff}+2\sigma = 0.9233$, showing that the system is more reactive with the fuel assemblies centered in the basket compartments.

Intact BWR fuel may be placed in DFCs. These containers are made of stainless steel, are 5.77 inches square, and have walls 0.031 inches thick. Cases are run with fuel maintaining normal geometry inside a DFC, one case also with a zircaloy fuel channel and another case without. The case with no channel gave a value of $k_{eff}+2\sigma$ of 0.9316, and the case with a fuel channel gave a value of $k_{eff}+2\sigma$ of 0.9312. These are statistically equal, as was observed for the cases without a DFC. This comparison also shows that the DFC slightly increases k_{eff} due to increased fuel interaction. The cases with the DFC meet the USL criteria of 0.9412.

A 10 CFR 71.55(d)(e) analysis is performed for a dry cask cavity with the cask in an infinite array. Although the cask cavity is dry, water continued to fill the pellet-clad gap and the cask was fully reflected. Two models were examined. One had the fuel in a normal configuration centered in the cells without channels. This gives $k_{eff}+2\sigma = 0.4702$. The other model assumed the fuel channel was present and that the fuel rods expanded to fill the channel under HAC conditions. This gave a value of $k_{eff}+2\sigma$ of 0.4591. These results show that the value of k_{eff} is very low for dry conditions and readily meets the USL.

2.6.3.3 BWR Defense in Depth Analysis

Defense in Depth HAC conditions for BWR fuel have yet to be determined. Structural integrity of low-burnup fuel (≤45 GWd/MTU) may be assumed in the accident condition. However, structural integrity of high-burnup fuel (>45 GWd/MTU) cannot be guaranteed. For the defense in depth analysis, it is conservatively assumed that all fuel is high-burnup. Pitch expansion will likely be part of the consideration. How much the fuel pins will separate depends on whether or not the fuel assembly is shipped within its zircaloy fuel channel. Two cases are run to determine the impact of expanded fuel on a fully flooded shipping cask. The first assumed the fuel would evenly expand to fill the entire cell containing the fuel. The results showed an unacceptable value of $k_{eff}+2\sigma$ of 1.0054. When the fuel is shipped in its fuel channel, however, the pins cannot separate as much. When this was studied it was found that when the pins expanded to fill the fuel channel, the value of keff+2σ was 0.9488, which meets the defense in depth USL value for HAC conditions of 0.9712. As such, it will be recommended that high-burnup BWR fuel be shipped with its fuel channel.

The HAC model is modified so that fuel with expanded pitch and a fuel channel are placed inside a DFC. The results gave a value of $k_{eff}+2\sigma$ of 0.9549, showing a small increase in reactivity due to higher fuel interaction. Modeling the same case without the channel but same fuel pitch gave essentially the same value of $k_{eff}+2\sigma$ of 0.9548. These values are well below the HAC limit of 0.9712. These cases indicate that the DFC may be used and meet USL criteria.

2.6.3.4 BWR Criticality Analysis Conclusion

The design of the 61 UNF BWR basket allows for the shipment of spent fuel assemblies. The analysis shows that the requirements set in 10 CFR 71 will be met for normal and accident conditions. To meet defense in depth accident criteria, though, high-burnup assemblies should be shipped with fuel channels. It is not necessary to ship low-burnup assemblies with fuel channels because low-burnup fuel will remain intact in an accident.

FIGURE 2.6-3: 61 BWR FUEL ASSEMBLY BASKET IN CASK

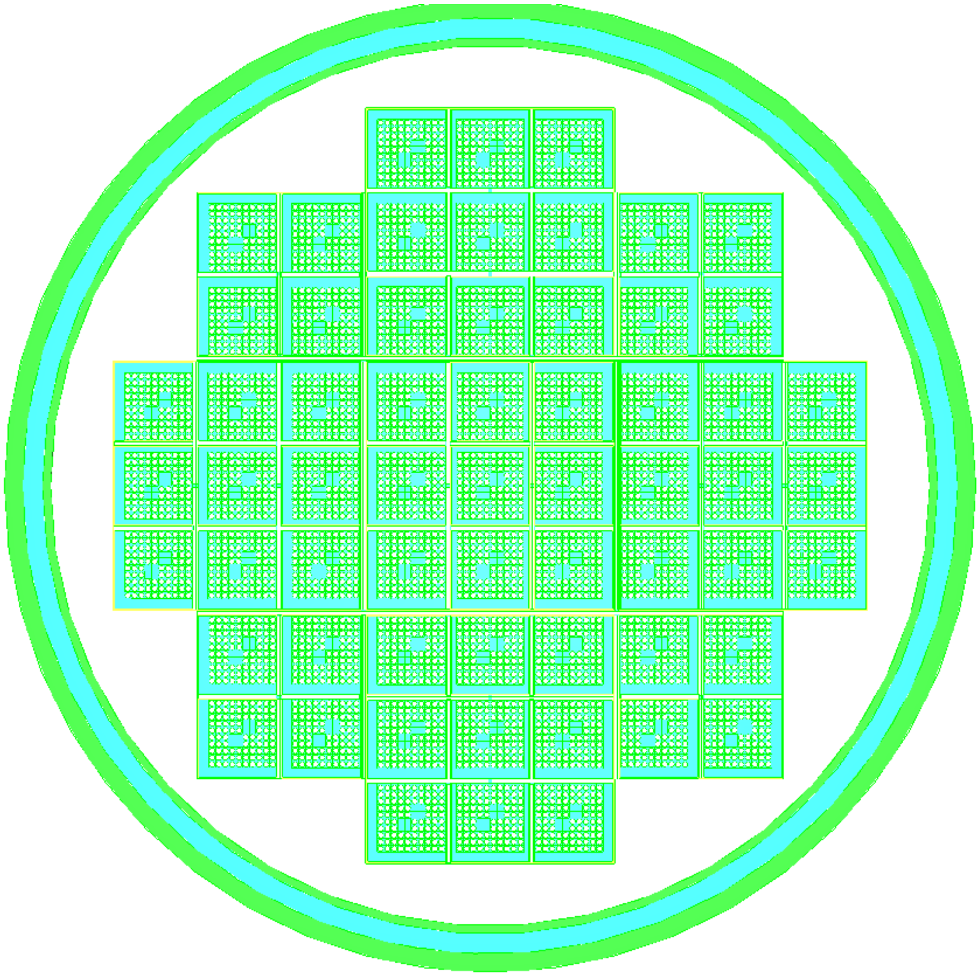


FIGURE 2.6-4: 61 BWR FUEL ASSEMBLIES WITH CHANNELS IN CASK

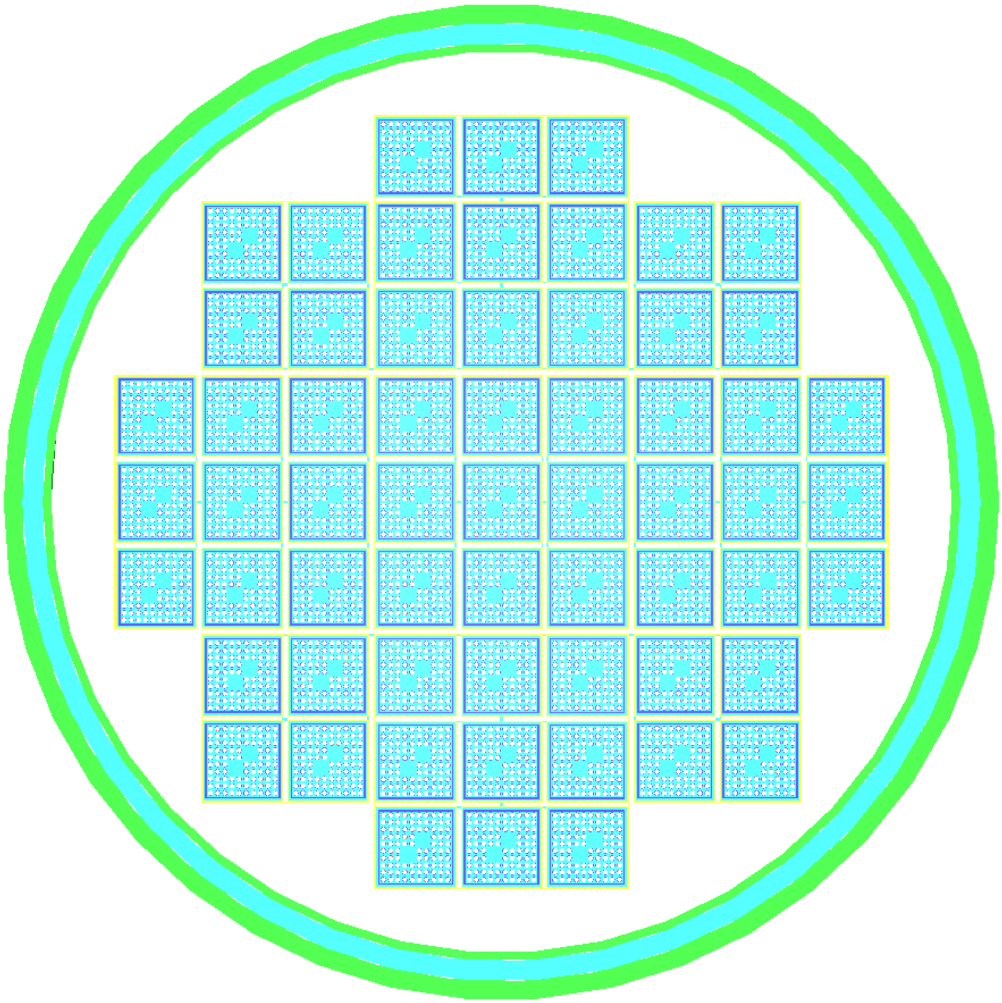
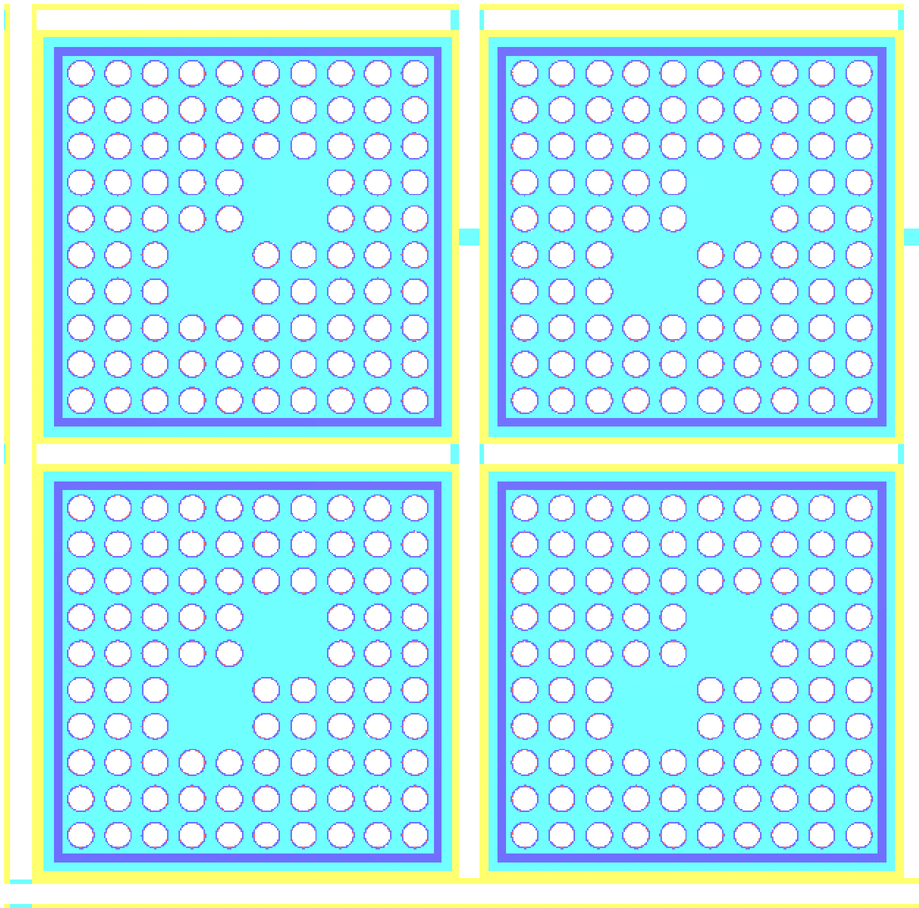


FIGURE 2.6-5: DETAIL OF MODEL SHOWING FUEL ASSEMBLY CHANNELS



2.7 Operating Procedures

This chapter contains loading and unloading procedures that are intended to show the general approach to operational activities for the 6625B-HB transportation cask. A separate Operations Manual (OM) will be prepared by the operator to describe the operational steps in greater detail. The OM, along with the information provided here, will be used to prepare the detailed site procedures that will address the particular operational considerations related to the 6625B-HB transportation cask and the site. Examples of specific site operations are:

- Haul path analysis based on site seismic conditions
- Requirement to use seismic restraints for the Cask
- Crane hardware interface / yoke / yoke Extensions

The operating procedures occur wet with four basket designs required to accommodate four payload configurations:

- BWR bare fuel assemblies
- BWR canned fuel assemblies
- PWR bare fuel assemblies
- PWR canned fuel assemblies

The operating procedures for PWR or BWR fuel assemblies are identical.

2.7.1 Package Loading

2.7.1.1 Preparation for Loading

- Perform a receipt inspection to check for any damages or irregularities upon arrival of the empty packaging. Verify that the records for the packaging are complete and accurate.
- Remove the security device, the impact limiter attachment bolts, and the associated hardware, as necessary. Remove the front and the rear impact limiters and remove the tie-downs and restraints.
- Clean the external surfaces of the cask, if necessary, to remove the road dirt.
- Rotate the cask from the horizontal to the vertical position. Lift and place the cask in the cask preparation area.
- Remove the outer lid bolts and the outer lid. Remove the seals from cask outer lid, vent, drain, and transport cover, and inspect the lid-sealing surface. Repeat these steps for the inner lid.
- Verify that the appropriate basket and/or damaged fuel can assemblies are installed in the cask, with no evident signs of damage to either. Verify that there is no foreign material in the cask.

2.7.1.2 Loading

Note: Prior to initiation and during loading operation for PWR fuel assemblies ensure that the boron concentration of the spent fuel pool water and the water added to the cavity of a loaded cask be greater than or equal to the pre-specified boron concentration to control the criticality of the content.

Move the cask to the cask-loading area using the lift beam attached to the top trunnions. Lower the cask into the cask loading pool. Load the pre-selected spent fuel assemblies or DFCs into the basket compartments following one of the options below:

Loading Bare Fuel into the Cask

- Obtain a list of bare fuel assemblies, transfer procedure, and cask loading diagram. Grapple the bare fuel assemblies one at a time and move them to the appropriate cell in the basket. Verify that the bare fuel assembly is seated properly in the basket. Verify the identity of the fuel assemblies loaded into the cask, and document the location of each fuel assembly/DFC on the cask-loading report.

Loading Canned Fuel into the Cask

- Remove the removable sleeve from the cask. Obtain a list of bare fuel assemblies, transfer procedure, and cask loading diagram.
    - Option 1: Selected fuel is installed in damaged fuel cans in the basket
      Remove damaged fuel can lid assemblies from the damaged fuel cans in the basket cells. Grapple the selected fuel assemblies one at a time and move them to the appropriate damaged fuel can in the basket. Install damaged fuel can lids on the can assemblies and latch in place using remote tools.
    - Option 2: Selected fuel assembly is installed in damaged fuel can in rack, then installed into basket
      Remove damaged fuel can lid assemblies from the can assemblies in the rack. Grapple the selected fuel assembly one at a time and move to the appropriate damaged fuel can in the rack. Install damaged fuel can lids on the can assemblies and latch in place using remote tool. Loading the fuel assemblies into the damaged fuel cans can be performed prior to lowering the cask in the pool.
      Grapple damaged fuel cans and lids containing selected fuel assemblies one at a time and move them to the appropriate cell in the basket.

  Note: The potential for fuel misloading is essentially eliminated through the implementation of procedural and administrative controls. The controls instituted to ensure that the selected spent fuel assemblies are placed into a known cell location within a basket will typically consist of the following:

    - A cask/DSC loading plan is developed to verify that the spent fuel assemblies meet the applicable burnup, enrichment and cooling time parameters.
    - The loading plan is independently verified and approved before the fuel load.
    - A fuel movement schedule is then written, verified and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance with the fuel movement schedule.

- Verify the identity of the DFC/fuel assembly combination loaded into the cask, and document the location of each fuel assembly/DFC on the cask-loading report.
- Configure the cask inner lid prior to installation so that water may be drained through the drain port and that helium can be used to fill the cask as the water is drained. Lower the inner lid placing it on the cask body flange with at least one lid penetration (drain or vent port) open.
- Lift the cask so that the top of the cask is above the pool water surface and install the lid bolts.
- Drain the water from the cask using the drain port. Helium shall be used to fill the cask as the water is drained from the cask cavity. In order to minimize internal hydrogen

accumulation, the cask should be drained completely within 18 hours of the start of draining. Draining may be done either before or after lifting the cask out of the pool depending on the maximum lift capacity.

- Disconnect the drain line and move the cask to the decontamination area.

2.7.1.3 Preparation for Transport

Note: During preparation for transport, worker exposure can be minimized by use of temporary shielding and by minimizing the exposure time and maximizing the distance, as well as using any measures to facilitate decontamination.

- Decontaminate the cask until acceptable surface contamination levels are obtained.
- Install the remaining inner lid bolts and tighten to the specified torque.
- Evacuate the cask cavity using the Vacuum Drying System (VDS) to remove the remaining moisture. The vacuum pump is connected to the vent port on the inner lid.

Note: Based on current NRC recommendations, the cask cavity must be evacuated to 4 mbar and hold for at least 30 minutes to adequately dry the cask cavity.

Typical vacuum drying takes approximately 12 to 24 hours. Presence of DFC may prolong the vacuum drying time since some small amount of water entraps between the bottom of the DFC and cask bottom plate.

- Isolate the vacuum pump, and backfill the cask cavity with helium (minimum 99.99 percent purity).
- Leak test the inner lid and inner vent and drain port cover seals. If the cask does not pass the leak test, evaluate the test method or return the cask to the pool and replace the lid seals.
- After positive leak test of the inner lid, install the outer lid and tighten the outer lid bolts to the specified torque. Evacuate the cavity between the inner and outer lid using the VDS and backfill it with helium (minimum 99.99 percent purity). Perform a leak test of the outer lid. If the outer lid does not pass the leak test, evaluate the test method or replace the outer lid seals.
- Lift the cask off and if necessary, rotate the cask from the vertical to the horizontal position. Place the cask on the transport skid/frame.
- Remove the trunnions and install the front and the rear impact limiters onto the cask.
- Install security seal and personnel barrier.
- Perform a final radiation and contamination survey to assure compliance with 10 CFR 71.47 and 71.87 and apply appropriate DOT labels and placards in accordance with 49 CFR 172.
- Prepare the final shipping documentation and release the loaded cask for shipment.

Simplified flow diagrams of cask loading operations are shown in **Figures 2.7-1 and 2.7-2**.

FIGURE 2.7-1: CASK LOADING OPERATIONS
BARE BWR & PWR FUEL LOADING STEPS (SIMPLIFIED FLOW DIAGRAM)

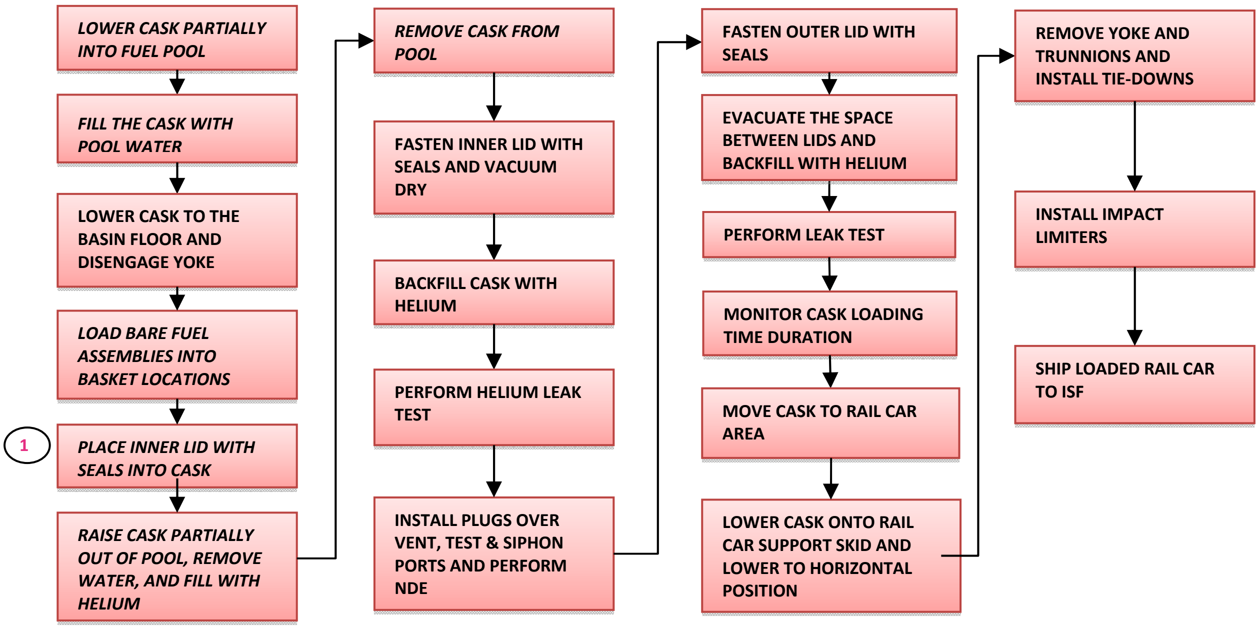
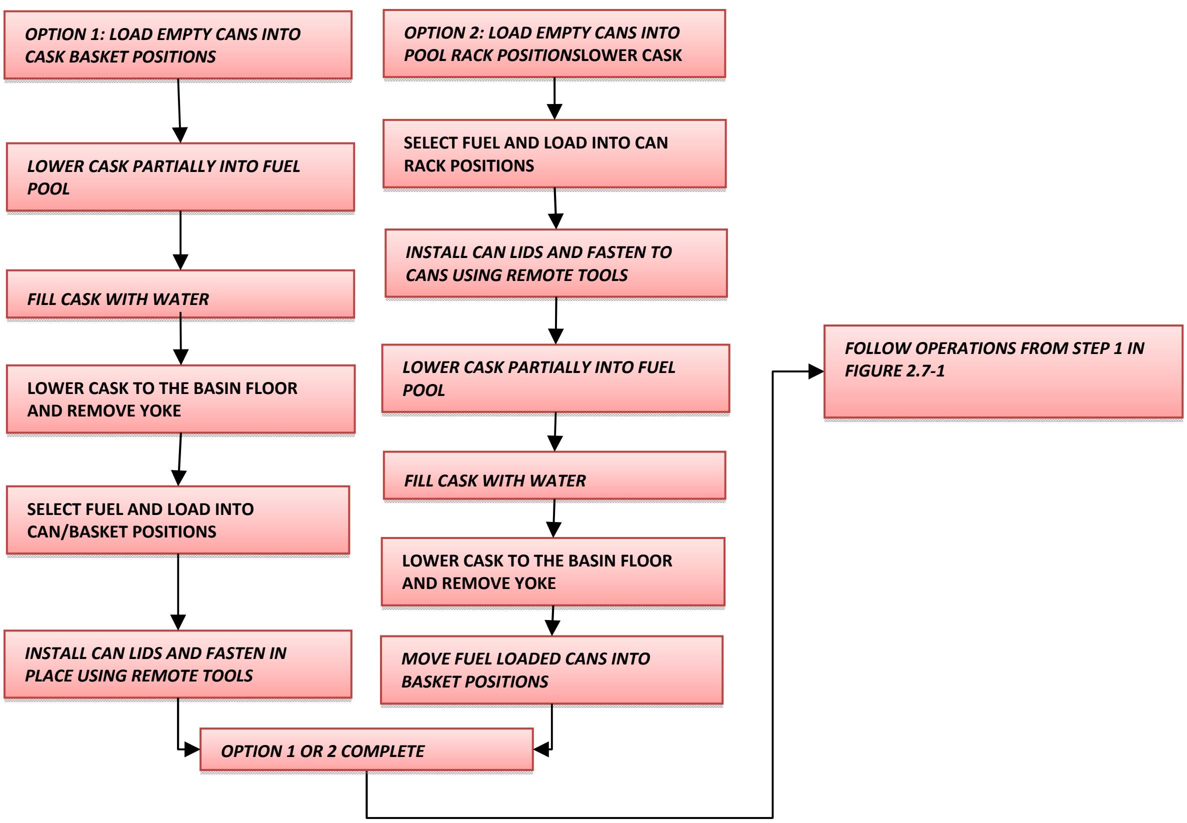


FIGURE 2.7-2: CASK LOADING OPERATIONS
CANNED BWR & PWR FUEL LOADING STEPS (SIMPLIFIED FLOW DIAGRAM)



2.7.2 Package Unloading

2.7.2.1 Receipt of Package from Carrier

- Upon arrival of the loaded cask, perform a receipt inspection of the cask to check for any damage or irregularities. Verify that the security seal is intact and perform a radiation survey.
- Remove the security seal.
- Remove the front and rear impact limiters as well as the thermal shield. Remove the tie down strap and trunnion support block caps.
- Attach the trunnions to the cask. Attach the lift yoke to the cask handling crane hook, and then engage the lift beam to the two upper (top) trunnions. Rotate the cask slowly from the horizontal to the vertical position.
- Lift the cask from the transport/shipping skid/frame, and place it in the decontamination area.

2.7.2.2 Preparation for Unloading

- Loosen and remove the outer lid bolts. Engage the lid lifting equipment and remove the outer cask lid.
- Remove the outer vent cover.
- Collect a cavity gas sample, through the vent port quick-disconnect coupling, if required.
- Analyze the gas sample for radioactive material and add necessary precautions based on the cavity gas sample results.

  Note: If degraded fuel is suspected, additional measures, appropriate for the specific conditions, are to be planned, reviewed, and approved by the appropriate plant personnel, as well as implemented to minimize worker exposures and radiological releases to the environment. These additional measures may include provision of filters, as well as respiratory protection and other methods to control releases and exposures to ALARA.

- Vent the cavity gas through the hose to the building ventilation system until atmospheric pressure is reached.
- Attach the fill and drain lines to the drain quick-disconnect coupling and the vent port adapter. Ensure that appropriate measures are in place for proper handling of steam.
- Loosen the inner lid bolts and remove all but six lid bolts, approximately equally spaced.
- Attach the cask to the crane using the lift beam. Attach the lid lifting equipment. Lift the cask and lower it into the SFP until the top surface is just above the water level.

  Note: If the maximum lift weight is not exceeded, the cask may be filled with pool water before lowering the cask into the pool or while the cask is partially submerged in the spent fuel pool.

2.7.2.3 Contents Removal

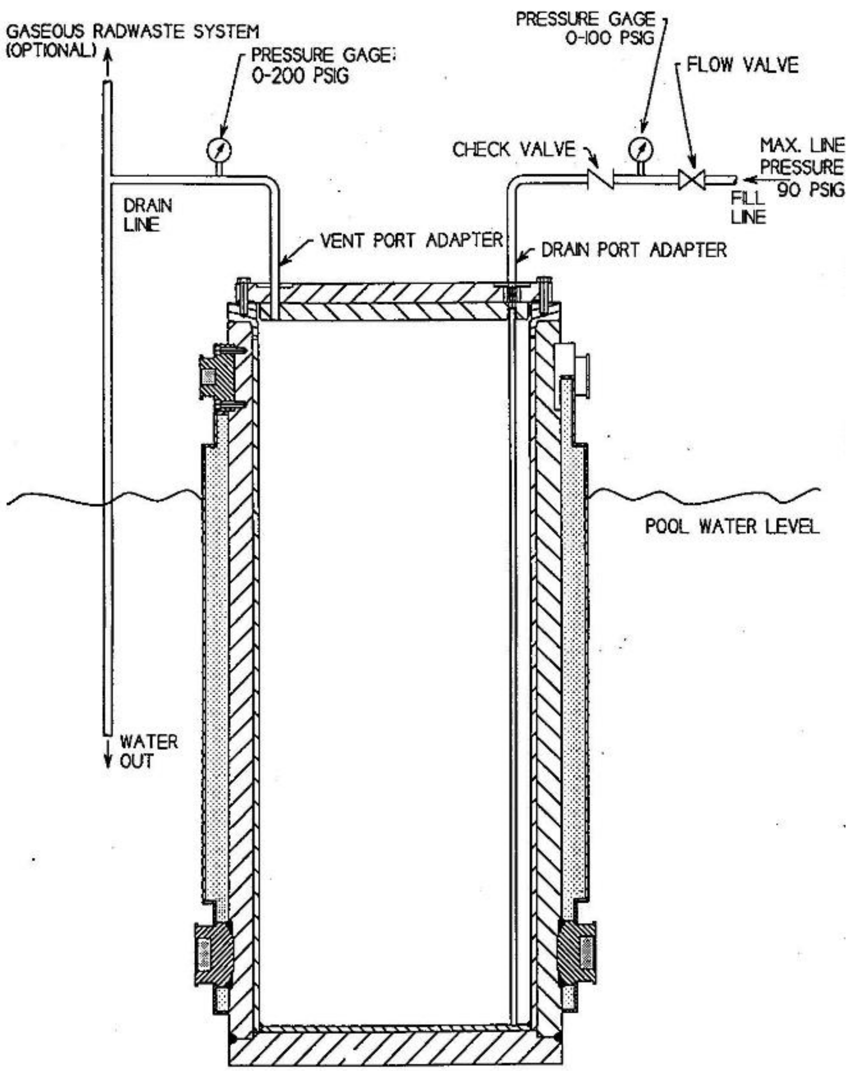
Note: Prior to initiation and during unloading operation for PWR fuel assemblies ensure that the boron concentration of the spent fuel pool water and the water added to the cavity of a loaded cask be greater than or equal to the pre-specified boron concentration to control the criticality of the content.

- Begin pumping pool or demineralized water into the cask through the drain port at a pre-specified rate while continuously monitoring the exit-pressure to avoid steam buildup.

  Note: Steam flushing may occur during unloading. In order to control the pressure within the cask cavity and prevent water/steam splashes, the reflooding rate must be controlled while the vent port is connected to the gaseous rad-waste system.

  Pressure measurement on the line to the rad-waste system is used to control the reflood rate. The set point for the pressure gage on the rad-waste line depends on the type of the connections and will be evaluated by operating site to ensure the internal cask cavity pressure remains (with a good margin) below the design pressure. A typical reflooding arrangement for a typical cask is shown in **Figure 2.7-3**.

FIGURE 2.7-3: TYPICAL REFLOODING SETUP



- When the cask is full of water, detach the fill and drain lines from the cask, remove the remaining six bolts, and lower the cask to the bottom of the pool/pit.
- Remove the cask lid. Remove the removable sleeve, if needed, and unload the spent fuel assemblies/DFCs in accordance with the site procedures.
- After completion of unloading the fuel assemblies/DFCs, lower the lid and place it on the cask body flange. At least one lid penetration must be completely open prior to installation of the lid.
- Lift the cask out of the pool.
- Using the drain port in the lid, drain the water from the cask.
- Move the cask to the decontamination area.

2.7.3 Preparation of Empty Package for Transport

- Decontaminate the cask until acceptable surface contamination levels are obtained.
- Place the removable sleeve into the cask cavity, if needed.
- Install the inner lid. Lubricate and install the inner lid bolts. Torque the bolts to pre-specified values.
- Evacuate the cask cavity using the VDS to remove the remaining moisture, and verify the dryness as follows:
    - Purge or evacuate the helium supply lines and evacuate the cask to 4 mbar (4 x 10^{-4} MPa) or less. Make provision to prevent or correct icing of the evacuation lines.
    - Isolate the vacuum pump. If, in a period of 30 minutes, the pressure does not exceed 4 mbar (4 x 10^{-4} MPa), the cask is adequately dried. Otherwise, repeat vacuum pumping until this criterion is met.
    - Backfill the evacuated cask cavity with helium (minimum 99.99 percent purity) to slightly above atmospheric pressure, remove the vacuum connector, and immediately install the quick disconnect fitting.
    - Attach the vacuum/backfill manifold to the vent port fitting, purge or evacuate the helium supply lines, and re-evacuate the cask to below 100 mbar.
    - Isolate the vacuum pump and backfill the cask cavity with an inert gas.
    - Install the cask outer lid. Evacuate the cavity between the two lids using the VDS and verify the dryness as described above.
    - Lift the cask off the decontamination pad. Rotate the cask from the vertical to the horizontal position.
    - Check if the surface dose rates and the surface contamination levels are within the regulatory limits.
    - Install the front and the rear impact limiters onto the cask.
    - Perform a final radiation and contamination survey to assure compliance with 10 CFR 71.47 and 71.87.
    - Install a transportation enclosure and apply appropriate DOT labels and placards in accordance with 49 CFR 172.
    - Prepare the final shipping documentation and release the empty cask for shipment.

2.8 Acceptance Tests and Maintenance Program

2.8.1 Acceptance Tests

The anticipated Acceptance Tests for a cask of this style are as described in this section.

2.8.1.1 Visual Inspections and Measurements

Visual inspections are performed to ensure that the cask conforms to the drawings and specifications. This includes Cleanliness Inspections, Visual Weld inspections per the ASME Code, inspection of Sealing Surfaces, and Dimensional Inspections.

2.8.1.2 Weld Inspections

All welding will be performed using qualified process and qualified personnel, according to the ASME Boiler and Pressure Vessel (B&PV) Code. Examination requirements meeting the ASME B&PV Code would be specified on the licensing drawings by personnel qualified in accordance with SNT-TC-1A.

2.8.1.3 Structural and Pressure Tests

A pressure test is performed on the cask assembly at a pressure above the greater of 1.25 times the design pressure per the requirements of ASME B&PV Code, Section III, Subsection NB Paragraph NB-6200 or NB-6300 or 1.50 times the maximum normal operating pressure, per 10 CFR 71.85(b). The test pressure is held for a minimum of 10 minutes. Testing may also be done in accordance with Section III, Division 3 Subsection WB, Paragraph WB-6200 or WB -6300.

The lifting trunnions shall be load tested to three times the design lift load to qualify them for single failure proof lift purposes.

The impact limiter lifting points shall be load tested to 1½ times their design load.

2.8.1.4 Leakage Tests

The containment boundary will be leakage rate tested during fabrication in accordance with ANSI N14.5. At final assembly the lids and seals including the drain and vents will also be leak tested in accordance with ANSI N14.5. The leakage rate tests will demonstrate leaktight containment by testing to 1×10^{-7} ref-cm^3/s. The leakage rate tests for the assembly will be duplicated on an annual basis or when the seal(s) are replaced.

2.8.1.5 Component Tests

2.8.1.5.1 Valves, Rupture Discs, and Fluid Transport Devices

There are no valves, rupture discs, or couplings in the 6625B-HB cask design.

2.8.1.5.2 Gaskets

Gaskets used for sealing the lids and other ports are elastomer seals that will be leakage rate tested when installed.

2.8.1.6 Shielding Tests

2.8.1.6.1 Neutron Shield Tests

The radial neutron shield is protected from damage and loss by a metal enclosure. The neutron shield material is expected to be a proprietary, borated, reinforced polymer that is resistant to high temperatures. The function of the resin is to shield against neutrons, which is primarily performed by the hydrogen and boron in the material. The boron also reduces n-γ reactions with hydrogen and thus, reduces the secondary gamma dose rate. The resin based on its density provides some gamma shielding.

The performance of the shielding can adequately be verified by chemical analysis and verification of the density. The chemical composition is verified by controlling the mixture and verifying the composition whenever new batches of the components are used. The density of the resin is verified with every mix batch.

2.8.1.6.2 Gamma Shield Tests

The gamma shield integrity of the cask body lead shielding will be verified with gamma scanning prior to the installation of the neutron shielding. The gamma scan will ensure that there are no voids or streaming paths prior to closure of the lead shield cavity.

2.8.1.7 Neutron Absorber Tests

The neutron absorber tests verify two functions of the material. The thermal conductivity of the material is qualified for the temperature range of interest. The production material that is utilized in the basket is verified for at least one temperature.

The B-10 areal density is verified by coupons removed from various areas of the sheets formed from each ingot, and tested with neutron transmission testing. The coupons are tested against known calibrated coupons. The limiting values ensure that there is 95 percent confidence, 95 percent certainty the boron 10 areal density is 100 percent of the minimum specified value.

2.8.1.8 Thermal Tests

Based on the known properties of the materials of construction of the 6625B-HB cask, no assembled thermal tests are necessary for acceptance of the package.

2.8.2 Maintenance Program

2.8.2.1 Structural

To verify continued compliance, after every 14 months of use, the lifting trunnions shall be subjected to either, a load test in accordance ANSI N14.6, or a dimensional testing, visual inspection, and nondestructive examination of critical areas including the bearing surfaces in accordance with Paragraph 6.3.1 of ANSI N 14.6.

2.8.2.2 Leak Tests

All containment closures, including the inner and outer lids, the vent port, and drain port, will be subjected to periodic maintenance pre-shipment testing in accordance with ANSI N14.5.

Periodic testing:

The seals shall be leakage rate tested to 1×10^{-7} ref-cm^3/s or better within 12 months prior to shipment

Pre-shipment:

Prior to each shipment the seals shall be leak tested to 1×10^{-3} ref-cm^3/s or better as permitted by ANSI N14.5 when the seals have not been replaced.

Maintenance:

After maintenance repair or replacement of any containment component including inner seals the component shall be leakage rate tested to 1 x 10^{-7} ref-cm^3/sec or better.

2.8.2.3 Component and Material Tests

2.8.2.3.1 Fasteners

All threaded fasteners and port plugs shall be inspected whenever removed, and annually, for deformed, stripped, corroded, or damaged threads. Damaged parts shall be evaluated for continued use and replaced as required.

At a minimum, the cask lid bolts shall be replaced at least every 250 shipments (round trip) to ensure adequate fatigue strength is maintained.

2.8.2.3.2 Impact limiters

A visual examination of the impact limiters prior to each shipment will be performed to ensure that the impact limiters have not been degraded between leakage test intervals. If there is no evidence of weld cracking or other damage that could result in water-intrusion, the wood will not be degraded. If there is visual damage; the impact limiter shall be removed from service, repaired, if possible, and inspected for degradation of the wood. Impact limiters shall be leakage tested once every five years to ensure that water has not entered the impact limiters. If the leakage test indicates that an impact limiter(s) have a leak, a humidity test will be performed to verify that there is no free water in the impact limiters.

3.0 COSTS

Costs estimates and their basis for the design and analysis, licensing, and fabrication of the 6625B-HB Cask, DFCs, and Auxiliary Loading Equipment are summarized below. As the provided unit prices are developed from actual rates, unit hours and/or pricing, which is considered proprietary to AREVA, only estimated monetary ranges for the summary unit prices are provided; no proprietary data is provided in the below estimate.

The estimated monetary range is defined as:

- "Low" descriptor of monetary range meaning lowest dollar value;
- "High" descriptor of monetary range meaning highest dollar value.

It should also be understood that the low and high unit prices do not reflect the impact of risk evaluation. Although it could be argued that lower cost could reflect higher risk or that higher cost reflects lower risk, in the data used for this evaluation, the fabricators that were used are fully qualified under a vendor qualification program, have audited/accepted/monitored quality assurance programs, and routinely perform similar fabrication projects successfully.

All monetary pricing data was originally in U.S. dollars; therefore, the unit prices below are in U.S. dollars with no conversion factors or approximations impacting their value. Finally, all unit price estimates for labor-based activities are based on actual hours applied against a 2015 rate structure.

The development of cost estimates for the 6625B-HB Cask is provided by design phase:

- Design / Analyses
  - Based on recent experience with two similar designs, minimal modifications were applied for modification to 6625B-HB current requirements.
  - The estimated monetary range was derived from actual cost data.
- Testing Costs
  - In the past, testing costs were reliant on scale testing to benchmark models; however, regulators currently do not accept scale testing for containment. Also, full scale testing is not readily available in the U.S. This places a heavy reliance on explicit dynamic modeling, which was estimated based on previous experience with recent similar designs. An allowance for some bench testing of specialty materials and one-half scale testing of the impact limiters was included in the licensing cost estimate and the licensing costs summary.
- Licensing Costs
  - Licensing cost is based on two recent NRC licensing activities of two similar designs with minimal modification for the complexity of the 6625B-HB cask. Licensing cost includes the NRC application review fees and interface, and estimated costs for both the NRC and AREVA response to requests for additional information (RAI). NRC licensing cost is based on actuals to derive an estimated range that is escalated to January 2015; other estimate ranges were derived from actual cost data.
- Fabrication Costs
  - Based on recent actual experience with two similar designs, three similar cask fabrication purchase orders and two similar cask fabrication quotes were reviewed for needed modification for current requirements. It was determined that the cask

requirements were similar enough that specific modifications were not needed to the reviewed purchase orders and quotes, and that new quotes would not yield any additional information or accuracy. A low and high range was then established from the purchase orders or quotes, and pricing escalated to January 2015. This same methodology was applied to the fabrication of the baskets, DFCs, and auxiliary loading equipment. Oversight management costs including product engineering, quality assurance, and project management support were then estimated derived from actual experience and added to the direct fabricator cost range. Other factors that influence fabrication cost of the 6625B-HB cask include:

- Material Selection and Availability
  - The primary structural material is ASME alloy steel Utilization of impact limiters
  - Gamma shielding material for top and bottom shield plugs is lead and neutron shielding surrounding outer shell of cask is a borated resin compound
  - Metal matrix composite neutron poison material is included in both basket designs
- Fabricator Availability
  - Multiple fabricators are available for fabrication of First-of-A-Kind (FOAK) and continuous production capacity, including available fabricators located outside the U.S.
  - An estimated production rate of four 6625B-HB casks per year is assumed based on current production rates, available capacity, supplier capability, and continuous material availability; no learning is assumed due to fabrication release lot size of one unit
  - Open capacity of fabricators was assumed to be four completed and delivered casks per year based on known self-placed limits on fabricators by their parent companies to meet long-term production and customer service strategies (Note: a dedicated production facility was not considered in this model as the investment analysis and facility conceptual design are beyond the scope of work of this TO)
  - Due to the job-shop manufacturing environment of the fabricators and that most all basket fabrication techniques are repetitive, start-up learning [37] for basket fabrication is considered with a minimal learning rate of 5 percent (95 percent learning curve) through the 16th unit, at which time learning would be considered minimal
  - Start-up learning for DFCs is considered a nominal learning rate of 10 percent (90 percent learning curve) and considered from the 10th unit (based on quotes received from potential fabricators) through production for a total of 10 casks' equivalent of production for both the PWR baskets (total of 240 DFC units) and BWR baskets (total of 640 DFC units)
  - Due to the job-shop nature of fabrication of the auxiliary loading equipment, the various fabricators that are routinely utilized and the

likelihood that fabrication will not occur under a continuous procurement, no learning was considered for these items

- Regulatory and security environment of the 6625B-HB cask and those casks used in the fabrication estimate are considered the same as current requirements and those of the referenced purchase orders and quotes; therefore, no additional physical or testing requirements were included in the estimates. The provided estimates do not consider any "Buy American" requirements.
  - Neither material-only nor material-and-fabrication requirements considered as recent experience has shown that U.S. fabricator pricing would trend towards the high end of the estimated unit pricing ranges.
  - There is a perceived reduced financial risk for utilization of U.S. fabricators vs reduced pricing from non-U.S. fabricators; however, as non-U.S. fabricators are utilized more frequently, this perception is equalizing and perceived risks are becoming more normalized.
  - Program commitment vs corporate investment is not considered. (Note: it is perceived by industry management that a large-scale, project-dedicated fabrication facility in the U.S. would be able to balance competitive fabrication automation, material purchasing initiatives, and required start-up investment to achieve similar or better unit pricing than non-U.S. cask fabricators; however, such a study is beyond the scope of this TO)

Following are estimated costs for the 6625B-HB Cask program adjusted to 2015 dollars. The cost basis and estimates for design and analysis, licensing, and fabrication have been reviewed and verified by AREVA TN North America—a leading supplier for transportable casks and supporting equipment—to substantiate their scope and reasonableness.

Task Order 17 Cost Estimate Summary

		Low Unit Price	High Unit Price
Estimated Costs for Design / Analyses			
Cask Body Design		$1,344,568	$1,568,663
Impact Limiters (includes testing 1/2-scale prototype)		$2,156,288	$2,515,669
PWR Basket		$941,198	$1,098,064
BWR Basket		$537,827	$627,465
Total Cask Design Cost		$4,979,881	$5,809,861
Damaged Fuel Canister		$74,699	$91,297
Needed Auxiliary Equipment Design		$806,741	$941,198
Includes: cradle, lift fixtures, up-righting system, impact limiter removal equipment, personnel barrier, drain & vent tools, vacuum drying equipment, etc.			
Total Cask, DFC and Aux Equip Design Costs		$5,861,321	$6,842,356
Estimated Cost for Licensing			
SAR		$603,928	$905,508
Response to RAIs		$201,387	$302,113
Total Contractor Cost		$805,315	$1,207,621
NRC Review Fees		$848,400	$1,060,500
Total Licensing Cost		$1,653,715	$2,268,121
Note: Assumes no hardware to fabricate or test			
Costs for Fabrication			
Cask Body		$2,877,297	$5,387,598
Impact Limiters (Set of 2)		$893,393	$1,116,741
Total Cask Cost without Basket		$3,770,690	$6,504,339
PWR Basket		$721,614	$902,017
Total Cask Cost w/ PWR Basket - 1st Unit		$4,492,304	$7,234,550
BWR Basket		$584,169	$730,211
Total Cask Cost w/BWR Basket - 1st Unit		$4,354,859	$7,234,550
Damaged Fuel Canisters			
PWR Damaged Fuel Canisters - single unit price		$2,749	$3,436
BWR Damaged Fuel Canisters - single unit price		$2,419	$3,024
Auxiliary Equipment		$652,864	$816,080
Includes: cradle, lifting fixtures, up-righting system, impact limiter removal equipment, personnel barrier, drain & vent tools, vacuum drying equipment, etc.			
Production Learning for Baskets (95% learning curve)[38]	Factor		
PWR Basket			
Total Cask Cost for PWR Basket - 1st Unit	1.0000	$4,492,304	$7,406,356
Total Cask Cost for PWR Basket - 2nd Unit	0.9750	$4,379,996	$7,221,197
Total Cask Cost for PWR Basket - 4th Unit	0.9436	$4,238,938	$6,988,638
Total Cask Cost for PWR Basket - 8th Unit	0.9075	$4,076,766	$6,721,268
Total Cask Cost for PWR Basket - 16th Unit	0.8692	$3,094,711	$6,487,605
BWR Basket			
Total Cask Cost for BWR Basket - 1st Unit	1.0000	$4,354,859	$7,234,550
Total Cask Cost for BWR Basket - 2nd Unit	0.9750	$4,245,988	$7,053,686
Total Cask Cost for BWR Basket - 4th Unit	0.9436	$4,109,245	$6,826,521
Total Cask Cost for BWR Basket - 8th Unit	0.9075	$3,952,035	$6,565,354
Total Cask Cost for BWR Basket - 16th Unit	0.8692	$3,785,243	$6,288,271
Production Learning for Damaged Fuel Canisters (90% learning curve)[38]			
DFC for PWR Basket	Factor		
DFC Unit Cost for PWR Basket - 10th Unit	1.0000	$7,559	$9,449
DFC Unit Cost for PWR Basket - 24th Unit (1 basket)	0.6337	$4,790	$5,988
DFC Unit Cost for PWR Basket - 240th Unit (10 baskets)	0..4966	$3,754	$4,693
DFC for BWR Basket	Factor		
DFC Unit Cost for BWR Basket - 10th Unit	1.0000	$7,559	$9,449
DFC Unit Cost for BWR Basket - 64th Unit (I basket)	0.5804	$4,388	$5,484
DFC Unit Cost for BWR Basket - 640th Unit (10 baskets)	0.4347	$3,286	$4,108

4.0 CONCEPT OF OPERATIONS

The concept of operations for the 6625B-HB, including the methodology, operational steps, associated estimated person hours, and personnel dose are described in this section. The analysis is based on the 6625B-HB cask system using a 24 PWR FA basket and bare design basis fuel as described in *Section 2.5*. The 6625B-HB BWR cask was shown to have essentially the same dose rate field as a function of distance and position around the cask as compared to the PWR cask, so the PWR cask dose rate field was used to compute dose rates around the BWR cask. The DFC material was assumed to have no impact on the shielding behavior of the cask system; therefore, the bare fuel dose rate field was also used to compute dose rates for the DFC cask analysis.

The described operations process is essentially identical for the 24P and 61B basket types, except for the actual time required to move 61 FAs into the BWR basket compared to the actual time required to move 24 FAs into the PWR basket. The use of DFCs will add processing time to the loading operations. Two options are available for loading DFCs into the 6625B-HB cask. Option 1 has the DFCs preloaded with selected FAs and the DFC lids installed prior to the commencement of the cask loading operation. Since the loading and lid installation is not performed as part of the cask loading evolution, the time and dose associated with loading and capping the DFCs is not accounted for with Option 1. Option 2 has the empty DFCs placed into the basket during the cask loading operation. Since the loading and lid installation is performed as part of the cask loading evolution, the time and dose associated with loading and capping the DFCs is accounted for with Option 2. Both options are detailed in this section for both PWR and BWR baskets.

Operational steps are developed based on the AREVA TN-68 [35] cask system, which allows for the use of proven operational steps that have been benchmarked against known dose measurements for actual loading campaigns. The following activities are performed to evaluate the operational dose rates, personnel hours, and shift requirements:

- The required steps for each operation have been identified; loading, transportation, unloading, and return of empty cask.
- The required time for each operational step has been estimated.
- The required number of workers to perform each operational step and the cumulative person hours needed to perform the operations has been estimated.
- The PWR shielding models have been developed in *Section 2.5*.
- The operational process dose rates and cumulative doses have been calculated using the shielding models and operations data described above

The results of the initial development of the operational steps, related person hours, shift requirements, and cumulative doses are included in the following tables which correspond to the operating procedures summarized in *Section 2.7*:

- **Table 4.0-1**: Summary of PWR Package Operating Cycle Cumulative Data
- **Table 4.0-2**: Summary of BWR Package Operating Cycle Cumulative Data
- **Table 4.0-3**: Prepare Empty Package for Loading
- **Table 4.0-4**: Package Loading: 24 Bare PWR Fuel Assemblies
- **Table 4.0-5**: Package Loading: PWR DFC Option 1
- **Table 4.0-6**: Package Loading: PWR DFC Option 2

- **Table 4.0-7**: Package Loading: 61 Bare BWR Fuel Assemblies
- **Table 4.0-8**: Package Loading: BWR DFC Option 1
- **Table 4.0-9**: Package Loading: BWR DFC Option 2
- **Table 4.0-10**: Prepare Loaded Package for Transport
- **Table 4.0-11**: Receive Loaded Package from Carrier
- **Table 4.0-12**: Prepare to Unload the Loaded Package
- **Table 4.0-13**: Remove PWR Contents
- **Table 4.0-14**: Remove BWR Contents
- **Table 4.0-15**: Prepare Internally Contaminated Empty Package for Transport

The total values for the entire package operating cycle as summarized in **Table 4.0-1** and **Table 4.0-2** result in the following:

- Total Dose
    - 1002 mRem for PWR Bare fuel
    - 1002 mRem for PWR DFC Option 1
    - 1003 mRem for PWR DFC Option 2
    - 1003 mRem for BWR Bare fuel
    - 1003 mRem for BWR DFC Option 1
    - 1004 mRem for BWR DFC Option 2
- Total person hours
    - 152 for PWR Bare fuel
    - 157 for PWR DFC Option 1
    - 171 for PWR DFC Option 2
    - 197 for BWR Bare fuel
    - 209 for BWR DFC Option 1
    - 245 for BWR DFC Option 2
- Number of eight hour shifts to complete the evolution
    - 9.9 Shifts for PWR Bare fuel
    - 10.2 Shifts for PWR DFC Option 1
    - 11.1 Shifts for PWR DFC Option 2
    - 12.7 Shifts for BWR Bare fuel
    - 13.4 Shifts for BWR DFC Option 1
    - 15.7 Shifts for BWR DFC Option 2

The following conclusions are based on the above evaluation information.

- DOSE:
    - Cumulative dose does not vary appreciably by basket type or DFC option (~1003 mRem average for all configurations)
    - Since the DFCs and their closure lids, and the associated fuel compartment basket spacers are handled either remotely or near an empty cask, there is no appreciable dose consequence to using either DFC option.
- TIME:
    - When comparing the loading of bare fuel to DFC Option 1 (preloading DFCs prior to commencing the cask loading evolution), the impact in required shift time is small and is primarily due to DFC lid installation.

- When comparing DFC Option 1 to DFC Option 2 (loading empty DFCs into the cask, then loading designated FAs into the DFCs), Option 2 increases shift time by:
  - ~1 shift for PWR
  - ~2.3 shifts for BWR

TABLE 4.0-1: SUMMARY OF PWR PACKAGE OPERATING CYCLE CUMULATIVE DATA

Section Number	2.7.1.1	2.7.1.2			2.7.1.3	Steps 2.7.1.1 through 2.7.1.3		
Section Description	Prepare Empty Package for Loading	Package Loading (24 PWR Bare Fuel Assemblies)	Package Loading (24 Preloaded PWR DFCs – DFC Option 1)	Package Loading (24 Empty PWR DFCs Loaded While in Cask – DFC Option 2)	Prepare Loaded Package for Transport	Total at Utility Bare Fuel	Total at Utility DFC Option 1	Total at Utility DFC Option 2
Total Dose (mRem)	0.1	58.5	58.6	58.9	546.0	604.7	604.8	605.1
Total Person Hours (# Persons*hours)	21.1	21.7	26.5	40.9	34.5	77.2	82.0	96.4
Total Clock Hours	8.3	11.3	13.7	20.9	17.7	37.3	39.7	46.9
Total Number of 8 hour Shifts	1.0	1.4	1.7	2.6	2.2	4.7	5.0	5.9
Section Number	2.7.2.1	2.7.2.2	2.7.2.3	2.7.3.0	Steps 2.7.2.1 - 2.7.3.0	Complete Process Totals		
Section Description	Receive Loaded Package from Carrier	Prepare to Unload The Loaded Package	Remove PWR Contents	Prepare Internally Contaminated Empty Package For Transport	Total at ISF	PWR Basket with Bare Fuel	PWR Basket with DFC Option 1	PWR Basket with DFC Option 2
Total Dose (mRem)	196.0	177.6	24.0	0.0	397.6	1002.2	1002.3	1002.6
Total Person Hours (# Persons*hours)	11.2	10.7	23.1	30.0	75.0	152.2	157.0	171.4
Total Clock Hours	6.3	6.1	14.3	15.2	41.9	79.1	81.5	88.7
Total Number of 8 hour Shifts	0.8	0.8	1.8	1.9	5.2	9.9	10.2	11.1

TABLE 4.0-2: SUMMARY OF BWR PACKAGE OPERATING CYCLE CUMULATIVE DATA

Section Number	2.7.1.1	2.7.1.2			2.7.1.3	Steps 2.7.1.1 through 2.7.1.3		
Section Description	Prepare Empty Package for Loading	Package Loading (61 BWR Bare Fuel Assemblies)	Package Loading (61 Preloaded BWR DFCs – DFC Option 1)	Package Loading (61 Empty BWR DFCs Loaded While in Cask – DFC Option 2)	Prepare Loaded Package for Transport	Total at Utility Bare Fuel	Total at Utility DFC Option 1	Total at Utility DFC Option 2
Total Dose (mRem)	0.1	58.9	59.2	59.9	546.0	605.1	605.4	606.1
Total Person Hours (# Persons*hours)	21.1	43.9	56.1	92.7	34.5	99.4	111.6	148.2
Total Clock Hours	8.3	22.4	28.5	46.8	17.7	48.4	54.5	72.8
Total Number of 8 hour Shifts	1.0	2.8	3.6	5.9	2.2	6.0	6.8	9.1
Section Number	2.7.2.1	2.7.2.2	2.7.2.3	2.7.3.0	Steps 2.7.2.1 - 2.7.3.0	Complete Process Totals		
Section Description	Receive Loaded Package from Carrier	Prepare to Unload The Loaded Package	Remove BWR Contents	Prepare Internally Contaminated Empty Package For Transport	Total at ISF	BWR Basket with Bare Fuel	BWR Basket with DFC Option 1	BWR Basket with DFC Option 2
Total Dose (mRem)	196.0	177.6	24.4	0.0	398.0	1003.1	1003.4	1004.1
Total Person Hours (# Persons*hours)	11.2	10.7	45.3	30.0	97.2	196.6	208.8	245.4
Total Clock Hours	6.3	6.1	25.4	15.2	53.0	101.3	107.4	125.7
Total Number of 8 hour Shifts	0.8	0.8	3.2	1.9	6.6	12.7	13.4	15.7

Table 4.0-3 details the operating steps for receiving the empty package at the utility site and preparing it for loading. The table includes the maximum estimated total average dose rates and person hours, and the maximum estimated average cumulative dose expected for completion of the evolution. **Table 4.0-3** corresponds to the operating procedures summary described in *Section 2.7.1.1*: *Preparation for Loading*.

TABLE 4.0-3: PREPARE EMPTY PACKAGE FOR LOADING: OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.1	**Prepare Empty Package for Loading**								
2.7.1.1.01	Upon arrival of the empty packaging, on its transport vehicle (rail or heavy haul trailer) and shipping frame, perform a receipt inspection to check for any damages or irregularities. Verify that the records for the packaging are complete and accurate.	Empty Transportation	3	0.5	1.5	Side	3	0.00	0.00
2.7.1.1.02	Remove the security device, the impact limiter attachment bolts, tie-rods, and the associated hardware, as necessary.	Empty Transportation	2	0.3	0.6	Side	1	0.00	0.00
2.7.1.1.03	Remove the front and the rear impact limiters.	Empty No Impact Limiters	6	1	6	Side	3	0.00	0.00
2.7.1.1.04	Remove the tie-down strap and trunnion support block caps.	Empty No Impact Limiters	2	0.3	0.6	Side	1	0.00	0.00
2.7.1.1.05	Clean the external surfaces of the cask, if necessary, to get rid of the road dirt.	Empty No Impact Limiters	2	0.3	0.6	Side	1	0.00	0.00
2.7.1.1.06	Attach the Top and Bottom Trunnions to the Cask.	Empty No Impact Limiters	2	0.5	1	Top Corner	1	0.00	0.00
			2	0.5	1	Bottom Corner	1	0.00	0.00
2.7.1.1.07	Attach the lift beam to the cask handling crane hook, and engage the lift beam to the two upper (top) trunnions.	Empty No Impact Limiters	1	1	1	Remote	N/A	0.00	0.00
2.7.1.1.08	Rotate the cask slowly from the horizontal to the vertical position.	Empty No Impact Limiters	5	0.1	0.5	Side	3	0.00	0.00
2.7.1.1.09	Lift the cask from the transport/shipping frame and place it in the cask preparation area.	Empty No Impact Limiters	5	0.2	1	Side	3	0.00	0.00

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.1.10	Disengage the lift beam from the cask.	Empty No Impact Limiters	5	0.1	0.5	Side	3	0.02	0.01
2.7.1.1.11	Remove the neutron shield pressure relief valve and install the plug in the neutron shield vent hole.	Empty No Impact Limiters	2	0.5	1	Corner	1	0.02	0.02
2.7.1.1.12	Remove the outer lid bolts and the outer lid.	Empty No Outer Lid	2	1.2	2.4	Corner	1	0.02	0.05
2.7.1.1.13	Remove the inner lid bolts and the inner lid.	Empty No Lids	2	1.2	2.4	Corner	1	0.02	0.05
2.7.1.1.14	Replace the inner lid seal using the retaining screws, and inspect the lid sealing surface. Check for defects in the seal contact areas that may prevent a proper seal. (This step may be performed at any time prior to installing the inner lid on the loaded cask).	Empty No Lids	2	0.2	0.4	Corner	1	0.02	0.01
2.7.1.1.15	Replace the seals in the vent, drain and transport covers, and inspect the sealing surfaces. Check for defects in the seal contact areas that may prevent a proper sealing. (This step may be performed at any time prior to installing covers on the loaded cask).	Empty No Lids	2	0.2	0.4	Corner	1	0.02	0.01
2.7.1.1.16	Remove the hold down ring from the cask cavity.	Empty No Lids	1	0.1	0.1	-	-	0.02	0.00
2.7.1.1.17	Verify that the basket is installed in the cask, with no evident signs of damage to either. Verify that there is no foreign material in the cask.	Empty No Lids	1	0.1	0.1	-	-	0.02	0.00
			Total Person Hours		21.10		Total Dose	Total (mRem)	0.15
								Total (Rem)	1.46E-04

Table 4.0-4 details the operating steps for moving the empty package into the SFP and loading it with 24 PWR bare fuel assemblies. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-4** corresponds to the operating procedures summary described in *Section 2.7.1.2: Loading*.

TABLE 4.0-4: PACKAGE LOADING (24 PWR BARE FUEL ASSEMBLIES): OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2	**Package Loading (24 PWR Bare Fuel Assemblies)**								
2.7.1.2.01	Move the cask to the cask loading area using the lift beam attached to the top trunnions.	Empty No Lids	5	0.1	0.5	Remote	N/A	0.02	0.01
2.7.1.2.02	Lower the cask into the cask loading pool while rinsing the exterior of the cask with demineralized water and filling the interior with demineralized or pool water.	Empty No Lids	2	0.1	0.2	Remote	N/A	0.02	0.00
2.7.1.2.03	Disengage the lift beam and move it aside.	Empty No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.04	Load the 24 pre-selected spent fuel assemblies into the basket compartments. Procedures shall be developed to ensure that the fuel loaded into the cask meets the fuel specifications in the authorized contents chapter of the SAR.	Loaded No Lids	2	7.2	14.4	Remote	N/A	0.02	0.29
2.7.1.2.05	Verify the identity of the 24 fuel assemblies loaded into the cask, and document the location of each fuel assembly on the cask loading report.	Loaded No Lids	2	1	2	Remote	N/A	0.02	0.04

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2.06	Configure the lid prior to installation so that water may be drained through the drain port and that helium can be used to back fill the cask as the water is drained. Using the lift beam and the lid lifting slings, lower the inner lid placing it on the cask body flange over the alignment pins with at least one lid penetration (drain or vent port) open.	Loaded No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.07	Engage the lift beam on the upper (top) trunnions, and lift the cask so that the top of the cask is above the water surface in the pool, and install some of the lid bolts. The lid bolts should be hand tight.	Loaded No Outer Lid	1	0.75	0.75	Top Corner	1	19.63	14.73
2.7.1.2.08	Using the drain port in the inner lid, drain the water from the cask. In order to minimize internal hydrogen accumulation, the cask should be drained completely within 18 hours of the start of draining. If this period is exceeded, the cask cavity should be inerted by injecting helium through the open lid penetration, while the draining continues. The cask is drained by connecting one end of a drain hose to the Hansen coupling in the drain port and routing the other to a pump. This must be done while lifting the cask out of the pool, unless the maximum lifting weight is not exceeded. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.	Loaded No Outer Lid	1.5	0.5	0.75	Top Corner	1	19.63	14.73
2.7.1.2.09	Disconnect the drain line.	Loaded No Outer Lid	1	0.25	0.25	Top Corner	0.5	36.62	9.15

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2.10	Move the cask to the decontamination area and disengage the lift beam.	Loaded No Outer Lid	2	1	2	Side	2	9.77	19.53
			Total Person Hours		21.65		Total Dose	Total (mRem)	58.49
								Total (Rem)	0.06

This section studies the use of PWR DFC loading Option 1, which has the 24 PWR DFCs preloaded with selected FAs and the DFC lids installed prior to the commencement of the cask loading operation. The time and dose associated with loading and installing the lids for the DFCs is not accounted for with Option 1, as these operations occur prior to the start of the cask loading evolution. **Table 4.0-5** details the operating steps for moving the empty package into the SFP and loading it with 24 preloaded and capped PWR DFCs. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-5** corresponds to the operating procedures summary described in *Section 2.7.1.2: Loading*.

TABLE 4.0-5: PACKAGE LOADING (24 PRELOADED PWR DFCS – DFC OPTION 1): OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2	**Package Loading (24 Preloaded PWR DFCs – DFC Option 1)**								
2.7.1.2.01	Remove the 24 removable sleeves from the basket in preparation for loading the DFCs.	Empty No Lids	2	2.4	4.8	Remote	N/A	0.02	0.10
2.7.1.2.02	Move the cask to the cask loading area using the lift beam attached to the top trunnions.	Empty No Lids	5	0.1	0.5	Remote	N/A	0.02	0.01
2.7.1.2.03	Lower the cask into the cask loading pool while rinsing the exterior of the cask with demineralized water and filling the interior with demineralized or pool water.	Empty No Lids	2	0.1	0.2	Remote	N/A	0.02	0.00
2.7.1.2.04	Disengage the lift beam and move it aside.	Empty No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.05	Load the 24 pre-selected, pre-loaded, closed DFCs into the basket compartments. Procedures shall be developed to ensure that the fuel loaded into the cask meets the fuel specifications in the authorized contents chapter of the SAR.	Loaded No Lids	2	7.2	14.4	Remote	N/A	0.02	0.29
2.7.1.2.06	Verify the identity of the 24 loaded DFCs loaded into the cask, and document the location of each loaded DFC on the cask loading report.	Loaded No Lids	2	1	2	Remote	N/A	0.02	0.04

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2.07	Configure the lid prior to installation so that water may be drained through the drain port and that helium can be used to back fill the cask as the water is drained. Using the lift beam and the lid lifting slings, lower the inner lid placing it on the cask body flange over the alignment pins with at least one lid penetration (drain or vent port) open.	Loaded No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.08	Engage the lift beam on the upper (top) trunnions, and lift the cask so that the top of the cask is above the water surface in the pool, and install some of the lid bolts. The lid bolts should be hand tight.	Loaded No Outer Lid	1	0.75	0.75	Top Corner	1	19.63	14.73
2.7.1.2.09	Using the drain port in the inner lid, drain the water from the cask. In order to minimize internal hydrogen accumulation, the cask should be drained completely within 18 hours of the start of draining. If this period is exceeded, the cask cavity should be inerted by injecting helium through the open lid penetration, while the draining continues. The cask is drained by connecting one end of a drain hose to the Hansen coupling in the drain port and routing the other to a pump. This must be done while lifting the cask out of the pool, unless the maximum lifting weight is not exceeded. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.	Loaded No Outer Lid	1.5	0.5	0.75	Top Corner	1	19.63	14.73
2.7.1.2.10	Disconnect the drain line.	Loaded No Outer Lid	1	0.25	0.25	Top Corner	0.5	36.62	9.15
2.7.1.2.11	Move the cask to the decontamination area and disengage the lift beam.	Loaded No Outer Lid	2	1	2	Side	2	9.77	19.53
			Total Person Hours		26.45		Total Dose	Total (mRem)	58.59
								Total (Rem)	0.06

This section studies the use of PWR DFC loading Option 2, which has the 24 empty PWR DFCs placed into the basket during the cask loading operation. The time and dose associated with loading and installing the lids for the DFCs is accounted for with Option 2. **Table 4.0-6** details the operating steps for moving the empty package into the SFP and loading it with 24 preloaded and capped PWR DFCs. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-6** corresponds to the operating procedures summary described in *Section 2.7.1.2: Loading*.

TABLE 4.0-6: PACKAGE LOADING (24 EMPTY PWR DFCS LOADED WHILE IN THE CASK – DFC OPTION 2): OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2	**Package Loading (24 Empty PWR DFCs Loaded While in the Cask – DFC Option 2)**								
2.7.1.2.01	Remove the 24 removable sleeves from the basket.	Empty No Lids	2	2.4	4.8	Remote	N/A	0.02	0.10
2.7.1.2.02	Insert the 24 empty DFCs into the fuel compartments of the basket.	Empty No Lids	2	2.4	4.8	Remote	N/A	0.02	0.10
2.7.1.2.03	Move the cask to the cask loading area using the lift beam attached to the top trunnions.	Empty No Lids	5	0.1	0.5	Remote	N/A	0.02	0.01
2.7.1.2.04	Lower the cask into the cask loading pool while rinsing the exterior of the cask with demineralized water and filling the interior with demineralized or pool water.	Empty No Lids	2	0.1	0.2	Remote	N/A	0.02	0.00
2.7.1.2.05	Disengage the lift beam and move it aside.	Empty No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.06	Load the 24 pre-selected fuel assemblies into the DFCs in the basket compartments. Procedures shall be developed to ensure that the fuel loaded into the cask meets the fuel specifications in the authorized contents chapter of the SAR.	Loaded No Lids	2	7.2	14.4	Remote	N/A	0.02	0.29
2.7.1.2.07	Install the 24 DFC lids onto the DFCs.	Loaded No Lids	2	4.8	9.6	Remote	N/A	0.02	0.19
2.7.1.2.08	Verify the identity of the 24 loaded DFCs loaded into the cask, and document the location of each loaded DFC on the cask loading report.	Loaded No Lids	2	1	2	Remote	N/A	0.02	0.04

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2.09	Configure the lid prior to installation so that water may be drained through the drain port and that helium can be used to back fill the cask as the water is drained. Using the lift beam and the lid lifting slings, lower the inner lid placing it on the cask body flange over the alignment pins with at least one lid penetration (drain or vent port) open.	Loaded No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.10	Engage the lift beam on the upper (top) trunnions, and lift the cask so that the top of the cask is above the water surface in the pool, and install some of the lid bolts. The lid bolts should be hand tight.	Loaded No Outer Lid	1	0.75	0.75	Top Corner	1	19.63	14.73
2.7.1.2.11	Using the drain port in the inner lid, drain the water from the cask. In order to minimize internal hydrogen accumulation, the cask should be drained completely within 18 hours of the start of draining. If this period is exceeded, the cask cavity should be inerted by injecting helium through the open lid penetration, while the draining continues. The cask is drained by connecting one end of a drain hose to the Hansen coupling in the drain port and routing the other to a pump. This must be done while lifting the cask out of the pool, unless the maximum lifting weight is not exceeded. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.	Loaded No Outer Lid	1.5	0.5	0.75	Top Corner	1	19.63	14.73
2.7.1.2.12	Disconnect the drain line.	Loaded No Outer Lid	1	0.25	0.25	Top Corner	0.5	36.62	9.15
2.7.1.2.13	Move the cask to the decontamination area and disengage the lift beam.	Loaded No Outer Lid	2	1	2	Side	2	9.77	19.53
			Total Person Hours		40.85		Total Dose	Total (mRem)	58.88
								Total (Rem)	0.06

Table 4.0-7 details the operating steps for moving the empty package into the SFP and loading it with 61 BWR bare fuel assemblies. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-7** corresponds to the operating procedures summary described in *Section 2.7.1.2: Loading*.

TABLE 4.0-7: PACKAGE LOADING (61 BWR BARE FUEL ASSEMBLIES): OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2	**Package Loading (61 BWR Bare Fuel Assemblies)**								
2.7.1.2.01	Move the cask to the cask loading area using the lift beam attached to the top trunnions.	Empty No Lids	5	0.1	0.5	Remote	N/A	0.02	0.01
2.7.1.2.02	Lower the cask into the cask loading pool while rinsing the exterior of the cask with demineralized water and filling the interior with demineralized or pool water.	Empty No Lids	2	0.1	0.2	Remote	N/A	0.02	0.00
2.7.1.2.03	Disengage the lift beam and move it aside.	Empty No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.04	Load the 61 pre-selected spent fuel assemblies into the basket compartments. Procedures shall be developed to ensure that the fuel loaded into the cask meets the fuel specifications in the authorized contents chapter of the SAR.	Loaded No Lids	2	18.3	36.6	Remote	N/A	0.02	0.73
2.7.1.2.05	Verify the identity of the 61 fuel assemblies loaded into the cask, and document the location of each fuel assembly on the cask loading report.	Loaded No Lids	2	1	2	Remote	N/A	0.02	0.04
2.7.1.2.06	Configure the lid prior to installation so that water may be drained through the drain port and that helium can be used to back fill the cask as the water is drained. Using the lift beam and the lid lifting slings, lower the inner lid placing it on the cask body flange over the alignment pins with at least one lid penetration (drain or vent port) open.	Loaded No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.07	Engage the lift beam on the upper (top) trunnions, and lift the cask so that the top of the cask is above the water surface in the pool, and install some of the lid bolts. The lid bolts should be hand tight.	Loaded No Outer Lid	1	0.75	0.75	Top Corner	1	19.63	14.73

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2.08	Using the drain port in the inner lid, drain the water from the cask. In order to minimize internal hydrogen accumulation, the cask should be drained completely within 18 hours of the start of draining. If this period is exceeded, the cask cavity should be inerted by injecting helium through the open lid penetration, while the draining continues. The cask is drained by connecting one end of a drain hose to the Hansen coupling in the drain port and routing the other to a pump. This must be done while lifting the cask out of the pool, unless the maximum lifting weight is not exceeded. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.	Loaded No Outer Lid	1.5	0.5	0.75	Top Corner	1	19.63	14.73
2.7.1.2.09	Disconnect the drain line.	Loaded No Outer Lid	1	0.25	0.25	Top Corner	0.5	36.62	9.15
2.7.1.2.10	Move the cask to the decontamination area and disengage the lift beam.	Loaded No Outer Lid	2	1	2	Side	2	9.77	19.53
			Total Person Hours		43.85		Total Dose	Total (mRem)	58.94
								Total (Rem)	0.06

This section studies the use of BWR DFC loading Option 1, which has the 61 BWR DFCs preloaded with selected FAs and the DFC lids installed prior to the commencement of the cask loading operation. The time and dose associated with loading and installing the lids for the DFCs is not accounted for with Option 1, as these operations occur prior to the start of the cask loading evolution. **Table 4.0-8** details the operating steps for moving the empty package into the SFP and loading it with 61 preloaded and capped BWR DFCs. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-8** corresponds to the operating procedures summary described in *Section 2.7.1.2: Loading*.

TABLE 4.0-8: PACKAGE LOADING (61 PRELOADED BWR DFCS – DFC OPTION 1): OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2	**Package Loading (61 Preloaded BWR DFCs – DFC Option 1)**								
2.7.1.2.01	Remove the 61 removable sleeves from the basket in preparation for loading the DFCs.	Empty No Lids	2	6.1	12.2	Remote	N/A	0.02	0.24
2.7.1.2.02	Move the cask to the cask loading area using the lift beam attached to the top trunnions.	Empty No Lids	5	0.1	0.5	Remote	N/A	0.02	0.01
2.7.1.2.03	Lower the cask into the cask loading pool while rinsing the exterior of the cask with demineralized water and filling the interior with demineralized or pool water.	Empty No Lids	2	0.1	0.2	Remote	N/A	0.02	0.00
2.7.1.2.04	Disengage the lift beam and move it aside.	Empty No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.05	Load the 61 pre-selected, pre-loaded, closed DFCs into the basket compartments. Procedures shall be developed to ensure that the fuel loaded into the cask meets the fuel specifications in the authorized contents chapter of the SAR.	Loaded No Lids	2	18.3	36.6	Remote	N/A	0.02	0.73
2.7.1.2.06	Verify the identity of the 61 loaded DFCs loaded into the cask, and document the location of each loaded DFC on the cask loading report.	Loaded No Lids	2	1	2	Remote	N/A	0.02	0.04
2.7.1.2.07	Configure the lid prior to installation so that water may be drained through the drain port and that helium can be used to back fill the cask as the water is drained. Using the lift beam and the lid lifting slings, lower the inner lid placing it on the cask body flange over the alignment pins with at least one lid penetration (drain or vent port) open.	Loaded No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2.08	Engage the lift beam on the upper (top) trunnions, and lift the cask so that the top of the cask is above the water surface in the pool, and install some of the lid bolts. The lid bolts should be hand tight.	Loaded No Outer Lid	1	0.75	0.75	Top Corner	1	19.63	14.73
2.7.1.2.09	Using the drain port in the inner lid, drain the water from the cask. In order to minimize internal hydrogen accumulation, the cask should be drained completely within 18 hours of the start of draining. If this period is exceeded, the cask cavity should be inerted by injecting helium through the open lid penetration, while the draining continues. The cask is drained by connecting one end of a drain hose to the Hansen coupling in the drain port and routing the other to a pump. This must be done while lifting the cask out of the pool, unless the maximum lifting weight is not exceeded. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.	Loaded No Outer Lid	1.5	0.5	0.75	Top Corner	1	19.63	14.73
2.7.1.2.10	Disconnect the drain line.	Loaded No Outer Lid	1	0.25	0.25	Top Corner	0.5	36.62	9.15
2.7.1.2.11	Move the cask to the decontamination area and disengage the lift beam.	Loaded No Outer Lid	2	1	2	Side	2	9.77	19.53
			Total Person Hours		56.05		Total Dose	Total (mRem)	59.18
								Total (Rem)	0.06

This section studies the use of BWR DFC loading Option 2, which has the 61 empty BWR DFCs placed into the basket during the cask loading operation. The time and dose associated with loading and installing the lids for the DFCs is accounted for with Option 2. **Table 4.0-9** details the operating steps for moving the empty package into the SFP and loading it with 61 preloaded and capped BWR DFCs. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-9** corresponds to the operating procedures summary described in *Section 2.7.1.2: Loading*.

TABLE 4.0-9: PACKAGE LOADING (61 EMPTY BWR DFCS LOADED WHILE IN THE CASK – DFC OPTION 2): OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2	**Package Loading (61 Empty BWR DFCs Loaded While In Cask – DFC Option 2)**								
2.7.1.2.01	Remove the 61 removable sleeves from the basket.	Empty No Lids	2	6.1	12.2	Remote	N/A	0.02	0.24
2.7.1.2.02	Insert the 61 empty DFCs into the fuel compartments of the basket.	Empty No Lids	2	6.1	12.2	Remote	N/A	0.02	0.24
2.7.1.2.03	Move the cask to the cask loading area using the lift beam attached to the top trunnions.	Empty No Lids	5	0.1	0.5	Remote	N/A	0.02	0.01
2.7.1.2.04	Lower the cask into the cask loading pool while rinsing the exterior of the cask with demineralized water and filling the interior with demineralized or pool water.	Empty No Lids	2	0.1	0.2	Remote	N/A	0.02	0.00
2.7.1.2.05	Disengage the lift beam and move it aside.	Empty No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.06	Load the 61 pre-selected fuel assemblies into the DFCs in the basket compartments. Procedures shall be developed to ensure that the fuel loaded into the cask meets the fuel specifications in the authorized contents chapter of the SAR.	Loaded No Lids	2	18.3	36.6	Remote	N/A	0.02	0.73
2.7.1.2.07	Install the 61 DFC lids onto the DFCs.	Loaded No Lids	2	12.2	24.4	Remote	N/A	0.02	0.49
2.7.1.2.08	Verify the identity of the 24 loaded DFCs loaded into the cask, and document the location of each loaded DFC on the cask loading report.	Loaded No Lids	2	1	2	Remote	N/A	0.02	0.04

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.2.09	Configure the lid prior to installation so that water may be drained through the drain port and that helium can be used to back fill the cask as the water is drained. Using the lift beam and the lid lifting slings, lower the inner lid placing it on the cask body flange over the alignment pins with at least one lid penetration (drain or vent port) open.	Loaded No Lids	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.1.2.10	Engage the lift beam on the upper (top) trunnions, and lift the cask so that the top of the cask is above the water surface in the pool, and install some of the lid bolts. The lid bolts should be hand tight.	Loaded No Outer Lid	1	0.75	0.75	Top Corner	1	19.63	14.73
2.7.1.2.11	Using the drain port in the inner lid, drain the water from the cask. In order to minimize internal hydrogen accumulation, the cask should be drained completely within 18 hours of the start of draining. If this period is exceeded, the cask cavity should be inerted by injecting helium through the open lid penetration, while the draining continues. The cask is drained by connecting one end of a drain hose to the Hansen coupling in the drain port and routing the other to a pump. This must be done while lifting the cask out of the pool, unless the maximum lifting weight is not exceeded. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.	Loaded No Outer Lid	1.5	0.5	0.75	Top Corner	1	19.63	14.73
2.7.1.2.12	Disconnect the drain line.	Loaded No Outer Lid	1	0.25	0.25	Top Corner	0.5	36.62	9.15
2.7.1.2.13	Move the cask to the decontamination area and disengage the lift beam.	Loaded No Outer Lid	2	1	2	Side	2	9.77	19.53
			Total Person Hours		92.65		Total Dose	Total (mRem)	59.91
								Total (Rem)	0.06

Table 4.0-10 details the operating steps for preparing the loaded cask for transport from the utility site to the interim spent fuel storage facility. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-10** corresponds to the operating procedures summary described in *Section 2.7.1.3: Preparation for Transport*.

TABLE 4.0-10: PREPARE LOADED PACKAGE FOR TRANSPORT: OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.3	**Prepare Loaded Package for Transport**								
2.7.1.3.01	Note: During preparation for transport, worker exposure can be minimized by use of temporary shielding and by minimizing the exposure time and maximizing the distance, as well as using any measures to facilitate decontamination. Decontaminate the cask until acceptable surface contamination levels are obtained.	Loaded No Outer Lid	2	1.25	2.5	Side	1	16.85	42.12
2.7.1.3.02	Install the remaining inner lid bolts and torque them to the appropriate first pass specification. Follow the torqueing sequence shown in the figures. Repeat the torqueing process following the sequence of the figures to torque all bolts to the appropriate second pass specification, then the third pass specification, and then the final pass specification.	Loaded No Outer Lid	2	1	2	Top Corner	0.5	36.62	73.23
2.7.1.3.03	Remove the plug from the neutron shield vent, and reinstall the pressure relief valve, making sure that it is operable and set.	Loaded No Outer Lid	1	0.25	0.25	Top Corner	0.5	36.62	9.15
2.7.1.3.04	Evacuate the cask cavity using the Vacuum Drying System (VDS) to remove the remaining moisture, and verify the dryness as follows:								
2.7.1.3.04A	Using a wand attached to the vacuum drying system, remove any excess water from the seal areas through the passageways at the overpressure drain and vent ports.	Loaded No Outer Lid	1	0.125	0.125	Top Corner	0.5	36.62	4.58
2.7.1.3.04B	Remove the quick disconnect from the drying port, and install the drain port cover.	Loaded No Outer Lid	1	1	1	Top Corner	0.5	36.62	36.62

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.3.04C	With the quick disconnect removed to improve evacuation, connect the VDS to a flanged vacuum connector installed over the vent port. Purge or evacuate the helium supply lines and evacuate the cask to the appropriate specification. Make provisions to prevent or correct any icing of the evacuation lines, if necessary.	Loaded No Outer Lid	1	0.125	0.125	Top Corner	0.5	36.62	4.58
2.7.1.3.04D	Isolate the vacuum pump. If, in a specified period, the pressure does not increase to a specified amount, the cask is adequately dried. Otherwise, repeat the vacuum pumping until this criterion is met within the specified time frame.	Loaded No Outer Lid	1	0.5	0.5	Side	1	16.85	8.42
2.7.1.3.04E	Backfill the evacuated cask cavity with helium (minimum 99.99% purity), to slightly above atmospheric pressure. Then, remove the vacuum connector and immediately install the quick disconnect fitting.	Loaded No Outer Lid	1	0.5	0.5	Side	1	16.85	8.42
2.7.1.3.04F	Attach the vacuum/backfill manifold to the vent port fitting, purge or evacuate the helium supply lines, and re-evacuate the cask to below the appropriate specification.	Loaded No Outer Lid	1	0.25	0.25	Top Corner	0.5	36.62	9.15
2.7.1.3.05	Isolate the vacuum pump, and backfill the cask cavity to the appropriate specification with helium (minimum 99.99% purity).	Loaded No Outer Lid	2	1	2	Side	2	9.77	19.53
2.7.1.3.06	Leak test the inner lid, inner vent and drain port cover seals. The maximum acceptable cask seal leak rate is specified somewhere. The leak test shall be performed in accordance with ANSI N14.5. For ports containing quick disconnects, purge the cavity below the cover with helium, at a minimum flow rate of some number of cubic feet per hour for at least some amount of time. Install the port cover. (A partial pressure of at least 50% helium will be obtained under the cover.) Install the vent and drain cover bolts and torque to the appropriate first pass value, then to the appropriate final pass value following the torqueing sequence shown in a figure prior to leak testing.	Loaded No Outer Lid	3	3	9	Top Corner	1	19.63	176.71
2.7.1.3.07	Install the outer lid and Torque the bolts to the value specified.	Loaded No Impact Limiters	2	0.25	0.5	Top Corner	0.5	30.24	15.12

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.3.08	Re-engage the lift beam to the upper (top) trunnions of the cask.	Loaded No Impact Limiters	2	0.2	0.4	Side	3	9.84	3.94
2.7.1.3.09	Move the transport vehicle into the loading position.	Loaded No Impact Limiters	1	0.2	0.2	Remote	N/A	0.02	0.00
2.7.1.3.10	Lift the cask off the decontamination pad, and place the rear trunnions on the rear trunnion supports of the transport frame.	Loaded No Impact Limiters	2	0.2	0.4	Side	3	9.84	3.94
2.7.1.3.11	Rotate the cask from the vertical to the horizontal position.	Loaded No Impact Limiters	2	0.1	0.2	Side	3	9.84	1.97
2.7.1.3.12	Install the lower (bottom) trunnion support caps and the tie-down strap.	Loaded No Impact Limiters	2	0.2	0.4	Side	1	16.96	6.78
2.7.1.3.13	Check if the surface dose rates and the surface contamination levels are within the regulatory limits. Install an optional shield ring adjacent to the top of neutron shield, if required, based on dose limits.	Loaded No Impact Limiters	2	0.5	1	All Around	2	17.52	17.52
2.7.1.3.14	Prior to installing the impact limiters, inspect them visually for damage. The impact limiters may not be used without repair if any wood has been exposed. Damage due to handling other than small dings and scratches must be evaluated for their effect on the performance during the hypothetical drop and puncture accidents.	Loaded No Impact Limiters	2	0.5	1	Remote	N/A	0.02	0.02
2.7.1.3.15	Install the thermal shield on the front end of the cask. Then remove the thermal shield lifting eye bolts.	Loaded No Impact Limiters	2	0.2	0.4	Top Corner	1	19.26	7.71
2.7.1.3.16	Remove the upper and lower trunnions from the cask.	Loaded No Impact Limiters	2	0.5	1	Top Corner	1	19.26	19.26
			2	0.5	1	Bottom Corner	1	22.09	22.09
2.7.1.3.17	Install the front and the rear impact limiters onto the cask. Lubricate the attachment bolts with Never-Seize or an equivalent and torque to the appropriate first pass value, and then to the appropriate final pass value.	Loaded Transportation	2	1	2	Top Corner	1	3.42	6.83
			2	1	2	Bottom Corner	1	3.70	7.39
2.7.1.3.18	Install the impact limiter attachment tie-rods between the front and the rear impact limiters.	Loaded Transportation	2	0.2	0.4	Side	1	10.53	4.21
2.7.1.3.19	Render the impact limiter lifting lugs inoperable by covering the lifting holes or installing a bolt inside the holes to prevent their inadvertent use.	Loaded Transportation	2	0.5	1	Side	1	10.53	10.53
2.7.1.3.20	Install security seal on one tie-rod and lock sleeve.	Loaded Transportation	1	0.2	0.2	Side	1	10.53	2.11

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.1.3.21	Install a transportation enclosure.	Loaded Transportation	2	1	2	Side	2	6.63	13.25
2.7.1.3.22	Check the temperature on all accessible surfaces to make sure that it is <185. F.	Loaded Transportation	1	0.5	0.5	All Around	1	7.71	3.86
2.7.1.3.23	Perform a final radiation and contamination survey to satisfy the shield test requirements and to assure compliance with 10CFR71.47 and 10CFR71.87.	Loaded Transportation	2	0.5	1	All Around	2	5.01	5.01
2.7.1.3.24	Apply appropriate DOT labels and Placards in accordance with 49CFR172. Prepare the final shipping documentation.	Loaded Transportation	2	0.2	0.4	All Around	2	5.01	2.00
2.7.1.3.25	Release the loaded cask for shipment.	Loaded Transportation	1	0.2	0.2	Remote	N/A	0.02	0.00
			Total Person Hours		34.45		Total Dose	Total (mRem)	546.04
								Total (Rem)	0.55

Table 4.0-11 details the operating steps for receiving the loaded package from the carrier. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-11** corresponds to the operating procedures summary described in *Section 2.7.2.1: Receipt of Package from Carrier*.

TABLE 4.0-11: RECEIVE LOADED PACKAGE FROM CARRIER: OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.2.1	**Receive Loaded Package from Carrier**								
2.7.2.1.01	Upon arrival of the loaded cask, perform a receipt inspection of the cask to check for any damage or irregularities. Verify that the security seal is intact, and perform a radiation survey.	Loaded Transportation	1	0.2	0.2	All Around	1	7.71	1.54
2.7.2.1.02	Verify that the records for the packaging are complete and accurate.	Loaded Transportation	1	0.2	0.2	Remote	N/A	0.02	0.00
2.7.2.1.03	Render the impact limiter lifting lugs operable by removing the covering on the lifting holes or the bolt inside the lifting holes, which prevented their inadvertent use.	Loaded Transportation	2	0.25	0.5	Side	0	21.03	10.51
2.7.2.1.04	Remove the security seal.	Loaded Transportation	1	0.25	0.25	Top Corner	1	3.42	0.85
2.7.2.1.05	Remove personnel barrier.	Loaded Transportation	3	0.5	1.5	Side	1	10.53	15.79
2.7.2.1.06	Remove the front and rear impact limiters.	Loaded No Impact Limiters	2	0.5	1	Top Corner	1	19.26	19.26
		Loaded No Impact Limiters	2	0.5	1	Bottom Corner	1	22.09	22.09
2.7.2.1.08	Remove the tie-rods and associated hardware.	Loaded No Impact Limiters	2	0.5	1	Side	1	16.96	16.96
2.7.2.1.09	Remove the tie down strap and trunnion support block caps.	Loaded No Impact Limiters	2	1	2	Side	1	16.96	33.92
2.7.2.1.10	Attach the Top and Bottom Trunnions to the Cask.	Loaded No Impact Limiters	2	0.5	1	Top Corner	1	19.26	19.26
			2	0.5	1	Bottom Corner	1	22.09	22.09
2.7.2.1.11	Attach the lift beam to the cask handling crane hook, and then engage the lift beam to the two upper (top) trunnions.	Loaded No Impact Limiters	2	0.1	0.2	Side	1	16.96	3.39

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.2.1.12	Rotate the cask slowly from the horizontal to the vertical position.	Loaded No Impact Limiters	1	0.2	0.2	Side	2	9.84	1.97
2.7.2.1.13	Lift the cask from the transport/shipping frame, and place it in the decontamination area.	Loaded No Impact Limiters	1	0.2	0.2	Side	3	9.84	1.97
2.7.2.1.14	Disengage the lift beam from the cask, and move the crane as well as the lift beam from the area.	Loaded No Impact Limiters	2	0.1	0.2	Side	1	16.96	3.39
2.7.2.1.15	Clean the external surfaces of the cask, if necessary, to get rid of the road dirt.	Loaded No Impact Limiters	1	0.5	0.5	All Around	1	36.30	18.15
2.7.2.1.16	Remove the neutron shield pressure relief valve, and install the plug in the neutron shield vent hole.	Loaded No Impact Limiters	1	0.25	0.25	Top Corner	1	19.26	4.82
			Total Person Hours		11.20		Total Dose	Total (mRem)	195.96
								Total (Rem)	0.20

Table 4.0-12 details the operating steps for preparing the loaded package to be unloaded. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-12** corresponds to the operating procedures summary described in *Section 2.7.2.2: Preparation for Unloading*.

TABLE 4.0-12: PREPARE TO UNLOAD THE LOADED PACKAGE: OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.2.2	**Prepare to Unload The Loaded Package**								
2.7.2.2.01	Remove the outer lid bolts, engage the lid lifting equipment and remove the outer cask lid.	Loaded No Outer Lid	2	1.2	2.4	Top Corner	1	19.63	47.12
2.7.2.2.02	Remove the vent cover.	Loaded No Outer Lid	2	0.2	0.4	Side	1	16.85	6.74
2.7.2.2.03	Collect a cavity gas sample, through the vent port quick-disconnect coupling, if required.	Loaded No Outer Lid	2	0.2	0.4	Remote	N/A	0.02	0.01
2.7.2.2.04	Analyze the gas sample for radioactive material, and add necessary precautions based on the cavity gas sample results. NOTE: If degraded fuel is suspected, additional measures, appropriate for the specific conditions, are to be planned, reviewed, and approved by the appropriate plant personnel, as well as implemented to minimize worker exposures and radiological releases to the environment. These additional measures may include provisions of filters, respiratory protection equipment, and other methods to control releases and exposures to ALARA.	Loaded No Outer Lid	2	0.5	1	Remote	N/A	0.02	0.02
2.7.2.2.05	In accordance with the site requirements, vent the cavity gas through the hose until atmospheric pressure is reached.	Loaded No Outer Lid	1	0.5	0.5	Top Corner	1	19.63	9.82
2.7.2.2.06	Remove the vent port quick-disconnect and the drain port cover. Attach the vent port adapter. Ensure that appropriate measures are in place for proper handling of steam.	Loaded No Outer Lid	1	0.5	0.5	Top Corner	1	19.63	9.82
2.7.2.2.07	Loosen the inner lid bolts and remove all but six lid bolts, approximately equally spaced.	Loaded No Outer Lid	2	1.2	2.4	Top Corner	1	19.63	47.12

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.2.2.08	Attach the cask to the crane using lift beam. Attach the lid lifting equipment.	Loaded No Outer Lid	2	1.2	2.4	Top Corner	1	19.63	47.12
2.7.2.2.09	Attach the fill and drain lines to the drain quick-disconnect coupling and the vent port adapter.	Loaded No Outer Lid	1	0.5	0.5	Top Corner	1	19.63	9.82
2.7.2.2.10	Lower the cask into the spent fuel pool cask pit, while spraying the exterior of the cask with demineralized water to minimize contamination. Lower the cask until the top surface is just above the water level. Note: The cask may be filled with some water before lowering the cask into the pool or while the cask is partially submerged in the spent fuel pool if the maximum lifting weight is not exceeded. Vent the cavity pressure, and then remove the drain port cover.	Loaded No Outer Lid	2	0.1	0.2	Remote	N/A	0.02	0.00
			Total Person Hours		10.70		Total Dose	Total (mRem)	177.59
								Total (Rem)	0.18

Table 4.0-13 details the operating steps for removing the 24 bare fuel assemblies or DFCs from the PWR package. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-13** corresponds to the operating procedures summary described in *Section 2.7.2.3: Contents Removal*.

TABLE 4.0-13: REMOVE PWR CONTENTS: OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.2.3	**Remove PWR Contents**								
2.7.2.3.01	Begin pumping pool or demineralized water into the cask through the drain port, at a rate of 1 gpm, while continuously monitoring the exit-pressure. Continue pumping the water at a rate of 1 gpm for at least eighty minutes. By this time, the water level in the cask will have reached the active fuel length.	Loaded No Outer Lid	1	1.4	1.4	Remote	N/A	0.02	0.03
2.7.2.3.02	The flow rate can then be gradually increased, while monitoring the pressure at the outlet. If the pressure gage reading exceeds some set pressure, close the inlet valve until the pressure falls below some other specified pressure. Re-flooding can then be resumed.	Loaded No Outer Lid	1	1.4	1.4	Remote	N/A	0.02	0.03
2.7.2.3.03	After verifying that a steady stream of water is coming from the vent line (by checking for bubbles or carefully lifting the hose out of the water), take a sample for chemical analysis.	Loaded No Outer Lid	1	0.2	0.2	Top Corner	1	19.63	3.93
2.7.2.3.04	When the cask is full of water, remove the hose from the drain port, and the hose and the vent port adapter from the vent port. Remove the remaining six lid bolts.	Loaded No Outer Lid	2	0.5	1	Top Corner	1	19.63	19.63
2.7.2.3.05	Lower the cask and place it on the bottom of the pool/pit while rinsing the lift beam with demineralized water.	Loaded No Outer Lid	1	0.5	0.5	Remote	N/A	0.02	0.01
2.7.2.3.06	Raise the lift beam from the cask, removing the cask lid.	Loaded No Lids	1	0.5	0.5	Remote	N/A	0.02	0.01
2.7.2.3.07	Unload the 24 spent fuel assemblies/DFCs in accordance with the site procedures.	Loaded No Lids	2	7.2	14.4	Remote	N/A	0.02	0.29

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.2.3.08	At least one lid penetration must be completely open (both cover and quick disconnect fitting removed) prior to installation of the lid. Using the lift beam and lid lifting slings, lower the lid placing it on the cask body flange, over the two alignment pins.	Empty No Outer Lid	1	0.5	0.5	Remote	N/A	0.02	0.01
2.7.2.3.09	Engage the lift beam on the upper (top) trunnions, and lift the cask out of the pool.	Empty No Outer Lid	1	0.5	0.5	Remote	N/A	0.02	0.01
2.7.2.3.10	Using the drain port in the lid, drain the water from the cask in accordance with the procedures. This is done while lifting the cask out of the pool, unless the maximum lifting capacity of the crane is not exceeded. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.	Empty No Outer Lid	2	1	2	Top Corner	1	0.02	0.04
2.7.2.3.11	Disconnect the drain line from the quick-disconnect couplings.	Empty No Outer Lid	1	0.5	0.5	Top Corner	1	0.02	0.01
2.7.2.3.12	Move the cask to the decontamination area, and disengage the lift beam.	Empty No Outer Lid	2	0.1	0.2	Remote	N/A	0.02	0.00
			Total Person Hours		23.10		Total Dose	Total (mRem)	24.00
								Total (Rem)	0.02

Table 4.0-14 details the operating steps for removing the 61 bare fuel assemblies or DFCs from the BWR package. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-14** corresponds to the operating procedures summary described in *Section 2.7.2.3: Contents Removal*.

TABLE 4.0-14: REMOVE BWR CONTENTS: OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.2.3	**Remove BWR Contents**								
2.7.2.3.01	Begin pumping pool or demineralized water into the cask through the drain port, at a rate of 1 gpm, while continuously monitoring the exit-pressure. Continue pumping the water at a rate of 1 gpm for at least eighty minutes. By this time, the water level in the cask will have reached the active fuel length.	Loaded No Outer Lid	1	1.4	1.4	Remote	N/A	0.02	0.03
2.7.2.3.02	The flow rate can then be gradually increased, while monitoring the pressure at the outlet. If the pressure gage reading exceeds some set pressure, close the inlet valve until the pressure falls below some other specified pressure. Re-flooding can then be resumed.	Loaded No Outer Lid	1	1.4	1.4	Remote	N/A	0.02	0.03
2.7.2.3.03	After verifying that a steady stream of water is coming from the vent line (by checking for bubbles or carefully lifting the hose out of the water), take a sample for chemical analysis.	Loaded No Outer Lid	1	0.2	0.2	Top Corner	1	19.63	3.93
2.7.2.3.04	When the cask is full of water, remove the hose from the drain port, and the hose and the vent port adapter from the vent port. Remove the remaining six lid bolts.	Loaded No Outer Lid	2	0.5	1	Top Corner	1	19.63	19.63
2.7.2.3.05	Lower the cask and place it on the bottom of the pool/pit while rinsing the lift beam with demineralized water.	Loaded No Outer Lid	1	0.5	0.5	Remote	N/A	0.02	0.01
2.7.2.3.06	Raise the lift beam from the cask, removing the cask lid.	Loaded No Lids	1	0.5	0.5	Remote	N/A	0.02	0.01
2.7.2.3.07	Unload the 61 spent fuel assemblies/DFCs in accordance with the site procedures.	Loaded No Lids	2	18.3	36.6	Remote	N/A	0.02	0.73

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.2.3.08	At least one lid penetration must be completely open (both cover and quick disconnect fitting removed) prior to installation of the lid. Using the lift beam and lid lifting slings, lower the lid placing it on the cask body flange, over the two alignment pins.	Empty No Outer Lid	1	0.5	0.5	Remote	N/A	0.02	0.01
2.7.2.3.09	Engage the lift beam on the upper (top) trunnions, and lift the cask out of the pool.	Empty No Outer Lid	1	0.5	0.5	Remote	N/A	0.02	0.01
2.7.2.3.10	Using the drain port in the lid, drain the water from the cask in accordance with the procedures. This is done while lifting the cask out of the pool, unless the maximum lifting capacity of the crane is not exceeded. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.	Empty No Outer Lid	2	1	2	Top Corner	1	0.02	0.04
2.7.2.3.11	Disconnect the drain line from the quick-disconnect couplings.	Empty No Outer Lid	1	0.5	0.5	Top Corner	1	0.02	0.01
2.7.2.3.12	Move the cask to the decontamination area, and disengage the lift beam.	Empty No Outer Lid	2	0.1	0.2	Remote	N/A	0.02	0.00
			Total Person Hours		45.30		Total Dose	Total (mRem)	24.44
								Total (Rem)	0.02

Table 4.0-15 details the operating steps for preparing the used internally contaminated and empty package for transportation back to a utility site. The table includes the maximum estimated total dose rates and person hours, and the maximum estimated cumulative dose expected for completion of the evolution. **Table 4.0-15** corresponds to the operating procedures summary described in *Section 2.7.3: Preparation of Empty Package for Transport*.

TABLE 4-15: PREPARE INTERNALLY CONTAMINATED EMPTY PACKAGE FOR TRANSPORT: OPERATIONAL STEPS, DOSE RATES, PERSON HOURS, AND CUMULATIVE DOSE

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.3.0	**Prepare Internally Contaminated Empty Package For Transport**								
2.7.3.0.01	Decontaminate the cask until acceptable surface contamination levels are obtained.	Empty No Outer Lid	2	1.25	2.5	Side	1	0.00	0.00
2.7.3.0.02	Place the removable sleeve into the cask cavity if needed.	Empty No Outer Lid	1	0.25	0.25	Top Corner	0.5	0.00	0.00
2.7.3.0.03	Lubricate and install the lid bolts and torque them to the specified 1st pass torque. Follow the proper torqueing sequence. Repeat the torqueing process torqueing the bolts to the specified 2nd pass torque. A circular pattern of torqueing may be used to eliminate further bolt movement.	Empty No Outer Lid	2	1	2	Top Corner	0.5	0.00	0.00
2.7.3.0.04	Remove the plug from the neutron shield vent, and reinstall the pressure relief valve, making sure that it is operable and set.	Empty No Outer Lid	1	0.25	0.25	Top Corner	0.5	0.00	0.00
2.7.3.0.05	Evacuate the cask cavity using the Vacuum Drying System (VDS) to remove the remaining moisture, and verify the dryness as follows:								
2.7.3.0.05A	Using a wand attached to the vacuum drying system, remove any excess water from the seal areas through the passageways at the overpressure drain and vent the ports.	Empty No Outer Lid	1	0.125	0.125	Top Corner	0.5	0.00	0.00
2.7.3.0.05B	Remove the quick disconnect from the drain port, and install the drain port cover.	Empty No Outer Lid	1	1	1	Top Corner	0.5	0.00	0.00

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.3.0.05C	With the quick-disconnect removed to improve evacuation, .connect the VDS to a flanged vacuum connector installed over the vent port. Purge or evacuate the helium supply lines and evacuate the cask to 4 millibar (4 x 104 MPa) or less. Make provision to prevent or correct icing of the evacuation lines.	Empty No Outer Lid	1	0.125	0.125	Top Corner	0.5	0.00	0.00
2.7.3.0.05D	Isolate the vacuum pump. If, in a period of 30 minutes, the pressure does not exceed some specified pressure, the cask is adequately dried. Otherwise, repeat vacuum pumping until this criterion is met.	Empty No Outer Lid	1	0.5	0.5	Side	1	0.00	0.00
2.7.3.0.05E	Backfill the evacuated cask cavity with helium (minimum 99.99% purity) to slightly above atmospheric pressure, remove the vacuum connector, and immediately install the quick disconnect fitting.	Empty No Outer Lid	1	0.5	0.5	Side	1	0.00	0.00
2.7.3.0.05F	Attach the vacuum/backfill manifold to the vent port fitting, purge or evacuate the helium supply lines, and re-evacuate the cask to below 100 millibar.	Empty No Outer Lid	1	0.25	0.25	Top Corner	0.5	0.00	0.00
2.7.3.0.06	Isolate the vacuum pump, and backfill the cask cavity with an inert gas.	Empty No Outer Lid	2	1	2	Side	2	0.00	0.00
2.7.3.0.07	Install the cask outer lid. Torque the bolts to the value specified on a drawing. Evacuate the cavity between the two lids using the VDS and verify the dryness as described in steps 2.7.3.0-A through 2.7.3.0-F above.	Empty No Impact Limiters	3	3	9	Top Corner	1	0.00	0.00
2.7.3.0.08	Re-engage the lift beam to the upper (top) trunnions of the cask.	Empty No Impact Limiters	2	0.25	0.5	Top Corner	0.5	0.00	0.00
2.7.3.0.09	Move the transport vehicle into the loading position.	Empty No Impact Limiters	2	0.2	0.4	Side	3	0.00	0.00
2.7.3.0.10	Lift the cask off the decontamination pad, and place the rear trunnions on the rear trunnion supports of the transport frame.	Empty No Impact Limiters	1	0.2	0.2	Remote	N/A	0.00	0.00
2.7.3.0.11	Rotate the cask from the vertical to the horizontal position.	Empty No Impact Limiters	2	0.2	0.4	Side	3	0.00	0.00
2.7.3.0.12	Install the front and rear trunnion tie-downs.	Empty No Impact Limiters	2	0.1	0.2	Side	3	0.00	0.00
2.7.3.0.13	Check if the surface dose rates and the surface contamination levels are within the regulatory limits.	Empty No Impact Limiters	2	0.2	0.4	Side	1	0.00	0.00

Sequence Number	Process Step	Cask Configuration	Number of Workers	Occupancy Time (hours)	Total Person Hours	Personnel Location Around Cask	Distance from Cask Surface (meters)	Total Dose Rate (mrem/hour)	Total Dose (mrem)
2.7.3.0.14	Install the thermal shield on the front end of the cask. Then remove the thermal shield lifting eye bolts.	Empty No Impact Limiters	2	0.2	0.4	Top Corner	1	0.00	0.00
2.7.3.0.15	Install the front and the rear impact limiters onto the cask. Lubricate the attachment bolts with Never-Seez or an equivalent. Follow the proper torqueing sequence to torque the bolts to the specified 1st pass torque value. Repeat the torqueing process torqueing the bolts to the specified 2nd pass torque. A circular pattern of torqueing may be used to eliminate further bolt movement.	Empty Transportation	2	1	2	Top Corner	1	0.00	0.00
			2	1	2	Bottom Corner	1	0.00	0.00
2.7.3.0.16	Install the impact limiter attachment tie-rods between the front and the rear impact limiters.	Empty Transportation	2	0.2	0.4	Side	1	0.00	0.00
2.7.3.0.17	Render the impact limiter lifting lugs inoperable, by covering the lifting holes or installing a bolt inside the holes to prevent their inadvertent use.	Empty Transportation	2	0.5	1	Side	1	0.00	0.00
2.7.3.0.18	Perform a final radiation and contamination survey to satisfy the shield test requirements and to assure compliance with 10CFR71.47 and 71.87.	Empty Transportation	2	0.5	1	All Around	2	0.00	0.00
2.7.3.0.19	Install a transportation enclosure.	Empty Transportation	2	1	2	Side	2	0.00	0.00
2.7.3.0.20	Apply appropriate DOT labels and Placards in accordance with 49CFR172, and prepare the final shipping documentation.	Empty Transportation	2	0.2	0.4	All Around	2	0.00	0.00
2.7.3.0.21	Release the empty cask for shipment.	Empty Transportation	1	0.2	0.2	Remote	N/A	0.00	0.00
			Total Person Hours		30.00		Total Dose	Total (mRem)	0.00
								Total (Rem)	0.00

5.0 MAINTENANCE REQUIREMENTS

This section defines an approach to planning and performing NRC required maintenance and maintenance performed to support operations. Specific acceptance testing and maintenance requirements as well as acceptance criteria as requested in the SOW are described in detail in *Section 2.8*. The committed maintenance and testing as listed in *Section 2.8* is the same as would be committed to the NRC in the licensing application.

5.1 Graded Approach

The NRC uses graded Quality Assurance for 10 CFR 71-licensed components. For cask systems licensed to 10 CFR 71, the complete cask system, as submitted to the NRC, is considered important to safety. The level of importance to safety for the various components is applied at the time of fabrication in accordance with the guidance found in Regulatory Guide 7.10 and other applicable NUREGs. The Quality classifications for the specific cask components are generated and justified at the time that fabrication drawings are developed. This graded approach to component classifications facilitates the development of appropriate maintenance requirements to which license holders can commit.

5.2 Maintaining Cask Operability and Reliability

Non-mandatory preventive maintenance is performed to maintain cask operability and reliability. Such maintenance tasks include visual inspections, general cask and equipment cleaning, proper covering and protection during storage, and preventive maintenance performed on the leak test equipment and supporting lifting equipment.

5.3 AREVA's Approach to Cask Maintenance

AREVA has over 40 years of experience in transportation cask maintenance including the maintenance of over 4000 casks at three dedicated facilities. These facilities, located at sites where UNF is loaded and/or unloaded, serve several purposes including:

- Keeping casks in compliance with the safety analysis report requirements
  - In France, UNF casks are maintained per the requirements of TS-R-1, which are similar to the maintenance requirements of 10 CFR 71.
- Minimizing delays due to malfunctions during loading and unloading activities
  - Specially trained maintenance personnel that are familiar with the casks are available on-site or can be dispatched to a remote location for technical support and to assist in problem resolution. Spare parts inventories and consumables are readily available for repairs and other emergent work.
- Extending the cask design life
  - In addition to required maintenance of SSC's, cask components are inspected and maintained by performing preventive maintenance at regular intervals or on an as-needed basis. This proactive approach reduces the risk of component failure while under operation. It is also a proven method to reduce repair costs by keeping the cask components on track for meeting or exceeding their design life.
- Reducing dose to workers by minimizing the buildup of internal contamination
  - Tailored techniques for decontamination, designed with ALARA principles in mind, are available for routine and non-routine use. Managing contamination levels reduces dose during loading and unloading operations and transport. Minimizing the

contamination level of the cask reduces the time and complexity of handling and shipping both the loaded and the empty cask. Ensuring that the levels stay well below the allowable DOT release levels for an empty package minimizes the effort to keep the cask in operation.

5.4 Cask Maintenance Program Example

The TN-12 cask is one example of a Type B cask regularly maintained by AREVA in France. Maintenance tasks of the TN-12 can be divided into four broad categories:

- Turnaround inspection on arrival and on departure from the facility
    - This inspection is primarily visual but also includes a leak-tightness test each time the cask is sealed.
- Basic maintenance every 3 years or every 15 transports
    - This includes a visual inspection of all components including body, impact limiters, trunnions, finned area, top of basket, and threads. It also includes dimensional and dye-penetrant inspection of trunnions, a check of selected trunnion bolts, and a fit check of each fuel element compartment within the cask. All gaskets are replaced.
- Main maintenance every 6 years or every 60 transports
    - This includes all basic maintenance as well as complete disassembly of the trunnions and their bolts, a check of the condition of neutron absorber in the basket walls, and a check of the shielding profile and thermal efficiency while loaded with UNF

Corrective maintenance performed as needed. Corrective maintenance may include bolt replacement, seal replacement and replacement of protective coatings on trunnion and lift components.

5.5 Cask Maintenance Facility Example

The following is a brief description of the AMEC, the cask maintenance facility located at AREVA's La Hague Reprocessing Plant.

With a surface area of 1,800 m^2 on four levels, AMEC is one of the world's largest maintenance facilities for UNF and HLW transport casks. AMEC is designed to accommodate La Hague cask maintenance requirements as well as the special needs of AREVA's worldwide customers in Japan, Germany, the Netherlands, Switzerland, Belgium and Sweden. AMEC can handle and process 18 different types of casks weighing up to 120 metric tons. Casks are routinely maintained according to a schedule laid out in the safety analysis report.

AMEC maintains a staff of one hundred operators who perform cask maintenance operations in the facility on a daily basis. Daily operations include maintenance requirements analysis, cask modifications, cask renovation, cask fleet management, chemical decontamination, electro-decontamination and adaptation of LWR casks to accommodate MOX fuels. Information from cask maintenance operations is provided to the design department to verify safety compliance, identify recurring operational issues and contribute to cask life extension.

5.6 Conclusions

Maintenance requirements are not limited to those agreed to in the licensing application approved and enforced by the regulator. Effective planning of non-mandatory preventive maintenance is proven to yield a variety of benefits during cask operation such as reduced cost, reduced worker dose, and increased reliability. Applying lessons learned from existing cask maintenance programs will result in a robust program that meets the needs of the regulator and the licensed operator.

6.0 KEY INFORMATION

This section provides links to the key information associated with the 6625B-HB as requested in item/activity 6 in the SOW: "Provide additional key information associated with each of the SNF transportation cask system design concepts, including information on dimensions, component masses, total mass for both fully loaded and unloaded conditions, maximum thermal loading, and estimated dose rates during normal conditions of transport. Also provide supporting analyses indicating that the transportation cask system, including the cask, impact limiters, and DFCs (when applicable), would be licensable and usable for transportation under 10 CFR Part 71."

Information on the nominal dimensions of the 6625B-HB transportation cask are identified in **Table 2.1-1**. **Table 2.1-2** contains the masses of the various components of the 6625B-HB, as well as the total unloaded ('empty') and loaded weight of the 6625B-HB. The total loaded weight in this table includes the 6625B-HB containing DFCs and PWR or BWR UNF. The total loaded weight of the 6625B-HB with DFCs is 265,202 lbs for the PWR system and 259,152 lbs for the BWR system.

The summary of temperatures for NCT and HAC are presented in **Table 2.3-2** for the Cask PWR basket and BWR basket for the thermal loading patterns shown in **Figure 2.1-6** (for PWR UNF) and **Figure 2.1-8** (for BWR UNF). *Section 2.3* provides a summary of the thermal analyses performed for the 6625B-HB for both NCT and HAC of transportation. These analyses provide the basis for demonstrating the 6625B-HB cask system is licensable and usable for transportation under 10 CFR 71.

The estimated dose rates during NCT and HAC are provided in **Table 2.5-21** and **Table 2.5-22**, respectively for PWR and BWR UNF for the loading patterns shown in **Figure 2.1-6** (for PWR UNF) and **Figure 2.1-8** (for BWR UNF). *Section 2.5* provides a summary of the shielding analyses performed for the 6625B-HB for both NCT and HAC of transportation. These analyses provide the basis for demonstrating the 6625B-HB cask system is licensable and usable for transportation under 10 CFR 71.

The sub-critical nature of the package for NCT and HAC are described in *Section 2.6* for the various configurations of the 6625B-HB. **Figure 2.6-2** presents the minimum required fuel assembly burnup as a function of the initial enrichment. Intact fuel assemblies that lie above the curve in this figure are acceptable for transport by the 6625B-HB. For the HAC conditions, moderator exclusion was credited. Additional defense-in-depth sub-critical analyses were performed for "reasonably" damaged UNF with fresh water moderation and for PWR UNF a mis-load condition.

Section 2.2.2 provides the structural assessment of the 6625B-HB cask system and includes supporting arguments for demonstrating that the 6625B-HB is licensable and usable for transportation under 10 CFR 71. *Section 2.2.3* provides the evaluation of the 6625B-HB cask impact limiters to ensure they mitigate the worst-case free drop conditions for both NCT and HAC as prescribed by 10 CFR 71. DFCs are only credited for ensuring the UNF contained within them remains within the confines of its basket position under all credible conditions. DFCs may also be credited for retrieving the UNF from the cask system.

In general, *Section 2* of this report is intended to cover all aspects necessary to ensure licensability of the 6625B-HB. *Section 2* has been designed to align with the chapters of a SAR as established in the SRP for transportation packages (NUREG-1617). This ensures information addressed in a SAR by an applicant to the NRC is included in this report and, upon completion of the final report, will provide reasonable assurance the 6625B-HB can be licensed by the NRC. Actual licensing of this cask system by the NRC would require completion of detailed design calculations, fabrication drawings, etc. that are outside the scope of this TO.

7.0 CASK SYSTEM REQUIREMENTS

This section provides links to the portions of the report that demonstrate that the cask system requirements specified in item/activity 7 in the SOW are satisfied. These requirements are in addition to those necessary to provide reasonable assurance that the 6625B-HB is capable of meeting 10 CFR 71 requirements. These requirements are as follows:

a. "*The system design concept, including impact limiters, will have a maximum width of 128 inches*." As shown in **Figure 2.1-3**, the 6625B-HB transportation cask system with impact limiters has an OD of 126 inches, which also equates to the maximum width of the package, and hence, less than the 128-inch requirement.
b. "*The system must allow for the transportation of high-burnup fuel (>45GWd/MTU) with a target of transporting fuel with an average assembly burnup of up to 62.5GWd/MT with up to 5.0 wt% enrichment and out-of-reactor cooling time of 5 years.*" As shown in **Table 2.1-7** for PWR UNF (Zone 3) and **Table 2.1-10** for BWR UNF (Zone 2), the 6625B-HB transportation cask system can handle 62.5 GWd/MTU with an out-of-reactor cooling time of 5 years and an initial minimum enrichment of 3.8 wt%. In addition, *Section 2.5.6* examines the ability of the 6625B-HB to handle alternative heat load configurations potentially more suitable for SFPs with an insufficient quantity of older or low-burnup UNF. The configurations analyzed in this section require short loading the 6625B-HB in order to load hotter UNF (i.e., younger, higher burnup UNF). **Figures 2.5-9** and **2.5-10** show this alternate configuration for PWR and BWR UNF, respectively.
c. "*Reasonable assurance that the design concepts can accommodate essentially the entire existing and future inventory of commercial light-water reactor SNF must be provided, without undue penalty (e.g. reduced cask capacity resulting in sub-optimization for the majority of anticipated shipments). Specific fuel designs or attributes (e.g. fuel length, assembly decay heat limits, or burnup limits) not allowed by the cask design concepts must be identified.*" *Section 2.1.2.2* provides a description of the contents of the PWR and BWR basket of the 6625B-HB transportation cask. **Table 2.1-4** and **Table 2.1-9** provide the intact PWR and BWR fuel specification, respectively, authorized for the 6625B-HB transportation cask. *Section 8.2* performs a trade study based on the length of fuel the 6625B-HB transportation cask is designed to handle. This study established that only the South Texas Project PWR fuel assemblies could not be shipped in the 6625B-HB. This study also noted that if the 6625B-HB were designed to accommodate fuel with an unirradiated length of ≤ 159.8 inches then more fuel assemblies could be placed into the cask and only the South Texas Project and Palo Verde PWR fuel could not be accommodated in this shorter cask system. Nevertheless, the 6625B-HB is designed to physically accommodate all but the South Texas Project PWR UNF and is thermally and radiologically designed to take high burnup UNF that has been cooled at least 5 years since service in a reactor. *Section 8.3* further expands the scope of UNF the 6625B-HB can handle by performing a trade study on the transport of certain types of damaged UNF and identifies the current limits to the damage this UNF can have undergone. Additional damaged and failed UNF likely can be handled by the 6625B-HB; however specific analyses must be performed to certify this specific UNF in this cask system.

   The main limitation for accommodating almost all of the commercial light water reactor UNF in the 6625B-HB is the 125 ton limit for SFP cranes. As demonstrated for the Duke

SFPs in Appendix A, a portion of the reactors in the U.S. do not have cranes that are allowed to lift 125 tons and hence, cannot handle a completely full 6625B-HB.

d. "*In addition to the NRC's regulations, design activities shall also consider applicable regulatory guides and recent licensing experience and actions related to transportation cask design, fabrication, and operations. Any applicable Department of Transportation (DOT) requirements and constraints of AAR S-2043 that may have an impact on cask design shall also be considered.*" As noted throughout *Section 2*, the 6625B-HB takes advantage of elements of previously and recently NRC licensed transportation cask systems, which includes the MP197HB that has been recently NRC licensed for the transportation of high burnup UNF. *Section 2.1.1.1* identifies the specific regulatory requirements applied to the design of the 6625B-HB transportation cask. Other regulations, codes, and standards applied to the 6625B-HB are identified throughout the text of *Section 2*, such as the DOT requirement related to labels and placards specified in *Section 2.7.1.3*. In addition, the 6625B-HB is designed to meet the requirements/constraints of AAR S-2043, primarily the 128 inch plate limit.
e. "*The cask system for DFCs in all positions will place constraints on capacity due to the size of the DFCs. The design concepts should satisfy all appropriate regulatory and operational limits, while maximizing capacity.*" *Section 2.2.1* describes the process used to maximize the capacity of the 6625B-HB, while other portions of *Section 2* demonstrate the 6625B-HB meet regulatory and operational limits (e.g., *Section 2.3* demonstrates the 6625B-HB meets regulatory thermal limits). Also as noted in *Section 8.1*, inclusion of the DFC into the 6625B-HB transportation cask did not reduce the quantity of UNF this cask system could contain; the main impact of the DFC is on the cost, operations, and accumulated doses.
f. "*The transportation casks shall be capable of being closed and reopened multiple times, so the cask can be reused for many shipments. The method for closing and reopening shall be described. Factors limiting the possible number of times that the cask can be reused shall be identified, along with possible means for extending life and reusability of the casks.*" As described in *Section 2.1.2.1.1*, the 6625B-HB is a bolted transportation cask system that allows it to be reused "multiple times" for transportation. *Section 2.4.1.4* describes the seals that are necessary for providing a containment boundary that can be re-established multiple times. The operations necessary to remove and reinstall the bolts are provided in *Section 2.7*. *Sections 2.8.1.3* and *2.8.1.4* discuss the structural and leak-tightness testing necessary for transportation casks, which allow for the reuse of the transportation cask. In general, if the 6625B-HB is maintained in accordance to the requirements identified in a Safety Analysis Report (as outlined in *Section 2* of this report), there is no limit on the number of times the 6625B-HB could be used. This is further supported by depletion analyses which have been performed on the effectiveness of the neutron shield and demonstrated it remains effective over a 100-year history in the presence of design basis UNF.
g. "*The loaded and closed DFCs shall also be capable of being reopened, to allow assembly repackaging, and the method for reopening shall be described.*" *Section 2.1.2.2* provides a description of the DFCs to be utilized in the 6625B-HB transportation cask. The method for reopening the DFC to allow for the retrieval of the UNF is described in *Sections 2.1.2.2.1* and *2.1.2.2.3*.
h. Consistent with current industry designs, the DFCs shall be vented at the top and bottom." As noted in Sections 2.1.2.2.1 and 2.1.2.2.3, the DFC for the 6625B-HB transportation cask is provided with screens at the bottom and top.

8.0 TRADE STUDY

8.1 Damaged Fuel Canister

The impact of using a DFC to transport UNF is examined in this section. While developing the 6625B-HB design, it was noted that the cask weight limit was the primary limiting design requirement. Once the weight limit had been established, it dictated a limiting diameter for the cask, which in turn established the maximum quantity of PWR fuel assemblies that could be placed within the cask. With this maximum quantity of PWR fuel assemblies (24 FAs), the thermal loading, assembly burn-up, and required cooling times were optimized to ensure this quantity of PWR fuel assemblies could be safely placed in the cask. The inclusion of DFCs did not have a penalty on the package capacity (i.e., no fuel assemblies had to be removed to meet the weight limit when DFCs were included). Without DFCs, the cask weight while maintaining the 180-inch minimum length requirement, could not be reduced to the point that the assembly quantity could be increased.

The next common (i.e., previously licensed) symmetric assembly array (required by thermal and criticality/shielding), is 32 FAs. Based on a maximum PWR assembly weight of 1,715 lbs, a weight reduction of 1,715 x 8 = 13,720 lbs would have been required, as well as an increase to the cask diameter to accommodate the additional assemblies. So the use of DFCs did not adversely impact the maximum loading of PWR fuel assemblies into the cask.

However, the use of DFCs does impact cask operations. DFCs require additional steps to be added to operations that account for the preloading of the cans into the cask (prior to fuel loading) or alternatively account for first loading the cans with fuel and then inserting them into the cask. In addition, the basket sleeves must be removed to accommodate the can in the cask basket and the can lids would need to be installed and locked after they had been loaded with fuel. There may also be an impact to the duration required for vacuum drying the cask loaded with DFCs. Similar steps in reverse would be required for unloading the DFCs or the fuel from the DFCs in the cask. These additional operations would likely increase the loading time and personnel dose.

The cost of fabricating a DFC is included in *Section 3.0*. Additional DFCs would need to be fabricated as spares. Additional costs would be associated with procuring lifting and rigging equipment designed to interface with DFC lids during cask loading and unloading operations. Equipment design work would need to be completed in conjunction with site-specific facilities to ensure proper interfaces between DFC lids, site grapples, site cranes, etc.

In conclusion, the use of DFCs poses no penalty to the number of fuel assemblies that can be loaded into a cask. However, it is likely to result in increased time required for loading and unloading, higher personnel exposure, and increased cost to operate the 6625B-HB cask system.

8.2 Fuel Length Trade Study

The 6625B-HB cask was designed to accommodate irradiated fuel assemblies up to a maximum length of 180-inches. This design requirement allows one cask design to accommodate all of the existing BWR and PWR fuel assemblies, only excluding the South Texas Project PWR fuel assemblies (these fuel assemblies are 199 inches in length, unirradiated).

Table 8.2-1 illustrates some of the tradeoffs between cavity length and fuel capacity, while maintaining the 125-ton maximum on-the-hook weight.

TABLE 8.2-1: TRADEOFFS BETWEEN CAVITY LENGTH AND FUEL CAPACITY

	F/A Length (in)	Cavity Length (in)	Cavity Diameter (in)	F/A Weight (lb)	DFC weight (lb)	# F/As (by weight)	#DFC's	Water in cavity?
PWR	180	182	66.25	1715	54.9	24	24	No
	165.7	168.8	67.19	1682	52.6	27	27	No
	165.7	168.8	67.19	1682	52.6	28	4	No
	165.7	167.7	67.19	1682	0	28	0	No
	159.8	162.9	70.00	1682	51.7	27	27	No
	159.8	162.9	70.00	1682	51.7	28	16	No
	146	149.1	70.0	1682	49.5	32	27	No
	147.5	150.6	70.0	1682	49.7	31	31	No
BWR	180	182.0	66.25	705	30.7	61	61	No
	176.2	178.2	66.25	705	0	61	0	Yes
	176.2	179.3	70.0	705	30.2	63	30	No
	171.2	173.3	70.0	705	0	68	0	No
	171.2	174.3	70.0	705	0	67	20	No

Note: 125-ton maximum on-the-hook weight is maintained in the above capacity table

When reading the above table, it should be noted that fuel basket geometries are typically rectangular arrays with a discrete number of fuel assembly combinations. Geometries with 24, 32, and 37 fuel assemblies are some of the known configurations for PWR baskets and 61 and 69 fuel assemblies for BWR baskets. When fewer fuel assemblies are shown for these discrete numbers, empty locations were assumed. In the columns where the DFC weight is shown as "0", DFCs are excluded. When the DFCs are excluded, the cavity length is also reduced since the thermal expansion of the DFC does not need to be accommodated and the thermal growth of the fuel assembly is less than that of the cask.

The fuel assembly lengths used in the study were taken from "Nuclear Fuels Storage and Transportation Planning Project Inventory Basis", FCRD-NFST-2013-000263, Rev. 1 [11.36].

The most interesting finding of this study is that the majority of the PWR fuel assemblies considered could be accommodated in a significantly shorter cask. This would lead to less material, significantly less weight, and reduced impact limiter size. A reduced size design could accommodate all PWR fuel assemblies with an unirradiated length ≤ 159.8 inches. This includes fuel in DFCs. The only additional fuel assemblies that would be excluded are Combustion Engineering 16 x 16, System 80 used only at Palo Verde.

Alternatively, instead of reducing size and weight, the cask body diameter could be increased to accommodate 28 FAs. This assumes no further optimization of fuel assembly spacing in the design modification. Due to the on-the hook weight restriction, this maximum loading would require that no more than four FAs be contained in DFCs.

Not all of the variations above have been fully investigated. Further study of FA spacing scenarios and arrangement within the cask body could lead to other efficiency gains.

8.3 Damaged and Failed Fuel

The thermal, shielding, and criticality analyses in the main body of the report are performed using intact fuel models for NCT. However, the utilities may have damaged or failed fuel that would be placed into DFCs. These damaged and failed fuels are defined as follows:

Damaged fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel assembly shall be handled by normal means. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.

Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.

Loading damaged fuel would have negligible effect on the thermal and shielding performance of the package because damaged fuel is capable of being handled by normal means and therefore would behave essentially the same as intact fuel. However, loading damaged fuel would have an impact on the criticality analysis, as fuel with missing fuel rods could have increased moderation and hence be more reactive than an intact fuel assembly.

Additionally, in the case of leaking fuel, decontamination of the cask and basket would be required extending the cask turn around time, require additional considerations with respect to the receiving fuel pool (to account for handling the contamination introduced to the destination pool) and in extreme cases, replacement of the basket. revised basket designs with features more amenable to decontamination or canisterization of leaking fuel assemblies might also be considered to reduce decontamination requirements and improve turn around times.

The criticality analysis for damaged fuel is limited to the 10 CFR 71.55(b) analysis, which is the fully-flooded single package with fuel in the as-loaded condition. The defense in depth analysis documented in *Section 2.6* demonstrates the acceptability of all fuel assemblies being damaged in an accident, although the defense in depth analysis uses a higher USL than the §71.55(b) analysis.

As a postulated bounding fuel damage, the pitch of the PWR fuel assemblies is allowed to increase until the fuel assembly is restrained by the DFC. For BWR fuel, the pitch is allowed to increase until restrained by the zircaloy fuel channel. The PWR analysis is performed for 8 damaged and 16 undamaged fuel assemblies with the damaged fuel assemblies in Zone 4. The BWR analysis is performed for 12 damaged and 49 undamaged fuel assemblies with the damaged fuel assemblies in Zone 4. The USL is met for each scenario.

Failed fuel includes severely damaged fuel assemblies, including loose fuel pellets. The thermal, shielding, and criticality models for intact fuel do not necessarily bound failed fuel. If it is assumed failed fuel may form a concentrated lump, the thermal load and source term could be concentrated over a smaller volume, which could impact the acceptability of the result. Also, more penalizing criticality geometry could be assumed for the §71.55(b) analysis. Therefore, a comprehensive analysis for failed fuel is beyond the scope of the current report.

9.0 SPECIAL FEATURES

This section identifies some special features, which could be introduced into the 6625B-HB cask design and operational concepts to allow for system optimization. However, based on the AREVA team's experience designing storage and transportation cask systems, the team believes the 6625B-HB represents a mature (licensable) design, utilizing many state-of the art design features and having been optimized through several design evolutions. Hence, the implementation of these special features into the design of the 6625B-HB could be a design, licensing, and/or fabrication challenge, requiring additional design iterations to incorporate and to balance against potential drawbacks.

The AREVA team held several team meetings to "brainstorm" features that could potentially lead to: increased capacity of the cask systems, reduced costs, reduced maintenance activities, increased operational efficiencies, reduced cumulative personnel doses, etc. In addition, potential lessons learned from the AREVA team's exercises towards the unloading of the SFPs at various Duke Energy reactor sites are documented in this section and included in the Appendix to this report.

The scope of this section is limited to general concepts and the discussion of any special features is necessarily qualitative in nature. The proposed features may not be compatible with one another and hence, should not be considered for simultaneous application as design enhancements and/or operational efficiencies. Benefits derived from one special feature may adversely impact another feature, licensability, operations, and/or cost; hence, requiring an assessment/analysis (e.g., a cost-benefit analysis) to be performed. The following discussions are not an attempt to quantify any trade-offs, but will raise them in a qualitative manner where applicable.

The special features in this section are organized first by listing any special design features, followed by improved operational features, although in some cases the design features may also indirectly yield operational benefits, or a special design feature may directly be proposed to improve operations. In general, these proposed special features are introduced using qualitative or semi-quantitative comparisons with the baseline cask system design discussed in the previous sections of this report.

Where possible, the proposed special features could be designed to avoid or minimize any modifications of reactor site procedures or at-reactor structures, systems, or components. However, some special features requiring modifications at reactor sites are still proposed where it could be easier to implement at the spent fuel receiving facility. In particular, any features that improve the throughput will have greater positive impacts at the receiving facility (e.g., multiple lines to process casks arriving from several reactors simultaneously). In particular, the proposed special features focus on the following general topics related to cask design and operations:

Cask system capacity optimization

Cask system weight optimization

Reduction in gamma and neutron radiation doses

Enhanced thermal transmission and dissipation

Improved criticality margins

Cask lift system alternatives

Automation of manual functions

Reduced decontamination and cask drying times

Increased cask handling flexibility

Other throughput improvements (cask loading/unloading)

9.1 Cask System Capacity Optimization

As described in *Section 8.2*, the 6625B-HB cask was designed to accommodate a minimum 180-inch long fuel assembly. This design requirement allowed one cask design to accommodate all of the existing BWR and PWR fuel assemblies, only excluding the South Texas Project PWR fuel assemblies. However, only the fuel assemblies used at Palo Verde require the full length of the cask. The majority of PWR fuel assemblies, which are less than 160 inches in unirradiated length, could be accommodated in a cask cavity that is approximately 20 inches shorter than the current baseline cask design. Based on the nominal overall cask length of 200.5 inches without impact limiters (**Table 2.1-2**), this condition represents a reduction of approximately 10% in required cask length for the majority of PWR fuel.

While the top and bottom shielding thicknesses would likely not be significantly reduced in the shorter length cask, the shortened cylindrical section of the cask body and basket could be proportionally reduced without any loss of side shielding or structural strength. The consequent reduction in the loaded weight of the cask (considering also the reduced weights of the shorter assemblies compared to the longer baseline design assemblies, each with a maximum nominal weight of 1,715 lbs) may allow a proportional increase in the external diameter sufficient to increase the cask capacity, while maintaining the required shielding and thermal properties.

In **Table 8.2-1**, the cask cavity diameter is projected to increase by less than one inch to accommodate as many as 4 additional PWR assemblies relative the baseline of 24. This number was compared to known basket geometries, and the tabulated values assumed empty basket locations in order to remain within the overall casks system on-hook weight limit of 125 tons, excluding water from the cavity. However, it may be possible to re-configure the basket geometry by deviating from the current cross-pattern design of the basket cells. By offsetting some of the basket rows by half a cell length, it may be possible to gain an additional cell in some of these rows while at the same time remaining within the overall basket outer diameter envelope. For example, it may be possible to increase capacity of the current 24 PWR basket design by 2 additional assemblies (rows of 2, 5, 6, 6, 5, 2 assembly channels) without a change in basket diameter (but the transition rail volume would be proportionally reduced). The increased weight of the two additional fuel assemblies would be more than offset by the weight savings from a shorter cask. In addition, the weight savings may be sufficient to offset the weight of the loaded cask with water, allowing the cask to be removed from the pool without having to first drain the water from the cavity. Current “egg-crate” basket designs and the associate fabrication process would require substantial modification to accommodate offset rows. Any offset in the rows will significantly affect the structural strength of the basket potentially requiring thicker basket walls and/or larger impact limiters to adequately reduce the loads of an impact. These changes may offset any gain from the changed geometry.

Rapid advances in additive manufacturing (aka, 3-D printing) that now include the ability to employ different metallic and non-metallic additives might be used in the future to facilitate construction and achieve this design objective. However, this needs to be balanced against the challenges associated with any additive manufacturing process, which include the need to qualify the resulting material as an ASME B&PV Code material and the ability to provide, and cost-effectively implement, adequate quality control. Nevertheless, in addition to increasing basket capacity within any given envelope, additional benefits could extend to optimizing basket structural and thermal dissipation features that may be derived from a more flexible "mono-body" construction. Additional dimensional, thermal, shielding, and weight analyses can be performed to determine whether the capacity could be further increased within the cask system weight constraints.

Meeting the above design, licensing, and fabrication challenges to deploy an increased-capacity cask that accommodates the majority of the PWR fuel assemblies at most reactor sites would still require the current baseline cask to be utilized for the longer Palo Verde fuel assemblies. The current baseline design would also be required to accommodate the majority of BWR assemblies, which are longer than the majority of PWR assemblies. If more than one cask body design is considered feasible, then the associated optimization studies could be extended to encompass the South Texas Project 199-inch long PWR assemblies, so that up to three-cask system designs could be considered:

A cask system optimized for South Texas Project/Palo Verde PWR assemblies (possibly with smaller assembly capacity to meet on-hook weight limits at these two facilities),

The current baseline cask system applied only to BWR assemblies (or slightly shorter cask length, based on longest BWR assemblies), and

The increased-capacity cask system as described above for the majority of PWR assemblies.

9.2 Cask System Weight Optimization

The on-hook cask system (including cask body/trunnions, basket, assemblies and/or DFCs, water, lids, and yoke) weights are shown in **Table 2.1-2**. For the baseline cask, the water volume is estimated to be 6,452 lbs for PWR assemblies and 11,304 lbs for BWR assemblies. Without the outer lid installed, the baseline BWR cask could be lifted with water in the cavity, but the hook limit would be exceeded in the case of the PWR cask.

The initial sizing calculations described in *Section 2.2.1* indicated that removal of water from the cask cavity would be required before a cask loaded with PWR assemblies could be removed from the pool. Draining the cask cavity requires considerable time and efforts, including attaching drain and fill-tubes to the drain and vent ports as well as pressurized gas through the fill port to enhance draining while the top of the cask is above the pool. During this operation, the cask remains suspended from the pool crane, with an associated increase in the cask drop risk. The ability to leave water in the cask cavity would greatly facilitate handling operations, and reduce risks.

As shown in *Section 9.1*, a shorter PWR cask could result in sufficient weight reduction to allow water to remain in the cavity, even with an increased fuel assembly capacity. This reduction occurs because some of the additional assembly mass would be offset by a lower water mass in the cask cavity, assuming a fixed inner cask diameter. For the longer PWR casks, reducing the cask diameter and assembly capacity could result in reduced cavity water mass as well, thus bringing the total hook weight to within the 125-ton limit, while allowing water to remain in the cavity. For the BWR cask, no water drainage would be required within the constraints on the baseline cask cavity dimensions. This configuration could be further optimized for an increased BWR assembly capacity if the baseline cask cavity length could also be shortened.

Table 2.1-2 considers the total hook weight with or without the outer lid. However, the outer lid, weighing 3,563 lbs, would not be installed until the inner lid has been installed, followed by full evacuation of the water, helium back-fill, port closure, and leak testing. Therefore, the outer lid mass is not considered to be a target for additional weight optimization, since the cavity water mass exceeds the outer lid mass for both PWR and BWR baseline casks.

Table 2.1-2 considers a lifting yoke weight of 7,500 lbs. The yoke is assumed to be sufficiently long to attach to the upper cask trunnions, while allowing the pool crane to lift and lower the cask to the bottom of the pool, after which it is detached from the cask and moved out of the way to enable cask loading. Most of the yoke mass is assumed to be made up of stainless steel arms and crossbar, along with the rigging that attaches to the crane hook. It may be possible to allow a heavier (and larger capacity cask) by reducing the yoke weight while maintaining the necessary margins of safety for lifting the cask by the trunnions. Strong, lightweight materials, such as titanium or Kevlar components, could be used to substitute all or some of the steel components of the yoke. In addition, a modified cask design that eliminates the need for trunnions (described in greater detail in *Section 9.6* below) may also result in a lower yoke weight. Yokes fabricated of these materials are likely to be more expensive than steel, and should be considered only to the extent that there is a significant weight reduction benefit relative to the baseline yoke.

Similarly, some cask shell weight reduction could be achieved by substituting a lighter material with similar structural, thermal and shielding properties as the baseline cask shell. For example, substituting the inner surfaces of the cask with titanium instead of stainless steel may allow a reduction in the inner shell thickness while maintaining the strength as steel. In addition, titanium can withstand high temperatures and is corrosion-resistant. However, these beneficial properties are offset in part by a lower radiation shielding efficiency (due to lower density and slightly lower atomic mass than steel), and lower thermal conductivity. In addition, titanium alloys are significantly more expensive than stainless steel (about 4 times more expensive than stainless steel by weight, partly offset by almost half the density of titanium compared to steel). Furthermore, titanium has numerous fabrication difficulties, including the difficulty of sealing it to the structural portion of the cask and the containment boundary. Alternatively, higher strength alloy steels could be used and then clad with stainless steel sheet or weld overlay.

Besides the steel structural elements in the cask body, a large part of the cask weight consists of the lead shielding required to reducing the gamma dose rate at the cask outer surface (see *Section 2.1.2.1.3*) to acceptable levels. The baseline design consists of a 3-inch thick lead annulus enclosed between the cask inner and outer cylindrical steel shells, a 4.5-inch thick lead disk enclosed in the cask bottom, and a 2.5-inch thick lead thickness in the cask inner lid. As described in *Section 9.3* below, substituting depleted uranium for lead would not only reduce the necessary thickness due to the greater density of uranium metal, but also result in a net mass reduction due to the greater shielding efficiency of uranium for high-energy gamma rays. The benefits gained from doing this substitution would have to be evaluated against the cost, difficulties in fabricating depleted uranium, and the necessary design changes that would be required to accommodate the chemical and physical properties of the material, including its brittle fracture characteristics.

For those reactor sites that have limited pool crane capacities or pool floor load limits that cannot accommodate even a reduced-weight standard cask design (i.e., below the 125 ton hook limit), one possible solution would be to use a transfer cask system. Such a system would be designed to insert a smaller number of assemblies than a standard capacity cask into a basket mounted inside a bolted canister, which is placed inside the transfer cask that has been lowered into the pool. After loading into the bolted canister, the inner lid would be placed over the top of the canister. This smaller inner lid would be sized and designed to seal the canister after removal from the pool. After removal from the pool, the bolted canister/transfer cask system would then undergo similar processing and handling, as performed on standard transport casks. Once completed, the transfer cask system would be placed over a standard-sized transport cask body. But in this case, the transport cask shell would include annular inserts/shield elements to accommodate the smaller-diameter bolted canister into the cask body, after which the standard-sized outer cover is bolted onto the top and the cask is transferred to the transport skid for loading onto the rail car.

9.3 Efficiencies in Controlling Gamma and Neutron Radiation Doses

As describe above, the primary material used to protect workers and the public from radiation exposure are the lead and steel layers in the cask body, bottom and lid. Lead is a relatively inexpensive and dense metal with a high atomic number, making it an ideal shielding material. But it has poor structural properties, low melting temperature, and heavy metal chemical toxicity characteristics that require it to be enclosed in steel layers to be effective in cask designs. The goal in cask design is to use the minimum amount of material for shielding while meeting the regulatory requirements. Other materials can be used, and could be evaluated against their cost and effects on fabrication and design.

Since shielding of high-energy gamma rays is both a function of density and atomic number, increasing both density and atomic number of the shielding material improves the shielding properties of the material. For example, steel (7.86 g/cc) has a half-value layer, defined as the shield thickness to reduce high-energy gamma levels by half, almost twice as thick as lead (11.3 g/cc) for high energy gammas such as those from Co-60. However, since lead is denser than steel, the same thickness of lead is 44 percent heavier than steel.

Depleted uranium metal, defined as containing more than 99.3 percent of the U-238 isotope by mass, has excellent shielding properties. It is only mildly radioactive, and as with lead, is more of heavy metal toxicity hazard. For Co-60 gammas, the half-value layer of uranium metal is almost half that of lead, making it an even more efficient shielding material.

An additional benefit of uranium shielding material relative to lead is a higher melting point (2070 °F vs 621.5 °F), but it has a lower thermal conductivity (27.5 $W \cdot m^{-1} \cdot K^{-1}$ vs 35.3 $W \cdot m^{-1} \cdot K^{-1}$). This reduction in thermal conductivity is offset by that fact that a lower thickness of uranium would be required for the equivalent shielding. Structural analysis would be required to compare the structural properties of lead and uranium, but in either case, most of the structural analysis relies on the strength of the steel shells. One drawback of uranium metal is its pyrophoricity (ability to spontaneously combust), making it a fabrication challenge prior to its encapsulation in the steel layers of the cask shell and lids. . Additional uranium metal effects on the design include brittle fracture characteristics, and the potential eutectics it forms with stainless steel.

Other factors to consider in replacing lead with uranium in the design are the availability and cost of depleted uranium as well as licensing challenges. Another material that could be considered (again, due to high density compared to lead) is tungsten.

Another location where a dense material (e.g., lead) was considered for use was as a 'filler' in a hollow basket transition rail. In the current design, the rails are solid aluminum instead of being hollow to enhance thermal conductivity. If the rails were to be filled with lead or another metal of high density/atomic number with good conductivity properties, their proximity to the source of gamma radiation could improve the shielding efficiency, requiring less shielding thickness in the cask body shell. Unfortunately, any time dissimilar metals are used together there exists the potential for gaps to develop between them and that greatly affects the heat transfer, potentially reducing much of the gain by changing materials. Additionally utilizing some of these dense materials (e.g., lead) in or near the basket is risky due to their low melting temperatures that are likely to be exceeded by the high temperatures potentially seen in and near the basket.

An additional potential drawback of using depleted uranium as a gamma shielding material results from its ability to absorb thermal neutrons, and become a secondary source of gamma radiation due to neutron activation. Analysis would be required to ensure that these secondary gammas could be adequately captured or attenuated by the uranium itself, or in the cask outer steel shell.

In addition to gamma shielding, the baseline design includes neutron shielding for neutrons emitted by the UNF. The primary neutron shielding consists of a 6-inch thick borated resin compound (for neutron absorption inside quarter-inch-thick copper containers to provide a heat conduction path), wrapped around the cask's outer steel structural shell. A final 0.25-inch thick steel shell encloses the array of resin/copper containers. Due to its reduced performance capabilities at high temperature, the materials characteristics have a significant influence on the effective post-accident dose rate. Therefore, continued research is underway to develop or locate materials that will survive the HAC events. Qualification of any material would need to be adequate to satisfy regulatory authorities.

Additional neutron absorption is provided by borated aluminum as described in *Section 2.1.2.2.1*, in the basket design.

9.4 Enhanced Thermal Transmission and Dissipation

The current design limits and restricts the thermal properties of individual assemblies to certain array locations inside the basket (see *Section 2.3*).

While being considered in past designs, active cooling systems (such as using coolant tubes embedded in the neutron absorbing material layer), no such transportation cask designs are licensed. Part of the reason for this is the difficulty of demonstrating that they can maintain their safety function during the HAC of 10 CFR § 71.73 (i.e., free drop, puncture, and fire) , and function unattended for a year as required by regulation. Current regulatory position does not accept active cooling systems that are required to meet regulatory limits.

Passive heat dissipation systems have been proposed and implemented in past designs, typically consisting of cooling fins running in parallel along the cask body. Such fins, if fabricated of good thermal conducting material, can aid in dissipating the heat load due to the increased surface area of the cask. Starting with the baseline design, it may be possible to increase the outer surface area of the copper-encapsulated borated resin tubes used for neutron dose reduction. However, this would present problems in encapsulating the resin tubes with the outer steel shell. In addition, such external cooling fins would present decontamination challenges after the cask is moved from pool to decontamination pit. Finally, even passive outer cooling systems present significant licensing challenges in analyzing the system for HAC events since any improved heat removal features also have the ability to transfer heat into the cask during the thermal event.

9.5 Improved Criticality Margins

The criticality analysis is presented in *Section 2.6*. Credit is given for the fixed neutron absorbers that are present in the basket in the form of borated aluminum plates. This material is ideal for long-term use in the radiation and thermal environments of the package.

If baskets can be fabricated using the additive manufacturing technique discussed in *Section 9.1*, increased boron loading could be considered in the material mix used in fabricating the baskets. Other options may include borated steel, cadmium, hafnium, or other solid neutron absorbing material. Trade-offs would include neutron capture efficiency, structural property changes, and effects on design, cost, weight or concentration, ease of fabrication, and material compatibility.

9.6 Cask Lifting Systems Alternatives

The baseline considers a bolted trunnion cask design, with two upper (lifting) trunnions and two lower (up/down-ending) trunnions. The upper trunnions are used to lift the cask in a vertical position so that it can be handled in the pool area, after being up-ended from its horizontal position on the rail car skid. The lower trunnions rest on cradle arms on the skid allowing the cask to rotate from the vertical to the horizontal position (along with the upper trunnions being used to lift or lower the top of the cask). Both the upper and lower trunnions are designed to be attached to the cask outer structural shell (under the neutron shield) with bolts, and are removed for shipping. The reason for removing the trunnions during shipping is so that the stroke of the impact limiters in not obstructed by the presence of the trunnions, and result in a hard point during a HAC free drop event. The bolting concept is used to minimize the amount of shielding removed to accommodate the trunnions attachment points.

Bolting and unbolting these trunnions is currently envisioned as a manual operation (although it could be automated, but with difficulty since it would have to be performed either on the rail car or a turning fixture per *Section 9.7*). Such a repeated operation requires some time to accomplish, resulting in a dose to the workers performing the removal from or installation of trunnions on a loaded cask. Some possible alternatives to a bolted trunnion design could include shoulder/shear-pin trunnion system, removable in-socket trunnions, or trunnion-less designs.

Alternative lifting and rotating concepts were investigated, however no concept providing tangible improvement was identified, as each lifting system was evaluated for: its effects on the cask shielding both during handling and during transport; its ability to meet both the 10CFR71 lifting requirements and the single proof lift requirements of ANSI N14.6; and the effect on the overall hook weight. Some concepts improved one aspect of these features, but at the cost of another feature (e.g., simplifying the interface on the cask resulted in increasing the weight of the yoke). Additional investigations may be merited, but the maturity of the existing design (having gone through multiple optimization iterations) makes the return on further enhancements likely limited.

9.7 Automation of Manual Functions

In the baseline concept of cask handling operations, some fairly labor intensive activities are performed in the vicinity of the cask surfaces, sometimes while the cask is loaded and the radiation levels contribute significantly to worker doses, both at the reactor site and at a central storage facility. These include bolting/unbolting trunnions as well as inner and outer lids, and attaching/detaching impact limiters to the cask body.

Options for reducing the bolted trunnions and the associated bolted neutron absorbers have already been considered in the activities covered by *Section 9.6*. However, if a bolted trunnion system is retained in the design, the time to bolt and unbolt the trunnions could be greatly reduced by using a multi-spindle bolting machine. These types of devices are commonly used in the manufacturing industry to simultaneously secure pieces of equipment with a pre-programmed bolting pattern and pre-set torque levels. Such a system could be suspended from gantry crane and be used both horizontally (as would be the case for the bolted trunnions/neutron absorber covers). Alternatively, a robotic arm could be programmed to individually tighten bolts, but multi-spindle devices have an advantage in that they can apply simultaneous motion to multiple bolts at the same time, avoiding use of a 'star-pattern' to apply uniform bolting pressure or the need for multi-step bolt tightening levels. This approach could be designed either to work on the rail car or on a turning stand.

Similarly, an automated multi-spindle cask lid bolting system combined with a robotic arm could also be designed. This system would be located in the work area close to the loading or unloading pool. The arm would allow removal of covers and the installation of venting and gas sampling tools, as well as contamination survey equipment. Multiple spindles could be used. Since it is unlikely that the inner and outer lids will have the same number of bolts, the optimized design would include the lowest common denominator of spindles, and the bolting annulus can either be designed to rotate along the circumference of the cask top (perhaps using the outer diameter of the cask as an attachment/reference point), or the spindles themselves could be track mounted on the annulus and move to each bolting position. Adjustable arms would be used to move the spindles to the correct radial position over the lid, and the spindles could be designed to be self-centering over the bolts. Bolt placement into or removal from the bolt holes could also be automated, but the initial insertion or final removal may still be accomplished manually. This equipment could be simplified by the use of cone headed bolts and capture bolts.

Other labor/exposure-intensive operations in the vicinity of the cask top is the attachment/disconnection of the drain and fill system as well as the vacuum drying and helium backfilling operation, contamination surveys, and decontamination, along with the associate monitoring and testing operations. Currently, prior to performing any of these operations, the top of the cask (including the lids) would need to be decontaminated to reduce exposure risks to workers performing the operations. Similar to the systems described above, a circular/radial decontamination system could be designed that completely cleans the cask lid area, and could be supplemented by an automated smear/counting system to verify that the contamination levels are within the acceptance criteria.

The cask side decontamination process could be automated by having a ring sprayer and robotic arm-type of system that could access all parts of the exterior to precision clean these surfaces and verify release criteria. This process could be combined with a wash down system as the cask exits the pool. An automated system could be used for these operations. Prior demonstrations programs under the OCRWM program demonstrated that a properly designed work station could eliminate almost all exposure during cask opening and closing operations while in the facility.

While each of these systems will come with an associated design and fabrication cost, several systems could be built and could be re-used at each reactor site, and multiple units at the central storage facility could be used to support parallel processing lines. At each location there could be multiple systems or most features built into one system. Over the life of the fuel transfer campaign, the savings in terms of time and collective dose could be substantial and sustain ambitious throughput goals.

9.8 Reduced Decontamination and Cask Drying Times

Another way to create time and dose savings is to reduce the cask decontamination and drying times. After removal from the pool, the cask undergoes a gross decontamination process and is placed in a decontamination pit where it may undergo multiple decontamination cycles to remove contamination that is trapped in pores of the cask steel surface. This type of contamination can 'weep' or seep to the cask surface over time.

The automated decontamination features described in *Section 9.7* can help accelerate the decontamination processing times, but cannot easily mitigate the seepage from the cask surfaces. If contact of the cask outer surfaces with the pool water can be avoided or minimized, this could reduce contamination levels that need to be remediated. Alternatively, surface contamination may be limited, at least over large external cask surfaces, by applying a reusable sleeve or skirt around the cask body, leaving only the top of the cask exposed to water. Another variation on this theme would be to wrap the cask surfaces with a disposable strippable plastic. After gross decontamination, this wrap could be stripped, leaving a clean cask surface, and disposed as solid radioactive waste (with incineration as a possible volume reduction technique).

9.9 Increased Cask Handling Flexibility

The baseline design envisions 5 casks delivered by rail car to each site for loading and return to a Central Storage Facility. Currently, cask loading and handling steps at a reactor site are postulated to occur with one cask at a time, requiring as long as one week to process a cask from removal from the rail car to loading back on the rail car. Many sites do not have sufficient space in the pool area or the crane capacity to accommodate more than one cask at a time, and procedures are written for sequential steps applied to a single cask. For those sites that have external areas that could accommodate multiple loaded casks, it may be possible to conduct overlapping operations to expedite the cask loading and handling operations. Such a system could consist of a temporary (and re-usable) weather protected cask staging area assembled near an accessible space, large enough to handle multiple casks (up to 5). A dedicated cask carrier (could be similar to straddle carriers used to move a loaded storage cask between the pool and the dry storage cask pad) could be designed to move the empty or loaded casks between rail car and this staging area, and to and from the pool area. Depending on the rail car/skid design, the carrier could be designed to move casks on skids, including the upending function and a capability to straddle the rail car to remove the cask/skid. The feasibility of such a system would depend on the available space at the various shipping and receiving facilities as well as the site specific licenses and the difficulty of expanding potentially contaminated work areas.

Such a cask carrier could be equipped to perform all cask-handling operations that do not require the cask to be inside the pool airlock enclosure, something that can be envisioned if some of the enhanced cask decontamination features can be implemented so that the cask surfaces are relatively clean prior to being removed from the pool area. The operating procedures could be modified to allow multiple casks to be processed outside of the pool area with a dedicated crew responsible for these steps, while the reactor site workers would focus on receiving the casks from the staging area, performing the pool-centered loading and decontamination operations and prior to releasing each loaded cask back to the staging area. This task sharing could made even more efficient if the 5 rail cars could all be unloaded at the same time (with all 5 casks delivered to the staging area, plus one additional space for a sixth cask) allowing the release of the rail cars to pick another 5 casks and return to the site. Once the shipping campaign is underway at each site, the 6^{th} empty cask could be delivered to the staging area, a loaded cask would then be moved from the staging area to the empty rail car, and so on until the train is full of rail cars with loaded casks and the staging area again contains only empty casks.

Several temporary staging buildings and cask carriers may be required if multiple reactor sites are loading spent fuel during the same time period. To facilitate mobility, the structure should be designed for quick assembly/disassembly and rail/truck transport of its components to each site. The cask carriers could also be designed to be dispatched (typically by rail due to weight and size) and would either have to be “collapsible” to fit the rail transportation envelope, or also be shipped to each site in parts easily assembled parts.

9.10 Other Throughput Improvements

At reactor sites, some time is spent detaching or attaching the impact limiters to the cask body. The baseline design discusses impact limiters that use cranes to remove and place the impact limiters, which are secured to the cask body with 12 bolts each. However, the cask carrier could be equipped to perform this function as well as automatic bolting, freeing up the site crane for other tasks.

Alternatively, these impact limiters could be rail- or roller-mounted onto the transportation skid. For this design, the impact limiters would not require any lifting, but would use the tracks to slide out of the way to remove or re-place the cask to or from the skid. In addition, an electric winch system could be incorporated into the impact limiter design: a pair of cables attached to each side of the impact limiters could be used to draw the impact limiters together and onto both ends of the cask, while the same winch system could be reversed (attaching the cables to the ends of the skid) to pull the impact limiters away from the cask. This system could be either completely removable or part of the skid and detached during transit.

The Central Storage Facility can be designed to handle multiple cask unloading lines operating simultaneously and in parallel. Elements not present at reactor sites might include trolley-based cask handling systems, among others. Adopting a bolted canister/transport overpack design could further enhance this increased operational flexibility, designed around the shipping cask elements. Having multiple/independent canister inserts may allow a reduced number of transport overpacks to be built and serviced. For example, the handling facility pool could consist of a submerged transport overpack pit and one or more canister pits, in this way decoupling the cask/assembly unloading operations.

9.11 Recommendations for special features implementation

Several of the potential special features described qualitatively in the previous sections could be implemented in either the short or medium term, while others may require more extensive feasibility and trade-off studies to explore their potential for (or exclusion from) implementation. Some examples of preliminary trade-off studies are presented in *Section 8* of this report.

One of the most promising areas that does not involve extensive cask design modifications is the use of automation of trunnion and lid bolting and unbolting operations, which can result in substantial time and dose savings to operators working in the vicinity of loaded casks. A feasibility study would help identify the various automation alternatives and evaluate their design, material, and fabrication costs relative to the resulting time and dose savings (including performing ALARA analysis).

Other areas for further investigation include the strategies to minimize contact of exposed cask surfaces (especially external surfaces) to contaminated pool water to facilitate decontamination and reduce cask drying times. A trade-off and feasibility in the use of either temporary sleeves or skirts, or more permanent solutions such as chemical coatings or other hydrophilic surface treatments (need to be demonstrated to be chemically compatible with receiving and shipping pools) could determine which option would result in the best combination of waste minimization, time/dose reduction, lowest cost, fabricability/maintainability, etc.

While some special features that increase throughput may be more feasible to implement in a new facility design (e.g. for a consolidated storage facility), a feasibility study could help determine which of these can also be implemented at reactor sites. This includes determining the feasibility to pre-stage and process casks outside the reactor pool area using a dedicated cask-handling crew (and associated temporary structures and cask moving equipment) that supplements site-based workers assigned to pool-based operations.

Longer term design-related special features amenable to feasibility and trade-off studies include the potential for additive manufacturing, incorporation of special light-weight materials for structural components and higher-density shielding materials that could reduce cask component size and/or weight, increase cask capacity, improve thermal or criticality margins, or any combination of these features. Designing and fabricating casks with these options will need to be compared to the current baseline design, balanced against associated costs, fabrication limitations, and licensing challenges.

10.0 CONCLUSIONS

Using AREVA's experience designing, licensing, procuring, and implementing UNF storage and transportation cask systems for commercial utilities and the DOE, the 6625B-HB transportation cask system was developed for the transportation of PWR and BWR UNF. The 6625B-HB transportation cask system is a reusable, rail-type transportation package currently designed to handle four different basket designs for bare and canistered (in DFCs) PWR and BWR UNF. The baskets are configured to contain 24 PWR UNF assemblies and 61 BWR UNF assemblies either bare or packaged in DFCs. The cask system utilizes bolted inner and outer lids allowing for reuse and providing credit for moderator exclusion under specific conditions.

To demonstrate reasonable assurance that the 6625B-HB transportation cask can be licensed by the NRC under 10 CFR 71, *Section 2* of this report is aligned with the layout of a transportation cask SAR as established in the SRP for transportation packages (NUREG-1617). This ensures information addressed in a SAR by an applicant to the NRC is included in this report and, provides reasonable assurance the 6625B-HB transportation cask can be licensed by the NRC. Actual licensing of this cask system by the NRC would require completion of detailed design calculations, fabrication drawings, etc. that are outside the scope of this TO.

Section 2 includes general design information of the 6625B-HB transportation cask design (*Section 2.1*) that provides the bases for the 10CFR71 licensing review and an overview of the key regulatory requirements that impacted the design activities performed in this TO. Specifically, the regulations related to thermal, shielding, containment, and criticality plus the subsequently established weight limit and minimum cavity length provide the key criteria impacting the design of the 6625B-HB. This section also provides details of the PWR and BWR basket designs, the specifications for the PWR and BWR fuel acceptable for transport in the 6625B-HB, and operational features of the 6625B-HB. *Section 2.1* also provides the general arrangement drawings for the 6625B-HB package.

These general arrangement drawings for the 6625B-HB were the product of an iterative process between several engineering disciplines (civil, mechanical, and nuclear) and the results of the iterative process are documented in *Sections 2.2* through *2.8*. The starting point was working from the weight limit and cavity length limit to establish an approximate cask body design. Once the cask body design was identified the basket designs were developed closely with the thermal model to ensure sufficient heat would be removed from the fuel to ensure component temperatures were not exceeded under NCT or HAC conditions. Concurrent with the thermal analyses, the radiation shielding analyses were performed to ensure dose rate requirements were satisfied. After design changes were made to meet the thermal and radiation shielding requirements, the weight of the cask system had to be re-evaluated to ensure weight limits were maintained. Once the weight, thermal, and shielding criteria were met, the structural evaluation (including the impact limiters) and the criticality analyses were performed. If they required any changes to the design, then another iteration was initiated, starting with the weight evaluation. Finally, the operating and maintenance requirements were evaluated for the designed cask and could result in yet another design iteration if changes were necessary to optimize or simplify operations and/or reduce burdens of maintenance requirements.

The most limiting criteria for the 6625B-HB involved the weight, thermal, and shielding requirements (essentially in that order of importance) and required the most iterative activities.

The results for the weight, thermal, and shielding assessments are documented *Sections 2.2.1*, *2.3*, and *2.5*, respectively. The structural and criticality criteria posed the next most restrictive set of limiting requirements and the results from their assessments are documented in *Sections 2.2.2* and *2.6*, respectively. The operating and maintenance requirements posed the least restrictive criteria on the design of the 6625B-HB, but this is really the product of the extensive, positive experience the AREVA team has had with loading cask systems at utility SFPs. *Sections 2.7* and *2.8* document the operating procedures and maintenance requirements for the 6625B-HB, respectively.

A summary comparison of various characteristics of each of the baskets designed for the 6625B-HB is provided in **Table 10-1**. In addition to the heat load zoning identified in this table, alternative configurations were considered to allow for the loading of UNF at a higher average heat output, but with fewer UNF assemblies placed in the cask (short loading). These alternative cases were examined to demonstrate the potential flexibility of the 6625B-HB to unload SFPs that may only have UNF at higher heat outputs than those identified for the base case for the 6625B-HB. Generally, if the total heat output of the UNF loaded into this cask falls below the 30.4 kW PWR limit and the 30.2 kW BWR limit through the short-loading of assemblies then, as confirmed through analyses, the 6625B-HB can generally be utilized to move this UNF.

In addition to materials provided to demonstrate reasonable assurance of licensability of the 6625B-HB, *Sections 3* through *9* address specific items/activities requested in the SOW. *Section 3.0* provides information on the estimated cost ranges associated with: design and analyses for licensing the 6625B-HB, licensing of the 6625B-HB, and fabrication of the FOAK and n^{th} of a kind 6625B-HB. *Section 4* takes the operations described in *Section 2.7* and performs a time-dose study to estimate total doses, total person-hours, total duration/clock time, and total number of shifts for preparation, loading, and unloading operations for PWR and BWR UNF in both bare and in DFC configurations. *Sections 5, 6,* and *7* cover material specifically requested in the SOW for material that is mostly covered in *Section 2*. *Section 8* contains three trade studies on: (1) the impacts DFCs have on the 6625B-HB design (e.g., could more UNF assemblies be placed into the 6625B-HB if DFCs were not used?); (2) the impact fuel length has on the 6625B-HB design (e.g., what are the penalties for making the 6625B-HB longer to accept a larger spectrum of UNF assemblies?); and (3) the ability of the 6625B-HB to load damaged and potentially failed UNF (as defined in *Section 2.1*). The results of these trade studies reveal: the 6625B-HB is already designed to accept UNF assemblies in DFCs in every position of the basket and given the dimensional and weight limitations, no further UNF assemblies can be placed into the cask even if all the DFCs were removed from the cask; increasing the length of the 6625B-HB would potentially reduce the number of UNF assemblies the cask can handle as the increased length increases the weight of the cask and hence, a proportional reduction in the diameter and/or quantity of UNF loaded would be needed; and the design of the 6625B-HB allows for multiple damaged UNF assemblies in DFCs to be loaded into it and qualitatively should be able to accept limited quantities of failed UNF once the failed UNF has been properly characterized and subsequently qualified through calculations.

Section 9 includes special features which could be introduced into the cask design and operational concepts to allow for system optimization. The features considered were those that could: increase the capacity of the 6625B-HB, decrease the weight of the 6625B-HB, reduce dose rates from the 6625B-HB, enhance thermal performance of the 6625B-HB, improve criticality margins of the 6625B-HB, provide alternative equipment for the handling of the

6625B-HB, automate operations to reduce cumulative doses and reduce operating durations, reduce decontamination and/or cask drying times, increase the flexibility for the handling of the 6625B-HB, and improve general throughputs associated with the 6625B-HB. The concepts identified in this section require further evaluation and could result in additional benefits to the unloading of SFPs using a cask system like the 6625B-HB.

Finally, *Appendix A* contains information regarding an exercise performed with Duke Energy to examine the ability of the 6625B-HB to empty the SFPs at the operating Duke reactor sites at a rate sufficient to eliminate the need to move any additional UNF into onsite dry storage systems, while at the same time trying to minimize any increase in the duration necessary to load cask systems (storage versus transportation) at the SFP. The results from this exercise informed several of the items/activities performed for this TO including the evaluation of: (1) the ability to efficiently perform loading operations of the 6625B-HB; (2) the effectiveness of the 6625B-HB for off-loading the contents of SFPs for both PWR and BWR fuel; (3) alternative loading patterns for UNF into the existing basket structure of the 6625B-HB; and (4) the need for mixing the loading of bare UNF and UNF in DFCs. The results from this exercise included identifying that almost all of the Duke UNF assemblies met the criteria of the 6625B-HB fuel qualification tables and hence could be loaded into the 6625B-HB without the need for further evaluation. Once the existing UNF in the Duke SFPs was assessed, predictions were then made for the future SFP inventory and its characteristics based on trends from the current core unloading activities. With the predicted SFP inventory, an assessment on the ability of the 6625B-HB to unload current and future inventories was performed to establish the total number of casks required to unloaded the SFPs. In addition, this assessment established how many short-loadings of the 6625B-HB would be required to empty these SFPs and if an alternative fuel loading pattern would be merited for the 6625B-HB (which was performed but with limited benefit). This assessment also identified the impact of the number of 6625B-HB's used to empty the SFP per year on the overall inventory within the SFP (e.g., how many 6625B-HBs need to be used to unload UNF from the SFP at a rate sufficient to ensure the number of UNF assemblies in the SFP remains below its capacity?). Assessments of this type identify vital information to the plant operators (e.g., can the transportation casks used to empty the SFP at an "adequate" rate be performed within the period of current loading activities of dry storage canisters/casks?) and to the supplier of the transportation casks and rail consists (e.g., how many casks need to be sent in a shipment to a reactor site for "optimal" removal of UNF from the SFPs to prevent the need for further loading of dry storage canisters/casks?), which allow them to make future plans (e.g., procurements of transportation/storage systems for UNF).

In addition to assessing the capability of the 6625B-HB to effectively and efficiently remove the UNF from the SFPs, an assessment was also performed on the capabilities of the infrastructure (e.g., crane and floor capacities) at the Duke SFPs to handle the 6625B-HB. This assessment revealed potential gaps in the ability of some of the plants to handle the weight of the loaded 6625B-HB, resulting in potentially short-loading the 6625B-HB or requiring a slimmed down version of the 6625B-HB. This type of assessment should be performed for all SFPs to establish actual weight and height restrictions for these facilities, which in turn would establish if alternative cask designs should be considered.

The final portion of *Appendix A* contains information on the quantity and type of damaged/failed UNF that exists in the Duke SFPs. This information was used to inform the trade study performed in *Section 8* on the loading of damaged and failed UNF into the 6625B-HB. Although the Duke plants are likely a fair representation of the remaining plants in the U.S., the pursuing of this data (e.g., through the GC-859 process) is important to establish: how much damaged and failed UNF the transportation cask should be designed to handle and the extent of the damage to the UNF to establish if this UNF can be placed in DFCs and to provide bounds for the analyses needed to demonstrate the cask is capable of handling this UNF (e.g., criticality).

TABLE 10-1: SUMMARY COMPARISON OF BASKET CONFIGURATION

Characteristic	Bare PWR UNF	PWR UNF in DFC	Bare BWR UNF	BWR UNF in DFC
Fuel Assembly Capacity (#)	24	24	61	61
Total Loaded Weight of Package for Transport (lbs.)	298,882	300,202	292,259	294,152
Maximum Width (inches)	126	126	126	126
Maximum Length with Impact Limiters (inches)	261.5	261.5	261.5	261.5
Maximum Length without Impact Limiters (inches)	200.5	200.5	200.5	200.5
Cask Outisde Diameter without Impact Limiters (inches)	93.25	93.25	93.25	93.25
Cask Cavity Inside Diameter (inches)	66.25	66.25	66.25	66.25
Cask Cavity Length (inches)	182.0	182.0	182.0	182.0
Hook weight 1 (lbs.) filled with water and inner lid installed	250,037	251,357	248,268	250,159
Hook weight 3 (lbs.) water removed and inner & outer lid installed	247,148	248,467	240,526	242,417
Maximum Heat Load (kW)	30.4	30.4	30.3	30.3
Maximum Heat Output per Assembly per Zone (kW)	0.9, 1.4, 2.1, 0.9	0.9, 1.4, 2.1, 0.9	0.33, 0.78, 0.45, 0.33	0.33, 0.78, 0.45, 0.33
Maximum Fuel Cladding Temperature – NCT (°F)	582	578	552	552
Maximum Fuel Cladding Temperature – HAC (°F)	860†			
Maximum Accessible Surface Temperature (°F)	126†			
Maximum Cavity Pressure – NCT (psig)	9.3‡		11.0‡	
Maximum Cavity Pressure – HAC (psig)	63.6‡		67.7‡	
Maximum Dose Rate 2m from Cask Surface – NCT (mrem/hr)	9.3‡		9.3‡	
Maximum Dose Rate at Surface of Impact Limiter – NCT (mrem/hr)	33.8‡		63.2‡	
Maximum Dose Rate 1m from Cask Surface – HAC (mrem/hr)	875.7‡		920.9‡	

Characteristic	Bare PWR UNF	PWR UNF in DFC	Bare BWR UNF	BWR UNF in DFC
Max. k_{eff} + 2σ - NCT (wet)	0.9144	0.9145	0.9252	0.9316
Max. k_{eff} + 2σ - HAC (wet)	0.9386	0.9386	0.9488	0.9549
Design/Analyses Costs for Total Cask System ($)	5,861,321 – 6,842,365			
Licensing Costs for Cask System ($)	1,653,715 - 2,268,121			
Fabrication Costs for Cask System ($)	5,145,168 – 8,222,436	5,211,144 – 8,304,900	5,007,723 – 8,050,630	5,155,282 – 8,235,094
Total Cumulative Dose for Loading & Unloading UNF (mrem)	1002.2	1002.6	1003.1	1004.1
Total Loading and Unloading Duration (hr)	79.1	88.7	101.3	125.7

† Listed values are maximum for all basket cases.

‡ DFCs were not modeled in the shielding analyses as the DFCs provide additional shielding and hence, the bare fuel models are considered to provide bounding values.

In conclusion, this report provides a summary of the analyses performed for the 6625B-HB transportation cask system which is designed to:

- Have reasonable assurance of receiving an NRC license under 10 CFR 71
- Be reusable (crediting regular maintenance and inspection activities)
- Meet Plate B criteria in AAR S-2043 (for rail transport)
- Contain either 24 PWR or 61 BWR intact fuel assemblies (bare or in DFCs)
- Handle high burnup UNF up to 62.5 GWd/MTU cooled a minimum of 5 years
- Handle the majority of the existing UNF inventory in the U.S.
- Have impact limiters that are identical at either end of the cask
- Optimize loading and unloading operations (fast as possible and ALARA)

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APPENDIX A: UTILITY STUDY

A.1 SFP and DSC Study

A.1.1 Overview

This appendix contains information collected from our utility partner for this TO, Duke Energy. Duke Energy currently has six sites with operating nuclear plants. In this appendix, data collected from Duke Energy on the contents of their SFPs, their approximate refueling schedule, and current dry storage cask loading activities at the SFPs was utilized to assess the capability of the 6625B-HB transportation cask to unload the SFPs and eliminate the need for any additional onsite dry storage cask-loading activities while at the same time trying to minimize any increase in the duration necessary to load cask systems at the SFP.

The material presented in this appendix was used to inform several of the other tasks contained within this report, which include:

- Evaluating the need and ability to more efficiently perform loading operations of the 6625B-HB;
- Establishing the effectiveness of the 6625B-HB for off-loading the contents of SFPs for both PWR and BWR fuel (e.g., will short loading occur?);
- Providing feedback on the zone loading of the 6625B-HB (e.g., are there too few or too many slots assigned to the highest heat load zones in the 6625B-HB?);
- Identifying the potential need for mixing the loading of bare UNF and UNF in DFCs.

Typical sample utility data for DSC loading data is provided in this appendix (see **Table A.1-4**). The sample DSC data sets and SFP FA inventory data are used for estimating typical UNF transport cask-loading values and the results are shown in **Table A.1-12**. The methodology used to determine the “6625B-HB Performance Assessment” efficiencies and loading values shown in the table is discussed in *Section A.1.3.1*. Loading timeline estimates for the utility SFP inventories are presented *Section A.1.4*.

In addition, this appendix contains utility data for weight assessments (*Appendix A.2*) and damaged fuel loading (*Appendix A.3*) used to inform other tasks contained within this report.

A.1.2 Qualification Criteria and Performance

Fuel qualification tables, criticality burnup curves, and resulting performance plots as discussed in the following sections are used to verify bounding conditions and estimate 6625B-HB “average percent full” loading efficiencies. The number of 6625B-HB loads required to empty the SFPs over the reactor life are then determined using the 6625B-HB “average percent full” efficiencies.

A.1.2.1 Fuel Qualification Table (FQT)

The PWR and BWR fuel qualification tables (FQT) used for the 6625B-HB are provided (see **Table A.1-1** and **Table A.1-2**). A linear interpolation approach was utilized for the number of years that each FA was in the SFP as a function of burnup (for details see **Figure A.1-1** and **Figure A.1-3**), since the FQT discrete step sizes are larger than those typically used with FA for a DSC. This is shown mathematically in each FQT by replacing the minimum fixed number of years a FA must cool (i.e., a constant) with the minimum cooling time as a linear function of burnup (i.e., $y(x) = m \cdot x + b \rightarrow$ Min Cooling Time (B) = $m \cdot B + b$, where “B” is the burnup, “m” is the linear slope, and “b” is the cooling time axis intercept). Without interpolating, several FAs were unnecessarily rejected, because of the large “jumps” between data points in the FQT. Interpolation more closely matches the true limit function of years cooled versus burnup, to more accurately assess if a FA is acceptable, while still remaining conservative (i.e., **Figure A.1-2**) shows that the linear interpolation only accepts those FA that are acceptable per the actual function limit). Treating the limits in this manner is less penalizing than treating them as a constant, when there are large “jumps” between the FQT data points. Therefore, it is the recommended approach. In general, this approach of linear interpolation is recommended for any of the requirement or performance evaluation curves that have large “jumps” between the evaluation points (provided a linear interpolation reasonably represents the true function between the discrete evaluation points and remains conservative for evaluation).

TABLE A.1-1: INTERPOLATED PWR FUEL QUALIFICATION TABLE (TYPICAL)

		Minimum Cooling Time (years)		
		Zone 1/4	Zone 2	Zone 3
Maximum Burnup (GWd/MTU)	Minimum Enrichment (%)	Heat ≤ 0.9 kW	Heat ≤ 1.4 kW	Heat ≤ 2.1 kW
≤ 30	≥ 1.8	≥ 5	≥ 5	≥ 5
≤ 37	≥ 2.3	≥ 0.2143B-1.4286	≥ 5	≥ 5
≤ 45	≥ 2.8	≥ 0.4375B-9.6875	≥ 5	≥ 5
≤ 53	≥ 3.3	≥ 0.7500B-23.750	≥ 0.1875B-3.4375	≥ 5
≤ 62.5	≥ 3.8	≥ 1.0526B-39.789	≥ 0.2632B-7.44749	≥ 5

FIGURE A.1-1: EXAMPLE FQT INTERPOLATED PWR YEARS COOLED VS BURNUP (ZONE 1/4)

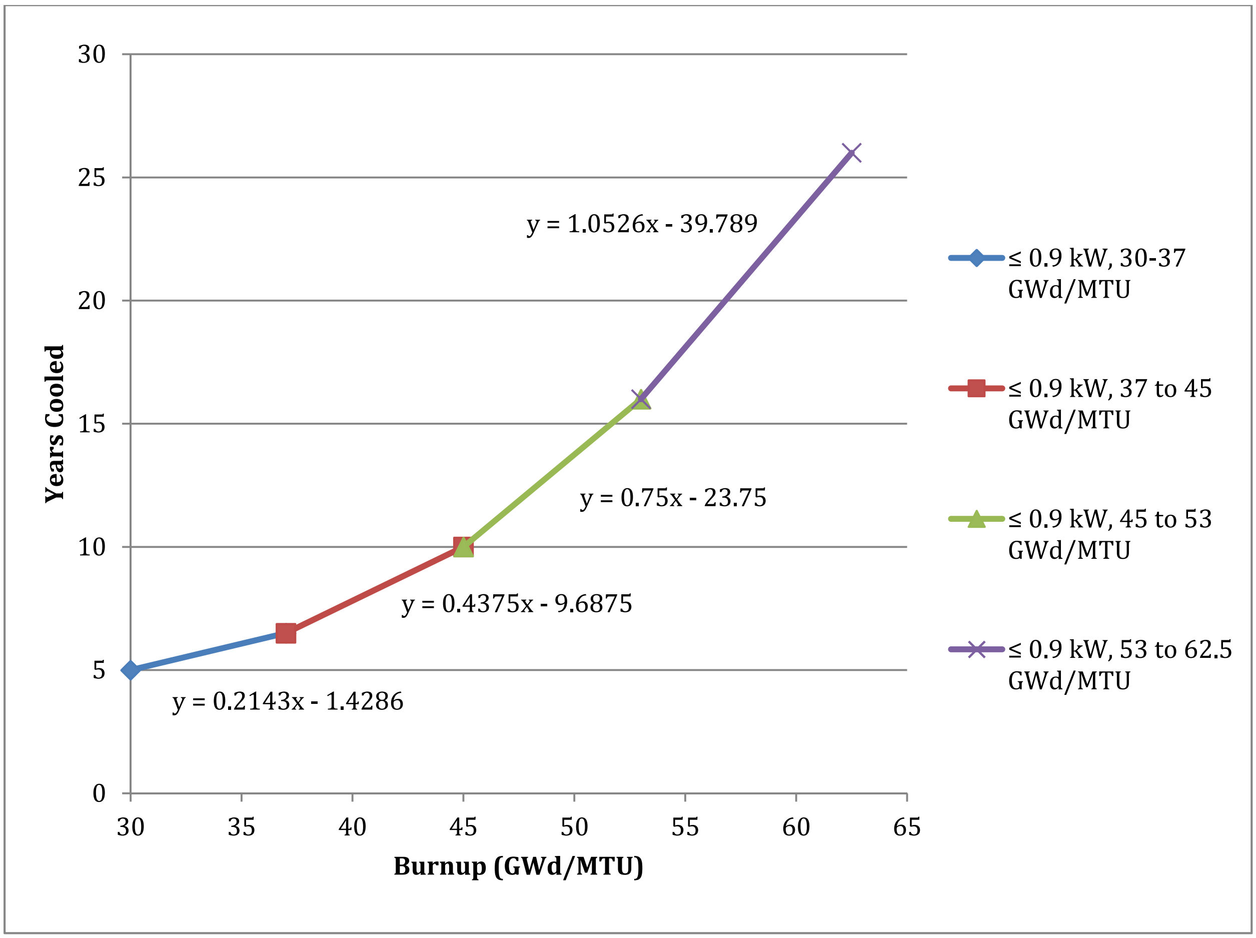


FIGURE A.1-2: PWR FQT INTERPOLATION DETAIL YEARS COOLED VS BURNUP (ZONE 1/4)

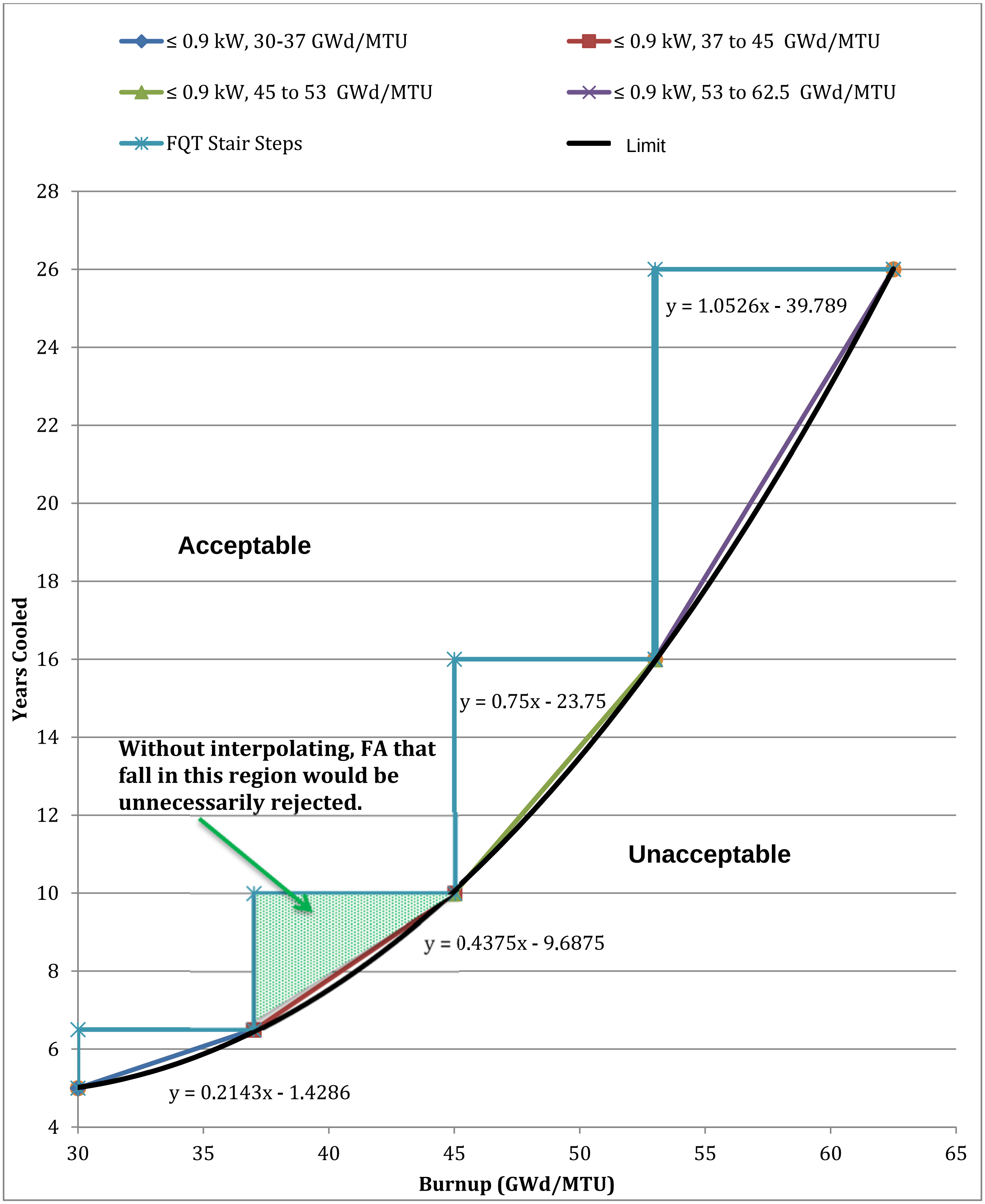
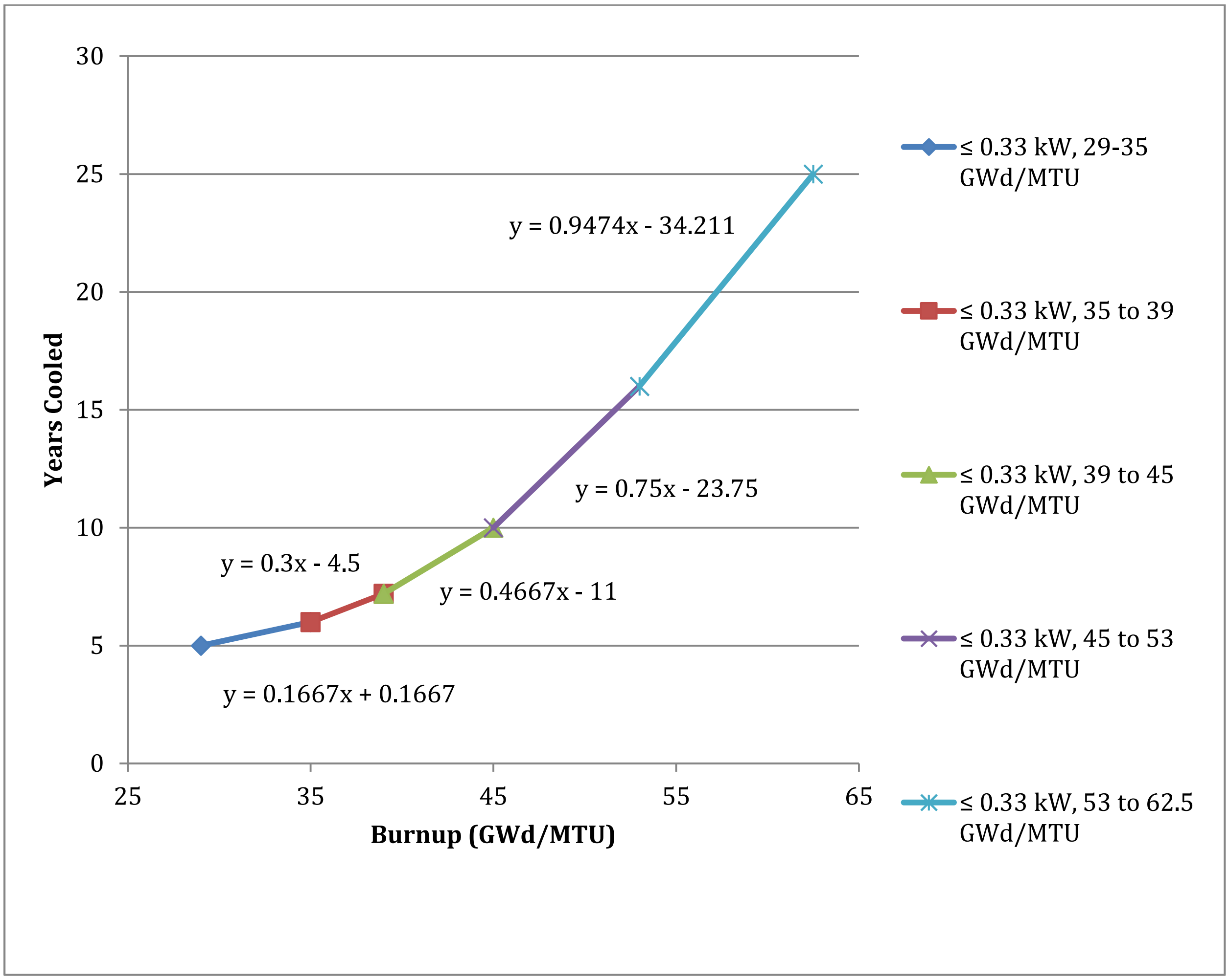


TABLE A.1-2: INTERPOLATED BWR FUEL QUALIFICATION TABLE (TYPICAL)

		Minimum Cooling Time (years)		
		Zone 1/4	Zone 2	Zone 3
Maximum Burnup (GWd/MTU)	Minimum Enrichment (%)	Heat ≤ 0.33 kW	Heat ≤ 0.78 kW	Heat ≤ 0.45 kW
≤ 29	≥ 1.5	≥ 5	≥ 5	≥ 5
≤ 35	≥ 2.2	≥ 0.1667B + 0.1667	≥ 5	≥ 5
≤ 39	≥ 2.4	≥ 0.3B - 4.5	≥ 5	≥ 5
≤ 45	≥ 2.8	≥ 0.4667B - 11	≥ 5	≥ 0.1667B - 1.5
≤ 53	≥ 3.3	≥ 0.75B - 23.75	≥ 5	≥ 0.25B - 5.25
≤ 62.5	≥ 3.8	≥ 0.9474B - 34.21	≥ 5	≥ 0.4737B - 17.11

FIGURE A.1-3: EXAMPLE FQT INTERPOLATED BWR YEARS COOLED VS BURNUP (ZONE 1/4)



A.1.2.2 Criticality Burnup Curves

Similar to the FQT, the criticality burnup curves are evaluated at specific enrichments and burnup values. Therefore, this method of linear interpolation between the discrete evaluation points (see Appendix A.1.2.1) is also recommended for the criticality burnup curves. A summary of how many FAs satisfy the burnup limit requirement is provided in **Table A.1-3**. Burnup and enrichment data from 242 PWR FAs were considered in this study. Of the 242 PWR FAs, 97.9% were identified as acceptable per the burnup limit criteria (i.e., a total of 5 or 2.1% were rejected). **Figure A.1-4** (see Appendix A.1.2.3) supports this result graphically. In general, this approach of linear interpolation is recommended for any of the requirement or performance evaluation curves that have large "jumps" between the evaluation points (provided a linear interpolation reasonably represents the true function between the discrete evaluation points and remains conservative for evaluation).

A.1.2.3 Fuel Qualification Criteria Performance Plots

Several performance plots of the Duke Energy FA data to various qualification criteria are provided to assess typical trends. The method of linear interpolation (as discussed in Appendix A-1.2.1) is also recommended for the performance plots.

- **Figure A.1-4**: PWR FA Allowable Burnup vs Enrichment (Plants A – D)
  - The FA data almost exclusively fall within the acceptable range (97.9% acceptance). There are a few FA's that are close to the limit (2.1%). However, these are all from Plant D and are from older DSC loading data (e.g., 2005). See *Section A.1.2.2* for additional discussion of the criticality burnup curves.
- **Figure A.1-5**: PWR FA Years Cooled vs Burnup (Plants A – D)
  - All the PWR FA's are within the acceptable range. A significant portion of the PWR FA's are tightly packed within Zone 2.
- **Figure A.1-6**: BWR FA Years Cooled vs Burnup (Plant E)
  - All the BWR FA's are within the acceptable range, with a generally even distribution of FA's between the zones.
- **Figure A.1-7**: PWR FA Design Type (Plants A – D)
  - This figure provides a distribution of the PWR FA design types used in Plants A - D. There is not a distribution plot provided for Plant E (BWR), since all the FAs in Plant E considered in this study are "GE 9x9" and "GE 10x10" .
- **Figure A.1-8**: PWR FA Design Type DistributionFIGURE A.1-8: PWR FA
  - This figure provides a combined distribution of each PWR FA type. Approximately 63% of the PWR FAs in this study are B&W FA designs.
- **Figure A.1-9**: PWR FA Initial Uranium Loading vs Burnup (Plants A – D)
  - All the PWR FA's are within the acceptable range, as bounded by the B&W 15 x 15 FA limit.
- **Figure A.1-10**: BWR FA Initial Uranium Loading vs Burnup (Plant E)
  - All the BWR FA's are within the acceptable range, as bounded by the GE 7x7 FA limit.
- **Figure A.1-11**: PWR FA Initial Uranium Loading vs Years Cooled (Plants A – D)
  - All the PWR FA's are within the acceptable range, as bounded by the B&W 15 x 15 FA limit.
- **Figure A.1-12**: BWR FA Initial Uranium Loading vs Years Cooled (Plant E)

  - All the BWR FA's are within the acceptable range, as bounded by the GE 7x7 FA limit.
- **Figure A.1-13**: PWR FA Length vs Years Cooled (Plants A – D)
  - All the PWR FA's are within the acceptable range.
- **Figure A.1-14**: BWR FA Length vs Years Cooled (Plant E)
  - All the BWR FA's are within the acceptable range.
- **Figure A.1-15**: PWR FA Weight vs Years Cooled (Plants A – D)
  - All the PWR FA's are within the acceptable range.
- **Figure A.1-16**: BWR FA Weight vs Years Cooled (Plant E)
  - All the BWR FA's are within the acceptable range.

FIGURE A.1-4: PWR FA ALLOWABLE BURNUP VS ENRICHMENT (PLANTS A – D)

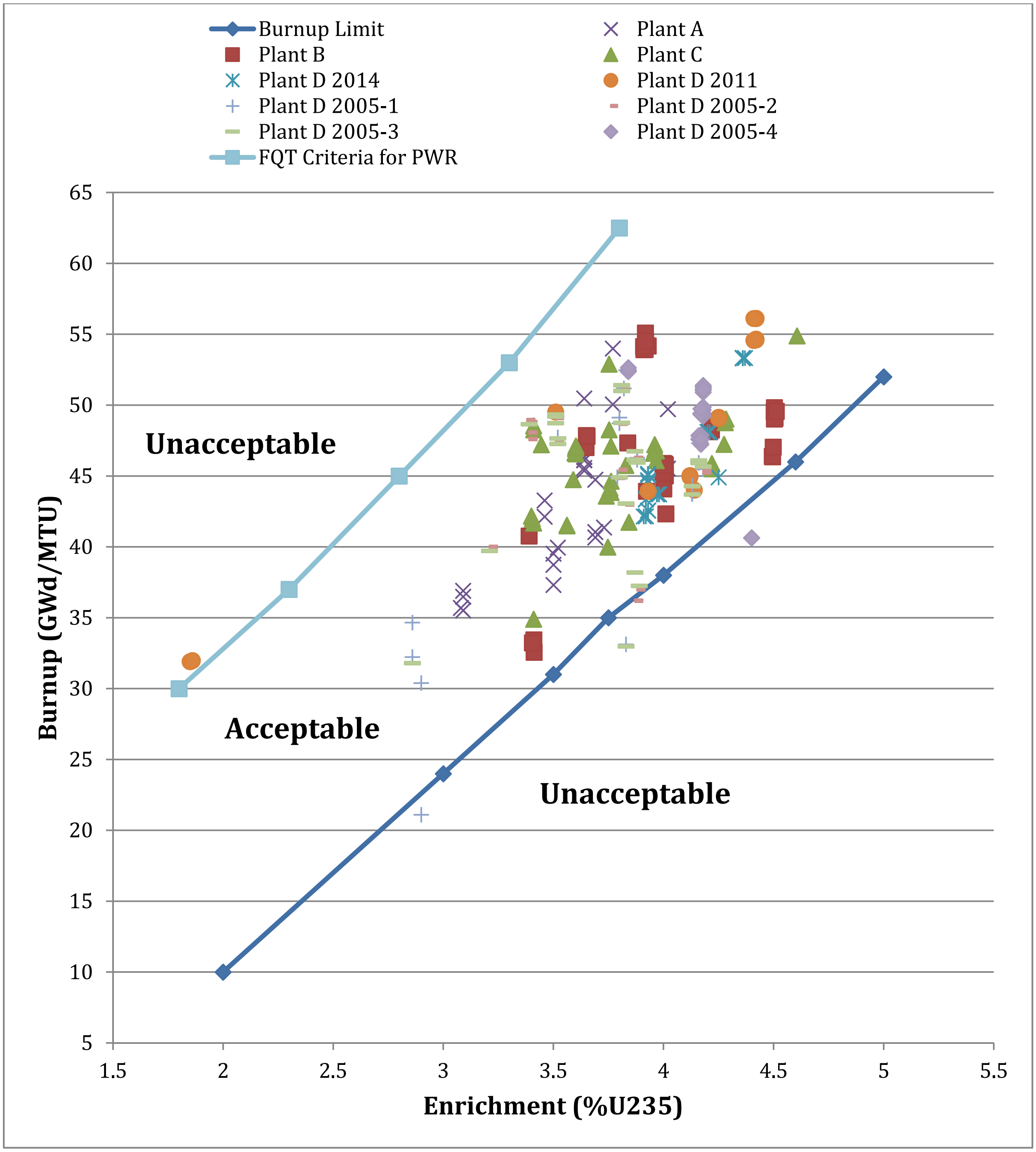


TABLE A.1-3: PWR FA (PLANTS A – D) BURNUP LIMIT PASS/FAIL

Total # PWR FA	Total # PWR FA Pass	Total % PWR FA Pass	Total # PWR FA Fail	Total % PWR FA Fail
242	237	97.9%	5	2.1%

FIGURE A.1-5: PWR FA YEARS COOLED VS BURNUP (PLANTS A – D)

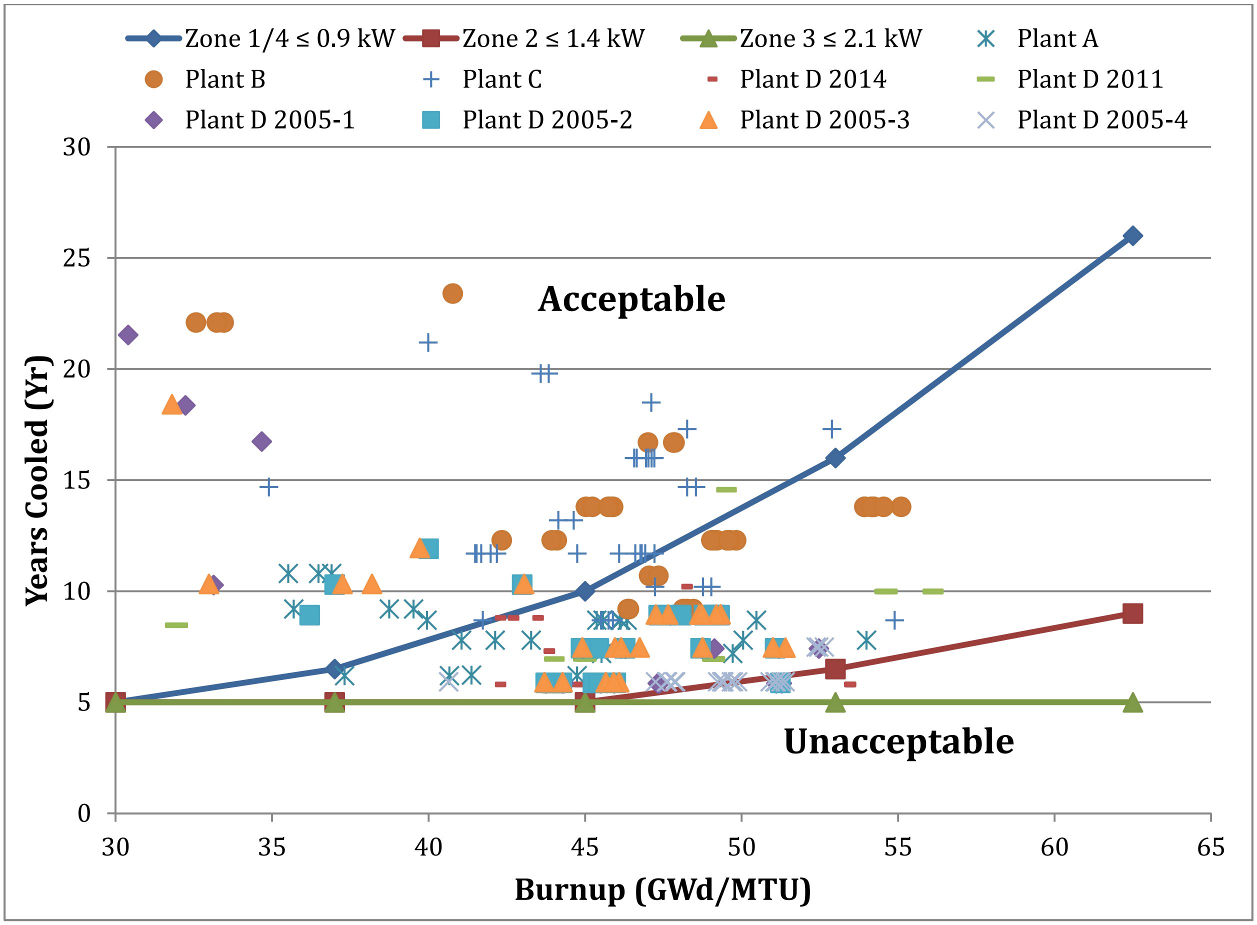


FIGURE A.1-6: BWR FA YEARS COOLED VS BURNUP (PLANT E)

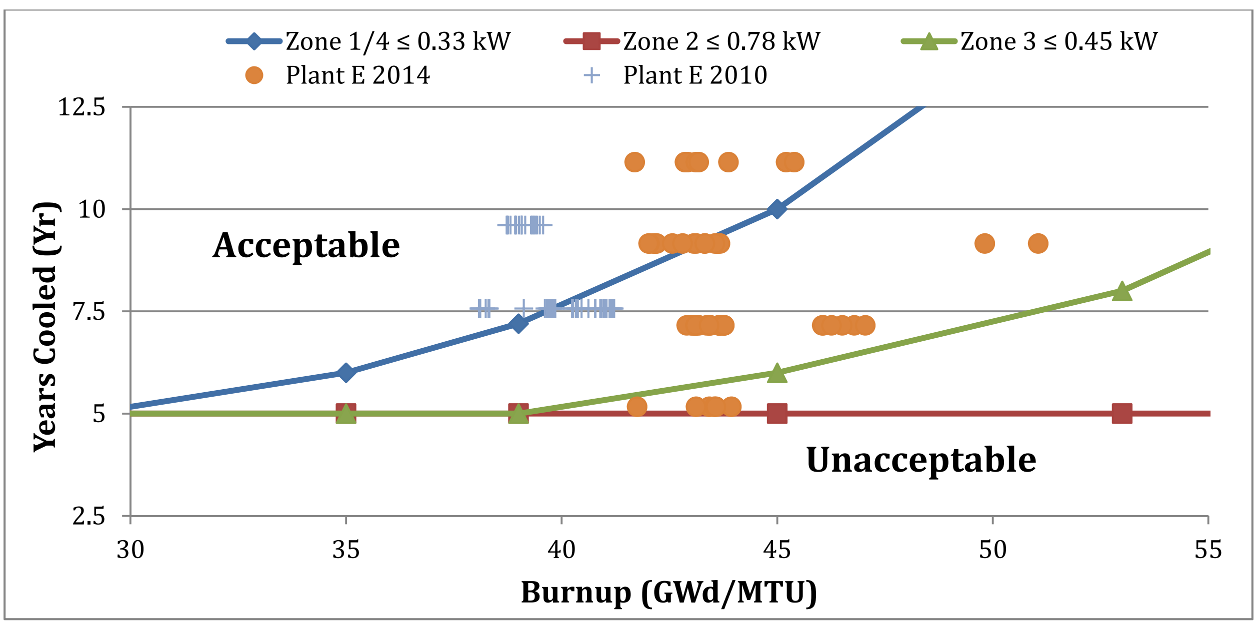


FIGURE A.1-7: PWR FA DESIGN TYPE (PLANTS A – D)

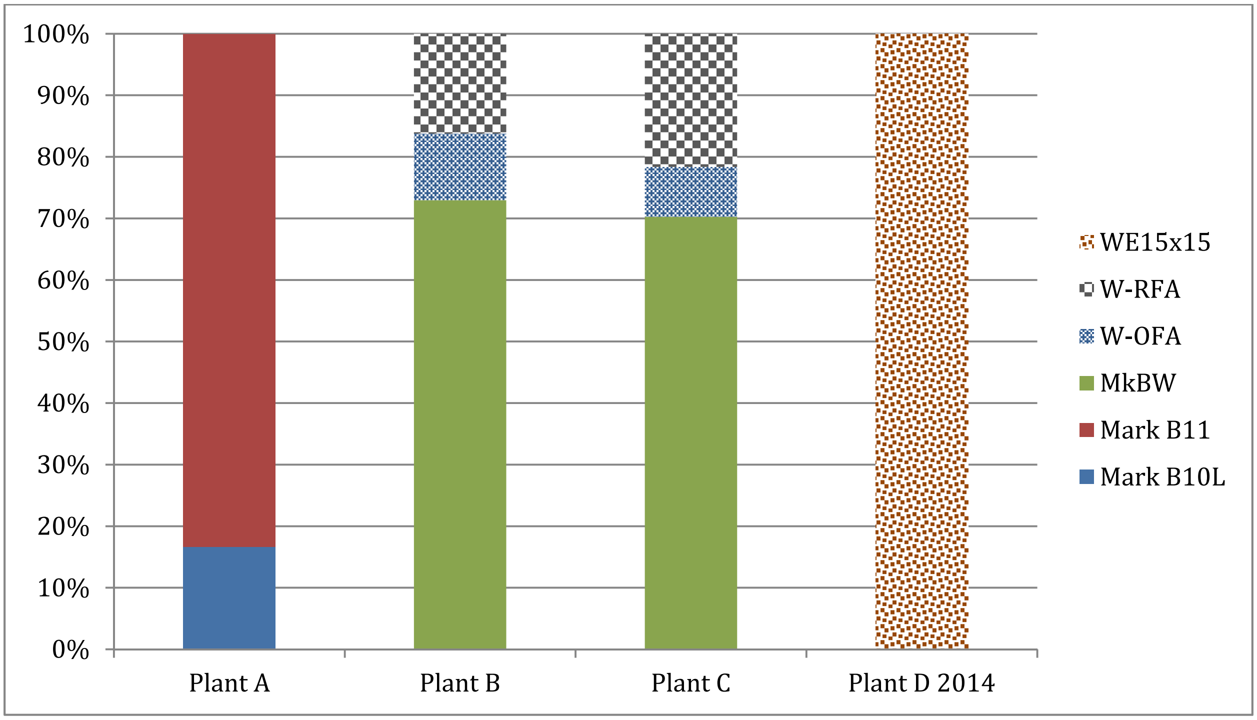


FIGURE A.1-8: PWR FA DESIGN TYPE DISTRIBUTION

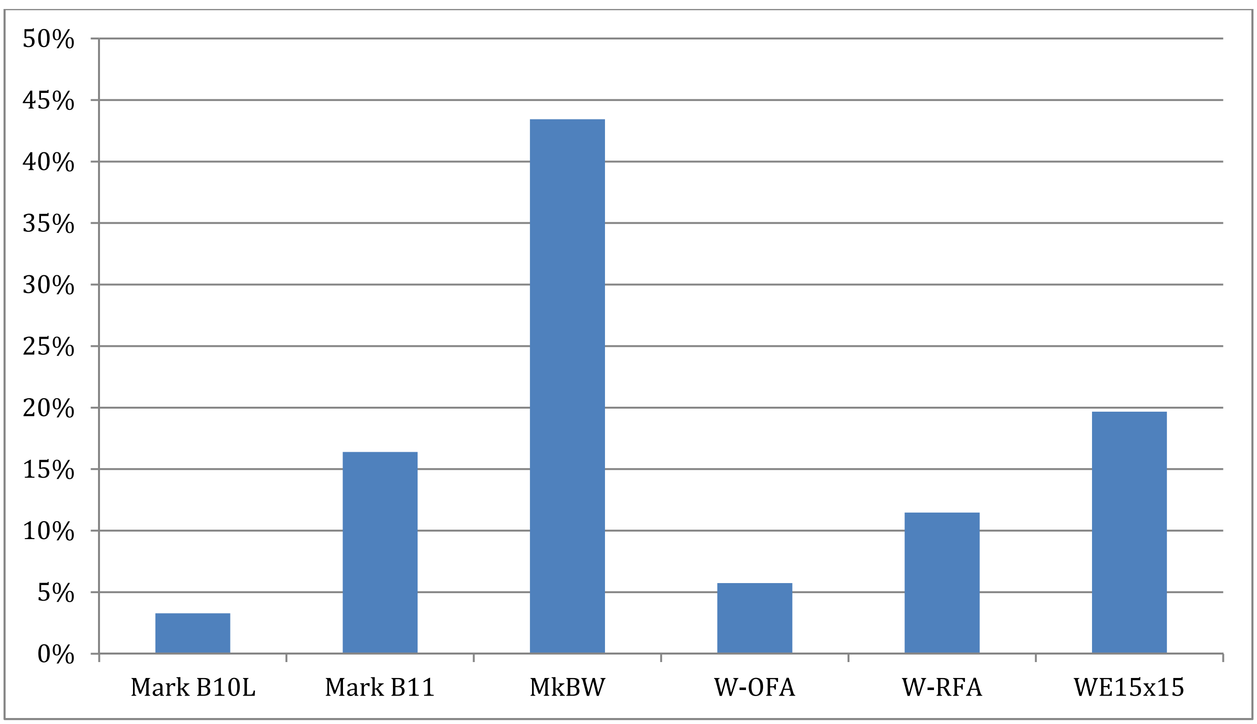


FIGURE A.1-9: PWR FA INITIAL URANIUM LOADING VS BURNUP (PLANTS A – D)

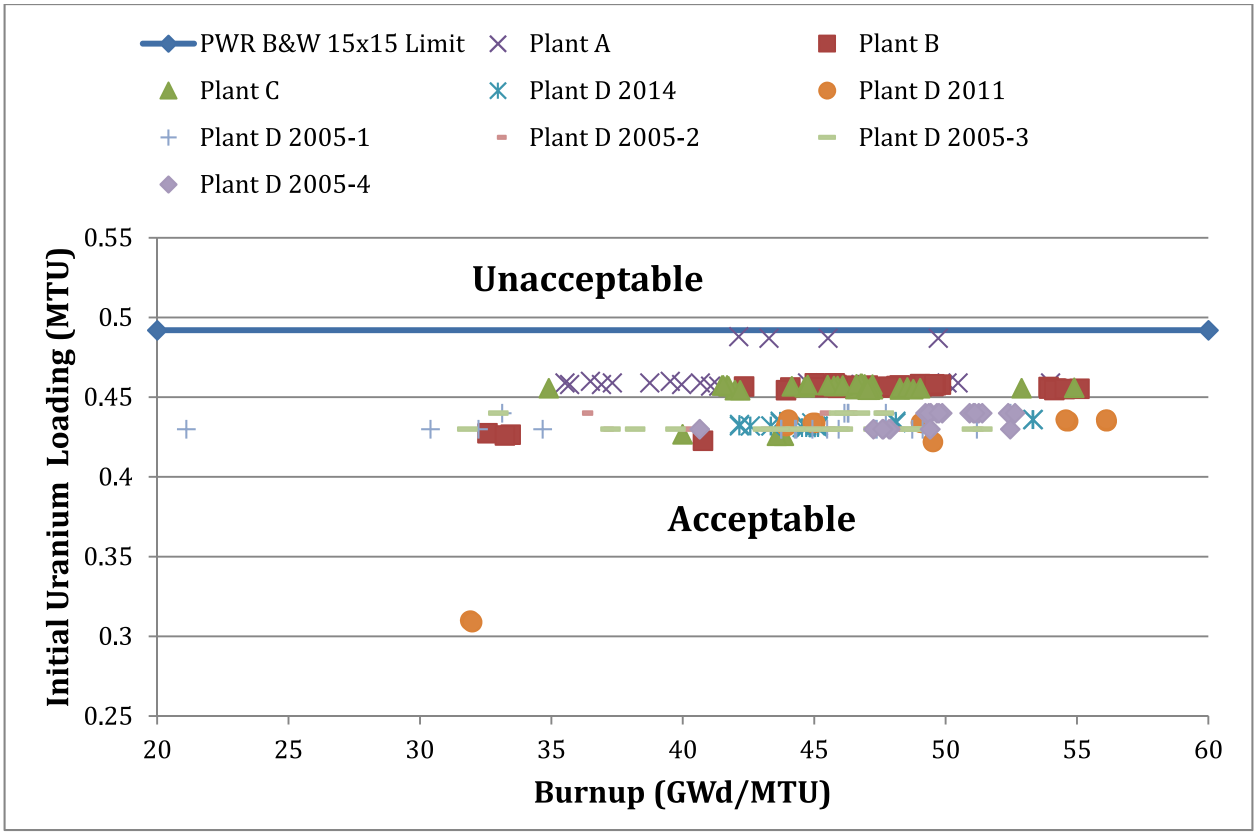


FIGURE A.1-10: BWR FA INITIAL URANIUM LOADING VS BURNUP (PLANT E)

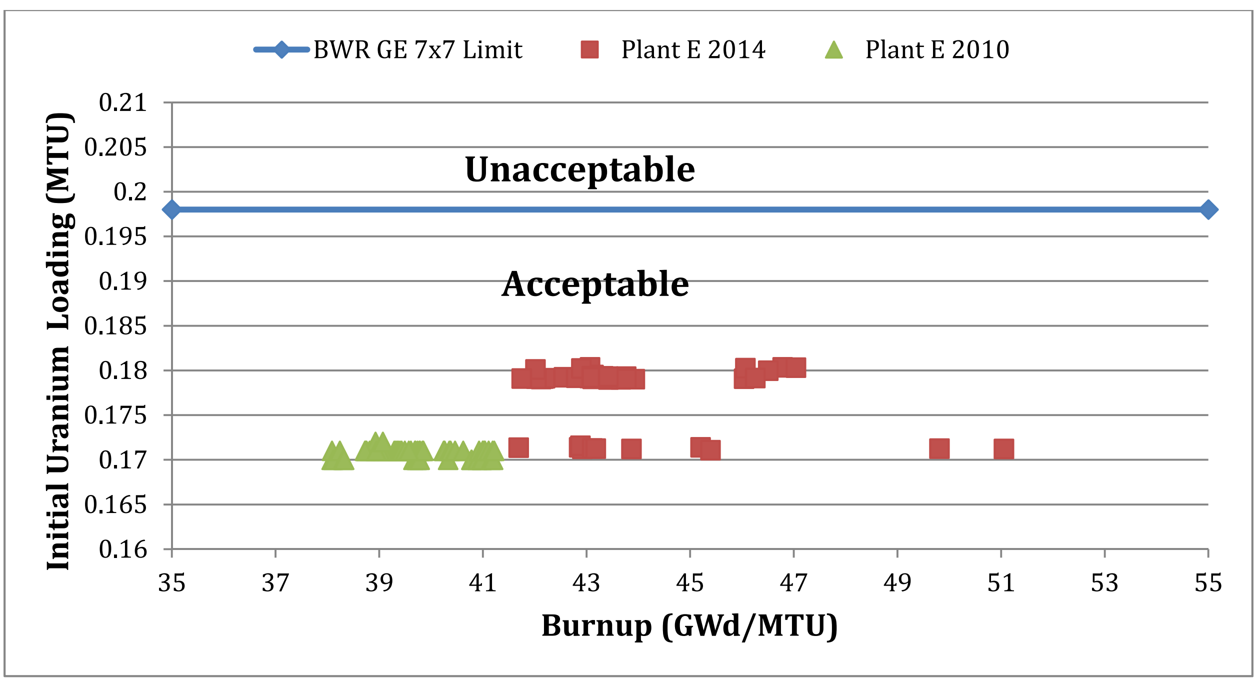


FIGURE A.1-11: PWR FA INITIAL URANIUM LOADING VS YEARS COOLED (PLANTS A – D)

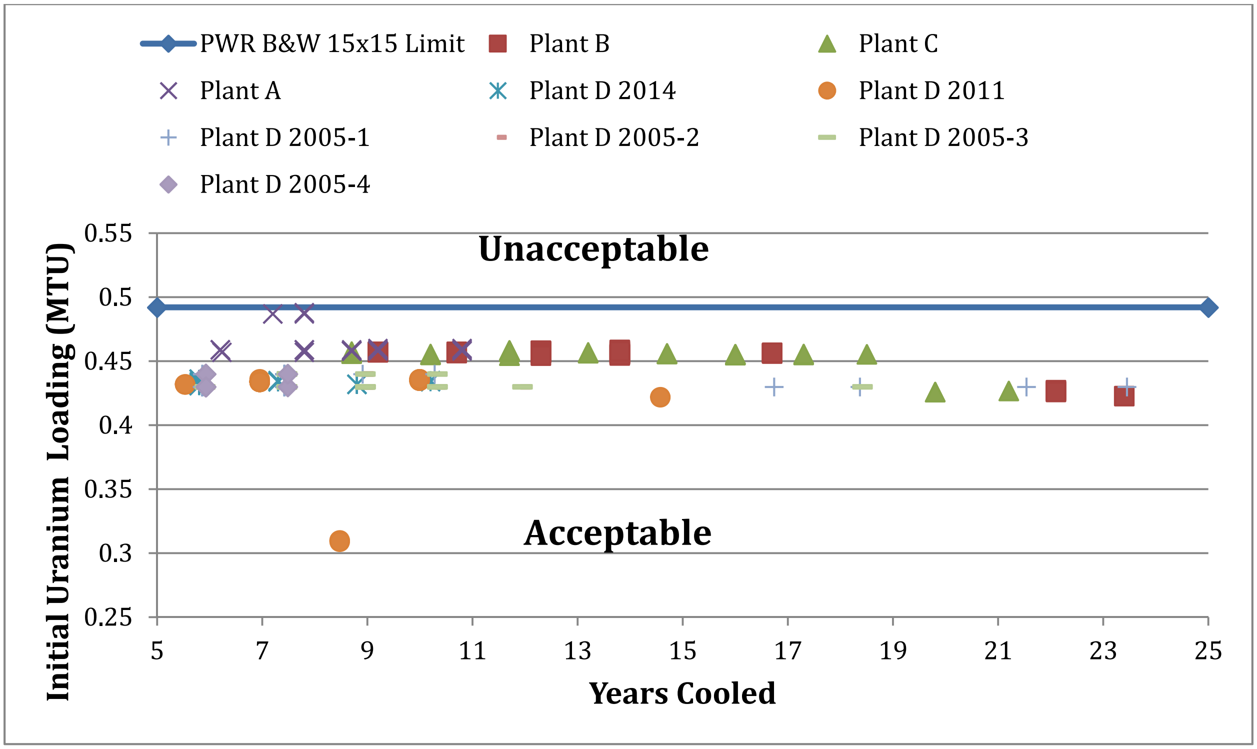


FIGURE A.1-12: BWR FA INITIAL URANIUM LOADING VS YEARS COOLED (PLANT E)

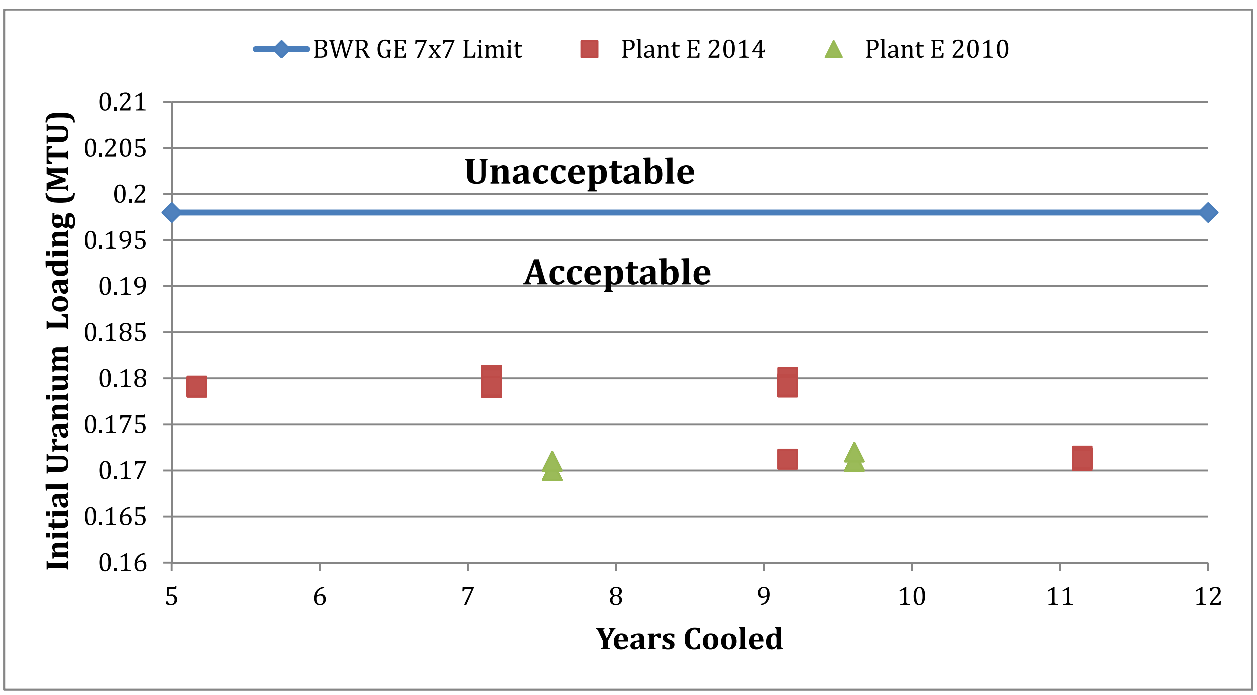


FIGURE A.1-13: PWR FA LENGTH VS YEARS COOLED (PLANTS A – D)

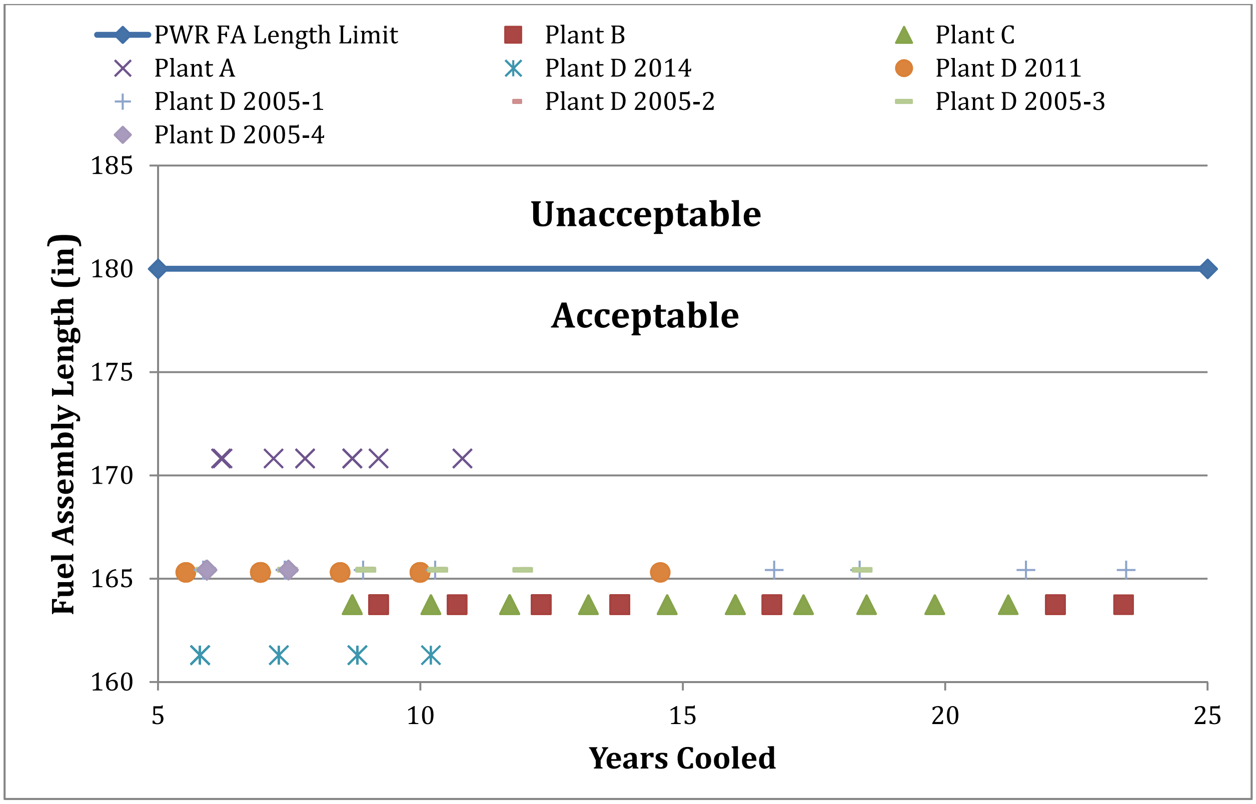


FIGURE A.1-14: BWR FA LENGTH VS YEARS COOLED (PLANT E)

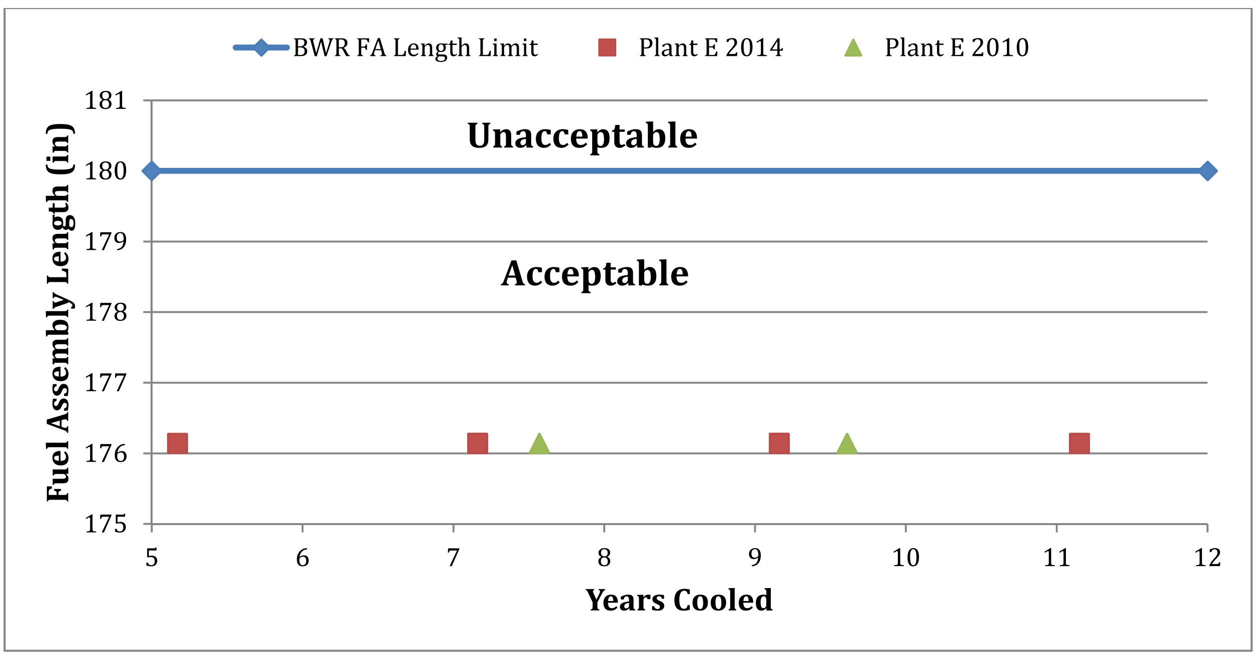


FIGURE A.1-15: PWR FA WEIGHT VS YEARS COOLED (PLANTS A – D)

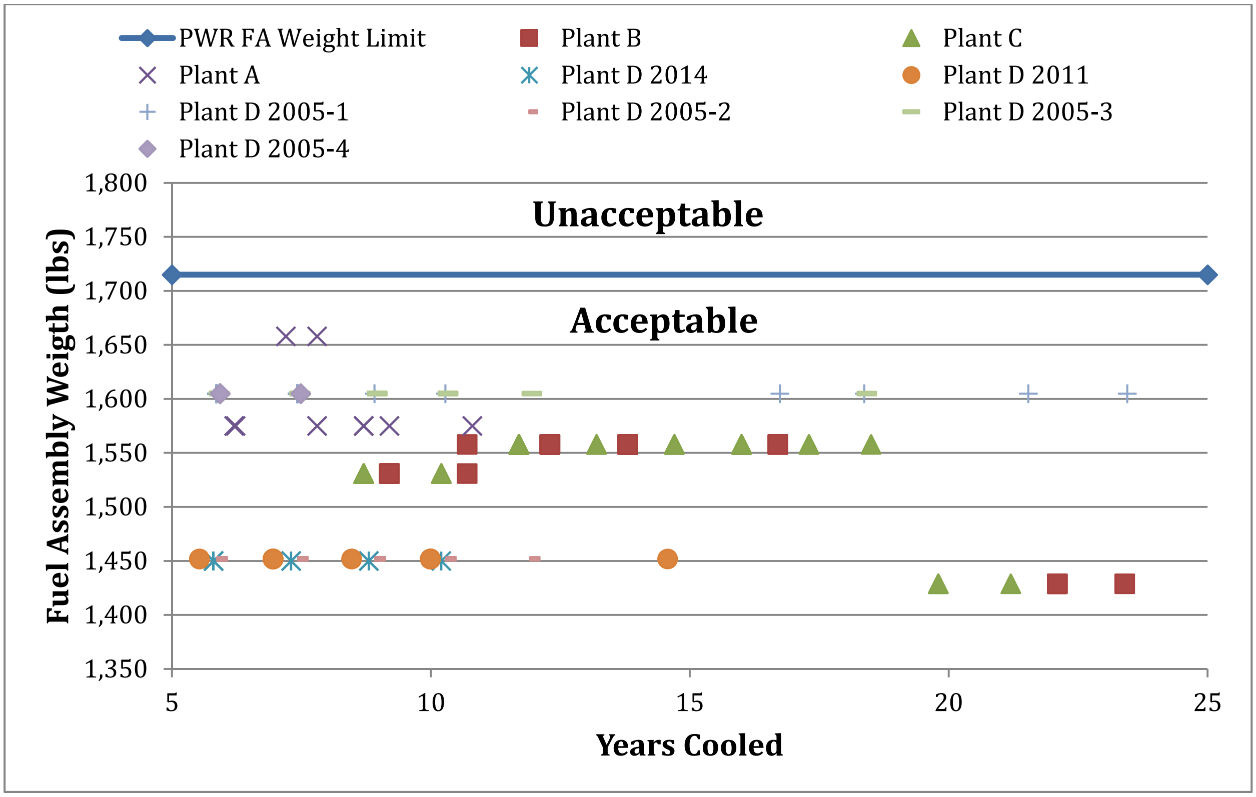
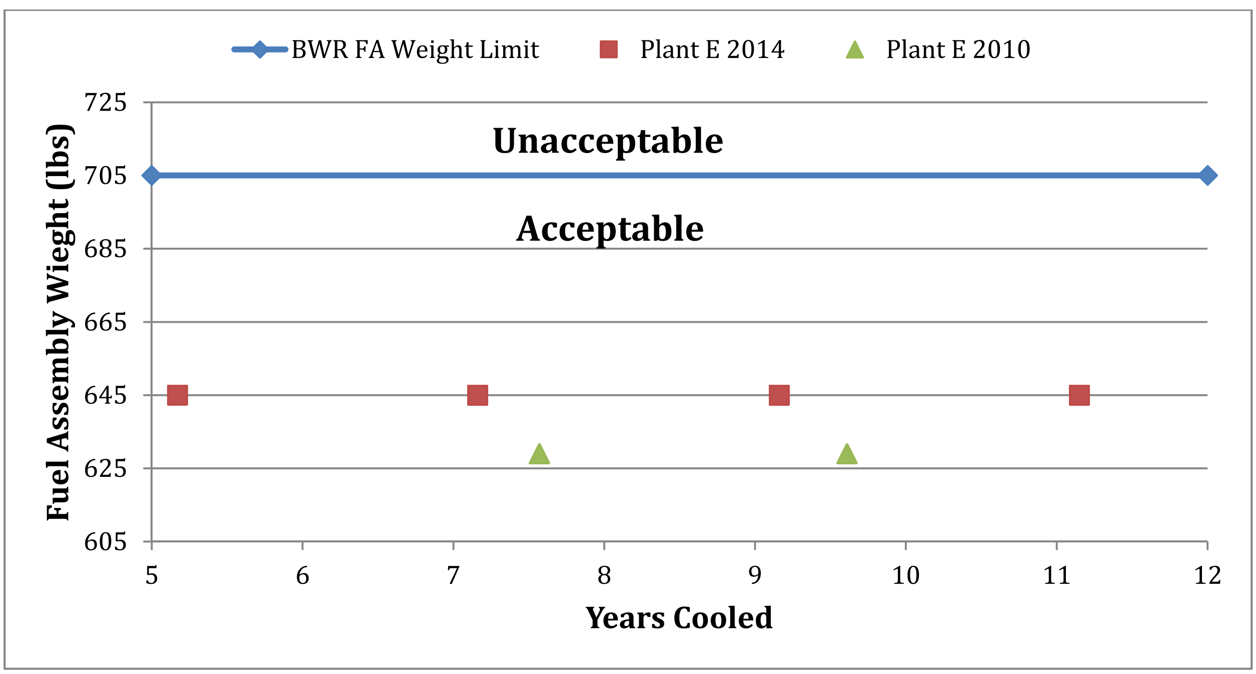


FIGURE A.1-16: BWR FA WEIGHT VS YEARS COOLED (PLANT E)



A.1.3 Efficiencies and Loadings (of 6625B-HB using Typical DSC Loading Data)

A.1.3.1 Methodology

The tables below (**Table A.1-5** through **Table A.1-11**) provide FA qualification (of the FAs used in the DSCs from several plants) for use with the 6625B-HB UNF transport casks. It is assumed that the FA data for the DSCs are representative samples of the actual FA distributions in the SFP for each plant, since this is largely driven by reactor core design. It is assumed that these FA distributions will be representative of future distributions during the life of the plant. The average percent that a 6625B-HB could be filled based on (a) the FA heat load zone configuration and (b) the fuel qualification table (see *Section 2.5* of report) is provided, as described in Equation 1 and shown with an example in Equation 2. In addition, an example of typical DSC FA loading data is provided in **Table A.1-4**.

Equation 1:

$$N_{\text{6625B-HB Full}} = N_{\text{FA,Z1/4}} \cdot R_{\text{FA,Z1/4}} + N_{\text{FA,Z2}} \cdot R_{\text{FA,Z2}} + N_{\text{FA,Z3}} \cdot R_{\text{FA,Z3}}$$

Where,

- $N_{\text{6625B-HB Full}}$ is the average percentage that a 6625B-HB could be filled with the FA's in a plant SFP (i.e., loading efficiency of 6625B-HB to transport actual UNF)
- $N_{\text{FA,Z1/4}}$ is the percentage that Zone 1 and Zone 4 are filled with qualified FA's in a plant SFP
- $N_{\text{FA,Z2}}$ is the percentage that Zone 2 is filled with qualified FA's in a plant SFP
- $N_{\text{FA,Z3}}$ is the percentage that Zone 3 is filled with qualified FA's in a plant SFP
- $R_{\text{FA,Z1/4}}$ is the ratio of the number of Zone 1 and Zone 4 FA compartments to the total number of FA compartments in the UNF transport cask
- $R_{\text{FA, Z2}}$ is the ratio of the number of Zone 2 FA compartments to the total number of FA compartments in the UNF transport cask
- $R_{\text{FA, Z3}}$ is the ratio of the number of Zone 3 FA compartments to the total number of FA compartments in the UNF transport cask

Example for Plant A:
Per the heat load zone configuration and the fuel qualification table (see *Section 2.5* of report):

- $N_{\text{FA,Z1/4}}$ is 58.3% = 0.583 (i.e., Zone 1/4 is only 58.3% full with available qualified SFP FA)
- $N_{\text{FA,Z2}}$ is 100.0% = 1.000 (i.e., Zone 2 is 100% full with available qualified SFP FA)
- $N_{\text{FA,Z3}}$ is 100.0% = 1.000 (i.e., Zone 3 is 100% full with available qualified SFP FA)
- $R_{\text{FA,Z1/4}}$ is 12/24 (since there are 12 compartments in Zone 1/4 and 24 total FA compartments)
- $R_{\text{FA,Z2}}$ is 8/24 (since there are 8 compartments in Zone 2 and 24 total FA compartments)
- $R_{\text{FA,Z3}}$ is 4/24 (since there are 4 compartments for Zone 3 FA and 24 total FA compartments)

Equation 2

$$N_{\text{6625B-HB Full}}(\text{Plant A}) = 0.583 \cdot \frac{12}{24} + 1.000 \cdot \frac{8}{24} + 1.000 \cdot \frac{4}{24} = 0.792 = 79.2\%$$

Alternate Heat Load Zoning Configurations:
An alternate heat load zone configuration and FQT is provided in *Section 2.5* of this report. This alternate design was considered in the efficiency estimation of the 6625B-HB to the plant DSC FA data to identify if a greater loading efficiency could be obtained in the "short loaded" cases. However, in general the "alternate" FQT did not perform as well as the standard FQT. The one exception is Plant D, where the "alternate" FQT proved advantageous (see **Table A.1-8** and **Table A.1-9**).

A.1.3.2 Typical Plant DSC Loading Data

Typical sample utility data for DSC loading data is provided in **Table A.1-4**.

TABLE A.1-4: EXAMPLE OF TYPICAL PWR DSC LOADING DATA

Load Year	DSC FA Location	Burnup (GWd/ MTU)	FA+BP Decay Heat (kW)	FA+ BPRA Length (in)	FA+ BPRA Weight (lb)	Years Cooled	Enrichment Wt (%)	Initial Uranium Loading (MTU)	FA Design
2014	1	41.368	0.988	170.826	1575	6.23	3.73	0.457	Mark B11
2014	2	40.666	0.983	170.826	1575	6.2	3.69	0.459	Mark B11
2014	3	46.11	0.929	170.826	1575	8.7	3.64	0.458	Mark B11
2014	4	38.74	0.717	170.826	1575	9.2	3.50	0.459	Mark B11
2014	5	39.951	0.768	170.826	1575	8.7	3.52	0.458	Mark B11
2014	6	37.316	0.887	170.826	1575	6.2	3.50	0.459	Mark B11
2014	7	44.743	1.104	170.826	1575	6.2	3.69	0.459	Mark B11
2014	8	41.053	0.869	170.826	1575	7.8	3.69	0.457	Mark B11
2014	9	35.691	0.654	170.826	1575	9.2	3.08	0.458	Mark B11
2014	10	36.481	0.604	170.826	1575	10.8	3.09	0.46	Mark B11
2014	11	43.275	0.991	170.826	1658	7.8	3.46	0.487	Mark B10L
2014	12	53.99	1.228	170.826	1575	7.8	3.77	0.459	Mark B11
2014	13	45.524	1.088	170.826	1658	7.2	4.02	0.487	Mark B10L
2014	14	42.126	0.963	170.826	1658	7.8	3.46	0.488	Mark B10L
2014	15	36.9	0.611	170.826	1575	10.8	3.09	0.458	Mark B11
2014	16	35.514	0.591	170.826	1575	10.8	3.09	0.459	Mark B11
2014	17	45.571	0.915	170.826	1575	8.7	3.64	0.458	Mark B11
2014	18	49.715	1.211	170.826	1658	7.2	4.02	0.487	Mark B10L
2014	19	45.373	0.902	170.826	1575	8.7	3.64	0.458	Mark B11
2014	20	39.521	0.739	170.826	1575	9.2	3.50	0.46	Mark B11
2014	21	46.139	0.927	170.826	1575	8.7	3.64	0.458	Mark B11
2014	22	46.342	0.929	170.826	1575	8.7	3.64	0.458	Mark B11
2014	23	50.474	1.037	170.826	1575	8.7	3.64	0.459	Mark B11
2014	24	50.051	1.119	170.826	1575	7.8	3.77	0.459	Mark B11

A.1.3.3 6625B-HB Loading Efficiency Study

The tables below (**Table A.1-5** through **Table A.1-11**) provide FA qualification (of the FAs used in the DSCs from several plants) for use with the 6625B-HB UNF transport casks. These efficiencies are based on the interpolated FQT, as discussed in Appendix A.1.2.

TABLE A.1-5: PLANT A 2014 PWR DSC FA DISTRIBUTION TO THE 6625B-HB-24

Heat Zone #	Heat Zone	# FA Allowed / Zone	% DSC FA Qualify	6625B-HB Zone % Full
1 or 4	0.0 < kW <= 0.9	12	29.2%	58.3%
2	0.9 < kW <= 1.4	8	95.8%	100.0%
3	1.4 < kW <= 2.1	4	95.8%	100.0%

Plant A 2014 DSC: 6625B-HB-24 Avg. % Full Per Zone Qualification of Actual Data = 79.2% (i.e., 19 FA)

TABLE A.1-6: PLANT B 2014 PWR DSC FA DISTRIBUTION TO THE 6625B-HB-24

Heat Zone #	Heat Zone	# FA Allowed / Zone	% DSC FA Qualify	6625B-HB Zone % Full
1 or 4	0.0 < kW <= 0.9	12	43.2%	86.5%
2	0.9 < kW <= 1.4	8	100.0%	100.0%
3	1.4 < kW <= 2.1	4	100.0%	100.0%

Plant B 2014 DSC: 6625B-HB-24 Avg. % Full Per Zone Qualification of Actual Data = 93.2% (i.e., 22 FA)

TABLE A.1-7: PLANT C 2014 PWR DSC FA DISTRIBUTION TO THE 6625B-HB-24

Heat Zone #	Heat Zone	# FA Allowed / Zone	% DSC FA Qualify	6625B-HB Zone % Full
1 or 4	0.0 < kW <= 0.9	12	56.8%	100.0%
2	0.9 < kW <= 1.4	8	97.3%	100.0%
3	1.4 < kW <= 2.1	4	100.0%	100.0%

Plant C 2014 DSC: 6625B-HB-24 Avg. % Full Per Zone Qualification of Actual Data = 100.0% (i.e., 24 FA)

TABLE A.1-8: PLANT D 2014 PWR DSC FA DISTRIBUTION TO THE 6625B-HB-24

Heat Zone #	Heat Zone	# FA Allowed / Zone	% DSC FA Qualify	6625B-HB Zone % Full
1 or 4	0.0 < kW <= 0.9	12	4.2%	8.3%
2	0.9 < kW <= 1.4	8	83.3%	100.0%
3	1.4 < kW <= 2.1	4	100.0%	100.0%

Plant D 2014 DSC: 6625B-HB-24 Avg. % Full Per Zone Qualification of Actual Data = 54.2% (i.e., 13 FA)

TABLE A.1-9: PLANT D 2014 PWR DSC FA DISTRIBUTION TO THE 6625B-HB-16 (ALTERNATE FQT)

Heat Zone #	Heat Zone	# FA Allowed / Zone	% DSC FA Qualify	6625B-HB Zone % Full
1 or 2	0.0 < kW <= 1.9	16	100.0%	100.0%

Plant D 2014 DSC: 6625B-HB-16 Avg. % Full Per Zone Qualification of Actual Data = 100.0% (i.e., 16 FA)

TABLE A.1-10: PLANT E 2014 BWR DSC FA DISTRIBUTION TO THE 6625B-HB-61

Heat Zone #	Heat Zone	# FA Allowed / Zone	% DSC FA Qualify	6625B-HB Zone % Full
1 or 4	0.0 < kW <= 0.33	21	4.2%	8.3%
2	0.45 < kW <= 0.78	16	83.3%	100.0%
3	0.33 < kW <= 0.45	24	100.0%	100.0%

Plant E 2014 DSC: 6625B-HB-61 Avg. % Full Per Zone Qualification of Actual Data = 77.0% (i.e., 47 FA)

TABLE A.1-11: PLANT E 2010 BWR DSC FA DISTRIBUTION TO THE 6625B-HB-61

Heat Zone #	Heat Zone	# FA Allowed / Zone	% DSC FA Qualify	6625B-HB Zone % Full
1 or 4	0.0 < kW <= 0.33	21	32.8%	95.2%
2	0.45 < kW <= 0.78	16	100.0%	100.0%
3	0.33 < kW <= 0.45	24	100.0%	100.0%

Plant E 2010 DSC: 6625B-HB-61 Avg. % Full Per Zone Qualification of Actual Data = 98.4% (i.e., 60 FA)

A.1.3.4 Plant Loading Study

The 6625B-HB average percent full efficiency and the determination of the number of loads required to empty an SFP with the designed 6625B-HB transportation cask using data from a utility's DSC and SPF capacities/loads is considered. The 6625B-HB number of loads is determined and shown in **Table A.1-12**. The entries of the table are as described below:

- Duke Energy Plants
  - *Plant*: a specific utility plant
  - *Unit*: the unit number of the utility plant
  - *Type*: the reactor type (PWR or BWR) of each unit
  - *Next Outage*: the estimated date of the most recent or next outage
  - *Renewed License Expiration*: the license expiration date for each unit
- Reactor Core Fuel Assembly Design Data (per NRC)
  - *#FA Per Core*: the number of fuel assemblies in the core (provided)
  - *%Core FA Replaced Each Fuel Cycle*: approximate percentage of new fuel assemblies loaded each cycle (provided)
  - *Approx. # FA Replaced / Fuel Cycle*: approximate number of new fuel assemblies loaded each cycle. This value is determined based on the product of the "*#FA Per Core*" and the "*%Core FA Replaced Each Fuel Cycle*".
  - *Plant Fuel Cycle (Months):* frequency of new fuel loadings (provided)
- SFP Data (as provided by Duke)
  - *SFP #*: SFP number at the utility plant
  - *FA Type*: the fuel assembly type (PWR or BWR) in each SFP
  - *FA Capacity*: fuel assembly capacity of each SFP
  - *FA Actual*: current approximate number of FA in each SFP
  - *% Full*: the total number of FA in the pool versus the capacity of the pool (provided)
- Plant Loading Operations Data (as provided by Duke)
  - *# DSCs*: the number of DSCs typically loaded during a SFP loading campaign

  - *Frequency (months)*: SFP DSC loading campaign frequency in months
- 6625B-HB Performance Assessment
  - *6625B-HB Avg. % Full Per Heat Load Zone Qualification of Actual DSC Data:* the average percentage that a 6625B-HB could be filled based on the FA heat load zone configuration (*Section 2.5* of report) and the fuel qualification table (*Section 2.5* of report) of actual DSC FA loading data. This value is determined using Equation 1 and an example is provided.
  - *#FA Locations / 6625B-HB:* the total number of FA slots in each basket
  - *Avg. #FA Loaded / 6625B-HB:* the average total number of FA slots filled in each basket, per the Equation 1, based on qualification of actual DSC FA loading data.
  - *Approx. # Plant Fuel Cycles Remaining:* estimated number of plant fuel cycles remaining between the next anticipated outage and the license expiration date for each plant.
  - *Approx. # FA Added to SPF Over Remaining Plant Life:* estimated number of FA that will be added to the SFP over the remaining plant life based on the quantity of FA added to the SFP at each outage and an entire reactor core offload at the end of plant life.
  - *# 6625B-HB Loads Required to Empty SFP Over Remaining Reactor Life:* the estimated total number of 6625B-HB loads required to the empty the SFP of each plant based on actual DSC FA loading data taking into account the heat load zone configuration and the fuel qualification table. This value is determined by dividing the sum of "*FA Actual*" and "*Approx. # FA Added to SPF Over Remaining Plant Life*" by the "Avg. *#FA Loaded / 6625B-HB*".

TABLE A.1-12: UTILITY DATA FOR SFP INVENTORY AND DSC LOADING DATA TO ESTIMATE TYPICAL UNF TRANSPORT CASK LOADING

Duke Energy Plants					Reactor Core FA Design Data (per NRC)				SFP Data (as provided by Duke)					Plant Loading Operations Data (as provided by Duke)		6625B-HB Performance Assessment					
Plant	Unit	Type	Next Outage	Renewed License Expiration	# FA Per Core	% Core FA Replaced Each Fuel Cycle	Approx. # FA Replaced / Fuel Cycle	Plant Fuel Cycle (Months)	SFP #	FA Type	FA Capacity	FA Actual	% Full	#DSCs	Frequency (Months)	6625B-HB Avg. % Full Per Heat Load Zone Qualification of Actual DSC Data	#FA Locations / 6625B-HB	Avg. #FA Loaded / 6625B-HB	Approx. # Plant Fuel Cycles Remaining	Approx. # FA Added to SPF Over Remaining Plant Life	# 6625B-HB Loads Required to Empty SFP Over Remaining Reactor Life
A	1	PWR	11/3/2014	2/6/2033	177	33%	59	24	1	PWR	1312	989	75.4%	5	36	79.2%	24	19	9	649	87
	2	PWR	10/17/2015	10/6/2033	177	33%	59	24	1	PWR	1312	989	75.4%	5	36	79.2%	24	19	9	649	87
	3	PWR	4/15/2014	7/19/2034	177	33%	59	24	2	PWR	825	478	57.9%	5	36	79.2%	24	19	10	708	63
B	1	PWR	11/21/2015	12/5/2043	193	33%	64	18	1	PWR	1421	1090	76.7%	4	36	93.2%	24	22	19	1345	111
	2	PWR	2/28/2015	12/5/2043	193	33%	64	18	2	PWR	1421	1138	80.1%	4	36	93.2%	24	22	19	1345	113
C	1	PWR	9/20/2014	6/12/2041	193	33%	64	18	1	PWR	1463	1247	85.2%	4	36	100.0%	24	24	18	1281	106
	2	PWR	9/12/2015	3/3/2043	193	33%	64	18	2	PWR	1463	1267	86.6%	4	36	100.0%	24	24	18	1281	107
D	2	PWR	5/9/2015	7/31/2030	157	33%	52	18	1	PWR	544	282	51.8%	4	36	66.7%	24	16	10	625	57
E	1	BWR	2/15/2016	9/8/2036	560	33%	187	24	1	BWR	1803	1178	65.3%	4	24	87.7%	61	53	10	2243	65
	N/A	N/A	N/A	N/A	0	0%	0	0	1	PWR	160	160	100.0%	4	24	100.0%	24	24	N/A	0	7
	2	BWR	2/14/2015	12/27/2034	560	40%	224	24	2	BWR	1839	987	53.7%	4	24	87.7%	61	53	10	2576	68
	N/A	N/A	N/A	N/A	0	0%	0	0	2	PWR	144	144	100.0%	4	24	100.0%	24	24	N/A	0	6
F	1	PWR	3/28/2015	10/24/2046	157	33%	52	18	1	PWR	2055	1778	86.5%	N/A	N/A	100.0%	24	24	21	1197	124
	N/A	N/A	N/A	N/A	0	0%	0	0	1	BWR	4507	4397	97.6%	N/A	N/A	100.0%	61	61	N/A	0	73

Notes:
[1] DSC for PWR is 24-slot basket and DSC for BWR is 61-slot basket. However, “Plant B” and “Plant C” DSC for PWR is a 37-slot basket.
[2] It is assumed that the PWR FAs in the SFP of Plant E satisfy the requirements to fill a 6625B-HB 100%
[3] It is assumed that the BWR and PWR FAs in the Plant F SFP satisfy the requirement to fill a 6625B-HB 100%
[4] Based on heat load zone configuration (see *Section 2.5* of report) and fuel qualification table (see *Section 2.5* of report) per Equation 1
[5] *Section 2.5* of the report provides the number of UNF Transport Cask compartments available for each heat load zone
[6] It is assumed that the Plant A Unit 3 SFP FA qualification distribution is the same as the Plant A Unit 1/2 SFP FA qualification distribution
[7] It is assumed that the Plant B Unit 2 SFP FA qualification distribution is the same as the Plant B Unit 1 SFP FA qualification distribution
[8] It is assumed that the Plant C Unit 2 SFP FA qualification distribution is the same as the Plant C Unit 1 SFP FA qualification distribution
[9] For “Plant D” the 6625B-HB uses the “alternate” design (16 FA) for short loading, since it could hold more FA than the standard 6625B-HB FQT design (see **Table A.1-8** and **Table A.1-9**)
[10] For “Plant E” the “6625B-HB Avg. % Full Per Heat Load Zone Qualification of Actual DSC Data” and “Avg. #FA Loaded / 6625B-HB” are an average of the values shown in **Table A.1-10** and **Table A.1-11**
[11] “Plant A” Units 1 and 2 share the same SFP (i.e., SFP #1)

A.1.3.5 Total 6625B-HB Loads (FAs Combined in a Single Location)

Table A.1-13 provides an approximate combined total number of PWR and BWR FAs that must be processed through a SPF based on the current inventory of FAs in each SPF and the estimated total number of FA to be added to each SFP from now until the end of current plant life (i.e., a total number of FA's, if they were all contained in one SFP). Additionally, this table provides that total number of 6625B-HB loads required to transport the combined total number of FA's, assuming that each 6625B-HB is completely full (i.e., 24 PWR FA or 61 BWR FA in each 6625B-HB load).

TABLE A.1-13: NUMBER OF 6625B-HB LOADS TO TRANSPORT ALL REMAINING DUKE ENERGY FA

FA Type	FA Total	6625B-HB Total (100% Full)	6625B-HB Total (Actual Avg. % Full per Table A.1-12
PWR	18642	777	868
BWR	11381	187	206

A.1.4 SFP Inventory Timeline Study

A timeline of each SFP FA inventory was constructed for each plant. Each SFP receives an input of FA's at the rate of the plant fuel cycle frequency and the quantity specified in **Table A.1-12** (e.g., Plant A SFP 1: 1/3 of the reactor core FA's, 59 FA, are moved to SFP 1 every 24 months). The FA's are removed (output) via either DSC's or transport casks (i.e., 6625B-HB) at the loading campaign frequency used by each plant (e.g., Plant A SFP 1: 5 DSC's every 36 months). Additionally, the transport casks (6625B-HB) are also loaded in campaigns with multiples of 5 casks per campaign (e.g., 5, 10, 15, 20… per loading campaign) to support efficiency of rail transport, which handles 5 casks each. In addition, loading campaigns at the current DSC frequency and quantity are analyzed. The transport casks (6625B-HB) are loaded with the average number of FA that qualify, based on the FA distribution in each SFP for the FA DSC data, as shown in **Table A.1-12** and discussed in **Equation 1** (e.g., a 6625B-HB PWR cask may have 24 FA slots, but on average only 19 FA qualify for loading in the cask). The timeline analysis plots (see **Figure A.1-17** through **Figure A.1-29**) show the SFP FA inventory for each plant SFP from the next scheduled (or most recently occurred) outage through the renewed license expiration date. In summary, the FA inventory is shown using three FA output scenarios:

1) Output with DSC at the current DSC frequency and quantity (e.g., 5 DSCs every 36 months, where each DSC holds 37 FA),
2) Output with the 6625B-HB at the current DSC frequency and quantity (e.g., 5 "6625B-HB" every 36 months, where each 6625B-HB can only hold 19 FA)
3) Output with the 6625B-HB at a specified frequency (that supports other plant SFP loading availability considering other plant operations) and quantity (e.g., 10 "6625B-HB" every 36 months, where each 6625B-HB can only hold 19 FA), so that SFP limits are maintained. The transport casks (6625B-HB) are loaded in campaigns with multiples of 5 casks per campaign (e.g., 5, 10, 15, 20… per loading campaign) to support efficiency of rail transport, which handles 5 casks each.

For each of the Duke Energy plants considered, based on the SFP FA data, loading timeline estimates are provided as follows:

- **Figure A.1-17**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant A, SFP 1)
  - Plant A, SFP 1: At the current DSC frequency (5 every 36 months), the 6625B-HB (19 FA / 6625B-HB) would result in the SFP FA capacity being exceeded. However, increasing the frequency to 10 "6625B-HB" casks every 36 months would not only stay below the SFP capacity limit, but would also reduce the current inventory.
- **Figure A.1-18**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant A, SFP 2)
  - Plant A, SFP 2: At the current DSC frequency (5 every 36 months), the 6625B-HB (19 FA / 6625B-HB) would not only stay below the SFP capacity limit, but would also reduce the current inventory.
- **Figure A.1-19**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant B, SFP 1)
  - Plant B, SFP 1: At the current DSC frequency (4 every 36 months), the 6625B-HB (22 FA / 6625B-HB) would result in the SFP FA capacity being reached at the end of the reactor license. However, increasing the frequency to 10 "6625B-HB" casks every 36 months would not only stay below the SFP capacity limit, but would also reduce the current inventory.
- **Figure A.1-20:** PWR FA Inventory – Loading via DSC & 6625B-HB (Plant B, SFP 2)
  - Plant B, SFP 2: At the current DSC frequency (4 every 36 months), the 6625B-HB (22 FA / 6625B-HB) would result in the SFP FA capacity being reached near the end of the reactor license. However, increasing the frequency to 10 "6625B-HB" casks every 36 months would not only stay below the SFP capacity limit, but would also reduce the current inventory.
- **Figure A.1-21**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant C, SFP 1)
  - Plant C, SFP 1: At the current DSC frequency (4 every 36 months), the 6625B-HB (24 FA / 6625B-HB) would result in the SFP FA capacity being reached near the end of the reactor license. However, increasing the frequency to 5 "6625B-HB" casks every 36 months would keep the FA inventory below the SFP capacity limit.
- **Figure A.1-22**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant C, SFP 2)
  - Plant C, SFP 2: At the current DSC frequency (4 every 36 months), the 6625B-HB (24 FA / 6625B-HB) would result in the SFP FA capacity being reached near the end of the reactor license. However, increasing the frequency to 5 "6625B-HB" casks every 36 months would keep the FA inventory below the SFP capacity limit.
- **Figure A.1-23**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant D, SFP 1)
  - Plant D, SFP 1: At the current DSC frequency (4 every 36 months), the 6625B-HB (16 FA / 6625B-HB) would result in the SFP FA capacity being reached at the end of the reactor license. However, increasing the frequency to 10 "6625B-HB" casks every 36 months would not only stay below the SFP capacity limit, but would also reduce the current inventory. For "Plant D" the 6625B-HB uses the "alternate" design (16 FA) for short loading, since it could hold more FA than the standard 6625B-HB FQT design (see **Table A.1-8**, **Table A.1-9**, and **Table A.1-12**)
- **Figure A.1-24**: BWR FA Inventory – Loading via DSC & 6625B-HB (Plant E, SFP 1)
  - Plant E, SFP 1: At the current DSC frequency (4 every 24 months), the 6625B-HB (53 FA / 6625B-HB) would not only stay below the SFP capacity limit, but would also reduce the current inventory. Therefore, increasing to 5 "6625B-HB" casks every 24 months would further reduce inventory.

- **Figure A.1-25**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant E, SFP 1)
  - Plant E, SFP 1: This reactor at this plant does not use PWR FA. However, there are some stored in its SFP. Therefore, the 6625B-HB (24 FA / 6625B-HB) would result in the SFP FA inventory being reduced at any loading frequency, since it is not anticipated that any additional PWR FA will be added to this SFP.
- **Figure A.1-26**: BWR FA Inventory – Loading via DSC & 6625B-HB (Plant E, SFP 2)
  - Plant E, SFP 2: At the current DSC frequency (4 every 24 months), the 6625B-HB (53 FA / 6625B-HB) would stay below the SFP capacity limit. Increasing to 5 "6625B-HB" casks every 24 months would further reduce inventory.
- **Figure A.1-27**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant E, SFP 2)
  - Plant E, SFP 2: This reactor at this plant does not use PWR FA. However, there are some stored in its SFP. Therefore, the 6625B-HB (24 FA / 6625B-HB) would result in the SPF FA inventory being reduced at any loading frequency, since it is not anticipated that any additional PWR FA will be added to this SFP.
- **Figure A.1-28**: PWR FA Inventory – Loading via DSC & 6625B-HB (Plant F, SFP 1)
  - Plant F, SFP 1: This plant is not currently moving any FA's to dry storage (i.e., no DSC loading is currently occurring). Therefore, without any action, the SFP capacity will be exceeded. At a frequency of 5 "6625B-HB" casks every 36 months (with 24 FA / 6625B-HB), the FA inventory would not only stay below the SFP capacity limit, but would also reduce the current FA inventory. Increasing to 10 "6625B-HB" casks every 36 months would further reduce inventory.
- **Figure A.1-29**: BWR FA Inventory – Loading via DSC & 6625B-HB (Plant F, SFP 1)
  - Plant E, SFP 2: This reactor at this plant does not use BWR FA. However, there are some stored in its SFP. Therefore, the 6625B-HB (61 FA / 6625B-HB) would result in the SFP FA inventory being reduced at any loading frequency, since it is not anticipated that any additional BWR FA will be added to this SFP.

The SFP FA and DSC/6625B-HB input and output quantities and the plant fuel cycles and DSC/6625B-HB frequencies for each Duke Energy plant are tabulated as follows. The tables listed below correspond to the figures listed above for each plant SFP (e.g., **Table A.1-14** is associated with **Figure A.1-17**). Therefore, the comments listed with **Figure A.1-17** through **Figure A.1-29** also pertain to the associated tables listed below for each plant SFP.

- **Table A.1-14**: PWR FA Input / Output Data (Plant A, SFP1)
- **Table A.1-15**: PWR FA Input / Output Data (Plant A, SFP 2)
- **Table A.1-16**: PWR FA Input / Output Data (Plant B, SFP 1)
- **Table A.1-17**: PWR FA Input / Output Data (Plant B, SFP 2)
- **Table A.1-18**: PWR FA Input / Output Data (Plant C, SFP 1)
- **Table A.1-19**: PWR FA Input / Output Data (Plant C, SFP 2)
- **Table A.1-20**: PWR FA Input / Output Data (Plant D, SFP 1)
- **Table A.1-21**: BWR FA Input / Output Data (Plant E, SFP 1)
- **Table A.1-22**: PWR FA Input / Output Data (Plant E, SFP 1)
- **Table A.1-23**: BWR FA Input / Output Data (Plant E, SFP 2)
- **Table A.1-24**: PWR FA Input / Output Data (Plant E, SFP 2)
- **Table A.1-25**: PWR FA Input / Output Data (Plant F, SFP 1)
- **Table A.1-26**: BWR FA Input / Output Data (Plant F, SFP 1)

FIGURE A.1-17: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT A, SFP 1)

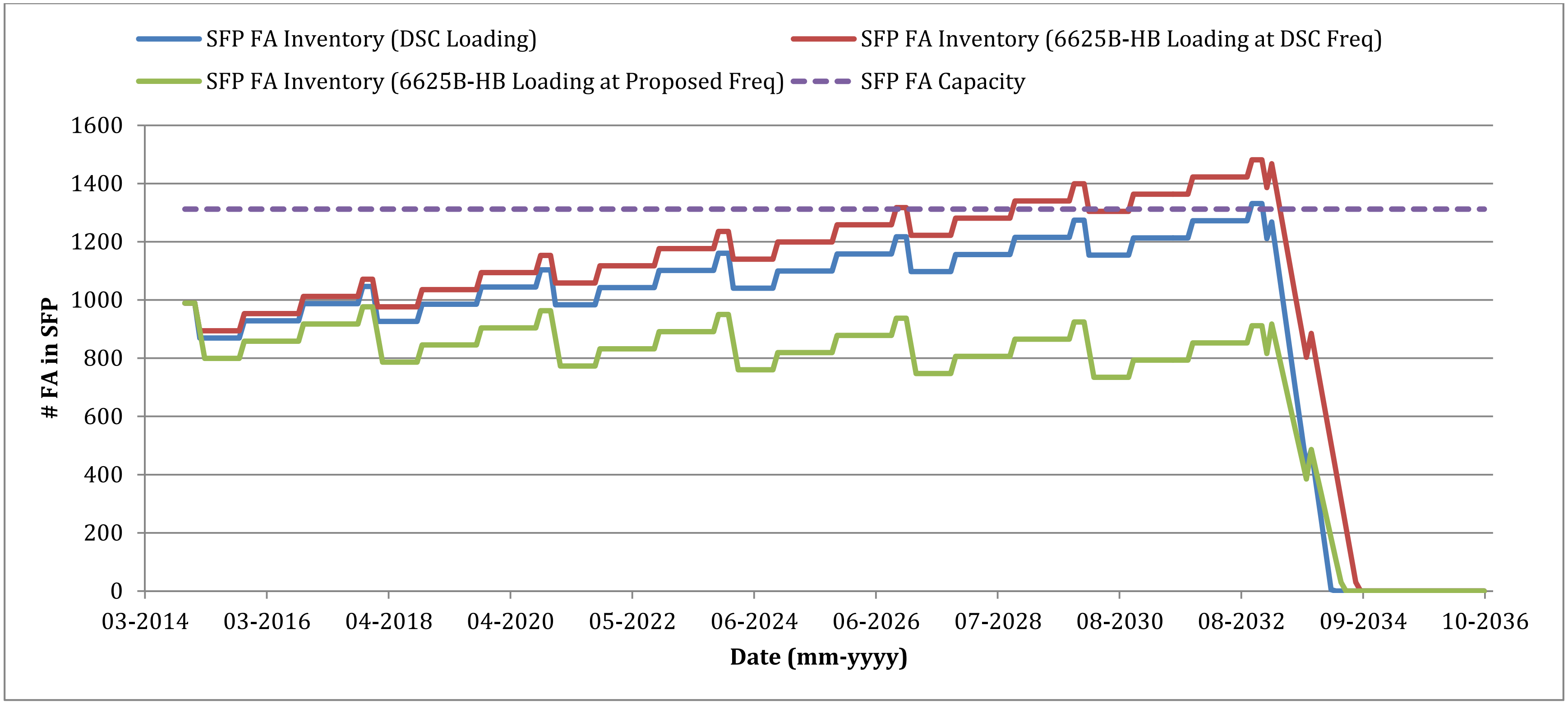


TABLE A.1-14: PWR FA INPUT / OUTPUT DATA (PLANT A, SFP 1)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In (Unit 1)	N/A	59	24
In (Unit 2)	N/A	59	24
Out (DSC)	5	120	36
Out (6625B-HB at DSC Freq)	5	95	36
Out (6625B-HB at Proposed Freq)	10	190	36

FIGURE A.1-18: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT A, SFP 2)

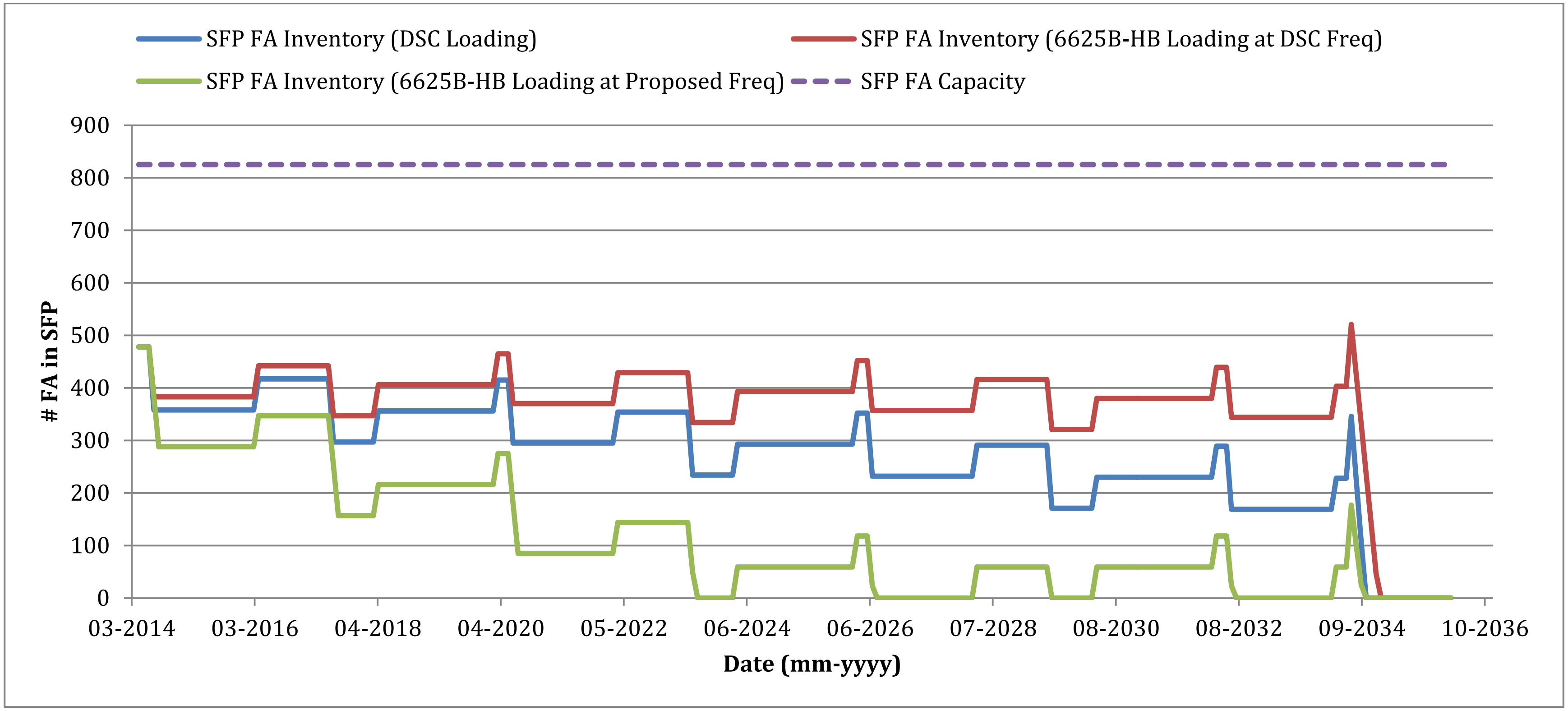


TABLE A.1-15: PWR FA INPUT / OUTPUT DATA (PLANT A, SFP 2)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	59	24
Out (DSC)	5	120	36
Out (6625B-HB at DSC Freq)	5	95	36
Out (6625B-HB at Proposed Freq)	10	190	36

FIGURE A.1-19: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT B, SFP 1)

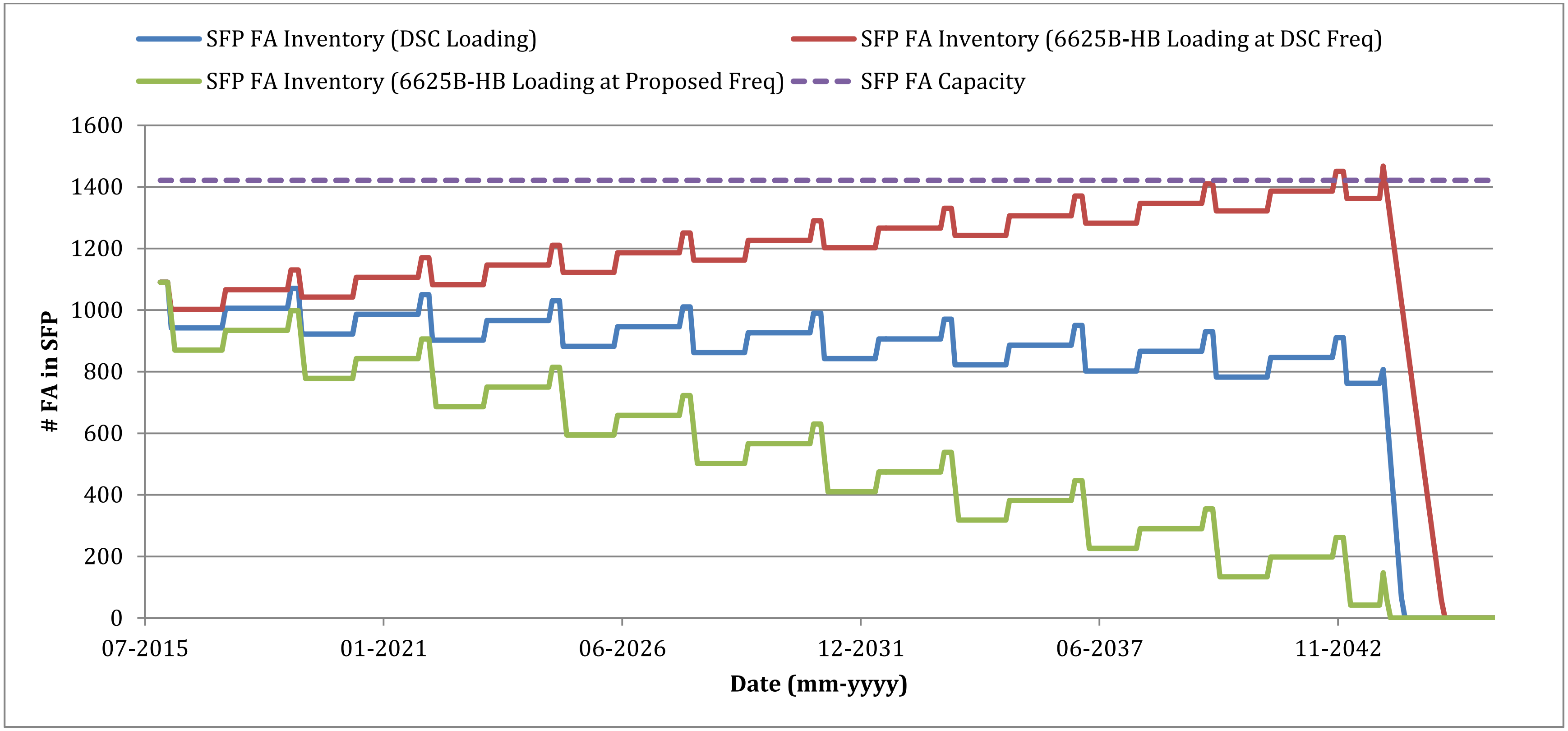


TABLE A.1-16: PWR FA INPUT / OUTPUT DATA (PLANT B, SFP 1)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	64	18
Out (DSC)	4	148	36
Out (6625B-HB at DSC Freq)	4	88	36
Out (6625B-HB at Proposed Freq)	10	220	36

FIGURE A.1-20: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT B, SFP 2)

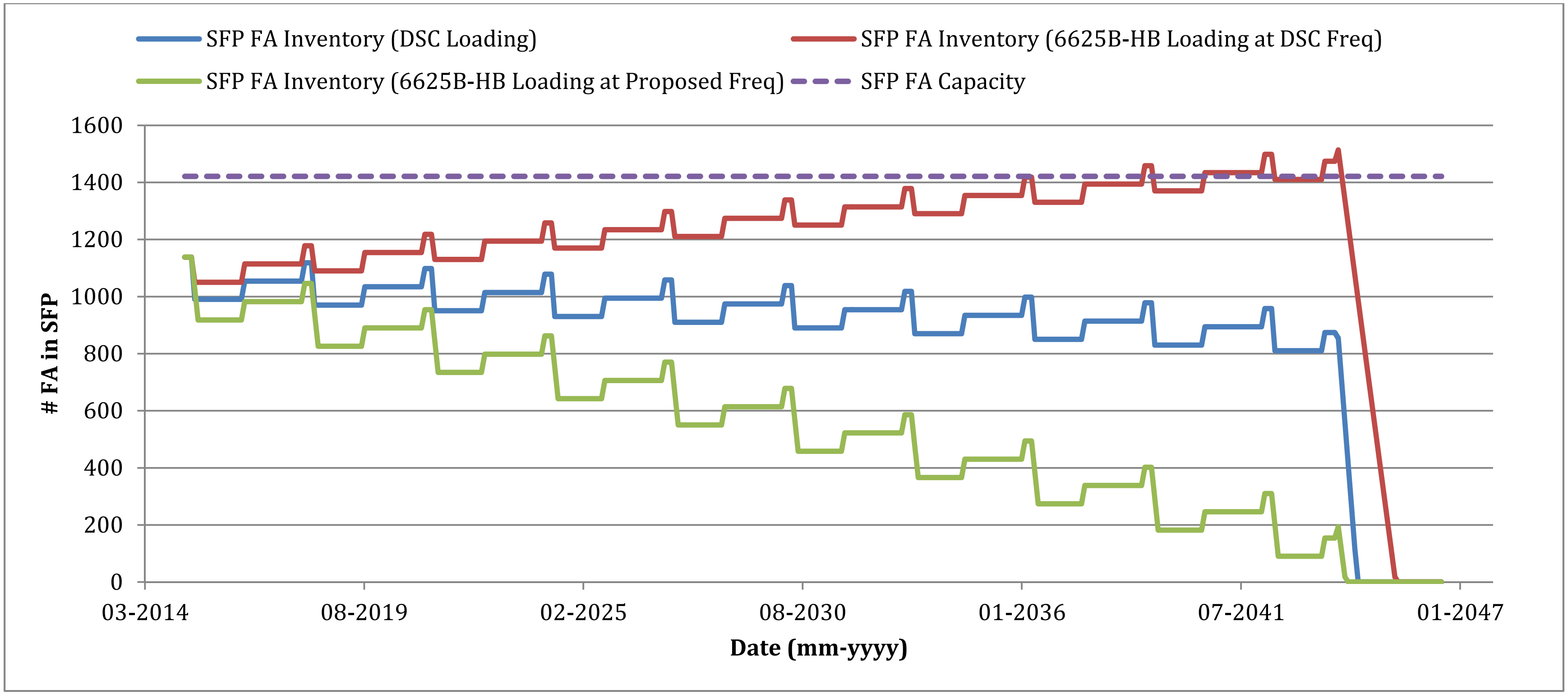


TABLE A.1-17: PWR FA INPUT / OUTPUT DATA (PLANT B, SFP 2)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	64	18
Out (DSC)	4	148	36
Out (6625B-HB at DSC Freq)	4	88	36
Out (6625B-HB at Proposed Freq)	10	220	36

FIGURE A.1-21: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT C, SFP 1)

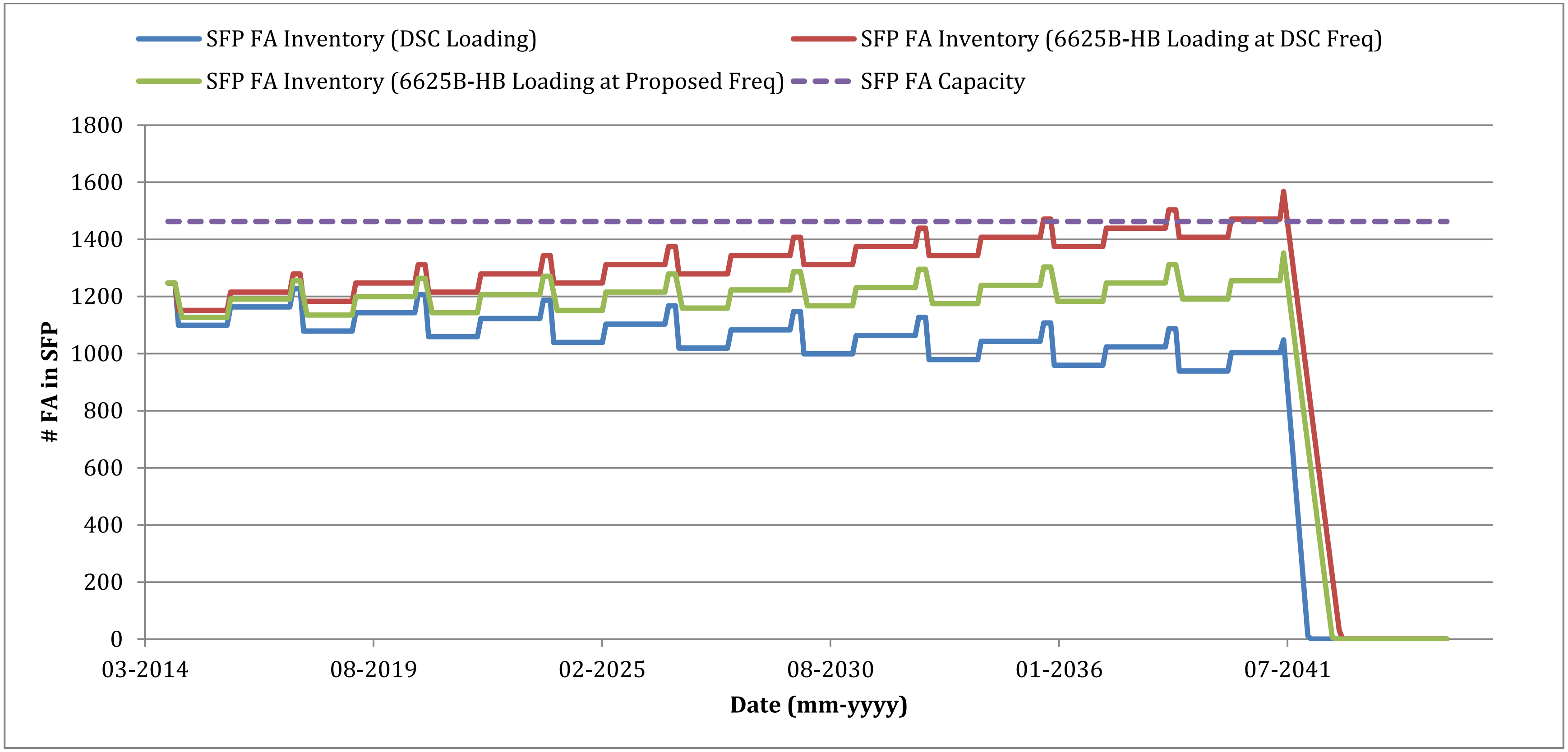


TABLE A.1-18: PWR FA INPUT / OUTPUT DATA (PLANT C, SFP 1)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	64	18
Out (DSC)	4	148	36
Out (6625B-HB at DSC Freq)	4	96	36
Out (6625B-HB at Proposed Freq)	5	120	36

FIGURE A.1-22: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT C, SFP 2)

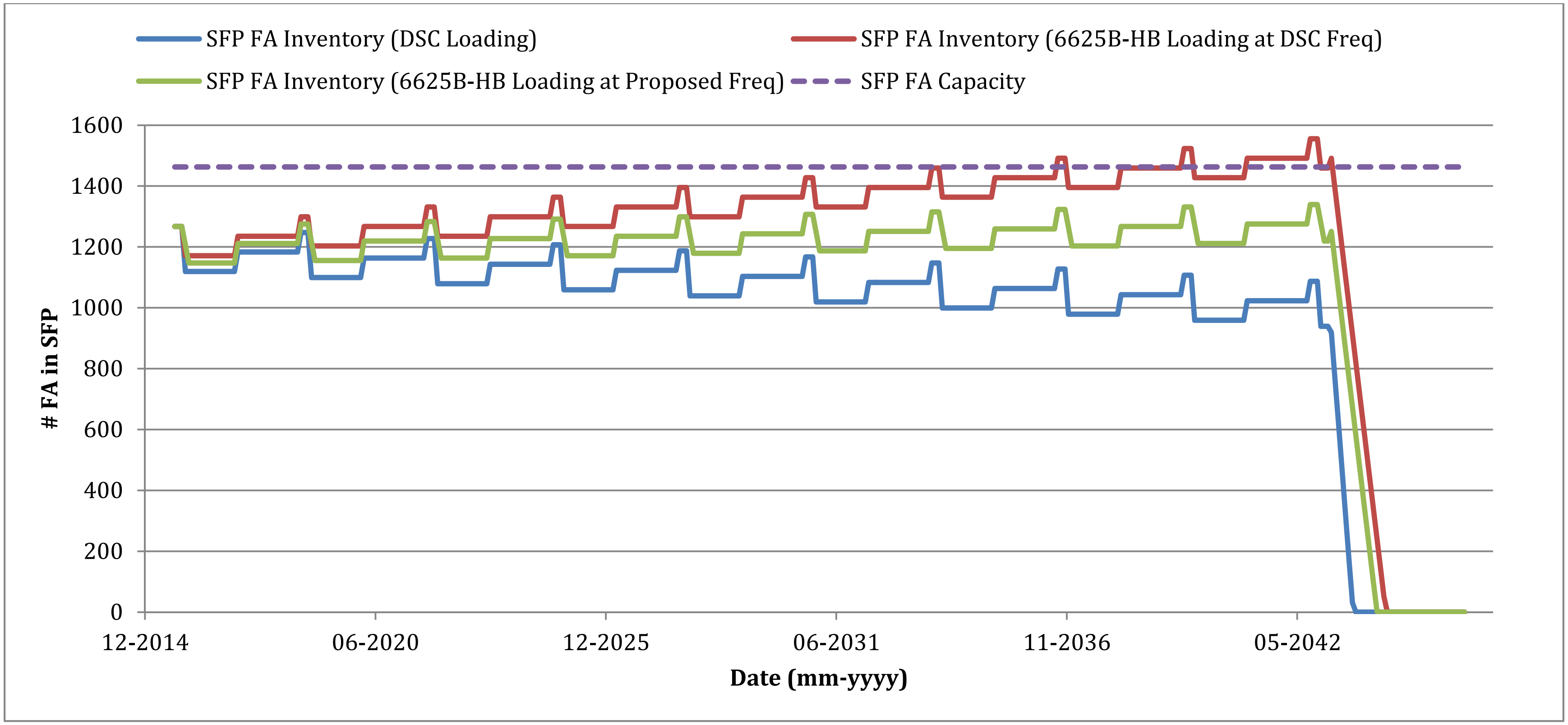


TABLE A.1-19: PWR FA INPUT / OUTPUT DATA (PLANT C, SFP 2)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	64	18
Out (DSC)	4	148	36
Out (6625B-HB at DSC Freq)	4	96	36
Out (6625B-HB at Proposed Freq)	5	120	36

FIGURE A.1-23: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT D, SFP 1)

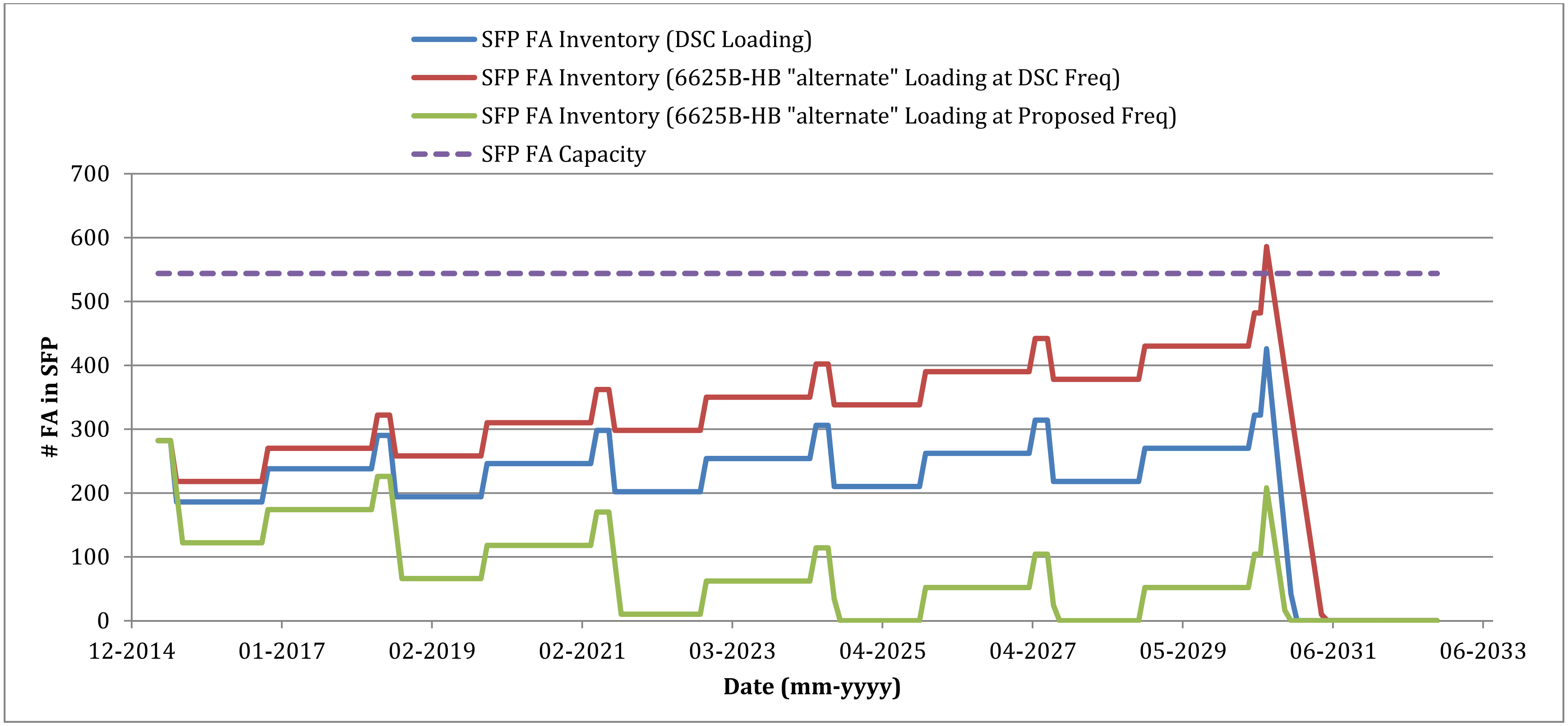


TABLE A.1-20: PWR FA INPUT / OUTPUT DATA (PLANT D, SFP 1)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	52	18
Out (DSC)	4	96	36
Out (6625B-HB at DSC Freq)	4	64	36
Out (6625B-HB at Proposed Freq)	10	160	36

FIGURE A.1-24: BWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT E, SFP 1)

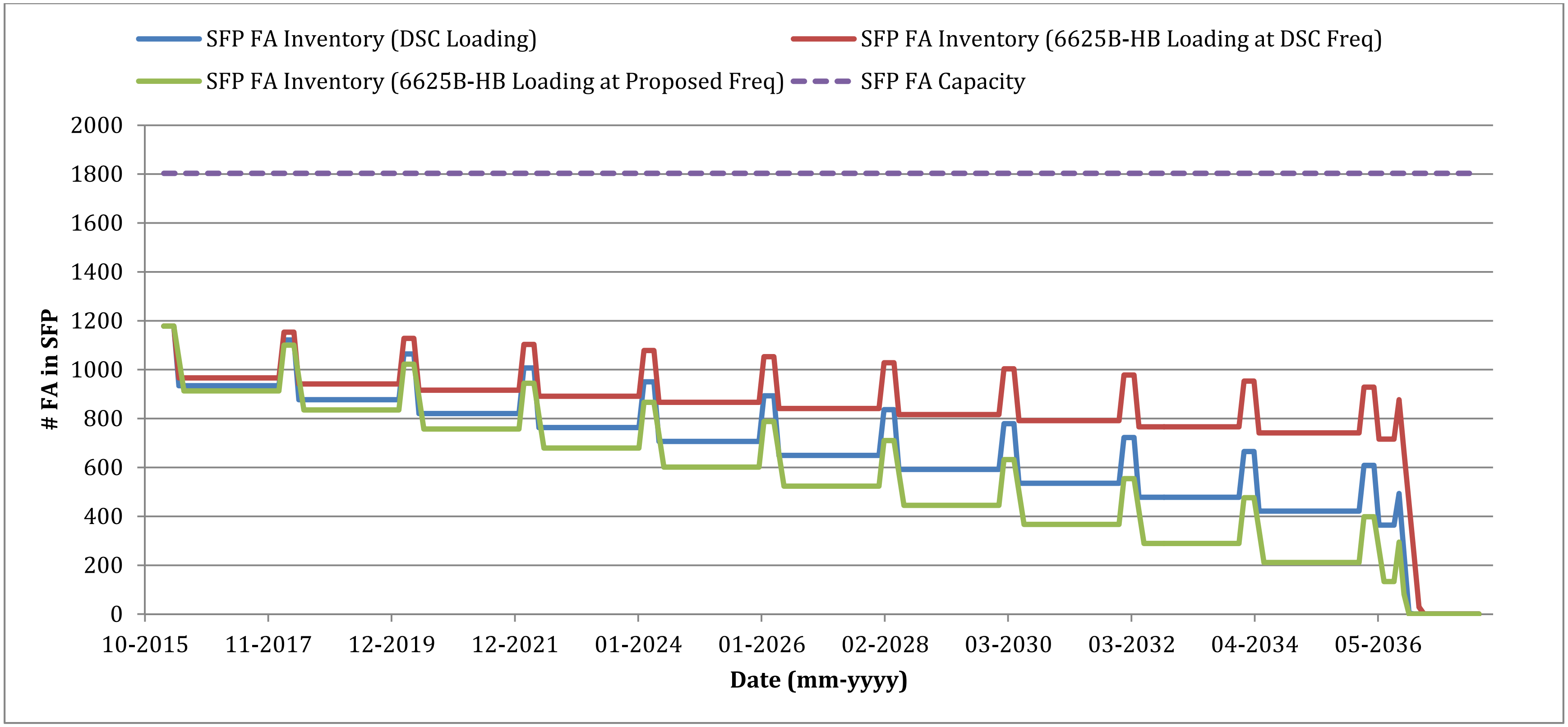


TABLE A.1-21: BWR FA INPUT / OUTPUT DATA (PLANT E, SFP 1)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	187	24
Out (DSC)	4	244	24
Out (6625B-HB at DSC Freq)	4	212	24
Out (6625B-HB at Proposed Freq)	5	265	24

FIGURE A.1-25: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT E, SFP 1)

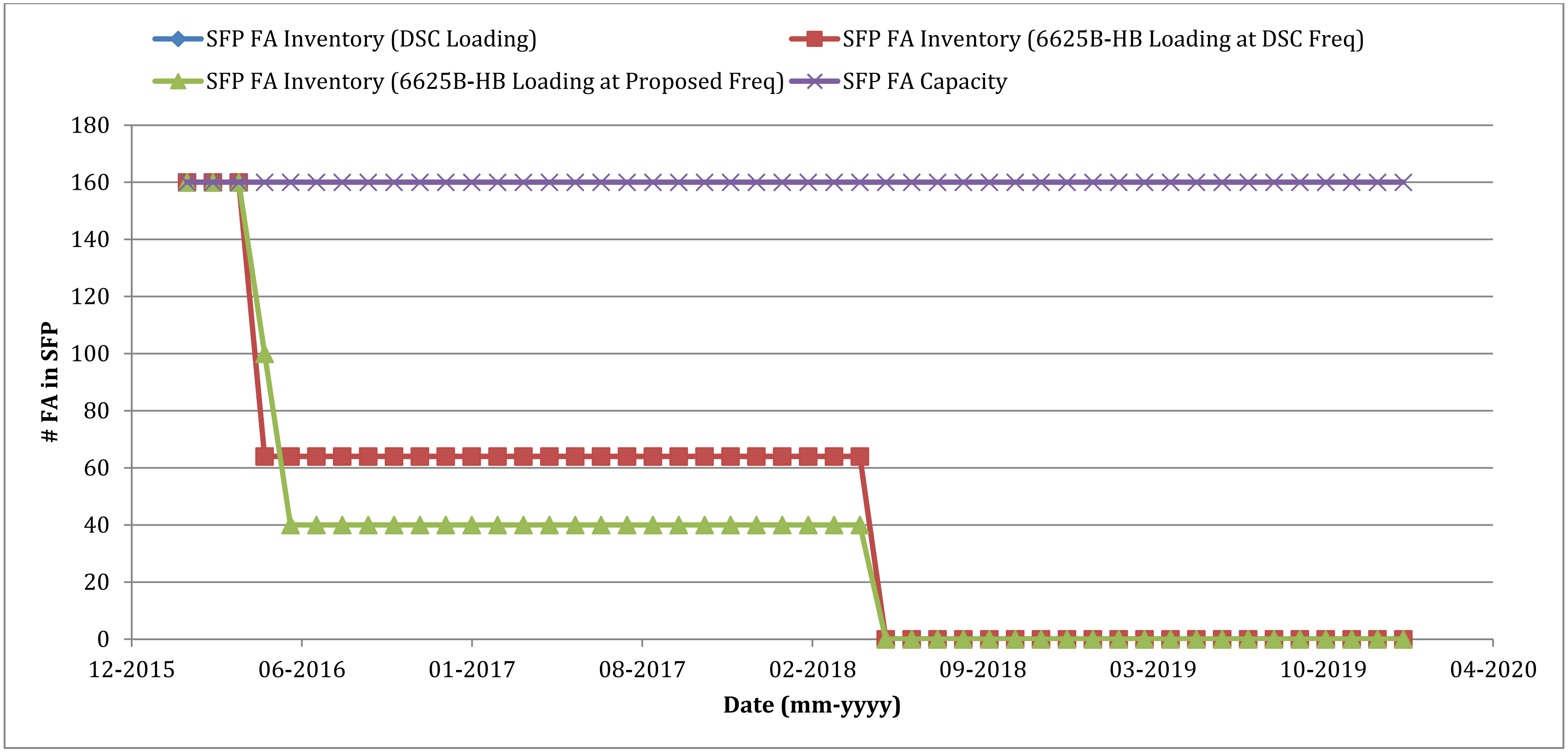


TABLE A.1-22: PWR FA INPUT / OUTPUT DATA (PLANT E, SFP 1)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A		
Out (DSC)	4	96	24
Out (6625B-HB at DSC Freq)	4	96	24
Out (6625B-HB at Proposed Freq)	5	120	24

FIGURE A.1-26: BWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT E, SFP 2)

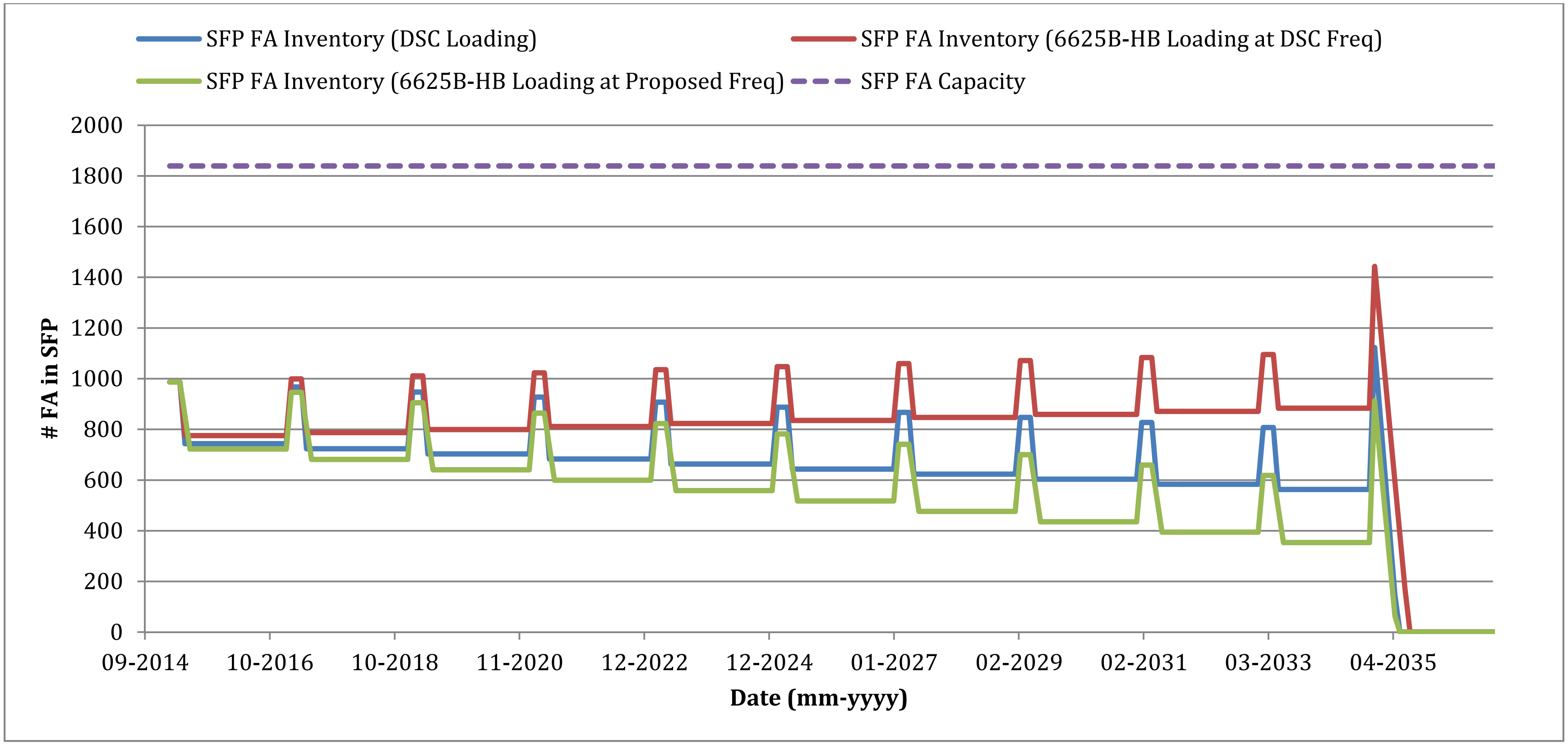


TABLE A.1-23: BWR FA INPUT / OUTPUT DATA (PLANT E, SFP 2)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	224	24
Out (DSC)	4	244	24
Out (6625B-HB at DSC Freq)	4	212	24
Out (6625B-HB at Proposed Freq)	5	265	24

FIGURE A.1-27: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT E, SFP 2)

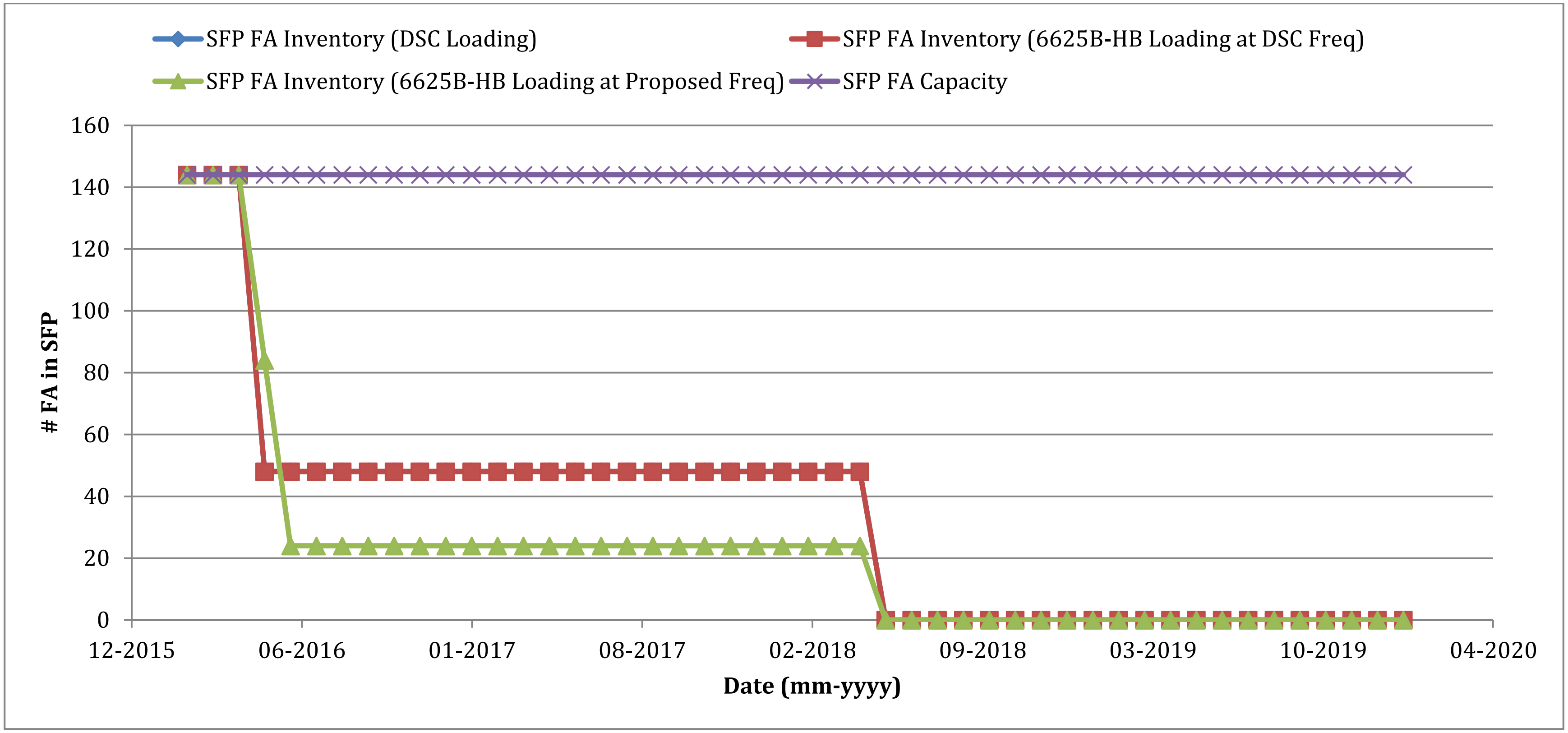


TABLE A.1-24: PWR FA INPUT / OUTPUT DATA (PLANT E, SFP 2)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A		
Out (DSC)	4	96	24
Out (6625B-HB at DSC Freq)	4	96	24
Out (6625B-HB at Proposed Freq)	5	120	24

FIGURE A.1-28: PWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT F, SFP 1)

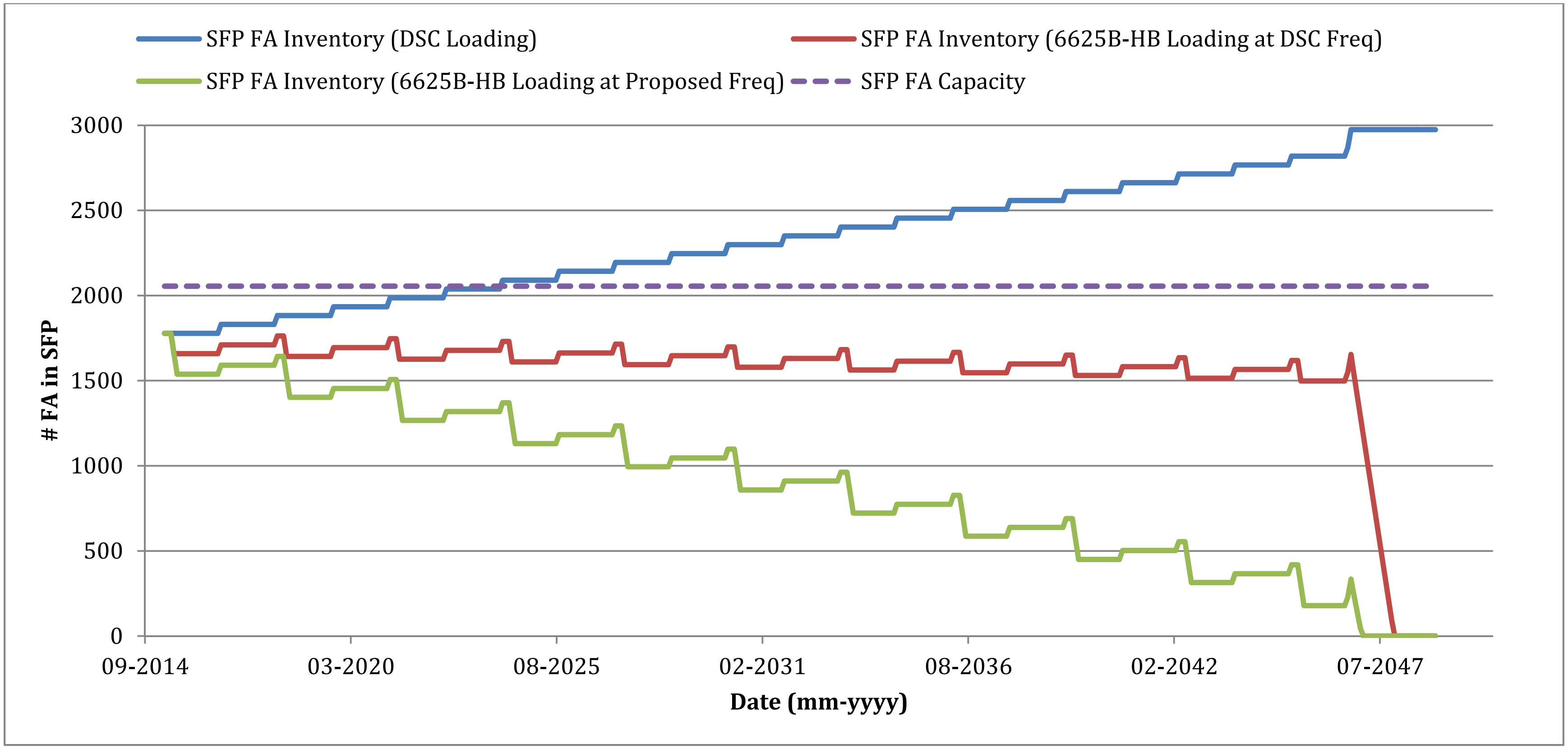


TABLE A.1-25: PWR FA INPUT / OUTPUT DATA (PLANT F, SFP 1)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A	52	18
Out (DSC)	5		36
Out (6625B-HB at DSC Freq)	5	120	36
Out (6625B-HB at Proposed Freq)	10	240	36

FIGURE A.1-29: BWR FA INVENTORY – LOADING VIA DSC & 6625B-HB (PLANT F, SFP 1)

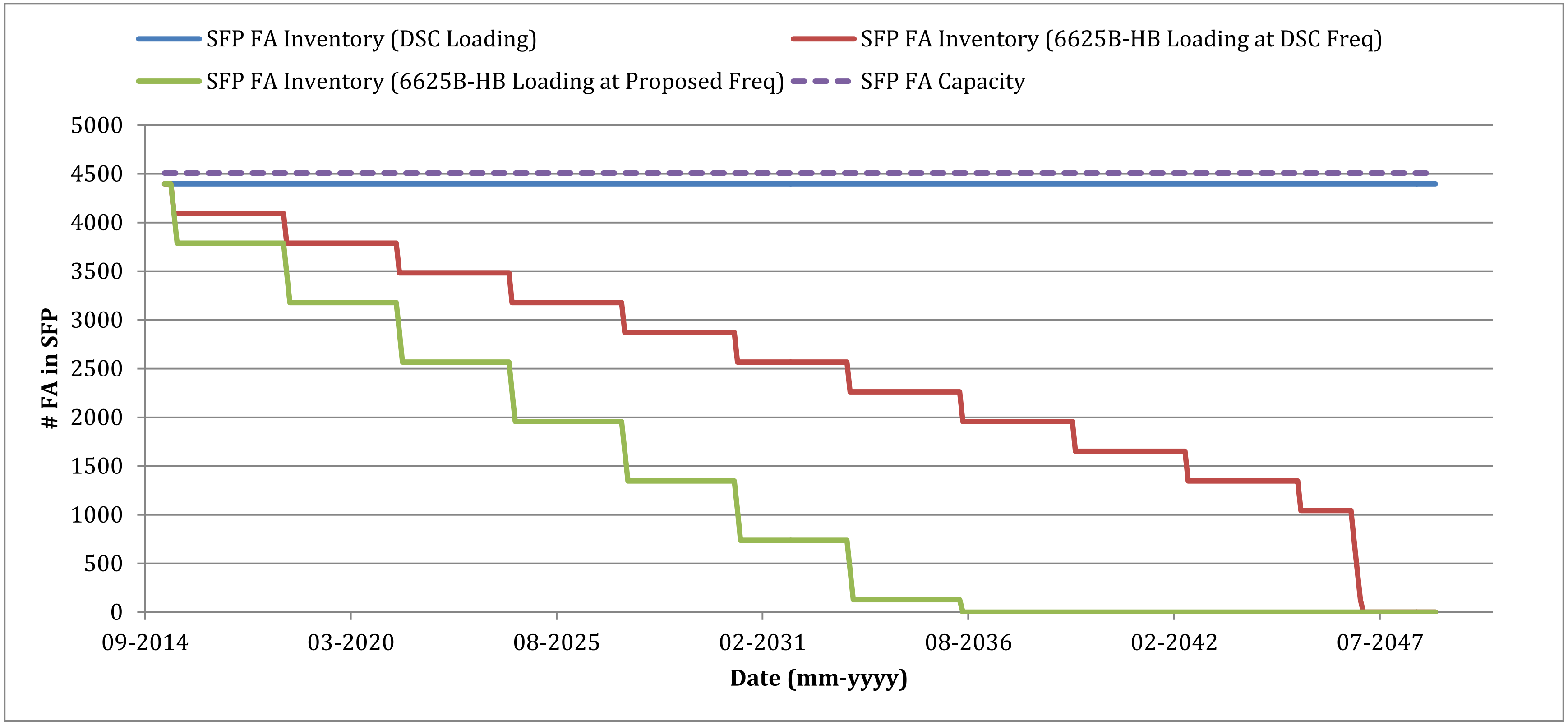


TABLE A.1-26: BWR FA INPUT / OUTPUT DATA (PLANT F, SFP 1)

SFP FA Activity	#DSC or 6625B-HB / Campaign	#FA / DSC or 6625B-HB Campaign	Cycle (months)
In	N/A		
Out (DSC)	5		36
Out (6625B-HB at DSC Freq)	5	305	36
Out (6625B-HB at Proposed Freq)	10	610	36

A.2 Weight Capacity Study for Lifting 6625B-HB

A weight analysis is provided in **Table A.2-1** that estimates the maximum anticipated hook weight based on the actual plant data and loading performance of the 6625B-HB per **Table A.1-12** and compares it to:

- The 6625B-HB design weight requirement (<125T),
- The administrative fuel loading crane capacity for each Duke Energy plant included in this study**,**
- The floor capacity for each Duke Energy plant included in this study.

The weight comparisons in **Table A.2-1** are based on the "Administrative Fuel Loading Crane Capacity" values that are dependent on site specific operating parameters/requirements. The floor capacity values are estimated typical values. Therefore, further research and investigation (outside the scope of this work) is required to analyze the specific floor areas utilized by the 6625B-HB cask.

Data regarding the lifting yoke designs and associated weights used at the Duke Energy plants is provided in **Table A.2-2**.

TABLE A.2-1: WEIGHT ANALYSIS (ESTIMATED MAXIMUM HOOK WEIGHT VS. PLANT LOADING CRANE AND FLOOR CAPACITY)

Plant	Admin Fuel Loading Crane Capacity (T)	Crane Rating (T)	Loading Floor Capacity (T)	Avg. FA Weight (T)	# FA Loaded / 6625B-HB (per FQT)	Total Avg. FA Weight (T)	Empty 6625B-HB Weight (T), see Section 2.1	Maximum Hook Weight (T)	Is The Max. Hook Weight < 125T?	Is The Max. Hook Weight < Plant Loading Floor Capacity?	Max. # FA Permitted, which Satisfy Plant Loading Floor Req.	Max. # FA Permitted, which Satisfy Plant Admin. FA Loading Crane Capacity Req.
Plant A	100	100.0	100	0.795	19	15.1	103.0	118.1	Yes	No	Exceeds Limit @ 0 FA	Exceeds Limit @ 0 FA
Plant B	125	125.0	125	0.770	22	16.9	103.0	120.0	Yes	Yes	24	24
Plant C	125	125.0	125	0.771	24	18.5	103.0	121.5	Yes	Yes	24	24
Plant D	110	125.0	110	0.764	16	12.2	103.0	115.2	Yes	No	9	9
Plant E	106	125.0	75	0.319	53	16.9	98.8	115.6	Yes	No	Exceeds Limit @ 0 FA	22
Plant F	97.5	150.0	97.5	0.795	24	19.1	103.0	122.1	Yes	No	Exceeds Limit @ 0 FA	Exceeds Limit @ 0 FA
Plant G	72	120.0	25	No Data	No Data	No Data	No Data	No Data	No Data	No Data	No Data	No Data

*A future new crane replacement for Plant G will be 130T capacity.

For the 6625B-HB design an adjustable hook yoke similar to that used for TN32 was assumed. Not all Duke Energy plants are able to use a J-type hook arrangement. Even though they use the TN system, Plant D and Plant E use an adjustable yoke where the arms flair out (e.g., hydraulic), because of space restrictions. Plant A uses a Sister-Hook. Plant B and Plant C use an NAC designed system and their yoke is similar in operation to the Plant D and Plant E yoke design. However, it is anticipated that Plant B and Plant C could use a J-type hook arrangement if needed.

TABLE A.2-2: LIFTING YOKE DESIGNS AND WEIGHTS FOR PLANTS A-F

Duke Energy Plant	Lifting Yoke Design Type	Lifting Yoke Weight (lbs)
6625B-HB (assumed yoke design and weight TN-32)	J-Hook	7500
Plant A	Sister-Hook	6550 (new light lift design 2100)
Plant B	NAC Design Similar to Plants D/E (J-Hook Assumed Possible)	5500
Plant C	NAC Design Similar to Plants D/E (J-Hook Assumed Possible)	5500
Plant D	Hydraulic Arm	9700
Plant E	Hydraulic Arm	9700
Plant F	J-Hook Assumed Possible	No Data Available

A.3 Estimation of Utility Damaged Fuel Assemblies

An analysis of the damaged fuel estimated to be in the SFP of the Duke Energy plants is provided in **Table A.3-1**. This information aids in identifying the potential need for loading of damaged fuel assemblies (e.g., identify how many damaged fuel slots are needed per 6625B-HB basket).

'Damaged fuel' is fuel that has been damaged, but can be handled by normal means. Damage is not considered severe. Damaged fuel may contain missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. Damaged fuel data will be addressed in other tasks contained within this report for the Duke Energy plants considered.

'Failed fuel', however, is fuel that has been severely damaged or fuel that cannot be handled by normal means. Failed fuel may contain ruptured fuel rods, severed fuel rods, or loose fuel pellets, which is outside the scope of this report.

Per this definition of failed and damaged fuel, there are some damaged FAs, as categorized and quantified in **Table A.3-1**, but these can be handled by normal means. The only minimal amount of "failed fuel" resides at Plant A, which is approximately 1.6% (around 24 FAs). The bounding (worst case) values in **Table A.3-1**, are summarized below:

- The worst case BWR could be assumed as Plant E SFP 2 with 6% damaged. This equates to approximately 4 ($61*0.06 = 3.636 \rightarrow 4$) damaged FA per the BWR (61 slot) 6625B-HB;
- The worst case PWR could be assumed as Plant A with 12.8% damaged. This equates to approximately 3 ($24*0.128 = 3.072 \rightarrow 3$) damaged FA per the PWR (24 slot) 6625B-HB.

TABLE A.3-1: DAMAGED FUEL ANALYSIS FOR PLANTS A-F

Damaged Fuel (Handled by normal means into DFC)					
Plant	Leaker	Physical Damage (i.e. Bent Fastener, Missing Tie Rod Nut, Rub Marks, Bowed, Dropped, Section missing between pins, Hard to seat, Spacer Damaged, etc..)	Total Damaged FAs	Total FAs	% Damaged
Plant A, SFP 1	-	-	61	989	6.2%
Plant A, SFP 2	-	-	61	478	12.8%
Plant B, SFP 1	-	-	6*	1090	0.5%
Plant B, SFP 2	-	-	6*	1138	0.5%
Plant C, SFP 1	-	-	6*	1247	0.5%
Plant C, SFP 1	-	-	6*	1267	0.5%
Plant D	5	11	16	282	6%
Plant E, SFP 1	14	12	26	1338	2%
Plant E, SFP 2	45	22	67	1131	6%
Plant F	25	81	106	6175	2%

*Estimated value

APPENDIX B: DOE CASK DATA TEMPLATE FOR 6625B-HB

Table B-1 is the DOE's "Cask Data Template" that has been filled out with data for the 6625B-HB conceptual transportation cask system designed in this report.

TABLE B-1: NON-CANISTER-BASED SYSTEM DATA

Cask: The shielded, self-contained, integrated system. A cask is typically bolted and can be used for storage, transfer and transport of fuel assemblies. Example: CASTOR-21	
Name	6625B-HB
Fabricator	TBD
Design/operation life	Per NRC License
Mode (storage, transportation, storage & transportation)	Transportation
Total assembly capacity	24 PWR / 61 BWR
- Zone 1 {if applicable}	4 PWR / 9 BWR
- Zone 2 {if applicable}	8 PWR / 16 BWR
- Zone 3 {if applicable}	4 PWR / 24 BWR
- Zone 4 {if applicable}	8 PWR / 12 BWR
Proposed Certificate of Compliance Limits	
Total thermal limit (kW)	30.4 BWR / 30.3 BWR
Thermal limit per cell (kW) {if applicable}	N/A
Thermal limit per zone {if applicable}	See zones
- Zone 1 (kW) {if applicable}	0.9 PWR / 0.33 BWR
- Zone 2 (kW) {if applicable}	1.4 PWR / 0.78 BWR
- Zone 3 (kW) {if applicable}	2.1 PWR / 0.45 BWR
- Zone 4 (kW) {if applicable}	0.9 PWR / 0.33 BWR
Drying procedures (vacuum, FHD, other)	vacuum drying
Criticality methodology for storage (boron credit, none)	N/A
Boron loading requirement in ppm {if applicable}	N/A
Max. enrichment for storage due to criticality requirements	N/A
Min. enrichment for storage due to shielding requirements	N/A
Min. cooling time for storage due to shielding & thermal requirements	N/A
Max. burnup for storage due to shielding & thermal requirements	N/A
Criticality methodology for transportation (burnup credit, none)	burnup credit (PWR)
Criticality loading curve {table or equation, if burnup credit}	See Table 2.6-1 (PWR)
Max. enrichment for transportation due to criticality requirements {if no burnup credit}	5% (BWR)
Shielding loading curve {table or equation}	See Tables 2.1-7 (PWR) and 2.1-10 (BWR)
Thermal loading curve {table or equation} {may be combined with shielding loading curve}	
High burnup fuel storage or transportation requirements (DFC, none)	N/A
Non-fuel hardware loading allowed (yes, no)	yes
Spacers allowed/required (required, allowed, none)	allowed
Restricted fuel class/type {if any}	Maximum irradiated assembly length of 180 inches

Physical Characteristics of DFC	
Outer length (cm)	457.2 Cavity
Outer width (cm)	N/A
Thickness (cm)	N/A
Physical Properties	
Length w/o impact limiters (cm)	509.3
Length w/ impact limiters (cm)	664.2
Diameter w/o impact limiters (cm)	236.9
Diameter w/ impact limiters (cm)	320.0
Cavity length (cm)	462.3
Cavity diameter (cm)	168.3
Top lid thickness including neutron shield (cm)	16.5 inner / 6.35 outer
Top neutron shield thickness (cm)	N/A
Bottom thickness including neutron shield (cm)	21.6
Bottom neutron shield thickness(cm)	N/A
Wall thickness including neutron shield (cm)	33.7 / 31.1 (ends)
Neutron shield side thickness (cm)	15.2 / 12.7 (ends)
Empty weight w/o impact limiters (lb.)	159,266
Empty weight w/ impact limiters (lb.)	183,500
Max. loaded weight w/o impact limiters - flooded (lb.)	254,920
Max. loaded weight w/o impact limiters - dry (lb.)	248,467
Max. loaded weight w/ impact limiters - dry (lb.)	265,202
Neutron shield type (type, none)	Vyal-B
Basket cell inner dimension (cm)	22.6 PWR / 15.2 BWR
Basket cell wall thickness (cm)	0.64 PWR / .34 BWR
Basket material	stainless steel/aluminum
Basket neutron poison material (type, none)	Enriched Boron Aluminum Alloy
Basket neutron poison B10 areal density {if any}	40.6 mg/cm^2 PWR 85.3 mg/cm^2 BWR
Flux trap (yes, no)	no
Overweight truck (yes, no)	rail car design

Unit Processing Times and Corresponding Dose	Time (hr)	Dose (mrem)
Cask loading for storage or transportation		
- Preparation	8.3	0.1
- SNF transfer from pool to cask	20.9 PWR 46.8 BWR	58.9 PWR 59.9 BWR
- Decontamination	1.25	42.12
- Drying	3.75	100.45
- Closing	4.25	265.06
- Loading onto vehicle {if applicable}	0.7	9.85
- Preparation for transport {if applicable}	7.7	128.57
Cask receipt and processing		
- Preparation	6.3	196.0
- Cask opening	6.1	177.6
- Fuel transfer from cask to pool	14.3 PWR 25.4 BWR	24.0 PWR 24.4 BWR
- Inspection	N/A	N/A
- Maintenance	N/A	N/A
Unit Costs (per cask)		
Cask purchase ($)	7,234,550	
Ancillary equipment - loading ($)	816,080	
Loading operation ($)	N/A	
Ancillary equipment - unloading ($)	N/A	
Unloading operation ($)	N/A	
Inspection ($)	N/A	
Maintenance ($)	N/A	
Refurbishment ($)	N/A	