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NAC-UMS

Universal Storage System

FINAL SAFETY ANALYSIS REPORT

for the UMS Universal Storage System

Docket No. 72-1015

NAC INTERNATIONAL

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1.0 GENERAL DESCRIPTION

NAC International Inc. (NAC) has designed a canister-based system for the storage and transportation of spent nuclear fuel. The system is designated the Universal MPC System® (UMS®). The storage component of the UMS® is designated the Universal Storage System. This Safety Analysis Report (SAR) demonstrates the ability of the Universal Storage System to satisfy the requirements of the U.S. Nuclear Regulatory Commission (NRC) for the storage of spent nuclear fuel as prescribed in Title 10 of the Code of Federal Regulations, Part 72 (10 CFR 72) [1], and NUREG-1536 [2]. The transportation component of the UMS® is designated the Universal Transportation System, which is addressed in the NAC Safety Analysis Report for the Universal Transport Cask, Docket No. 71-9270 [3].

The Universal Storage System primary components consist of the Transportable Storage Canister, Vertical Concrete Cask, and a transfer cask. The Transportable Storage Canister is designed and fabricated to meet the requirements for transport in the Universal Transport Cask (part of the Universal Transportation System) and to be compatible with the U.S. Department of Energy (DOE) MPC Design Procurement Specification [4], so as not to preclude the possibility of permanent disposal in a deep Mined Geological Disposal System.

In long-term storage, the Transportable Storage Canister is installed in a Vertical Concrete Cask, which provides passive radiation shielding and natural convection cooling. The Vertical Concrete Cask also provides protection during storage for the Transportable Storage Canister under adverse environmental conditions. The cask employs a double-welded closure design to preclude loss of contents and to preserve the general health and safety of the public during long-term storage of spent fuel.

The transfer cask is used to move the Transportable Storage Canister from the work stations where the canister is loaded and closed to the Vertical Concrete Cask. It is also used to transfer the canister from the Vertical Concrete Cask to the Universal Transport Cask for transport.

This Safety Analysis Report is formatted in accordance with U.S. NRC Regulatory Guide 3.61 [5]. This chapter provides a general description of the major components of the Universal Storage System and a description of system operation. Definition of terminology used throughout this report is summarized in Table 1-1. The term “concrete cask” or “cask” is routinely used to refer to the Vertical Concrete Cask. The term “Transportable Storage Canister” or “canister” is used to refer to both the PWR and BWR canisters where the discussion is common to both configurations. Discussion of features unique to each of the PWR and BWR configurations is handled in subsections, as appropriate, within each chapter.

Table 1.5-1 provides a compliance matrix to the regulatory requirements and acceptance criteria specified in NUREG-1536. This matrix describes how the Universal Storage System Safety Analysis Report addresses and demonstrates compliance with each requirement and criterion listed in NUREG-1536. Table B3-1 in Appendix B of the CoC Number 1015 Technical Specifications provides a list of the exceptions to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Table 1-1 Terminology

Universal Storage System	The storage component of the Universal MPC System (UMS®) designed by NAC for the storage and transportation of spent nuclear fuel.
Universal Transport Cask	The packaging consisting of a Universal Transport Cask body with a closure lid and energy-absorbing impact limiters. The Universal Transport Cask is used to transport a Transportable Storage Canister containing spent fuel. The cask body provides the primary containment boundary during transport.
Air Pad Rig Set (Air Pallet)	A device used to lift the Vertical Concrete Cask by using high volume air.
Assembly Defect	Any change in the physical as-built condition of the assembly, with the exception of normal in-reactor changes such as elongation from irradiation growth or assembly bow. Example of assembly defects include: (a) missing rods, (b) broken or missing grids or grid straps (spacer), and (c) missing or broken grid springs, etc. An assembly with a defect is damaged only if it cannot meet its fuel-specific and system-related functions.
Breached Spent Fuel Rod	Spent fuel with cladding defects that permit the release of gas from the interior of the fuel rod. A fuel rod breach may be a minor defect (i.e., hairline crack or pinhole), allowing the rod to be classified as undamaged, or be a gross breach requiring a damaged fuel classification.
Confinement System	The components of the Transportable Storage Canister intended to retain the radioactive material during storage.
Consolidated Fuel	A nonstandard fuel configuration in which the individual undamaged fuel rods from one or more fuel assemblies are placed in a single container or a lattice structure that is dimensionally similar to a fuel assembly. Consolidated Fuel is stored in a Maine Yankee Fuel Can.
Contents	Twenty-four PWR fuel assemblies, or fifty-six BWR fuel assemblies. The fuel assemblies may be configured as Site Specific Fuel. The fuel assemblies are contained in a Transportable Storage Canister.

Table 1-1 Terminology (Continued)

Damaged Fuel

Spent nuclear fuel (SNF) that cannot fulfill its fuel-specific or system-related function. Damaged fuel must be placed in a Maine Yankee Fuel Can unless otherwise noted. Spent fuel is classified as damaged under the following conditions.

1) There is visible deformation of the rods in the SNF assembly.

   Note: This is not referring to the uniform bowing that occurs in the reactor; this refers to bowing that significantly opens up the lattice spacing.

2) Individual fuel rods are missing from the assembly and the missing rods are not replaced by dummy rods that displace a volume equal to, or greater than, the original fuel rods.

   Note: Maine Yankee fuel assemblies with missing fuel rods, not replaced by filler rods, do not require placement into a Maine Yankee Fuel Can, but must be preferentially loaded.

3) The SNF assembly has missing, displaced or damaged structural components such that either:

   - Radiological and/or criticality safety is adversely affected (e.g., significantly changed rod pitch); or
   - The assembly cannot be handled by normal means (i.e., crane and grapple).

   Note: PWR assemblies with the following structural defects meet UMS system-related functional requirements and are, therefore, classified as undamaged.

   - Grid, grid strap, and/or grid strap spring damage in PWR assemblies such that the unsupported length of the fuel rod does not exceed 60 inches.

Table 1-1 Terminology (Continued)

Damaged Fuel (cont'd)	4) Any SNF assembly that contains fuel rods for which reactor operating records (or other records or tests) cannot support the conclusion that they do not contain gross breaches. Note: Breached fuel rods with minor cladding defects (i.e, pinhole leaks or hairline cracks that will not permit significant release of particulate matter from the spent fuel rod) meet UMS system-related functional requirements and are, therefore, classified as undamaged. 5) The SNF assembly is no longer in the form of an intact fuel bundle (e.g., consists of or contains debris such as loose fuel pellets or rod segments).
Fuel Basket (Basket)	The structure located within the Transportable Storage Canister that provides structural support, criticality control, and primary heat transfer paths for the fuel assemblies.
- Support Disk	The primary lateral load-bearing component of the fuel basket. The PWR support disk is a circular stainless steel plate with 24 square holes machined in a symmetrical pattern. The BWR support disk is a circular carbon steel plate with 56 square holes machined in a symmetrical pattern. Each square hole is a location for a fuel tube.
- Heat Transfer Disk	A circular aluminum plate with 24 (PWR basket) or 56 (BWR basket) square holes machined in a symmetrical pattern. The heat transfer disk enhances heat transfer in the fuel basket.
- Fuel Tube	A stainless steel tube having a square cross-section. One fuel tube is inserted through each square hole in the support disks and heat transfer disks. Fuel assemblies are loaded into the fuel tubes. A fuel tube may have neutron absorber material enclosed by a stainless steel sheet on one or more of its external faces, depending on fuel type and the position of the fuel tube in the basket.
- Tie Rod	A stainless steel rod used to align, retain, and support the support disks and the heat transfer disks in the fuel basket structure. The tie rods extend from the top weldment to the bottom weldment.

Table 1-1 Terminology (Continued)

- Spacer	Installed on the tie rod between the support disks (BWR only) or between the support disks and top and bottom weldments (BWR and PWR) to properly position the disks and provide axial support for the support disks.
- Split Spacer	Spacers installed on the tie rod between the support disks and the heat transfer disks to properly position the disks and provide axial support for the support disks and the heat transfer disks.
Grossly Breached Spent Fuel Rod	A breach in the spent fuel cladding that is larger than a pinhole or hairline crack. A gross cladding breach may be established by visual examination with the capability to determine if the fuel pellet can be seen through the cladding, or through a review of reactor operating records indicating the presence of heavy metal isotopes.
Heavy Haul Trailer	The trailer used to transport the empty or loaded Vertical Concrete Cask.
High Burnup Fuel	A fuel assembly meeting the definition of a standard fuel assembly with an assembly average burnup between 45,000 and 60,000 MWd/MTU. Maximum peak average rod burnup is limited to 62,500 MWd/MTU.
Intact Fuel (Assembly or Rod)	Any fuel that can fulfill all fuel-specific and system-related functions and that is not breached.
Maine Yankee Fuel Can	A specially designed stainless steel screened can sized to hold an undamaged fuel assembly, consolidated fuel, or damaged fuel. The can screens permit draining and drying, while precluding the release of gross particulates into the canister cavity. The Maine Yankee Fuel Can may only be loaded in a Class 1 Canister.
Margin of Safety	An analytically determined value defined as the "factor of safety" minus 1. Factor of safety is also analytically determined, and is defined as the allowable stress or displacement of a material divided by its actual (calculated) value.
NS-4-FR or NS-3	Solid hydrogenous materials with neutron absorption capabilities.

Table 1-1 Terminology (Continued)

Site Specific Fuel	Spent fuel configurations that are unique to a site or reactor due to the addition of other components or reconfiguration of the fuel assembly at the site. It includes fuel assemblies, which hold nonfuel-bearing components, such as control components or instrument and plug thimbles, or which are modified as required by expediency in reactor operations, research and development or testing. Modification may consist of individual fuel rod removal, fuel rod replacement of similar or dissimilar material or enrichment, the installation, removal or replacement of burnable poison rods, or containerizing damaged (failed) fuel. Site specific fuel includes irradiated fuel assemblies designed with variable enrichments and/or axial blankets, fuel that is consolidated and fuel that exceeds design basis fuel parameters.
Shield Lid	A thick stainless steel disk that is located directly above the fuel basket. The shield lid comprises the first part of a double-welded closure system for the Transportable Storage Canister. The shield lid provides a containment/confinement boundary for storage and shielding for the contents.
- Drain Port	A penetration located in the shield lid to permit draining of the canister cavity.
- Vent Port	A penetration located in the shield lid to aid in draining and in vacuum drying and backfilling the canister with helium.
- Port Cover	The stainless steel covers that close the vent and drain ports, and that are welded in place following draining, drying, and backfilling operations.
-Quick Disconnect	The valved nipple used in the vent and drain ports to facilitate operations.

Table 1-1 Terminology (Continued)

Standard Fuel	Irradiated fuel assemblies with a burnup less than, or equal to, 45,000 MWd/MTU and having the same configuration as when originally fabricated consisting generally of the end fittings, fuel rods, guide tubes, and integral hardware. For PWR fuel, a flow mixer (thimble plug), an in-core instrument thimble, a burnable poison rod insert, or a solid stainless steel rod insert is considered to be a component of standard fuel. For BWR fuel, the channel is considered to be integral hardware. The design basis fuel characteristics and analysis are based on the standard fuel configuration.
Structural Lid	A thick stainless steel disk that is positioned on top of the shield lid and welded to the canister. The structural lid is the second part of a double-welded closure system for the Transportable Storage Canister. The structural lid provides a confinement boundary for storage, shielding for the contents, and canister lifting/handling capability.
Transfer Adapter	A carbon steel plate assembly that is positioned on to the top of the transport or concrete cask to facilitate installation and alignment of the transfer cask. It also provides the operating mechanism for the transfer cask bottom doors.
Transfer Cask	A shielded lifting device for handling of the Transportable Storage Canister during loading of spent fuel, canister closure operations, and transfer of the canister into or out of the Vertical Concrete Cask during storage, or into or out of the Universal Transport Cask during transportation. The transfer cask incorporates bottom doors that permit the vertical loading of the storage and transport casks. The transfer cask is provided in either the standard or the advanced configuration. The advanced configuration has a higher weight capacity.
- Transfer Cask Lifting Trunnions	Four low alloy steel trunnions used to lift and move the transfer cask in a vertical orientation.

Table 1-1 Terminology (Continued)

Transportable Storage Canister (Canister)	The stainless steel cylindrical shell, bottom end plate, shield lid, and structural lid that cont ain the fuel basket structure and the contents.
Undamaged Fuel	Spent nuclear fuel that can meet all fuel specific and system-related functions. Undamaged Fuel is spent nuclear fuel that is not Damaged Fuel, as defined herein, and does not contain assembly structural defects that adversely affect radiological and/or criticality safety. As such, Undamaged Fuel may contain: a) Breached spent fuel rods (i.e, rods with minor defects up to hairline cracks or pinholes) but can not contain grossly breached fuel rods; b) Grid, grid strap, and/or grid spring damage in PWR assemblies, provided that the unsupported length of the fuel rod does not exceed 60 inches.
Vertical Concrete Cask (Concrete Cask)	A concrete cylinder that contains the Transportable Storage Canister during storage. The Vertical Concrete Cask is formed around a steel inner liner and base and is closed by a shield plug and lid.
- Shield Plug	A thick carbon steel plug, which also contains a neutron shield material, installed in the top end of the Vertical Concrete Cask to reduce skyshine radiation.
- Lid	A thick carbon steel plate that serves as the bolted closure for the Vertical Concrete Cask. The lid precludes access to the canister and provides additional radiation shielding.
- Liner	A thick carbon steel shell that forms the annulus of the concrete cask. The liner serves as the inner form during concrete pouring and provides radiation shielding of the canister contents.
- Base	A carbon steel weldment that contains the air inlets, the concrete cask jacking points and the pedestal that supports the canister inside of the concrete cask.

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1.1 Introduction

The Universal Storage System is a spent fuel dry storage system that uses a Vertical Concrete Cask and a stainless steel Transportable Storage Canister with a double welded closure to safely store spent fuel. The Transportable Storage Canister is stored in the central cavity of the Vertical Concrete Cask and is compatible with the Universal Transport Cask for future off-site shipment. The concrete cask provides radiation shielding and contains internal air flow paths that allow the decay heat from the canister contents to be removed by natural air circulation around the canister wall. The Universal Storage System is designed and analyzed for a 50-year service life.

The principal components of the Universal Storage System are the canister, the concrete cask, and the transfer cask. The loaded canister is moved to and from the concrete cask by using the transfer cask. The transfer cask provides radiation shielding while the canister is being closed and sealed and while the canister is being transferred. The canister is placed in the concrete cask by positioning the transfer cask with the loaded canister on top of the concrete cask and lowering the canister into the concrete cask. Figure 1.1-1 depicts the major components of the Universal Storage System in such a configuration.

The Universal Storage System is designed to safely store up to 24 PWR or up to 56 BWR spent fuel assemblies. The fuel specifications and parameters that serve as the design basis are presented in Tables 2.1.1-1 and 2.1.2-1 for PWR and BWR fuel assemblies, respectively. The spent fuel considered in the design basis includes fuel assemblies that have different overall lengths. The range of overall lengths of the PWR fuel assembly population is grouped into three classes. To accommodate the three classes, the Universal Storage System principal components—the transportable storage canister, transfer cask and vertical concrete cask—are provided in three different lengths. Similarly, BWR fuel assemblies are grouped into two classes, which are also accommodated by two different lengths of the principal components. The class designations of these principal components, and corresponding lengths, are shown on the License Drawings. The identification of representative fuel assemblies, by class, is shown in Tables 6.2-1 and 6.2-2 for PWR and BWR fuel, respectively. Fuel assemblies were grouped to facilitate licensing evaluations. Bounding configurations were evaluated and no restriction is placed on the loading of a given fuel assembly type into a particular UMS® canister class.

The inclusion of nonfuel-bearing components or fixtures in a fuel assembly can increase its overall length, resulting in the need to use the next longer class of Universal Storage System components. Stainless steel spacers may be used in a given class of canister to allow loading of fuel that is significantly shorter than the canister length. The BWR fuel assembly classes are evaluated for the effects of the zirconium alloy channel that surrounds the fuel assembly in reactor operations.

In addition to the design basis fuel, fuel that is unique to a certain reactor site, referred to as site specific fuel, is also evaluated. Site specific fuel consists of fuel assemblies that are configured differently, or have different parameters (such as enrichment or burnup), than the design basis fuel assemblies. These site specific fuel configurations result from conditions that occurred during reactor operations, from participation in research and development programs (testing programs intended to improve reactor operations), or from the insertion of control components or other items within the fuel assembly.

Site specific spent fuels are described in Section 1.3.2. These site specific fuel configurations are either shown to be bounded by the design basis fuel analysis, or are separately evaluated. Unless specifically excepted, site specific fuel must also meet the conditions for the design basis fuel presented in Section 1.3.1.

Three canister classes accommodate the PWR fuel assemblies, and two canister classes accommodate the BWR fuel assemblies. Each of the five canisters is stored in a concrete cask of specific length designed to accommodate the specific canister. The fuel is loaded into the appropriate canister prior to movement of the canister into the concrete cask. Figure 1.1-2 depicts a Transportable Storage Canister containing a PWR spent fuel basket. A canister containing a BWR spent fuel basket is shown in Figure 1.1-3.

The system design and analyses are performed in accordance with 10 CFR 72, ANSI/ANS 57.9 [6] and the applicable sections of the ASME Boiler and Pressure Vessel Code and the American Concrete Institute Code [7].

Figure 1.1-1 Major Components of the Universal Storage System (in Vertical Concrete Cask Loading Configuration)

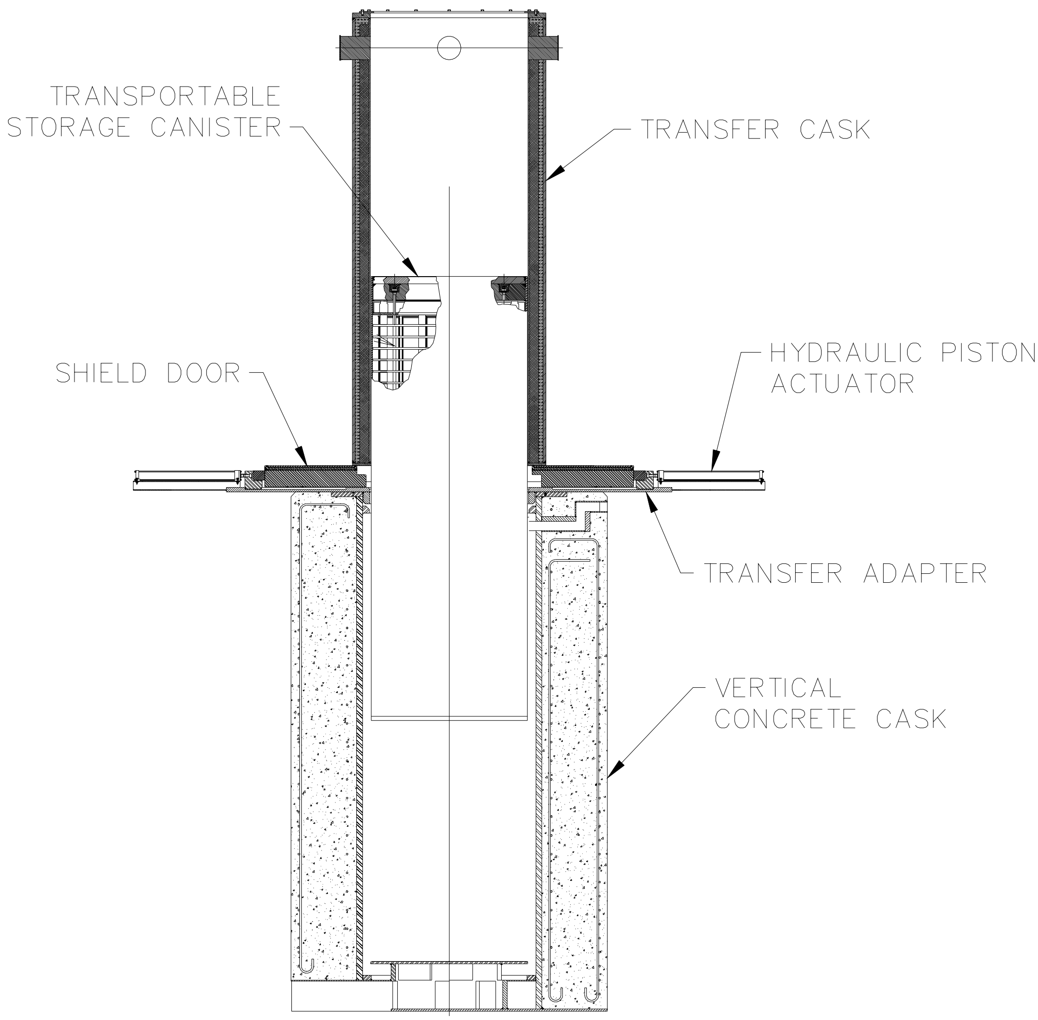


Figure 1.1-2 Transportable Storage Canister Containing PWR Spent Fuel Basket

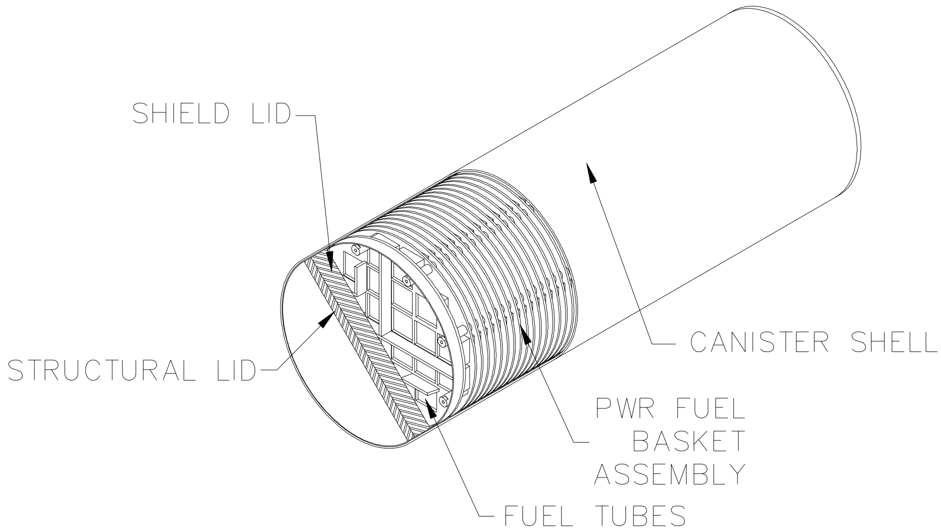
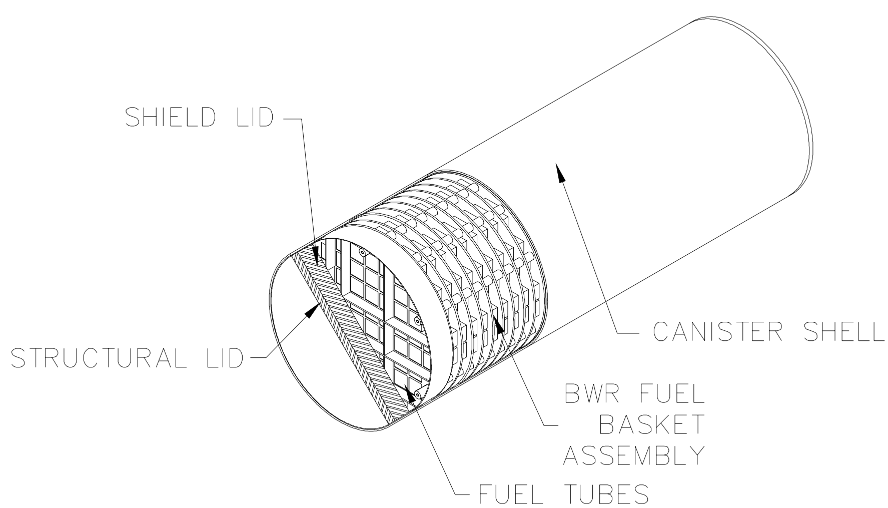


Figure 1.1-3 Transportable Storage Canister Containing BWR Spent Fuel Basket



1.2 General Description of the Universal Storage System

The Universal Storage System provides long-term storage of any of three classes of PWR fuel or two classes of BWR fuel, and subsequent transport using a Universal Transport Cask (Docket 71-9270). During long-term storage, the system provides an inert environment; passive shielding, cooling, and criticality control; and a confinement boundary closed by welding. The structural integrity of the system precludes the release of contents in any of the design basis normal conditions and off-normal or accident events, thereby assuring public health and safety during use of the system.

1.2.1 Universal Storage System Components

The design and operation of the principal components of the Universal Storage System and the associated ancillary equipment are described in the following sections. The weights of the principal components are provided in Section 3.2.

The Universal Storage System consists of three principal components:

- Transportable Storage Canister (including PWR or BWR fuel basket),
- Vertical Concrete Cask, and
- Transfer Cask.

The design characteristics of these components are presented in Table 1.2-1.

Ancillary equipment needed to use the Universal Storage System are:

- Automated or manual welding equipment;
- An air pallet or hydraulic roller skid (used to move the concrete cask on and off the heavy haul trailer and to position the concrete cask on the storage pad);
- Suction pump, vacuum drying, helium backfill and leak detection equipment;
- A heavy haul trailer or transporter (for transport of concrete cask to the storage pad);
- An adapter plate and hardware to position the transfer cask with respect to the storage or transport cask; and
- A lifting yoke for the transfer cask and lifting slings for the canister and canister lids.

In addition to these items, the system requires utility services (electric, helium, air and water), common tools and fittings, and miscellaneous hardware.

1.2.1.1 Transportable Storage Canister

Three classes of Transportable Storage Canisters accommodate the PWR fuel assemblies, and two classes of Transportable Storage Canisters accommodate the BWR fuel assemblies. The canister is designed to be transported in the Universal Transport Cask. Transport conditions establish the design basis load conditions for the canister, except for canister lifting. The transport load conditions produce higher stresses in the canister than would be produced by the storage load conditions. Consequently, the canister design is conservative with respect to storage conditions. The evaluation of the canister for transport conditions is documented in the Safety Analysis Report for the Universal Transport Cask, Docket No. 71-9270.

The Transportable Storage Canister consists of a stainless steel canister that contains the fuel basket structure and contents. The canister is defined as confinement for the spent fuel during storage and is provided with a double welded closure system. The welded closure system prevents the release of contents in any design basis normal, off-normal or accident condition. The basket assembly in the canister provides the structural support and primary heat transfer path for the fuel assemblies while maintaining a subcritical configuration for all normal conditions of storage, off-normal events and hypothetical accident conditions. The PWR and BWR fuel basket assemblies are discussed in Section 1.2.1.2.

The major components of the Transportable Storage Canister are the shell and bottom, basket assembly, shield lid, and structural lid. The canister and the shield and structural lids provide a confinement boundary during storage, shielding, and lifting capability for the basket. The Transportable Storage Canister design parameters for the storage of the five classes of fuel are provided in Table 1.2-2.

The canister consists of a cylindrical, 5/8-inch thick Type 304L stainless steel shell with a 1.75-inch thick Type 304L stainless steel bottom plate and a Type 304 stainless steel shield lid support ring. A basket assembly is placed inside the canister. The shield lid assembly is a 7-inch thick Type 304 stainless steel disk that is positioned on the shield lid support ring above the basket assembly. The shield lid is welded to the canister after the canister is loaded and moved to the workstation for completion of canister closure activities. Two penetrations through the shield lid are provided for draining, vacuum drying, and backfilling the canister with helium. The drain pipe is threaded into the shield lid after the canister is moved to the workstation. The vent penetration in the shield lid is used to aid water removal and for vacuum drying and backfilling the canister with helium. After the shield lid is welded in place, it is pressure and leakage tested to ensure no credible leakage of the confinement boundary during storage.

The structural lid is a 3-inch thick Type 304L stainless steel disk positioned on top of the shield lid and welded to the shell after the shield lid is welded in place and the canister is drained, dried, and backfilled with helium. Removable lifting fixtures, installed in the structural lid, are used to lift and lower the loaded canister.

The Transportable Storage Canister is designed to the requirements of the ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Division I, Subsection NB [8]. It is fabricated and assembled in accordance with the requirements of Subsection NB to the maximum extent practicable, consistent with the conditions of use. Exceptions to the ASME Code are noted in Table B3-1 in Appendix B.

A summary of the canister fabrication specifications is presented in Table 1.2-3. As shown in that table, the field installed welds joining the shield and structural lids to the canister shell are not full penetration welds. The shield lid weld is dye penetrant inspected on the root and final cover pass. The structural lid weld is either ultrasonically inspected when completed or it is dye penetrant inspected on the root and final cover passes and on each 3/8-inch intermediate layer. These inspections assure weld integrity in accordance with the requirements of ASME Code Section V, Articles 5 and 6 [9], as appropriate. The weld joining the shield lid to the canister shell is pressure tested and leak tested as described in Section 8.1.1. The structural and shield lid welds are made with the aid of a backing ring (also called a spacer ring) or shims, which cannot be removed when the weld is completed. There are no detrimental effects that result from the presence of the spacer ring or shims, and no structural credit is taken for their presence.

The design of the transportable storage canister and its fabrication controls would allow the canister to be ASME Code stamped in accordance with the ASME Code Section III, if desired.

1.2.1.2 Fuel Baskets

The transportable storage canister contains a fuel basket which positions and supports the stored fuel in normal, off-normal and accident conditions. As described in the following sections, the design of the basket is similar for the PWR and BWR configurations. The fuel basket for each fuel type is designed and fabricated to the requirements of the ASME Code, Section III, Division I, Subsection NG [10]. However, the basket assembly is not Code stamped and no reports relative to Code stamping are prepared. Consequently, an exception is taken to Article NG-8000, Nameplates, Stamping and Reports.

1.2.1.2.1 PWR Fuel Basket

The PWR fuel basket is contained within the transportable storage canister. It is constructed of stainless steel, but incorporates aluminum disks for enhanced heat transfer. The fuel basket design is a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks. The basket design parameters for the storage of the three classes of PWR fuel are provided in Table 1.2-4. The Class 1, 2 or 3 fuel baskets incorporate 30, 32 or 34 support disks, respectively. The disks are retained by a top nut and supported by spacers on tie rods at eight locations. The top nut is torqued at installation to provide a solid load path in compression between the support disks. The support disks are fabricated of SA-693, Type 630, 17-4 PH stainless steel. The disks are spaced axially at 4.92 inches center-to-center and contain square holes for the fuel tubes.

The top and bottom weldments are fabricated from Type 304 stainless steel and are geometrically similar to the support disks. The tie rods and top nuts are fabricated from SA-479, Type 304 stainless steel. The top nut is fabricated from a 3.5-in.-diameter bar, and the spacers are fabricated from a 2.5-in. pipe XXS, Type 304 stainless steel. The fuel tubes are fabricated from A-240, Type 304 stainless steel and support an enclosed neutron absorber sheet on each of the four sides. The neutron absorber provides criticality control in the basket. No credit is taken for the fuel tubes for structural strength of the basket or support of the fuel assemblies.

Each PWR fuel basket has a capacity of 24 PWR fuel assemblies in an aligned configuration in 8.80-inch square fuel tubes. The holes in the top weldment are 8.75-inch square. The holes in the bottom weldment are 8.65-inch square. The basket design traps the fuel tube between the top and bottom weldments, thereby preventing axial movement of the fuel tube. The support disk configuration includes webs between the fuel tubes with variable widths depending on location.

The PWR basket design incorporates Type 6061-T651 aluminum alloy heat transfer disks to enhance heat transfer in the basket. Twenty-nine heat transfer disks are contained in the Class 1 basket. Class 2 and 3 fuel baskets contain 31 and 33 disks, respectively. The heat transfer disks are spaced and supported by the tie rods and spacers, which also support and locate the support disks. The heat transfer disks, located at the center of the axial spacing between the support disks, are sized to eliminate contact with the canister inner shell due to differential thermal expansion.

The Transportable Storage Canister is designed to facilitate filling with water and subsequent draining. Water fills and drains freely between the basket disks through three separate paths. One path is the gaps that exist between the disks and canister shell. The second path is through the gaps between the fuel tubes and disk that surrounds the fuel tubes. The third path is through three 1.3-inch diameter holes in each of the disks that are intended to provide additional paths for water flow between disks. The basket bottom weldment supports the fuel tubes above the canister bottom plate. The fuel tubes are open at the top and bottom ends, allowing the free flow of water from the bottom of the fuel tube. The bottom weldment is positioned by supports 1.0 inch above the canister bottom to facilitate water flow to the drain line. These design features ensure that water flows freely in the basket so that the canister fills and drains evenly.

1.2.1.2.2 BWR Fuel Basket

Like the PWR fuel basket, the BWR basket is contained within the stainless steel Transportable Storage Canister. The BWR fuel basket is also a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks (40 disks for the Class 4 fuel basket and 41 disks for the Class 5 fuel basket). The basket design parameters for the storage of the two classes of BWR fuel are provided in Table 1.2-4. The support disks are retained by cylindrical spacers on tie rods at six locations. The top nut is torqued at installation to provide a solid load path in compression between the support disks. The support disks are fabricated of SA-533, Type B, Class 2 carbon steel and are coated with electroless nickel to inhibit corrosion and the formation of combustible gases during fuel loading. The disks are spaced axially at 3.8-inch center-to-center and contain square holes for the fuel tubes.

The top and bottom weldments are fabricated from Type 304 stainless steel, and are geometrically similar to the support disks. The fuel tubes are also fabricated from Type 304 stainless steel. Three types of tubes are designed to contain one BWR fuel assembly: tubes with neutron absorber on two sides, tubes with neutron absorber one side, and tubes with no neutron absorber. No credit is taken for the fuel tubes for structural strength of the basket or support of the fuel assemblies.

Each BWR fuel basket has a capacity of 56 BWR fuel assemblies in an aligned configuration. The fuel tubes in 52 positions have an inside square dimension of 5.90 inches. The inside dimension of the four fuel tubes located in the outside corners of the basket array is 6.05-inches square. The holes in the top weldment are 5.75 inches by 5.75 inches, except for the four enlarged holes, which are 5.90 inches-square. The holes in the bottom weldment are 5.63-inches square. The basket design traps the fuel tube between the top and bottom weldments, thereby preventing axial movement of the fuel tube. The support disk webs between the fuel tubes are 0.65-inch wide. The BWR fuel basket design also incorporates 17 Type 6061-T651 aluminum alloy heat transfer disks similar in design and function of those in the PWR baskets.

The BWR canister is also designed to facilitate filling with water and subsequent draining. Water fills and drains freely between the basket disks through three separate paths. One path is the gaps that exist between the disks and canister shell. The second path is through the gaps between the fuel tubes and disk that surrounds the fuel tubes. The third path is through three 1.3-inch diameter holes in each of the disks that are intended to provide additional paths for water flow between disks. The basket bottom weldment supports the fuel tubes above the canister bottom plate. The fuel tubes are open at the top and bottom ends, allowing the free flow of water from the bottom of the fuel tube. The bottom weldment is positioned by supports 1.0 inch above the canister bottom to facilitate water flow to the drain line. These design features ensure that water flows freely in the basket so that the canister fills and drains evenly.

1.2.1.3 Vertical Concrete Cask

The Vertical Concrete Cask is the storage overpack for the Transportable Storage Canister. Five concrete casks of different lengths are designed to store five canisters of different lengths containing one of three classes of PWR or of two classes of BWR fuel assemblies. The concrete cask provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage. Table 1.2-5 lists the principal physical design parameters of the concrete cask.

The concrete cask is a reinforced concrete (Type II Portland cement) structure with a structural steel inner liner. The concrete wall and steel liner provide the neutron and gamma radiation shielding to reduce the average contact dose rate to less than 50 millirem per hour for design basis PWR or BWR fuel. Inner and outer reinforcing steel (rebar) assemblies are contained within the concrete. The reinforced concrete wall provides the structural strength to protect the canister and its contents in natural phenomena events such as tornado wind loading and wind driven missiles. The concrete cask incorporates reinforced chamfered corners at the edges to facilitate construction. The concrete cask is shown in Figure 1.2-1.

The Vertical Concrete Cask forms an annular air passage to allow the natural circulation of air around the canister to remove the decay heat from the spent fuel. The air inlets and outlets are steel-lined penetrations that take nonplanar paths to the concrete cask cavity to minimize radiation streaming. A baffle assembly directs inlet air upward and around the pedestal that supports the canister. The weldment structure includes the baffle assembly configuration, as shown in Drawing 790-561. The decay heat is transferred from the fuel assemblies to the tubes in the fuel basket and through the heat transfer disks to the canister wall. Heat flows by radiation and convection from the canister wall to the air circulating through the concrete cask annular air passage and is exhausted through the air outlets. This passive cooling system is designed to maintain the peak cladding temperature of the zirconium alloy-clad fuel well below acceptable limits during long-term storage. This design also maintains the bulk concrete temperature below 150°F and localized concrete temperatures below 200°F in normal operating conditions.

The top of the Vertical Concrete Cask is closed by a shield plug and lid. The shield plug is approximately 5 inches thick and incorporates carbon steel plate as gamma radiation shielding, and NS-4-FR or NS-3 as neutron radiation shielding. A carbon steel lid that provides additional gamma radiation shielding is installed and bolted in place above the shield plug. The shield plug and lid reduce skyshine radiation and provide a cover and seal to protect the canister from the environment and postulated tornado missiles. At the option of the user, a tamper-indicating seal wire and seal may be installed on two of the concrete cask lid bolts. An optional supplemental shielding fixture, shown in Drawing 790-613, may be installed in the air inlets to reduce the radiation dose rate at the base of the cask.

Fabrication of the concrete cask involves no unique or unusual forming, concrete placement, or reinforcement requirements. The concrete portion of the concrete cask is constructed by placing concrete between a reusable, exterior form and the inner metal liner. Reinforcing bars are used near the inner and outer concrete surfaces, to provide structural integrity. The inner liner and base of the concrete cask are shop fabricated. The principal fabrication specifications for the concrete cask are shown in Table 1.2-6.

1.2.1.4 Transfer Cask

The transfer cask is a heavy lifting device, which is designed, fabricated, and load-tested to meet the requirements of NUREG-0612 [11] and ANSI N14.6 [12]. The transfer cask can be provided in either a Standard or Advanced configuration. Canister handling, fuel loading and canister closing are operationally identical for either transfer cask configuration.

The transfer cask provides biological shielding when it contains a loaded canister and is used for the vertical transfer of the canister between work stations and the concrete cask, or transport cask. Five transfer casks of either configuration, having different lengths, are designed to handle the five canisters of different lengths containing one of three classes of PWR fuel assemblies or two classes of BWRfuel assemblies. In addition, a Transfer Cask Extension may be used to extend the operational height, when using the standard transfer cask. This height extension allows a transfer cask designed for a specific canister class to be used with the next longer canister.

The transfer cask design incorporates a top retaining ring, which is bolted in place to prevent a loaded canister from being inadvertently lifted through the top of the transfer cask. The transfer cask has retractable bottom shield doors. During loading operations, the doors are closed and secured by door lock bolts/lock pins, so they cannot inadvertently open. During unloading, the doors are retracted using hydraulic cylinders to allow the canister to be lowered into a concrete cask for storage or into a transport cask. A typical transfer cask is shown in Figure 1.2-2. The principal design parameters of the transfer casks are shown in Table 1.2-7.

To minimize the potential for contamination of a canister or the inside of the transfer cask during loading operations in the spent fuel pool, clean water is circulated in the annular gap between the transfer cask interior surface and the canister exterior surface. Clean water is processed or filtered pool water, or any water external to the spent fuel pool that is compatible. The transfer cask has eight supply and two discharge lines passing through its wall. Normally, two of the lines are connected to allow clean water under pressure to flow into and through the annular gap to minimize potential for the intrusion of pool water when the canister is being loaded. Lines not used for clean water supply may be capped. The eight supply lines can also be used for the introduction of forced air at the bottom of the transfer cask to achieve cooling of the canister contents. This allows the canister to remain in the transfer cask for an extended period, if necessary, during canister closing operations.

Standard and Advanced Transfer Casks

The Standard and Advanced transfer casks are designed for lifting and handling in the vertical orientation only. The Standard transfer cask may be used to lift canisters weighing up to 88,000 pounds. The Advanced transfer cask is similar to the Standard transfer cask, except that the Advanced transfer cask incorporates a trunnion support plate that allows the Advanced transfer cask to lift canisters weighing up to 98,000 pounds. The Standard and Advanced transfer casks have four lifting trunnions, which allow for redundant load path lifting. Both transfer casks incorporate a multiwall (steel/lead/NS-4-FR/steel) design, and both designs have a maximum empty weight of approximately 121,500 pounds. The Standard and Advanced transfer cask designs are shown in Drawing 790-560.

1.2.1.5 Auxiliary Equipment

This section presents a brief description of the principal auxiliary equipment needed to operate the Universal Storage System in accordance with its design.

1.2.1.5.1 Transfer Adapter

The transfer adapter is a carbon steel table that is positioned on the top of the Vertical Concrete Cask or the Universal Transport Cask and mates the transfer cask to either of those casks. It has a large center hole that allows the Transportable Storage Canister to be raised or lowered through the plate into or out of the transfer cask. Rails are incorporated in the transfer adapter to guide and support the bottom shield doors of the transfer cask when they are in the open position. The transfer adapter also supports the hydraulic system and the actuators that open and close the transfer cask bottom doors.

1.2.1.5.2 Air Pad Rig Set

The air pad rig set (air pad set) is a commercially available device, sometimes referred to as an air pallet. When inflated, the air pad rig set lifts the concrete cask by using high volume air flow. The air pads employ a continuous, regulated air flow and a control system that equalizes lifting heights of the four air pads by regulating compressed air flow to each of the air pads. The compressed air supply creates an air film between the inflated air cushion and the supporting surface. The thin film of air allows the concrete cask to be lifted and moved. Once lifted, the cask can be moved by a suitable towing vehicle, such as a commercial tug or forklift.

1.2.1.5.3 Automatic Welding System

The automatic welding system consists of commercially available components with a customized weld head. The components include a welding machine, a remote pendant, a carriage, a drive motor and welding wire motor, and the weld head. The system is designed to make at least one weld pass automatically around the canister after its weld tip is manually positioned at the proper location. As a result, radiation exposure during canister closure is much less than would be incurred from manual welding.

1.2.1.5.4 Draining and Drying System

The draining and drying system consists of a suction pump and a vacuum pump. The suction pump is used to remove free water from the canister cavity. The vacuum pump is a two-stage unit for drying the interior of the canister. The first stage is a large capacity or "roughing" pump intended to remove free water not removed by the suction pump. The second stage is a vacuum pump used to evacuate the canister interior of the small amounts of remaining moisture and establish the vacuum condition.

1.2.1.5.5 Lifting Jacks

Hydraulic jacks are installed at jacking pads in the air inlets at the bottom of the concrete cask to lift the cask so that the air pad set can be installed or removed. Four hydraulic jacks are provided, along with a control panel, an electric hydraulic oil pump, an oil reservoir tank and all hydraulic lines and fittings. The jacks are used to lift the cask approximately three inches. This permits installation of the air pad rig set under the concrete cask.

1.2.1.5.6 Heavy-Haul Trailer

The heavy-haul trailer is used to move the Vertical Concrete Cask. A special trailer is designed for transport of the empty or loaded concrete cask. The design incorporates a jacking system that facilitates raising the concrete cask to allow installation of the air pad set used to move the cask onto the storage pad. The trailer incorporates both reinforcing to increase the trailer load-bearing area and design features that reduce its turning radius. However, any commercial double-drop-frame trailer having a deck height approximately matching that of the storage pad could be used.

1.2.1.5.7 Transporter

A cask transporter may also be used to move an empty or loaded Vertical Concrete Cask. The typical design incorporates a vertical lifting system that raises the concrete cask using the Vertical Concrete Cask lifting lugs. The transporter may be a self-propelled, towed or pushed design.

1.2.1.5.8 Helium Leak Test Equipment

A helium leak detector and leak test fixtures are required to verify the integrity of the welds of the canister shield lid. The helium leak detector is the mass spectrometer type.

1.2.1.5.9 Rigging and Slings

Load rated rigging attachments and slings are provided for major components. The rigging attachments are swivel hoist rings that allow attachment of the slings to the hook. All slings are commercially purchased to have adequate safety margin to meet the requirements of ANSI N14.6 and NUREG-0612. The slings include a concrete cask lid sling, concrete cask shield plug sling, canister shield lid sling, loaded canister transfer sling (also used to handle the structural lid), and a canister retaining ring sling. The appropriate rings or eye bolts are provided to accommodate each sling and component. Note: A cask user may utilize other slings, as needed, to perform the numerous required lifts of the UMS® components, provided that the slings meet all applicable safety requirements.

The transfer cask lifting yoke is specially designed and fabricated for lifting the transfer cask. It is designed to meet the requirements of ANSI N14.6 and NUREG-0612. It is designed as a special lifting device for critical loads. The transfer cask lifting yoke is initially load tested to 300 percent of the maximum service load.

1.2.1.5.10 Transfer Cask Extension

A transfer cask extension may be used to extend the operational height of a transfer cask by approximately 10 inches. This height extension allows a transfer cask designed for a specific canister class to be used with the next longer canister. The extension is stainless steel.

1.2.1.5.11 Temperature Instrumentation

The concrete casks may be equipped with temperature-monitoring equipment to measure the outlet air temperature. The Technical Specification requires either daily temperature measurements or daily visual inspection for inlet and outlet blockage to ensure the cask remains operable.

1.2.1.6 Universal Transport Cask

The Universal Transport Cask is designed to transport the Transportable Storage Canister. The canister, which may contain PWR or BWR spent fuel, is positioned in the Universal Transport Cask cavity by axial spacer(s) at the bottom of the cavity. A Class 1, 2 or 5 canister is located by one spacer. A Class 4 canister is located by four spacers. A Class 3 canister has no spacers. The spacer(s) are required because the Universal Transport Cask cavity length is 192.5 inches, while the lengths of the canisters for different classes of fuel vary from 175.3 inches to 192.0 inches.

The transport configuration of the Universal Transport Cask is shown in Figure 1.2-3. The Universal Transport Cask is assigned 10 CFR 71 [13] Docket No. 71-9270 [3].

1.2.2 Operational Features

The principal activities associated with the use of the Universal Storage System are closing the canister and loading the canister in the concrete cask. The transfer cask is designed to meet the requirements of these operations. The transfer cask holds the canister during loading with fuel; provides biological shielding during closing of the canister; and provides the means by which the loaded canister is moved to, and installed in, the concrete cask.

The canister consists of five principal components: the canister shell (side wall and bottom); the shield lid; the vent port; the drain port (together with the vent and drain port covers); and the structural lid. A drain tube extends from the shield lid drain port to the bottom of the canister. The location of the drain and vent ports is shown in Figure 8.1.1-1. The vent and drain ports allow the draining, vacuum drying, and backfilling with helium necessary to provide a dry, inert atmosphere for the contents. The vent and drain port covers, the shield lid, the canister shell, and the joining welds form the primary confinement boundary. A secondary confinement boundary is formed over the shield lid by the structural lid and the weld that joins it to the canister shell. The primary and secondary boundaries are shown in Figures 7.1-1 and 7.1-2.

The structural lid contains the drilled and tapped holes for attachment of the swivel hoist rings used to lift the loaded canister. The drilled and tapped holes are filled with bolts or plugs to avoid collecting debris, and to preclude the possibility of radiation streaming from the holes, when the hoist rings are not installed.

The step-by-step procedures for the operation of the Universal Storage System are presented in Chapter 8.0. The following is a list of the principal activities. This list assumes that the empty canister is installed in the transfer cask for spent fuel pool loading (see Figure 1.2-4).

- Lift the transfer cask over the pool and start the flow of clean or filtered pool water to the transfer cask annulus and canister. After the annulus and canister fill, lower the cask to the bottom of the pool.

- Load the selected spent fuel assemblies into the canister and set the shield lid.
- Raise the transfer cask from the pool. Decontaminate the transfer cask exterior as it clears the pool surface. Drain the annulus. Place the transfer cask in the decontamination area.

  Note: As an alternative, some sites may choose to perform welding operations for closure of the canister in a cask loading pit with water around the canister (below the trunnions) and in the annulus. This alternative provides additional shielding during the closure operation.
- Weld the shield lid to the canister shell. Inspect and pressure test the weld. Drain the pool water from the canister. Attach the vacuum system to the drain line, and operate the system to achieve a vacuum. Verify the cavity dryness.
- Reduce the vacuum pressure and backfill with helium to 1 atmosphere.
- Install the vent and drain port covers and weld them to the shield lid. Inspect the port cover welds. Helium leak check the shield lid to canister shell weld and port cover welds.
- Install the structural lid and weld it to the canister shell. Inspect the structural lid weld. Install the hoist rings and attach the canister lifting slings. (Note: Alternative canister lifting system designs may be utilized based on a site-specific analysis and evaluation.) Install the transfer adapter on the concrete cask.
- Lift the transfer cask to the top of the concrete cask and set it on the transfer adapter. (See Figure 1.2-5). Ensure that the bottom door hydraulic actuators are engaged.
- Attach the canister lifting slings or an alternative lifting device to the crane hook and lift the canister off of the transfer cask bottom doors.
- Open the bottom doors of the transfer cask.
- Lower the canister into the concrete cask (see Figure 1.2-6). Remove the canister lifting equipment.
- Remove the transfer cask and transfer adapter.
- Install the shield plug and lid on the concrete cask.
- Move the loaded concrete cask to the storage pad.
- Using the air pad rig set and a towing vehicle, move the concrete cask to its designated location on the storage pad.
- Perform initial surveillance verification of cask heat rejection capability.

During storage operations, the concrete cask operability is verified on a daily basis as specified in the Technical Specifications.

The removal operations are essentially the reverse of these steps, except that weld removal and cool down of the contents is required.

The auxiliary equipment needed to operate the Universal Storage System is described in Section 1.2.1.5. Other items required are miscellaneous hardware, connection hose and fittings, and hand tools typically found at a reactor site.

Figure 1.2-1 Vertical Concrete Cask

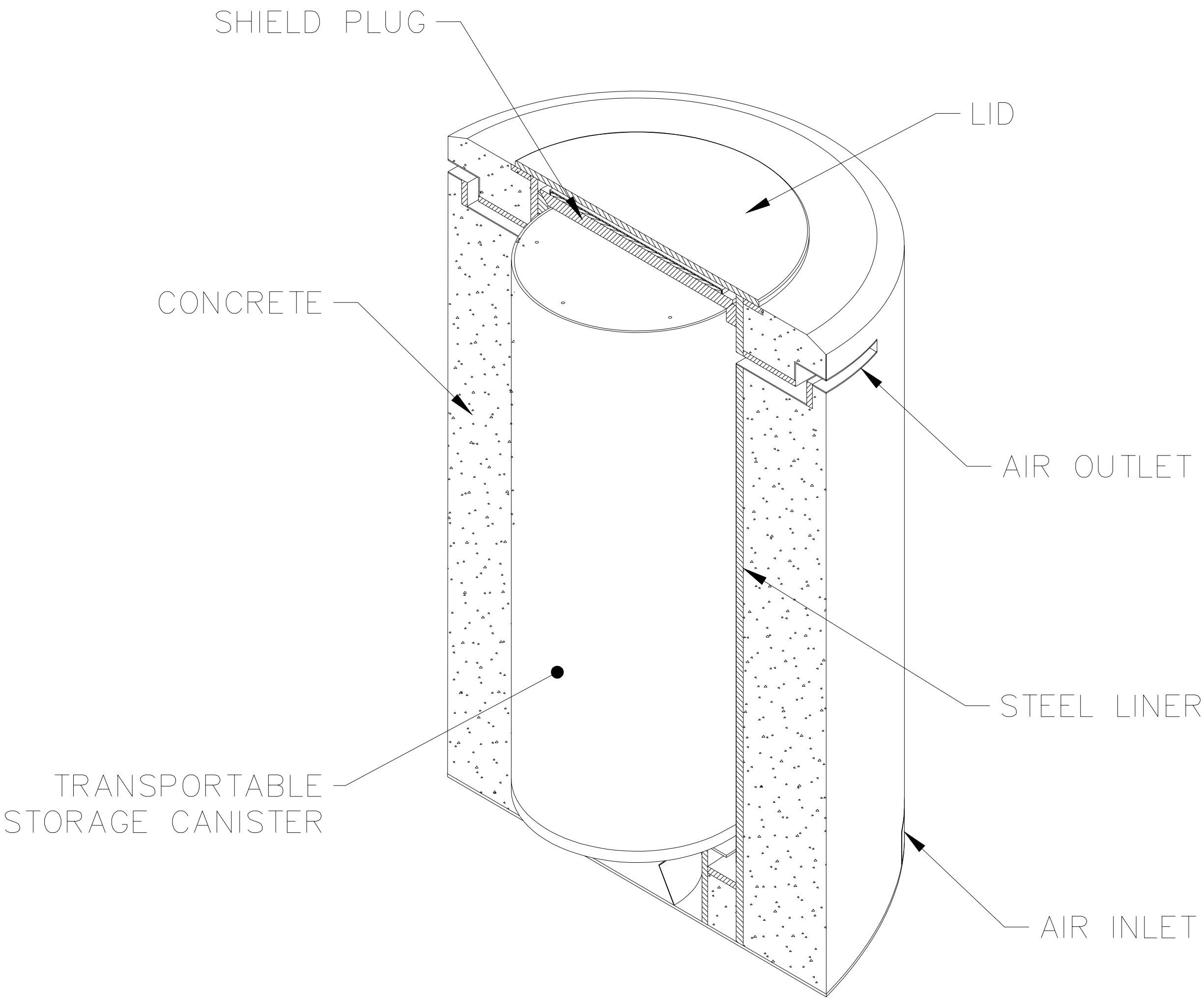
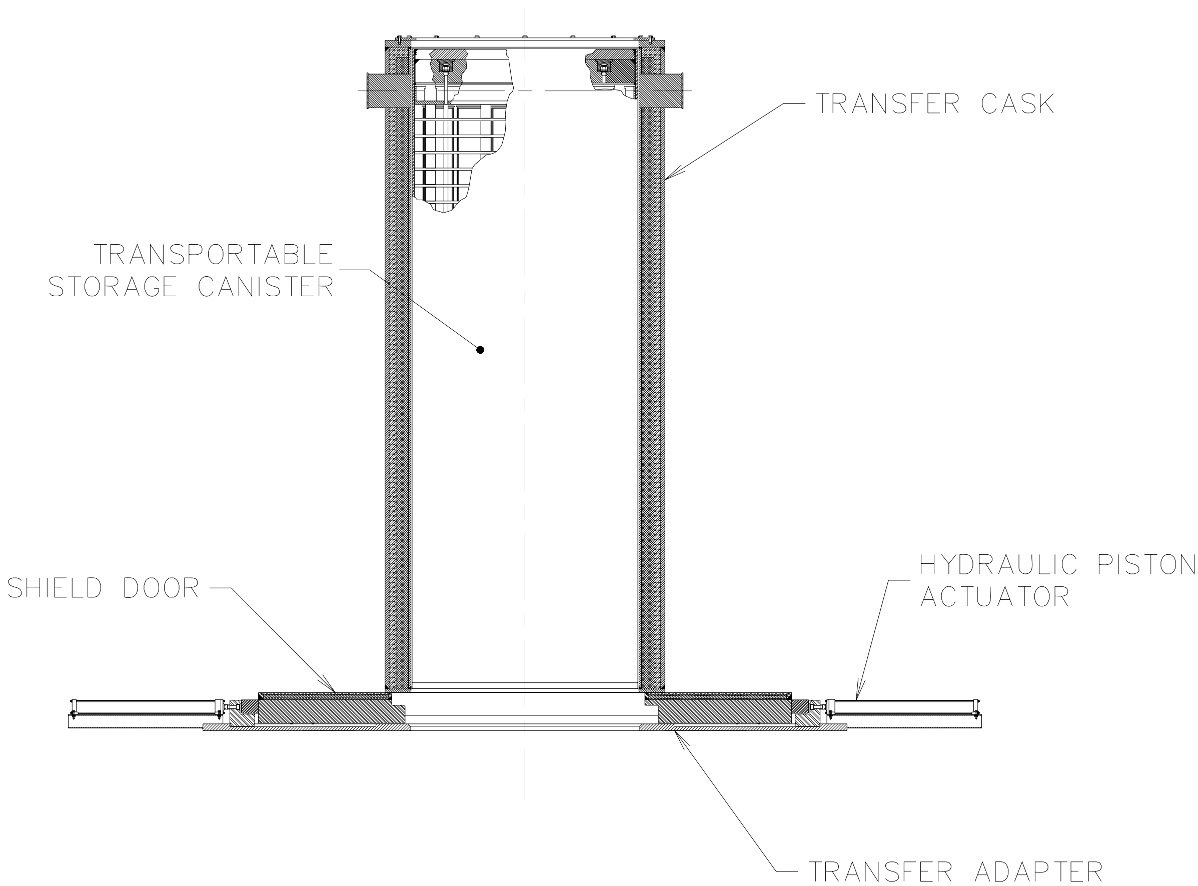


Figure 1.2-2 Transfer Cask



Typical Transfer Cask with Transfer Adapter

Figure 1.2-3 Transport Configuration of the Universal Transport Cask

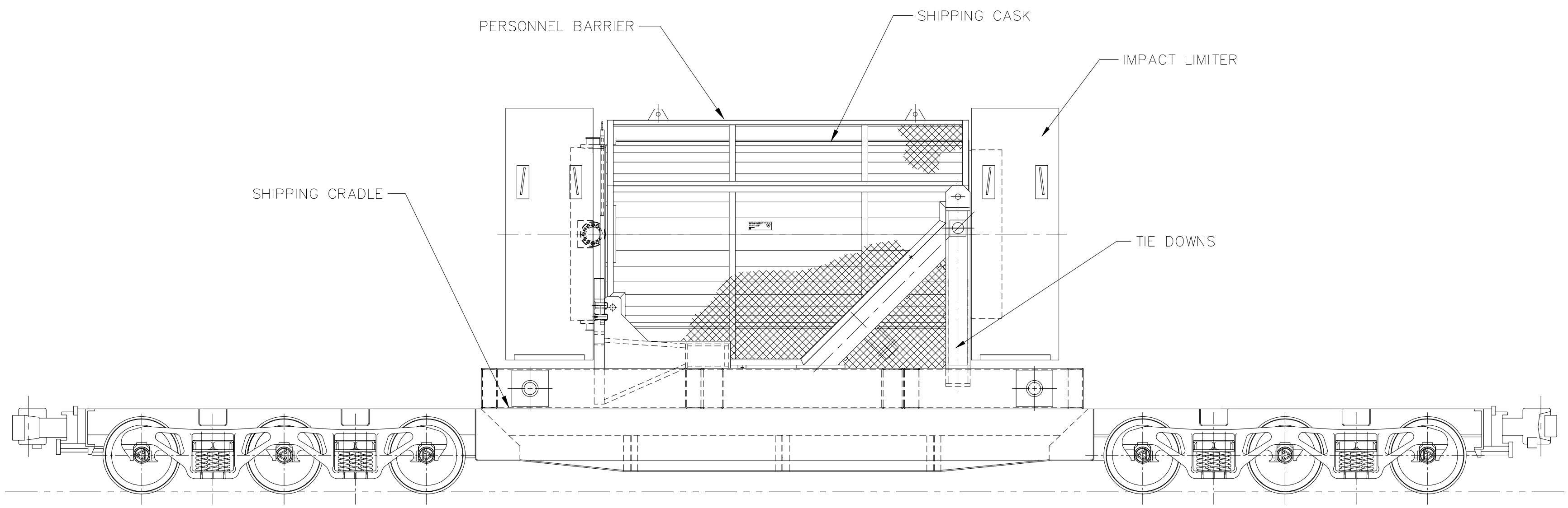


Figure 1.2-4 Transfer Cask and Canister Arrangement

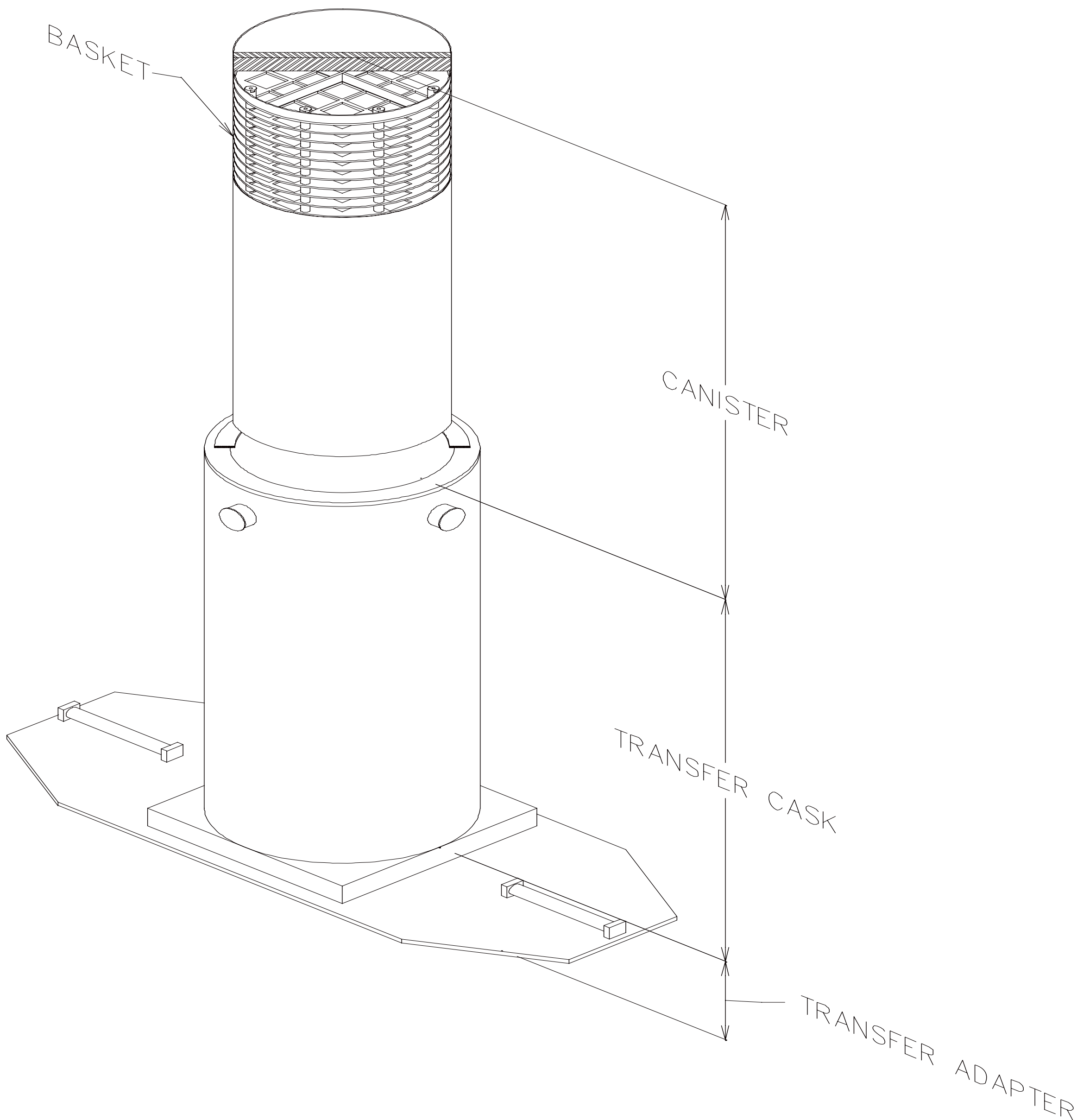


Figure 1.2-5 Vertical Concrete Cask and Transfer Cask Arrangement

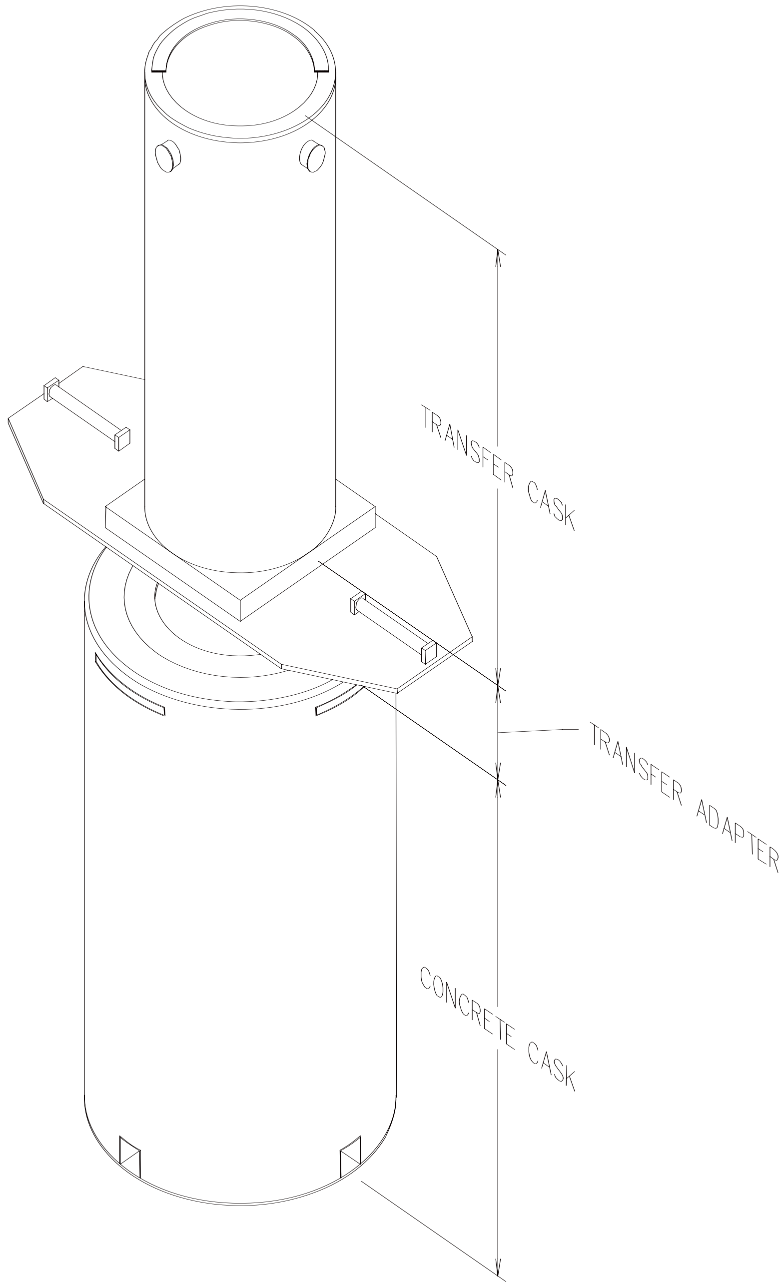


Figure 1.2-6 Major Component Configuration for Loading the Vertical Concrete Cask

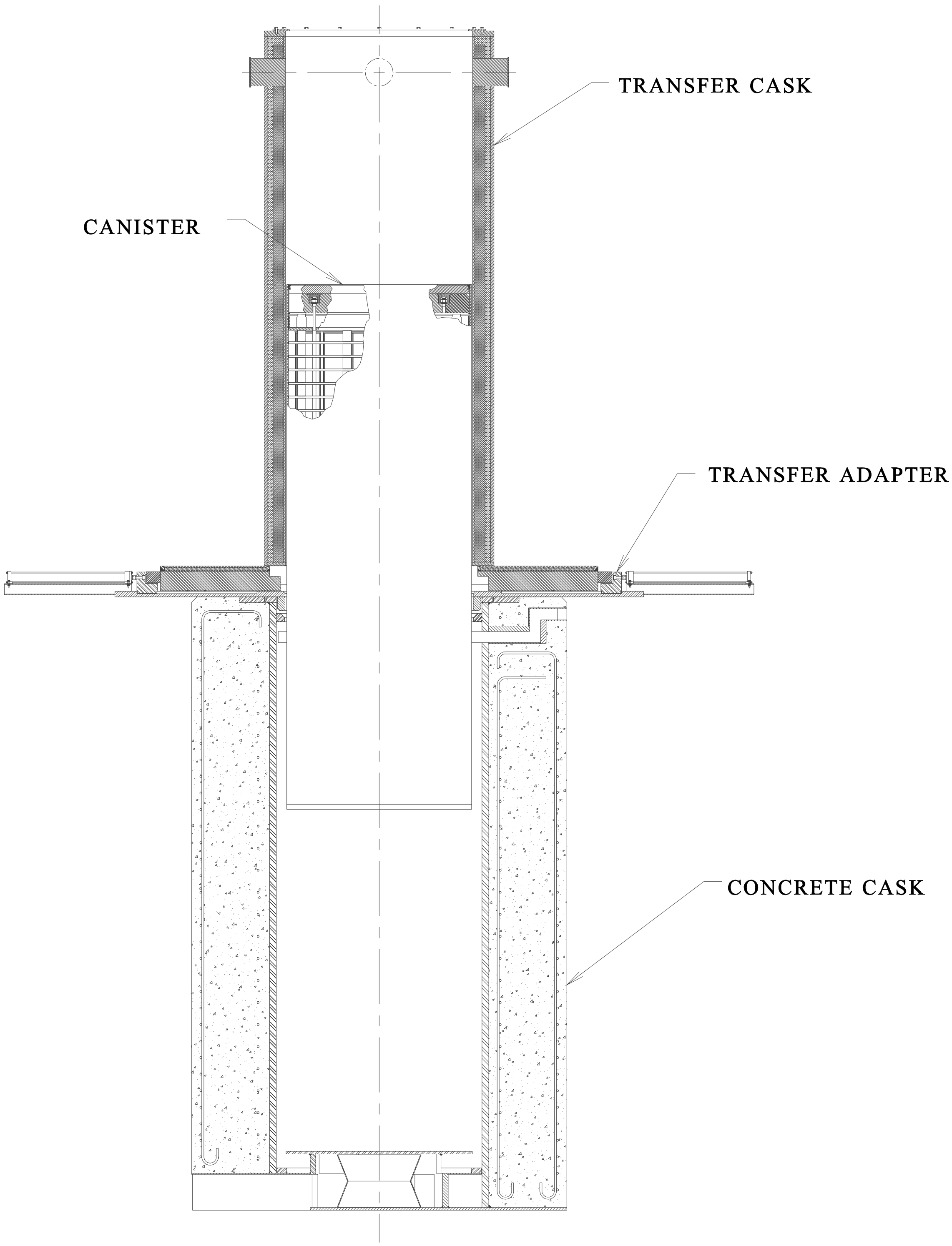


Table 1.2-1 Design Characteristics of the UMS® Universal Storage System

Design Characteristic	Value (in.)	Material
Transportable Storage Canister		
Shell thickness	0.625	Type 304L Stainless Steel
Shell bottom thickness	1.75	Type 304L Stainless Steel
Shield lid thickness	7	Type 304 Stainless Steel
Structural lid thickness	3	Type 304L Stainless Steel
Canister Fuel Basket		
Top weldment PWR thickness	1.25	Type 304 Stainless Steel
Bottom weldment PWR thickness	1.0	Type 304 Stainless Steel
Top and bottom weldment BWR thickness	1.0	Type 304 Stainless Steel
Support disks thickness		
- PWR	0.5	Type 17-4 PH Stainless Steel
- BWR	0.625	SA-533, Type B Class 2 Carbon Steel
Heat transfer disk thickness	0.5	Type 6061-T651 Aluminum Alloy
Fuel tube dimensions		
- PWR (inside)	8.8 × 8.8	Type 304 Stainless Steel Enclosing neutron absorber
- BWR Standard (inside)	5.9 × 5.9	Type 304 Stainless Steel Enclosing neutron absorber
- BWR Over-Sized Fuel (inside)	6.05 × 6.05	Type 304 Stainless Steel Enclosing neutron absorber
Spacer(s) diameter	2.875	Type 304 Stainless Steel
Tie rod diameter		
- PWR	1-5/8	Type 304 Stainless Steel
- BWR	1-5/8	Type 304 Stainless Steel

Table 1.2-1 Design Characteristics of the UMS® Universal Storage System (Continued)

Design Characteristic	Value (in.)	Material
Standard and Advanced Transfer Cask		
Outer Shell	1.25 × 85.3 dia.	ASTM A588 Low Alloy Steel
Inner Shell	0.75 × 67.8 dia.	ASTM A588 Low Alloy Steel
Retaining Ring	0.75 × 77.1 dia.	ASTM A588 Low Alloy Steel
Trunnions	10.0 dia.	A350 LF2 Low Alloy Steel
Bottom Plate	1.0 thick plate	ASTM A588 Low Alloy Steel
Top Plate	2.0 thick plate	ASTM A588 Low Alloy Steel
Shield Doors	9.0 thick	A350 LF2 Low Alloy Steel and NS-4-FR
Door Rails	9.4 × 6.5	A350 LF2 Low Alloy Steel
Gamma Shield	4.0 thick	ASTM B29, Chemical Copper Grade Lead
Neutron Shield	2.75 thick	NS-4-FR, Solid Synthetic Polymer
Transfer Adapter		
Base Plate	2.0 thick plate	ASTM A36 Carbon Steel
Locating Ring	2.75 wide × 73.75	ASTM A36 Carbon Steel

Table 1.2-1 Design Characteristics of the UMS® Universal Storage System (Continued)

Design Characteristic	Value (in.)	Material
Vertical Concrete Cask		
Weldment Structure		
Shell	2.5 thick × 79.50 dia	ASTM A36 Carbon Steel
Top Flange	2.0 thick × 101.40 dia.	ASTM A36 Carbon Steel
Support Ring	2.5 thick × 74.50 dia.	ASTM A36 Carbon Steel
Base Plate	2.0 thick × 67.50 dia.	ASTM A36 Carbon Steel
Concrete Cask		
Concrete Shell	28.3 thick × 136 dia.	Type II Portland Cement
Shield Plug (NS-4-FR)	5.13 × 74.0 dia.	ASTM A36 Carbon Steel and NS-4-FR
Shield Plug (NS-3)	5.63 × 74.0 dia.	ASTM A36 Carbon Steel and NS-3
Cask Lid	1.50 thick × 85.6 dia.	ASTM A36 Carbon Steel
Rebar	Various Lengths	ASTM A615, GR 60, ASTM A615, GR75, and A-706 Carbon Steel

Table 1.2-2 Major Physical Design Parameters of the Transportable Storage Canister

Canister Parameter	Value
Canister Shell	
Outside Diameter (in.)	67.1
Thickness (in.)	0.625
Overall Length (in.)	
Class 1 (PWR)	175.1
Class 2 (PWR)	184.2
Class 3 (PWR)	191.8
Class 4 (BWR)	185.6
Class 5 (BWR)	190.4
Capacity (No. of fuel assemblies)	
Classes 1 – 3 (PWR)	24
Classes 4 – 5 (BWR)	56
Maximum Heat Load (kW)	
PWR	23.0
BWR	23.0
Maximum Long-Term Fuel Cladding Temperature – 5-year cooled fuel (°F [°C])	
Classes 1 – 3 (PWR)	752 (400)
Classes 4 – 5 (BWR)	752 (400)
Internal Atmosphere	Helium

Table 1.2-3 Transportable Storage Canister Fabrication Specification Summary

Materials

- All material shall be in accordance with the referenced drawings and meet the applicable ASME code sections.

Welding

- All welds shall be in accordance with the referenced drawings.
- All filler metals shall be appropriate ASME materials.
- All welders and welding operators shall be qualified in accordance with ASME Section IX [14].
- All welding procedures shall be written and qualified in accordance with ASME Section IX.
- All welds specified to be visually examined shall be examined as specified in ASME Section V, Article 9 with acceptance per ASME Code Section VIII [15], UW-35 and UW-36.
- All welds specified to be dye penetrant examined shall be examined in accordance with the requirements of ASME Section V, Article 6, with acceptance in accordance with ASME Section III, NB-5350.
- All personnel performing examinations shall be qualified in accordance with the NAC International Quality Assurance program and SNT-TC-1A [16].
- All welds specified to be radiographed shall be examined in accordance with the requirements of ASME Code Section V, Article 2, with acceptance per ASME Code Section III, NB 5320.
- All welds specified to be ultrasonically examined shall be examined per ASME Code Section V, Article 5, with acceptance per ASME Code Section III, NB-5330.

Fabrication

- All cutting, welding, and forming shall be in accordance with ASME Code Section III, NB-4000 unless otherwise specified. Code stamping is not required.
- All surfaces shall be cleaned to a surface cleanness classification C or better as defined in ANSI N45.2.1 [17], Section 2.
- All fabrication tolerances shall meet the requirements of the referenced drawings after fabrication.
- Fit-up testing of a "dummy" fuel assembly into each fuel tube and insertion of the completed basket into the canister shell is required. Verification of the basket overall length and diameter is required.

Packaging

- Packaging and shipping shall be in accordance with ANSI N45.2.2 [18].

Quality Assurance

- The canister shall be fabricated under a quality assurance program that meets 10 CFR 72 Subpart G and 10 CFR 71 Subpart H.
- The supplier's quality assurance program must be accepted by the licensee prior to initiation of work.
- A Certificate of Conformance shall be issued by the fabricator stating that the canister meets the specifications and drawings.

Table 1.2-4 Major Physical Design Parameters of the Fuel Basket

Basket Parameter	Value
Basket Assembly Length, in.	
Class 1 (PWR)	162.6
Class 2 (PWR)	171.7
Class 3 (PWR)	179.3
Class 4 (BWR)	173.1
Class 5 (BWR)	177.9
Basket Assembly Diameter, in.	65.5
Number of Support Disks	
Class 1 (PWR)	30
Class 2 (PWR)	32
Class 3 (PWR)	34
Class 4 (BWR)	40
Class 5 (BWR)	41
Number of Heat Transfer Disks	
Class 1 (PWR)	29
Class 2 (PWR)	31
Class 3 (PWR)	33
Class 4 (BWR)	17
Class 5 (BWR)	17
Number of Fuel Tubes	
Classes 1 – 3 (PWR)	24 (with neutron absorber on all four sides)
Classes 4 – 5 (BWR)	56 (42 with neutron absorber on two sides; 11 with neutron absorber on one side; and 3 with no neutron absorber)
Number of Tie Rods	
Classes 1 – 3 (PWR)	8
Classes 4 – 5 (BWR)	6

Table 1.2-5 Major Physical Design Parameters of the Vertical Concrete Cask

Parameter	Value
Height (in.)	
Class 1 (PWR)	209.2
Class 2 (PWR)	218.3
Class 3 (PWR)	225.9
Class 4 (BWR)	219.7
Class 5 (BWR)	224.5
Outside diameter (in.)	136.0
Nominal weight (lbs), Without Canister	
(140 pcf concrete)	223,500
Class 1 (PWR)	232,300
Class 2 (PWR)	239,700
Class 3 (PWR)	233,700
Class 4 (BWR)	238,400
Class 5 (BWR)	
Shielding (side wall)	
Concrete thickness (in.)	28.2
Steel liner thickness (in.)	2.5
Radiation dose rate (mrem/hr):	
Side surface	< 50 (average)
Top surface	<50 (average)
Air inlet/outlet	< 100 (average)
Air flow at design heat load (lb-m)/sec	1
Material of construction	
Concrete	Type II Portland Cement
Reinforcing steel	A615 Grade 60
Steel liner	A36 Carbon Steel
Service life (years)	50
Maximum concrete temperatures for normal operation (°F)	150 (bulk) 200 (local)

Table 1.2-6 Vertical Concrete Cask Construction Specification Summary

Note: The American Society for Testing and Materials (ASTM) approved revisions of the ASTM standards referenced in this table that are in effect at the time of product/test procurement shall be invoked in meeting FSAR requirements.

Materials

- Concrete mix shall be in accordance with the requirements of ACI 318 and ASTM C94 [19].
- Type II Portland Cement, ASTM C150 [20].
- Fine aggregate ASTM C33 [21] or C637 [22].
- Coarse aggregate ASTM C33.
- Admixtures
  - Water Reducing and Superplasticizing ASTM C494 [23].
  - Pozzolanic Admixture (Loss on Ignition 6% or less) ASTM C618 [24].
- Compressive Strength 4000 psi per ACI 318.
- Specified Air Entrainment per ACI 318.
- All steel components shall be of material as specified in the referenced drawings.

Welding

- Visual inspection of all welds shall be performed to the requirements of AWS D1.1, Section 8.6.1 [25].

Construction

- A minimum of two concrete samples for each concrete cask shall be taken in accordance with ASTM C172 [26] and ASTM C31 [27] for the purpose of obtaining concrete slump, density, air entrainment, and compressive strength values. The two samples shall not be taken from the same batch or truckload.
- Test specimens shall be tested in accordance with ASTM C39 [28].
- Formwork shall be in accordance with ACI 318.
- All sidewall formwork shall remain in place in accordance with ACI 318.
- Grade, type, and details of all reinforcing steel shall be in accordance with the referenced drawings.
- Embedded items shall conform to ACI 318 and the referenced drawings.
- The placement of concrete shall be in accordance with ACI 318.
- Surface finish shall be in accordance with ACI 318.

Quality Assurance

The concrete cask shall be constructed under a quality assurance program that meets 10 CFR 72 Subpart G. The quality assurance program must be accepted by NAC International and the licensee prior to initiation of the work.

Table 1.2-7 Major Physical Design Parameters of the Transfer Casks

Parameter	Transfer Cask Configuration	
	Standard	**Advanced**
Inside Diameter (in.)	67.8	67.8
Outside Diameter (in.)	85.3	85.3
Cavity Height (nominal) (in.)		
Class 1	177.3	177.3
Class 2	186.4	186.4
Class 3	194.0	194.0
Class 4	187.8	187.8
Class 5	192.6	192.6
Empty Weight (nominal) (lbs)		
Class 1	112,300	112,300
Class 2	117,300	117,300
Class 3	121,500	121,500
Class 4	118,100	118,100
Class 5	120,700	120,700
Allowable Canister Weight	≤ 88,000	≤ 98,000

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1.3 Universal Storage System Contents

The Universal Storage System is designed to store up to 24 PWR fuel assemblies or up to 56 BWR fuel assemblies. The design basis fuel contents are subject to the limits presented in Section 1.3.1. Site specific contents are described in Section 1.3.2. The site specific contents are either shown to be bounded by the evaluation of the design basis fuel, or are separately evaluated to establish limits which are maintained by administrative controls.

1.3.1 Design Basis Spent Fuel

The Universal Storage System is evaluated based on a set of fuel assembly parameters that establish bounding conditions for the system. The bounding fuel parameters are provided in Table 2.1.1-1 for PWR fuel and in Table 2.1.2-1 for BWR fuel. Fuel assembly designs having parameters bounded by those in Tables 2.1.1-1 and 2.1.2-1 are acceptable for loading. Four different assembly array sizes: 14 × 14, 15 × 15, 16 × 16 and 17 × 17, produced by several different fuel vendors, were evaluated in the development of the PWR design basis spent fuel description. Three different arrays: 7 × 7, 8 × 8 and 9 × 9, produced by several different fuel vendors were evaluated in the development of the BWR design basis spent fuel description.

The Universal Storage System fuel limits are:

1. The characteristics of the PWR and BWR fuel to be stored shall be in accordance with Tables 2.1.1-1 and 2.1.2-1, respectively.

2. The total decay heat of the PWR fuel shall not exceed 23.0 kW.

3. The total decay heat of the BWR fuel shall not exceed 23.0 kW.

4. The maximum initial enrichment shall not exceed 5.0 wt % ^{235}U for PWR and 4.8 wt % ^{235}U for BWR fuel assemblies.

5. The maximum initial enrichment of the PWR fuel is based on a pool/canister water boron content of at least 1,000 parts per million for some fuel parameter combinations. The maximum initial enrichment of the BWR fuel is defined as the maximum initial peak planar-average enrichment. The initial peak planar-average enrichment is the maximum initial peak planar-average enrichment at any height along the axis of the fuel assembly. The initial peak planar-average may be higher than the bundle average enrichment value that appears in fuel design or plant documents. Unenriched fuel assemblies are not evaluated and are not included as a proposed content.

6. The maximum PWR fuel assembly burnup (MWD/MTU) and minimum cooling time (years) shall be as defined by Table 2.1.1-2.

7. The maximum BWR fuel assembly burnup (MWD/MTU) and minimum cooling time (years) shall be as defined by Table 2.1.2-2.

8. Radiation levels shall not exceed the requirements of 10 CFR 72.104 and 10 CFR 72.106.

9. An inert atmosphere shall be maintained within the canister where spent fuel is stored.

10. Stainless steel spacers may be used to axially position PWR fuel assemblies that are shorter than the canister cavity length to facilitate handling.

11. Flow mixers (thimble plugs), in-core instrument thimbles, burnable poison rods or solid stainless steel rods may be placed in PWR guide tubes as long as the maximum fuel assembly weights listed in Table 2.1.1-1 are not exceeded and no credit for soluble boron is taken.

1.3.2 Site Specific Spent Fuel

This section describes fuel assembly characteristics and configurations, which are unique to specific reactor sites. These site specific content configurations result from conditions that occurred during reactor operations, participation in research and development programs (testing programs intended to improve reactor operations), and from the placement of control components or other items within the fuel assembly.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation of the configuration.

Unless specifically excepted, site specific fuel must meet all of the conditions specified for the design basis fuel presented in Section 1.3.1 above. Site specific fuels are also described in Section 2.1.3.

1.3.2.1 Maine Yankee Site Specific Spent Fuel

The configurations of Maine Yankee site specific fuel assemblies that have been evaluated and found to be acceptable contents are:

- Fuel assemblies with up to 176 fuel rods removed from the assembly lattice.
- Fuel assemblies with fuel rods replaced with stainless steel rods, solid zirconium alloy rods or fuel rods enriched to 1.95 wt % ^{235}U.
- Fuel assemblies with burnable poison rods replaced with hollow zirconium alloy tubes.
- Fuel assemblies that are variably enriched with a maximum fuel rod enrichment of 4.21 wt % ^{235}U and that also have a maximum planar average enrichment of 3.99 wt % ^{235}U.
- Fuel assemblies with variable enrichment and/or annular axial blankets.
- Fuel assemblies with a control element assembly inserted.
- Fuel assemblies with an instrument thimble inserted in the center guide tube.
- Fuel assemblies with up to two fuel rods inserted in any or all of the guide tubes.
- Fuel assemblies with inserted nonfuel components, including start-up sources.
- Consolidated fuel.
- Fuel assemblies having up to 100% of the rods damaged in each assembly.
- Fuel assemblies having a burnup of greater than 45,000 MWD/MTU but less than 50,000 MWD/MTU.

These site specific fuel configurations are evaluated against the limits established for the UMS® Storage System based on the design basis fuel. The site specific fuel is either shown to be bounded by the evaluation of the design basis fuel or is separately evaluated to establish limits which are maintained by preferential loading administrative controls. Where applicable to specific configurations, the preferential loading controls are described in Section 2.1.3.1.1. The preferential loading controls take advantage of design features of the UMS® Storage System to allow the loading of fuel configurations that may have higher burnup or additional hardware or fuel source material that is not specifically considered in the design basis fuel evaluation.

The Transportable Storage Canister loading procedures will indicate that the loading of a fuel configuration with removed fuel or poison rods, damaged or consolidated fuel in a Maine Yankee fuel can, or fuel with burnup greater than 45,000, but less than 50,000, MWD/MTU is administratively controlled in accordance with Section 2.1.3.1 and Table 2.1.3.1-1. As shown in the table, only one consolidated fuel lattice is loaded in any single canister. Preferential loading positions in the canister basket are shown in Figure 2.1.3.1-1.

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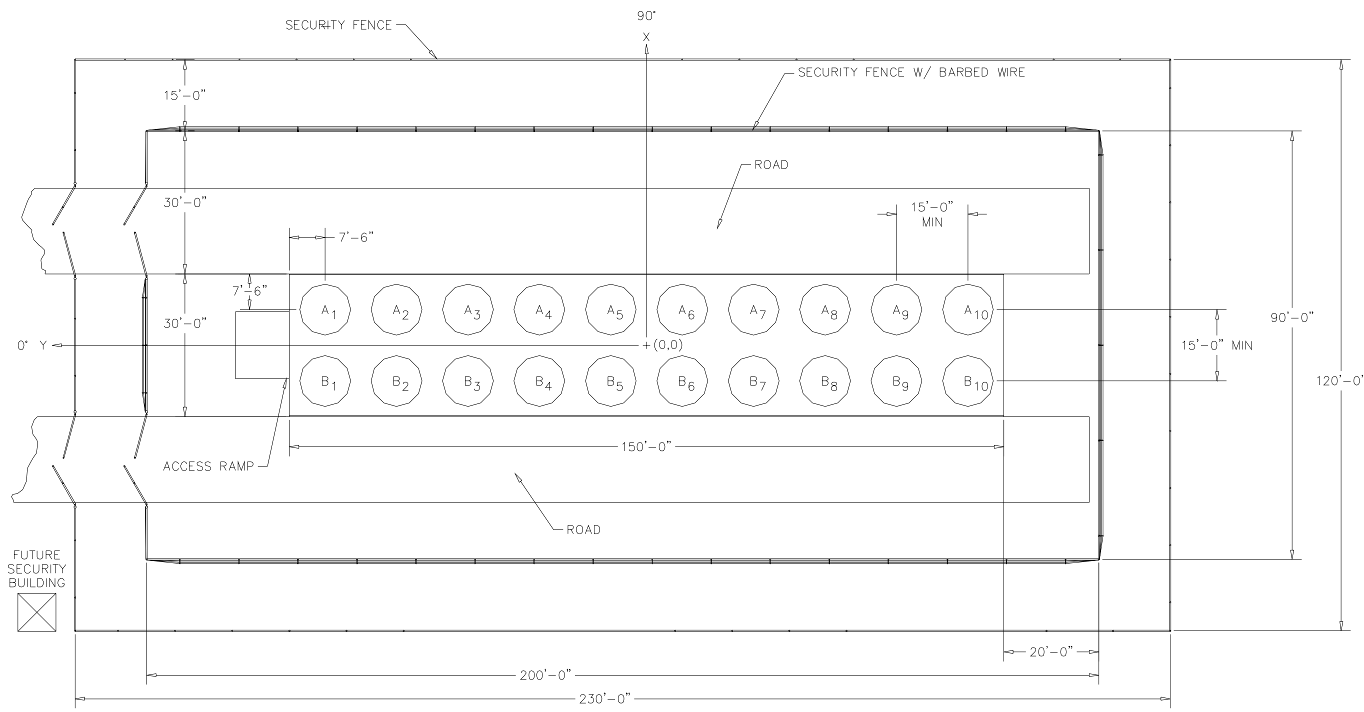
1.4 Generic Vertical Concrete Cask Arrays

A typical ISFSI storage pad layout for 20 concrete casks is provided in Figure 1.4-1. As shown in this figure, roads parallel the sides of the pad to facilitate transfer of the concrete cask from the transporter to the designated storage position on the pad. Loaded concrete casks are placed in the vertical position on the pad in a linear array. Array sizes could accommodate from 1 to more than 200 casks. Figure 1.4-1 shows typical spacing and representative site dimensions. Actual spacing and dimensions are dependent on the general site layout, access roads, site boundaries, and transfer equipment selection, but must conform to the spacing or dimension requirements established in Section 8.1.3 of the Operating Procedures.

The reinforced concrete foundation is capable of sustaining the transient loads from the air pad and the general loads of the stored casks. If necessary, the pad can be constructed in phases to specifically meet utility-required expansions.

Figure 1.4-1 Typical ISFSI Storage Pad Layout



1.5 UMS® Universal Storage System Compliance with NUREG-1536

The design of the UMS® Universal Storage System meets the regulatory requirements and acceptance criteria specified in NUREG-1536 as shown in Table 1.5-1. This table provides a compliance matrix that shows the specified regulatory requirements and acceptance criteria of NUREG-1536, and the location in the UMS® Universal Storage System Safety Analysis Report where each of the requirements or criteria are addressed.

Table 1.5-1 NUREG-1536 Compliance Matrix

Chapter 1 – General Description			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
1. General Description and Operational Features	The application must present a general description and discussion of the DCSS, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. [10 CFR Part 72.24(b)]	The applicant should provide a broad overview and a general, nonproprietary description (including illustrations) of the DCSS, clearly identifying the functions of all components and providing a list of those components classified by the applicant as being "important to safety."	A general description of the system is provided in Section 1.2. Quality category classifications are provided in Table 2.3-1.
2. Drawings	Structures, systems, and components (SSCs) important to safety must be described in sufficient detail to enable reviewers to evaluate their effectiveness. [10 CFR Part 72.24(c)(3)]	The applicant should provide non-proprietary drawings of the storage system, of sufficient detail, that an interested party can ascertain its major design features and general operations.	Drawings of the system are provided in Section 1.8.
3. DCSS Contents	The applicant must provide specifications for the contents expected to be stored in the DCSS (normally spent fuel). These specifications may include, but not be limited to, type of spent fuel (i.e., boiling-water reactor (BWR), pressurized-water reactor (PWR), or both), maximum allowable enrichment of the fuel before any irradiation, burnup (i.e., megawatt-days/metric ton Uranium), minimum acceptable cooling time of the spent fuel before storage in the DCSS (aged at least 1 year), maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (i.e., undamaged or damaged assembly or consolidated fuel rods), weight and nature of nonspent fuel contents, and inert atmosphere requirements. [10 CFR Part 72.2(a)(1) and 10 CFR Part 72.236(a)]	The applicant should characterize the fuel and other radioactive wastes expected to be stored in the DCSS. If the potential exists that the DCSS will be used to store degraded fuel, the SAR should include a discussion of how the sub-criticality and retrievability requirements will be maintained.	A description of the contents to be stored is presented in Section 2.1, and Tables 2.1.1-1 and 2.1.2-1.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 1 – General Description			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
4. Qualifications of the Applicant	The application must include the technical qualifications of the applicant to engage in the proposed activities. Qualifications should include training and experience. [10 CFR Part 72.24(j), 10 CFR Part 72.28(a)]	The reviewer should ensure that the applicant has clearly identified the roles and responsibilities that the DCSS designer, vendor, and other agents, such as potential licensees, fabricators, and contractors will have in the review process. Verify that the applicant has provided clear evidence demonstrating that they are qualified to engage in the proposed activities. In addition, verify that the applicant has delineated the responsibilities for all those who will be involved in the construction and operation of the DCSS if known. The reviewer should ensure that the applicant has specifically defined activities which they will not perform.	Applicant qualifications are discussed in Section 1.6.
5. Quality Assurance	The safety analysis report (SAR) must include a description of the applicant's quality assurance (QA) program, with reference to implementing procedures. This description must satisfy the requirements of 10 CFR Part 72, Subpart G, and must be applied to DCSS SSC that are important to safety throughout all design, fabrication, construction, testing, operations, modifications and decommissioning activities. These implementing procedures need not be explicitly included in the application. [10 CFR Part 72.24(n)]	Verify that the applicant has described the proposed QA program, citing the applicable implementing procedures. This description should satisfy all requirements of 10 CFR Part 72, Subpart G, that apply to the design, fabrication, construction, testing, operation, modification, and decommissioning of the DCSS SSCs that are important to safety.	Applicant QA program is presented in Chapter 13.
6. Consideration of 10 CFR Part 71 Requirements Regarding Transportation	If the DCSS under consideration has previously been reviewed and certified for use as a transportation cask, the application must include a copy of the Certificate of Compliance issued for the DCSS under 10 CFR Part 71, including drawings and other documents referenced in the certificate. [10 CFR 72.230(b)]	If the DCSS under review has previously been evaluated for use as a transportation cask, the submittal should include the Part 71 Certificate of Compliance and associated documents.	The transport application for issuance of a Part 71 Certificate of Compliance is discussed in Section 1.0.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
1. Structures, Systems, and Components (SSC) Important to Safety	The applicant must identify all SSC that are important to safety, and describe the relationships of non-important to safety SSC on overall DCSS performance. [10 CFR 72.24(c)(3) and 72.44(d)] The applicant must specify the design bases and criteria all SSC that are important to safety. [10 CFR 72.24(c)(1), 72.24(c)(2), 72.120(a), and 72.236(b)]	The applicant should discuss the general configuration of the DCSS, and should provide an overview of specific components and their intended functions. In addition, the applicant should identify those components deemed to be important to safety, and should address the safety functions of those components in terms of how they meet the general design criteria and regulatory requirements discussed above. Additional information concerning specific functional requirements for individual DCSS components are addressed in the subsequent chapters of this SRP.	The quality category classification of system components are described in Table 2.3-1. The design bases and criteria for the system are specified in Table 2-1. Detailed design criteria are presented in Section 2.2.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
2. Design Bases for Structures, Systems, and Components Important to Safety **a. Spent Fuel Specifications**	The applicant must provide the range of specifications for the spent fuel to be stored in the DCSS. These specifications should include, but are not to be limited to: the type of spent fuel (i.e., boiling-water reactor (BWR), pressurized-water reactor (PWR), or both); content, weight, dimensions and configurations of the fuel; maximum allowable enrichment of the fuel before any irradiation; maximum fuel burnup (i.e., megawatt-days/mtu); minimum acceptable cooling time of the spent fuel before storage in the DCSS (aged at least 1 year); maximum heat load to be dissipated; maximum spent fuel elements to be loaded; spent fuel condition (i.e., undamaged or damaged assembly or consolidated fuel rods); and any inerting atmosphere requirements. [10 CFR 72.2(a)(1) and 72.236(a)]	Detailed descriptions of each of the items listed below are generally found in specific sections of the SAR; however, a brief description of these areas, including a summary of the analytical techniques used in the design process, should also be captured in Section 2 of the SAR. This description gives reviewers a perspective on how specific DCSS components interact to meet the regulatory requirements of 10 CFR Part 72. This discussion should be non-proprietary since it may be used to familiarize interested persons with the design features and bounding conditions of operation of a given DCSS. The applicant should define the range and types of spent fuel or other radioactive materials that the DCSS is designed to store. In addition, these specifications should include, but are not to be limited to, the type of spent fuel (i.e., boiling-water reactor (BWR), pressurized-water reactor (PWR), or both), weights of the stored materials, dimensions & configurations of the fuel, maximum allowable enrichment of the fuel before any irradiation, burnup (i.e., megawatt-days/mtu), minimum acceptable cooling time of the spent fuel before storage in the DCSS (aged at least 1 year), maximum heat designed to be dissipated, maximum number of spent fuel elements, condition of the spent fuel (i.e., undamaged or damaged assembly or consolidated fuel rods), inerting atmosphere requirements, and the maximum amount of fuel permitted for storage in the DCSS. For DCSSs that will be used to store radioactive materials other than spent fuel, that is, activated components associated with a spent fuel assembly (e.g., control rods, BWR fuel channels), the applicant should specify the types and amounts of radionuclides, heat generation and the relevant source strengths and radiation energy spectra permitted for storage in the DCSS.	Specifications of the spent fuel contents are provided in Section 2.1. Specific physical parameters of the design basis fuel are listed in Tables 2.1.1-1 and 2.1.2-1.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Bases for Structures, Systems, and Components Important to Safety **b. External Conditions**	The design bases for SSC important to safety must reflect an appropriate consideration of environmental conditions associated with normal operations, as well as design considerations for both normal and accident conditions and the effects of natural phenomena events. [10 CFR 72.122(b)]	The SAR should define the bounding conditions under which the DCSS is expected to operate. Such conditions include both normal and off-normal environmental conditions, as well as accident conditions. In addition, the applicant should consider the effects of natural events, such as tornadoes, earthquakes, floods, and lightning strikes. The effects of such events are addressed in individual chapters of the SRP (e.g., the effects of an earthquake on the DCSS structural components are addressed in Chapter 3, "Structural Analysis").	The environmental conditions and natural phenomena considered as design bases are described in Section 2.2.
3. Design Criteria for Safety Protection Systems **a. General**	The DCSS must be designed to safely store the spent fuel for a minimum of 20 years and to permit maintenance as required. [10 CFR 72.236(g)] SSC important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed. [10 CFR 72.122(a)] The applicant must identify all codes and standards applicable to the SSC. [10 CFR 72.24(c)(4)]	The SAR should define an expected lifetime for the cask design. The staff has accepted a minimum of 20 years as consistent with the licensing period. The applicant should also briefly describe the proposed quality assurance (QA) program, and applicable industry codes and standards, that will be applied to the design, fabrication, construction, and operation of the DCSS. In establishing normal and off-normal conditions applicable to the design criteria for DCSS designs, applicants should account for actual facility operating conditions. Design considerations should therefore reflect normal operational ranges, including any seasonal variations or effects.	The codes and standards of design and construction of the system and the design life are specified in Table 2-1, and discussed in Section 3.1.2.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **b. Structural**	SSC that are important to safety must be designed to accommodate the combined loads of normal operations, accidents, and natural phenomena events with an adequate margin of safety. [10 CFR 72.24(c)(3), 72.122(b), and 72.122(c)]	The SAR should define how the DCSS structural components are designed to accommodate combined normal, off-normal, and accident loads, while protecting the DCSS contents from significant structural degradation, criticality, and loss of confinement, while preserving retrievability. This discussion is generally a summary of the analytical techniques and calculational results from the detailed analysis discussed in SAR Section 3 and should be presented in a non-proprietary forum.	A discussion of the structural design criteria are presented in Section 2.2. Combined loadings are addressed specifically in Section 2.2.5, and in Tables 2.2-1 and 2.2-2.
	The design-basis earthquake must be equivalent to or exceed the safe shutdown earthquake of a nuclear plant at sites evaluated under 10 CFR Part 100. [10 CFR 72.102(f)]		The design-basis earthquake is specified in Section 2.2.3 in accordance with 10 CFR 72.102 criteria.
	The DCSS must maintain confinement of radioactive material within the limits of 10 CFR Part 72 and Part 20, under normal, off-normal, and credible accident conditions. [10 CFR 72.236(l)]		Analyses show that the system maintains adequate margins of safety during normal (Section 3.4.4.1), off-normal (Section 11.1) and accident condition (Section 11.2) events, therefore, confinement of the radioactive material is assured.
	The DCSS must be designed and fabricated so that the spent fuel is maintained in a subcritical condition all under all credible normal, off-normal, and accident conditions. [10 CFR 72.124(a) and 72.236(c)]		As the system maintains adequate structural margins of safety during normal, off-normal and accident condition events, criticality control is assured based on the analyses presented in Chapter 6.
	The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures, or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. [10 CFR 72.122(h)(1)]		The maximum allowable cladding temperatures are specified in Tables 2-1 and 4.1-3. The temperature results for the fuel cladding listed in Tables 4.1-4 and 4.1-5 show that the allowable cladding temperatures are not exceeded. Therefore, the fuel cladding is protected against degradation during storage.
	Storage systems must be designed to allow ready retrieval of spent fuel waste for further processing or disposal. [10 CFR 72.122(l)]		As described in Section 1.2, the system is designed to be readily retrievable and transported off site as necessary for further processing or disposal.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **c. Thermal**	Each spent fuel storage or handling system must be designed with a heat removal capability having testability and reliability consistent with its importance to safety. [10 CFR 72.128(a)(4)]	The applicant should provide a general discussion of the proposed heat removal mechanisms, including the reliability and verifiability of such mechanisms and any associated limitations. All heat removal mechanisms should be passive and independent of intervening actions under normal and off-normal conditions.	The verification of the heat removal capability of the storage system is described in Section 2.3.3.2. The reliability of the heat removal system is demonstrated in Chapter 4. Routine surveillance of the concrete cask is described in Section 2.3.3.2 to verify continuing operability.
	The DCSS must be designed to provide adequate heat removal capacity without active cooling systems. [10 CFR 72.236(f)]		As shown by the results of the thermal evaluation of the system reported in Tables 4.1-4 and 4.1-5, the storage system provides adequate heat removal through the passive cooling design features described in Section 1.2.1.3.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **d. Shielding/Confinement/ Radiation Protection**	The proposed DCSS design must provide radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 72.106. [10 CFR 72.126(a), 72.128(a)(2), 72.128(a)(3), and 72.236(d)]	The applicant should describe those features of the cask that protect occupational workers and members of the public against direct radiation dosages and releases of radioactive material, and minimize the dose after any off-normal or accident conditions.	The confinement design features are described in Section 2.3.2.1, while the radiation shielding design features are described in Section 2.3.5.
	During normal operations and other anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to (1) planned discharges to the general environment of radioactive materials except radon and its decay products, (2) direct radiation from operations of the ISFSI or monitored retrievable storage (MRS), and (3) any other radiation from uranium fuel cycle operations within the region. [10 CFR 72.24(d), 72.104(a), and 72.236(d)]		Section 10.4 presents the necessary minimum site boundary distances from an array of loaded storage systems to meet the controlled area dose limits.
	Any individual located at or beyond the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design-basis accident. The minimum distance from the spent fuel handling and storage facilities to the nearest boundary of the controlled area shall be 100 meters. [10 CFR 72.24(d), 72.24(m), 72.106(b), and 36(d)]		As stated in Section 10.2.2, there is no postulated accident condition that would result in a release of radioactive materials. Therefore, the accident dose limit is met.
	The DCSS must be designed to provide redundant sealing of confinement systems. [10 CFR 72.236(e)]		The redundant sealing features of the confinement system are presented in Section 2.3.2.1.
	Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. [10 CFR 72.122(h)(4) and 72.128(a)(1)]		As described in Section 2.3.1, the system is fully welded and can operate through all postulated normal, off-normal, and accident events while confining of the stored radioactive material. Therefore, continuous monitoring is not required.
	The DCSS design must include inspections, instrumentation and/or control (I&C) systems to monitor the SSC that are important to safety over anticipated ranges for normal and off-normal operation. In addition, the applicant must identify those control systems that must remain operational under accident conditions. [10 CFR 72.122(i)]		Appendix A, Section A 3.1.6 and Section A 5.4 specify the surveillance requirements for the system under normal conditions and after an accident, respectively. These activities are specified to ensure that the system is operated within its design parameters at all times.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **e. Criticality**	Spent fuel transfer and storage systems must be designed to remain subcritical under all credible conditions. [10 CFR 72.124(a) and 72.236(c)] When practicable, the DCSS must be designed on the basis of favorable geometry, permanently fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design shall allow for positive means to verify their continued efficacy. [10 CFR 72.124(b)]	The SAR should address the mechanisms and design features that enable the DCSS to maintain spent fuel in a subcritical condition under normal, off-normal, and accident conditions.	The criticality safety design criteria for the system are presented in Section 2.3.4.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **f. Operating Procedures**	The DCSS must be compatible with wet or dry spent fuel loading and unloading procedures. [10 CFR 72.236(h)] Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)] The DCSS must be designed to minimize the quantity of radioactive waste generated. [10 CFR 72.24(f) and 72.128(a)(5)] The applicant must describe equipment and processes proposed to maintain control of radioactive effluents. [10 CFR 72.24(l)(2)] To the extent practicable, the DCSS must be designed to facilitate decontamination. [10 CFR 72.236(I)] The applicant must establish operational restrictions to meet the limits defined in 10 CFR Part 20 and to ensure that radioactive materials in effluents and direct radiation levels associated with ISFSI operations will remain as low as is reasonably achievable (ALARA). [10 CFR 72.24(e) and 72.104(b)]	The applicant should provide potential licensees with guidance regarding the content of normal, off-normal, and accident response procedures. Cautions regarding both loading, unloading, and other important procedures should be mentioned here. Applicants may choose to provide model procedures to be used as an aid for preparing detailed site-specific procedures.	The operating procedures for the system are presented in Chapter 8, and include procedures for wet loading and unloading operations. Discussion is provided on the development of operating procedures for dry cask handling facilities. The procedures include methods for retrieving the spent fuel after storage for off-site transport or for return to the spent fuel pool. The decommissioning considerations of the system are described in Section 2.4. Operation of the system generates no radioactive waste, other than a limited amount of protective clothing and tools used during loading operations that could be easily disposed or decontaminated. The radiation protection design features of the system are presented in Section 2.3.5. Operating procedures for the system include provisions for controlling potential effluents from the system. The canister is designed to facilitate decontamination, as described in Section 2.3.5.3. Fuel assembly specifications are provided in Appendix B, Section B 2.1.1 to ensure that doses from direct radiation are maintained ALARA. There are no radioactive effluents from the canister or concrete cask in storage operations.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **g. Acceptance Tests and Maintenance**	The DCSS design must permit testing and maintenance as required. [10 CFR 72.236(g)] SSC that are important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function to be performed. [10 CFR 72.24(c), 72.122(a), 72.122(f), and 72.128(a)(1)]	The applicant should identify the general commitments and industry codes and standards used to derive acceptance, maintenance, and periodic surveillance tests used to verify the capability of DCSS components to perform their designated functions. In addition, the applicant should discuss the methods used to assess the need for such tests with regard to specific components.	The acceptance tests and maintenance program for the system are provided in Chapter 9, including the associated commitments to industry standards and/or NRC regulations.
3. Design Criteria for Safety Protection Systems **g. Acceptance Tests and Maintenance**	The DCSS design must permit testing and maintenance as required. [10 CFR 72.236(g)] SSC that are important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function to be performed. [10 CFR 72.24(c), 72.122(a), 72.122(f), and 72.128(a)(1)]	The applicant should identify the general commitments and industry codes and standards used to derive acceptance, maintenance, and periodic surveillance tests used to verify the capability of DCSS components to perform their designated functions. In addition, the applicant should discuss the methods used to assess the need for such tests with regard to specific components.	The acceptance tests and maintenance program for the system are provided in Chapter 9, including the associated commitments to industry standards and/or NRC regulations.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **h. Decommissioning**	The DCSS must be compatible with wet or dry unloading facilities. [10 CFR 72.236(h)] The DCSS must be designed for decommissioning. Provisions must be made to facilitate decontamination of structures and equipment and to minimize the quantity of radioactive wastes, contaminated equipment, and contaminated materials at the time the ISFSI is permanently decommissioned. [10 CFR 72.24(f), 72.130, and 72.236(I)] The applicant must provide information concerning the proposed practices and procedures for decontaminating the site and facilities and for disposing of residual radioactive materials after all spent fuel has been removed. Such information must provide reasonable assurance that decontamination and decommissioning will adequately protect the health and safety of the public. [10 CFR 72.24(q) and 72.30(a)]	Casks should be designed for ease of decontamination and eventual decommissioning. The applicant should describe the features of the design that support these two activities.	Decommissioning of the system is discussed in Section 2.4.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 3 – Structural Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Structures, Systems, and Components Important to Safety	Structures, systems, and components (SSC) important to safety must meet the regulatory requirements established in 10 CFR 72.24(c)(3) and (4), as well as 10 CFR 72.122(a), (b), and (c).	
	10 CFR 72.24(c)(3) Contents of Application: Descriptions of Components Important to Safety	Component descriptions are provided in Section 1.2. Description of the structural design is provided in Section 3.1.1.
	10 CFR 72.24(c)(4) Contents of Application: Applicable Codes and Standards	The applicable codes and standards are specified in Table 2-1 and Sections 3.1.1 and 3.1.2.
	10 CFR 72.122(a) Overall Requirements: Quality Standards	The quality standards of the system are provided in Table 2.3-1.
	10 CFR 72.122(b) Overall Requirements: Protection Against Environmental Conditions and natural Phenomena	The system is evaluated structurally for normal operating loads in Sections 3.4.4 and 3.4.5. Off-normal and accident loads are evaluated in Sections 11.1 and 11.2, respectively.
	10 CFR 72.122(c) Overall Requirements: Protection Against Fires and Explosions	The system is evaluated for fire and explosive loadings in Section 11.2.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 3 – Structural Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
2. Radiation, Shielding, Confinement, and Subcriticality	Radiation shielding, confinement, and subcriticality must meet the regulatory requirements defined in 10 CFR 72.24(d); 10 CFR 72.124(a); and 10 CFR 72.236(c), (d), and (l).	
	10 CFR 72.24(d) Contents of Application: Margins of Safety / Mitigation of Accident Consequences	The margins of safety for normal conditions are listed in Sections 3.4.4.1 and 3.4.4.2. Off-normal and accident condition margins of safety are presented in Sections 11.1 and 11.2, respectively. Adequate safety margins are maintained for all events, ensuring the mitigation of accident consequences, and maintaining the shielding, confinement, and criticality analyses presented in the SAR.
	10 CFR 72.124(a) Criteria for Nuclear Criticality Safety: Design for Criticality Safety	The nuclear criticality safety design of the system is discussed in Sections 2.3.4 and 6.1.
	10 CFR 72.236(c) Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Subcritical Configuration	Subcriticality of the system is demonstrated in Section 6.4.3.
	10 CFR 72.236(d) Specific Requirements for Spent Fuel Storage Cask Approval: Radiation Protection	Radiation protection of the system is demonstrated in Sections 5.4, 10.3 and 10.4.
	10 CFR 72.236(l) Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Confinement	Confinement of the spent fuel is discussed in Sections 7.2 and 7.3.
3. Removal of Spent Fuel	As stated in 10 CFR 72.122(f) and (h)(l), the storage system design must allow ready retrieval of spent fuel without posing operational safety problems.	The system is not adversely affected by normal, off-normal, or accident condition events as demonstrated in Sections 3.4.4.1, 3.4.4.2, 11.1 and 11.2. Operating procedures for removing spent fuel from the system are presented in Sections 8.2 and 8.3.
4. Design Basis Earthquake	As stated in 10 CFR 72.102(f), the design-basis earthquake (DBE) must be equal to or greater than the safe-shutdown earthquake (SSE) of nuclear plant sites previously evaluated under 10 CFR Part 100 or, in the case of sites licensed before the implementation of 10 CFR Part 100, developed under Topic III-2 of the Systematic Evaluation Program (SEP).	As described in Section 2.2.3.1, the system is designed for a seismic event that meets the regulatory requirements.
5. Minimum Lifetime	As stated in 10 CFR 72.24(c) and 10 CFR 72.236(g), the analysis and evaluation of the structural design and performance must demonstrate that the cask system will allow storage of spent fuel for a minimum of 20 years with an adequate margin of safety.	Section 1.1 and Table 2-1 specify a 50-year design life for the system.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 3 – Structural Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
6. Reinforced Concrete Structures	Reinforced concrete structures may have a role in shielding, form ventilation passages and weather enclosures, and providing protection against natural phenomena and accidents. The pertinent regulations include 10 CFR 72.24(c) and 10 CFR 72.182(b) and (c).	
	10 CFR 72.24(c) Contents of Application: Design Criteria, Design Bases, Component Descriptions, Codes and Standards	A general description of the Vertical Concrete Cask (VCC) is provided in Section 1.2.1.3. The design criteria for the VCC is presented in Table 2-1. The design bases considered in the structural evaluation of the VCC are presented in Section 2.2.5.1.
	10 CFR 72.182(b) Design for Physical Protection: Design Bases / Design Criteria	This requirement is applicable to the ISFSI, not the storage system.
	10 CFR 72.182(c) Design for Physical Protection: Security System Description	This requirement is applicable to the ISFSI, not the storage system.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Confinement Cask **a. Steel Confinement Cask**	The structural design, fabrication, and testing of the confinement system and its redundant sealing system should comply with an acceptable code or standard, such as Section III of the Boiler and Pressure Vessel Code (B&PV) promulgated by the American Society of Mechanical Engineers (ASME). (The NRC has accepted use of either Subsection NB or Subsection NC of this code.) Other design codes or standards may be acceptable depending on their application. i. The NRC staff evaluates the proposed limitations on allowable stresses and strains in the confinement cask, reinforced concrete components, system components important to safety, and other components subject to review, by comparison with those specified in applicable codes and standards. Where certain proposed load combinations will exceed the accepted limits for localized points on the structure, the applicant should provide adequate justification to show that the deviation will not affect the functional integrity of the structure. ii. The NRC has accepted the use of applicable subsections of the ASME B&PV Code, Division 1, for components used within the confinement cask but not integrated with it. This includes the “basket” structure used in casks to restrain and position multiple fuel elements.	As specified in Section 3.1.2, the canister and basket structure are designed in accordance with the ASME Code, Section III, Division I, 1995 Edition. The canister is designed in accordance with Subsection NB of the ASME Code, while the basket structure is designed in accordance with Subsection NG criteria. A list of exceptions from the ASME code is provided in Appendix B, Table B3-1.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
b. Concrete Containments	i. ACI 359 (also designated as Section III, Division 2, of the ASME B&PV Code, Subsection CC) constitutes an acceptable standard for prestressed and reinforced concrete that is an integral component of a radioactive material containment vessel that must withstand internal pressure in operation or testing. ii. If ACI 359 pertains to a given ISFSI structure, it applies to all aspects of the design, material selection, fabrication, and construction of that structure. The NRC has not accepted the proposed substitution of elements from ACI 318 or ACI 349 for any portion of ACI 359 with regard to the structure of an ISFSI. ISFSI structures to which ACI 359 applies shall also meet the minimum functional requirements of ANSI/ANS-57.9 for subject areas not specifically addressed in ACI 359.	The UMS system does not utilize concrete containment vessels. Thus, ACI-359 is not applicable.
2. Reinforced Concrete (RC) Structures Important to Safety, but not within the Scope of ACI 359	The NRC accepts the use of ACI 349 for the design, material selection and specification, and construction of all reinforced concrete structures that are not addressed within the scope of ACI 359. However, in such instances, the design, material selection and specification, and construction must also meet any additional or more stringent requirements given in ANSI/ANS-57.9, as incorporated by reference in NRC Regulatory Guide (RG) 3.60. Section V of this chapter provides additional guidance regarding specific review procedures.	As stated in Section 3.1.2, the Vertical Concrete Cask is designed in accordance with ACI-349 and ANSI/ANS-57.9.
3. Other Reinforced Concrete Structures Subject to Approval	The NRC accepts the use of either ACI 318 or ACI 349 for reinforced concrete structures that are subject to approval but are not important to safety. Section V of this chapter provides additional guidance regarding specific review procedures.	The UMS system has no concrete structures other than that addressed in #2 above.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
4. Other System Components Important to Safety	The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and standards cited therein) as the basic reference for ISFSI structures important to safety that are not designed in accordance with the Section III of the ASME B&PV Code. However, both the lifting equipment design and the devices for lifting system components that are important to safety must comply with American National Standards Institute (ANSI) Standard N14.6. The NRC accepts the load combinations shown in Table 3-1 for structures not designed under either Section III of the ASME B&PV Code or ACI 359. These load combinations are based upon ANSI/ANS-57.9, with supplemental definition of terms and combinations. The principal codes and standards include the following references that may apply to steel structures and components: a. American Institute of Steel Construction (AISC), "Specification for Structural Steel Buildings — Allowable Stress Design and Plastic Design" b. AISC, "Load and Resistance Factor Design Specification for Structural Steel Buildings" c. American Welding Society, "Structural Welding Code Steel," AWS D1.1 d. American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7 [however, note that load combinations established on the basis of ANSI/ANS-57.9 (DCSS SRP Table 3-1) are to be used] e. ACI 349-85, Appendix B, for embedments or 10.14 for composite compression sections, as applicable, when constructed of structural steel embedded in reinforced concrete	The lifting devices of the UMS system are evaluated in accordance with NUREG-0612 and ANSI N14.6, as specified in Section 3.1.2.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
5. Other Components Subject to NRC Approval	For structural design and construction of other components subject to NRC approval, the principal codes and standards include the following: a. ASCE 7 b. Uniform Building Code (UBC) c. AISC, "Specification for Structural Steel Buildings—Allowable Stress Design and Plastic Design" d. AISC "Code of Standard Practice for Steel Buildings and Bridges" e. ASME B&PV Code, Section VIII	Not applicable. All components of the system subject to NRC approval are covered by the acceptance criteria specified in the previous sections.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 4 – Thermal Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Minimum Lifetime	10 CFR Part 72 requires an analysis and evaluation of DCSS thermal design and performance to demonstrate that the cask will permit safe storage of the spent fuel for a minimum of 20 years.	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Tables 4.1-4 and 4.1-5 demonstrate that the system's temperatures are maintained within their allowable limits.
2. Spent Fuel Cladding Protection	The spent fuel cladding must be protected against degradation that may lead to gross ruptures.	Tables 4.1-4 and 4.1-5 demonstrate that the fuel cladding temperatures are maintained within allowable limits.
3. Thermal Structures, Systems, and Components	Thermal structures, systems, and components important to safety must be described in sufficient detail to permit evaluation of their effectiveness. Applicable thermal requirements are identified, in part, in 10 CFR 72.24(c)(3), 72.24(d), 72.122(h)(1), 72.122(l), 72.128(a)(4), 72.236(f), 72.236(g), and 72.236(h). 10 CFR 72.24(c)(3) Contents of Application: Descriptions of Components Important to Safety 10 CFR 72.24(d) Contents of Application: Margins of Safety / Mitigation of Accident Consequences 10 CFR 72.122(h)(1) Overall Requirements: Confinement Barriers and Systems 10 CFR 72.122(l) Overall Requirements: Retrievability 10 CFR 72.128(a)(4) Criteria for Spent Fuel Storage and Handling: Testable Heat Removal Capacity 10 CFR 72.236(f) Specific Requirements for Spent Fuel Storage Cask Approval: Passive Heat Removal 10 CFR 72.236(g) Specific Requirements for Spent Fuel Storage Cask Approval: Minimum 20-year Lifetime 10 CFR 72.236(h) Specific Requirements for Spent Fuel Storage Cask Approval: Wet/Dry Loading and Unloading Compatibility	The discussion of the thermal design features of the system is presented in Section 4.1. Tables 4.1-4 and 4.1-5 demonstrate that the temperatures of SSCs are maintained within allowable limits for all components of the system, including the fuel cladding. Therefore, the system is not adversely affected by normal, off-normal, or accident condition events. The temperatures of the system are maintained within allowable limits, and do not preclude retrieval of spent fuel from the system. As specified in LCO A 3.1.6, the air temperatures at the outlet vents and ISFSI ambient are measured to verify proper operation of the concrete cask's heat removal system following the start of storage operations. Section 1.1 and Table 2-1 specify a 50-year design life for the system. Tables 4.1-4 and 4.1-5 demonstrate that the system's temperatures are maintained within their allowable limits. The operating procedures for the system, presented in Chapter 8, include procedures for wet and dry loading and unloading operations. A discussion is provided for development of dry loading and unloading procedures for dry cask handling facilities.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 4 – Thermal Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Long-term Cladding Temperatures	Fuel cladding (zirconium alloy) temperature at the beginning of dry cask storage should generally be below the allowable temperature of 400°C (752°F) per ISG-11, Rev. 2.	As shown in Tables 4.1-4 and 4.1-5, the fuel cladding temperatures are maintained below allowable temperature limits for zirconium alloy-clad fuel as determined in accordance with ISG 11, Rev. 2.
2. Short Term Cladding Temperatures	Fuel cladding temperature should generally be maintained below 570°C (1058°F) for short-term, off-normal and accident conditions (PNL 4835). For fuel transfer operations (e.g., vacuum drying of the cask or dry transfer), the temperature should generally be maintained below 400°C (752°F). (ISG-11, Rev 2)	As shown in Tables 4.1-4 and 4.1-5, the fuel cladding temperatures are maintained below 570°C (1058°F) for short-term, off-normal and accident conditions. For transfer operations, the fuel cladding temperatures are maintained below 400°C (752°F).
3. Maximum Internal Pressure	The maximum internal pressure of the cask should remain within its design pressures for normal, off-normal, and accident conditions assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods.	The maximum normal condition pressure calculation is presented in Section 4.4.5. The accident condition pressure calculation is presented in Section 11.2.1. The off-normal condition is bounded by the accident condition, which assumes 100% failure of the cladding.
4. Maximum Material Temperatures	Cask and fuel materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions in order to enable components to perform their intended safety functions.	Tables 4.1-4 and 4.1-5 demonstrate that the temperatures are maintained within allowable limits for all components of the system, including the fuel cladding. Therefore, the system is not adversely affected by normal, off-normal, or accident condition events.
5. Fuel Cladding Protection	The spent fuel cladding is the primary structural component that is used to ensure that the spent fuel is contained in a known geometric configuration.	As concluded in ISG-11, Rev. 2, creep under normal conditions of storage will not cause gross rupture of the cladding, and the geometric configuration of the spent fuel will be preserved provided that the maximum cladding temperature does not exceed 400°C (752°F).
6. Long-Term Cladding Damage	Creep is the dominant mechanism for cladding deformation under normal conditions of storage. The relatively high temperatures, differential pressures, and corresponding hoop stress on the cladding will result in permanent creep deformation of the cladding over time.	A temperature limit of 400°C (752°F) for normal conditions of storage and for short-term storage operations will limit cladding hoop stresses and creep and limit the amount of soluble hydrogen available to form radial hydrides. (ISG-11, Rev. 2)
7. Passive Cooling	The cask system should be passively cooled. [10 CFR 72.236(f)]	As stated in Sections 1.2 and 4.1, the system is passively cooled.
8. Thermal Operating Limits	The thermal performance of the cask should be within the allowable design criteria specified in SAR Section 2 (e.g., materials, decay heat specifications) and SAR Section 3 (e.g., thermal stress analysis) for normal, off-normal, and accident conditions.	The thermal stress analyses of the canister and Vertical Concrete Cask for normal conditions are provided in Sections 3.4.4.1.1 and 3.4.4.2.3, respectively. The system is evaluated for off-normal thermal loading in Section 11.1.2, and the system is analyzed for accident thermal loading in Sections 11.2.6, 11.2.7 and 11.2.13.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 5 – Shielding Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Shielding System Description	10 CFR Part 72 requires that spent fuel radioactive waste storage and handling systems be designed with suitable shielding to provide adequate radiation protection under both normal and accident conditions. Consequently, the DCSS application must describe the shielding structures, systems, and components (SSCs) important to safety in sufficient detail to allow the NRC staff to thoroughly evaluate their effectiveness. It is the responsibility of the vendor, the facility owner, and the NRC staff to analyze such SSCs with the objective of assessing the impact of direct radiation doses on public health and safety.	A general description of the system is provided in Section 1.2, with a detailed description of the shielding features of the system provided in Section 5.1.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 5 – Shielding Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
2. Protection During Accidents	In addition, SSCs important to safety must be designed to withstand the effects of both credible accidents and severe natural phenomena without impairing their capability to perform their safety functions. The applicable shielding requirements are identified, in part, in 10 CFR 72.24(c)(3), 72.24(d), 72.104(a), 72.106(b), 72.122(b), 72.122(c), 72.128(a)(2), and 72.236(d).	
	10 CFR 72.24(c)(3) Contents of Application: Descriptions of Components Important to Safety	A description of the shielding components of the system is provided in Section 5.1.
	10CFR 72.24(d) Contents of Application: Margins of Safety / Mitigation of Accident Consequences	The design basis dose rates for accident conditions are listed in Section 10.2.2. Specific details of the dose rate due to the tip-over accident are presented in Section 11.2.12.
	10 CFR 72.104(a) Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS: Annual Site Boundary Dose Limit	The controlled area boundary dose calculations and minimum site boundary distances are presented in Section 10.4.
	10 CFR 72.106(b) Controlled Area of an ISFSI or MRS: Design Basis Accident Site Boundary Dose Limit	The accident condition dose rates are discussed in Section 10.2.2.
	10 CFR, 72.122(b) Overall Requirements: Protection Against Environmental Conditions and Natural Phenomena	Evaluation of the system for off-normal and accident condition events is provided in Sections 11.1 and 11.2. The radiological consequences of each event are addressed.
	10 CFR 72.122(c) Overall Requirements: Protection Against Fires and Explosions	The radiological consequences of a fire accident are provided in Section 11.2.6. The radiological consequences of an explosion are provided in Section 11.2.5.
	10 CFR 72.128(a)(2) Criteria for Spent Fuel … Storage and Handling: Radiation Protection	The dose rate results demonstrating the radiation protection features of the system are presented in Section 5.1.
	10 CFR 72.236(d) Specific Requirements for Spent Fuel Storage Cask Approval: Radiation Protection	As described above, the normal condition controlled area boundary dose rates are provided in Section 10.4. The accident condition doses are discussed in Section 10.2.2.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 5 – Shielding Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Minimum Distance from Controlled Area Boundary	The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The "controlled area" is defined in 10 CFR 72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.	As described in Section 10.4, the minimum allowable controlled area boundary distance for a single cask is 100 meters.
2. Controlled Area Boundary Dose Limits	The cask vendor must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed DCSS are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.	Section 10.4 presents the controlled area boundary dose rate evaluation for a typical array configuration. The minimum allowable controlled area boundary distance is 100 meters without taking credit for shielding provided by any intermediate structures or topography.
3. ALARA	Dose rates from the cask must be consistent with a well-established "as low as reasonably achievable" (ALARA) program for activities in and around the storage site.	The dose rates for the system are presented in Section 5.1. These dose rates are within the allowables specified in Section 10.2.1, which are consistent with ALARA principles.
4. Maximum Accident Controlled Area Boundary Dose	After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ.	Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem.
5. Occupational Dose Limits	The proposed shielding features must ensure that the DCSS meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20, Subparts C and D.	Occupational dose estimates for typical loading operations are provided in Section 10.3. In practice, occupational doses would be controlled on a site-specific basis by the operator of the ISFSI.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 6 – Criticality Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
	Spent fuel storage systems must be designed to remain subcritical unless at least two unlikely independent events occur. Moreover, the spent fuel cask must be designed to remain subcritical under all credible conditions. Regulations specific to nuclear criticality safety of the cask system are specified in 10 CFR 72.124 and 72.236(c). Other pertinent regulations include 10 CFR 72.24(c)(3), 72.24(d), and 72.236(g). Normal and accident conditions to be considered are also identified in 10 CFR Part 72.	
	10 CFR 72.24(c)(3) Contents of Application: Descriptions of Components Important to Safety	A general description of the system is provided in Section 1.2, with a detailed description of the criticality safety features of the system provided in Section 6.1.
	10 CFR 72.24(d) Contents of Application: Margins of Safety / Mitigation of Accident Consequences	Section 6.4 presents the results of the criticality evaluation of the transfer cask and storage cask.
	10 CFR 72.124 Criteria for Nuclear Criticality Safety	The criteria for criticality safety are provided in Sections 2.3.4 and 6.1.
	10 CFR 72.236(c) Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Subcritical Configuration	Section 6.4 presents the results of the criticality evaluation of the storage cask for the most credible reactive conditions.
	10 CFR 72.236(g) Specific Requirements for Spent Fuel Storage Cask Approval: Minimum 20-year Lifetime	Section 1.1 and Table 2-1 specify a 50-year design life for the system.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 6 – Criticality Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Subcriticality Margin	The multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.	As stated in Section 6.1, the maximum allowable multiplication factor (k_s) for the system is 0.95, including adjustment for all biases and uncertainties, as calculated in Section 6.5.
2. Double Contingency	At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.	As stated in Section 6.1, the criticality analyses are performed for the most reactive credible configuration of the cask, at the highest enrichment, without credit for fuel burnup, and at the most reactive internal water moderator density, even though it is stated that water intrusion is not a credible event.
3. Criticality Design Features	When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.	As stated in Section 6.1, the criticality safety of the design is based on geometry and fixed neutron poisons. Recently proposed rule changes (Federal Register, June 9, 1998) include discussion clarifying the 10 CFR 72.124(b) requirement to verify the "continued efficacy" of neutron poisons as applicable only to wet storage systems, and not to dry, provided that the effectiveness of the poisons is demonstrated at the outset. Verification of the neutron absorbing materials effectiveness is discussed in Section 9.1.
4. Conservative Assumptions	Criticality safety of the cask system should not rely on use of the following credits: a. burnup of the fuel b. fuel-related burnable neutron absorbers c. more than 75 percent for fixed neutron absorbers when subject to standard acceptance tests.	Section 6.1 provides a list of conservative assumptions that are used in the criticality safety evaluation. No fuel burnup is assumed, and only 75% of the minimum ^{10}B loading on the neutron absorber plates is used. Also, no integral fuel burnable neutron absorbers, nor fission product neutron poisons, are considered in the analysis.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 7 – Confinement Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Description of Structures, Systems, and Components Important to Safety	The SAR must describe the confinement structures, systems, and components (SSCs) important to safety in sufficient detail to facilitate evaluation of their effectiveness. [10 CFR 72.24(c)(3) and 10 CFR 72.24(l)]	A general description of the system is provided in Section 1.2, with a detailed description of the confinement features of the system provided in Section 7.1.
2. Protection of Spent Fuel Cladding	The design must adequately protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures during storage, or the fuel must be confined through other means such that fuel degradation during storage will not pose operational safety problems with respect to removal of the fuel from storage. [10 CFR 72.122(h)(1)]	As described in Sections 7.2.1 and 7.3, the integrity of the canister is maintained under normal and accident conditions. Therefore, the inert helium atmosphere is maintained in the canister, protecting the fuel cladding against degradation.
3. Redundant Sealing	The cask design must provide redundant sealing of the confinement boundary. [10 CFR 72.236(e)]	As described in Section 7.1.3.2, the canister is sealed after loading by means of a redundant closure system.
4. Monitoring of Confinement System	Storage confinement systems must allow continuous monitoring, such that the licensee will be able to determine when to take corrective action to maintain safe storage conditions. [10 CFR 72.122(h)(4) and 10 CFR 72.128(a)(1)]	The canister is a fully welded class 1 component designed and fabricated in accordance with ASME Code, Section III, Subsection NB. It is closed with a fully welded redundant closure system. Therefore, in accordance with previous regulatory guidance, monitoring of the confinement is not required.
5. Instrumentation	The design must provide instrumentation and controls to monitor systems that are important to safety over anticipated ranges for normal and off-normal operation. In addition, the applicant must identify those control systems that must remain operational under accident conditions. [10 CFR 72.122(l)]	As monitoring is not required, there is no instrumentation and controls required.
6. Release of Nuclides to the Environment	The applicant must estimate the quantity of radionuclides expected to be released annually to the environment. [10 CFR 72.24(l)(1)]	As described in Sections 7.2.1 and 7.3, there is no credible leakage from the confinement boundary during all postulated normal and accident condition events. Therefore, no release of radionuclides to the environment is credible.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 7 – Confinement Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
7. Evaluation of Confinement System	The applicant must evaluate the cask and its systems important to safety, using appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR 72.236(l) and 10 CFR 72.24(d)] In addition, SSCs important to safety must be designed to withstand the effects of credible accidents and severe natural phenomena without impairing their capability to perform safety functions. [10 CFR 72.122(b)]	The confinement system is analyzed for normal conditions in Sections 3.4.2 and 3.4.4.1, and for off-normal, and accident conditions in Sections 11.1 and 11.2, respectively. The confinement capability of the canister closure is verified by helium leakage testing of the shield lid-to-canister shell weld following fuel loading as specified in Section 8.1 and the Technical Specifications.
8. Annual Dose Limit in Effluents and Direct Radiation from an Independent Spent Fuel Storage Installation (ISFSI)	During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ. [10 CFR 72.104(a)]	The site boundary dose calculations and minimum site boundary distances are presented in Section 10.4.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 7 – Confinement Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Redundant Sealing	The cask design must provide redundant sealing of the confinement boundary sealing surface. Typically, this means that field closures of the confinement boundary must either have double seal welds or double metallic O-ring seals.	As described in Section 7.1.3.2, the canister is sealed after loading by means of a redundant lid closure system.
2. Code Compliance	The confinement design must be consistent with the regulatory requirements, as well as the applicant's "General Design Criteria" reviewed in Chapter 2 of this SRP. The NRC staff has accepted construction of the primary confinement barrier in conformance with Section III, Subsections NB or NC, of the Boiler and Pressure Vessel (B&PV) Code promulgated by the American Society of Mechanical Engineers (ASME). (This code defines the standards for all aspects of construction, including materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components.) In such instances, the staff has relied upon Section III to define the minimum acceptable margin of safety; therefore, the applicant must fully document and completely justify any deviations from the specifications of Section III. In some cases after careful and deliberate consideration, the staff has made exceptions to this requirement.	The codes and standards utilized for the confinement system design are specified in Section 7.1.1. ASME Code, Section III, Subsection NB is utilized for the design and fabrication of the canister.
3. Maximum Allowable Leakage Rates	The applicant must specify the maximum allowed leakage rates for the total primary confinement boundary and redundant seals. (Applicants frequently display this information in tabular form, including the leakage rate of each seal.) In addition, the applicant's leakage analysis should be consistent with the principles specified in the "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials" (ANSI N14.5). Generally, the allowable leakage rate must be evaluated for its radiological consequences and its effect on maintaining the necessary inert atmosphere within the cask.	As specified in Sections 7.2.1 and 7.3, leakage from the confinement system under normal, off-normal, and accident conditions is not credible.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 7 – Confinement Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
4. Monitoring and Surveillance	The applicant should describe the proposed monitoring capability and/or surveillance plans for mechanical closure seals. In instances involving welded closures, the staff has previously accepted that no closure monitoring system is required. This practice is consistent with the fact that other welded joints in the confinement system are not monitored. However, the lack of a closure monitoring system has typically been coupled with a periodic surveillance program that would enable the licensee to take timely and appropriate corrective actions to maintain safe storage conditions after closure degradation. The discussion in (a) below taken from chapter 2 of this SRP expands on the requirement for continuous monitoring. (a) Continuous Monitoring The Office of the General Counsel (OGC) has developed an opinion as to what constitutes “continuous monitoring” as required in 10 CFR Part 72.122(h)(4). The staff, in accordance with that opinion has concluded that both routine surveillance programs and active instrumentation meets the intent of “continuous monitoring.” Cask vendors may propose, as part of the SAR, either active instrumentation and/or surveillance to show compliance with 10 CFR Part 72.122(h)(4). The reviewer should note that some DCSS designs may contain a component or feature whose continued performance over the licensing period has not been demonstrated to staff with a sufficient level of confidence. Therefore the staff may determine that active monitoring instrumentation is required to provide for the detection of component degradation or failure. This particularly applies to components whose failure immediately affects or threatens public health and safety. In some cases the vendor or staff in order to demonstrate compliance with 10 CFR Part 72.122(h)(4), may propose a technical specification requiring such instrumentation as part of the initial use of a cask system. After initial use, and if warranted and approved by staff, such instrumentation may be discontinued or modified.	The system utilizes welded closures, as specified in Section 7.1. Therefore, no monitoring system is required.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 7 – Confinement Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
5. Non-Reactive Environment	The cask must provide a non-reactive environment to protect fuel assemblies against fuel cladding degradation, which might otherwise lead to gross rupture. Measures for providing a non-reactive environment within the confinement cask typically include drying, evacuating air and water vapor, and backfilling with a non-reactive cover gas (such as helium). For dry storage conditions, experimental data have not demonstrated an acceptably low oxidation rate for UO_2 spent fuel, over the 20-year licensing period, to permit safe storage in an air atmosphere. Therefore, to reduce the potential for fuel oxidation and subsequent cladding failure, an inert atmosphere (e.g., helium cover gas) has been used for storing UO_2 spent fuel in a dry environment. (See Chapter 8 of this SRP for more detailed information on the cover gas filling process.) Note that other fuel types, such as graphite fuels for the high-temperature gas-cooled reactors (HTGRs), may not exhibit the same oxidation reactions as UO_2 fuels and, therefore, may not require an inert atmosphere. Applicants proposing to use atmospheres other than inert gas should discuss how the fuel and cladding will be protected from oxidation.	As described in Sections 7.0 and 7.1.1, the confinement system is vacuum dried, the dryness verified, and then backfilled with inert helium gas during loading operations.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 8 – Operating Procedures		
Area	**Regulatory Requirement**	**Description of Compliance**
Health and Safety	1. The applicant must develop operating procedures that adequately protect health and minimize danger to life or property. [10 CFR 72.40(a)(5)]	Operating procedures are provided in Chapter 8. Notes and Cautions are listed among the steps to emphasize steps important to maintaining health and safety.
ALARA	2. The applicant must establish operational restrictions to meet the regulatory requirements of 10 CFR Part 20 and objective limits that are as low as is reasonably achievable (ALARA) for radioactive materials in effluents and direct radiation levels associated with ISFSI operations. [10 CFR 72.104(b) and 10 CFR 72.24(e)]	Section 8.0 specifies that the procedures are developed to maintain occupational dose ALARA. Automated welding systems and temporary shielding are utilized to minimize worker dose during canister loading operations. Appendix A, Section A 3.2.2 specifies maximum external dose rates to maintain reasonable dose level within a cask array for routine surveillance and inspection activities.
Control of Radioactive Effluents	3. The applicant must describe all equipment and processes used to maintain control of radioactive effluents. [10 CFR 72.24(l)(2)]	As described in Section 8.0, there are no radioactive effluents in routine operations other than pool water and helium gas that are removed from the canister. These effluents are routinely handled in Licensee operations.
Written Procedures	4. The general licensee shall conduct activities related to storage of spent fuel in accordance with written procedures. [10 CFR 72.212(b)(9)] 5. Vendors seeking approval of a cask design shall ensure that written procedures and appropriate tests are established before initial use of the casks. In addition, the vendor must provide a copy of these procedures and tests to each prospective cask user. [10 CFR 72.234(f)]	Written procedures for the system are provided in Chapter 8. These procedures are intended to provide general operational guidance for use of the system. These procedures will be used by an ISFSI operator to develop detailed, site specific procedures for use of the system.
Wet or Dry Loading and Unloading Facilities	6. The cask must be compatible with wet or dry spent fuel loading and unloading facilities. [10 CFR 72.236(h)]	The system design is compatible with both wet or dry loading and unloading facilities.
Decontamination Features	7. To the extent practicable, the design of the cask must facilitate decontamination. [10 CFR 72.236(i)]	The canister is designed to facilitate decontamination as described in Section 2.3.5.3. As described in Section 8.1.1, the annulus between the canister and transfer cask is filled with clean water prior to placement in the fuel pool to minimize the potential for contamination of the surface of the canister.
Ready Retrieval of Spent Fuel	8. The design of storage systems must allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)]	The procedure provided in Section 8.2 and 8.3 specify the steps necessary for retrieval of the spent fuel from the system for further processing or disposal.

Chapter 8 – Operating Procedures		
Area	**Regulatory Requirement**	**Description of Compliance**
Radioactive Waste Generation	9. The design of the cask must minimize the quantity of radioactive waste generated. [10 CFR 72.128(a)(5) and 10 CFR 72.24(f)]	Operation of the system generates no radioactive waste, other than a limited amount of protective clothing and tools used during loading operations that could be easily disposed or decontaminated.
Inspection, Maintenance, and Testing	10. The design of structures, systems, and components (SSCs) that are important to safety must permit inspection, maintenance, and testing. [10 CFR 72.122(f)]	Appendix A of the CoC Number 1015 Technical Specifications specifies the inspection and maintenance activities required for the system.
Scope of Application	1. Major operating procedures apply to the principal activities expected to occur during dry cask storage. The expected scope of activities for the SAR operating procedure descriptions is described in Section II, "Areas of Review" (*of the SRP*), as well as Section 8 of Regulatory Guide 3.61. Operating procedure descriptions should be submitted to address the cask design features and planned operations.	The operating procedures provided in Chapter 8 cover all planned operations of the system, including loading of spent fuel, placement of the system at the site, and unloading of the system.
Process Control and Hazard Mitigation	2. Operating procedure descriptions should identify measures to control processes and mitigate potential hazards that may be present during planned normal operations. Section V, "Review Procedures" (*of the SRP*), discusses previously identified processes and potential hazards.	The operating procedures provided in Chapter 8 include Notes and Cautions to indicate steps important to mitigate potential hazards.
Operating Controls and Limits	3. Operating procedure descriptions should ensure conformance with the applicable operating controls and limits described in the technical specifications provided in SAR Chapter 12.	The operating controls and limits specified in Chapter 12 are included with the appropriate procedures in Chapter 8.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
Operational Planning	4. Operating procedure descriptions should reflect planning to ensure that operations will fulfill the following acceptance criteria:	
	a. Occupational radiation exposures will remain ALARA	As stated in Section 8.0, the operating procedures are developed to support maintaining occupational doses ALARA.
	b. Effective measures will be taken to preclude potential unplanned and uncontrolled releases of radioactive materials	Sections 8.1.1 and 8.3 include steps to preclude releases of radioactive material during loading and unloading operations.
	c. Offsite dose rates will be maintained within the limits of 10 CFR Part 20 and 10 CFR 72.104 for normal operations, and 10 CFR 72.106 for accident conditions.	Section 10.4 presents the site boundary dose rate evaluation, including the minimum controlled area boundary distance needed to meet an annual dose limit of 25 mrem for normal conditions. Section 10.2.2 indicates that the accident condition controlled area boundary dose will not exceed 5 rem to any organ.
	In addition, the operating procedure descriptions should support and be consistent with the bases used to estimate radiation exposures and total doses. (Refer to Chapter 10 of this SRP).	The operating procedures specified in Chapter 8, and the previous cask loading and unloading experience of NAC, support the calculation of occupational dose rates presented in Section 10.3.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
Surveillance, Maintenance, and Contingency Plans	5. Operating procedure descriptions should include provisions for the following activities:	
	a. testing, surveillance, and monitoring of the stored material and casks during storage and loading and unloading operations	The testing and inspection requirements during loading and unloading operations are specified in Section 8.1 and 8.3. Section 9.2 specifies the inspection and maintenance activities required for the system during storage. The limits established in Appendix A, Section A3.0 and Appendix B, Section B3.0, are provided to ensure that the spent fuel is protected during loading and unloading operations.
	b. maintenance of casks and cask functions during storage	Normal operational maintenance and surveillance activities are specified in Section 9.2. These activities include contingency actions that may be required as a result of the inspection.
	c. contingency actions triggered by inspections, checks, observations, instrument readings, and so forth. (Some of these may involve off-normal conditions addressed in SAR Section 11.)	
Cladding Protection	6. As required by 10 CFR 72.122(h)(1), the operating procedure descriptions should facilitate reducing the amount of water vapor and oxidizing material within the confinement cask to an acceptable level to protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures.	As specified in Appendix A, Sections 3.1.2 and 3.1.3, the canister is vacuum dried to eliminate water, the cavity dryness is verified, and the cavity is then backfilled with inert helium gas during fuel loading operations to protect the fuel cladding against oxidation.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Testing and Maintenance	a. The SAR must describe the applicant's program for preoperational testing and initial operations. [10 CFR 72.24(p)]	Section 9.1 presents the acceptance testing for the system.
	b. The cask design must permit maintenance as required. [10 CFR 72.236(g)]	Section 9.2 presents the maintenance activities for the system.
	c. Structures, systems, and components (SSCs) important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. [10 CFR 72.122(a), 10 CFR 72.122(f), 10 CFR 72.128(a)(1), and 10 CFR 72.24(c)]	The acceptance tests and maintenance activities presented in Sections 9.1 and 9.2 are performed to verify compliance with the design bases and criteria, and that the system continues to perform as designed.
	d. The applicant or licensee must establish a test program to ensure that all required testing is performed to meet applicable requirements and acceptance criteria. In addition, at least 30 days before the receipt of spent fuel, the licensee must submit to the NRC a report concerning the pre-operational test acceptance criteria and test results. [10 CFR 72.162 and 10 CFR 72.82(e)]	The testing and maintenance provided in Sections 9.1 and 9.2 are intended to be used by an ISFSI user in the development of site-specific programs.
	e. The applicant or licensee must evaluate the cask and its systems important to safety, using appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR 72.236(l)]	The acceptance tests presented in Section 9.1 demonstrate that the system will maintain confinement of the spent fuel under normal, off-normal, and accident conditions.
	f. The applicant or licensee must inspect the cask to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce confinement effectiveness. [10 CFR 72.236(j)]	As described in Section 9.1.1, the canister is visually and non-destructively examined prior to use.
	g. The applicant must perform, and make provisions that permit the Commission to perform, tests that the Commission deems necessary or appropriate. [10 CFR 72.232(b)]	Provisions shall be made, as necessary, to facilitate additional NRC imposed testing as required.
	h. The general licensee must accurately maintain the record provided by the cask supplier showing any maintenance performed on each cask. This record must include evidence that any maintenance and testing have been conducted under an NRC-approved quality assurance (QA) program. [10 CFR 72.212(b)(8)]	Records of maintenance activities would be maintained by the ISFSI user, and thus are not applicable.
	The applicant or licensee must assure that the casks are conspicuously and durably marked with a model number, unique identification number, and the empty weight. [10 CFR 72.236(k)]	As specified in Section 9.1.8, each system is to be marked with the model number, unique cask number, empty system weight, and additional information.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
2. Resolution of Issues Concerning Adequacy or Reliability	The SAR must identify all SSCs important to safety for which the applicant cannot demonstrate functional adequacy and reliability through previous acceptable evidence. For this purpose, acceptable evidence may be established in any of the following ways: • prior use for the intended purpose • reference to widely accepted engineering principles • reference to performance data in related applications In addition, the SAR should include a schedule showing how the applicant or licensee will resolve any associated safety questions before the initial receipt of spent fuel. [10 CFR 72.24(i)]	As described in Sections 3.1 and 3.3, the design of the system is based on industry standard codes and standards for materials and margins of safety. The acceptance tests specified in Section 9.1 are performed to demonstrate the adequacy of each fabricated system in accordance with applied Codes and Standards. The system does not rely on any materials or design standards that lack acceptable evidence of functional adequacy.
3. Cask Identification	The applicant or licensee must conspicuously and durably mark the cask with a model number, unique identification number, and empty weight. [10 CFR 72.236(k)]	As specified in Section 9.1.8, each system is to be marked with the model number, unique cask number, empty system weight, and additional information.
Confinement System	American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel (B&PV) Code," Section III, Subsection NB or NC "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment" (ANSI N14.5-1997)	As specified in Section 3.1.2, the canister is designed in accordance with the ASME Code, Section III, Subsection NB. Exceptions to the Code are provided in Appendix B, Table B3-1. The shield lid is helium leakage tested to ensure no credible leakage from the confinement boundary using the evacuated envelope test method in accordance with ANSI N14.5 and ASME Code, Section V, Section 7.1.1.3.
Confinement Internals (e.g., basket)	ASME B&PV Code, Section III, Subsection NG	As specified in Section 3.1.2, the basket structure is designed in accordance with the ASME Code, Section III, Subsection NG.
Metal Cask Overpack	ASME B&PV Code, Section VIII	Not applicable.
Concrete Cask Overpack	American Concrete Institute (ACI) Standards 318 and 349, as appropriate	As stated in Section 3.1.2, the concrete cask is designed in accordance with ACI-349 and ANSI/ANS-57.9.
Other Metal Structures	ASME B&PV Code, Section III, Subsection NF American Institute of Steel Construction (AISC), "Manual of Steel Construction"	Not applicable.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 10 – Radiation Protection		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Effluent and Direct Radiation	Criteria for radioactive material released due to effluents and direct radiation from an ISFSI or MRS are contained 10 CFR 72.104. 10 CFR 72.104 Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS	The controlled area boundary dose calculations and minimum controlled area boundary distances are presented in Section 10.4.
2. Occupational Exposures	Criteria for Occupational Exposures are contained in 10 CFR 20.1201, 10 CFR 20.1207, 10 CFR 20.1208, and 10 CFR 20.1301 10 CFR 20.1201 Occupational Dose Limits for Adults 10 CFR 20.1207 Occupational Dose Limits for Minors 10 CFR 20.1208 Dose to an Embryo/Fetus 10 CFR 20.1301 Dose Limits for Individual Members of the Public	Estimated occupational doses for typical loading operations are provided in Section 10.3. In practice, occupational doses would be controlled on a site-specific basis by the operator of the ISFSI.
3. Public Exposures	Criteria for public exposures under normal and accident conditions are contained within. [10 CFR 72.104 and 10 CFR 72.106] 10 CFR 72.104 Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS 10 CFR 72.106 Controlled Area of an ISFSI or MRS	The controlled area boundary dose calculations and minimum site boundary distances are presented in Section 10.4. Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, exclusive of skin.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 10 – Radiation Protection		
Area	**Regulatory Requirement**	**Description of Compliance**
4. ALARA	Criteria for ALARA are contained within 10 CFR 20.1101, 10 CFR 72.24(e), 10 CFR 72.104(b), and 10 CFR 72.126(a)	
	10 CFR 20.1101 Radiation Protection Programs 10 CFR 72.24(e) Contents of Application: ALARA Features	The description of the radiation protection and ALARA considerations of the system are provided in Section 10.1.
	10 CFR 72.104(b) Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS: Operational Restrictions	The design basis for radiation protection is presented in Section 10.2.
	10 CFR 72.126(a) Criteria for Radiological Protection: Exposure Control	Operational methods utilized to provide radiation protection are discussed in Section 10.1.3.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 10 – Radiation Protection		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Design Criteria	Limitations on dose rates associated with direct radiation from the cask are established on the basis of the shielding and confinement evaluations in order to satisfy the regulatory requirements for public dose limits. As stated in 10 CFR Part 72.104, during normal operations and anticipated occurrences, the annual dose equivalent to a real individual located beyond the controlled area, must not exceed the limits discussed below.	The dose rate design criteria are specified in Section 10.2.1.
2. Occupational Exposures	a. dose limits for adults: 5 rem/yr (total effective dose equivalent) b. dose limits for minors: 0.5 rem/yr c. dose to an embryo or fetus (declared pregnant woman): 0.5 rem during entire pregnancy	Estimated occupational doses for typical loading operations are provided in Section 10.3. In practice, occupational doses would be controlled on a site-specific basis by the operator of the ISFSI.
3. Public Exposures	**a. Normal Conditions:** whole body: 25 mrem/yr thyroid: 75 mrem/yr other organ: 25 mrem/yr	The controlled area boundary dose calculations and minimum controlled area boundary distances under normal conditions are presented in Section 10.4.
	These doses include the cumulative effects of other nuclear fuel cycle facilities that may be at the same location as the storage system (i.e., the nuclear power plant) and apply to the limiting real individual of the general public residing at a permanent location nearest the facility.	Contribution to the controlled area boundary dose rate from other facilities co-located with the ISFSI are beyond the scope of the SAR, and are addressed on a site-specific basis by the ISFSI operator.
	b. Accident Conditions and Natural Phenomenon Events 5 rem to the whole body or any organ of any individual located at or beyond the nearest boundary of the controlled area.	Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, exclusive of skin.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 10 – Radiation Protection		
Area	**Regulatory Requirement**	**Description of Compliance**
4. ALARA	As a minimum, the proposed ALARA policy must fulfill the following criteria: a. To the extent practicable, the applicant should employ procedures and engineering controls that are founded upon sound radiation protection principles. b. Any design change should account for radiation protection, technological, and economical considerations. c. The applicant should have a written policy statement reflecting management commitment to maintain occupational and public exposures to radiation and radioactive material ALARA.	The description of the ALARA considerations of the system are provided in Section 10.1. The operating procedures provided in Chapter 8 are developed to keep occupational doses ALARA.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 11 – Accident Analysis		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Credible Accident and Natural Phenomena	Structures, systems, and components (SSC) important to safety must be designed to withstand credible accidents and natural phenomena without impairing their ability to perform safety functions. [10 CFR 72.24(d)(2); 10 CFR 72.122(b)(2), (3), (d), and (g)]	Analyses of the system for a variety of postulated off-normal and accident conditions are presented in Sections 11.1 and 11.2, respectively.
2. Controlled Area Boundary Dose	During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ as a result of exposure to the sources listed in the regulations. [10 CFR 72.104(a); 10 CFR 72.236(d); and 10 CFR 72.24(d)]	The controlled area boundary dose calculations and minimum controlled area boundary distances under normal conditions are presented in Section 10.4.
3. Design Basis Accident Dose	Dose Limits for Design-Basis Accidents require that any individual located on or beyond the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident. [10 CFR 72.106(b); 10 CFR 72.24(m); and 10 CFR 72.24(d)(2)]	Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, exclusive of skin.
4. Criticality Control	The spent fuel must be maintained in a subcritical condition under credible conditions. [10 CFR 72.236(c) and 10 CFR 72.124(a)]	Section 6.4 presents the results of the criticality evaluation of the storage cask for the most credible reactive conditions, including the consequences of the off-normal and accident condition events evaluated in Sections 11.1 and 11.2, respectively.
5. Confinement Control	The cask and its systems important to safety must be evaluated, using appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under credible accident conditions. [10 CFR 72.236(l)]	As stated in Section 7.3, the confinement system maintains its integrity for all credible off-normal and accident conditions.
6. Ready Retrieval of Spent Fuel	Storage systems must allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)]	The off-normal and accident condition analyses presented in Sections 11.1 and 11.2 demonstrate that the spent fuel contents are protected during off-normal and accident conditions. Therefore, retrieval of the spent fuel from the system is not impacted by these postulated events.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 11 – Accident Analysis		
Area	**Regulatory Requirement**	**Description of Compliance**
7. Monitoring Systems	Instrumentation and control systems must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation. Those instruments and control systems that must remain operational under accident conditions must be identified in the Safety Analysis Report. [10 CFR 72.122(i)]	Daily surveillance of the concrete cask is performed to verify continued thermal operability of the system. The confinement system is fully welded per Appendix A, Section A3.1.5, to assure no credible leakage from the confinement boundary. No seal monitoring is required.
8. Surveillance	Where instrumentation and control systems are not appropriate, storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. [72.122(h)(4)]	No active, continuous monitoring systems are required. Licensee radiological monitoring programs assure ISFSI operations meet 10 CFR 72.104 and 72.106 requirements.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 11 – Accident Analysis		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Dose Limits for Off-Normal Events	During normal operations and anticipated occurrences, the requirements specified in 10 CFR Part 20 must be met. In addition the annual dose equivalent to any individual located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to the following sources: a. planned discharges to the general environment of radioactive materials (with the exception of radon and its decay products) b. direct radiation from operations of the independent spent fuel storage installation (ISFSI) c. any other cumulative radiation from uranium fuel cycle operations (i.e., nuclear power plant) in the affected area	The controlled area boundary dose calculations and minimum controlled area boundary distances under normal conditions are presented in Section 10.4. No off-normal events are postulated that would result in a controlled area boundary dose in excess of the normal condition analysis.
2. Dose Limit for Design-Basis Accidents	Any individual located at or beyond the nearest controlled area boundary must not receive a dose greater than 5 rem to the whole body or any organ from any design-basis accident.	Section 10.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, exclusive of skin.
3. Criticality	The spent fuel must be maintained in a subcritical condition under credible conditions (i.e., k_{eff} equal to or less than 0.95). At least two unlikely, independent, and concurrent or sequential changes must be postulated to occur in the conditions essential to nuclear criticality safety before a nuclear criticality accident is possible (double contingency).	Section 6.4 presents the results of the criticality evaluation of the storage cask for the most credible reactive conditions, including the consequences of the off-normal and accident condition events evaluated in Sections 11.1 and 11.2, respectively. As stated in Section 6.1, the criticality analyses are performed for the most reactive credible configuration of the cask, at the highest enrichment, without credit for fuel burnup, and at the most reactive internal water moderator density, even though it is stated that water intrusion is not a credible event.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 11 – Accident Analysis		
Area	**Acceptance Criteria**	**Description of Compliance**
4. Confinement	The cask and its systems important to safety must be evaluated, using appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under credible accident conditions.	As stated in Section 7.3, the confinement system maintains its integrity for all credible off-normal and accident conditions.
5. Retrievability	Retrievability is the capability to return the stored radioactive material to a safe condition without endangering public health and safety. This generally means ensuring that any potential release of radioactive materials to the environment or radiation exposures is not in excess of the limits in 10 CFR 20 or 10 CFR 72.122(h)(5). ISFSI and MRS storage systems must be designed to allow ready retrieval of the stored spent fuel or high level waste (MRS only) for compliance with 10 CFR 72.122(l).	The off-normal and accident condition analyses presented in Sections 11.1 and 11.2 demonstrate that the spent fuel contents are protected during off-normal and accident conditions. Therefore, retrieval of the spent fuel from the system is not impacted by these postulated events.
6. Instrumentation	The SAR must identify all instruments and control systems that must remain operational under accident conditions.	The system does not utilize instrumentation and control systems, but utilizes routine inspection and surveillance to verify proper operation of the system.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 12 – Operating Controls and Limits	
Regulatory Requirement	**Description of Compliance**
1. General Requirement for Technical Specifications	
The applicant shall propose technical specifications (complete with acceptable bases and adequate justification). These specifications must include the following five areas [10 CFR 72.44(c), 10 CFR 72.24(g), and 10 CFR 72.26]:	
a. functional/operating limits, monitoring instruments, and limiting controls	Functional and operating limits are specified in Appendix A, Section 3.0 and in Appendix B, Sections B2.0 and B3.0.
b. limiting conditions	Limiting conditions for operation are specified in Appendix A, Section A3.0.
c. surveillance requirements	Surveillance requirements are specified in Appendix A, Section A3.0.
d. design features	Design features are specified in Appendix B, Section B3.0.
e. administrative controls	Administrative controls are specified in Appendix A, Section A5.0.
Subpart E, "Siting Evaluation Factors," and Subpart F, "General Design Criteria," to 10 CFR Part 72, provide the bases for the cask system design and, hence, are applicable as bases for appropriate technical specifications.	
2. Specific Requirements for Technical Specifications — Storage Cask Approval	
As a condition of approval, the design, fabrication, testing, and maintenance of a spent fuel DCSS must comply with the requirements of 10 CFR 72.236. [10 CFR 72.234(a)] 10 CFR 72.236 Specific Requirements for Spent Fuel Storage Cask Approval	The operating controls, limits, and surveillance activities specified in Appendix A are intended to ensure that the system is maintained within its design basis through all normal, off-normal, and accident conditions.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 12 – Operating Controls and Limits	
Regulatory Requirement	**Description of Compliance**
The applicant must provide specifications for the spent fuel to be stored in the DCSS. At a minimum, these specifications should include, but not be limited to the following details [10 CFR 72.236(a)]: a. type of spent fuel (i.e., BWR, PWR, or both) b. maximum allowable enrichment of the fuel prior to any irradiation c. burn-up (i.e., megawatt-days/MTU) d. minimum acceptable cooling time of the spent fuel prior to storage in the DCSS (minimum 1 year) e. maximum heat that the DCSS system is designed to dissipate f. maximum spent fuel loading limit weights and dimensions g. condition of the spent fuel (i.e., undamaged or damaged assembly or consolidated fuel rods) h. inerting atmosphere requirements	Specifications for the spent fuel contents are provided in Appendix B, Tables B2-1 through B2-5. As specified in Appendix A, LCO 3.1.3, the canister is backfilled with helium gas to maintain an inert atmosphere for the spent fuel.
The applicant must provide design bases and design criteria for structures, systems, and components (SSCs) important to safety. [10 CFR 72.236(b)]	The design bases and criteria for the system are specified in Section 2.2.
The applicant must design and fabricate the DCSS so that the spent fuel will be maintained in a subcritical condition under credible conditions. [10 CFR 72.236(c)]	As shown in Section 6.4, the spent fuel is maintained in a subcritical configuration under all credible configurations.
The applicant must provide radiation shielding and confinement features that are sufficient to meet the requirements in 10 CFR 72.104 and 72.106 regarding radioactive material in effluents, direct radiation, and area control. [10 CFR 72.236(d) and 10 CFR Part 20] 10 CFR 72.104 Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS 10 CFR 72.106 Controlled Area of an ISFSI or MRS	The maximum external dose rates for the system are specified in Appendix A, Section A3.2.2. These limits are established to ensure that, for the minimum controlled area boundary distance presented in Section 10.4, the controlled area boundary annual dose will be maintained within allowable limits.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 12 – Operating Controls and Limits	
Regulatory Requirement	**Description of Compliance**
The applicant must design the DCSS to meet the following criteria:	
• Provide redundant sealing of confinement systems. [10 CFR 72.236(e)]	The redundant sealing features of the confinement system are presented in Section 2.3.2.1 and Chapter 7.
• Provide adequate heat removal capacity without active cooling systems. [10 CFR 72.236(f)]	As shown in Table 4.1-4, the system provides adequate heat removal through the passive cooling design features described in Section 4.1.
• Safely store the spent fuel for a minimum of 20 years and permit maintenance as required. [10 CFR 72.236(g)]	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Routine maintenance is permitted as specified by Section 9.2.
• Facilitate decontamination to the extent practicable. [10 CFR 72.236(i)]	Decommissioning of the system is discussed in Section 2.4.
The DCSS must be compatible with wet or dry spent fuel loading and unloading facilities. [10 CFR 72. 236(h)]	The operating procedures for the system are presented in Chapter 8, and include procedures for wet and dry loading and unloading operations.
The applicant must inspect the DCSS to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness. [10 CFR 72.236(j)]	As described in Section 9.1.1, the canister is visually and non-destructively examined prior to use.
The applicant must evaluate the DCSS, and its systems important to safety, using appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR 72.236(l)]	The canister is analyzed for normal conditions in Section 3.4.4.1, and for off-normal and accident conditions in Sections 11.1 and 11.2, respectively. Because the canister maintains adequate positive margins of safety, the system will reasonably maintain confinement under all credible conditions.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 13 – Quality Assurance	
Regulatory Requirement	**Description of Compliance**
According to 10 CFR 72.24, "Contents of Application: Technical Information," the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, "Quality Assurance," with regard to the QA program to be applied to the design, fabrication, construction, testing, and operation of the DCSS SSCs important to safety. Moreover, Subpart G states that the licensee shall establish the QA program at the earliest practicable time consistent with the schedule for accomplishing the activities.	A synopsis of the NAC Quality Assurance Program is presented in Section 13.2. This program description is consistent with the 18 criteria specified in Subpart G. The NAC Quality Assurance Program is approved by the NRC under 10 CFR 71, Subpart H.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 13 – Quality Assurance		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Quality Assurance Organization	The SAR should describe (and illustrate in an appropriate chart) the organizational structure, interrelationships, and areas of functional responsibility and authority for all organizations performing quality- and safety-related activities, including both the applicant's organization and principal contractors, if applicable. Persons or organizations responsible for ensuring that an appropriate QA program has been established and verifying that activities affecting quality have been correctly performed should have sufficient authority, access to work areas, and organizational freedom to carry out that responsibility.	The QA organization is described in Section 13.2.1. An organizational chart is provided in Figure 13.2-1.
2. Quality Assurance Program	The SAR should provide acceptable evidence that the applicant's proposed QA program will be well-documented, planned, implemented, and maintained to provide the appropriate level of control over activities and SSCs, consistent with their relative importance to safety.	The implementation of the QA program is described in Section 13.2.2.
3. Design Control	The SAR should describe the approach that the applicant will use to define, control, and verify the design and development of the DCSS. An effective design control program will provide assurance that the proposed DCSS will be appropriately designed and tested and will perform its intended function.	Design control is described in Section 13.2.3.
4. Procurement Document Control	Documents used to procure SSCs or services should include or reference applicable design bases and other requirements necessary to ensure adequate quality. To the extent necessary, these procurement documents should require that suppliers have a QA program consistent with the quality level of the SSCs or services to be procured.	Procurement document control is described in Section 13.2.4.
5. Instructions, Procedures, and Drawings	The SAR should define the applicant's proposed procedures for ensuring that activities affecting quality will be prescribed by, and performed in accordance with, documented instructions, procedures, or drawings of a type appropriate for the circumstances.	Procedures, instructions and drawings are described in Section 13.2.5.
6. Document Control	The SAR should define the applicant's proposed procedures for preparing, issuing, and revising documents that specify quality requirements or prescribe activities affecting quality. These procedures should provide adequate control to ensure that only the latest documents are used. In addition, the applicant's authorized personnel should carefully review and approve the accuracy of all documents and associated revisions before they are released for use.	Document control is described in Section 13.2.6.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 13 – Quality Assurance		
Area	**Acceptance Criteria**	**Description of Compliance**
7. Control of Purchased Material, Equipment, and Services	The SAR should define the applicant's proposed procedures for controlling purchased material, equipment, and services to ensure conformance with specified requirements.	Control of purchased items and services is described in Section 13.2.7.
8. Identification and Control of Materials, Parts, and Components	The SAR should define the applicant's proposed provisions for identifying and controlling materials, parts, and components to ensure that incorrect or defective SSCs are not used.	Identification and control of material, parts and components are described in Section 13.2.8.
9. Control of Special Processes	The SAR should describe the controls that the applicant will establish to ensure the acceptability of special processes (such as welding, heat treatment, nondestructive testing, and chemical cleaning) and that they are performed by qualified personnel using qualified procedures and equipment.	Control of special processes is described in Section 13.2.9.
10. Licensee Inspection	The SAR should define the applicant's proposed provisions for inspection of activities affecting quality to verify conformance with instructions, procedures, and drawings.	Inspection is described in Section 13.2.10.
11. Test Control	The SAR should define the applicant's proposed provisions for tests to verify that SSCs conform to specified requirements and will perform satisfactorily in service. The applicant should specify test requirements in written procedures, including provisions for documenting and evaluating test results. In addition, the applicant should establish qualification programs for test personnel.	Test control is described in Section 13.2.11.
12. Control of Measuring and Test Equipment	The SAR should define the applicant's proposed provisions to ensure that tools, gauges, instruments, and other measuring and testing devices are properly identified, controlled, calibrated, and adjusted at specified intervals.	Control of measuring and test equipment is described in Section 13.2.12.
13. Handling, Storage, and Shipping Control	The SAR should define the applicant's proposed provisions to control the handling, storage, shipping, cleaning, and preservation of SSCs in accordance with work and inspection instructions to prevent damage, loss, and deterioration.	Handling, storage and shipping are described in Section 13.2.13.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 13 – Quality Assurance		
Area	**Acceptance Criteria**	**Description of Compliance**
14. Inspection, Test, and Operating Status	The SAR should define the applicant's proposed provisions to control the inspection, test, and operating status of SSCs to prevent inadvertent use or bypassing of inspections and tests.	Inspection, test, and operating status are described in Section 13.2.14.
15. Nonconforming Materials, Parts, or Components	The SAR should define the applicant's proposed provisions to control the use or disposition of nonconforming materials, parts, or components.	Control of nonconforming items is described in Section 13.2.15.
16. Corrective Action	The SAR should define the applicant's proposed provisions to ensure that conditions adverse to quality are promptly identified and corrected and that measures are taken to preclude recurrence.	Corrective action is described in Section 13.2.16.
17. Quality Assurance Records	The SAR should define the applicant's proposed provisions for identifying, retaining, retrieving, and maintaining records that document evidence of the control of quality for activities and SSCs important to safety.	Records are described in Section 13.2.17.
18. Audits	The SAR should define the applicant's proposed provisions for planning, scheduling, and conducting audits to verify compliance with all aspects of the QA program, and to determine the effectiveness of the overall program. The SAR should clearly identify responsibilities and procedures for conducting audits, documenting and reviewing audit results, and designating management levels to review and assess audit results. In addition, the SAR should describe the applicant's provisions for incorporating the status of audit recommendations in management reports.	Audits are described in Section 13.2.18.

Table 1.5-1 NUREG-1536 Compliance Matrix (continued)

Decommissioning		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Facility Design Features	The ISFSI or MRS must be designed for decommissioning. Provisions must be made to facilitate decontamination of structures and equipment, minimize the quantity of radioactive wastes and contaminated equipment, and facilitate the removal of radioactive wastes and contaminated materials at the time the ISFSI or MRS is permanently decommissioned. [10 CFR 72.130.]	The design of the ISFSI facility is site-specific, and thus not applicable to a DCSS. Decommissioning considerations are discussed in Section 2.4.
2. Cask Design Features	The cask must be designed to facilitate decontamination to the extent practicable. [10 CFR 72.236(i).]	The decontamination features of the system are discussed in Section 2.4.
3. Financial / Records	The requirements for financial assurance and record keeping associated with decommissioning are found in 10 CFR 72.30. 10 CFR 72.30 Financial Assurance and Recordkeeping for Decommissioning	Financial assurance and record keeping issues are site-specific, and thus not applicable to a DCSS.
4. License Termination	The requirements for terminating an ISFSI license and decommissioning ISFSI sites and buildings are found in 10 CFR 72.54, including the requirements for submitting the final decommissioning plan.	ISFSI license termination is a site-specific issue, and thus not applicable to a DCSS.

Decommissioning	
Acceptance Criteria	**Description of Compliance**
1. Decontamination of buildings and equipment, as specified in RG 1.86.	The decontamination features of the system are discussed in Section 2.4.
2. Classification and disposal of wastes, as contained in 10 CFR 61.55.	Not applicable.

1.6 Identification of Agents and Contractors

The prime contractor for the Universal Storage System design is NAC. All design, analysis, licensing, and procurement activities are performed by NAC in accordance with its approved Quality Assurance Program, as described in Chapter 13. Fabrication of the steel components will be by qualified vendors. A qualified concrete contractor will perform construction of the concrete cask. All vendors and contractors will be selected and their performance monitored in accordance with the NAC Quality Assurance Program. All UMS® fabrication and assembly activities will be performed in accordance with quality assurance programs that meet the requirements of 10 CFR 72, Subpart G.

NAC as a contractor, or the licensee, may perform construction of the ISFSI and UMS® loading operations on site in accordance with the NAC or licensee quality assurance program, as appropriate. The licensee will perform decommissioning of the ISFSI in accordance with the licensee quality assurance program.

NAC was founded as a private corporation in 1968, with the primary focus of tracking, inspecting, handling, storing, and transporting spent nuclear fuel. NAC is a wholly owned subsidiary of USEC, Inc., since completion of its acquisition in November 2004. NAC is recognized in the industry as an expert in all aspects of the design, licensing and operation of spent fuel handling, inspection, storage and transport equipment, as well as in the management of spent fuel inventories.

Within the past 15 years, NAC has completed fabrication or has under construction the following transportation and/or storage systems.

Part 71 (Transport Casks)	**Part 72 (Storage System Casks and Components)**
NAC-LWT	UMS®/MPC transfer casks
TRUPACT-II	NAC-I28 S/T metal casks
RH-TRU 72B	NAC-I26 S/T metal cask
NAC-STC	UMS®/MPC TSCs
	UMS®/MPC concrete casks

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1.7 References

1. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Part 72, Title 10, January 1996.

2. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

3. NAC Document No. EA790-SAR-001, "Safety Analysis Report for the UMS® Universal Transport Cask," Docket No. 71-9270, April 1997.

4. Department of Energy, "Multi-Purpose Canister (MPC) Subsystem Design Procurement Specification," Document No. DBG000000-01717-6300-00001, Rev. 6, June 1996.

5. Nuclear Regulatory Commission, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Concrete Cask," Regulatory Guide 3.61, February 1989.

6. ANSI/ANS-57.9-1992, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)," American Nuclear Society, May 1992.

7. American Concrete Institute, "Building Code Requirements for Structural Concrete," (ACI 318-95) and Commentary (ACI 318R-95), October 1995.

8. ASME Boiler and Pressure Vessel Code, Division I, Section III, Subsection NB, "Class 1 Components," 1995 Edition with 1995 Addenda.

9. ASME Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination," 1995 Edition with 1995 Addenda.

10. ASME Boiler and Pressure Vessel Code, Division I, Section III, Subsection NG, "Core Support Structures," 1995 Edition with 1995 Addenda.

11. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980.

12. ANSI N14.6-1993, "American National Standard for Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More," American National Standards Institute, Inc., June 1993.

13. Code of Federal Regulations, "Packaging and Transportation of Radioactive Materials," Part 71, Title 10, April 1996.

14. ASME Boiler and Pressure Vessel Code, Section IX, "Qualification Standards for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators," July 1995.

15. ASME Boiler and Pressure Vessel Code, Section VIII, "Rules for Construction of Pressure Vessels," 1995 Edition with 1995 Addenda.

16. Recommended Practice No. SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing," The American Society for Nondestructive Testing, Inc., edition as invoked by the applicable ASME Code.

17. ANSI N45.2.1, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

18. ANSI N45.2.2-1978, "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants."

19. American Society for Testing and Materials, "Standard Specification for Ready-Mixed Concrete," ASTM C 94.

20. American Society for Testing and Materials, "Standard Specification for Portland Cement," ASTM C 150.

21. American Society for Testing and Materials, "Standard Specification for Concrete Aggregates," ASTM C 33.

22. American Society for Testing and Materials, "Specification for Aggregates for Radiation-Shielding Concrete," ASTM C 637.

23. American Society for Testing and Materials, "Standard Specification for Chemical Admixtures for Concrete," ASTM C 494.

24. American Society for Testing and Materials, “Specification for Fly Ash and Raw or Calcined Natural Pozzolan for Use as a Mineral Admixture in Portland Cement Concrete,” ASTM C 618.

25. American Welding Society, “Structural Welding Code Steel,” AWS D1.1-96, 1996.

26. American Society for Testing and Materials, “Standard Practice for Sampling Freshly Mixed Concrete,” ASTM C 172.

27. American Society for Testing and Materials, “Method of Making and Curing Concrete Test Specimens in the Field,” ASTM C 31.

28. American Society for Testing and Materials, “Standard Test Method for Compressive Strength of Cylindrical Concrete Specimens,” ASTM C 39.

29. Nuclear Regulatory Commission, “Cladding Considerations for the Transport and Storage of Spent Fuel,” Interim Staff Guidance-11, Revision 2.

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1.8 License Drawings

This section presents the list of License Drawings for the Universal Storage System.

1.8.1 License Drawings for the UMS® Universal Storage System

Drawing Number	Title	Revision No.	No. of Sheets
790-501	Canister/Basket Assembly Table, NAC-UMS®	3	1
790-559	Assembly, Transfer Adapter, NAC-UMS®	7	4
790-560	Assembly, Standard Transfer Cask (TFR), NAC-UMS®	17	7
790-561	Weldment, Structure, Vertical Concrete Cask (VCC), NAC-UMS®	15	4
790-562	Reinforcing Bar and Concrete Placement, Vertical Concrete Cask (VCC), NAC-UMS®	18	7
790-563	Lid, Vertical Concrete Cask (VCC), NAC-UMS®	6	1
790-564	Shield Plug, Vertical Concrete Cask (VCC), NAC-UMS®	8	3
790-565	Nameplate, Vertical Concrete Cask (VCC), NAC-UMS®	5	1
790-570	Fuel Basket Assembly, 56 Element BWR, NAC-UMS®	4	2
790-571	Bottom Weldment, Fuel Basket, 56 Element BWR, NAC-UMS®	3	1
790-572	Top Weldment, Fuel Basket, 56 Element BWR, NAC-UMS®	4	1
790-573	Support Disk and Misc. Basket Details, 56 Element BWR, NAC-UMS®	8	1
790-574	Heat Transfer Disk, Fuel Basket, 56 Element BWR, NAC-UMS®	3	1
790-575	BWR Fuel Tube, NAC-UMS®	10	2
790-581	PWR Fuel Tube, NAC-UMS®	9	2
790-582	Shell Weldment, Canister, NAC-UMS®	12	2
790-583	Assembly, Drain Tube, Canister, NAC-UMS®	8	1
790-584	Details, Canister, NAC-UMS®	20	3
790-585	Transportable Storage Canister (TSC), NAC-UMS®	22	3
790-587	Spacer Shim, Canister, NAC-UMS®	1	1
790-590	Loaded Vertical Concrete Cask (VCC), NAC-UMS®	7	2
790-591	Bottom Weldment, Fuel Basket, 24 Element PWR, NAC-UMS®	6	2

License Drawings (continued)

Drawing Number	Title	Revision No.	No. of Sheets
790-592	Top Weldment, Fuel Basket, 24 Element PWR, NAC-UMS®	8	1
790-593	Support Disk and Misc. Basket Details, 24 Element PWR, NAC-UMS®	7	2
790-594	Heat Transfer Disk, Fuel Basket, 24 Element PWR, NAC-UMS®	2	1
790-595	Fuel Basket Assembly, 24 Element PWR, NAC-UMS®	10	2
790-605	BWR Fuel Tube, Over-Sized Fuel, NAC-UMS®	11	2
790-613	Supplemental Shielding, VCC Inlets, NAC-UMS®	2	1
790-617	Door Stop, NAC-UMS®	4	2

1.8.2 Site Specific Spent Fuel License Drawings

Drawing Number	Title	Revision No.	No. of Sheets
412-501	Spent Fuel Can Assembly, Maine Yankee (MY), NAC-UMS®	4	2
412-502	Fuel Can Details, Maine Yankee (MY), NAC-UMS®	6	6

R-135508

DCR No.	REVISION
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A

COMPONENT (DRAWING NUMBER)	PWR CLASS 1		PWR CLASS 2		PWR CLASS 3		BWR CLASS 4		BWR CLASS 5		COMPONENT REQUIRED FOR:
	DRAWING	ASSEMBLY	DRAWING	ASSEMBLY	DRAWING	ASSEMBLY	DRAWING	ASSEMBLY	DRAWING	ASSEMBLY	
ASSEMBLY, UTC, OVERPACK, CASK BODY ASS'Y (790500)	502	99	502	99	502	99	502	99	502	99	
CASK BODY, TRANSPORT CASK, NAMEPLATE (790509)	509	–	509	–	509	–	509	–	509	–	
CASK BODY, TRANSPORT CASK, PRIMARY TRUNNION (790505)	505	–	505	–	505	–	505	–	505	–	
LID ASSEMBLY, CASK, LID (790503)	503	99	503	99	503	99	503	99	503	99	
LID ASSEMBLY, CASK, PORT COVER PLATE (790503)	504	99	504	99	504	99	504	99	504	99	TRANSPORT
IMPACT LIMITER ASSEMBLY, UPPER, CASK, (790506)	506	99	506	99	506	99	506	99	506	99	
IMPACT LIMITER ASSEMBLY, LOWER, CASK, (790507)	507	99	507	99	507	99	507	99	507	99	
TRANSPORT CASK SPACER WELDMENT (790520)	520	98	520	99	–	–	520	1	520	1	
PACKAGE ASSEMBLY, UNIVERSAL TRANSPORT CASK (UTC) (790516)	585	95	585	96	585	99	585	97	585	98	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, BOTTOM WELDMENT (790570)	–	–	–	–	–	–	571	99	571	99	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TOP WELDMENT (790570)	–	–	–	–	–	–	572	99	572	99	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, SUPPORT DISK (790570)	–	–	–	–	–	–	573	1	573	1	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TUBE (2-SIDED) (790570)	–	–	–	–	–	–	575	99	575	98	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TOP NUT (790570)	–	–	–	–	–	–	573	4	573	4	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TIE ROD (790570)	–	–	–	–	–	–	573	5	573	6	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TUBE (NO BORAL) (790570)	–	–	–	–	–	–	575	95	575	94	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TUBE (1 SIDE BORAL) (790570) OVER-SIZED FUEL	–	–	–	–	–	–	605	97	605	96	(3)
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TUBE (NO BORAL) (790570) OVER-SIZED FUEL	–	–	–	–	–	–	605	95	605	94	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TUBE (1 SIDE BORAL) (790570)	–	–	–	–	–	–	575	97	575	96	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TUBE (2 SIDE BORAL) (790570) OVER-SIZED FUEL	–	–	–	–	–	–	605	99	605	98	(3)
FUEL BASKET ASS'Y. 56 ELEMENT BWR, TOP SPACER (790570)	–	–	–	–	–	–	573	7	573	7	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, SPLIT SPACER (790570)	–	–	–	–	–	–	573	8	573	8	
FUEL BASKET ASS'Y. 56 ELEMENT BWR, HEAT TRANSFER DISK (790570)	–	–	–	–	–	–	574	1	574	1	
TSC, SHELL WELDMENT (790585)	582	95	582	96	582	99	582	97	582	98	
TSC, FUEL BASKET ASS'Y (790585)	595	99	595	98	595	97	570	99	570	98	
TSC, DRAIN TUBE ASS'Y (790585)	583	95	583	96	583	99	583	97	583	98	
TSC, LID SUPPORT RING (790585)	584	6	584	6	584	6	584	6	584	6	BOTH
TSC, SHIELD LID ASS'Y (790585)	584	99	584	99	584	99	584	99	584	99	
TSC, COVER (790585)	584	5	584	5	584	5	584	5	584	5	
TSC, STRUCTURAL LID (790585)	584	4	584	4	584	4	584	4	584	4	
TSC, BACKING RING (790585)	584	7	584	7	584	7	584	7	584	7	
TSC, KEY (790585)	584	8	584	8	584	8	584	8	584	8	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, BOTTOM WELDMENT (790595)	591	99	591	99	591	99	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, TOP WELDMENT (790595)	592	97	592	98	592	99	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, SUPPORT DISK (790595)	593	1	593	1	593	1	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, TUBE (790595)	581	99	581	98	581	97	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, SPACER (790595)	593	3	593	3	593	3	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, SPLIT SPACER (790595)	593	2	593	2	593	2	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, TOP NUT (790595)	593	4	593	4	593	4	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, TIE ROD (790595)	593	5	593	6	593	7	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, HEAT TRANSFER DISK (790595)	594	1	594	1	594	1	–	–	–	–	
FUEL BASKET ASS'Y. 24 ELEMENT PWR, TOP SPACER (790595)	593	8	593	8	593	8	–	–	–	–	
ASSEMBLY, TRANSFER ADAPTER, NAC-UMS (790559)	559	99	559	99	559	99	559	99	559	99	
ASSEMBLY, TRANSFER CASK (TFR) NAC-UMS (790560)	560	99	560	98	560	95	560	97	560	96	
WELDMENT, STRUCTURE, VERTICAL CONCRETE CASK (VCC) LAYOUT (790561)	561	95	561	96	561	99	561	97	561	98	
REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) LAYOUT (790562)	562	95	562	96	562	99	562	97	562	98	
LID, VERTICAL CONCRETE CASK (VCC) NAC-UMS (790563)	563	99	563	99	563	99	563	99	563	99	STORAGE
SHIELD PLUG, VERTICAL CONCRETE CASK (VCC) NAC-UMS (790564)	564	99	564	99	564	99	564	99	564	99	
LOADED VERTICAL CONCRETE CASK (VCC) NAC-UMS (790590)	590	95	590	96	590	99	590	97	590	98	

1. THE COMPONENT REQUIRED FOR COLUMN INDENTIFIES THOSE COMPONENTS REQUIRED FOR EITHER TRANSPORTATION OR STORAGE PURPOSES.

NOTE

95	96	97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
ASSY	ASSY	ASSY	ASSY	ASSY						

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3-12		6-18	±.06
		OVER 12		OVER 18	±.09
	ANGULARITY	.X	±.1	ANGLES ±.5°	
	PERPENDICULARITY	ALL UNSPECIFIED TOOL RADII: .01 - .03			
	PARALLELISM	BREAK ALL SHARP CORNERS .01 - .03			
	CONCENTRICITY	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			
	TRUE POSITION	NEXT ASSEMBLY:			
		DRAWING TYPE: LICENSE			

GROUP	NAME	DATE
PREPARER	R. Walker	5-9-00
CHECKER	[signature]	5-9-00
PROJECT MANAGER	[signature]	5/10/00
DIRECTOR DESIGN AND ANALYSIS	[signature]	5-11-00
DIRECTOR LICENSING	J.A. Thompson	5/10/00
VICE PRESIDENT QUALITY	B. Barrett for	5/18/00

NAC INTERNATIONAL

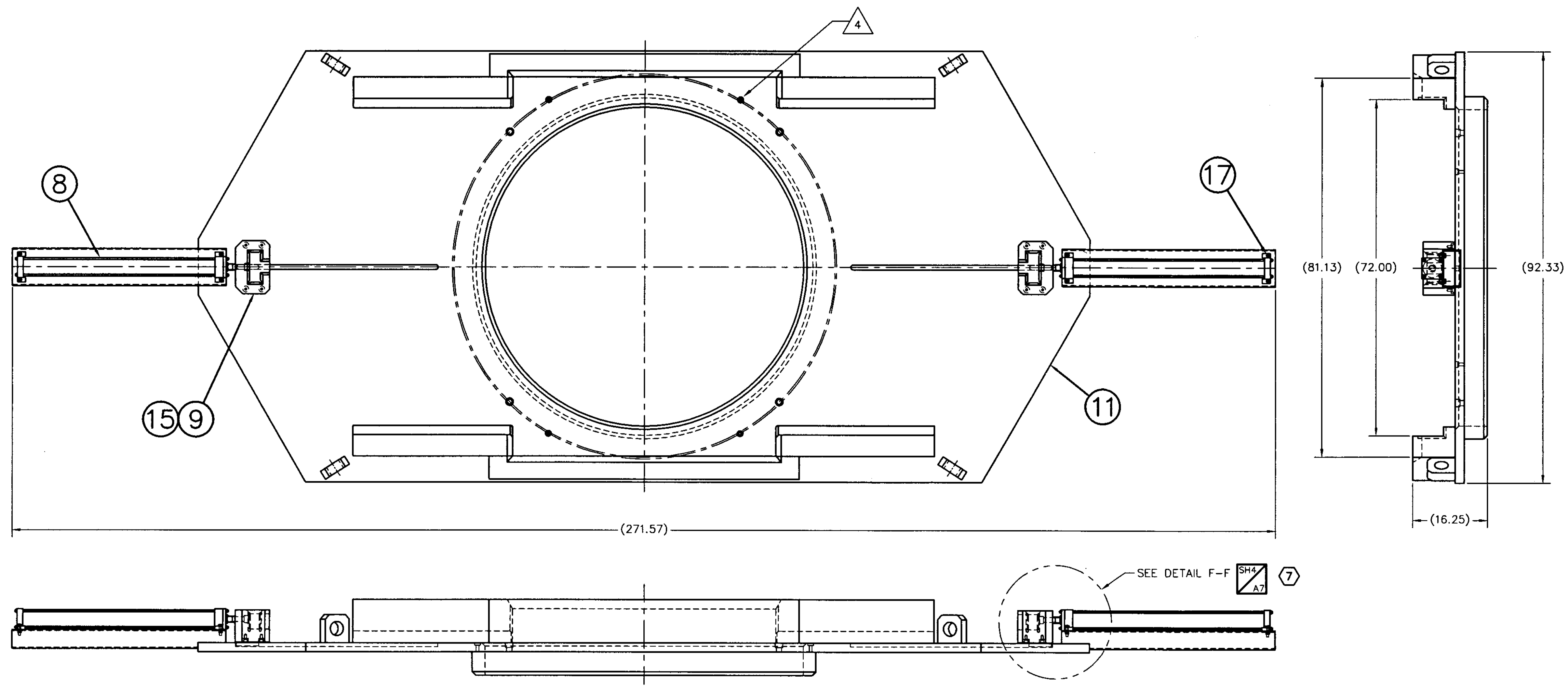
CANISTER / BASKET ASSEMBLY TABLE NAC-UMS®

PROJECT 790 | DRAWING 501 | REV 3

SCALE NONE | EST.WT. | SH 1 OF 1

R 152693

DCR No.	REVISION
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A,2B
4	INC DCR 3A
5	INC DCR 4A,4B,4C,4D,4E
6	INC DCR 5A,5B
7	INC DCR 6A



99 ADAPTER ASSEMBLY
WT: 12,210#

95	96	97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
				A/R	20	SET SCREW (OPTIONAL)	ALLOY STEEL	COML		SIZE AND LENGTH BY FIELD
				A/R	19	CYLINDER STOP (OPTIONAL)	CARBON STEEL	COML		2 DIA SCH 40 PIPE (LENGTH BY FIELD)
			4		18	GUIDE SEGMENT	CARBON STEEL	ASTM A36		PLATE/BAR
				8	17	CYLINDER BOLT	CARBON STEEL	COML		1/2-13 UNC-2A X 2 HEX HD BOLT
					16	(DELETED)				
				2	15	CONNECTOR BODY BOLT	CARBON STEEL	COML		1-14 UNS-3A X 3 1/2 SHCS
		4			14	WEAR PAD BOLT	CARBON STEEL	COML		1/2-13 UNC-3A X 1 SHCS
		4			13	WEAR PAD	BEARING BRONZE			1 1/4 DIA BAR
		1			12	CONNECTOR BODY	CARBON STEEL	ASTM A36		PLATE/BAR
				1	11	ADAPTER WELDMENT			790-559-98	
			8		10	CYLINDER NUT	CARBON STEEL	COML		1/2-13 UNC HEX NUT
				2	9	CONNECTOR ASSY			790-559-97	
				2	8	DOOR CYLINDER	STEEL	COML		PARKER #4"C-3LU19A40"
			4		7	LIFT LUG	CARBON STEEL	ASTM A36		2 PLATE
			2		6	SUPPORT	CARBON STEEL	ASTM A500/GRADE B		8 X 4 RECT. TUBE W/ 1/4 WALL
			2		5	SIDE SHIELD	CARBON STEEL	ASTM A36		PLATE/BAR
			2		4	DOOR RAIL	CARBON STEEL	ASTM A36		PLATE/BAR
			2		3	DOOR RAIL	CARBON STEEL	ASTM A36		PLATE/BAR
			1		2	LOCATING RING	CARBON STEEL	ASTM A36		PLATE/BAR
			1		1	BASE PLATE	CARBON STEEL	ASTM A36		2 PLATE
ASSY	ASSY	ASSY	ASSY	ASSY						
QUANTITY										

7. GRIND TOP SURFACE OF ITEM 1, AS REQUIRED, FOR PROPER FIT-UP AND FUNCTION OF TRANSFER CASK DOORS. PRIMARY FIT-UP REQUIREMENT IS TRANSFER CASK DOOR RAILS SHOULD BE WITHIN 1/16" OF ITEMS 3 AND 4 WHEN THE TRANSFER CASK IS ENGAGED WITH THE ADAPTER. SECONDARY FIT-UP REQUIREMENT IS TO ACHIEVE A NOMINAL 1/16" GAP BETWEEN THE TRANSFER CASK DOORS AND ITEM 1 UPON OPERATION.

6 (triangle) OPTIONAL MODIFICATION TO EXISTING ASSEMBLIES, INSTALL 4X ITEM 18 EQUALLY SPACED AS SHOWN IN DETAIL C-C.

5. OPTIONAL: SEAL WELD ALL OPEN SEAMS FOR ITEMS 3, 4 AND 5.

4 (triangle) THE .60 HOLES AT ZONE F4, ARE OPTIONAL, AS IS ATTACHMENT HARDWARE BETWEEN THE VCC AND THE TRANSFER ADAPTER.

3. REMOVE ANY OIL AND GREASE FROM ALL SURFACES IN ACCORDANCE WITH SSPC-SP 1. COMMERCIAL BLAST CLEAN PER SSPC-SP6. APPLY CARBOLINE 890 COATING OR KEELER & LONG E-SERIES EPOXY ENAMEL PER MANUFACTURERS APPLICATION INSTRUCTIONS.

2. MAG. PARTICLE (MT) ALL SHOP WELDS PER ASME SECTION V, ARTICLE 7. ACCEPTANCE PER ASME ASME SECTION VIII, APPENDIX 6.

1. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH AWS D1.1 OR ASME SECT IX.

NOTES:

8. TOUCH UP PAINTING MAY USE COMMERCIALLY AVAILABLE RUST INHIBITING ENAMEL AND BE APPLIED IN ACCORDANCE WITH SITE APPROVED PROCEDURES.

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3-12		6-18	±.06
	ANGULARITY	OVER 12		OVER 18	±.09
	PERPENDICULARITY	.X	±.1	ANGLES ±.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .01 - .03			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .01 - .03			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

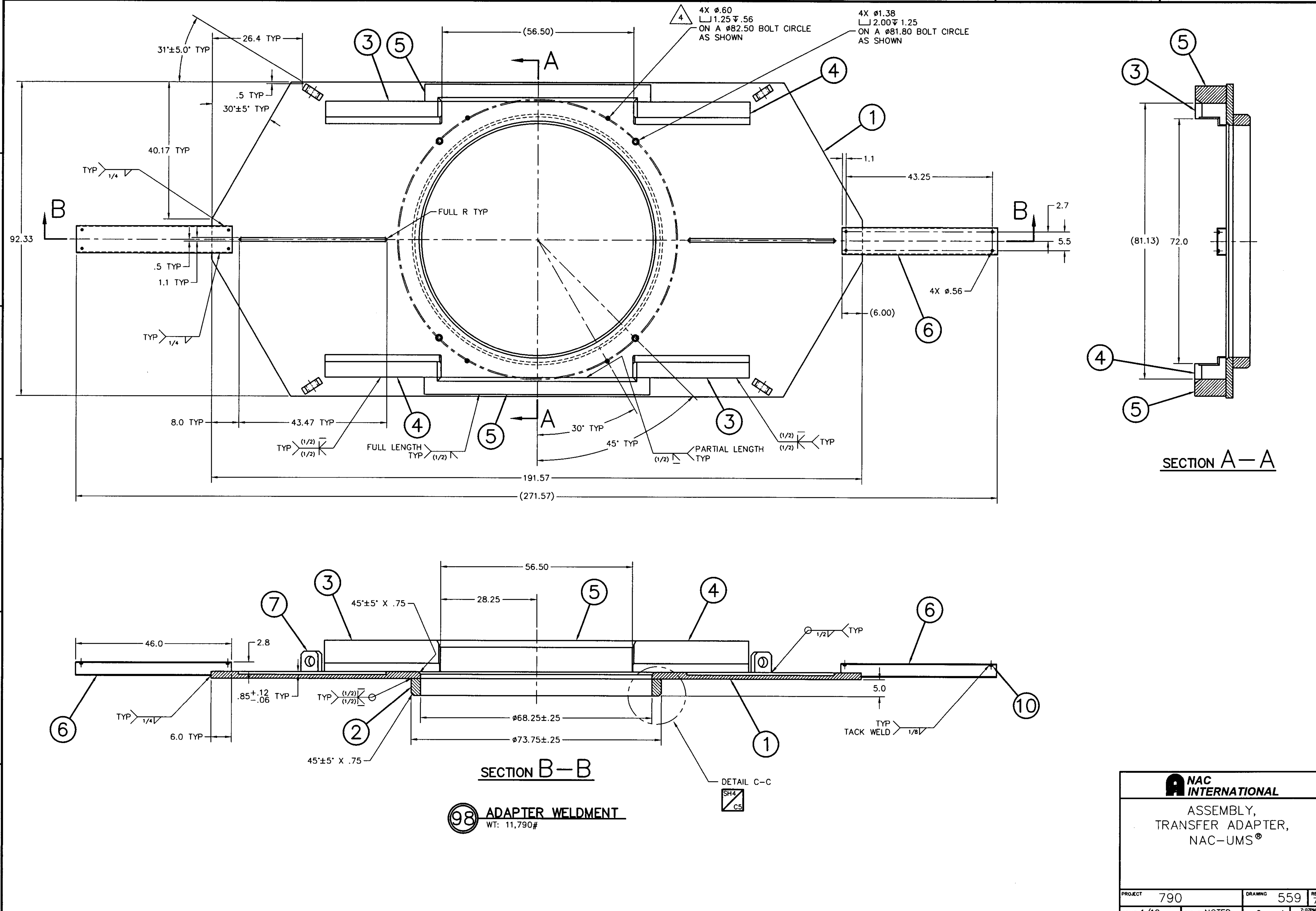
NEXT ASSEMBLY:
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	6-5-03
CHECKER	R. Walker	6-9-03
PROJECT MANAGER	JRBoyer for JB	6-10-03
DIRECTOR DESIGN AND ANALYSIS	Thomas [signature]	6-10-03
DIRECTOR, LICENSING	Anthony L Patko for TCT	6-11-03
VICE PRESIDENT, QUALITY	BF Barrett for RHS	6-11-03

NAC INTERNATIONAL

ASSEMBLY, TRANSFER ADAPTER, NAC-UMS®

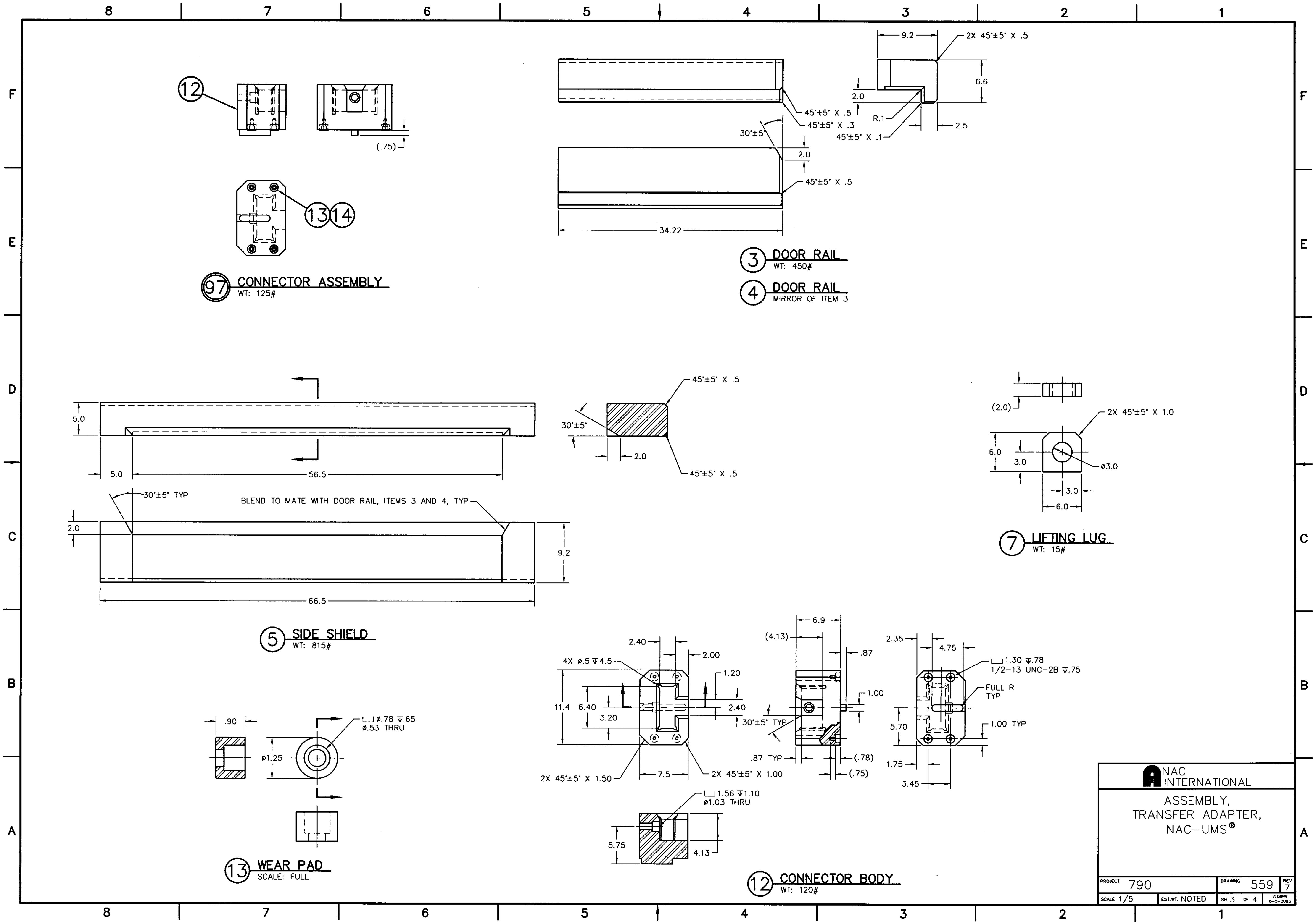
PROJECT 790 | DRAWING 559 | REV 7
SCALE 1/10 | EST.WT. NOTED | SH 1 OF 4 | 7:07PM 6-5-2003



NAC INTERNATIONAL

ASSEMBLY, TRANSFER ADAPTER, NAC-UMS®

PROJECT 790	DRAWING 559	REV 7
SCALE 1/10	EST.WT. NOTED	SH 2 OF 4



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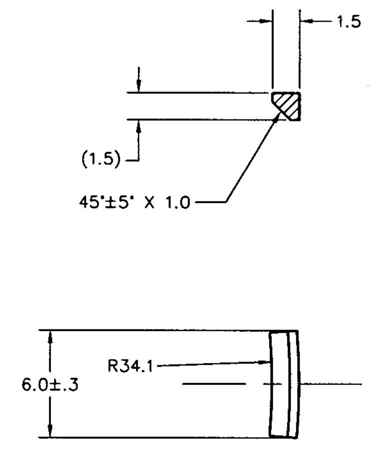
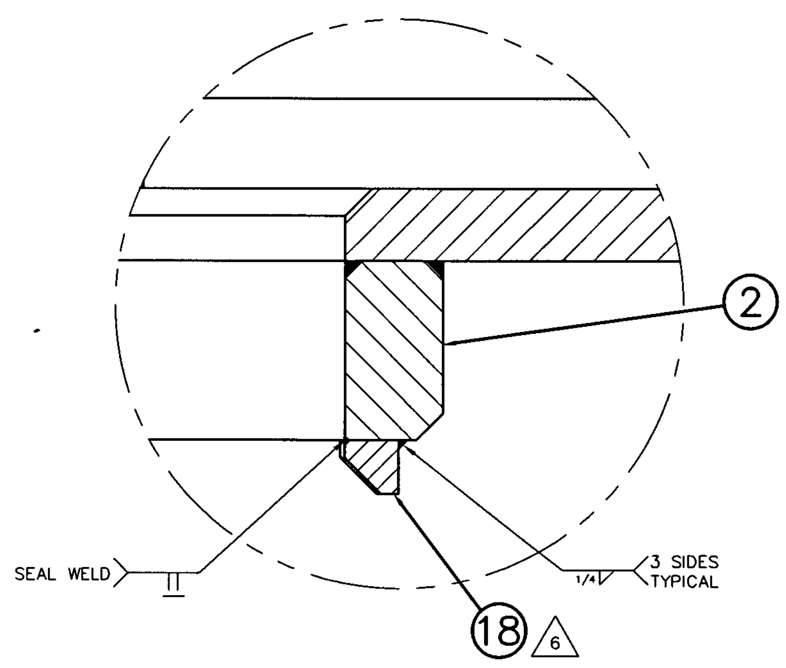
ASSEMBLY,
TRANSFER ADAPTER,
NAC-UMS®

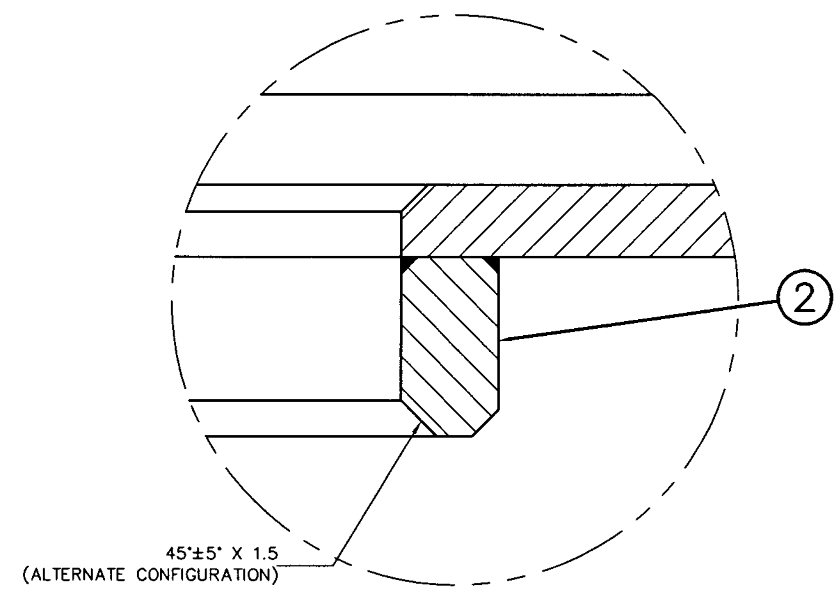
PROJECT 790		DRAWING 559	REV 7
SCALE 1/5	EST.WT. NOTED	SH 3 OF 4	7:08PM 6-5-2003



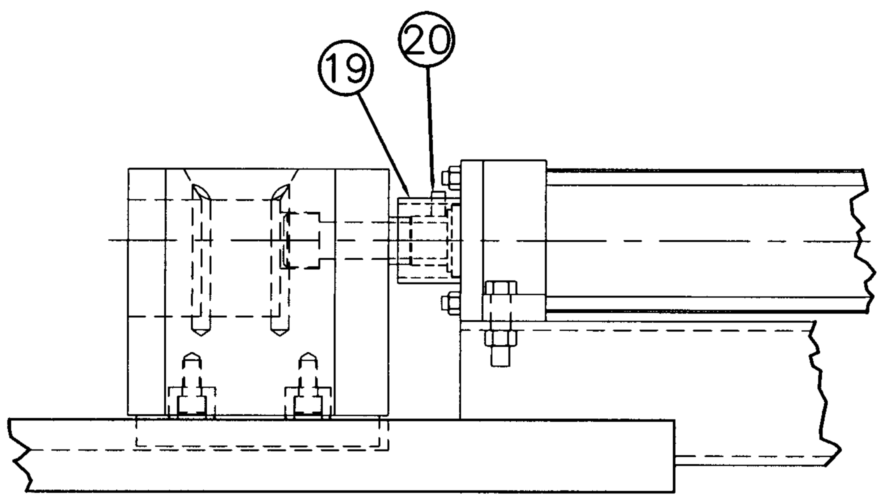
18 GUIDE SEGMENT
SCALE: 1/1



6 SH2 A4 DETAIL C—C
(OPTIONAL MODIFICATION TO EXISTING ASSEMBLIES)



SH2 A4 DETAIL C—C
(ALTERNATE CONFIGURATION, NEW FABRICATION)



7 DETAIL F—F SH1 D3
SCALE: 2/5
(OPTIONAL ASSEMBLY)
TYPICAL 2 PLACES

NAC INTERNATIONAL

ASSEMBLY,
TRANSFER ADAPTER,
NAC-UMS®

PROJECT 790		DRAWING 559	REV 7
SCALE 1/4	EST.WT. NOTED	SH 4 OF 4	7:08PM 6-5-2003

R-202406

REV	CHANGE
0	INITIAL ISSUE
1	INC. DCR 0A,0B,0C
2	INC. DCR 1A
3	INC. DCR 2A
4	INC. DCR 3A
5	INC. DCR 4A
6	INC. DCR 5A
7	INC. DCR 6A,6B
8	INC. DCR 7A,7B,7C,7D,7E
9	INC. DCR 8A,8B
10	INC. DCR 9A,9B,9C,9D,9E
11	INC. DCR 10A
12	INC. DCR 11A,11B,11C,11D
13	INC. DCR 12A
14	INC DCR 13A,13B
15	INC DCR 14A
16	INC DCR 15A
17	INC DCR 16A,16B

NOTES:

1. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH AWS D1.1 OR ASME SECT. IX.

2. VISUALLY INSPECT (VT) ALL WELDS IN ACCORDANCE WITH ASME SECTION V, ARTICLE 9. ACCEPTANCE PER ASME SECTION III, NG-5360. AFTER LOAD TESTING, MAG. PARTICLE TEST (MT) ALL ACCESSIBLE LOAD BEARING WELDS IN ACCORDANCE WITH ASME SECTION V, ARTICLE 7. ACCEPTANCE PER ASME SECTION III, NF-5340.

3. BACKING BARS ARE OPTIONAL.

4. THE TRUNNIONS (ITEM 12) SHALL BE FLUSH WITH THE INSIDE DIAMETER OF THE INNER SHELL (ITEMS 2, 3, 4, 5 OR 6).

5. THE TRUNNIONS (ITEM 12) SHALL BE COAXIAL TO EACH OTHER WITHIN .100.

6. (Δ) TRANSFER DRILL LOCATIONS FOR ITEM 19, DOOR LOCK BOLTS, IN ITEMS 17 AND 18, SHIELD DOORS.

7. REMOVE ANY OIL AND/OR GREASE FROM ALL SURFACES IN ACCORDANCE WITH SSPC-SP 1. COMMERCIAL BLAST CLEAN PER SSPC-SP 10. APPLY CARBOLINE 890 OR KEELER & LONG E-SERIES EPOXY ENAMEL COATING PER MANUFACTURERS APPLICATION INSTRUCTIONS.

8. THE MATING SURFACES BETWEEN THE DOOR RAILS AND THE SHIELD DOORS SHALL BE COATED WITH A SPENT FUEL POOL COMPATIBLE LUBRICANT, SUCH AS NEOLUBE.

9. (Δ) TYPICAL FOR SEAM AND GIRTH WELDS, NUMBER AND LOCATION OPTIONAL. SEAM WELDS SHALL BE OFFSET.

10. (Δ) MATCH DRILL LOCATIONS FROM ITEM 20. TAP 3/4-10 UNC-2B X 1.5 DEEP.

11. DELETED

12. (Δ) 1/4" MAX GAP AT THE TOP OF NEUTRON SHIELDING. NEUTRON SHIELD TO CONTAIN 0.6 WEIGHT PERCENT B_4C MIN.

13. (Δ) FOR 92 ASSEMBLY, FASTEN ITEM 41, TRANSFER CASK EXT. TO CASK USING ITEM 42, TRANSFER ADAPTER SHCS. BOLT ITEM 20, RETAINING RING, TO TRANSFER ADAPTER USING ITEM 38, RETAINING RING BOLT.

14. (Δ) MACHINE ID OF ITEM 43, SHIELDING RING, FOR SLIP FIT W/INNER SHELL. MACHINE THICKNESS TO ALLOW FOR UP TO .3 OF CRUSH IN LEAD AND PRESS UNTIL FLUSH WITH TOP OF INNER SHELL PRIOR TO WELDING. MACHINE OD TO 77.1. ITEM 43 MAY BE USED AS THE BACKING BAR FOR UPPER, INNER SHELL WELD.

15. (Δ) STEEL STAMP/ENGRAVE AS SHOWN WITH 2.0 HIGH LETTERS APPROXIMATELY .03 DEEP AND FILL WITH BLACK WEATHER RESISTANT PAINT. IF ITEM 21 COMPONENTS ARE USED, CHANGE TEXT TO READ "ADVANCED TRANSFER CASK", AND ADD TEXT TO NOTE "MAX. ALLOWABLE CANISTER WEIGHT: 98,000 LBS.". ACTUAL ASSEMBLED WEIGHT TO BE INSERTED IN PLACE OF XX,XXX AT TIME OF FABRICATION. ALTERNATIVELY, INFORMATION MAY BE STEEL STAMPED/ENGRAVED ONTO ITEM 25 AND SEAL WELDED ALL AROUND TO THE OUTER SHELL.

16. (Δ) CONTROL OVERALL HEIGHT AS SHOWN. PLACE A ROW OF FLAT BOTTOM BRICKS IN THE CAVITY FIRST AND A ROW OF BRICKS WITH FLAT TOPS IN THE CAVITY AS THE LAST ROW.

17. COMMERCIAL GRADE LEAD WOOL MAY BE USED TO FILL OPEN SPACES AT THE TRUNNIONS AS BEST POSSIBLE.

18. (Δ) GRIND TRANSITION CHAMFERS ON THE LEADING AND TRAILING EDGES OF THE TRANSFER CASK DOOR RAIL (ITEM 16) 1/8" X 30-45 DEGREES.

19. (Δ) GRIND TRANSITION CHAMFERS ON THE LEADING AND TRAILING EDGES OF THE 2.3" X 2.0" SHOULDER 1/8" X 30-45 DEGREES.

20. (Δ) LOCATE AT ASSEMBLY, WITH DOORS CENTERED, FULLY CLOSED AND WITH THE DOWEL PIN IN CONTACT WITH THE TRANSFER CASK BASE/OUTER SHELL. MAINTAIN A .25" MINIMUM OUTER EDGE DISTANCE. DRILL AND REAM FOR PRESS-FIT 1.0 DEEP INTO ITEM 29.

21. (Δ) IF ITEM 21 COMPONENTS ARE INSTALLED, THE MAXIMUM CANISTER ALLOWABLE WEIGHT IS 98,000 LBS. IF ITEM 21 COMPONENTS ARE NOT INSTALLED, THE MAXIMUM CANISTER ALLOWABLE WEIGHT IS 88,000 LBS.

22. (Δ) STAMP/ENGRAVE HEAD OF BOLT TO INDICATE 17-4 PH MATERIAL.

23. (Δ) AT THE USERS OPTION EITHER: FOUR (4) OF ITEM 19, OR TWO (2) OF ITEM 47 (ONE (1) IN EACH DOOR) OR TWO (2) OF ITEM 48 (ONE (1) FOR EACH DOOR) SHALL BE USED. IF ITEMS 19 AND 47 ARE NOT INVOKED OPTIONS, THE HOLES FOR THEIR IMPLEMENTATION MAY BE OMITTED.

24. FOR INNER AND OUTER SHELL (ITEMS 2-6, 7-11) AREAS ADJACENT TO SEAM, GIRTH, TOP PLATE (ITEM 15) AND BASE PLATE (ITEM 1) WELDS, AN ADDITIONAL .015" BELOW ASTM MATERIAL SPECIFICATION MINIMUM IS ALLOWED. AREAS ADJACENT TO THE TRUNNION WELDS SHALL MEET THE ASTM MATERIAL SPECIFICATION.

25. (Δ) ENDS OF ITEM 16 DOOR RAIL MAY BE SEAL WELDED TO ITEM 1 BASE PLATE.

26. AT THE OPTION OF THE USER, THE DOWEL PINS (ITEM 46) MAY BE SEAL WELDED TO THE SHIELD DOORS (ITEMS 17 AND 18).

27. AT THE OPTION OF THE USER, THE DOWEL PINS (ITEM 46) MAY BE REMOVED AND THE DRILLED HOLES WELDED CLOSED. THE WELDING OPERATION SHALL BE DOCUMENTED AS A WELD REPAIR TO THE SHIELD DOORS (ITEMS 17 AND 18).

(17) 28. (Δ) ITEMS 49, 50, 51 AND 52 ARE OPTIONAL COMPONENTS.

91 ASSY	92 ASSY	93 ASSY	94 ASSY	95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
	A/R	A/R							52	LIFT PLATE B	CARBON STEEL	COML		PLATE/BAR
	A/R	A/R							51	LIFT PLATE A	CARBON STEEL	COML		PLATE/BAR
	A/R	A/R							50	DOOR PLUG	ST. STL.	COML		PLATE/BAR
				A/R	A/R	A/R	A/R	A/R	49	WEAR STRIP	NITRONIC 30	COML		SHEET/STRIP
				2	2	2	2	2	48	DOOR STOP			790-617-99	
				A/R	A/R	A/R	A/R	A/R	47	DOOR LOCK BOLT	17-4 PH	ASTM A564		BAR
				4	4	4	4	4	46	DOWEL PIN	ST. STL.	COML		DIA 5/8 X 2 LG.
1									45	FILL/DRAIN LINE PIPE	304 ST. STL.	ASTM A312		1 IN. SCH 40 PIPE
1									44	FILL/DRAIN LINE PLATE	304 ST. STL.	ASTM A240		11 GAUGE SHEET
				1	1	1	1	1	43	SHIELDING RING	CARBON STEEL	ASTM A36/A105/A516 GR.70		PLATE
	32								42	TRANSFER ADAPTER SHCS	HIGH ALLOY STEEL	ASTM A193 GR. B6		3/4-10 UNC-3A 9.0 LG
	1								41	TRANSFER CASK EXTENSION	CARBON STEEL	ASTM A516 GR. 70		PLATE
	1								40	ASSEMBLY, TRANSFER CASK			790-560-99	
		1	1						39	CONNECTOR	LOW ALLOY STEEL	ASTM A588		4 PLATE
				32	32	32	32	32	38	RETAINING RING BOLT	HIGH ALLOY STEEL	ASTM A193 GR. B6		3/4-10 UNC-2A 2.0 LG
				4	4	4	4	4	37	SCUFF PLATE	304 ST. STL.	ASTM A240		11 GAUGE SHEET
				3760	3720	3640	3600	3440	36	GAMMA SHIELD BRICK	LEAD (PB)	ASTM B29		CHEMICAL COPPER GRADE
				10	10	10	10	10	35	FILL/DRAIN LINE			790-560-91	
		1							34	NS COVER PLATE B	LOW ALLOY STEEL	ASTM A588		1/2 PLATE
			1						33	NS COVER PLATE A	LOW ALLOY STEEL	ASTM A588		1/2 PLATE
		2							32	NS BOUNDARY	LOW ALLOY STEEL	ASTM A588		1 1/2 X 2 RECT. BAR
			2						31	NS BOUNDARY	LOW ALLOY STEEL	ASTM A588		1 1/2 X 2 RECT. BAR
		1	1						30	NS BOUNDARY	LOW ALLOY STEEL	ASTM A588		1 1/2 X 2 RECT. BAR
		2	2						29	NS BOUNDARY	LOW ALLOY STEEL	ASTM A588		1 1/2 X 2 RECT. BAR
		1	1						28	NS BOUNDARY	LOW ALLOY STEEL	ASTM A588		1 X 2 RECT. BAR
		1							27	BOTTOM PLATE B	LOW ALLOY STEEL	A350 LF2		PLATE
			1						26	BOTTOM PLATE A	LOW ALLOY STEEL	A350 LF2		PLATE
				1	1	1	1	1	25	NAMEPLATE	ST. STL.	COML		11-16 GAUGE SHEET
				A/R	A/R	A/R	A/R	A/R	24	PAINT		COML		SEE NOTE 15
				A/R	A/R	A/R	A/R	A/R	23	LEAD WOOL	Pb	COML		SEE NOTE 17
				A/R	A/R	A/R	A/R	A/R	22	COATING SYSTEM		COML		SEE NOTE 7
				4	4	4	4	4	21	SUPPORT PLATE	LOW ALLOY STEEL	ASTM A588		3/4 PLATE
				1	1	1	1	1	20	RETAINING RING	LOW ALLOY STEEL	ASTM A588		3/4 PLATE
				A/R	A/R	A/R	A/R	A/R	19	DOOR LOCK BOLT	304 ST. STL.	ASTM A276		1 1/8 HEX BAR
				1	1	1	1	1	18	SHIELD DOOR B			790-560-93	
				1	1	1	1	1	17	SHIELD DOOR A			790-560-94	
				2	2	2	2	2	16	DOOR RAIL	LOW ALLOY STEEL	A350 LF2		BAR
				1	1	1	1	1	15	TOP PLATE	LOW ALLOY STEEL	ASTM A588		PLATE
		A/R	A/R	A/R	A/R	A/R	A/R	A/R	14	NEUTRON SHIELD	NS-4-FR	COML		
				4	4	4	4	4	13	TRUNNION CAP	LOW ALLOY STEEL	ASTM A588		3/8 PLATE
				4	4	4	4	4	12	TRUNNION	LOW ALLOY STEEL	A350 LF2		ROUND BAR
				1					11	OUTER SHELL	LOW ALLOY STEEL	ASTM A588		1 1/4 PLATE
					1				10	OUTER SHELL	LOW ALLOY STEEL	ASTM A588		1 1/4 PLATE
						1			9	OUTER SHELL	LOW ALLOY STEEL	ASTM A588		1 1/4 PLATE
							1		8	OUTER SHELL	LOW ALLOY STEEL	ASTM A588		1 1/4 PLATE
								1	7	OUTER SHELL	LOW ALLOY STEEL	ASTM A588		1 1/4 PLATE
				1					6	INNER SHELL	LOW ALLOY STEEL	ASTM A588		3/4 PLATE
					1				5	INNER SHELL	LOW ALLOY STEEL	ASTM A588		3/4 PLATE
						1			4	INNER SHELL	LOW ALLOY STEEL	ASTM A588		3/4 PLATE
							1		3	INNER SHELL	LOW ALLOY STEEL	ASTM A588		3/4 PLATE
								1	2	INNER SHELL	LOW ALLOY STEEL	ASTM A588		3/4 PLATE
				1	1	1	1	1	1	BOTTOM PLATE	LOW ALLOY STEEL	ASTM A588		1 PLATE

QUANTITY

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3-12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6-18	±.03
	OVER 18	±.06
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY

ALL DIMENSIONS ARE IN INCHES

BORDER SIZE: F (40 X 28)

ALL UNSPECIFIED TOOL RADII: .015 - .030

BREAK ALL SHARP CORNERS .015 - .030

MACHINED SURFACES TO BE ▽ OR BETTER

NEXT ASSEMBLY: N/A

DRAWING TYPE: LICENSE

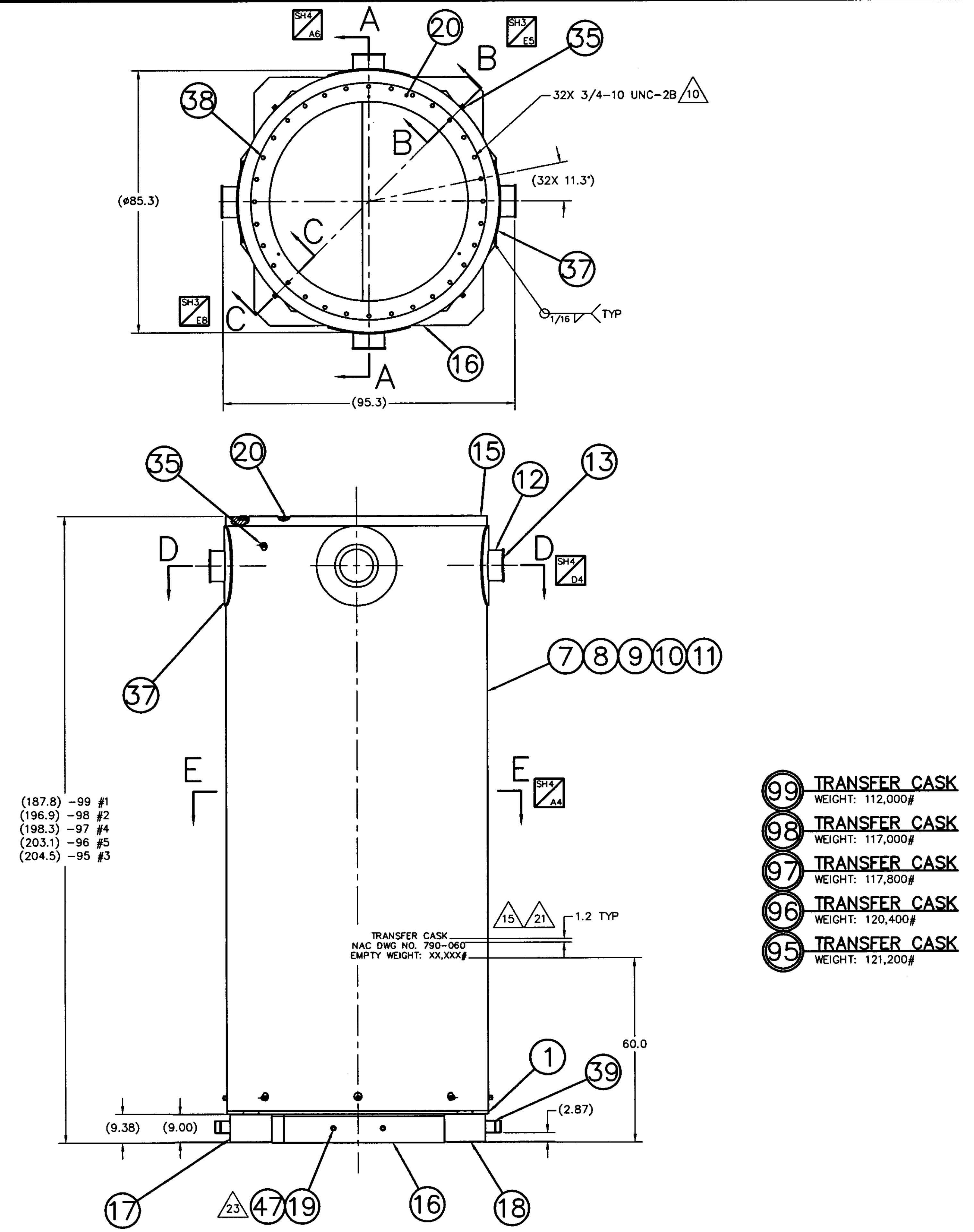
GROUP	NAME	DATE
PREPARER	R. Walker	6-25-04
CHECKER	[signature]	10-25-04
PROJECT MANAGER	[signature]	10-27-04
ENGINEERING	[signature]	10-29-04
LICENSING	[signature]	11/02/04
QUALITY	[signature]	11/03/04

NAC INTERNATIONAL

ASSEMBLY, STANDARD TRANSFER CASK (TFR) NAC-UMS®

PROJECT 790 | DRAWING 560 | REV 17

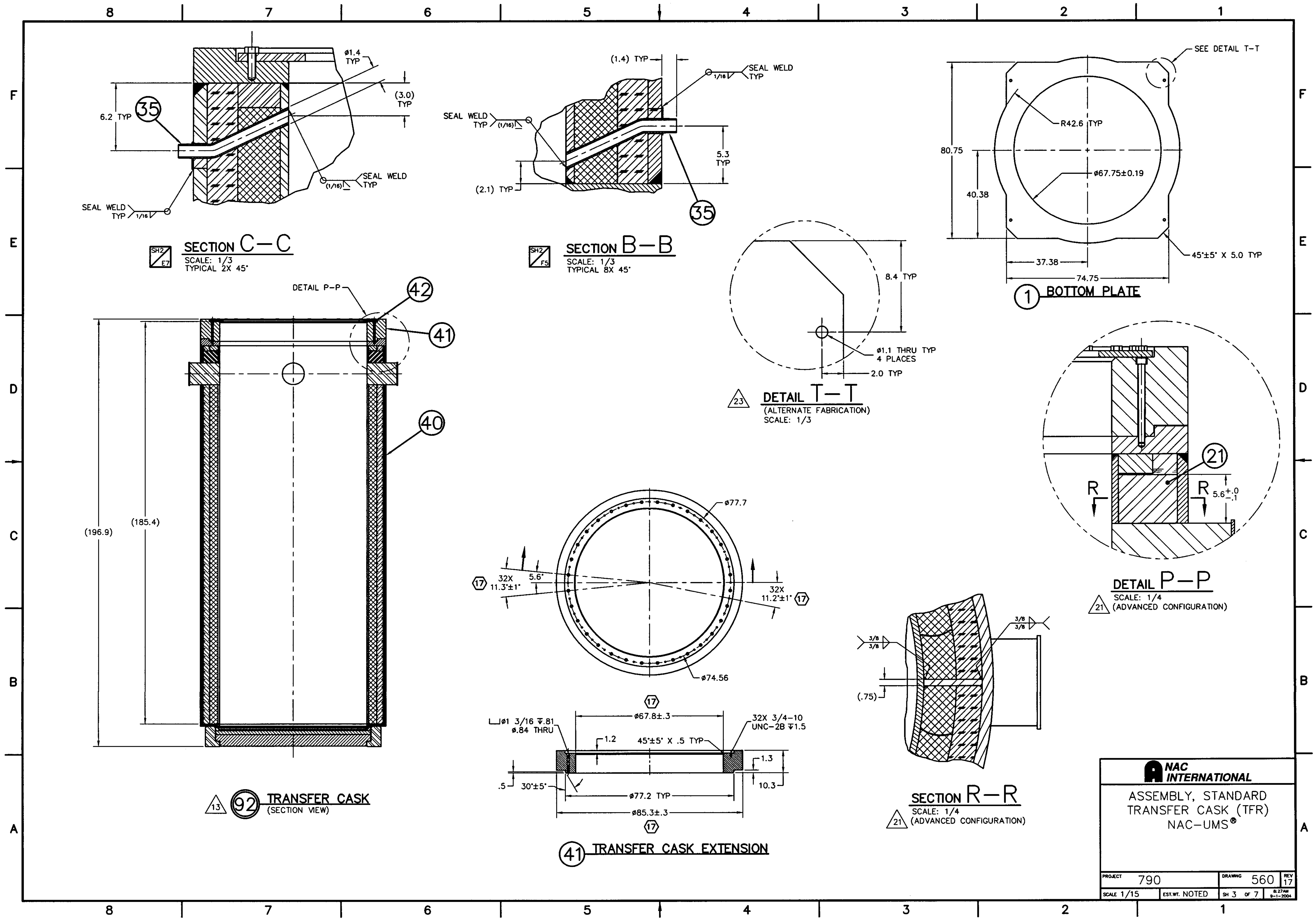
SCALE 1/1 | WEIGHT NOTED | SH 1 OF 7 | 4:04PM 10-22-2004

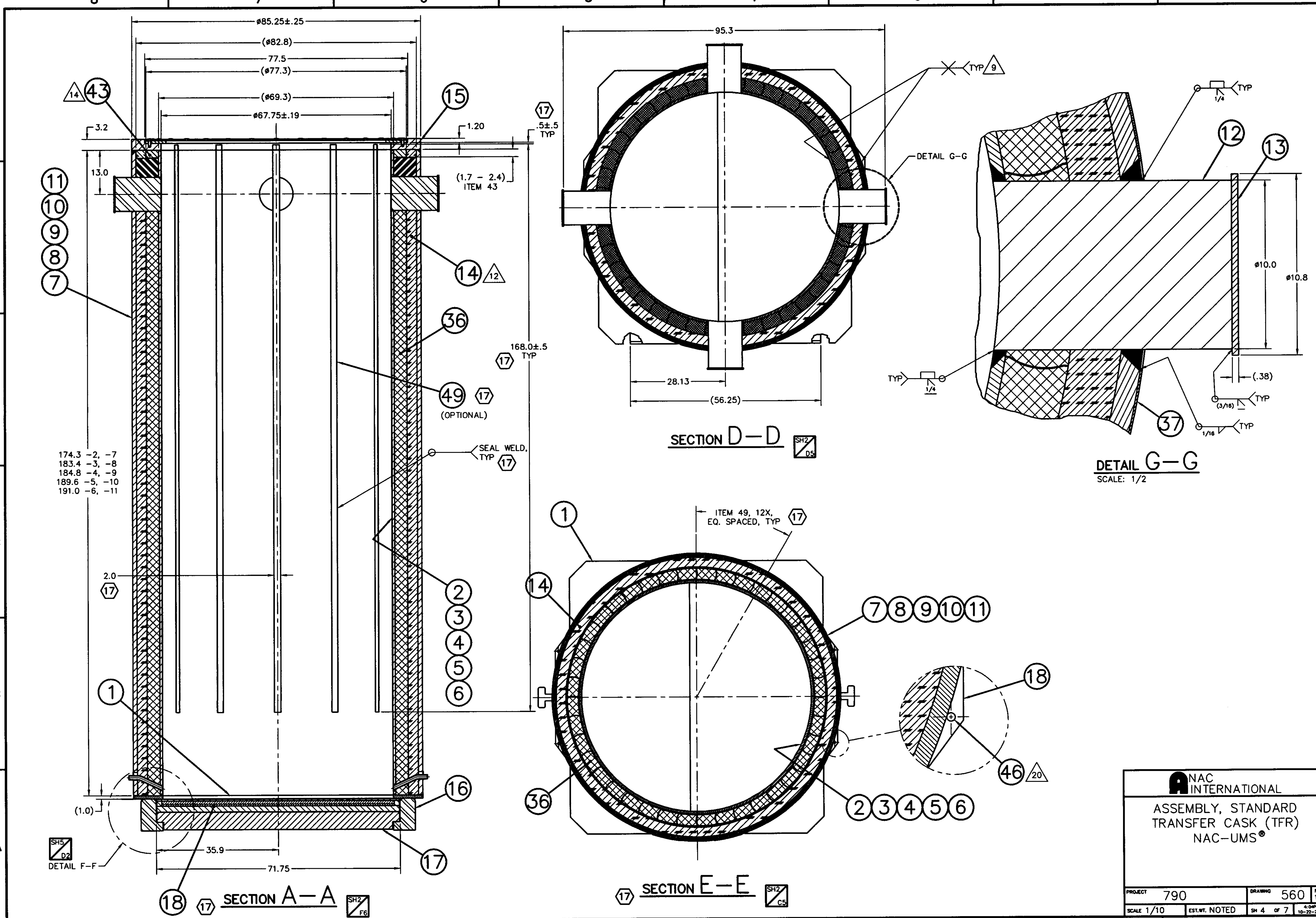


NAC INTERNATIONAL

ASSEMBLY, STANDARD
TRANSFER CASK (TFR)
NAC-UMS®

PROJECT 790	DRAWING 560	REV 17
SCALE 1/15	EST.WT. NOTED	SH 2 OF 7


SECTION C—C
SCALE: 1/3
TYPICAL 2X 45°
SECTION B—B
SCALE: 1/3
TYPICAL 8X 45°
SEAL WELD TYP
1 BOTTOM PLATE
R42.6 TYP
ø67.75±0.19
80.75
40.38
37.38
74.75
45°±5° X 5.0 TYP
SEE DETAIL T-T
DETAIL T—T
(ALTERNATE FABRICATION)
SCALE: 1/3
ø1.1 THRU TYP
4 PLACES
8.4 TYP
2.0 TYP
DETAIL P—P
SCALE: 1/4
(ADVANCED CONFIGURATION)
DETAIL P-P
92 TRANSFER CASK
(SECTION VIEW)
(196.9)
(185.4)
ø77.7
ø74.56
32X 11.3°±1°
32X 11.2°±1°
ø67.8±.3
ø77.2 TYP
ø85.3±.3
32X 3/4-10 UNC-2B ∓1.5
45°±5° X .5 TYP
30°±5°
41 TRANSFER CASK EXTENSION
SECTION R—R
SCALE: 1/4
(ADVANCED CONFIGURATION)
NAC INTERNATIONAL
ASSEMBLY, STANDARD
TRANSFER CASK (TFR)
NAC-UMS®
PROJECT 790
DRAWING 560
REV 17
SCALE 1/15
EST.WT. NOTED
SH 3 OF 7



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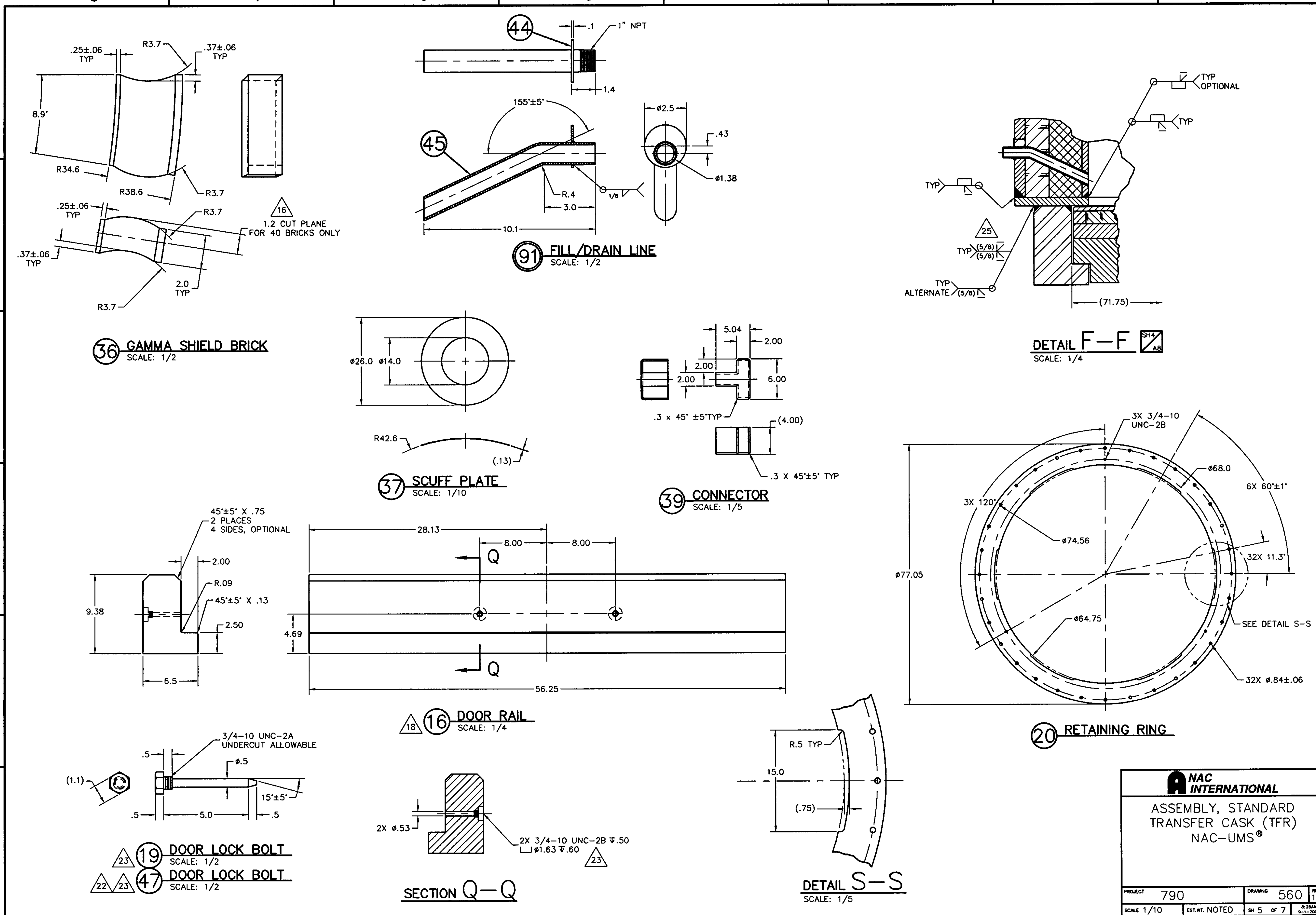
ASSEMBLY, STANDARD TRANSFER CASK (TFR) NAC–UMS®

PROJECT 790	DRAWING 560	REV 17
SCALE 1/10	EST.WT. NOTED	SH 4 OF 7

4:04PM 10-22-2004


36 GAMMA SHIELD BRICK
SCALE: 1/2
1.2 CUT PLANE FOR 40 BRICKS ONLY
91 FILL/DRAIN LINE
SCALE: 1/2
1" NPT
DETAIL F—F
SCALE: 1/4
TYP OPTIONAL
TYP ALTERNATE
37 SCUFF PLATE
SCALE: 1/10
39 CONNECTOR
SCALE: 1/5
.3 x 45° ±5°TYP
.3 X 45°±5° TYP
20 RETAINING RING
3X 3/4-10 UNC-2B
SEE DETAIL S-S
32X ø.84±.06
45°±5° X .75
2 PLACES
4 SIDES, OPTIONAL
16 DOOR RAIL
SCALE: 1/4
3/4-10 UNC-2A
UNDERCUT ALLOWABLE
19 DOOR LOCK BOLT
SCALE: 1/2
47 DOOR LOCK BOLT
SCALE: 1/2
SECTION Q—Q
2X 3/4-10 UNC-2B ↧.50
⌴ ø1.63 ↧.60
DETAIL S—S
SCALE: 1/5
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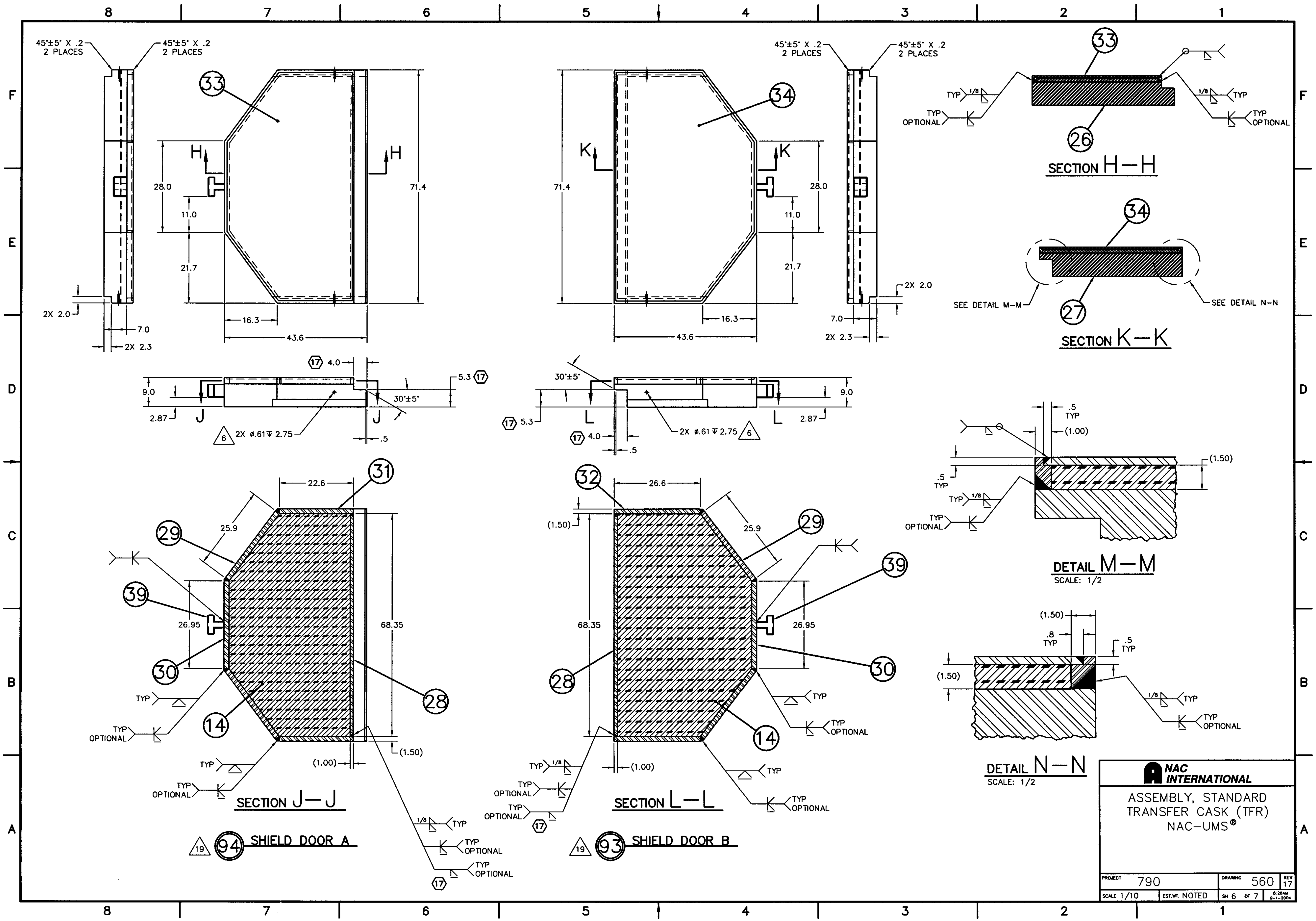
ASSEMBLY, STANDARD
TRANSFER CASK (TFR)
NAC-UMS®
PROJECT 790
DRAWING 560
REV 17
SCALE 1/10
EST.WT. NOTED
SH 5 OF 7


SECTION H—H
SECTION K—K
SEE DETAIL M—M
SEE DETAIL N—N
DETAIL M—M
SCALE: 1/2
DETAIL N—N
SCALE: 1/2
SECTION J—J
94 SHIELD DOOR A
SECTION L—L
93 SHIELD DOOR B
45°±5° X .2
2 PLACES
2X ø.61 ⊽ 2.75
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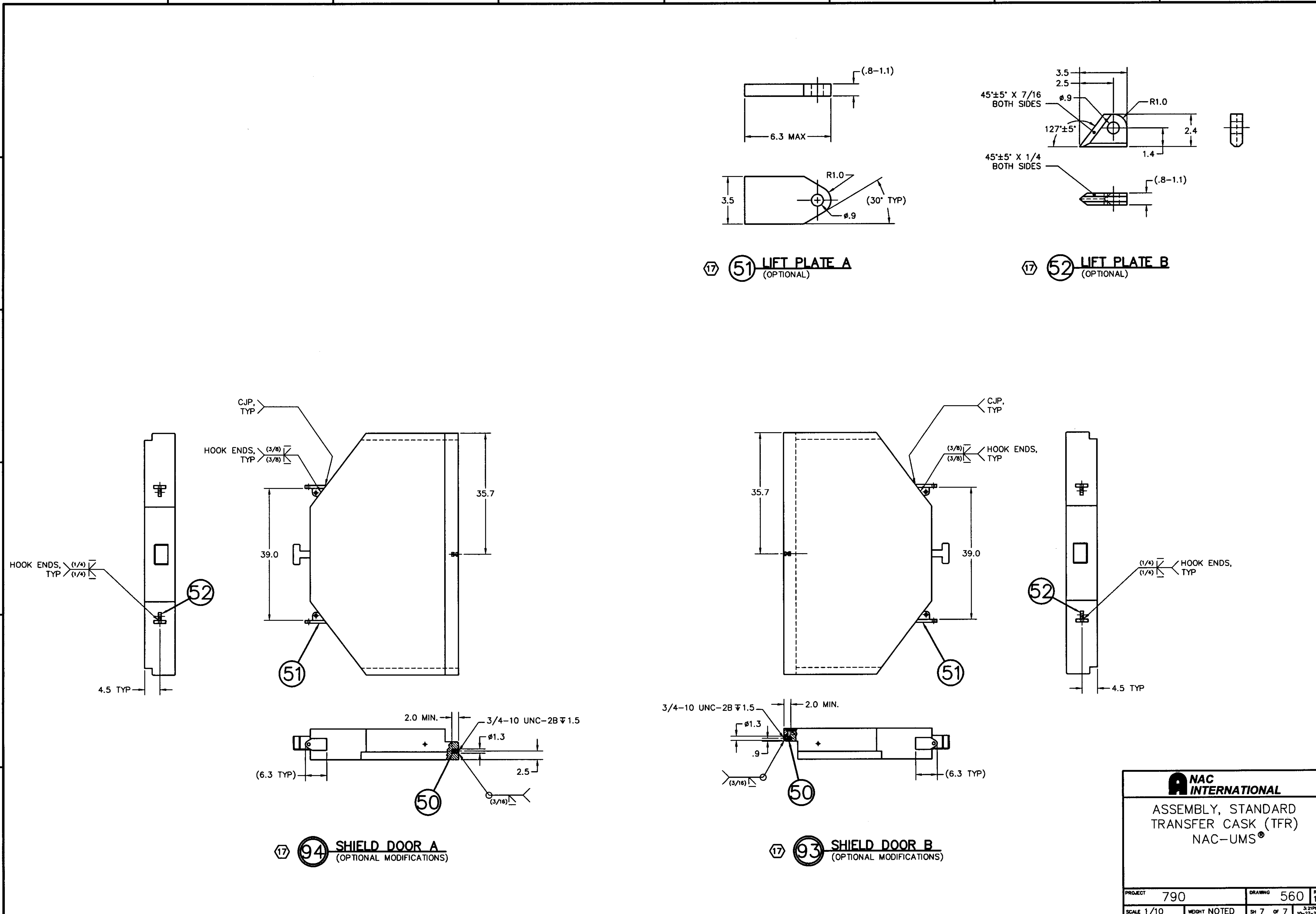
ASSEMBLY, STANDARD
TRANSFER CASK (TFR)
NAC-UMS®
PROJECT 790
DRAWING 560
REV 17
SCALE 1/10
EST.WT. NOTED
SH 6 OF 7

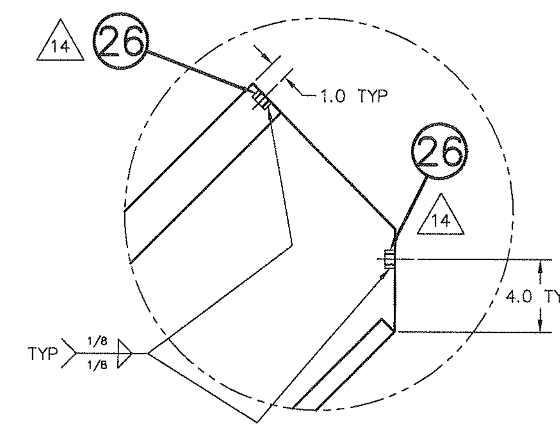


NAC INTERNATIONAL

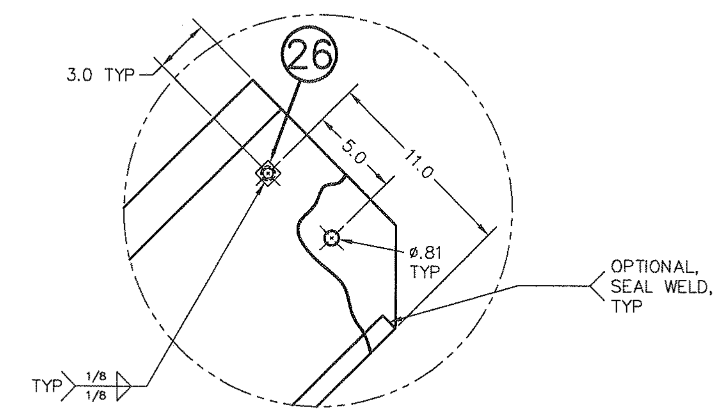
ASSEMBLY, STANDARD TRANSFER CASK (TFR) NAC–UMS®

PROJECT 790	DRAWING 560	REV 17
SCALE 1/10	WEIGHT NOTED	SH 7 OF 7	3:21PM 10-27-2004

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A,5C,5D,5E
7	INC DCR 6A
8	INC DCR 7A,7B,7C,7D
9	INC DCR 8A
10	INC DCR 9A,9B,9C,9D
11	INC DCR 10A,10B,10C,10D,10E
12	INC DCR 11A
13	INC DCR 12A
14	INC DCR 13A, 13B
15	INC DCR 14A



DETAIL E—E SH3 D5



DETAIL E—E SH3 D5
OPTIONAL CONFIGURATION
(NUTS ARE LOCATED ON CONCRETE SIDE/OUTSIDE OF INLET TUNNEL)

18. ASSEMBLY 94 SHOWS ALTERNATIVE METHOD FOR INSTALLATION OF THE COVER (ITEM 36) IN PLACE OF THAT SHOWN ON DRAWING #790-590.

17. ITEM# 12, MAY BE FABRICATED FROM MULTIPLE PLATES. SEAM(S) BETWEEN PLATE SECTIONS MUST BE FULL PENETRATION WELDED.

16. PIPE, TUBING OR SOLID BAR MAY BE USED FOR BOM ITEM 35. PIPE SHOULD BE ASTM A53, GR. B/ASTM A106, GR. B, 3 1/2 NOMINAL DIA., XXH. TUBE SHOULD BE ASTM A519, 4" X 5/8" WALL. SOLID BAR SHOULD BE 4" DIA., ASTM A36.

15. TOP FLANGE, MAY BE ROTATED 45° AT THE OPTION OF THE USER/FABRICATOR TO LOCATE 2X 1-1/2 HOLES ON SAME AXIS AS TWO OPPOSING LOWER INLET VENT CENTERLINES.

14. ALTERNATE DESIGN ELIMINATES NUTS, BILL OF MATERIAL ITEM 26.,

13. AT FABRICATOR OPTION, DELETE DETAIL C-C AND CORRESPONDING BILL OF MATERIAL ITEMS 4, 5, 6, 7, AND 8.

12. SHIM PLATES MAY BE UTILIZED TO FACILITATE FIELD WELDING OPERATIONS ON ONE OR BOTH SIDES OF THE SUPPLEMENTAL SHIELDING WELDMENT. LOCATE WELDS APPROX. AS SHOWN.

11. DELETED

10. DELETED

9. LOCATE FACE OF ITEM 26 FLUSH WITH 68" RADIUS PROFILE.

8. BAFFLE WELDMENT MAY BE FABRICATED USING MULTIPLE SECTIONS. ALL SEAMS BETWEEN SECTIONS MUST BE FULL PENETRATION WELDED UPON ASSEMBLY.

6. ITEMS MAY BE FABRICATED USING MULTIPLE SECTIONS. ALL SEAMS BETWEEN SECTIONS MUST BE SEAL WELDED UPON ASSEMBLY.

5. ALTERNATE FABRICATION FOR ITEMS 22/24 AND 18/23, MULTIPLE COMPONENTS MAY BE FORMED FROM ONE PIECE OF PLATE. 1 X THICKNESS MIN. I.D. BEND RADIUS.

4. TYPICAL FOR SEAM AND GIRTH WELDS, NUMBER AND LOCATION OPTIONAL. ADJACENT SECTIONS WITH SEAM WELDS SHALL BE OFFSET.

(14) 3. PREPARE ALL SURFACES IN ACCORDANCE WITH SSPC-SP1. PREPARE AND COAT ALL SURFACES WITH A HEAT-RESISTANT PRIMER PER MANUFACTURER'S APPLICATION INSTRUCTIONS. APPLY A HEAT RESISTANT COATING PER MANUFACTURER'S APPLICATION INSTRUCTIONS ON ALL SURFACES THAT WILL NOT BE COMPLETELY SHIELDED FROM THE ELEMENTS BY CONCRETE.

2. VISUAL (VT) INSPECT ALL WELDS.

1. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH AWS D1.1 OR ASME SECTION IX.

NOTES:

92 ASSY	93 ASSY	94 ASSY	95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
		3						37	DOWEL PIN	ST. STL.	COML		DIA 1/2 X 3/4 LG.
		1						36	COVER	304 ST. STL.	ASTM A240		1/4 PLATE x 67.5 DIA
		16						35	PIPE/TUBE/BAR (OPTIONAL)	CARBON STEEL	SEE NOTE 16		SEE NOTE 16
		4						34	SUPPLEMENTAL SHIELDING			790-613-99	
								33	DELETED				
A/R	A/R	A/R	A/R	A/R	A/R	A/R	A/R	32	COATINGS		COML		SEE NOTE 3
	2							31	LIFTING NUT	CARBON STEEL	COML		1-8 UNC-2B HEAVY HEX NUT
			1					30	SHELL	CARBON STEEL	ASTM A36		2 1/2 PLATE
				1				29	SHELL	CARBON STEEL	ASTM A36		2 1/2 PLATE
					1			28	SHELL	CARBON STEEL	ASTM A36		2 1/2 PLATE
						1		27	SHELL	CARBON STEEL	ASTM A36		2 1/2 PLATE
	2	16						26	SCREEN TAB	ST. STL.	COML		5/8-11 UNC-2B SQUARE NUT
2								25	BAFFLE	CARBON STEEL	ASTM A36		1/4 PLATE
	1							24	OUTLET BACK	CARBON STEEL	ASTM A36		3/8 PLATE
	2							23	OUTLET SIDE	CARBON STEEL	ASTM A36		3/8 PLATE
	1							22	OUTLET TOP	CARBON STEEL	ASTM A36		3/8 PLATE
	1							21	OUTLET BOTTOM	CARBON STEEL	ASTM A36		3/8 PLATE
	1							20	SHIELD PLATE	CARBON STEEL	ASTM A36		2 PLATE
	1							19	TOP	CARBON STEEL	ASTM A36		2 PLATE
	1							18	BOTTOM	CARBON STEEL	ASTM A36		3/8 PLATE
		56						17	NELSON STUD	CARBON STEEL	COML		3/4 X 6 3/16 S3L
		1						16	BASE PLATE	CARBON STEEL	ASTM A36		2 PLATE
		1						15	STAND	CARBON STEEL	ASTM A36		2 PLATE
		4						14	INLET TOP	CARBON STEEL	ASTM A36		2 1/4 PLATE
		8						13	INLET SIDE	CARBON STEEL	ASTM A36		3/8 PLATE
		1						12	BOTTOM	CARBON STEEL	ASTM A36		1 PLATE
		4						11	SHIELD RING	CARBON STEEL	ASTM A36		2 PLATE
		4						10	COVER	CARBON STEEL	ASTM A36		3/8 PLATE
		1						9	BAFFLE WELDMENT			790-561-92	
			4	4	4	4	4	8	JAM NUT	CARBON STEEL	COML		1-8UNC-2B
			4	4	4	4	4	7	JACK NUT	CARBON STEEL	COML		1-8UNC-2B
			4	4	4	4	4	6	JACK SCREW	CARBON STEEL	COML		1-8UNC-2A X 5 LG
			8	8	8	8	8	5	JACK GUSSET	CARBON STEEL	ASTM A36		3/8 PLATE
			4	4	4	4	4	4	JACK BASE	CARBON STEEL	ASTM A36		1 1/2 PLATE/BAR
			1	1	1	1	1	3	SUPPORT RING	CARBON STEEL	ASTM A36/A105		2 1/2 X 3 RECT. BAR/FORGING/PLATE
			1	1	1	1	1	2	TOP FLANGE	CARBON STEEL	ASTM A36		2 PLATE
							1	1	SHELL	CARBON STEEL	ASTM A36		2 1/2 PLATE

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

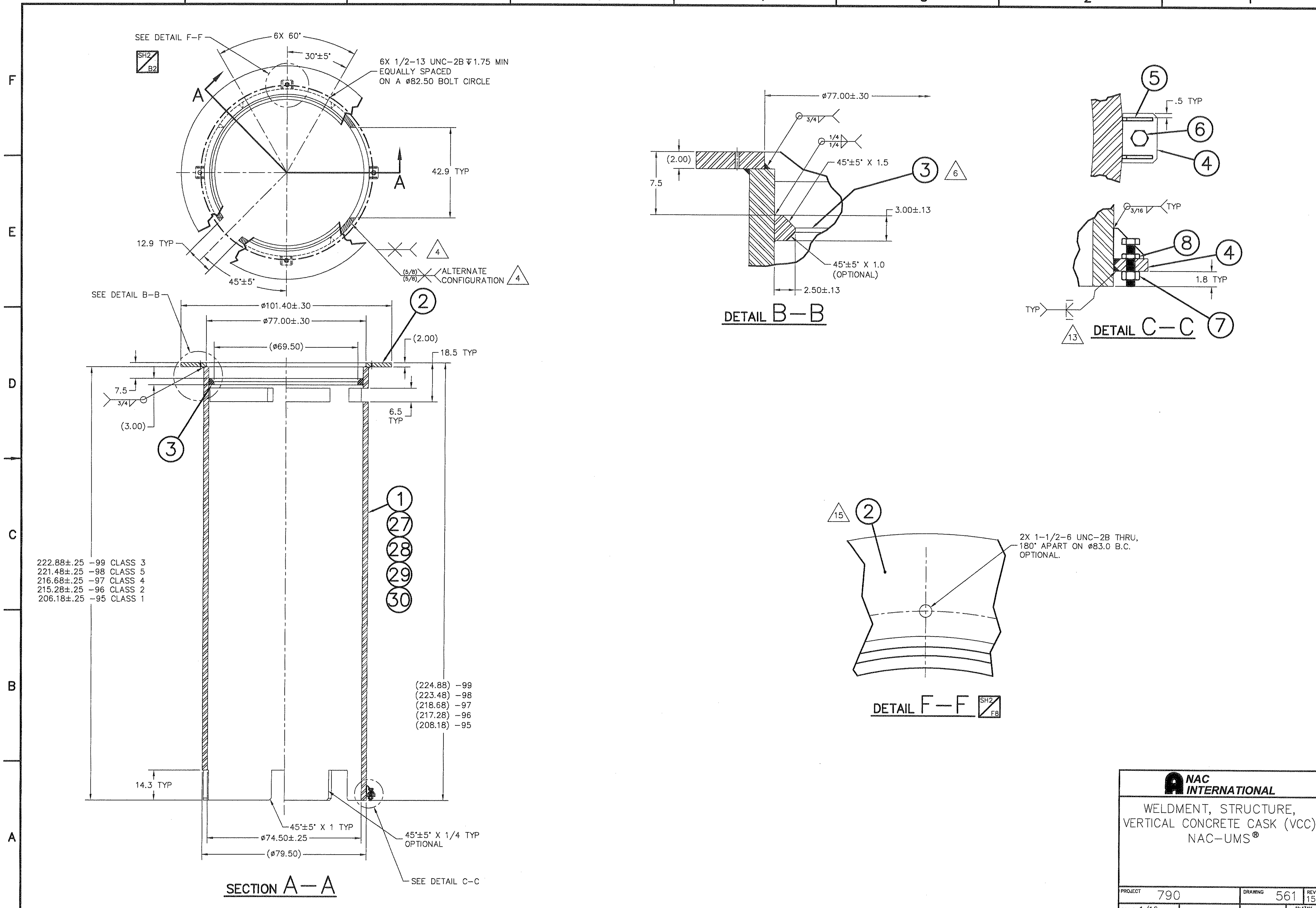
NEXT ASSEMBLY: 790-562
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	11/4/10
CHECKER	[signature]	11/04/10
PROJECT MANAGER	[signature]	11/4/10
ENGINEERING	[signature]	11/4/10
LICENSING	[signature]	11/4/10
QUALITY	[signature]	11/4/10

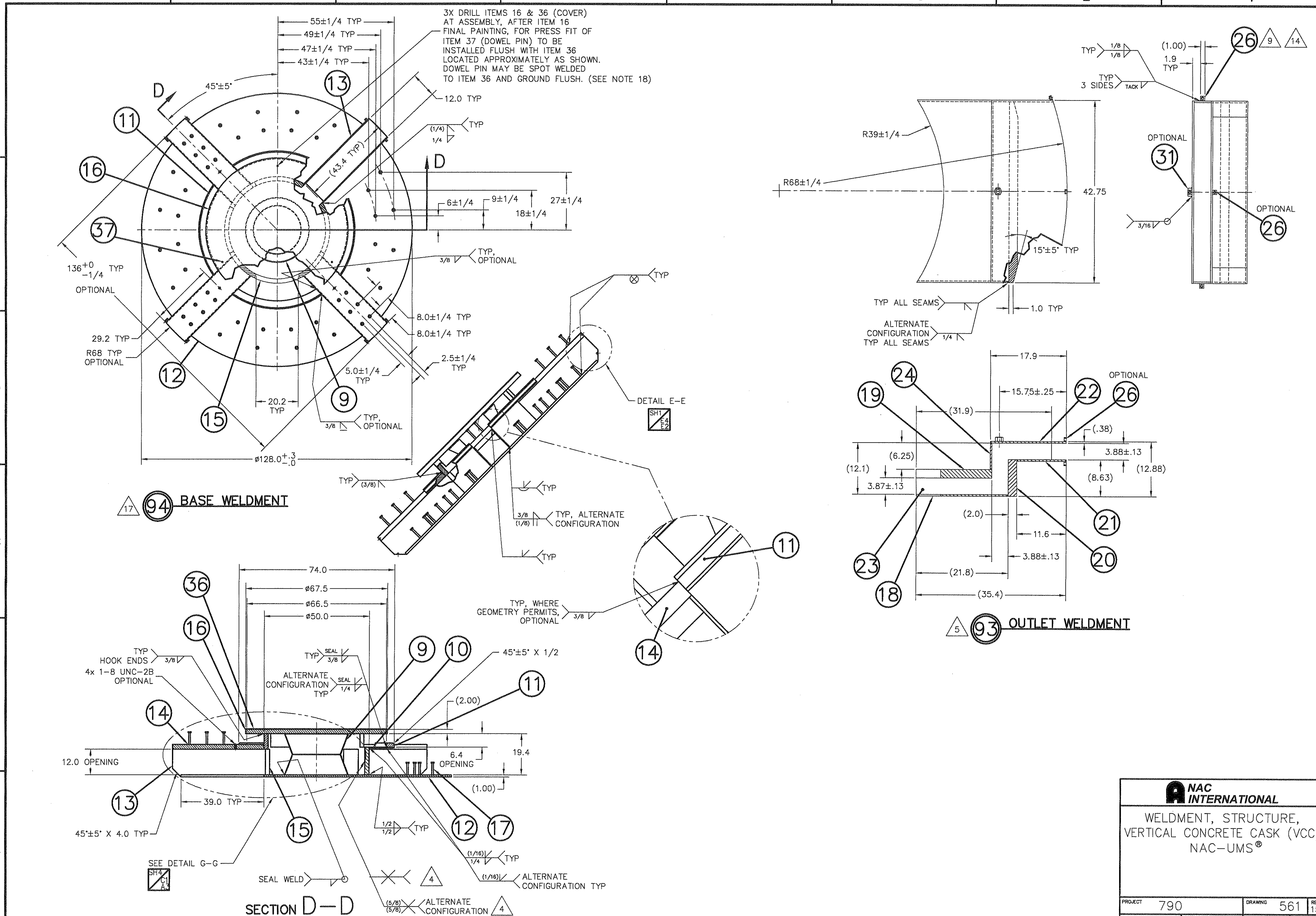
NAC INTERNATIONAL

WELDMENT, STRUCTURE, VERTICAL CONCRETE CASK (VCC) NAC-UMS®

PROJECT 790 | DRAWING 561 | REV 15
SCALE | EST.WT. | SH 1 OF 4 | 11:41AM 11-1-2010



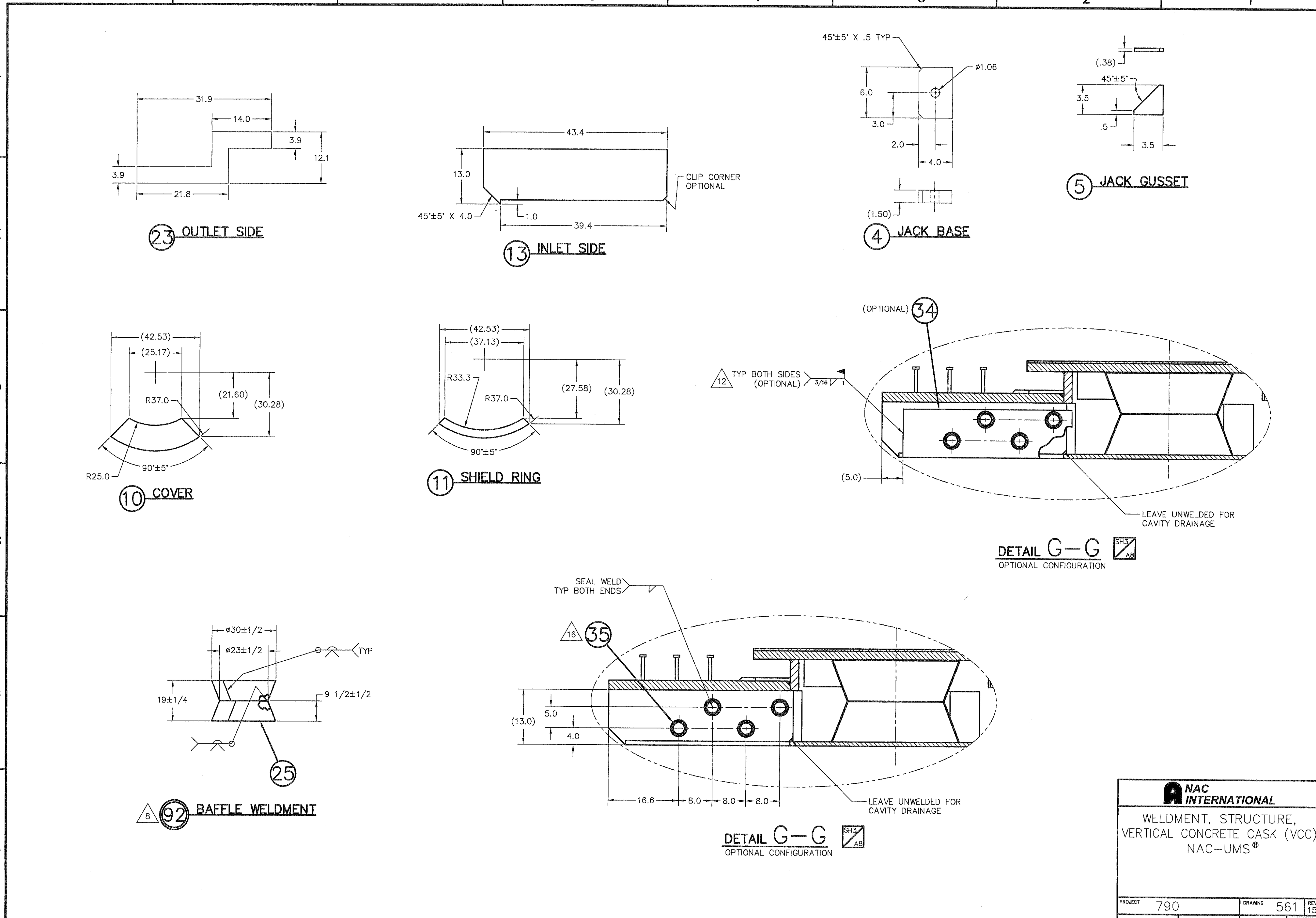
SEE DETAIL F-F
6X 60°
30°±5°
6X 1/2-13 UNC-2B ▽1.75 MIN EQUALLY SPACED ON A ø82.50 BOLT CIRCLE
42.9 TYP
12.9 TYP
45°±5°
ALTERNATE CONFIGURATION
SEE DETAIL B-B
ø101.40±.30
ø77.00±.30
(ø69.50)
(2.00)
18.5 TYP
7.5
(3.00)
6.5 TYP
222.88±.25 -99 CLASS 3
221.48±.25 -98 CLASS 5
216.68±.25 -97 CLASS 4
215.28±.25 -96 CLASS 2
206.18±.25 -95 CLASS 1
(224.88) -99
(223.48) -98
(218.68) -97
(217.28) -96
(208.18) -95
14.3 TYP
45°±5° X 1 TYP
ø74.50±.25
(ø79.50)
45°±5° X 1/4 TYP OPTIONAL
SEE DETAIL C-C
SECTION A—A
45°±5° X 1.5
3.00±.13
45°±5° X 1.0 (OPTIONAL)
2.50±.13
DETAIL B—B
.5 TYP
1.8 TYP
DETAIL C—C
2X 1-1/2-6 UNC-2B THRU, 180° APART ON ø83.0 B.C. OPTIONAL.
DETAIL F—F
NAC INTERNATIONAL
WELDMENT, STRUCTURE, VERTICAL CONCRETE CASK (VCC) NAC-UMS®
PROJECT 790
DRAWING 561
REV 15
SCALE 1/16
SH 2 OF 4



NAC INTERNATIONAL

WELDMENT, STRUCTURE,
VERTICAL CONCRETE CASK (VCC)
NAC-UMS®

PROJECT	DRAWING	REV
790	561	15
	SH 3 OF 4	11:45AM 11-1-2010



NAC INTERNATIONAL

WELDMENT, STRUCTURE, VERTICAL CONCRETE CASK (VCC) NAC-UMS®

PROJECT	DRAWING	REV
790	561	15
	SH 4 OF 4	11:47AM 11-1-2010

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A,5B,5C,5D,5E
7	INC DCR 6A,6B,6C
8	INC DCR 7A,7B,7C,7D
9	INC DCR 8A,8B,8C,8D,8E
10	INC DCR 9A
11	INC DCR 10A,10B,10C,10D
12	INC DCR 11A,11B,11C,11D,11E
13	INC DCR 12A,12B,12C,12D
14	INC DCR 13A,13B,13C
15	INC DCR 14A
16	INC DCR 15A
17	INC DCR 16A, 16B
18	INC DCR 17A,17B,17C

REBAR LENGTHS (TOLERANCE ±1)

ITEM 1	ITEM 2	ITEM 3	ITEM 4	ITEM 5	ITEM 6	CASK ASSEMBLY	CLASS
191.0	201.5	177.5	185.5	172.5	187.5	95	PWR 1
200.1	210.6	186.6	194.6	181.4	196.6	96	PWR 2
201.5	212	188	196	182.8	198	97	BWR 4
206.3	216.8	192.8	200.8	187.6	202.8	98	BWR 5
207.7	218.2	194.2	202.2	189	204.2	99	PWR 3

92 ASSY	93 ASSY	94 ASSY	95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
			A/R	A/R	A/R	A/R	A/R	49	SCREEN	304 ST. STL.	COML		4X4 MILLING GR. .035" WIRE CLOTH
			8	8	8	8	8	48	RETAINER PLATE	304 ST. STL.	ASTM A240		14 GA. – 18 GA. SHEET
1								47	SPACER	CARBON STEEL	ASTM A537, CL 2		2 PLATE
8								46	REBAR	CARBON STEEL	ASTM A706		#11
12								45	WASHER	COML	SEE NOTE 24		SEE NOTE 24
12								44	NUT	COML	SEE NOTE 24		SEE NOTE 24
6								43	THREADED REBAR	COML	SEE NOTE 24		SEE NOTE 24
			4	4	4	4	4	42	SUPPLEMENTAL COVER	304 ST. STL.	ASTM A240		18 GA. SHEET
			28	28	28	28	28	41	WASHER	ST. STL.	COML		1/4 I.D.
			24	24	24	24	24	40	SCREEN BOLT	ST. STL.	COML		1/4–20 UNC X 1/2 LG. HEX HEAD BOLT
			24	24	24	24	24	39	CONCRETE ANCHOR	ST. STL.	COML		HILTI #00336430 (HDI 1/4" (SS303)
			32	32	32	32	32	38	WASHER	ST. STL.	COML		3/8 TYPE B, SERIES W
			32	32	32	32	32	37	SCREEN SCREW	ST. STL.	COML		HEX HEAD CAP SCREW
			32	32	32	32	32	36	CONCRETE ANCHOR	ST. STL.	COML		SEE NOTE 32
								35	SUPPLEMENTAL SHIELDING			790–613–99	
								34	LIFTING ANCHOR			790–562–92	
8								33	REBAR	CARBON STEEL	ASTM A615 GR 75		#11
1								32	BASE PLATE	CARBON STEEL	ASTM A537, CL 2		2 PLATE
2								31	LIFTING LUG	CARBON STEEL	ASTM A537, CL 2		2 PLATE
			1	1	1	1	1	30	NAME PLATE			790–565–1	
			4	4	4	4	4	29	SCREW	18–8 ST. STL.	COML.		DROP–IN ANCHOR FASTENER
			4	4	4	4	4	28	CONCRETE ANCHOR	ZINC–ALLOY	COML.		DROP–IN ANCHOR
								27	NOT USED				
								26	NOT USED				
			4	4	4	4	4	25	INLET SCREEN			790–562–93	
			4	4	4	4	4	24	OUTLET SCREEN			790–562–94	
			1					23	STRUCTURE WELDMENT			790–561–95	
				1				22	STRUCTURE WELDMENT			790–561–96	
					1			21	STRUCTURE WELDMENT			790–561–97	
						1		20	STRUCTURE WELDMENT			790–561–98	
			24	24	24	24	24	19	SCREEN SCREW	ST. STL.	COML.		5/8–11 UNC–2A X 1.0 LG. HEX BOLT
			24	24	24	24	24	18	FLAT WASHER	ST. STL.	COML.		5/8 DIA, TYPE B, SERIES W
	A/R	A/R						17	STRIP	304 ST. STL.	ASTM A240		18 GA. SHEET
	A/R	A/R						16	SCREEN	304 ST. STL.	COML.		4X4 MILLING GR. WIRE CLOTH
			A/R	A/R	A/R	A/R	A/R	15	CONCRETE SHELL	CONCRETE	COML.		TYPE 2 PORTLAND CEMENT
			4	4	4	4	4	14	OUTLET WELDMENT			790–561–93	
							1	13	STRUCTURE WELDMENT			790–561–99	
			1	1	1	1	1	12	BASE WELDMENT			790–561–94	
			4	4	4	4	4	11	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			4	4	4	4	4	10	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			1	1	1	1	1	9	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			45	47	48	49	49	8	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			23	24	24	25	25	7	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			12	12	12	12	6	6	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			8	8	8	8	8	5	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			28	28	28	28	28	4	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			20	20	20	20	20	3	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			16	16	16	16	16	2	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)
			8	8	8	8	8	1	REBAR	CARBON STEEL	ASTM A615/A615M		#6(19)

QUANTITY

NOTES:

1. REINFORCEMENT FABRICATION SHALL COMPLY WITH ACI 318/318R AND 349/349R STANDARDS. REINFORCEMENT PLACEMENT SHALL BE IN ACCORDANCE WITH ACI 117–90 TOLERANCES.
2. ALL REINFORCEMENT BARS MAY BE FIELD CUT, BENT AND PLACED OR TIED FOR CLEARANCE AS APPROVED BY NAC ENGINEERING.
3. ALL BENDING OF REBARS SHALL BE PERFORMED COLD, ALL HOOKS ARE STANDARD UNLESS OTHERWISE DETAILED.
4. A 3 INCH CONCRETE COVER SHALL BE MAINTAINED FOR REINFORCEMENT AT THE EXTERIOR CONCRETE SURFACES, 2 INCH CONCRETE COVER BETWEEN THE CASK LINER AND THE REINFORCEMENT, AND 3/4 INCH CONCRETE COVER BETWEEN THE OTHER NON–EXPOSED SURFACES AND THE REINFORCEMENT UNLESS OTHERWISE NOTED, IN ACCORDANCE WITH THE TOLERANCES AS ALLOWED BY ACI 117–90.
5. ALL REBAR OVERLAPS TO BE 33 INCHES MINIMUM. REBAR OVERLAPS IN ADJACENT HOOPS SHALL NOT BE VERTICALLY ALIGNED.
6. CONCRETE SHALL DEVELOP A COMPRESSIVE STRENGTH (F ') OF 4000 PSI USING TYPE 2 PORTLAND CEMENT, 1 1/2 IN. MAXIMUM SIZE AGGREGATE. CONCRETE DENSITY SHALL BE 140 PCF MINIMUM. (18) OPTIONAL 1.5 PCY FIBERMESH MAY BE ADDED.
7. DURING FABRICATION SEGMENTED HOOPS MAY BE USED IN LIEU OF WHOLE HOOPS.
8. INSTALL PLUGS INTO SCREEN ATTACHMENT POINTS PRIOR TO FORMING
9. INSTALL PER MANUFACTURERS INSTRUCTIONS. OTHER STAINLESS STEEL DROP IN ANCHORS ARE ACCEPTABLE FOR USE IN PLACE OF ITEM 28. IF ALTERNATIVE ANCHORS ARE USED THEN THE SCREW/BOLT (ITEM 29) SHALL BE STAINLESS STEEL AND MATE WITH THE ANCHOR.
10. COAT ALL EXPOSED CONCRETE SURFACES WITH AN APPROVED SEALER.
11. ITEM 17 MAY BE CONSTRUCTED FROM SINGLE PIECE PLATE. FIELD MODIFICATION OF HOLES IN ITEM 17 TO ALLOW FIT–UP IS PERMISSIBLE AS LONG AS SCREENS ARE NOT DAMAGED. ITEMS 16 AND 17 MAY BE TRIMMED IN THE FIELD TO ALLOW FIT–UP.
12. LOCATION OF HORIZONTAL REINFORCEMENT (HOOP BARS) CAN BE REVERSED WITH THE VERTICAL REINFORCEMENT, BOTH INSIDE AND OUTSIDE REINFORCEMENT CURTAIN.
13. ITEM 18, FLAT WASHER, MAY BE REPLACED AS NEEDED WITH A 5/8 DIA, STAINLESS STEEL, BEVEL WASHER.
14. ITEM 5 MAY BE INSTALLED WITH HOOK AT TOP OF CASK. ALTERNATELY, THE 90° BEND MAY BE SUBSTITUTED WITH A 180° HOOK. LENGTH OF BAR MAY BE SAME AS ITEM 3 AS SHOWN ON THE REBAR LENGTH TABLE.
15. SECURE THE AIR OUTLETS TO PREVENT BOTH UPWARD DISPLACEMENT DURING CONCRETE PLACEMENT AND DOWNWARD DISPLACEMENT DUE TO FABRICATION PRIOR TO CONCRETE PLACEMENT.
16. SHIM PLATES MAY BE UTILIZED TO FACILITATE FIELD WELDING OPERATIONS ON ONE OR BOTH SIDES OF THE SUPPLEMENTAL SHIELDING WELDMENT. LOCATE WELDS APPROX. AS SHOWN.
17. ITEM 10 (REBAR) MAY REST ON THE TOP OF THE OUTLET WELDMENT LIFTING NUT OR MAY BE LOCATED TANGENT TO THE DIAMETER OF ITEM 9.
18. ITEM 36 (CONCRETE ANCHOR) AND 37 (SCREEN SCREW) AND 38 (WASHER) TO BE USED WHEN ALTERNATE ASSEMBLIES –93 AND –94 ARE USED.
19. (18) AFTER COMPLETION OF LOAD TESTING AND POST LOAD TEST NDE, THE LIFT ANCHORS, FOR APPROXIMATELY 18" ON THE LIFT END, SHALL BE PREPARED IN ACCORDANCE WITH SSPC–SP1. BLAST ALL SURFACES AND APPLY A HEAT–RESISTANT COATING PER MANUFACTURER'S APPLICATION INSTRUCTIONS. COATING SHOULD BE WRAPPED OVER ALL EDGES TO ENSURE COMPLETE COVERAGE.
20. WELD EFFECTIVE THROAT MAY BE REDUCED BY 1/8 AT THE EDGE OF TAPPED HOLE.
21. ENGAGE THREAD OF ITEM 33 (REBAR) FLUSH TO THE TOP OF ITEM 32 (BASE PLATE).
22. APPLY SILICONE CAULK BETWEEN THE EDGE OF ITEM 34 AND THE TOP FLANGE OF THE LINER WELDMENT, IF THE GAP IS 3/8 OR LESS.
23. ITEMS 39, 40, AND 41 ARE APPLICABLE FOR ALTERNATE HOLE LOCATIONS ON ASSEMBLY 94 ALTERNATE FABRICATION–C AND THE 9/16" HOLES LOCATED ON EITHER SIDE. ITEMS 36, 37, AND 38 MAY BE USED FOR HOLES LOCATED AT THE TOP AND BOTTOM CENTER OF ASSEMBLY 94 ALTERNATE FABRICATION–C.
24. ITEM 43, 1 1/4 NOM. 120 KSI WILLIAMS ALL THREAD–BAR (R71) ITEM 44, WILLIAMS HEX NUT (R73–10) ITEM 45, WILLIAMS HARDENED WASHER (R9F–10)
25. ITEM 44 TO BE HAND TIGHTENED AND TURNED ONE LAND. ITEM 45 TO BE MODIFIED AS REQUIRED FOR FIT–UP.
26. AT THE USE OF ASSY 92 (ALTERNATE CONFIGURATION–D) TWO OF ITEM 6 MAY BE CUT TO FIT.
27. ITEM 32 AND NELSON STUDS ON THE BASE WELDMENT MAY COME IN CONTACT WITH REINFORCING STEEL, HOWEVER, A CLEAR COVER OF 3/4" SHALL BE MAINTAINED BETWEEN ITEM 31 AND REBAR.
28. HOLES ARE TO BE LOCATED AND DRILLED IN THE FIELD AS NEEDED.
29. SEAL WELD OPEN SEAMS.
30. CONCRETE ANCHOR ADHESIVE SUCH AS HILTI ADHESIVE HIT HY 150 OR EQUIVALENT MAY BE USED TO PREVENT THE CONCRETE ANCHOR FROM BECOMING LOOSE OR PULLING OUT.
31. THE NAC–UMS VERTICAL CONCRETE CASK NUMBER MAY BE FORMED INTO THE CONCRETE SHELL. THE FORMED NUMBERS SHALL BE APPROXIMATELY 12" X 12" NOT TO EXCEED .4" DEEP.
32. HILTI #00045788 OR #00336431, (HDI 3/8" SS303) MAY BE USED.
33. (18) 1" DIAMETER HOLE OPTIONAL ON THE CENTERLINE OF THE LIFT ANCHOR BASE PLATE (32) AND SPACER PLATE (47).

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5–94 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3–12	±.005	6–18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 – .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 – .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE 125 OR BETTER			

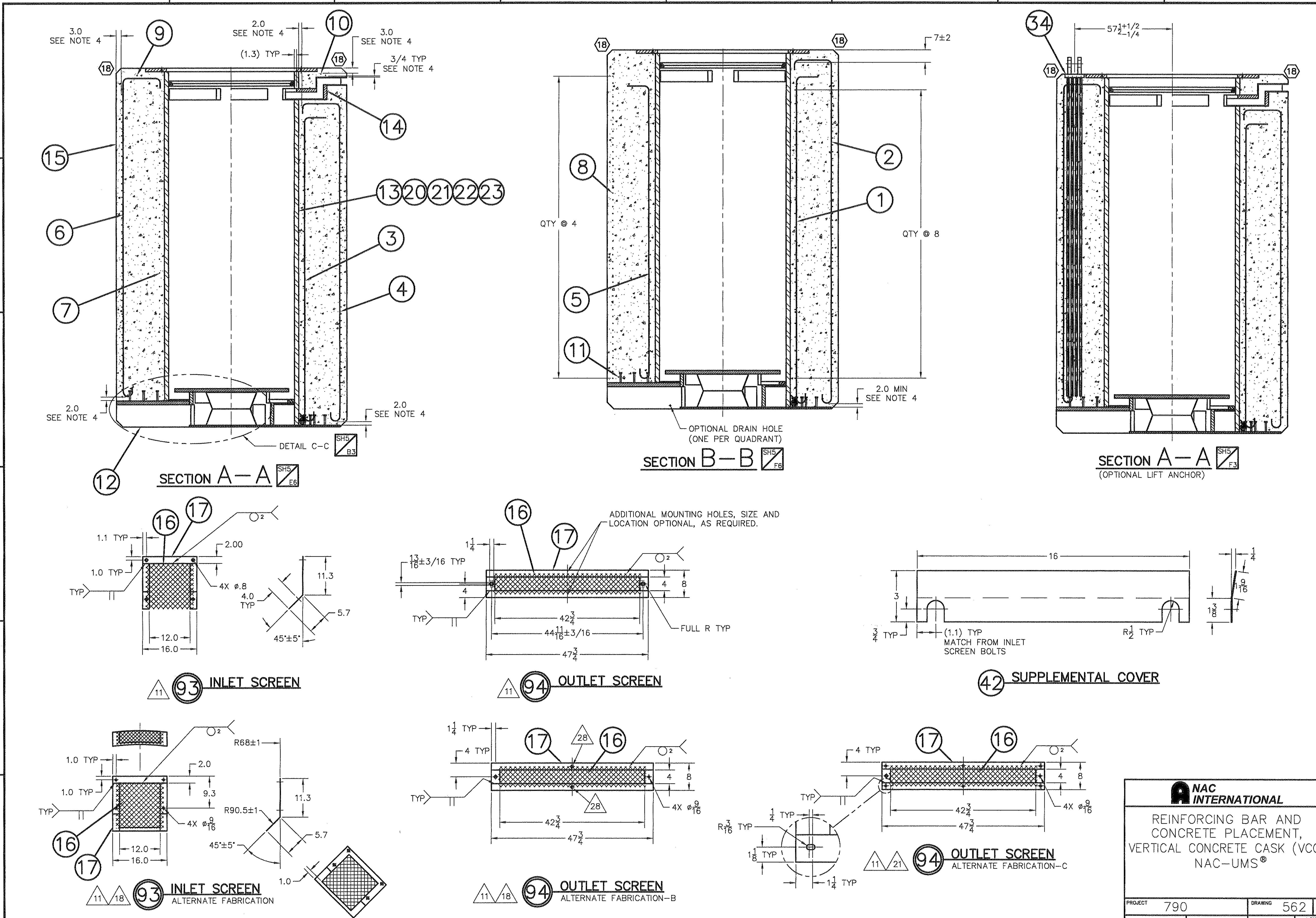
NEXT ASSEMBLY: 790–590
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	11/4/10
CHECKER	[signature]	11/04/10
PROJECT MANAGER	[signature]	11/4/10
ENGINEERING	[signature]	11/4/10
LICENSING	[signature]	11/4/10
QUALITY	[signature]	11/4/10

NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) NAC–UMS®

PROJECT 790 | DRAWING 562 | REV 18
SCALE | EST.WT. | SH 1 OF 7 | 1:37PM 11–1–2010

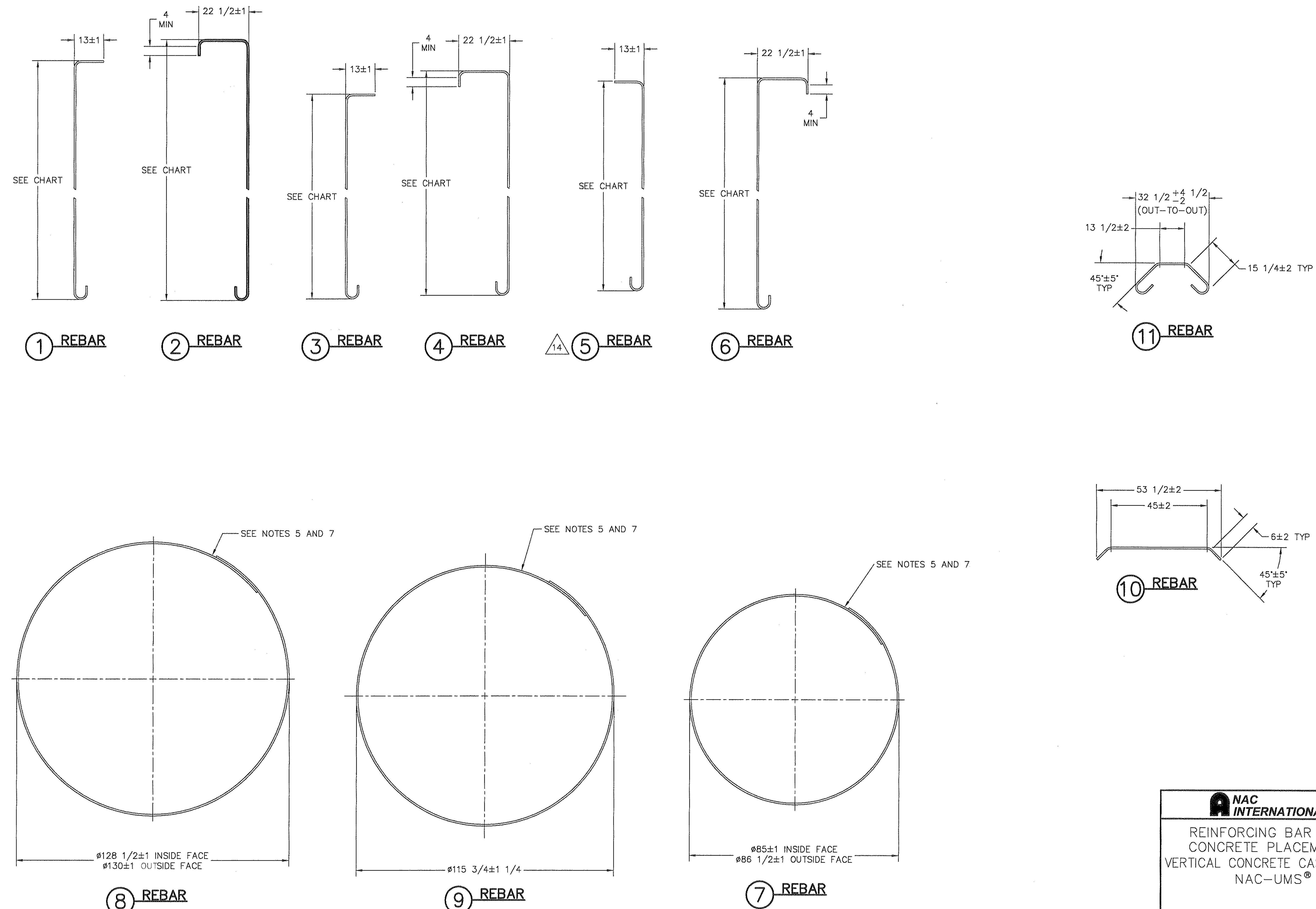


NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) NAC-UMS®

PROJECT	DRAWING	REV
790	562	18

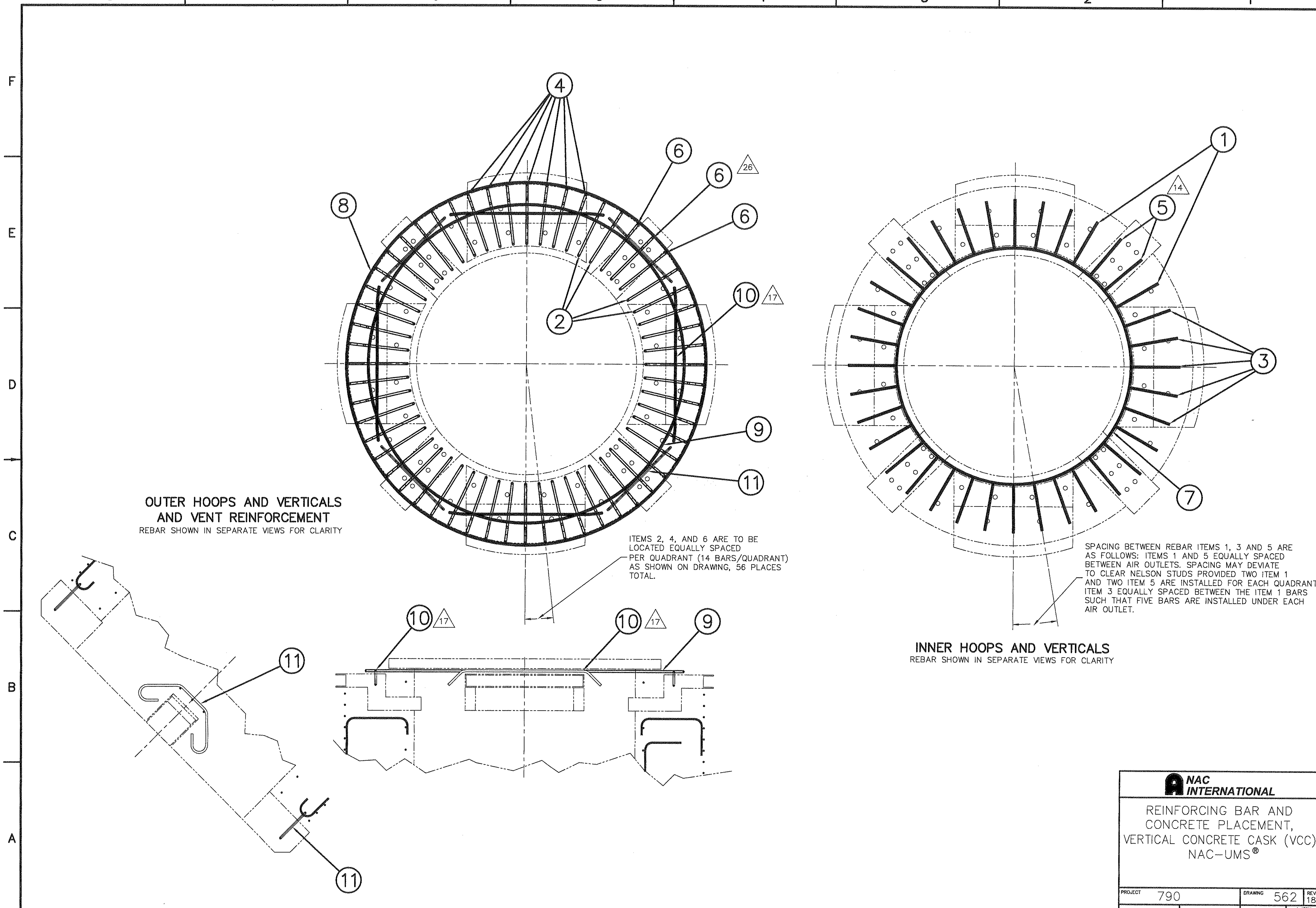
SH 2 OF 7

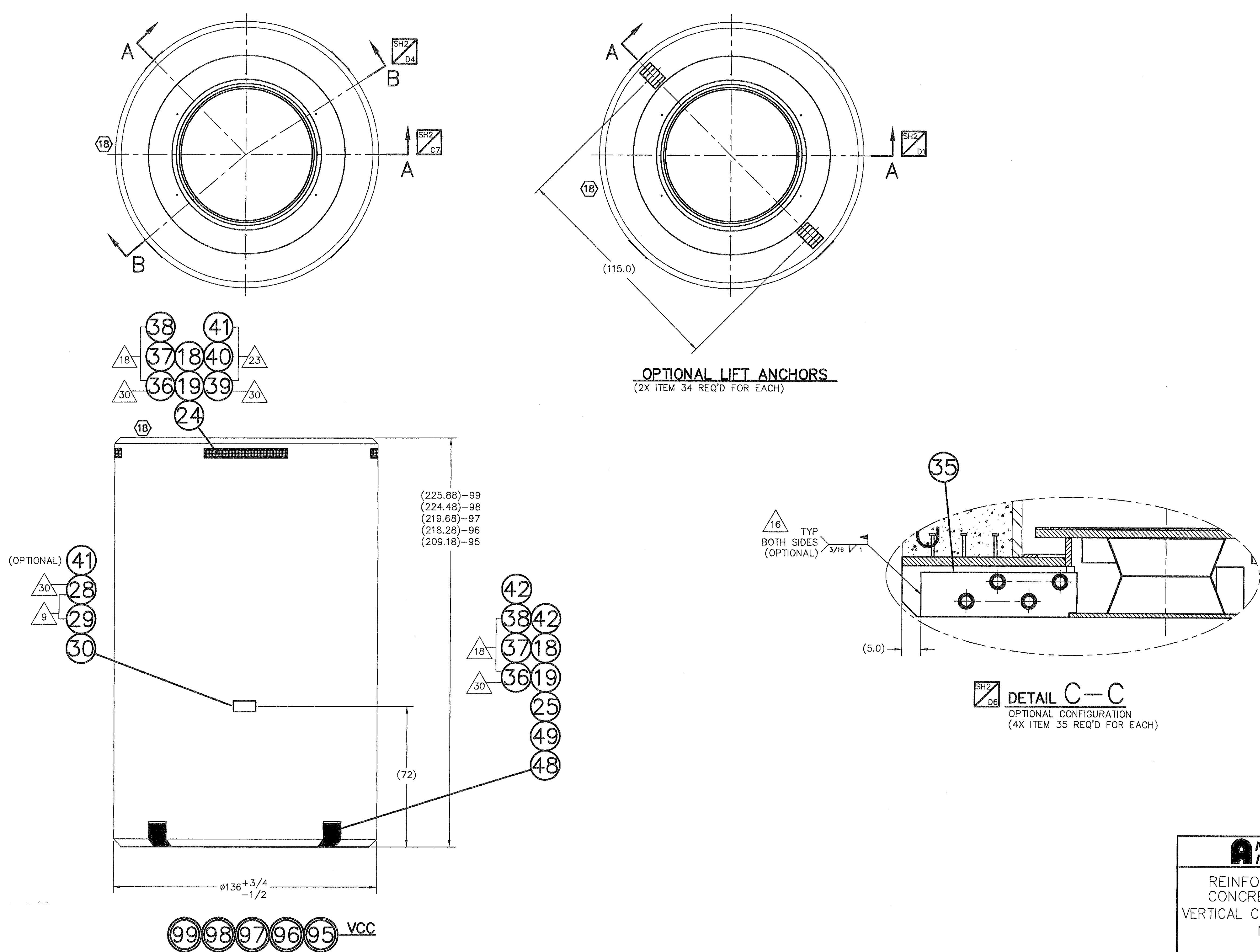


NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) NAC-UMS®

PROJECT 790	DRAWING 562	REV 18	
	SH 3 OF 7	1:40PM 11-1-2010	

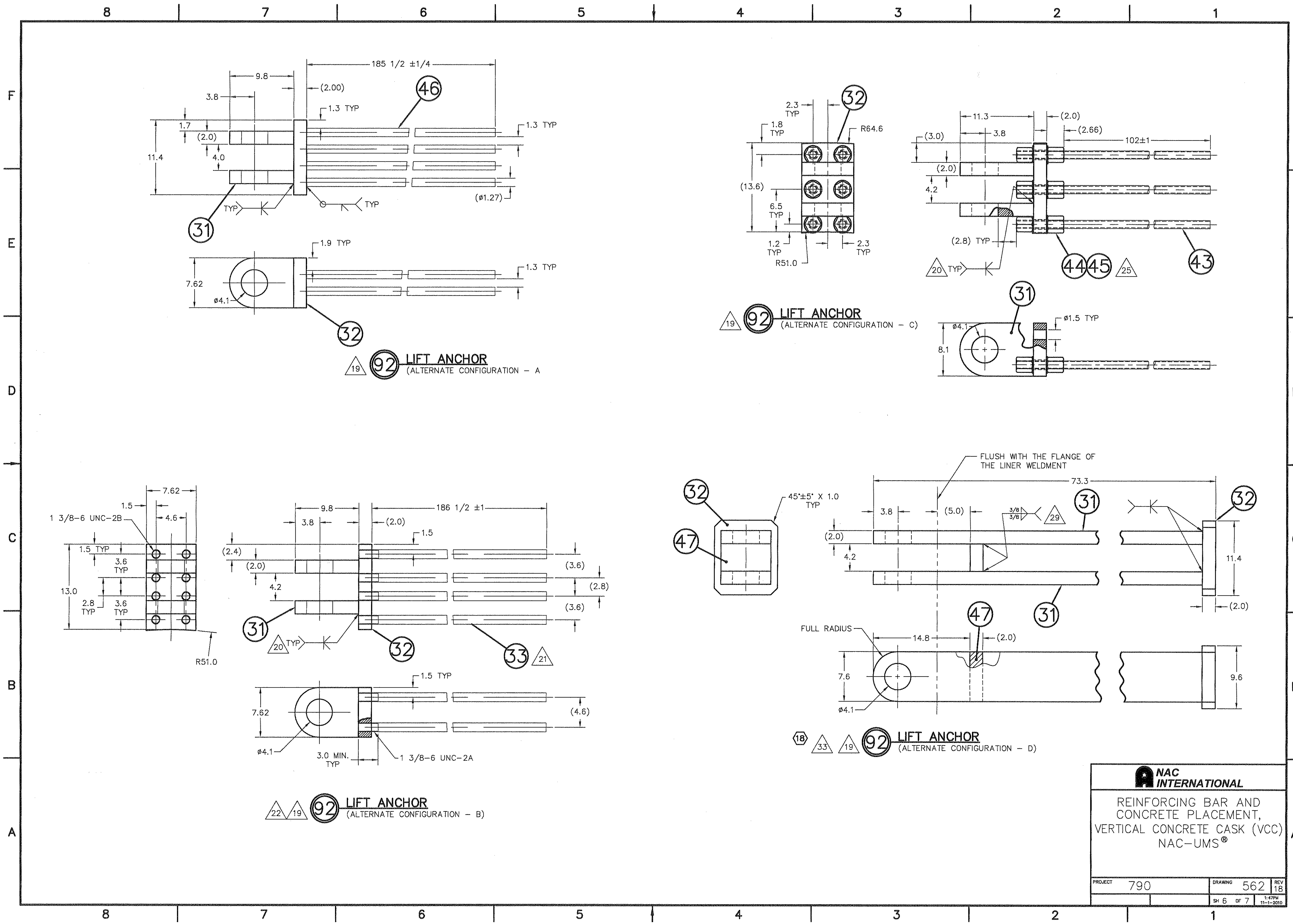

OUTER HOOPS AND VERTICALS
AND VENT REINFORCEMENT
REBAR SHOWN IN SEPARATE VIEWS FOR CLARITY
ITEMS 2, 4, AND 6 ARE TO BE LOCATED EQUALLY SPACED PER QUADRANT (14 BARS/QUADRANT) AS SHOWN ON DRAWING, 56 PLACES TOTAL.
INNER HOOPS AND VERTICALS
REBAR SHOWN IN SEPARATE VIEWS FOR CLARITY
SPACING BETWEEN REBAR ITEMS 1, 3 AND 5 ARE AS FOLLOWS: ITEMS 1 AND 5 EQUALLY SPACED BETWEEN AIR OUTLETS. SPACING MAY DEVIATE TO CLEAR NELSON STUDS PROVIDED TWO ITEM 1 AND TWO ITEM 5 ARE INSTALLED FOR EACH QUADRANT. ITEM 3 EQUALLY SPACED BETWEEN THE ITEM 1 BARS SUCH THAT FIVE BARS ARE INSTALLED UNDER EACH AIR OUTLET.
NAC INTERNATIONAL
REINFORCING BAR AND
CONCRETE PLACEMENT,
VERTICAL CONCRETE CASK (VCC)
NAC-UMS®
PROJECT 790
DRAWING 562
REV 18
SH 4 OF 7
1:43PM 11-1-2010

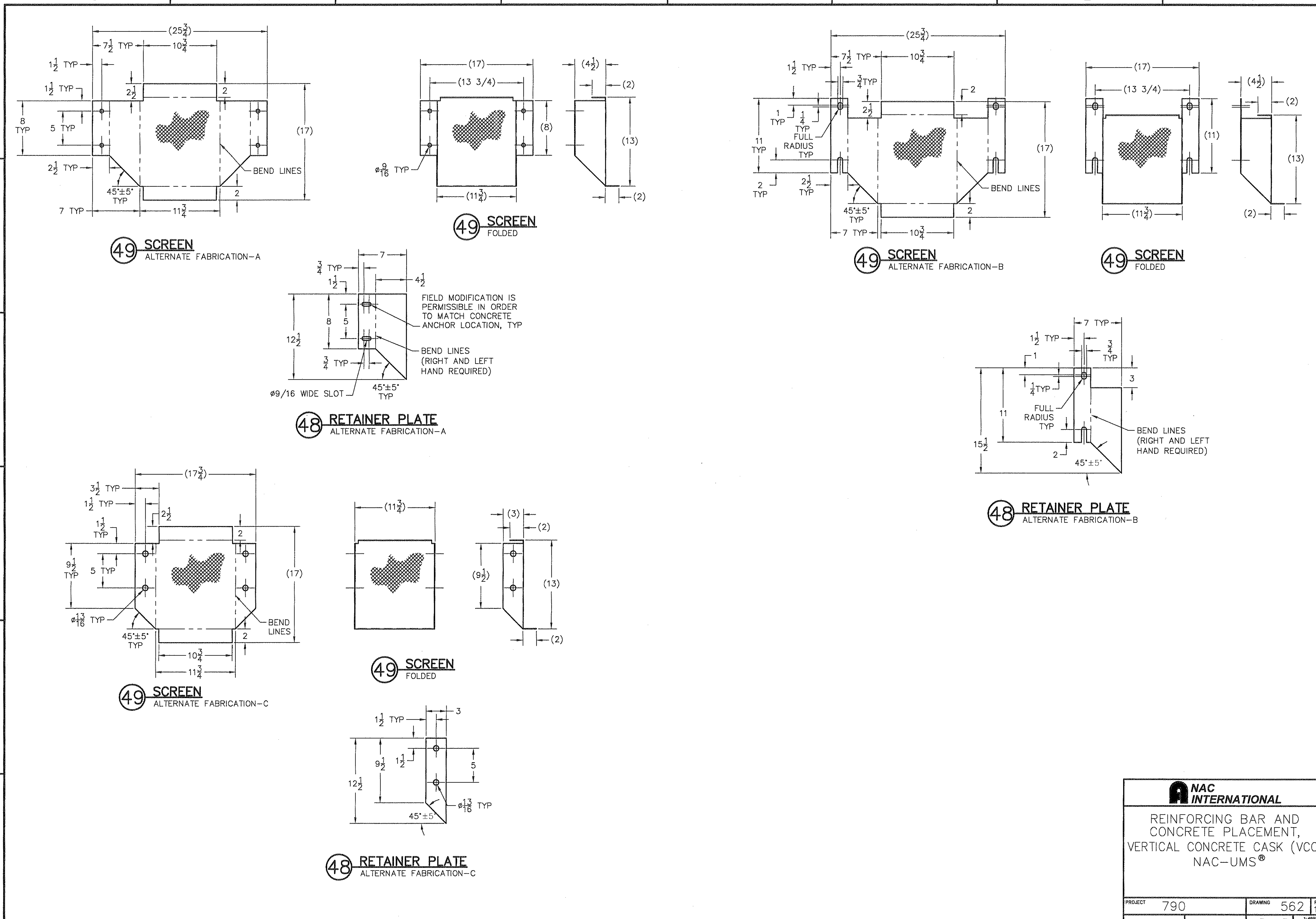


NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) NAC–UMS®

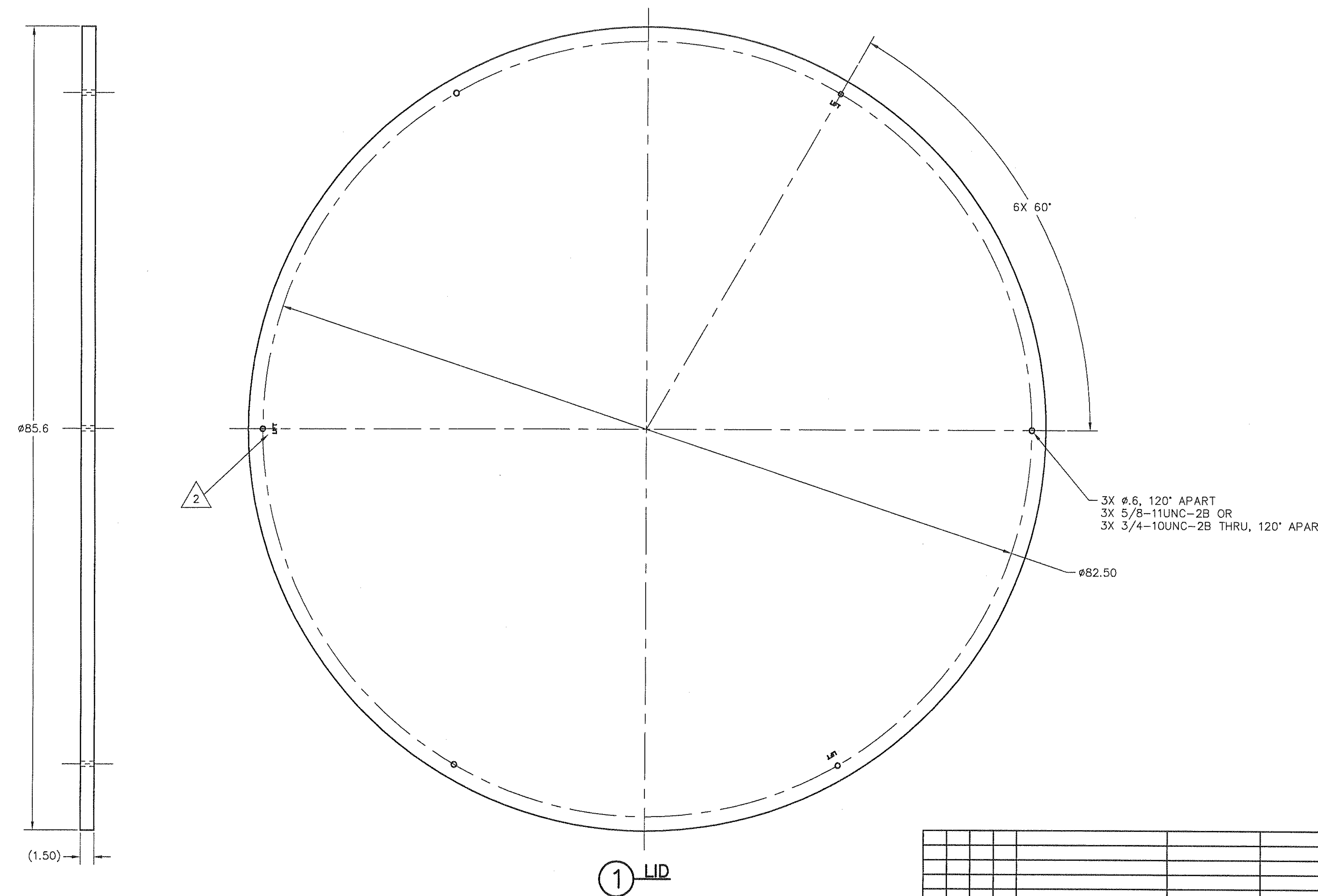
PROJECT	DRAWING	REV
790	562	18
	SH 5 OF 7	1:45PM 11-1-2010


LIFT ANCHOR
(ALTERNATE CONFIGURATION – A
LIFT ANCHOR
(ALTERNATE CONFIGURATION – C)
LIFT ANCHOR
(ALTERNATE CONFIGURATION – B)
LIFT ANCHOR
(ALTERNATE CONFIGURATION – D)
FLUSH WITH THE FLANGE OF THE LINER WELDMENT
FULL RADIUS
NAC INTERNATIONAL
REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) NAC–UMS®
PROJECT 790
DRAWING 562
REV 18
SH 6 OF 7
1:47PM 11-1-2010



49 SCREEN
ALTERNATE FABRICATION-A
49 SCREEN
FOLDED
48 RETAINER PLATE
ALTERNATE FABRICATION-A
FIELD MODIFICATION IS PERMISSIBLE IN ORDER TO MATCH CONCRETE ANCHOR LOCATION, TYP
BEND LINES (RIGHT AND LEFT HAND REQUIRED)
ø9/16 WIDE SLOT
49 SCREEN
ALTERNATE FABRICATION-B
49 SCREEN
FOLDED
48 RETAINER PLATE
ALTERNATE FABRICATION-B
49 SCREEN
ALTERNATE FABRICATION-C
49 SCREEN
FOLDED
48 RETAINER PLATE
ALTERNATE FABRICATION-C
BEND LINES
FULL RADIUS TYP
NAC INTERNATIONAL
REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) NAC-UMS®
PROJECT 790
DRAWING 562
REV 18
SH 7 OF 7
1:49PM 11-1-2010

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A,3B
5	INC DCR 4A
6	INC DCR 5A



1 LID

NOTES:

⑥ 1. PREPARE ALL SURFACES IN ACCORDANCE WITH SSPC-SP1. BLAST ALL SURFACES AND APPLY A HEAT-RESISTANT COATING PER MANUFACTURER'S APPLICATION INSTRUCTIONS. WRAP COATING OVER ALL EDGES TO ENSURE COMPLETE COVERAGE.

△2 STEEL STAMP 3/8" HIGH X .03 DEEP, "LIFT" APPROX. AS SHOWN. THREE PLACES AT THE TAPPED HOLE LOCATIONS.

3. IT IS PERMISSIBLE TO PAINT THE VCC NUMBER ON THE LID USING WEATHER RESISTANT CONTRASTING COLOR PAINT.

97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
	1	1		LID	CARBON STEEL	ASTM A36		1 1/2 PLATE
ASSY	ASSY	ASSY						
QUANTITY								

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3-12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6-18	±.03
	OVER 18	±.06
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY

ALL DIMENSIONS ARE IN INCHES

BORDER SIZE: F (40 X 28)

ALL UNSPECIFIED TOOL RADII: .015 - .030

BREAK ALL SHARP CORNERS .015 - .030

MACHINED SURFACES TO BE 125 OR BETTER

NEXT ASSEMBLY: 790-590

DRAWING TYPE: LICENSE

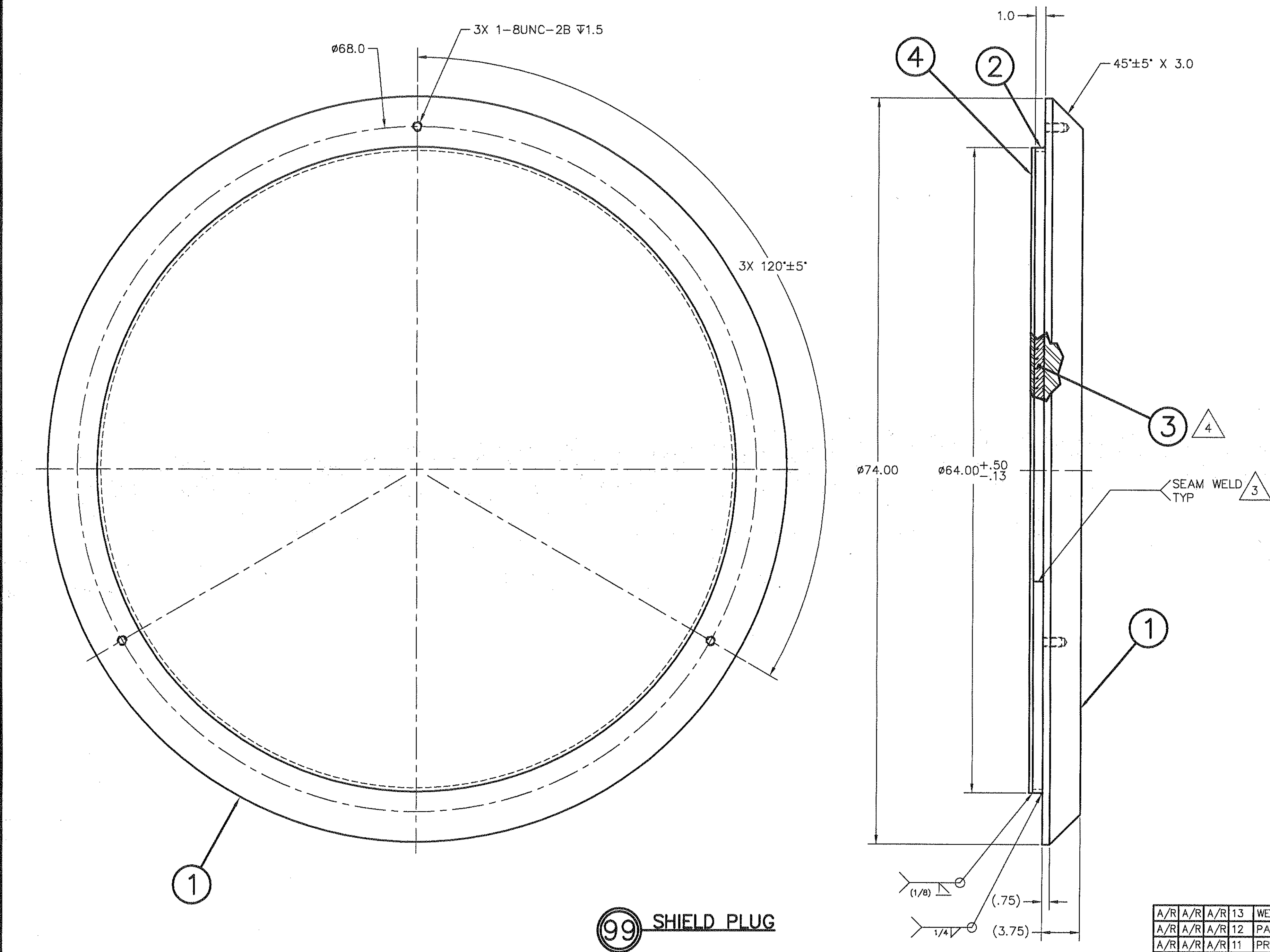
GROUP	NAME	DATE
PREPARER	[signature]	11/4/10
CHECKER	[signature]	11/09/10
PROJECT MANAGER	[signature]	11/4/10
ENGINEERING	[signature]	11/4/10
LICENSING	[signature]	11/4/10
QUALITY	[signature]	11/4/10

NAC INTERNATIONAL

LID,
VERTICAL CONCRETE CASK (VCC)
NAC-UMS®

PROJECT 790 | DRAWING 563 | REV 6

SCALE 1/5 | WEIGHT 2449# | SH 1 OF 1 | 2:07PM 11-1-2010



REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A,3B
5	INC DCR 4A,4B
6	INC DCR 5A,5B
7	INC DCR 6A
8	INC DCR 7A

NOTES:

(8) 1. PREPARE ALL SURFACES IN ACCORDANCE WITH SSPC-SP1. BLAST ALL SURFACES AND APPLY A HEAT-RESISTANT COATING PER MANUFACTURER'S APPLICATION INSTRUCTIONS. WRAP COATING OVER ALL EDGES TO ENSURE COMPLETE COVERAGE.

(2) NEUTRON SHIELD TO CONTAIN 0.6 WEIGHT PERCENT B_4C MINIMUM.

(3) SEAM WELD, GEOMETRY, NUMBER AND LOCATION OPTIONAL

(4) NEUTRON SHIELD MATERIAL TO BE POURED TO DEPTH SHOWN +00/-.13.

5. IT IS PERMISSIBLE TO PAINT THE VCC NUMBER ON THE SHIELD PLUG USING WEATHER RESISTANT CONTRASTING COLOR PAINT.

97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
A/R	A/R	A/R	13	WEATHER RESISTANT PAINT		COML		SEE NOTE 5
A/R	A/R	A/R	12	PAINT		COML		SEE NOTE 1
A/R	A/R	A/R	11	PRIMER		COML		SEE NOTE 1
1			10	CENTER BOSS	CARBON STEEL	ASTM A36		DIA. 2 BAR
3			9	LIFTING BOSS	CARBON STEEL	ASTM A36		DIA. 2 BAR
1			8	NS COVER	CARBON STEEL	ASTM A36		3/8 PLATE
1			7	NS RETAINING RING	CARBON STEEL	ASTM A36		3/8 BAR/PLATE
	1		6	NS RETAINING RING	CARBON STEEL	ASTM A36		3/8 BAR/PLATE
A/R	A/R		5	NEUTRON SHIELD	NS-3	COML		
	1	1	4	NS COVER	CARBON STEEL	ASTM A36		3/8 PLATE
A/R		A/R	3	NEUTRON SHIELD	NS-4-FR	COML		
		1	2	NS RETAINING RING	CARBON STEEL	ASTM A36		3/8 BAR/PLATE
1	1	1	1	SHIELD PLUG	CARBON STEEL	ASTM A36		3 3/4 PLATE
ASSY	ASSY	ASSY						
QUANTITY								

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94. UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3-12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6-18	±.03
	OVER 18	±.06
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY

ALL DIMENSIONS ARE IN INCHES

BORDER SIZE: F (40 X 28)

ALL UNSPECIFIED TOOL RADII: .015 - .030

BREAK ALL SHARP CORNERS .015 - .030

MACHINED SURFACES TO BE ⩔ OR BETTER

NEXT ASSEMBLY: 790-590

DRAWING TYPE: LICENSE

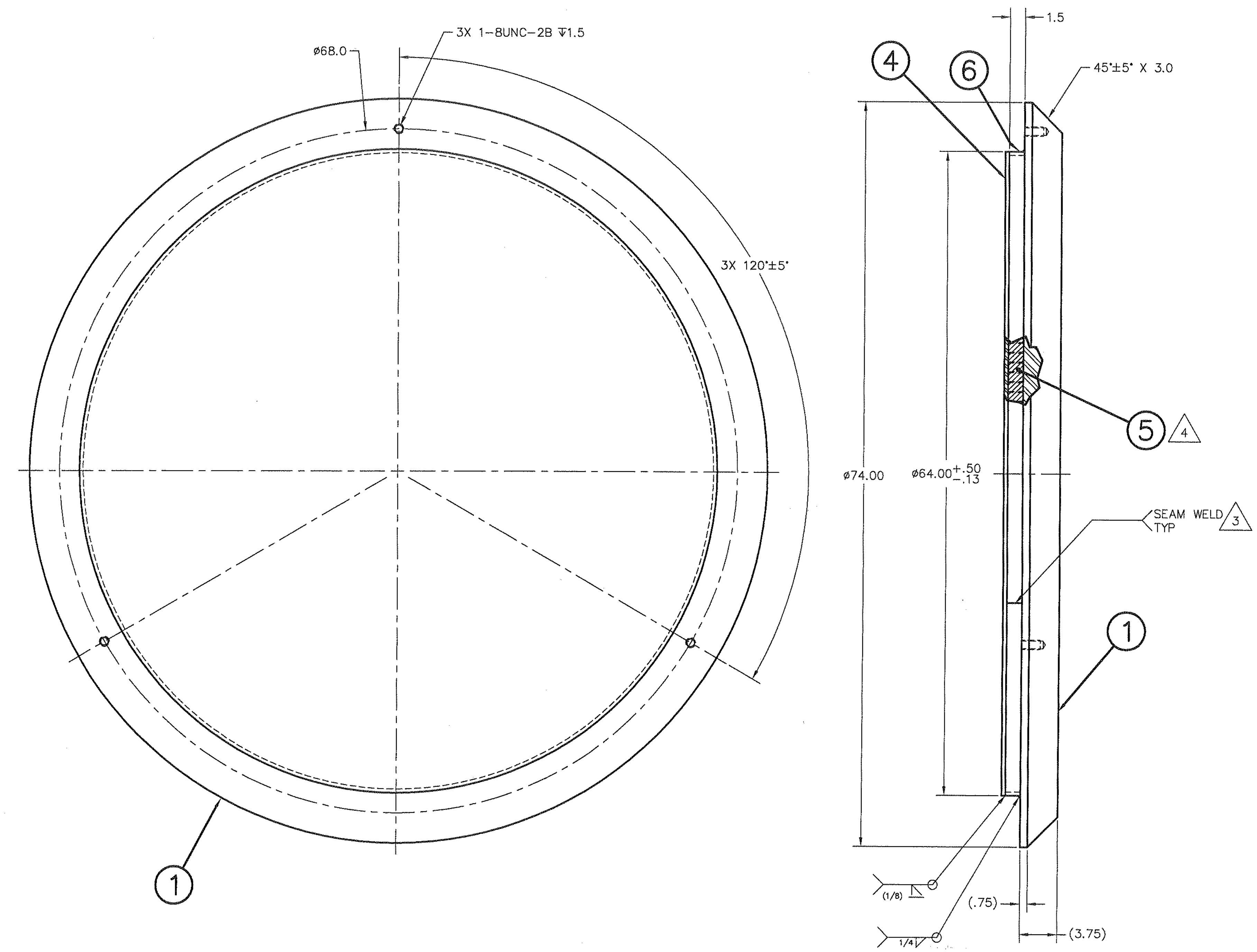
GROUP	NAME	DATE
PREPARER		11/4/10
CHECKER		11/04/10
PROJECT MANAGER		11/4/10
ENGINEERING		11/4/10
LICENSING		11/4/10
QUALITY		11/4/10

NAC INTERNATIONAL

SHIELD PLUG,
VERTICAL CONCRETE CASK (VCC)
NAC-UMS®

PROJECT 790 | DRAWING 564 | REV 8

SCALE 1/5 | WEIGHT 4845# | SH 1 OF 3 | 2:41PM 11-1-2010

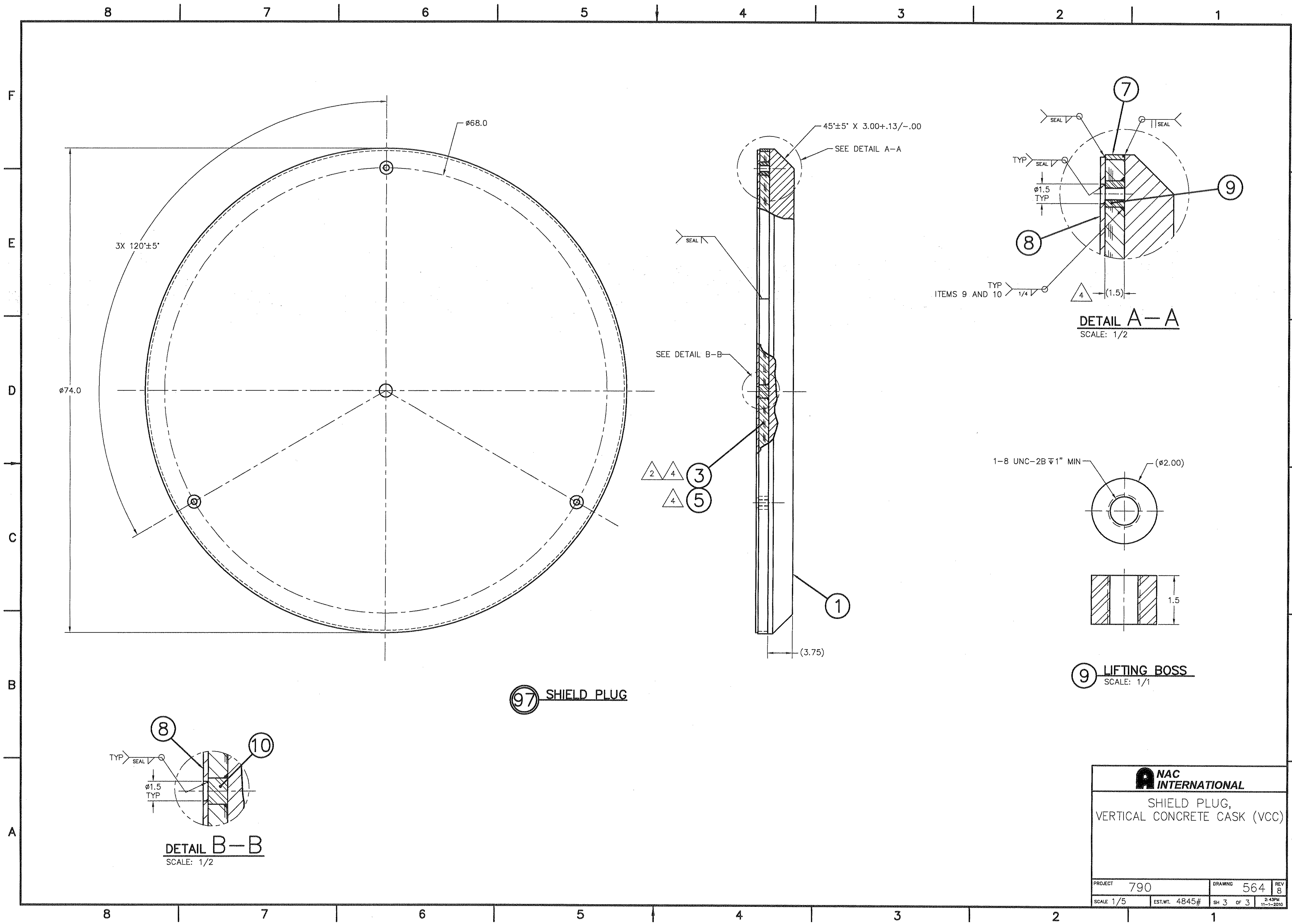


98 SHIELD PLUG

NAC INTERNATIONAL

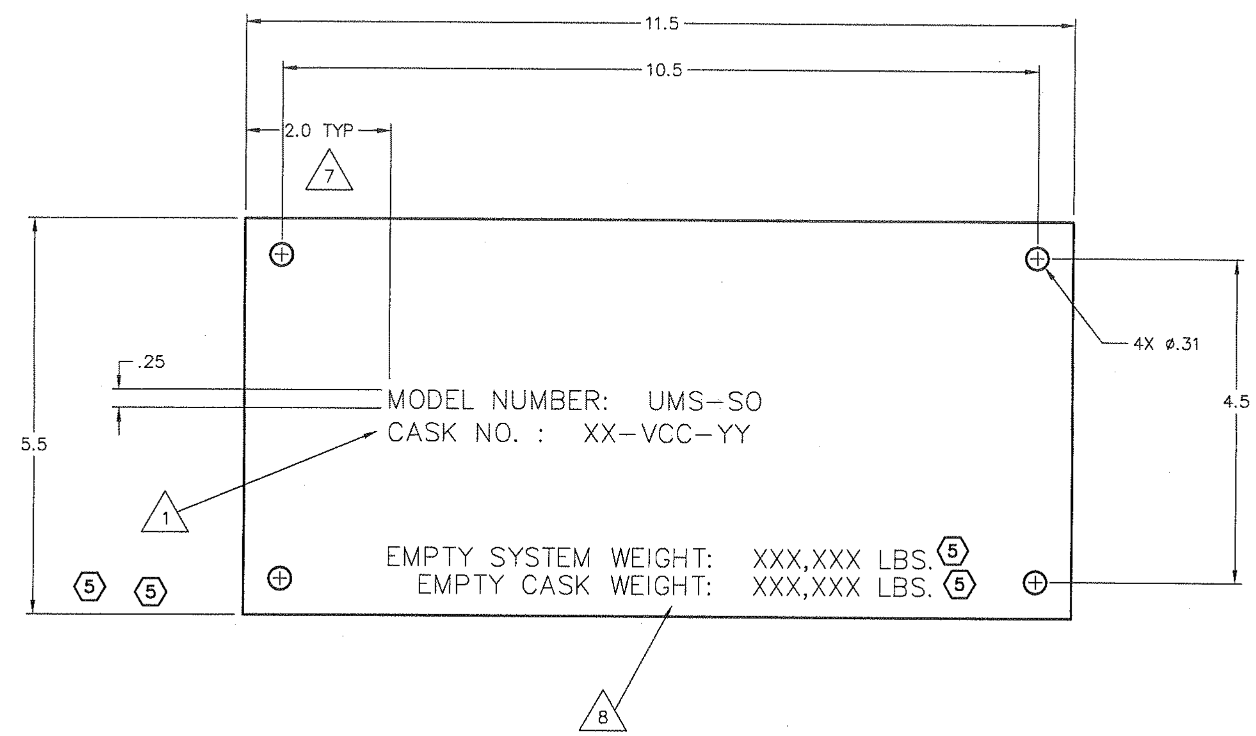
SHIELD PLUG,
VERTICAL CONCRETE CASK (VCC)

PROJECT 790		DRAWING 564	REV 8
SCALE 1/5	EST.WT. 4845#	SH 2 OF 3	2:48PM 11-1-2010



ø68.0
3X 120°±5'
ø74.0
45°±5' X 3.00+.13/-.00
SEE DETAIL A-A
SEAL
SEE DETAIL B-B
3
5
1
(3.75)
97 SHIELD PLUG
7
9
8
TYP
ø1.5 TYP
ITEMS 9 AND 10
1/4
(1.5)
DETAIL A—A
SCALE: 1/2
1-8 UNC-2B ⊽1" MIN
(ø2.00)
1.5
9 LIFTING BOSS
SCALE: 1/1
10
DETAIL B—B
SCALE: 1/2
NAC INTERNATIONAL
SHIELD PLUG,
VERTICAL CONCRETE CASK (VCC)
PROJECT 790
DRAWING 564
REV 8
SCALE 1/5
EST.WT. 4845#
SH 3 OF 3
2:43PM 11-1-2010

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A,2B,2C,2D
4	INC DCR 3B
5	INC DCR 4A



1 VCC NAMEPLATE

NOTES:

1. (Δ1) EACH NAMEPLATE FOR ITS RESPECTIVE CASK TO BE UNIQUELY IDENTIFIED, WHERE XX IS A UNIQUE ID FOR EACH CUSTOMER SITE AND YY IS A UNIQUE CONSECUTIVE NUMBER BEGINNING WITH 01 FOR EACH CUSTOMER SITE.
2. STEEL STAMP/ENGRAVE AS SHOWN, APPROXIMATELY .03 DEEP AND FILL WITH BLACK WEATHER RESISTANT PAINT.
3. (Δ3) SHEET METAL MAY VARY FROM 10 GAUGE TO 18 GAUGE.
4. DELETED
5. ADDITIONAL INFORMATION MAY BE ADDED TO THE NAMEPLATE AT THE USERS/NAC DISCRETION.
6. NAMEPLATE ATTACHMENT HOLES MAY BE FIELD MODIFIED TO ADJUST FOR MINOR MISALIGNMENT OF CONCRETE ANCHOR LOCATIONS IN THE VCC.
7. (Δ7) ENGRAVINGS TO BE LEFT-JUSTIFIED. OFFSET CAN BE ADJUSTED DOWN TO 1.25" OFFSET BY ENGRAVER TO FIT LETTERING, TEXT WRAPPING IS PERMISSIBLE. DO NOT RUN-OUT LETTERING THROUGH ANCHOR LOCATIONS.
8. (Δ8) NOMINAL WEIGHT FROM FSAR TABLE 3.2-1 OR 3.2-2 (AS APPROPRIATE) UNLESS OTHERWISE PROVIDED.

97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
			1	VCC NAMEPLATE	304 ST. STL.	ASTM A240		SEE NOTE 3 (Δ3)

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ∇ OR BETTER			

NEXT ASSEMBLY: 790-562

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER		11/1/10
CHECKER		11/04/10
PROJECT MANAGER		11/4/10
DIRECTOR DESIGN AND ANALYSIS		11/4/10
DIRECTOR, LICENSING		11/4/10
VICE PRESIDENT, QUALITY		11/4/10

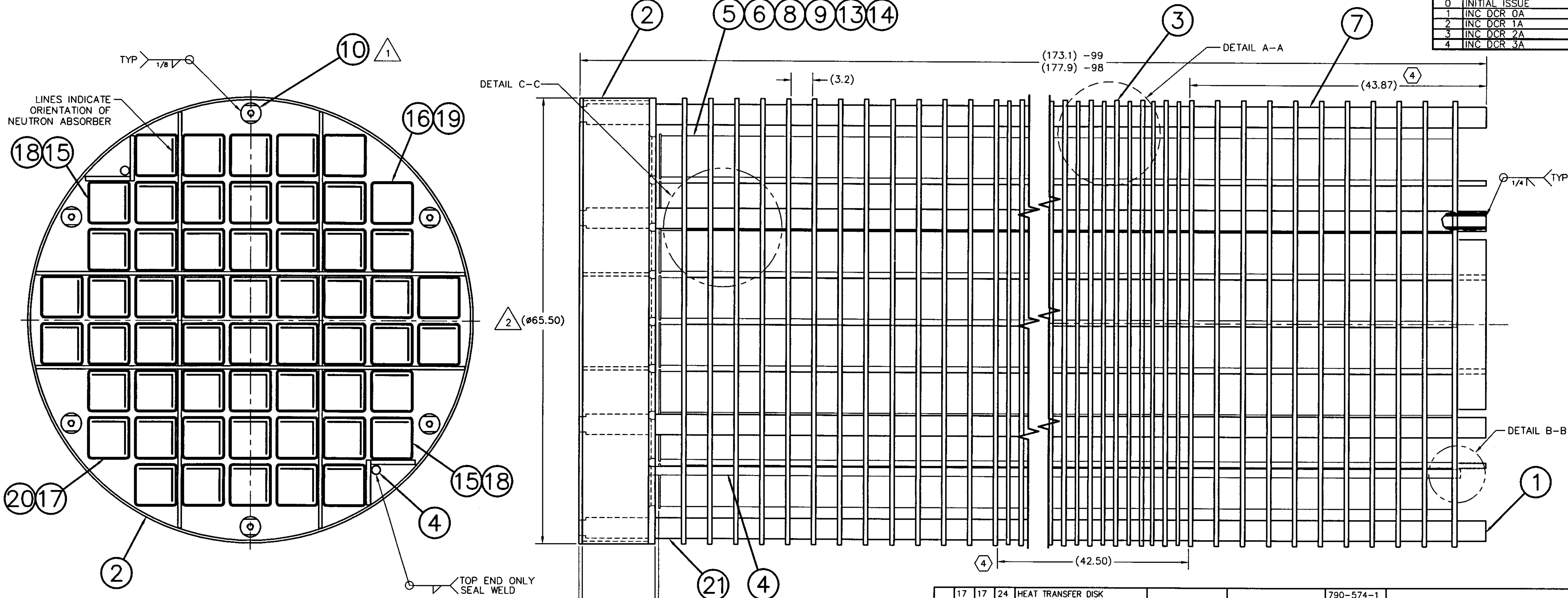
NAC INTERNATIONAL

NAMEPLATE
VERTICAL CONCRETE CASK (VCC)
NAC-UMS®

PROJECT 790 | DRAWING 565 | REV 5
SCALE FULL | EST.WT. | SH 1 OF 1 | 3:03PM 11-1-2010

R-148483

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A



99 FUEL BASKET ASSEMBLY

98 FUEL BASKET ASSEMBLY

97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
	17	17	24	HEAT TRANSFER DISK			790-574-1	
	204	204	23	FLAT WASHER	ST. STL.	COML ANSI B.18.22.1		2 1/2 TYPE A PLAIN WASHER
	204	204	22	SPLIT SPACER			790-573-8	
	6	6	21	TOP SPACER			790-573-7	
	1		20	TUBE (2 SIDE BORAL)			790-605-98	
	1		19	TUBE (NO BORAL)			790-605-94	
	2		18	TUBE (1 SIDE BORAL)			790-605-96	
		1	17	TUBE (2 SIDE BORAL)			790-605-99	
		1	16	TUBE (NO BORAL)			790-605-95	
		2	15	TUBE (1 SIDE BORAL)			790-605-97	
	2		14	TUBE (NO BORAL)			790-575-94	
		2	13	TUBE (NO BORAL)			790-575-95	
	6		12	TIE ROD			790-573-6	
		6	11	TIE ROD			790-573-5	
	6	6	10	TOP NUT			790-573-4	
	9		9	TUBE (1-SIDED)			790-575-96	
		9	8	TUBE (1-SIDED)			790-575-97	
	144	138	7	SPACER			790-573-3	
	41		6	TUBE (2-SIDED)			790-575-98	
		41	5	TUBE (2-SIDED)			790-575-99	
	1	1	4	DRAIN TUBE SLEEVE	304 ST.STL.	ASTM A249/A213		2 DIA TUBE WITH .035 WALL
	41	40	3	SUPPORT DISK			790-573-1	
	1	1	2	TOP WELDMENT			790-572-99	
	1	1	1	BOTTOM WELDMENT			790-571-99	
ASSY	ASSY	ASSY						
QUANTITY								

NOTES:

1. LUBRICATE WITH A SPENT FUEL POOL COMPATIBLE LUBRICANT SUCH AS NEOLUBE AND TORQUE NUTS TO 50±10 FT-LBS PRIOR TO WELDING.
2. ASSEMBLY -99, -98 TO BE FREELY INSERTED INTO CANISTER, DURING FINAL ASSEMBLY. ALTERNATE ASSEMBLY IN CANISTER OPTIONAL.
3. THIS DIMENSION IS TO BE HELD BY ADJUSTMENT OF THE TOP SPACER (ITEM 21). LONGER OR SHORTER PARTS MAY BE USED.

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE 125/OR BETTER			

NEXT ASSEMBLY: 790-585

DRAWING TYPE: LICENSE

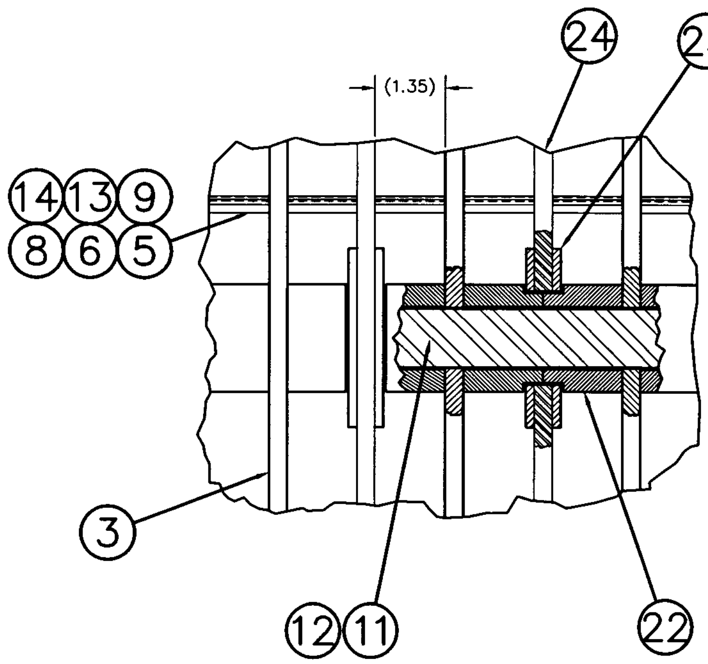
GROUP	NAME	DATE
PREPARER	[signature]	7/24/02
CHECKER	S. Graham	7-24-02
PROJECT MANAGER	[signature]	7/24/02
DIRECTOR DESIGN AND ANALYSIS	Thomas Anfum	7-25-02
DIRECTOR, LICENSING	[signature] for TCT	7/26/02
VICE PRESIDENT, QUALITY	[signature] / RHS	8/5/02

NAC INTERNATIONAL

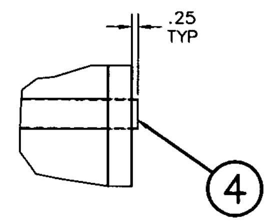
FUEL BASKET ASSEMBLY, 56 ELEMENT BWR NAC-UMS®

PROJECT 790 | DRAWING 570 | REV 4

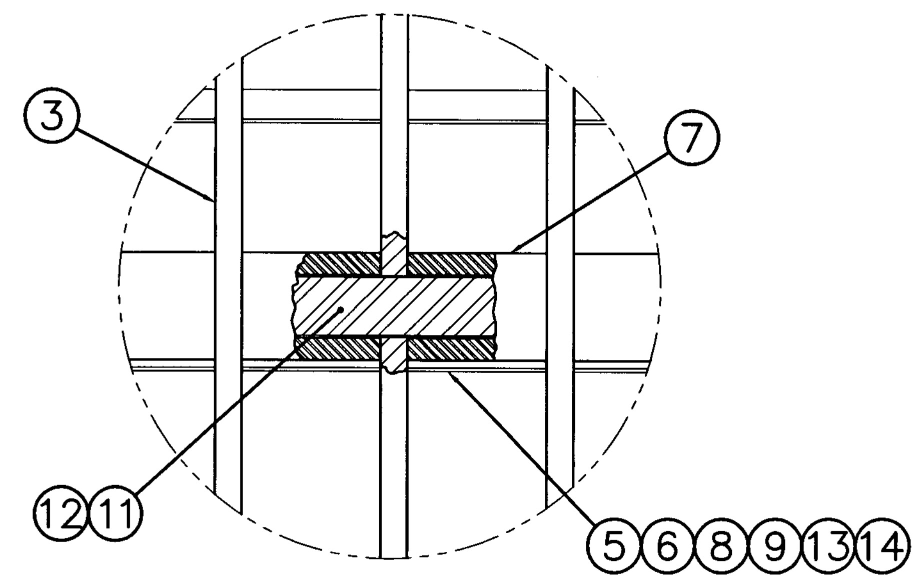
SCALE 1/6 | EST.WT. | SH 1 OF 2 | 4:20PM 7-24-2002



DETAIL A—A
SCALE: 1/2



DETAIL B—B



DETAIL C—C
SCALE: 1/2

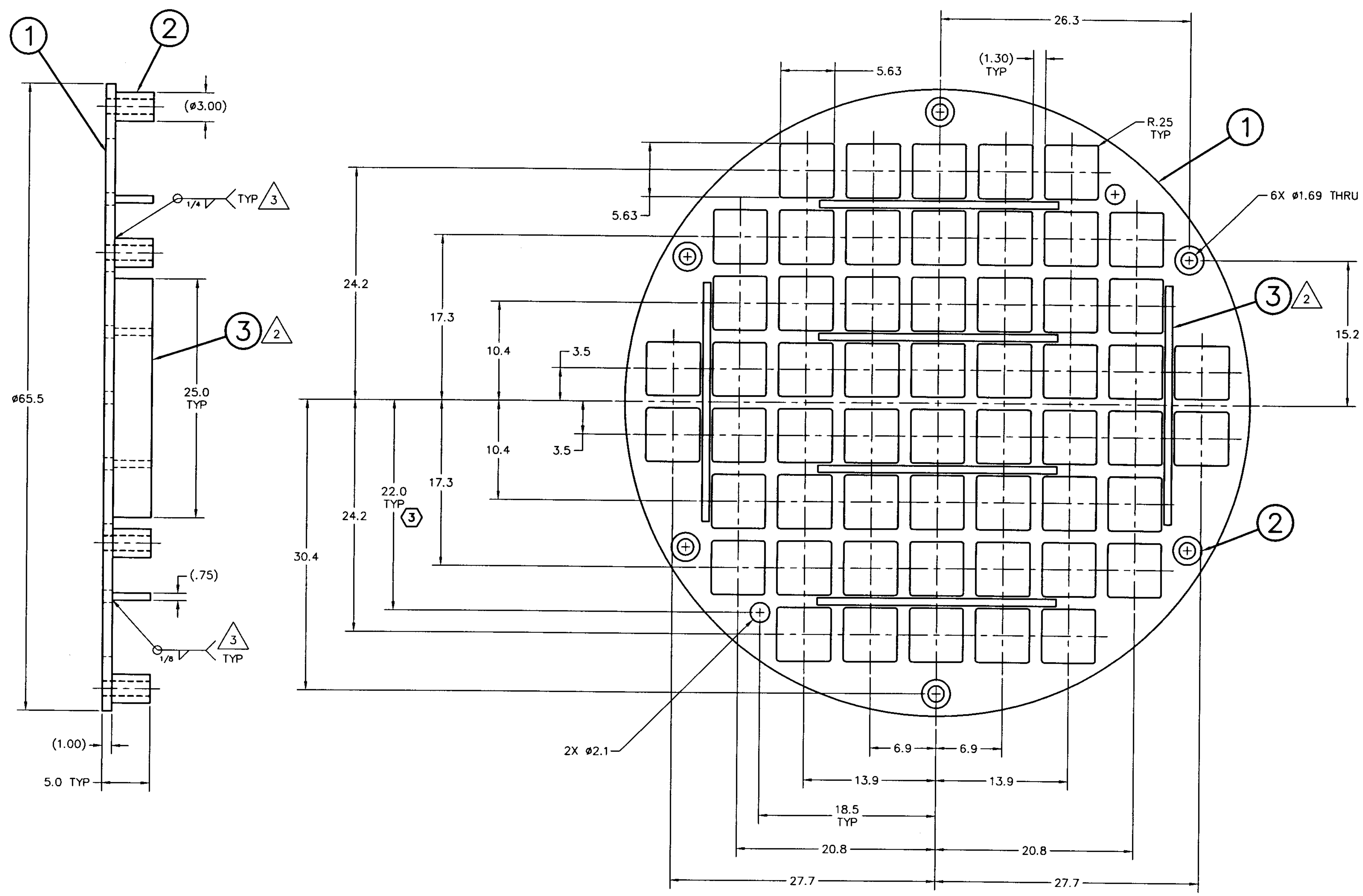
NAC INTERNATIONAL

FUEL BASKET ASSEMBLY,
56 ELEMENT BWR
NAC-UMS®

PROJECT 790		DRAWING 570	REV 4
SCALE 1/6	EST.WT.	SH 2 OF 2	12:33PM 7-19-2002

R-146693

DCR No.	REVISION
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A



BOTTOM VIEW

(99) BOTTOM WELDMENT

3. LIQUID PENETRANT (PT) EXAMINED PER ASME SECTION V, ARTICLE 6. ACCEPTANCE PER ASME SECTION III, ARTICLE NG-5350.

2. CENTERED APPROX. ON WEB.

1. ALL WELDING PROCEDURES AND QUALIFICATION TO BE IN ACCORDANCE WITH ASME SECTION IX.

NOTES:

95	96	97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
				6	3	SUPPORT	304 ST.STL.	ASME SA240/SA479		3/4 PLATE/BAR
				6	2	PAD	304 ST.STL.	ASME SA479		3 DIA BAR
				1	1	PLATE	304 ST.STL.	ASME SA240		1 PLATE
ASSY	ASSY	ASSY	ASSY	ASSY						

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
⏥	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3-12		6-18	±.06
∠	ANGULARITY	OVER 12		OVER 18	±.09
⊥	PERPENDICULARITY	.X	±.1	ANGLES ±.5°	
//	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .01 - .03			
◎	CONCENTRICITY	BREAK ALL SHARP CORNERS .01 - .03			
⊕	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE 125 OR BETTER			

NEXT ASSEMBLY: 790-570

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	4/5/02
CHECKER	[signature]	4-11-02
PROJECT MANAGER	[signature]	4/17/02
DIRECTOR DESIGN AND ANALYSIS	[signature]	4-23-02
DIRECTOR, LICENSING	[signature]	4/23/02
VICE PRESIDENT, QUALITY	[signature]	4/25/02

NAC INTERNATIONAL

BOTTOM WELDMENT, FUEL BASKET, 56 ELEMENT BWR NAC-UMS®

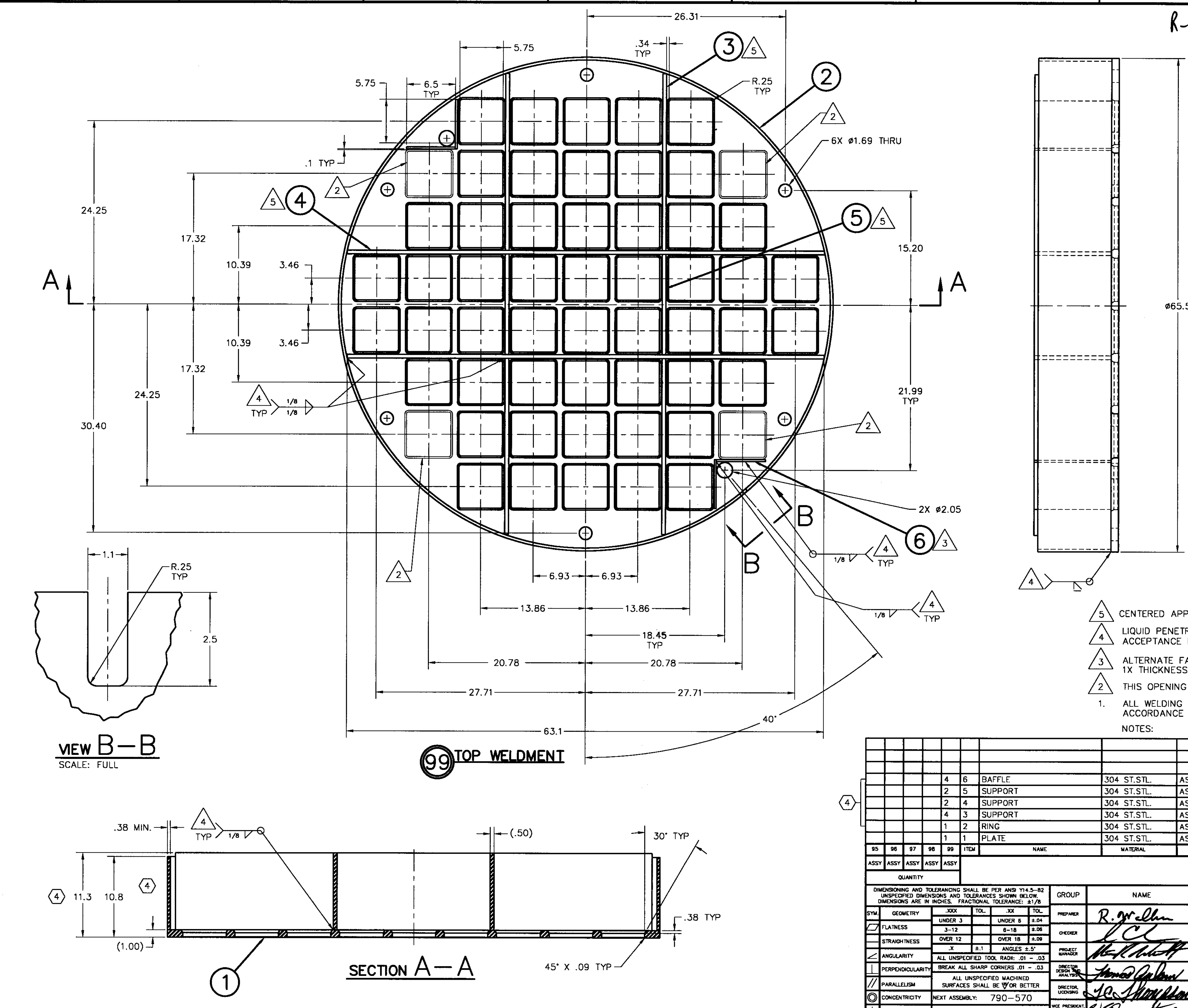
PROJECT	DRAWING	REV
790	571	3
SCALE 1/5	EST.WT.	SH 1 OF 1

4:20PM 4-5-2002

R-133548

~~R-133448~~ 8/11/04

DCR No.	REVISION
0	INITIAL ISSUE
1	INC DCR SAR-001-0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A



5. CENTERED APPROX. ON WEB.
4. LIQUID PENETRANT (PT) EXAMINED PER ASME SECTION V, ARTICLE 6. ACCEPTANCE PER ASME SECTION III, ARTICLE NG-5350.
3. ALTERNATE FABRICATION BY FORMING ONE PIECE, 1X THICKNESS MIN. I.D. BEND RAD.
2. THIS OPENING SHALL BE 5.90 X 5.90
1. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH ASME SECTION IX.

NOTES:

95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
				4	6	BAFFLE	304 ST.STL.	ASME SA240/SA479		1/4 PLATE/BAR
				2	5	SUPPORT	304 ST.STL.	ASME SA240/SA479		1/2 PLATE/BAR
				2	4	SUPPORT	304 ST.STL.	ASME SA240/SA479		1/2 PLATE/BAR
				4	3	SUPPORT	304 ST.STL.	ASME SA240/SA479		1/2 PLATE/BAR
				1	2	RING	304 ST.STL.	ASME SA240/SA479		PLATE/BAR
				1	1	PLATE	304 ST.STL.	ASME SA240		1 PLATE

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3–12		6–18	±.06
	ANGULARITY	OVER 12		OVER 18	±.09
	PERPENDICULARITY	.X	±.1	ANGLES ±.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .01 – .03			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .01 – .03			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

NEXT ASSEMBLY: 790-570

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. [illegible]	8-5-99
CHECKER	[illegible]	8/6/99
PROJECT MANAGER	[illegible]	8/9/99
DIRECTOR DESIGN AND ANALYSIS	[illegible]	8/9/99
DIRECTOR, LICENSING	[illegible]	8/11/99
VICE PRESIDENT, QUALITY	[illegible] for	8/12/99

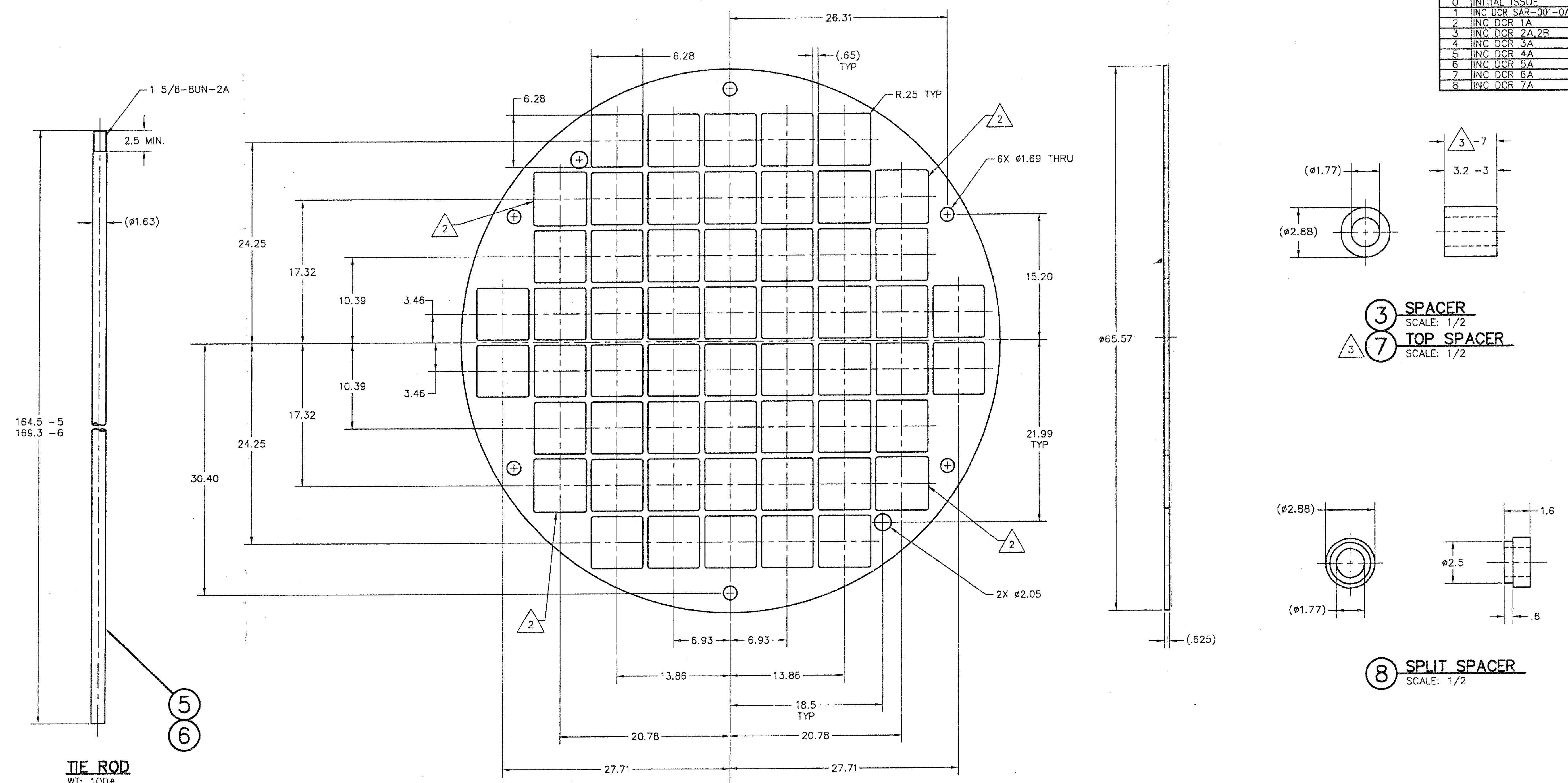
NAC INTERNATIONAL

TOP WELDMENT, FUEL BASKET, 56 ELEMENT BWR NAC-UMS®

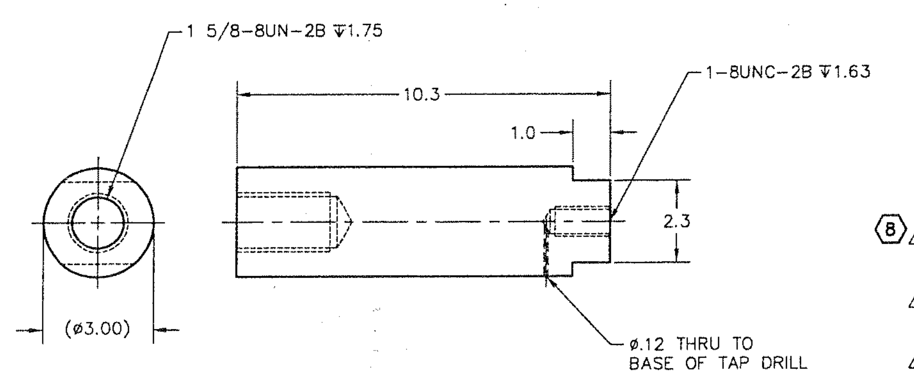
PROJECT 790 | DRAWING 572 | REV 4

SCALE 1/5 | EST.WT. | SH 1 OF 1 | 3:30PM 8-5-99

NAC DCRM 03



1 SUPPORT DISK



4 TOP NUT
SCALE: 1/2

DCR No.	REVISION
0	INITIAL ISSUE
1	INC DCR SAR-001-0A
2	INC DCR 1A
3	INC DCR 2A,2B
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A
7	INC DCR 6A
8	INC DCR 7A

8 4 (DELETED)

3 LENGTH TO BE DETERMINED AT ASSEMBLY.

2 THIS OPENING SHALL BE 6.43 X 6.43

1 ELECTROLESS NICKEL PLATED IN ACCORDANCE WITH ASTM SPECIFICATION B733-97, SC3 TYPE V, CLASS 1.

NOTES:

95	96	97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
					8	SPLIT SPACER	304 ST. STL.	ASME SA312		2 1/2 PIPE XXS
					7	TOP SPACER	304 ST. STL.	ASME SA312		2 1/2 PIPE XXS
					6	TIE ROD	304 ST. STL.	ASME SA479		1 5/8 DIA BAR
					5	TIE ROD	304 ST. STL.	ASME SA479		1 5/8 DIA BAR
					4	TOP NUT	304 ST. STL.	ASME SA479		3 DIA BAR
					3	SPACER	304 ST. STL.	ASME SA312		2 1/2 PIPE XXS
					2					
					1	SUPPORT DISK	CARBON STEEL	ASME SA533 TY B CL 2		5/8 PLATE
ASSY	ASSY	ASSY	ASSY	ASSY						

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3-12		6-18	±.06
	ANGULARITY	OVER 12		OVER 18	±.09
	PERPENDICULARITY	.X	±.1	ANGLES ±.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .01 - .03			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .01 - .03			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

NEXT ASSEMBLY: 790-570
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	9-6-06
CHECKER	[signature]	9-6-06
PROJECT MANAGER	[signature]	9/7/06
DIRECTOR DESIGN AND ANALYSIS	[signature]	9-7-06
DIRECTOR, LICENSING	[signature]	9-7-06
VICE PRESIDENT, QUALITY	[signature]	9/18/06

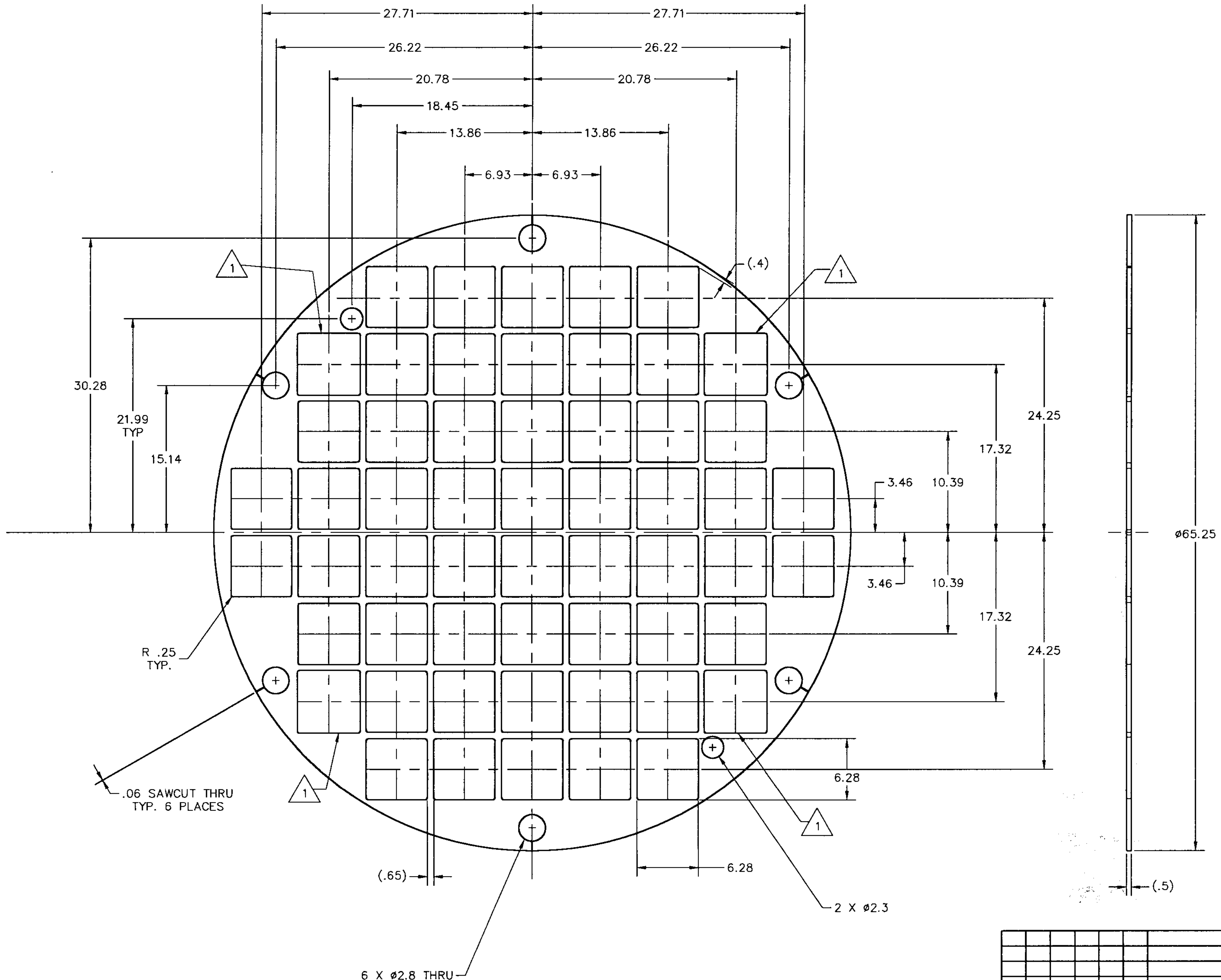
NAC INTERNATIONAL

SUPPORT DISK AND MISC BASKET DETAILS, 56 ELEMENT BWR NAC-UMS®

PROJECT 790 | DRAWING 573 | REV 8
SCALE 1/5 | EST.WT. | SH 1 OF 1 | 3:26PM 9-6-2006

R-133799

DCR No.	REVISION
0	INITIAL ISSUE
1	INC DCR SAR-001-0A
2	INC DCR 1A
3	INC DCR 2A



① HEAT TRANSFER DISK
WT: 57#

NOTES:

1. THIS OPENING IS 6.43 X 6.43

95	96	97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
					1	HEAT TRANSFER DISK	6061-T651 AL.	ASME SB209		1/2 PLATE
ASSY	ASSY	ASSY	ASSY	ASSY						

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82
UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW.
DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3-12		6-18	±.06
	ANGULARITY	OVER 12		OVER 18	±.09
	PERPENDICULARITY	.X	±.1	ANGLES ±.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII .01 - .03			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .01 - .03			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE 125 OR BETTER			

NEXT ASSEMBLY: 790-570
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. Walker	10-8-99
CHECKER	[signature]	10/8/99
PROJECT MANAGER	[signature]	10/8/99
DIRECTOR DESIGN AND ANALYSIS	Thomas Dalane	10/8/99
DIRECTOR, LICENSING	T.O. Thompson	10/8/99
VICE PRESIDENT, QUALITY	B. Barnett	10/8/99

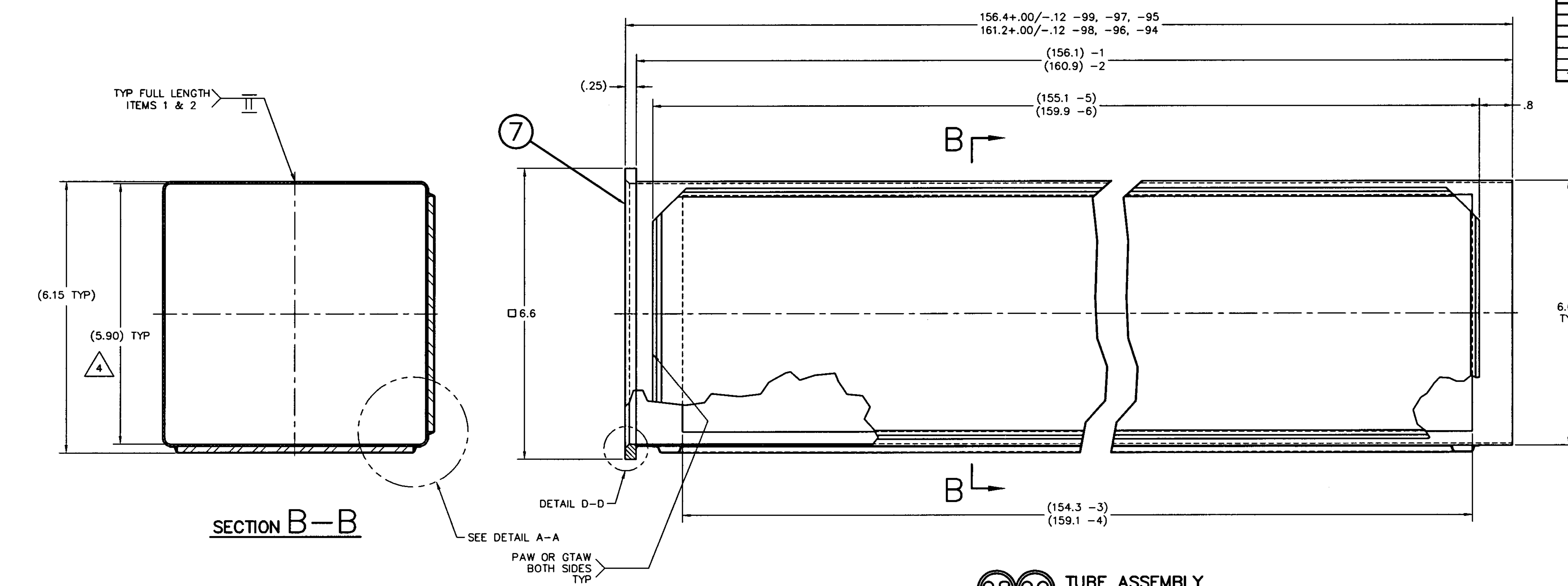
NAC INTERNATIONAL

HEAT TRANSFER DISK
FUEL BASKET,
56 ELEMENT BWR
NAC-UMS®

PROJECT 790 | DRAWING 574 | REV 3
SCALE 1/5 | EST.WT. | SH 1 OF 1 | 9:11AM 10-8-99

R_2022,03

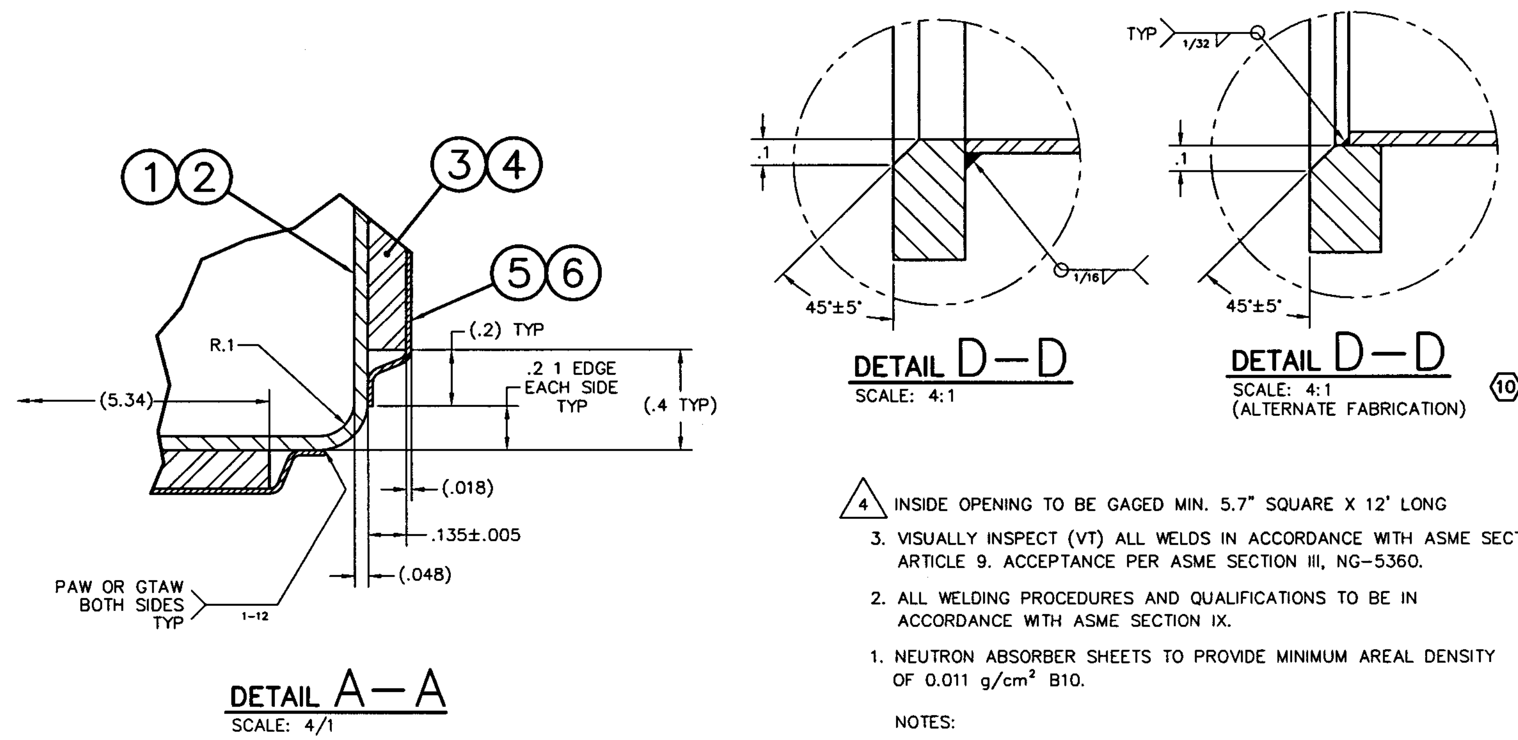
REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A
7	INC DCR 6A
8	INC DCR 7A
9	INC DCR 8A
10	INC DCR 9A



98 99 TUBE ASSEMBLY
NEUTRON ABSORBER (2) SIDES

96 97 TUBE ASSEMBLY
NEUTRON ABSORBER (1) SIDE

94 95 TUBE ASSEMBLY
NO NEUTRON ABSORBER



DETAIL A—A
SCALE: 4/1

DETAIL D—D
SCALE: 4:1

DETAIL D—D
SCALE: 4:1
(ALTERNATE FABRICATION)

4 INSIDE OPENING TO BE GAGED MIN. 5.7" SQUARE X 12' LONG

3. VISUALLY INSPECT (VT) ALL WELDS IN ACCORDANCE WITH ASME SECTION V, ARTICLE 9. ACCEPTANCE PER ASME SECTION III, NG-5360.

2. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH ASME SECTION IX.

1. NEUTRON ABSORBER SHEETS TO PROVIDE MINIMUM AREAL DENSITY OF 0.011 g/cm² B10.

NOTES:

94 ASSY	95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
1	1	1	1	1	1	7	FLANGE	304 ST. STL.	ASTM A240		1/4 PLATE
		1		2		6	CLADDING	304 ST. STL.	ASTM A240		26 GA SHEET
			1		2	5	CLADDING	304 ST. STL.	ASTM A240		26 GA SHEET
		1		2		4	NEUTRON ABSORBER	BORAL	SEE NOTE 1		.135 SHEET
			1		2	3	NEUTRON ABSORBER	BORAL	SEE NOTE 1		.135 SHEET
1		1		1		2	TUBING	304 ST. STL.	ASTM A240		18 GA SHEET
	1		1		1	1	TUBING	304 ST. STL.	ASTM A240		18 GA SHEET

QUANTITY (10)

UNLESS OTHERWISE STATED
DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.
ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

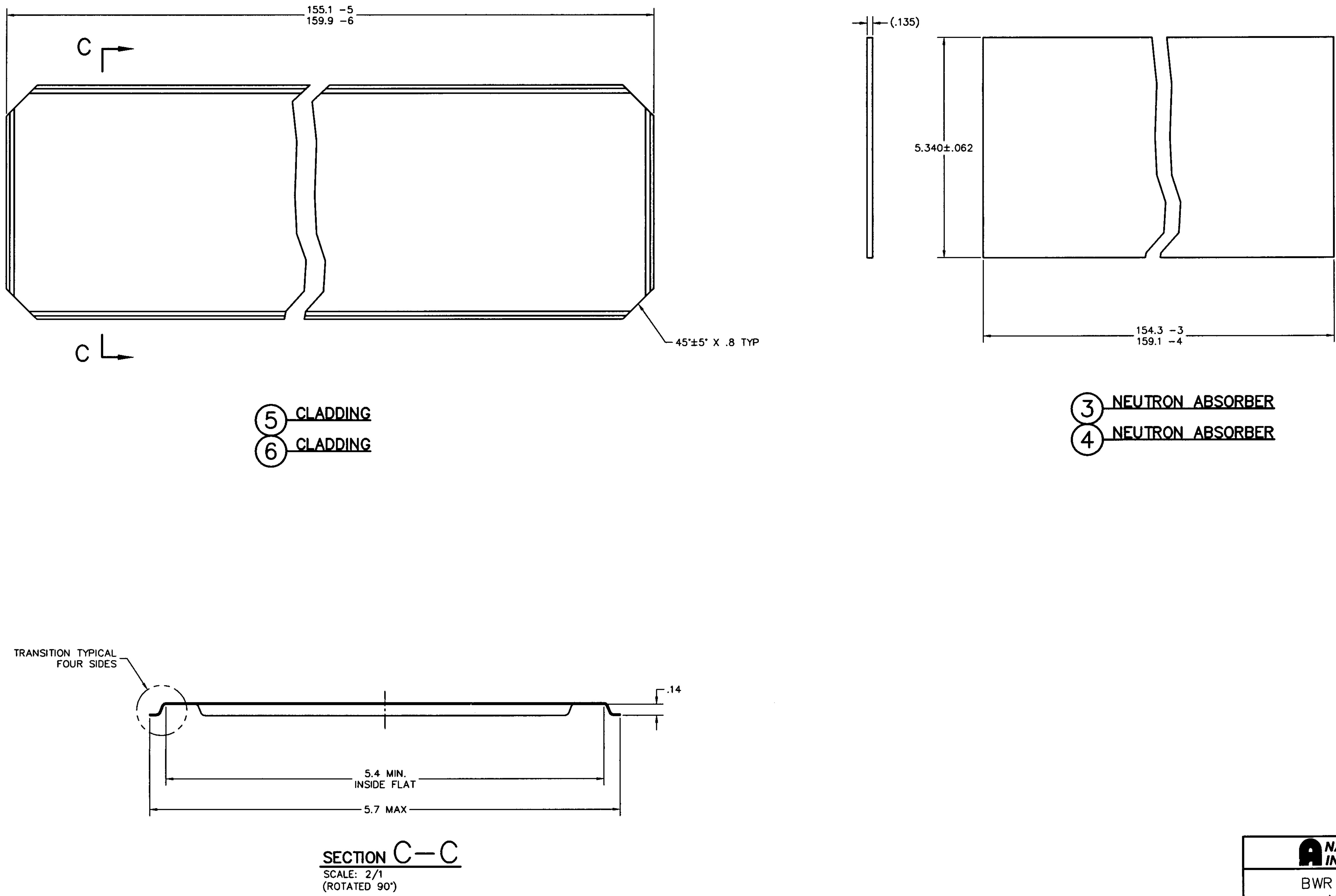
.XXX	UNDER 3	±.003	WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
	3–12	±.005	
	OVER 12	±.010	ALL DIMENSIONS ARE IN INCHES
.XX	UNDER 6	±.02	BORDER SIZE: F (40 X 28)
	6–18	±.03	ALL UNSPECIFIED TOOL RADII: .015 – .030
	OVER 18	±.06	BREAK ALL SHARP CORNERS .015 – .030
.X	ALL	±.1	MACHINED SURFACES TO BE ▽ OR BETTER
FRACTIONAL	±1/8		NEXT ASSEMBLY: 790-570
ANGLES ±0.5°			DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. Walker	9-22-04
CHECKER	[signature]	9-28-04
PROJECT MANAGER	[signature]	9/23/04
ENGINEERING	[signature]	9/24/04
LICENSING	[signature]	9-23-04
QUALITY	[signature]	9/23/04

NAC INTERNATIONAL

BWR FUEL TUBE, NAC–UMS®

PROJECT 790 | DRAWING 575 | REV 10
SCALE 1/1 | WEIGHT | SH 1 OF 2 | 9:54AM 9-22-2004



NAC INTERNATIONAL

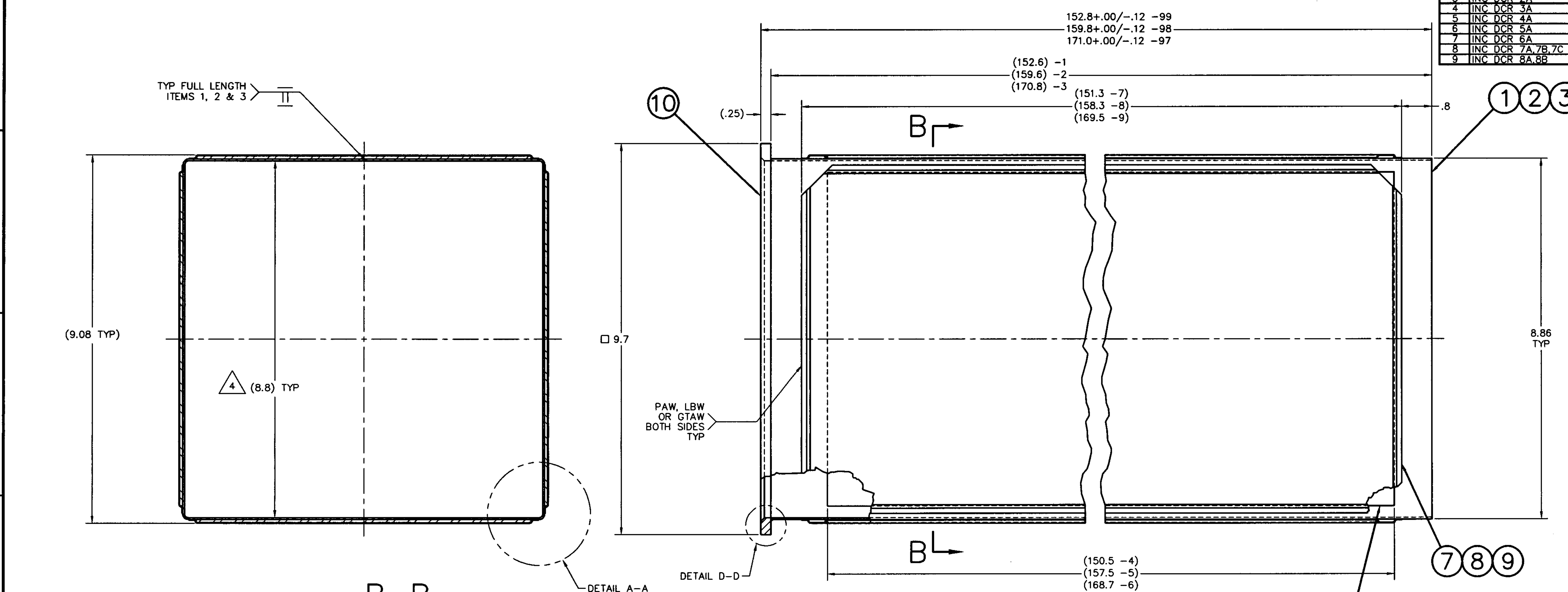
BWR FUEL TUBE, NAC-UMS®

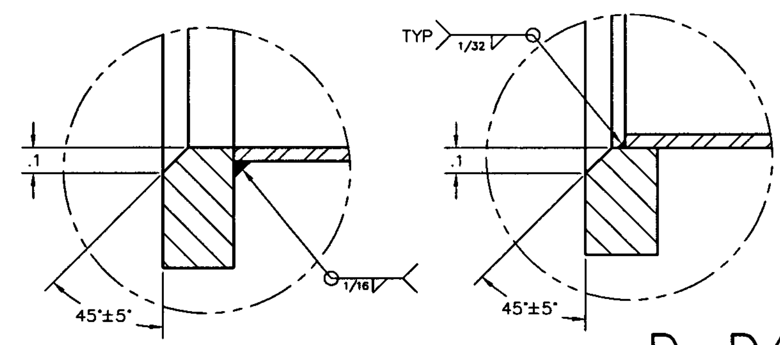
PROJECT 790 | DRAWING 575 | REV 10

SCALE 1/1 | WEIGHT | SH 2 OF 2 | 9:54AM 9-22-2004

L1 202204

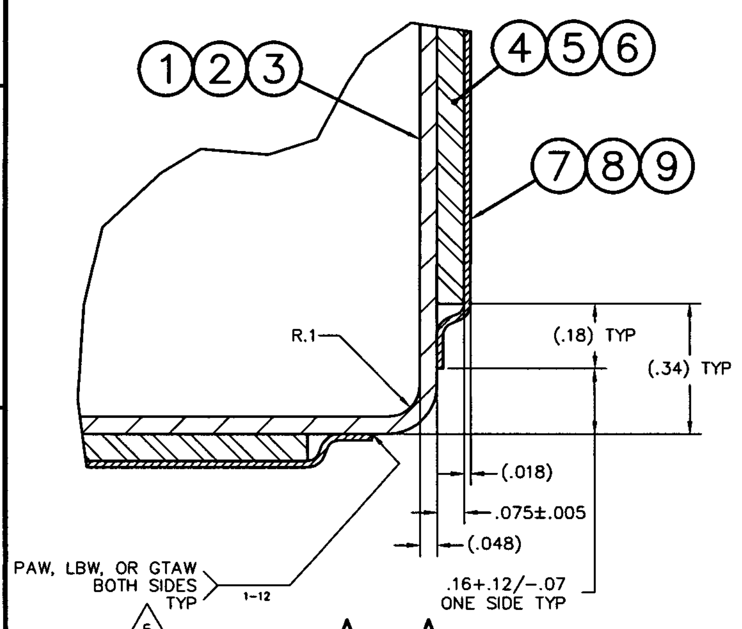
REV	CHANGE
0	INITIAL ISSUE
1	INC. DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A
7	INC DCR 6A
8	INC DCR 7A,7B,7C
9	INC DCR 8A,8B





DETAIL D–D
SCALE: 4:1

DETAIL D–D
SCALE: 4:1
(ALTERNATE FABRICATION)



DETAIL A–A
SCALE: 5/1

99 TUBE ASSEMBLY

98 TUBE ASSEMBLY

97 TUBE ASSEMBLY

97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
1	1	1	10	FLANGE	304 ST.STL.	ASTM A240		1/4 PLATE
4			9	CLADDING	304 ST.STL.	ASTM A240		26 GA SHEET
	4		8	CLADDING	304 ST.STL.	ASTM A240		26 GA SHEET
		4	7	CLADDING	304 ST.STL.	ASTM A240		26 GA SHEET
4			6	NEUTRON ABSORBER	BORAL	SEE NOTE 1		.075 SHEET
	4		5	NEUTRON ABSORBER	BORAL	SEE NOTE 1		.075 SHEET
		4	4	NEUTRON ABSORBER	BORAL	SEE NOTE 1		.075 SHEET
1			3	TUBING	304 ST.STL.	ASTM A240		18 GA SHEET
	1		2	TUBING	304 ST.STL.	ASTM A240		18 GA SHEET
		1	1	TUBING	304 ST.STL.	ASTM A240		18 GA SHEET
ASSY	ASSY	ASSY						
QUANTITY								

(9)

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M–94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3–12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6–18	±.03
	OVER 18	±.06
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY

ALL DIMENSIONS ARE IN INCHES

BORDER SIZE: F (40 X 28)

ALL UNSPECIFIED TOOL RADII: .015 – .030

BREAK ALL SHARP CORNERS .015 – .030

MACHINED SURFACES TO BE ∇ OR BETTER

NEXT ASSEMBLY: 790–595

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. Walker	9-22-04
CHECKER	[signature]	9-23-04
PROJECT MANAGER	[signature]	9/23/04
ENGINEERING	[signature]	9/23/04
LICENSING	J.C. Thompson	9-23-04
QUALITY	B. Barrett for KH	9/23/04

NAC INTERNATIONAL

PWR FUEL TUBE, NAC–UMS®

PROJECT 790 | DRAWING 581 | REV 9

SCALE FULL | WEIGHT | SH 1 OF 2 | 8:54AM 9-22-2004

5 ALTERNATIVELY, CLADDING MAY BE ATTACHED TO TUBE VIA A CONTINUOUS WELD.

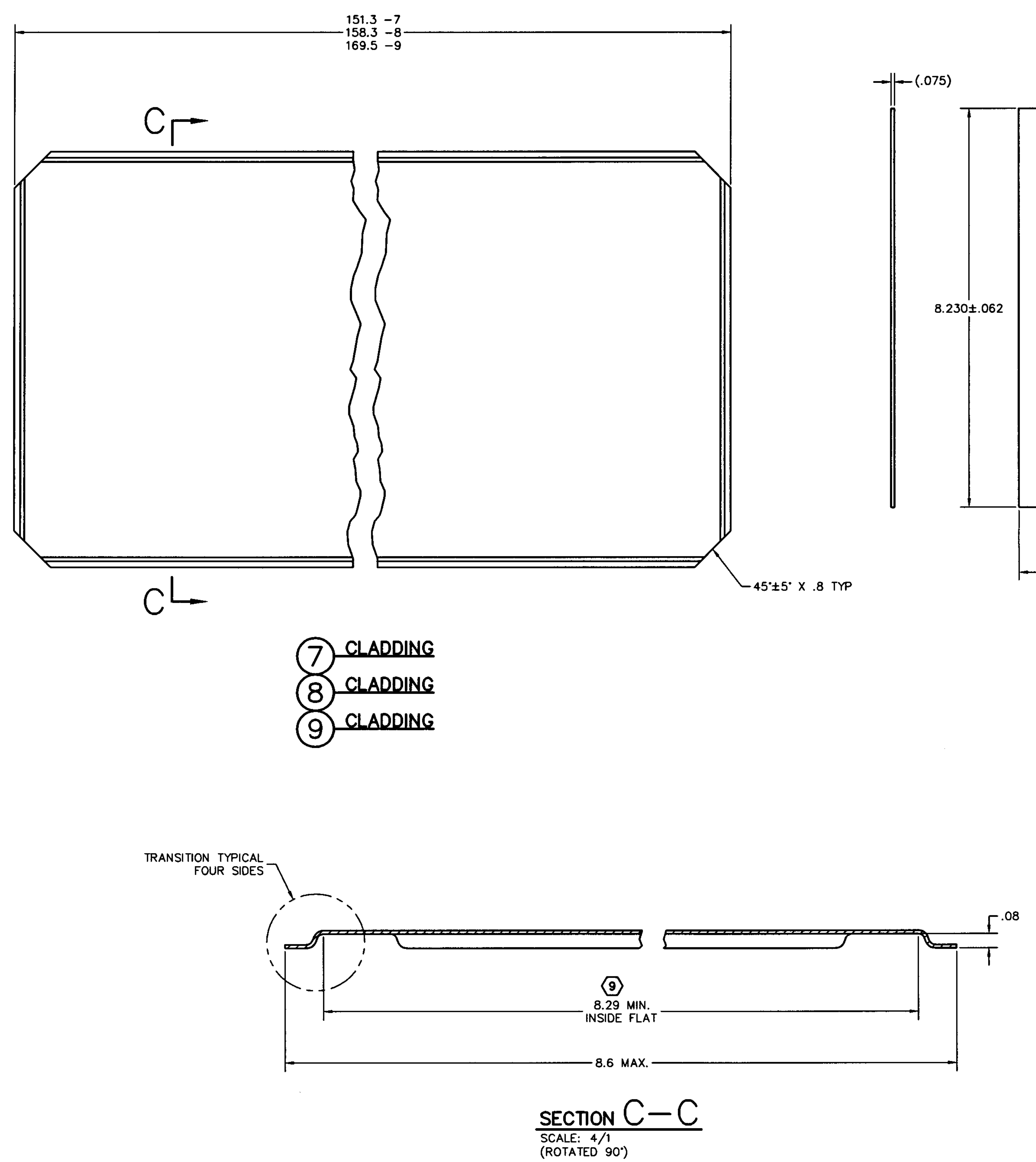
4 INSIDE OPENING TO BE GAGED MIN. 8.65" SQUARE X 12' LONG.

3. VISUALLY INSPECT (VT) ALL WELDS IN ACCORDANCE WITH ASME SECTION V, ARTICLE 9. ACCEPTANCE PER ASME SECTION III, NG–5360.

2. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH ASME SECTION IX.

1. NEUTRON ABSORBER SHEETS TO PROVIDE MINIMUM AREAL DENSITY OF .025 g/cm² B10.

NOTES:



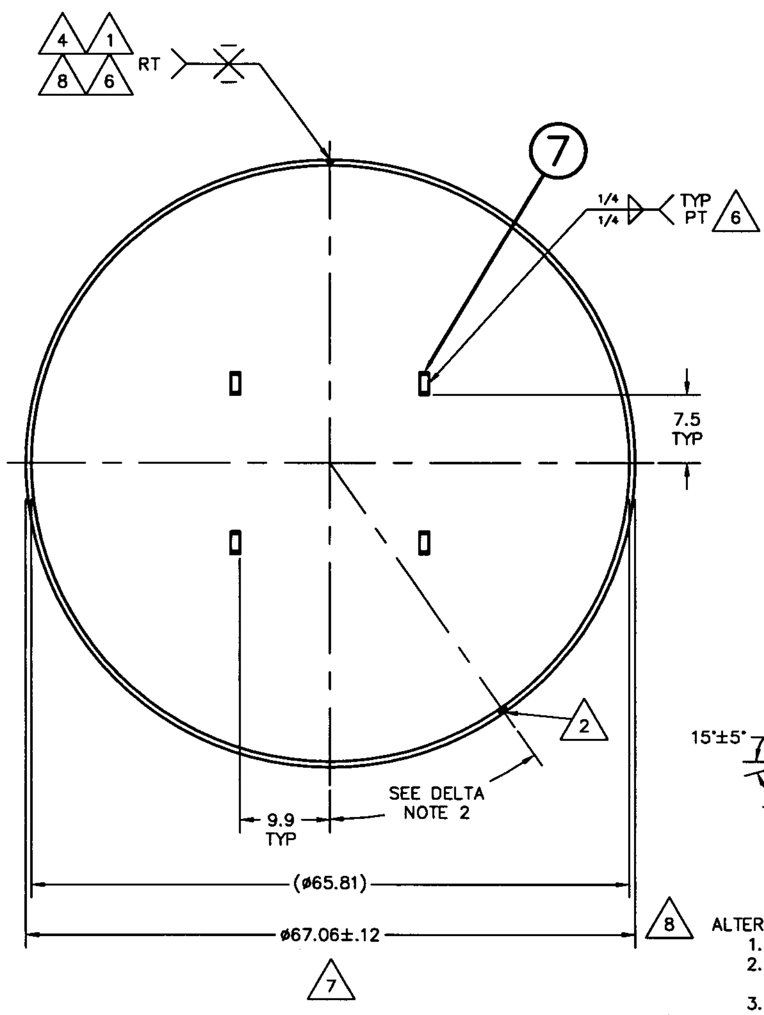
NAC INTERNATIONAL

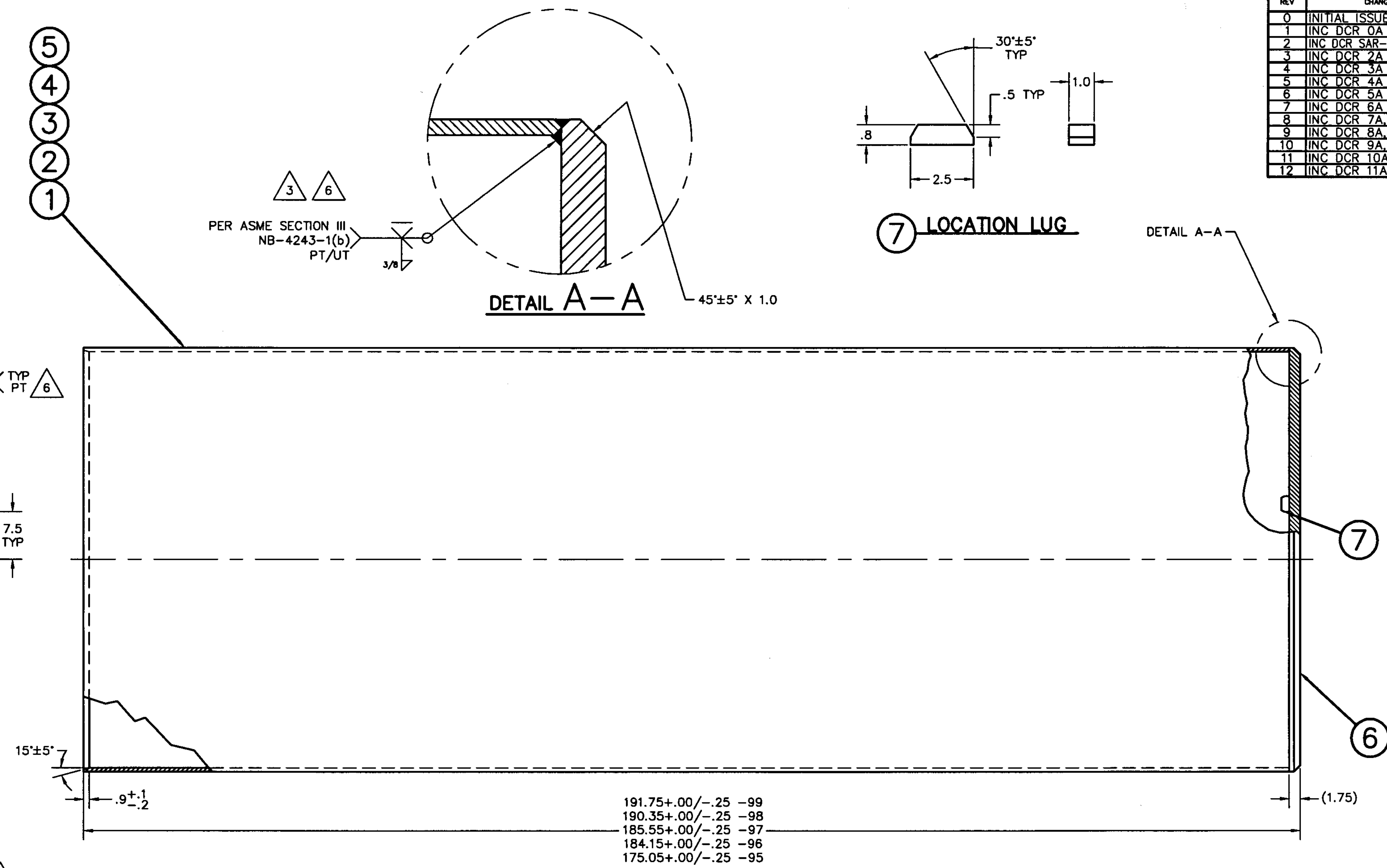
PWR FUEL TUBE, NAC-UMS®

PROJECT	DRAWING	REV
790	581	9
SCALE FULL	SH 2 OF 2	8:54AM 9-22-2004

R-203047

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR SAR-001-0A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A
7	INC DCR 6A
8	INC DCR 7A,7B
9	INC DCR 8A,8B,8C
10	INC DCR 9A,9B
11	INC DCR 10A
12	INC DCR 11A





99 SHELL WELDMENT

98 SHELL WELDMENT

97 SHELL WELDMENT

96 SHELL WELDMENT

95 SHELL WELDMENT

8. ALTERNATE WELD ALLOWED PROVIDED THE FOLLOWING CRITERIA ARE SATISFIED:
   1. ALTERNATE WELD JOINT ACHIEVES COMPLETE JOINT PENETRATION.
   2. ALTERNATE WELD JOINT MEETS ALL REQUIREMENTS OF SECTION III, SUBSECTION NB.
   3. ALTERNATE WELD JOINT IS APPROVED BY NAC.

(12) 7. TOLERANCES APPLIED TO LOCALIZED AREAS OF SHELL DIAMETER ARE +.21"/-.24", OR ALTERNATIVELY, THE MINIMUM INSIDE DIAMETER OF THE SHELL WELDMENT, WITH THE SHELL WELDMENT VERTICAL, SHALL BE THE AS-BUILT DIAMETER OF THE SHIELD LID ASSEMBLY PLUS .01". LOCALIZED AREAS BEING FLATSPOTS, BULGES, ETC. AT NO POINT SHALL THERE BE ANY INTERFERENCE BETWEEN THE INTERFACING COMPONENTS AND THE SHELL.

6. LIQUID PENETRANT EXAMINED PER ASME SECTION V, ARTICLE 6. ACCEPTANCE PER SECTION III, ARTICLE NB-5350.

5. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH SECTION IX.

4. RADIOGRAPHIC EXAMINED PER ASME SECTION V, ARTICLE 2. ACCEPTANCE PER SECTION III, ARTICLE NB-5320.

3. ULTRASONIC EXAMINED PER ASME SECTION V, ARTICLE 5. ACCEPTANCE PER SECTION III, DIVISION 1, ARTICLE NB-5330.

2. OPTIONAL ENGRAVED STRIPE – FOR ASSY-99,96,95 ANGLE IS 35°±5° FOR ASSY-98,97 ANGLE IS 40°±5° ENGRAVE STRIPE 1.0 WIDE X .03 DEEP AND FILL WITH WEATHER RESISTANT YELLOW PAINT.

1. TYPICAL FOR SEAM AND GIRTH WELDS, NUMBER AND LOCATION OPTIONAL. ADJACENT SECTIONS WITH SEAM WELDS SHALL BE OFFSET.

NOTES:

95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
4	4	4	4	4	7	LOCATION LUG	304 ST.STL.	ASTM A240/A276		PLATE/BAR
1	1	1	1	1	6	BOTTOM	304L ST.STL.	ASME SA240		1 3/4 PLATE
				1	5	SHELL	304L ST.STL.	ASME SA240		5/8 PLATE
			1		4	SHELL	304L ST.STL.	ASME SA240		5/8 PLATE
		1			3	SHELL	304L ST.STL.	ASME SA240		5/8 PLATE
	1				2	SHELL	304L ST.STL.	ASME SA240		5/8 PLATE
1					1	SHELL	304L ST.STL.	ASME SA240		5/8 PLATE

QUANTITY

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003	WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
	3-12	±.005	ALL DIMENSIONS ARE IN INCHES
	OVER 12	±.010	BORDER SIZE: F (40 X 28)
.XX	UNDER 6	±.02	ALL UNSPECIFIED TOOL RADII: .015 – .030
	6-18	±.03	BREAK ALL SHARP CORNERS .015 – .030
	OVER 18	±.06	MACHINED SURFACES TO BE ▽ OR BETTER
.X	ALL	±.1	NEXT ASSEMBLY: 790-585
FRACTIONAL ±1/8			DRAWING TYPE: LICENSE
ANGLES ±0.5°			

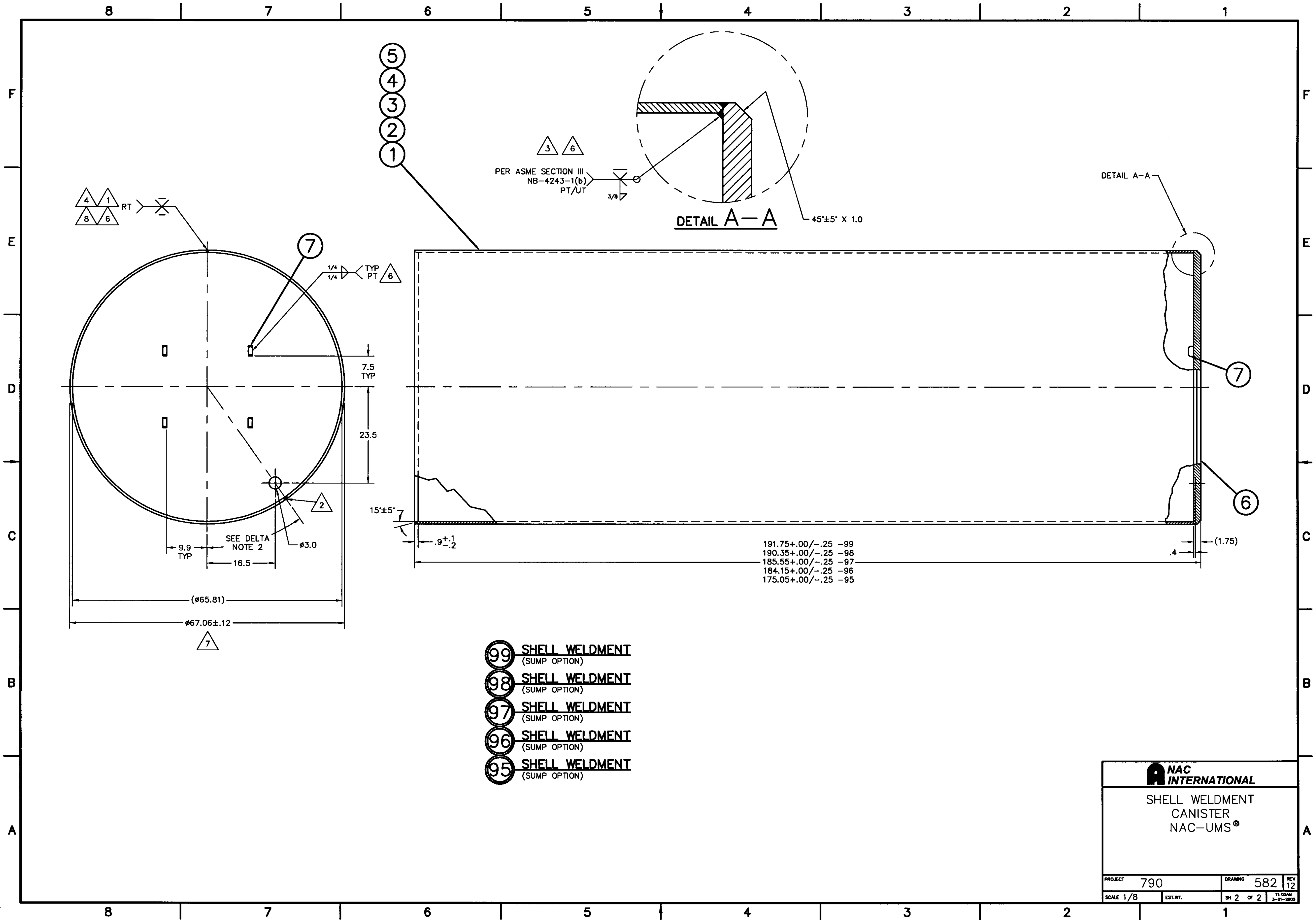
GROUP	NAME	DATE
PREPARER	R. Walker	3-21-05
CHECKER	[signature]	3-21-05
PROJECT MANAGER	[signature]	3/22/05
ENGINEERING	[signature]	3/24/05
LICENSING	[signature]	3/24/05
QUALITY	[signature]	3/25/05

NAC INTERNATIONAL

SHELL WELDMENT
CANISTER
NAC-UMS®

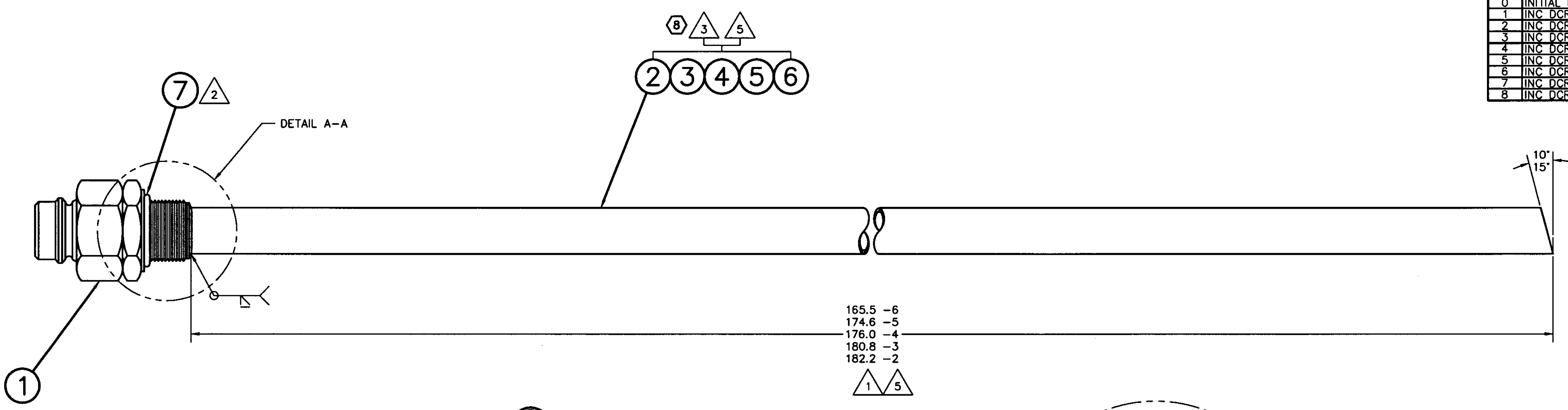
PROJECT 790 | DRAWING 582 | REV 12

SCALE 1/8 | WEIGHT | SH 1 OF 2 | 11:05AM 3-21-2005

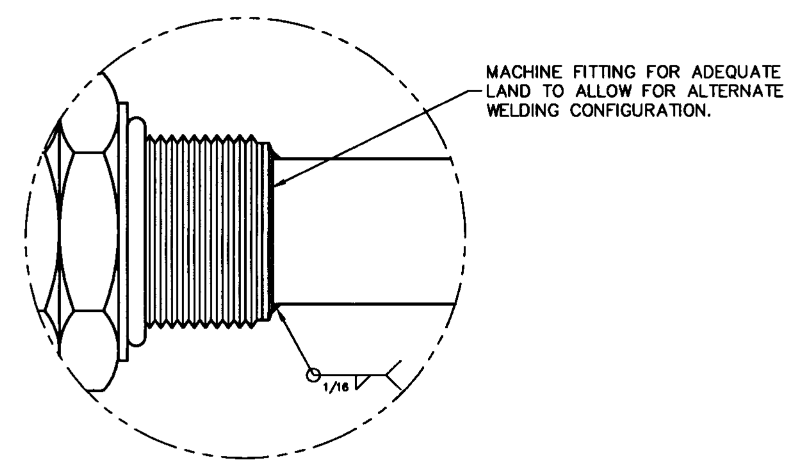

DETAIL A–A
45°±5° X 1.0
PER ASME SECTION III
NB-4243-1(b)
PT/UT
3/8
3
6
DETAIL A-A
RT
4
1
8
6
1/4
1/4
TYP
PT
6
7.5
TYP
23.5
2
SEE DELTA
NOTE 2
ø3.0
9.9
TYP
16.5
(ø65.81)
ø67.06±.12
7
15°±5°
.9+.1/-.2
191.75+.00/-.25 -99
190.35+.00/-.25 -98
185.55+.00/-.25 -97
184.15+.00/-.25 -96
175.05+.00/-.25 -95
.4
(1.75)
99 SHELL WELDMENT (SUMP OPTION)
98 SHELL WELDMENT (SUMP OPTION)
97 SHELL WELDMENT (SUMP OPTION)
96 SHELL WELDMENT (SUMP OPTION)
95 SHELL WELDMENT (SUMP OPTION)
NAC INTERNATIONAL
SHELL WELDMENT
CANISTER
NAC-UMS®
PROJECT 790
DRAWING 582
REV 12
SCALE 1/8
EST.WT.
SH 2 OF 2
11:05AM 3-21-2005

R-203048

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A,5B,5C
7	INC DCR 6A,6B,6C
8	INC DCR 7A



99 DRAIN TUBE ASSEMBLY
98 DRAIN TUBE ASSEMBLY
97 DRAIN TUBE ASSEMBLY
96 DRAIN TUBE ASSEMBLY
95 DRAIN TUBE ASSEMBLY



DETAIL A—A
SCALE: 2:1
(ALTERNATE WELDING CONFIG.)

NOTES:

(5) WHEN USED WITH OPTIONAL COUNTERBORE DETAIL ON DRAWING 790-582, TRIAL FIT AND ADJUST LENGTH IF NEEDED TO ENSURE A .12+/-.1 CLEARANCE BETWEEN END OF TUBE AND BOTTOM OF THE .38 DEEP COUNTERBORE IN THE BOTTOM PLATE.

4. (DELETED)

(8) (3) MINOR SURFACE INDENTATIONS AND/OR DIMENSIONAL DEVIATIONS ARE ACCEPTABLE FOR ITEMS 2-6, TUBE.

(2) ITEM 7, SEAL TO BE SPECIFIED BY USER: ELASTOMER SEAL (VITON OR EPDM) OR A METAL SEAL.

(1) WHEN THE OPTIONAL SUMP SHOWN ON DRAWING 790-582 IS USED THESE DIMENSIONS SHALL BE: (165.9) -6
(175.0) -5
(176.4) -4
(181.2) -3
(182.6) -2

(8) (3) (5)

95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
1	1	1	1	1	7	SEAL	SEE NOTE 2	COML		SEE NOTE 2
1					6	TUBE	304 ST. STL.	COML		1 OD X .035 WALL TUBE
	1				5	TUBE	304 ST. STL.	COML		1 OD X .035 WALL TUBE
		1			4	TUBE	304 ST. STL.	COML		1 OD X .035 WALL TUBE
			1		3	TUBE	304 ST. STL.	COML		1 OD X .035 WALL TUBE
				1	2	TUBE	304 ST. STL.	COML		1 OD X .035 WALL TUBE
1	1	1	1	1	1	NIPPLE	ST. STL.	COML		QUICK DISCONNECT COUPLING

QUANTITY

UNLESS OTHERWISE STATED
DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.
ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3-12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6-18	±.03
	OVER 18	±.06
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
ALL DIMENSIONS ARE IN INCHES
BORDER SIZE: F (40 X 28)
ALL UNSPECIFIED TOOL RADII: .015 - .030
BREAK ALL SHARP CORNERS .015 - .030
MACHINED SURFACES TO BE 125 OR BETTER
NEXT ASSEMBLY: 790-585
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER		3-21-05
CHECKER		3/22/05
PROJECT MANAGER		3/22/05
ENGINEERING		3/24/05
LICENSING		3/24/05
QUALITY		3/25/05

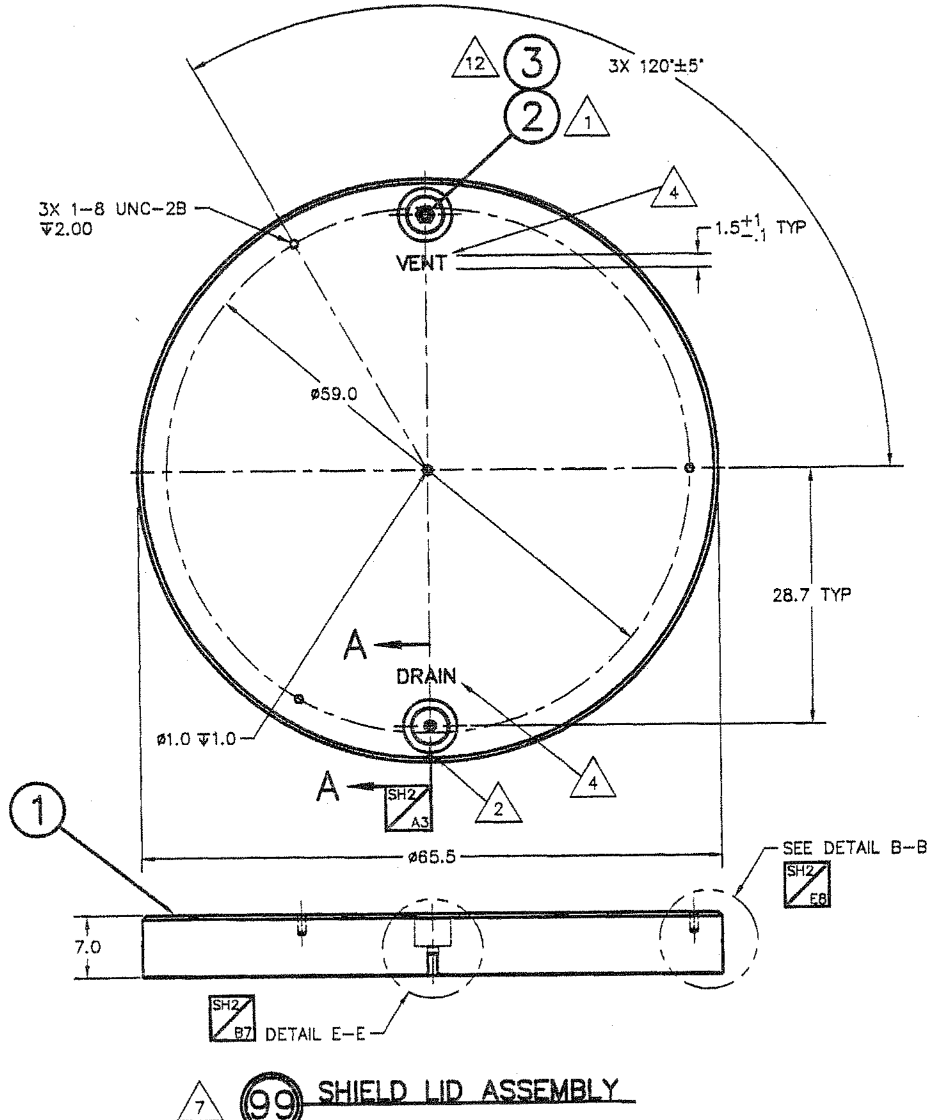
NAC INTERNATIONAL

ASSEMBLY, DRAIN TUBE, CANISTER NAC-UMS®

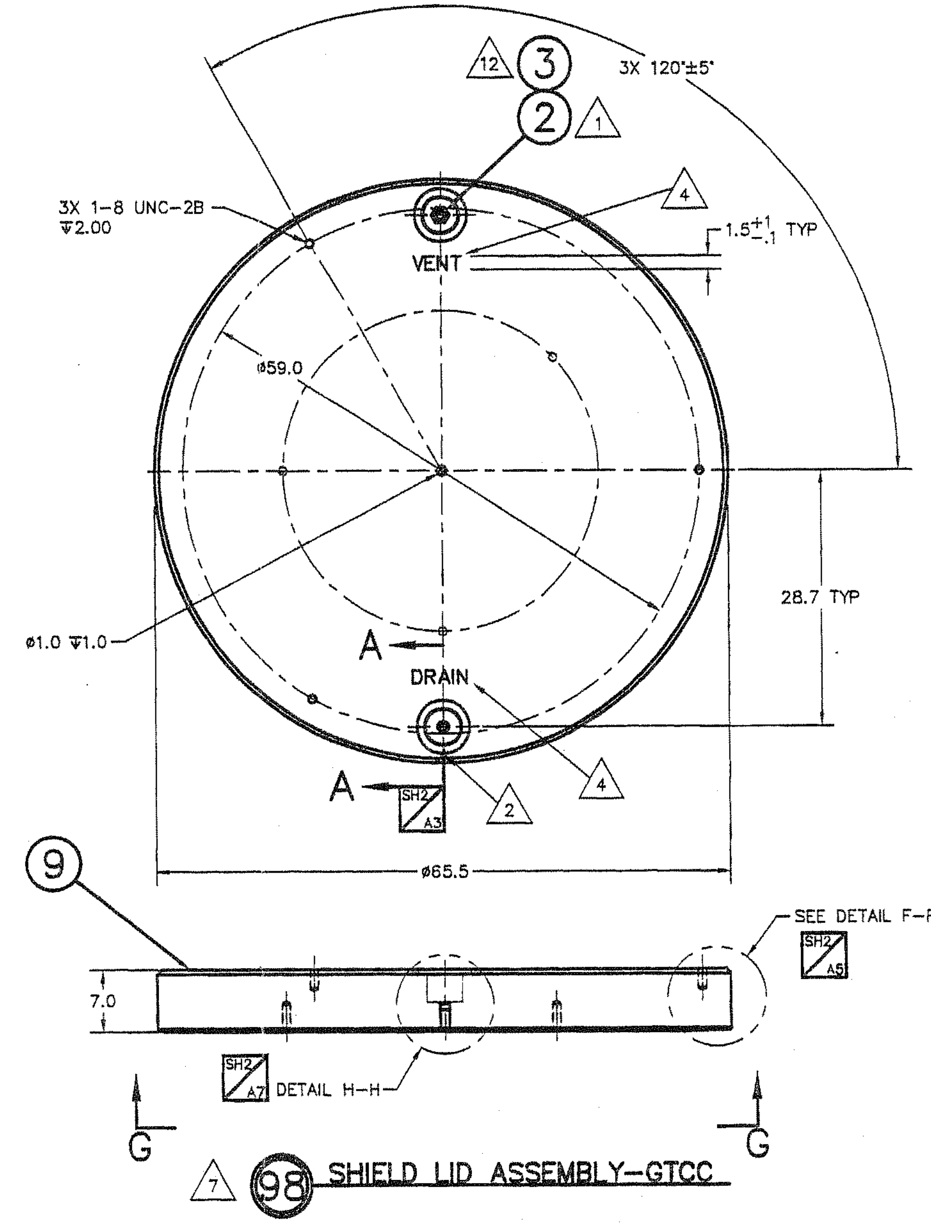
PROJECT 790 | DRAWING 583 | REV 8
SCALE 1/1 | WEIGHT | SH 1 OF 1 | 11:32AM 3-21-2005

R- 207964

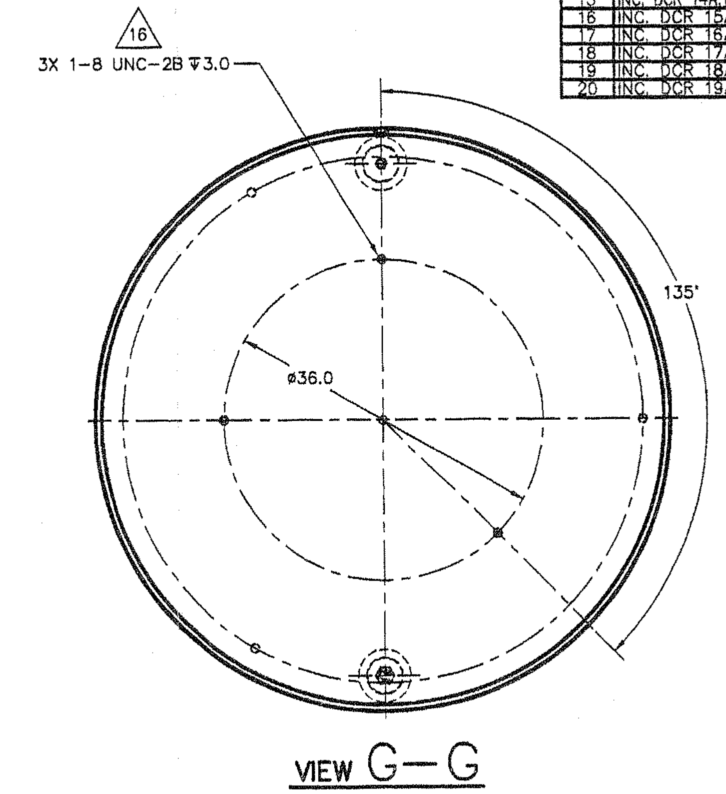
REV	CHANGE
0	INITIAL ISSUE
1	INC. DCR 0A
2	INC. DCR 1A
3	INC. DCR 2A
4	INC. DCR 3A
5	INC. DCR 4A
6	INC. DCR 5A
7	INC. DCR 6A
8	INC. DCR 7A
9	INC. DCR 8A
10	INC. DCR 9A,9B,9C
11	INC. DCR 10A,10B,10C,10D,10E
12	INC. DCR 11A,11B,11C
13	INC. DCR 12A,12B,12C
14	INC. DCR 13A,13B,13C,13D,13E,13F
15	INC. DCR 14A,14B,14C,14D
16	INC. DCR 15A
17	INC. DCR 16A
18	INC. DCR 17A
19	INC. DCR 18A
20	INC. DCR 19A



7 (99) SHIELD LID ASSEMBLY



7 (98) SHIELD LID ASSEMBLY-GTCC



VIEW G-G

8. WELD PREPARATION SHALL BE DETERMINED BY THE FABRICATOR BASED UPON THE WELD PROCESS USED. SEE DRAWING 790-585 AND 790-612 FOR EFFECTIVE THROAT SIZE OF WELD.

7. FOR ITEMS 1, 4 AND 9, THE SA182 FORGING MATERIAL MUST HAVE YIELD AND ULTIMATE STRENGTHS EQUAL TO OR GREATER THAN THE SA240 PLATE MATERIAL.

6. MAY BE FABRICATED USING MULTIPLE SECTIONS. ALL SEAMS BETWEEN SECTIONS MUST BE FULL PENETRATION WELDED UPON ASSEMBLY.

5. STEEL STAMP/ENGRAVE 1/2" HIGH THE FOLLOWING SEQUENCE LOCATED APPROX. AS SHOWN. "UMS-TSC-XXX-YYY" WHERE XXX IS INDICATED ON THE PURCHASE ORDER FOR A PARTICULAR PROJECT CODE, OR OTHER UNIQUE IDENTIFYING NUMBER, AND THE YYY IS A SEQUENTIAL SERIES OF NUMBERS STARTING WITH 001. "CALCULATED EMPTY CANISTER WEIGHT = ZZ,ZZZ LBS." WHERE ZZ,ZZZ IS INDICATED ON THE PURCHASE ORDER. DIRECTLY BELOW THIS NUMBER IS AN OPEN AREA FOR EACH CUSTOMER TO ADD ANY REQUIRED INFORMATION THEY CHOOSE.

4. METAL STAMP/ENGRAVE CHARACTERS APPROX. AS SHOWN X .03 DEEP, FILL WITH BLACK WEATHER RESISTANT PAINT.

3. LOCATION OF KEY SLOT IS TYPICAL ONE PLACE ONLY.

2. ENGRAVE DELTA .5" PER SIDE, OR CIRCLE .5" DIAMETER, X .03" DEEP, NOT TO INFRINGE ON THE WELD BEVEL, AND FILL WITH WEATHER RESISTANT BLACK PAINT.

1. LUBRICATE WITH A SPENT FUEL COMPATIBLE LUBRICANT SUCH AS NEOLUBE. INSTALL FITTING BY HAND UNTIL METAL TO METAL CONTACT IS ACHIEVED THEN TORQUE TO (115±5 FT-LBS FOR ELASTOMER SEALS) OR (135±15 FT-LBS FOR METAL SEALS).

NOTES:

16. ALTERNATIVELY, THE LOCATION OF THESE THREE SHIELD COVER ATTACHMENT HOLES CAN BE MATCHED DRILLED WITH THE SHIELD COVER.

15. SMALL INDENTATIONS AND IMPERFECTIONS ARE ALLOWED IN THE LAND OR GROOVE OF THE STRUCTURAL OR SHIELD LID WELD PREP, PROVIDING THAT THEY DO NOT ADVERSELY AFFECT THE WELDING PROCEDURES OR WELD PROCEDURE QUALIFICATIONS.

14. (DELETED)

13. SURFACE FINISH REQUIREMENTS ARE OPTIONAL FOR DETAIL J-J.

12. ITEM 3, SEAL TO BE SPECIFIED BY USER: ELASTOMER SEAL (VITON OR EPDM) OR A METAL SEAL.

11. TOOL MARKS AND OTHER MARKS ARE ACCEPTABLE ON ALL UNSPECIFIED MACHINED SURFACES AS LONG AS REQUIRED THICKNESS/DIAMETER OF ITEMS ARE MET.

10. ITEM 6, LID SUPPORT TO BE FIT-UP AND WELDED TO SHELL INSIDE DIAMETER IN A MANNER TO ASSURE MAINTENANCE OF THE 1" NOMINAL GAP.

9. MINIMUM OF .125 OF MATERIAL IS REQUIRED TO BE UNDERNEATH BOLT HOLE.

Note	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
7	1			9	SHIELD LID-GTCC	304 ST. STL.	ASME SA240/SA182		PLATE/FORGING
				8	KEY	304 ST. STL.	ASTM A240/A276		1/2 PLATE/BAR
6				7	SPACER RING	304 ST. STL.	ASME SA479/SA240		1/2 X 1/2 SQ. BAR/PLATE
				6	LID SUPPORT RING	304 ST. STL.	ASME SA479/SA240		1/2 X 1/2 SQ. BAR/PLATE
7				5	COVER	304 ST. STL.	ASME SA479		BAR
				4	STRUCTURAL LID	304L ST. STL.	ASME SA240/SA182		PLATE/FORGING
	1	1		3	SEAL	SEE NOTE 12	COML		SEE NOTE 12
	1	1		2	NIPPLE	ST. STL.	COML		QUICK DISCONNECT COUPLING
7		1		1	SHIELD LID	304 ST. STL.	ASME SA240/SA182		PLATE/FORGING

QUANTITY

UNLESS OTHERWISE STATED
DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94. UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.
ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3-12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6-18	±.03
	OVER 18	±.05
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
ALL DIMENSIONS ARE IN INCHES
BORDER SIZE: F (40 X 28)
ALL UNSPECIFIED TOOL RADII: .015 - .030
BREAK ALL SHARP CORNERS .015 - .030
MACHINED SURFACES TO BE ∇ OR BETTER
NEXT ASSEMBLY: 790-585
DRAWING TYPE: LICENSE

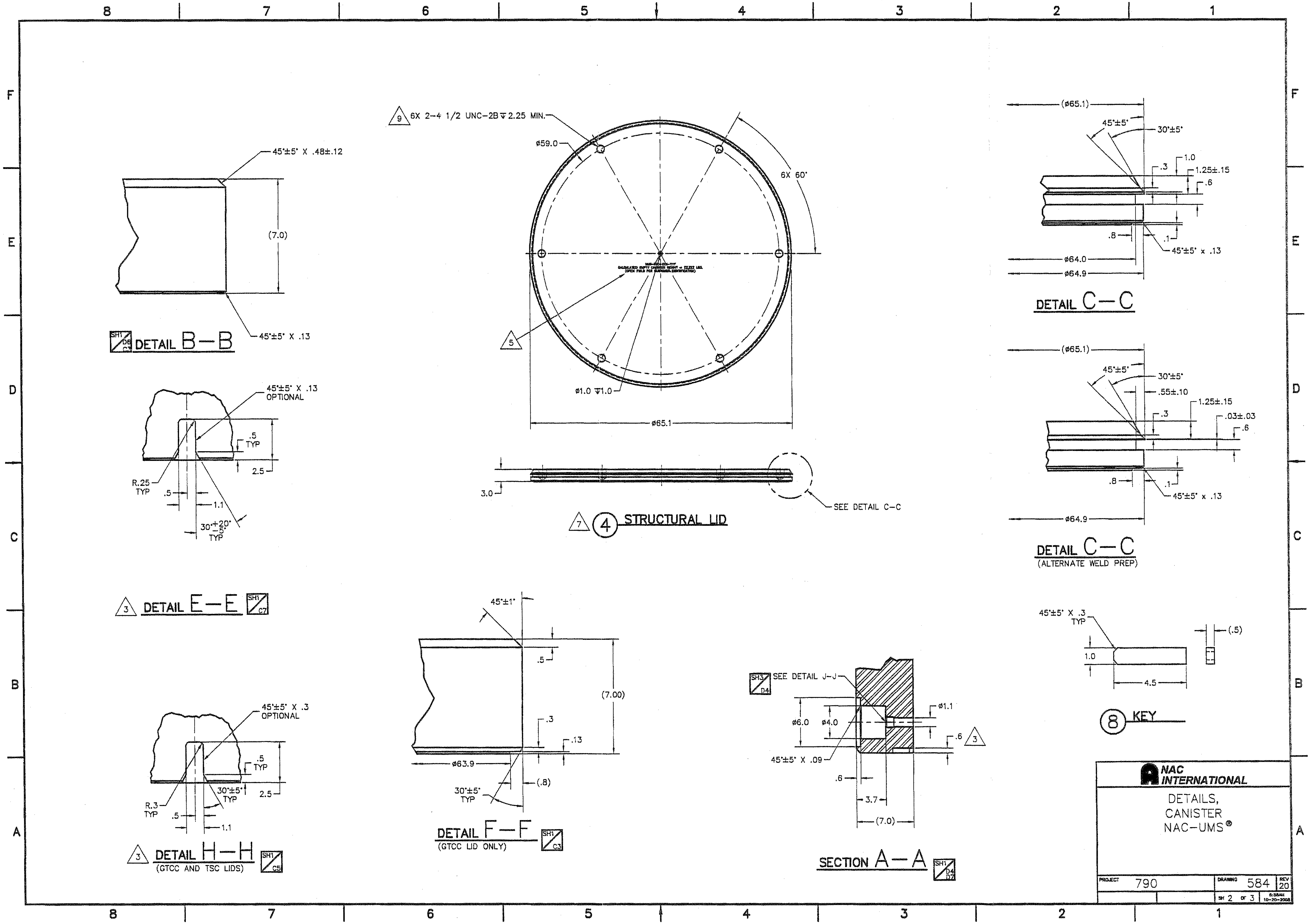
(20)

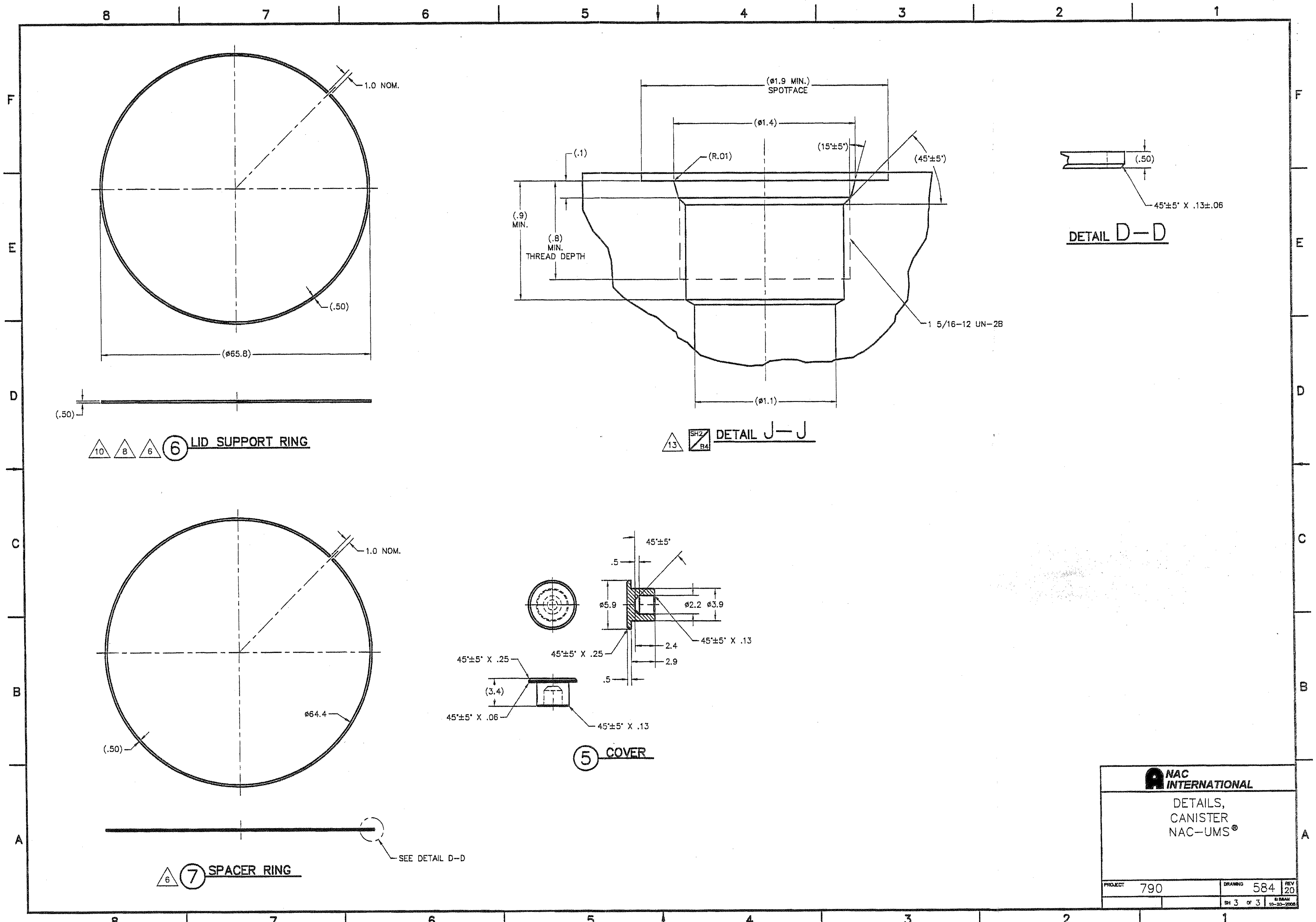
GROUP	NAME	DATE
PREPARER	[signature]	10/20/08
CHECKER	[signature]	10/21/08
PROJECT MANAGER	[signature]	10/22/08
ENGINEERING	[signature]	10-22-08
LICENSING	[signature]	10/24/08
QUALITY	[signature]	10/31/08

NAC INTERNATIONAL

DETAILS,
CANISTER
NAC-UMS®

PROJECT 790 | DRAWING 584 | REV 20
SCALE | WEIGHT | SH 1 OF 3 | 0:57AM 10-20-2008


6X 2-4 1/2 UNC-2B ⌴ 2.25 MIN.
ø59.0
6X 60°
ø1.0 ⌴1.0
ø65.1
3.0
SEE DETAIL C-C
4 STRUCTURAL LID
DETAIL B-B
45°±5' X .48±.12
(7.0)
45°±5' X .13
DETAIL E-E
45°±5' X .13
OPTIONAL
.5
TYP
2.5
R.25
TYP
1.1
30°+20° -5°
TYP
DETAIL H-H
(GTCC AND TSC LIDS)
45°±5' X .3
OPTIONAL
30°±5'
TYP
R.3
TYP
DETAIL F-F
(GTCC LID ONLY)
45°±1°
(7.00)
.13
ø63.9
(.8)
30°±5'
TYP
SECTION A-A
SEE DETAIL J-J
ø6.0
ø4.0
ø1.1
45°±5' X .09
.6
3.7
(7.0)
DETAIL C-C
(ø65.1)
45°±5'
30°±5'
1.0
1.25±.15
.3
.6
.8
.1
45°±5' x .13
ø64.0
ø64.9
DETAIL C-C
(ALTERNATE WELD PREP)
.55±.10
.03±.03
8 KEY
45°±5' X .3
TYP
(.5)
4.5
NAC INTERNATIONAL
DETAILS,
CANISTER
NAC-UMS®
PROJECT 790
DRAWING 584
REV 20
SH 2 OF 3



1.0 NOM.
(.50)
(ø65.8)
(.50)
10 8 6 (6) LID SUPPORT RING
1.0 NOM.
ø64.4
(.50)
SEE DETAIL D-D
6 (7) SPACER RING
(ø1.9 MIN.) SPOTFACE
(ø1.4)
(R.01)
(15°±5°)
(45°±5°)
(.1)
(.9) MIN.
(.8) MIN. THREAD DEPTH
1 5/16-12 UN-2B
(ø1.1)
13 SH2 B4 DETAIL J—J
(.50)
45°±5° X .13±.06
DETAIL D—D
45°±5°
.5
ø5.9
ø2.2 ø3.9
45°±5° X .25
45°±5° X .13
2.4
2.9
45°±5° X .25
(3.4)
45°±5° X .06
45°±5° X .13
(5) COVER
NAC INTERNATIONAL
DETAILS, CANISTER NAC-UMS®
PROJECT 790
DRAWING 584
REV 20
SH 3 OF 3
10-20-2006

l-231616

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR SAR-001-0A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A
7	INC DCR 6A
8	INC DCR 7A,7B,7C
9	INC DCR 8A
10	INC DCR 9A,9B,9C
11	INC DCR 10A,10B,10C,10D,10E
12	INC DCR 11A
13	INC DCR 12A,12B,12C
14	INC DCR 13B,13C,13D,13E
15	INC DCR 14A,14B,14C
16	INC DCR 15A
17	INC DCR 16A
18	INC DCR 17A
19	INC DCR 18A
20	INC DCR 19A
21	INC DCR 20A
22	INC DCR 21A

12. DELETED

11. FIELD INSTALLED CODE NAMEPLATES SHALL BE LOCATED ON THE STRUCTURAL LID AS DIRECTED BY THE CLIENT. CODE NAMEPLATES TO BE WELDED TO THE STRUCTURAL LID WITH FOUR FILLET TACK WELDS 1/4 INCH LONG (MINIMUM), ONE ADJACENT TO EACH CORNER OF THE PLATE. FILLET WELD SIZE TO BE EQUAL TO THE THICKNESS OF THE NAMEPLATE.

10 (flag). AT THE OPTION OF THE USER, SPACER SHIMS (DRAWING 790-587) MAY BE USED TO ADJUST THE STACK HEIGHT OF THE STRUCTURAL LID AND ENSURE THAT THE TOP OF THE LID IS AT EITHER LEVEL WITH THE TOP OF THE CANISTER OR EXTENDS PAST THE TOP EDGE OF THE CANISTER, SO LONG AS THE THE MIN. EFFECTIVE WELD THROAT IS MAINTAINED FOR THE CLOSURE WELD.

9 (flag). AT THE OPTION OF THE USER, STAINLESS STEEL (ASTM/ASME A/SA240, TYPE 304/304L) SHIMS OF APPROPRIATE THICKNESS MAY BE USED IN THE WELDING OF THE SHIELD LID (ITEM 17) TO THE SHELL WELDMENT (ITEMS 1-5).

8 (flag). THE USE OF THE STRUCTURAL LID AND SHIELD LID THREADED PLUGS AND DOWEL PIN (ITEM 22, 23 AND 24) ARE OPTIONAL AT DISCRETION OF USER. IF USED, ITEMS 22 AND 24 MAY BE FIELD MODIFIED IF REQUIRED TO FIT FLUSH OR BELOW THE TOP SURFACE OF THE RESPECTIVE LID DURING INSTALLATION. ITEM 24 MAY BE GROUND TO FACILITATE A PRESS FIT DURING INSTALLATION. THE STRUCTURAL LID THREADED PLUGS (ITEM 23), IF USED DURING STORAGE OPERATION, SHALL BE REMOVED PRIOR TO TRANSPORT (790-516).

7 (flag). PT FINAL SURFACE. EXAMINED PER ASME SECTION V, ARTICLE 6. ACCEPTANCE PER ASME SECTION III, ARTICLE NB-5350.

6 (flag). AT THE OPTION OF THE USER, STAINLESS STEEL SHIMS (ASME SA240/479, TYP 304L) OF APPROPRIATE THICKNESS MAY BE USED IN THE WELDING OF THE STRUCTURAL LID (ITEM 19) TO THE SHELL WELDMENT (ITEM 1-5).

5 (flag). PT ROOT AND FINAL SURFACE. SHOULD ROOT AND FINAL SURFACE BE ONE AND THE SAME (i.e. SINGLE PASS WELD) THEN ONLY ONE PT EXAMINATION IS REQUIRED. EXAMINED PER ASME SECTION V, ARTICLE 6, ACCEPTANCE PER ASME SECTION III, ARTICLE NB-5350.

4 (flag). AFTER INSTALLING BASKET (ITEMS 6-10), POSITION KEY (ITEM 21) IN SLOT AT TOP OF BASKET, AS SHOWN, AND TACK WELD KEY TO SIDE OF SHELL (ITEMS 1-5). INSTALL LID SUPPORT RING (ITEM 16), AS SHOWN, AND WELD TO SHELL AND KEY. INSTALL SHIELD ASSEMBLY (ITEM 17), LOCATING ON KEY.

3 (flag). LUBRICATE WITH A SPENT FUEL COMPATIBLE LUBRICANT SUCH AS NEOLUBE. INSTALL FITTING BY HAND UNTIL METAL TO METAL CONTACT IS ACHIEVED THEN TORQUE TO (115±5 FT-LBS FOR ELASTOMER SEALS) OR (135±15 FT-LBS FOR METAL SEALS). LENGTH MAY BE ADJUSTED IN THE FIELD TO ALLOW FOR STACK UP CONDITIONS.

2 (flag). PT ROOT, EACH SUCCESSIVE 3/8 INCH WELD THICKNESS, AND FINAL SURFACE. EXAMINE PER ASME SECTION V, ARTICLE 6. ACCEPTANCE PER ASME SECTION III, ARTICLE NB-5350; OR, UT PER ASME SECTION V, ARTICLE 5. ACCEPT PER ASME SECTION III, ARTICLE NB-5330 IN CONJUNCTION WITH PT EXAMINATION PER NOTE 5.

1. PACKAGE SHOWN FULLY ASSEMBLED. WELD SYMBOLS ARE FOR REFERENCE ONLY. REFERENCE PROCEDURE FOR OPERATIONAL ASSEMBLY SEQUENCE.

NOTES:

(22) 21 (flag). FOR ASSEMBLIES 98 AND 97 (BWR) WELD SIZE IS (1/2), FOR ASSY 99, 96, AND 95 (PWR) MINIMUM WELD DEPTH OF BEVEL IS .36" FOR A WELD SIZE OF (3/8) NOMINAL TO (5/16) MINIMUM.

20 (flag). FOR ASSEMBLIES 98 AND 97 (BWR) WELD SIZE IS (7/8), FOR ASSY 99, 96, AND 95 (PWR) WELD SIZE IS (3/4).

19. DELETED

18 (flag). WELD NOT TO EXCEED TOP SURFACE OF ITEM 17 (SHIELD LID ASSY)

17. AFTER SHIELD LID WELDING, TSC SHELLS MAY BE RE-ROUNDED BY MECHANICAL METHODS OR TOOLING. IF RE-ROUNDING IS PERFORMED, THE EXAMINATION OF THE SHIELD LID FINAL WELD SURFACE, PRESSURE TEST AND LEAK TEST SHALL BE PERFORMED, AFTER ALL RE-ROUNDING ACTIVITIES ARE COMPLETED.

16. AT THE OPTION OF THE USER, STAINLESS STEEL (ASTM/ASME A/SA240 TYPE 304/304L) SHIMS OF APPROPRIATE SIZE AND THICKNESS MAY BE USED IN THE SPACER RING GROOVE IN THE STRUCTURAL LID TO ENSURE THE BAR FITS TIGHTLY AGAINST THE TOP OF THE GROOVE AT THE BOTTOM OF THE WELD JOINT.

15. THE STRUCTURAL LID AND INSTALLED SPACER RING CAN BE FIELD DRESSED IN LOCALIZED AREAS TO ELIMINATE INTERFERENCE WITH THE CANISTER SHELL DURING INSTALLATION OF THE LID.

14. FOR ASME CODE STAMPED TSC's, THE GROOVE WELD FROM THE TOP OF THE STRUCTURAL LID ROOT PASS WELD TO THE SHORTEST DISTANCE OF THE FINISHED WELD PROFILE SHALL BE AT LEAST 0.72", FOR MAINE YANKEE CANISTER ONLY.

13. INSPECTION OF THE COLD STACK-UP OF THE SHIELD AND STRUCTURAL LIDS SHOULD BE PERFORMED PRIOR TO COMMENCEMENT OF FIELD WELDING. FOR REFERENCE PURPOSES, THE STRUCTURAL LID SHOULD NOT EXCEED THE TOP EDGE OF THE CANISTER SHELL BY MORE THAN .180". THIS INSPECTION MAY EITHER BE PERFORMED AT THE FABRICATOR'S FACILITY OR IN THE FIELD. FOLLOWING COMPLETION OF THE STRUCTURAL LID WELDING, THE TOP SURFACE OF THE STRUCTURAL LID SHALL BE FLUSH OR ABOVE THE EDGE OF THE TSC SHELL.

95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
2	2	2	2	2	24	DOWEL PIN	ST. STL.	COML.		1.0 DIA X 1.0 LONG
6	6	6	6	6	23	STRUCTURAL LID PLUG	304 ST. STL.	ASTM A276		2 DIA BAR
3	3	3	3	3	22	SHIELD LID PLUG	304 ST. STL.	ASTM A276		1 DIA BAR
1	1	1	1	1	21	KEY			790-584-8	
1	1	1	1	1	20	SPACER RING			790-584-7	
1	1	1	1	1	19	STRUCTURAL LID			790-584-4	
2	2	2	2	2	18	COVER			790-584-5	
1	1	1	1	1	17	SHIELD LID ASSY			790-584-99	
1	1	1	1	1	16	LID SUPPORT RING			790-584-6	
				1	15	DRAIN TUBE ASSY			790-583-99	
			1		14	DRAIN TUBE ASSY			790-583-98	
		1			13	DRAIN TUBE ASSY			790-583-97	
	1				12	DRAIN TUBE ASSY			790-583-96	
1					11	DRAIN TUBE ASSY			790-583-95	
1					10	FUEL BASKET ASSY			790-595-99	PWR -- CANISTER CLASS 1
	1				9	FUEL BASKET ASSY			790-595-98	PWR - CANISTER CLASS 2
		1			8	FUEL BASKET ASSY			790-570-99	BWR - CANISTER CLASS 4
			1		7	FUEL BASKET ASSY			790-570-98	BWR - CANISTER CLASS 5
				1	6	FUEL BASKET ASSY			790-595-97	PWR - CANISTER CLASS 3
				1	5	SHELL WELDMENT			790-582-99	
			1		4	SHELL WELDMENT			790-582-98	
		1			3	SHELL WELDMENT			790-582-97	
	1				2	SHELL WELDMENT			790-582-96	
1					1	SHELL WELDMENT			790-582-95	

QUANTITY (flag 8 applies to items 22, 23, 24)

UNLESS OTHERWISE STATED
DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.
ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3-12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6-18	±.03
	OVER 18	±.06
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

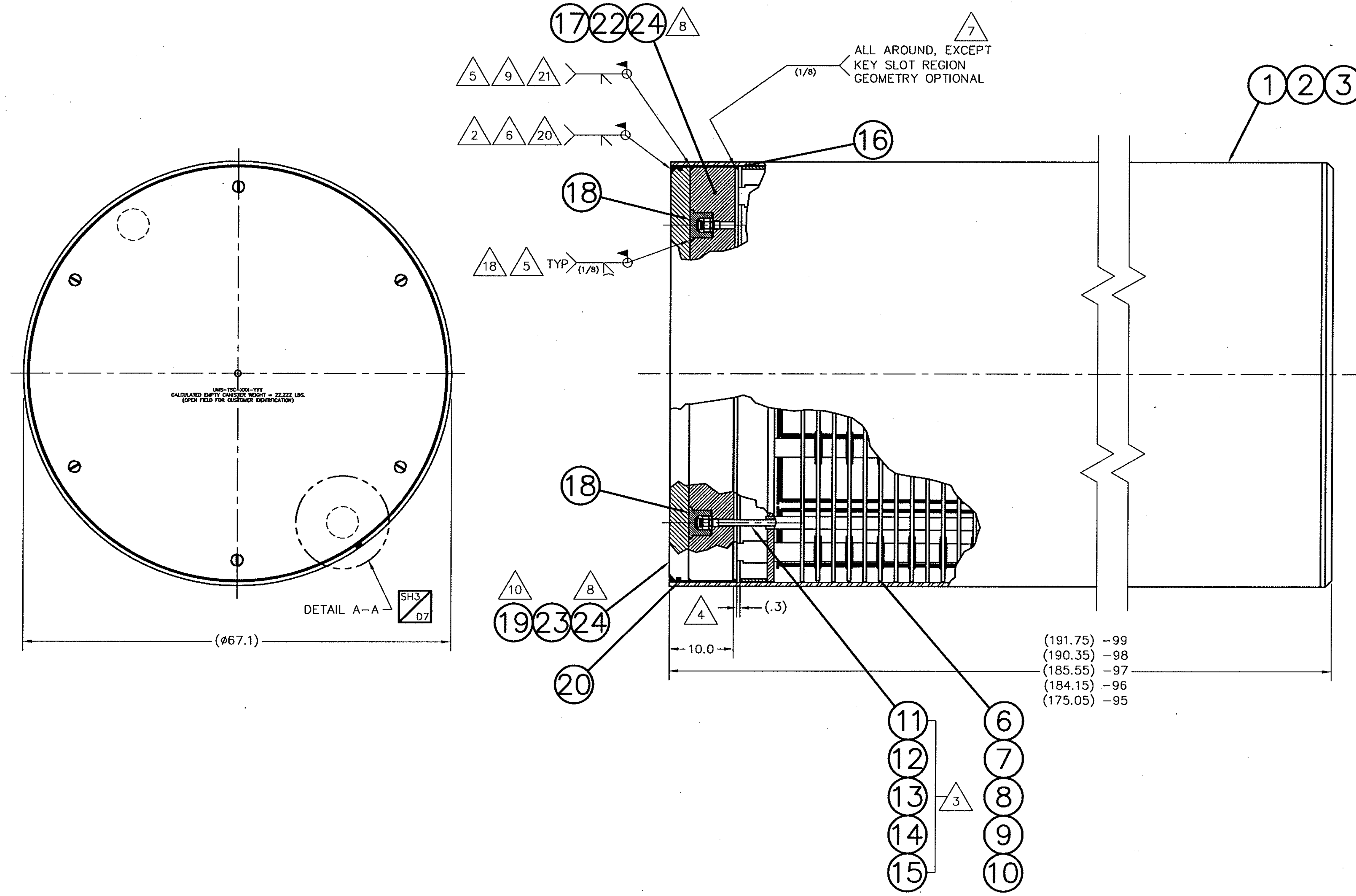
WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
ALL DIMENSIONS ARE IN INCHES
BORDER SIZE: F (40 X 28)
ALL UNSPECIFIED TOOL RADII .015 - .030
BREAK ALL SHARP CORNERS .015 - .030
MACHINED SURFACES TO BE ∇ OR BETTER
NEXT ASSEMBLY: 790-590/516
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	3-11-16
CHECKER	[signature]	3-11-16
PROJECT MANAGER	[signature]	3/14/16
ENGINEERING	[signature]	3-11-16
LICENSING	[signature]	3/14/16
QUALITY	[signature]	3-24-16

NAC INTERNATIONAL

TRANSPORTABLE STORAGE CANISTER, (TSC) NAC-UMS®

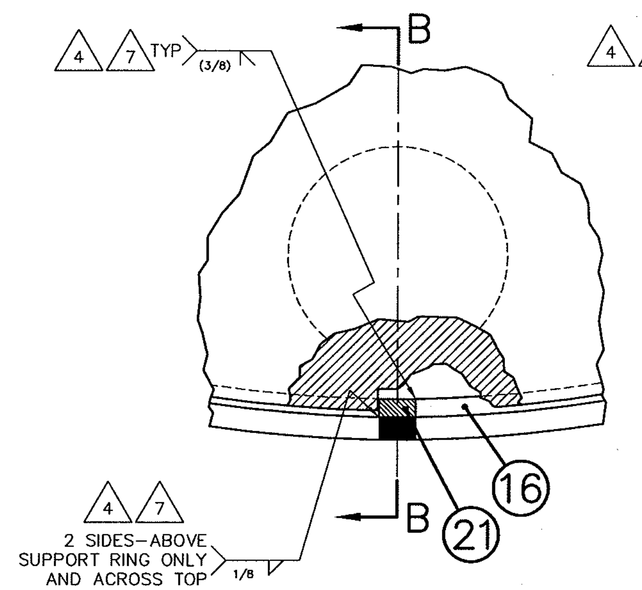
PROJECT 790 | DRAWING 585 | REV 22
SCALE | WEIGHT | SH 1 OF 3 | 1:41PM 3-11-2016



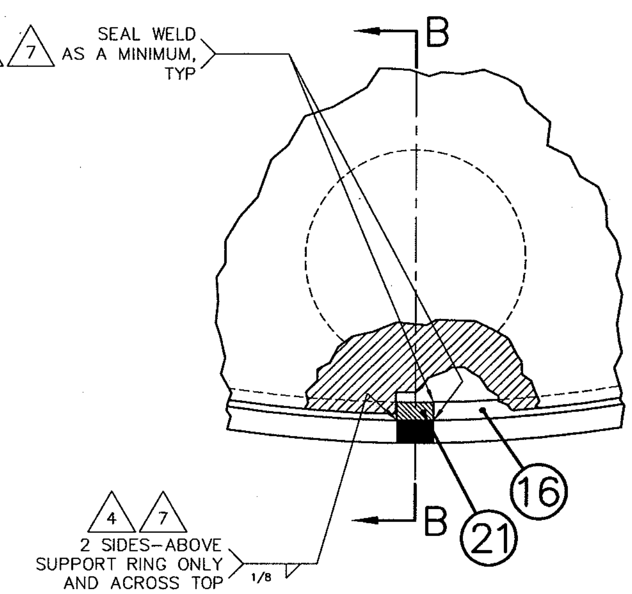
NAC INTERNATIONAL

TRANSPORTABLE STORAGE CANISTER, (TSC) NAC–UMS®

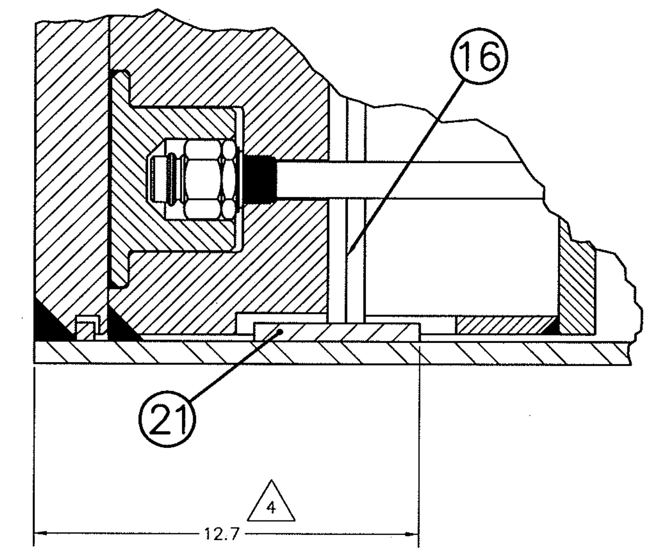
PROJECT	DRAWING	REV
790	585	22
	SH 2 OF 3	1:43PM 3-11-2010



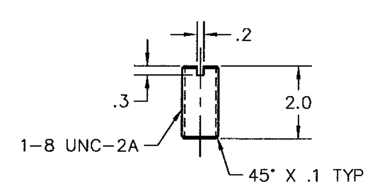
SH2 C6 DETAIL A—A
VIEW ROTATED
OPTIONAL ENGRAVED STRIPE SHOWN



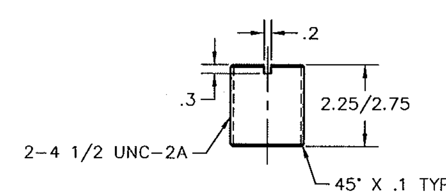
SH2 C6 DETAIL A—A
VIEW ROTATED
ALTERNATE CONFIGURATION
OPTIONAL ENGRAVED STRIPE SHOWN



SECTION B—B



8 22 SHIELD LID PLUG



8 23 STRUCTURAL LID PLUG

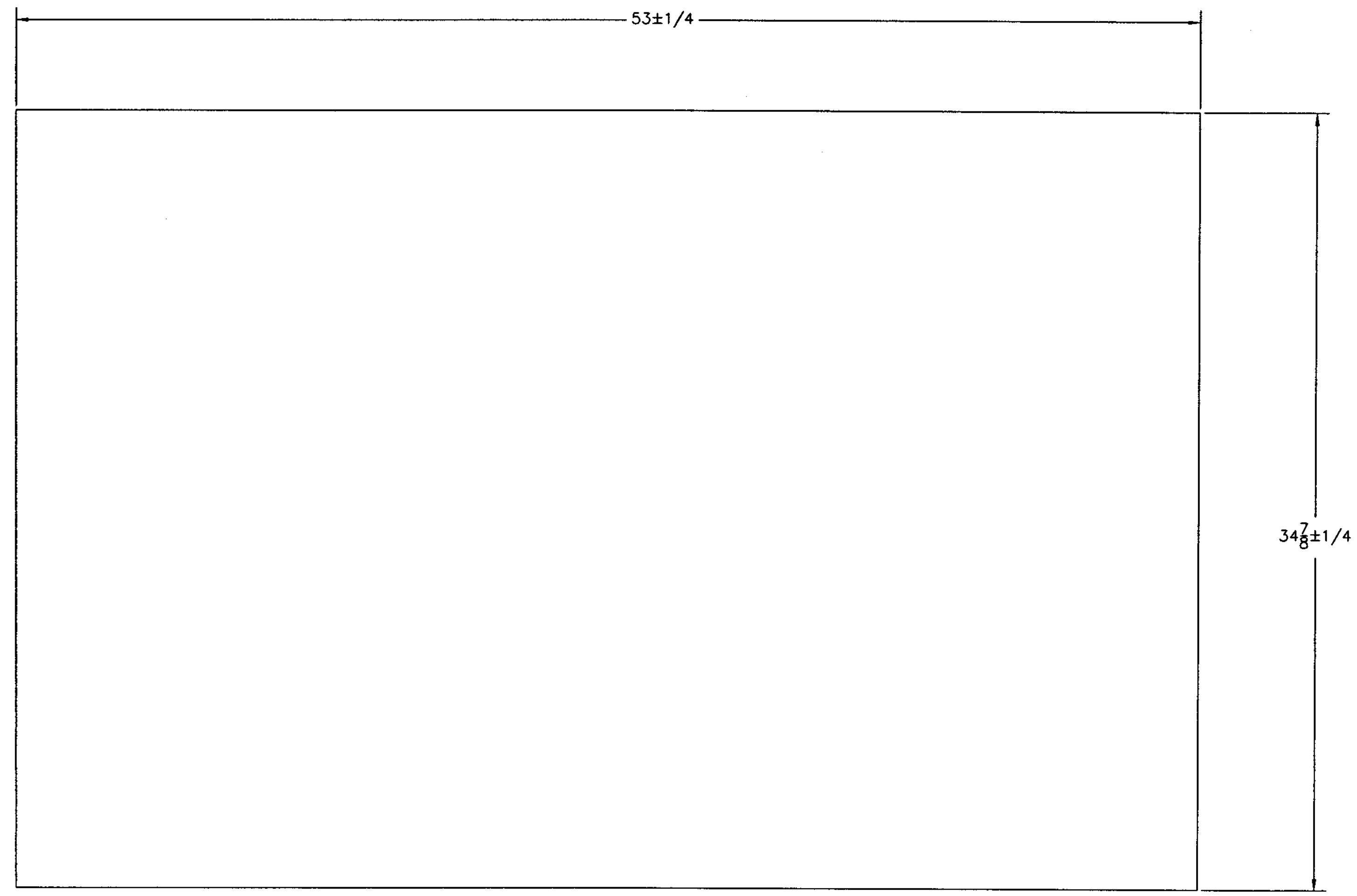
NAC INTERNATIONAL

TRANSPORTABLE STORAGE CANISTER, (TSC) NAC-UMS®

PROJECT	DRAWING	REV
790	585	22
	SH 3 OF 3	1:42PM 3-11-2010

R- 145831

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A,0B



(1) SPACER SHIM #1
WEIGHT: 14#

(2) SPACER SHIM #2
WEIGHT: 29#

(3) SPACER SHIM #3
WEIGHT: 43#

(4) SPACER SHIM #4
WEIGHT: 58#

(5) SPACER SHIM #5
WEIGHT: 23#

(6) SPACER SHIM #6
WEIGHT: 9#

NOTES:

1. DELETED (1)
2. MATERIAL MAY BE ANY 300 SERIES STAINLESS STEEL.
3. ALTERNATE SPACER SHIM DIMENSIONS OF 48"±1/4 X 38"±1/4 MAY BE USED FOR ALL THICKNESS LISTED ON B.O.M. (1)
4. EDGES OF SHEET MAY BE FIELD DRESSED TO ELIMINATE CURLS, BURRS, ETC. (1)

97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
			6	SPACER SHIM #6	SEE NOTE 2	COML		26 GAUGE SHEET (.018)
			5	SPACER SHIM #5	SEE NOTE 2	COML		18 GAUGE SHEET (.048)
			4	SPACER SHIM #4	SEE NOTE 2	COML		11 GAUGE SHEET (.120)
			3	SPACER SHIM #3	SEE NOTE 2	COML		13 GAUGE SHEET (.090)
			2	SPACER SHIM #2	SEE NOTE 2	COML		16 GAUGE SHEET (.060)
			1	SPACER SHIM #1	SEE NOTE 2	COML		22 GAUGE SHEET (.029)

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

NEXT ASSEMBLY: 790-585/-612
DRAWING TYPE: LICENSE

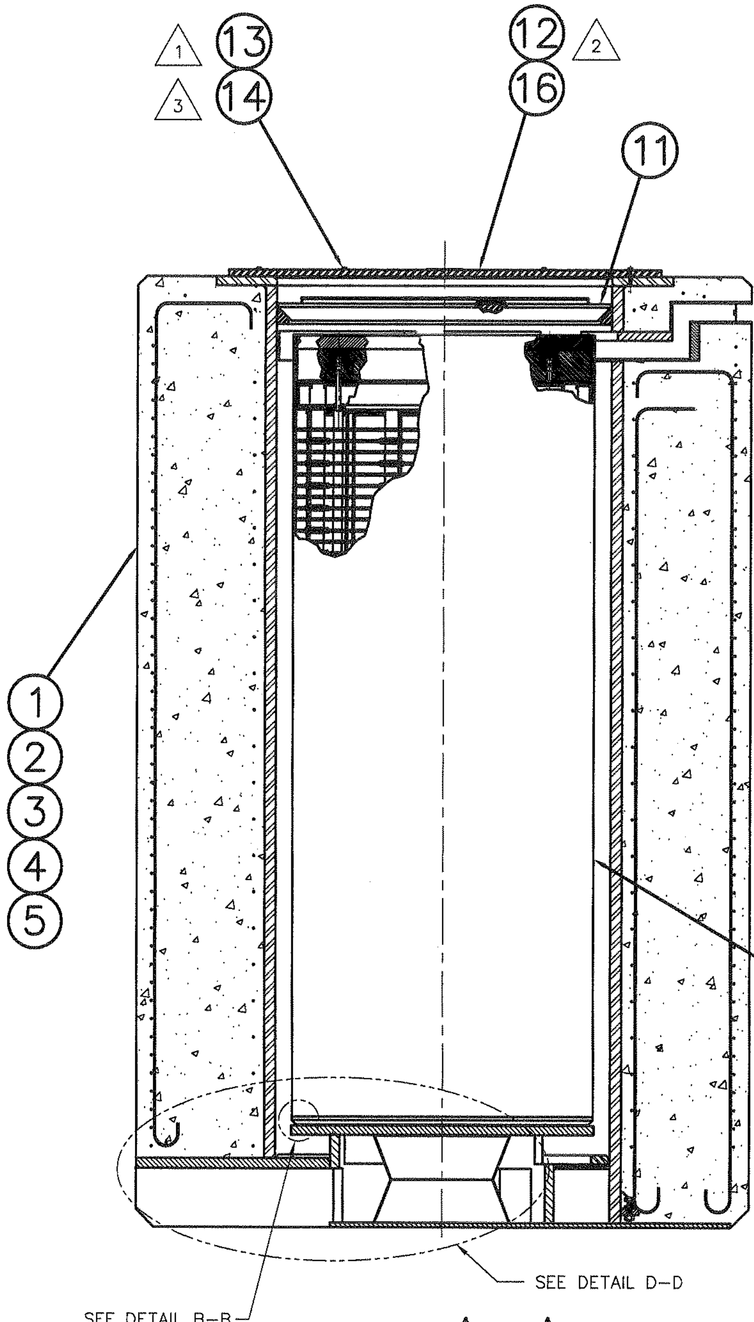
GROUP	NAME	DATE
PREPARER	[signature]	3/13/02
CHECKER	[signature]	3-14-02
PROJECT MANAGER	[signature]	3/15/02
DIRECTOR DESIGN AND ANALYSIS	[signature]	3-22-02
DIRECTOR LICENSING	[signature]	3/25/02
VICE PRESIDENT, QUALITY	[signature]	3/27/02

NAC INTERNATIONAL

SPACER SHIM, CANISTER NAC-UMS®

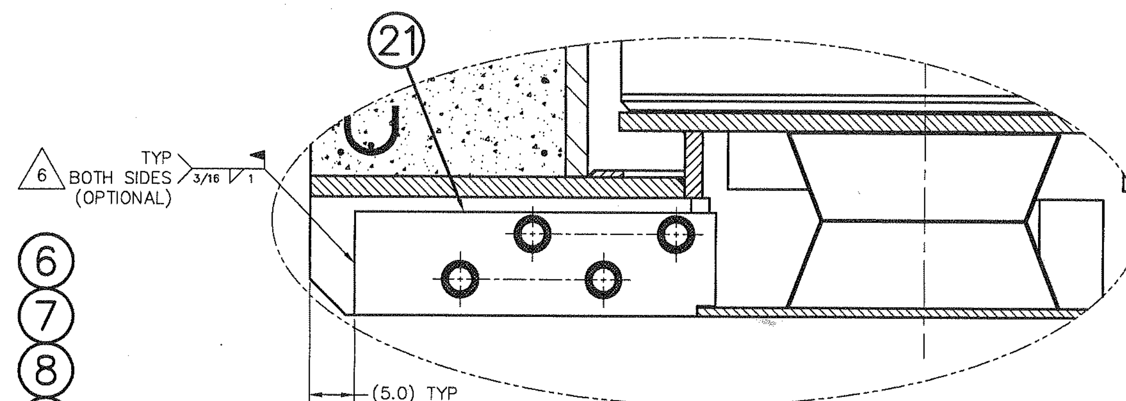
PROJECT 790 | DRAWING 587 | REV 1

SCALE 1/3 | EST.WT. NOTED | SH 1 OF 1 | 1:58PM 3-13-2002



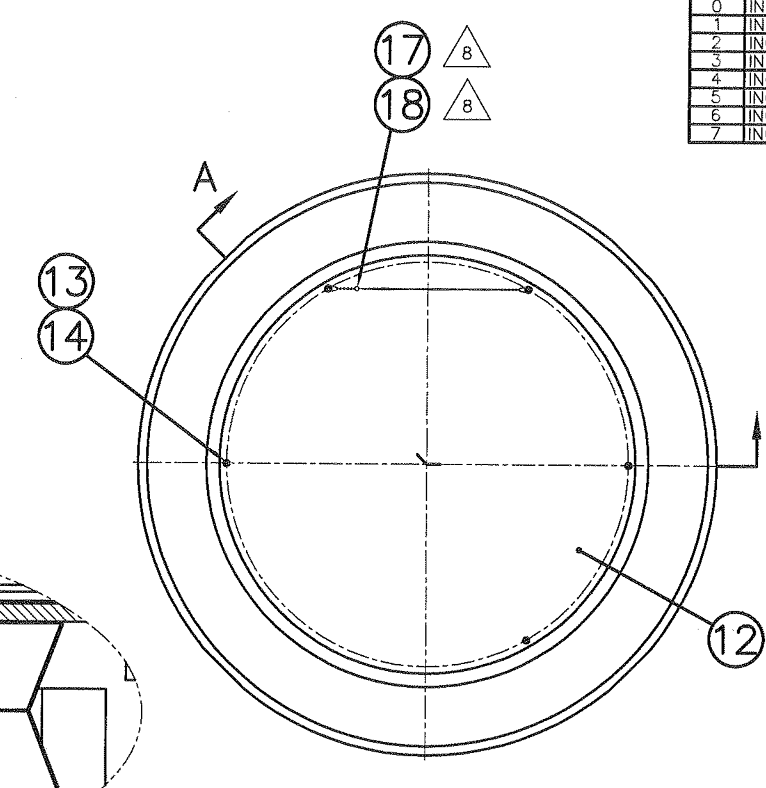
SECTION A—A

99	LOADED VERTICAL CONCRETE CASK CLASS 3 PWR
98	LOADED VERTICAL CONCRETE CASK CLASS 5 BWR
97	LOADED VERTICAL CONCRETE CASK CLASS 4 BWR
96	LOADED VERTICAL CONCRETE CASK CLASS 2 PWR
95	LOADED VERTICAL CONCRETE CASK CLASS 1 PWR

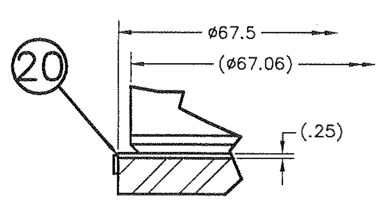


DETAIL D—D

SCALE: 1/2
OPTIONAL CONFIGURATIONS
(4X ITEM 21 REQ'D FOR EACH)



TOP VIEW



DETAIL B—B

SCALE: 1/4

NOTES:

(7) 1. AT THE OPTION OF THE USER, DRILL A 1/16 DIAMETER HOLE THRU THE MIDDLE OF THE BOLT HEAD, FROM THE MIDDLE OF ONE FLAT TO THE OPPOSITE FLAT. FOR A MINIMUM OF 2 BOLTS PER ASSEMBLY.

(7) 2. AT THE OPTION OF THE USER, ONE CIRCULAR LAYER OF SEAL TAPE (ITEM 16) MAY BE APPLIED ON THE FLANGE OF THE VCC JUST INSIDE THE LID BOLT CIRCLE.

3. AT CONSTRUCTOR'S OPTION, AN ADDITIONAL WASHER MAY BE ADDED TO FACILITATE LID TO LID BOLT FIT-UP.

4. GAP AS REQUIRED FOR FIT-UP.

5. ITEM 13 TO BE 1/2-13 UNC-2A X 3-1/4 LG. HEX HD. WITH A MINIMUM THREAD LENGTH OF 1.75 OR 1/2-13 UNC-2A X 2-1/2 LG. HEX HD.

6. SHIM PLATES MAY BE UTILIZED TO FACILITATE FIELD WELDING OPERATIONS ON ONE OR BOTH SIDES OF THE SUPPLEMENTAL SHIELDING WELDMENT. LOCATE WELDS APPROX. AS SHOWN.

7. AT THE OPTION OF THE USER, 790-564 (ASSEMBLY 99, 98, OR 97) MAY BE USED.

8. THE INSTALLATION AND USE OF SECURITY SEAL AND SEAL WIRE (ITEMS 17 AND 18) IS AT OPTION OF THE USER.

(7) 9. AT THE OPTION OF THE USER, THE COVER (ITEM 15) MAY BE INSTALLED AS SPECIFIED AS AN ALTERNATIVE ON DRAWING 790-561. IF ALTERNATIVE ASSEMBLY OF THE COVER (ITEM 15) IS PERFORMED, ITEMS 19 AND 20 ARE NOT USED.

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A,1B,1C,1D,1E
3	INC DCR 2A
4	INC DCR 3A,3B
5	INC DCR 4A
6	INC DCR 5A,5B,5C
7	INC DCR 6A

94 ASSY	95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
						21	SUPPLEMENTAL SHIELDING			790-613-99	
	1	1	1	1	1	20	COVER PLATE			790-590-94	
3						19	TAB	304 ST. STL.	ASTM A240/A276/A479		1/4 PLATE/BAR
	A/R	A/R	A/R	A/R	A/R	18	SEAL WIRE		1/32 DIA WIRE		AMERICAN CASTING CO.
	1	1	1	1	1	17	SECURITY SEAL		1.0 DIA METAL CUP SEAL		AMERICAN CASTING CO.
	A/R	A/R	A/R	A/R	A/R	16	SEAL TAPE	SEMI-RIGID RUBBER	COML		MCMASTER-CARR #8622K23
1						15	COVER	304 ST. STL.	ASTM A240		1/4 PLATE
	6	6	6	6	6	14	WASHER	ST. STL.	COML		1/2 FLAT WASHER
	6	6	6	6	6	13	LID BOLT	ST. STL.	COML		SEE NOTE 5
	1	1	1	1	1	12	CASK LID			790-563-99	
	1	1	1	1	1	11	SHIELD PLUG			SEE NOTE 7	
	1					10	TSC ASSEMBLY			790-585-95	
		1				9	TSC ASSEMBLY			790-585-96	
			1			8	TSC ASSEMBLY			790-585-97	
				1		7	TSC ASSEMBLY			790-585-98	
					1	6	TSC ASSEMBLY			790-585-99	
	1					5	VCC ASSEMBLY			790-562-95	
		1				4	VCC ASSEMBLY			790-562-96	
			1			3	VCC ASSEMBLY			790-562-97	
				1		2	VCC ASSEMBLY			790-562-98	
					1	1	VCC ASSEMBLY			790-562-99	

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94
UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW.
DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE 250 OR BETTER			

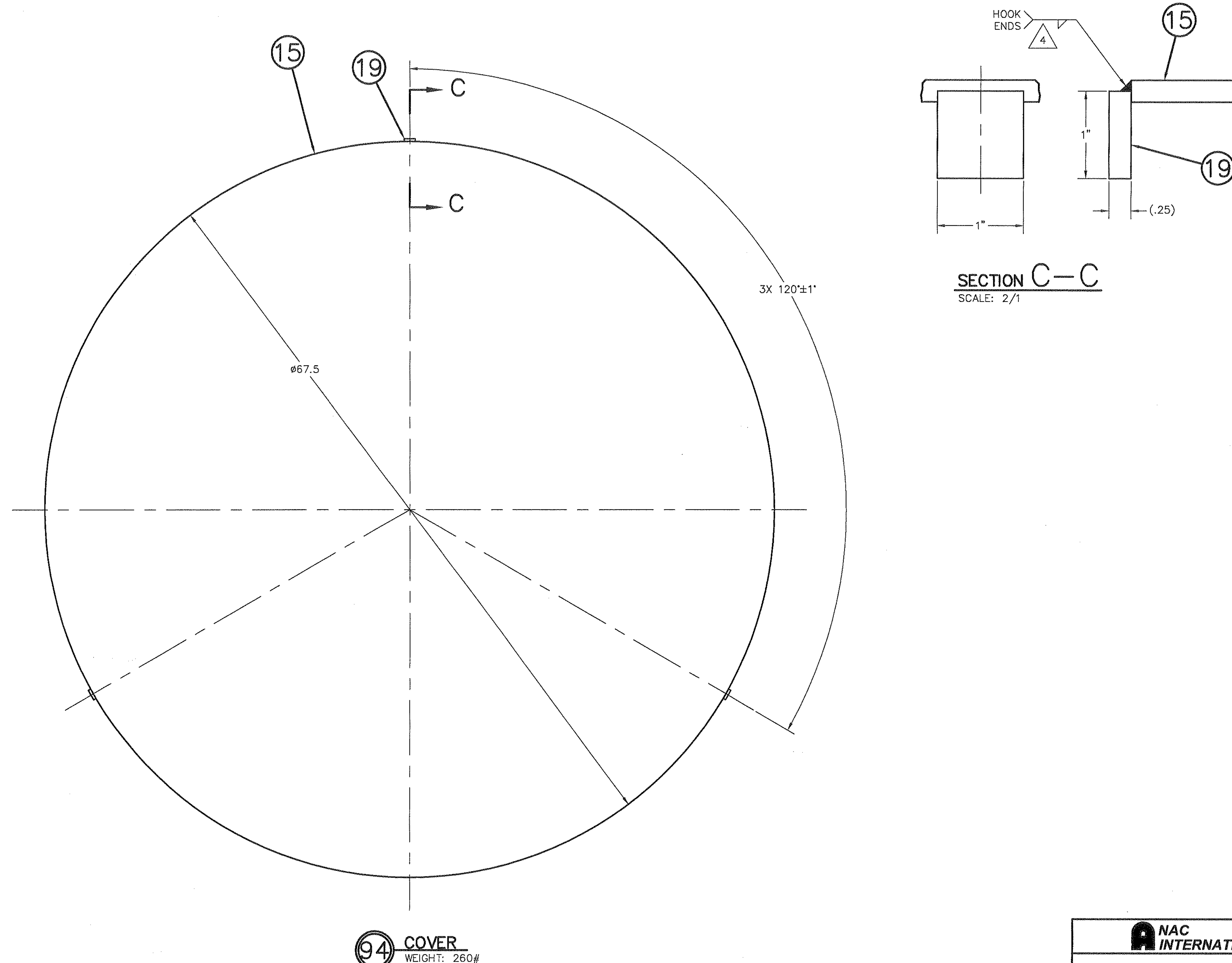
NEXT ASSEMBLY: N/A
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	11/4/10
CHECKER	[signature]	11/04/10
PROJECT MANAGER	[signature]	11/4/10
DIRECTOR, DESIGN AND ANALYSIS	[signature]	11/4/10
DIRECTOR, LICENSING	[signature]	11/4/10
VICE PRESIDENT, QUALITY	[signature]	11/4/10

NAC INTERNATIONAL

LOADED VERTICAL CONCRETE CASK (VCC) NAC-UMS®

PROJECT 790 | DRAWING 590 | REV 7
SCALE 1/16 | EST.WT. | SH 1 OF 2 | 4:10PM 11-1-2010



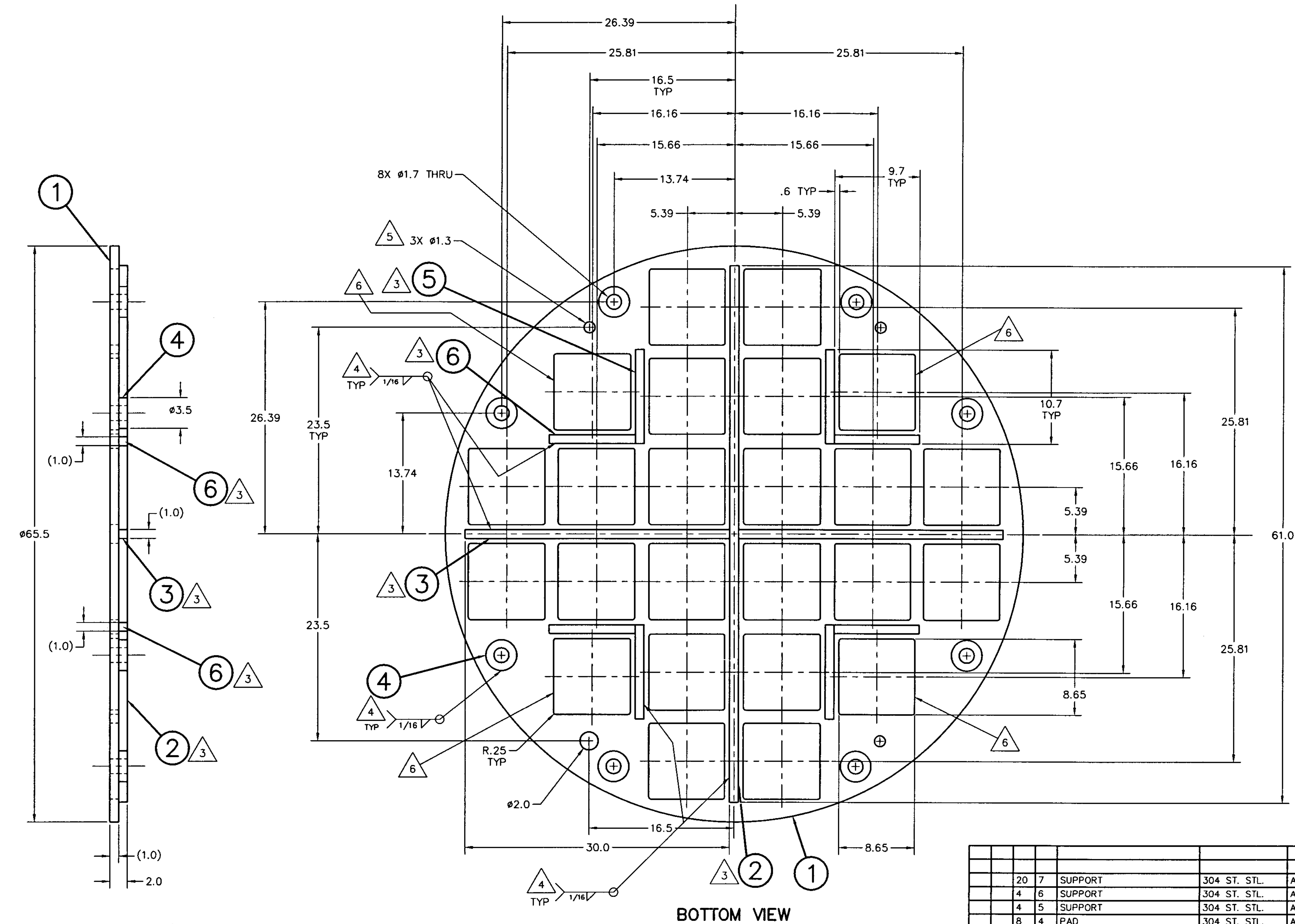
NAC INTERNATIONAL

LOADED VERTICAL
CONCRETE CASK (VCC)
NAC-UMS®

PROJECT 790		DRAWING 590	REV 7
SCALE 1/4	EST.WT. NOTED	SH 2 OF 2	4:26PM 11-1-2010

R-152483

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR SAR-001-0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A,3B,3C
5	INC DCR 4A,4B,4C
6	INC DCR 5B



NOTES:

1. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH ASME SECTION IX.
2. "SEAL WELD" ALL OPEN SEAMS.
3. (triangle) CENTERED APPROX. ON WEB.
4. (triangle) LIQUID PENETRANT (PT) EXAMINED PER ASME SECTION V, ARTICLE 6. ACCEPTANCE PER ASME SECTION III, ARTICLE NG-5350.
5. (triangle) THE 3X ø1.3 HOLES MAY BE REPLACED WITH HOLES OF ø2.0.
6. (triangle) WHEN A MAINE YANKEE FUEL CAN (SHOWN ON NAC DRAWING 412-501) MAY BE INCLUDED AS CONTENTS OF THE CANISTER, THESE HOLE LOCATIONS SHALL HAVE OPENINGS OF 8.70 X 8.70 WITH AN INSIDE CORNER RADIUS OF .13 MAXIMUM.
7. (triangle) ITEM 7 (SUPPORT) ARE TO BE CUT TO THE LENGTH OF 8.8±.1.
8. IN ANY SINGLE QUADRANT OF THE BOTTOM WELDMENT, THE ORIENTATION OF ITEMS 5 AND 6 MAY BE REVERSED.

97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
		20	7	SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
		4	6	SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
		4	5	SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
		8	4	PAD	304 ST. STL.	ASME SA240/SA479		PLATE/BAR
		2	3	SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
		1	2	CENTER SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
		1	1	BOTTOM DISK	304 ST. STL.	ASME SA240		1 PLATE
ASSY	ASSY	ASSY						
QUANTITY								

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM	GEOMETRY	.XXX	TOL	.XX	TOL
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

NEXT ASSEMBLY: 790-595

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	4/8/03
CHECKER	[signature]	4.8.03
PROJECT MANAGER	[signature]	4/10/03
DIRECTOR DESIGN AND ANALYSIS	[signature]	5-7-03
DIRECTOR LICENSING	Anthony L Patko for TCT	5/7/03
VICE PRESIDENT QUALITY	[signature]	5/12/03

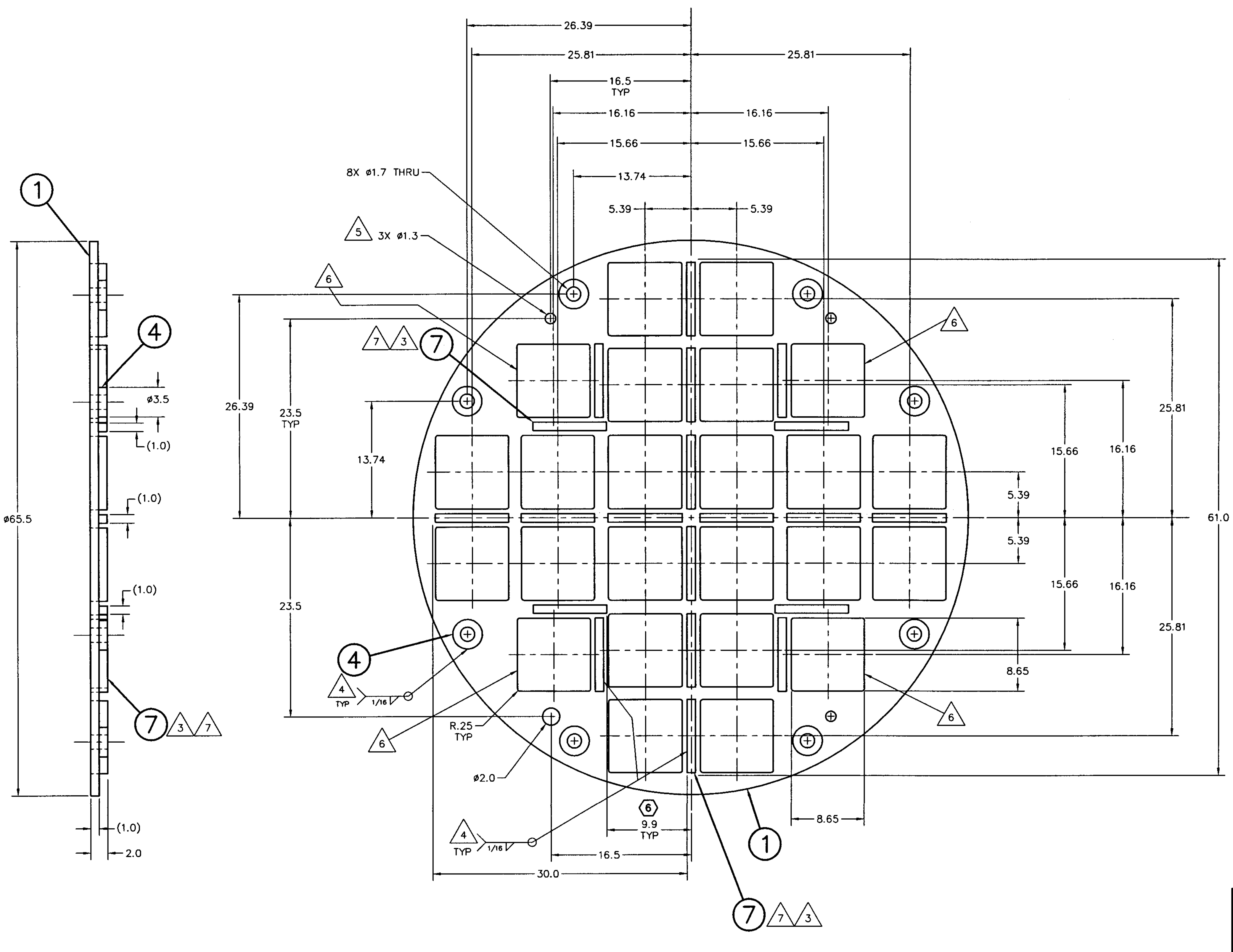
NAC INTERNATIONAL

BOTTOM WELDMENT, FUEL BASKET, 24 ELEMENT PWR NAC-UMS®

PROJECT 790 | DRAWING 591 | REV 6

SCALE 1/5 | EST.WT. 513# | SH 1 OF 2 | 4:17PM 4-8-2003

NAC DCRM 01



BOTTOM VIEW

99 BOTTOM WELDMENT
OPTIONAL CONFIGURATION

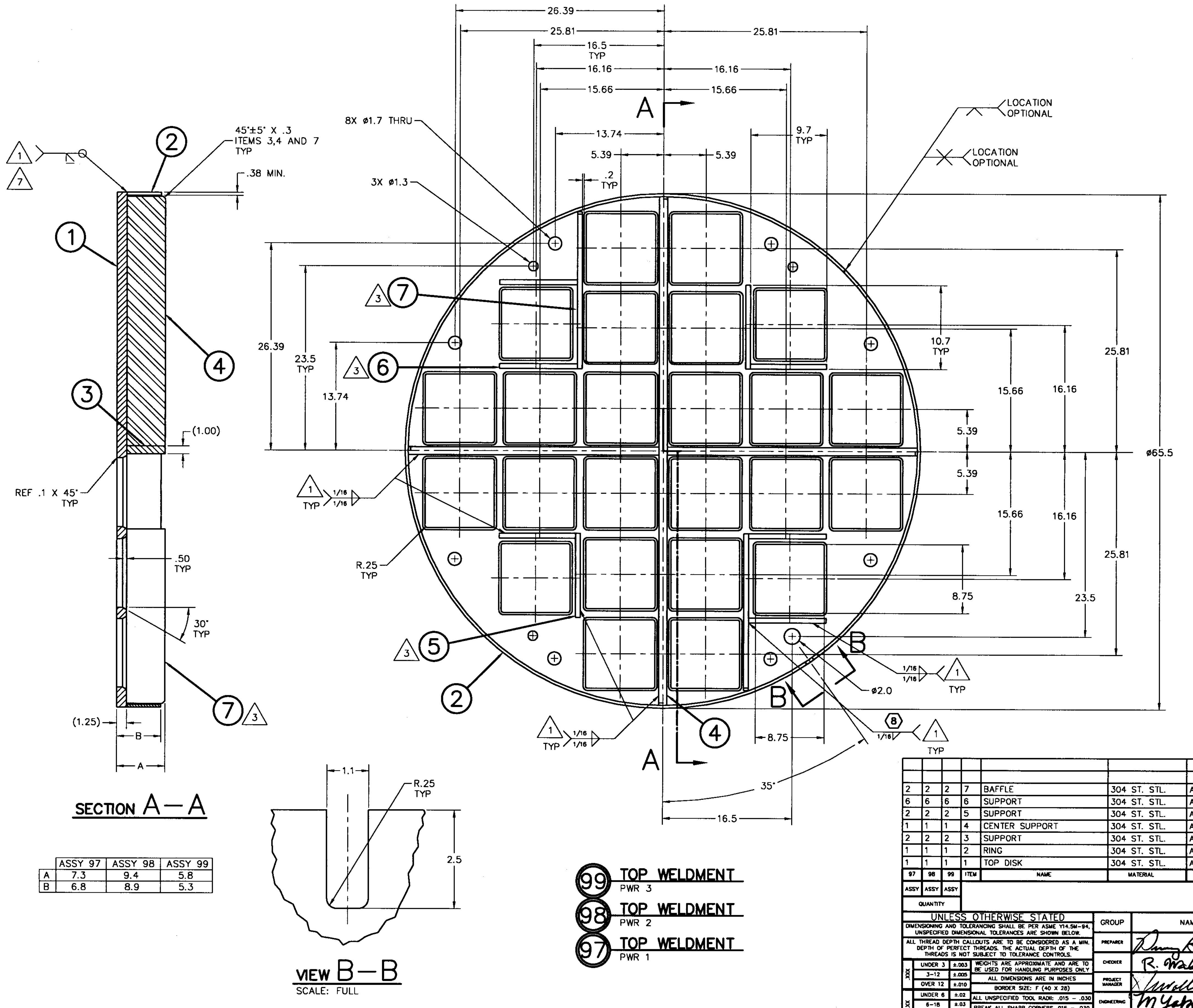
NAC INTERNATIONAL

BOTTOM WELDMENT,
FUEL BASKET,
24 ELEMENT PWR
NAC-UMS®

PROJECT 790	DRAWING 591	REV 6
SCALE 1/5	EST.WT. 513#	SH 2 OF 2	10:56AM 4-7-2003

ℓ-200873

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR SAR-001-0A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A,4B
6	INC DCR 5A
7	INC DCR 6A,6B,6C,6D
8	INC DCR 7A



	ASSY 97	ASSY 98	ASSY 99
A	7.3	9.4	5.8
B	6.8	8.9	5.3

NOTES:

1. (△) LIQUID PENETRANT (PT) EXAMINED PER ASME SECTION V, ARTICLE 6. ACCEPTANCE PER ASME SECTION III, ARTICLE NG-5350.
2. "SEAL WELD" ALL OPEN SEAMS.
3. (△) ITEMS 5, 6 AND 7 CENTERED APPROX. ON WEB.
4. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH ASME SECTION IX.
5. TOLERANCE FOR FUEL TUBE OPENINGS, DIMENSIONS 5.39, 15.66, 16.16, AND 25.81 IS ±.04.
6. MINIMUM THICKNESS OF ITEM 2 MAY BE REDUCED TO .355 FOR A LENGTH OF UP TO 31 INCHES MEASURED ALONG THE OUTER CIRCUMFERENCE.
7. (△) AS AN ALTERNATIVE TO THE FULL-PENETRATION, ALL-AROUND GROOVE WELD, A 1/16" ALL-AROUND GROOVE WELD ON THE OUTSIDE OF ITEM 2 AND 1/16" ALL-AROUND FILLET WELD ON THE INSIDE OF ITEM 2 MAY BE USED.

97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
2	2	2	7	BAFFLE	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
6	6	6	6	SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
2	2	2	5	SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
1	1	1	4	CENTER SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
2	2	2	3	SUPPORT	304 ST. STL.	ASME SA240/SA479		1 PLATE/BAR
1	1	1	2	RING	304 ST. STL.	ASME SA240/SA479		PLATE/BAR
1	1	1	1	TOP DISK	304 ST. STL.	ASME SA240		1 1/4 PLATE
ASSY	ASSY	ASSY						
QUANTITY								

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94. UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

XXX	UNDER 3	±.003	WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
	3–12	±.005	ALL DIMENSIONS ARE IN INCHES
	OVER 12	±.010	BORDER SIZE: F (40 X 28)
XX	UNDER 6	±.02	ALL UNSPECIFIED TOOL RADII: .015 – .030
	6–18	±.03	BREAK ALL SHARP CORNERS .015 – .030
	OVER 18	±.06	MACHINED SURFACES TO BE ▽ OR BETTER
X	ALL	±.1	NEXT ASSEMBLY: 790-595
FRACTIONAL ±1/8			
ANGLES ±0.5°			DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	12-9-03
CHECKER	R. Walker	12-15-03
PROJECT MANAGER	[signature]	1/6/04
ENGINEERING	[signature]	1/26/04
LICENSING	[signature]	12/16/03
QUALITY	[signature]	2/2/04

NAC INTERNATIONAL

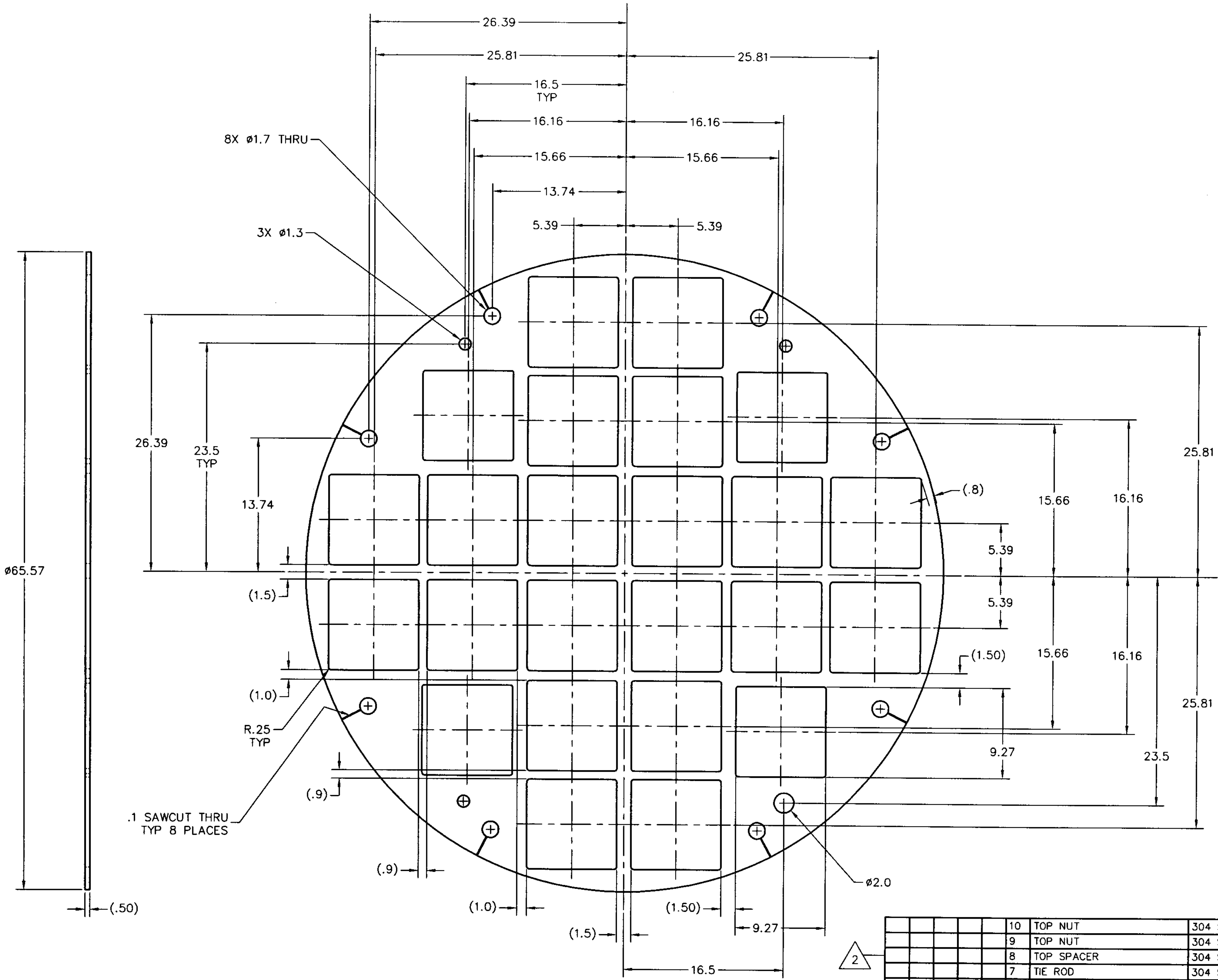
TOP WELDMENT,
FUEL BASKET,
24 ELEMENT PWR
NAC-UMS®

PROJECT 790 | DRAWING 592 | REV 8

SCALE 1/5 | WEIGHT 725# | SH 1 OF 1 | 10:03AM 12-9-2003

R-200874

DCR No.	REVISION
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR SAR-001-0A
3	INC DCR 2A
4	INC DCR 3A,3B
5	INC DCR 4A
6	INC DCR 5A,5B
7	INC DCR 6A



① SUPPORT DISK
WT: 181#

NOTES:

1. HEAT TREAT TO CONDITION H1150.
2. LENGTH TO BE DETERMINED AT ASSEMBLY.
3. AT FABRICATOR OPTION, CHAMFER ON ID OF SPLIT SPACER UP TO .09" X .09" MAY BE UTILIZED.
4. DELETED (7)

95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
					10	TOP NUT	304 ST. STL.	ASME SA479		3 1/2 DIA BAR
					9	TOP NUT	304 ST. STL.	ASME SA479		3 1/2 DIA BAR
					8	TOP SPACER	304 ST. STL.	ASME SA312		2 1/2 PIPE XXS
					7	TIE ROD	304 ST. STL.	ASME SA479		1 5/8 DIA BAR
					6	TIE ROD	304 ST. STL.	ASME SA479		1 5/8 DIA BAR
					5	TIE ROD	304 ST. STL.	ASME SA479		1 5/8 DIA BAR
					4	TOP NUT	304 ST. STL.	ASME SA479		3 1/2 DIA BAR
					3	SPACER	304 ST. STL.	ASME SA312		2 1/2 PIPE XXS
					2	SPLIT SPACER	304 ST. STL.	ASME SA312		2 1/2 PIPE XXS
					1	SUPPORT DISK	17-4PH ST. STL.	ASME SA693, TYPE 630		1/2 PLATE

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3		UNDER 6	±.04
	STRAIGHTNESS	3-12		6-18	±.06
		OVER 12		OVER 18	±.09
	ANGULARITY	.X	±.1	ANGLES ±.5°	
	PERPENDICULARITY	ALL UNSPECIFIED TOOL RADII: .01 - .03			
	PARALLELISM	BREAK ALL SHARP CORNERS .01 - .03			
	CONCENTRICITY	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			
	TRUE POSITION				

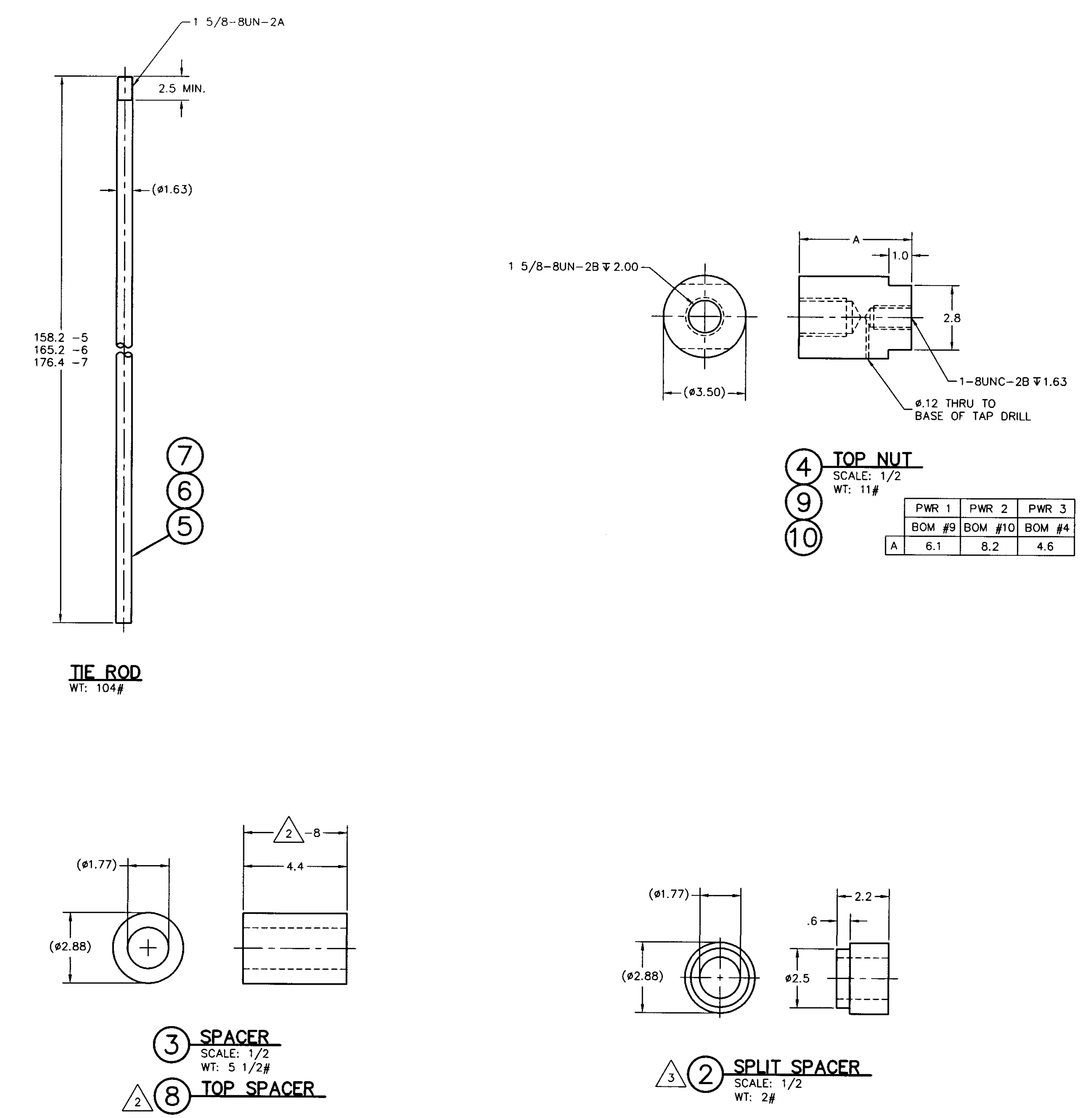
NEXT ASSEMBLY: 790-595
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	1-26-04
CHECKER	R. Wallace	1-27-04
PROJECT MANAGER	[signature]	1/27/04
DIRECTOR DESIGN AND ANALYSIS	[signature]	1/29/04
DIRECTOR, LICENSING	[signature]	1/30/04
VICE PRESIDENT, QUALITY	[signature]	2/3/04

NAC INTERNATIONAL

SUPPORT DISK AND MISC. BASKET DETAILS, 24 ELEMENT PWR NAC-UMS®

PROJECT 790 | DRAWING 593 | REV 7
SCALE 1/5 | EST.WT. | SH 1 OF 2 | 9:08AM 1-26-2004



	PWR 1	PWR 2	PWR 3
	BOM #9	BOM #10	BOM #4
A	6.1	8.2	4.6

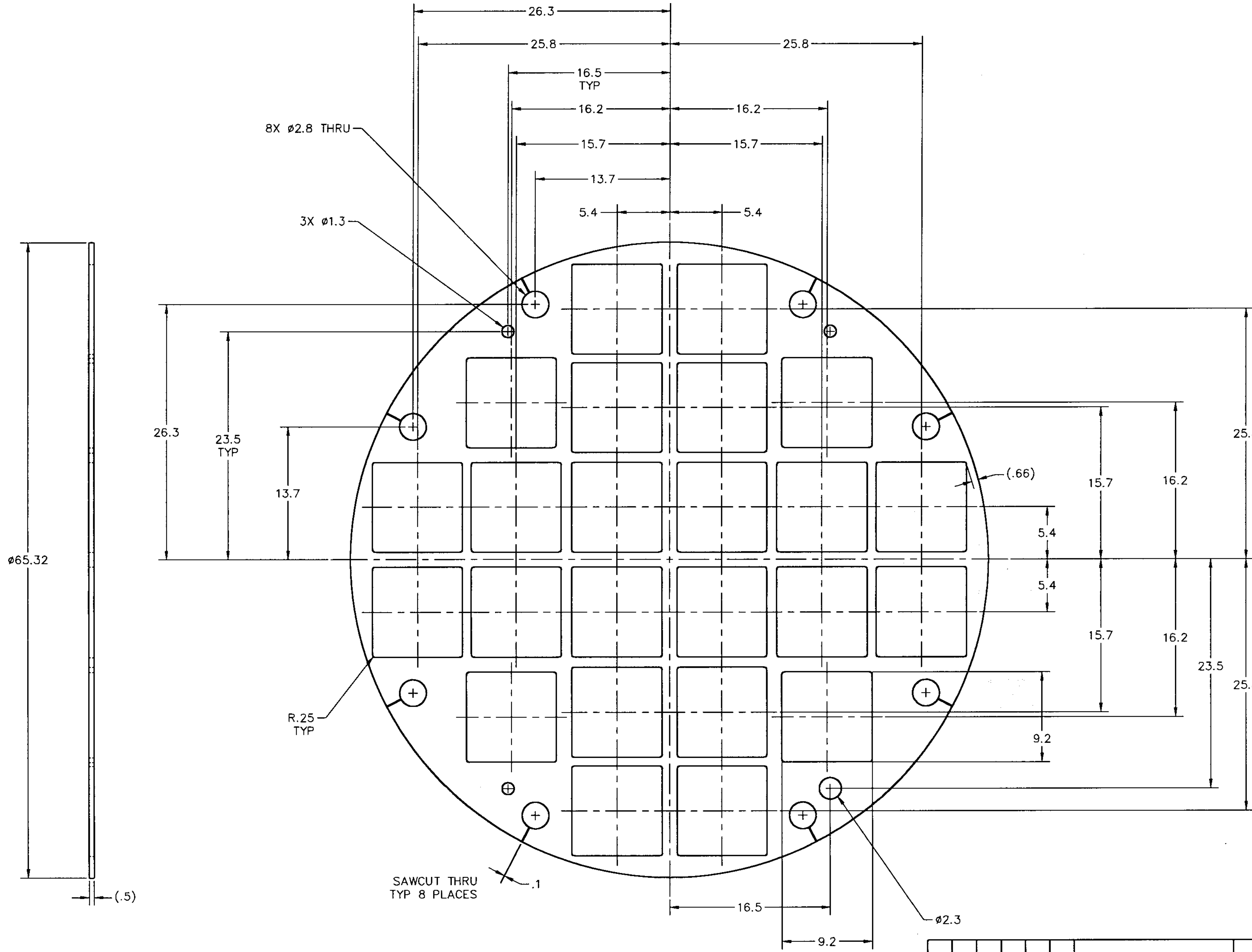
NAC INTERNATIONAL

SUPPORT DISK AND
MISC. BASKET DETAILS,
24 ELEMENT PWR
NAC-UMS®

PROJECT 790		DRAWING 593	REV 7
SCALE 1/5	EST.WT.	SH 2 OF 2	8:10AM 1-26-2004

R-133800

DCR No.	REVISION
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A



1 HEAT TRANSFER DISK
WT: 62#

95	96	97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
					1	HEAT TRANSFER DISK	6061-T651 AL.	ASME SB209		1/2 PLATE
ASSY	ASSY	ASSY	ASSY	ASSY						

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ANSI Y14.5-82. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
⏥	FLATNESS	UNDER 3		UNDER 6	±.04
—	STRAIGHTNESS	3-12		6-18	±.06
∠	ANGULARITY	OVER 12		OVER 18	±.09
⊥	PERPENDICULARITY	.X	±.1	ANGLES ±.5°	
//	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .01 – .03			
◎	CONCENTRICITY	BREAK ALL SHARP CORNERS .01 – .03			
⌖	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE 125/OR BETTER			

NEXT ASSEMBLY: 790-595
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. Graham	10-8-99
CHECKER	[signature]	10/8/99
PROJECT MANAGER	[signature]	10/8/99
DIRECTOR, DESIGN AND ANALYSIS	[signature]	10/8/99
DIRECTOR, LICENSING	[signature]	10/8/99
VICE PRESIDENT, QUALITY	B. Barrett for	10/8/99

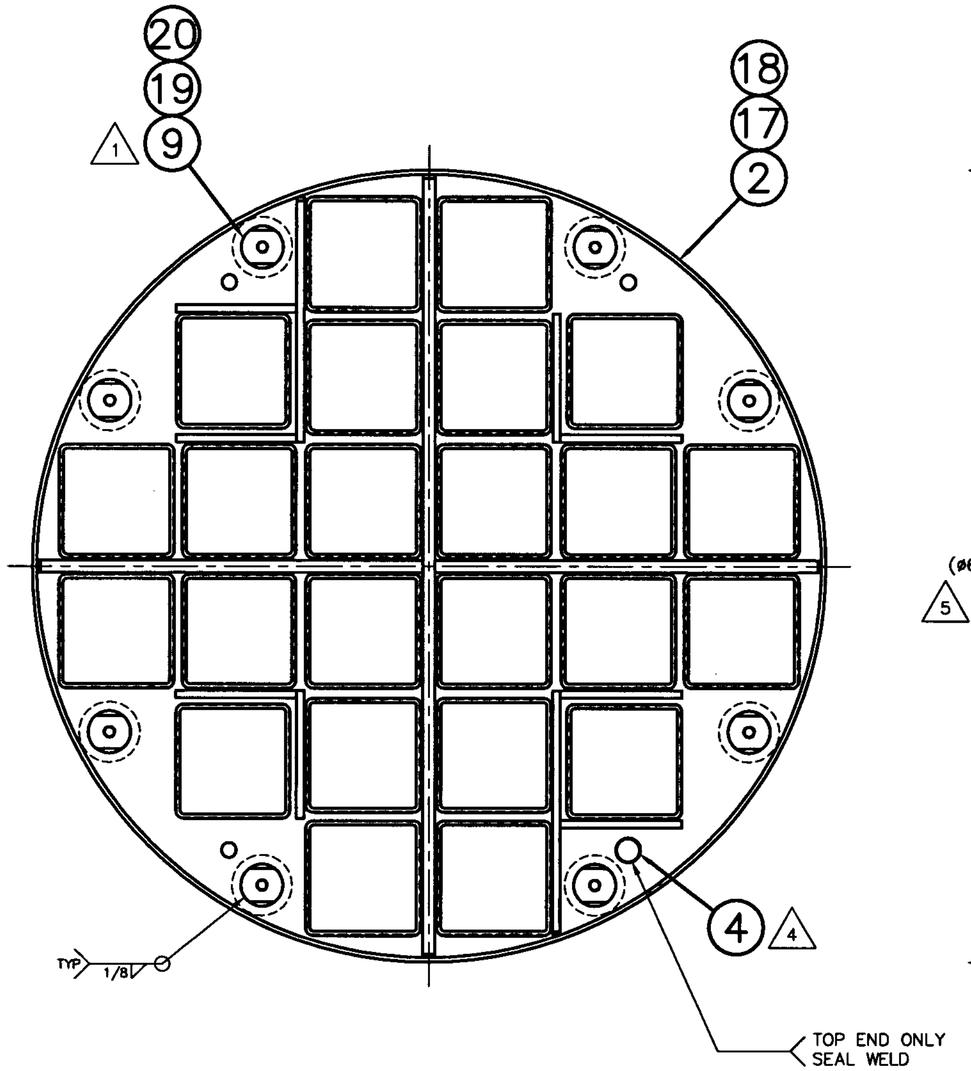
NAC INTERNATIONAL

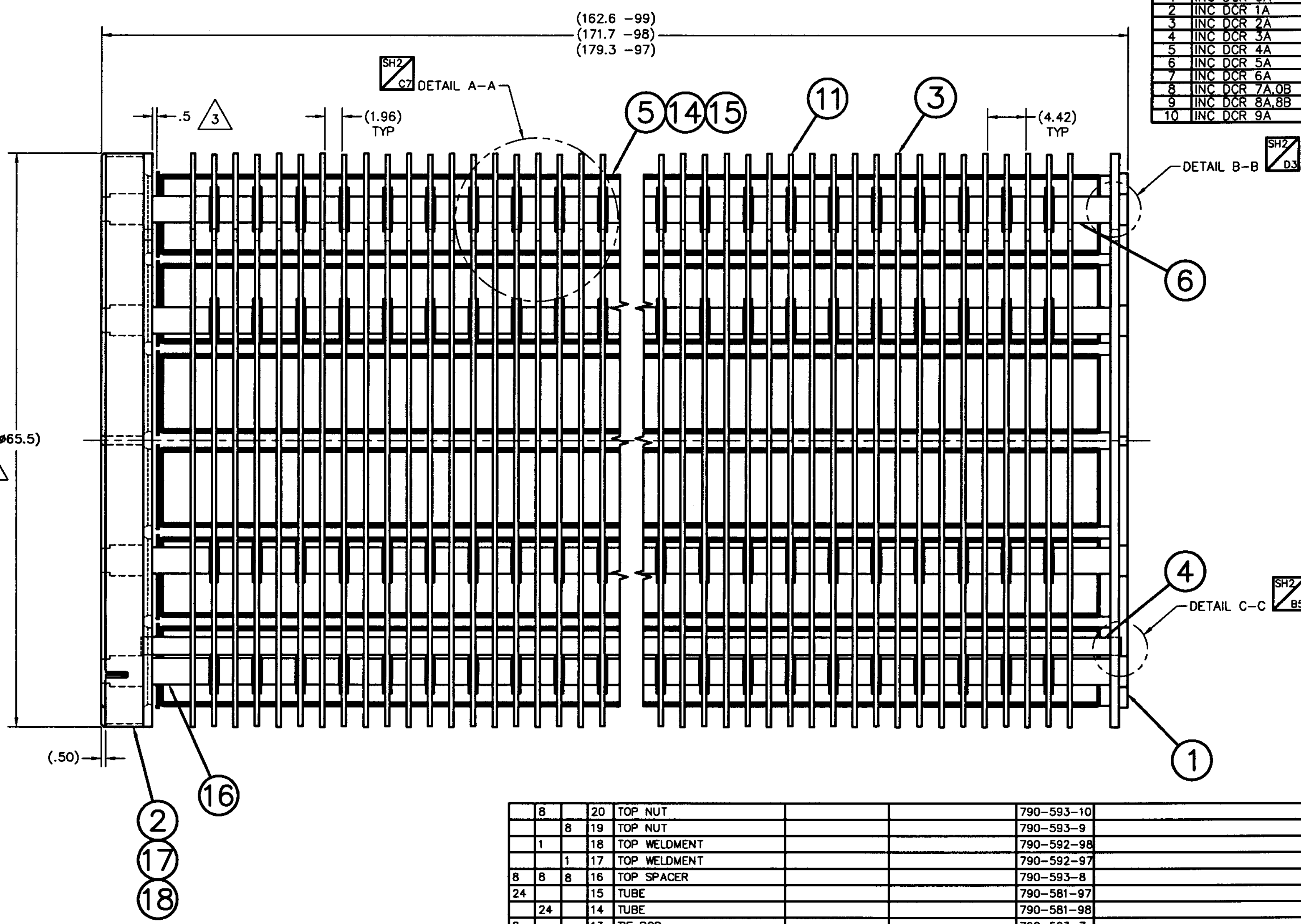
HEAT TRANSFER DISK,
FUEL BASKET,
24 ELEMENT PWR
NAC-UMS®

PROJECT	790	DRAWING	594	REV 2
SCALE 1/5	EST.WT.	SH 1 OF 1	9:12AM 10-8-99	

R-20305,1

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A
7	INC DCR 6A
8	INC DCR 7A,0B
9	INC DCR 8A,8B
10	INC DCR 9A





99 FUEL BASKET ASSEMBLY

98 FUEL BASKET ASSEMBLY

97 FUEL BASKET ASSEMBLY

97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
	8		20	TOP NUT			790-593-10	
		8	19	TOP NUT			790-593-9	
	1		18	TOP WELDMENT			790-592-98	
		1	17	TOP WELDMENT			790-592-97	
8	8	8	16	TOP SPACER			790-593-8	
24			15	TUBE			790-581-97	
	24		14	TUBE			790-581-98	
8			13	TIE ROD			790-593-7	
	8		12	TIE ROD			790-593-6	
33	31	29	11	HEAT TRANSFER DISK			790-594-1	
		8	10	TIE ROD			790-593-5	
8			9	TOP NUT			790-593-4	
528	496	464	8	FLAT WASHER	ST. STL.	COML ANSI B18.22.1		2 1/2 TYPE A PLAIN WASHER
528	496	464	7	SPLIT SPACER			790-593-2	
8	8	8	6	SPACER			790-593-3	
		24	5	TUBE			790-581-99	
1	1	1	4	DRAIN TUBE SLEEVE	304 ST. STL.	COML		2 DIA TUBE WITH .035 OR .065 WALL
34	32	30	3	SUPPORT DISK			790-593-1	
1			2	TOP WELDMENT			790-592-99	
1	1	1	1	BOTTOM WELDMENT			790-591-99	

QUANTITY

5 ASSEMBLY -99, -98, -97 TO BE FREELY INSERTED INTO CANISTER, DURING FINAL ASSEMBLY. ALTERNATE ASSEMBLY IN CANISTER OPTIONAL.

4 ITEM 4 LENGTH TO EXTEND BEYOND ITEM 2, 17, OR 18 SURFACE BY .25 +.02/-.25.

3 THIS DIMENSION IS TO BE HELD BY ADJUSTMENT OF THE TOP SPACERS (ITEM 16). LONGER OR SHORTER PARTS MAY BE USED. THIS DIMENSION APPLIES TO THE GAP BETWEEN THE TALLEST FUEL TUBE AND THE TOP WELDMENT ONLY.

2 (DELETED)

1 LUBRICATE WITH A SPENT FUEL POOL COMPATIBLE LUBRICANT SUCH AS NEOLUBE AND TORQUE NUTS TO 50±10 FT-LBS PRIOR TO WELDING.

NOTES:

10 6. IF AFTER FINAL ASSEMBLY, THE 1-8 UNC-2B THREADS OF ITEMS 9, 19 AND 20 (TOP NUT) ARE DAMAGED, THEY WILL NOT REQUIRE REPLACEMENT IF A MINIMUM OF 4 UNDAMAGED EQUALLY SPACED TOP NUTS REMAIN.

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM.	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

NEXT ASSEMBLY: 790-585

DRAWING TYPE: LICENSE

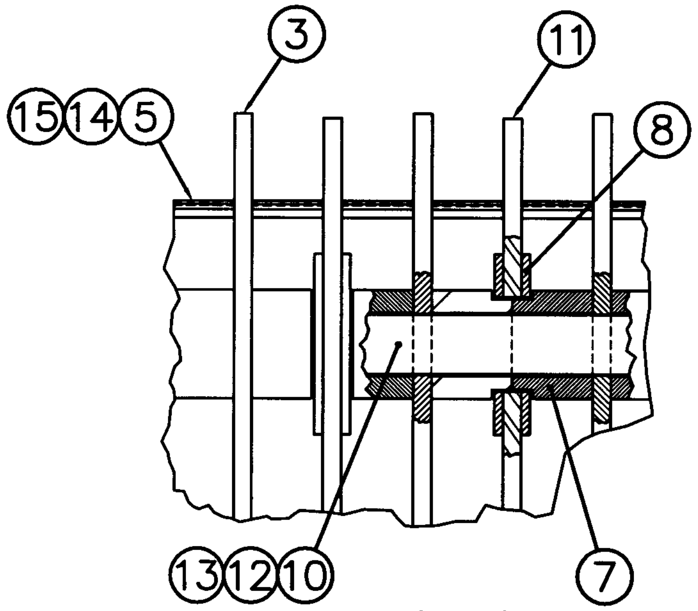
GROUP	NAME	DATE
PREPARER	R. Weldon	3-21-05
CHECKER	[signature]	3-21-05
PROJECT MANAGER	[signature]	3/21/05
DIRECTOR DESIGN AND ANALYSIS	[signature]	3/24/05
DIRECTOR, LICENSING	[signature]	3/24/05
VICE PRESIDENT, QUALITY	[signature]	3/25/05

NAC INTERNATIONAL

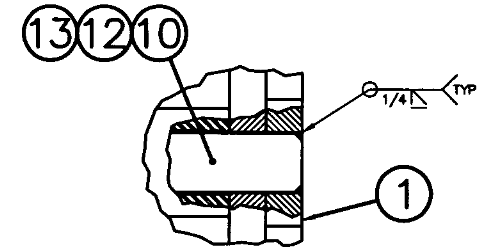
FUEL BASKET ASSEMBLY, 24 ELEMENT PWR NAC-UMS®

PROJECT 790 | DRAWING 595 | REV 10

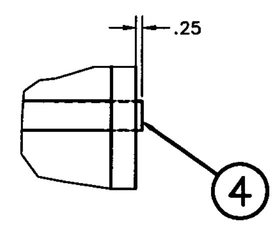
SCALE 1/6 | EST.WT. | SH 1 OF 2 | 4:30PM 3-21-2005



DETAIL A—A
SCALE: 1/2



DETAIL B—B
SCALE: 1/2



DETAIL C—C
SCALE: 1/2

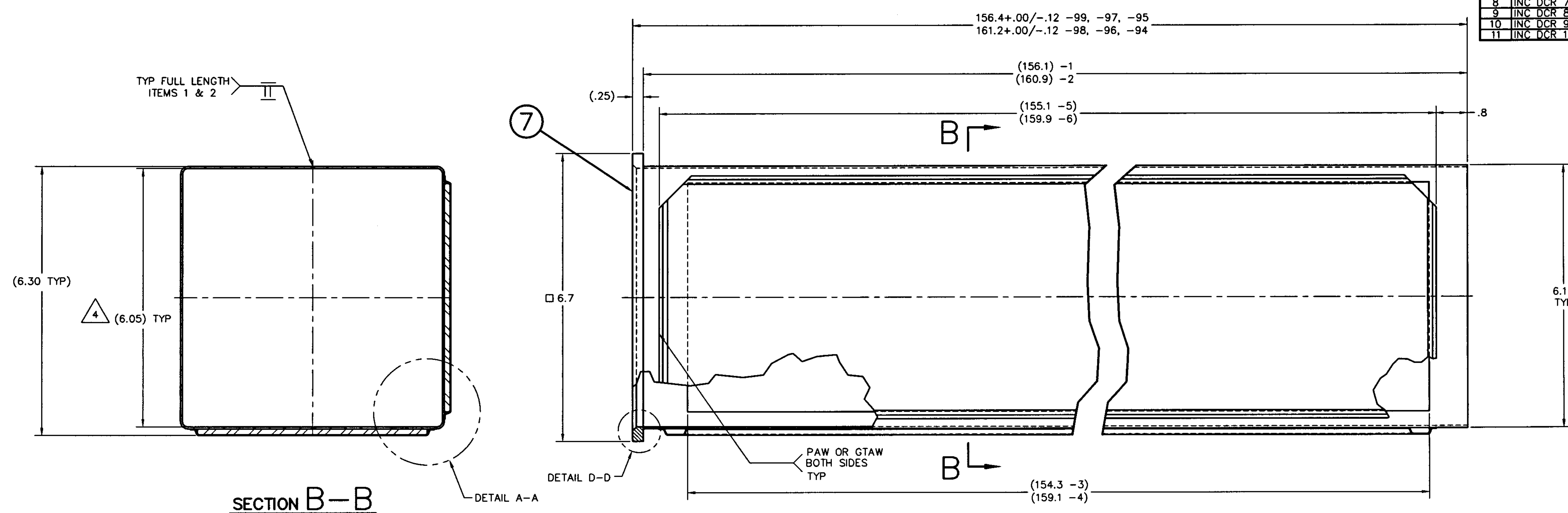
NAC INTERNATIONAL

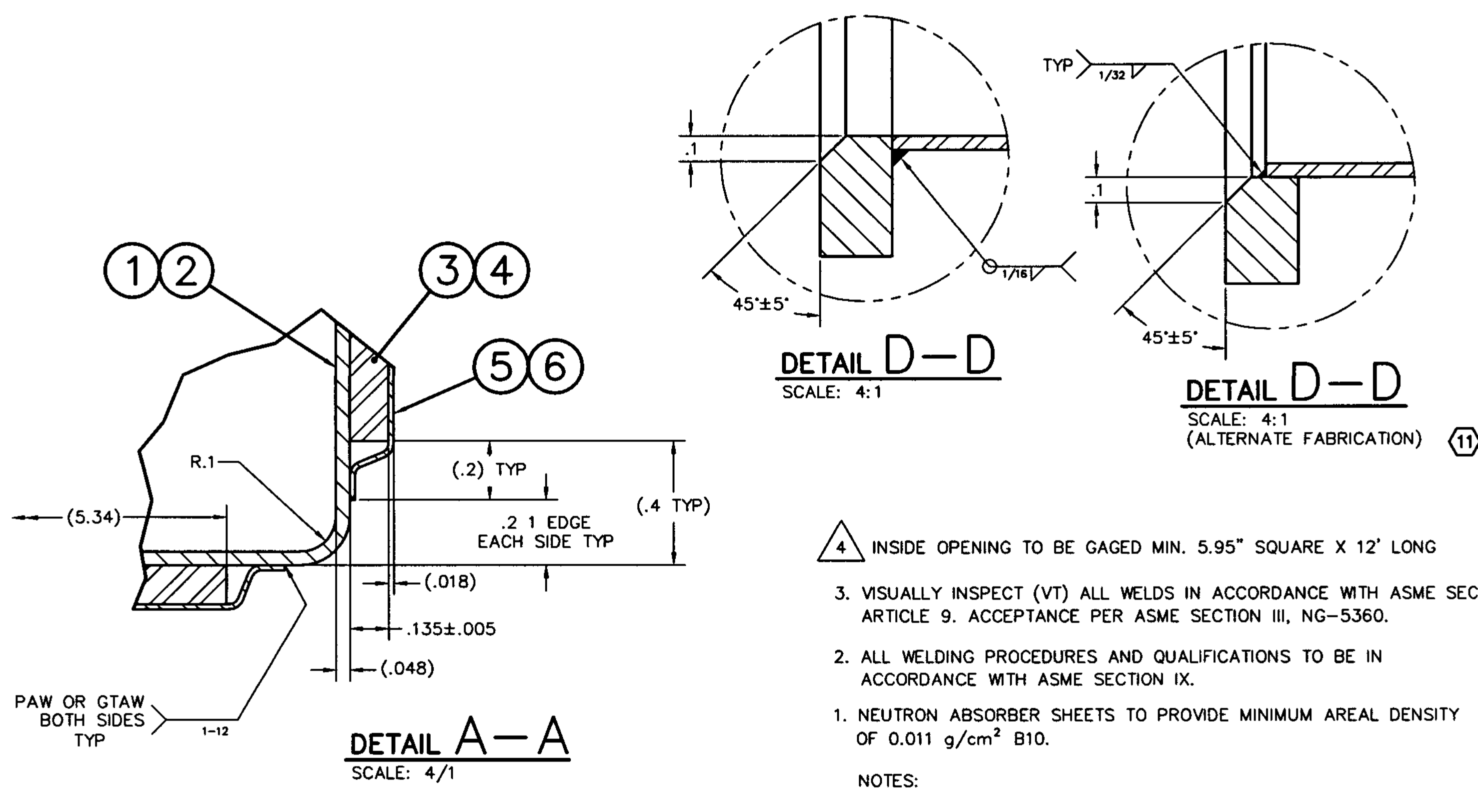
FUEL BASKET ASSEMBLY,
24 ELEMENT PWR
NAC-UMS®

PROJECT 790		DRAWING 595	REV 10
SCALE 1/6	EST.WT.	SH 2 OF 2	10:18AM 3-21-2005

R-202205

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A
5	INC DCR 4A
6	INC DCR 5A
7	INC DCR 6A
8	INC DCR 7A
9	INC DCR 8A
10	INC DCR 9A
11	INC DCR 10A





98 99 **TUBE ASSEMBLY** NEUTRON ABSORBER (2) SIDES

96 97 **TUBE ASSEMBLY** NEUTRON ABSORBER (1) SIDE

94 95 **TUBE ASSEMBLY** NO NEUTRON ABSORBER

94 ASSY	95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
1	1	1	1	1	1	7	FLANGE	304 ST.STL.	ASTM A240		1/4 PLATE
		1		2		6	CLADDING	304 ST.STL.	ASTM A240		26 GA SHEET
			1		2	5	CLADDING	304 ST.STL.	ASTM A240		26 GA SHEET
		1		2		4	NEUTRON ABSORBER	BORAL	SEE NOTE 1		.135 SHEET
			1		2	3	NEUTRON ABSORBER	BORAL	SEE NOTE 1		.135 SHEET
1		1		1		2	TUBING	304 ST.STL.	ASTM A240		18 GA SHEET
	1		1		1	1	TUBING	304 ST.STL.	ASTM A240		18 GA SHEET

QUANTITY

(11)

4 INSIDE OPENING TO BE GAGED MIN. 5.95" SQUARE X 12' LONG

3. VISUALLY INSPECT (VT) ALL WELDS IN ACCORDANCE WITH ASME SECTION V, ARTICLE 9. ACCEPTANCE PER ASME SECTION III, NG-5360.

2. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH ASME SECTION IX.

1. NEUTRON ABSORBER SHEETS TO PROVIDE MINIMUM AREAL DENSITY OF 0.011 g/cm² B10.

NOTES:

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

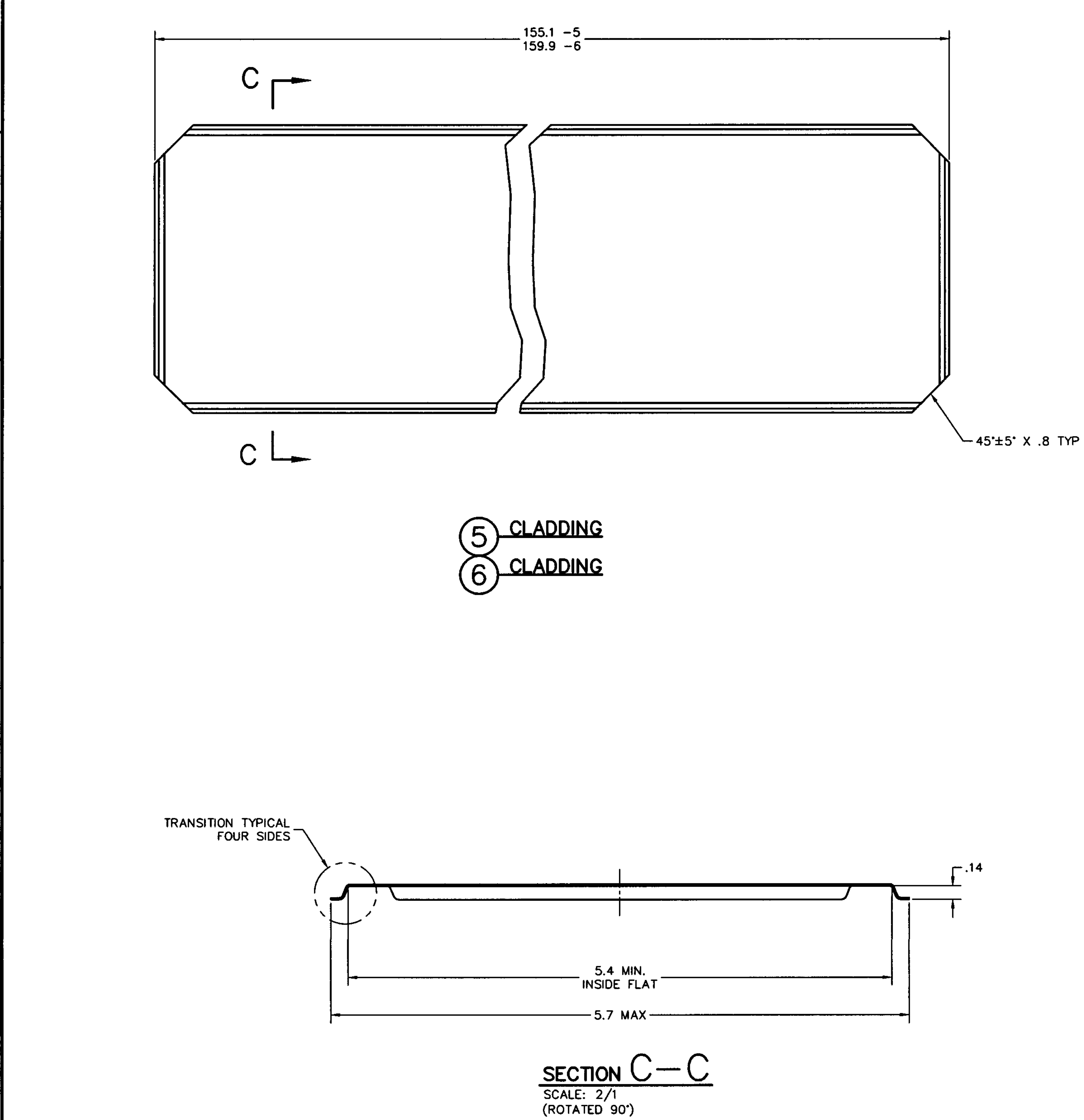
.XXX	UNDER 3	±.003	WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
	3-12	±.005	ALL DIMENSIONS ARE IN INCHES
	OVER 12	±.010	BORDER SIZE: F (40 X 28)
.XX	UNDER 6	±.02	ALL UNSPECIFIED TOOL RADII: .015 - .030
	6-18	±.03	BREAK ALL SHARP CORNERS .015 - .030
	OVER 18	±.06	MACHINED SURFACES TO BE ▽ OR BETTER
.X	ALL	±.1	NEXT ASSEMBLY: 790-570
FRACTIONAL ±1/8			
ANGLES ±0.5°			DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. Walker	9-22-04
CHECKER	[signature]	9-23-04
PROJECT MANAGER	[signature]	9/23/04
ENGINEERING	[signature]	9/23/04
LICENSING	J.C. Thompson	9-23-04
QUALITY	[signature]	9/23/04

NAC INTERNATIONAL

BWR FUEL TUBE, OVER-SIZED FUEL, NAC-UMS®

PROJECT 790		DRAWING 605	REV 11
SCALE 1/1	WEIGHT	SH 1 OF 2	9:54AM 9-22-2004



NAC INTERNATIONAL

BWR FUEL TUBE, OVER-SIZED FUEL, NAC-UMS®

PROJECT 790		DRAWING 605	REV 11
SCALE 1/1	WEIGHT	SH 2 OF 2	9-22-2004

R- 149867

REV	CHANGE
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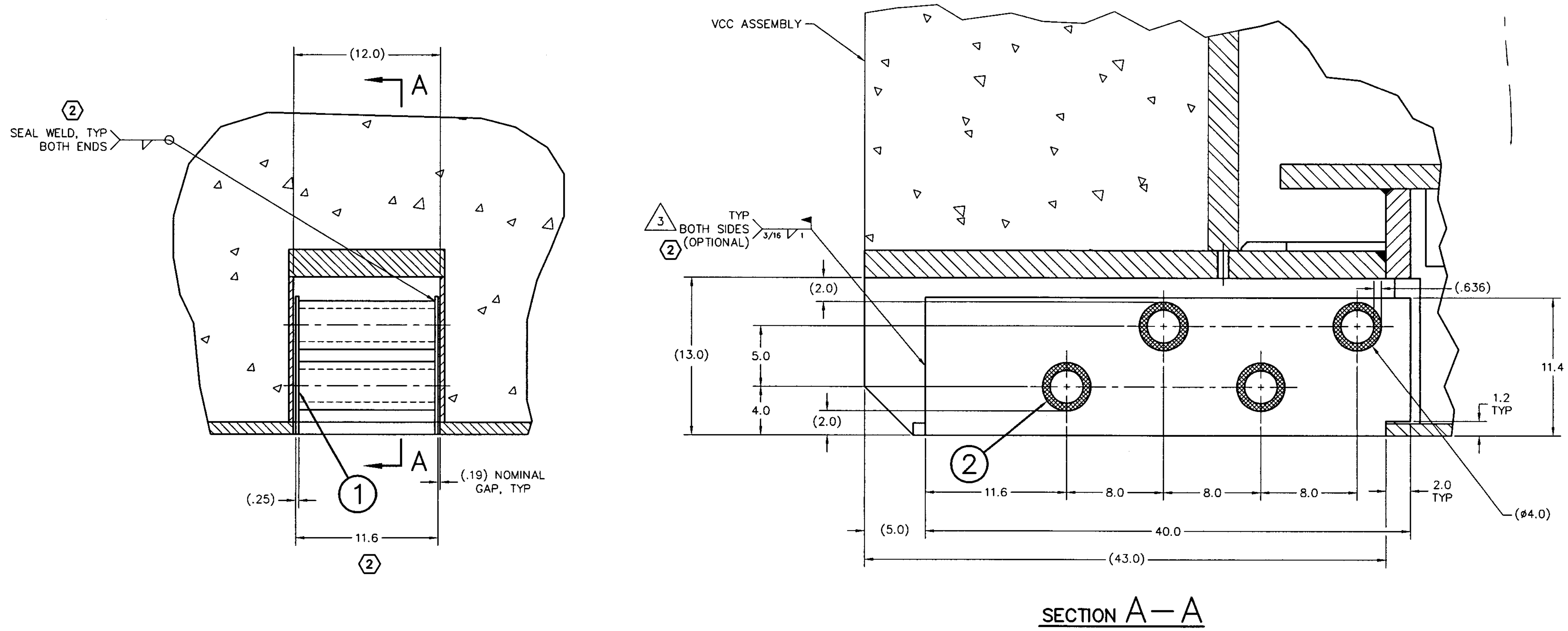
2	INC DCR 1A,1B,1C



99 SUPPLEMENTAL SHIELDING
WEIGHT: 147#

NOTES:

1. PREPARE ALL SURFACES IN ACCORDANCE W/ SSPC-SP1. COMMERCIAL BLAST CLEAN PER SSPC-SP6. APPLY KEELER & LONG KOLOR-POXY PRIMER No. 3200 TO ALL SURFACES PER MANUFACTURERS APPLICATION INSTRUCTIONS. APPLY ACRYTHANE ENAMEL Y-1-SERIES TOP COATING PER MANUFACTURERS APPLICATION INSTRUCTIONS.

(2) △2 PIPE, TUBING OR SOLID BAR MAY BE USED FOR BOM ITEM 2. PIPE SHOULD BE ASTM A53, GR. B/ASTM A106, GR. B, 3 1/2 NOMINAL DIA., XXH. TUBE SHOULD BE ASTM A519, 4" X 5/8" WALL. SOLID BAR SHOULD BE 4" DIA., ASTM A36.

△3 SHIM PLATES MAY BE UTILIZED TO FACILITATE FIELD WELDING OPERATIONS ON ONE OR BOTH SIDES OF THE SUPPLEMENTAL SHIELDING WELDMENT. LOCATE WELDS APPROX. AS SHOWN.

97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
		A/R	4	SHIMS	CARBON STEEL	ASTM A36		BY FIELD
		A/R	3	PAINT				SEE NOTE 1
		4	2	PIPE/TUBE/BAR	CARBON STEEL	SEE NOTE 2		SEE NOTE 2
		2	1	SIDE PLATE	CARBON STEEL	ASTM A36		1/4 PLATE

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽ OR BETTER			

NEXT ASSEMBLY: 790-561/562
DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	[signature]	10/23/02
CHECKER	[signature]	10-23-02
PROJECT MANAGER	[signature]	10-23-02
DIRECTOR DESIGN AND ANALYSIS	[signature]	10-23-02
DIRECTOR, LICENSING	[signature]	10/24/02
VICE PRESIDENT, QUALITY	[signature]	10/30/02

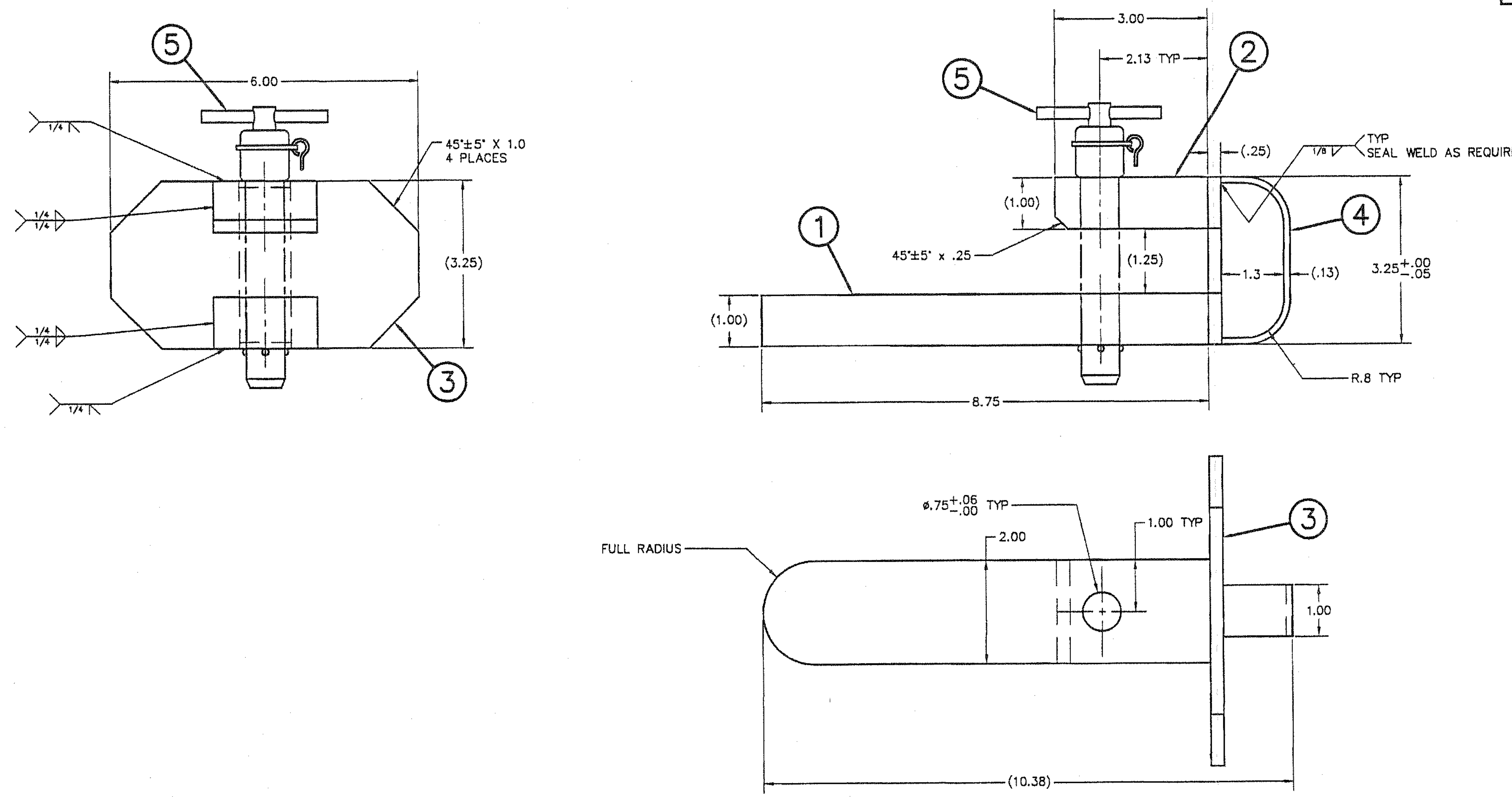
NAC INTERNATIONAL

SUPPLEMENTAL SHIELDING, VCC INLETS, NAC-UMS®

PROJECT 790 | DRAWING 613 | REV 2
SCALE 1/4 | EST.WT. NOTED | SH 1 OF 1 | 1:24PM 10-23-2002

R-20,7965

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A,1B
3	INC DCR 2A
4	INC DCR 3A



99 DOOR STOP

97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
		1	6	ATTACHMENT SCREW	ST. STL.	COML		10-32 UNC-2A 1/2 LG
		1	5	LOCK PIN	ST. STL.	COML		SEE NOTE 2
		1	4	HANDLE	304 ST. STL.	ASTM A240/A276		1/8 PLATE/BAR
		1	3	BACK PLATE	304 ST. STL.	ASTM A240		1/4 PLATE
		1	2	TOP PLATE	304 ST. STL.	ASTM A240/A276		1 PLATE/BAR
		1	1	BOTTOM PLATE	304 ST. STL.	ASTM A240/A276		1 PLATE/BAR

QUANTITY

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94. UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

.XXX	TOL.	.XX	TOL.
UNDER 3	±.003	UNDER 6	±.02
3–12	±.005	6–18	±.03
OVER 12	±.010	OVER 18	±.06
.X	±.1	ANGLES ±0.5°	

ALL UNSPECIFIED TOOL RADII: .015 – .030
BREAK ALL SHARP CORNERS .015 – .030
ALL UNSPECIFIED MACHINED SURFACES SHALL BE 125 OR BETTER
NEXT ASSEMBLY: N/A
DRAWING TYPE: LICENSE

SYM. GEOMETRY: FLATNESS, STRAIGHTNESS, ANGULARITY, PERPENDICULARITY, PARALLELISM, CONCENTRICITY, TRUE POSITION

GROUP	NAME	DATE
PREPARER	[signature]	10/20/08
CHECKER	[signature]	10/21/08
PROJECT MANAGER	[signature]	10/22/08
ENGINEERING	[signature]	10/23/08
LICENSING	[signature]	10/23/08
QUALITY	[signature]	10/31/08

NAC INTERNATIONAL

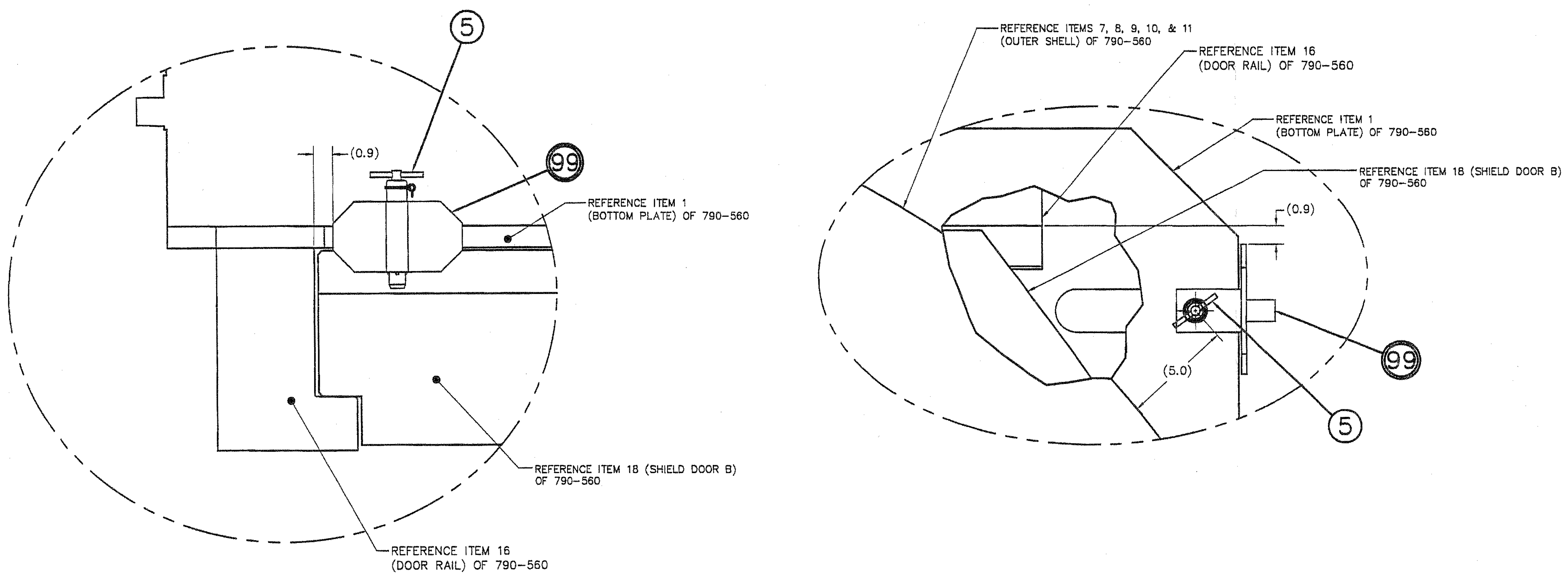
DOOR STOP
NAC-UMS®

PROJECT 790 | DRAWING 617 | REV 4
SCALE | EST.WT. | SH 1 OF 2

2. CARRLANE #CL-12-BPDT-3.25-S-4BALL-C OR McMASTER-CARR #93750A910

△1 ITEM 6 TO BE FIELD LOCATED AND INSTALLED FOR CABLE.

NOTE:



NAC INTERNATIONAL

DOOR STOP
NAC-UMS®

PROJECT 790	DRAWING 617	REV 4
	SH 2 OF 2	7:11AM 10-20-2008

K 150504

REV	CHANGE
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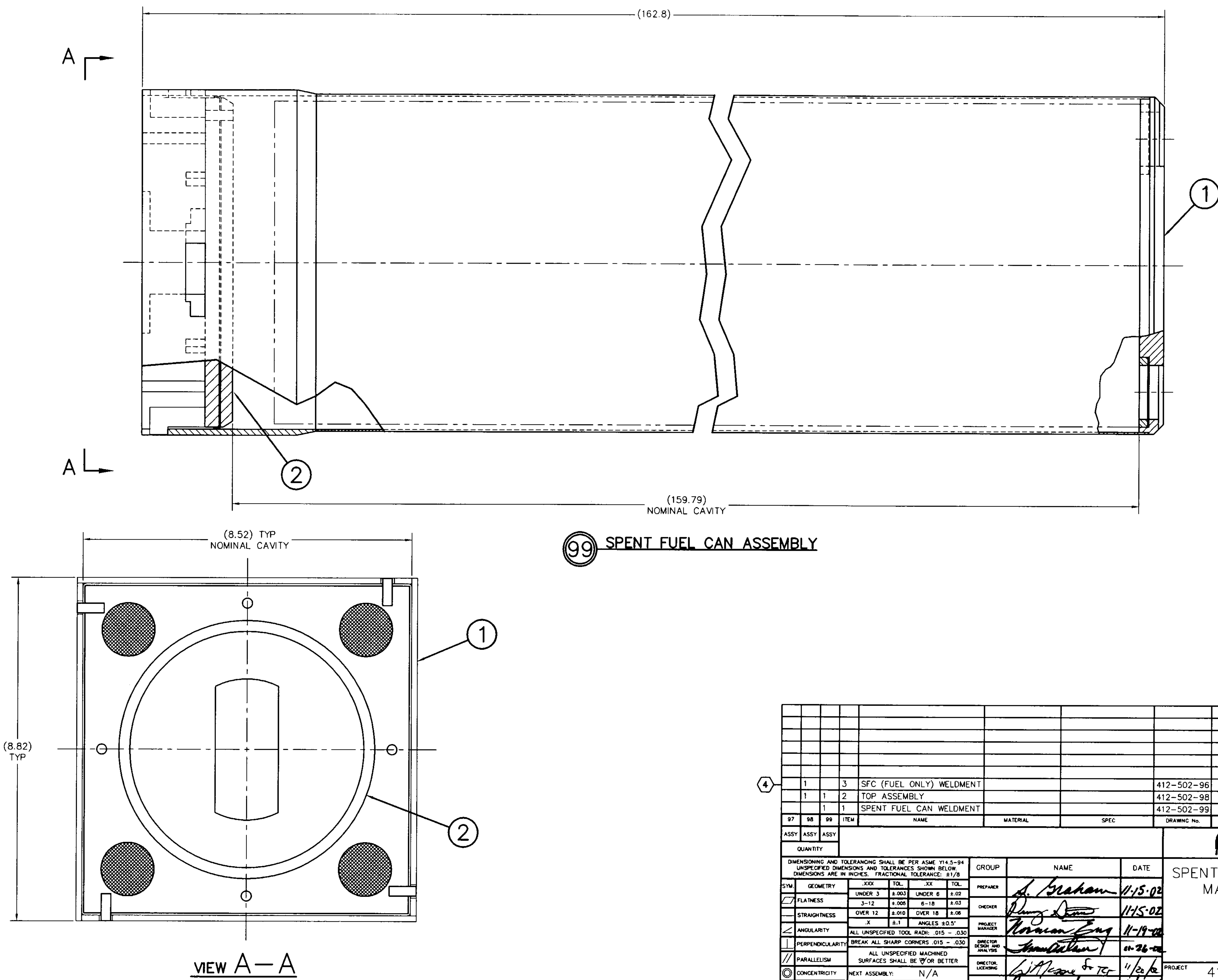
2	INC DCR 1A
3	INC DCR 2A
4	INC DCR 3A



99 SPENT FUEL CAN ASSEMBLY

VIEW A–A

97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
1			3	SFC (FUEL ONLY) WELDMENT			412-502-96	
1	1		2	TOP ASSEMBLY			412-502-98	
	1	1	1	SPENT FUEL CAN WELDMENT			412-502-99	
ASSY	ASSY	ASSY						
QUANTITY								

(4)

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM	GEOMETRY	.XXX	TOL.	.XX	TOL.
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	.X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII: .015 – .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 – .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE ▽/OR BETTER			

NEXT ASSEMBLY: N/A

DRAWING TYPE: LICENSE

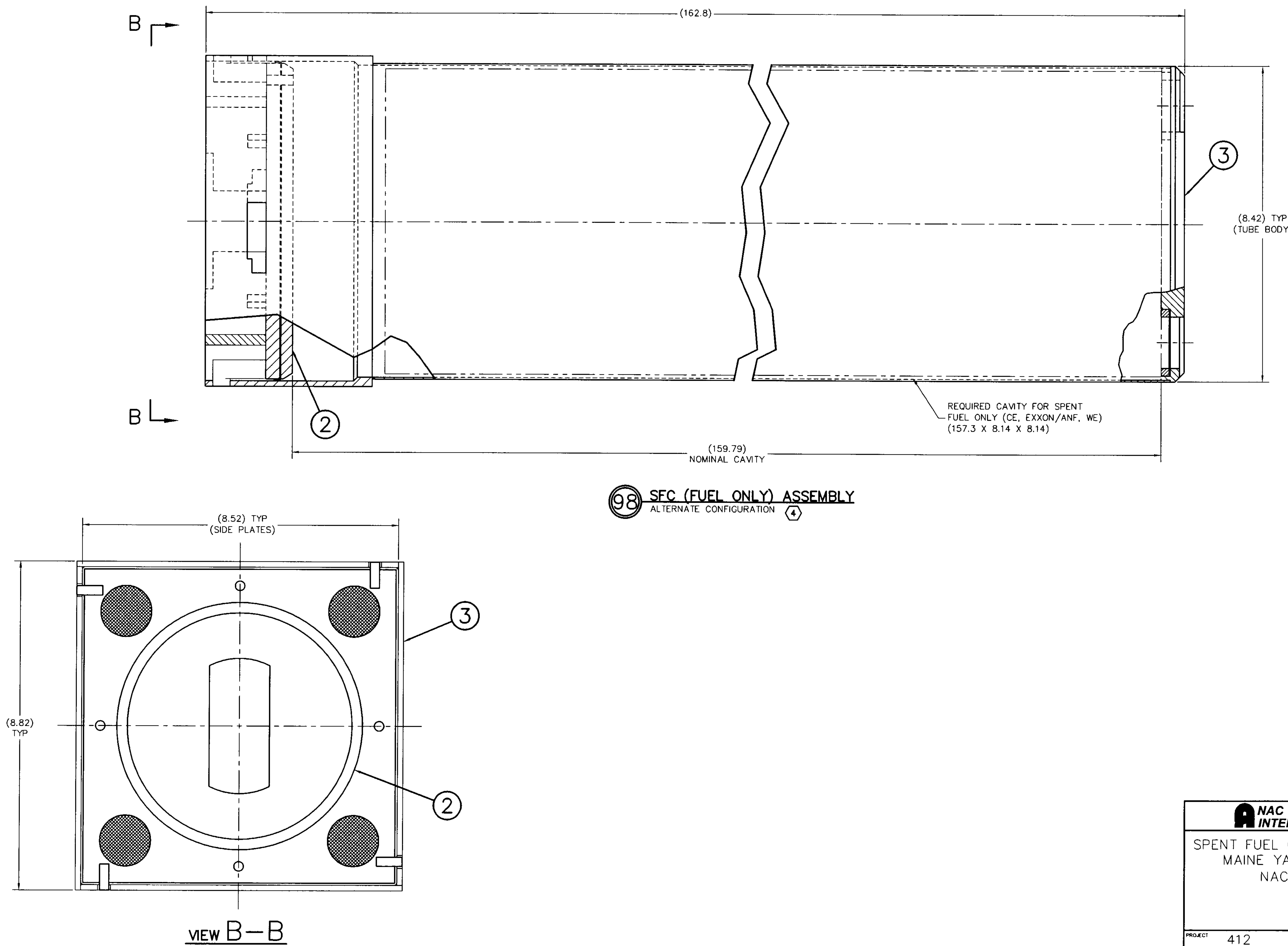
GROUP	NAME	DATE
PREPARER	S. Graham	11-15-02
CHECKER	Danny Dunn	11-15-02
PROJECT MANAGER	Norman Eng	11-19-02
DIRECTOR DESIGN AND ANALYSIS	[illegible]	11-26-02
DIRECTOR, LICENSING	W. Moore for TC	11/20/02
VICE PRESIDENT, QUALITY	[illegible]	11/26/02

NAC INTERNATIONAL

SPENT FUEL CAN ASSEMBLY
MAINE YANKEE (MY)
NAC-UMS®

PROJECT 412 | DRAWING 501 | REV 4

SCALE FULL | EST.WT. 133# | SH 1 OF 2 | 10:49AM 11-15-2002



NAC INTERNATIONAL

SPENT FUEL CAN ASSEMBLY
MAINE YANKEE (MY)
NAC-UMS®

PROJECT 412		DRAWING 501	REV 4
SCALE FULL	EST.WT. 133#	SH 2 OF 2	11:50AM 11-15-2002

R-203040

REV	CHANGE
0	INITIAL ISSUE
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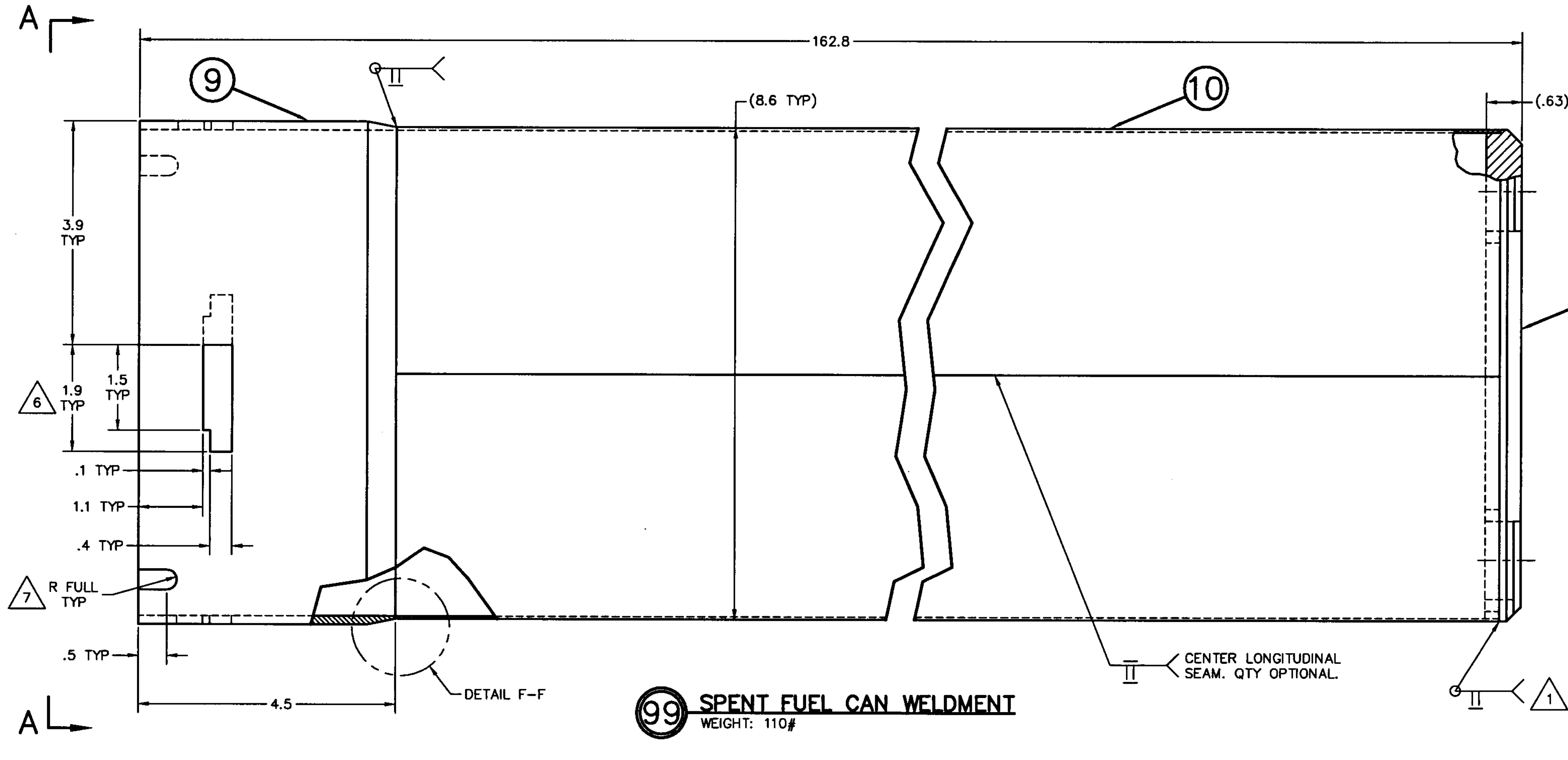
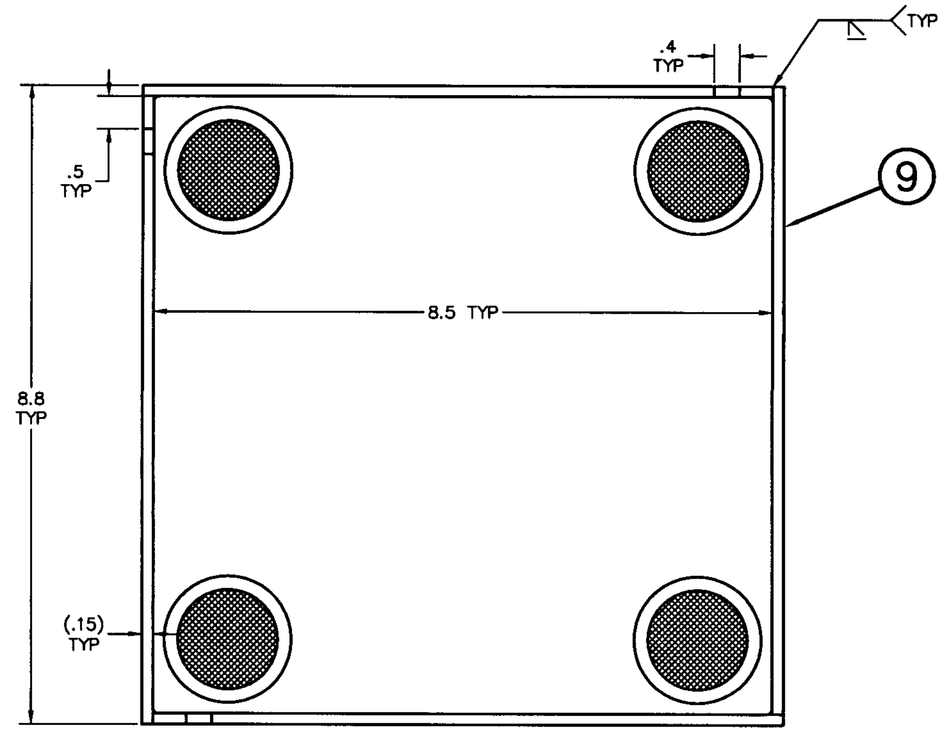
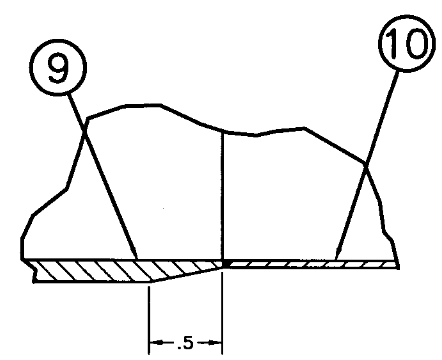
3	INC DCR 2A,2B
4	INC DCR 3A,3B,3C
5	INC DCR 4A
6	INC DCR 5A



99 SPENT FUEL CAN WELDMENT
WEIGHT: 110#



VIEW A—A



DETAIL F—F
SCALE: 2/1

6 /10\ MINOR EDGE SURFACE AND/OR DIMENSIONAL DEVIATIONS ARE ACCEPTABLE FOR ITEM 4, WIPER.

9. VISUALLY INSPECT (VT) ALL WELDS IN ACCORDANCE WITH ASME SECTION V, ARTICLE 9. ACCEPTANCE PER ASME SECTION III, NG-5360.

8. ALL WELDING PROCEDURES AND QUALIFICATIONS TO BE IN ACCORDANCE WITH ASME SECTION IX.

/7\ CORNERS MAY BE ROUNDED PROVIDED THE SLOT OPENING DOES NOT EXCEED 0.5

/6\ BREAK ALL INTERIOR EDGES OF SLOT.

5. FOR ITEM 4 WIPER MATERIAL IS 6061-T6 ALUMINUM ASTM B209, .032 SHEET OR ITEM 4 ALTERNATE WIPER IS ST. STL. COMMERCIAL, .005-.008 (.127-.203MM) SHEET/STRIP, MATERIAL FOR ST. STL. WIPER TO HAVE A MINIMUM YIELD STRENGTH OF 100KSI.

4. THE CORRESPONDING DESIGN DRAWING FOR THE ALTERNATE CONFIGURATION IS 412-109.

/3\ MACHINE ITEM 20, BOTTOM ASSEMBLY, FOR SLIP FIT OF ITEM 19, TUBE BODY.

/2\ MACHINE ITEM 1, FOR LIGHT PRESS FIT WITH ITEM 8 OR 17, BOTTOM PLATE. THE THICKNESS OF ITEM 1 IS MAINTAINED SO THAT IT IS FLUSH WITH ITEM 8 OR 17 WHEN THE SCREENS ARE IN PLACE AND THE TWO ITEMS ARE WELDED TOGETHER.

/1\ MACHINE ITEM 11, BOTTOM ASSEMBLY, FOR SLIP FIT OF ITEM 10, TUBE BODY.

NOTES:

95 ASSY	96 ASSY	97 ASSY	98 ASSY	99 ASSY	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION
	1				20	BOTTOM ASSEMBLY			412-502-95	
	1				19	TUBE BODY	304 ST. STL.	ASME SA240		18 GAGE SHEET
	4				18	SIDE PLATE	304 ST. STL.	ASME SA240/SA479		PLATE/BAR
1					17		304 ST. STL.	ASME SA240		5/8 PLATE
			4		16	DOWEL PIN	ST. STL.	COML		1/4 DIA X 3/4 LONG
			A/R		15	BACKING SCREEN	ST. STL.	COML		NEWARK ZQ1 16 X 16 .023
			A/R		14	FILTER SCREEN	ST. STL.	COML		NEWARK PAC 250 X 250 X .0016
			1		13	SUPPORT RING	304 ST. STL.	ASME SA312		6" SCH 40 PIPE
			1		12	LIFT TEE	304 ST. STL.	ASME SA240/SA479		PLATE/BAR
				1	11	BOTTOM ASSEMBLY			412-502-97	
				1	10	TUBE BODY	304 ST. STL.	ASME SA240		18 GAGE SHEET
				4	9	SIDE PLATE	304 ST. STL.	ASME SA240/SA479		PLATE/BAR
		1			8	BOTTOM PLATE	304 ST. STL.	ASME SA240		5/8 PLATE
A/R		A/R			7	BACKING SCREEN	ST. STL.	COML		NEWARK ZQ1 16 X 16 .023
A/R		A/R			6	FILTER SCREEN	ST. STL.	COML		NEWARK PAC 250 X 250 X .0016
			1		5	LID BOTTOM	304 ST. STL.	ASME SA240		5/16 PLATE
			1		4	WIPER	SEE NOTE 5	SEE NOTE 5		SEE NOTE 5
			4		3	LID GUIDE	304 ST. STL.	ASME SA240/SA479		1/4 PLATE/BAR
			1		2	LID PLATE	304 ST. STL.	ASME SA240		3/8 PLATE
4		4			1	COLLAR	304 ST. STL.	ASME SA240/SA479		PLATE/BAR

QUANTITY

UNLESS OTHERWISE STATED
DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94. UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.
ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

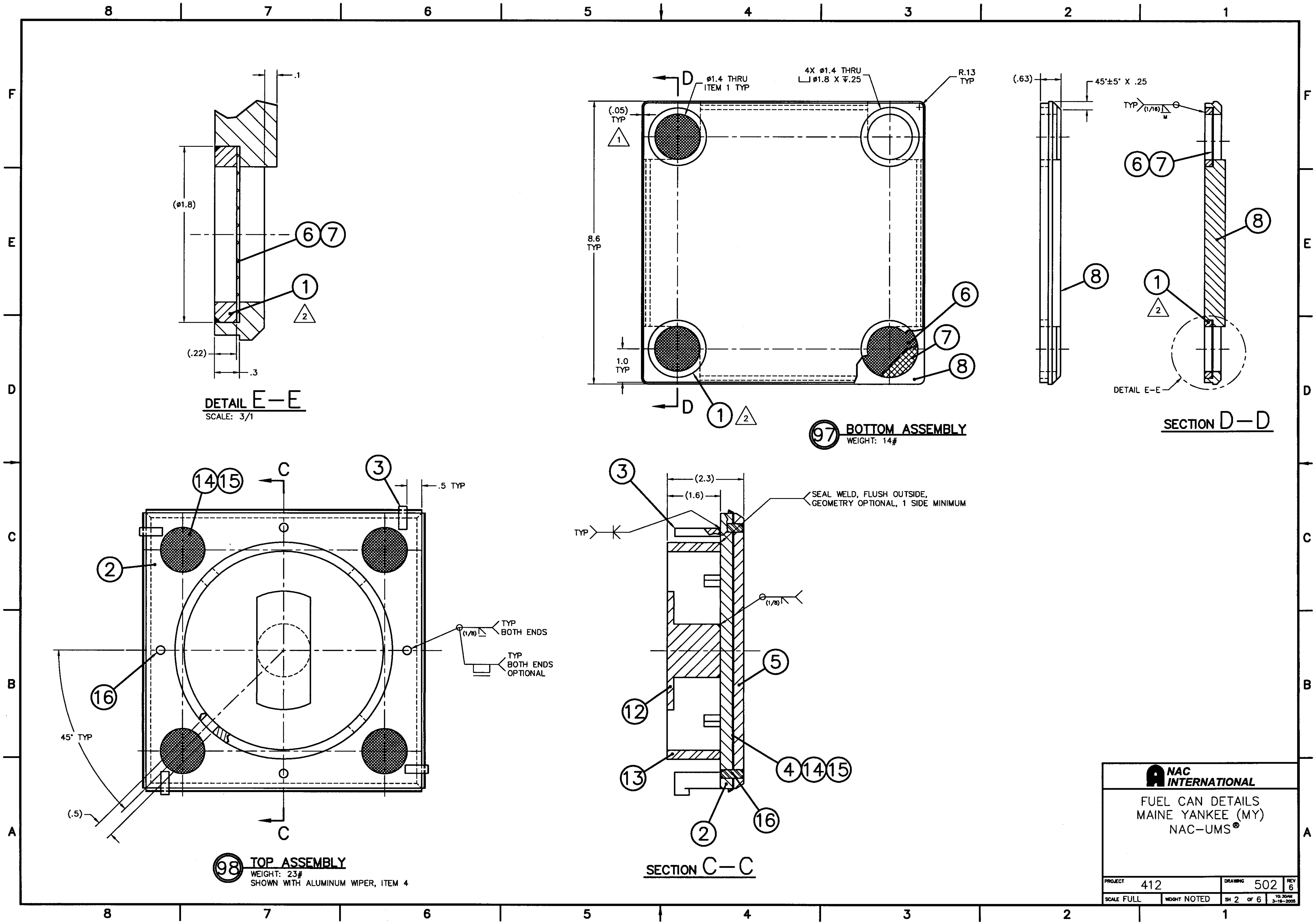
XXX' UNDER 3	±.003	WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
3-12	±.005	ALL DIMENSIONS ARE IN INCHES
OVER 12	±.010	BORDER SIZE: F (40 X 28)
XX' UNDER 6	±.02	ALL UNSPECIFIED TOOL RADII: .015 - .030
6-18	±.03	BREAK ALL SHARP CORNERS .015 - .030
OVER 18	±.06	MACHINED SURFACES TO BE ∇ OR BETTER
X' ALL	±.1	NEXT ASSEMBLY: 412-501
FRACTIONAL	±1/8	
ANGLES	±0.5°	DRAWING TYPE: LICENSE

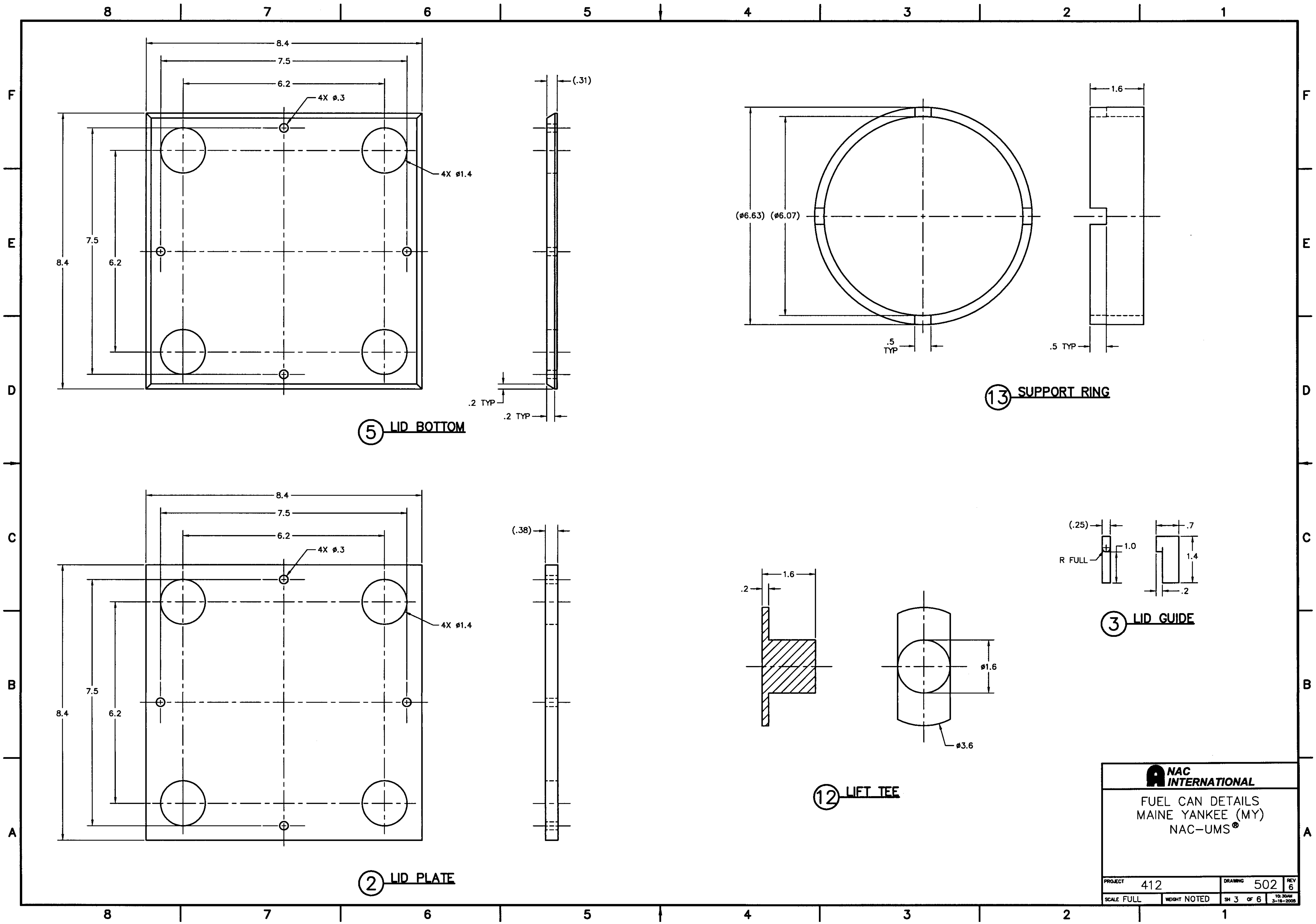
GROUP	NAME	DATE
PREPARER	[signature]	3-16-05
CHECKER	[signature]	3/18/05
PROJECT MANAGER	[signature]	3/22/05
ENGINEERING	[signature]	3/24/05
LICENSING	[signature]	3/24/05
QUALITY	[signature]	3/25/05

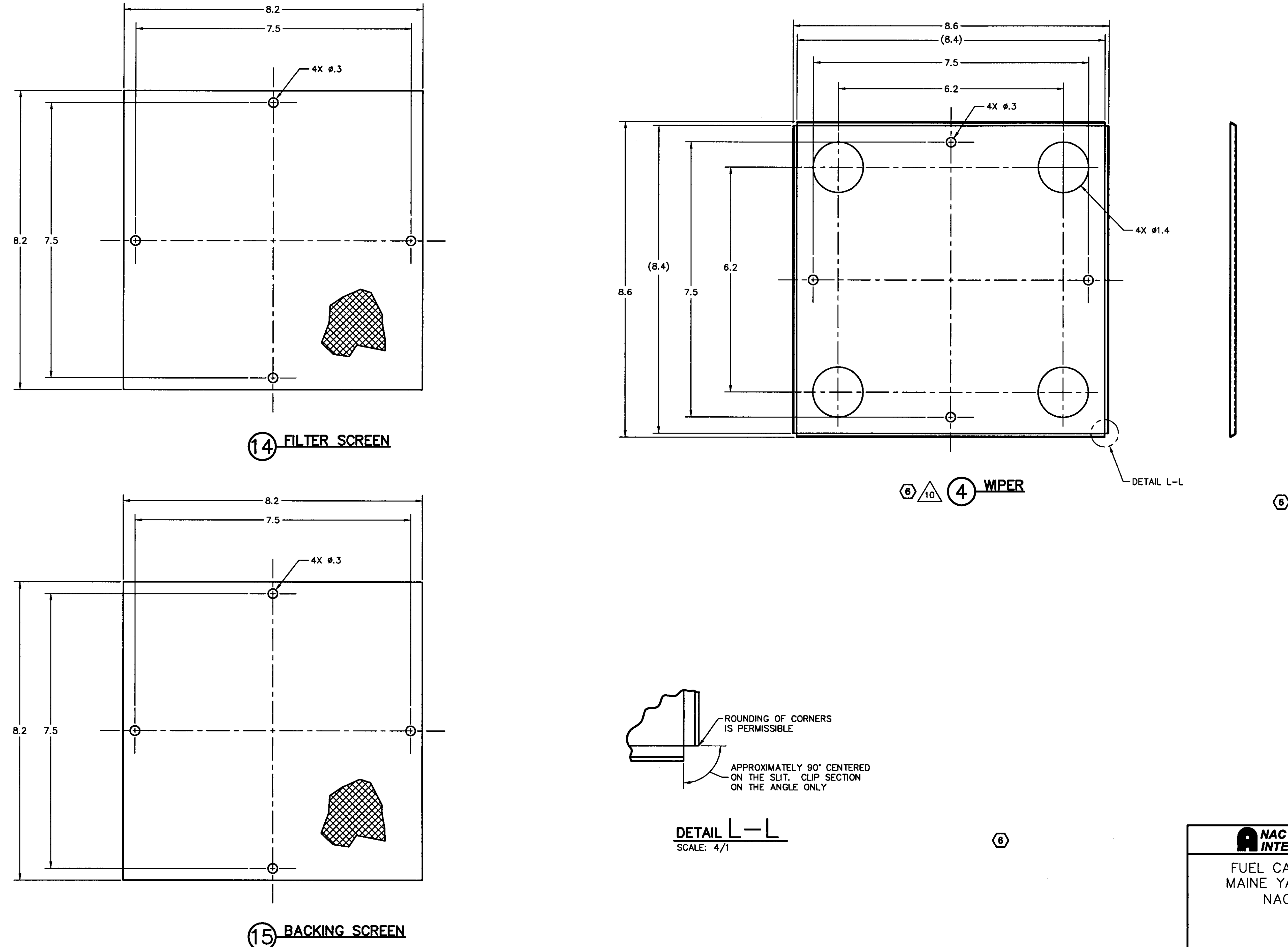
NAC INTERNATIONAL

FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC-UMS®

PROJECT 412 | DRAWING 502 | REV 6
SCALE FULL | WEIGHT NOTED | SH 1 OF 6 | 10:28AM 3-16-2005


DETAIL E—E
SCALE: 3/1
(ø1.8)
(.22)
.3
.1
BOTTOM ASSEMBLY
WEIGHT: 14#
ø1.4 THRU
ITEM 1 TYP
4X ø1.4 THRU
ø1.8 X ↧.25
R.13
TYP
(.05)
TYP
8.6
TYP
1.0
TYP
(.63)
45°±5° X .25
TYP
(1/16)
DETAIL E-E
SECTION D—D
TOP ASSEMBLY
WEIGHT: 23#
SHOWN WITH ALUMINUM WIPER, ITEM 4
.5 TYP
45° TYP
(.5)
TYP
BOTH ENDS
TYP
BOTH ENDS
OPTIONAL
SECTION C—C
(2.3)
(1.6)
SEAL WELD, FLUSH OUTSIDE,
GEOMETRY OPTIONAL, 1 SIDE MINIMUM
(1/8)
NAC INTERNATIONAL
FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC-UMS®
PROJECT 412
DRAWING 502
REV 6
SCALE FULL
WEIGHT NOTED
SH 2 OF 6

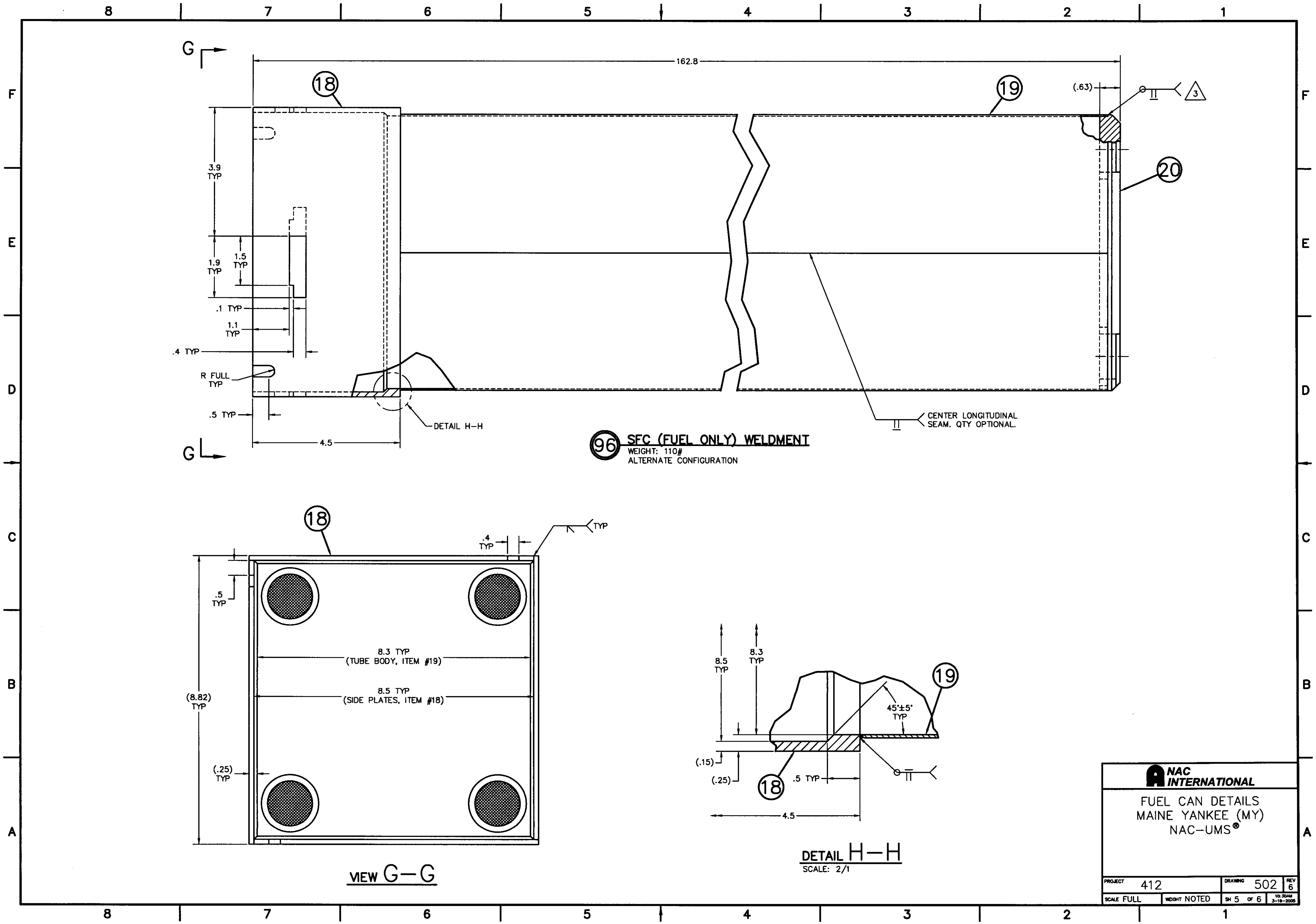

8.4
7.5
6.2
4X ø.3
4X ø1.4
(.31)
.2 TYP
.2 TYP
5 LID BOTTOM
(.38)
2 LID PLATE
(ø6.63) (ø6.07)
.5 TYP
.5 TYP
1.6
13 SUPPORT RING
(.25)
1.0
R FULL
.7
1.4
.2
3 LID GUIDE
1.6
.2
ø1.6
ø3.6
12 LIFT TEE
NAC INTERNATIONAL
FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC-UMS®
PROJECT 412
DRAWING 502
REV 6
SCALE FULL
WEIGHT NOTED
SH 3 OF 6



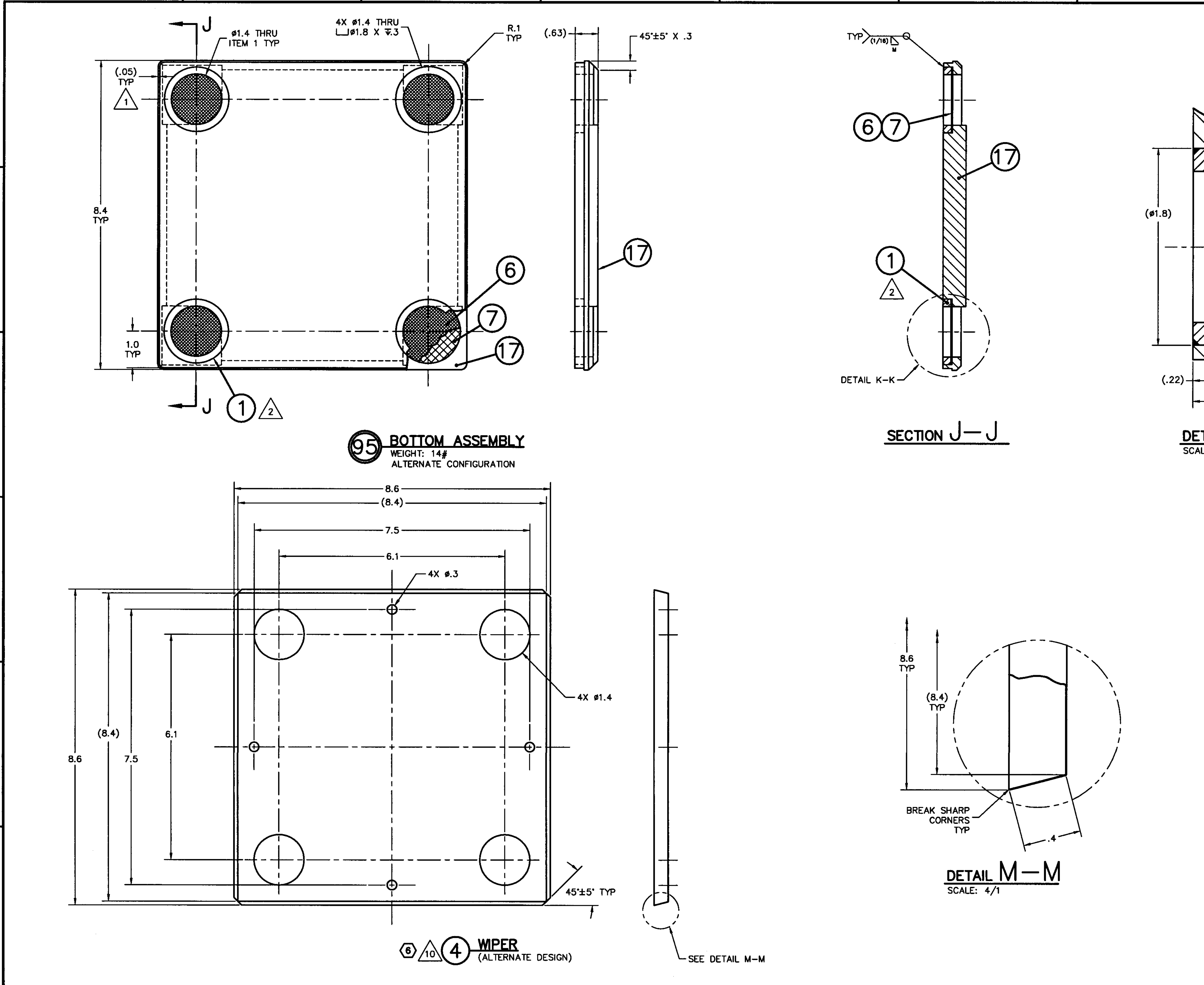
NAC INTERNATIONAL

FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC–UMS®

PROJECT 412	DRAWING 502	REV 6
SCALE FULL	WEIGHT NOTED	SH 4 OF 6 · 10:29AM 3-16-2005



162.8
(.63)
3
18
19
20
3.9
TYP
1.9
TYP
1.5
TYP
.1 TYP
1.1
TYP
.4 TYP
R FULL
TYP
.5 TYP
4.5
DETAIL H-H
CENTER LONGITUDINAL
SEAM. QTY OPTIONAL.
96 SFC (FUEL ONLY) WELDMENT
WEIGHT: 110#
ALTERNATE CONFIGURATION
.4
TYP
TYP
.5
TYP
8.3 TYP
(TUBE BODY, ITEM #19)
8.5 TYP
(SIDE PLATES, ITEM #18)
(8.82)
TYP
(.25)
TYP
VIEW G—G
8.5
TYP
8.3
TYP
45°±5°
TYP
(.15)
(.25)
.5 TYP
DETAIL H—H
SCALE: 2/1
NAC INTERNATIONAL
FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC–UMS®
PROJECT 412
DRAWING 502
REV 6
SCALE FULL
WEIGHT NOTED
SH 5 OF 6
10:30AM
3-16-2005



NAC INTERNATIONAL

FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC-UMS®

PROJECT	412	DRAWING	502	REV 6
SCALE FULL	WEIGHT NOTED	SH 6 OF 6	9:18AM 3-25-2005	

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List of Tables

2.0 PRINCIPAL DESIGN CRITERIA

The Universal Storage System is a canister-based spent fuel dry storage cask system that is designed to be compatible with the Universal Transportation System. It is designed to store a variety of PWR and BWR fuel assemblies. This chapter presents the design bases, including the principal design criteria, limiting load conditions, and operational parameters of the Universal Storage System. The principal design criteria are summarized in Table 2-1.

Table 2-1 Summary of Universal Storage System Design Criteria

Parameter	Criteria
Design Life	50 years
Design Code - Confinement	ASME Code, Section III, Subsection NB [1] for confinement boundary
Design Code - Nonconfinement	
Basket	ASME Code, Section III, Subsection NG [2] and NUREG/CR-6322 [3]
Vertical Concrete Cask	ACI-349 [4], ACI-318 [5]
Transfer Cask	ANSI N14.6 [6] and NUREG-0612 [7]
Maximum Weight:	
Canister with Design Basis PWR Fuel Assembly (dry, including inserts) (Class 2)	72,900 lbs.
Canister with Design Basis BWR Fuel (dry) (Class 5)	75,600 lbs.
Vertical Concrete Cask (loaded) (Class 5)	323,900 lbs.
Transfer Cask (Class 3)	121,500 lbs.
Thermal:	
Maximum Fuel Cladding Temperature:	
PWR Fuel	752°F (400°C) for Normal and Transfer [25] 1058°F (570°C) Off-Normal and Accident [21]
BWR Fuel	752°F (400°C) for Normal and Transfer [25] 1058°F (570°C) Off-Normal and Accident [21]
Ambient Temperature:	
Normal (average annual ambient)	76°F
Off-Normal (extreme cold; extreme hot)	-40°F; 106°F
Accident	133°F
Concrete Temperature:	
Normal Conditions	≤ 150°F (bulk); ≤ 200°F (local) [24]
Off-Normal/Accident Conditions	≤ 350°F local/ surface [4]
Cavity Atmosphere	Helium

Table 2-1 Summary of Universal Storage System Design Criteria (Continued)

Radiation Protection/Shielding	**Criteria**
Concrete Cask Side Wall Contact Dose Rate	< 50 mrem/hr. (avg)
Concrete Cask Top Lid Contact Dose Rate	< 50 mrem/hr. (avg)
Concrete Cask Air Inlet/Outlet Dose Rate	< 100 mrem/hr. (avg)
Owner Controlled Area Boundary Dose [11] Normal/Off-Normal Conditions Accident Whole Body Dose	 25 mrem (Annual Whole Body) 5 rem (Whole Body)

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2.1 Spent Fuel To Be Stored

The Universal Storage System is designed to safely store up to 24 PWR spent fuel assemblies, or up to 56 BWR spent fuel assemblies, contained within a Transportable Storage Canister. On the basis of fuel assembly length and cross-section, the fuel assemblies are grouped into three classes of PWR fuel assemblies and two classes of BWR fuel assemblies. The class of the fuel assemblies is shown in Tables 6.2-1 and 6.2-2 for PWR and BWR fuel, respectively, and is based primarily on overall length.

The PWR and BWR fuel having the parameters shown in Tables 2.1.1-1 and 2.1.2-1, respectively, may be stored in the Universal Storage System. As shown in Table 2.1.1-1, the evaluation of PWR fuel includes fuel having thimble plugs and burnable poison rods in guide tube positions. In addition, solid stainless steel rods may be inserted into guide tube positions as long as the fuel assembly weight limits in Table 2.1.1-1 are not exceeded and no soluble boron credit is taken. As shown in Table 2.1.2-1, the BWR fuel evaluation includes fuel with a zirconium alloy channel. Any empty fuel rod position must be filled with a solid filler rod fabricated from either zirconium alloy or Type 304 stainless steel, or may be solid neutron absorber rods inserted for in-core reactivity control prior to reactor operation.

In addition to the design basis fuel, fuel that is unique to a reactor site, referred to as site specific fuel, is also evaluated. Site specific fuel consists of fuel assemblies that are configured differently, or have different parameters (such as enrichment or burnup), than the design basis fuel assemblies.

Site specific fuel is described in Section 2.1.3.

Site specific fuel is shown to be bounded by the fuel parameters shown in Tables 2.1.1-1 or 2.1.2-1, or it is separately evaluated.

The minimum initial enrichment limits are shown in Tables 2.1.1-2 and 2.1.2-2 for PWR and BWR fuel, respectively. The minimum enrichment limits exclude the loading of fuel assemblies enriched to less than 1.9 wt.% ^{235}U, including unenriched fuel assemblies, into the Transportable Storage Canister. However, fuel assemblies with unenriched axial end-blankets may be loaded into the canister.

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2.1.1 PWR Fuel Evaluation

The parameters of the PWR fuel assemblies that may be loaded in the transportable storage canister (canister) are shown in Table 2.1.1-1. The maximum initial enrichment limit represents the maximum fuel rod enrichment limit for variably enriched PWR assemblies. Each canister may contain up to 24 undamaged PWR fuel assemblies.

The design of the Universal Storage System is based on certain reference fuel assemblies that maximize the source terms used for the shielding and criticality evaluation, and that maximize the weight used in the structural evaluation. These reference fuel assemblies are described in the chapters appropriate to the condition being evaluated. The principal characteristics and parameters of a reference fuel, such as fuel volume, initial enrichment, cool time and burnup, do not represent limiting or bounding values. Bounding values for a fuel class are established based primarily on how principal parameters are combined and on the loading conditions or restrictions established for a class of fuel based on its parameters.

The maximum decay heat load for the storage of all types of PWR fuel assemblies is 23.0 kW (0.958 kW/assembly), except in cases where preferential loading is employed.

The minimum cool time is based on the maximum decay heat load (23.0 kW) and the dose rate limits for the concrete and transfer casks and is presented in Section 5.5. PWR fuel must be loaded in accordance with Table 2.1.1-2.

Site specific fuel that does not meet the enrichment and burnup limits of this section and Table 2.1.1-1 is separately evaluated in Section 2.1.3 to establish loading limits.

Table 2.1.1-1 PWR Fuel Assembly Characteristics

Fuel Class[1, 2]	**14 × 14**	**14 × 14**	**15 × 15**	**15 × 15**	**15 × 15**	**16 × 16**	**17 × 17**
Fissile Isotopes	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Max Initial Enrichment (wt % ^{235}U)[3]	5.0	5.0	4.6	4.4	4.2	4.8	4.3
Max Initial Enrichment (wt % ^{235}U)[4]	5.0	5.0	5.0	5.0	5.0	5.0	5.0
Number of Fuel Rods	176	179	204	208	216	236	264
Number of Water Holes	5	17	21	17	9	5	25
Max Assembly Average Burnup (MWd/MTU)[8]	60,000	60,000	60,000	60,000	60,000	60,000	60,000
Min Cool Time (years)	5	5	5	5	5	5	5
Min Average Enrichment (wt % ^{235}U)	1.9	1.9	1.9	1.9	1.9	1.9	1.9
Cladding Material	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy
Nonfuel Hardware[5]	FM, T, BPR	FM, T, BPR	FM, T, BPR	FM, T, BPR	FM, T, BPR	FM, T, BPR	FM, T, BPR
Max Weight (lb) per Storage Location[6]	1,602	1,602	1,602	1,602	1,602	1,602	1,602
Max Decay Heat (Watts) per Storage Location[7]	958.3	958.3	958.3	958.3	958.3	958.3	958.3
Fuel Condition	Undamaged	Undamaged	Undamaged	Undamaged	Undamaged	Undamaged	Undamaged

General Notes:

1. Fuel, except Maine Yankee fuel, must be loaded in accordance with Table 2.1.1-2.
2. Maine Yankee fuel must be loaded in accordance with Tables 2.1.3.1-4 and 2.1.3.1-5, as appropriate.
3. Maximum initial enrichment without boron credit. Represents the maximum fuel rod enrichment for variably enriched assemblies. Assemblies meeting this limit may contain a flow mixer (FM) (thimble plug), an ICI thimble (T), a burnable poison rod insert (BPR), or a solid stainless steel rod insert.
4. Maximum initial enrichment with taking credit for a minimum soluble boron concentration of 1000 ppm in the spent fuel pool water. Represents the maximum fuel rod enrichment for variably enriched assemblies. Assemblies meeting this limit may contain a flow mixer (thimble plug).
5. Assemblies may not contain control element assemblies, except as permitted for site specific fuel.
6. Weight includes the weight of nonfuel-bearing components, including solid stainless steel rods inserted into guide tube positions.
7. Maximum decay heat may be higher for site-specific fuel configurations, which control fuel loading position.
8. Peak average rod burnup is limited to 62,500 MWd/MTU.

Table 2.1.1-2 Minimum Cooling Time Versus Burnup/Initial Enrichment for PWR Fuel

Minimum Initial Enrichment	**Assembly Average Burnup ≤30 GWD/MTU Minimum Cooling Time [years]**				**30< Assembly Average Burnup ≤35 GWD/MTU Minimum Cooling Time [years]**			
wt % ^{235}U (E)	**14×14**	**15×15**	**16×16**	**17×17**	**14×14**	**15×15**	**16×16**	**17×17**
1.9 ≤ E < 2.1	5	5	5	5	7	7	5	7
2.1 ≤ E < 2.3	5	5	5	5	7	6	5	6
2.3 ≤ E < 2.5	5	5	5	5	6	6	5	6
2.5 ≤ E < 2.7	5	5	5	5	6	6	5	6
2.7 ≤ E < 2.9	5	5	5	5	6	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5	5	5	5
3.7 ≤ E < 3.9	5	5	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5	5	5
E ≥ 4.9	5	5	5	5	5	5	5	5

Minimum Initial Enrichment	**35< Assembly Average Burnup ≤40 GWD/MTU Minimum Cooling Time [years]**				**40< Assembly Average Burnup ≤45 GWD/MTU Minimum Cooling Time [years]**			
wt % ^{235}U (E)	**14×14**	**15×15**	**16×16**	**17×17**	**14×14**	**15×15**	**16×16**	**17×17**
1.9 ≤ E < 2.1	10	10	7	10	15	15	11	15
2.1 ≤ E < 2.3	9	9	6	9	14	13	9	13
2.3 ≤ E < 2.5	8	8	6	8	12	12	8	12
2.5 ≤ E < 2.7	8	7	6	7	11	11	7	11
2.7 ≤ E < 2.9	7	7	6	7	10	10	7	10
2.9 ≤ E < 3.1	7	6	6	7	9	9	7	9
3.1 ≤ E < 3.3	6	6	6	6	9	8	7	8
3.3 ≤ E < 3.5	6	6	6	6	8	8	7	8
3.5 ≤ E < 3.7	6	6	6	6	7	8	7	7
3.7 ≤ E < 3.9	6	6	6	6	7	8	7	7
3.9 ≤ E < 4.1	6	6	6	6	7	7	7	7
4.1 ≤ E < 4.3	5	6	6	6	6	7	7	7
4.3 ≤ E < 4.5	5	6	6	6	6	7	7	7
4.5 ≤ E < 4.7	5	6	5	6	6	7	6	7
4.7 ≤ E < 4.9	5	6	5	6	6	7	6	7
E ≥ 4.9	5	6	5	6	6	7	6	7

Table 2.1.1-2 Minimum Cooling Time Versus Burnup/Initial Enrichment for PWR Fuel (continued)

Minimum Initial Enrichment	45< Assembly Average Burnup ≤50 GWd/MTU Minimum Cooling Time [years]				50< Assembly Average Burnup ≤55 GWd/MTU Minimum Cooling Time [years]			
wt % ^{235}U (E)	14×14	15×15	16×16	17×17	14×14	15×15	16×16	17×17
1.9 ≤ E < 2.1	21	21	18	21	27	27	25	27
2.1 ≤ E < 2.3	19	19	16	19	25	25	23	25
2.3 ≤ E < 2.5	17	17	14	17	23	24	21	24
2.5 ≤ E < 2.7	16	16	12	16	21	22	19	22
2.7 ≤ E < 2.9	14	14	11	14	20	20	17	20
2.9 ≤ E < 3.1	13	13	9	13	18	18	15	18
3.1 ≤ E < 3.3	12	12	9	12	17	17	13	17
3.3 ≤ E < 3.5	11	11	9	11	15	15	12	15
3.5 ≤ E < 3.7	10	10	8	10	14	14	11	14
3.7 ≤ E < 3.9	9	10	8	9	13	13	11	13
3.9 ≤ E < 4.1	9	10	8	9	12	13	11	12
4.1 ≤ E < 4.3	8	10	8	9	11	13	10	12
4.3 ≤ E < 4.5	8	9	8	9	10	13	10	12
4.5 ≤ E < 4.7	7	9	8	9	10	12	10	12
4.7 ≤ E < 4.9	7	9	8	9	9	12	10	12
E ≥ 4.9	7	9	8	9	9	12	10	11

Minimum Initial Enrichment	55< Assembly Average Burnup ≤60 GWd/MTU Minimum Cooling Time [years]							
wt % ^{235}U (E)	14×14	15×15	16×16	17×17				
1.9 ≤ E < 2.1	33	34	32	34				
2.1 ≤ E < 2.3	31	32	30	32				
2.3 ≤ E < 2.5	29	30	28	30				
2.5 ≤ E < 2.7	28	28	26	28				
2.7 ≤ E < 2.9	26	26	24	26				
2.9 ≤ E < 3.1	24	24	22	24				
3.1 ≤ E < 3.3	22	23	20	23				
3.3 ≤ E < 3.5	21	21	18	21				
3.5 ≤ E < 3.7	19	19	17	20				
3.7 ≤ E < 3.9	18	18	15	18				
3.9 ≤ E < 4.1	17	18	14	17				
4.1 ≤ E < 4.3	15	17	14	16				
4.3 ≤ E < 4.5	14	17	14	16				
4.5 ≤ E < 4.7	13	17	14	16				
4.7 ≤ E < 4.9	12	17	13	16				
E ≥ 4.9	12	16	13	15				

2.1.2 BWR Fuel Evaluation

The parameters of the BWR fuel assemblies that may be loaded in the transportable storage canister (canister) are shown in Table 2.1.2-1. Each canister may contain up to 56 undamaged BWR fuel assemblies.

The design of the Universal Storage System is based on certain reference fuel assemblies that maximize the source terms used for the shielding and criticality evaluation, and that maximize the weight used in the structural evaluation. These reference fuel assemblies are described in the chapters appropriate to the condition being evaluated. The principal characteristics and parameters of a reference fuel, such as fuel volume, initial enrichment, cool time and burnup, do not represent limiting or bounding values. Bounding values for a fuel class are established based primarily on how principal parameters are combined and on the loading conditions or restrictions established for a class of fuel based on its parameters.

The maximum canister decay heat load for the storage of all types of BWR fuel assemblies is 23.0 kW (0.411 kW/assembly).

The minimum cooling time determination is based on the maximum decay heat load (23.0 kW) and the dose rate limits for the concrete and transfer casks and is presented in Section 5.5. BWR fuel must be loaded in accordance with Table 2.1.2-2.

Table 2.1.2-1 BWR Fuel Assembly Characteristics

Fuel Class[1]	**7 × 7**	**7 × 7**	**8 × 8**	**8 × 8**	**8 × 8**	**9 × 9**	**9 × 9**
Fissile Isotopes	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Max Initial Enrichment (wt % ^{235}U)[1]	4.5	4.7	4.5	4.7	4.8	4.5	4.6
Number of Fuel Rods	48	49	60	62	63	74	79
Number of Water Holes	1[4]	0	1/4[5]	2	4	2/7[5]	2
Max Assembly Average Burnup (MWd/MTU)	45,000	45,000	45,000	45,000	45,000	45,000	45,000
Min Cool Time (years)	5	5	5	5	5	5	5
Min Average Enrichment (wt % ^{235}U)	1.9	1.9	1.9	1.9	1.9	1.9	1.9
Cladding Material	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy
Nonfuel Hardware[2]	Channel	Channel	Channel	Channel	Channel	Channel	Channel
Max Channel Thickness (mil)	120	120	120	120	120	120	120
Max Weight (lb) per Storage Location[3]	702	702	702	702	702	702	702
Max Decay Heat (Watts) per Storage Location	410.7	410.7	410.7	410.7	410.7	410.7	410.7
Fuel Condition	Undamaged	Undamaged	Undamaged	Undamaged	Undamaged	Undamaged	Undamaged

General Notes:
1. Fuel must be loaded in accordance with Table 2.1.2-2.
2. Each BWR fuel assembly may have a zirconium alloy channel or be unchanneled, but cannot have a stainless steel channel.
3. Weight includes the weight of the channel.
4. Solid fill or water rod.
5. Water rods may occupy more than one fuel lattice location.

Table 2.1.2-2 Minimum Cooling Time Versus Burnup/Initial Enrichment for BWR Fuel

Minimum Initial Enrichment	**Assembly Average Burnup ≤30 GWD/MTU Minimum Cooling Time [years]**			**30< Assembly Average Burnup ≤35 GWD/MTU Minimum Cooling Time [years]**		
wt % ^{235}U (E)	**7×7**	**8×8**	**9×9**	**7×7**	**8×8**	**9×9**
1.9 ≤ E < 2.1	5	5	5	8	7	7
2.1 ≤ E < 2.3	5	5	5	6	6	6
2.3 ≤ E < 2.5	5	5	5	6	5	6
2.5 ≤ E < 2.7	5	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5	5
3.7 ≤ E < 3.9	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5
E ≥ 4.9	5	5	5	5	5	5
Minimum Initial Enrichment	**35< Assembly Average Burnup ≤40 GWD/MTU Minimum Cooling Time [years]**			**40< Assembly Average Burnup ≤45 GWD/MTU Minimum Cooling Time [years]**		
wt % ^{235}U (E)	**7×7**	**8×8**	**9×9**	**7×7**	**8×8**	**9×9**
1.9 ≤ E < 2.1	16	14	15	26	24	25
2.1 ≤ E < 2.3	13	12	12	23	21	22
2.3 ≤ E < 2.5	11	9	10	20	18	19
2.5 ≤ E < 2.7	9	8	8	18	16	17
2.7 ≤ E < 2.9	8	7	7	15	13	14
2.9 ≤ E < 3.1	7	6	6	13	11	12
3.1 ≤ E < 3.3	6	6	6	11	10	10
3.3 ≤ E < 3.5	6	5	6	9	8	9
3.5 ≤ E < 3.7	6	5	6	8	7	7
3.7 ≤ E < 3.9	6	5	5	7	6	7
3.9 ≤ E < 4.1	5	5	5	7	6	7
4.1 ≤ E < 4.3	5	5	5	7	6	6
4.3 ≤ E < 4.5	5	5	5	6	6	6
4.5 ≤ E < 4.7	5	5	5	6	6	6
4.7 ≤ E < 4.9	5	5	5	6	6	6
E ≥ 4.9	5	5	5	6	6	6

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2.1.3 Site Specific Spent Fuel

This section describes site specific spent fuel, i.e., fuel assemblies that are configured differently or that have different fuel parameters, such as enrichment or burnup, than the fuel assemblies considered in the design basis. The site specific fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, testing programs intended to improve reactor operations or from the insertion of control components or other items within the fuel assembly.

Site specific spent fuel configurations are either shown to be bounded by the design basis fuel analysis or are separately evaluated. Unless specifically excepted, site specific spent fuel must also meet the conditions specified for the fuel considered in the design basis that is described in Sections 2.1.1 and 2.1.2.

2.1.3.1 Maine Yankee Site Specific Spent Fuel

The standard Maine Yankee site specific fuel is a Combustion Engineering PWR 14×14 assembly that is included in those fuel assemblies considered in the design basis fuel parameters described in Table 2.1.1-1. Maine Yankee spent fuel assemblies are categorized as undamaged or damaged as defined in Table 1-1. All damaged fuel and certain undamaged fuel configurations are placed in a Maine Yankee fuel can for storage in the Transportable Storage Canister. Each canister may contain up to 24 Maine Yankee assemblies, including up to 4 Maine Yankee Fuel Cans.

The estimated Maine Yankee site specific spent fuel inventory is shown in Section B2.0 of Appendix B. As noted, certain fuel configurations are preferentially loaded to take advantage of the design features of the Transportable Storage Canister and basket to allow the loading of fuel that does not specifically conform to the design basis spent fuel. Loading positions are shown in Figure 2.1.3.1-1.

The evaluated fuel includes those standard fuel assemblies modified by the installation or removal of fuel or nonfuel-bearing components. The three principal types of modifications are:

- The removal of fuel rods without replacement.
- The replacement of removed fuel rods or burnable poison rods with rods of another material, such as stainless steel, or with fuel rods of a different enrichment.
- The insertion of control elements, nonfuel items including start-up sources, or instrument or plug segments, in guide tube positions.

Site specific spent fuel also includes fuel assemblies that are uniquely designed to support reactor physics. These fuel assemblies include those that are variably enriched or that are variably enriched with annular axial blankets. Generally, these fuel assemblies (described in Sections 6.6.1.2.2 and 6.6.1.2.3) are bounded by the evaluation of the design basis fuel.

As described in Section 2.1.3.1.6, certain of the site-specific spent fuel configurations, including damaged and consolidated fuel loaded in Maine Yankee fuel cans, must be preferentially loaded in corner positions of the fuel basket. A fuel assembly with a burnup between 45,000 and 50,000 MWD/MTU must be preferentially loaded in a peripheral fuel position in the basket.

2.1.3.1.1 Damaged Fuel Lattices

There are two lattices for damaged fuel rods in the current Maine Yankee fuel inventory, designated CF1 and CA3, that are loaded in Maine Yankee fuel cans. CF1 is a lattice having roughly the same dimensions as a standard fuel assembly. It is a 9×9 array of tubes, some of which contain damaged fuel rods. CA3 is a previously used fuel assembly lattice that has had all of the rods removed, and into which, damaged fuel rods have been inserted. The CF1 and CA3 lattices are placed in a Maine Yankee fuel can for storage. No credit is taken for the lattice structures in the criticality, structural, or thermal analysis.

2.1.3.1.2 Maine Yankee Consolidated Fuel

The Maine Yankee fuel inventory includes two consolidated fuel lattices, which house undamaged fuel rods taken from three fuel assemblies. Each lattice is a 17×17 array formed using stainless steel grids and top and bottom stainless steel end fittings. Four solid stainless steel connector rods connect the end fittings. The top end fitting is designed so that the lattice can be handled by the standard fuel assembly lifting fixture (grapple). These lattices were not used in the reactor and the stainless steel hardware is not activated.

One of these lattices contains 283 fuel rods and 2 rod position vacancies. The other contains 172 fuel rods, with the 76 stainless steel dummy rods in the outer periphery of the lattice.

The consolidated fuel is placed in a Maine Yankee fuel can for storage. No credit is taken for the lattice structures in the criticality, structural, or thermal analysis.

2.1.3.1.3 Maine Yankee Spent Fuel with Inserted Integral Hardware or Nonfuel Items

Certain Maine Yankee fuel assemblies have either a Control Element Assembly or an Instrument Segment inserted in the fuel assembly. These components add to the gamma radiation source term of the standard fuel assembly.

A Maine Yankee Control Element Assembly (CEA) consists of five control rods mounted on a Type 304 stainless steel spider assembly. The five control rods are inserted in the fuel assembly guide tubes when the CEA is inserted in the fuel assembly. When fully inserted, the control element spider rests on the fuel assembly upper end fitting. The rods are fabricated from Inconel 625 or stainless steel and encapsulate B_4C as the primary neutron poison material. Fuel assemblies with a control element installed must be loaded into a Class 2 canister because of the additional height that the control element spider adds to the fuel assembly overall length. A CEA plug may also be inserted in a fuel rod. The CEA plug installs in the same position on the top of the fuel assembly, but the plug rods are only about 10 inches in length. These plugs are used to control water flow in the guide tubes. Fuel assemblies with CEA plugs installed must be loaded in a Class 2 canister.

Some standard fuel assemblies have an in-core instrument (ICI) thimble inserted in the center guide tube of the fuel assembly. The detector material and lead wire have been removed from the ICI assembly. The thimble top end and tube are primarily zirconium alloy. When installed, the instrument thimble does not add to the overall fuel assembly length. Consequently, fuel assemblies with ICI thimbles are loaded in the Class 1 canister.

The non-fuel inventory includes a segment of an ICI instrument thimble approximately 24 inches long. This segment is loaded in the corner guide tube position of an undamaged fuel assembly. The fuel assembly with the ICI segment installed must have a CEA flow plug installed to close the top of the corner guide tube, capturing the segment between the CEA flow plug and the bottom end plate of the fuel assembly. The ICI segment may be installed in a fuel assembly that also holds CEA finger tips in other corner guide tube positions. Because of the CEA fuel plug, the fuel assembly must be installed in a Class 2 canister.

The nonfuel inventory also includes five startup sources. One of the startup sources is unirradiated.

The startup sources include three Pu-Be sources and two Sb-Be sources that are installed in the center guide tubes of fuel assemblies that subsequently must be loaded in one of the four corner fuel positions of the basket. Each source is designed to fit in the center guide tube of an assembly, and only one startup source may be loaded in any fuel assembly. All five of these startup sources contain Sb-Be pellets, which are 50% Be by volume. One of the three Pu-Be sources is unirradiated and evaluation of this source is based on a "fresh" source material assumption.

2.1.3.1.4 Maine Yankee Spent Fuel with Unique Design

Certain Maine Yankee fuel assemblies were uniquely designed to accommodate reactor physics. These assemblies incorporate variable radial enrichment and axial blankets.

Two batches of fuel used at Maine Yankee contain variably enriched fuel rods. The maximum fuel rod enrichment of one batch is 4.21 wt % ^{235}U with the variably enriched rods enriched to 3.5 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.99 wt % ^{235}U. For the other batch, the maximum fuel rod enrichment is 4.0 wt % ^{235}U, with the variably enriched rods enriched to 3.4 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.92 wt % ^{235}U.

One batch of variably enriched fuel also incorporates axial end blankets with fuel pellets that have a center hole, referred to as annular fuel pellets. Annular fuel pellets are used in the top and bottom 5% of the active fuel length of each fuel rod in this batch.

2.1.3.1.5 Maine Yankee Fuel Can

Fuel assemblies classified as damaged that exceed the limits for loading as undamaged fuel and certain undamaged fuel configurations are loaded in a Maine Yankee fuel can, which is shown in Drawings 412-501 and 412-502. The fuel can may be loaded only in a corner position (positions numbered 3, 6, 19 and 22 in Figure 2.1.3.1-1) in the basket of a Class 1 canister. The fuel can analysis assumes the failure of 100% of the fuel rods held in the fuel can.

The fuel can is sized to accommodate a fuel assembly and must be loaded in a corner position of the fuel basket. As shown in the drawings, the can is provided in two configurations. Both cans are 162.8 inches in length and, in the top 4.5 inches, have an external square dimension of 8.8 inches. One configuration of the fuel can body has an internal square dimension of 8.52 inches and an external square dimension of 8.62 inches. The corresponding dimensions of the second configuration are 8.3 and 8.4 inches, respectively. The smaller cross-section allows the use of the fuel can in a basket in which the corner fuel loading positions of the bottom weldment are not enlarged. The fuel cans are closed on the bottom end by a 0.63-inch thick plate that is welded to the can shell. The plate has drilled holes in each corner to allow water to drain from the can. A screen covers the holes to preclude the release of gross particulates from the fuel can. A lid having an overall depth dimension of 2.38 inches closes the can. The lid is not secured to the can shell, but is held in place when the shield lid is installed in the canister. The lid also has four drilled and screened holes. The damaged fuel is inserted in the fuel can and the lid is installed. Slots in the can shell allow the loaded can to be lifted and installed in the basket. Alternately, the fuel can may be inserted in a basket corner position before the damaged fuel assembly is inserted in the fuel can. Since the fuel can lid is held in place by the canister shield lid, the fuel can may be used only in the Class 1 canister.

A Maine Yankee fuel can containing fuel debris with greater than 20 Curies of plutonium, requires double containment for transport conditions in accordance with 10 CFR 71.63 (b).

The Maine Yankee fuel can design and fabrication specification summary is provided in Table 2.1.3.1-2. The major physical design parameters of the Maine Yankee fuel can are provided in Table 2.1.3.1-3. The structural evaluation of the Maine Yankee site specific fuel configurations is provided in Section 3.6.1

2.1.3.1.6 Maine Yankee Site Specific Spent Fuel Preferential Loading

The estimated Maine Yankee site specific spent fuel inventory is shown in Table 2.1.3.1-1. (Note that the population of fuel in a given configuration may change based on future spent fuel inspection or survey.) As shown in this table, certain fuel configurations are preferentially loaded to take advantage of the design features of the Transportable Storage Canister and basket to allow the loading of fuel that does not specifically conform to the design basis spent fuel. The designated preferential loading positions are shown in Figure 2.1.3.1-1.

Fuel with missing fuel rods, fuel with fuel rods that have been replaced by rods of other material, consolidated fuel lattices and damaged fuel are preferentially loaded in corner positions of the basket, numbered 3, 6, 19 and 22 in Figure 2.1.3.1-1. The requirements for preferential loading schemes using the corner positions result primarily from shielding or criticality evaluations of the designated fuel configurations.

Preferential loading is also used for spent fuel having a burnup between 45,000 and 50,000 MWd/MTU. This fuel is assigned to peripheral basket locations, which are the outer 12 fuel loading positions shown in Figure 2.1.3.1-1. Locating the high burnup fuel in the peripheral basket locations reduces the maximum temperatures of these assemblies.

High burnup fuel (45,000 – 50,000 MWd/MTU) may be loaded as undamaged fuel provided that ISG-11, Revision 2 [25] temperature limits are met. The 752°F (400°C) ISG-11, Revision 2 fuel temperature limit is met as shown in Table 4.1-4.

Fuel assemblies with a control element inserted will be loaded in a Class 2 canister and basket for storage and transport due to the increased length of the assembly with the control element installed. However, these assemblies are not restricted as to loading position within the basket.

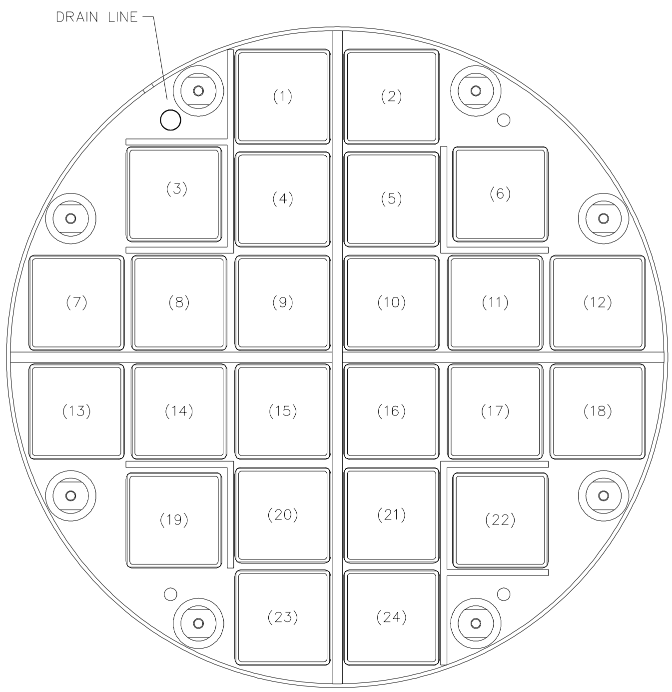
Fuel assemblies with a startup source in the center guide tube position must be loaded in one of the basket corner positions. A fuel assembly may not hold more than one startup source.

The loading position of fuel assemblies holding the CEA finger tips and/or the ICI segment in a fuel assembly corner guide tube position is not controlled; however, these fuel assemblies must have a CEA flow plug to ensure these items are captured within the guide tube(s).

2.1.3.1.7 Maine Yankee High Burnup Fuel

There are ninety (90) Maine Yankee fuel assemblies that have achieved a burnup between 45,000 and 50,000 MWD/MTU. As described in Section 2.1.3.1.6, these fuel assemblies are preferentially loaded in the 12 peripheral fuel loading positions in the basket. The high burnup assemblies are similar to the other Maine Yankee fuel planned to be placed in dry storage (i.e., those with burnup less than 45,000 MWD/MTU), but have design differences that support the high burnup objective.

Figure 2.1.3.1-1 Preferential Loading Diagram for Maine Yankee Site Specific Spent Fuel



Note: Locations numbered 3, 6, 19 and 22 are corner positions.
Locations numbered 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23 and 24 are periphery positions.
Locations numbered 4, 5, 8, 11, 14, 17, 20 and 21 are intermediate positions.
Locations numbered 9, 10, 15 and 16 are center positions.

Table 2.1.3.1-1 Maine Yankee Site Specific Fuel Population

Site-Specific Spent Fuel Configurations[1]	**Est. Number of Assemblies**[2]
Standard Fuel	1,434
Inserted Control Element Assembly (CEA)	168
Inserted In-Core Instrument (ICI) Thimble	138
Consolidated Fuel	2
Fuel Rod Replaced by Rod Enriched to 1.95 wt %	3
Fuel Rod Replaced by Stainless Steel Rod or Zirconium Alloy Rod	18
Fuel Rods Removed	10
Variable Enrichment	72
Variable Enrichment and Axial Blanket	68
Burnable Poison Rod Replaced by Hollow Zirconium Alloy Rod	80
Damaged Fuel in Maine Yankee Fuel Can	12
Burnup between 45,000 and 50,000 MWD/MTU	90
Maine Yankee Fuel Can	As Required
Inserted Startup Source	5
Inserted CEA Fingertips or ICI String Segment	1

1. The loading of the site-specific fuel is controlled by the requirement of Appendix B, Section B 2.0, of the CoC Number 1015 Technical Specifications.
2. The number of fuel assemblies in some categories may vary depending on future fuel inspections.

Table 2.1.3.1-2 Maine Yankee Fuel Can Design and Fabrication Specification Summary

Design

- The Maine Yankee Fuel Can shall be designed in accordance with ASME Code, Section III, Subsection NG except for: 1) the noted exceptions of Table B3-1 for fuel basket structures; and 2) the Maine Yankee Fuel Can may deform under accident conditions of storage.
- The Maine Yankee Fuel Can will have screened vents in the lid and base plate. Stainless steel meshed screens (250×250) shall cover all openings.
- The Maine Yankee Fuel Can shall limit the release of material from damaged fuel assemblies and fuel debris to the canister cavity.
- The Maine Yankee Fuel Can lifting structure and lifting tool shall be designed with a minimum factor of safety of 3.0 on material yield strength.

Materials

- All material shall be in accordance with the referenced drawings and meet the applicable ASME Code sections.
- All structural materials are ASME SA 240, Type 304 stainless steel.

Welding

- All welds shall be in accordance with the referenced drawings.
- The final surface of all welds (first unit) shall be liquid penetrant examined in accordance with ASME Code Section V, Article 6, with acceptance in accordance with ASME Code Section III, NG-5350. Subsequent units shall be visually examined in accordance with ASME Code Section V, Article 9, with acceptance in accordance with ASME Code Section III, NG-5360.

Fabrication

- All cutting, welding, and forming shall be in accordance with ASME Code Section III, NG-4000.

Acceptance Testing

- The Maine Yankee Fuel Can (first unit) and handling tool shall be load tested and visually inspected at the completion of fabrication.

Quality Assurance

- The Maine Yankee Fuel Can shall be constructed under a quality assurance program that meets 10 CFR 72 Subpart G. The quality assurance program must be accepted by NAC International and the licensee prior to initiation of the work.
- A Certificate of Conformance (or Compliance) shall be issued by the fabricator stating that the component meets the specifications and drawings.

Table 2.1.3.1-3 Major Physical Design Parameters of the Maine Yankee Fuel Can

Parameter	**Value**
Overall Length (in.)	162.8
Inside Cross Section (in.)	8.5 × 8.5 or 8.3 × 8.3
Outside Cross Section (in.) (1)	8.6 × 8.6 or 8.4 × 8.4
Can Wall Thickness	18 Gauge (0.048 in.)
Internal Cavity Length (in.)	160.0
Empty Weight (nominal) (lbs.)	130

Note (1) The top of the Maine Yankee Fuel Can is located above the top weldment of the fuel basket when it is installed. The outside top cross-section is 8.82 × 8.82 in. at the top 4.5 inches to allow for lid engagement and fuel can lifting.

Table 2.1.3.1-4 Loading Table for Maine Yankee Fuel without Nonfuel Material

Enrichment	Burnup ≤ 30 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU – Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	7	7	5
2.1 ≤ E < 2.3	6	6	5
2.3 ≤ E < 2.5	6	6	5
2.5 ≤ E < 2.7	5	6	5
2.7 ≤ E < 2.9	5	6	5
2.9 ≤ E < 3.1	5	6	5
3.1 ≤ E < 3.3	5	6	5
3.3 ≤ E < 3.5	5	6	5
3.5 ≤ E < 3.7	5	6	5
3.7 ≤ E ≤ 4.2	5	6	5

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern: interior basket locations; allowable heat decay = 0.867 kW per assembly
3. "Preferential" loading pattern: periphery basket locations; allowable heat decay = 1.05 kW per assembly

Table 2.1.3.1-4 Loading Table for Maine Yankee Fuel without Nonfuel Material (continued)

Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time [years] for[1]		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	11	11	6
2.1 ≤ E < 2.3	9	9	6
2.3 ≤ E < 2.5	8	8	6
2.5 ≤ E < 2.7	7	7	6
2.7 ≤ E < 2.9	7	7	6
2.9 ≤ E < 3.1	6	7	6
3.1 ≤ E < 3.3	6	7	5
3.3 ≤ E < 3.5	6	7	5
3.5 ≤ E < 3.7	6	7	5
3.7 ≤ E ≤ 4.2	6	7	5
Enrichment	45 < Burnup ≤ 50 GWD/MTU - Minimum Cool Time [years] for[1]		
	Standard[1]	Preferential (I)[2]	Preferential (P)[3]
1.9 ≤ E < 2.1	Not allowed	Not allowed	7
2.1 ≤ E < 2.3	Not allowed	Not allowed	7
2.3 ≤ E < 2.5	Not allowed	Not allowed	7
2.5 ≤ E < 2.7	Not allowed	Not allowed	7
2.7 ≤ E < 2.9	Not allowed	Not allowed	7
2.9 ≤ E < 3.1	Not allowed	Not allowed	7
3.1 ≤ E < 3.3	Not allowed	Not allowed	7
3.3 ≤ E < 3.5	Not allowed	Not allowed	6
3.5 ≤ E < 3.7	Not allowed	Not allowed	6
3.7 ≤ E ≤ 4.2	Not allowed	Not allowed	6

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern: interior basket locations; allowable heat decay = 0.867 kW per assembly
3. "Preferential" loading pattern: periphery basket locations; allowable heat decay = 1.05 kW per assembly

+Table 2.1.3.1-5 Loading Table for Maine Yankee Fuel Containing a CEA

	≤ 30 GWD/MTU Burnup - Minimum Cool Time in Years for				
Enrichment	**No CEA (Class 2)**	**5 Year CEA**	**10 Year CEA**	**15 Year CEA**	**20 Year CEA**
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time in Years for				
Enrichment	**No CEA (Class 2)**	**5 Year CEA**	**10 Year CEA**	**15 Year CEA**	**20 Year CEA**
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time in Years for				
Enrichment	**No CEA (Class 2)**	**5 Year CEA**	**10 Year CEA**	**15 Year CEA**	**20 Year CEA**
1.9 ≤ E < 2.1	7	7	7	7	7
2.1 ≤ E < 2.3	6	6	6	6	6
2.3 ≤ E < 2.5	6	6	6	6	6
2.5 ≤ E < 2.7	5	6	5	5	5
2.7 ≤ E < 2.9	5	6	5	5	5
2.9 ≤ E < 3.1	5	6	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time in Years for				
Enrichment	**No CEA (Class 2)**	**5 Year CEA**	**10 Year CEA**	**15 Year CEA**	**20 Year CEA**
1.9 ≤ E < 2.1	11	11	11	11	11
2.1 ≤ E < 2.3	9	9	9	9	9
2.3 ≤ E < 2.5	8	8	8	8	8
2.5 ≤ E < 2.7	7	7	7	7	7
2.7 ≤ E < 2.9	7	7	7	7	7
2.9 ≤ E < 3.1	6	6	6	6	6
3.1 ≤ E < 3.3	6	6	6	6	6
3.3 ≤ E < 3.5	6	6	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6
3.7 ≤ E ≤ 4.2	6	6	6	6	6

Note: The No CEA (Class 2) column is provided for comparison. Fuel assemblies without a CEA insert may not be loaded in a Class 2 canister.

2.2 Design Criteria for Environmental Conditions and Natural Phenomena

This section presents the design criteria for site environmental conditions and natural phenomena applied in the design basis analysis of the UMS® Universal Storage System. These criteria reflect conditions and phenomena to which the Storage System could be exposed during the period of storage. The system is designed to withstand the loads imposed by these environmental conditions and natural phenomena. Analyses to demonstrate that the design basis system meets the design criteria defined in this section are presented in the appropriate chapters of this Safety Analysis Report.

The use of the UMS® Universal Storage System at a specific site requires that the site either meet the design criteria of this section or be separately evaluated against the site specific conditions to ensure the acceptable performance of the UMS® Universal Storage System. Site specific evaluations are incorporated in designated sections of each chapter of this Safety Analysis Report. Site specific evaluations for environmental conditions and natural phenomena are presented in Section 11.2.15.

2.2.1 Tornado and Wind Loadings

The Vertical Concrete Casks are typically placed outdoors on an unsheltered reinforced concrete storage pad at an ISFSI site. This storage condition exposes the casks to tornado and wind loading.

2.2.1.1 Applicable Design Parameters

The design basis tornado and wind loading is defined based on Regulatory Guide 1.76 [9] Region 1 and NUREG-0800 [10]. The tornado and wind loading criteria are:

Tornado and Wind Condition	**Limit**
Rotational Wind Speed, mph	290
Translational Wind Speed, mph	70
Maximum Wind Speed, mph	360
Radius of Max. Wind Speed, ft.	150
Pressure Drop, psi	3.0
Rate of Pressure Drop, psi/sec	2.0

2.2.1.2 Determination of Forces on Structures

Tornado wind forces on the Vertical Concrete Cask are calculated by multiplying the dynamic wind pressure by the frontal area of the cask normal to the wind direction. Wind forces are applied to the cask in the wind direction. No streamlining is assumed. The evaluation of wind loading and tornado missile effects on the cask is presented in Section 11.2.11. The total design basis wind loading on the projected area of the cask is determined in Section 11.2.11. The cask is demonstrated to remain stable under design basis tornado wind loading in conjunction with impact from a high energy tornado missile.

2.2.1.3 Tornado Missiles

The design basis tornado missile impacts are defined in Paragraph 4, Subsection III, Section 3.5.1.4 of NUREG-0800 [10]. The design basis tornado is considered to generate three types of missiles that impact the cask at normal incidence:

1. Massive Missile - (Deformable w/high kinetic energy)	Weight = 4,000 lbs Frontal Area = 20 sq.-ft
2. Penetration Missile - (Rigid hardened steel)	Weight = 280 lbs Diameter = 8.0 in
3. Protective Barrier Missile - (Solid steel sphere)	Weight = 0.15 lbs Diameter = 1.0 in

Each missile is assumed to impact the cask at a velocity of 126 miles per hour, horizontal to the ground, which is 35 percent of the maximum wind speed of 360 miles per hour. For missile impacts in the vertical direction, the assumed missile velocity is (0.7)(126) = 88.2 miles per hour.

The detailed analysis of the Vertical Concrete Cask for missile impacts applies the laws of conservation of momentum and conservation of energy to determine the rigid body response of the concrete cask. Each missile impact is evaluated, and all missiles are assumed to impact in a manner that produces the maximum damage to the cask. The tornado and wind driven missile impact evaluation is presented in Section 11.2.11.

2.2.2 Water Level (Flood) Design

The Vertical Concrete Cask may be exposed to a flood during storage on an unsheltered concrete storage pad at an ISFSI site. The source and magnitude of the probable maximum flood depend on specific site characteristics.

2.2.2.1 Flood Elevations

The Vertical Concrete Cask is evaluated in Section 11.2.9 for a maximum flood water depth of 50 feet above the base of the cask. The flood water velocity is assumed to be 15 feet per second. Results of the evaluation show that under design basis flood conditions, the cask does not float, tip, or slide on the storage pad, and that the confinement function is maintained.

2.2.2.2 Phenomena Considered in Design Load Calculations

The occurrence of flooding at an ISFSI site is dependent upon the specific site location and the surrounding geographical features, natural and man-made. Some possible sources of a flood at an ISFSI site are: (1) overflow from a river or stream due to unusually heavy rain, snow-melt runoff, a dam or major water supply line break caused by a seismic event (earthquake); (2) high tides produced by a hurricane; and (3) a tsunami (tidal wave) caused by an underwater earthquake or volcanic eruption.

Flooding at an ISFSI site is highly improbable because of the extensive environmental impact studies that are performed during the selection of a site for a nuclear facility.

2.2.2.3 Flood Force Application

The evaluation of the Universal Storage System for a flood condition determines a maximum allowable flood water current velocity and a maximum allowable flood water depth. The criteria employed in the determination of the maximum allowable values are that a cask sliding or tip-over will not occur, and that the canister material yield strength is not exceeded. The evaluation of the effects of flood conditions on the system is presented in Section 11.2.9.

The force of the flood water current on the cask is calculated as a function of the current velocity by multiplying the dynamic water pressure by the frontal area of the cask that is normal to the current direction. The dynamic water pressure is calculated using Bernoulli's equation relating fluid velocity and pressure. The force of the flood water current is limited such that the overturning moment on the cask will be less than that required to tip the cask over.

2.2.2.4 Flood Protection

The inherent strength of the reinforced concrete cask provides a substantial margin of safety against any permanent deformation of the cask for a credible flood event at an ISFSI site. Therefore, no special flood protection measures for the cask are necessary. The evaluation presented in Section 11.2.9 shows that for the design basis flood, the allowable stresses in the canister are not exceeded.

2.2.3 Seismic Design

An ISFSI site may be subject to seismic events (earthquakes) during its lifetime. The seismic response spectra experienced by the cask depends upon the geographical location of the specific site and the distance from the epicenter of the earthquake. The only significant effect of a seismic event on the vertical concrete cask is a possible tip-over or a collision of two casks. However, tip-over does not occur during the design basis earthquake. For sites not implementing a friction limitation, it is possible for two casks to collide due to sliding. Seismic response of the cask is presented in Section 11.2.8.

2.2.3.1 Input Criteria

The transportable storage canister and vertical concrete cask are designed and analyzed by applying a seismic acceleration or a maximum resultant horizontal planar velocity of the ISFSI pad.

2.2.3.2 Seismic - System Analyses

The analysis for the earthquake condition applied to nuclear facilities is provided in Section 11.2.8.2. Evaluations of the consequences of a hypothetical tip-over event or a collision of two vertical concrete casks are provided in Section 11.2.12.

2.2.4 Snow and Ice Loadings

The criterion for determining design snow loads is based on ANSI/ASCE 7-93 [12], Section 7.0. Flat roof snow loads apply and are calculated from the following formula:

$$p_f = 0.7C_eC_tIp_g$$

where:

p_f = flat roof snow load (psf)
C_e = Exposure factor = 1.0
C_t = Thermal factor = 1.2
I = Importance factor = 1.2
p_g = ground snow load, (psf) = 100

The numerical values of C_e, C_t, I and p_g are obtained from Tables 18, 19, 20 and Figure 7, respectively, of ANSI/ASCE 7-93.

The exposure factor, C_e, accounts for wind effects. The site of the Universal Storage System is assumed to be a location typical for siting Category C, which is defined to be "locations in which snow removal by wind cannot be relied on to reduce roof loads because of terrain, higher structures, or several trees nearby."

The thermal factor, C_t, accounts for the importance of buildings and structures in relation to public health and safety. The Universal Storage System is conservatively classified as Category III.

Ground snow loads for the contiguous United States are given in Figures 5, 6 and 7 of ANSI/ ASCE 7-93. A worst case value of 100 lbs per square ft is assumed.

Based on the above, the design criterion for snow and ice loads is:

Flat Roof Snow Load, p_f = (0.7) (1.0) (1.2) (1.2) (100) = 100.8 psf

This load is bounded by the weight of the loaded transfer cask on the top of the concrete cask shell and by the tornado missile loading on the concrete cask lid. The snow load is considered in the load combinations described in Section 3.4.4.2.2.

2.2.5 Combined Load Criteria

Each normal, off-normal and accident condition has a combination of load cases that defines the total combined loading for that condition. The individual load cases considered include thermal, seismic, external and internal pressure, missile impacts, drops, snow and ice loads, and/or flood water forces.

The load conditions to be evaluated for storage casks are identified in 10 CFR 72[11] and ANSI/ANS-57.9 [13].

2.2.5.1 Load Combinations and Design Strength - Vertical Concrete Cask

The load combinations specified in ANSI/ANS 57.9 for concrete structures are applied to the concrete casks as shown in Table 2.2-1. The live loads are considered to vary from 0 percent to 100 percent to ensure that the worst-case condition is evaluated. In each case, use of 100 percent of the live load produces the maximum load condition. The steel liner of the concrete cask is a stay-in-place form and it provides radiation shielding. The concrete cask is designed to the requirements of ACI 349 [4].

In calculating the design strength of concrete in the Vertical Concrete Cask body, nominal strength values are multiplied by a strength reduction factor in accordance with Section 9.3 of ACI 349.

2.2.5.2 Load Combinations and Design Strength - Canister and Basket

The canister is designed in accordance with the 1995 edition of the ASME Code, Section III, Subsection NB [1] for Class 1 components. The basket structure is designed in accordance with

ASME Code, Section III, Subsection NG [2]. Structural buckling of the basket is evaluated in accordance with NUREG/CR-6322 [3].

The load combinations for all normal, off-normal, and accident conditions and corresponding service levels are shown in Table 2.2-2. The table, therefore, defines the canister design and service loadings. Levels A and D service limits are used for normal and accident conditions, respectively. Levels B and C service limits are used for off-normal conditions. The analysis methods of the ASME Code are employed. Stress intensities caused by pressure, temperature, and mechanical loads are combined before comparing them to ASME code allowables. The Code allowables are listed in Table 2.2-3.

2.2.5.3 Design Strength - Transfer Cask

The transfer cask is a special lifting device. It is designed and fabricated to the requirements of ANSI N14.6 [6] and NUREG 0612 [7] for the lifting trunnions and supports, and ANSI/ANS-57.9 [13] for the remainder of the structure. The criteria are:

1. The combined shear stress or maximum tensile stress during the lift (with 10 percent dynamic load factor) shall be ≤ $S_y/6$ and $S_u/10$ for a nonredundant load path, or shall be ≤ $S_y/3$ and $S_u/5$ for redundant load paths.

2. The ferritic steel material used for the load bearing members of the transfer cask shall satisfy the material toughness requirements of ANSI N14.6, paragraph 4.2.6.

Load testing of the transfer cask is described in Section 2.3.3.1.

2.2.6 Environmental Temperatures

A normal, long-term annual average design ambient temperature of 76°F is selected to bound most annual average temperatures seen by a cask over its lifetime. This temperature is based on the maximum average annual temperature in the 48 contiguous United States, specifically, Miami, FL., at 75.6°F [14], and is, therefore, used so as to bound existing and potential ISFSI sites.

The 76°F normal temperature is used as the base for thermal evaluations. The evaluation of this environmental condition is discussed along with the thermal analysis models in Chapter 4.0. The thermal stress evaluation for the normal operating conditions is provided in Section 3.4.4. Normal temperature fluctuations are bounded by the severe ambient temperature cases that are evaluated as off-normal and accident conditions.

Off-normal, severe environmental conditions are defined as -40°F with no solar loads and 106°F with solar loads. An extreme environmental condition of 133°F with maximum solar loads is evaluated as an accident case (11.2.7) to show compliance with the maximum heat load case required by ANSI/ANS-57.9. Thermal performance is also evaluated assuming half-blockage of the concrete cask air inlets and the complete blockage of the air inlets and outlets. Thermal analyses for these cases are presented in Sections 11.1.2 and 11.2.13. The evaluation based on ambient temperature conditions is presented in Section 4.4.

The design basis temperatures used in the Universal Storage System analysis are shown below. Solar insolance is as specified in 10 CFR 71.71 [15] and Regulatory Guide 7.8 [16].

Condition	Ambient Temperature	Solar Insolance
Normal	76°F	yes
Off-Normal - Severe Heat	106°F	yes
Off-Normal - Severe Cold	-40°F	no
Accident - Extreme Heat	133°F	yes

Table 2.2-1 Load Combinations for the Vertical Concrete Cask

Load Combination	Condition	Dead	Live	Wind	Thermal	Seismic	Tornado/ Missile	Drop/ Impact	Flood
1	Normal	1.4D	1.7L						
2	Normal	1.05D	1.275L		$1.275T_o$				
3	Normal	1.05D	1.275L	1.275W	$1.275T_o$				
4	Off-Normal and Accident	D	L		T_a				
5	Accident	D	L		T_o	E_{ss}			
6	Accident	D	L		T_o			A	
7	Accident	D	L		T_o				F
8	Accident	D	L		T_o		W_t		

Load Combinations are from ANSI/ANS-57.9 [13] and ACI 349 [4].

D	=	Dead Load	T_a	=	Off- Normal or Accident Temperature
L	=	Live Load	E_{ss}	=	Design Basis Earthquake
W	=	Wind	W_t	=	Tornado/Tornado Missile
T_o	=	Normal Temperature	A	=	Drop/Impact
F	=	Flood			

Table 2.2-2 Load Combinations for the Transportable Storage Canister

LOAD		**NORMAL**			**OFF-NORMAL**					**ACCIDENT**					
ASME Service Level		A			B		C			D					
Load Combinations		1	2	3	1	2	3	4	5	1	2	3	4	5	6
Dead Weight	Canister with fuel	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Thermal	In Storage Cask 76° F Ambient	X		X				X		X	X	X	X	X	
	In Transfer Cask 76° F Ambient		X		X		X								X
	In Storage Cask -40°F or 106°F Ambient					X			X						
Internal Pressure	Normal	X	X	X			X	X	X	X	X	X	X		
	Off-Normal				X	X									
	Accident													X	X
Handling Load	Normal		X	X	X										
	Off-Normal						X	X	X						
Drop/Impact	Accident									X					
Seismic	Accident										X				
Flood	Accident											X			
Tornado	Accident												X		

Table 2.2-3 Structural Design Criteria for Components Used in the Transportable Storage Canister

	Component	Criteria
1.	Normal Operations: Service Level A Canister: ASME Section III, Subsection NB [1] Basket: ASME Section III, Subsection NG [2] Lifting Devices: ANSI N14.6 [6] and NUREG 0612 [7]	$P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$ $P_L + P_b + Q \leq 3S_m$ Redundant load path: combined shear or max. tensile stress $\leq S_u/5$ and $S_y/3$
2.	Off-Normal Operations: Service Level B Canister: ASME Section III, Subsection NB	$P_m < 1.1\ S_m$ and $P_L + P_b < 1.65\ S_m$
3.	Off-Normal Operations: Service Level C Canister: ASME Section III, Subsection NB Basket: ASME Section III, Subsection NG Note: Subsection NB allowables for Service Level C are conservatively applied to the basket.	Subsection NB Allowables: $P_m < 1.2\ S_m$ or S_y (whichever is greater) and $P_L + P_b < 1.8\ S_m$ or $1.5\ S_y$ (whichever is less)
4.	Accident Conditions, Service Level D Canister: ASME Section III, Subsection NB Basket: ASME Section III, Subsection NG	$P_m \leq 2.4\ S_m$ or $0.7\ S_u$ (whichever is less) and $P_L + P_b \leq 3.6\ S_m$ or $1.05\ S_u$ (whichever is less)
5.	Basket Structural Buckling	NUREG/CR-6322 [3]

Symbols:

S_m = material design stress intensity
S_u = material ultimate strength
S_y = material yield strength
P_L = primary local membrane stress
P_m = primary general membrane stress
P_b = primary bending stress

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2.3 Safety Protection Systems

The Universal Storage System relies upon passive systems to ensure the protection of public health and safety, except in the case of fire or explosion. As discussed in Section 2.3.6, fire and explosion events are effectively precluded by site administrative controls that prevent the introduction of flammable and explosive materials. The use of passive systems provides protection from mechanical or equipment failure.

2.3.1 General

The Universal Storage System is designed for safe, long-term storage of spent nuclear fuel. The system will withstand all of the evaluated normal, off-normal, and postulated accident conditions without release of radioactive material or excessive radiation exposure to workers or the general public. The major design considerations that are incorporated in the Universal Storage System to assure safe, long-term fuel storage are:

1. Continued containment in postulated accidents.
2. Thick concrete and steel biological shield.
3. Passive systems that ensure reliability.
4. Inert helium atmosphere to provide corrosion protection for fuel cladding and enhanced heat transfer for the stored fuel.

Each component of the Universal Storage System is classified with respect to its function and corresponding effect on public safety. In accordance with Regulatory Guide 7.10 [17], each system component is assigned a quality category classification and then "important to safety" items are further categorized based on importance to safety into Category A, B, C, or NQ as shown in Table 2.3-1. The quality category classification is based on review of each component's function and the assessment of the consequences of its failure following the guidelines of NUREG/CR-6407 [18]. The quality category classification categories are defined as follows:

- Category A - Components critical to safe operations whose failure or malfunction could directly result in conditions adverse to safe operations, integrity of spent fuel, or public health and safety.

- Category B - Components with major impact on safe operations whose failure or malfunction could indirectly result in conditions adverse to safe operations, integrity of spent fuel, or public health and safety.

- Category C - Components whose failure would not significantly reduce the packaging effectiveness and would not likely result in conditions adverse to safe operations, integrity of spent fuel, or public health and safety.

- Category NQ - Non quality components have no impact on safety.

As discussed in the following sections, the Universal Storage System design incorporates features addressing the above design considerations to assure safe operation during loading, handling, and storage of spent nuclear fuel.

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 Confinement Barriers and Systems

The radioactivity that the Universal Storage System must confine originates from the spent fuel assemblies to be stored and residual contamination that may remain inside the canister as a result of contact with water in the fuel pool where the canister loading is conducted. The system is designed to confine this radioactive material.

The Transportable Storage Canister is closed by welding. The shield lid weld is pressure tested. All of the field-installed shield lid welds are liquid penetrant examined following the root and final weld passes. The shield lid welds are leak tested. The installation of the canister structural lid, which provides a redundant closure over the shield lid and port covers, is accomplished by multi-pass welding that is either: 1) progressively liquid penetrant examined; or 2) ultrasonically examined in conjunction with a liquid penetrant examination of the final weld surface. The longitudinal and girth welds of the canister shell are full penetration welds that are radiographically examined during fabrication. The weld that joins the bottom plate to the canister shell is ultrasonically and liquid penetrant examined during fabrication.

The canister welds are an impenetrable boundary to the release of fission gas products during the period of storage. There are no evaluated normal, off-normal, or accident conditions that result in the breach of the canister and the subsequent release of fission products. The canister is designed to withstand a postulated drop accident in the UMS® Universal Transport Cask without precluding the subsequent removal of the fuel (i.e., the fuel tubes do not deform such that they bind the fuel assemblies).

Personnel radiation exposure during handling and closure of the canister is minimized by the following steps:

1. Placing the shield lid on the canister while the transfer cask and canister are under water in the fuel pool.
2. Decontaminating the exterior of the transfer cask prior to draining the canister or performing canister closure operations with the transfer cask partially submerged to preserve the shielding benefit of the water.
3. Using temporary shielding.
4. Using a retaining ring on the transfer cask to ensure that the canister is not raised out of the shield provided by the transfer cask.
5. Placing a shielding ring over the annular gap between the transfer cask and the canister.

2.3.2.2 Cask Cooling

The loaded Vertical Concrete Cask is passively cooled. Cool (ambient) air enters at the bottom of the concrete cask through four inlet vents. Heated air exits through the four outlets at the top of the cask. Radiant heat transfer also occurs from the canister shell to the concrete cask liner. Consequently, the liner also heats the convective air flow. Conduction does not play a substantial role in heat removal from the canister surface. This natural circulation of air inside the Vertical Concrete Cask, in conjunction with radiation from the canister surface, maintains the fuel cladding temperature and all of the concrete cask component temperatures below their design limits. The cask cooling system is described in detail in Sections 4.1 and 4.4.

2.3.3 Protection by Equipment and Instrumentation Selection

The Universal Storage System is a passive storage system that does not rely on equipment or instruments to preserve public health or safety and to meet its safety functions in long-term storage. The system employs support equipment and instrumentation to facilitate operations. These items, and the actions taken to assure performance, are described below.

2.3.3.1 Equipment

The equipment that is important-to-safety employed in the use and operation of the Universal Storage System is the transfer cask and the lifting yoke used to lift the transfer cask. The transfer cask is provided in the standard and advanced configurations. The lifting yoke is designed to meet the requirements of ANSI N14.6 and NUREG-0612 and is designed as a special lifting device for critical loads. Both lifting yokes are proof load tested to 300% of design load when fabricated. The lifting yokes have no welds in the lifting load path. Following the load test, the bolted connections are disassembled, and the components are inspected for deformation. Permanent deformation of components is not acceptable. The lifting yoke is inspected for visible defects prior to each use and is inspected annually.

The transfer cask is used to move the empty and loaded Transportable Storage Canister in all of the operations that precede the installation of the loaded canister in the Vertical Concrete Cask. The transfer cask is evaluated as a lifting component. The principal design criteria of the transfer cask are presented in Section 2.2.5.3, above. The transfer cask design meets the requirements of ANSI N14.6 and NUREG-0612. The standard and advanced transfer casks both have two pairs of lifting trunnions. Each pair is designed as a special lifting device for critical loads, but both pairs may be used together in order to provide a redundant load path. Each pair of transfer cask trunnions is load tested to 300% of the maximum calculated service load. The service load includes the transfer cask weight, the loaded canister, and water in the canister. Following the load test, the trunnion welds and other welds in the load path are inspected for indications of cracking or deformation. The principal load bearing welds and the transfer cask lifting trunnions are evaluated in Section 3.4.3.3.

The transfer cask bottom shield doors support the canister from the bottom during handling of the canister. The shield doors are also load tested to 300% of the maximum calculated service load. The service load includes the weight of the loaded canister and water in the canister. Following the load test, the load bearing surface areas of the doors, rails, and attachment welds are examined for evidence of cracking or deformation.

The transfer cask welds are subjected to a liquid penetrant examination, performed in accordance with the ASME Code, Section V, Article 6. Acceptance criteria is in accordance with the ASME Code, Paragraph NF-5350.

Any evidence of permanent deformation, cracking, galling of bearing surfaces, or unacceptable liquid penetrant examination results is cause for rejection. Any identified defects must be repaired and the load test repeated prior to final acceptance.

2.3.3.2 Protection by Instrumentation

No instrumentation is required for the safe storage operations of the UMS®. A remote temperature-monitoring system may be used to measure the outlet air temperature of the concrete casks in long-term storage. The outlet and ISFSI ambient air temperatures can be monitored daily as a check of the continuing thermal performance of the concrete cask. Alternately, a daily visual inspection for blockage and integrity of the air inlet and air outlet screens of all concrete casks may be performed. Following any natural phenomena event, such as an earthquake or tornado, the concrete casks shall be inspected for damage and air inlet and air outlet blockage.

2.3.4 Nuclear Criticality Safety

The Universal Storage System design includes features to ensure that nuclear criticality safety is maintained (i. e., the cask remains subcritical) under normal, off-normal, and accident conditions. The design of the canister and fuel basket is such that, under all conditions, the highest neutron multiplication factor (k_{eff}) is less than 0.95. The criticality evaluation for the design basis fuel is presented in Section 6.4.

2.3.4.1 Control Methods for Prevention of Criticality

Criticality control in the PWR basket is achieved using a neutron flux trap configuration. Individual fuel assemblies are surrounded by four neutron absorber sheets, one on each side of the assembly, that provide absorption of moderated neutrons. The assemblies are separated by a gap that is filled with water during hypothetical accident conditions when the canister is flooded. Fast neutrons escaping one fuel assembly are moderated in the gap between the assemblies and absorbed by the neutron absorber material surrounding the assemblies. The minimum loading of the neutron absorber sheets is 0.025 g $^{10}B/cm^2$. The sheets are mechanically supported by the fuel tube structure to ensure that the neutron absorber sheets remain in place during the design basis normal, off-normal, and accident events.

Individual fuel assemblies in the BWR basket are separated from adjacent assemblies by a single neutron absorber sheet between fuel assemblies. Of the total 56 fuel tubes, 42 tubes contain neutron absorber sheets on two sides of the tubes, 11 tubes contain neutron absorber sheets on one side, and the remaining 3 tubes contain no neutron absorber sheets. The arrangement of the fuel tubes ensures that there is at least one neutron absorber sheet between adjacent fuel assemblies. Although this configuration of water gaps and neutron absorber sheets does not form a classic neutron flux trap, the design ensures that there is sufficient absorption of moderated neutrons by the neutron absorber to maintain criticality control in the basket (k_{eff} < 0.95). The minimum loading of the neutron absorber sheets in the BWR fuel tubes is 0.011 g $^{10}B/cm^2$. The neutron absorber sheets are mechanically supported by the fuel tube structure to ensure that the sheets remain in place during the design basis normal, off-normal, and accident events.

The efficiency of the neutron absorber sheets in preserving nuclear criticality safety is demonstrated by the criticality results presented in Section 6.4.3.

The principal criticality design criterion is that k_{eff} remain below 0.95 under all conditions. Assumptions made in the analyses used to demonstrate conformance to this criterion include:

1. Fuel assembly with maximum ^{235}U loading (95% theoretical density);
2. 75 percent of the nominal ^{10}B loading in the neutron absorber sheet;
3. Infinite array of casks in the X-Y (horizontal) plane;
4. Infinite fuel length with no inclusion of end leakage effects;
5. No credit taken for structural material present in the assembly; and,
6. No credit taken for fuel burnup or for the buildup of fission product neutron poisons.

Use of administrative controls of fuel burnup levels, neutron absorption properties of the burned fuel, and the presence of steel shell of the canister provide further criticality controls in the Universal Storage System.

2.3.4.2 Error Contingency Criteria

The calculated values of k_{eff} include error contingencies and calculation and modeling biases. The standards and regulations of criticality safety require that k_{eff}, including uncertainties, k_s, be less than 0.95. The bias and 95/95 uncertainty are applied to the calculation of k_s by using:

$$k_s = k_{nom} + 0.0052 + [(0.0087)^2 + (2\sigma_{MC})^2]^{1/2} \leq 0.95$$

where:

k_{nom} = the nominal k_{eff} for the cask, and
σ_{MC} = the Monte Carlo uncertainty.

The calculation of error contingencies and uncertainties is presented in Section 6.4.

2.3.4.3 Verification Analyses

The CSAS25 criticality analysis sequence is benchmarked through a series of calculations based on 63 critical experiments. These experiments span a range of fuel enrichments, fuel rod pitches, poison sheet characteristics, shielding materials, and geometries that are typical of light water reactor fuel in a cask. To achieve accurate results, three-dimensional models, as close to the actual experiment as possible, are used to evaluate the experiments. The results of the benchmark calculations are provided in Section 6.5.

2.3.5 Radiological Protection

The Universal Storage System, in keeping with the As Low As Is Reasonably Achievable (ALARA) philosophy, is designed to minimize, to the extent practicable, operator radiological exposure.

2.3.5.1 Access Control

Access to a Universal Storage System ISFSI site is controlled by a peripheral fence to meet the requirements of 10 CFR 72 and 10 CFR 20 [19]. Access to the storage area, and its designation as to the level of radiation protection required, are established by site procedure. The storage area is surrounded by a fence, having lockable truck and personnel access gates. The fence has intrusion-detection features as determined by the site procedure.

2.3.5.2 Shielding

The Universal Storage System is designed to limit the dose rates as follows:

- external surface dose (gamma and neutron) to less than 50 mrem/hr (average) on the Vertical Concrete Cask sides.
- external surface dose to less than 50 mrem/hr (average) on the Vertical Concrete Cask top.
- a maximum of 100 mrem/hr (average) at the Vertical Concrete Cask air inlets and outlets.
- the supplemental shielding at the top of the canister shield lid reduces personnel exposure during canister closure operations.

Sections 72.104 and 72.106 of 10 CFR 72 set whole body dose limits for an individual located beyond the controlled area at 25 millirems per year (whole body) during normal operations and 5 rems (5,000 millirems) from any design basis accident. The analyses showing the actual Universal Storage System doses, and dose rates, are included in Chapters 5.0, 10.0 and 11.0.

2.3.5.3 Ventilation Off-Gas

The Universal Storage System is passively cooled by radiation and natural convection heat transfer at the outer surface of the concrete cask and in the canister-concrete cask annulus. The bottom of the cask is conservatively assumed to be an adiabatic surface. In the canister-concrete cask annulus, air enters the air inlets, flows up between the canister and concrete cask liner in the annulus, and exits the air outlets. The air flow in the annulus is due to the buoyancy effect created by the heating of the air by the canister and concrete cask liner walls. The details of the passive ventilation system design are provided in Chapter 4.0.

The surface of the canister is exposed to cooling air when the canister is placed in the concrete cask. If the surface is contaminated, the possibility exists that contamination could be carried aloft by the cooling air stream. Therefore, during fuel loading, the spent fuel pool water is excluded from the canister exterior by filling the transfer cask/canister annular gap with clean water as the transfer cask is being lowered into the fuel pool. Clean water is injected into the gap during the entire time the transfer cask is submerged. These steps minimize the potential for the intrusion of contaminated water into the canister annular gap.

Once the transfer cask is removed from the pool, a smear survey is taken of the exterior surface of the canister near the top. While no contamination is expected to be found, it is possible that the surface could be contaminated. The allowable upper limit for surface contamination of the canister and transfer cask is provided in LCO 3.2.1 in Appendix A. As described in LCO 3.2.1, if this limit is exceeded, steps to decontaminate the canister surface must be taken and continued until the contamination is less than the allowable limit.

To facilitate decontamination, the canister is fabricated so that its exterior surface is smooth. There are no corners or pockets that could trap and hold contamination.

There are no radioactive releases during normal operations. Also, there are no credible accidents that cause significant releases of radioactivity from the Universal Storage System and, hence, there are no off-gas system requirements for the system during normal storage operation. The only time an off-gas system is required is during the canister drying phase. During this operation, the reactor off-gas system or a HEPA filter system is used.

2.3.5.4 Radiological Alarm Systems

No radiological alarms are required on the Universal Storage System. Justification for this is provided in Chapter 5.0 (Shielding), 10.0 (Radiological Protection), and 11.0 (Accident Analysis).

Typically, total radiation exposure due to the ISFSI installation is determined by the use of Thermo-Luminescent Detectors (TLDs) mounted at convenient locations on the ISFSI fence. The TLDs are read quarterly to provide a record of boundary dose.

2.3.6 Fire and Explosion Protection

Fire and explosion protection of the Universal Storage System is provided primarily by administrative controls applied at the site, which preclude the introduction of any explosive and any excessive flammable materials into the ISFSI area.

2.3.6.1 Fire Protection

A major ISFSI fire is not considered credible, since there is very little material near the casks that could contribute to a fire. The concrete cask is largely impervious to incidental thermal events. Administrative controls are put in place to ensure that the presence of combustibles is minimized. A hypothetical fire event is evaluated as an accident condition in Section 11.2.6. The fire event evaluated is a 1475°F fire of 8 minutes duration. This condition is considered to be highly conservative.

2.3.6.2 Explosion Protection

The Universal Storage System is analyzed to ensure its proper function under an over-pressure condition. As described in Section 11.2.5, in the evaluated 22 psig over-pressure condition, stresses in the canister remain below allowable limits and there is no loss of confinement. These results are conservative, as the canister is protected from direct over-pressure conditions by the concrete cask.

For the same reasons as for the fire condition, a severe explosion on an ISFSI site is not considered credible. The evaluated over-pressure is considered to bound any explosive over-pressure resulting from an industrial explosion at the boundary of the owner-controlled area.

2.3.7 Ancillary Structures

The loading, transfer and transport of the UMS® System requires the use of auxiliary equipment as described in Section 2.3.3 and may require the use of an ancillary structure, referred to as a "Canister Handling Facility." The Canister Handling Facility is an especially designed and engineered structure separate from the 10 CFR 50 facilities at the site. The Canister Handling Facility, if required, would provide a housing for a lifting crane, service air and water, a radiation control area, auxiliary equipment storage and support services and work areas related to canister handling and transfer. Transfer operations could include temporary holding of a loaded canister in the transfer cask to allow repair of a concrete cask, transfer of a canister from one concrete cask to another, or transfer from a concrete cask to a transport cask.

The design of the Canister Handling Facility would meet the requirements of the Universal Storage System described in Approved Contents and Design Features presented in Appendix B of the CoC Number 1015 Technical Specifications, in addition to those requirements established by the site.

The design, analysis, fabrication, operation and maintenance of the Canister Handling Facility would be performed in accordance with the quality assurance program requirements of the site general licensee, or the site-specific licensee of the ISFSI. The Canister Handling Facility would be classified as Important to Safety or Not Important to Safety in accordance with the guidelines of NUREG-6407.

Table 2.3-1 Quality Category Classification of Universal Storage System Components

Drawing No.	Description	Item No.	Component	Function	Quality Category
790-559	Assembly, Transfer Adapter	20	Set Screw	Operations	NQ
		19	Cylinder Stop	Operations	NQ
		18	Guide Segment	Operations	C
		17	Cylinder Bolt	Operations	C
		15	Connector Body Bolt	Operations	C
		14	Wear Pad Bolt	Operations	NQ
		13	Wear Pad	Operations	NQ
		12	Connector Body	Operations	C
		10	Cylinder Nut	Operations	C
		8	Door Cylinder	Operations	C
		7	Lift Lug	Operations	C
		6	Support	Operations	C
		5	Side Shield	Operations	C
		3, 4	Door Rail	Operations	C
		2	Locating Ring	Operations	C
		1	Base Plate	Operations	C
790-560	Assembly, Transfer Cask	52	Lift Plate B	Operations	NQ
		51	Lift Plate A	Operations	NQ
		50	Door Plug	Operations	NQ
		49	Wear Strip	Operations	NQ
		47	Door Lock Bolt	Operations	C
		46	Dowel Pin	Operations	NQ
		45	Fill/Drain Line Pipe	Operations	C
		44	Fill/Drain Line Plate	Operations	C
		43	Shielding Ring	Shielding	B
		42	Transfer Adapter SHCS	Shielding	B
		41	Transfer Cask Extension	Shielding	B
		39	Connector	Operations	C
		38	Retaining Ring Bolt	Operations	B

Table 2.3-1 Quality Category Classification of Universal Storage System Components (continued)

Drawing No.	Description	Item No.	Component	Function	Quality Category
790-560 (Continued)	Assembly, Transfer Cask	37	Scuff Plate	Operations	NQ
		36	Gamma Shield Brick	Shielding	B
		33-34	Neutron Shield Cover Plate	Operations	C
		28-32	Neutron Shield Boundary	Structural	C
		26-27	Bottom Plate	Structural	B
		25	Stainless Steel Sheet	Operations	NQ
		24	Paint	Operations	NQ
		23	Lead Wool	Operations/Shielding	NQ
		22	Coating	Operations	C
		21	Support Plate	Operations	B
		20	Retaining Ring	Operations	B
		19	Door Lock Bolt	Operations	C
		16	Door Rail	Operations	B
		15	Top Plate	Structural	B
		14	Neutron Shield	Shielding	B
		13	Trunnion Cap	Operations	C
		12	Trunnion	Structural	B
		7-11	Outer Shell	Structural	B
		2-6	Inner Shell	Structural	B
		1	Bottom Plate	Structural	B
790-561	Weldment, Structure, Vertical Concrete Cask	37	Dowel Pin	Operations	NQ
		36	Cover	Operations	C
		35	Pipe/Tube/Bar	Shielding	B
		32	Coatings	Operations	NQ
		31	Lifting Nut	Operations	NQ

Table 2.3-1 Quality Category Classification of Universal Storage System Components (continued)

Drawing No.	Description	Item No.	Component	Function	Quality Category
790-561 (Continued)	Weldment, Structure, Vertical Concrete Cask	26	Screen Table	Structural	C
		25	Baffle	Heat Transfer	B
		18-24	Outlet (4)	Heat Transfer	B
		20	Shield Plate	Shielding	B
		17	Nelson Stud	Structural	B
		16	Base Plate	Structural	B
		15	Stand	Structural	B
		13-14	Inlet (4)	Heat Transfer	B
		12	Bottom	Structural	B
		11	Shield Ring	Shielding	B
		10	Cover	Operations	B
		4-8	Jack (Leveling)	Operations	NQ
		3	Support Ring	Structural	C
		2	Top Flange	Structural	B
		1, 27-30	Shell	Shielding/Structural	B
790-562	Reinforcing Bar And Concrete Placement	48	Retainer Plate	Operations	NQ
		47	Spacer	Operations	B
		45	Washer	Operations	B
		44	Nut	Operations	B
		43	Threaded Rebar	Operations	B
		42	Supplemental Cover	Operations	NQ
		32	Base Plate	Structural	B
		31	Lift Lug	Structural	B
		29	Lag Screw	Operations	NQ

Table 2.3-1 Quality Category Classification of Universal Storage System Components (continued)

Drawing No.	Description	Item No.	Component	Function	Quality Category
790-562 (Continued)	Reinforcing Bar And Concrete Placement	28, 36, 39	Concrete Anchor	Operations	NQ
		16-19, 40-41, 49	Screen/Strip/Screw/Washer	Operations	NQ
		15	Concrete Shell	Shielding/ Structural	B
		1-11, 33, 46	Reinforcing Bar	Structural	B
790-563	Lid, Vertical Concrete Cask	1	Lid	Structural/Operations	B
790-564	Shield Plug, Vertical Concrete Cask	13			
		12			
		11			
		10			
		9			
		4, 8	Neutron Shield Cover	Shielding/Operations	B
		3, 5	Neutron Shield	Shielding	B
		2, 6, 7	NS Retaining Ring	Structural	B
		1	Shield Plug	Shielding	B
790-565	Nameplate, Vertical Concrete Cask	1	Nameplate	Operations	NQ
790-570	BWR Fuel Basket	23	Flat Washer	Structural	C
		4	Drain Tube Sleeve	Operations	C
790-571	Bottom Weldment, BWR Fuel Basket	3	Support	Structural	A
		2	Pad	Structural	A
		1	Plate	Structural	A

Table 2.3-1 Quality Category Classification of Universal Storage System Components (Continued)

Drawing No.	Description	Item No.	Component	Function	Quality Category
790-572	Top Weldment, BWR Fuel Basket	6	Baffle	Structural	A
		3-5	Support		
		2	Ring		
		1	Plate		
790-573	Support Disk and BWR Basket Details	8	Split Spacer	Structural	A
		7	Top Spacer	Structural	A
		5, 6	Tie Rod	Structural	A
		4	Top Nut	Structural	A
		3	Spacer	Structural	A
		1	Support Disk	Structural	A
790-574	Heat Transfer Disk, BWR	1	Heat Transfer Disk	Thermal	A
790-575	BWR Fuel Tube	10	Flange	Structural	A
		7-9	Cladding	Criticality Control	A
		4-6	Neutron Absorber	Criticality Control	A
		1-3	Tubing	Structural	A
790-581	PWR Fuel Tube	10	Flange	Structural	A
		7-9	Cladding	Criticality Control	A
		4-6	Neutron Absorber	Criticality Control	A
		1-3	Tubing	Structural	A
790-582	Canister, Shell	7	Location Lug	Operations	C
		6	Bottom	Structural/Confinement	A
		1-5	Shell	Structural/Confinement	A

Table 2.3-1 Quality Category Classification of Universal Storage System Components (Continued)

Drawing No.	Description	Item No.	Component	Function	Quality Category
790-583	Drain Tube Assembly	7	Metal Boss Seal	Operations	C
		2-6	Tube	Operations	C
		1	Nipple	Operations	C
790-584	Canister Details	8	Key	Operations	C
		7	Spacer Ring	Structural	C
		6	Lid Support Ring	Structural	B
		5	Cover	Confinement/Operations	B
		4	Structural Lid	Structural	A
		3	Metal Boss Seal	Operations	C
		2	Nipple	Operations	C
		1	Shield Lid	Shielding/Confinement	B
790-585	Transportable Storage Canister	24	Dowel Pin	Operations	NQ
		23	Structural Lid Plug	Operations	NQ
		22	Shield Lid Plug	Operations	NQ
790-587	Spacer Shim, Canister	1-6	Spacer Shims #1 - #6	Operations	C
790-590	Loaded Vertical Concrete Cask	19	Tab	Operations	NQ
		18	Seal Wire	Operations	C
		17	Security Seal	Operations	C
		16	Seal Tape (Optional)	Operations	NQ
		15	Cover	Operations	C
		14	Washer (Lid Bolt)	Operations	NQ
		13	Lid Bolt	Operations	B

Table 2.3-1 Quality Category Classification of Universal Storage System Components (Continued)

Drawing No.	Description	Item No.	Component	Function	Quality Category
790-591	Bottom Weldment, PWR Basket	3, 5-7	Support	Structural	A
		4	Pad	Structural	A
		2	Support	Structural	A
		1	Bottom Disk	Structural	A
790-592	Top Weldment, PWR Basket	7	Baffle	Structural	A
		3, 5-6	Support	Structural	A
		4	Center Support	Structural	A
		2	Ring	Structural	A
		1	Top Disk	Structural	A
790-593	Support Disk and Details, PWR	8	Top Spacer	Structural	A
		5-7	Tie Rod	Structural	A
		4, 9, 10	Top Nut	Structural	A
		3	Spacer	Structural	A
		2	Split Spacer	Structural	A
		1	Support Disk	Structural	A
790-594	Heat Transfer Disk, PWR	1	Heat Transfer Disk	Thermal	A
790-595	PWR Fuel Basket	8	Flat Washer	Structural	C
		4	Drain Tube Sleeve/Tube	Operations	C
790-605	BWR Fuel Tube, Over-Sized	7	Flange	Structural	A
		5-6	Cladding	Criticality Control	A
		3-4	Neutron Absorber	Criticality Control	A
		1-2	Tubing	Structural	A

Table 2.3-1 Quality Category Classification of Universal Storage System Components (Continued)

Drawing No.	Description	Item No.	Component	Function	Quality Category
790-613	Supplemental Shielding, VCC Inlets	4	Shims	Operations	NQ
		3	Paint	Operations	NQ
		2	Pipe	Shielding	B
		1	Side Plate	Shielding	B
790-617	Door Stop	6	Attachment Screw	Operations	NQ
		5	Lock Pin	Operations	NQ
		4	Handle	Operations	NQ
		3	Back Plate	Operations	NQ
		2	Top Plate	Operations	NQ
		1	Bottom Plate	Operations	NQ
412-502	Maine Yankee (MY) Fuel Can Details, NAC-UMS®	16	Dowel Pin	Operations	C
		13	Support Ring	Structural/Operations	B
		12	Lift Tee	Structural/Operations	B
		10, 19	Tube Body	Structural/Criticality	A
		9, 18	Side Plate	Structural/Criticality	A
		8	Bottom Plate	Structural/Criticality	A
		7, 15	Backing Screen	Operations	C
		6, 14	Filter Screen	Confinement	B
		5	Lid Bottom	Structural/Criticality	A
		4	Wiper	Operations	C
		3	Lid Guide	Operations	C
		2	Lid Plate	Structural/Criticality	A
		1	Lid Collar	Confinement	A

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2.4 Decommissioning Considerations

The principal elements of the Universal Storage System are the Vertical Concrete Cask and the Transportable Storage Canister.

The concrete cask provides biological shielding and physical protection for the contents of the canister during long-term storage. The concrete that provides biological shielding is not expected to become contaminated during the period of use, as it does not come into contact with other contaminated objects or surfaces. The concrete cask is not expected to become surface contaminated during use, except through incidental contact with other contaminated surfaces. Incidental contact could occur at the interior surface (liner) of the concrete cask, the top surface that supports the transfer cask during loading and unloading operations, and the base plate of the concrete cask that supports the canister. All of these surfaces are made of carbon steel, and it is anticipated that these surfaces could be decontaminated as necessary for decommissioning.

Activation of the carbon steel liner, concrete, support plates, and reinforcing bar could occur due to neutron flux from the stored fuel. Since the neutron flux rate is low, only minimal activation of carbon steel in the concrete cask is expected to occur. The activity concentrations from activation of storage cask components are listed in Tables 2.4-1 through 2.4-4. Tables 2.4-1 and 2.4-2 provide the activation summaries of the concrete cask and canister for the design-basis PWR fuel, while Tables 2.4-3 and 2.4-4 provide the summaries for the design-basis BWR fuel. These tables include the radiologically significant isotopes, together with a total concentration of all activated nuclides in the respective component. The total concentrations listed include activities of radionuclides, which do not have any substantial contribution to radiation dose and are not specifically identified by 10 CFR 61 waste classification. In particular, the isotope contributing the majority of the carbon steel total curie activity is ^{55}Fe, which decays following electron capture and is not of radiological concern.

Decommissioning of the concrete cask will involve the removal of the canister, and the subsequent disassembly of the concrete cask. It is expected that the concrete will be broken up, and steel components segmented, to reduce volume. Any contaminated or activated items are expected to qualify for near-surface disposal as low specific activity material. The activity concentrations from activation of concrete cask components resulting from the design basis PWR and BWR fuel assemblies are listed in Tables 2.4-1 and 2.4-3, respectively.

The Transportable Storage Canister is designed and fabricated to be suitable for use as part of the waste package for permanent disposal in a deep Mined Geological Disposal System (i.e., it meets the requirements of the DOE MPC Design Procurement Specification [20]). The canister is fabricated from materials having high long-term corrosion resistance, and it contains no paints or coatings that could adversely affect its permanent disposal. Consequently, decommissioning of the canister will occur only if the fuel contained in the canister had to be removed, or if current requirements for disposal were to change. Decommissioning of the canister will require that the closure welds at the canister structural lid, shield lid, and shield lid port covers be cut, so that the spent fuel can be removed. Removal of the contents of the canister will require that the canister be returned to a spent fuel pool or dry unloading facility, such as a hot cell. Closure welds can be cut either manually or with automated equipment, with the procedure being essentially the reverse of that used to initially close the canister.

Following removal of its contents, the canister interior is expected to have significant contamination, and the bottom of the canister may contain "crud" or other residual material. Some effort may be required to remove the surface contamination prior to disposal; however, in practice, it will not be absolutely necessary to decontaminate the canister internals. Since the canister internal contamination will consist only of by-product materials, any contaminated canister and internal components are expected to qualify for near-surface disposal as low specific activity waste without internal contamination. Any required internal decontamination is facilitated, should it become necessary, by the smooth surfaces of the canister and the basket, and by the design that precludes the presence of crud traps. Since the neutron flux rate from the stored fuel is low, only minimal activation of the canister is expected to occur. The activity concentrations from activation of canister components resulting from the design basis PWR and BWR fuel assemblies are listed in Tables 2.4-2 and 2.4-4, respectively.

The unloaded canister can also qualify as a strong, tight container for other waste. In this case, the canister can be filled, within weight limits, with other qualified waste, closed, and transported to a near-surface disposal site. Use of the canister for this purpose can reduce decommissioning costs by avoiding decontamination, segmenting, and repackaging.

The storage pad, fence, and supporting utility fixtures are not expected to require decontamination as a result of use of the Universal Storage System. The design of the cask and canister precludes the release of contamination from the contents over the period of use of the system. Consequently, these items may be reused or disposed of as locally generated clean waste.

Table 2.4-1 Activity Concentration Summary for the Concrete Cask - PWR Design Basis Fuel (Ci/m^3)

Isotope[1]	Concrete Shell	Shell Liner	Shield Plug	Lid	Cover Plate	Bottom	Base Plate
^{14}C	--	--	2.35E-08	--	--	--	--
^{45}Ca	4.62E-06	--	--	--	--	--	--
^{54}Mn	5.13E-08	6.97E-02	1.34E-03	1.63E-04	3.17E-06	5.56E-02	1.88E-02
^{55}Fe	2.30E-05	1.22E+00	2.12E-01	5.49E-02	3.85E-05	7.15E-01	2.27E-01
^{60}Co	1.95E-06	3.43E-04	7.22E-05	1.38E-05	1.54E-05	2.71E-04	8.58E-05
^{63}Ni	--	--	--	--	2.02E-02	--	--
Total	3.09E-05	1.30E+00	2.15E-01	5.54E-02	2.06E-02	7.77E-01	2.48E-01

1. 40-year activation, 1-week cooling.

Table 2.4-2 Activity Concentration Summary for the Canister – PWR Design Basis Fuel (Ci/m^3)

Isotope[1]	Wall	Shield Lid	Structural Lid	Bottom
^{54}Mn	9.94E-05	3.32E-04	4.42E-06	1.00E-04
^{55}Fe	7.94E-04	8.26E-04	3.67E-04	1.05E-03
^{60}Co	3.15E-04	3.31E-04	1.47E-04	4.22E-04
^{59}Ni	3.54E-07	3.67E-07	1.64E-07	4.66E-07
^{63}Ni	4.17E-01	4.33E-01	1.93E-01	5.49E-01
Total	4.27E-01	4.43E-01[2]	1.97E-01[2]	5.63E-01[2]

1. 40-year activation, 1 -week cooling.
2. ^{32}P accounts for most of the unlisted total activity.

Table 2.4-3 Activity Concentration Summary for the Concrete Cask – BWR Design Basis Fuel (Ci/m^3)

Isotope[1]	Concrete Shell	Shell Liner	Shield Plug	Lid	Cover Plate	Bottom	Base Plate
^{14}C	--	--	3.57E-08	--	--	--	--
^{45}Ca	7.91E-06	--	--	--	--	--	--
^{54}Mn	7.74E-08	1.07E-01	1.97E-03	2.39E-04	1.37E-06	7.06E-02	2.40E-02
^{55}Fe	3.93E-05	2.10E00	3.23E-01	8.29E-02	2.08E-05	1.13E-04	3.52E-01
^{60}Co	3.33E-06	5.93E-04	1.10E-04	2.08E-05	8.35E-06	4.26E-04	1.33E-04
^{63}Ni	--	--	--	--	1.09E-02	--	--
Total	5.28E-05	2.22E00	3.27E-01	8.37E-02	1.12E-02	1.21E00	3.79E-01

1. 40-year activation, 1-week cooling.

Table 2.4-4 Activity Concentration Summary for the Canister – BWR Design Basis Fuel (Ci/m^3)

Isotope[1]	Wall	Shield Lid	Structural Lid	Bottom
^{54}Mn	1.53E-04	4.89E-05	6.51E-06	1.26E-04
^{55}Fe	1.39E-03	1.26E-06	5.57E-04	1.68E-03
^{60}Co	5.52E-04	5.04E-04	2.22E-04	6.73E-04
^{59}Ni	6.21E-07	5.60E-07	2.48E-07	7.46E-07
^{63}Ni	7.31E-01	6.60E-01	2.92E-01	8.79E-01
Total	7.49E-01	6.76E-01	2.99E-01	9.00E-01

1. 40-year activation, 1-week cooling.

2.5 References

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3.0 STRUCTURAL EVALUATION

This chapter describes the design and analysis of the principal structural components of the Universal Storage System under normal operating conditions. It demonstrates that the Universal Storage System meets the structural requirements for confinement of contents, criticality control, radiological shielding, and contents retrievability required by 10 CFR 72 [1] for the design basis normal operating conditions. Off-normal and accident conditions are evaluated in Chapter 11.0.

3.1 Structural Design

The Universal Storage System includes five configurations to accommodate three classes of PWR and two classes of BWR fuel assemblies. The five classes of fuel are determined primarily by the overall length of the fuel assembly. The allocation of a fuel design to a UMS class is shown in Tables 2.1.1-1 and 2.1.2-1 for PWR and BWR fuel, respectively.

The three major components of the Universal Storage System are the vertical concrete cask; the transportable storage canister (canister), and the transfer cask (see Figure 3.1-1). These components are provided in five different lengths to accommodate the five classes of fuel. They also have different weights, as shown in Table 3.2-1 for the PWR configurations, and in Table 3.2-2 for the BWR configurations. The weight differences reflect the differences in length of components and fuel, and differences in basket design between the PWR and BWR configurations.

The principal structural members of the vertical concrete cask are the reinforced concrete shell and steel liner. The principal structural members of the canister are the structural lid, shell, bottom plate, the welds joining these components, and the fuel basket assembly. For the transfer cask, the trunnions, the inner and outer steel walls, the bottom shield doors, and the shield door support rails, are the principal structural components.

The evaluations presented in this chapter are based on the bounding or limiting configuration of the UMS System for the condition being evaluated. In most cases, the bounding condition evaluates the heaviest configuration of the five classes. For each evaluated condition, the bounding configuration applied is identified. Margins of safety greater than ten are generally stated in the analyses as "+Large." Numerical values are shown for Margins of safety that are less than ten.

3.1.1 Discussion

The transportable storage canister is designed to be transported in the Universal Transport Cask (USNRC Docket Number 71-9270 [2]. Consequently, the canister diameter is same for each of the five configurations. The outside diameter of the vertical concrete cask is established by the shielding requirement for the design basis fuel used for the shielding evaluation. The shielding required for the design basis fuel is conservatively applied to the five concrete cask configurations.

Vertical Concrete Cask

The vertical concrete cask is a reinforced concrete cylinder with an outside diameter of 136 in. and an overall height (including the lid) ranging from 210.68 in. to 227.38 in., depending upon the configuration. The internal cavity of the concrete cask is lined by a 2.5-inch thick carbon steel inner shell having an inside diameter of 74.5 in. The support ring for the concrete cask shield plug at the top of the inner shell limits the available contents diameter to less than 69.5 in. The inner shell thickness is primarily determined by radiation shielding requirements, but is also related to the need to establish a practical limit for the diameter of the concrete shell. The concrete shell is constructed using Type II Portland Cement and has a nominal density of 140 lb/ft^3 and a nominal compressive strength of 4000 psi. The inner and outer rebar assemblies are formed by vertical hook bars and horizontal hoop bars.

A ventilation air-flow path is formed by inlets at the bottom of the cask, the annular space between the cask inner shell and the canister, and outlets near the top of the cask. The passive ventilation system operates by natural convection as cool air enters the bottom inlets, is heated by the canister, and exits from the top outlets.

A shield plug that consists of 4.125 inches of carbon steel and either a 1-inch thick layer of NS-4-FR or a 1.5-inch thick layer of NS-3 neutron shield material enclosed by the carbon steel is installed in the concrete cask cavity above the canister. The plug is supported by a support ring welded to the inner shell. The 1.5-in. thick carbon steel lid provides a cover to protect the canister from adverse environmental conditions and postulated tornado driven missiles. The shield plug and lid provide shielding to reduce the skyshine radiation. When the lid is bolted in place, the shield plug is secured between the lid and the shield plug support ring.

Transportable Storage Canister

The transportable storage canister consists of a cylindrical shell assembly closed at its top end by an inner shield lid and an outer structural lid. The canister forms the confinement boundary for the basket assembly that contains the PWR or BWR spent fuel. Three canister classes accommodate the PWR fuel assemblies (Tables 2.1.1-1) and two canister classes accommodate the BWR fuel assemblies (Table 2.1.2-1). The canister is fabricated from Type 304L stainless steel. The canister shield lid is 7-in. thick, SA-240 Type 304 stainless steel, and the structural lid is 3.0-in. thick SA-240, Type 304L stainless steel. SA-182 Type 304 stainless steel may be substituted for the SA-240 Type 304 stainless steel used in the shield lid, provided that the SA-182 material has yield and ultimate strengths equal to or greater than those of the SA-240 material. Similarly, SA-182 Type 304L stainless steel may be substituted for the SA-240 Type 304L stainless steel used in the structural lid, provided that the SA-182 material has yield and ultimate strengths equal to or greater than those of the SA-240 material. Both lids are welded to the canister shell to close the canister. The minimum weld sizes for the PWR canister are 0.75 inch for the structural lid and 0.375 inch for the shield lid. For analysis purposes, bounding PWR canister results are reported except for the BWR canister tip-over evaluation (Section 11.2.12.3.2). The minimum weld sizes for the BWR canister are 0.875 inch for the structural lid and 0.5 inch for the shield lid. The shield lid is supported by a support ring. The structural lid is supported, prior to welding, by the shield lid. A groove is machined into the structural lid circumference to accept a spacer ring. The spacer ring facilitates welding of the structural lid to the canister shell. The bottom of the canister is a 1.75-in. thick SA-240, Type 304L stainless steel plate that is welded to the canister shell. The canister is also described in Section 1.2.1.1.

The fuel basket assembly is provided in two configurations — one for up to 24 PWR fuel assemblies and one for up to 56 BWR fuel assemblies. The PWR basket is comprised of Type 17-4 PH stainless steel support disks, Type 6061-T651 aluminum alloy heat transfer disks, and Type 304 stainless steel fuel tubes equipped with a neutron absorber and stainless steel cover. The remaining structural components are Type 304 stainless steel. The BWR basket is comprised of SA-533 carbon steel support disks coated with electroless nickel, Type 6061-T651 aluminum alloy heat transfer disks, and fuel tubes constructed of the same materials as the PWR tubes. The remaining structural components of the BWR basket are Type 304 stainless steel. The basket assemblies are more fully described in Section 1.2.1.2.

The fuel basket support disks, heat transfer disks, and fuel tubes, together with the top and bottom weldments, are positioned by tie rods (with spacers and washers) that extend the length of the basket and hold the assembly together. The support disks provide structural support for the fuel tubes. They also help to remove heat from the fuel tubes. The heat transfer disks provide the primary heat removal capability and are not considered to be structural components. The heat transfer disks are sized so that differential thermal expansion does not result in disk contact with the canister shell. The number of heat transfer disks and support disks varies depending upon the length of the fuel to be confined in the basket. The fuel tubes house the spent fuel assemblies. The top and bottom weldments provide longitudinal support for the fuel tubes. The fuel tubes are fabricated from Type 304 stainless steel. No structural credit is taken for the presence of the fuel tubes in the basket assembly analysis. The walls of each PWR fuel tube support a sheet of neutron absorber material that is covered by stainless steel. No structural credit is taken in the basket assembly analysis for the neutron absorber sheet or its stainless steel cover. The PWR assembly fuel tubes have a nominal inside dimension of 8.8-inches square and a composite wall thickness of 0.14 inch. The BWR assembly fuel tubes have a nominal inside dimension of 5.9-inches square and a composite wall thickness of 0.20 inch. Depending upon its location in the basket assembly, an individual BWR fuel tube may support neutron absorber material on one or two sides. Certain fuel tubes located on the outer edge of the basket do not have neutron absorber material. The fuel tubes have been evaluated to ensure that the neutron absorber material remains in place under normal conditions and design basis off-normal and accident events.

Four over-sized fuel storage positions are located on the periphery of the BWR basket to provide additional space for BWR fuel assemblies with channels that have been reused, since reused channels are expected to have increased bowing or bulging. Normal BWR fuel assemblies may also be stored in these locations.

As mentioned above, five classes of transportable storage canisters are provided for the storage of PWR and BWR spent fuel. The analysis is based on the identification of bounding conditions and the application of those conditions to determine the maximum stresses.

The canister is designed to be transported in the Universal Transport Cask. Transport conditions establish the design basis loading, except for lifting, because the hypothetical accident transport conditions produce higher stresses in the canister and basket than do the design basis storage conditions. Consequently, the canister and basket design is conservative with respect to storage conditions. The evaluation of the canister and basket assembly for transport conditions is documented in the Safety Analysis Report for the Universal Transport Cask [2].

Transfer Cask

The transfer cask, with its lifting yoke, is primarily a lifting device used to move the canister. It provides biological shielding when it contains a loaded canister. The transfer cask is provided in the Standard configuration for canisters weighing up to 88,000 lbs, or in the Advanced configuration for canisters weighing up to 98,000 lbs. The transfer cask configurations have identical operational features. The transfer cask is a heavy lifting device that is designed, fabricated and load-tested to the requirements of NUREG-0612 [8] and ANSI N14.6 [9]. The transfer cask design incorporates a top retaining ring, which is bolted in place to prevent a loaded canister from being inadvertently lifted through the top of the transfer cask. The transfer cask has retractable bottom shield doors. During loading operations, the doors are closed and secured by bolts/pins, so they cannot inadvertently open. During unloading, the doors are retracted using hydraulic cylinders to allow the canister to be lowered into the storage or transport cask. The principal design parameters of the transfer casks are shown in Table 1.2-7.

Both transfer cask configurations are provided in five different lengths to accommodate the canisters containing one of the three classes of PWR fuel assemblies or two classes of BWR fuel assemblies.

The transfer cask is used for the vertical transfer of the canister between work stations and the concrete cask, or transport cask. It incorporates a multiwall (steel/lead/NS-4-FR/steel) design to provide radiation shielding.

Component Evaluation

The following components are evaluated in this chapter:

- canister lifting devices,
- canister shell, bottom, and structural lid,
- canister shield lid support ring,
- fuel basket assembly,
- transfer cask trunnions, shells, retaining ring, bottom doors, and support rails,
- vertical concrete cask body, and
- concrete cask steel components (reinforcement, inner shell, lid, bottom plate, bottom, etc.).

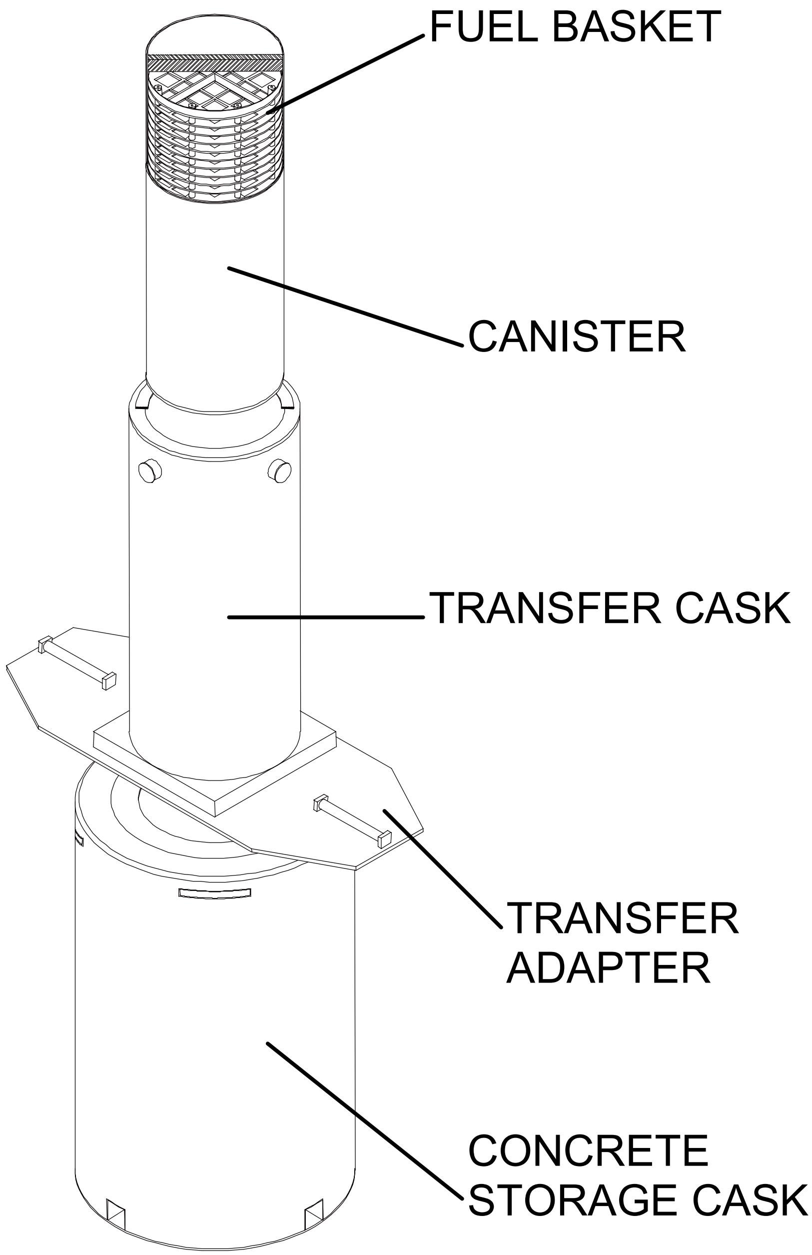
Other Universal Storage System components shown on the license drawings in Chapter 1 are included as loads in the evaluation of the components listed above, as appropriate.

The structural evaluations in this chapter demonstrate that the Universal Storage System components meet their structural design criteria and are capable of safely storing the design basis PWR or BWR spent fuel.

3.1.2 Design Criteria

The Universal Storage System structural design criteria are described in Section 2.2. Load combinations for normal, off-normal, and accident loads are evaluated in accordance with ANSI/ANS 57.9 [3] and ACI-349 [4] for the concrete cask (see Table 2.2-1), and in accordance with the ASME Code, Section III, Division I, Subsection NB [5] for Class 1 components of the canister (see Table 2.2-2). The basket is evaluated in accordance with ASME Code, Section III, Subsection NG [6], and NUREG-6322 [7]. The transfer cask and the lifting yoke are lifting devices that are designed to NUREG-0612 [8] and ANSI N14.6 [9].

Figure 3.1-1 Principal Components of the Universal Storage System



Note: Standard transfer cask shown.

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3.2 Weights and Centers of Gravity

The weights and centers of gravity (CGs) for the Universal Storage System PWR configuration and components are summarized in Table 3.2-1. Those for the BWR configuration are summarized in Table 3.2-2. The weights and CGs presented in this section are calculated on the basis of nominal design dimensions.

Table 3.2-1 Universal Storage System Weights and CGs – PWR Configuration

Description	Class 1		Class 2		Class 3	
	Calculated Weight (lb)	Center of Gravity[1]	Calculated Weight (lb)	Center of Gravity[1]	Calculated Weight (lb)	Center of Gravity[1]
Fuel Contents (including inserts)	37,700	—	38,500	—	35,600	—
Poison Rods (Inserts)	(1,400)	—	(1,400)	—	—	—
Concrete Cask Lid	2,500	—	2,500	—	2,500	—
Concrete Cask Shield Plug	4,900	—	4,900	—	4,900	—
Canister (empty, w/o lids)	8,400	—	8,700	—	9,000	—
Canister Structural Lid	3,000	—	3,000	—	3,000	—
Canister Shield Lid	7,000	—	7,000	—	7,000	—
Transfer Adapter Plate	11,200	—	11,200	—	11,200	—
Transfer Cask Lifting Yoke[4]	6,000	—	6,000	—	6,000	—
Water in Canister	14,000	—	14,800	—	15,800	—
Basket	14,900	—	16,000	—	16,500	—
Canister (with basket, without fuel or lids)	23,300	—	24,700	—	25,500	—
Canister (with fuel, and shield and structural lids)	70,600	—	72,900	—	70,800	—
Concrete Cask (empty, with shield plug and lid; includes optional lift lugs) – 140 pcf concrete	223,500	—	232,300	—	239,700	—
Concrete Cask (with loaded Canister and lids; includes optional lift lugs)[2] – 140 pcf concrete	294,100	108.8	305,100	113.1	310,400	117.1
Concrete Cask with Lift Anchors (empty, with shield plug and lid) – 148 pcf concrete	232,600	—	241,700	—	249,400	—
Concrete Cask with Lift Anchors (with loaded Canister and lids)[2] – 148 pcf concrete	303,300	108.7	314,600	112.9	320,200	117.0
Transfer Cask (empty)[3]	112,300	—	117,300	—	121,500	—
Transfer Cask and Canister, basket (empty, without lids)[3]	135,500	—	141,900	—	146,900	—
Transfer Cask and Canister (with fuel, water and shield lid)[3]	193,900	—	201,900	—	205,000	—
Transfer Cask and Canister (with fuel, dry with lids)[3]	182,900	—	190,100	—	192,200	—

General Note: All weights are rounded up. Therefore, assembly weights cannot be computed using rounded value of component weights.

1. Weights and CGs are calculated from nominal design dimensions.
2. Center of gravity is measured from the bottom of the concrete cask.
3. Standard or Advanced Transfer Cask.
4. Transfer cask lifting yoke weight for specific sites may vary from listed weight. The site-specific yoke weight should be used for site-specific applications.

Table 3.2-2 Universal Storage System Weights and CGs – BWR Configuration

Item Description	Class 4		Class 5	
	Calculated Weight (lb)	Center of Gravity[1]	Calculated Weight (lb)	Center of Gravity[1]
Fuel Contents (Including channels)	39,400	—	39,400	—
Concrete Cask Lid	2,500	—	2,500	—
Concrete Cask Shield Plug	4,900	—	4,900	—
Canister (empty, w/o lids)	8,800	—	9,000	—
Canister Structural Lid	3,000	—	3,000	—
Canister Shield Lid	7,000	—	7,000	—
Transfer Adapter Plate	11,200	—	11,200	—
Transfer Cask Lifting Yoke[4]	6,000	—	6,000	—
Water in Canister	15,100	—	15,200	—
Basket	17,200	—	17,600	—
Canister (with basket, without fuel or lids)	25,900	—	26,500	—
Canister (with fuel, and shield and structural lids)	75,000	—	75,600	—
Concrete Cask (empty, with shield plug and lid, includes optional lift lugs) – 140 pcf concrete	233,700	—	238,400	—
Concrete Cask (with loaded Canister and lids, includes optional lift lug)[2] – 140 pcf concrete	308,700	113.7	313,900	115.8
Concrete Cask (empty, with shield plug and lid, includes optional lift lugs) – 148 pcf concrete	243,200	—	248,000	—
Concrete Cask (with loaded Canister and lids, includes optional lift lug)[2] – 148 pcf concrete	319,000	113.6	323,900	115.7
Transfer Cask (empty)[3]	118,000	—	120,700	—
Transfer Cask and Canister (empty, without lids)[3]	143,900	—	147,200	—
Transfer Cask and Canister (with fuel, water and shield lid)[3]	205,100	—	208,400	—
Transfer Cask and Canister (with fuel, dry with lids)[3]	193,000	—	196,200	—

General Note: All weights are rounded up. Therefore, assembly weights cannot be computed using rounded values of component weights.

1. Weights and CGs are calculated from nominal design dimensions.
2. Center of gravity is measured from the bottom of the concrete cask.
3. Standard or Advanced Transfer Cask
4. Transfer cask lifting yoke weight for specific sites may vary from listed weight. The site-specific yoke weight should be used for site-specific applications.

Table 3.2-3 Calculated Under-Hook Weights for the Standard Transfer Cask

Configuration	PWR Class 1	PWR Class 2	PWR Class 3	BWR Class 4	BWR Class 5
Transfer cask (empty)	112,300	117,300	121,500	118,000	120,700
Transfer cask, canister (empty, without lids) and yoke[1]	141,400	147,800	152,700	149,800	153,000
Transfer cask; loaded canister wet (fuel, water and shield lid); and yoke[1]	199,800	207,800	210,900	211,000	214,300
Transfer cask, loaded canister dry (fuel and lids) and yoke[1]	188,700	196,000	198,000	198,900	202,100

General Note: All weights are rounded to the next 100 lb.

1. Transfer cask lifting yoke weight for specific sites may vary from listed weight. The site-specific yoke weight should be used for site-specific applications.

3.3 Mechanical Properties of Materials

The mechanical properties of steels used in the fabrication of the Universal Storage System components are presented in Tables 3.3-1 through 3.3-10. The primary steels, Type 304 and Type 304L stainless steel, were selected because of their high strength, ductility, resistance to corrosion and brittle fracture, and metallurgical stability for long-term storage.

3.3.1 Primary Component Materials

The steels and aluminum alloy used in the fabrication of the canister and basket are:

Canister shell	ASME SA-240, Type 304L stainless steel
Canister bottom plate	ASME SA-240, Type 304L stainless steel
Canister shield lid	ASME SA-240, Type 304 stainless steel
Canister structural lid	ASME SA-240, Type 304L stainless steel
Support disks	
PWR basket	ASME SA-693, Type 630, 17-4 PH stainless steel
BWR basket	ASME SA-533, Type B class 2 carbon steel
Heat transfer disks	ASME SB-209, Type 6061-T651 aluminum alloy
Spacers	ASME SA-312, Type 304 stainless steel
Tie rods	ASME SA-479, Type 304 stainless steel
Basket end weldments	ASME SA-240, Type 304 stainless steel
Fuel tubes	ASTM A240, Type 304 stainless steel

SA-182 Type 304 stainless steel may be substituted for SA-240 Type 304 stainless steel for the shield lid provided that the SA-182 material has yield and ultimate strengths greater than or equal to those of the SA-240 material. SA-182 Type 304L stainless steel may be substituted for SA-240 Type 304L stainless steel for the structural lid provided that the SA-182 material has yield and ultimate strengths greater than or equal to those of the SA-240 material.

Steels used in the fabrication of the vertical concrete cask are:

Inner shell	ASTM A36 carbon steel
Pedestal and base	ASTM A36 carbon steel
Reinforcing bar	ASTM A615, Grade 60 carbon steel
	ASTM A615, Grade 75 carbon steel
	ASTM A706 carbon steel

The steels used in the fabrication of the transfer cask are:

Inner shell	ASTM A588 low alloy steel
Outer shell	ASTM A588 low alloy steel
Bottom plate	ASTM A588 low alloy steel
Top plate	ASTM A588 low alloy steel
Retaining ring	ASTM A588 low alloy steel
Trunnions	ASTM A350, LF2 low alloy steel
Shield doors and rails	ASTM A350, LF2 low alloy steel
Retaining ring bolts	ASTM A193, Grade B6 high alloy steel

The mechanical properties of the 6061-T651 aluminum heat transfer disks in the fuel basket are shown in Table 3.3-11. The mechanical properties of the concrete are listed in Table 3.3-12. Table 3.3-13 provides the mechanical properties of NS-4-FR and NS-3. The mechanical properties of carbon steel (SA-516, Grade 70) are shown in Table 3.3-14.

Table 3.3-1 Mechanical Properties of SA-240 and A-240, Type 304 Stainless Steel

Property	**Value**									
Temperature (°F)	-40	-20	70	200	300	400	500	750	800	900
Ultimate strength, S_u (ksi)*	75.0	75.0	75.0	71.0	66.0	64.4	63.5	63.1	62.7	61.0
Yield strength, S_y (ksi)*	30.0	30.0	30.0	25.0	22.5	20.7	19.4	17.3	16.8	16.2
Design Stress Intensity, S_m (ksi)*	20.0	20.0	20.0	20.0	20.0	18.7	17.5	15.6	15.2	—
Modulus of Elasticity, E ($\times 10^3$ ksi)*	28.7	28.7	28.3	27.6	27.0	26.5	25.8	24.4	24.1	23.5
Alternating Stress @ 10 cycles (ksi)**	718.0	718.0	708.0	690.5	675.5	663.0	645.5	610.4	—	—
Alternating Stress @ 10^6 cycles (ksi)**	28.7	28.7	28.3	27.6	27.0	26.5	25.8	24.4	—	—
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/°F)*	8.13	8.19	8.46	8.79	9.00	9.19	9.37	9.76	9.82	—
Poisson's Ratio*	0.31									
Density*	503 lbm/ft^3 (0.291 lbm/in^3)									

General Note: SA-182, Type 304 stainless steel may be substituted for SA-240, Type 304 stainless steel provided that the SA-182 material yield and ultimate strengths are equal to or greater than those of the SA-240 material. The SA-182 forging material and the SA-240 plate material are both Type 304 austenitic stainless steels. Austenitic stainless steels do not experience a ductile-to-brittle transition for the range of temperatures considered in this Safety Analysis Report. Therefore, fracture toughness is not a concern.

* ASME Code, Section II, Part D [10].

** ASME Code, Appendix I [11].

Table 3.3-2 Mechanical Properties of SA-479, Type 304 Stainless Steel

Property	Value							
Temperature (ºF)	-40	-20	70	200	300	400	500	750
Ultimate strength, $S_{u,}$ (ksi) ***	—	75.0	75.0	71.0	66.0	64.4	63.5	63.1
Yield strength, $S_{y,}$ (ksi) ***	—	30.0	30.0	25.0	22.5	20.7	19.4	17.3
Design Stress Intensity, $S_{m,}$(ksi) *	20.0	20.0	20.0	20.0	20.0	18.7	17.5	15.6
Modulus of Elasticity ($\times 10^3$ ksi) *	28.8	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Alternating Stress @ 10 cycles (ksi) **	720	718	708	683	675	663	645	610
Alternating Stress @ 10^6 cycles (ksi) **	28.8	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/ºF) *	—		8.46	8.79	9.00	9.19	9.37	9.76
Poisson's Ratio*	0.31							
Density*	503 lbm/ft³ (0.291 lbm/in³)							

* ASME Code, Section II, Part D [10].

** ASME Code, Appendix I [11].

*** Calculated based on Design Stress Intensity:

$$\left(\frac{S_{m-temp}}{S_{m\,70^\circ}}\right) S_{u70} = S_{u-temp}$$

Table 3.3-3 Mechanical Properties of SA-240, Type 304L Stainless Steel

Property	**Value**							
Temperature (°F)	-40	-20	70	200	300	400	500	750
Ultimate strength, $S_{u,}$ (ksi) *	70.0	70.0	70.0	66.2	60.9	58.5	57.8	55.9
Yield strength, $S_{y,}$ (ksi) *	25.0	25.0	25.0	21.4	19.2	17.5	16.4	14.7
Design Stress Intensity, $S_{m,}$(ksi) *	16.7	16.7	16.7	16.7	16.7	15.8	14.8	13.3
Modulus of Elasticity ($\times 10^3$ ksi) *	28.7	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Alternating Stress @ 10 cycles (ksi) **	718.0	718.0	708.0	690.5	675.5	663.0	645.5	610.4
Alternating Stress @ 10^6 cycles (ksi) **	28.7	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/°F) **	8.13	8.19	8.46	8.79	9.00	9.19	9.37	9.76
Poisson's Ratio*	0.31							
Density*	503 lbm/ft^3(0.291 lbm/in^3)							

General Note: SA-182, Type 304L stainless steel may be substituted for SA-240 Type 304L stainless steel provided that the SA-182 material yield and ultimate strengths are equal to or greater than those of the SA-240 material. The SA-182 forging material and the SA-240 plate material are both Type 304L austenitic stainless steels. Austenitic stainless steels do not experience a ductile-to-brittle transition for the range of temperatures considered in this Safety Analysis Report. Therefore, fracture toughness is not a concern.

* ASME Code, Section II, Part D [10].

** ASME Code, Appendix I [11].

Table 3.3-4 Mechanical Properties of SA-564 and SA-693, Type 630, 17-4 PH Stainless Steel

Property	Value								
Temperature (ºF)	-40	-20	70	200	300	400	500	650	800
Ultimate strength, S_u, (ksi) *	135.0	135.0	135.0	135.0	135.0	131.4	128.5	125.7	105.3***
Yield strength, S_y, (ksi) *	105.0	105.0	105.0	97.1	93.0	89.8	87.0	83.6	77.7***
Design Stress Intensity, S_m,(ksi) *	45.0	45.0	45.0	45.0	45.0	43.8	42.8	41.9	35.1
Modulus of Elasticity ($\times 10^3$ ksi) *	28.7	28.7	28.3	27.6	27.0	26.5	25.8	25.1	24.1
Alternating Stress @ 10 cycles (ksi) **	401.8	401.8	396.2	386.4	378.0	371.0	361.2	341.6	--
Alternating Stress @ 10^6 cycles (ksi) **	19.1	19.1	18.9	18.4	18.0	17.7	17.2	16.3	--
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/ºF) **	—		5.89	5.90	5.90	5.91	5.91	5.93	5.96
Poisson's Ratio*	0.31								
Density*	503 lbm/ft³ (0.291 lbm/in³)								

* ASME Code, Section II, Part D [10].

** ASME Code, Appendix I [11].

*** MIL-HDBK-5G [15].

Table 3.3-5 Mechanical Properties of A-36 Carbon Steel

Property	Value							
Temperature (°F)	100	200	300	400	500	600	650	700
Ultimate strength, $S_{u,}$ (ksi) ***	58.0	58.0	58.0	58.0	—	—	—	—
Yield strength, $S_{y,}$ (ksi) *	36.0	32.8	31.9	30.8	29.1	26.6	26.1	25.9
Design Stress Intensity, $S_{m,}$(ksi) *	19.3	19.3	19.3	19.3	19.3	17.7	17.4	17.3
Modulus of Elasticity, E ($\times 10^3$ ksi) *	29.0	28.8	28.3	27.7	27.3	26.7	26.1	25.5
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/°F) *	5.53	5.89	6.26	6.61	6.91	7.17	7.30	7.41
Poisson's Ratio*	0.31							
Density**	0.284 lbm/in^3							

* ASME Code, Section II, Part D [10].

** Metallic Materials Specification Handbook [12].

*** ASME Code Case, Nuclear Components, N-71-17 [13].

Table 3.3-6 Mechanical Properties of A615, Grade 60, A615, Grade 75 and A706 Reinforcing Steel

Property	A615, Grade 60	A615, Grade 75	A706
Ultimate Strength ** (ksi)	90.0	100.0	80.0
Yield Strength ** (ksi)	60.0	75.0	60.0
Coefficient of Thermal Expansion,* α (in/in/° F)	6.1×10^{-6}	6.1×10^{-6}	6.1×10^{-6}
Density [12] lbm/in^3	0.284	0.284	0.284

* Metallic Materials Specification Handbook [12].

** Annual Book of ASTM Standards [14].

Table 3.3-7 Mechanical Properties of SA-533, Type B, Class 2 Carbon Steel

Property	Value							
Temperature (ºF)	-20	70	200	300	400	500	750	800
Ultimate strength S_u, (ksi) *	90.0	90.0	90.0	90.0	90.0	90.0	87.2	81.8
Yield strength, $S_{y,}$ (ksi) *	70.0	70.0	65.5	64.5	63.2	62.3	59.3	58.3
Design Stress Intensity, $S_{m,}$(ksi) *	30.0	30.0	30.0	30.0	30.0	30.0	—	—
Modulus of Elasticity E, ($\times 10^3$ ksi) *	29.9	29.2	28.5	28.0	27.4	27.0	24.6	23.9
Alternating Stress @ 10 cycles (ksi) **	465.0	465.0	453.8	435.0	436.3	429.9	391.7	—
Alternating Stress @ 10^6 cycles (ksi) **	15.8	15.8	15.4	15.2	14.8	14.6	13.3	—
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/ºF) *	—	7.02	7.25	7.43	7.58	7.70	8.00	8.05
Poisson's Ratio *	0.31							
Density *	503 lbm/ft^3(0.291 lbm/in^3)							

* ASME Code, Section II, Part D [10].

** ASME Code, Section III, Appendix I [11].

Table 3.3-8 Mechanical Properties of A-588, Type A or B Low Alloy Steel

Property	Value							
Temperature (ºF)	100	200	300	400	500	600	650	700
Ultimate strength, $S_{u,}$ (ksi) ***	70.0	70.0	70.0	70.0	70.0	70.0	70.0	70.0
Yield strength, $S_{y,}$ (ksi) ***	50.0	47.5	45.6	43.0	41.8	39.9	38.9	37.9
Design Stress Intensity, $S_{m,}$ (ksi) ***	23.3	23.3	23.3	23.3	23.3	23.3	23.3	23.3
Modulus of Elasticity E, ($\times 10^3$ ksi) *	29.0	28.8	28.3	27.7	27.3	26.7	26.1	25.5
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/ºF) *	5.53	5.89	6.26	6.61	6.91	7.17	7.30	7.41
Poisson's Ratio*	0.31							
Density **	0.284 lbm/in^3							

* ASME Code, Section II, Part D [10].

** Metallic Materials Specification Handbook [12].

*** ASME Code Cases, Nuclear Components, NC-71-17, Tables 1, 2, 3, 4, and 5 for material thickness ≤ 4 inches [13].

Table 3.3-9 Mechanical Properties of SA-350/A-350, Grade LF 2, Class 1 Low Alloy Steel

Property	**Value**					
Temperature (ºF)	70	200	300	400	500	700
Ultimate strength, S_u, (ksi) *	70.0	70.0	70.0	70.0	70.0	70.0
Yield strength, S_y (ksi) *	36.0	32.8	31.9	30.8	29.1	25.9
Design Stress Intensity, S_m (ksi) *	23.3	21.9	21.3	20.6	19.4	17.3
Modulus of Elasticity, E, (× 10^3 ksi) *	29.2	28.5	28.0	27.4	27.0	25.3
Coefficient of Thermal Expansion α (× 10^{-6} in/in/°F) *	—	5.89	6.26	6.61	6.91	7.41
Alternating Stress at 10^6 cycles (ksi) **	12.5	12.2	11.9	11.7	11.5	10.8
Alternating Stress at 10 cycles (ksi) **	580.0	566.0	556.1	544.2	536.3	502.5
Poisson's Ratio *	0.31					
Density *	0.279 lbm/in^3					

* ASME Code, Section II, Part D [10].

** ASME Code, Appendix I [11].

Table 3.3-10 Mechanical Properties of SA-193, Grade B6, High Alloy Steel Bolting Material

Property	**Value**							
Temperature (°F)	-40	-20	70	200	300	400	500	600
Ultimate Stress, S_u (ksi) *, ***	No Value Given	110.0	110.0	104.9	101.5	98.3	95.6	92.9
Yield Stress, S_y (ksi) *, ***	No Value Given	85.0	85.0	81.1	78.1	76.0	73.9	71.8
Design Stress Intensity, S_m (ksi) *	28.3	28.3	28.3	27.0	26.1	25.3	24.6	23.9
Modulus of Elasticity, E (ksi) *	30.1E+ 03	30.1E+ 03	29.2E+ 03	28.5E+ 03	27.9E+ 03	27.3E+ 03	26.7E+ 03	26.1E+03
Alternating Stress @ 10 cycles (ksi) **	1104.4	1100.0	1085.0	1058.0	1035.0	1015.0	989.0	935.3
Alternating Stress @ 10^6 cycles (ksi) **	13.0	12.9	12.7	12.4	12.2	11.9	11.6	11.0
Coefficient of Thermal Expansion, α (in/in/°F) *	5.73E-06	5.76E-06	5.92E-06	6.15E-06	6.30E-06	6.40E-06	6.48E-06	6.53E-06
Poisson's Ratio *	← 0.31 →							
Density *	← 503 lbm/ft³(0.291 lbm/in³) →							

* ASME Code, Section II, Part D [10].

** ASME Code, Appendix I [11].

*** Calculated based on Design Stress Intensity:

$$\left(\frac{S_{m-temp}}{S_{m\,70^\circ}}\right)S_{u70} = S_{u-temp}$$

.

Table 3.3-11 Mechanical Properties of 6061-T651 Aluminum Alloy

Property	Value								
Temperature (ºF)	70	100	200	300	400	500	600	700	750
Ultimate strength, S_u (ksi) **	42.0	40.7	38.2	31.5	17.2	6.7	3.4	2.1	--
Yield strength, Sy (ksi) **	35.0	33.9	32.2	26.9	14.0	5.3	2.5	1.4	1.4
Design Stress Intensity S_m (ksi) *	10.5	10.5	10.5	8.4	4.4	--	--	--	--
Modulus of Elasticity, E ($\times 10^3$ ksi) *	10.0	9.9	9.6	9.2	8.7	8.1	7.0	--	--
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/°F) *	—	12.6	12.91	13.22	13.52	13.7	14.3	--	--
Poisson's Ratio *	0.33								
Density *	0.098 lbm/in^3								

* ASME Code, Section II, Part D [10].

** Military Handbook MIL-HDBK-5G [15].

Table 3.3-12 Mechanical Properties of Concrete

Property	Value					
Temperature (ºF)	70	100	200	300	400	500
Compressive Strength (psi) *	4000	4000	4000	3800	3600	3400
Modulus of Elasticity, ($\times 10^3$ ksi) *	—	3.64	3.38	3.09	3.73	3.43
Coefficient of Thermal Expansion, α ($\times 10^{-6}$ in/in/ºF) *	5.5					
Density *	140 lbm/ft^3					

* Handbook of Concrete Engineering [16].

Table 3.3-13 Mechanical Properties of NS-4-FR and NS-3

NS-4-FR	**Temperature (°F)**			
Property (units) *	**86**	**158**	**212**	**302**
Coefficient of Thermal Expansion (in/in/°F)	2.22E-5	4.72E-5	5.88E-5	5.74E-5
Compressive Modulus of Elasticity (ksi)	561			
Density (lbm/in^3)	0.0607			

NS-3	
Property (units) *	**Value**
Coefficient of Thermal Expansion (in/in/°F) at 150°F	7.78×10^{-6}
Compressive Modulus of Elasticity (ksi)	163
Density (lbm/in^3)	0.0636

* GESC Product Data [17].

Table 3.3-14 Mechanical Properties of SA-516, Grade 70 Carbon Steel

Property	Value						
Temperature (°F)	70	200	300	400	500	700	800
Ultimate Tensile Stress S_u (ksi) *	70.0	70.0	70.0	70.0	70.0	70.0	64.3
Yield Stress, S_y (ksi) *	38.0	34.6	33.7	32.6	30.7	27.4	25.3
Design Stress Intensity, S_m (ksi) *	23.3	23.1	22.5	21.7	20.5	18.3	—
Modulus of Elasticity (ksi) *	29.5E+3	28.8E+3	28.3E+3	27.7E+3	27.3E+3	25.5E+3	24.2E+3
Alternating Stress @ 10 cycles (ksi) **	580.0	552.8	543.0	531.5	523.7	477.0	—
Alternating Stress @ 10^6 cycles (ksi) **	12.5	11.9	11.7	11.5	11.3	10.3	—
Coefficient of Thermal Expansion, α (in/in/ °F) *	—	5.89E-6	6.26E-6	6.61E-6	6.91E-6	7.41E-6	7.59E-6
Thermal Conductivity (BTU/hr-in°F) *	1.9	2.0	2.0	2.0	2.0	1.9	1.8
Poisson's Ratio*	0.31						
Density*	482 lbm/ft^3 (0.279 lbm/in^3)						

* ASME Code, Section II, Part D [10].

** ASME Code, Appendix I [11].

3.3.2 Fracture Toughness Considerations

The primary structural materials of the NAC-UMS® Transportable Storage Canister and basket are a series of stainless steels. These stainless steel materials do not undergo a ductile-to-brittle transition in the temperature range of interest for the NAC-UMS® System. Therefore, fracture toughness is not a concern for these materials.

The optional lift anchors for the NAC-UMS® Vertical Concrete Cask are fabricated from A-537, Class 2, and A-706 ferritic steels. Since there are eight rebars (A-706) for each lift anchor, the rebars are not considered fracture-critical components because multiple, redundant load paths exist, in the same manner that bolted systems are considered in Section 5 of NUREG/CR-1815. Therefore, brittle fracture evaluation of the rebar material is not required. The lifting lug and base plate of the lift anchors are designed as 2-inch thick, A 537 Class 2, steel plates in accordance with ANSI N14.6. Applying the fracture toughness requirements of ASME Code Section III, Subsection NF-2311(b)13 and Figure NF-2311(b)-1, the minimum allowable design metal temperature is –5°F (Curve D, 2-inch nominal thickness). The Vertical Concrete Cask lift anchors are restricted to be used only when the surrounding air temperatures are greater than, or equal to, 0°F (Section 12(B 3.4)(9)), so impact testing of the material is not required.

The NAC-UMS® BWR basket support disks are 0.625-inch thick, SA 533, Type B, Class 2, ferritic steel plate. Per ASME Code Section III, Subsection NG-2311(a)(1), impact testing of material with a nominal section thickness of 5/8 inch (16 mm) and less is not required. The limitation of the plate thickness for testing is consistent with the limitation of the minimum specimen thickness to force the results of the impact testing to correspond to plane strain conditions. Specimen thicknesses are selected to permit the conditions adjacent to the crack tip to correspond to plane strain conditions (Section 2.10.2 in Reference [42]). A graph of component stress ratios versus dimensionless lengths (Reference [42], Figure 2.38) shows that for a plate thickness of 5/8 inch, the stress state in front of the crack tip at a distance of 1/16 inch is essentially a plane stress condition. Plane stress conditions result in deviatory stress components that are significantly larger than for a plane strain condition. Larger deviatory stress components increase the plasticity zone, thereby eliminating the ability of the plate to experience conditions of brittle fracture.

3.4 General Standards

3.4.1 Chemical and Galvanic Reactions

The materials used in the fabrication and operation of the Universal Storage System are evaluated to determine whether chemical, galvanic or other reactions among the materials, contents, and environments can occur. All phases of operation — loading, unloading, handling, and storage — are considered for the environments that may be encountered under normal, off-normal, or accident conditions. Based on the evaluation, no potential reactions that could adversely affect the overall integrity of the vertical concrete cask, the fuel basket, the transportable storage canister or the structural integrity and retrievability of the fuel from the canister have been identified. The evaluation conforms to the guidelines of NRC Bulletin 96-04 [18].

3.4.1.1 Component Operating Environment

Most of the component materials of the Universal Storage System are exposed to two typical operating environments: 1) an open canister containing fuel pool water or borated water with a pH of 4.5 and spent fuel or other radioactive material; or 2) a sealed canister containing helium, but with external environments that include air, rain water/snow/ice, and marine (salty) water/air. Each category of canister component materials is evaluated for potential reactions in each of the operating environments to which those materials are exposed. These environments may occur during fuel loading or unloading, handling or storage, and include normal, off-normal, and accident conditions.

The long-term environment to which the canister's internal components are exposed is dry helium. Both moisture and oxygen are removed prior to sealing the canister. The helium displaces the oxygen in the canister, effectively precluding chemical corrosion. Galvanic corrosion between dissimilar metals in electrical contact is also inhibited by the dry environment inside the sealed canister. NAC's operating procedures provide two helium backfill cycles in series separated by a vacuum-drying cycle during the preparation of the canister for storage. Therefore, the sealed canister cavity is effectively dry and galvanic corrosion is precluded.

The control element assembly, thimble plugs and nonfuel components—including start-up sources and instrument segments—are nonreactive with the fuel assembly. By design, the control components and nonfuel components are inserted in the guide tubes of a fuel assembly. During reactor operation, the control and nonfuel components are immersed in acidic water having a high flow rate and are exposed to significantly higher neutron flux, radiation and pressure than will exist in dry storage. The control and nonfuel components are physically placed in storage in a dry, inert atmosphere in the same configuration as when used in the reactor. Therefore, there are no adverse reactions, such as gas generation, galvanic or chemical reactions or corrosion, since these components are nonreactive with the zirconium alloy guide tubes and fuel rods. There are no aluminum or carbon steel parts, and no gas generation or corrosion occurs during prolonged water immersion (20 – 40 years). Thus, no adverse reactions occur with the control and nonfuel components over prolonged periods of dry storage.

3.4.1.2 Component Material Categories

The component materials are categorized in this section for their chemical and galvanic corrosion potential on the basis of similarity of physical and chemical properties and component functions. The categories are stainless steels, nonferrous metals, carbon steel, coatings, concrete, and criticality control materials. The evaluation is based on the environment to which these categories could be exposed during operation or use of the canister.

The canister component materials are not reactive among themselves, with the canister's contents, nor with the canister's operating environments during any phase of normal, off-normal, or accident condition, loading, unloading, handling, or storage operations. Since no reactions will occur, no gases or other corrosion by-products will be generated.

The control component and nonfuel component materials are those that are typically used in the fabrication of fuel assemblies, i.e., stainless steels, Inconel 625, and zirconium alloy, so no adverse reactions occur in the inert atmosphere that exists in storage. The control element assembly, thimble plugs and nonfuel components—including start-up sources or instrument segments to be inserted into a fuel assembly—are nonreactive among themselves, with the fuel assembly, or with the canister's operating environment for any storage condition.

3.4.1.2.1 Stainless Steels

No reaction of the canister component stainless steels is expected in any environment except for the marine environment, where chloride-containing salt spray could potentially initiate pitting of the steels if the chlorides are allowed to concentrate and stay wet for extended periods of time (weeks). Only the external canister surface could be so exposed. The corrosion rate will, however, be so low that no detectable corrosion products or gases will be generated. The Universal Storage System has smooth external surfaces to minimize the collection of such materials as salts.

Galvanic corrosion between the various types of stainless steels does not occur because there is no effective electrochemical potential difference between these metals. No coatings are applied to the stainless steels. An electrochemical potential difference does exist between austenitic (300 series) stainless steel and aluminum. However, the stainless steel becomes relatively cathodic and is protected by the aluminum.

The canister confinement boundary uses Type 304L stainless steel for all components, except the shield lid, which is made of Type 304 stainless steel. Type 304L resists chromium-carbide precipitation at the grain boundaries during welding and assures that degradation from intergranular stress corrosion will not be a concern over the life of the canister. Fabrication specifications control the maximum interpass temperature for austenitic steel welds to less than 350°F. The material will not be heated to a temperature above 800°F, other than by welding thermal cutting. Minor sensitization of Type 304 stainless steel that may occur during welding will not affect the material performance over the design life because the storage environment is relatively mild.

Based on the foregoing discussion, no potential reactions associated with the stainless steel canister or basket components are expected to occur.

3.4.1.2.2 Nonferrous Metals

Aluminum is used as a heat transfer component in the Universal Storage System spent fuel basket, and aluminum components in electrical contact with austenitic stainless steel could experience corrosion driven by electrochemical Electromotive Force (EMF) when immersed in water. The conductivity of the water is the dominant factor. BWR fuel pool water is demineralized and is not sufficiently conductive to promote detectable corrosion for these metal couples. PWR pool water, however, does provide a conductive medium. The only aluminum components that will be in contact with stainless steel and exposed to the pool water are the alloy 6061-T651 heat transfer disks in the fuel basket.

Aluminum produces a thin surface film of oxidation that effectively inhibits further oxidation of the aluminum surface. This oxide layer adheres tightly to the base metal and does not react readily with the materials or environments to which the fuel basket will be exposed. The volume of the aluminum oxide does not increase significantly over time. Thus, binding due to corrosion product build-up during future removal of spent fuel assemblies is not a concern. The borated water in a PWR fuel pool is an oxidizing-type acid with a pH on the order of 4.5. However, aluminum is generally passive in pH ranges down to about 4 [19]. Data provided by the Aluminum Association [20] shows that aluminum alloys are resistant to aqueous solutions (1-15%) of boric acid (at 140°F). Based on these considerations and the very short exposure of the aluminum in the fuel basket to the borated water, oxidation of the aluminum is not likely to occur beyond the formation of a thin surface film. No observable degradation of aluminum components is expected as a result of exposure to BWR or PWR pool water at temperatures up to 200°F, which is higher than the permissible fuel pool water temperature.

Aluminum is high on the electromotive potential table, and it becomes anodic when in electrical contact with stainless or carbon steel in the presence of water. BWR pool water is demineralized and is not sufficiently conductive to promote detectable corrosion for these metal couples. PWR pool water is sufficiently conductive to allow galvanic activity to begin. However, exposure time of the aluminum components to the PWR pool environment is short. The long-term storage environment is sufficiently dry to inhibit galvanic corrosion.

From the foregoing discussion, it is concluded that the initial surface oxidation of the aluminum component surfaces effectively inhibits any potential galvanic reactions.

Heat transfer disks fabricated from 6061-T651 aluminum alloy are used in the NAC-UMS® Universal Storage System PWR and BWR fuel baskets to augment heat transfer from the spent fuel through the basket structure to the canister exterior. Vendor and Nuclear Regulatory Commission safety evaluations of the NUHOMS Dry Spent Fuel Storage System (Docket No. 72-1004) have concluded that combustible gases, primarily hydrogen, may be produced by a chemical reaction and/or radiolysis when aluminum or aluminum flame-sprayed components are immersed in spent fuel pool water. The evaluations further concluded that it is possible, at higher temperatures (above 150 - 160°F), for the aluminum/water reaction to produce a hydrogen concentration in the canister that approaches or exceeds the Lower Flammability Limit (LFL) for hydrogen of 4 percent. The NRC Inspection Reports No. 50-266/96005 and 50-301/96005 dated July 01, 1996, for the Point Beach Nuclear Plant concluded that hydrogen generation by radiolysis was insignificant relative to other sources.

Thus, it is reasonable to conclude that small amounts of combustible gases, primarily hydrogen, may be produced during UMS® Storage System canister loading or unloading operations as a result of a chemical reaction between the 6061-T6 aluminum heat transfer disks in the fuel basket and the spent fuel pool water. The generation of combustible gases stops when the water is removed from the cask or canister and the aluminum surfaces are dry.

A galvanic reaction may occur at the contact surfaces between the aluminum disks and the stainless steel tie rods and spacers in the presence of an electrolyte, like the pool water. The galvanic reaction ceases when the electrolyte is removed. Each metal has some tendency to ionize, or release electrons. An EMF associated with this release of electrons is generated between two dissimilar metals in an electrolytic solution. The EMF between aluminum and stainless steel is small and the amount of corrosion is directly proportional to the EMF. Loading operations generally take less than 24 hours, a large portion of which has the canister immersed in and open to the pool water after which the electrolyte (water) is drained and the cask or canister is dried and back-filled with helium, effectively halting any galvanic reaction.

The potential chemical or galvanic reactions do not have a significant detrimental effect on the ability of the aluminum heat transfer disks to perform their function for all normal and accident conditions associated with dry storage.

Loading Operations

After the canister is removed from the pool and during canister closure operations, an air space is created inside the canister beneath the shield lid by the drain-down of the water in the canister so that the shield-lid-to-canister-shell weld can be performed. The resulting air space is at least 3 inches in depth. As there is some clearance between the inside diameter of the canister shell and the outside diameter of the shield lid, it is possible that gases released from a chemical reaction inside the canister could accumulate beneath the shield lid. A bare aluminum surface oxidizes when exposed to air, reacts chemically in an aqueous solution, and may react galvanically when in contact with stainless steel in the presence of an aqueous solution.

The reaction of aluminum in water, which results in hydrogen generation, proceeds as:

$$2\ Al + 3\ H_20 \Rightarrow Al_2O_3 + 3\ H_2$$

The aluminum oxide (Al_2O_3) produces the dull, light gray film that is present on the surface of bare aluminum when it reacts with the oxygen in air or water. The formation of the thin oxide film is a self limiting reaction as the film isolates the aluminum metal from the oxygen source acting as a barrier to further oxidation. The oxide film is stable in pH neutral (passive) solutions, but is soluble in borated PWR spent fuel pool water. The oxide film dissolves at a rate dependent upon the pH of the water, the exposure time of the aluminum in the water, and the temperatures of the aluminum and water.

PWR spent fuel pool water is a boric acid and demineralized water solution. BWR spent fuel pool water does not contain boron and typically has a neutral pH (approximately 7.0). The pH, water chemistry, and water temperature vary from pool to pool. Since the reaction rate is largely dependent upon these variables, it may vary considerably from pool to pool. Thus, the generation rate of combustible gas (hydrogen) that could be considered representative of spent fuel pools in general is very difficult to accurately calculate, but the reaction rate would be less in the neutral pH BWR pool.

The BWR basket configuration incorporates carbon steel support plates that are coated with electroless nickel. The coating protects the carbon steel during the comparatively short time that the canister is immersed in, or contains, water. The coating is described in Section 3.8.3. The coating is non-reactive with the BWR pool water and does not off-gas or generate gases as a result of contact with the pool water. Consequently, there are no flammable gases that are generated by the coating. A coating is not used in PWR basket configurations.

To ensure the safe loading and unloading of the UMS® transportable storage canister, the loading and unloading procedures detailed in Chapter 8 provide for the monitoring for hydrogen gas from before initiating shield lid welding operations through completion of the root pass of the shield lid-to-shell weld. The monitoring system shall be capable of detecting hydrogen concentrations at < 60% of the lower flammability limit of hydrogen (4%), i.e., H_2 concentration of 2.4%. The hydrogen detector will be connected to the vent port opening to allow the detector to draw gas samples from the free volume below the shield lid. The detector shall be operated to verify acceptable flammable gas levels (i.e., < 2.4%) prior to initiation of the weld through completion of the root pass. If H_2 levels are detected in concentrations equal to or greater than 2.4%, welding operations shall be immediately suspended until the hydrogen concentrations are returned to acceptable levels. When H_2 concentrations exceed the limit, the free volume under the lid will be either evacuated by the vacuum pump on the Vacuum Drying System (VDS), thereby drawing in ambient air into the volume through the shield lid to shell weld gap, or gas can be flushed through the free volume through the vent port.

Upon completion of the root pass, the hydrogen detector can be disconnected from the vent as any possible ignition source is isolated from the cavity free volume. The cavity will continue to be vented to atmosphere through the vent port. Following completion of the shield lid welding and examinations, the canister is drained, dried and backfilled with helium. Once the canister is drained and dry, the source of combustible gas production is removed.

The vacuum pump shall exhaust to a system or area where hydrogen flammability is not an issue. Once the root pass weld is completed, there is no further likelihood of a combustible gas burn because the ignition source is isolated from the combustible gas. Once welding of the shield lid has been completed, the canister is drained, vacuum dried and backfilled with helium.

No hydrogen is expected to be detected prior to, or during, the welding operations. During the completion of the shield lid to canister shell root pass, the hydrogen gas detector is attached to the vent port and continuously operates. During operation, the detector maintains a negative pressure in the canister, drawing air into the canister at the circumference of the shield lid. This ensures that hydrogen gas does not enter the weld area. The mating surfaces of the support ring and inner lid are machined to provide a good level fit-up, but are not machined to provide a metal-to-metal seal. Consequently, additional exit paths for the combustible gases exist at the circumference of the shield lid. Once the canister is dry, no combustible gases form within the canister.

Unloading Operations

It is not expected that the canister will contain a measurable quantity of combustible gases during the time period of storage. The canister is vacuum dried and backfilled with helium immediately prior to being welded closed. There are only minor mechanisms by which hydrogen is generated after the canister is dried and sealed.

As shown in Section 8.3, the principal steps in opening the canister are the removal of the structural lid, the removal of the vent and drain port covers, and the removal of the shield lid. These steps are expected to be performed by cutting or grinding. The design of the canister precludes monitoring for the presence of combustible gases prior to the removal of the structural lid and the vent or drain port covers. Following removal of the vent port cover, a vent line is connected to the vent port quick disconnect. The vent line incorporates a hydrogen gas detector which is capable of detecting hydrogen at a concentration of 2.4% (60% of its lower flammability limit of 4%). The pressurized gases (expected to be greater than 96% helium) in the canister are expected to carry combustible gases out of the vent port. If the exiting gases in the vent line contain no hydrogen at concentrations above 2.4%, the drain port cover weld is cut and the cover removed. If levels of hydrogen gas above 2.4% concentration are detected in the vent line, then the vacuum system is used to remove all residual gas prior to removal of the drain port cover. During the removal of the drain port cover, the hydrogen gas detector is attached to the vent port to ensure that the hydrogen gas concentration remains below 2.4%. Following removal of the drain port cover, the canister is filled with water using the vent and drain ports. Prior to cutting the shield lid weld, 70 gallons of water are removed from the canister to permit the removal of the shield lid. Monitoring for hydrogen would then proceed as described for the loading operations.

3.4.1.2.3 Carbon Steel

Carbon steel support disks are used in the BWR basket configuration. There is a small electrochemical potential difference between carbon steel (SA-533) and aluminum and stainless steel. When in contact in water, these materials exhibit limited electrochemically-driven corrosion. BWR pool water is demineralized and is not sufficiently conductive to promote detectable corrosion for these metal couples. In addition, the carbon steel support disks are coated with electroless nickel to protect the carbon steel surface during exposure to air or to spent fuel pool water, further reducing the possibility of corrosion. Once the canister is loaded, the water is drained from the cavity, the air is evacuated, and the canister is backfilled with helium and sealed. Removal of the water and the moisture eliminates the catalyst for galvanic corrosion. The canister operating procedures (see Chapter 8) provide two backfill cycles in series separated by a vacuum drying cycle during closing of the canister. The displacement of oxygen by helium effectively inhibits corrosion.

The transfer cask structural components are fabricated primarily from ASTM A-588 and A-36 carbon steel. The exposed carbon steel components are coated with either Keeler & Long E-

Series Epoxy Enamel or Carboline 890 to protect the components during in-pool use and to provide a smooth surface to facilitate decontamination.

The concrete shell of the vertical concrete cask contains an ASTM A36 carbon steel liner, as well as other carbon steel components. The exposed surfaces of the base of the concrete cask and the liner are coated with Keeler & Long Y-1-Series Acrylic Urethane Enamel or PPG METALHIDE® 97-694 Series Primer or PPG DIMETCOTE® 9 Primer and PPG PITT-THERM® 97-724 Series Top Coating to provide protection from weather-related moisture and direct sunlight.

No potential reactions associated with the BWR basket carbon steel disks, the transfer cask components or vertical concrete cask components are expected to occur.

3.4.1.2.4 Coatings

The exposed carbon steel surfaces of the transfer cask and the transfer cask adapter plate are coated with either Carboline 890 or Keeler & Long E-Series Epoxy Enamel. The technical specifications for these coatings are provided in Sections 3.8.1 and 3.8.2, respectively. These coatings are approved for Nuclear Service Level 2 use. Load bearing surfaces (i.e., the bottom surface of the trunnions and the contact surfaces of the transfer cask doors and rails) are not painted, but are coated with an appropriate nuclear grade lubricant, such as Neolube®. The exposed metal surfaces of the vertical concrete cask are coated with Keeler & Long Kolor-Poxy Primer No. 3200 and Acrythane Enamel Y-1 Series top coating or PPG METALHIDE® 97-694 Series Primer or PPG DIMETCOTE® 9 Primer and PPG PITT-THERM® 97-724 Series top coating. The technical specifications for these coatings are provided in Sections 3.8.4, 3.8.5, 3.8.6 and 3.8.7, respectively.

Carbon steel support disks used in the BWR canister basket are coated with electroless nickel. The coating is applied in accordance with ASTM B733-SC3, Type V, Class 1[37]. As described in Section 3.8.3, the electroless nickel coating process uses a chemical reducing agent in a hot aqueous solution to deposit nickel on a catalytic surface. The deposited nickel coating is a hard alloy of uniform thickness of 25 μm (0.001 inch), containing from 4% to 12% phosphorus. Following its application, the nickel coating combines with oxygen in the air to form a passive oxide layer that effectively eliminates free electrons on the surface that would be available to cathodically react with water to produce hydrogen gas. Consequently, the production of hydrogen gas in sufficient quantities to facilitate combustion is highly unlikely.

3.4.1.2.5 Concrete

The vertical concrete storage cask is fabricated of 4000 psi, Type 2 Portland cement that is reinforced with vertical and circumferential carbon steel rebar. Quality control of the proportioning, mixing, and placing of the concrete, in accordance with the NAC fabrication specification, will make the concrete highly resistant to water. The concrete shell is not expected to experience corrosion, or significant degradation from the storage environment through the life of the cask.

3.4.1.2.6 Criticality Control Material

The criticality control material is boron carbide mixed in an aluminum alloy matrix. Sheets of this material are affixed to one or more sides of the designated fuel tubes and enclosed by a welded stainless steel sheet. The material resists corrosion similar to aluminum, and is protected by an oxide layer that forms shortly after fabrication and inhibits further interaction with the stainless steel. Consequently, no potential reactions associated with the aluminum-based criticality control material are expected.

3.4.1.2.7 Neutron Shielding Material

The neutron shielding materials, NS-3 and NS-4-FR, consist primarily of aluminum, carbon, oxygen and hydrogen. NS-4-FR is used in the transfer cask and either NS-3 or NS-4-FR may be used in the shield plug of the vertical concrete storage cask to provide radiation shielding. The acceptable performance of the materials has been demonstrated by use and testing. The materials have been used for over 10 years in licensed storage casks in the United States and in licensed casks in Japan, Spain and the United Kingdom. There are no reports that the shielding effectiveness of the materials has degraded in these applications, demonstrating the long-term reliability for the purpose of shielding neutrons from personnel and the environment. There are no potential reactions associated with the polymer structure of the materials and the stainless steel or carbon steel in which it is encapsulated during use.

The chemistry of the materials (e.g., the way the elements are bonded to one another) contributes significantly to the fire-retardant capability. Approximately 90% of the off-gassing that does occur consists of water vapor.

The thermal performance of NS-4-FR has been demonstrated by long-term functional stability tests of the material at temperatures from -40°F to 338°F. These tests included specimens open to the atmosphere and enclosed in a cavity at both constant and cyclic thermal loads. The tests evaluated material loss though off-gassing and material degradation. The results of the tests demonstrate that, in the temperature range of interest, the NS-4-FR does not exhibit loss of material by off-gassing, does not generate any significant gases, and does not suffer degradation or embrittlement. Further, the tests demonstrated that encased material, as it is used in the NAC-UMS®, performed significantly better than exposed material. Consequently, the formation of flammable gases is not a concern.

Radiation exposure testing of NS-4-FR in reactor pool water demonstrated no physical deterioration of the material and no significant loss of hydrogen (less than 1%). The tests also demonstrated that the NS-4-FR retains its neutron shield capability over the cask's 50-year design life with substantial margin. The radiation testing has shown that detrimental embrittlement and loss of hydrogen from the material do not occur at dose rates (9×10^{14} n/cm^2) that exceed those that would occur assuming the continuous storage of design basis fuel for a 50-year life (estimated to be 1.7×10^{12} cm^2/yr). Consequently, detrimental deterioration or embrittlement due to radiation flux does not occur.

Since the NS-4-FR in the NAC-UMS® transfer cask is sandwiched between the shell and the lead shield and enclosed within a welded steel shell where the shell seams are welded to top and bottom plates with full penetration or fillet welds, it will maintain its form over the expected lifetime of the transfer cask's radiation exposure. The material's placement between the lead shield and the outer shell does not allow the material to redistribute within the annulus.

The NS-3 and NS-4-FR shield material is similarly enclosed in the storage cask shield plug, since a disk of NS-3 or NS-4-FR is captured in a cavity formed by a carbon steel ring and two carbon steel plates. This material cannot redistribute within this volume.

3.4.1.3 General Effects of Identified Reactions

No potential chemical, galvanic, or other reactions have been identified for the Universal Storage System. Therefore, no adverse conditions, such as the generation of flammable or explosive quantities of combustible gases or an increase in neutron multiplication in the fuel (criticality) because of boron precipitation, can result during any phase of canister operations for normal, off-normal, or accident conditions.

3.4.1.4 Adequacy of the Canister Operating Procedures

Based on this evaluation, which results in no identified reactions, it is concluded that the Universal Storage System operating controls and procedures presented in Chapter 8.0 are adequate to minimize the occurrence of hazardous conditions.

3.4.1.5 Effects of Reaction Products

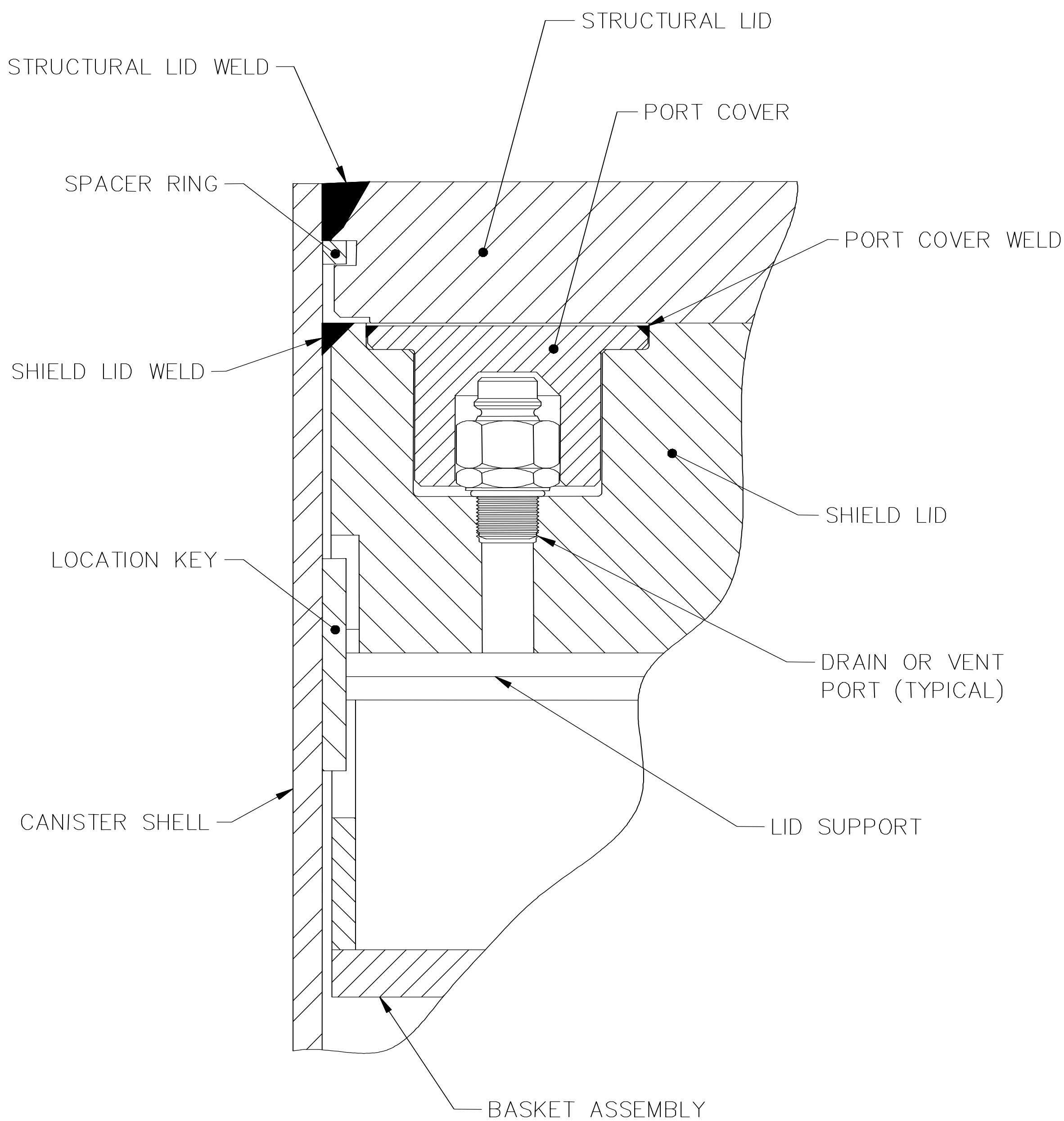
No potential chemical, galvanic, or other reactions have been identified for the Universal Storage System. Therefore, the overall integrity of the canister and the structural integrity and retrievability of the spent fuel are not adversely affected for any operations throughout the design basis life of the canister. Based on the evaluation, no change in the canister or fuel cladding thermal properties is expected, and no corrosion of mechanical surfaces is anticipated. No change in basket clearances or degradation of any safety components, either directly or indirectly, is likely to occur since no potential reactions have been identified.

3.4.2 Positive Closure

The Universal Storage System employs a positive closure system composed of multi-pass welds to join the canister shield lid and the canister structural lid to the shell. The penetrations to the canister cavity through the shield lid are sealed by welded port covers. The welded canister closure system (see Figure 3.4.2-1) precludes the possibility of inadvertent opening of the canister.

The top of the vertical concrete cask is closed by a bolted lid that weighs approximately 2,500 lbs. The weight of the lid, its inaccessibility, and the presence of the bolts effectively preclude inadvertent opening of the lid.

Figure 3.4.2-1 Universal Storage System Welded Canister Closure



3.4.3 Lifting Devices

The UMS® is designed to allow for efficient and safe handling of the system's components at cask user facilities using various lifting and handling equipment. The transfer cask is handled by a lift yoke attached to the two lifting trunnions. The canister is handled by a suitable lifting system, such as slings and hoist rings, attached to threaded holes in the top of the structural lid. The concrete cask can be lifted and moved by the use of jacks and air pads installed under the inlets or by a vertical cask hauler connected to the optional lifting lugs.

The designs of the UMS® Universal Storage System and Universal Transport System components address the concerns identified in U.S. NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment" (April 11, 1996) as follows:

(1) The UMS® lifting and handling components satisfy the requirements of NUREG-0612 and ANSI N14.6 for safety factors on redundant or nonredundant load paths as described in this chapter.

(2) Transfer or transport cask lifting in the spent fuel pool or cask loading pit or transfer or transport cask lifting and movement above the spent fuel pool operating floor will be addressed on a plant-specific basis.

The transfer cask is provided in either the Standard configuration for canisters weighing up to 88,000 lbs or in the Advanced configuration for canisters weighing up to 98,000 lbs. The two configurations have identical operating features. The transfer casks are lifted by trunnions located near the top of each cask. The Standard transfer cask trunnions are attached by full-penetration welds to both the inner and the outer shells (Figure 3.4.3-1). The Advanced transfer cask trunnions are similarly attached, but incorporate a trunnion support plate at each trunnion for the additional load. The transfer casks are each designed as a heavy-lifting device that satisfies the requirements of NUREG-0612 and ANSI N14.6 for lifting the fully loaded canister of fuel and water, together with the shield lid, which is the maximum weight of the transfer cask during a lifting operation with a given configuration.

The transportable storage canister remains within the transfer cask during all preparation, loading, canister closure, and transfer operations. The canister is lifted using two redundant sets of lifting slings and hoist rings. The hoist rings thread into the structural lid to lift the loaded canister and to lower it into the concrete cask after the shield doors are opened. The hoist rings, shown in Figure 3.4.3-2, are also used for any subsequent lifting of the loaded dry canister. Alternative canister lifting system designs may be utilized based on a site-specific analysis and evaluation.

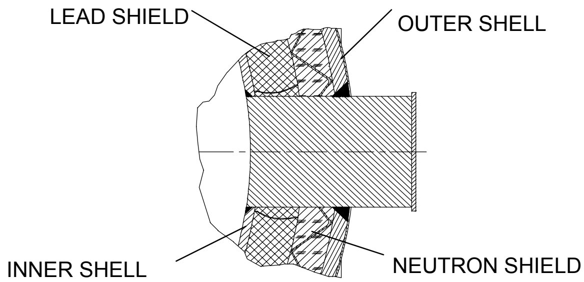
The vertical concrete cask is moved by means of a system of air pads. The cask is raised approximately 4 inches. by four lifting jacks placed at the jacking pads located near the end of each air inlet. A system consisting of 4 air pads is then inserted under the concrete cask. The cask is lowered onto the uninflated air pads, the jacks are removed, and the air pads are inflated to lift the concrete cask and position it as required on the storage pad or transport vehicle. When positioning is complete, the jacks are used to support the cask as the air pads are removed.

As an option, the loaded concrete cask may also be lifted and moved using lifting lugs at the top of the cask. The top lifting lugs are described in Section 3.4.3.1.3.

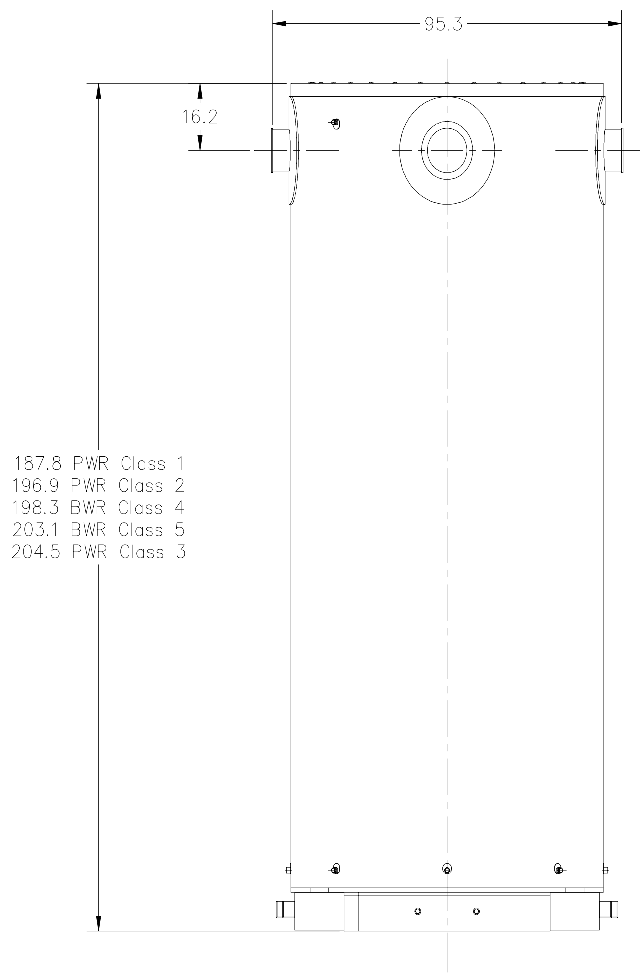
The structural evaluations in this section consider the bounding conditions for each aspect of the analysis. Generally, the bounding condition for lifting devices is represented by the heaviest component, or combination of components, of each configuration. The bounding conditions used in this section are:

Section	Evaluation	Bounding Condition	Configuration
3.4.3.1	Concrete Cask Lifting Jacks	Heaviest loaded Concrete Cask + 10% dynamic load factor	BWR Class 5
	Pedestal Loading	Heaviest loaded Canister + 10% dynamic load factor	BWR Class 5
	Concrete Cask Air Pads (Lifting)	Heaviest loaded Concrete Cask	BWR Class 5
	Concrete Cask Top Lifting Lugs (Lifting)	Heaviest loaded Concrete Cask + 10% dynamic load factor	BWR Class 5
3.4.3.2	Canister Lift	Heaviest loaded Canister + 10% dynamic load factor	BWR Class 5
3.4.3.3	Standard Transfer Cask Lift	Heaviest loaded Transfer Cask + 10% dynamic load factor	BWR Class 5
3.4.3.3.4	Standard Transfer Cask Shield Doors and Rails	Heaviest loaded Canister + water, shield doors and 10% dynamic load factor	BWR Class 5

Figure 3.4.3-1 Standard Transfer Cask Lifting Trunnion

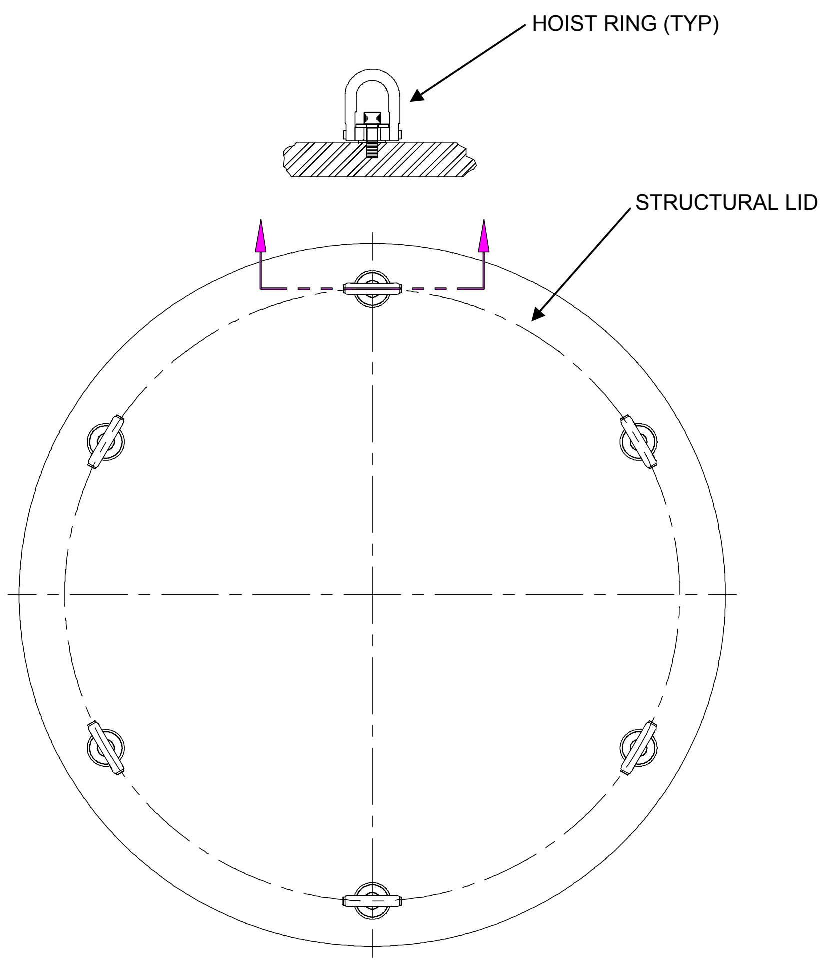


TRUNNION REGION DETAIL



Dimensions in inches

Figure 3.4.3-2 Canister Hoist Ring Design



3.4.3.1 Vertical Concrete Cask Lift Evaluation

The vertical concrete cask may be lifted and moved using an air pad system under the base of the cask or four lifting lugs provided at the top of the cask.

Lifting jacks installed at jacking points in the air inlet channels are used to raise the cask so that the air pads can be inserted under the cask. The lifting jacks use a synchronous lifting system to equally distribute the hydraulic pressure among four hydraulic jack cylinders. The calculated weight of the heaviest, loaded concrete cask to be lifted by the jacking system, the BWR Class 5 configuration, is 323,900 pounds with loaded canister and lids (center of gravity is measured from the bottom of the concrete cask). A bounding weight of 330,000 pounds is used for the evaluation in this section.

The lifting lugs are analyzed in accordance with ANSI N14.6 and ACI-349.

3.4.3.1.1 Bottom Lift By Hydraulic Jack

To ensure that the concrete bearing stress at the jack locations due to lifting the cask does not exceed the allowable stress, the area of the surface needed to adequately spread the load is determined in this section. The allowable bearing capacity of the concrete at each jack location is:

$$U_b = \phi f_c' A = \frac{(0.7)(4{,}000)\pi d^2}{4} = 2{,}199.1\ d^2,$$

where:

ϕ = 0.7 strength reduction factor for bearing,
f_c' = 4,000 psi concrete compressive strength,
$A = \frac{\pi d^2}{4}$, concrete bearing area (d = bearing area diameter).

The concrete bearing strength must be greater than the cask weight multiplied by a load reduction factor, $L_f = 1.4$.

$$2{,}199.1\ d^2 > \frac{L_f \times W}{n} = \frac{1.4(330{,}000\ \text{lb})}{4} \Rightarrow d > 7.25\ \text{in.},$$

where:

n = the number of jacks, 4
W = the weight of the vertical concrete cask, 330,000 lb.
L_f = the load factor, 1.4

The diameter obtained in the above equation corresponds to the minimum permissible area over which the load must be distributed. The force exerted by the jack is applied through the 2.25-in. - thick steel air inlet top plate. This increases the effective diameter of the load acting on the concrete surface from a 4.125-in. diameter jack cylinder to about 8.625 in., assuming a 45° angle for the cone of influence.

The bearing stress at each jack location with a bearing area of $\frac{\pi \times 8.625^2}{4} \approx 58.4\ \text{in}^2$ is:

$$\sigma = \frac{P}{A} = \frac{(1.4)(330{,}000\ \text{lb})}{4(58.4\ \text{in}^2)} = 1{,}978\ \text{psi}$$

The allowable bearing stress is:

$$\sigma = \phi f_c = (0.7)(4{,}000\ \text{psi}) = 2{,}800\ \text{psi}$$

The Margin of Safety is:

$$MS = \frac{2{,}800}{1{,}978} - 1 = +\,0.42$$

Bottom Plate Flexure

During a bottom lift of the concrete cask, the weight of the loaded canister, the pedestal, and the air inlet system are transferred to the bottom plate. As the load is applied, the bottom plate flexes, tending to separate from the concrete. Nelson studs are used to tie the concrete to the bottom plate and prevent separation.

Thirty-two 3/4 in. diameter × 6 3/16-in. long Nelson studs are used in the concrete cask. The shear capacity of each stud is about 23.9 kips [21]. The total load capacity of the studs is:

Capacity = 32 studs × 23.86 kips/stud = 763.5 kips.

The allowable load, P_u, with a load factor of 2.0, as specified in the manufacturer's design data [21], is:

$$P_u = \frac{763.5 \text{ kips}}{2.0} = 381.8 \text{ kips}$$

The total calculated load applied to the concrete cask bottom plate is 75,600 pounds.

Loaded Canister + Pedestal Assembly = 95,000* + 11,000 = 106,000 lb

*Note a conservative value of 95,000 lb. is used for evaluation.

The total load applied to the storage cask bottom plate (including a 10% dynamic load factor) is:

106,000 × 1.1 = 116,600 lb

Therefore, the margin of safety is:

$$MS = \frac{381.8 \text{ kip}}{116.6 \text{ kip}} - 1 = +2.3$$

Base Weldment

This analysis evaluates a bounding configuration of the standard design of the pedestal support structure for static loads. The analysis conservatively assumes a loaded canister with a bounding weight of 95,000 pounds. The pedestal assembly weight is 11,000 pounds. The base plate is modeled with a thickness of 2 inches, the stand (pedestal ring) is 2 inches thick, and the baffle is 1/4 inch thick. To bound the maximum pedestal weight, the densities of the base plate and baffle are increased to simulate a 4-inch plate and 2-inch plate, respectively.

A half-symmetry model of the base weldment (pedestal) is built using the ANSYS preprocessor (see Figure 3.4.3.1-1). The model is constructed of 8-node brick elements (SOLID45). Symmetry conditions (UY=0) are applied along the plane of symmetry (X-Z plane). The total load is simulated by increasing the density of the base plate. The total pressure applied to the model is:

$$F = 95{,}000 \text{ lb} \times 1.1 \text{ g},$$

where, a 10% dynamic load factor is applied to account for handling loads.

To determine the baffle assembly's contribution to the support of the pedestal, gap elements (CONTAC52) are added between the upper truncated cone and the base plate. Two analyses are performed. The first assumes that a gap of 1/4 inch exists between the truncated cone and base plate. The second analysis assumes zero gap.

The following table provides a summary of maximum nodal stresses compared to the allowable stresses for SA-36 carbon steel. For conservatism, the nodal stress (membrane + bending) is compared to the membrane allowable (S_m).

Stress Location	Maximum Nodal Stress (psi)	Allowable, S_m (psi)	Margin of Safety
	1/4-inch Gap		
Pedestal Ring	10214.3	19300.0	0.89
Baffle	107.3	19300.0	>10
Base Plate	1021.4	19300.0	>10
	Zero Gap		
Pedestal Ring	8225.5	19300.0	1.35
Baffle	6283.0	19300.0	2.07
Base Plate	790.8	19300.0	>10

As shown in the table, the maximum nodal stress occurs in the pedestal ring when the gap is set to 1/4-inch and does not close. When the gap is set to zero, a portion of the load is distributed to the baffle. In all cases, the maximum nodal stress is less than the allowable.

3.4.3.1.2 Bottom Support by Air Pads

The concrete cask is supported by air pads in each of 4 quadrants during transport. The layout of the air pads (four 60 in. × 60 in. or 48 in. × 48 in. square pads) are designed to clear the air inlet locations by approximately 4 inches to allow for hydraulic jack access.

The air pad system maximum height is 6.0 in. (3-in. maximum lift, plus 3.0-in. overall height when deflated). The air pad system has a rated lift capacity of 560,000 pounds for the 60 in. × 60 in. pads and 360,000 pounds for the 48 in. × 48 in. pads. The air pads must supply sufficient force to overcome the weight of the concrete cask under full load plus a lift load factor of 1.1. The weight of the heaviest storage configuration, the BWR class 5 system, is about 313,900 pounds. The air pad evaluation uses a conservative weight of 320,000 pounds. The required lift load is 1.1 × (320,000 lb) = 352,000 pounds. Since the available lift force is greater than the load, the air pads are adequate to lift the concrete cask. Considering the minimum air pad capacity of 360,000 pounds, the lifting force margin of safety is:

MS = (360,000 / 352,000) -1 = + 0.02.

3.4.3.1.3 Top Lift By Lifting Lugs

A set of four lifting lugs is provided at the top of the vertical concrete cask so that the cask, with a loaded transportable storage canister, may be lifted from the top end. Similar to the bottom lift, the BWR Class 5 configuration maximum weight is used in the analysis of the lifting lugs.

The steel components of the lifting lugs are analyzed in accordance with ANSI N14.6. The development length of the rebar embedded in the concrete is analyzed in accordance with ACI-349-85[4].

Lifting Lug Axial Load

The maximum loaded concrete cask weight is about 324,000 pounds. A bounding weight of 325,000 pounds is used in this analysis. Assuming a 10% dynamic load factor, the load (P) on each lug is:

$$P = \frac{325{,}000(1.1)}{4} = 89{,}375 \text{ lb}$$

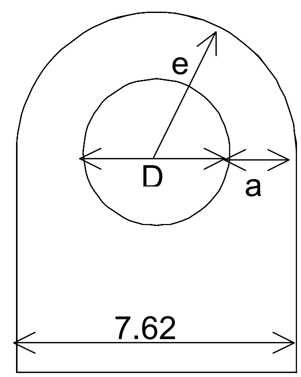
For the analysis, P is taken as 89,500 pounds. The lugs are evaluated for adequate strength under a uniform axial load in accordance with the method described in Section 9.3 of AFFDL-TR-69-42 [32].

The bearing stresses and loads for lug failure involving bearing, shear-tearout, and hoop tension are determined using an allowable load coefficient (K). Actual lug failures may involve more than one failure mode, but such interaction effects are accounted for in the value of K.

The allowable lug yield bearing stress (F_{byL}) is:

$$F_{byL} = K\frac{a}{D}(F_y) \quad \text{(for e/D < 1.5)}$$

$$= 43.13 \text{ ksi}$$



Lifting lug

where:

K = allowable axial load coefficient [32]

=1.65 for e/D = 0.94

e = 7.6/2 = 3.8 in

D = 4.063 in

e /D = 3.8/4.063 = 0.94 (< 1.5)

a $= e - \frac{D}{2}$ = 1.77 in

F_y = 60 ksi = lug yield tensile strength for ASME SA537, Class 2 carbon steel

The lug yield bearing load (P_{byL}) for lug failure in bearing, shear-out, or hoop tension is:

$$P_{byL} = F_{byL} \times D \times t$$

$$= 350.47 \text{ kips}$$

where:

t = lug thickness = 2.0 in

The lug yield load capacity (350.47 kips) divided by the lug maximum load (89.5 kips) is:

$$FS_y = \frac{350.47}{89.5} = 3.92 > 3$$

Therefore, the design criterion of a minimum factor of safety (FS) of 3 on the basis of material yield strength is met.

The lug allowable ultimate bearing load (P_{buL}) for lug failure in bearing, shear-out, or hoop tension is:

$$P_{buL} = 1.304 \times F_{byL} \times D \times t \qquad (\text{if } F_u > 1.304\ F_y)$$

$$= 457.02 \text{ kips}$$

where:

$$\frac{F_u}{F_y} = \frac{80 \text{ ksi}}{60 \text{ ksi}} = 1.33 > 1.304$$

t = lug thickness = 2.0 in

F_u = lug ultimate tensile strength = 80 ksi for ASME SA537, Class 2 carbon steel

The lug ultimate load capacity (457.02 kips) divided by the lug maximum load (89.5 kips) is:

$$FS_u = \frac{457.02}{89.5} = 5.11 > 5$$

Therefore, the design criterion of a minimum factor of safety (FS) of 5 on the basis of material yield strength is met.

The tensile stress (σ) in the net cross-sectional area is:

$$\sigma = \frac{P}{A} = \frac{89.5 \text{ kips}}{7.08 \text{ in.}^2} = 12.64 \text{ ksi}$$

where:

P = the load on each lug
A = the net cross sectional area ($2 \times a \times t = 7.08$ in.^{2})

The factor of safety based on material yield strength $(FS_y)_t$ is:

$$(FS_y)_t = \frac{F_y}{\sigma} = \frac{60 \text{ ksi}}{12.64 \text{ ksi}} = 4.75 > 3$$

Therefore, the design criterion of a minimum factor of safety (FS) of 3 on the basis of material yield strength is met.

The factor of safety based on material ultimate strength $(FS_u)_t$ is:

$$(FS_u)_t = \frac{F_u}{\sigma} = \frac{80 \text{ ksi}}{12.64 \text{ ksi}} = 6.33 > 5$$

Therefore, the design criterion of a minimum factor of safety (FS) of 5 on the basis of material ultimate strength is met.

Embedded Plate

The load path from the lugs through the embedded plate and to the embedded reinforcing steel is symmetrical, with the edges of the lifting lugs being very near the axial center line of the reinforcing steel. Therefore, no significant bending moments are introduced into the embedded plate. The embedded plate cross-sectional area is more than double that of the lugs; therefore, the tensile strength of the plate is adequate by inspection.

Concrete Anchors

Each embedded plate has two lifting lugs, therefore, the load (P_{pl}) on each embedded plate is 2 × 89,500 lb or

$$P_{pl} = 179{,}000 \text{ lbs}$$

Four alternate configurations are provided for the anchorage of the lifting lugs to concrete:

Lift Anchor Configuration A – Welded Rebar (ASTM A706)

The required cross-sectional area of reinforcing steel (A_s) on the basis of yield strength (S_y = 60 ksi) is:

$$A_s = \frac{P_{pl}}{S_y} = \frac{179\text{kips}}{60\text{ksi}} = 2.98 \text{ in}^2$$

Eight #11 reinforcing steel bars are selected to anchor the embedded plate to the concrete cask concrete shell. The cross-sectional area for each #11 bar is 1.56 in^2 [41]. Therefore, the total area (A_t) resisting the tensile load is:

$$A_t = 8 \times 1.56 \text{ in}^2 = 12.48 \text{ in}^2$$

The reinforcing steel actual cross-sectional area (12.48 in.^{2}) divided by the required cross-sectional area (2.98 in^2) is:

$$FS = \frac{12.48}{2.98} = 4.19 > 3$$

Therefore, the design criterion of a minimum factor of safety (FS) of 3 on the basis of material yield strength is met.

The required cross-sectional area of reinforcing steel (A_s) on the basis of ultimate strength (S_u = 80 ksi) is:

$$A_s = \frac{P_{pl}}{S_u} = \frac{179\text{kips}}{80\text{ksi}} = 2.24 \text{ in}^2$$

The reinforcing steel actual cross-sectional area (12.48 in.^{2}) divided by the required cross-sectional area (2.24 in^2) is:

$$FS = \frac{12.48}{2.24} = 5.57 > 5$$

Therefore, the design criterion of a minimum factor of safety (FS) of 5 on the basis of material ultimate strength is met.

<u>Lift Anchor Configuration B – Threaded Rebars (ASTM A615)</u>

The required cross-sectional area of reinforcing steel (A_s) on the basis of yield strength for Grade 75 is:

$$A_s = \frac{P_{pl}}{S_y}$$

$$= 2.39 \text{ in}^2$$

where:

P_{pl} = 179 kips

$S_y = 75$ ksi

Eight #11 reinforcing steel bars are selected to anchor the embedded plate to the concrete cask concrete shell. The bars are to be threaded 1-3/8 (6 UNC 2A). The tensile stress area for each #11 threaded bar is 1.155 in^2 [40]. Therefore, the total area (A_t) resisting the tensile load is:

$$A_t = 8 \times 1.155 \text{ in.}^2 = 9.24 \text{ in}^2$$

The reinforcing steel actual cross-sectional area divided by the required cross-sectional area is:

$$FS = \frac{9.24}{2.39} = 3.87 > 3$$

Therefore, the design criterion of a minimum factor of safety (FS) of 3 on the basis of material yield strength is met.

The required cross-sectional area of reinforcing steel (A_s) on the basis of ultimate strength for Grade 75 is:

$$A_s = \frac{P_{pl}}{S_u}$$

$$= 1.79 \text{ in}^2$$

where:

$P_{pl} = 179$ kips

$S_u = 100$ ksi

The reinforcing steel actual cross-sectional area divided by the required cross-sectional area is:

$$FS = \frac{9.24}{1.79} = 5.16 > 5$$

Therefore, the design criterion of a minimum factor of safety (FS) of 5 on the basis of material ultimate strength is met.

Thread Engagement

Based on the Machinery's Handbook [40], the shear area of the 1-3/8 (6 UNC 2A) bolt hole internal threads (A_n) is calculated as:

$$A_n = 3.1416 n L_e D_s \min\left[\frac{1}{2n} + 0.57735\left(D_s \min - E_n \max\right)\right] = 6.53 \text{ in}^2$$

and the shear area for the external threads of the plate, A_s, is calculated as:

$$A_s = 3.1416 n L_e K_n \max\left[\frac{1}{2n} + 0.57735\left(E_s \min - K_n \max\right)\right] = 4.68 \text{ in}^2$$

where:

n = 6, threads per inch

L_e = plate thickness (= 2.0 in),

but not less than bolt thread engagement length

$$= \frac{2A_t}{3.1416 K_n \max[0.5 + 0.57735 n (E_s \min - K_n \max\}]}$$

= 1.0 in (Use L_e = 2.0 in)

$D_s \min$ = 1.3544 in, minimum major diameter−external thread

$E_n \max$ = 1.2771 in, maximum pitch diameter−internal thread

$K_n \max$ = 1.225 in, maximum minor diameter−internal thread

$E_s \min$ = 1.2563 in, minimum pitch diameter−external thread

A_t = 1.155 in^2, tensile stress area

The minimum shear area of 4.68 in^2 controls. Hence, the shear stress, τ, in the bolt hole threads is:

$$\tau = \frac{W / n}{A_s} = 4.78 \text{ ksi}$$

where:

W = 179.0 kips

n = number of rebar = 8

A_s = 4.68 in^2

The factors of safety for ASTM A615 (Grade 75) rebar allowables (S_y = 75 ksi, S_u = 100 ksi), which meet the NUREG criteria for redundant systems, are:

$$FS_y = \frac{0.6S_y}{\tau} = 9.41 > 3$$

$$FS_u = \frac{0.5S_u}{\tau} = 10.46 > 5$$

Lift Anchor Configuration C – Williams All-Thread-Bars

The required cross-sectional area of reinforcing steel (A_s) on the basis of yield strength (S_y = 120 ksi) is:

$$A_s = \frac{P_{pl}}{S_y} = 1.49 \text{ in}^2$$

where P_{pl} = 179.0 kips

S_y = 120.0 ksi

Six 1-1/4" Grade-150 Williams All-Thread-Bar are selected to anchor the embedded plate to the concrete cask shell. The cross-sectional area for each bar is 1.25 in^2. Therefore, the total area (A_t) resisting the tensile load is:

$A_t = 6 \times 1.25 \text{ in}^2 = 7.5 \text{ in}^2$

The reinforcing steel actual cross-sectional area (7.5 in^2) divided by the required cross-sectional area on the basis of yield strength, (1.49 in^2) is:

$$FS_{yield} = \frac{A_t}{A_s} = 5.03 > 3$$

Therefore, the design criterion of a minimum factor of safety (FS) of 3 on the basis of material yield strength is met.

The required cross-sectional area of reinforcing steel (A_s) on the basis of ultimate strength (S_u = 150 ksi) is:

$$A_s = \frac{P_{pl}}{S_u} = 1.19 \text{ in}^2$$

where P_{pl} = 179.0 kips
S_u = 150.0 ksi

The reinforcing steel actual cross-sectional area (7.5 in^2) divided by the required cross-sectional area on the basis of yield strength, (1.19 in^2) is:

$$FS_{ultimate} = \frac{A_t}{A_s} = 6.30 > 5$$

Therefore, the design criterion of a minimum factor of safety (FS) of 5 on the basis of material ultimate strength is met.

Thread Engagement

Based on the Machinery's Handbook [40], the shear area for the internal threads of the 1-1/4" nut, A_n, is calculated as:

$$A_n = 3.1416 n L_e D_s \min\left[\frac{1}{2n} + 0.57735\left(D_s \min - E_n \max\right)\right] = 6.30 \text{ in}^2$$

and the shear area for the external threads of the rebar, A_s, is calculated as:

$$A_s = 3.1416 n L_e K_n \max\left[\frac{1}{2n} + 0.57735\left(E_s \min - K_n \max\right)\right] = 5.54 \text{ in}^2$$

where

n = 4, number of threads per inch,

L_e = 2.5 in (overall height of hex nut),

but not less than the thread engagement length

$$= \frac{2A_t}{3.1416 K_n \max[0.5 + 0.57735 n (E_s \min - K_n \max)]}$$

= 1.19 in (Use L_e = 2.5 inches)

A_t = 1.32 in^2, tensile stress area (non-standard),

$D_s\min$ = 1.399 in, minimum major diameter – external thread (rebar),

$E_n\max$ = 1.3674 in, maximum pitch diameter – internal thread (nut),

$K_n\max$ = 1.2898 in, maximum minor diameter – internal thread (nut), and

$E_s\min$ = 1.31 in, minimum pitch diameter – external thread (rebar)

The minimum shear area of 5.54 in^2 controls. Hence, the shear stress, τ, in the bolt hole threads is:

$$\tau = \frac{W/n}{A_s} = 5.38 \text{ ksi}$$

where:

$W = 179.0$ kips
n = number of rebar = 6
$A_s = 5.54\ in^2$

The factor of safety for the rebar based on the yield strength (120.0 ksi) is:

$$FS_{yield} = \frac{0.6S_y}{\tau} = 13.38 > 3$$

The factor of safety for the bar based on the ultimate strength (150.0 ksi) is:

$$FS_{ultimate} = \frac{0.5S_u}{\tau} = 13.94 > 5$$

Development Length of Welded Bars, Threaded Bars, and Williams All-Thread-Bars (Configurations A, B, and C)

The development length (l_d) is the length of embedded reinforcing steel required to develop the design strength of the reinforcing steel at a critical section. The required reinforcing steel development length (l_d) for bars in tension in accordance with Section 12.2 of the ASME Code, Code Cases – Nuclear Components [13] shall be:

$$l_d = \text{larger of } \frac{0.04A_bS_y}{\sqrt{f_c'}} \text{ or } 0.0004d_bS_y \text{ or 12 inches}$$

where

d_b = diameter of rebar (1.41 inch for # 11, 1.411 inch for Williams)

A_b = tensile stress area of rebar (1.56 in^2 for # 11, 1.32 in^2 for Williams)

S_y = yield strength of the reinforcing steel (60 ksi for A706, 75 ksi for A615, 120 ksi for Williams)

f_c' = concrete design strength = 4,000 psi

The development lengths for the different diameter and strength rebars are given in the following table:

Development Lengths

S_y	Reinforcing Steel Bar	$l_d = \frac{0.04A_bS_y}{\sqrt{f_c'}}$	$l_d = 0.0004d_bS_y$	l_d =12 in	Max. l_d
60 ksi	# 11 (A706)	59.2	33.8	12.0	59.2
75 ksi	# 11 (A615)	74.0	42.3	12.0	74.0
120 ksi	Williams	100.2	67.7	12.0	100.2

The actual length of the regular reinforcing steel provided is 185.5 inches and that of the Williams threaded bars is 102 inches. These lengths are greater than the maximum required length given in the preceding table.

Lift Anchor Configuration D – Steel Plates

Each vertical plate has one lifting lug; therefore, the load on each vertical plate is 89,500 pounds. The required cross-sectional area of vertical steel plates on the basis of yield (60 ksi) and ultimate (80 ksi) strengths is:

$A_{yield} = 89.5/60 = 1.49 \text{ in}^2$

$A_{ultimate} = 89.5/80 = 1.12 \text{ in}^2$

The vertical steel plates are welded to the embedded base plate, which acts as an anchor to the vertical concrete cask shell. The actual tensile stress area of the vertical steel plates is 15.2 in^2 (7.6x2.0).

The factors of safety measured as the actual plate areas divided by the required plate areas on the basis of yield and ultimate strengths are given as:

$$FS\ (yield) = \frac{15.2}{1.49} = 10.2 > 3$$

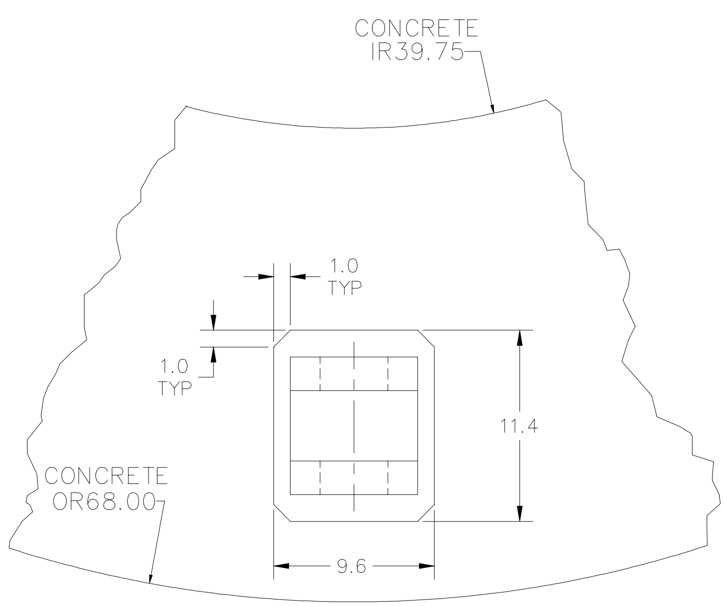
$$FS\ (ultimate) = \frac{15.2}{1.12} = 13.6 > 5$$

Therefore, the design criteria of a minimum Factor of Safety of 3 on the basis of material yield strength and a minimum Factor of Safety of 5 on the basis of material ultimate strength are met.

The depth of the shear area of the concrete section in tension is evaluated according to Sections 11.1 and 11.3 of ACI 349-85 [4] as follows.

Conservatively, using the shear plane at the edge of the base plate and discounting the face of the plate towards the outer surface of the vertical concrete cask (see the following figure), the shear perimeter is:

$$P = 2\times(11.4-2)+(9.6-2) + 2\times\sqrt{2} = 29.23 \text{ inch}$$



The maximum applied load is W = 89.5×2 = 179 kips. The effective shear area is A_{shear} = P×D (D is the depth of the shear area). The shear strength provided by concrete (V_n) is conservatively taken as $2\sqrt{f_c^{'}}\,A_{shear}$. Using the relationship $V_u \le \Phi V_n$ (Φ = 0.85 for shear [13], and V_u is the applied load), the required depth of the shear area (D) is determined as:

$$D = \frac{W}{\Phi 2\sqrt{f_c^{'}}P} = 57.0 \text{ inch} < 61.5 \text{ inch}$$

where $f_c^{'}$ = 4,000 psi.

The actual depth of the shear area (61.5 inch) is adequate since it is greater than the required depth calculated above.

Welds

The lifting lugs are welded to the embedded plate with full penetration welds developing the full strength of the attached lugs.

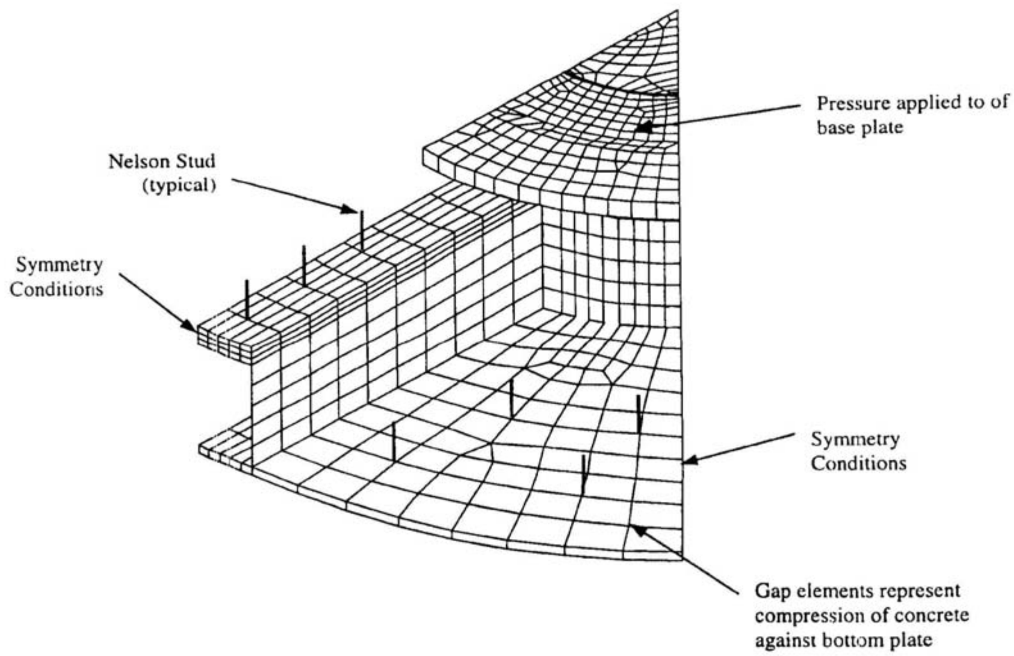
The vertical plate is welded to the base plate with full penetration welds developing the full strength of the vertical plate.

Therefore, all welds are adequate.

Nelson Studs

During a top end lift, the weight of the canister and pedestal applies a tensile load to the Nelson studs. Using the BWR Class 5 configuration, 75,600-pound canister weight (77,000 pounds used in this analysis), an ANSYS finite element model is used to obtain the maximum load on the Nelson studs. The model, shown in the following figure, represents one-eighth of the pedestal. The weight of the canister is applied as a pressure load to the top of the 2-inch base plate. The load is reacted through the Nelson studs and gap elements between the pedestal and the concrete. Using a 10% dynamic load factor, the maximum load on a Nelson stud is 14,272 pounds.

In accordance with ACI-349-85 [4], the design pullout strength of the concrete (P_d) for any embedment is based on a uniform tensile stress acting on an effective stress area which is defined by the projected area of stress cones radiating toward the attachment from the bearing edge of the anchor heads. The effective area shall be limited by overlapping stress cones, by the intersection of the cones with concrete surfaces, by the bearing area of anchor heads, and by the overall thickness of the concrete. A 45° inclination angle is used for the stress cones.



Pedestal Finite Element Model

The maximum pullout strength of the concrete (P_d) is defined by the equation

$$P_d = 4 \times \phi \times \sqrt{f_c^{'}} \times A_{cp}$$

where:

ϕ - strength reduction factor = 0.85

$f_c^{'}$ - concrete compression strength = 4,000 psi

A_{cp} - projected surface area of stress cones for Nelson studs

The maximum load occurs in the six Nelson studs located on the top of the air inlet. A_{cp} for the six Nelson studs equals 419.2 inch^2. Therefore, P_d equals:

$$P_d = 4 \times 0.85 \times \sqrt{4000} \times 419.2 = 90{,}143 \text{ lb.}$$

The total load on the six Nelson studs is 27,508 pounds.

The margin of safety for the concrete is:

$$MS = \frac{90{,}143}{27{,}508} - 1 = +2.28$$

For a single stress cone, the maximum load is 14,272 pounds. The corresponding pull-out strength is:

$$P_d = 4 \text{ x } 0.85 \text{ x } 117.8 \text{ x } \sqrt{4{,}000} = 25{,}331 \text{ lbs.}$$

where the projected surface area for a single stress cone (Acp) of a single Nelson stud is 117.8.

The margin of safety for a single Nelson stud is:

$$MS = \frac{25{,}331}{14{,}272} - 1 = +0.77$$

The cross-sectional area of the Nelson studs is:

$$A_s = \frac{\pi}{4} \times 0.75^2 = 0.44\,\text{in}^2$$

The allowable load per stud is:

$$P_s = 0.44 \times 55{,}000 = 24{,}200\,\text{lbs}$$

where 55,000 psi is the ultimate tensile strength for ASTM A108 Grade 1010 through 1020 low carbon steel [14].

The margin of safety for the Nelson stud is:

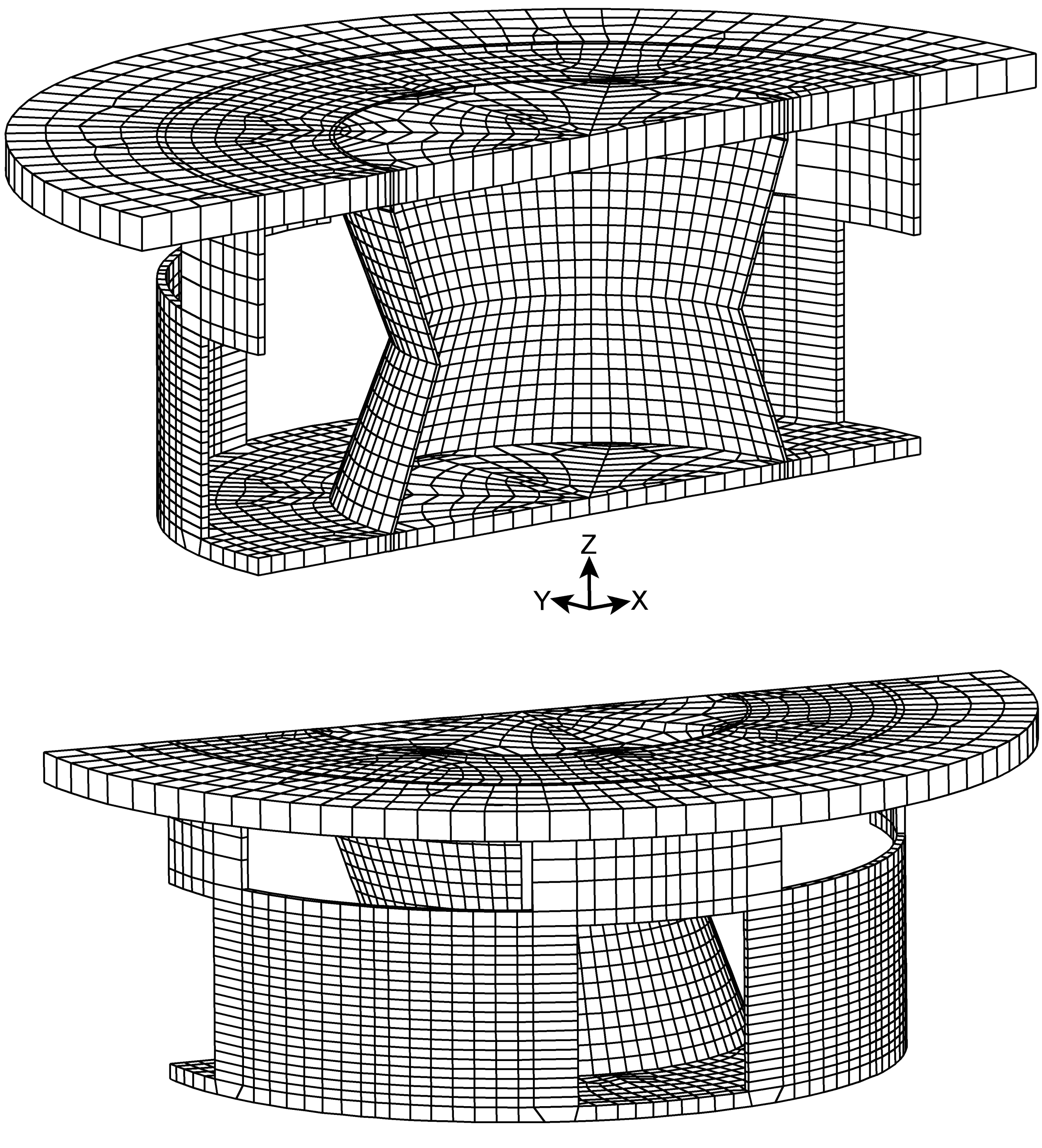
$$MS = \frac{24{,}200}{14{,}272} - 1 = +0.70$$

Vertical Concrete Cask Pedestal

Using the same ANSYS Finite Element Model that was used for the Nelson Stud analysis, an analysis of the pedestal was performed. The maximum nodal stress intensity for the pedestal is 5,785 psi. From Tables 4.1-4 and 4.1-5, the maximum canister temperature is 376°F. For A36 steel, the allowable stress (S_m) is 19,300 psi. The margin of safety is, conservatively:

$$MS = \frac{19{,}300}{5{,}785} - 1 = +2.34$$

Figure 3.4.3.1-1 Base Weldment Finite Element Model



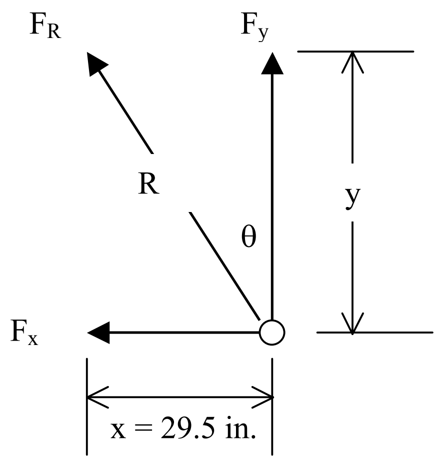
3.4.3.2 Canister Lift

The adequacy of the canister lifting devices is demonstrated by evaluating the hoist rings, the canister structural lid, and the weld that joins the structural lid to the canister shell against the criteria in NUREG-0612 [8] and ANSI N14.6 [9]. The lifting configuration for the PWR and BWR canisters consists of six hoist rings threaded into the structural lid at equally spaced angular intervals. The hoist rings are analyzed as a redundant system with two three-legged lifting slings. For redundant lifting systems, ANSI N14.6 requires that load-bearing members be capable of lifting three times the load without exceeding the tensile yield strength of the material and five times the load without exceeding the ultimate tensile strength of the material. The canister lid is evaluated for lift conditions as a redundant system that demonstrates a factor of safety greater than three based on yield strength and a factor of safety greater than five based on ultimate strength. The canister lift analysis is based on a load of 76,000 lb, which bounds the weight of the heaviest loaded canister configuration, plus a dynamic load factor of 10 %. Alternative canister lifting system designs may be used based on a site-specific analysis and evaluation.

The canister lifting configuration is shown in the following figure, where: x is the distance from the canister centerline to the hoist ring center line (29.5 inches); F_y is the vertical component of force on the hoist ring; F_x is the horizontal component of force on the hoist ring; R is the sling length; and, F_R is the maximum allowable force on the hoist ring (30,000 lbs.). The angle θ is the angle from vertical to the sling. The vertical load, F_y, assuming a 10% dynamic load factor, is:

$$F_y = \frac{76{,}000 \text{ lbs x } 1.1}{3 \text{ lift points}} = 27{,}867 \text{ lbs}$$

The hoist rings are American Drill Bushing Company, Model 23200 Safety Engineered Hoist Rings, rated at 30,000 lbs., (or comparable ring from an alternative manufacture) with a safety factor of 5 on ultimate strength.



Calculating the maximum angle, θ, that will limit F_R to 30,000 lb:

$$\theta = \cos^{-1}\left(\frac{F_y}{F_R}\right) = \cos^{-1}\left(\frac{27{,}867}{30{,}000}\right) = 21.7 \text{ deg}$$

The minimum sling length, R, is

$$R = \frac{x}{\sin\theta} = \frac{29.5}{\sin 21.7^{o}} = 79.8 \text{ in.}$$

An 80-in. sling places the master link about 75 in. above the top of the canister (y = R cos θ = 80 cos 21.7º = 74.3 inches).

A minimum distance of 75 inches between the master link and the top of the canister is specified in Sections 8.1.2 and 8.2.

From the Machinery's Handbook [24], The shear area, A_n, in the structural lid bolt hole threads is calculated as

$$A_n = 3.1416\ n\ L_e\ D_s\min\left[\frac{1}{2n} + 0.57735\left(D_s\min - E_n\max\right)\right]$$

$$= 3.1416(4.5)(2.0 \text{ in.})(1.9751 \text{ in.})\left[\frac{1}{2(4.5)} + 0.57735(1.9751 \text{ in.} - 1.8681 \text{ in.})\right]$$

$$= 9.654 \text{ in}^2$$

where:

n	=	4.5 threads per in,
L_e	=	2.0-in. bolt thread engagement length
$D_s$min	=	1.9751 in., minimum major diameter of class 2A bolt threads
$E_n$max	=	1.8681 in., maximum pitch diameter of class 2B lid threads

The shear stress, τ, in the structural lid bolt hole threads is calculated as:

$$\tau = \frac{F_y}{A_n} = \frac{27{,}867 \text{ lb}}{9.654 \text{ in}^2} = 2{,}887 \text{ psi}$$

The canister structural lid is constructed of SA240, Type 304L stainless steel. Using shear allowables of 0.6 S_y and 0.5 S_u at a temperature of 300°F, the shear stress of 2,887 psi results in factors of safety of:

$$(\text{F.S.})_y = \frac{0.6 \text{ x } 19{,}200 \text{ psi}}{2{,}887 \text{psi}} = 4.0 > 3$$

$$(\text{F.S.})_u = \frac{0.5 \text{ x } 60{,}900 \text{ psi}}{2{,}887 \text{psi}} = 10.5 > 5$$

The criteria of NUREG-0612 and ANSI N14.6 for a redundant systems are met. Therefore, the 2.0-inch length of thread engagement is adequate.

The total weight of the heaviest loaded transfer cask (Class 5 BWR) is approximately 208,400 pounds. Three (3) times the design weight of the loaded canister is (3 × 76,000) 228,000 lbs, which is greater than the weight of the heaviest loaded transfer cask. Consequently, the preceding analysis bounds the inadvertently lifting of the transfer cask by the canister, since the canister lid and the hoist rings do not yield.

The structural adequacy of the canister structural lid and weld is evaluated using a finite element model of the upper portion of the canister. As shown in Figure 3.4.3.2-1, the model represents one-half of the upper section of the canister, including the structural and shield lids. The model uses gap/spring elements to simulate contact between adjacent components. Specifically, contact between the canister structural and shield lids is modeled using COMBIN40 combination elements in the axial (UY) degree of freedom. Simulation of the spacer ring is accomplished using a ring of COMBIN40 gap/spring elements connecting the shield lid and the canister in the axial direction at the lid lower outside radius. CONTAC52 elements are used to model the interaction between the structural lid and canister shell and the shield lid and canister shell just below the respective lid weld joints. The size of the CONTAC52 gaps was determined from nominal dimensions of contacting components. The COMBIN40 elements used between the structural and shield lids, and for the spacer ring, were assigned small gap sizes of 1×10^{-8} in. All gap/spring elements are assigned a stiffness of 1×10^{8} lb/in.

Boundary conditions were applied to enforce symmetry at the cut boundary of the model (in the x-y plane). All nodes on the x-y symmetry plane were restrained perpendicular to the symmetry plane (UZ). In addition, the nodes in the x-z plane at the bottom of the model were restrained in the axial direction (UY).

The lifting configuration for the canister consists of six hoist rings bolted to the structural lid at equally spaced angular intervals. To simulate the lifting of the canister, point loads equal to one-sixth of the total loaded canister weight plus a dynamic loading factor of 10% were applied to the model as forces at the lift locations while restraining the model at its base in the axial direction. Because of the symmetry conditions of the model, the forces applied to nodes on the symmetry plane were one-half of that applied at the other locations. The nodal point forces applied to the model as depicted in Figure 3.4.3.2-1 are calculated (including a dynamic load factor of 10%) as

$$W/6 = (76{,}000 \text{ lb} \times 1.1)/6 = 13{,}934 \text{ lb}$$

$$W/12 = (76{,}000 \text{ lb} \times 1.1)/12 = 6{,}967 \text{ lb}$$

To evaluate the canister lid welds during lift conditions, linearized sectional stresses are taken across the weld. The sections are shown in Figure 3.4.3.2-1. Stress results are compared to material allowables at a temperature of 300°F. For conservatism, the weld allowable is taken as the base material. The following table is a summary of the weld stress results.

Section	Component Description	Material	Stress Intensity $P_m + P_b$ (psi)	Factor of Safety on Yield	Factor of Safety on Ultimate
1	Structural Lid Weld	304L SS	1,678	11.4	36.3
2	Canister shell	304L SS	3,083	6.2	19.8
3	Shield Lid Weld	304 SS	1,794	10.7	33.9
4	Canister shell	304L SS	2,491	7.7	24.4
5	Canister shell	304L SS	1,305	14.7	46.7

The maximum nodal stress intensity outside the weld region of 2,608 psi occurs in the structural lid. The nodal stress results are presented graphically in Figure 3.4.3.2-2. The corresponding factors of safety are:

$$(\text{F.S.})_{\text{yield}} = \frac{\text{yield strength}}{\text{maximum nodal stress intensity}} = \frac{19{,}200 \text{ psi}}{2{,}608 \text{ psi}} = 7.4 \ (> 6)$$

$$(\text{F.S.})_{\text{ultimate}} = \frac{\text{ultimate strength}}{\text{maximum nodal stress intensity}} = \frac{60{,}900\ \text{psi}}{2{,}608\ \text{psi}} = 23.4\ (>10)$$

Therefore, the canister meets the criteria of NUREG-0612 and ANSI N14.6 for nonredundant systems.

Figure 3.4.3.2-1 Canister Lift Finite Element Model

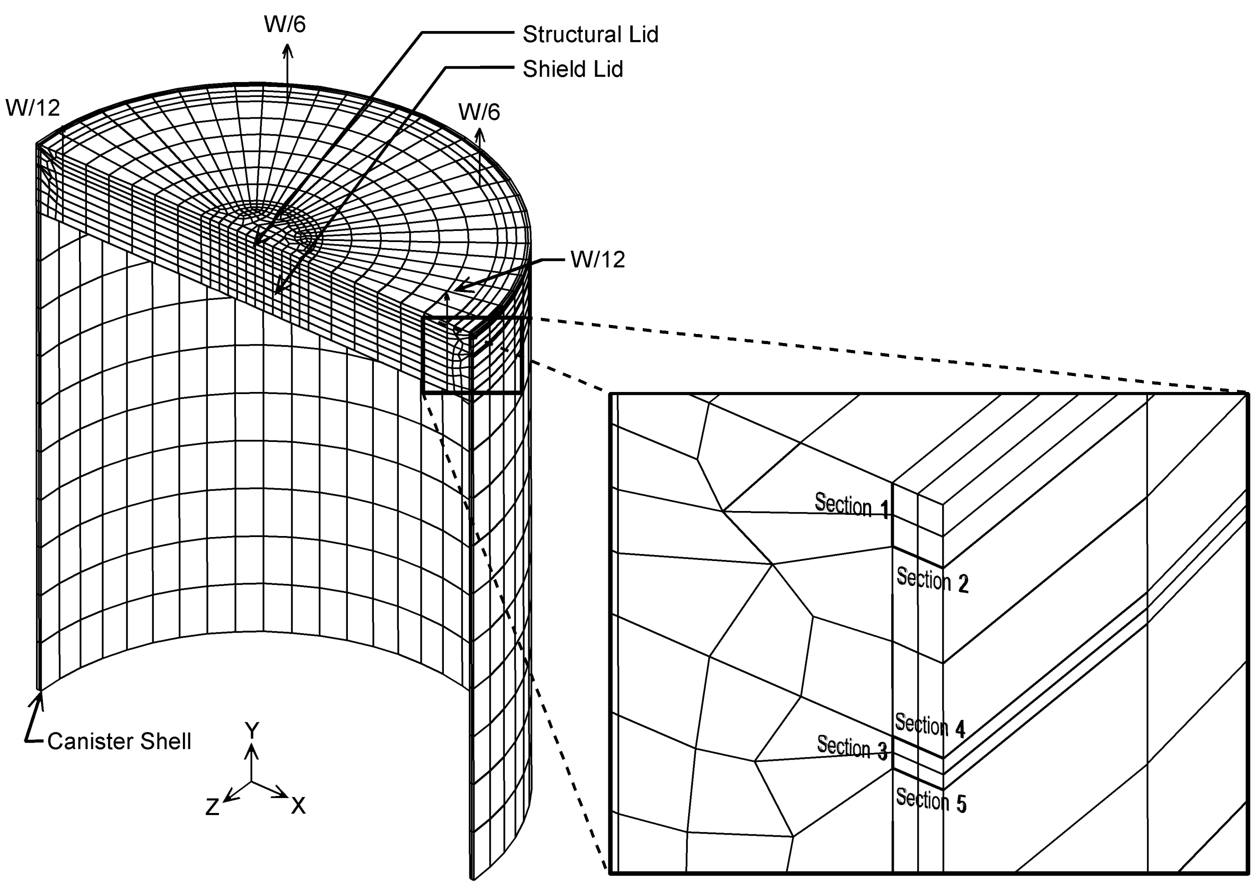
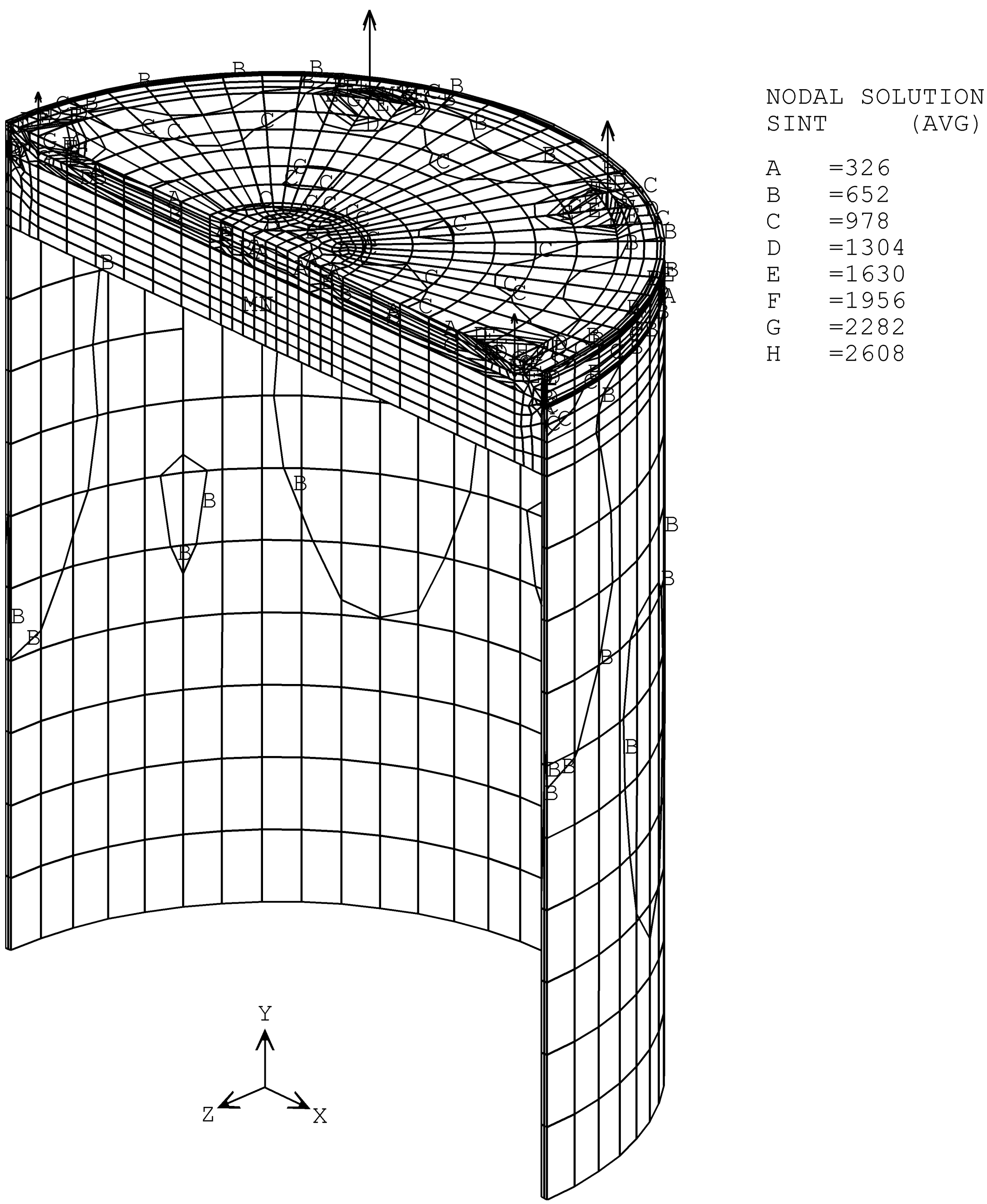


Figure 3.4.3.2-2 Canister Lift Model Stress Intensity Contours (psi)



3.4.3.3 Standard Transfer Cask Lift

The evaluation of the standard transfer cask presented here shows that the design meets NUREG-0612 [8] and ANSI N14.6 [9] requirements for nonredundant lift systems. The adequacy of the standard transfer cask is shown by evaluating the stress levels in all of the load-path components against the NUREG-0612 criteria.

3.4.3.3.1 Standard Transfer Cask Shell and Trunnion

The adequacy of the trunnions and the cask shell in the region around the trunnions during lifting conditions is evaluated in this section in accordance with NUREG-0612 and ANSI N14.6.

A three-dimensional finite element model is used to evaluate the lifting of a fully loaded standard transfer cask. Because of symmetry, it was necessary to model only one-quarter of the standard transfer cask, including the trunnions and the shells at the trunnion region. Note that the optional stiffener plates above the trunnions (between the two shells) are not included in the model. The model represents the bounding configuration without the stiffener plates. The lead and the NS-4-FR between the inner and outer shells of the standard transfer cask are neglected, since they are not structural components. SOLID95 (20 noded brick element) and SHELL93 (8 noded shell element) elements are used to model the trunnion and shells, respectively. Due to the absence of rotation degrees of freedom for the SOLID95 elements, BEAM4 elements perpendicular to the shells are used at the interface of the trunnion and the shells to transfer moments from the SOLID95 elements to SHELL93 elements. The finite element model is shown in Figure 3.4.3.3-1.

The total weight of the heaviest loaded standard transfer cask (Class 5 BWR) is calculated at approximately 208,400 pounds. A conservative load of 210,000 lb., plus a 10% dynamic load factor, is used in the model. The load used in the quarter-symmetry model is $(210{,}000 \times 1.1)/4 = 57{,}750$ lb. The load is applied upward at the trunnion as a "surface load" whose location is determined by the lifting yoke dimensions. The model is restrained along two planes of symmetry with symmetry boundary conditions. Vertical restraints are applied to the bottom of the model to resist the force applied to the trunnion.

The maximum temperature in the standard transfer cask shell/trunnion region is conservatively evaluated as 300°F. For the ASTM A-588 shell material, the yield strength, S_y, is 45.6 ksi, and the ultimate strength, S_u, is 70 ksi. The trunnions are constructed of ASTM A-350 carbon steel, Grade LF2, with a yield stress of 31.9 ksi and an ultimate stress of 70 ksi. The standard impact test temperature for ASTM A-350, Grade LF2 is -50°F. The NDT temperature range is -70°F to -10°F for ASTM A-588 with a thickness range of 0.625 in. to 3 in. [25]. Therefore, the minimum service temperature for the trunnion and shells is conservatively established as 0°F (50°F higher than the NDT test temperature, in accordance with Section 4.2.6 of ANSI N14.6 [9].

Table 3.4.3.3-1 through Table 3.4.3.3-4 provide summaries of the top 30 maximum stresses for both surfaces of the outer shell and inner shell (see Figure 3.4.3.3-2 and Figure 3.4.3.3-3 for node locations for the outer shell and inner shell, respectively). Stress contour plots for the outer shell are shown in Figure 3.4.3.3-4 and Figure 3.4.3.3-5. Stress contours for the inner shell are shown in Figure 3.4.3.3-6 and Figure 3.4.3.3-7. As shown in Table 3.4.3.3-1 through Table 3.4.3.3-4, all stresses, except local stresses, meet the NUREG-0612 and ANSI N14.6 criteria. That is, a factor of safety of 6 applies on material yield strength and 10 applies on material ultimate strength. The high local stresses, as defined in ASME Code Section III, Article NB-3213.10, which are relieved by slight local yielding, are not required to meet the 6 and 10 safety factor criteria [see Ref. 9, Section 4.2.1.2].

The localized stresses occur at the interfaces of the trunnion with the inner and outer shells. The size of the areas are less than 4.1 inches and 4.0 inches for the inner and outer shell, respectively. In accordance with ASME Code, Article NB-3213.10, the area of localized stresses cannot be larger than:

$$1.0\sqrt{Rt}$$

where:

R is the minimum midsurface radius
t is the minimum thickness in the region considered

Based on this formula, the size limitations for local stress regions are 5.1 inches (>4.06 inches) and 7.3 inches (>4.00 inches) for the inner and outer shells, respectively.

For the trunnion, the maximum tensile bending stress and average shear stresses occur at the interface with the outer shell. The linearized stresses through the trunnion are 3,377 psi in bending and 1,687 psi in shear. Comparing these stresses to the material allowable yield and ultimate strength (A350, Grade LF2), the factor of safety on yield strength is 9.4 (which is >6) and on ultimate strength is 20.7 (which is >10).

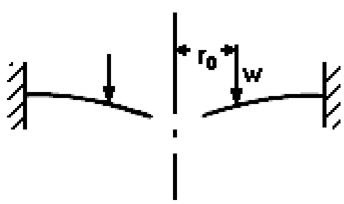
3.4.3.3.2 Retaining Ring and Bolts

The standard transfer cask uses a retaining ring bolted to the top flange to prevent inadvertent lifting of the canister out of the transfer cask, which could increase the radiation exposure to nearby workers. In the event that the loaded transfer cask is inadvertently lifted by attaching to the canister eyebolts instead of the transfer cask trunnions, the retaining ring and bolts have sufficient strength to support the weight of the heaviest transfer cask, plus a 10% dynamic load factor.

Retaining Ring

To qualify the retaining ring, the equations for annular rings are used (Roark [26], Table 24, Case 1e). The retaining ring is represented as shown in the sketch below. The following sketch assists in defining the variables used to calculate the stress in the retaining ring and bolts. The model assumes a uniform annular line load *w* applied at radius r_o.

The boundary conditions for the model are outer edge fixed, inner edge free with a uniform annular line load w at radius r_o.



The material properties and parameters for the analysis are:

Plate dimensions:	
thickness:	t = 0.75 in
outer radius (bolt circle):	a = 37.28 in
outer radius (outer edge):	c = 38.52 in
inner radius:	b = 32.37 in
Weight of bounding transfer cask:	wt = 124,000 lb × 1.1
Radial location of applied load:	r_o = 33.53 in
Material:	ASTM A-588
Modulus of elasticity:	$E = 28.3 \times 10^6$ psi
Poisson's ratio:	ν = 0.31
Number of bolts:	Nb = 32
Radial length of applied load:	$L_r = 2\pi r_o$
	L_r = 210.675 in
Applied unit load:	$w \equiv \frac{wt}{L_r}$
	w = -647.44 psi

The shear modulus is:

$$G = \frac{E}{2 \cdot (1 + v)}$$

$$G = 1.08 \times 10^7 \text{ psi}$$

D is a plate constant used in determining boundary values; it is also used in the general equations for deflection, slope, moment and shear. K_{sb} and K_{sro} are tangential shear constants used in determining the deflection due to shear:

$$D = \frac{E \cdot t^3}{12 \cdot (1 - v^2)}$$

$D = 1.101 \times 10^6$ lb-in

Tangential shear constants, K_{sb} and K_{sro}, are used in determining the deflection due to shear:

$$K_{sb} = K_{sro} = -1.2 \cdot \frac{r_o}{a} \cdot \ln\left(\frac{a}{r_o}\right)$$
$$= -0.114$$

Radial moment M_{rb} and M_{ra} at points b and a (inner and outer radius, respectively) are:

M_{rb} (b,0) = 0 lb-in/in

M_{ra} (a,0) = 2207.86 lb-in/in

Transverse moment M_{tb} and M_{ta}, at points b and a (inner and outer radius, respectively) due to bending are:

M_{tb} (b,0) = -122.64 lb-in./in.

M_{ta} (a,0) = 684.44 lb-in./in.

The calculated shear stresses, τ_b and τ_a, at points b and a (inner and outer radius, respectively) are:

$\tau_b = 0$ psi

$$\tau_a = \frac{wt}{2\pi At}$$

τ_a = -776.42 psi

The calculated radial bending stresses, σ_{rb} and σ_{ra}, at points b and a (inner and outer radius) are:

$$\sigma_{r(i)} = \frac{6M_{r(i)}}{t^2}$$

σ_{rb} = 0 psi

σ_{ra} = 23,550 psi

The calculated transverse bending stresses, σ_{tb} and σ_{ta}, at points b and a (inner and outer radius) are:

$$\sigma_{t(i)} = \frac{6M_{t(i)}}{t^2}$$

σ_{tb} = -1308.2 psi

σ_{ta} = 7,300.7 psi

The principal stresses at the outer radius are:

σ_{1a} = 23,590 psi

σ_{2a} = 7,263.6 psi

σ_{3a} = 0 psi

The stress intensity, SI_a, at the outer radius ($P_m + P_b$) is:

$SI_a = \sigma_{1a} - \sigma_{3a}$

SI_a = 23,590 psi

The principal stresses at the inner radius are:

σ_{1b} = 0 psi

σ_{2b} = - 1308.2 psi

σ_{3b} = 0 psi

The stress intensity, SI_b, at the inner radius ($P_m + P_b$) is:

$$SI_b = \sigma_{1b} - \sigma_{2b}$$
$$SI_b = 1308.2 \text{ psi}$$

The maximum stress intensity occurs at the outer radius of the retaining ring. For the off-normal condition, the allowable stress intensity is equal to the lesser of 1.8 S_m and 1.5 S_y. For ASTM A-588, the allowable stress intensity at 300°F is 1.8(23.3) = 41.94 ksi. The calculated stress of 23.59 ksi is less than the allowable stress intensity and the margin of safety is:

$$MS = \frac{41.94}{23.59} - 1 = 0.78$$

Retaining Ring / Canister Bearing

The bearing stress, S_{brg}, between the retaining ring and canister is calculated as:

Weight of Transfer Cask (TFR) = 124,000 × 1.1 = 136,400 lbs.
Area of contact between retaining ring and canister:

$$A = \pi\left(33.53^2 - 32.37^2\right) = 240 \text{ in}^2$$

$$S_{brg} = \frac{136{,}400}{240} = 568 \text{ psi}$$

Bearing stress allowable is S_y. For ASTM A-588, the allowable stress at 300°F is 45.6 ksi. The calculated bearing stress is well below the allowable stress with a large margin of safety.

Shearing stress of Retaining Plate under the Bolt Heads

The shearing stress of the retaining plate under the bolt head is calculated as:

Outside diameter of bolt head d_b = 1.125 in.

Total shear area under bolt head = π (1.125) × 32 × 0.75

= 84.82 in^2.

Shear stress of retaining plate, τ_p, under bolt head is:

$$\tau_p = \frac{136{,}400}{84.82} = 1608 \text{ psi}$$

Conservatively, the shear allowable for normal conditions is used.

$$\tau_{allowable} = (0.6)\,(S_m) = (0.6)\,(23.3 \text{ ksi}) = 13.98 \text{ ksi}$$

The Margin of Safety is: $\frac{13{,}980}{1{,}608} - 1 = +\text{large}$

<u>Bolt Edge Distance</u>

Using Table J3.5 “Minimum Edge Distance, in.” of Section J3 from “Manual of Steel Construction Allowable Stress Design,”[23] the required saw-cut edge distance for a 0.75 inch bolt is 1.0 inch. As shown below, the edge distance for the bolts meets the criteria of the Steel Construction Manual.

$$\frac{77.04 - 74.56}{2} = 1.24 \text{ in} > 1.0 \text{ in}$$

<u>Retaining Ring Bolts</u>

The load on a single bolt, F_F, due to the reactive force caused by inadvertently lifting the canister, is:

$$F_F = \frac{wt}{N_b} = 4{,}262 \text{ lb}$$

where:

N_b = number of bolts, 32, and

wt = the weight of the cask, plus a 10% load factor, 124,000 lb × 1.1 = 136,400 lb.

The load on each bolt, F_M, due to the bending moment, is:

$$F_M = \left(\frac{2 \cdot \pi \cdot a}{N_b}\right) \cdot \left(\frac{\sigma \cdot t^2}{6 \cdot L}\right)$$

F_M = 12,929 lb

where:

a = the outer radius of the bolt circle, 37.28 in.,

t = the thickness of the ring, 0.75 in.,

σ = the radial bending stress at point *a*, σ_{ra} = 23,550 psi, and

L = the distance between the bolt center line and ring outer edge, c - a = 1.25 in.

The total tension, F, on each bolt is

$$F = F_F + F_M = 17{,}191 \text{ lb}$$

Knowing the bolt cross-sectional area, A_b, the bolt tensile stress is calculated as:

$$\sigma_t = \frac{F}{A_b} = 38{,}912 \text{ psi}$$

where:

$A_b = 0.4418 \text{ in}^2$

For off-normal conditions, the allowable primary membrane stress in a bolt is $2S_m$. The allowable stress for SA-193 Grade B6 bolts is 54 ksi at 120°F, the maximum temperature of the transfer cask top plate. The margin of safety for the bolts is

$$MS = \frac{54{,}000}{38{,}912} - 1 = +0.38$$

Since the SA-193 Grade B6 bolts have higher strength than the top plate, the shear stress in the threads of the top plate is evaluated. The yield and ultimate strengths for the top plate ASTM A-588 material at a temperature of 120°F are:

S_y = 49.5 ksi

S_u = 70.0 ksi

From Reference 27, the shear area for the internal threads of the top plate, A_n , is calculated as:

$$A_n \quad = \quad 3.1416\, n\, L_e\, D_s min\left[\frac{1}{2n} + 0.57735\left(D_s min - E_n max\right)\right] = 1.525 \text{ in}^2$$

where:

D = 0.7482 in., basic major diameter of bolt threads,
n = 10, number of bolt threads per inch,
$D_s min$ = 0.7353 in., minimum major diameter of bolt threads,
$E_n max$ = 0.6927 in., maximum pitch diameter of lid threads, and
L_e = 1.625-0.74=0.885 in., minimum thread engagement.

The shear stress (τ_n) in the top plate is:

$$\tau_n = \frac{F}{A_n} = \frac{17{,}191 \text{ lb}}{1.525 \text{ in}^2} = 11{,}273 \text{ psi}$$

Where the total tension, F, on each bolt is

$$F = F_F + F_M = 17{,}191 \text{ lb}$$

The shear allowable for normal conditions is conservatively used:

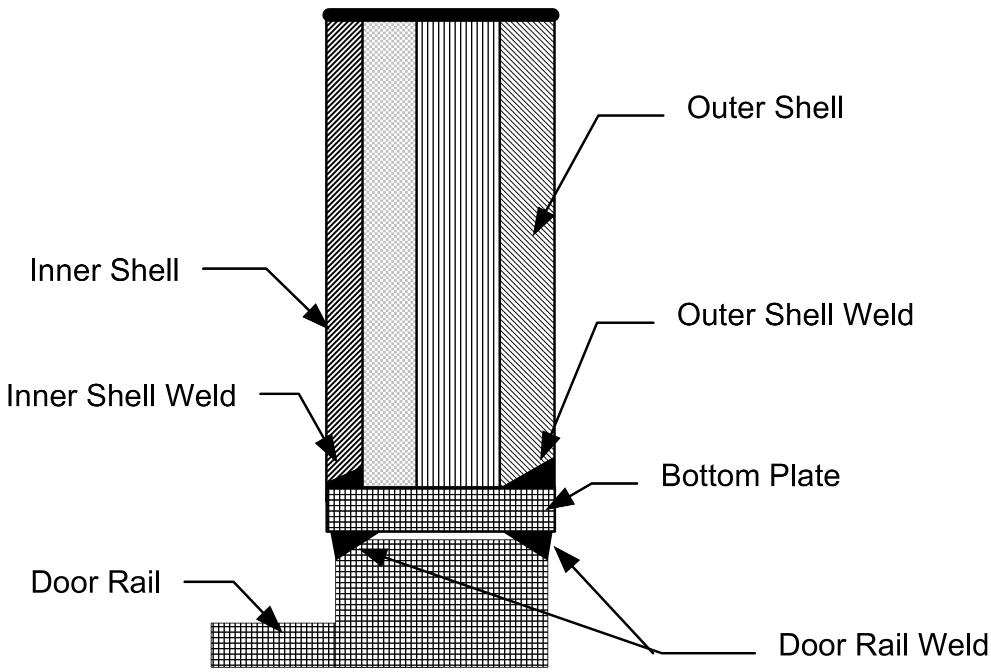
$$\tau_{allowable} = (0.6)\,(S_m) = (0.6)\,(23.3 \text{ ksi}) = 13.98 \text{ ksi}$$

The Margin of Safety is: $\frac{13{,}980}{11{,}273} - 1 = +0.24$

Therefore, the threads of the top plate will not fail in shear.

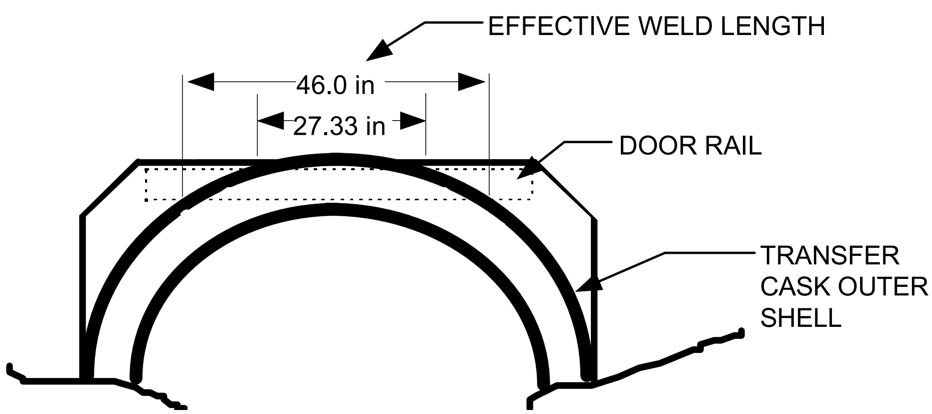
3.4.3.3.3 Bottom Plate Weld Analysis

The bottom plate is connected to the outer and inner shell of the transfer cask by full penetration welds. The weight of a loaded canister along with the shield door rail structure is transmitted from the bottom plate to the shell via the full penetration weld. For conservatism, only the length of the weld directly under the shell is considered effective in transmitting a load.



The weld connecting the outer and inner shell to the bottom plate has a length of approximately

l_w = (27.33 in. + 46.0 in.)/2 in. = 36.66 in.



Stresses occurring in the outer shell to bottom plate weld are evaluated using a weight, W = 131,800 lb × 1.1 = 145,000 lb, which bounds the weight of the heaviest loaded canister, the weight of the water, and the weight of the shield doors and rails, with a 10% dynamic load factor.

The door rail structure and canister load will be transmitted to both the inner and outer shell via full penetration welds. The thickness of the two shells and welds are different; however, for conservatism, this evaluation assumes both shell welds are 0.75 in. groove welds.

Weld effective area = (36.66 in.)(0.75 in. + 0.75 in.) = 54.99 in^2

$$\sigma_{axial} = \frac{P}{A} = \frac{(145{,}000\ \text{lb})/(2)}{54.99\ \text{in}^2} = 1{,}318\ \text{psi}$$

For the bottom plate material (ASTM A-588) at a bounding temperature of 400°F, the yield and ultimate stresses are:

S_y = 43.0 ksi

S_u = 70.0 ksi

$$FS_{yield} = \frac{43.0}{1.32} = +32.6 > 6$$

$$FS_{ultimate} = \frac{70.0}{1.32} = +53.0 > 10$$

Thus, the welds in the bottom plate meet the ANSI N14.6 and NUREG-0612 criteria for nonredundant systems.

3.4.3.3.4 Standard Transfer Cask Shield Door Rails and Welds

This section demonstrates the adequacy of the transfer cask shield doors, door rails, and welds in accordance with NUREG-0612 and ANSI N14.6, which require safety factors of 6 and 10 on material yield strength and ultimate strength, respectively, for nonredundant lift systems.

The shield door rails support the weight of a wet, fully loaded canister and the weight of the shield doors themselves. The shield doors are 9.0-in. thick plates that slide on the door rails. The rails are 9.38 in. deep × 6.5 in. thick and are welded to the bottom plate of the transfer cask. The doors and the rails are constructed of A-588 and A-350 Grade LF 2 low alloy steel, respectively.

The design weight used in this evaluation, W = 131,800 × 1.1 ≈ 145,000 pounds, is an assumed value that bounds the weight of the heaviest loaded canister, the weight of the water in the canister and the weight of the shield doors and rails. A 10% dynamic load factor is included to ensure that the evaluation bounds all normal operating conditions. This evaluation shows that the door rail structures and welds are adequate to support the design input.

Allowable stresses for the material are taken at 400°F, which bounds the maximum temperature at the bottom of the transfer cask under normal conditions. The material properties of A-588 and A-350 Grade LF 2 low alloy steel are provided in Tables 3.3-8 and 3.3-9, respectively. The standard impact test temperature for ASTM A-350, Grade LF2 is -50°F. The NDT temperature range is -70°F to -10°F for ASTM A-588 with a thickness range of 0.625 in. to 3 in. [28]. Therefore, the minimum service temperature for the trunnion and shells is conservatively established as 0°F (50°F higher than the NDT test temperature, in accordance with Section 4.2.6 of ANSI N14.6 [9]. For conservatism, the stress allowables for A-350 Grade LF 2 are used for all stress calculations.

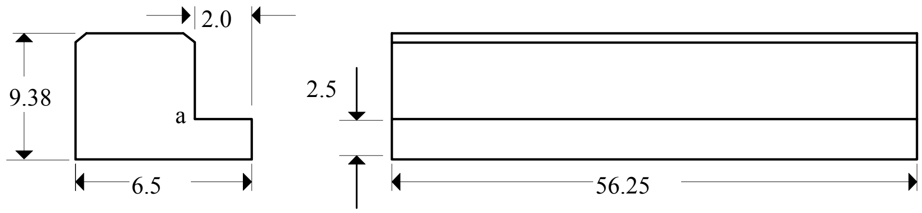
<u>Stress Evaluation for Door Rail</u>

Each rail is assumed to carry a uniformly distributed load equal to 0.5W. The shear stress in each door rail bottom plate due to the applied load, W, is:

$$\tau = \frac{W}{A} = \frac{145{,}000\ \text{lb}}{281.25\ \text{in}^2} = 516\ \text{psi}$$

where:

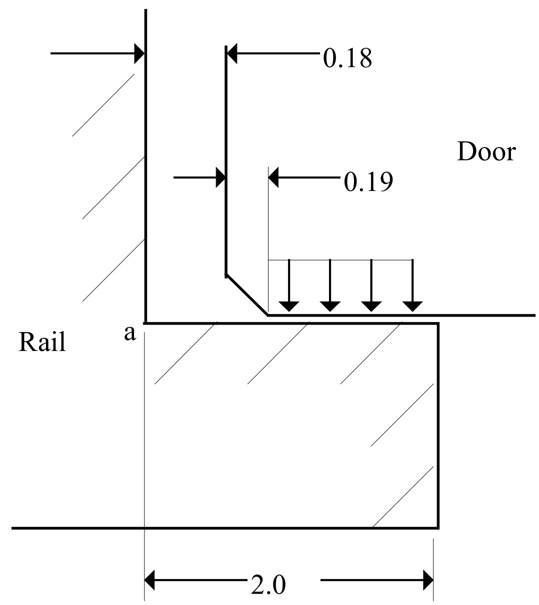
$A = 2.5\ \text{in.} \times 56.25\ \text{in. length/rail} \times 2\ \text{rails} = 281.25\ \text{in}^2$.



The bending stress in each rail bottom section due to the applied load of W is:

$$\sigma_b = \frac{6M}{bt^2} = \frac{6 \times 86{,}275}{56.25 \times 2.5^2} = 1{,}472 \text{ psi},$$

where:



M = moment at *a*,

$$= \frac{W}{2} \times \mathcal{L} = \frac{145{,}000 \text{ lb.}}{2} \times 1.19 \text{ in.}$$

= 86,275 in-lb,

and,

$$\mathcal{L} = 2 - \frac{2 - (0.18 + 0.19)}{2}$$

$\mathcal{L}$ = 1.19 in., applied load moment arm.

The maximum principal stress in the bottom section of the rail is:

$$\sigma = \left(\frac{\sigma_b}{2}\right) + \sqrt{\left(\frac{\sigma_b}{2}\right)^2 + \tau^2}$$

= 1,635 psi

The acceptability of the rail design is evaluated by comparing the allowable stresses to the maximum calculated stresses, considering the safety factors of NUREG-0612 and ANSI N14.6. For the yield strength criteria:

$$\frac{30{,}800\ \text{psi}}{1{,}635\ \text{psi}} = 18.8 > 6$$

For the ultimate strength criteria,

$$\frac{70{,}000\ \text{psi}}{1{,}635\ \text{psi}} = 42.8 > 10$$

The safety factors meet the criteria of NUREG-0612. Therefore, the rails are structurally adequate.

Stress Evaluation for the Shield Doors

The shield doors consist of a layer of NS-4-FR neutron shielding material sandwiched between low alloy steel plates (Note: steel bars are also welded on the edges of the doors so that the neutron shielding material is fully encapsulated). The door assemblies are 9-inch thick at the center and 6.75-inch thick at the edges, where they slide on the support rails. The stepped edges of the two door leaves are designed to interlock at the center and are, therefore, analyzed as a single plate that is simply supported on two sides.

The shear stress at the edge of the shield door where the door contacts the rail is:

$$\tau = \frac{W}{2 \times A_S} = \frac{145{,}000\ \text{lb}}{2 \times (49.2\ \text{in.} \times 4.75\ \text{in.})} = 310\ \text{psi}$$

where:

A = the total shear area, 4.75 in. thick × 49.2 in. long. Note that the effective thickness at the edge of the doors is taken as 4.75 in. because the neutron shield material and the cover plate are assumed to carry no shear load. The shear stress at the center of the doors approaches 0 psi.

The moment equation for the simply-supported beam with uniform loading is:

$$M = 72{,}500\ X - 2{,}031(X)(0.5\ X) = 72{,}500\ X - 1{,}015\ X^2$$

The maximum bending moment occurs at the center of the doors, X = 35.7 in. The bending moment at this point is:

$$M = 72{,}500 \text{ lb} \times (35.7 \text{ in.}) - 1{,}015 \text{ lb/in.} \times (35.7 \text{ in})^2$$
$$M = 12.95 \times 10^5 \text{ in.-lb.}$$

The maximum bending stress, σ_{max}, at the center of the doors, is

$$\sigma_{ax} = \frac{Mc}{I} = \frac{12.95 \times 10^5 \text{ in.} - \text{lb} \times 5.5 \text{ in.}}{2{,}378 \text{ in.}^4} = 2{,}995 \text{ psi}$$

where:

$$c = \frac{h}{2} = \frac{7 \text{ in.}}{2} + 2 \text{ in.} = 5.5 \text{ in.}, \text{ and}$$

$$I = \frac{bh^3}{12} = \frac{83.2 \text{ in.} \times 7^3 \text{ in}}{12} = 2378 \text{ in}^4.$$

The acceptability of the door design is evaluated by comparing the allowable stresses to the maximum calculated stresses. As shown above, the maximum stress occurs for bending.

For the yield strength criteria,

$$\frac{30{,}800 \text{ psi}}{2{,}995 \text{ psi}} = 10.3 > 6$$

For the ultimate strength criteria,

$$\frac{70{,}000 \text{ psi}}{2{,}995 \text{ psi}} = 23.4 > 10$$

The safety factors satisfy the criteria of NUREG-0612. Therefore, the doors are structurally adequate.

Door Rail Weld Evaluation

The door rails are attached to the bottom of the transfer cask by 0.625-in. partial penetration bevel groove welds that extend the full length of the inside and outside of each rail. If the load is conservatively assumed to act at a point on the inside edge of the rail, the load, P, on each rail is,

$$P = \frac{W}{2} = \frac{145{,}000 \text{ lb}}{2} = 72{,}500 \text{ lb}$$

Summing moments about the inner weld location:

$$0 = P \times a - F_o \times (b) = 72{,}500 \text{ lb} \times 1.19 \text{ in.} - F_o\,(4.5 \text{ in.}), \text{ or}$$

$$F_o = 19{,}172 \text{ lb}$$

Summing forces:

$$F_i = F_o + P = 19{,}172 \text{ lb} + 72{,}500 \text{ lb} = 91{,}672 \text{ lb}$$

The effective area of the inner weld is $0.625 \text{ in} \times 56.25 \text{ in. long} = 35.16 \text{ in}^2$

The shear stress, τ, in the inner weld is

$$\tau = \frac{91{,}672 \text{ lb}}{35.16 \text{ in}^2} = 2{,}607 \text{ psi}$$

The factors of safety are

$$\frac{30{,}800 \text{ psi}}{2{,}607 \text{ psi}} = 11.8 > 6 \qquad \text{(for yield strength criteria)}$$

$$\frac{70{,}000 \text{ psi}}{2{,}607 \text{ psi}} = 26.8 > 10 \qquad \text{(for ultimate strength criteria)}$$

The safety factors meet the criteria of NUREG-0612.

3.4.3.3.5 PWR Class 1 Standard Transfer Cask with Transfer Cask Extension

The PWR Class 1 standard transfer cask, baseline weight of 112,300 lb. empty, can be equipped with a Transfer Cask extension to accommodate the loading of a PWR Class 2 canister. The purpose of the extended transfer cask configuration is to permit the loading of PWR Class 1 fuel assemblies with Control Element Assemblies inserted into a PWR Class 2 canister; the length of the control element assemblies requires the use of the longer PWR Class 2 canister. The weight of the transfer cask extension is 5,500 pounds. Therefore, the total weight of the PWR Class 1 transfer cask with extension would be:

$$W_{TC} = 112{,}300 + 5{,}500 = 117{,}800 \text{ lbs}$$

Standard Transfer Cask Shell and Trunnion

From the analysis in Section 3.4.3.3.1 for the Transfer Cask Shell and Trunnion, the heaviest loaded transfer cask weight used in the analysis was 210,000 pounds (Class 5 BWR). The total weight of the loaded transfer cask with extension is:

$$W_{TC\text{-}L} = 193{,}900 + 5{,}500 = 199{,}400 \text{ lbs}$$

where:

193,900 lbs = the weight of a PWR Class 1 transfer cask and canister (with fuel, water, and shield lid)

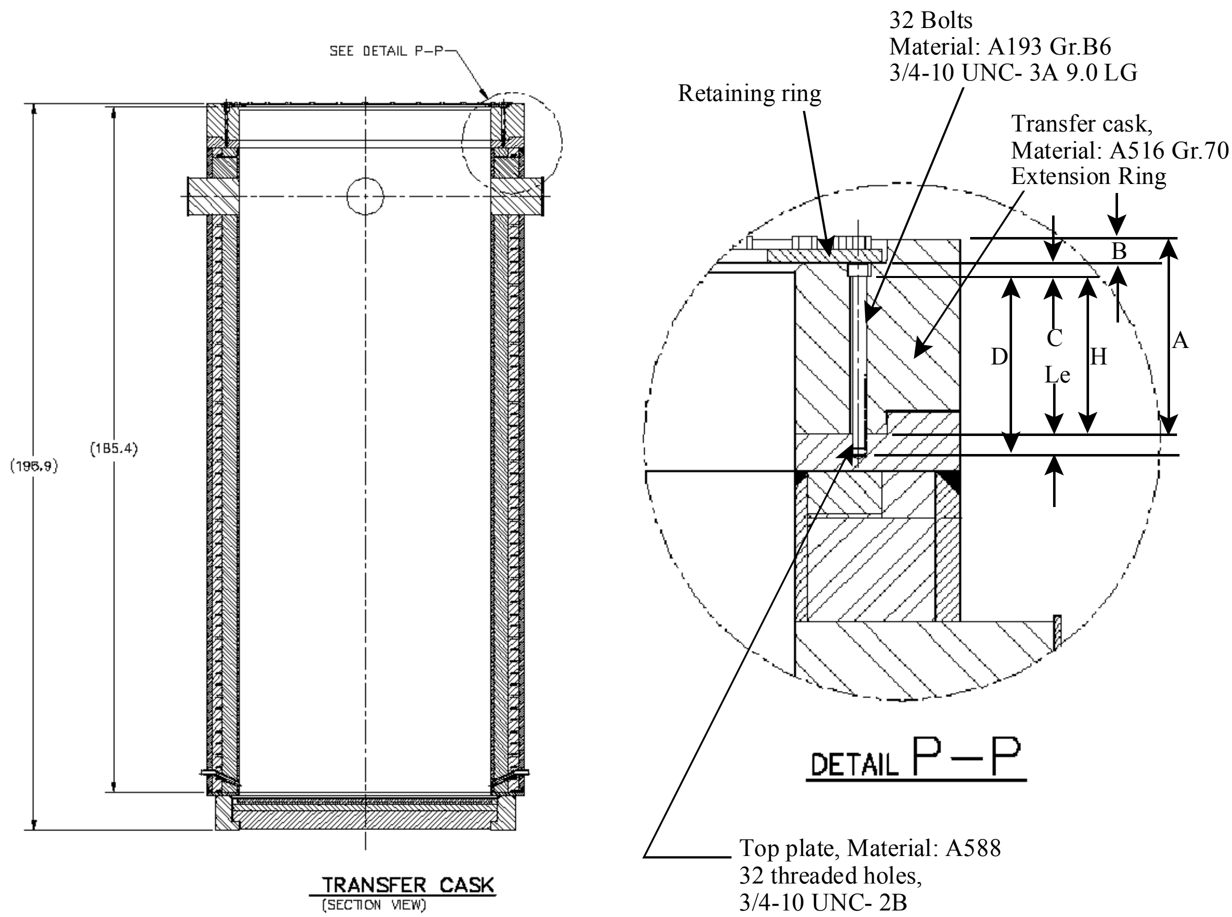
The Class 5 BWR transfer cask configuration bounds the PWR Class 1 transfer cask with extension; therefore, no additional handling analysis is required for the transfer cask shell and trunnions.

Retaining Ring and Bolts

From Section 3.4.3.3.2, the bounding transfer cask weight used was 124,000 pounds. As stated above, the weight of the PWR Class 1 transfer cask with extension is 117,800 pounds; therefore, the existing analysis in Section 3.4.3.3.2 bounds the PWR Class 1 transfer cask with extension and no additional analysis is required.

Standard Transfer Cask Extension Attachment Bolts

The transfer cask extension is attached to the transfer cask by 32 bolts that are identical to the retaining ring bolts with the exception of bolt length. The transfer cask, the top plate, the retaining ring and the extension ring are shown in the following figure. The bolts are only loaded if the transfer cask is accidentally lifted by the retaining ring. In this condition, the only load experienced by the extension bolts is the weight of the transfer cask. The weight of the canister is transferred directly through the lift rig attached to the structural lid.



Referring to the preceding figure, the bolt engagement is calculated as follows:

A = 10.3 in. = extension ring thickness
B = 1.2 in = retaining ring seat recess depth
C = 0.81 in. = bolt head counter bore depth
D = 9 in. = bolt body length

The thickness (H) of the extension ring under the bolt head is calculated as:

$$H = A - B - C = 10.3 - 1.2 - 0.81 = 8.29 \text{ in.}$$

The thread engagement length, L_e, in the top plate is:

$$L_e = D - H = 9 - 8.29 = 0.71 \text{ in.}$$

The extension attachment bolts are 9.0 inches long. Since the thickness of the extension ring under the bolt head is 8.29 inches, the prying action is negligible for the transfer cask extension attachment bolts during an inadvertent lift of the transfer cask via the retaining ring during a canister handling operation. The PWR Class 1 Transfer Cask with extension weighs approximately 7,000 pounds less than the bounding analysis weight. A bounding load of 124,000 pounds is conservatively used for this analysis.

The total load (P) applied to each extension bolt is the weight of the transfer cask divided by the number of bolts:

$$P = \frac{(124{,}000)(1.1)}{32} = 4{,}263 \text{ lbs per bolt}$$

The multiplication factor of 1.1 accounts for the dynamic load factor (DLF). From "Machinery's Handbook" [27], the shear area of the external threads (A_s) in the bolt is calculated as:

$$A_s = (3.1416)\, n\, L_e K_n \max\left[\frac{1}{2n} + 0.57735\left(E_s \min - K_n \max\right)\right] = 0.89 \text{ in}^2$$

and the shear area (A_n) for the internal threads of the bolt is calculated as:

$$A_n = (3.1416)\, n\, L_e D_s \min\left[\frac{1}{2n} + 0.57735\left(D_s \min - E_n \max\right)\right] = 1.244 \text{ in}^2$$

where:

$K_n \max = 0.663$ in = maximum minor diameter- internal thread for 3/4 10-UNC-2B

$E_s \min = 0.6806$ in = minimum pitch diameter-external thread for 3/4 10-UNC-3A

$D_s \min = 0.7371$ in = minimum major diameter-external thread for 3/4 10-UNC-3A

E_n max = 0.6927 in = maximum pitch diameter-internal thread for 3/4 10-UNC-2B
L_e = 0.71 in. = length of thread engagement
n = 10 = number of thread per inch

The shear stress (τ_s) on the threads of the bolt is:

$$\tau_s = \frac{4263}{0.89} = 4{,}791 \text{ psi}$$

The allowable stress of ASTM A193 GR B6 at 120°F for pure shear is used.

$$\tau_{allowable} = (0.6)\ (S_m) = (0.6)\ (27.8 \text{ ksi}) = 16.68 \text{ ksi}$$

The margin of safety is $\frac{16.68}{4.79} - 1 = +\ 2.48$

The shear stress (τ_n) on the threads in the bolt hole is:

$$\tau_n = \frac{4263}{1.244} = 3{,}427 \text{ psi}$$

The allowable stress of ASTM A-588 at 120°F for pure shear is used.

$$\tau_{allowable} = (0.6)\ (S_m) = (0.6)\ (23.3 \text{ ksi}) = 13.98 \text{ ksi}$$

The margin of safety is $\frac{13.98}{3.427} - 1 = +\ 3.08$

Figure 3.4.3.3-1 Finite Element Model for Standard Transfer Cask Trunnion and Shells

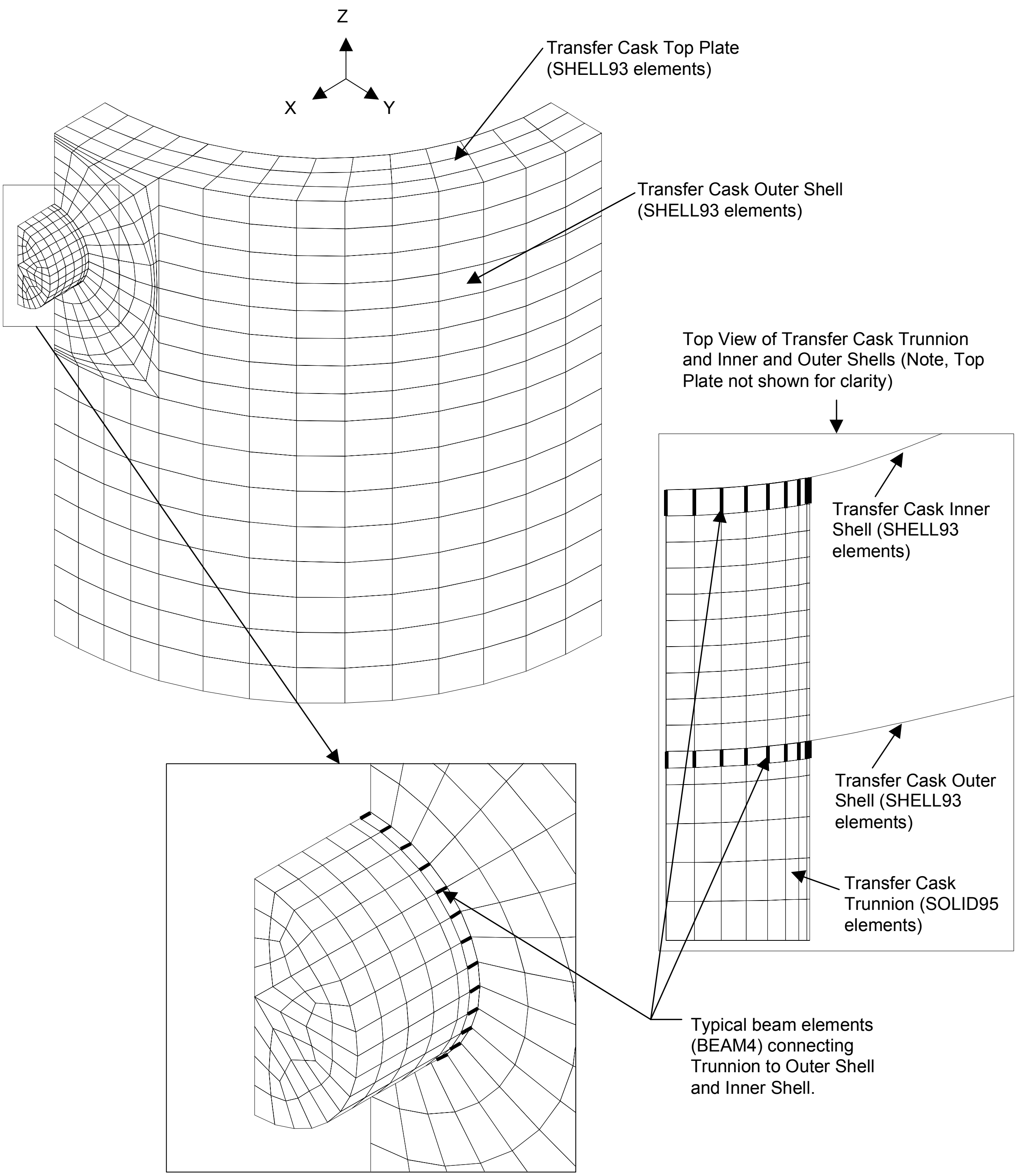


Figure 3.4.3.3-2 Node Locations for Standard Transfer Cask Outer Shell Adjacent to Trunnion

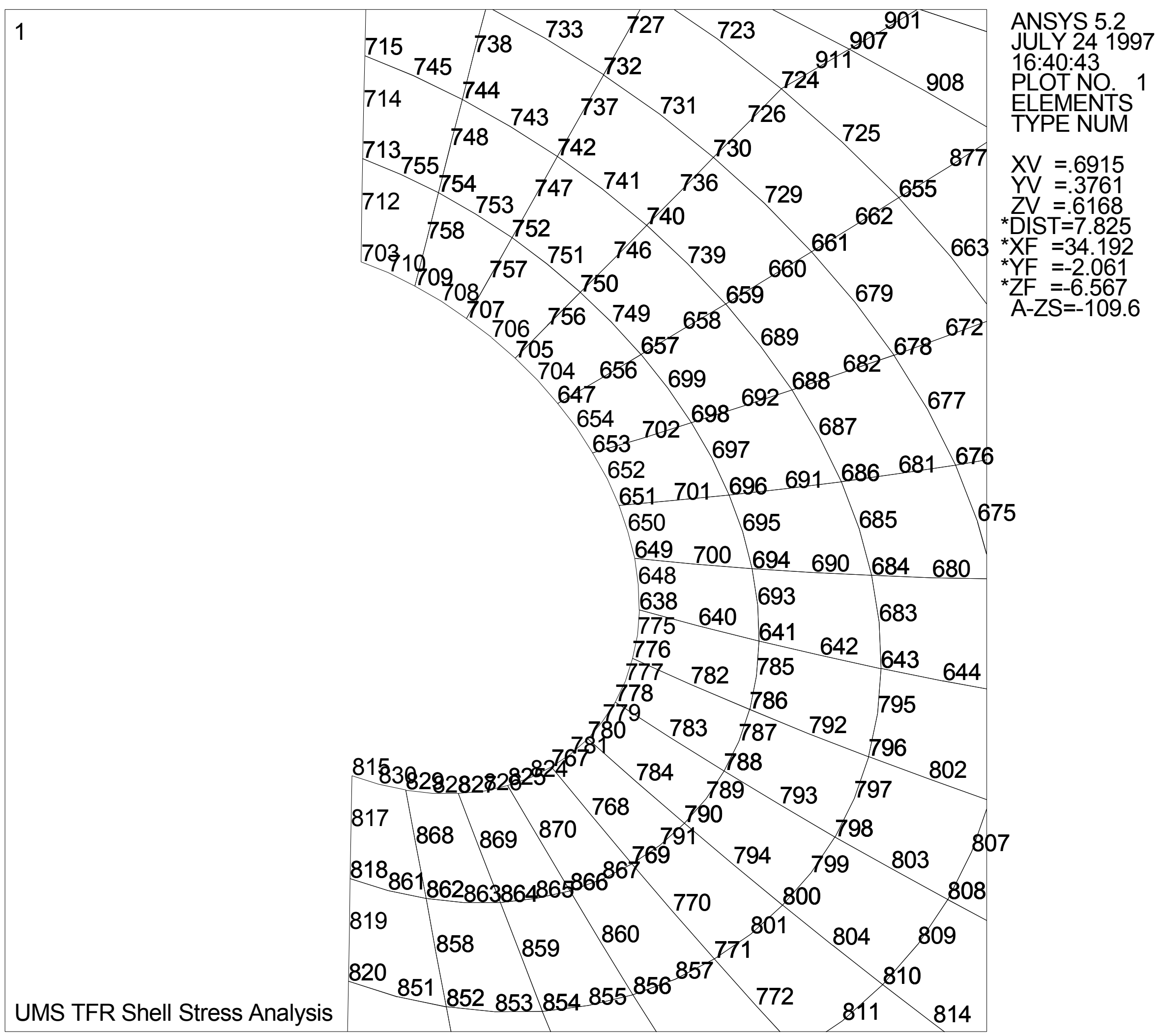
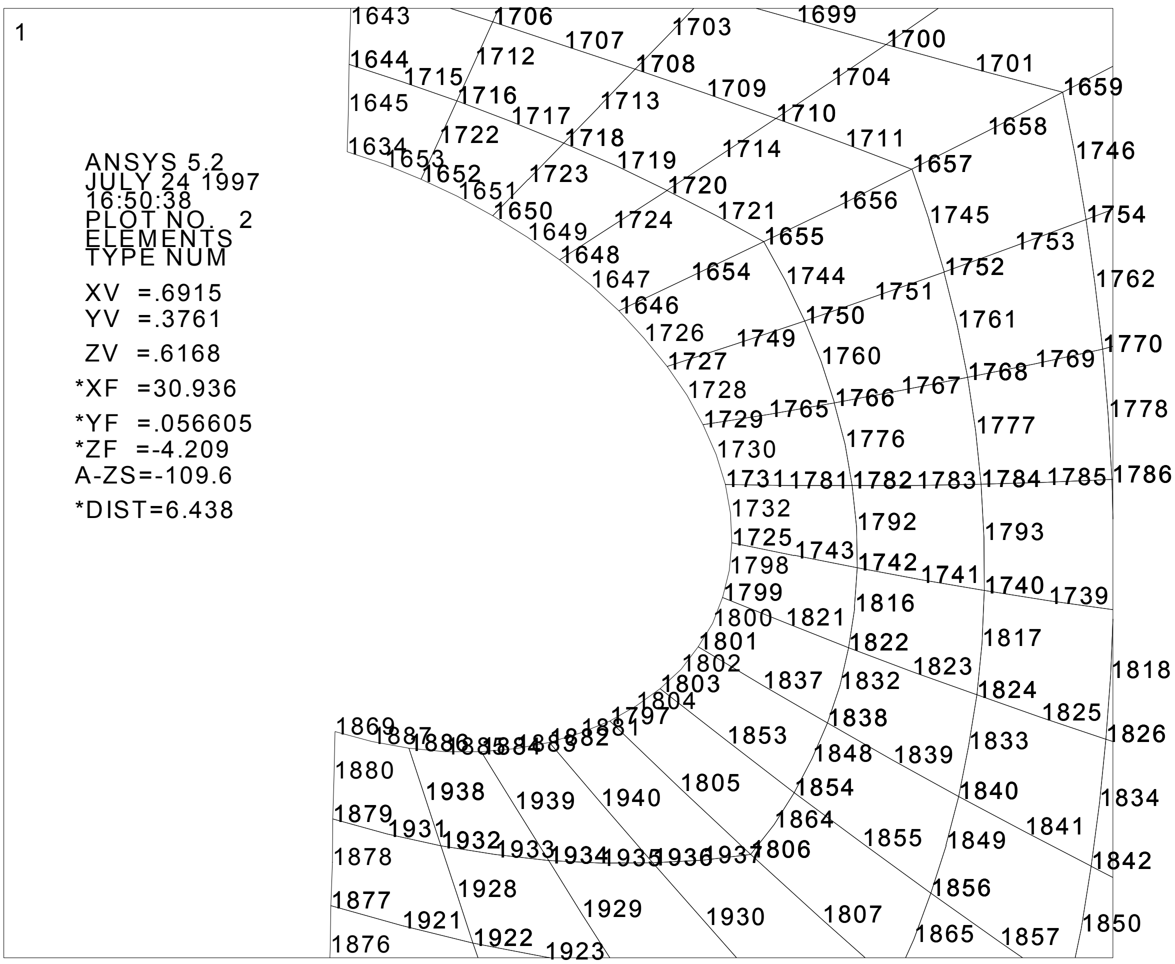


Figure 3.4.3.3-3 Node Locations for Standard Transfer Cask Inner Shell Adjacent to Trunnion



UMS TFR Shell Stress Analysis

Figure 3.4.3.3-4 Stress Intensity Contours (psi) for Standard Transfer Cask Outer Shell Element Top Surface

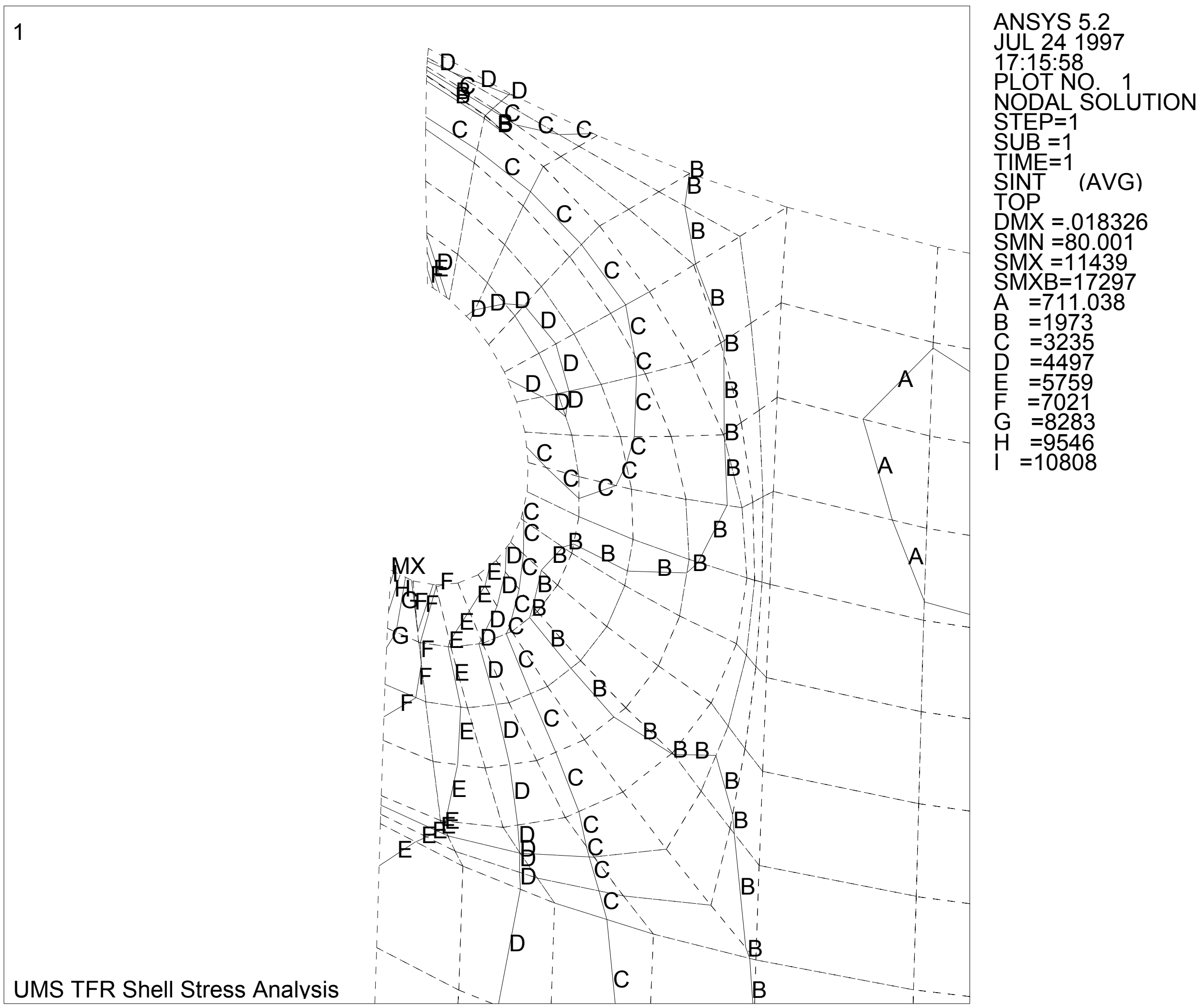


Figure 3.4.3.3-5 Stress Intensity Contours (psi) for Standard Transfer Cask Outer Shell Element Bottom Surface

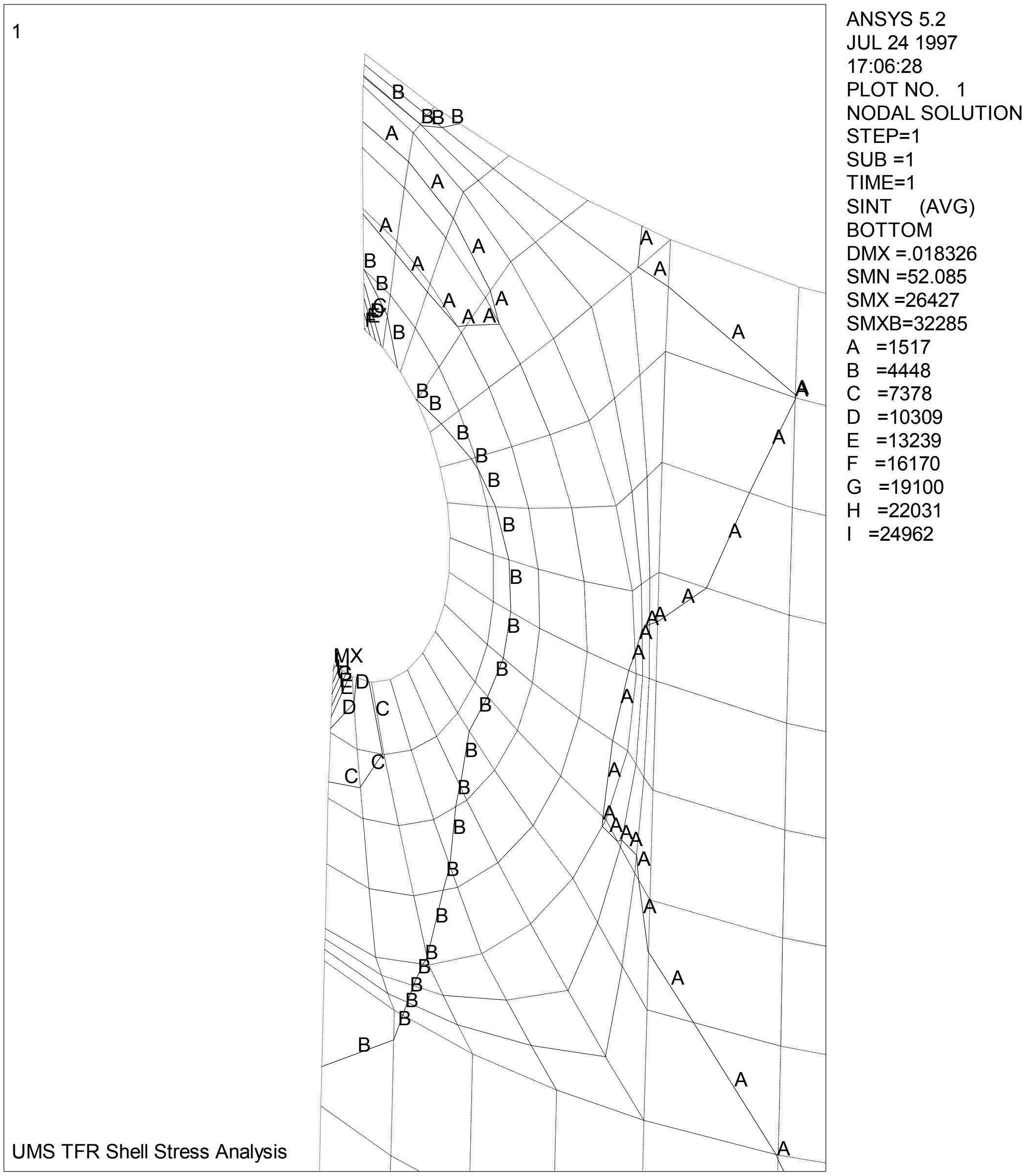


Figure 3.4.3.3-6 Stress Intensity Contours (psi) for Standard Transfer Cask Inner Shell Element Top Surface

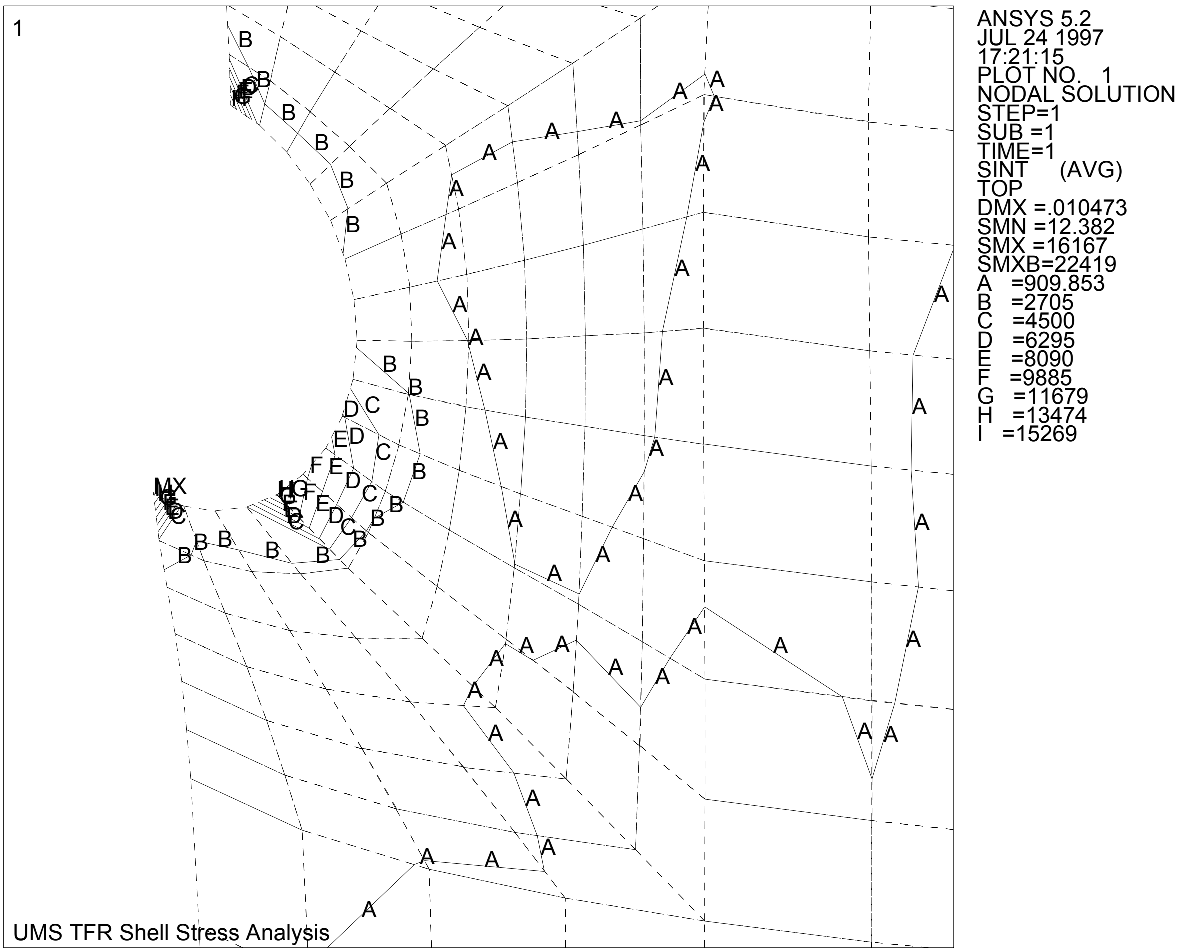


Figure 3.4.3.3-7 Stress Intensity Contours (psi) for Standard Transfer Cask Inner Shell Element Bottom Surface

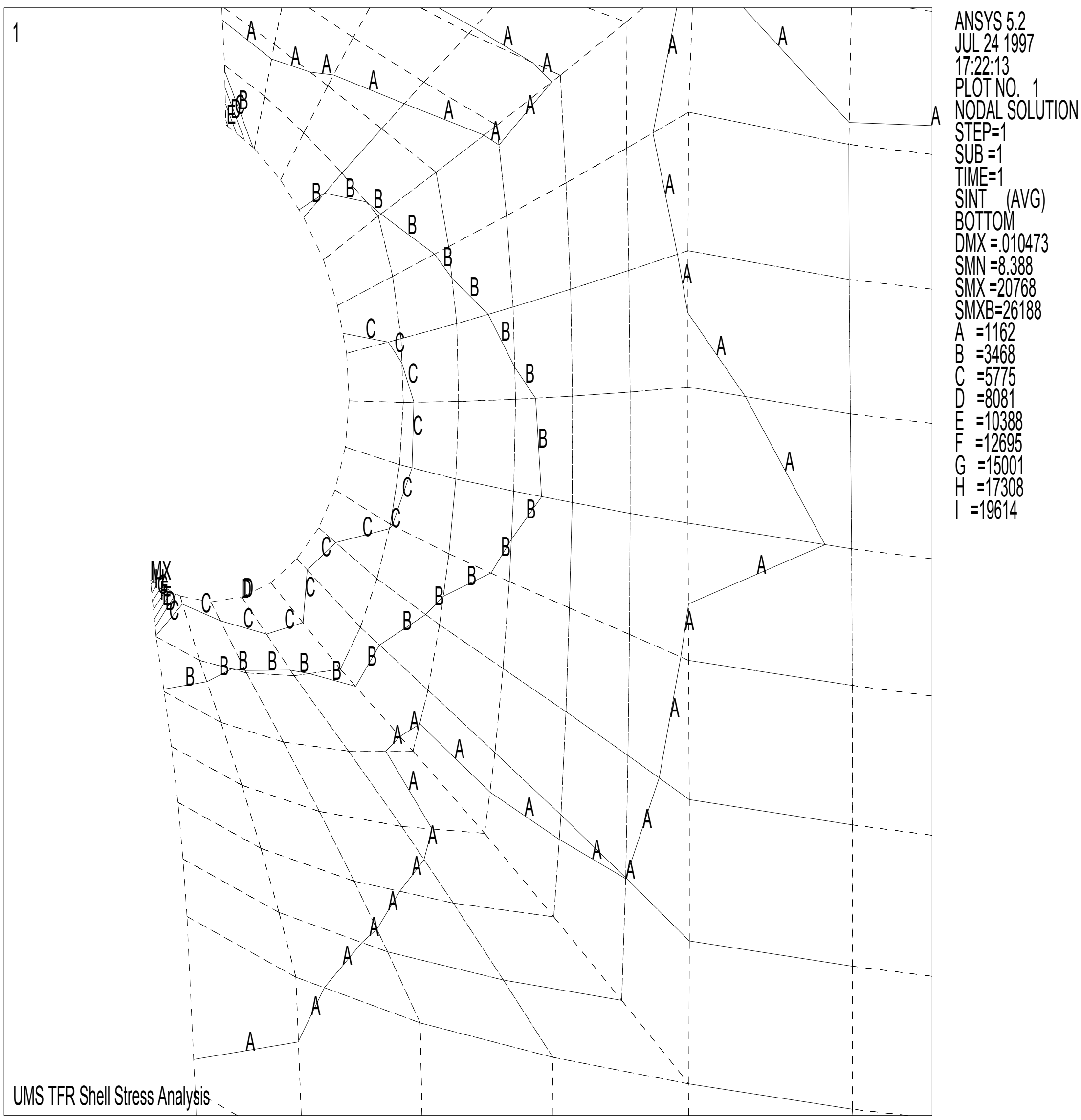


Table 3.4.3.3-1 Top 30 Stresses for Standard Transfer Cask Outer Shell Element Top Surface

Node[1]	Principal Stresses(psi)			Nodal S.I. (psi)	F.S. on Yield S_y/S.I.[2]	F.S. on Ultimate (S_u/S.I.)[2]
	S1	S2	S3			
815	3521.5	-288.8	-7917.2	11439.0	N/A[3]	N/A[3]
818	5092.6	-4.7	-3640.3	8732.9	N/A	N/A
703	7056.8	719.0	-995.8	8052.5	N/A	N/A
820	4315.2	-2.5	-3128.0	7443.2	N/A	N/A
862	4091.0	3.8	-3005.9	7096.9	N/A	N/A
827	4908.7	8.5	-2161.6	7070.3	N/A	N/A
825	4727.4	39.0	-2214.8	6942.2	6.6	10.1
852	4134.8	0.7	-2756.8	6891.6	6.6	10.2
822	3927.3	-0.3	-2788.6	6716.0	6.8	10.4
829	3525.9	-15.5	-3132.6	6658.6	6.8	10.5
767	4010.9	111.0	-2445.3	6456.2	7.1	10.8
842	3806.4	0.2	-2475.5	6281.9	7.3	11.1
816	3607.1	-0.1	-2644.0	6251.1	7.3	11.2
943	3547.6	-0.1	-2638.2	6185.8	7.4	11.3
941	3495.7	-0.1	-2626.5	6122.2	7.4	11.4
2	3430.3	0.0	-2609.0	6039.3	7.6	11.6
832	3497.2	0.2	-2341.5	5838.7	7.8	12.0
964	3412.4	0.3	-2271.0	5683.3	8.0	12.3
864	3625.6	15.6	-2002.0	5627.7	8.1	12.4
854	3683.9	3.6	-1853.7	5537.7	8.2	12.6
954	3335.5	0.3	-2199.9	5535.4	8.2	12.6
8	3251.5	0.1	-2132.4	5383.9	8.5	13.0
780	2941.0	173.8	-2411.8	5352.8	8.5	13.1
871	5250.1	2907.8	-23.4	5273.6	8.6	13.3
47	2848.5	0.0	-2367.8	5216.3	8.7	13.4
844	3470.2	2.3	-1701.8	5172.0	8.8	13.5
657	2272.2	-18.5	-2625.5	4897.7	9.3	14.3
57	2781.3	-0.3	-2093.2	4874.5	9.4	14.4
705	3143.0	-323.9	-1675.6	4818.6	9.5	14.5
834	3227.7	1.9	-1578.1	4805.7	9.5	14.6

Notes:

1. See Figure 3.4.3.3-2 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.3-2 Top 30 Stresses for Standard Transfer Cask Outer Shell Element Bottom Surface

Node[1]	Principal Stresses(psi)			Nodal S.I. (psi)	F.S. on Yield S_y/S.I. [2]	F.S. on Ultimate (S_u/S.I.) [2]
	S1	S2	S3			
815	26042.0	1368.5	-385.3	26427.0	N/A[3]	N/A[3]
703	433.6	-1196.0	-16049.0	16482.0	N/A	N/A
829	11257.0	4762.2	-25.6	11283.0	N/A	N/A
818	9377.2	1335.4	-11.0	9388.2	N/A	N/A
862	8650.9	2600.4	-13.1	8663.9	N/A	N/A
638	3906.5	-37.6	-3390.4	7296.9	N/A	N/A
864	7245.0	2309.2	-13.3	7258.4	N/A	N/A
776	5054.5	156.6	-1993.6	7048.1	N/A	N/A
649	2372.4	-306.3	-4436.1	6808.5	6.7	10.3
827	6731.4	2737.4	-15.4	6746.9	6.8	10.4
820	6699.0	2463.6	-1.6	6700.6	6.8	10.4
778	5550.7	521.4	-837.7	6388.4	7.1	11.0
852	6375.9	2277.2	-3.5	6379.4	7.1	11.0
709	78.1	-4994.3	-6150.1	6228.2	7.3	11.2
825	6070.4	2367.2	-42.8	6113.2	7.5	11.5
651	1180.6	-998.2	-4879.3	6060.0	7.5	11.6
780	5703.3	1363.7	-312.2	6015.5	7.6	11.6
866	5998.4	1528.3	-1.7	6000.1	7.6	11.7
767	5772.1	2120.8	-131.9	5904.0	7.7	11.9
871	20.8	-416.7	-5855.7	5876.6	7.8	11.9
854	5737.9	1707.3	-4.5	5742.4	7.9	12.2
822	5656.1	1990.6	-0.3	5656.4	8.1	12.4
653	689.6	-2286.6	-4882.7	5572.3	8.2	12.6
842	5453.5	1832.8	-0.8	5454.3	8.4	12.8
873	20.0	-243.1	-5388.0	5408.0	8.4	12.9
769	5322.5	815.7	1.0	5321.5	8.6	13.2
641	3174.6	1.8	-1987.0	5161.6	8.8	13.6
786	3830.7	0.4	-1282.9	5113.5	8.9	13.7
694	2454.1	4.2	-2655.5	5109.6	8.9	13.7
816	5070.5	1851.7	-0.1	5070.6	9.0	13.8

Notes:

1. See Figure 3.4.3.3-2 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.3-3 Top 30 Stresses for Standard Transfer Cask Inner Shell Element Top Surface

Node[1]	Principal Stresses(psi)			Nodal S.I. (psi)	F.S. on Yield S_y/S.I.[2]	F.S. on Ultimate (S_u/S.I.)[2]
	S1	S2	S3			
1869	1765.2	-503.6	-14402.0	16167.0	N/A[3]	N/A[3]
1797	11044.0	-108.1	-2767.4	13811.0	N/A	N/A
1634	1615.7	-326.8	-12092.0	13708.0	N/A	N/A
1803	10114.0	3278.4	-293.2	10407.0	N/A	N/A
1801	8800.8	3432.8	-213.3	9014.1	N/A	N/A
1799	6238.1	3249.0	-161.2	6399.3	7.1	10.9
1882	728.3	-2351.9	-3701.0	4429.3	10.3	15.8
1633	4070.8	551.7	-1.6	4072.3	11.2	17.2
1879	350.0	-116.5	-3650.0	4000.0	11.4	17.5
1725	3690.7	2859.1	-166.8	3857.5	11.8	18.1
1648	485.8	-261.7	-3244.6	3730.5	12.2	18.8
1652	137.0	-1003.2	-3529.2	3666.2	12.4	19.1
1886	101.9	-2993.0	-3541.1	3643.1	12.5	19.2
1644	962.4	-24.8	-2674.1	3636.5	12.5	19.2
1650	433.9	11.7	-3137.7	3571.6	12.8	19.6
1884	416.6	-1841.5	-3125.6	3542.1	12.9	19.8
1666	3474.7	386.0	-0.3	3475.0	13.1	20.1
1822	3435.6	2108.1	-17.9	3453.6	13.2	20.3
1646	311.6	-945.1	-2960.5	3272.1	13.9	21.4
1838	3148.2	2452.5	-35.3	3183.5	14.3	22.0
1636	3157.0	750.3	-2.3	3159.3	14.4	22.2
1676	2879.2	707.8	-2.4	2881.6	15.8	24.3
1742	2725.1	1367.2	-8.9	2734.0	16.7	25.6
1727	308.8	-540.4	-2300.1	2608.9	17.5	26.8
1668	2486.6	121.0	-10.4	2496.9	18.3	28.0
1854	2393.3	2044.3	-55.4	2448.7	18.6	28.6
1731	2185.5	1530.9	-262.9	2448.4	18.6	28.6
1936	152.0	-126.5	-2235.5	2387.5	19.1	29.3
1638	2372.8	486.1	-2.7	2375.6	19.2	29.5
1120	4.2	-759.8	-2344.0	2348.2	19.4	29.8

Notes:
1. See Figure 3.4.3.3-3 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.3-4 Top 30 Stresses for Standard Transfer Cask Inner Shell Element Bottom Surface

Node[1]	Principal Stresses(psi)			Nodal S.I. (psi)	F.S. on Yield S_y/S.I.[2]	F.S. on Ultimate (S_u/S.I.)[2]
	S1	S2	S3			
1869	18955.0	554.4	-1812.1	20768.0	N/A[3]	N/A[3]
1634	10094.0	530.6	-887.6	10982.0	N/A	N/A
1882	7550.5	886.3	-631.4	8181.8	N/A	N/A
1797	1147.8	143.2	-5927.0	7074.8	N/A	N/A
1731	2320.8	-75.8	-4368.2	6689.0	6.8	10.5
1884	6149.9	517.9	-483.4	6633.3	6.9	10.6
1725	1242.9	-392.2	-5118.9	6361.8	7.2	11.0
1729	3117.2	52.5	-3023.5	6140.7	7.4	11.4
1803	474.7	-3926.6	-5631.6	6106.3	7.5	11.5
1886	5973.5	2440.1	-81.0	6054.5	7.5	11.6
1801	457.4	-3130.0	-5557.0	6014.4	7.6	11.6
1742	1965.5	-0.9	-4026.8	5992.3	7.6	11.7
1782	2451.4	-0.2	-3512.8	5964.2	7.6	11.7
1799	543.1	-1622.2	-5294.3	5837.4	7.8	12.0
1822	1595.1	4.2	-4233.9	5829.0	7.8	12.0
1766	2666.8	-1.0	-2994.6	5661.4	8.1	12.4
1879	5157.5	127.0	-284.2	5441.6	8.4	12.9
1727	3646.3	282.8	-1615.2	5261.4	8.7	13.3
1838	1426.6	25.3	-3770.7	5197.3	8.8	13.5
1740	2367.5	-2.5	-2661.6	5029.1	9.1	13.9
1784	2285.8	-0.7	-2712.6	4998.4	9.1	14.0
1750	2342.2	-6.7	-2516.2	4858.4	9.4	14.4
1646	3727.5	676.6	-1129.4	4856.9	9.4	14.4
1806	3417.2	95.3	-827.4	4244.6	10.7	16.5
1824	2109.9	-2.3	-2106.6	4216.5	10.8	16.6
1768	1813.3	-0.4	-2337.6	4150.9	11.0	16.9
1854	1304.9	49.1	-2746.8	4051.6	11.3	17.3
1738	2231.7	1.0	-1617.9	3849.6	11.8	18.2
1786	1897.7	0.5	-1860.4	3758.2	12.1	18.6
1932	3722.3	1449.3	-8.2	3730.5	12.2	18.8

Notes:

1. See Figure 3.4.3.3-3 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

3.4.3.4 Advanced Transfer Cask Lift

The Advanced transfer cask and Standard transfer cask are identical in design, except that the Advanced transfer cask incorporates a 0.75-inch thick support plate positioned above each of the trunnions between the inner shell and the outer shell. The support plate allows the Advanced transfer cask to lifting canisters weighing up to 98,000, whereas the Standard transfer cask is limited to canisters weighing up to 88,000 pounds. The 0.75-inch thick support plate is welded to the inner and outer shells of the Advanced transfer cask, adding significant rigidity to the shell-trunnion juncture to resist the loads applied during the lifting operation of the transfer cask. The welds attaching the support plate to the shells are 0.375-inch double-sided fillet welds at each end of the plate. The support plate is not attached to the trunnion, which prevents any significant shear force from being developed in the welds. The Advanced transfer cask analysis is conservatively based on a transfer cask contents weight of 103,000 pounds.

The evaluation of the Advanced transfer cask presented here shows that the design meets NUREG-0612 [8] and ANSI N14.6 [9] requirements for nonredundant lift systems. The adequacy of the standard transfer cask is shown by evaluating the stress levels in all of the load-path components against the NUREG-0612 criteria.

3.4.3.4.1 Advanced Transfer Cask Shell and Trunnion

The adequacy of the trunnions and the cask shell in the region around the trunnions during lifting conditions is evaluated in this section in accordance with NUREG-0612 and ANSI N14.6.

A three-dimensional finite element model is used to evaluate the lifting of a fully loaded Advanced transfer cask. Because of symmetry, it was necessary to model only one-quarter of the Advanced transfer cask, including the trunnions and the shells at the trunnion region. The stiffener plate above the trunnions (between the two shells) is included in the model. The lead and the NS-4-FR between the inner and outer shells of the Advanced transfer cask are neglected, since they are not structural components. SOLID95 (20 noded brick element) and SHELL93 (8 noded shell element) elements are used to model the trunnion and shells, respectively. Due to the absence of rotational degrees of freedom for the SOLID95 elements, BEAM4 elements perpendicular to the shells are used at the interface of the trunnion and the shells to transfer moments from the SOLID95 elements to SHELL93 elements. The finite element model is shown in Figure 3.4.3.4-1.

The total weight of the heaviest loaded Advanced transfer cask (Advanced Class 3 PWR) is calculated at approximately 217,300 pounds. A conservative load of 225,000 lb., plus a 10% dynamic load factor, is used in the model. The 225,000-pound load corresponds to an assumed transfer cask contents weight of 103,000 pounds. The load used in the quarter-symmetry model is (225,000 × 1.1)/4 = 61,875 pounds. The load is applied upward at the trunnion as a "surface load" whose location is determined by the lifting yoke dimensions. The model is restrained along two planes of symmetry with symmetry boundary conditions. Vertical restraints are applied to the bottom of the model to resist the force applied to the trunnion.

The maximum temperature in the Advanced transfer cask shell/trunnion region is conservatively evaluated as 300°F. For the ASTM A-588 shell material, the yield strength, S_y, is 45.6 ksi, and the ultimate strength, S_u, is 70 ksi. The trunnions are constructed of ASTM A-350 carbon steel, Grade LF2, with a yield stress of 31.9 ksi and an ultimate stress of 70 ksi. The standard impact test temperature for ASTM A-350, Grade LF2 is -50°F. The NDT temperature range is -70°F to -10°F for ASTM A-588 with a thickness range of 0.625 in. to 3 in. [25]. Therefore, the minimum service temperature for the trunnion and shells is conservatively established as -10°F (40°F higher than the NDT test temperature, in accordance with Section 4.2.6 of ANSI N14.6 [9]).

Table 3.4.3.4-1 through Table 3.4.3.4-6 provide summaries of the top 30 maximum combined stresses (Equivalent von Misses stresses) for both surfaces of the outer shell, inner shell, and stiffener plate (see Figure 3.4.3.4-2 through Figure 3.4.3.4-4 for node locations for the outer shell, inner shell, and stiffener plate, respectively). Stress contour plots for the outer shell are shown in Figure 3.4.3.4-5 and Figure 3.4.3.4-6. Stress contours for the inner shell are shown in Figure 3.4.3.4-7 and Figure 3.4.3.4-8. Stress contours for the stiffener plate are shown in Figures 3.4.3.4-9 and 3.4.3.4-10. As shown in Table 3.4.3.4-1 through Table 3.4.3.4-6, all stresses, except local stresses, meet the NUREG-0612 and ANSI N14.6 criteria. That is, a factor of safety of 6 applies on material yield strength and 10 applies on material ultimate strength. The high local stresses, as defined in ASME Code Section III, Article NB-3213.10, which are relieved by slight local yielding, are not required to meet the 6 and 10 safety factor criteria [see Reference 9, Section 4.2.1.2].

The localized stresses occur at the interfaces of the trunnion with the inner and outer shells. In accordance with ASME Code, Article NB-3213.10, the area of localized stresses cannot be larger than:

$$1.0\sqrt{Rt}$$

where:

R is the minimum midsurface radius

t is the minimum thickness in the region considered

Based on this formula, the maximum distance from the discontinuity to the local high stress is less than 5.1 inches for the inner shell and 7.3 inches for the outer shell.

For the trunnion, the maximum tensile bending stress and average shear stresses occur at the interface with the outer shell. The linearized stresses through the trunnion are 4,260 psi in bending and 1,871 psi in shear. Comparing these stresses to the material allowable yield and ultimate strength (A350, Grade LF2), the factor of safety on yield strength is 7.5 (which is >6) and on ultimate strength is 16.4 (which is >10).

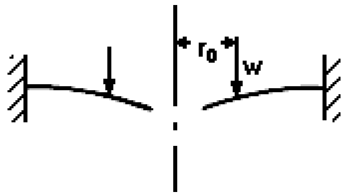
3.4.3.4.2 Advanced Transfer Cask Retaining Ring and Bolts

The Advanced transfer cask uses a retaining ring bolted to the top flange to prevent inadvertent lifting of the canister out of the transfer cask, which could increase the radiation exposure to nearby workers. In the event that the loaded transfer cask is inadvertently lifted by attaching to the canister eyebolts instead of the transfer cask trunnions, the retaining ring and bolts have sufficient strength to support the weight of the heaviest transfer cask, plus a 10% dynamic load factor.

Retaining Ring

To qualify the retaining ring, the equations for annular rings are used (Roark [26], Table 24, Case 1e). The retaining ring is represented as shown in the sketch below. The following sketch assists in defining the variables used to calculate the stress in the retaining ring and bolts. The model assumes a uniform annular line load *w* applied at radius r_o.

The boundary conditions for the model are outer edge fixed, inner edge free with a uniform annular line load w at radius r_o.



The material properties and parameters for the analysis are:

Plate dimensions:	
thickness:	$t = 0.75$ in
outer radius (bolt circle):	$a = 37.28$ in
outer radius (outer edge):	$c = 38.52$ in
inner radius:	$b = 32.37$ in
Weight of bounding transfer cask:	$wt = 124{,}000 \text{ lb} \times 1.1$
Radial location of applied load:	$r_o = 33.53$ in
Material:	ASTM A-588
Modulus of elasticity:	$E = 28.3 \times 10^6$ psi
Poisson's ratio:	$\nu = 0.31$
Number of bolts:	Nb = 32
Radial length of applied load:	$L_r = 2\pi r_o$
	$L_r = 210.675$ in
Applied unit load:	$w \equiv \frac{wt}{L_r}$
	$w = 647.44$ psi

The shear modulus is:

$$G = \frac{E}{2 \cdot (1 + v)}$$

$$G = 1.08 \times 10^7 \text{ psi}$$

D is a plate constant used in determining boundary values; it is also used in the general equations for deflection, slope, moment and shear. K_{sb} and K_{sro} are tangential shear constants used in determining the deflection due to shear:

$$D = \frac{E \cdot t^3}{12 \cdot (1 - v^2)}$$

$$D = 1.101 \times 10^6 \quad \text{lb-in}$$

Tangential shear constants, K_{sb} and K_{sro}, are used in determining the deflection due to shear:

$$K_{sb} = K_{sro} = -1.2 \cdot \frac{r_o}{a} \cdot \ln\left(\frac{a}{r_o}\right)$$

$$= -0.114$$

Radial moment M_{rb} and M_{ra} at point s b and a (inner and outer radius, respectively) are:

M_{rb} (b,0) = 0 lb-in/in

M_{ra} (a,0) = 2207.86 lb-in/in

Transverse moment M_{tb} and M_{ta}, at points b and a (inner and outer radius, respectively) due to bending are:

M_{tb} (b,0) = -122.64 lb-in./in.

M_{ta} (a,0) = 684.44 lb-in./in.

The calculated shear stresses, τ_b and τ_a, at points b and a (inner and outer radius, respectively) are:

$\tau_b = 0$ psi

$$\tau_a = \frac{wt}{2\pi At}$$

$\tau_a = -776.42$ psi

The calculated radial bending stresses, σ_{rb} and σ_{ra}, at points b and a (inner and outer radius) are:

$$\sigma_{r(i)} = \frac{6M_{r(i)}}{t^2}$$

$\sigma_{rb} = 0$ psi

$\sigma_{ra} = 23{,}550$ psi

The calculated transverse bending stresses, σ_{tb} and σ_{ta}, at points b and a (inner and outer radius) are:

$$\sigma_{t(i)} = \frac{6M_{t(i)}}{t^2}$$

$\sigma_{tb} = -1308.2$ psi

$\sigma_{ta} = 7{,}300.7$ psi

The principal stresses at the outer radius are:

σ_{1a} = 23,590 psi
σ_{2a} = 7,263.6 psi
σ_{3a} = 0 psi

The stress intensity, SI_a, at the outer radius ($P_m + P_b$) is:

$SI_a = \sigma_{1a} - \sigma_{3a}$
SI_a = 23,590 psi

The principal stresses at the inner radius are:

σ_{1b} = 0 psi
σ_{2b} = -1308.2 psi
σ_{3b} = 0 psi

The stress intensity, SI_b, at the inner radius ($P_m + P_b$) is:

$SI_b = \sigma_{1b} - \sigma_{2b}$
SI_b = 1308.2 psi

The maximum stress intensity occurs at the outer radius of the retaining ring. For the off-normal condition, the allowable stress intensity is equal to the lesser of 1.8 S_m and 1.5 S_y. For ASTM A-588, the allowable stress intensity at 300°F is 1.8(23.3) = 41.94 ksi. The calculated stress of 23.59 ksi is less than the allowable stress intensity and the margin of safety is:

$$MS = \frac{41.94}{23.59} - 1 = +0.78$$

Retaining Ring / Canister Bearing

The bearing stress, S_{brg}, between the retaining ring and canister is calculated as:

Weight of Transfer Cask (TFR) = 124,000 × 1.1 = 136,400 lbs.

Area of contact between retaining ring and canister:

$$S_{brg} = \frac{136{,}400}{240} = 568 \text{ psi}$$

$$A = \pi\left(33.53^2 - 32.37^2\right) = 240 \text{ in}^2$$

Bearing stress allowable is S_y. For ASTM A-588, the allowable stress at 300°F is 45.6 ksi. The Calculated bearing stress is well below the allowable stress with a large margin of safety.

<u>Shearing Stress of Retaining Plate under the Bolt Heads</u>

The shearing stress of the retaining plate under the bolt head is calculated as:

Outside diameter of bolt head d_b = 1.125 in.

Total shear area under bolt head = π (1.125) × 32 × 0.75

= 84.82 in^2.

$\tau_p = \frac{136{,}400}{84.82} = 1608 \text{ psi}$ Shear stress of retaining plate, τ_p, under bolt head is:

$$\tau_p = \frac{136{,}400}{84.82} = 1{,}608 \text{ psi}$$

Conservatively, the shear allowable for normal conditions is used.

$$\tau_{allowable} = (0.6)\,(S_m) = (0.6)\,(23.3 \text{ ksi}) = 13.98 \text{ ksi}$$

The Margin of Safety is: $\frac{13{,}980}{1{,}608} - 1 = +\text{large}$

<u>Bolt Edge Distance</u>

$\frac{77.04 - 74.56}{2} = 1.24 \text{ in} > 1.0 \text{ in}$ Using Table J3.5 "Minimum Edge Distance, in." of Section J3 from "Manual of Steel Construction Allowable Stress Design,"[23] the required saw-cut edge distance for a 0.75 inch bolt is 1.0 inch. The edge distance for the bolts that meets the criteria of the Steel Construction Manual is:

$$\frac{77.04 - 74.56}{2} = 1.24\,\text{in} > 1.0\,\text{in}$$

Retaining Ring Bolts

The load on a single bolt, F_F, due to the reactive force caused by inadvertently lifting the canister, is:

$$F_F = \frac{wt}{N_b} = 4{,}262 \text{ lb}$$

where:

N_b = number of bolts, 32, and

wt = the weight of the cask, plus a 10% load factor, 124,000 lb × 1.1 = 136,400 lb.

The load on each bolt, F_M, due to the bending moment, is:

$$F_M = \left(\frac{2 \cdot \pi \cdot a}{N_b}\right) \cdot \left(\frac{\sigma \cdot t^2}{6 \cdot L}\right)$$

F_M = 12,929 lb

where:

a = the outer radius of the bolt circle, 37.28 in.,

t = the thickness of the ring, 0.75 in.,

σ = the radial bending stress at point *a*, σ_{ra} = 23,550 psi, and

L = the distance between the bolt centerline and ring outer edge, c - a = 1.25 in.

The total tension, F, on each bolt is

$$F = F_F + F_M = 17{,}191 \text{ lb}$$

Knowing the bolt cross-sectional area, A_b, the bolt tensile stress is calculated as:

$$\sigma_t = \frac{F}{A_b} = 38{,}912 \text{ psi}$$

where:

$A_b = 0.4418 \text{ in}^2$

For off-normal conditions, the allowable primary membrane stress in a bolt is $2S_m$. The allowable stress for SA-193 Grade B6 bolts is 54 ksi at 120°F, the maximum temperature of the transfer cask top plate. The margin of safety for the bolts is

$$MS = \frac{54{,}000}{38{,}912} - 1 = +0.38$$

Since the SA-193 Grade B6 bolts have higher strength than the top plate, the shear stress in the threads of the top plate is evaluated. The yield and ultimate strengths for the top plate ASTM A-588 material at a temperature of 120°F are:

S_y = 49.5 ksi
S_u = 70.0 ksi

From Reference 27, the shear area for the internal threads of the top plate, A_n , is calculated as:

$$A_n = 3.1416\, n\, L_e\, D_s\min\left[\frac{1}{2n} + 0.57735\left(D_s\min - E_n\max\right)\right] = 1.525 \text{ in}^2$$

where:

D = 0.7482 in., basic major diameter of bolt threads,
n = 10, number of bolt threads per inch,
$D_s\min$ = 0.7353 in., minimum major diameter of bolt threads,
$E_n\max$ = 0.6927 in., maximum pitch diameter of lid threads, and
L_e = 1.625-0.74=0.885 in., minimum thread engagement.

The shear stress (τ_n) in the top plate is:

$$\tau_n = \frac{F}{A_n} = \frac{17{,}191 \text{ lb}}{1.525 \text{ in}^2} = 11{,}273 \text{ psi}$$

Where the total tension, F, on each bolt is

$$F = F_F + F_M = 17{,}191 \text{ lb}$$

The shear allowable for normal conditions is conservatively used:

$\tau_{allowable} = (0.6)\ (S_m) = (0.6)\ (23.3\ ksi) = 13.98\ ksi$

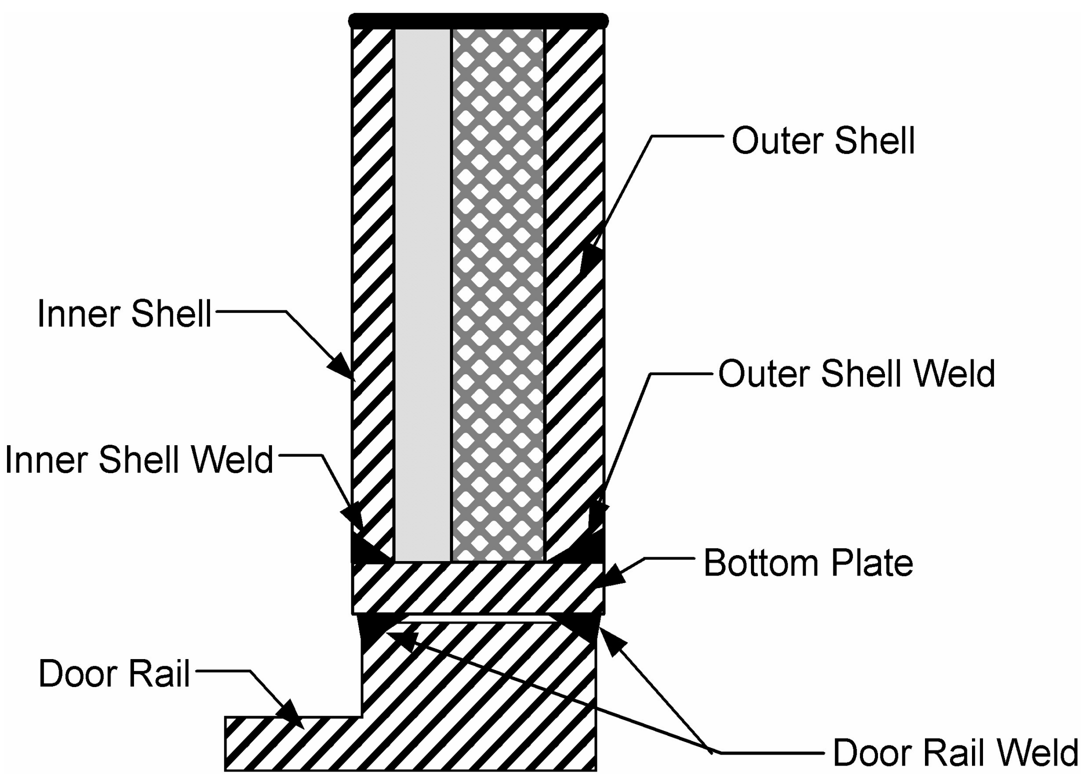
The Margin of Safety is:

$$MS = \frac{13,980}{11,273} - 1 = +0.24$$

Therefore, the threads of the top plate will not fail in shear.

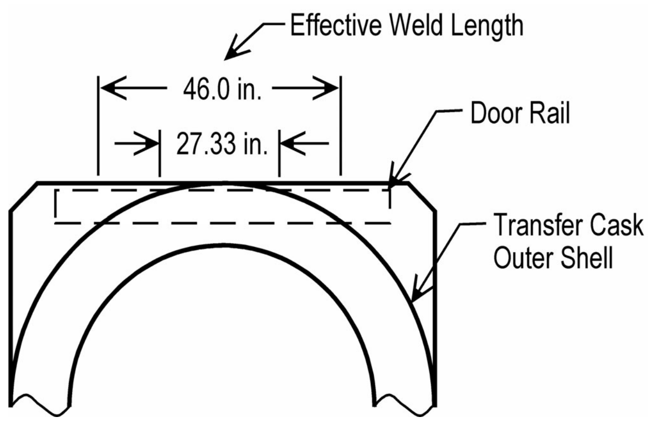
3.4.3.4.3 Advanced Transfer Cask Bottom Plate Weld Analysis

The bottom plate is connected to the outer and inner shell of the transfer cask by full penetration welds. The weight of a loaded canister along with the shield door rail structure is transmitted from the bottom plate to the shell via the full penetration weld. For conservatism, only the length of the weld directly under the shell is considered effective in transmitting a load.



The weld connecting the outer and inner shell to the bottom plate has a length of approximately

l_w = (27.33 in. + 46.0 in.)/2 in. = 36.66 in.



Stresses occurring in the outer shell to bottom plate weld are evaluated using a weight, W = 131,800 lb × 1.1 = 145,000 lb, which bounds the weight of the heaviest loaded canister, the weight of the water, and the weight of the shield doors and rails, with a 10% dynamic load factor.

The door rail structure and canister load will be transmitted to both the inner and outer shell via full penetration welds. The thickness of the two shells and welds are different; however, for conservatism, this evaluation assumes both shell welds are 0.75 in. groove welds.

Weld effective area = (36.66 in.)(0.75 in. + 0.75 in.) = 54.99 in^2

$$\sigma_{axial} = \frac{P}{A} = \frac{(145{,}000\ \text{lb})/(2)}{54.99\ \text{in}^2} = 1{,}318\ \text{psi}$$

For the bottom plate material (ASTM A-588) at a bounding temperature of 400°F, the yield and ultimate stresses are:

$$FS_{yield} = \frac{43.0}{1.32} = +32.6 > 6$$

$$FS_{ultimate} = \frac{70.0}{1.32} = +53.0 > 10$$

where:

S_y = 43.0 ksi

S_u = 70.0 ksi

Thus, the welds in the bottom plate meet the ANSI N14.6 and NUREG-0612 criteria for nonredundant systems.

3.4.3.4.4 Advanced Transfer Cask Shield Door Rails and Welds

This section demonstrates the adequacy of the transfer cask shield doors, door rails, and welds in accordance with NUREG-0612 and ANSI N14.6, which require safety factors of 6 and 10 on material yield strength and ultimate strength, respectively, for nonredundant lift systems.

The shield door rails support the weight of a wet, fully loaded canister and the weight of the shield doors themselves. The shield doors are 9.0-in. thick plates that slide on the door rails. The rails are 9.38 in. deep × 6.5 in. thick and are welded to the bottom plate of the transfer cask. The doors and the rails are constructed of A-588 and A-350 Grade LF 2 low alloy steel, respectively.

The design weight used in this evaluation, $W = 131{,}800 \times 1.1 \approx 145{,}000$ pounds, is an assumed value that bounds the weight of the heaviest loaded canister, the weight of the water in the canister and the weight of the shield doors and rails. A 10% dynamic load factor is included to ensure that the evaluation bounds all normal operating conditions. This evaluation shows that the door rail structures and welds are adequate to support the design input.

Allowable stresses for the material are taken at 400°F, which bounds the maximum temperature at the bottom of the transfer cask under normal conditions. The material properties of A-588 and A-350 Grade LF 2 low alloy steel are provided in Tables 3.3-8 and 3.3-9, respectively. The standard impact test temperature for ASTM A-350, Grade LF2 is -50°F. The NDT temperature range is -70°F to -10°F for ASTM A-588 with a thickness range of 0.625 in. to 3 in. [28]. Therefore, the minimum service temperature for the trunnion and shells is conservatively established as 0°F (50°F higher than the NDT test temperature, in accordance with Section 4.2.6 of ANSI N14.6 [9]. For conservatism, the stress allowables for A-350 Grade LF 2 are used for all stress calculations.

Stress Evaluation for Door Rail

Each rail is assumed to carry a uniformly distributed load equal to 0.5W. The shear stress in each door rail bottom plate due to the applied load, W, is:

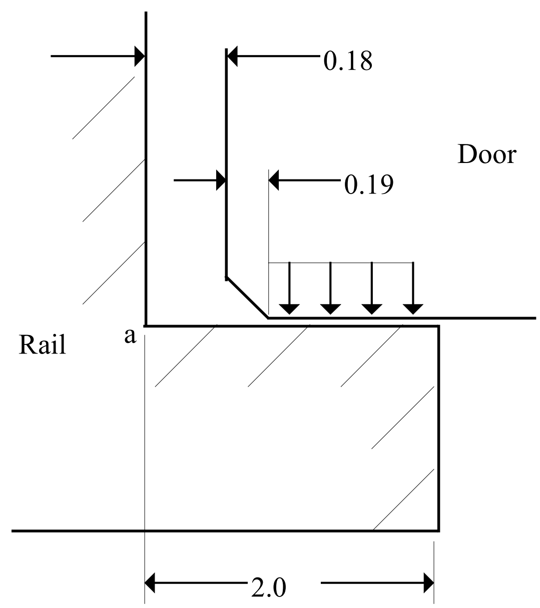
$$\tau = \frac{W}{A} = \frac{145{,}000 \text{ lb}}{281.25 \text{ in}^2} = 516 \text{ psi}$$

where:

$A = 2.5$ in. × 56.25 in. length/rail × 2 rails = 281.25 in^2.

The bending stress in each rail bottom section due to the applied load of W is:

$$\sigma_b = \frac{6M}{bt^2} = \frac{6 \times 86{,}275}{56.25 \times 2.5^2} = 1{,}472 \text{ psi},$$



where:

M = moment at *a*,

$$= \frac{W}{2} \times \ell = \frac{145{,}000 \text{ lb.}}{2} \times 1.19 \text{ in.}$$

= 86,275 in-lb,

and,

$$\ell = 2 - \frac{2 - (0.18 + 0.19)}{2}$$

ℓ = 1.19 in., applied load moment arm.

The maximum principal stress in the bottom section of the rail is:

$$\sigma = \left(\frac{\sigma_b}{2}\right) + \sqrt{\left(\frac{\sigma_b}{2}\right)^2 + \tau^2}$$
$$= 1{,}635 \text{ psi}$$

The acceptability of the rail design is evaluated by comparing the allowable stresses to the maximum calculated stresses, considering the safety factors of NUREG-0612 and ANSI N14.6. For the yield strength criteria:

$$\frac{30{,}800 \text{ psi}}{1{,}635 \text{ psi}} = 18.8 > 6$$

For the ultimate strength criteria,

$$\frac{70{,}000 \text{ psi}}{1{,}635 \text{ psi}} = 42.8 > 10$$

The safety factors meet the criteria of NUREG-0612. Therefore, the rails are structurally adequate.

Stress Evaluation for the Shield Doors

The shield doors consist of a layer of NS-4-FR neutron shielding material sandwiched between low alloy steel plates (Note: steel bars are also welded on the edges of the doors so that the neutron shielding material is fully encapsulated). The door assemblies are 9-inch thick at the center and 6.75-inch thick at the edges, where they slide on the support rails. The stepped edges of the two door leaves are designed to interlock at the center and are, therefore, analyzed as a single plate that is simply supported on two sides.

The shear stress at the edge of the shield door where the door contacts the rail is:

$$\tau = \frac{W}{2 \times A_S} = \frac{145{,}000 \text{ lb}}{2 \times (49.2 \text{ in.} \times 4.75 \text{ in.})} = 310 \text{ psi}$$

where:

A = the total shear area, 4.75 in. thick × 49.2 in. long. Note that the effective thickness at the edge of the doors is taken as 4.75 in. because the neutron shield material and the cover plate are assumed to carry no shear load. The shear stress at the center of the doors approaches 0 psi.

The moment equation for the simply-supported beam with uniform loading is:

$$M = 72{,}500\ X - 2{,}031(X)(0.5\ X) = 72{,}500\ X - 1{,}015\ X^{2}$$

The maximum bending moment occurs at the center of the doors, X = 35.7 in. The bending moment at this point is:

$$M = 72{,}500 \text{ lb} \times (35.7 \text{ in.}) - 1{,}015 \text{ lb/in.} \times (35.7 \text{ in})^2$$
$$M = 12.95 \times 10^5 \text{ in.-lb.}$$

The maximum bending stress, σ_{max}, at the center of the doors, is

$$\sigma_{ax} = \frac{Mc}{I} = \frac{12.95 \times 10^5 \text{ in.} - \text{lb} \times 5.5 \text{ in.}}{2{,}378 \text{ in.}^4} = 2{,}995 \text{ psi}$$

where:

$$c = \frac{h}{2} = \frac{7 \text{ in.}}{2} + 2 \text{ in.} = 5.5 \text{ in.}, \text{ and}$$

$$I = \frac{bh^3}{12} = \frac{83.2 \text{ in.} \times 7^3 \text{ in}}{12} = 2{,}378 \text{ in}^4.$$

The acceptability of the door design is evaluated by comparing the allowable stresses to the maximum calculated stresses. As shown above, the maximum stress occurs for bending.

For the yield strength criteria,

$$\frac{30{,}800 \text{ psi}}{2{,}995 \text{ psi}} = 10.3 > 6$$

For the ultimate strength criteria,

$$\frac{70{,}000\ \text{psi}}{2{,}995\ \text{psi}} = 23.4 > 10$$

The safety factors satisfy the criteria of NUREG-0612. Therefore, the doors are structurally adequate.

Door Rail Weld Evaluation

The door rails are attached to the bottom of the transfer cask by 0.75-in. partial penetration bevel groove welds that extend the full length of the inside and outside of each rail. If the load is conservatively assumed to act at a point on the inside edge of the rail, the load, P, on each rail is,

$$P = \frac{W}{2} = \frac{145{,}000\ \text{lb}}{2} = 72{,}500\ \text{lb}$$

Summing moments about the inner weld location:

$$0 = P \times a - F_o \times (b) = 72{,}500\ \text{lb} \times 1.19\ \text{in.} - F_o\ (4.5\ \text{in.}),\ \text{or}$$

$$F_o = 19{,}172\ \text{lb}$$

Summing forces:

$$F_i = F_o + P = 19{,}172\ \text{lb} + 72{,}500\ \text{lb} = 91{,}672\ \text{lb}$$

The effective area of the inner weld is $0.75\ \text{in} \times .707 \times 56.25\ \text{in. long} = 29.83\ \text{in}^2$

The shear stress, τ, in the inner weld is

$$\tau = \frac{91{,}672\ \text{lb}}{29.83\ \text{in}^2} = 3{,}073\ \text{psi}$$

The factors of safety are

$$\frac{30{,}800\ \text{psi}}{3{,}073\ \text{psi}} = 10.0 > 6 \qquad \text{(for yield strength criteria)}$$

$$\frac{70{,}000\ \text{psi}}{3{,}073\ \text{psi}} = 22.8 > 10 \qquad \text{(for ultimate strength criteria)}$$

The safety factors meet the criteria of NUREG-0612.

Figure 3.4.3.4-1 Advanced Transfer Cask Finite Element Model

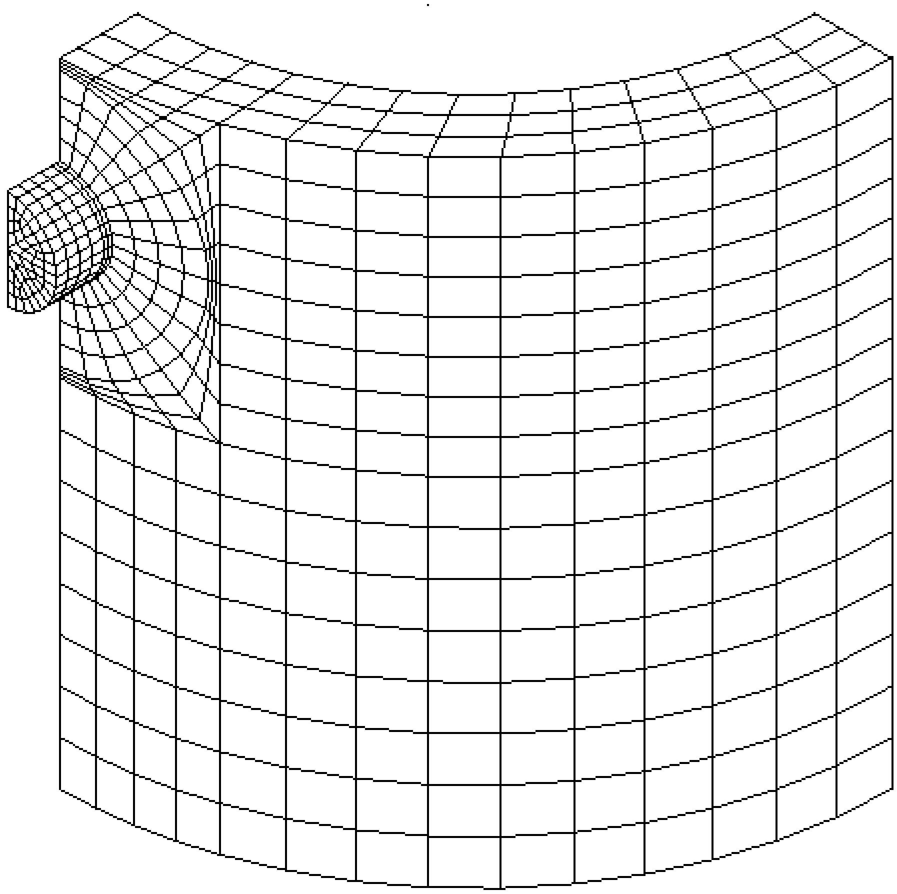


Figure 3.4.3.4-2 Node Locations for Advanced Transfer Cask Outer Shell Adjacent to Trunnion

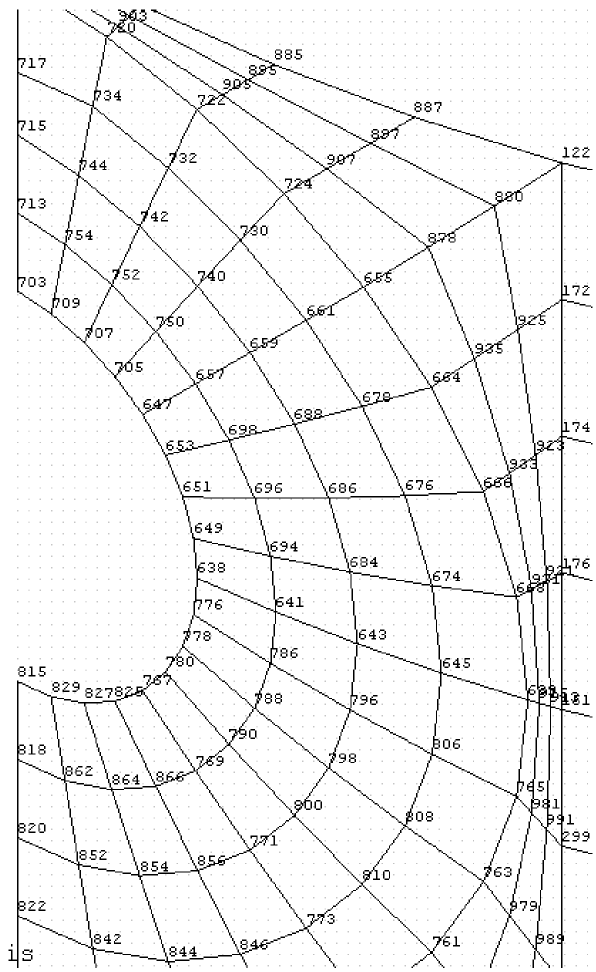


Figure 3.4.3.4-3 Node Locations for Advanced Transfer Cask Inner Shell Adjacent to Trunnion

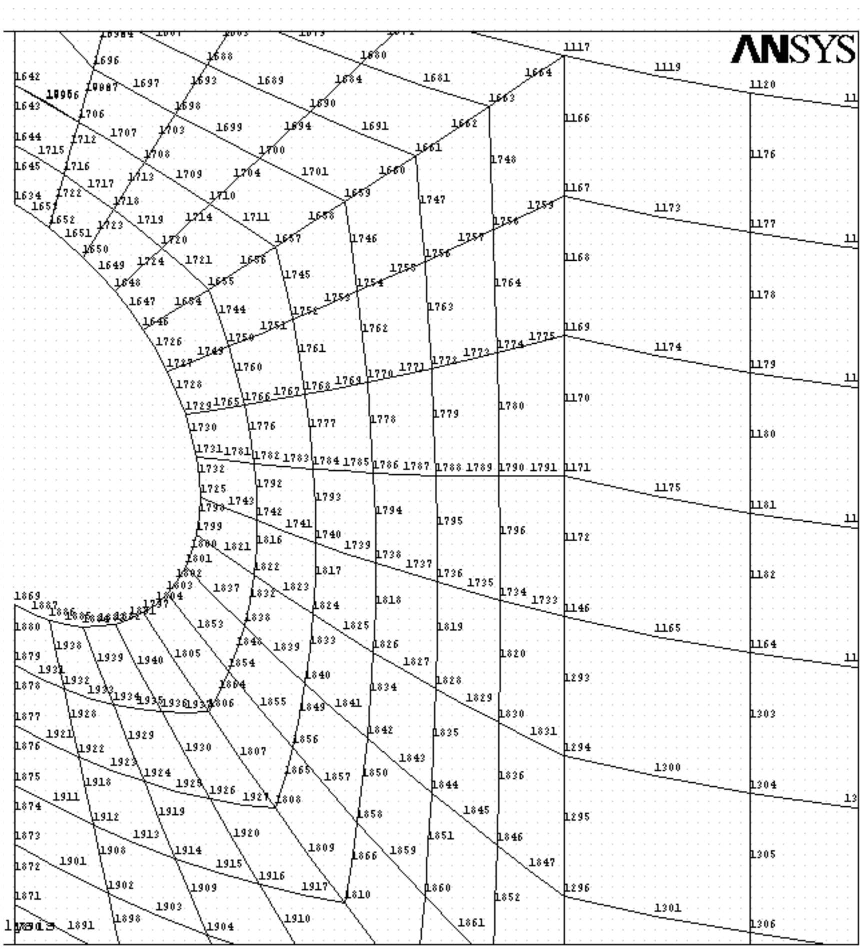


Figure 3.4.3.4-4 Node Locations for Advanced Transfer Cask Stiffener Plate Above Trunnion

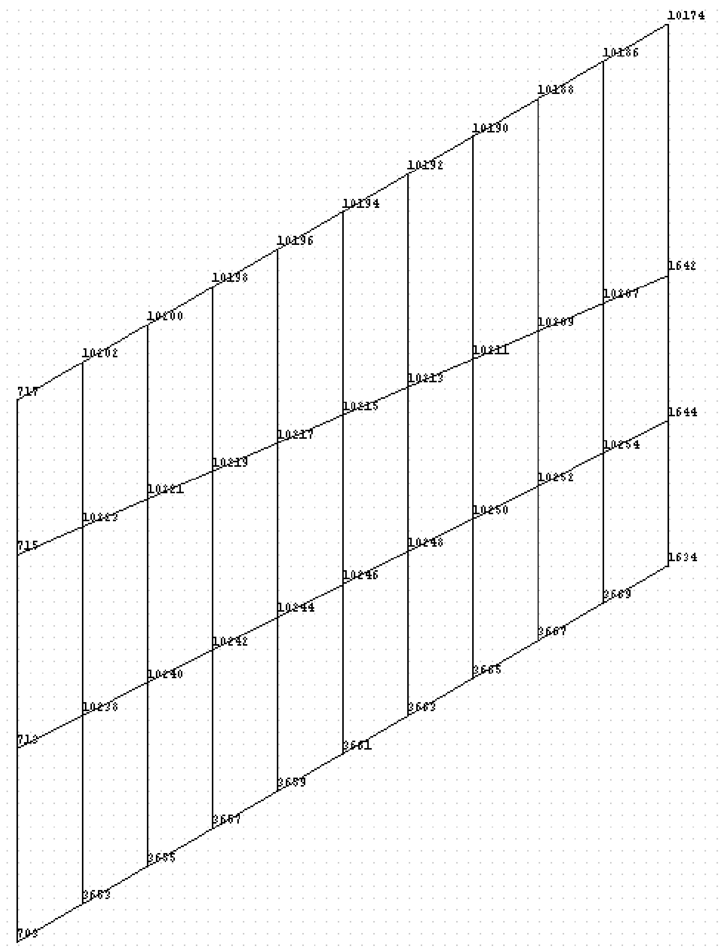


Figure 3.4.3.4-5 Stress Intensity Contours (psi) for Advanced Transfer Cask Outer Shell Element Top Surface

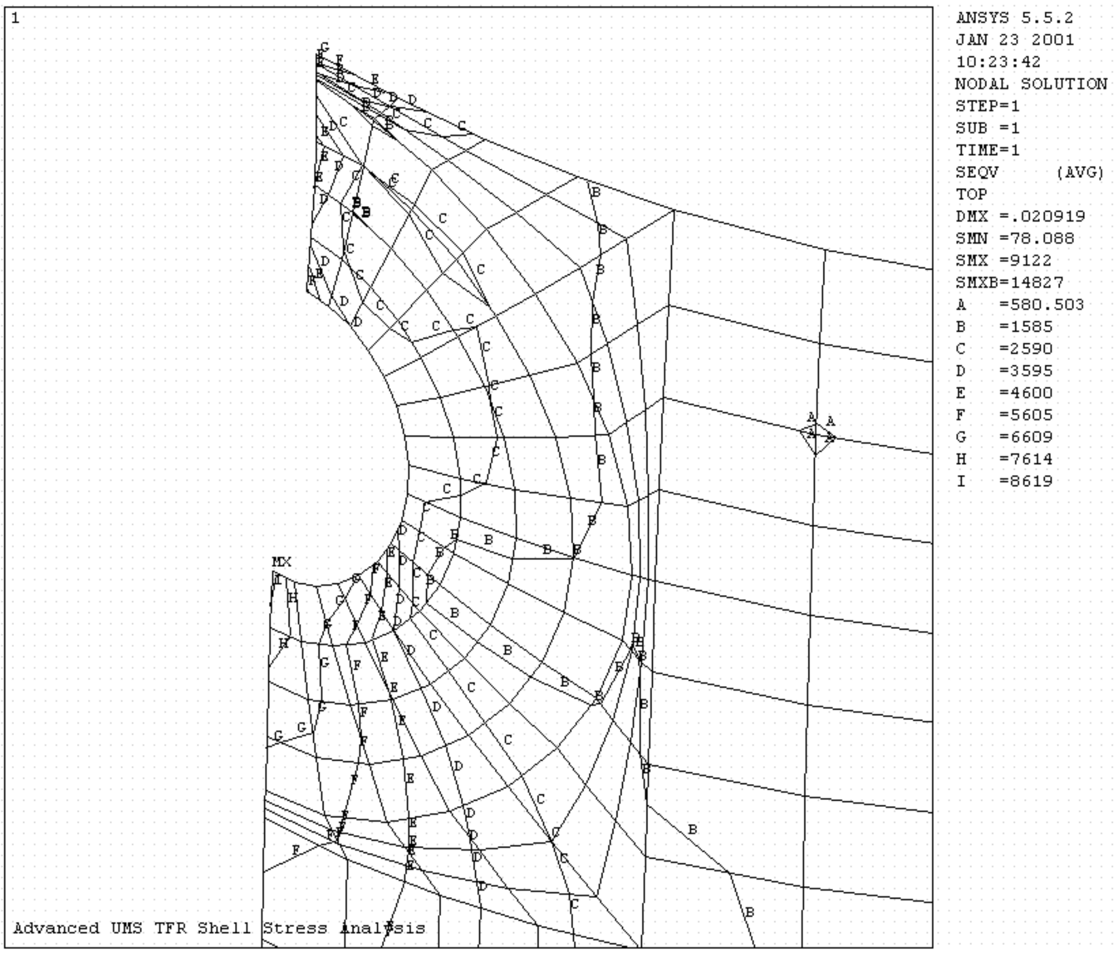


Figure 3.4.3.4-6 Stress Intensity Contours (psi) for Advanced Transfer Cask Outer Shell Element Bottom Surface

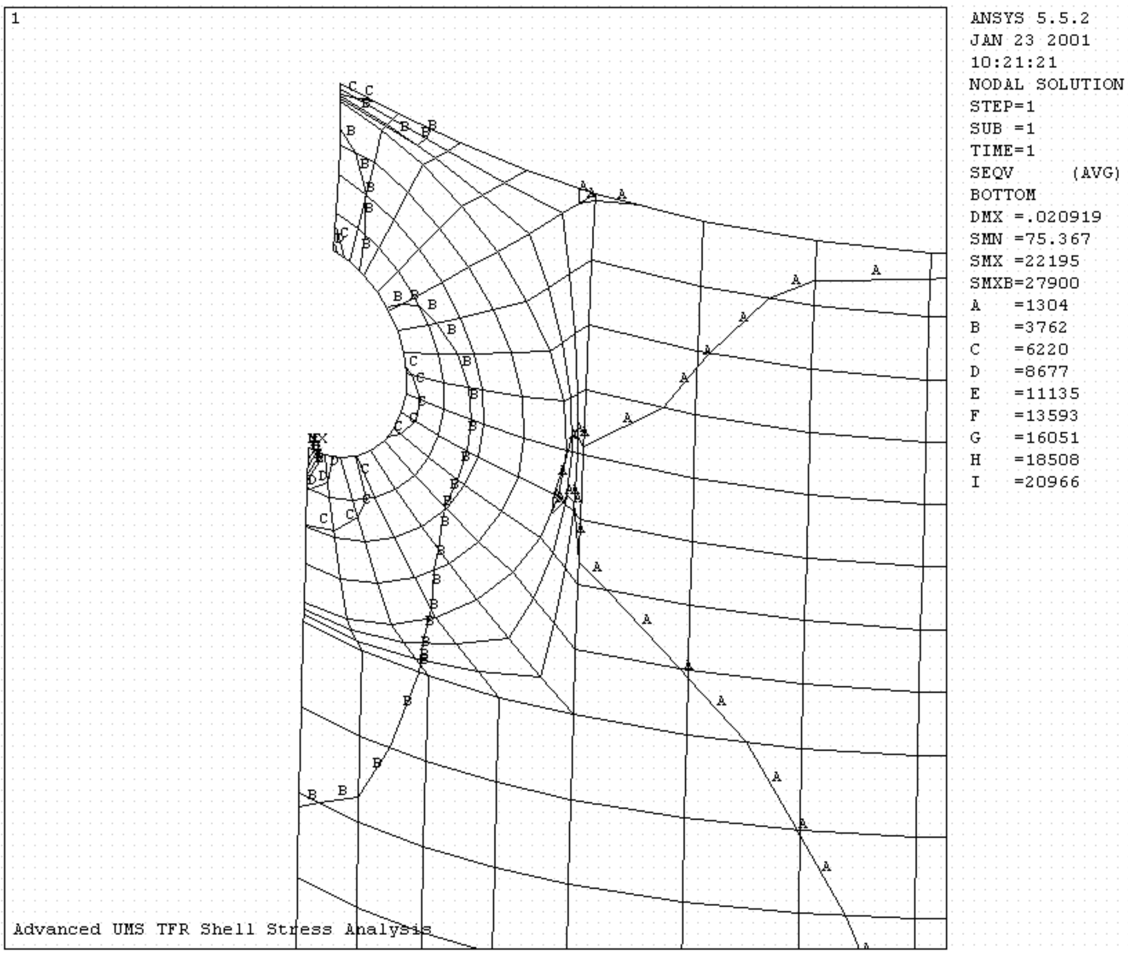


Figure 3.4.3.4-7 Stress Intensity Contours (psi) for Advanced Transfer Cask Inner Shell Element Top Surface

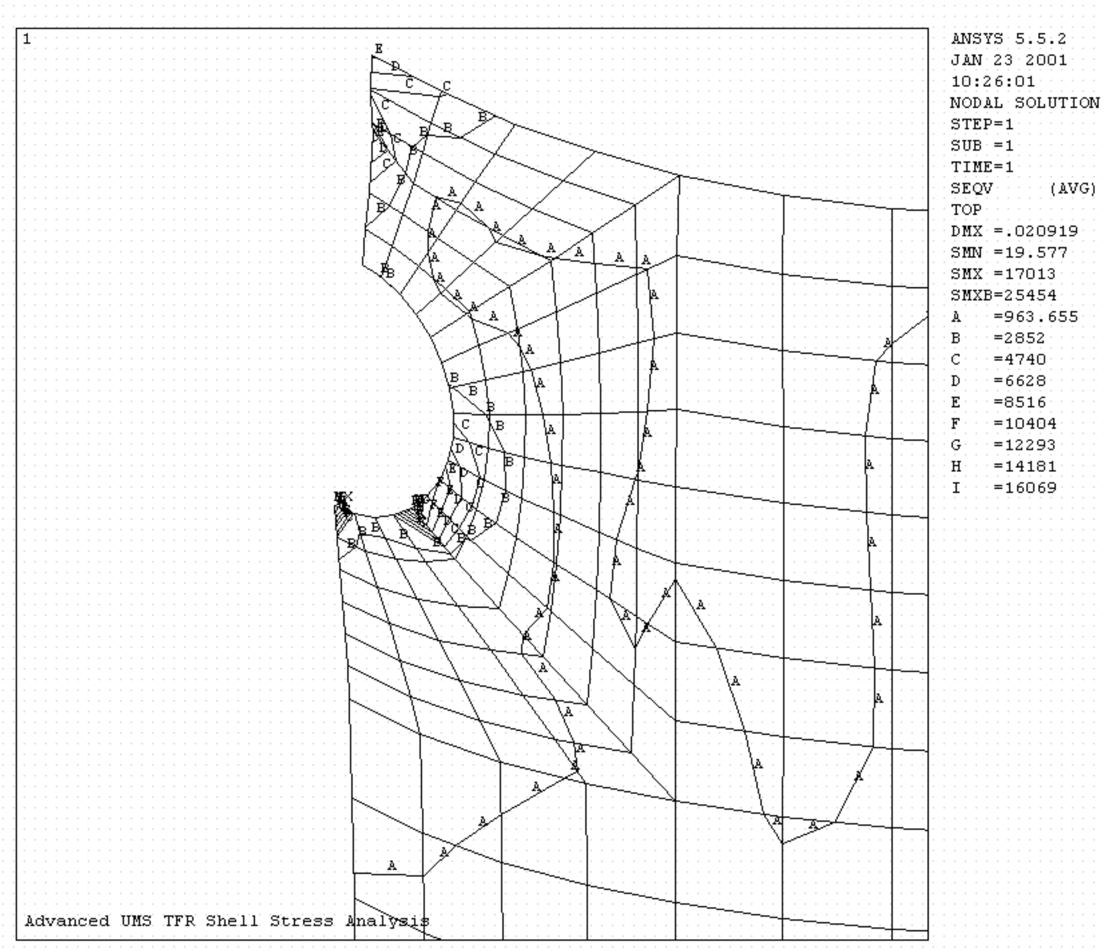


Figure 3.4.3.4-8 Stress Intensity Contours (psi) for Advanced Transfer Cask Inner Shell Element Bottom Surface

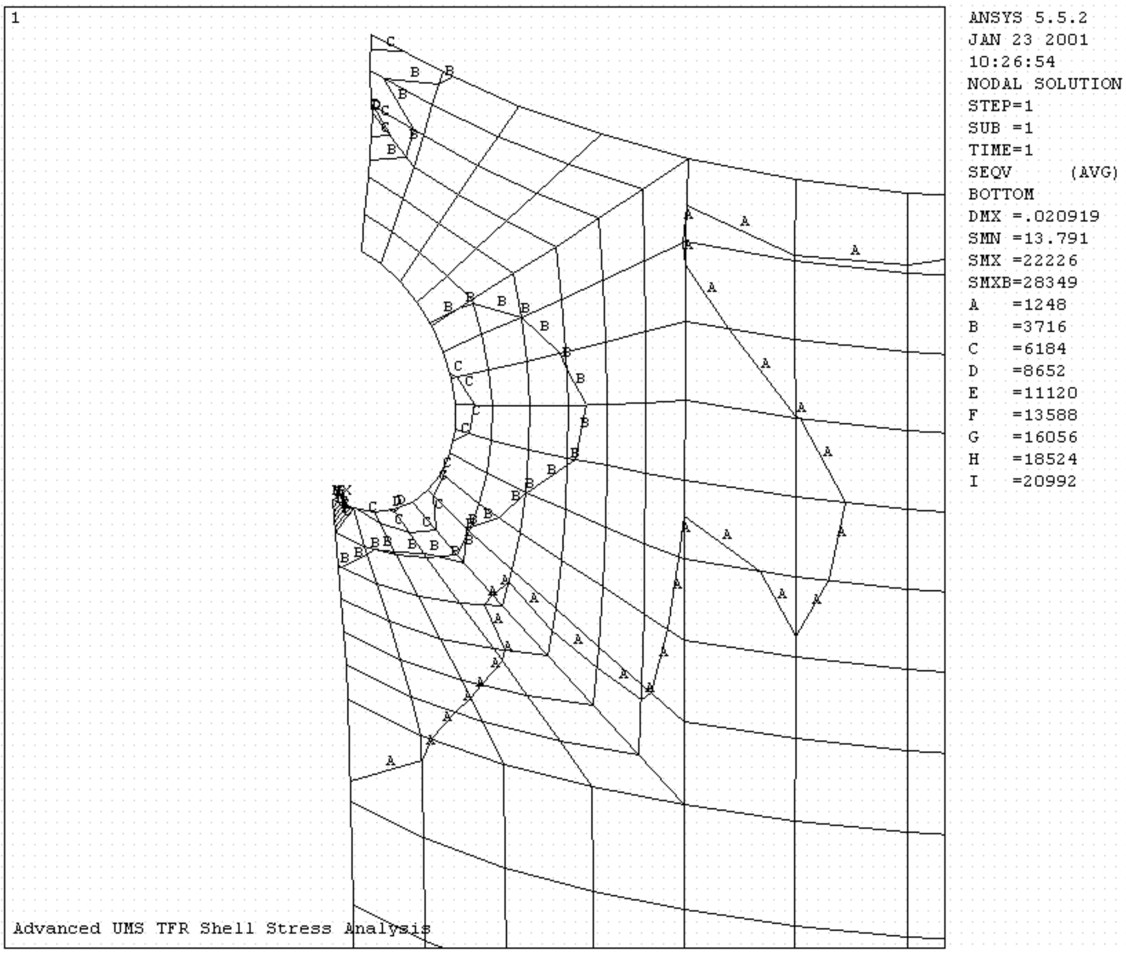


Figure 3.4.3.4-9 Stress Intensity Contours (psi) for Advanced Transfer Cask Stiffener Plate Element Top Surface

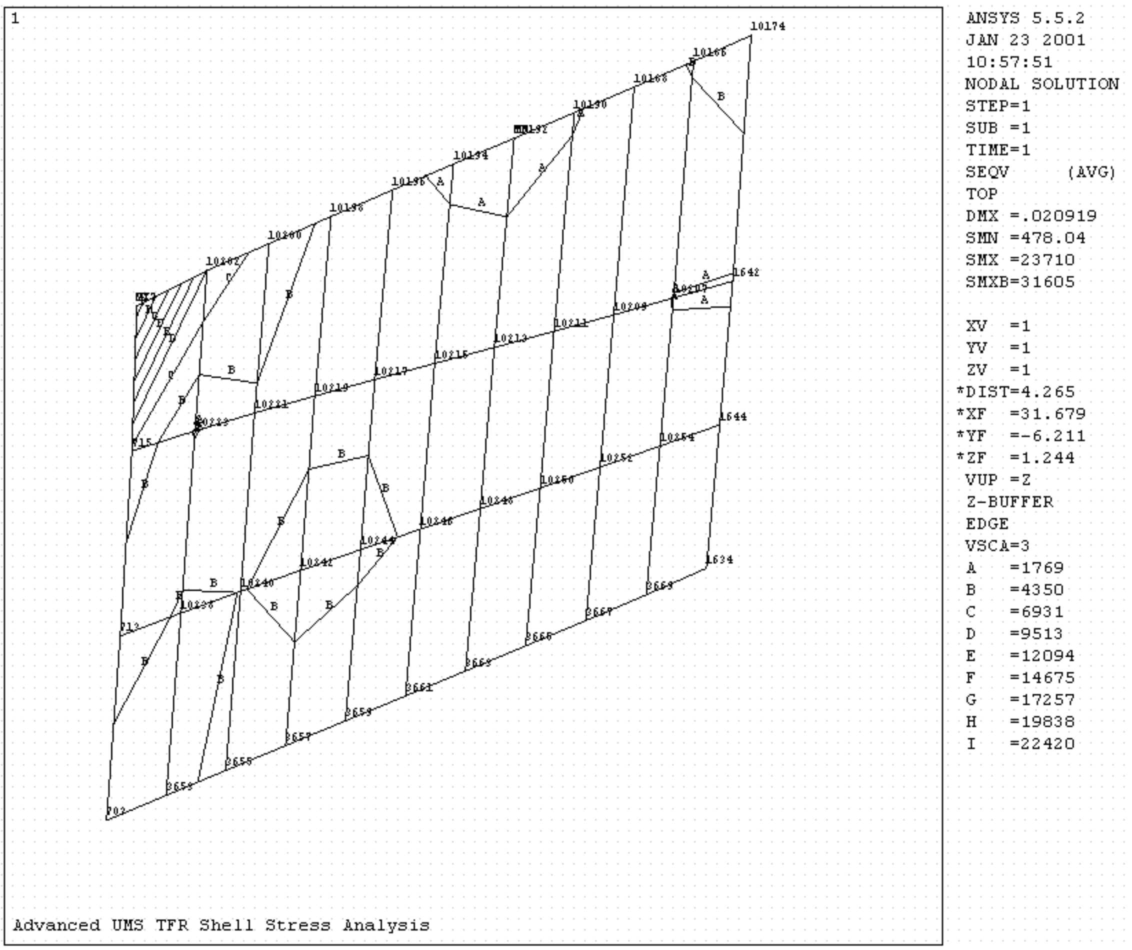


Figure 3.4.3.4-10 Stress Intensity Contours (psi) for Advanced Transfer Cask Stiffener Plate Element Bottom Surface

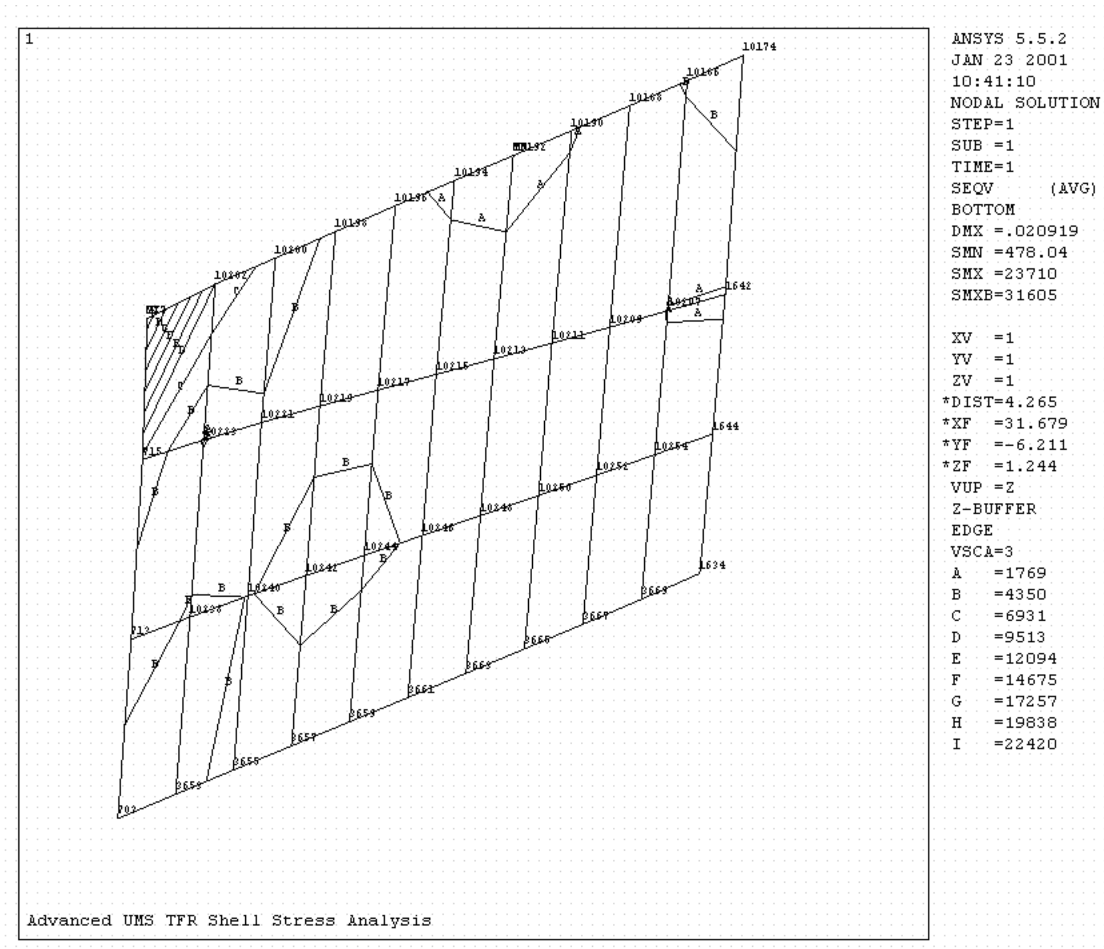


Table 3.4.3.4-1 Top 30 Stresses for Advanced Transfer Cask Outer Shell Element Top Surface

Node[1]	Principal Stresses (psi)			Nodal Von Mises Stresses	FS on Yield (S_y)[2]	FS on Ultimate (S_u)[2]
	S1	S2	S3			
815	3826.5	-213.1	-6617.3	9121.6	NA[3]	NA[3]
818	5489.8	-5.1	-4049.8	8293.3	NA	NA
820	4864.3	-1.9	-3634.4	7385.9	6.2	NA
827	5467.3	9.9	-2896.6	7354.7	6.2	NA
862	4847.2	3.3	-3530.7	7285.0	6.3	NA
825	5331.6	46.7	-2843.8	7180.6	6.4	NA
852	4708.0	0.3	-3275.8	6951.2	6.6	10.1
829	4163.6	-32.5	-3761.8	6867.6	6.6	10.2
871	7593.2	2376.9	-104.9	6805.5	6.7	10.3
822	4395.9	-0.3	-3328.5	6710.8	6.8	10.4
767	4460.2	129.3	-3077.8	6552.3	7.0	10.7
842	4289.1	0.2	-3001.4	6346.5	7.2	11.0
816	3994.4	-0.1	-3172.3	6220.2	7.3	11.3
943	3923.2	-0.1	-3167.1	6152.0	7.4	11.4
864	4384.5	9.7	-2590.9	6105.7	7.5	11.5
941	3858.7	-0.1	-3154.5	6083.8	7.5	11.5
2	3777.8	0.0	-3137.0	5997.0	7.6	11.7
832	3896.2	0.2	-2847.8	5864.0	7.8	11.9
854	4294.6	2.8	-2380.3	5858.9	7.8	11.9
703	3796.2	403.9	-2924.7	5820.5	7.8	12.0
964	3797.0	0.3	-2769.6	5710.0	8.0	12.3
873	6270.5	2019.3	-108.3	5625.3	8.1	12.4
954	3706.0	0.2	-2688.5	5561.1	8.2	12.6
844	3986.4	2.1	-2173.4	5410.7	8.4	12.9
8	3604.7	0.0	-2610.7	5405.5	8.4	12.9
780	3173.5	201.6	-3062.0	5402.0	8.4	13.0
47	3082.8	0.0	-2836.0	5127.3	8.9	13.7
717	5482.2	2416.5	-302.7	5012.8	9.1	14.0
834	3658.3	1.9	-2009.6	4977.0	9.2	14.1
866	3876.4	2.7	-1685.2	4939.0	9.2	14.2

Notes:
1. See Figure 3.4.3.4 -2 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.4-2 Top 30 Stresses for Advanced Transfer Cask Outer Shell Element Bottom Surface

Node[1]Node	Principal Stresses (psi)Principal Stresses (psi)			Nodal Von Mises Stresses	FS on Yield $(S_y)^2$	FS on Ultimate $(S_u)^2$
	S1	S2	S3			
815	23117.0	2218.6	-178.6	22195.0	NA[3]	NA[3]
829	11968.0	5735.9	-18.9	10383.0	NA	NA
703	1423.8	-967.7	-9354.8	9804.1	NA	NA
818	9713.1	2279.0	-6.7	8802.4	NA	NA
871	94.9	-1212.6	-8374.4	7897.2	NA	NA
862	8885.1	3223.8	-10.3	7798.7	NA	NA
638	5557.4	114.7	-2341.4	7001.6	6.5	10.0
827	8016.0	3977.8	-8.5	6949.4	6.6	10.1
776	6510.7	508.4	-1100.0	6947.6	6.6	10.1
873	96.4	-763.0	-7125.6	6833.0	6.7	10.2
864	7722.7	2933.9	-9.0	6759.1	6.7	10.4
778	6789.6	1430.4	-446.1	6503.7	7.0	10.8
649	4028.8	-83.7	-3465.5	6500.5	7.0	10.8
820	7069.8	2942.0	-1.3	6152.4	7.4	11.4
825	7053.3	3670.3	-38.6	6143.9	7.4	11.4
780	6781.3	2682.2	-224.7	6096.5	7.5	11.5
875	100.3	-280.5	-6043.9	5963.0	7.6	11.7
767	6767.8	3530.1	-113.1	5962.5	7.6	11.7
852	6770.6	2764.7	-2.8	5898.5	7.7	11.9
866	6665.5	2211.1	-0.6	5881.0	7.8	11.9
651	2424.8	-291.8	-4029.7	5613.0	8.1	12.5
769	6045.8	1502.0	0.4	5451.9	8.4	12.8
854	6169.1	2215.3	-3.8	5415.8	8.4	12.9
715	42.8	-4696.1	-5838.1	5401.2	8.4	13.0
822	6062.5	2413.7	-0.2	5286.6	8.6	13.2
790	5610.4	835.1	-1.7	5244.1	8.7	13.3
717	356.3	-4113.9	-5392.7	5228.2	8.7	13.4
788	5221.9	112.3	-2.8	5168.2	8.8	13.5
842	5860.1	2239.1	-0.7	5122.3	8.9	13.7
786	4723.6	-2.7	-633.4	5071.1	9.0	13.8

Notes:

1. See Figure 3.4.3.4 -2 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.4-3 Top 30 Stresses for Advanced Transfer Cask Inner Shell Element Top Surface

Node[1]	Principal Stresses (psi)			Nodal Von Mises Stresses (	FS on Yield $(S_y)^2$	FS on Ultimate $(S_u)^2$
	S1	S2	S3			
1869	2012.7	-552.0	-16137.0	17013.0	NA[3]	NA[3]
1797	13166.0	-115.4	-3089.1	14991.0	NA	NA
1803	12734.0	4058.3	-311.4	11501.0	NA	NA
1801	11627.0	4490.2	-214.9	10327.0	NA	NA
10174	663.9	-6733.1	-10256.0	9653.6	NA	NA
1633	9836.1	2649.9	-31.5	8837.4	NA	NA
1799	8856.4	4640.1	-144.2	7799.9	NA	NA
1638	743.8	-1547.7	-6362.4	6282.1	7.3	11.1
1725	5909.8	4672.1	-118.6	5514.7	8.3	12.7
1666	5438.4	1119.1	-33.4	4996.2	9.1	14.0
1882	783.0	-2383.4	-4495.2	4601.4	9.9	15.2
1636	4276.1	128.5	-576.2	4541.2	10.0	15.4
1822	4908.4	3039.9	-24.4	4313.6	10.6	16.2
1879	385.8	-127.4	-4147.3	4299.6	10.6	16.3
1731	4586.7	3239.6	-100.4	4179.6	10.9	16.7
1642	370.6	-17.7	-3713.0	3904.0	11.7	17.9
1838	4243.3	3272.0	-43.6	3893.2	11.7	18.0
1742	4389.1	2373.4	-14.5	3818.2	11.9	18.3
1886	99.4	-3236.9	-4024.2	3791.7	12.0	18.5
1884	444.8	-1827.8	-3719.9	3611.7	12.6	19.4
1676	3632.1	460.2	-25.6	3440.7	13.3	20.3
1854	3092.7	2724.2	-63.9	2989.4	15.3	23.4
1729	3305.4	1609.1	-110.0	2957.8	15.4	23.7
1652	2282.5	-2.8	-959.2	2884.9	15.8	24.3
1650	1868.2	46.8	-1388.3	2826.8	16.1	24.8
1644	576.4	-30.5	-2481.9	2804.5	16.3	25.0
1782	3124.2	1561.5	-8.7	2713.2	16.8	25.8
1120	4.1	-1046.1	-2882.0	2530.1	18.0	27.7
1648	1619.2	131.3	-1221.8	2461.3	18.5	28.4
1122	3.6	-824.2	-2582.3	2287.2	19.9	30.6

Notes:

1. See Figure 3.4.3.4-3 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.4-4 Top 30 Stresses for Advanced Transfer Cask Inner Shell Element Bottom Surface

Node[1]	Principal Stresses (psi)			Nodal Von Mises Stresses	FS on Yield (S_y)[2]	FS on Ultimate (S_u)[2]
	S1	S2	S3			
1869	21448.0	632.6	-1960.3	22226.0	NA[3]	NA[3]
1882	8980.2	1059.9	-688.5	8923.9	NA	NA
10174	8109.4	7767.7	-819.1	8762.6	NA	NA
1797	1665.5	195.3	-7189.2	8218.8	NA	NA
1633	34.4	-2893.9	-8886.1	7875.8	NA	NA
1884	7160.2	652.9	-518.7	7165.3	6.4	NA
1731	1798.5	-157.2	-5950.2	6979.4	6.5	10.0
1803	501.1	-4651.7	-6891.7	6565.9	6.9	10.7
1725	819.9	-847.5	-6386.4	6534.2	7.0	10.7
1729	2571.3	-23.3	-4710.5	6392.4	7.1	11.0
1801	451.5	-4185.3	-6697.8	6282.0	7.3	11.1
1886	6799.4	2900.0	-79.4	5975.0	7.6	11.7
1879	5957.9	215.5	-205.1	5963.8	7.6	11.7
1638	5833.9	814.9	-647.9	5888.3	7.7	11.9
1799	450.0	-2722.7	-6304.4	5853.1	7.8	12.0
1742	1683.1	-3.7	-4630.1	5661.5	8.1	12.4
1822	1331.5	5.0	-4781.8	5569.8	8.2	12.6
1782	2155.7	-3.5	-4010.0	5418.9	8.4	12.9
1727	2988.1	35.3	-2969.9	5159.9	8.8	13.6
1766	2423.2	-1.0	-3317.9	4992.0	9.1	14.0
1784	2724.4	-2.3	-2938.1	4905.0	9.3	14.3
1838	1172.3	36.4	-4115.5	4821.3	9.5	14.5
1740	2640.2	-5.1	-2772.6	4688.0	9.7	14.9
1768	2402.5	-0.5	-2701.2	4422.5	10.3	15.8
1806	4006.7	141.6	-771.3	4393.2	10.4	15.9
1750	2260.4	0.0	-2725.3	4323.9	10.5	16.2
1666	18.8	-2100.4	-4642.2	4042.1	11.3	17.3
1786	2648.7	1.1	-1951.0	3998.6	11.4	17.5
1636	418.4	-283.5	-3777.7	3892.9	11.7	18.0
1646	2917.4	117.6	-1523.0	3888.9	11.7	18.0

Notes:

1. See Figure 3.4.3.4-3 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.4-5 Top 30 Stresses for Advanced Transfer Cask Stiffener Plate Element Top Surface

Node[1]	**Principal Stresses (psi)**			**Nodal Von Mises Stresses**	**FS on Yield (S_y)**[2]	**FS on Ultimate (S_u)**[2]
	S1	**S2**	**S3**			
717	21871.0	0.0	-3327.1	23710.0	NA[3]	NA[3]
10202	10380.0	2355.3	0.0	9425.9	NA	NA
703	0.0	-2611.0	-7384.1	6485.5	7.0	10.8
715	2540.9	0.0	-4590.8	6260.7	7.3	11.2
10174	0.0	-331.0	-6322.5	6163.7	7.4	11.4
10200	5583.6	0.0	-327.2	5754.1	7.9	12.2
3653	0.0	-234.7	-5162.0	5048.8	9.0	13.9
10238	1540.5	0.0	-3784.2	4745.8	9.6	14.7
10186	0.0	-353.6	-4708.3	4541.8	10.0	15.4
10242	2112.4	0.0	-3100.6	4541.5	10.0	15.4
10244	2350.2	0.0	-2849.7	4510.2	10.1	15.5
10240	1634.9	0.0	-3271.9	4327.5	10.5	16.2
10246	2401.3	0.0	-2507.3	4251.3	10.7	16.5
10217	2848.4	0.0	-2000.3	4220.5	10.8	16.6
10219	3030.0	0.0	-1779.4	4211.7	10.8	16.6
10215	2588.3	0.0	-2137.9	4099.2	11.1	17.1
3657	1182.4	0.0	-3351.0	4073.1	11.2	17.2
10221	3249.6	0.0	-1287.2	4049.6	11.3	17.3
10213	2350.3	0.0	-2163.4	3910.1	11.7	17.9
10198	3889.7	51.5	0.0	3864.2	11.8	18.1
10248	2329.7	0.0	-2066.7	3809.6	12.0	18.4
3659	1493.7	0.0	-2771.5	3748.6	12.2	18.7
3655	0.0	-122.0	-3793.1	3733.6	12.2	18.7
10211	2126.4	0.0	-2015.8	3587.7	12.7	19.5
3661	1862.2	0.0	-2213.6	3534.1	12.9	19.8
3669	3134.7	0.0	-384.7	3343.7	13.6	20.9
3663	2090.4	0.0	-1721.4	3306.3	13.8	21.2
10250	2173.3	0.0	-1540.2	3231.5	14.1	21.7
3665	2305.8	0.0	-1283.8	3150.4	14.5	22.2
10209	1909.7	0.0	-1598.9	3042.5	15.0	23.0

Notes:
1. See Figure 3.4.3.4-4 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

Table 3.4.3.4-6 Top 30 Stresses for Advanced Transfer Cask Stiffener Plate Element Bottom Surface

Node[1]	Principal Stresses (psi)			Nodal Von Mises Stresses	FS on Yield $(S_y)^2$	FS on Ultimate $(S_u)^2$
	S1	S2	S3			
717	21871.0	0.0	-3327.1	23710.0	NA[3]	NA[3]
10202	10380.0	2355.3	0.0	9425.9	NA	NA
703	0.0	-2611.0	-7384.1	6485.5	7.0	10.8
715	2540.9	0.0	-4590.8	6260.7	7.3	11.2
10174	0.0	-331.0	-6322.5	6163.7	7.4	11.4
10200	5583.6	0.0	-327.2	5754.1	7.9	12.2
3653	0.0	-234.7	-5162.0	5048.8	9.0	13.9
10238	1540.5	0.0	-3784.2	4745.8	9.6	14.7
10186	0.0	-353.6	-4708.3	4541.8	10.0	15.4
10242	2112.4	0.0	-3100.6	4541.5	10.0	15.4
10244	2350.2	0.0	-2849.7	4510.2	10.1	15.5
10240	1634.9	0.0	-3271.9	4327.5	10.5	16.2
10246	2401.3	0.0	-2507.3	4251.3	10.7	16.5
10217	2848.4	0.0	-2000.3	4220.5	10.8	16.6
10219	3030.0	0.0	-1779.4	4211.7	10.8	16.6
10215	2588.3	0.0	-2137.9	4099.2	11.1	17.1
3657	1182.4	0.0	-3351.0	4073.1	11.2	17.2
10221	3249.6	0.0	-1287.2	4049.6	11.3	17.3
10213	2350.3	0.0	-2163.4	3910.1	11.7	17.9
10198	3889.7	51.5	0.0	3864.2	11.8	18.1
10248	2329.7	0.0	-2066.7	3809.6	12.0	18.4
3659	1493.7	0.0	-2771.5	3748.6	12.2	18.7
3655	0.0	-122.0	-3793.1	3733.6	12.2	18.7
10211	2126.4	0.0	-2015.8	3587.7	12.7	19.5
3661	1862.2	0.0	-2213.6	3534.1	12.9	19.8
3669	3134.7	0.0	-384.7	3343.7	13.6	20.9
3663	2090.4	0.0	-1721.4	3306.3	13.8	21.2
10250	2173.3	0.0	-1540.2	3231.5	14.1	21.7
3665	2305.8	0.0	-1283.8	3150.4	14.5	22.2
10209	1909.7	0.0	-1598.9	3042.5	15.0	23.0

Notes:

1. See Figure 3.4.3.4-4 for node locations.
2. S_y = 45,600 psi, S_u = 70,000 psi.
3. Local stresses that are relieved by local material yielding. Therefore, stress design factors of 6 and 10 on material yield and ultimate strength are not applicable (ANSI N14.6, Section 4.2.1.2).

3.4.4 Normal Operating Conditions Analysis

The Universal Storage System is evaluated using individual finite element models for the fuel basket, canister, and vertical concrete cask. Because the individual components are free to expand without interference, the structural finite element models need not be connected.

3.4.4.1 Canister and Basket Analyses

The evaluations presented in this Section are based on consideration of the bounding conditions for each aspect of the analysis. Generally, the bounding condition is represented by the component, or combination of components, of each configuration that is the heaviest. The bounding thermal condition is established by the configuration having the largest thermal gradient in normal use. Some cases require the evaluation of both a PWR and a BWR configuration because of differences in the design of these systems. For reference, the bounding case used in each of the structural evaluations is:

Section	Aspect Evaluated	Bounding Condition	Configuration
3.4.4.1.1	Canister Thermal Stress	Largest temperature gradient	Temperature[a] distribution
3.4.4.1.2	Canister Dead Weight	Heaviest loaded canister	BWR Class 5
3.4.4.1.3	Canister Pressure	Bounding pressure 15 psig, smallest canister	PWR Class 1 BWR Class 4
3.4.4.1.4	Canister Handling	Shortest canister dimensions w/ heaviest canister load [b]	PWR Class 1 BWR Class 5
3.4.4.1.5	Canister Load Combinations	Bounding pressure 15 psig + shortest canister dimensions w/ heaviest loaded canister[b] (handling) + shortest canister dimensions w/ heaviest loaded canister [b] (dead load) largest temperature gradient (thermal)	PWR Class 3 PWR Class 1 BWR Class 5 PWR Class 1 BWR Class 5 Temperature[a] distribution
3.4.4.1.6	Canister Fatigue	Bounding thermal excursions (58°F)	Not Applicable
3.4.4.1.7	Canister Pressure Test	Loaded canister (smallest canister)	PWR Class 1
3.4.4.1.8	PWR Basket Support Disk	Loaded PWR Canister	PWR fuel basket
	BWR Basket Support Disk	Loaded BWR Canister	BWR fuel basket [c]
3.4.4.1.9	PWR Basket Weldment	Loaded PWR Canister	PWR Class 2
	BWR Basket Weldment	Loaded BWR Canister	BWR Class 5
3.4.4.1.10	PWR Fuel Tube	Loaded PWR Canister (Longest)	PWR Class 3
	BWR Fuel Tube	Loaded BWR Canister (Longest)	BWR Class 5
3.4.4.1.11	Canister Closure Weld	Same as 3.4.4.1.5	Same as 3.4.4.1.5

[a] See Section 3.4.4.1.1 for an explanation of the composite temperature distribution used in the analyses. The shortest canister, PWR Class 1, has the fewest number of fuel basket support disks.

[b] When combined with the heaviest fuel assembly/fuel basket weight (BWR Class 5), the load per support disk or weldment disk is maximized.

[c] The evaluation of the BWR basket uses the analysis presented in the UMS Transport SAR [2].

3.4.4.1.1 Canister Thermal Stress Analysis

A three-dimensional finite element model of the canister was constructed using ANSYS SOLID45 elements. By taking advantage of the symmetry of the canister, the model represents one-half (180° section) of the canister including the canister shell, bottom plate, structural lid, and shield lid. Contact between the structural and shield lids is modeled using COMBIN40 combination elements in the axial (UY) degree of freedom. Simulation of the spacer ring is accomplished using a ring of COMBIN40 gap/spring elements connecting the shield lid and the canister in the axial direction at the lid lower outside radius. In addition, CONTAC52 elements are used to model the interaction between the structural lid and the canister shell and between the shield lid and canister shell, just below the respective lid weld joints as shown in Figure 3.4.4.1-2. The size of the CONTAC52 gaps is determined from nominal dimensions of contacting components. The gap size is defined by the "Real Constant" of the CONTAC52 element. Due to the relatively large gaps resulting from the nominal geometry, these gaps remain open during all loadings considered. The COMBIN40 elements used between the structural and shield lids and for the spacer ring are assigned small gap sizes of 1×10^{-8} in. All gap/spring elements are assigned a stiffness of 1×10^{8} lb/in. The three-dimensional finite element model of the canister used in the thermal stress evaluation is shown in Figure 3.4.4.1-1 through Figure 3.4.4.1-3.

The model is constrained in the Z-direction for all nodes in the plane of symmetry. For the stability of the solution, one node at the center of the bottom plate is constrained in the Y-direction, and all nodes at the centerline of the canister are constrained in the X-direction. The directions of the coordinate system are shown in Figure 3.4.4.1-1.

This model represents a "bounding" combination of geometry and loading that envelopes the Universal Storage System PWR and BWR canisters. Specifically, the shortest canister (PWR Class 1) and minimum weld sizes (0.75-inch structural lid weld and 0.375-inch shield lid weld) are modeled in conjunction with the heaviest fuel and fuel basket combination (BWR Class 5). By using the shortest canister (PWR Class 1), which has the fewest number of support disks, in combination with the weight of the heaviest loaded fuel basket, the load per support disk and weldment disk is maximized. Thus, the analysis yields very conservative results relative to the expected performance of the actual canister configurations.

The finite element thermal stress analysis is performed with canister temperatures that envelope the canister temperature gradients for off-normal storage (106°F and -40°F ambient temperatures) and transfer conditions for all canister configurations. Prior to performing the thermal stress analysis, the steady-state temperature distribution is determined using temperature data from the storage and transfer thermal analyses (Chapter 4.0). This is accomplished by converting the SOLID45 structural elements of the canister model to SOLID70 thermal elements and using the material properties from the thermal analyses. Nodal temperatures are applied at six key locations for the steady state heat transfer analysis — top-center of the structural lid, top-outer diameter of the structural lid, bottom-center of the shield lid, bottom-center of the bottom plate, bottom-outer diameter of the bottom plate, and mid-elevation of the canister shell.

Two temperature distributions are used in the structural analyses to envelope the worst-case allowable temperatures and temperature gradients experienced by all PWR and BWR canister configurations under storage and transfer conditions. The temperatures at the key locations are:

Top center of the structural lid	= 160
Top outer diameter of the structural lid	= 150
Bottom center of the shield lid	= 200
Bottom center of the bottom plate	= 300
Bottom outer diameter of the bottom plate	= 200
Mid-elevation of the canister shell	= 600

Temperatures used for determining allowable stress values were selected to envelope the maximum temperatures experienced by the canister components during storage and transfer conditions. Allowable stress values for the structural/shield lid region were taken at 220°F, those for the center of the bottom plate were taken at 300°F, those for the outer radius of the bottom plate at 220°F, and those for the canister shell at 550°F.

The temperatures for all nodes in the canister model are obtained by the solution of the steady state thermal conduction problem. The key temperature differences, ΔT, of the worst-case

PWR and BWR canisters in the radial and axial directions and those used in the canister thermal stress analysis are:

	Maximum ΔT (°F)							
	Top of Structural Lid (Radial)		Bottom Plate (Radial)		Shield and Structural Lid (Axial)		Canister Shell (Axial)	
Condition	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
Storage, Normal 76°F ambient	3	3	3	7	6	8	267	299
Storage, Off-Normal 106°F ambient	4	3	3	7	6	8	266	298
Storage, Off-Normal, -40°F ambient	3	3	4	7	5	7	264	296
Storage, Off-Normal Half Inlets Blocked 76°F	4	3	3	7	6	8	265	296
Transfer, 76°F ambient	10	4	69	64	16	7	396	388
Parameters used for Canister Thermal Stress Analysis	10		100		40		450	

The resulting maximum (secondary) thermal stresses in the canister are summarized in Table 3.4.4.1-1. The sectional stresses at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations of the stress sections are shown in Figure 3.4.4.1-4. After solving for the canister temperature distribution, the thermal stress analysis was performed by converting the SOLID70 elements back to SOLID45 structural elements.

3.4.4.1.2 Canister Dead Weight Load Analysis

The canister is structurally analyzed for dead weight load using the finite element model described in Section 3.4.4.1.1. The canister temperature distribution discussed in Section 3.4.4.1.1 is used in the dead load structural analysis to evaluate the material properties at temperature. The fuel and fuel basket assembly contained within the canister are not explicitly modeled but are included in the analysis by applying a uniform pressure load representing their combined weight to the top surface of the canister bottom plate. The nodes on the bottom surface of the bottom plate are restrained in the axial direction in conjunction with the constraints described in Section 3.4.4.1.1. The evaluation is based on the weight of the BWR Class 5 canister, which has the highest weight, and the length of the PWR Class 1 canister, which is the shortest configuration and has minimum weld sizes (0.75-inch structural lid weld and 0.375-inch shield lid weld). An acceleration of 1g is applied to the model in the axial direction (Y) to simulate the dead load.

The resulting maximum canister dead load stresses are summarized in Table 3.4.4.1-2 and Table 3.4.4.1-3 for primary membrane and primary membrane plus bending stresses, respectively. The sectional stresses at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations for the stress sections are shown in Figure 3.4.4.1-4.

The lid support ring is evaluated for the dead load condition using classical methods. The ring, which is made of ASTM A-479, Type 304 stainless steel, is welded to the inner surface of the canister shell to support the shield lid. For conservatism, a temperature of 400°F, which is higher than the anticipated temperature at this location, is used to determine the material allowable stress. The total weight, W, imposed on the lid support ring is conservatively considered to be the weight of the auxiliary shielding and the shield lid. A 10% load factor is also applied to ensure that the analysis bounds all normal operating loads. The stresses on the support ring are the bearing stresses and shear stresses at its weld to the canister shell.

The bearing stress $\sigma_{bearing}$ is:

$$\sigma_{bearing} = \frac{W}{area} = \frac{14{,}200\ \text{lb}}{102.6\ \text{in}^2} = 138\ \text{psi}$$

where:

W = (7,000 lb + 5,890 lb) × 1.1 = 14,179 lb, use 14,200 lb
where the weight of the auxiliary shielding (W_s) can be comprised of three 2-inch-thick stainless steel plates resting on the shield lid, or

W_s = .291 x (π/4) x 65.5^2 x 6 = 5,883 lb, use 5,890 lb

$$A = \frac{\pi}{4}(D^2 - (D-2t)^2)\,in^2 = 102.6\ in^2$$

D = lid support ring diameter = 65.81 in.

t = radial thickness of support ring = 0.5 in.

The yield strength, S_y, for A-479, Type 304 stainless steel = 20,700 psi, and the ultimate allowable tensile stress, S_u = 64,400 psi at 400°F. The allowable bearing stress is 1.0 S_y per ASME Code, Section III, Subsection NB. The acceptability of the support ring design is evaluated by comparing the allowable stresses to the maximum calculated stress:

$$MS = \frac{20{,}700\ psi}{138\ psi} - 1 = +Large$$

Therefore, the support ring is structurally adequate.

The attachment weld for the lid support ring is a 1/8-in. partial penetration groove weld. The total shear force on the weld is considered to be the weight of the shield lid, the structural lid, and the lid support ring. The total effective area of each weld is A_{eff} = .125 × π × 65.81 in. = 25.8 in^2. The average shear stress in the weld is:

$$\sigma_w = \frac{W}{A_{eff}} = \frac{14{,}200\ lb}{25.8\ in^2} = 550\ psi$$

The allowable stress on the weld is 0.30 × the nominal tensile strength of the weld material [Ref.23, Table J2.5]. The nominal tensile strength of E308-XX filler material is 80,000 psi [Ref.28, SFA-5.4, Table 5]. However, for conservatism, S_y and S_u for the base metal, are used. The acceptability of the support ring weld is evaluated by comparing the allowable stress to the maximum calculated stress:

$$MS = \frac{0.3 \times 20{,}700\ psi}{550\ psi} - 1 = +Large$$

Therefore, the support ring attachment weld is structurally adequate.

3.4.4.1.3 Canister Maximum Internal Pressure Analysis

The canister is structurally analyzed for a maximum internal pressure load using the finite element model and temperature distribution and restraints described in Section 3.4.4.1.1. A maximum internal pressure of 15 psig is applied as a surface load to the elements along the internal surface of the canister shell, bottom plate, and shield lid. This pressure bounds the calculated pressure of 7.1 psig that occurs in the smallest canister, PWR Class 1, under normal conditions. The PWR Class 1 canister internal pressure bounds the internal pressures of the other four canister configurations because it has the highest quantity of fission-gas-to-volume ratio.

The resulting maximum canister stresses for maximum internal pressure load are summarized in Table 3.4.4.1-9 and Table 3.4.4.1-10 for primary membrane and primary membrane plus primary bending stresses, respectively. The sectional stresses at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations of the stress sections are shown in Figure 3.4.4.1-4.

3.4.4.1.4 Canister Handling Analysis

The canister is structurally analyzed for handling loads using the finite element model and conditions described in Section 3.4.4.1.1. Normal handling is simulated by restraining the model at nodes on the structural lid simulating three lift points and applying a 1.1g acceleration, which includes a 10% dynamic load factor, to the model in the axial direction. The canister is lifted at six points; however, a three-point lifting configuration is conservatively used in the handling analysis. Since the model represents a one-half section of the canister, the three-point lift is simulated by restraining two nodes 120° apart (one node at the symmetry plane and a second node 120° from the first) along the bolt diameter at the top of the structural lid in the axial direction. Additionally, the nodes along the centerline of the lids and bottom plate are restrained in the radial direction, and the nodes along the symmetry face are restrained in the direction normal to the symmetry plane.

The maximum stresses during canister handling occur when the heaviest weight canister (BWR class 5) is analyzed with the minimum structural lid weld (PWR class canister with 0.75-inch structural lid weld). Therefore, this analysis bounds all handling configurations.

The resulting maximum stresses in the canister are summarized in Table 3.4.4.1-4 and Table 3.4.4.1-5 for primary membrane and primary membrane plus primary bending stresses, respectively. The sectional stresses at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations of the stress sections are shown in Figure 3.4.4.1-4.

3.4.4.1.5 Canister Load Combinations

The canister is structurally analyzed for combined thermal, dead, maximum internal pressure, and handling loads using the finite element model and the conditions described in Section 3.4.4.1.1. Loads are applied to the model as discussed in Sections 3.4.4.1.1 through 3.4.4.1.4. A maximum internal pressure of 15.0 psi is used in conjunction with a positive axial acceleration of 1.1g. Two nodes 120° apart (one node at the symmetry plane and a second node 120° from the first) are restrained along the bolt diameter at the top of the structural lid in the axial direction. Additionally, the nodes along the centerline of the lids and bottom plate are restrained in the radial direction, and the nodes along the symmetry face are restrained in the direction normal to the symmetry plane.

The resulting maximum stresses in the canister for combined loads are summarized in Table 3.4.4.1-6, Table 3.4.4.1-7, and Table 3.4.4.1-8, for primary membrane, primary membrane plus primary bending, and primary plus secondary stresses, respectively. The sectional stresses at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations for the stress sections are shown in Figure 3.4.4.1-4.

As shown in Table 3.4.4.1-6 through Table 3.4.4.1-8, the canister maintains positive margins of safety for the combined load conditions.

3.4.4.1.6 Canister and Basket Fatigue Evaluation

The purpose of this section is to evaluate whether an analysis for cyclic service is required for the Universal Storage System components. The requirements for analysis for cyclic operation of components designed to ASME Code criteria are presented in ASME Section III, Subsection NB-3222.4 [5] for the canister and Subsection NG-3222.4 [6] for the fuel basket. Guidance for components designed to AISC standards is in the Manual of Steel Construction, Table A-K4.1 [23].

During storage conditions, the canister is housed in the vertical concrete cask. The concrete cask is a shielded, reinforced concrete overpack designed to hold a canister during long-term storage conditions. The cask is constructed of a thick inner steel shell surrounded by 28 in. of reinforced concrete. The cask inner shell is not subjected to cyclic mechanical loading. Thermal cycles are limited to changes in ambient air temperature. Because of the large thermal mass of the concrete cask and the relatively minor changes in ambient air temperature (when compared to the steady state heat load of the cask contents), fatigue as a result of cycles in ambient air is not significant, and no further fatigue evaluation of the inner shell is required.

ASME criteria for determining whether cyclic loading analysis is required are comprised of six conditions, which, if met, preclude the requirement for further analysis:

1. Atmospheric to Service Pressure Cycle
2. Normal Service Pressure Fluctuation
3. Temperature Difference — Startup and Shutdown
4. Temperature Difference — Normal Service
5. Temperature Difference — Dissimilar Materials
6. Mechanical Loads

Evaluation of these conditions follows.

Condition 1 — Atmospheric to Service Pressure Cycle

This condition is not applicable. The ASME Code defines a cycle as an excursion from atmospheric pressure to service pressure and back to atmospheric pressure. Once sealed, the canister remains closed throughout its operational life, and no atmospheric to service pressure cycles occur.

Condition 2 — Normal Service Pressure Fluctuation

This condition is not applicable. The condition establishes a maximum pressure fluctuation as a function of the number of significant pressure fluctuation cycles specified for the component, the design pressure, and the allowable stress intensity of the component material. Operation of the canister is not cyclic, and no significant cyclic pressure fluctuation is anticipated.

Condition 3 — Temperature Difference — Startup and Shutdown

This condition is not applicable. The Universal Storage System is a passive, long-term storage system that does not experience cyclic startups and shutdowns.

Condition 4 — Temperature Difference — Normal and Off-Normal Service

The ASME Code specifies that temperature excursions are not significant if the change in ΔT between two adjacent points does not experience a cyclic change of more than the quantity:

$$\Delta T = \frac{S_a}{2E\alpha} = 58^\circ F,$$

where, for Type 304L stainless steel,

S_a = 28,200 psi, the value obtained from the fatigue curve for service cycles < 10^6,
E = 26.5×10^6 psi, modulus of elasticity at 400 °F,
α = 9.19×10^{-6} in./in.-°F.

Because of the large thermal mass of the canister and the concrete cask and the relatively constant heat load produced by the canister's contents, cyclic changes in ΔT greater than 58 °F will not occur.

Condition 5 — Temperature Difference Between Dissimilar Materials

The canister and its internal components contain several materials. However, the design of all components considers thermal expansion, thus precluding the development of unanalyzed thermal stress concentrations.

Condition 6 — Mechanical Loads

This condition does not apply. Cyclic mechanical loads are not applied to the vertical concrete cask and canister during storage conditions. Therefore, no further cyclic loading evaluation is required.

The criteria ASME Code Subsections NB-3222.4 and NG-3222.4 are met, and no fatigue analysis is required.

3.4.4.1.7 Canister Pressure Test

The canister is designed and fabricated to the requirements of ASME Code, Subsection NB, to the extent possible. A 35 psia (35 – 14.7 = 20.3 psig) hydrostatic pressure test is performed in accordance with the requirements of ASME Code Subsection NB-6220 [5]. The pressure test is performed after the shield lid to canister shell weld is completed. The test pressure slightly exceeds 1.25 × design pressure (1.25 × 15 psig = 18.75 psig). Considering head pressure for the tallest canister (191.75 × 0.036 = 6.9 psig), the maximum canister pressure developed during the pneumatic pressure test is bounded by using 27.2 psig in the structural evaluation for the canister test pressure.

The ASME Code requires that the pressure test loading comply with the following criteria from Subsection NB-3226:

(a) P_m shall not exceed $0.9S_y$ at test temperature. For convenience, the stress intensities developed in the analysis of the canister due to a normal internal pressure of 15 psig (Tables 3.4.4.1-9 and 3.4.4.1-10) are ratioed to demonstrate compliance with this requirement. From Table 3.4.4.1-9, the maximum primary stress intensity, P_m, is 2.24 ksi. The canister material is ASME SA-240, Type 304L stainless steel, and the test temperature will be less than 200°F for the design basis heat load of 23 kW (Figures 4.4.3-5 and 4.4.3-6). Since yield strength decreases with increasing temperature, for purposes of this calculation, the minimum material yield strength at the bounding canister temperature of 200°F is used for the structural critical limit.

$(P_m)_{test}$ = (27.2/15)(2.24 ksi) = 4.1 ksi, which is < 0.9 S_y = 0.9 (21.4 ksi) = 19.3 ksi

Thus, criterion (a) is met.

(b) For P_m <$0.67S_y$ (see criterion a), the primary membrane plus bending stress intensity, P_m + P_b, shall be ≤ $1.35S_y$. From Table 3.4.4.1-10, $P_m + P_b$ = 7.36 ksi.

$(P_m + P_b)_{test}$ = (27.2/15)×(7.36 ksi) = 13.3 ksi, which is ≤ $1.35S_y$ = 28.9 ksi (1.35×21.4 ksi).

Thus, criterion (b) is met.

(c) The external pressure shall not exceed 135% of the value determined by the rules of NB-3133. The exterior of the canister is at atmospheric pressure at the time the pressure test is conducted. Therefore, this criterion is met.

(d) For the 1.25 Design Pressure pneumatic test of NB-6221, the stresses shall be calculated and compared to the limits of criteria (a), (b), and (c). This calculation and the fatigue evaluation of (e) need not be revised unless the actual hydrostatic test pressure exceeds 1.25 Design Pressure by more than 6%.

The test pressure (20.3 psig) slightly exceeds 1.25 × Design Pressure (18.75). However, the stresses used in this evaluation are ratioed to the test pressure. Thus, the stresses at the test pressure are calculated.

(e) Tests, with the exception of the first 10 hydrostatic tests in accordance with NB-6220, shall be considered in the fatigue evaluation of the component.

The canisters are not reused, and the hydrostatic test will be conducted only once. Thus, the pressure test is not required to be considered in the fatigue analysis.

The canister hydrostatic pressure tests comply with all NB-3226 criteria. These results bound the performance of a pneumatic pressure test performed in accordance with NB-6220, since the pneumatic pressure test pressure is lower (1.2 × the design pressure or 1.2 × 15 psig = 18 psig).

3.4.4.1.8 Fuel Basket Support Disk Evaluation

The PWR and BWR fuel baskets are described in detail in Sections 1.2.1.2.1 and 1.2.1.2.2, respectively. The design of the basket is similar for the PWR and BWR configurations. The major components of the BWR basket are shown in Figure 3.4.4.1-5. The structural evaluation for the PWR and BWR support disks for the normal conditions of storage is presented in the following sections. Note that the canister may be handled in a vertical or horizontal position. The evaluation is performed for the governing configuration in which the canister is handled in a vertical position. During normal conditions, the support disk is subjected to its self-weight only (in canister axial direction) and is supported by the tie rods/spacers at 8 locations for PWR configuration and 6 locations for the BWR configuration. To account for the condition when the canister is handled, a handling load, defined as 10 percent of the dead load, is considered. Finite element analyses using the ANSYS program are performed for the support disk for PWR and BWR configurations, respectively. In addition to the dead load and handling load (10% of dead load), thermal stresses are also considered based on conservative temperatures that envelop those experienced by the support disk during normal, off-normal (106°F and –40°F ambient temperatures) and transfer conditions. The stress criteria is defined according to ASME Code, Section III, Subsection NG. For the normal condition of storage, the Level A allowable stresses from Subsection NG as shown below are used.

Stress Category	Normal (Level A) Allowable Stresses
P_m	S_m
P_m+P_b	1.5 S_m
P+Q	3.0 S_m

3.4.4.1.8.1 PWR Support Disk

As shown in Figure 3.4.4.1-6, a finite element model is generated to analyze the PWR fuel basket support disks. The model is constructed using the ANSYS three-dimensional SHELL63 elements and corresponds to a single support disk with a thickness of 0.5 inch. The only loading on the model is the inertial load (1.1g) that includes the dead load and handling load in the out-of-plane direction (Global Z) for normal conditions of storage. The model is constrained in eight locations in the out-of-plane direction to simulate the supports of the tie rods/spacers.

Note that a full model is generated because this model is also used for the evaluation of the support disk for the off-normal handling condition (Section 11.1.3) in which non-symmetric loading (side load) is present. In addition, this model is used for the evaluation of a support disk for the 24-inch end drop accident condition of the vertical concrete cask (Section 11.2.4).

The model accommodates thermal expansion effects by using the temperature data from the thermal analysis and the coefficient of thermal expansion. Prior to performing the structural analyses, the temperature distribution in the support disk is determined by executing a steady-state thermal conduction analysis. This is accomplished by converting the SHELL63 structural elements to SHELL57 thermal elements. A maximum temperature of 700°F is applied to the nodes at the center slot of the disk model, and a minimum temperature 275°F is applied to the nodes around the outer circumferential edge of the disk, thus providing a bounding temperature delta of 425°F for the support disk. All other nodal temperatures are then obtained by the steady state conduction solution. Note that the applied temperatures are conservatively selected to envelope the maximum temperature, as well as the maximum radial temperature gradient (ΔT) of the disk for all normal, off-normal and accident conditions of storage and for transfer conditions. For normal conditions of storage, the support disk is evaluated using stress allowables at 800°F.

To evaluate the most critical regions of the support disk, a series of cross sections are considered. The locations of these sections on a PWR support disk are shown in Figures 3.4.4.1-7 and

3.4.4.1-8. Table 3.4.4.1-11 lists the cross sections versus Point 1 and Point 2, which spans the cross section of the ligament in the plane of the support disk.

The stress evaluation for the support disk is performed according to ASME Code, Section III, Subsection NG. According to this subsection, linearized stresses of cross sections of the structure are to be compared against the allowable stresses. The stress evaluation results for the support disks for normal condition are presented in Tables 3.4.4.1-12 and 3.4.4.1-13. The tables list the 40 highest P_m+P_b and P+Q stress intensities with large margins of safety. The Level A allowable stresses, $1.5S_m$ and $3S_m$ of the 17-4PH stainless steel at corresponding nodal temperatures, are used for the P_m+P_b and P+Q stresses, respectively. Note that the P_m stresses for the support disk for normal conditions are essentially zero since there are no loads in the plane of the support disk. Stress allowables for the section cuts are taken at 800°F.

3.4.4.1.8.2 BWR Support Disk

Similar to the evaluation for the PWR fuel basket support disk, a finite element model is generated to analyze the BWR fuel basket support disks, as shown in Figure 3.4.4.1-12. The model is constructed using the ANSYS three-dimensional SHELL63 elements and corresponds to a single support disk with a thickness of 5/8 inch. The only loading on the model is the inertial load (1.1g) that includes the dead load and handling load in the out-of-plane direction (Global Z) for normal conditions of storage. The model is constrained in six locations in the out-of-plane direction to simulate the supports of the tie rods/spacers.

The model accommodates thermal expansion effects by using the temperature data from the thermal analysis and the coefficient of thermal expansion. The temperature distribution in the BWR support disk is determined using the same method used in Section 3.4.4.1.8.1 for the PWR support disk. A maximum temperature of 700°F is applied to the nodes at the center of the disk model, and a minimum temperature of 300°F is applied to the nodes around the outer circumferential edge of the disk, thus providing a bounding temperature delta of 400°F for the support disk. All other nodal temperatures are then obtained by the steady state conduction solution. Note that the applied temperatures are conservatively selected to envelope the maximum temperature, as well as the maximum radial temperature gradient (ΔT) of the disk for all normal, off-normal, and accident conditions of storage and for transfer conditions. For normal conditions of storage, the support disk is evaluated using stress allowables at 800°F.

To evaluate the most critical regions of the support disk, a series of cross sections are considered. The locations of these sections on a BWR support disk are shown in Figures 3.4.4.1-13 through 3.4.4.1-16. Table 3.4.4.1-14 lists the cross sections versus Point 1 and Point 2, which spans the cross section of the ligament in the plane of the support disk.

The stress evaluation results for the BWR support disks for normal condition are presented in Tables 3.4.4.1-15 and 3.4.4.1-16. The tables list the 40 highest P_m+P_b and P+Q stress intensities with large margins of safety. The Level A allowable stresses from ASME Code, Section III, Subsection NG, 1.5S_m and 3.0S_m of the SA533 carbon steel at corresponding nodal temperatures, are used for the P_m+P_b and P+Q stresses, respectively. Note that the P_m stresses for the support disk for normal conditions are essentially zero, since there is no loads in the plane of the support disk.

3.4.4.1.9 Fuel Basket Weldments Evaluation

The PWR and BWR fuel basket weldments are evaluated for normal storage conditions using the finite element method. In addition to the dead load of the weldment, a 10% dynamic load factor is considered to account for handling loads. Therefore, a total acceleration of 1.1g is applied to the weldment model in the out of plane direction. Thermal stresses for the basket weldments are determined using the method presented in Sections 3.4.4.1.8.1 and 3.4.4.1.8.2 for the PWR and BWR support disks, respectively. The temperatures used in the model to establish the weldment temperature gradient are:

Basket Weldment	Temperature at Center of Weldment (°F)	Temperature at Edge of Weldment (°F)
PWR Top	600	275
PWR Bottom	325	175
BWR Top	525	225
BWR Bottom	475	200

These temperatures are conservatively selected to envelop the maximum temperature and the maximum radial temperature gradient of the weldments for all normal and off-normal conditions of storage. The results of the structural analyses for dead load, handling load, and thermal load are summarized in Table 3.4.4.1-17.

3.4.4.1.9.1 PWR Fuel Basket Weldments

The PWR top and bottom weldment plates are 1.25 and 1.0-in. thick Type 304 stainless steel plate, respectively. The weldments support their own weight plus the weight of up to 24 PWR fuel assembly tubes. An ANSYS finite element analysis was prepared for both plates because the support location for each weldment is different. Both models use the SHELL63 elements, which permits out-of-plane loading. The finite element models for the top and bottom weldments are shown in Figures 3.4.4.1-8 and 3.4.4.1-9, respectively. Note that the corner baffles are conservatively omitted in the top weldment model. The load from the fuel tube on the bottom weldment is represented as point forces applied to the nodes at the periphery of the fuel assembly slots. An average point force is applied. The application of the nodal loads at the slot periphery is accurate because the tube weight is transmitted to the edge of the slot, which provides support to the fuel tubes while in the vertical position.

The maximum stress intensity and the margin of safety for the weldments are shown in Table 3.4.4.1-17. Note that the nodal stress intensity is conservatively used for the evaluation. The Pm stresses for the weldments for normal conditions are essentially zero since there are no loads in the plane of the weldments. The weldments satisfy the stress criteria in the ASME Code Section III, Subsection NG [6].

3.4.4.1.9.2 BWR Fuel Basket Weldments

In the BWR fuel basket transport analysis, the responses of the top and bottom weldment plates to normal storage conditions are evaluated in conjunction with the thermal expansion stress. The weldment plates are 1.0-in. thick Type 304 stainless steel. The weldments support their own weight and the weight of up to 56 BWR fuel assembly tubes. A finite element analysis was performed for the top and bottom plates because the support for each weldment differs depending upon the location of the welded ribs for each. Both models use SHELL63 elements, which permit out-of-plane loading. The finite element models for the top and bottom weldments are shown in Figure 3.4.4.1-18 and Figure 3.4.4.1-19, respectively. The load from the fuel tube on the bottom weldment is represented as average point forces applied to the nodes at the periphery of the fuel assembly slots because the tube weight is transmitted to the edge of the slot in the end-impact condition.

The maximum stress intensity and the margin of safety for the weldments are shown in Table 3.4.4.1-17. Note that the nodal stress intensity is conservatively used for the evaluation. The P_m stresses for the weldments for normal conditions are essentially zero since there are no loads in the plane of the weldments. The weldments satisfy the stress criteria in the ASME Code Section III, Subsection NG [6].

3.4.4.1.10 Fuel Tube Analysis

Under normal storage conditions, the fuel tubes, Figure 3.4.4.1-9 (PWR) and Figure 3.4.4.1-17 (BWR), support only their own weight. The fuel assemblies are supported by the canister bottom plate, not by the fuel tubes. Thermal stresses are considered to be negligible since the tubes are free to expand axially and radially. The handling load is taken as 10% of the dead load.

The weight of the fuel tube, with a load of 1.1g (to account for both the dead load and handling load) is carried by the tube cross-section. The cross sectional area of a PWR fuel tube is:

$$\text{Area} = (8.9\text{ in})^2 - (8.9\text{ in.} - 2 \times 0.048\text{ in.})^2 = 1.7\text{ in}^2$$

The bounding weight of the heaviest PWR fuel tube is about 200 pounds. Considering a g-load of 1.1, the maximum compressive and bearing stress in the fuel tube is about 129 psi (200 lb × 1.1 / 1.7 in^2). Limiting the compressive stress level in the tube to the material yield strength ensures the tube remains in position in storage conditions. The yield strength of Type 304 stainless steel is 17,300 psi at a conservatively high temperature of 750°F.

$$MS = 17{,}300/129 - 1 = +\text{Large}$$

The minimum cross-sectional area of a BWR fuel tube and oversized fuel tube is:

$$\text{Area} = (5.996\text{ in})^2 - (5.9969\text{ in.} - 2 \times 0.048\text{ in.})^2 = 1.14\text{ in}^2$$

The bounding weight of the heaviest BWR fuel tube and oversized fuel tube is about 100 pounds. Considering a g-load of 1.1, the maximum compressive and bearing stress in the fuel tube is about 96 psi (100 lb × 1.1 / 1.14 in.^{2}). Limiting the compressive stress level in the tube to the material yield strength ensures the tube remains in position in storage conditions. The yield strength of Type 304 stainless steel is 17,300 psi at a conservatively high temperature of 750°F.

Margin of Safety = 17,300/96 - 1 = +Large

Thus, the tubes are structurally adequate under normal storage and handing conditions.

3.4.4.1.11 Canister Closure Weld Evaluation

The minimum closure weld for the canister is a 0.75-inch groove weld between the structural lid and the canister shell. The evaluation of this weld incorporates a 0.8 stress reduction factor in accordance with NRC Interim Staff Guidance (ISG) No. 15, Revision 0. The use of this factor is in accordance with ISG No. 15, since the strength of the weld material (E308) is greater than that of the base material (Type 304 or 304L stainless steel).

The stresses for the canister closure weld are evaluated using sectional stresses as permitted by Subsection NB of the ASME Code. The location of the section for the canister closure weld evaluation is shown in Figure 3.4.4.1-4 and corresponds to Section 13. The governing P_m, P_m+ P_b, and P + Q stress intensities for Section 13, and the associated allowables, are listed in Table 3.4.4.1-6, Table 3.4.4.1-7, and Table 3.4.4.1-8, respectively. The factored allowables, incorporating the 0.8 stress reduction factor, and the resulting controlling Margins of Safety are shown below.

This evaluation confirms that the canister closure weld is acceptable for normal operation conditions.

Stress Category	**Analysis Stress Intensity (ksi)**	**0.8 × Allowable Stress (ksi)**	**Margin of Safety**
P_m	1.90	13.36	6.03
$P_m + P_b$	2.67	20.04	6.51
P + Q	6.93	40.08	4.78

Critical Flaw Size for the Canister Closure Weld

The closure weld for the canister is comprised of multiple weld beads using a compatible weld material for Type 304L stainless steel. An allowable (critical) flaw evaluation has been performed to determine the critical flaw size in the weld region. The result of the flaw evaluation is used to define the minimum flaw size, which must be identifiable in the nondestructive examination of the weld. Due to the inherent toughness associated with Type 304L stainless steel, a limit load analysis is used in conjunction with a J-integral/tearing modulus approach.

The safety factor used in this evaluation is that defined in Section XI of the ASME Code [43].

The stress component used in the evaluation for the critical flaw size is the radial stress component in the weld region of the structural lid. For the normal operation condition, in accordance with ASME Code Section XI, a safety factor of 3 is required. For the purpose of identifying the stress for the flaw evaluation, the weld region corresponding to Section 13 in Figure 3.4.4.1-4 is considered. The radial stress corresponds to SX in Tables 3.4.4.1-1 through 3.4.4.1-10. The maximum reported radial tensile stress is 1.55 ksi.

To perform the flaw evaluation, a 10 ksi stress is conservatively used, resulting in a significantly larger actual safety factor than the required safety factor of 3. Using a 10 ksi stress as the basis for the evaluation of the structural lid weld, the critical flaw size is 0.44 inch for a flaw that extends 360 degrees around the circumference of the structural lid weld. Stress components for the circumferential (Z) and axial (Y) directions are also reported in Tables 3.4.4.1-1 through 3.4.4.1-10, which would be associated with flaws oriented in the radial or horizontal directions, respectively. As shown in Table 3.4.4.1-7 at Section No. 13 (the structural lid weld), the maximum tensile stress reported for these components (SY and SZ) is 1.8 ksi, which is also enveloped by the value of 10 ksi used in the critical flaw evaluation for stresses in the radial direction.

The 360-degree flaw employed for the circumferential direction is considered to be bounding with respect to any partial flaw in the weld, which could occur in the radial and horizontal directions. Therefore, using a minimum detectable flaw size of 0.375 inch is acceptable, since it is less than the very conservatively determined 0.44-inch critical flaw size.

The Type 304L stainless steel structural lid may be forged (SA-182 material), or fabricated from plate (SA-240 material). Since the forged material is required to have ultimate and yield strengths that are equal to, or greater than, the plate material, the critical flaw size determination is applicable to both materials.

Figure 3.4.4.1-1 Canister Composite Finite Element Model

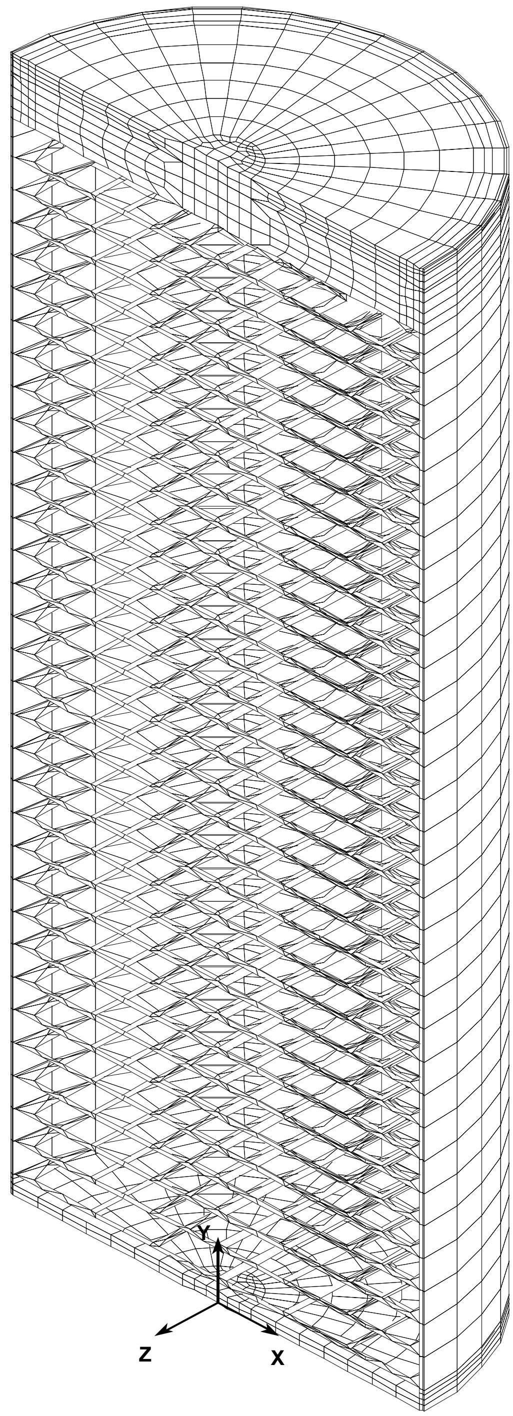


Figure 3.4.4.1-2 Weld Regions of Canister Composite Finite Element Model at Structural and Shield Lids

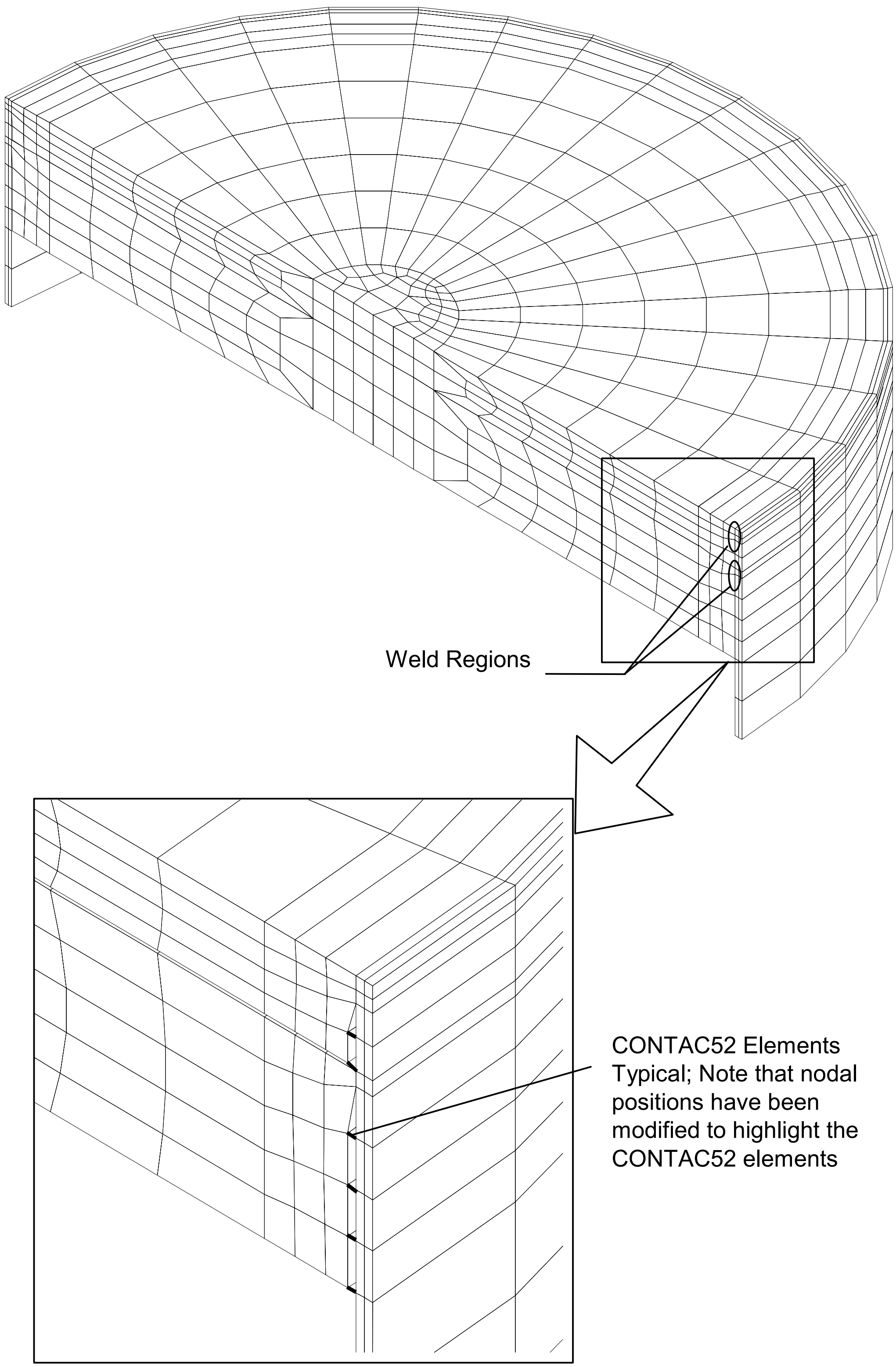


Figure 3.4.4.1-3 Bottom Plate of the Canister Composite Finite Element Model

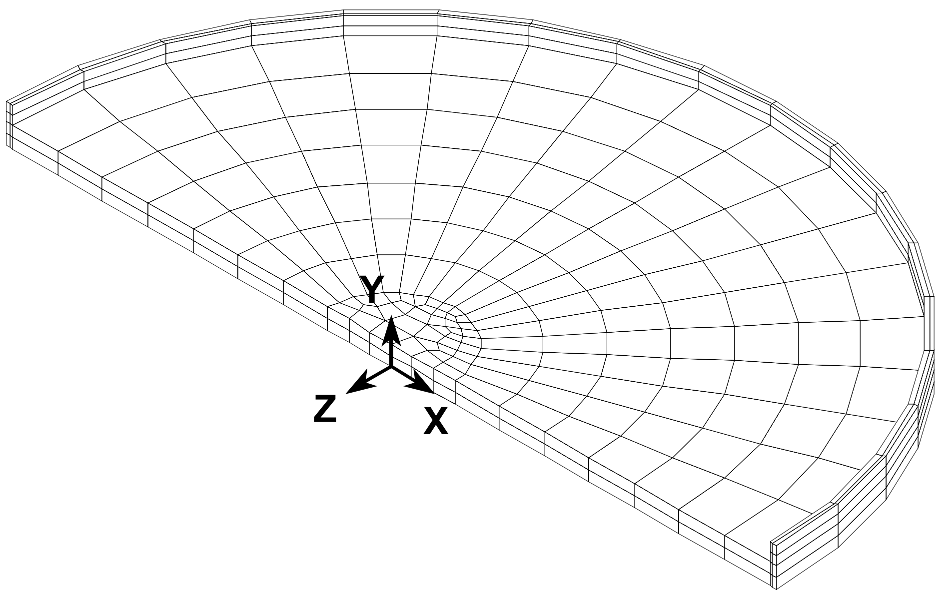
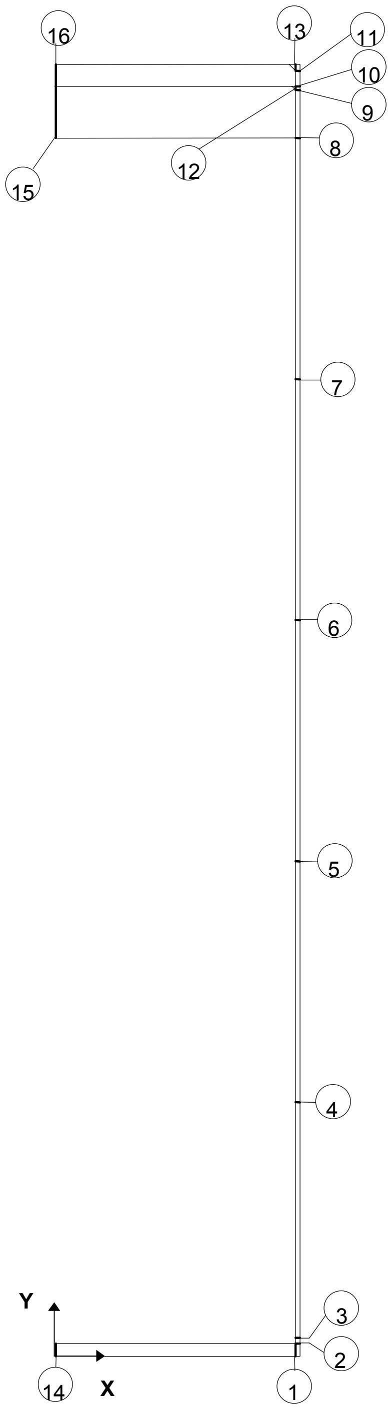
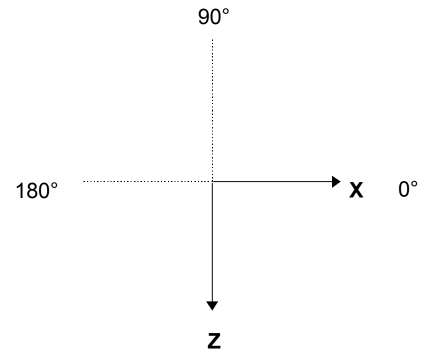


Figure 3.4.4.1-4 Locations for Section Stresses in the Canister Composite Finite Element Model





Top View of Axis

Section Coordinates at Z=0 and X>0

Axial Section	Node 1 X	Node 1 Y	Node 2 X	Node 2 Y
1	32.905	0.0	32.905	1.75
2	32.905	1.75	33.53	1.75
3	32.905	2.50	33.53	2.50
4	32.905	34.45	33.53	34.45
5	32.905	67.15	33.53	67.15
6	32.905	99.85	33.53	99.85
7	32.905	132.55	33.53	132.55
8	32.905	165.25	33.53	165.25
9	32.905	171.75	33.53	171.75
10	32.905	172.25	33.53	172.25
11	32.905	174.37	33.53	174.37
12	32.905	171.75	32.905	172.25
13	32.905	174.37	32.905	175.25
14	0.1	0.0	0.1	1.75
15	0.1	165.25	0.1	172.25
16	0.1	172.25	0.1	175.25

Figure 3.4.4.1-5 BWR Fuel Assembly Basket Showing Typical Fuel Basket Components

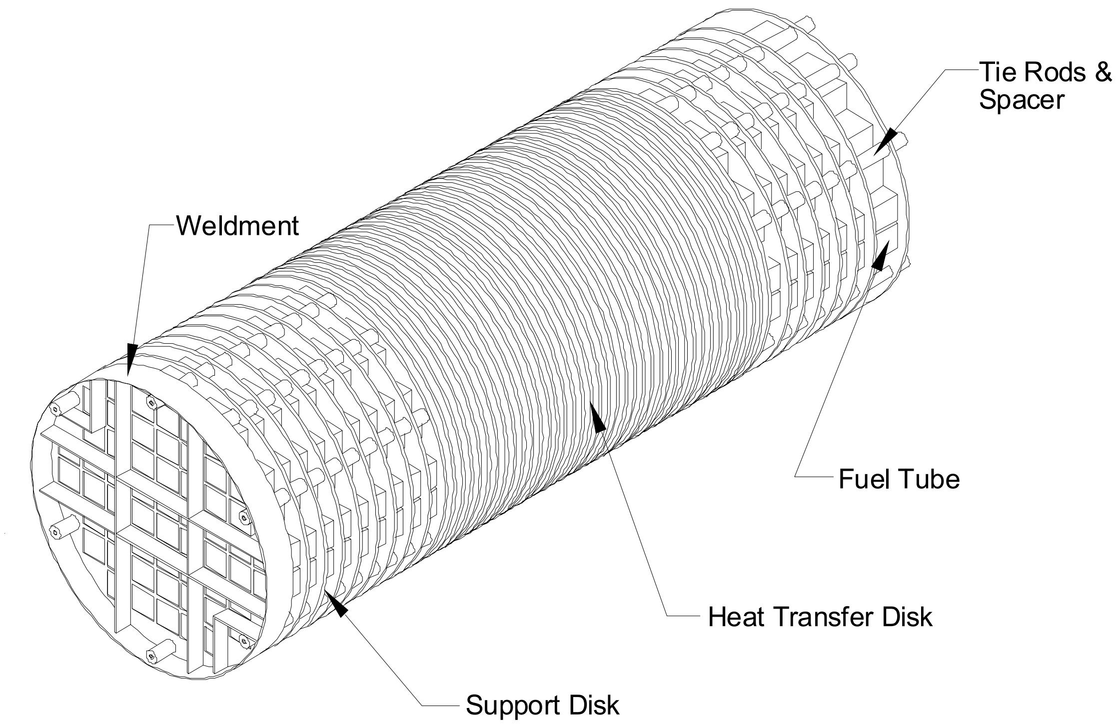


Figure 3.4.4.1-6 PWR Fuel Basket Support Disk Finite Element Model

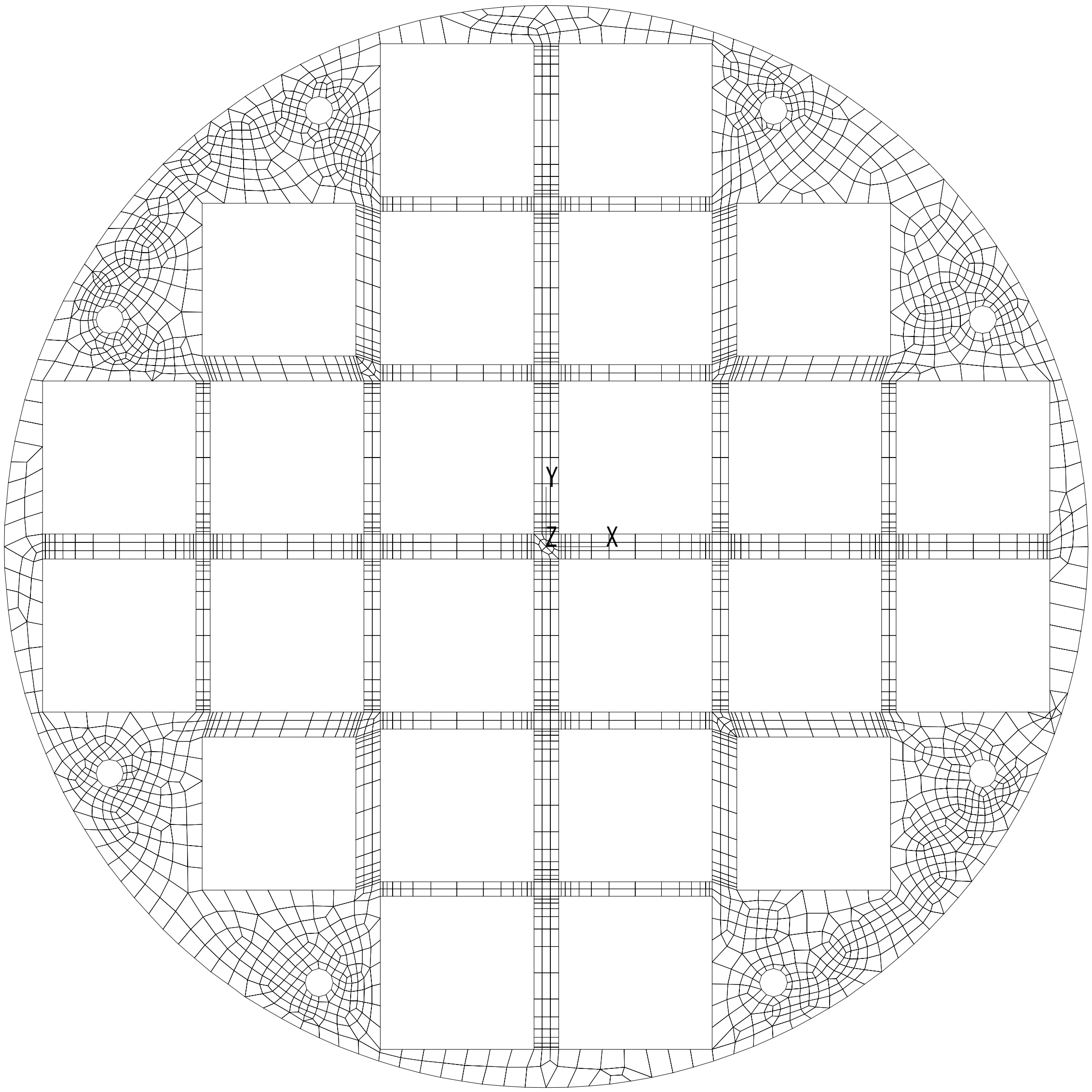


Figure 3.4.4.1-7 PWR Fuel Basket Support Disk Sections for Stress Evaluation (Left-Half)

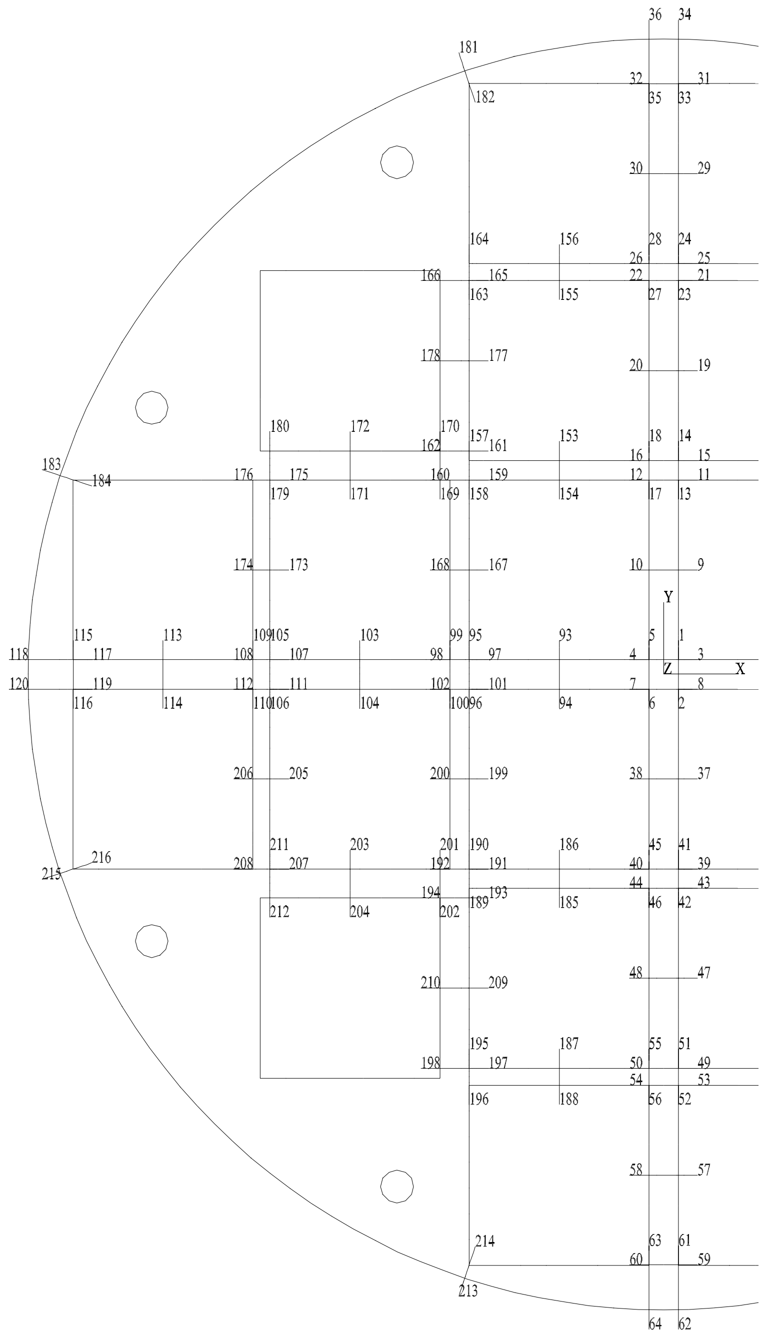


Figure 3.4.4.1-8 PWR Fuel Basket Support Disk Sections for Stress Evaluation (Right-Half)

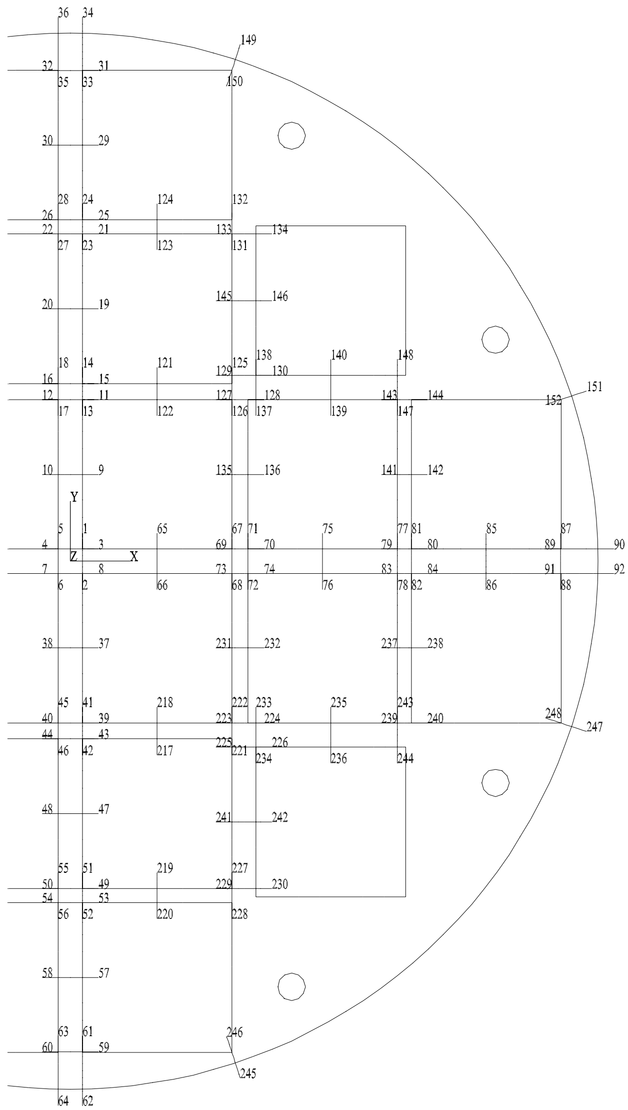


Figure 3.4.4.1-9 PWR Class 3 Fuel Tube Configuration

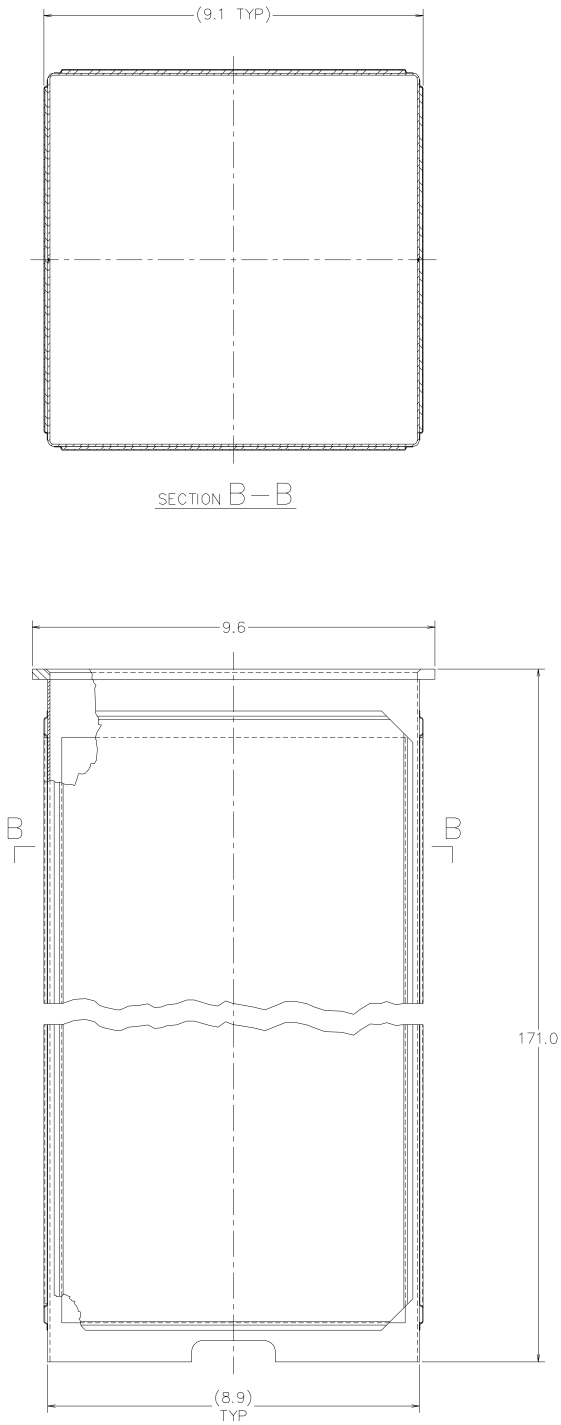


Figure 3.4.4.1-10 PWR Top Weldment Plate Finite Element Model

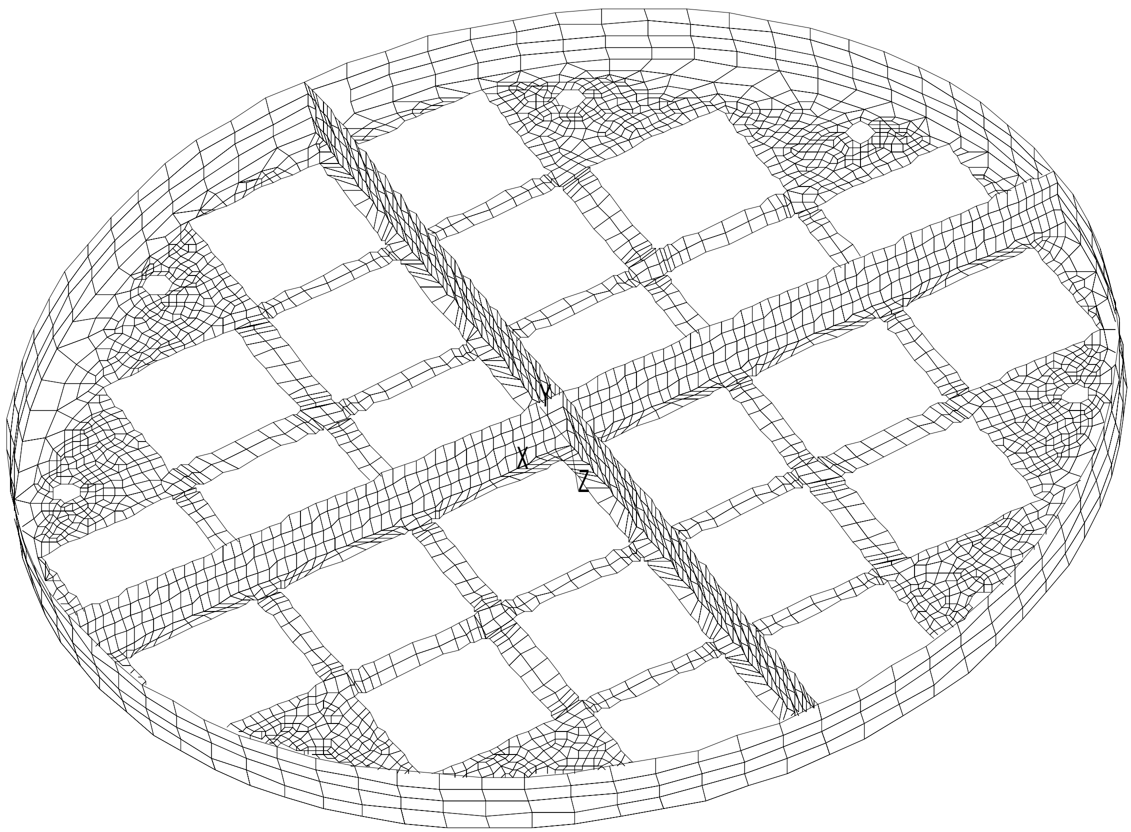


Figure 3.4.4.1-11 PWR Bottom Weldment Plate Finite Element Model

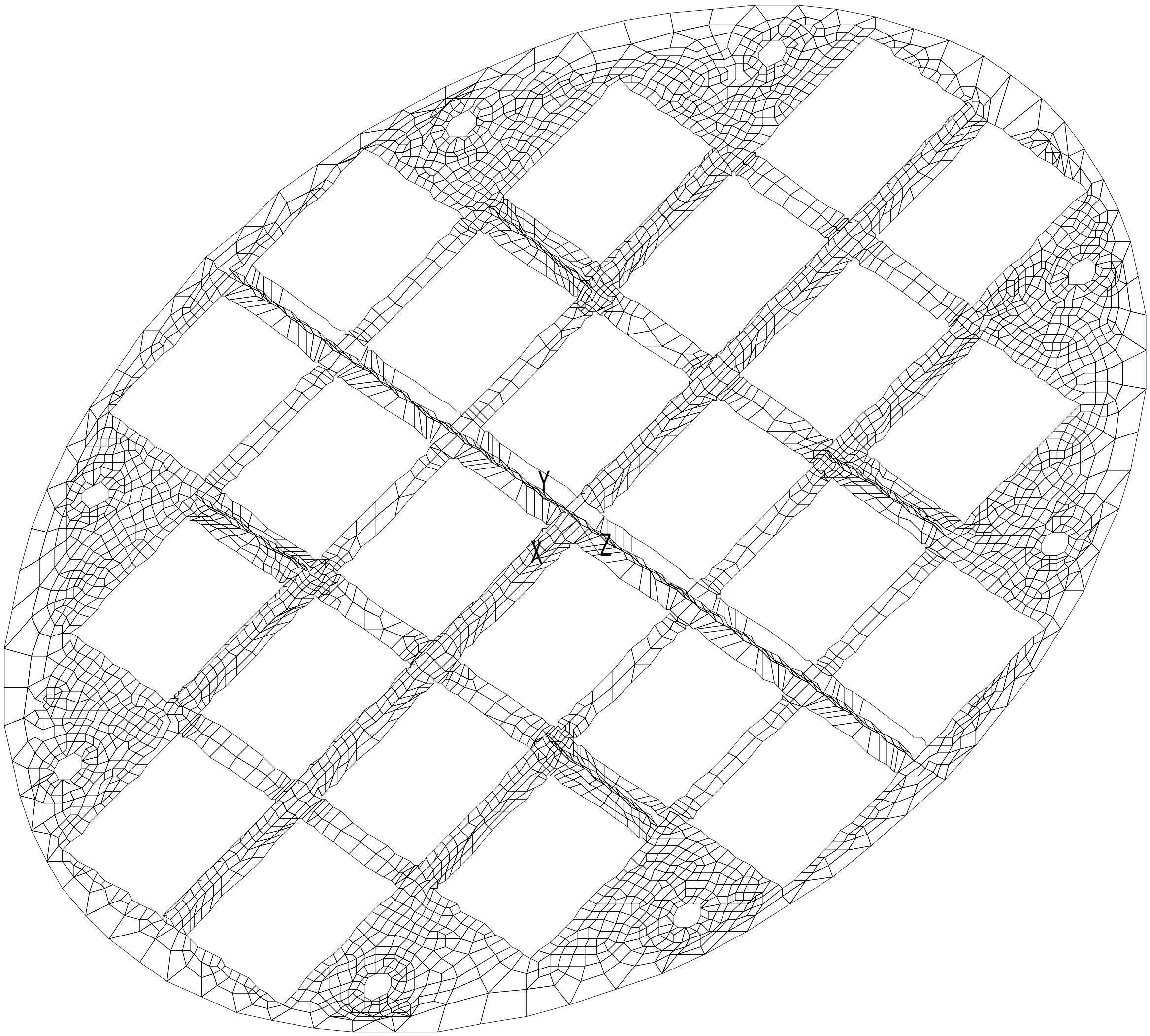


Figure 3.4.4.1-12 BWR Fuel Basket Support Disk Finite Element Model

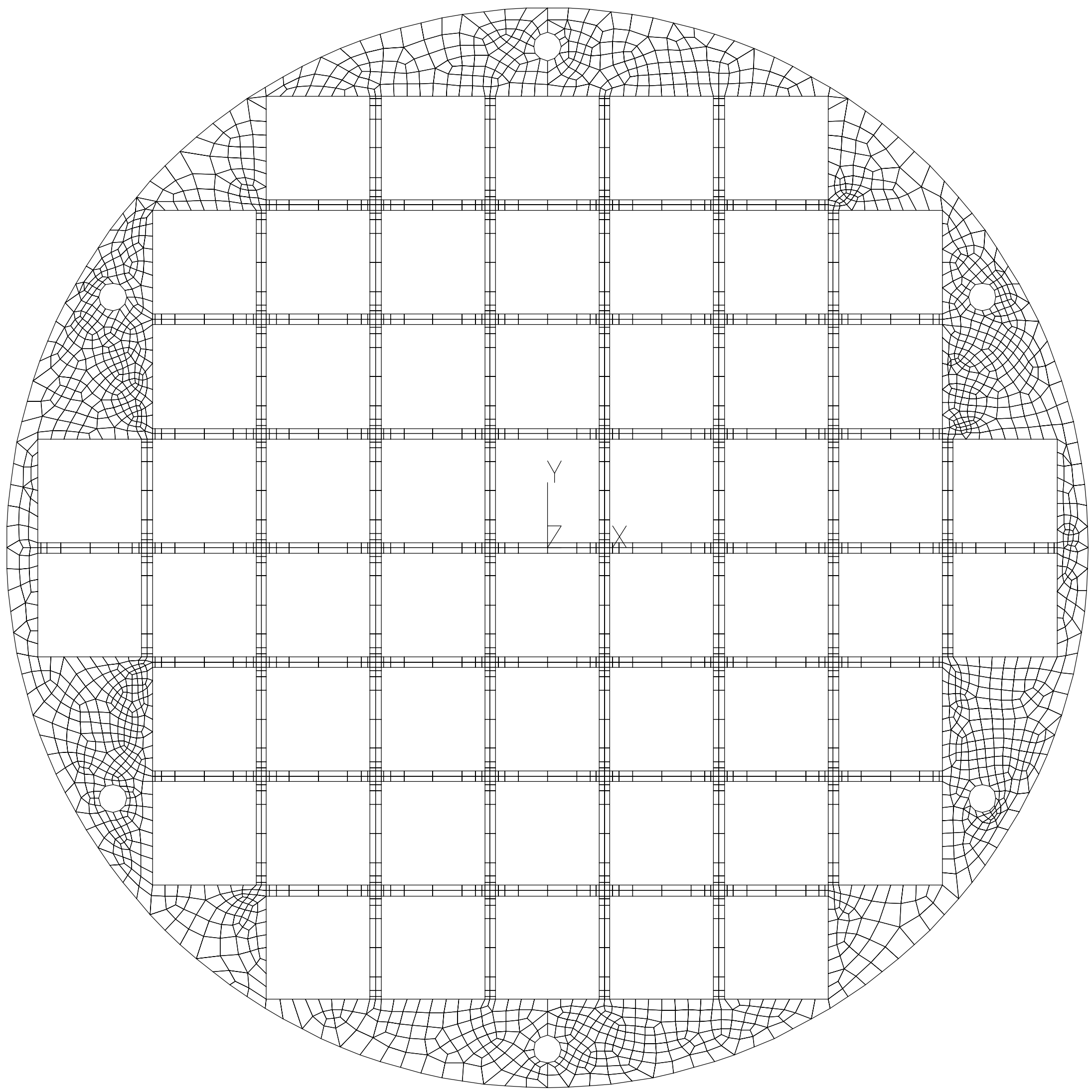


Figure 3.4.4.1-13 BWR Fuel Basket Support Disk Sections for Stress Evaluation (Quadrant I)

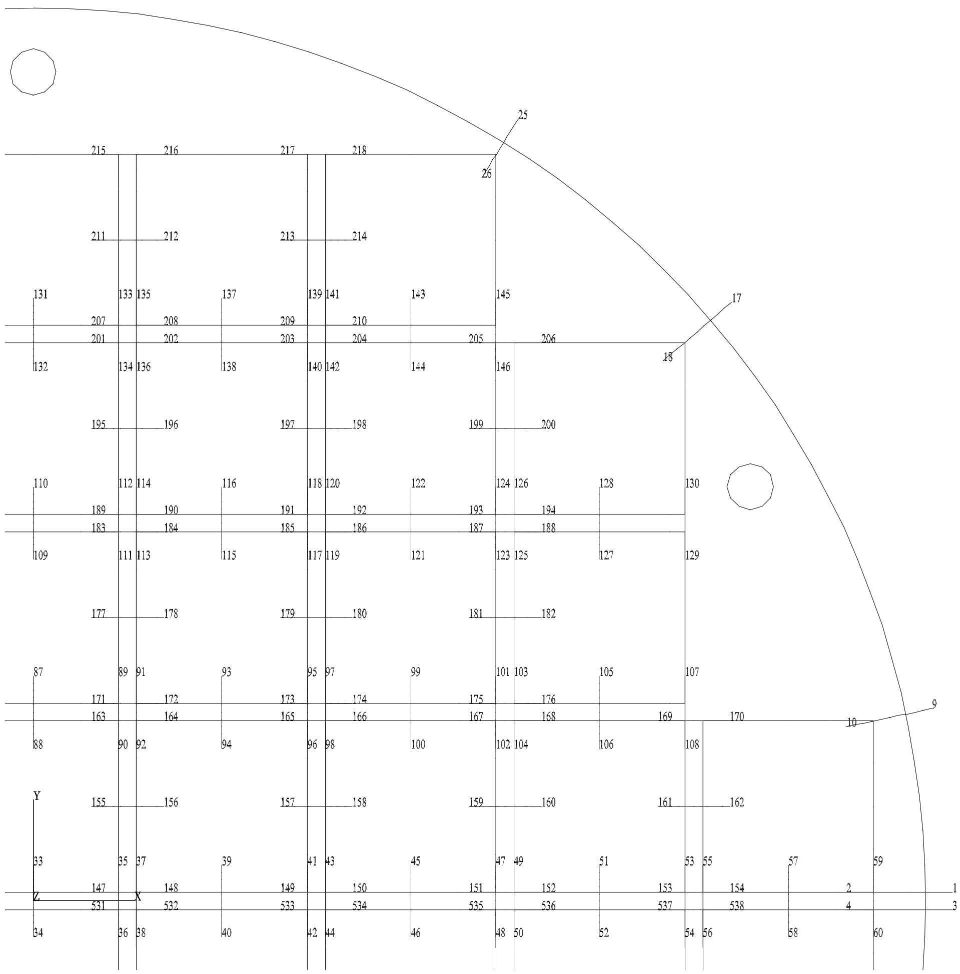


Figure 3.4.4.1-14 BWR Fuel Basket Support Disk Sections for Stress Evaluation (Quadrant II)

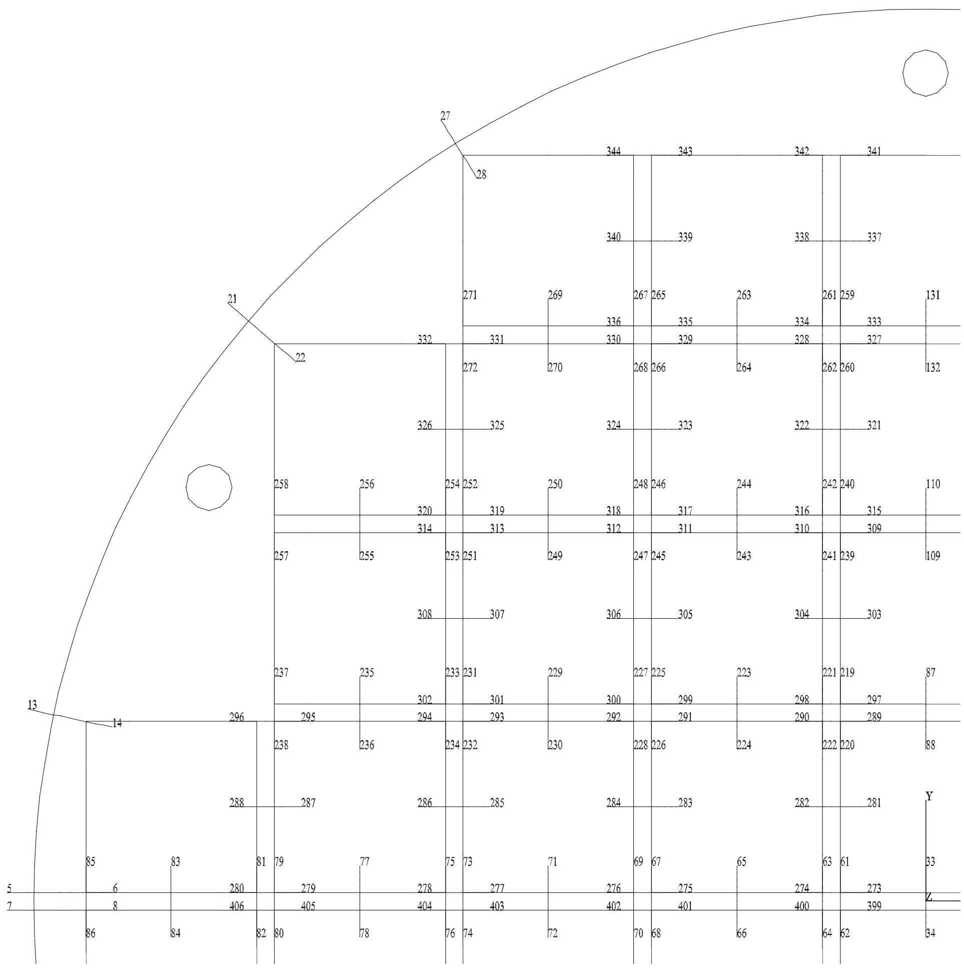


Figure 3.4.4.1-15 BWR Fuel Basket Support Disk Sections for Stress Evaluation (Quadrant III)

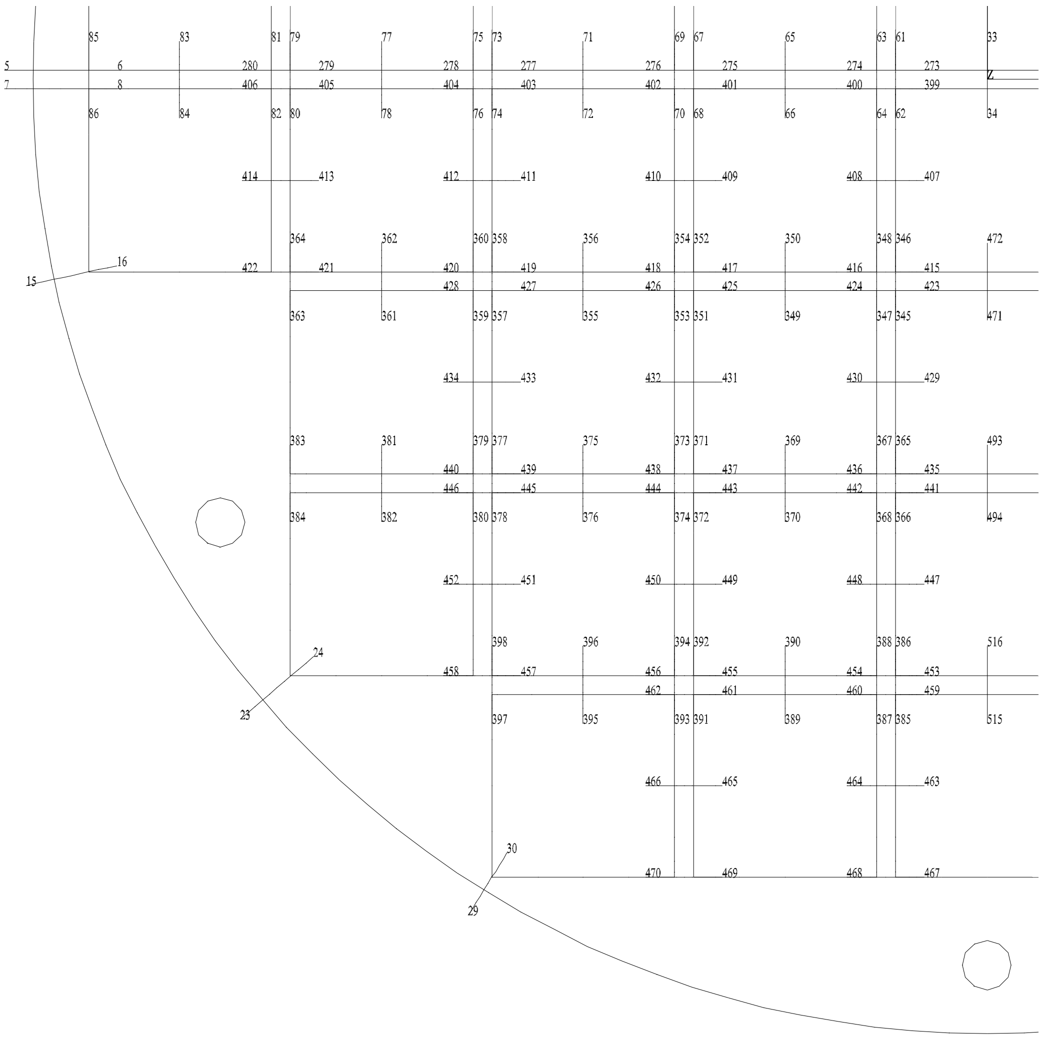


Figure 3.4.4.1-16 BWR Fuel Basket Support Disk Sections for Stress Evaluation (Quadrant IV)

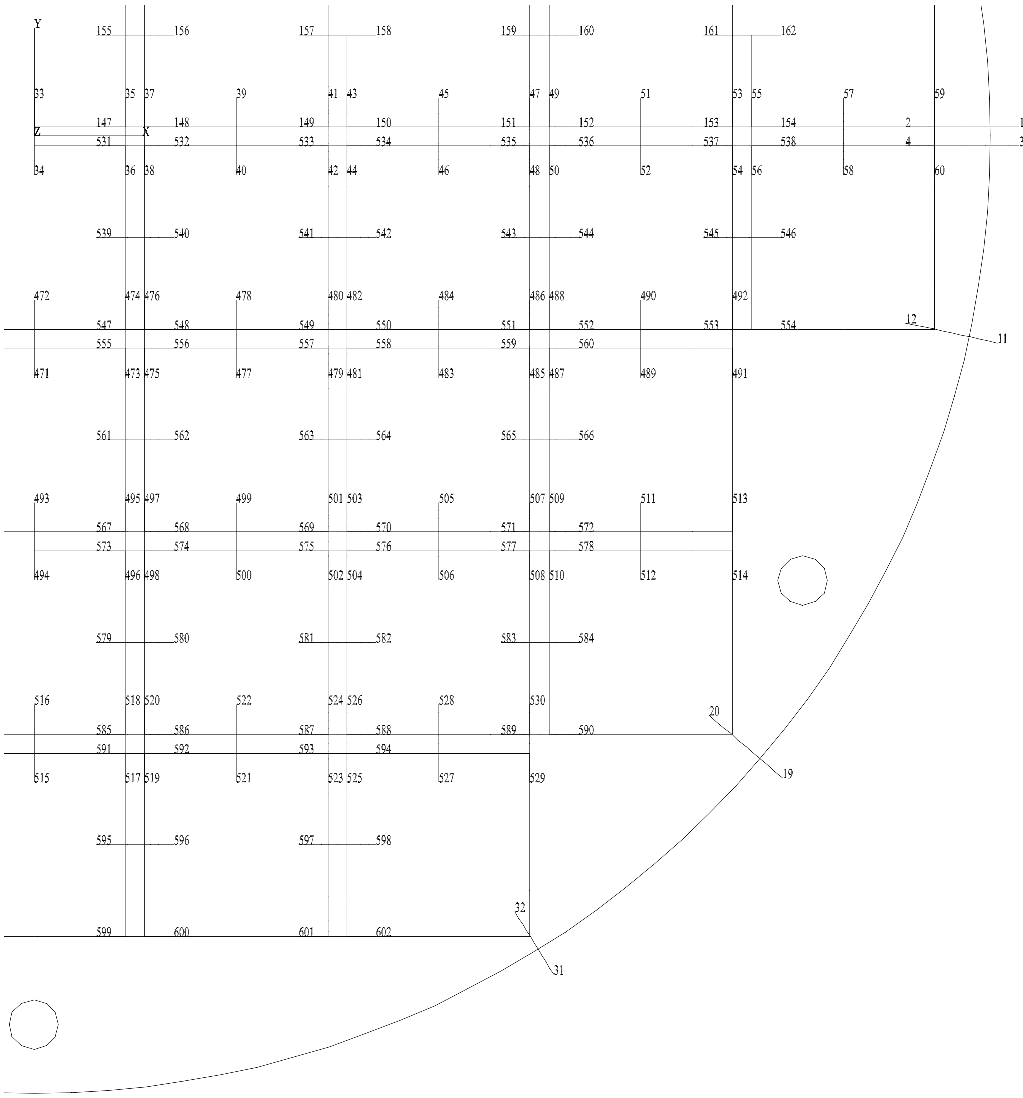


Figure 3.4.4.1-17 BWR Class 5 Fuel Tube Configuration

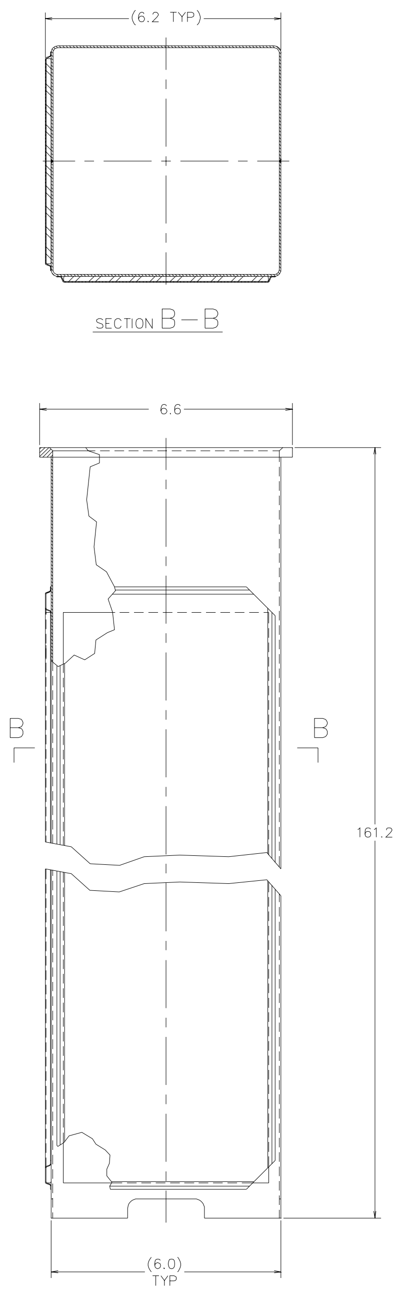


Figure 3.4.4.1-18 BWR Top Weldment Plate Finite Element Model

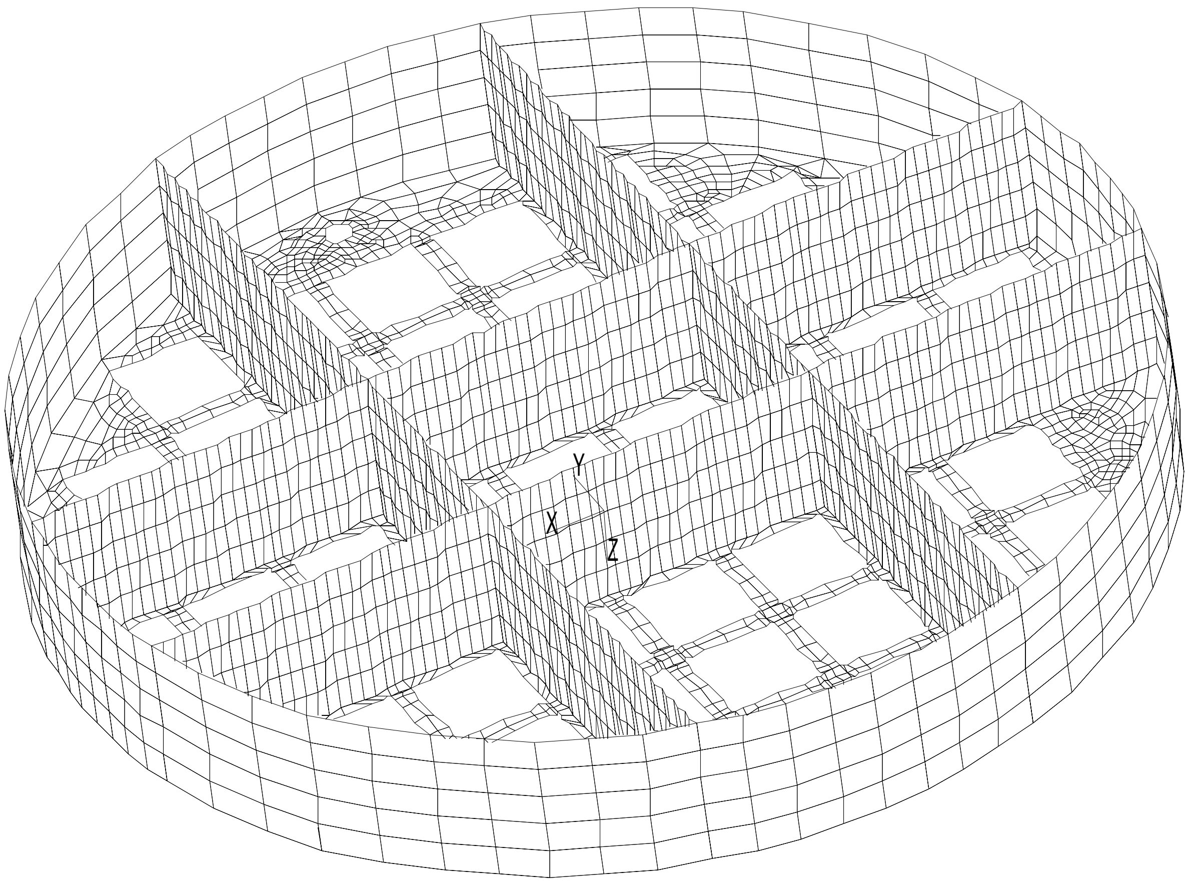
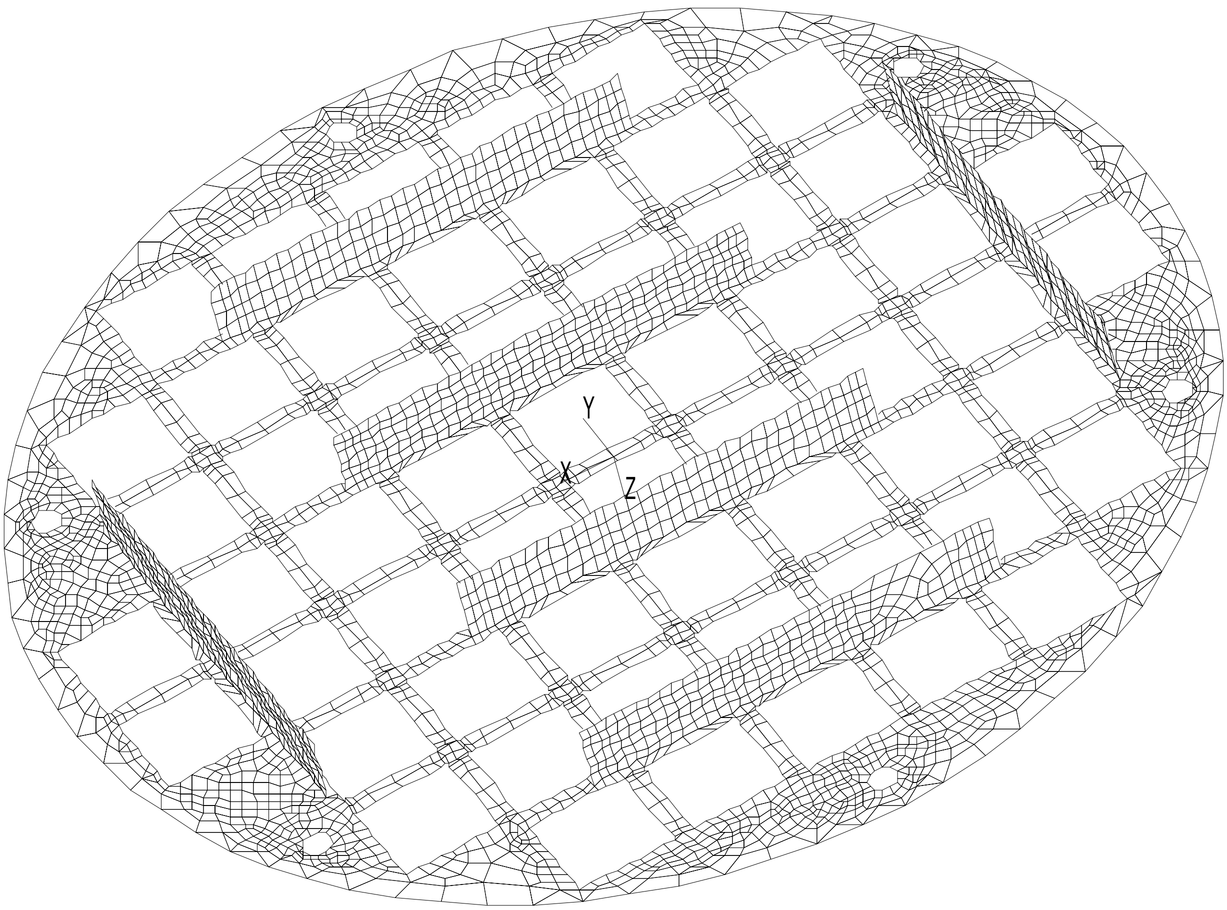


Figure 3.4.4.1-19 BWR Bottom Weldment Plate Finite Element Model



(Figure Inverted to Show Weldment Stiffeners)

Table 3.4.4.1-1 Canister Secondary (Thermal) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	-0.29	1.18	0.05	-0.13	-0.03	-0.10	1.52
2	0.16	0.48	-2.23	-0.03	-0.03	-0.18	2.72
3	-0.27	1.43	3.09	-0.14	0.02	0.07	3.37
4	0.00	0.00	-0.02	0.00	0.01	0.00	0.03
5	0.00	-0.05	0.09	0.00	-0.01	-0.01	0.14
6	0.00	-0.06	0.19	0.00	0.01	0.01	0.24
7	0.00	0.00	0.01	0.00	-0.01	0.00	0.03
8	0.00	-0.01	0.08	0.00	-0.01	0.00	0.10
9	3.58	1.49	1.59	0.03	0.15	1.31	3.31
10	-6.18	-2.32	-0.84	-0.22	-0.03	-0.87	5.63
11	1.80	-1.80	-8.02	-0.27	-0.09	0.74	9.96
12	-6.18	-2.32	-0.84	-0.22	-0.03	-0.87	5.63
13	-4.26	-0.79	1.43	0.27	-0.06	0.53	5.82
14	-23.43	-22.06	-14.19	0.72	1.42	-0.10	9.85
15	-7.92	-7.44	-6.62	0.20	0.49	0.00	1.64
16	0.28	0.29	-0.08	0.00	0.00	0.00	0.37

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-2 Canister Dead Weight Primary Membrane (P_m) Stresses (ksi), $P_{internal}$ = 0 psig

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0.00	-0.01	-0.05	0.00	0.00	-0.01	0.05
2	0.01	-0.02	-0.11	0.00	0.00	-0.01	0.12
3	0.00	-0.03	-0.13	0.00	0.00	0.00	0.12
4	0.00	0.00	-0.12	0.00	0.00	0.00	0.12
5	0.00	0.00	-0.11	0.00	0.00	0.00	0.11
6	0.00	0.00	-0.10	0.00	0.00	0.00	0.10
7	0.00	0.00	-0.09	0.00	0.00	0.00	0.09
8	0.00	0.01	-0.07	0.00	0.00	0.00	0.08
9	-0.01	-0.04	-0.04	0.00	0.00	-0.01	0.03
10	0.03	-0.02	-0.02	0.00	0.00	0.00	0.05
11	-0.03	-0.02	0.01	0.00	0.00	-0.01	0.04
12	0.01	-0.01	0.03	0.00	0.00	0.01	0.04
13	0.01	-0.02	-0.03	0.00	0.00	0.00	0.04
14	0.00	0.00	-0.02	0.00	0.00	0.00	0.02
15	0.00	0.00	0.00	0.00	0.00	0.00	0.01
16	0.00	0.00	0.00	0.00	0.00	0.00	0.00

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-3 Canister Dead Weight Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi), $P_{internal}$ = 0 psig

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0.01	-0.01	-0.06	0.00	0.00	-0.01	0.07
2	0.01	-0.03	-0.14	0.00	0.00	0.00	0.15
3	0.00	-0.03	-0.13	0.00	0.00	0.00	0.13
4	0.00	0.00	-0.12	0.00	0.00	0.00	0.12
5	0.00	0.00	-0.11	0.00	0.00	0.00	0.11
6	0.00	0.00	-0.10	0.00	0.00	0.00	0.10
7	0.00	0.00	-0.09	0.00	0.00	0.00	0.09
8	0.00	0.00	-0.09	0.00	0.00	0.00	0.09
9	-0.01	-0.05	-0.08	0.00	0.00	-0.01	0.08
10	0.02	-0.05	-0.10	0.00	0.00	-0.01	0.11
11	-0.02	0.01	0.08	0.00	0.00	-0.01	0.11
12	0.05	0.01	0.05	0.00	0.00	0.02	0.06
13	0.05	0.00	-0.01	0.00	0.00	-0.01	0.07
14	0.00	0.00	-0.02	0.00	0.00	0.00	0.02
15	0.07	0.07	0.00	0.00	0.00	0.00	0.07
16	-0.03	-0.03	0.00	0.00	0.00	0.00	0.03

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-4 Canister Normal Handling With No Internal Pressure Primary Membrane (P_m) Stresses, (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0.12	0.70	1.80	-0.05	-0.01	-0.26	1.76
2	1.17	-1.69	-1.15	0.22	-0.02	-0.27	2.92
3	-0.20	-2.63	0.53	0.22	0.04	0.48	3.42
4	0.00	0.01	0.51	0.00	0.00	0.00	0.51
5	0.00	0.00	0.55	0.00	0.00	0.00	0.55
6	0.01	-0.01	0.62	0.00	-0.01	0.00	0.62
7	0.01	-0.01	0.73	0.00	-0.01	0.00	0.74
8	0.02	-0.03	1.11	0.00	-0.07	0.00	1.15
9	0.05	0.40	1.56	-0.03	-0.15	0.07	1.53
10	-0.29	0.36	1.93	-0.07	-0.21	0.09	2.26
11	-0.68	0.74	1.05	-0.11	-0.13	-0.58	2.10
12	-0.13	0.52	2.01	-0.10	-0.10	0.17	2.19
13	0.34	0.99	-0.40	-0.16	-0.03	-0.61	1.79
14	0.29	0.29	-0.01	0.00	0.14	-0.02	0.41
15	-0.01	-0.01	-0.03	0.00	0.00	0.00	0.02
16	0.00	0.01	-0.05	0.00	-0.01	-0.01	0.06

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-5 Canister Normal Handling With No Internal Pressure Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	1.32	-0.05	4.36	0.07	-0.02	-0.02	4.42
2	0.57	-3.98	-8.37	0.38	-0.04	-0.60	9.05
3	-0.85	0.53	11.91	-0.08	0.04	0.62	12.82
4	0.00	-0.05	0.50	-0.01	0.00	0.00	0.56
5	0.00	-0.14	0.51	0.01	-0.01	0.00	0.65
6	0.01	-0.19	0.56	0.02	-0.01	0.00	0.75
7	0.01	-0.21	0.66	0.02	-0.01	0.00	0.88
8	0.03	-0.16	1.06	0.01	-0.05	0.00	1.23
9	-0.09	0.34	1.69	0.00	-0.21	-0.02	1.81
10	-0.46	0.64	2.87	-0.13	-0.13	0.20	3.38
11	-1.00	0.69	1.11	-0.12	-0.20	-1.02	2.98
12	-0.50	0.57	2.66	0.00	0.00	0.18	3.19
13	1.55	1.54	-0.83	-0.25	0.07	-0.25	2.67
14	6.60	6.61	0.18	0.00	0.13	-0.03	6.43
15	0.10	0.11	-0.06	0.00	0.01	0.00	0.17
16	0.25	0.27	-0.06	-0.02	-0.01	0.00	0.34

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-6 Summary of Canister Normal Handling plus Normal Internal Pressure Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.22	1.27	3.26	-0.09	-0.03	-0.47	3.19	16.70	4.2
2	2.15	-2.90	-2.11	0.38	-0.04	-0.48	5.16	16.70	2.2
3	-0.38	-4.49	0.95	0.38	0.08	0.90	5.94	16.70	1.8
4	0.00	0.80	0.90	-0.07	0.00	0.00	0.91	16.15	16.7
5	0.00	0.78	0.94	-0.07	0.00	0.00	0.94	14.94	14.8
6	0.01	0.78	1.01	-0.07	-0.01	0.00	1.01	14.81	13.7
7	0.01	0.78	1.12	-0.07	-0.01	0.00	1.12	15.93	13.2
8	0.02	0.49	1.49	-0.05	-0.07	-0.01	1.48	16.70	10.3
9	0.02	0.52	1.81	-0.04	-0.15	0.05	1.81	16.70	8.2
10	-0.33	0.46	2.10	-0.08	-0.21	0.02	2.47	16.70	5.8
11	-0.42	1.00	0.97	-0.12	-0.12	-0.51	1.77	16.70	8.5
12	-0.18	0.57	2.04	-0.11	-0.10	0.09	2.25	16.70	6.4
13	0.26	1.36	-0.05	-0.21	0.00	-0.57	1.90	16.70	7.8
14	0.53	0.53	-0.01	0.00	0.25	-0.04	0.74	16.70	21.6
15	-0.05	-0.05	-0.01	0.00	0.00	0.00	0.04	16.70	371.4
16	0.04	0.04	0.00	0.00	-0.01	0.00	0.04	16.70	418.2

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level A is used for material allowable stresses.

Table 3.4.4.1-7 Summary of Canister Normal Handling, Plus Normal Pressure Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	2.44	0.07	7.90	0.12	-0.04	-0.04	7.83	25.05	2.2
2	1.04	-7.06	-15.17	0.67	-0.07	-1.08	16.41	25.05	0.5
3	-1.56	1.19	21.42	-0.17	0.08	1.14	23.10	25.05	0.1
4	0.00	0.87	0.96	-0.08	0.00	0.00	0.96	24.23	24.2
5	0.01	0.90	0.98	-0.08	0.00	0.00	0.98	22.41	21.9
6	0.01	0.95	1.07	-0.08	0.00	0.00	1.07	22.22	19.8
7	0.01	0.97	1.20	-0.09	-0.01	0.00	1.19	23.90	19.0
8	0.01	0.63	1.60	-0.06	-0.08	-0.01	1.60	25.05	14.7
9	-0.08	0.52	2.12	-0.02	-0.21	0.01	2.23	25.05	10.2
10	-0.48	0.72	2.93	-0.14	-0.13	0.10	3.44	25.05	6.3
11	-0.68	1.22	1.89	-0.15	-0.19	-1.00	3.29	25.05	6.6
12	-0.52	0.63	2.65	-0.14	-0.12	0.08	3.21	25.05	6.8
13	1.08	1.80	-0.72	-0.31	0.12	-0.19	2.67	25.05	8.4
14	11.68	11.70	0.33	0.00	0.22	-0.05	11.38	25.05	1.2
15	-0.25	-0.25	-0.02	0.00	0.00	0.00	0.24	25.05	103.2
16	0.82	0.81	0.02	0.01	-0.01	0.00	0.80	25.05	30.2

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level A is used for material allowable stresses.

Table 3.4.4.1-8 Summary of Maximum Canister Normal Handling, plus Normal Pressure, plus Secondary (P + Q) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	3.73	2.84	11.32	-0.02	-0.05	0.12	8.48	50.10	4.9
2	1.25	-6.84	-18.35	0.67	-0.11	-1.23	19.81	50.10	1.5
3	-1.82	2.83	25.12	-0.33	0.10	1.22	27.07	50.10	0.9
4	0.00	0.87	0.97	-0.08	-0.01	0.00	0.98	48.46	48.7
5	-0.01	0.88	1.01	0.08	-0.02	-0.01	1.03	44.83	42.7
6	0.00	0.55	1.14	-0.05	-0.02	0.01	1.14	44.44	38.0
7	0.01	0.98	1.21	-0.09	0.00	0.00	1.21	47.79	38.6
8	0.01	0.62	1.68	-0.06	-0.07	-0.01	1.67	50.10	29.0
9	1.12	1.23	3.64	-0.02	-0.07	1.29	3.61	50.10	12.9
10	-6.72	-1.69	1.79	-0.36	-0.15	-0.79	8.69	50.10	4.8
11	2.15	-2.10	-9.58	-0.31	-0.14	0.89	11.89	50.10	3.2
12	-6.72	-1.69	1.79	-0.36	-0.15	-0.79	8.69	50.10	4.8
13	-5.08	-0.78	1.71	0.34	-0.09	0.62	6.93	50.10	6.2
14	-13.21	-12.96	-0.16	0.20	-0.05	-0.02	13.16	50.10	2.8
15	-8.25	-7.78	-6.63	0.20	0.49	0.00	1.90	50.10	25.4
16	0.01	0.06	-0.52	0.02	-0.05	0.00	0.59	50.10	83.3

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level A is used for material allowable stresses.

Table 3.4.4.1-9 Canister Normal Internal Pressure Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0.12	0.70	1.80	-0.05	-0.01	-0.26	1.76
2	1.17	-1.69	-1.15	0.22	-0.02	-0.27	2.92
3	-0.20	-2.63	0.53	0.22	0.04	0.48	3.42
4	0.00	0.01	0.51	0.00	0.00	0.00	0.51
5	0.00	0.00	0.55	0.00	0.00	0.00	0.55
6	0.01	-0.01	0.62	0.00	-0.01	0.00	0.62
7	0.01	-0.01	0.73	0.00	-0.01	0.00	0.74
8	0.02	-0.03	1.11	0.00	-0.07	0.00	1.15
9	0.05	0.40	1.56	-0.03	-0.15	0.07	1.53
10	-0.29	0.36	1.93	-0.07	-0.21	0.09	2.26
11	-0.68	0.74	1.05	-0.11	-0.13	-0.58	2.10
12	-0.13	0.52	2.01	-0.10	-0.10	0.17	2.19
13	0.34	0.99	-0.40	-0.16	-0.03	-0.61	1.79
14	0.29	0.29	-0.01	0.00	0.14	-0.02	0.41
15	-0.01	-0.01	-0.03	0.00	0.00	0.00	0.02
16	0.00	0.01	-0.05	0.00	-0.01	-0.01	0.06

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-10 Canister Normal Internal Pressure Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	1.32	-0.05	4.36	0.07	-0.02	-0.02	4.42
2	0.57	-3.98	-8.37	0.38	-0.04	-0.60	9.05
3	-0.85	0.53	11.91	-0.08	0.04	0.62	12.82
4	0.00	-0.05	0.50	-0.01	0.00	0.00	0.56
5	0.00	-0.14	0.51	0.01	-0.01	0.00	0.65
6	0.01	-0.19	0.56	0.02	-0.01	0.00	0.75
7	0.01	-0.21	0.66	0.02	-0.01	0.00	0.88
8	0.03	-0.16	1.06	0.01	-0.05	0.00	1.23
9	-0.09	0.34	1.69	0.00	-0.21	-0.02	1.81
10	-0.46	0.64	2.87	-0.13	-0.13	0.20	3.38
11	-1.00	0.69	1.11	-0.12	-0.20	-1.02	2.98
12	-0.50	0.57	2.66	0.00	0.00	0.18	3.19
13	1.55	1.54	-0.83	-0.25	0.07	-0.25	2.67
14	6.60	6.61	0.18	0.00	0.13	-0.03	6.43
15	0.10	0.11	-0.06	0.00	0.01	0.00	0.17
16	0.25	0.27	-0.06	-0.02	-0.01	0.00	0.34

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 3.4.4.1-11 Listing of Sections for Stress Evaluation of PWR Support Disk

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
1	1	2	0.75	0.75	0.75	-0.75
2	3	4	0.75	0.75	-0.75	0.75
3	5	6	-0.75	0.75	-0.75	-0.75
4	7	8	-0.75	-0.75	0.75	-0.75
5	9	10	0.75	5.39	-0.75	5.39
6	11	12	0.75	10.02	-0.75	10.02
7	13	14	0.75	10.02	0.75	11.02
8	15	16	0.75	11.02	-0.75	11.02
9	17	18	-0.75	10.02	-0.75	11.02
10	19	20	0.75	15.66	-0.75	15.66
11	21	22	0.75	20.29	-0.75	20.29
12	23	24	0.75	20.29	0.75	21.17
13	25	26	0.75	21.17	-0.75	21.17
14	27	28	-0.75	20.29	-0.75	21.17
15	29	30	0.75	25.81	-0.75	25.81
16	31	32	0.75	30.44	-0.75	30.44
17	33	34	0.75	30.44	0.75	32.74
18	35	36	-0.75	30.44	-0.75	32.74
19	37	38	0.75	-5.39	-0.75	-5.39
20	39	40	0.75	-10.02	-0.75	-10.02
21	41	42	0.75	-10.02	0.75	-11.02
22	43	44	0.75	-11.02	-0.75	-11.02
23	45	46	-0.75	-10.02	-0.75	-11.02
24	47	48	0.75	-15.66	-0.75	-15.66
25	49	50	0.75	-20.29	-0.75	-20.29
26	51	52	0.75	-20.29	0.75	-21.17
27	53	54	0.75	-21.17	-0.75	-21.17
28	55	56	-0.75	-20.29	-0.75	-21.17
29	57	58	0.75	-25.81	-0.75	-25.81
30	59	60	0.75	-30.44	-0.75	-30.44
31	61	62	0.75	-30.44	0.75	-32.74
32	63	64	-0.75	-30.44	-0.75	-32.74
33	65	66	5.39	0.75	5.39	-0.75
34	67	68	10.02	0.75	10.02	-0.75
35	69	70	10.02	0.75	11.02	0.75
36	71	72	11.02	0.75	11.02	-0.75
37	73	74	10.02	-0.75	11.02	-0.75
38	75	76	15.66	0.75	15.66	-0.75
39	77	78	20.29	0.75	20.29	-0.75
40	79	80	20.29	0.75	21.17	0.75
41	81	82	21.17	0.75	21.17	-0.75
42	83	84	20.29	-0.75	21.17	-0.75
43	85	86	25.81	0.75	25.81	-0.75
44	87	88	30.44	0.75	30.44	-0.75
45	89	90	30.44	0.75	32.74	0.75

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.

Table 3.4.4.1-11 Listing of Sections for Stress Evaluation of PWR Support Disk (Continued)

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
46	91	92	30.44	-0.75	32.74	-0.75
47	93	94	-5.39	0.75	-5.39	-0.75
48	95	96	-10.02	0.75	-10.02	-0.75
49	97	98	-10.02	0.75	-11.02	0.75
50	99	100	-11.02	0.75	-11.02	-0.75
51	101	102	-10.02	-0.75	-11.02	-0.75
52	103	104	-15.66	0.75	-15.66	-0.75
53	105	106	-20.29	0.75	-20.29	-0.75
54	107	108	-20.29	0.75	-21.17	0.75
55	109	110	-21.17	0.75	-21.17	-0.75
56	111	112	-20.29	-0.75	-21.17	-0.75
57	113	114	-25.81	0.75	-25.81	-0.75
58	115	116	-30.44	0.75	-30.44	-0.75
59	117	118	-30.44	0.75	-32.74	0.75
60	119	120	-30.44	-0.75	-32.74	-0.75
61	121	122	5.39	11.02	5.39	10.02
62	123	124	5.39	20.29	5.39	21.17
63	125	126	10.02	11.02	10.02	10.02
64	127	128	10.02	10.02	11.02	10.02
65	129	130	10.02	11.52	11.52	11.52
66	131	132	10.02	20.29	10.02	21.17
67	133	134	10.02	20.29	11.52	20.29
68	135	136	10.02	5.39	11.02	5.39
69	137	138	11.52	10.02	11.52	11.52
70	139	140	16.16	10.02	16.16	11.52
71	141	142	20.29	5.39	21.17	5.39
72	143	144	20.29	10.02	21.17	10.02
73	145	146	10.02	16.16	11.52	16.16
74	147	148	20.29	10.02	20.29	11.52
75	149	150	10.24	31.11	10.02	30.44
76	151	152	31.11	10.24	30.44	10.02
77	153	154	-5.39	11.02	-5.39	10.02
78	155	156	-5.39	20.29	-5.39	21.17
79	157	158	-10.02	11.02	-10.02	10.02
80	159	160	-10.02	10.02	-11.02	10.02
81	161	162	-10.02	11.52	-11.52	11.52
82	163	164	-10.02	20.29	-10.02	21.17
83	165	166	-10.02	20.29	-11.52	20.29
84	167	168	-10.02	5.39	-11.02	5.39
85	169	170	-11.52	10.02	-11.52	11.52
86	171	172	-16.16	10.02	-16.16	11.52
87	173	174	-20.29	5.39	-21.17	5.39
88	175	176	-20.29	10.02	-21.17	10.02
89	177	178	-10.02	16.16	-11.52	16.16
90	179	180	-20.29	10.02	-20.29	11.52

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.

Table 3.4.4.1-11 Listing of Sections for Stress Evaluation of PWR Support Disk (Continued)

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
91	181	182	-10.24	31.11	-10.02	30.44
92	183	184	-31.11	10.24	-30.44	10.02
93	185	186	-5.39	-11.02	-5.39	-10.02
94	187	188	-5.39	-20.29	-5.39	-21.17
95	189	190	-10.02	-11.02	-10.02	-10.02
96	191	192	-10.02	-10.02	-11.02	-10.02
97	193	194	-10.02	-11.52	-11.52	-11.52
98	195	196	-10.02	-20.29	-10.02	-21.17
99	197	198	-10.02	-20.29	-11.52	-20.29
100	199	200	-10.02	-5.39	-11.02	-5.39
101	201	202	-11.52	-10.02	-11.52	-11.52
102	203	204	-16.16	-10.02	-16.16	-11.52
103	205	206	-20.29	-5.39	-21.17	-5.39
104	207	208	-20.29	-10.02	-21.17	-10.02
105	209	210	-10.02	-16.16	-11.52	-16.16
106	211	212	-20.29	-10.02	-20.29	-11.52
107	213	214	-10.24	-31.11	-10.02	-30.44
108	215	216	-31.11	-10.24	-30.44	-10.02
109	217	218	5.39	-11.02	5.39	-10.02
110	219	220	5.39	-20.29	5.39	-21.17
111	221	222	10.02	-11.02	10.02	-10.02
112	223	224	10.02	-10.02	11.02	-10.02
113	225	226	10.02	-11.52	11.52	-11.52
114	227	228	10.02	-20.29	10.02	-21.17
115	229	230	10.02	-20.29	11.52	-20.29
116	231	232	10.02	-5.39	11.02	-5.39
117	233	234	11.52	-10.02	11.52	-11.52
118	235	236	16.16	-10.02	16.16	-11.52
119	237	238	20.29	-5.39	21.17	-5.39
120	239	240	20.29	-10.02	21.17	-10.02
121	241	242	10.02	-16.16	11.52	-16.16
122	243	244	20.29	-10.02	20.29	-11.52
123	245	246	10.24	-31.11	10.02	-30.44
124	247	248	31.11	-10.24	30.44	-10.02

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.

Table 3.4.4.1-12 $P_m + P_b$ Stresses for PWR Support Disk - Normal Conditions (ksi)

Section[1]	**Sx**	**Sy**	**Sxy**	**Stress Intensity**	**Allow. Stress[2]**	**Margin of Safety**
66	0.7	0.3	0.3	0.8	52.7	64.8
72	0.3	0.7	0.3	0.8	52.7	64.8
120	0.3	0.7	-0.3	0.8	52.7	64.8
82	0.7	0.3	-0.3	0.8	52.7	64.8
12	-0.4	0.2	0.0	0.6	52.7	86.8
28	-0.4	0.2	0.0	0.6	52.7	86.8
26	-0.4	0.2	0.0	0.6	52.7	86.8
54	0.2	-0.4	0.0	0.6	52.7	86.8
14	-0.4	0.2	0.0	0.6	52.7	86.8
42	0.2	-0.4	0.0	0.6	52.7	86.8
40	0.2	-0.4	0.0	0.6	52.7	86.8
56	0.2	-0.4	0.0	0.6	52.7	86.8
90	0.4	0.1	-0.2	0.5	52.7	104.3
67	0.1	0.4	0.2	0.5	52.7	104.3
99	0.1	0.4	0.2	0.5	52.7	104.3
106	0.4	0.1	0.2	0.5	52.7	104.3
122	0.4	0.1	-0.2	0.5	52.7	104.3
74	0.4	0.1	0.2	0.5	52.7	104.3
83	0.1	0.4	-0.2	0.5	52.7	104.3
115	0.1	0.4	-0.2	0.5	52.7	104.3
88	0.2	0.2	-0.3	0.5	52.7	104.3
114	0.2	0.2	-0.3	0.5	52.7	104.3
104	0.2	0.2	0.2	0.5	52.7	104.3
98	0.2	0.2	0.2	0.5	52.7	104.3
4	-0.2	-0.4	-0.1	0.4	52.7	130.6
2	-0.2	-0.4	-0.1	0.4	52.7	130.6
3	-0.4	-0.2	-0.1	0.4	52.7	130.6
1	-0.4	-0.2	-0.1	0.4	52.7	130.6
37	-0.1	-0.4	0.1	0.4	52.7	130.6
35	-0.1	-0.4	-0.1	0.4	52.7	130.6
7	-0.4	-0.1	-0.1	0.4	52.7	130.6
49	-0.1	-0.4	0.1	0.4	52.7	130.6
51	-0.1	-0.4	-0.1	0.4	52.7	130.6
23	-0.4	-0.1	-0.1	0.4	52.7	130.6
21	-0.4	-0.1	0.1	0.4	52.7	130.6
9	-0.4	-0.1	0.1	0.4	52.7	130.6
11	-0.2	0.2	-0.1	0.4	52.7	130.6
25	-0.2	0.2	-0.1	0.4	52.7	130.6
53	0.2	-0.2	0.1	0.4	52.7	130.6
39	0.2	-0.2	0.1	0.4	52.7	130.6

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.
2. Stress allowables are taken at 800°F.

Table 3.4.4.1-13 $P_m + P_b + Q$ Stresses for the PWR Support Disk - Normal Conditions (ksi)

Section[1]	Sx	Sy	Sxy	Stress Intensity	Allow. Stress[2]	Margin of Safety
44	-6.9	-29.3	6.1	30.8	105.3	2.42
58	-6.9	-29.3	6.1	30.8	105.3	2.42
75	23.5	2.2	-4.3	24.3	105.3	3.33
107	23.5	2.2	-4.2	24.3	105.3	3.33
108	2.1	23.3	-4.2	24.1	105.3	3.37
76	2.1	23.2	-4.1	24.0	105.3	3.39
123	20.6	2.0	5.4	22.1	105.3	3.76
124	1.9	20.6	5.4	22.1	105.3	3.76
92	1.8	20.6	5.3	22.0	105.3	3.79
91	20.5	1.9	5.4	22.0	105.3	3.79
7	-20.1	-6.7	-2.3	20.5	105.3	4.14
23	-20.1	-6.7	-2.3	20.5	105.3	4.14
49	-6.6	-20.0	2.3	20.4	105.3	4.16
37	-6.6	-20.0	2.3	20.4	105.3	4.16
9	-20.0	-6.7	2.3	20.4	105.3	4.16
21	-20.0	-6.7	2.3	20.4	105.3	4.16
35	-6.7	-20.0	-2.3	20.4	105.3	4.16
51	-6.7	-20.0	-2.3	20.4	105.3	4.16
17	20.6	-0.4	-1.2	21.1	105.3	3.99
32	20.6	-0.4	-1.2	21.1	105.3	3.99
45	-0.5	19.9	-1.4	20.7	105.3	4.09
60	-0.5	19.9	-1.4	20.7	105.3	4.09
80	-7.7	-19.5	2.4	19.9	105.3	4.29
112	-7.7	-19.5	2.4	19.9	105.3	4.29
31	19.6	-0.4	1.6	20.3	105.3	4.19
18	19.6	-0.4	1.6	20.3	105.3	4.19
79	-19.4	-7.6	2.3	19.9	105.3	4.29
111	-19.4	-7.6	2.3	19.9	105.3	4.29
95	-19.0	-7.7	-2.2	19.4	105.3	4.43
63	-19.0	-7.7	-2.2	19.4	105.3	4.43
96	-7.7	-18.8	-2.2	19.3	105.3	4.46
64	-7.7	-18.8	-2.2	19.3	105.3	4.46
59	-2.0	16.6	0.4	18.6	105.3	4.66
46	-2.0	16.6	0.4	18.6	105.3	4.66
30	-10.5	-11.3	4.5	15.3	105.3	5.88
16	-10.5	-11.3	4.5	15.3	105.3	5.88
6	-11.1	-9.3	-4.1	14.4	105.3	6.31
20	-11.1	-9.3	-4.1	14.4	105.3	6.31
48	-9.3	-11.0	-4.1	14.3	105.3	6.36
34	-9.3	-11.0	-4.1	14.3	105.3	6.36

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.
2. Stress allowables are taken at 800°F.

Table 3.4.4.1-14 Listing of Sections for Stress Evaluation of BWR Support Disk

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
1	1	2	32.74	0.33	30.85	0.33
2	3	4	32.74	-0.33	30.85	-0.33
3	5	6	-32.74	0.33	-30.85	0.33
4	7	8	-32.74	-0.33	-30.85	-0.33
5	9	10	32.03	6.85	30.85	6.6
6	11	12	32.03	-6.85	30.85	-6.6
7	13	14	-32.03	6.85	-30.85	6.6
8	15	16	-32.03	-6.85	-30.85	-6.6
9	17	18	24.87	21.30	23.89	20.46
10	19	20	24.87	-21.30	23.89	-20.46
11	21	22	-24.87	21.30	-23.89	20.46
12	23	24	-24.87	-21.30	-23.89	-20.46
13	25	26	17.27	27.83	17.00	27.39
14	27	28	-17.27	27.83	-17.00	27.39
15	29	30	-17.27	-27.83	-17.00	-27.39
16	31	32	17.27	-27.83	17.00	-27.39
17	33	34	0	0.33	0	-0.33
18	35	36	3.14	0.33	3.14	-0.33
19	37	38	3.79	0.33	3.79	-0.33
20	39	40	6.93	0.33	6.93	-0.33
21	41	42	10.07	0.33	10.07	-0.33
22	43	44	10.72	0.33	10.72	-0.33
23	45	46	13.86	0.33	13.86	-0.33
24	47	48	17	0.33	17	-0.33
25	49	50	17.65	0.33	17.65	-0.33
26	51	52	20.78	0.33	20.78	-0.33
27	53	54	23.92	0.33	23.92	-0.33
28	55	56	24.57	0.33	24.57	-0.33
29	57	58	27.71	0.33	27.71	-0.33
30	59	60	30.85	0.33	30.85	-0.33
31	61	62	-3.14	0.33	-3.14	-0.33
32	63	64	-3.79	0.33	-3.79	-0.33
33	65	66	-6.93	0.33	-6.93	-0.33
34	67	68	-10.07	0.33	-10.07	-0.33
35	69	70	-10.72	0.33	-10.72	-0.33
36	71	72	-13.86	0.33	-13.86	-0.33
37	73	74	-17	0.33	-17	-0.33
38	75	76	-17.65	0.33	-17.65	-0.33
39	77	78	-20.78	0.33	-20.78	-0.33
40	79	80	-23.92	0.33	-23.92	-0.33
41	81	82	-24.57	0.33	-24.57	-0.33
42	83	84	-27.71	0.33	-27.71	-0.33
43	85	86	-30.85	0.33	-30.85	-0.33
44	87	88	0	7.25	0	6.6
45	89	90	3.14	7.25	3.14	6.6
46	91	92	3.79	7.25	3.79	6.6
47	93	94	6.93	7.25	6.93	6.6
48	95	96	10.07	7.25	10.07	6.6
49	97	98	10.72	7.25	10.72	6.6
50	99	100	13.86	7.25	13.86	6.6

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.

Table 3.4.4.1-14 Listing of Sections for Stress Evaluation of BWR Support Disk (Continued)

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
51	101	102	17	7.25	17	6.6
52	103	104	17.65	7.25	17.65	6.6
53	105	106	20.78	7.25	20.78	6.6
54	107	108	23.92	7.25	23.92	6.6
55	109	110	0	13.53	0	14.18
56	111	112	3.14	13.53	3.14	14.18
57	113	114	3.79	13.53	3.79	14.18
58	115	116	6.93	13.53	6.93	14.18
59	117	118	10.07	13.53	10.07	14.18
60	119	120	10.72	13.53	10.72	14.18
61	121	122	13.86	13.53	13.86	14.18
62	123	124	17	13.53	17	14.18
63	125	126	17.65	13.53	17.65	14.18
64	127	128	20.78	13.53	20.78	14.18
65	129	130	23.92	13.53	23.92	14.18
66	131	132	0	21.11	0	20.46
67	133	134	3.14	21.11	3.14	20.46
68	135	136	3.79	21.11	3.79	20.46
69	137	138	6.93	21.11	6.93	20.46
70	139	140	10.07	21.11	10.07	20.46
71	141	142	10.72	21.11	10.72	20.46
72	143	144	13.86	21.11	13.86	20.46
73	145	146	17	21.11	17	20.46
74	147	148	3.14	0.33	3.79	0.33
75	149	150	10.07	0.33	10.72	0.33
76	151	152	17	0.33	17.65	0.33
77	153	154	23.92	0.33	24.57	0.33
78	155	156	3.14	3.46	3.79	3.46
79	157	158	10.07	3.46	10.72	3.46
80	159	160	17	3.46	17.65	3.46
81	161	162	23.92	3.46	24.57	3.46
82	163	164	3.14	6.6	3.79	6.6
83	165	166	10.07	6.6	10.72	6.6
84	167	168	17	6.6	17.65	6.6
85	169	170	23.92	6.6	24.57	6.6
86	171	172	3.14	7.25	3.79	7.25
87	173	174	10.07	7.25	10.72	7.25
88	175	176	17	7.25	17.65	7.25
89	177	178	3.14	10.39	3.79	10.39
90	179	180	10.07	10.39	10.72	10.39
91	181	182	17	10.39	17.65	10.39
92	183	184	3.14	13.53	3.79	13.53
93	185	186	10.07	13.53	10.72	13.53
94	187	188	17	13.53	17.65	13.53
95	189	190	3.14	14.18	3.79	14.18
96	191	192	10.07	14.18	10.72	14.18
97	193	194	17	14.18	17.65	14.18
98	195	196	3.14	17.32	3.79	17.32
99	197	198	10.07	17.32	10.72	17.32
100	199	200	17	17.32	17.65	17.32

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.

Table 3.4.4.1-14 Listing of Sections for Stress Evaluation of BWR Support Disk (Continued)

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
101	201	202	3.14	20.46	3.79	20.46
102	203	204	10.07	20.46	10.72	20.46
103	205	206	17	20.46	17.65	20.46
104	207	208	3.14	21.11	3.79	21.11
105	209	210	10.07	21.11	10.72	21.11
106	211	212	3.14	24.25	3.79	24.25
107	213	214	10.07	24.25	10.72	24.25
108	215	216	3.14	27.39	3.79	27.39
109	217	218	10.07	27.39	10.72	27.39
110	219	220	-3.14	7.25	-3.14	6.6
111	221	222	-3.79	7.25	-3.79	6.6
112	223	224	-6.93	7.25	-6.93	6.6
113	225	226	-10.07	7.25	-10.07	6.6
114	227	228	-10.72	7.25	-10.72	6.6
115	229	230	-13.86	7.25	-13.86	6.6
116	231	232	-17	7.25	-17	6.6
117	233	234	-17.65	7.25	-17.65	6.6
118	235	236	-20.78	7.25	-20.78	6.6
119	237	238	-23.92	7.25	-23.92	6.6
120	239	240	-3.14	13.53	-3.14	14.18
121	241	242	-3.79	13.53	-3.79	14.18
122	243	244	-6.93	13.53	-6.93	14.18
123	245	246	-10.07	13.53	-10.07	14.18
124	247	248	-10.72	13.53	-10.72	14.18
125	249	250	-13.86	13.53	-13.86	14.18
126	251	252	-17	13.53	-17	14.18
127	253	254	-17.65	13.53	-17.65	14.18
128	255	256	-20.78	13.53	-20.78	14.18
129	257	258	-23.92	13.53	-23.92	14.18
130	259	260	-3.14	21.11	-3.14	20.46
131	261	262	-3.79	21.11	-3.79	20.46
132	263	264	-6.93	21.11	-6.93	20.46
133	265	266	-10.07	21.11	-10.07	20.46
134	267	268	-10.72	21.11	-10.72	20.46
135	269	270	-13.86	21.11	-13.86	20.46
136	271	272	-17	21.11	-17	20.46
137	273	274	-3.14	0.33	-3.79	0.33
138	275	276	-10.07	0.33	-10.72	0.33
139	277	278	-17	0.33	-17.65	0.33
140	279	280	-23.92	0.33	-24.57	0.33
141	281	282	-3.14	3.46	-3.79	3.46
142	283	284	-10.07	3.46	-10.72	3.46
143	285	286	-17	3.46	-17.65	3.46
144	287	288	-23.92	3.46	-24.57	3.46
145	289	290	-3.14	6.6	-3.79	6.6
146	291	292	-10.07	6.6	-10.72	6.6
147	293	294	-17	6.6	-17.65	6.6
148	295	296	-23.92	6.6	-24.57	6.6
149	297	298	-3.14	7.25	-3.79	7.25
150	299	300	-10.07	7.25	-10.72	7.25

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.

Table 3.4.4.1-14 Listing of Sections for Stress Evaluation of BWR Support Disk (Continued)

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
151	301	302	-17	7.25	-17.65	7.25
152	303	304	-3.14	10.39	-3.79	10.39
153	305	306	-10.07	10.39	-10.72	10.39
154	307	308	-17	10.39	-17.65	10.39
155	309	310	-3.14	13.53	-3.79	13.53
156	311	312	-10.07	13.53	-10.72	13.53
157	313	314	-17	13.53	-17.65	13.53
158	315	316	-3.14	14.18	-3.79	14.18
159	317	318	-10.07	14.18	-10.72	14.18
160	319	320	-17	14.18	-17.65	14.18
161	321	322	-3.14	17.32	-3.79	17.32
162	323	324	-10.07	17.32	-10.72	17.32
163	325	326	-17	17.32	-17.65	17.32
164	327	328	-3.14	20.46	-3.79	20.46
165	329	330	-10.07	20.46	-10.72	20.46
166	331	332	-17	20.46	-17.65	20.46
167	333	334	-3.14	21.11	-3.79	21.11
168	335	336	-10.07	21.11	-10.72	21.11
169	337	338	-3.14	24.25	-3.79	24.25
170	339	340	-10.07	24.25	-10.72	24.25
171	341	342	-3.14	27.39	-3.79	27.39
172	343	344	-10.07	27.39	-10.72	27.39
173	345	346	-3.14	-7.25	-3.14	-6.6
174	347	348	-3.79	-7.25	-3.79	-6.6
175	349	350	-6.93	-7.25	-6.93	-6.6
176	351	352	-10.07	-7.25	-10.07	-6.6
177	353	354	-10.72	-7.25	-10.72	-6.6
178	355	356	-13.86	-7.25	-13.86	-6.6
179	357	358	-17	-7.25	-17	-6.6
180	359	360	-17.65	-7.25	-17.65	-6.6
181	361	362	-20.78	-7.25	-20.78	-6.6
182	363	364	-23.92	-7.25	-23.92	-6.6
183	365	366	-3.14	-13.53	-3.14	-14.18
184	367	368	-3.79	-13.53	-3.79	-14.18
185	369	370	-6.93	-13.53	-6.93	-14.18
186	371	372	-10.07	-13.53	-10.07	-14.18
187	373	374	-10.72	-13.53	-10.72	-14.18
188	375	376	-13.86	-13.53	-13.86	-14.18
189	377	378	-17	-13.53	-17	-14.18
190	379	380	-17.65	-13.53	-17.65	-14.18
191	381	382	-20.78	-13.53	-20.78	-14.18
192	383	384	-23.92	-13.53	-23.92	-14.18
193	385	386	-3.14	-21.11	-3.14	-20.46
194	387	388	-3.79	-21.11	-3.79	-20.46
195	389	390	-6.93	-21.11	-6.93	-20.46
196	391	392	-10.07	-21.11	-10.07	-20.46
197	393	394	-10.72	-21.11	-10.72	-20.46
198	395	396	-13.86	-21.11	-13.86	-20.46
199	397	398	-17	-21.11	-17	-20.46
200	399	400	-3.14	-0.33	-3.79	-0.33

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.

Table 3.4.4.1-14 Listing of Sections for Stress Evaluation of BWR Support Disk (Continued)

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
201	401	402	-10.07	-0.33	-10.72	-0.33
202	403	404	-17	-0.33	-17.65	-0.33
203	405	406	-23.92	-0.33	-24.57	-0.33
204	407	408	-3.14	-3.46	-3.79	-3.46
205	409	410	-10.07	-3.46	-10.72	-3.46
206	411	412	-17	-3.46	-17.65	-3.46
207	413	414	-23.92	-3.46	-24.57	-3.46
208	415	416	-3.14	-6.6	-3.79	-6.6
209	417	418	-10.07	-6.6	-10.72	-6.6
210	419	420	-17	-6.6	-17.65	-6.6
211	421	422	-23.92	-6.6	-24.57	-6.6
212	423	424	-3.14	-7.25	-3.79	-7.25
213	425	426	-10.07	-7.25	-10.72	-7.25
214	427	428	-17	-7.25	-17.65	-7.25
215	429	430	-3.14	-10.39	-3.79	-10.39
216	431	432	-10.07	-10.39	-10.72	-10.39
217	433	434	-17	-10.39	-17.65	-10.39
218	435	436	-3.14	-13.53	-3.79	-13.53
219	437	438	-10.07	-13.53	-10.72	-13.53
220	439	440	-17	-13.53	-17.65	-13.53
221	441	442	-3.14	-14.18	-3.79	-14.18
222	443	444	-10.07	-14.18	-10.72	-14.18
223	445	446	-17	-14.18	-17.65	-14.18
224	447	448	-3.14	-17.32	-3.79	-17.32
225	449	450	-10.07	-17.32	-10.72	-17.32
226	451	452	-17	-17.32	-17.65	-17.32
227	453	454	-3.14	-20.46	-3.79	-20.46
228	455	456	-10.07	-20.46	-10.72	-20.46
229	457	458	-17	-20.46	-17.65	-20.46
230	459	460	-3.14	-21.11	-3.79	-21.11
231	461	462	-10.07	-21.11	-10.72	-21.11
232	463	464	-3.14	-24.25	-3.79	-24.25
233	465	466	-10.07	-24.25	-10.72	-24.25
234	467	468	-3.14	-27.39	-3.79	-27.39
235	469	470	-10.07	-27.39	-10.72	-27.39
236	471	472	0	-7.25	0	-6.6
237	473	474	3.14	-7.25	3.14	-6.6
238	475	476	3.79	-7.25	3.79	-6.6
239	477	478	6.93	-7.25	6.93	-6.6
240	479	480	10.07	-7.25	10.07	-6.6
241	481	482	10.72	-7.25	10.72	-6.6
242	483	484	13.86	-7.25	13.86	-6.6
243	485	486	17	-7.25	17	-6.6
244	487	488	17.65	-7.25	17.65	-6.6
245	489	490	20.78	-7.25	20.78	-6.6
246	491	492	23.92	-7.25	23.92	-6.6
247	493	494	0	-13.53	0	-14.18
248	495	496	3.14	-13.53	3.14	-14.18
249	497	498	3.79	-13.53	3.79	-14.18
250	499	500	6.93	-13.53	6.93	-14.18

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.

Table 3.4.4.1-14 Listing of Sections for Stress Evaluation of BWR Support Disk (Continued)

Section Number[1]	Point 1	Point 2	Point 1		Point 2	
			X	Y	X	Y
251	501	502	10.07	-13.53	10.07	-14.18
252	503	504	10.72	-13.53	10.72	-14.18
253	505	506	13.86	-13.53	13.86	-14.18
254	507	508	17	-13.53	17	-14.18
255	509	510	17.65	-13.53	17.65	-14.18
256	511	512	20.78	-13.53	20.78	-14.18
257	513	514	23.92	-13.53	23.92	-14.18
258	515	516	0	-21.11	0	-20.46
259	517	518	3.14	-21.11	3.14	-20.46
260	519	520	3.79	-21.11	3.79	-20.46
261	521	522	6.93	-21.11	6.93	-20.46
262	523	524	10.07	-21.11	10.07	-20.46
263	525	526	10.72	-21.11	10.72	-20.46
264	527	528	13.86	-21.11	13.86	-20.46
265	529	530	17	-21.11	17	-20.46
266	531	532	3.14	-0.33	3.79	-0.33
267	533	534	10.07	-0.33	10.72	-0.33
268	535	536	17	-0.33	17.65	-0.33
269	537	538	23.92	-0.33	24.57	-0.33
270	539	540	3.14	-3.46	3.79	-3.46
271	541	542	10.07	-3.46	10.72	-3.46
272	543	544	17	-3.46	17.65	-3.46
273	545	546	23.92	-3.46	24.57	-3.46
274	547	548	3.14	-6.6	3.79	-6.6
275	549	550	10.07	-6.6	10.72	-6.6
276	551	552	17	-6.6	17.65	-6.6
277	553	554	23.92	-6.6	24.57	-6.6
278	555	556	3.14	-7.25	3.79	-7.25
279	557	558	10.07	-7.25	10.72	-7.25
280	559	560	17	-7.25	17.65	-7.25
281	561	562	3.14	-10.39	3.79	-10.39
282	563	564	10.07	-10.39	10.72	-10.39
283	565	566	17	-10.39	17.65	-10.39
284	567	568	3.14	-13.53	3.79	-13.53
285	569	570	10.07	-13.53	10.72	-13.53
286	571	572	17	-13.53	17.65	-13.53
287	573	574	3.14	-14.18	3.79	-14.18
288	575	576	10.07	-14.18	10.72	-14.18
289	577	578	17	-14.18	17.65	-14.18
290	579	580	3.14	-17.32	3.79	-17.32
291	581	582	10.07	-17.32	10.72	-17.32
292	583	584	17	-17.32	17.65	-17.32
293	585	586	3.14	-20.46	3.79	-20.46
294	587	588	10.07	-20.46	10.72	-20.46
295	589	590	17	-20.46	17.65	-20.46
296	591	592	3.14	-21.11	3.79	-21.11
297	593	594	10.07	-21.11	10.72	-21.11
298	595	596	3.14	-24.25	3.79	-24.25
299	597	598	10.07	-24.25	10.72	-24.25
300	599	600	3.14	-27.39	3.79	-27.39
301	601	602	10.07	-27.39	10.72	-27.39

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.

Table 3.4.4.1-15 $P_m + P_b$ Stresses for BWR Support Disk - Normal Conditions (ksi)

Section[1]	Sx	Sy	Sxy	Stress Intensity	Allow. Stress[2]	Margin of Safety
129	1.0	0.3	0.2	1.0	40.5	39.5
54	1.0	0.2	0.2	1.0	40.5	39.5
171	0.2	1.0	0.1	1.0	40.5	39.5
300	0.2	1.0	0.1	1.0	40.5	39.5
65	0.9	0.3	-0.2	1.0	40.5	39.5
192	0.9	0.3	-0.2	1.0	40.5	39.5
257	0.8	0.4	-0.3	1.0	40.5	39.5
234	0.2	0.9	-0.1	1.0	40.5	39.5
108	0.2	0.9	-0.1	1.0	40.5	39.5
119	0.9	0.2	-0.2	1.0	40.5	39.5
246	0.9	0.2	-0.2	0.9	40.5	44.0
182	0.9	0.2	0.2	0.9	40.5	44.0
103	0.3	0.3	0.2	0.5	40.5	80.0
229	0.2	0.3	0.2	0.5	40.5	80.0
109	-0.1	0.4	0.0	0.5	40.5	80.0
77	0.2	-0.3	0.1	0.5	40.5	80.0
203	0.2	-0.3	0.1	0.5	40.5	80.0
140	0.2	-0.3	-0.1	0.5	40.5	80.0
295	0.2	0.3	-0.2	0.5	40.5	80.0
269	0.2	-0.3	-0.1	0.5	40.5	80.0
166	0.2	0.3	-0.2	0.5	40.5	80.0
301	-0.1	0.4	0.0	0.5	40.5	80.0
172	-0.1	0.4	0.0	0.5	40.5	80.0
134	0.0	0.2	-0.2	0.5	40.5	80.0
263	0.0	0.2	-0.2	0.5	40.5	80.0
197	0.0	0.2	0.2	0.5	40.5	80.0
71	0.0	0.2	0.2	0.5	40.5	80.0
235	-0.1	0.4	0.0	0.5	40.5	80.0
27	0.3	-0.2	-0.1	0.5	40.5	80.0
165	-0.2	-0.1	-0.2	0.5	40.5	80.0
228	-0.2	-0.1	0.2	0.5	40.5	80.0
294	-0.2	-0.1	-0.2	0.5	40.5	80.0
40	0.3	-0.2	0.1	0.5	40.5	80.0
102	-0.2	-0.1	0.2	0.5	40.5	80.0
73	0.1	0.3	0.2	0.5	40.5	80.0
199	0.1	0.3	0.2	0.5	40.5	80.0
124	-0.4	-0.1	-0.2	0.4	40.5	100.3
252	-0.4	-0.1	-0.2	0.4	40.5	100.3
60	-0.4	-0.1	0.2	0.4	40.5	100.3
187	-0.4	-0.1	0.2	0.4	40.5	100.3

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.
2. Stress allowables are taken at 800°F.

Table 3.4.4.1-16 $P_m + P_b + Q$ Stresses for BWR Support Disk - Normal Conditions (ksi)

Section[1]	Sx	Sy	Sxy	Stress Intensity	Allow. Stress[2]	Margin of Safety
30	-8.8	-16.9	2.7	17.7	81.0	3.58
15	14.2	5.0	-6.4	17.4	81.0	3.66
43	-9.0	-16.6	2.7	17.4	81.0	3.66
13	14.0	5.1	-6.4	17.4	81.0	3.66
16	15.1	4.2	5.1	17.1	81.0	3.74
14	15.0	4.3	5.1	17.1	81.0	3.74
1	-1.8	14.0	-1.0	15.8	81.0	4.13
2	-1.8	14.0	-1.0	15.8	81.0	4.13
3	-1.8	13.9	-0.9	15.7	81.0	4.16
4	-1.8	13.9	-0.9	15.7	81.0	4.16
268	-7.4	-15.3	1.9	15.7	81.0	4.16
139	-7.4	-15.2	1.9	15.6	81.0	4.19
202	-7.4	-15.2	-1.9	15.6	81.0	4.19
76	-7.4	-15.2	-1.9	15.6	81.0	4.19
295	-0.6	-15.5	1.0	15.6	81.0	4.19
166	-0.5	-15.5	0.9	15.5	81.0	4.23
229	-0.8	-15.3	-1.0	15.4	81.0	4.26
103	-0.8	-15.3	-0.9	15.3	81.0	4.29
289	-4.4	-14.5	1.2	14.6	81.0	4.55
223	-4.5	-14.4	-1.2	14.6	81.0	4.55
160	-4.4	-14.4	1.2	14.5	81.0	4.59
97	-4.5	-14.4	-1.2	14.5	81.0	4.59
276	-5.6	-14.0	1.3	14.2	81.0	4.70
147	-5.6	-14.0	1.3	14.2	81.0	4.70
210	-5.5	-13.9	-1.3	14.1	81.0	4.74
84	-5.5	-13.9	-1.3	14.1	81.0	4.74
269	-6.7	-13.5	1.7	13.8	81.0	4.87
77	-6.5	-13.5	-1.6	13.8	81.0	4.87
140	-6.7	-13.5	1.7	13.8	81.0	4.87
203	-6.6	-13.5	-1.6	13.8	81.0	4.87
266	-8.3	-12.9	2.0	13.7	81.0	4.91
137	-8.3	-12.9	2.0	13.7	81.0	4.91
74	-8.2	-12.8	-2.0	13.6	81.0	4.96
18	-12.6	-7.2	2.4	13.6	81.0	4.96
200	-8.2	-12.8	-2.0	13.5	81.0	5.00
31	-12.6	-7.2	2.4	13.5	81.0	5.00
199	-13.0	-6.4	-1.5	13.3	81.0	5.09
73	-12.9	-6.3	-1.5	13.2	81.0	5.14
34	-12.4	-6.2	2.2	13.1	81.0	5.18
21	-12.4	-6.2	2.2	13.1	81.0	5.18

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.
2. Stress allowables are taken at 800°F.

Table 3.4.4.1-17 Summary of Maximum Stresses for PWR and BWR Fuel Basket Weldments - Normal Conditions (ksi)

Component	Stress Category	Maximum Stress Intensity[1]	Node Temperature (°F)	Stress Allowable[2]	Margin of Safety
PWR Top Weldment	$P_m + P_b$	0.5	297	28.1	+Large
	$P_m + P_b + Q$	52.4	292	56.1	0.07
PWR Bottom Weldment	$P_m + P_b$	0.6	179	30.0	+Large
	$P_m + P_b + Q$	20.9	175	60.0	+1.87
BWR Top Weldment	$P_m + P_b$	0.8	226	26.3	+Large
	$P_m + P_b + Q$	14.2	383	52.5	+Large
BWR Bottom Weldment	$P_m + P_b$	0.9	269	26.7	+Large
	$P_m + P_b + Q$	36.6	203	53.4	0.64

1. Nodal stresses are from the finite element analysis.
2. Conservatively, stress allowables are taken at 400°F for the PWR top weldment, 300°F for the PWR bottom weldment, 500°F for the BWR top weldment, and 300°F for the BWR bottom weldment.

3.4.4.2 Vertical Concrete Cask Analyses

The stresses in the concrete cask are evaluated in this section for normal conditions of storage. The evaluation for the steel base plate at the bottom of the cask is presented in Section 3.4.3.1. The stresses in the concrete due to dead load, live load, and thermal load are calculated in this section. The evaluations for off-normal and accident loading conditions are presented in Chapter 11.0. The radial dimensions of the concrete cask are the same for all cask configurations, only the height of the cask varies. Thus, the temperature differences through the concrete for all cask configurations vary only as a function of the heat source. Using the model described in this section, thermal analyses were run for both the maximum BWR and PWR heat loads for normal, off-normal, and accident conditions. The results of these analyses showed that the maximum temperature differences across the concrete cask wall occurred under normal operating conditions (76ºF, with a 1.275 load factor) for the BWR casks and under accident conditions (133ºF, with a load factor of 1.0) for the PWR casks. Thus, the structural analyses in this chapter use the temperature gradients from the BWR cask at 76ºF and the analyses in Chapter 11 use the temperature differences for the PWR cask at 133ºF. A summary of calculated stresses for the load combinations defined in Table 2.2-1 is presented in Table 3.4.4.2-1. As shown in Table 3.4.4.2-2, the concrete cask meets the structural requirements of ACI-349-85 [4].

The structural evaluation of the Universal Storage System is based on consideration of the bounding conditions for each aspect of the analysis. Generally, the bounding condition is represented by the component, or combination of components, of each configuration that is the heaviest. For reference, the bounding case used in each of the structural evaluations is presented in the following table.

Section	Aspect Evaluated	Bounding Condition	Configuration
3.4.4.2.1	Dead Load	Heaviest concrete cask	PWR Class 3
3.4.4.2.2	Live Load	Heaviest loaded transfer cask	BWR Class 5
	Snow Load	Same for all configurations	Not Applicable
3.4.4.2.3	Thermal Load	Highest temperature gradient under normal conditions	BWR Class 4

3.4.4.2.1 Dead Load

The concrete cask dead load evaluation is based on the PWR Class 3 concrete cask, which is the heaviest concrete cask. The weight used in this analysis bounds the calculated weight of the PWR Class 3 concrete cask, as shown in Tables 3.2-1 and 3.2-2. The dead load of the cask concrete is resisted by the lower concrete surface only. The concrete compression stress due to the weight of the concrete cask is:

$$\sigma_v = -W/A = -26.1 \text{ psi (compression)}$$

(30.0 psi conservatively used in the loading combination, Table 3.4.4.2-1)

where:

W = 250,000 lb concrete cask bounding dead weight (maximum calculated weight = 249,400 lb)
OD = 136 in. concrete exterior diameter
ID = 79.5 in. concrete interior diameter
A = $\pi (OD^2 - ID^2) / 4 = 9{,}563$ in.^{2}

This evaluation of stress at the base of the concrete conservatively considers the weight of the empty concrete cask, rather than the concrete alone. The weight of the canister is not supported by the concrete.

3.4.4.2.2 Live Load

The concrete cask is subjected to two live loads: the snow load and the weight of the fully loaded transfer cask resting atop the concrete cask. These loads are conservatively assumed to be applied to the concrete portion of the cask. No loads are assumed to be taken by the concrete cask's steel liner. The loads from the canister and its contents are transferred to the steel support inside the concrete cask and are not applied to the concrete. The stress in the steel support is evaluated in Section 3.4.3.1. Under these conditions, the only stress component is the vertical compression stress.

Snow Load

The calculated snow load and the resulting stresses are the same for all five of the concrete cask configurations because the top surface areas are the same for all configurations. The snow load on the concrete cask is determined in accordance with ANSI/ASCE 7-93 [30].

The uniformly distributed snow load on the top of the concrete cask, P_f , is

$$P_f = 0.70\ C_e\ C_t\ I\ P_g \quad = \quad 101 \text{ lbf/ft}^2$$

The concrete cask top area,

$$A_{top} = \pi\ (D/2)^2 = 14{,}527 \text{ in.}^2 = 101 \text{ ft}^2$$

The maximum snow load, F_s, is,

$$F_s = P_f \times A_{top} = 101 \text{ lbf/ft}^2 \times (101 \text{ ft}^2) = 10{,}201 \text{ lbf.}$$

The snow load is uniformly distributed over the top surface of the concrete cask. This load is negligible.

Transfer Cask Load

The live load of the heaviest loaded transfer cask is bounded by the weight used in this analysis, which is much greater than the weight of the maximum postulated snow load. Consequently, the stress due to the snow load is bounded by the stress due to the weight of the heaviest transfer cask. As with the snow load, the calculated transfer cask load, and the resulting stresses, are the same for all five of the concrete cask configurations because the top surface areas are the same for all configurations.

W $\approx$ 215,000 lb-bounding transfer cask weight (fully loaded)
D = 136 in.-concrete exterior diameter
ID = 79.5 in.-concrete interior diameter
A $= \pi (D^2 - ID^2)/4 = 9563$ in.^{2}

Compression stress at the base of the concrete is:

$\sigma_v = W/A = -22.5$ psi ≈ -25.0 psi (compressive)
(25.0 psi conservatively used in loading combination, Table 3.4.4.2-1)

3.4.4.2.3 Thermal Load

A three dimensional finite element model, shown in Figure 3.4.4.2-1, comprised of SOLID45, LINK8 (elements which support uniaxial loads only—no bending), and CONTAC52 elements was used to determine the stresses in the concrete cask due to thermal expansion. The SOLID45 elements represented the concrete while the LINK8 elements were used to represent the hoop and the vertical reinforcement bars. The model of the reinforcement bars is shown in Figure 3.4.4.2-2. The concrete cask has two sets of vertical reinforcement. At the inner radius of the concrete cask, there are 36 sets of vertical reinforcement, while at the outer radius, 56 sets of vertical reinforcement are used. The finite element model is a 1/56th circumferential model (or 360/56 = 6.42°), and the vertical reinforcement is modeled at the angular center of the model. To compensate for the smaller number of reinforcement elements at the inner radial location, the cross sectional area of the LINK8 elements were factored by 36/56. The cross sectional area of the LINK8s at the outer radial location corresponds to a Number 6 reinforcement bar, which has a 0.75-in. diameter and a cross sectional area of 0.44 in^2. LINK8s are also employed for the hoop reinforcements. The hoop reinforcements at the inner radial location are modeled 8-in. on center, while the outer hoop reinforcements are modeled on 4-in. centers. The nodal locations of the SOLID45 elements also correspond to the reinforcement locations to allow for the correct placement of the LINK8 elements in the model.

To allow the reinforcement to contain the tension stiffness of the concrete, the SOLID45 elements having nodes at a specified horizontal plane were separated by a small vertical distance

(0.1 in.) and were connected by CONTAC52 elements. The model contains three horizontal planes located at points ¼, ½, and ¾ of the axial length of the model. The CONTAC52 elements transmit compression across the horizontal planes, which allows the concrete elements to be subjected to compression. The LINK8 elements maintain a continuous connection from top to bottom. The structural boundary conditions are shown in Figure 3.4.4.2-3. The side of the model at 0° is restrained from translation in the circumferential direction. At 6.4°, the circumferential reinforcing bar (LINK8) elements extend beyond the model boundary and are also restrained at their ends from circumferential translation. The remaining nodes at 6.4° are attached to the CONTAC52 elements that only support compressive loading. The steel inner liner is radially coupled to the concrete, since for the thermal conditions analyzed, the steel will expand more than the concrete. The boundary conditions used simulate a complete fracture of the concrete at the 6.4° plane and between each of the axial sections of the model.

Analysis of the thermal loads and conditions for all cask configurations showed that maximum temperature gradient across the concrete wall of the cask under normal conditions, 62.42°F, occurs for the BWR configuration. Thus, the steady-state, three-dimensional thermal conduction analysis used the surface temperature boundary conditions for the 76°F normal operating condition to determine the temperature field throughout the model. These temperatures were applied with a load factor of 1.275 along the steel liner interior and concrete shell.

After the thermal solution was obtained, the thermal model was converted to a structural model. The nodal temperatures developed from the heat transfer analysis became the thermal load boundary conditions for the structural model.

The membrane stresses occurring in each individual circumferential reinforcement bar (rebar) varied on the basis of the rebar location along the longitudinal axis of the cask. The maximum circumferential tensile stress, 6,423 psi, occurred in the outer rebar, 56.4 in. from the base of the concrete cask.

The membrane stresses occurring in the vertical rebar varied on the basis of the radial location within the concrete shell. The maximum vertical tensile stress, 5,338 psi, occurred in the outer rebar 140.3 in. from the base of the cask.

The maximum allowable stress in the ASTM A-706 rebar material is:

$F_c = 60{,}000$ psi

The maximum allowable stress for the rebar assembly in the concrete cask shell is:

$$\sigma_{rebar} = \phi\, F_c = (0.9)(60{,}000 \text{ psi}) = 54{,}000 \text{ psi}$$

where:

F_c = 60,000 psi, the allowable stress on the rebar, and

ϕ = 0.90, a load reduction factor based on the rebar configuration.

Thus, the margin of safety of the rebar in the BWR cask under normal operating conditions is

$$MS = \frac{54{,}000\text{psi}}{6{,}423\text{psi}} - 1 = +7.4$$

The concrete component of the shell carries the compressive loads in both the circumferential and the vertical direction. The maximum calculated compressive stress, which occurs 144 in. from the base of the cask, is 116 psi in the circumferential direction. The maximum compressive concrete stress in the vertical direction is 653 psi, which occurs 136.34 in. from the base of the cask.

Tensile stresses were examined in both the axial and circumferential directions. Two vertical planes (at 0° and at 6.4° for circumferential stress) and three horizontal planes (bottom, middle and top, for axial stress) were examined at each of the four concrete sections modeled. The locations of the planes where the stress evaluations are performed are shown in Figures 3.4.4.2-4 and 3.4.4.2-5. The appropriate element stress is examined at each plane to determine if the stress is tensile or compressive. If the stress is tensile, the component stress and face area of that element are used to calculate an average concrete stress on the plane. If compressive, the element results are excluded from the calculation. Experimental studies show that the tensile strength of concrete is 8% to 15% of the concrete compressive strength [35]. Using a compressive strength of 4,000 psi and an 8% factor, an allowable tensile strength of 320 psi is used in the evaluation.

The results of the evaluation, presented in Tables 3.4.4.2-3 and 3.4.4.2-4, show that maximum tensile stress in the concrete is 143 psi and 243 psi, for the normal and accident conditions, respectively. These maximum stresses are less than the allowable stress (320 psi). Consequently, no cracking of the concrete will occur.

Applying the ACI 349-85 load reduction factor, the allowable bearing stress on the concrete shell is,

$$\sigma_{bearing} = \phi f_c' = (0.70)(4{,}000) = 2{,}800 \text{ psi}$$

where:

ϕ, the strength reduction factor for the concrete shell = 0.70

f_c', the nominal concrete compressive strength = 4,000 psi

The maximum 76°F normal operating thermally induced stress of 653 psi represents a margin of safety of

$$MS = \frac{2{,}800\text{psi}}{653\text{psi}} - 1 = +3.3$$

Figure 3.4.4.2-1 Concrete Cask Thermal Stress Model

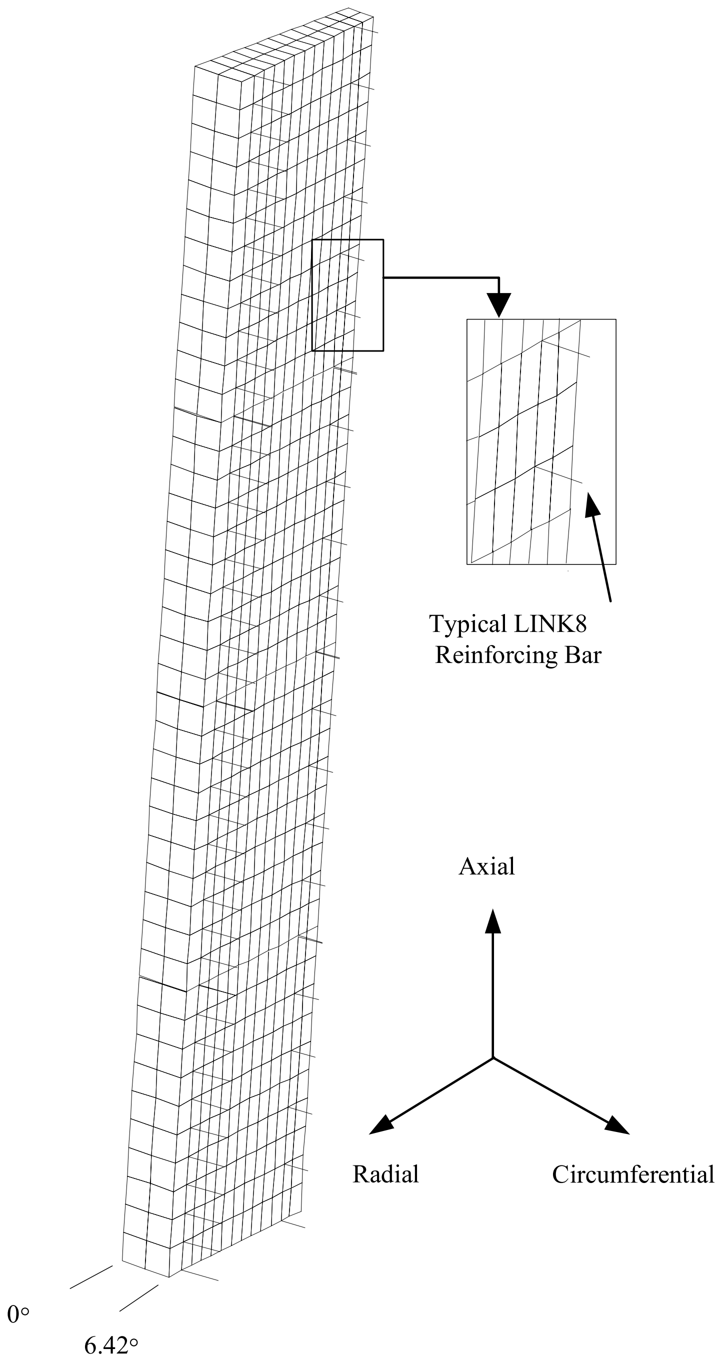


Figure 3.4.4.2-2 Concrete Cask Thermal Stress Model - Vertical and Horizontal Rebar Detail

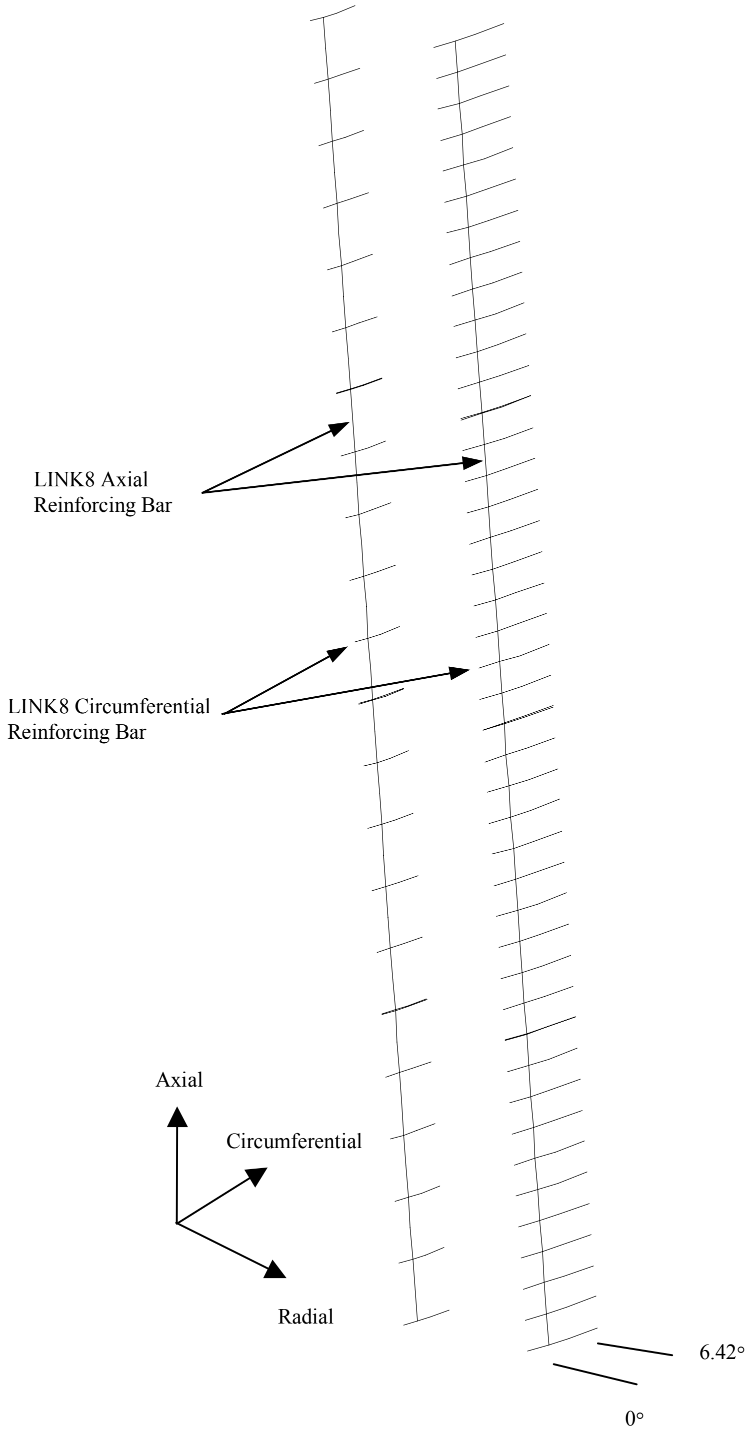
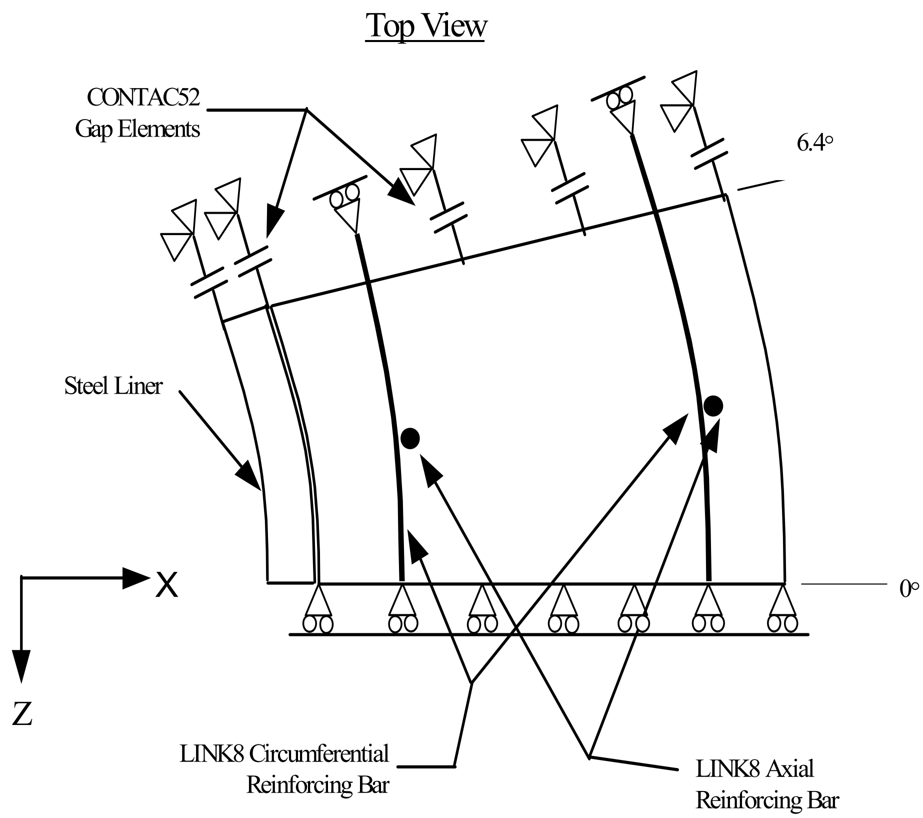


Figure 3.4.4.2-3 Concrete Cask Thermal Model Boundary Conditions



Note: CONTAC52 GAP Elements allow radial translation but don't transmit tensile loading

Figure 3.4.4.2-4 Concrete Cask Thermal Model Axial Stress Evaluation Locations

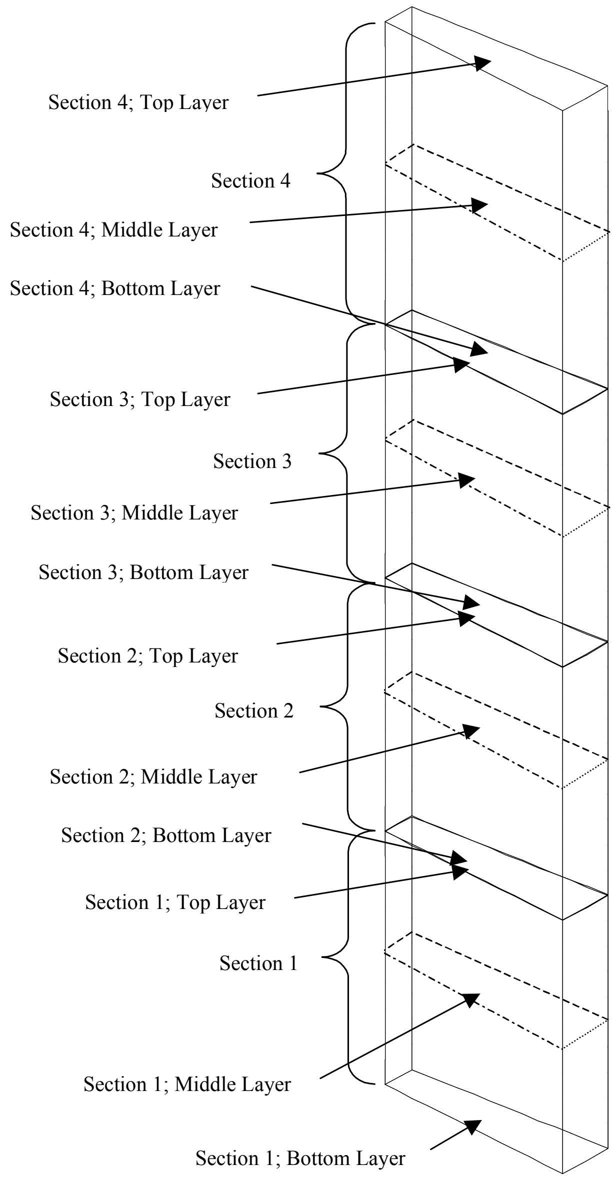


Figure 3.4.4.2-5 Concrete Cask Thermal Model Circumferential Stress Evaluation Locations

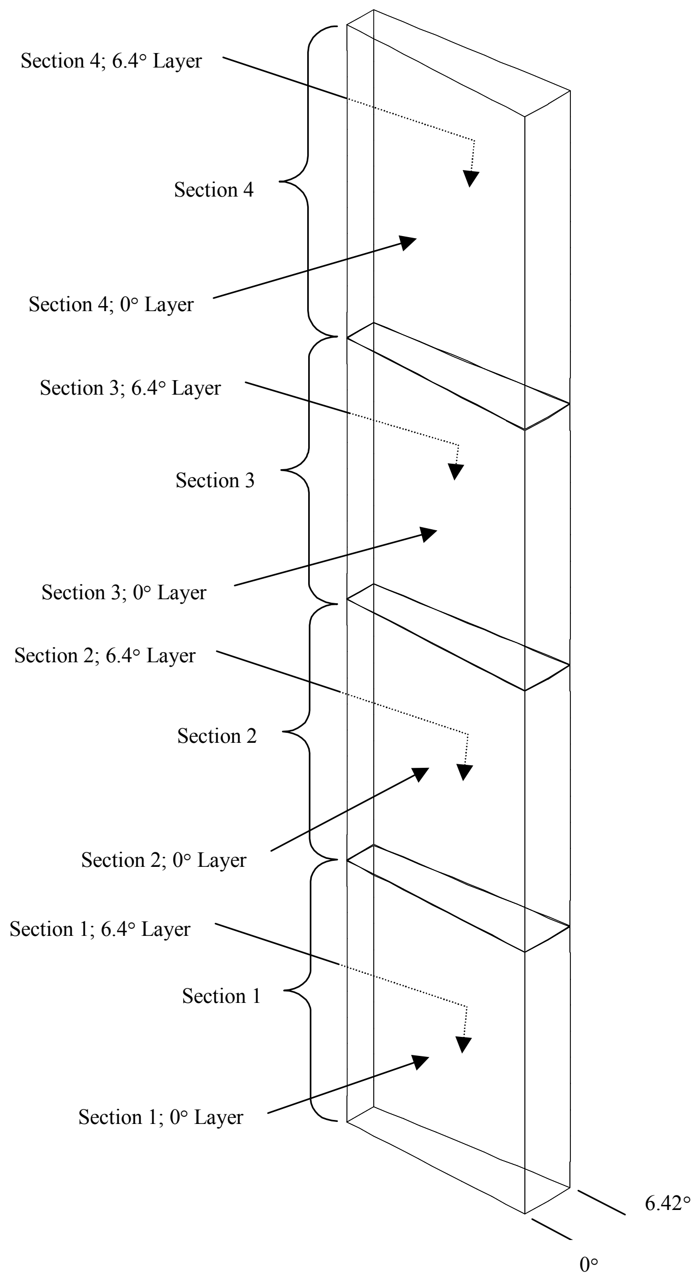


Table 3.4.4.2-1 Summary of Maximum Stresses for Vertical Concrete Cask Load Combinations

Load Comb[a]	Stress Direction	Stress [b] (psi)							
		Dead	Live	Wind [c]	Thermal [d]	Seismic [e]	Tornado [f]	Flood [g]	Total
Concrete Outside Surface:									
1	Vertical	-42.0	-43.0	—	—	—	—	—	-85.0
2	Vertical	-32.0	-32.0	—	—	—	—	—	-64.0
3	Vertical	-32.0	-32.0	-26.0	—	—	—	—	-90.0
4	Vertical	-30.0	-25.0	—	—	—	—	—	-55.0
5	Vertical	-30.0	-25.0	—	—	-135.0	—	—	-190.0
7	Vertical	-30.0	-25.0	—	—	—	—	-20.0	-75.0
8	Vertical	-30.0	-25.0	—	—	—	-20.0	—	-75.0
Concrete Inside Surface:									
1	Vertical	-42.0	-43.0	—	---	—	—	—	-85.0
	Circumferential	0.0	0.0	—	---	—	—	—	0.0
2	Vertical	-32.0	-32.0	—	-833.0	—	—	—	-897.0
	Circumferential	0.0	0.0	—	-147.0	—	—	—	-147.0
3	Vertical	-32.0	-32.0	-26.0	-833.0	—	—	—	-923.0
	Circumferential	0.0	0.0	0.0	-143.0	—	—	—	-143.0
4	Vertical	-30.0	-.30.0	—	-.721.0	—	—	—	-776.0
	Circumferential	0.0	0.0	—	-103.0	—	—	—	-103.0
5	Vertical	-.30.0	-30.0	—	-.653.0	-100.0	—	—	-808.0
	Circumferential	0.0	0.0	—	-116.0	—	—	—	-116.0
7	Vertical	-30.0	-30.0	—	-653.0	—	—	-20.0	-728.0
	Circumferential	0.0	0.0	—	-116.0	—	—	—	-116.0
8	Vertical	-30.0	-30.0	—	-653.0	—	-20.0	—	-728.0
	Circumferential	0.0	0.0	—	-116.0	—	—	—	-116.0

[a] Load combinations are defined in Table 2.2-1. See Sections 11.2.4 and 11.2.12 for evaluations of drop/impact and tipover conditions for load combination No. 6.

[b] Positive stress values indicate tensile stresses and negative values indicate compressive stresses.

[c] Stress results from Section 11.2.11 (tornado) are conservatively used with a load factor of 1.275.

[d] Tensile stresses (at concrete outside surface) are taken by the steel reinforcing bars and therefore are not shown in this Table. Stress Results for T_a (load combination #4) are obtained from Section 11.2.7.

[e] Stress results are obtained from Section 11.2.8.

[f] Stress results are obtained from Section 11.2.11 (tornado wind).

[g] Stress results are obtained from Section 11.2.9.

Table 3.4.4.2-2 Maximum Concrete and Reinforcing Bar Stresses

	Calculated (psi)	Allowable[1] (psi)	Margin of Safety
Concrete	923	2,800	+2.03
Reinforcing Bar			
Normal - vertical	5,338	54,000	+9.1
- hoop	6,423	54,000	+7.4
Accident[2] - vertical	6,619	54,000	+7.2
- hoop	7,869	54,000	+5.9

1 Allowable compressive stress for concrete is (0.7)(4,000 psi)=2,800 psi, where 0.7 is the strength reduction factor per ACI-349-85, Section 9.3; 4,000 psi is the nominal concrete strength. Allowable stress for reinforcing bar is determined in the calculation in this ACI Section.

2 Results are obtained from Section 11.2.7.

Table 3.4.4.2-3 Concrete Cask Average Concrete Axial Tensile Stresses

	Normal Conditions			Accident Conditions		
Stress Location	Calculated Stress (psi)	Allowable Stress (psi)	M.S.	Calculated Stress (psi)	Allowable Stress (psi)	M.S.
Section 1; Bottom Layer	38	320	7.4	149	320	1.1
Section 1; Middle Layer	27	320	10.8	46	320	6.0
Section 1; Top Layer	10	320	+Large	6	320	+Large
Section 2; Bottom Layer	85	320	2.7	133	320	1.4
Section 2; Middle Layer	42	320	6.6	90	320	2.6
Section 2; Top Layer	19	320	15.8	44	320	6.3
Section 3; Bottom Layer	77	320	3.2	120	320	1.7
Section 3; Middle Layer	66	320	3.8	136	320	1.4
Section 3; Top Layer	72	320	3.4	119	320	1.7
Section 4; Bottom Layer	37	320	7.6	65	320	3.9
Section 4; Middle Layer	59	320	4.4	116	320	1.8
Section 4; Top Layer	143	320	1.2	244	320	0.31

Table 3.4.4.2-4 Concrete Cask Average Concrete Hoop Tensile Stresses

	Normal Conditions			Accident Conditions		
Stress Location	Calculated Stress (psi)	Allowable Stress (psi)	M.S.	Calculated Stress (psi)	Allowable Stress (psi)	M.S.
Section 1; 0° Layer	29	320	10.0	50	320	5.4
Section 1; 6.42° Layer	28	320	10.4	43	320	6.4
Section 2; 0° Layer	57	320	4.6	89	320	2.6
Section 2; 6.42° Layer	59	320	4.4	85	320	2.8
Section 3; 0° Layer	87	320	2.7	114	320	1.8
Section 3; 6.42° Layer	85	320	2.8	108	320	2.0
Section 4; 0° Layer	61	320	4.2	80	320	3.0
Section 4; 6.42° Layer	58	320	4.5	74	320	3.3

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3.4.5 Cold

Severe cold environments are evaluated in Section 11.1.1. Stress intensities corresponding to thermal loads in the canister are evaluated by using a finite element model as described in Section 3.4.4.1. The thermal stresses that occur in the canister as a result of the maximum off-normal temperature gradients in the canister are bounded by the analysis of extreme cold in Section 11.1.1.

The PWR canister and basket are fabricated from stainless steel and aluminum, which are not subject to a ductile-to-brittle transition in the temperature range of interest. The BWR canister and basket are fabricated from stainless steel, aluminum, with carbon steel support disks. The carbon steel support disk thickness, 5/8 in., is selected to preclude brittle fracture at the design basis low temperature (-40°F). However, low temperature handling limits do apply to the transfer cask.

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3.5 Fuel Rods

The Universal Storage System is designed to limit fuel cladding temperatures to levels below those where zirconium alloy degradation is expected to lead to fuel clad failure. As shown in Chapter 4, fuel cladding temperature limits for PWR and BWR fuel have been established at 380°C based on 5-year cooled fuel for normal conditions of storage and 570°C for short term off-normal and accident conditions.

As shown in Table 4.1-4 and 4.1-5, the calculated maximum fuel cladding temperatures are well below the temperature limits for all design conditions of storage.

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3.6 Structural Evaluation of Site Specific Spent Fuel

This section presents the structural evaluation of fuel assemblies or configurations, which are unique to specific reactor sites or which differ from the UMS® Storage System design basis fuel. These site specific configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel that is classified as damaged. Damaged fuel includes fuel rods with cladding that exhibit defects greater than pinhole leaks or hairline cracks.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation.

3.6.1 Structural Evaluation of Maine Yankee Site Specific Spent Fuel for Normal Operating Conditions

This section describes the structural evaluation for site specific spent fuel configurations. As described in Sections 1.3.2.1 and 2.1.3.1, the inventory of site specific spent fuel configurations includes fuel classified as undamaged, undamaged with additional fuel and nonfuel-bearing hardware, consolidated fuel and fuel classified as damaged. Damaged fuel is separately containerized in one of the two configurations of the Maine Yankee Fuel Can.

3.6.1.1 Maine Yankee Undamaged Spent Fuel

The description for Maine Yankee site specific fuel is in Section 1.3.2.1. The standard spent fuel assembly for the Maine Yankee site is the Combustion Engineering (CE) 14×14 fuel assembly. Fuel of the same design has also been supplied by Westinghouse and by Exxon. The standard 14×14 fuel assemblies are included in the population of the design basis PWR fuel assemblies for the UMS® Storage System (see Table 2.1.1-1). The structural evaluation for the UMS® transport system loaded with the standard Maine Yankee fuels is bounded by the structural evaluations in Chapter 3 for normal conditions of storage and Chapter 11 for off-normal and accident conditions of storage.

With the Control Element Assembly (CEA) inserted, the weight of a standard CE 14×14 fuel assembly is 1,360 pounds. This weight is bounded by the weight of the design basis PWR fuel assembly (37,608/24 = 1,567 lbs) used in the structural evaluations (Table 3.2-1). The fuel configurations with removed fuel rods, with fuel rods replaced by solid stainless steel or zirconium alloy rods, or with poison rods replaced by hollow zirconium alloy tubes, all weigh less than the standard CE 14×14 fuel assembly. The configuration with instrument thimbles installed in the center guide tube position weighs less than the standard assembly with the installed control element assembly. Consequently, this configuration is also bounded by the weight of the design basis fuel assembly. Since the weight of any of these fuel assembly configurations is bounded by the design basis fuel assembly weight, no additional analysis of these configurations is required.

The two consolidated fuel lattices are each constructed of 17×17 stainless steel fuel grids and stainless steel end fittings, which are connected by 4 stainless steel support rods. One of the consolidated fuel lattices has 283 fuel rods with 2 empty positions. The other has 172 fuel rods, with the remaining positions either empty or holding stainless steel rods. The calculated weight for the heaviest of the two consolidated fuel lattices is 2,100 pounds. Only one consolidated fuel lattice can be loaded into any one canister. The weight of the site specific 14×14 fuel assembly plus the CEA is approximately 1,360 lbs. Twenty-three (23) assemblies (at 1,360 lbs each) in addition to the consolidated fuel assembly (at approximately 2,100 lbs) would result in a total weight of 33,380 pounds.

Therefore, the design basis UMS® PWR fuel weight of 37,608 lbs bounds the site specific fuel and consolidated fuel by 12%. The evaluations for the Margin of Safety for the dead weight load of the fuel and the lifting evaluations in Section 3.4.4 bound the Margins of Safety for the Maine Yankee site specific fuel.

3.6.1.2 Maine Yankee Damaged Spent Fuel

The Maine Yankee fuel can, shown in Drawings 412-501 and 412-502, is provided to accommodate Maine Yankee damaged fuel. The fuel can fits within a standard PWR basket fuel tube. The primary function of the Maine Yankee fuel can is to confine the fuel material within the can to minimize the potential for dispersal of the fuel material into the canister cavity volume.

The Maine Yankee fuel can is designed to hold an undamaged fuel assembly, a damaged fuel assembly, a fuel assembly with a burnup between 45,000 and 50,000 MWd/MTU and having a cladding oxidation layer thickness greater than 80 microns, or consolidated fuel in the Maine Yankee fuel inventory.

The fuel can is provided in two configurations that differ only in the square cross-section of the body of the fuel can. Both fuel can configurations have walls made of 0.048-inch thick Type 304 stainless steel sheet (18 gauge), have a total length of 162.8 inches and both have a bottom plate that is 0.63 inches thick. Four holes in the plates, screened with a Type 304 stainless steel wire screen (250 openings/inch × 250 openings/inch mesh), permit water to be drained from the can during loading operations. Since the bottom surface of the fuel can rests on the canister bottom plate, additional slots are machined in the fuel can (extending from the holes to the side of the bottom assembly) to allow the water to be drained from the can. At the top of the can, the wall thickness is increased to 0.15-inches to permit the can to be handled. Slots in the top assembly side plates allow the use of a handling tool to lift the can and contents. To confine the contents within the can, the top assembly consists of a 0.88-inch thick plate with screened drain holes identical to those in the bottom plate. Once the can is loaded, the can and contents are inserted into the basket, where the can may be supported by the sides of the fuel assembly tube, which are backed by the structural support disks. Alternately, the empty fuel can may be placed in the basket prior to having the designated contents inserted in the fuel can. The two configurations have different cross-sections in the can body. The first configuration has a square minimum internal width of 8.52 inches. The second has a square minimum internal width of 8.3 inches. This smaller internal width is conservatively used in the load handling analysis.

In normal operation, the can is in a vertical position. The weight of the fuel can contents is transferred through the bottom plate of the can to the canister bottom plate, which is the identical load path for undamaged fuel. The only loading in the vertical direction is the weight of the can and the top assembly. The lifting of the can with its contents is also in the vertical direction.

Classical hand calculations are used to qualify the stresses in the Maine Yankee fuel can.

A conservative bounding temperature of 600°F is used for the evaluation of the fuel can for normal conditions of storage. A temperature of 300°F is used for the lifting components at the top of the fuel can and for the lifting tool.

Calculated stresses are compared to allowable stresses in accordance with ASME Code, Section III, Subsection NG. The ASME Code, Section III, Subsection NG allowable stresses used for stress analysis are:

Property	600°F	300°F
S_u	63.3 ksi	66.0 ksi
S_y	18.6 ksi	22.5 ksi
S_m	16.7 ksi	20.0 ksi
E	25.2×10^3 ksi	27.0×10^3 ksi

The Maine Yankee fuel can is evaluated for dead weight and handling loads for normal conditions of storage. Since the can is not restrained, it is free to expand. Therefore, the thermal stress is considered to be negligible.

The Maine Yankee fuel can lifting components and handling tools are designed with a safety factor of 3.0 on material yield strength.

3.6.1.2.1 Dead Weight and Handling Loading Evaluation

The weight of the Maine Yankee fuel can is 130 pounds. The maximum compressive stress acting in the tube of the fuel can is due to its own weight in addition to that of the top assembly. A 10% dynamic load factor is applied to the fuel can weight for an applied load of 143 pounds to account for loads due to handling. Based on the minimum cross-sectional area of $(8.42)^2 - (8.32)^2 = 1.674\ \text{in}^2$, the margin of safety at 300°F is:

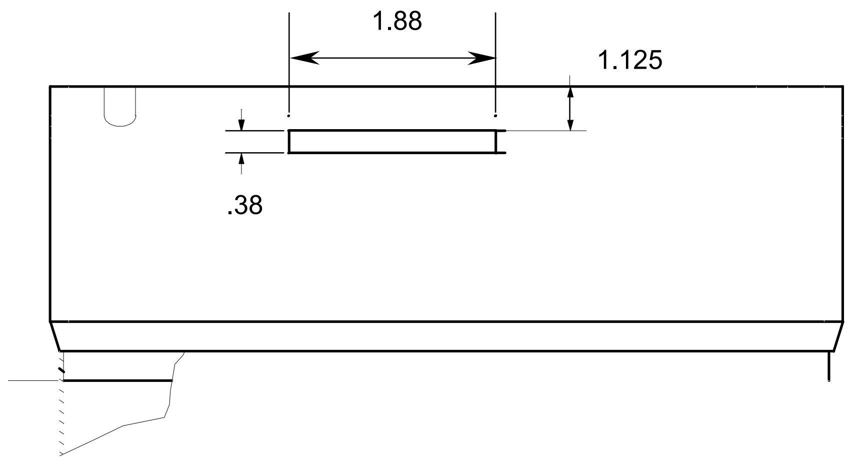
$$\text{M.S.} = 20{,}000/(143/1.674) - 1$$
$$\text{M.S.} = +\text{Large}$$

3.6.1.2.2 Lifting Evaluation

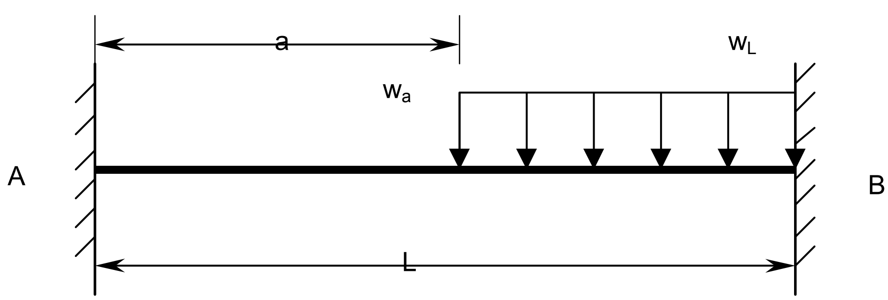
Based on the loaded weight of the fuel can, the lift evaluation does not require the use of the design criteria of ANSI N14.6 or NUREG-0612. However, for purposes of conservatism and good engineering practice, a factor of safety of three on material yield strength is used for the stress evaluations for the lift condition. Since a combined stress state results from the loading and the calculated stresses are compared to material yield strength, the Von Mises stress is computed.

Side Plates

The side plates will be subjected to bending, shear, and bearing stresses because of interaction with the lifting tool during handling operations. The lifting tool engages the 1.875-inch × 0.38-inch lifting slots with lugs that are 1-inch wide and lock into the four lifting slots. For this evaluation, the handling load is the weight of the consolidated fuel assembly (2,100 lbs design weight) plus the Maine Yankee fuel can weight (130 lbs), amplified by a dynamic load factor of 10%. Although the four slots are used to lift the can, the analysis assumes that the entire design load is shared by only two lift slots.



The stress in the side plate above the slot is determined by analyzing the section above the slot as a 0.15-inch wide × 1.875-inch long × 1.125-inch deep beam that is fixed at both ends. The lifting tool lug is 1 inch wide and engages the last 1 inch of the slot. The following figure represents the configuration to be evaluated:



where:

a = 0.875 in.
L = 1.875 in.
w_a = w_L = (2,230 lbs/2)(1.10)/1.0 in. = 613.3 lbs/in, use 620 lbs/in.

Reactions and moments at the fixed ends of the beam are calculated per Roark's Formula, Table 3, Case 2d.

The reaction at the left end of the beam (R_A) is:

$$R_A = \frac{w_a}{2L^3}(L-a)^3(L+a)$$

$$= \frac{620}{2(1.875)^3}(1.875-0.875)^3(1.875+0.875) = 129.3 \text{ lbs}$$

The moment at the left end of the beam (M_A) is:

$$M_A = \frac{-w_a}{12L^2}(L-a)^3(L+3a)$$
$$= \frac{-620}{12(1.875)^2}(1.875-0.875)^3(1.875+3(0.875)) = -66.1 \text{ lbs}\cdot\text{in.}$$

The reaction at the right end of the beam (R_B) is:

$$R_B = w_a(L-a) - R_A = 620(1.875-0.875) - 164.2 = 490.7 \text{ lbs}$$

The moment at the right end of the beam (M_B) is:

$$M_B = R_A L + M_A - \frac{w_a}{2}(L-a)^2$$
$$= 129.3(1.875) + (-66.1) - \frac{620}{2}(1.875-0.875)^2 = -133.7 \text{ lbs}\cdot\text{in.}$$

The maximum bending stress (σ_b) in the side plate is:

$$\sigma_b = \frac{Mc}{I} = \frac{133.7(0.5625)}{0.0178} = 4{,}224 \text{ psi}$$

The maximum shear stress (τ) occurs at the right end of the slot:

$$\tau = \frac{R_B}{A} = \frac{490.7}{1.125(0.15)} = 2{,}908 \text{ psi}$$

The Von Mises stress (σ_{max}) is:

$$\sigma_{max} = \sqrt{\sigma_b^2 + 3\tau^2} = \sqrt{4{,}224^2 + 3(2{,}908)^2} = 6{,}573 \text{ psi}$$

The yield strength (S_y) for Type 304 stainless steel is 22,500 psi at 300°F. The factor of safety is calculated as:

$$FS = \frac{22{,}500}{6{,}573} = 3.4 > 3$$

The design condition requiring a safety factor of 3 on material yield strength is satisfied.

Tensile Stress

The tube body will be subjected to tensile loads during lifting operations. The load (P) includes the can contents (2,100 lbs design weight), the tube body weight (78.77 lbs), and the bottom assembly weight (12.98 lbs) for a total of 2,191.8 pounds. A load of 2,200 lbs with a 10% dynamic load factor is used for the analysis.

The tensile stress (σ_t) is then:

$$\sigma_t = \frac{1.1P}{A} = \frac{1.1(2{,}200 \text{ lb})}{1.674 \text{ in.}^2} = 1{,}446 \text{ psi}$$

where:

A = tube cross-section area = $8.42^2 - 8.32^2 = 1.674$ in^2

The factor of safety (FS) based on the yield strength at 600°F (18,600 psi) is:

$$FS = \frac{18{,}600 \text{ psi}}{1{,}446} = 12.9 > 3$$

Weld Evaluation

The welds joining the tube body to the bottom weldment and to the side plates are full penetration welds (Type III, paragraph NG-3352.3). In accordance with NG-3352-1, the weld quality factor (n) for a Type III weld with visual surface inspection is 0.5.

The weld stress (σ_w) is:

$$\sigma_w = \frac{1.1(P)}{A} = \frac{1.1(2{,}200)}{1.674} = 1{,}446 \text{ psi}$$

where:

P = the combined weight of the tube body, bottom weldment, and can contents
A = cross sectional area of thinner member joined

The factor of safety (FS) is:

$$FS = \frac{n \cdot S_y}{\sigma_w} = \frac{0.5(18{,}600 \text{ psi})}{1{,}446 \text{ psi}} = +6.4 > 3$$

3.7 References

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3.8 Carbon Steel Coatings Technical Data

This section presents the technical data sheets for Carboline 890, Keeler & Long E-Series Epoxy Enamel, Keeler & Long Kolor-Poxy Primer No. 3200, Acrythane Enamel Y-1 Series top coating, PPG METALHIDE® 97-694 Series Primer or PPG DIMETCOTE® 9 Primer and PPG PITT-THERM® 97-724 Series top coating. These coatings are applied to protect exposed carbon steel surfaces of the transfer cask and the vertical concrete cask. Also provided is a description of the electroless nickel coating that is applied to the BWR support disks. Each coating meets the service and performance requirements that are established for the coating by the design and service environment of the component to be covered.

Performance requirements for the coatings of the carbon steel components used in the primary containment facility (Service Level 1) include the transfer cask and the BWR support disks. These components are exposed to similar environments and require that the coatings meet the following conditions:

- be applied to carbon steel
- be submersible for up to a week in clean water
- are rated Service Level 1 (EPRI TR-106160 for paints)
- do not contain zinc (boric acid pool condition)
- have a service temperature of at least 200°F in water and 600°F in a dry environment (applicable to basket materials)
- generate no hydrogen, or minimal hydrogen, when submersed in water (in-pool service)
- have no, or limited, special processes required for proper application or curing
- have a service environment in a high radiation field (basket material service)

Either Carboline 890 or Keeler & Long E-Series Epoxy Enamel may be used on the exposed carbon steel surfaces of the transfer cask and the transfer cask extension. These coatings are listed in EPRI TR 106160, "Coating Handbook for Nuclear Power Plants," June 1996 [36], as meeting the requirements for Service Level 1 or 2.

Electroless nickel coating is used on the carbon steel BWR support disks to provide a submersible, passive protective finish. This coating has a history of acceptance and successful performance in similar service conditions.

Vertical Concrete Cask carbon steel coatings provide service outside containment and are subject to radiation, heat loads and decontamination. These coatings are defined as Service Level 2 applications. Coatings identified for Service Level 1 are acceptable for Service Level 2 applications. Following initial shop application, alternate coatings to those listed previously may be used in routine maintenance for protection of the exposed Vertical Concrete Cask carbon steel surfaces.

No coating characteristics that may enhance the performance of the coated components (such as better emissivity) are considered in the analyses of these components. Therefore, no adverse effect on system performance results from incidental scratching or flaking of the coating, and no touchup of the coating on the BWR support disks or the storage cask liner is required.

3.8.1 Carboline 890

product data sheet

carboline®

CARBOLINE® 890

VOC plus

9860

SELECTION DATA

GENERIC TYPE: Two component, cross-linked epoxy.

GENERAL PROPERTIES: CARBOLINE 890 is a high solids, high gloss, high build epoxy topcoat that can be applied by spray, brush, or roller. The cured film provides a tough, cleanable and esthetically pleasing surface. Available in a wide variety of clean, bright colors. Features include:

- Good flexibility and lower stress upon curing than most epoxy coatings.
- Very good weathering resistance for a high gloss epoxy.
- Very good abrasion resistance.
- Excellent performance in wet exposures.
- Meets the most stringent VOC (Volatile Organic Content) regulations.

RECOMMENDED USES: Recommended where a high performance, attractive, chemically resistant epoxy topcoat is desired. Offers outstanding protection for interior floors, walls, piping, equipment and structural steel or as an exterior coating for tank farms, railcars, structural steel and equipment in various corrosive environments. Recommended industrial environments include Chemical Processing, Offshore Oil and Gas, Food Processing and Pharmaceutical, Water and Waste Water Treatment, Pulp and Paper, Power Generation among others. May be used as a two coat system direct to metal or concrete for Water and Municipal Waste Water immersion. CARBOLINE 890 has been accepted for use in areas controlled by USDA regulations for incidental food contact. Consult Carboline Technical Service Department for other specific uses.

NOT RECOMMENDED FOR: Strong acid or solvent exposures, or immersion service other than recommended.

TYPICAL CHEMICAL RESISTANCE:

Exposure	Immersion	Splash and Spillage	Fumes
Acids	NR	Very Good	Very Good
Alkalies	NR	Excellent	Excellent
Solvents	NR	Very Good	Excellent
Salt Solutions	Excellent	Excellent	Excellent
Water	Excellent	Excellent	Excellent

*NR = Not recommended

TEMPERATURE RESISTANCE:
Continuous: 200° F (93° C)
Non-continuous: 250° F (121° C)

At 300° F, coating discoloration and loss of gloss is observed, without loss of film integrity.

SUBSTRATES: Apply over suitably prepared metal, concrete, or other surfaces as recommended.

COMPATIBLE COATINGS: May be applied directly over inorganic zincs, weathered galvanizing, catalyzed epoxies, phenolics or other coatings as instructed. A test patch is recommended before use over existing coatings. May be used as a tiecoat over inorganic zincs. A mist coat of CARBOLINE 890 is required when applied over inorganic zincs to minimize bubbling. May be topcoated to upgrade weathering resistance. Not recommended over chlorinated rubber or latex coatings. Consult Carboline Technical Service Department for specific recommendations.

April 91 Replaces Oct. 90

SPECIFICATION DATA

THEORETICAL SOLIDS CONTENT OF MIXED MATERIAL:*

	By Volume
CARBOLINE 890	75%±2%

VOLATILE ORGANIC CONTENT:*
As Supplied: 1.78 lbs./gal.(214 gm/liter)
Thinned: The following are nominal values utilizing:

% Thinned	Fluid Ounces/Gal.	Pounds/ Gallon	Grams/ Liter
CARBOLINE Thinner # 2 (spray application)			
10%	12.8	2.26	271
CARBOLINE Thinner #33 (brush & roller application)			
12%	16	2.38	285

*Varies with color

RECOMMENDED DRY FILM THICKNESS PER COAT:
4-6 mils(100-150 microns).
5-7 mils (125-175 microns) DFT for a more uniform gloss over inorganic zincs.
Dry film thicknesses in excess of 10 mils(250 microns) per coat are not recommended. Excessive film thickness over inorganic zinc may increase damage during shipping or erection.

THEORETICAL COVERAGE PER MIXED GALLON:
1203 mil sq. ft. (30 sq. m/l at 25 microns)
241 sq. ft. at 5 mils(6.0 sq. m/l at 125 microns)
Mixing and application losses will vary and must be taken into consideration when estimating job requirements.

STORAGE CONDITIONS: Store Indoors
Temperature: 40-110° F (4-43° C)
Humidity: 0-100%

SHELF LIFE: Twenty-four months minimum when stored at 75° F (24° C).

COLORS: Available in Carboline Color Chart colors. Some colors may require two coats for adequate hiding. Colors containing lead or chrome pigments are not USDA acceptable. Consult your local Carboline representative or Carboline Customer Service for availability.

* See notice under DRYING TIMES.

GLOSS: High gloss (Epoxies lose gloss and eventually chalk in sunlight exposure).

ORDERING INFORMATION

Prices may be obtained from your local Carboline Sales Representative or Carboline Customer Service Department.

APPROXIMATE SHIPPING WEIGHT:

	2 Gal. Kit	10 Gal. Kit
CARBOLINE 890	29 lbs. (13 kg)	145 lbs. (66 kg)
THINNER #2	8 lbs. in 1's (4 kg)	39 lbs. in 5's (18 kg)
THINNER #33	9 lbs. in 1's (4 kg)	45 lbs. in 5's (20 kg)

FLASHPOINT: (Pensky-Martens Closed Cup)

CARBOLINE 890 Part A	73° F (23° C)
CARBOLINE 890 Part B	71° F (22° C)
THINNER #2	24° F (– 5° C)
THINNER #33	98° F (37° C)

APPLICATION INSTRUCTIONS
CARBOLINE® 890

These instructions are not intended to show product recommendations for specific service. They are issued as an aid in determining correct surface preparation, mixing instructions and application procedure. It is assumed that the proper product recommendations have been made. These instructions should be followed closely to obtain the maximum service from the materials.

0986

SURFACE PREPARATION: Remove oil or grease from surface to be coated with clean rags soaked in CARBOLINE Thinner #2 or Surface Cleaner #3 (refer to Surface Cleaner #3 instructions) in accordance with SSPC-SP 1.

Steel: Normally applied over clean, dry recommended primers. May be applied directly to metal. For immersion service, abrasive blast to a minimum Near White Metal Finish in accordance with SSPC-SP10, to a degree of cleanliness in accordance with NACE #2 to obtain a 1.5-3 mil (40-75 micron) blast profile. For non-immersion, abrasive blast to a Commercial Grade Finish in accordance with SSPC-SP6, to a degree of cleanliness in accordance with NACE #3 to obtain a 1.5-3 mil (40-75 micron) blast profile.

Concrete: Apply over clean, dry recommended surfacer or primer. Can be applied directly to damp(not visibly wet) or dry concrete where an uneven surface can be tolerated. Remove laitance by abrasive blasting or other means.

Do not coat concrete treated with hardening solutions unless test patches indicate satisfactory adhesion. Do not apply coating unless concrete has cured at least 28 days at 70° F (21° C) and 50% RH or equivalent time.

MIXING: Mix separately, then combine and mix in the following proportions:

	2 Gal. Kit	10 Gal. Kit
CARBOLINE 890 Part A	1 gallon	5 gallons
CARBOLINE 890 Part B	1 gallon	5 gallons

THINNING: For spray applications, may be thinned up to 10% (12.8 fl. oz./gal.) by volume with CARBOLINE Thinner #2.

For brush and roller application may be thinned up to 12 % (16 fl. oz./gal.) by volume with CARBOLINE Thinner #33.

Refer to Specification Data for VOC information.

Use of thinners other than those supplied or approved by Carboline may adversely affect product performance and void product warranty, whether express or implied.

POT LIFE: Three hours at 75° F (24° C) and less at higher temperatures. Pot life ends when material loses film build.

APPLICATION CONDITIONS:

	Material	Surfaces	Ambient	Humidity
Normal	60-85° F (16-29° C)	60-85° F (16-29° C)	60-90° F (16-32° C)	0-80%
Minimum	50° F (10° C)	50° F (10° C)	50° F (10° C)	0%
Maximum	90° F (32° C)	125° F (52° C)	110° F (43° C)	80%

Do not apply when the surface temperature is less than 5° F (or 3° C) above the dew point.

Special thinning and application techniques may be required above or below normal conditions.

SPRAY: This is a high solids coating and may require slight adjustments in spray techniques. Wet film thicknesses are easily and quickly achieved. The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco.

Conventional: Pressure pot equipped with dual regulators, 3/8" I.D. minimum material hose, .070" I.D. fluid tip and appropriate air cap.

Airless:
Pump Ratio: 30:1 (min.)*
GPM Output: 3.0 (min.)
Material Hose: 3/8"I.D.(min.)
Tip Size: .017-.021"
Output psi: 2100-2300
Filter Size: 60 mesh

*Teflon packings are recommended and are available from the pump manufacturer.

BRUSH OR ROLLER: Use medium bristle brush, or good quality short nap roller, avoid excessive rebrushing and rerolling. Two coats may be required to obtain desired appearance, hiding and recommended DFT. For best results, tie-in within 10 minutes at 75° F (24° C).

DRYING TIMES: These times are at 5 mils (125 microns) dry film thickness. Higher film thicknesses will lengthen cure times.

Dry to Touch 2 1/2 hours at 75° F (24° C)
Dry to Handle 6 1/2 hours at 75° F (24° C)

Temperature	Dry to Topcoat**	Final Cure
50° F (10° C)	24 hours	3 days
60° F (16° C)	16 hours	2 days
75° F (24° C)	8 hours	1 day
90° F (32° C)	4 hours	16 hours

**When recoating with CARBOLINE 890, recoat times will be drastically reduced. Contact Carboline Technical Service for specific recommendation.

Recommended minimum cure before immersion service is 5 days at 75° F (24° C).

EXCESSIVE HUMIDITY OR CONDENSATION ON THE SURFACE DURING CURING MAY RESULT IN SURFACE HAZE OR BLUSH; ANY HAZE OR BLUSH MUST BE REMOVED BY WATER WASHING BEFORE RECOATING.

CLEANUP: Use CARBOLINE Thinner #2.

CAUTION: READ AND FOLLOW ALL CAUTION STATEMENTS ON THIS PRODUCT DATA SHEET AND ON THE MATERIAL SAFETY DATA SHEET FOR THIS PRODUCT.

CAUTION: CONTAINS FLAMMABLE SOLVENTS. KEEP AWAY FROM SPARKS AND OPEN FLAMES. IN CONFINED AREAS WORKMEN MUST WEAR FRESH AIRLINE RESPIRATORS. HYPERSENSITIVE PERSONS SHOULD WEAR GLOVES OR USE PROTECTIVE CREAM. ALL ELECTRIC EQUIPMENT AND INSTALLATIONS SHOULD BE MADE AND GROUNDED IN ACCORDANCE WITH THE NATIONAL ELECTRICAL CODE. IN AREAS WHERE EXPLOSION HAZARDS EXIST, WORKMEN SHOULD BE REQUIRED TO USE NONFERROUS TOOLS AND TO WEAR CONDUCTIVE AND NONSPARKING SHOES.

carboline®

350 Hanley Industrial Ct • St. Louis, MO 63144-1599
an RPM company • 314-644-1000

3.8.2 Keeler & Long E-Series Epoxy Enamel

March, 1995

SSU-1

KEELER & LONG INC.

HEADQUARTERS:
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Watertown, CT 06795
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PROTECTIVE COATING SYSTEMS FOR NUCLEAR POWER PLANTS

INTRODUCTION

In the 1960's Keeler & Long made the commitment to develop Protective Coating Systems for Nuclear Power Plants. Coating Systems were developed and qualified in accordance with accepted standards, with emphasis upon their usage and specification for NEW construction projects. These systems were applied directly to either concrete or carbon steel substrates utilizing ideal surface preparation.

Presently, there is a necessity to apply these same coating systems or newly formulated systems over the original systems or over substrates which cannot be ideally prepared. Several years ago, Keeler & Long initiated a test program in order to test and qualify systems in conjunction with competitors products and/or with methods of preparation which are considered less than ideal. This test program provides OPERATING Nuclear Plants with qualified methods of preparation and a variety of qualified mixed coating systems.

HISTORY

In 1967, we embarked upon a testing program in order to comply with standards being prepared by the experts in the field and under the jurisdiction of The American National Standards Institute (ANSI). Earlier testing had involved research in order to determine the radiation tolerance and the decontamination properties of a variety of generic coating types including zinc rich, alkyds, chlorinated rubbers, vinyls, latex emulsions, and epoxies. This testing was conducted by various independent laboratories, such as Oak Ridge National Laboratory, Idaho Nuclear, and The Western New York Nuclear Research Center. It was concluded from these tests that almost any generic coating type would produce satisfactory radiation resistance and decontaminability.

Upon completion of the first ANSI Standards, however, it became evident that only Epoxy Coatings would meet the specific minimum acceptance criteria set forth in these standards. The single most important change from the earlier testing was the inclusion of a test which simulates the operation of the emergency core cooling system. This test is referred to as the Loss of Coolant Accident (LOCA) or the Design Basis Accident Condition (DBA). The test involves a high pressure, high temperature, alkaline, immersion environment.

Simultaneous with the preparation of these standards, we prepared to test Epoxy Systems in order to comply with the requirements. First hand knowledge of these standards was available since our personnel assisted in the development of these documents. Equipment was designed and built by our laboratory in order to conduct in-house DBA tests. The required physical and chemical tests were either conducted by us or by universities through research grants.

In 1972, the testing program was taken a step further in order to establish more credibility. The Franklin Institute of Philadelphia constructed an apparatus in order to simulate various Design Basis Accident Conditions and we prepared blocks and panels for an independent evaluation. The test results were among the "First" from an independent source, and these tests substantiated more than two years of in-house testing.

The Franklin Institute tests, along with our in-house testing program, were used as a basis for qualification until 1976. During this period also the following ANSI standards were revised and/or developed:

> **ANSI N5.9-1967** "Protective Coatings (Paints) for the Nuclear Industry" (Rev. ANSI N512-1974)
>
> **ANSI N101.2-1972** "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities"
>
> **ANSI N101.4-1972** "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"

Simultaneously, we developed a written Quality Assurance Program in compliance with ANSI N101.4 - 1972, Appendix B 10CFR50 of the Federal Register, and ANSI N45.2-1971 "Quality Assurance Program Requirements For Nuclear Power Plants".

In 1976, Oak Ridge national Laboratory (ORNL) established a testing program in order to conduct Radiation, Decontamination, and DBA tests under one roof. Keeler & Long, under contract with ORNL, conducted a series of tests in compliance with the parameters established by a major engineering firm and the ANSI standards. These tests, and similar series of tests conducted two years later in 1978, became the basis for the qualification of several of our concrete and carbon steel coating systems. From 1978 to the present day we have continued to qualify through ORNL and several other independent testing agencies any modifications to existing formulas and any changes in surface preparation or application requirements. We have also maintained an in-house testing program used to screen new products as well as modifications of existing systems. Furthermore, progress has continued in the revision of the ANSI standards during this time frame. Revision of these documents is presently under the jurisdiction of the American Society for Testing and Materials (ASTM) as outlined in D3842-80 "Standard Guide for Selection of Test Methods for Coatings Used in Light-Water Nuclear Power Plants".

The future dictates significantly less construction of new Nuclear Plants and much more emphasis upon the repair and maintenance of existing facilities. Our commitment remains the same as it was in 1965; that is, to meet the coating requirements of Nuclear Power Plants.

NUCLEAR COATINGS

SSU-1

Level One Coating Systems

The following Coating Systems are qualified for Coating Service Level One of a Nuclear Power Plant. "Coating Service Level One pertains to those systems applied to structures, systems and other safety related components which are essential to the prevention of, or the mitigation of the consequences of postulated accidents that could cause undue risk to the health and safety of the public."

SYSTEM IDENTIFICATION	COATING SYSTEMS	DRY FILM THICKNESS RANGE
CARBON STEEL COATING SYSTEMS		
System S-1		
Primer	No. 6548/7107 EPOXY WHITE PRIMER	3.0 - 14.0 mils DFT
Finish	No. E-1 SERIES EPOXY ENAMEL	2.5 - 6.0 mils DFT
System S-10		
Primer	No. 6548/7107 EPOXY WHITE PRIMER	5.0 - 12.0 mils DFT
Finish	No. D-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 6.0 mils DFT
System S-11		
Primer/Finish	No. 6548/7107 EPOXY WHITE PRIMER	8.0 - 18.0 mils DFT
System S-12		
Primer/Finish	No. 4500 EPOXY SELF-PRIMING SURFACING ENAMEL	5.0 - 18.0 mils DFT
System S-14 (FLOORS ONLY)		
Finish	No. 5000 EPOXY SELF-LEVELING FLOOR COATING	10.0 - 25.0 mils DFT
System S-15		
Primer	No. 6548/7107 EPOXY WHITE PRIMER	2.5 - 6.0 mils DFT
Finish	No. 9600 N KEELOCK	5.0 - 8.0 mils DFT
CONCRETE COATING SYSTEMS		
System KL-2		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548-S EPOXY SURFACER	Flush - 50.0 mils DFT
Finish	No. E-1 SERIES EPOXY ENAMEL	2.5 - 6.0 mils DFT
System KL-8		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548-S EPOXY SURFACER	Flush - 50.0 mils DFT
Finish	No. D-1 SERIES EPOXY HI-BUILD ENAMEL	4.0 - 8.0 mils DFT
System KL-9		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548/7107 EPOXY WHITE PRIMER	5.0 - 10.0 mils DFT
Finish	No. D-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 8.0 mils DFT
System KL-10		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 4000 EPOXY SURFACER	Flush - 50.0 mils DFT
Finish	No. D-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 6.0 mils DFT
System KL-12		
Curing Compound/Sealer	No. 4129 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer/Finish	No. 4500 EPOXY SELF-PRIMING SURFACING ENAMEL	10.0 - 50.0 mils DFT
System KL-14 (FLOORS ONLY)		
Primer/Sealer	No. 6129 EPOXY CLEAR PRIMER/SEALER	1.5 - 2.5 mils DFT
Finish	No. 5000 EPOXY SELF-LEVELING FLOOR COATING	35.0 - 50.0 mils DFT

SUMMARY OF QUALIFICATION TEST RESULTS

KEELER & LONG maintains a complete file of Nuclear Test Reports which substantiate the specification of the carbon steel and concrete coating systems listed in this bulletin. This file was initiated in the early 1970's and provides complete qualification in accordance with ANSI Standards N512 and N101.2. Results for radiation tolerance, decontamination, and the Design Basis Accident Condition are reported as performed by independent Laboratories. Also reported are the chemical and physical tests which were conducted by the Keeler & Long Laboratory in compliance with the ANSI Standards.

TEST REPORT REFERENCE

K&L COATING SYSTEM	SUBSTRATE	KEELER & LONG TEST REPORT NO. 76-0728-1	78-0810-1	85-0404	85-0524	90-0227	93-0818	93-0601
S-1	Steel	*	*					
S-10	Steel		*					
S-11	Steel		*					
S-12	Steel			*				
S-14	Steel					*		
S-15	Steel						*	
KL-2	Concrete	*	*					
KL-8	Concrete	*						
KL-9	Concrete	*	*					
KL-10	Concrete		*					
KL-12	Concrete				*			
KL-14	Concrete					*		*

This information is presented as accurate and correct, in good faith, to assist the user in application. No warranty is expressed or implied. No liability is assumed.

KOLORS KEELER & LONG INC. SSPC STEEL STRUCTURES PAINTING COUNCIL SUSTAINING MEMBER

E.340

HEADQUARTERS:
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856 Echo Lake Road
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EPOXY ENAMEL
E-SERIES

GENERIC TYPE: POLYAMIDE EPOXY

PRODUCT DESCRIPTION: A two component, polyamide epoxy enamel formulated to provide excellent chemical resistance, as well as being extremely resistant to abrasion and direct impact, for interior exposures.

RECOMMENDED USES: As a topcoat for concrete and steel surfaces subject to radiation, decontamination, and loss-of-coolant accidents in Coating Service Level I Areas of nuclear power plants.

NOT RECOMMENDED FOR: Areas other than the above, as the J-SERIES can be utilized in Coating Service Level II and III Areas, as well as Balance of Plant, of nuclear power plants, with attendant cost savings.

COMPATIBLE UNDERCOATS: Epoxy White Primer
Epoxy Surfacer

PRODUCT CHARACTERISTICS:

Solids by Volume:	53% ± 3%
Solids by Weight:	66% ± 3%
Recommended Dry Film Thickness:	2.0 - 2.5 mils
Theoretical Coverage:	425 Sq. Ft./Gallon @ 2.0 mils DFT
Finish:	Full Gloss (E-1), Semi-Gloss (E-2)
Available Colors:	White, light tints, and dark red
Drying Time @ 72°F	
To Touch:	4 Hours
To Handle:	8 Hours
To Recoat:	48 Hours
VOC Content:	3.4 Pounds/Gallon 407 Grams/Liter

June, 1994

TECHNICAL BULLETIN

E-SERIES

E 340

TECHNICAL DATA

PHYSICAL DATA:	Weight per gallon:	10.2 ± 0.5 (pounds)
	Flash Point (Pensky-Martens):	85°F ± 2°
	Shelf Life:	1 Year
	Pot Life @ 72°F:	8 Hours
	Temperature Resistance:	350°F
	Viscosity @ 77°F:	85 ± 5 (Krebs Units)
	Gloss (60° meter):	95 ± 5 (E-1)
	Storage Temperature:	55 - 95°F
	Mixing Ratio (Approx. by Volume):	4:1
APPLICATION DATA:	Application Procedure Guide:	APG-2
	Wet Film Thickness Range:	4.0 - 5.0 mils
	Dry Film Thickness Range:	2.0 - 2.5 mils
	Temperature Range:	55 - 120°F
	Relative Humidity:	80% Maximum
	Substrate Temperature:	Dew Point + 5°F
	Minimum Surface Preparation:	Primed
	Induction Time @ 72°F:	1 Hour
	Recommended Solvent	
	@ 50 - 85°F:	No. 4093
	@ 86 - 120°F:	No. 2200

Application Methods

Air Spray	
Tip Size:	.055"
Pressure:	30 - 60 PSIG
Thin:	1.0 - 2.0 Pts/Gal
Airless Spray	
Tip Size:	.011" - .017"
Pressure:	2500 - 3000 PSIG
Thin:	0.5 - 1.5 Pts/Gal
Brush or Roller	
Thin:	1.0 - 2.0 Pts/Gal

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KOLORS

SUSTAINING MEMBER

3.8.3 Description of Electroless Nickel Coating

This section provides a description of the electroless Nickel coating process as prepared by the ASM Committee on Nickel Plating. The electroless Nickel coating is used to provide corrosion protection of the BWR carbon steel support disks during the short time period from placement of the BWR canister in the spent fuel pool to the time of completion of vacuum drying and inerting with helium. The coating is applied in accordance with ASTM B733-SC3, Type V, Class 1 [37].

Electroless nickel is a nickel/phosphorus alloy that is produced by the use of a chemical reducing agent a hot aqueous solution to deposit nickel on a catalytic surface without the use of an electric current. The chemical reduction process produces a uniform, predicable coating thickness. Adhesion of the nickel coating to properly cleaned carbon steel is excellent with reported bond strength in the range of 40 to 60 ksi [38].

Electroless nickel coating is highly corrosion resistant because of its non-porous structure that seals off the coated surface from the environment. During the time following completion of the coating of the UMS BWR support disk until actual use, the nickel surface bonds with oxygen atoms in the air to create a passive nickel oxide layer on the surfaces of the support disk. Thus, very few free electrons are available on the surface to cathodically react with water and produce hydrogen gas. Test data for electroless nickel coated steel have been reported to show corrosion rates from 1 to 2 μm per year in water [39].

The coating classification of SC3 provides a minimum thickness of 25 μm (0.001 inch).

Nonelectrolytic Nickel Plating

*By the ASM Committee on Nickel Plating**

THREE METHODS may be employed for depositing nickel coatings without the use of electric current:

1 Immersion plating
2 Chemical reduction of nickelous oxide at 1600 to 2000 F
3 Autocatalytic chemical reduction of nickel salts by hypophosphite anions in an aqueous bath at 190 to 205 F ("electroless" nickel plating).

All three methods are, under certain limited conditions, useful substitutes for nickel electroplating; they are particularly useful in applications in which electroplating is impracticable or impossible because of cost or technical difficulties. Of the three methods, electroless nickel plating is in widest use, and is the method to which the most attention is devoted in this article.

Immersion Plating

The composition and operating conditions of an aqueous immersion plating bath are as follows:

Nickel chloride ($NiCl_2 \cdot 6H_2O$)	80 oz per gal
Boric acid (H_3BO_3)	4 oz per gal
pH	3.5 to 4.5
Temperature	160 F

When using this bath, it is desirable, but not mandatory, to move the work at a rate of about 16 ft per min.

This solution is capable of depositing a very thin (about 0.025 mil) and uniform coating of nickel on steel in periods of up to 30 min. The coating is porous and possesses only moderate adhesion, but these conditions can be improved by heating the coated part at 1200 F for 45 min in a nonoxidizing atmosphere. (Higher temperatures will promote diffusion of the coating.)

High-Temperature Chemical-Reduction Coating

By the reduction of a mixture of nickelous oxide and dibasic ammonium phosphate in hydrogen or other reducing atmosphere at 1600 to 2000 F, a nickel coating can be deposited without the use of electric current. This method (U. S. Patent 2,633,631) consists of applying a slurry of the two chemicals to all or selected surfaces of the workpiece, drying the slurry in air, and performing the chemical reduction at elevated temperature. No special tanks or other plating facilities are required. Some diffusion of nickel and phosphorus into the basis metal occurs at elevated temperature; when the coating is applied to steel, it will consist of nickel, iron, and about 3% phosphorus. The slurry may be used for brazing.

Electroless Nickel Plating

The electroless nickel plating process employs a chemical reducing agent (sodium hypophosphite) to reduce a nickel salt (such as nickel chloride) in hot aqueous solution and to deposit nickel on a catalytic surface. The deposit obtained from an electroless nickel solution is an alloy containing from 4 to 12% phosphorus and is quite hard. (As indicated later in this article, the hardness of the as-plated deposit can be increased by heat treatment.) Because the deposit is not dependent on current distribution, it is uniform in thickness, regardless of the shape or size of the plated surface.

Electroless nickel deposits may be applied to provide the basis metal with resistance to corrosion or wear, or for the buildup of worn areas. Typical applications of electroless nickel for these purposes are given in Table 1, which also indicates plate thicknesses and postplating heat treatments.

Surface Cleaning. In general, the methods employed for cleaning and preparing metal surfaces for electroless nickel plating are the same as those used for conventional electroplating. Heavy oxides are removed mechanically, and oils and grease are removed by vapor degreasing. A typical precleaning cycle might consist of alkaline cleaning (either agitated soak or anodic) and acid pickling, both followed by water rinsing.

Prior to electroless plating, the surfaces of all stainless steel parts must be chemically activated in order to obtain satisfactory adhesion of the plate. One activating treatment consists of immersing the work for about 3 min in a hot (200 F) solution containing equal volumes of water and concentrated sulfuric acid. Another treatment consists of immersing the work for 2 to 3 min in the following solution at 160 F:

Sulfuric acid (66° Bé)	25% by volume
Hydrochloric acid (18° Bé)	5% by volume
Ferric chloride hexahydrate	0.53 oz per gal

Pretreatments that are unique to electroless nickel plating include:

1 A strike copper plate must be applied to parts made of or containing lead, tin, cadmium or zinc, to insure adequate coverage and to prevent contamination of the electroless solution.
2 Massive parts are preheated to bath temperature to avoid delay in the deposition of nickel from the hot electroless bath.

Bath Characteristics. A simplified equation that describes the formation of electroless nickel deposits is:

$$NiSO_4 + NaH_2PO_2 + H_2O \xrightarrow[\text{catalyst}]{\text{heat}} Ni + NaH_2PO_3 + H_2SO_4$$

The essential requirements for any electroless nickel solution are:

1 A salt to supply the nickel
2 A hypophosphite salt to provide chemical reduction
3 Water
4 A complexing agent
5 A buffer to control pH
6 Heat
7 A catalytic *surface to be plated.*

Detailed discussions of the chemical characteristics of electroless baths, and of the critical concentration limits of the various reactants, can be found in several of the references listed at the end of this article.

Both alkaline (pH, 7.5 to 10) and acid (pH, 4.5 to 6) *electroless nickel* baths are used in industrial production. Although the acid baths are easier to maintain and are more widely used, the *alkaline baths are reported to have* greater compatibility with sensitive substrates (such as magnesium, silicon and aluminum).

Catalysis. *Nickel and hypophosphite* ions can exist together in a dilute solution without interaction, but will react on a catalytic surface to form a deposit. Furthermore, the surface of the deposit is also catalytic to the reaction, so that the catalytic process continues until any reasonable plate thickness is applied. This autocatalytic effect is the principle upon which all electroless nickel solutions are based.

Metals that catalyze the plating reaction are members of group VIII in the periodic table, which group includes nickel, cobalt and palladium. A deposit will begin to form on surfaces of these metals by simple contact with the solution. Other metals, such as aluminum or low-alloy steel, first form an

* See page 432 for committee list.

Table 1. Typical Applications of Electroless Nickel Plating

Part and basis metal	Typical plate thickness, mils	Postplating heat treatment(a)
Plate Applied for Corrosion Resistance		
Valve body, cast iron	5.0	None
Printing rolls, cast iron	1.0	None
Electronic chassis, 1010 steel	1.0	None
Railroad tank cars, 1020 steel	3.5	1 hr at 1150 F
Reactor vessels, 1020 steel	4.0	1 hr at 1150 F
Pressure vessel, 4130 steel	1.5	3 hr at 350 F
Tubular shaft, 4340 steel	1.5	3 hr at 375 F
Plate Applied for Wear Resistance		
Centrifugal pump, steel	1.0	2 hr at 400 F
Plastic extrusion dies, steel	2.0	2 hr at 375 F
Printing-press bed, steel	1.0	None
Valve inserts, steel	0.5	2 hr at 1150 F
Hydraulic pistons, 4340 steel	1.0	1 hr at 750 F
Screws, 410 stainless	0.2	None
Stator and rotor blades, 410 stainless	0.8 to 1.0	1 hr at 750 F
Spray nozzles, brass	0.5	None
Plate Applied for Buildup of Worn Areas		
Carburized gear (bearing journal)	0.8 to 1.0	5 hr at 275 F
Splined shaft (ID spline), 16-25-6 stainless	0.5	1 hr at 750 F
Connecting arm (dowel-pin holes), type 410	5.0	1 hr at 750 F

(a) Heat treatments above 450 F should be carried out in an inert or reducing atmosphere.

immersion deposit of nickel on their surfaces, which then catalyzes the reaction; still others, such as copper, require a galvanic nickel deposit in order to be plated. Such a galvanic nickel deposit can be formed by the plating solution itself, if the copper is in contact with steel or aluminum.

Plastics, glass, ceramics and other nonmetallics also can be plated, if their surfaces can be made catalytic. This usually is done by the application of traces of a strongly catalytic metal to the nonmetallic surface by chemical or mechanical means.

There is, however, a group of metals that not only do not display any catalytic action, but also interfere with all plating activity. The salts of these metals, if dissolved in a solution even in comparatively small amounts, are poisons and stop the plating reaction on all metals, thus necessitating the discarding of the solution and the formulation of a new one. Examples of these anticatalysts are Pb, Sn, Zn, Cd, Sb, As and Mo.

Paradoxically, the deliberate introduction of extremely minute traces of poisons has been practiced by a number of users of electroless nickel, with the intent of stabilizing the solution. Being an inherently metastable mixture, electroless nickel solutions are likely to decompose spontaneously, with the nickel and hypophosphite reacting on trace amounts of solid impurities present in any plating bath. In order to minimize this problem, a poisoning element is added in trace concentrations of parts per million (or per trillion) to the original make-up of the solution. The poison is adsorbed on the solid impurities in quantities large enough to destroy their catalytic nature. This selective adsorption on catalytic centers decreases the concentration of the catalytic poison to a level below the critical threshold, so that normal deposition of nickel is not impeded, although the rate of deposition is somewhat reduced. The deliberate introduction of catalytic poisons for the purpose of stabilization is covered by several patents, including U. S. Patents 2,762,723 and 2,847,327.

Table 2. Alkaline Electroless Nickel Baths

Constituent or condition	Bath 1	Bath 2	Bath 3
Composition, Grams per Liter			
Nickel chloride	30	45	30
Sodium hypophosphite	10	11	10
Ammonium chloride	50	50	50
Sodium citrate	..	100	..
Ammonium citrate	..	..	65
Ammonium hydroxide	to pH	to pH	to pH
Operating Conditions			
pH	8 to 10	8.5 to 10	8 to 10
Temperature, F	195 to 205	195 to 205	195 to 205
Plating rate (approx), mil per hr	0.3	0.4	0.3

Alkaline Baths. Most alkaline baths in commercial use today are based on the original formulations developed by Brenner and Riddell. They contain a nickel salt, sodium hypophosphite, ammonium hydroxide, and an ammonium salt; they may also contain sodium citrate or ammonium citrate. The ammonium salt serves to complex the nickel and buffer the solution. Ammonium hydroxide is used to maintain the pH between 7.5 and 10. Table 2 gives the compositions and operating conditions of three alkaline electroless baths.

At the operating temperatures of these baths (about 200 F), ammonia losses are considerable. Thorough ventilation and frequent adjustment of pH are required. The alkaline solutions are inherently unstable and are particularly sensitive to the poisoning effects of anticatalysts such as lead, tin, zinc, cadmium, antimony, arsenic and molybdenum — even when these elements are present in only trace quantities. However, when depletion occurs, these solutions undergo a definite color change from blue to green, indicating the need for addition of ammonium hydroxide.

Acid baths are more widely used in commercial installations than alkaline baths. Essentially, acid baths contain a nickel salt, a hypophosphite salt, and a buffer; some solutions also contain a chelating agent. Frequently, wetting agents and stabilizers also are added.

These baths are more stable than alkaline solutions, are easier to control, and usually provide a higher plating rate. Except for the evaporation of water, there is no loss of chemicals when acid baths are heated to their operating range. Table 3 gives the compositions and operating conditions of several acid electroless baths.

Solution Control. In order to assure optimum results and consistent plating rates, the composition of the plating solution should be kept relatively constant; this requires periodic analyses for the determination of pH, nickel content, and phosphite and hypophosphite concentrations. The rate at which these analyses should be made depends on the quantity of work being plated and the volume and type of solution being used. The following methods have been employed:

pH — Standard electrometric method

Nickel — Any one of the colorimetric, gravimetric or volumetric methods is satisfactory; the cyanide method is probably the most popular.

Phosphite — A 10-ml sample of the plating solution is combined with 20 ml of a 5% solution of sodium bicarbonate and cooled in an ice bath. Next, 50 ml of 0.1*N* iodine solution is added and the flask containing this mixture is stoppered and permitted to stand for 2 hr at room temperature. Then the flask is cooled for 15 min in ice water, after which it is unstoppered, the mixture is acidified with acetic acid, and the excess iodine is titrated with 0.1*N* sodium thiosulfate, with starch as an indicator. Determination is then made as follows:

$$NaH_2PO_3 \text{ per liter} = \frac{\text{net ml of } 0.1N \text{ iodine} \times 6.3}{\text{ml of plating solution}}$$

Hypophosphite (U. S. Patent 2,697,651) — A 25-ml sample of the plating solution is diluted to 1 liter. A 5-ml aliquot of the

Table 3. Acid Electroless Nickel Plating Baths(a)

Constituent or condition	Bath 4	Bath 5	Bath 6	Bath 7	Bath 8	Bath 9
Composition, Grams per Liter						
Nickel chloride	30	..	..	30	..	30
Nickel sulfate	..	21	20	..	15	..
Sodium hypophosphite	10	24	27	10	14	12
Sodium acetate	..	..	..	..	13	..
Sodium hydroxyacetate	50	..	..	10	..	..
Sodium succinate	..	..	16	..	..	..
Lactic acid (80%)	..	34 ml	..	..	..	..
Propionic acid (100%)	..	2.2 ml	..	..	..	10
Operating Conditions						
pH	4 to 6	4.3 to 4.6	4.5 to 5.5	4 to 6	5 to 6	4.5 to 5.5
Temperature, F	190 to 210	203	200 to 210	190 to 210	190 to 210	190 to 210
Plating rate (approx), mil per hr	0.5	1.0	1.0	0.4	0.7	0.6

(a) Baths 4 and 7 are covered by U. S. Patent 2,532,283 (a public patent assigned to the National Bureau of Standards); bath 5, by U. S. Patents 2,822,293 and 2,822,294, and bath 6 by U. S. Patents 2,658,841 and 2,658,842.

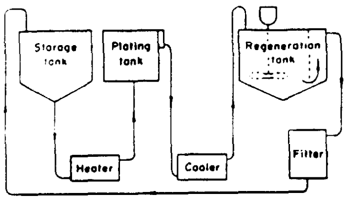


Fig. 1. Schematic of continuous-type system for electroless nickel plating. See text.

dilution is combined with 10 ml of a 10% solution of ammonium molybdate and 10 ml of fresh 6% sulfurous acid. The sample is covered and heated to boiling, and a deep blue color develops. The sample is cooled and diluted to 100 ml, and transmittance at a wave length of 440 microns is determined. The calibration curve on semilog paper is linear.

Hypophosphite (alternative method) — A 5-ml sample of the plating solution is mixed in a beaker with 5 ml of methyl orange solution made up of 1 gram of methyl orange in 1 liter of water. In another beaker is placed 15 ml of an acid solution made up by (*a*) dissolving 40 grams of sodium metabisulfate in 200 ml of water, (*b*) slowly adding the sodium metabisulfate solution to a cold solution of 82 ml of sulfuric acid in 650 ml of water, and then (*c*) diluting this mixture with water to 1 liter. When the acid solution and the solution containing the sample and methyl orange reach a temperature of 77 F in a thermostat, the two solutions are mixed. The time between mixing and the disappearance of the red color is recorded. The hypophosphite concentration is a function of this time and is read from a concentration-time curve made from known standards.

Equipment Requirements. The precleaning and post-treating equipment for an electroless nickel line is comparable to that employed in conventional electrodeposition. The plating tank itself, however, is unique.

The preferred plating tank for batch operations is constructed of stainless steel or aluminum and is lined with a coating of an inert material, such as tetrafluoroethylene or a phenolic-base organic. The size and shape of the tank are usually dictated by the parts to be plated, but the surface area of the plating solution should not be so large that excessive heat loss occurs as a result of evaporation.

A large heat-transfer area and a low temperature gradient are necessary between the heating medium and the plating solution. This combination provides for a reasonable heat-up time without local hot spots that could decompose the solution. It is accepted practice to surround the plating tank with a hot-water jacket or to immerse it in a tank containing hot water. Heating jackets using low-pressure steam also have been used successfully. The use of immersed steam coils is not favored, however, because it entails the sacrifice of a large amount of working area in the tank.

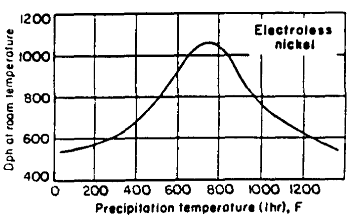
Accessory equipment required or recommended for the tank includes:

1 An accurate temperature controller
2 A filter to remove any suspended solids
3 A pH meter
4 An agitator to prevent gas streaking
5 On small tanks, a cover, to minimize heat loss and exclude foreign particles.
6 On large tanks, a separate small tank to dissolve and filter additives before they are put into the plating tank.

Considerably more equipment is required for a continuous-type system, such as that shown in Fig. 1. The bath is prepared and stored in a separate tank and flows through a heater (which raises its temperature to 205 F) into the plating tank. From the plating tank, the solution is pumped through a cooler, which decreases its temperature to 175 F or below, and then to an agitated regeneration tank, where reagents are added in controlled amounts to restore the solution to its original composition. The solution is then directed past a vertical underflow baffle and out of the regeneration tank to a filter, and then returned to storage.

In externally heated continuous-type systems such as the one shown in Fig. 1, the plating tank and other components of the system that come in contact with the plating solution are constructed of type 304 stainless steel and are not lined or coated; these components are periodically deactivated by chemical treatment. Details of this type of system are covered by several patents, including U. S. Patents 2,941,902; 2,658,839 and 2,874,073.

Properties of the Deposit. Electroless nickel is a hard, lamellar, brittle, uniform deposit. As plated, the hardness



Effect of temperature of 1-hr precipitation heat treatment on room-temperature hardness of a typical electroless nickel deposit (Eberbach tester, 100-gram load). Above 450 F, heat treatment was in an inert atmosphere.

Fig. 2. Heat treatment of coating

varies over a considerable range (425 to 575 dph), depending primarily on phosphorus content, which ranges from 4 to 12%. This hardness can be increased by a precipitation heat treatment. As indicated in Fig. 2, which shows temperature-hardness relationships for a typical deposit, by heating at 750 F for ½ to 1 hr, hardness can be increased to about 1000 dph.

The corrosion resistance of electroless nickel deposits is superior to that of electrodeposited nickel of comparable thickness, but this superiority varies with exposure conditions. Outdoor exposure and salt spray corrosion data indicate that about 25% more resistance is given a steel panel by electroless nickel than by electrolytic.

Table 4. Physical Properties of Electroless Nickel Deposits

Property	Value
Specific gravity	7.8 to 8.5
Melting point	1635 to 1850 F
Electrical resistivity	60 microhm-cm
Thermal expansion	13 × 10⁻⁶ per °C
Thermal conductivity	0.0105 to 0.0135 cal/cm sec/°C

Table 5. Costs for Electroless Nickel Plating (Example 2) (a)

Cost factor	Cost per year(b)
Original investment	$18,000
Fixed costs:	
Depreciation (10 years)	$ 1,800
Insurance	450
Floor space (200 sq ft)	192
Repairs and maintenance	450
Variable costs:	
Raw material	6,100
Utilities	740
Labor costs:	
Direct	10,400
Indirect	2,630
Total	$22,762
Total cost per hr	$9.48
Total cost per sq ft coated to 1 mil	$1.00

(a) Exclusive of costs for: overhead and administration; racking, cleaning and unracking; and preplating and postplating processes. (b) Based on deposition of 1 mil on 0.1-sq-ft parts at rate of 0.8 mil per hr (capacity: 117 pieces, or 9.4 sq-ft/mil, per hr), on a schedule of 10 hr per day, 20 days per month, 2400 hr per year.

Some of the physical properties of electroless nickel are listed in Table 4.

Advantages and Limitations. Some advantages of electroless nickel are:

1 Good resistance to corrosion and wear
2 Excellent uniformity
3 Solderability and brazability
4 Good oxidation resistance.

Limitations of electroless nickel are:

1 High cost
2 Brittleness
3 Poor welding characteristics
4 Lead, tin, cadmium and zinc must be copper strike plated before electroless nickel can be applied
5 Slower plating rate (in general), as compared to electrolytic methods
6 Full brightness in deposit cannot be obtained without extreme brittleness.

Cost. Electroless nickel is considerably more expensive than electrodeposited nickel. Actual costs for electroless nickel plating, as reported by two users, are given in the following examples.

Example 1. Based on the experience of one manufacturing plant, it costs $1.20 to deposit an electroless nickel coating 1 mil thick on a square foot of surface area: 37¢ for chemicals, 59¢ for labor, and 24¢ for equipment and maintenance.

Example 2. Another manufacturing plant reports that it costs $1 per sq ft to plate a 1-mil thickness of electroless nickel on specific parts with a surface area of 0.1 sq ft, on the basis of data obtained over a one-year period (2400 working hours). An analysis of their costs is given in Table 5.

Selected References

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G. Gutzeit, Industrial Nickel Coating by Chemical Catalytic Reduction, *Trans Inst Metal Finishing*, 33, 383-423 (1955-1956), and *Corrosion Technol*, 3, 208 (1956)

G. Gutzeit, An Outline of the Chemistry Involved in the Process of Catalytic Nickel Deposition from Aqueous Solution, *Plating*, Oct 1959, p 1158-1164; Nov 1959, p 1275-1278; Dec 1959, p 1377-1378; Jan 1960, p 63-70

C. H. de Minjer and A. Brenner, Studies on Electroless Nickel Plating, *Plating*, December 1957, p 1297-1305

Symposium on Electroless Nickel Plating (Catalytic Deposition of Nickel-Phosphorus Alloys by Chemical Reduction in Aqueous Solution), ASTM STP No. 265 (1959)

3.8.4 Keeler & Long Kolor-Poxy Primer No. 3200

E.140

KEELER & LONG INC

HEADQUARTERS:
P. O. Box 460
856 Echo Lake Road
Watertown, CT 06795
Tel (860) 274-6701
Fax (860) 274-5857

KOLOR-POXY PRIMER
No. 3200

GENERIC TYPE: POLYAMIDE EPOXY

PRODUCT DESCRIPTION: A two component, high solids, polyamide epoxy primer/topcoat formulated to provide a high-build; abrasion, impact and chemical resistant coating.

RECOMMENDED USES: As a high-build primer for steel and concrete surfaces exposed to a wide range of conditions. No. 3200 is certified by the National Sanitation Foundation (NSF) and Ministry of Environment (Ontario and Saskatchewan, CN)** for application to the interior of potable water tanks.* No. 3200 is also accepted by the USDA for application to incidental food contact surfaces.

NOT RECOMMENDED FOR: Immersion in strong acids.

COMPATIBLE TOPCOATS:

Kolor-Poxy Primers and Enamels	Kolor-Sil Enamels
Kolor-Poxy Hi-Solids Primer	Acrythane Enamels
Kolor-Poxy Hi-Build Enamels	Kolorane Enamels
Poly-Silicone Enamels	Tri-Polar Silicone Enamels
Hydro-Poxy Enamels	

PRODUCT CHARACTERISTICS:

Solids by Volume:	66% ± 3%
Solids by Weight:	82% ± 3%
Recommended Dry Film Thickness:	2.5 - 6.0 mils
Theoretical Coverage:	350 Sq. Ft./Gallon @ 3.0 mils DFT
Finish:	Flat
Available Colors:	White and tints
Drying Time @ 72°F	
To Touch:	4 Hours
To Handle:	8 Hours
To Recoat:	24 Hours
To Immersion:	10 Days
VOC Content:	2.52 Pounds/Gallon
	302 Grams/Liter

*
White or light gray only
5000 gallon tanks or larger
Up to four coats - Total DFT 24 mils maximum
Use No. 3700 Thinner up to 25% by volume

** Substrate temperature; 45°F (70°C) minimum during cure. Thorough rinse required after final cure.

June, 1994

TECHNICAL BULLETIN

No. 3200 E-140

TECHNICAL DATA

PHYSICAL DATA:	Weight per gallon:	13.6 ± 0.5 (pounds)
	Flash Point (Pensky-Martens):	85°F
	Shelf Life:	2 Years
	Pot Life @ 72°F:	8 Hours
	Temperature Resistance:	350°F
	Viscosity @ 77°F:	87 ± 5 (Krebs Units)
	Gloss (60° meter):	6 ± 5
	Storage Temperature:	50 - 95°F
	Mixing Ratio (Approx. by Volume):	4:1
APPLICATION DATA:	Application Procedure Guide:	APG-3
	Wet Film Thickness Range:	3.8 - 9.1 mils
	Dry Film Thickness Range:	2.5 - 6.0 mils
	Temperature Range:	50 - 120°F
	Relative Humidity:	80% Maximum
	Substrate Temperature:	Dew Point + 5°F
	Minimum Surface Preparation:	SSPC-SP6, SP10, SP5
	Induction Time @ 72°F:	45 Minutes
	Recommended Solvent	
	@ 50 - 85°F:	No. 3700
	@ 86 - 120°F:	No. 2200

Application Methods

Air Spray	
Tip Size:	.055" - .073"
Pressure:	30 - 60 PSIG
Thin:	1.0 - 2.0 Pts/Gal
Airless Spray	
Tip Size:	.015" - .019"
Pressure:	2500 PSIG
Thin:	0.5 - 1.5 Pts/Gal
Brush or Roller	
Thin:	0.5 - 1.5 Pts/Gal

KEELER & LONG INC.

P. O. Box 460, 856 Echo Lake Road
Watertown, CT 06795
Tel: (860) 274-6701 Fax: (860) 274-5857

KOLORS

SSPC SUSTAINING MEMBER

3.8.5 Acrythane Enamel Y-1 Series Top Coating

U.150

KEELER & LONG INC

HEADQUARTERS:
P. O. Box 460
856 Echo Lake Road
Watertown, CT 06795
Tel (860) 274-6701
Fax (860) 274-5857

ACRYTHANE ENAMEL
Y-1-SERIES

GENERIC TYPE: ACRYLIC URETHANE

PRODUCT DESCRIPTION: A two component, acrylic urethane high-gloss enamel formulated to provide maximum appearance and protective qualities when exposed to an exterior environment. It produces the ultimate in long term color and gloss retention.

RECOMMENDED USES: As a topcoat for exterior structural steel, tanks, piping, conveyors, equipment, and other similar surfaces, as well as interior and exterior concrete surfaces.

NOT RECOMMENDED FOR: Immersion service; splash and spillage of strong acids and alkalies.

COMPATIBLE UNDERCOATS:
Kolorane Aluminum Primer
Kolorane Zinc Rich Primer
Kolor-Poxy Primers and Enamels
Kolor-Poxy Hi-Solids Primer
Acrythane Intermediate Primer
Kolor-Poxy Surfacer

PRODUCT CHARACTERISTICS:

Solids by Volume:	52% ± 5%
Solids by Weight:	67% ± 5%
Recommended Dry Film Thickness:	2.0 - 4.0 mils
Theoretical Coverage:	278 Sq. Ft./Gallon @ 3.0 mils DFT
Finish:	Full Gloss
Available Colors:	Unlimited
Drying Time @ 72°F	
To Touch:	6 Hours
To Handle:	12 Hours
To Recoat:	24 Hours
VOC Content:	< 3.5 Pounds/Gallon < 420 Grams/Liter

June, 1995

TECHNICAL BULLETIN

Y-SERIES U.150

TECHNICAL DATA

PHYSICAL DATA:

Weight per gallon:	10.5 ± 0.5 (pounds)
Flash Point (Pensky-Martens):	85°F
Shelf Life:	1 Year
Pot Life @ 72°F:	6 Hours
Temperature Resistance:	250°F
Viscosity @ 77°F:	75 ± 5 (Krebs Units)
Gloss (60° meter):	90 ± 5 (Y-1)
Storage Temperature:	45 - 95°F
Mixing Ratio (Approx. by Volume):	4.2:1 (White only)

APPLICATION DATA:

Application Procedure Guide:	APG-5
Wet Film Thickness Range:	3.5 - 7.0 mils
Dry Film Thickness Range:	2.0 - 4.0 mils
Temperature Range:	45 - 100°F
Relative Humidity:	80% Maximum
Substrate Temperature:	Dew Point + 5°F
Minimum Surface Preparation:	Primed
Induction Time @ 72°F:	None
Recommended Solvent	
@ 45 - 85°F:	No. 1200
@ 86 - 100°F:	No. 0700

Application Methods

Air Spray	
Tip Size:	.055"
Pressure:	30 - 60 PSIG
Thin:	0.5 - 2.0 Pts/Gal
Airless Spray	
Tip Size:	.011" - .015"
Pressure:	2000 - 2500 PSIG
Thin:	0.0 - 1.5 Pts/Gal
Brush or Roller	Recommended only with limitations
Thin (No. 0700):	0.5 - 1.5 Pts/Gal

KEELER & LONG INC.
P. O. Box 460, 856 Echo Lake Road
Watertown, CT 06795
Tel: (860) 274-6701 Fax: (860) 274-5857

KOLORS

SSPC STEEL STRUCTURES PAINTING COUNCIL
SUSTAINING MEMBER

3.8.6 PPG METALHIDE® 97-694 Series Primer

PPG METALHIDE® 97-694 Series

HPC/Industrial Maintenance

METALHIDE® 2000 Inorganic Zinc Rich Coating

GENERAL DESCRIPTION

Heavy duty corrosion resistant primer for ferrous metal surfaces in industrial environments. Provides galvanic protection similar to galvanizing. Particularly suited as a lining for the interior, and as a primer to be topcoated for the exterior of tanks containing organic solvents, gasoline, and other fuels. It is also excellent for application in coastal, marine, and other offshore environments.

RECOMMENDED USES

Ferrous Metal

FEATURES AND BENEFITS

Provides galvanic corrosion protection
Excellent resistance to organic solvents
Can be handled with slings in 5-6 hours (77°F at 50% relative humidity)
Class B Slip Coefficent under ASTM A-325

PACKAGING

1-Gallon (3.78L)
3-Gallon (11.3L)
5-Gallon (18.9L)

Not all products are available in all sizes. Not all containers are full-filled.

TINTING AND BASE INFORMATION

97-694	Liquid Component A - Red
97-695	Liquid Component A - Green
97-697P	Powder Component

DO NOT TINT.

PRODUCT DATA

PRODUCT TYPE: Inorganic self-curing ethyl silicate-metallic zinc
GLOSS: Matte
VOC*: 3.88 lbs./gal. (466 g/L)
COVERAGE: 330 to 500 sq. ft./gal. (31 to 46 sq.m/3.78L)

WEIGHT/GALLON*: 20.3 lbs. (9.2 kg) +/- 0.3 lbs. (136 g)
WEIGHT SOLIDS*: 80.3% +/- 2%
Results will vary by color, thinning and other additives.
*Product data calculated on mixed 97-695/97-697P.
Dry Film Thickness*: 2 to 5 mils not to exceed 8 mils on spot readings

POT LIFE: 16 hours
MIX RATIO: Mix as packaged. See mixing instructions.
IN SERVICE TEMPERATURE: 750°F (399°C) Dry heat; 140°F (60°C) Wet heat
DRYING TIME@ 77°F (25°C); 50% relative humidity.
To Touch: 15 minutes
To Handle: 4 hours
To Recoat: 24 hours
Drying times listed may vary depending on temperature, humidity, color and air movement.

CLEAN UP: 97-727 PPG Thinner
FLASH POINT: 97-695 60°F (15.6°C)

I23

METALHIDE® 97-694 Series

METALHIDE® 2000 Inorganic Zinc Rich Coating HPC/Industrial Maintenance

GENERAL SURFACE PREPARATION

Remove all paint, mill scale, and rust. The surface to be coated must be dimensionally stable, dry, clean, and free of oil, grease, and other foreign materials. WARNING! If you scrape, sand, or remove old paint, you may release lead dust or fumes. LEAD IS TOXIC. EXPOSURE TO LEAD DUST OR FUMES CAN CAUSE SERIOUS ILLNESS, SUCH AS BRAIN DAMAGE, ESPECIALLY IN CHILDREN. PREGNANT WOMEN SHOULD ALSO AVOID EXPOSURE. Wear a properly fitted NIOSH-approved respirator and prevent skin contact to control lead exposure. Clean up carefully with a HEPA vacuum and a wet mop. Before you start, find out how to protect yourself and your family by contacting the USEPA National Lead Information Hotline at 1-800-424-LEAD or log on to www.epa.gov/lead. In Canada contact a regional Health Canada office. Follow these instructions to control exposure to other hazardous substances that may be released during surface preparation.

STEEL: Non-Immersion Service -- The minimum surface preparation for ferrous metal substrates is SSPC-SP6 Commercial Blast cleaning. Service life of coating is in direct proportion to surface preparation. Immersion Service -- Near White Metal Blast SSPC-SP10 is mandatory for ferrous metals. The surface to be coated must be clean, dry, and well prepared to receive the coating. For specific recommendations, see your PITTSBURGH® Paints dealer or call 1-800-441-9695.

RECOMMENDED PRIMERS

Self priming on properly prepared surfaces.

MIXING AND APPLICATIONS INFORMATION

MIXING INSTRUCTIONS: Mix the 97-694 or 695 opaque liquid base using a mechanical mixer until no pigment remains at the bottom of the container. Transfer to a large container to facilitate mixing, and slowly sift in the zinc dust, 97-697P under continuous agitation. Mix until blend is uniform and free of lumps. Strain through a 30-60 mesh screen. DO NOT MIX IN REVERSE ORDER. Maintain constant agitation during use to prevent zinc dust from settling. The liquid component and the mixed paint must be protected from moisture. Relatively small amounts of contamination will cause gelation.

Changes in application equipment, pressures and/or tip sizes may be required on ambient temperatures and application conditions.

Airless Spray: Pressure 1500 psi, tip 0.017" - 0.021" Filter: 30 mesh

Conventional Spray: Fluid Nozzle: DeVilbiss MBC-510 gun, with 64 air cap with E tip and needle, or comparable equipment. Atomization Pressure: 55 - 70 Fluid Pressure: Can not specify, dependent on numerous factors.

Spray equipment must be handled with due care and in accordance with manufacturer's recommendation. High-pressure injection of coatings into the skin by airless equipment may cause serious injury.

Brush: Not recommended

Roller: Not recommended

Thinning: Thinning not normally required. If thinning is desired do not thin more than 12% with 97-727.

MIXING AND APPLICATIONS INFORMATION (cont.)

Permissible temperatures during application:

Material:	50 to 90°F	10 to 32°C
Ambient:	0 to 100°F	-18 to 38°C
Substrate:	0 to 140°F	-18 to 60°C

LIMITATIONS OF USE

Apply in good weather when air and surface temperatures are between 50°F (10°C) and 100°F (37.8°C) with maximum relative humidity of 85%. Optimum paint temperatures is 70°F (21°C) - 80°F (26.7°C). Surface temperatures must be at least 5°F (3°C) above the dew point. Dew or rain on product while uncured may cause surface to blush and brown and may impair its cure and intercoat adhesion. Do not expose container to temperatures greater than 135°F (57°C). While this product will lose gloss and chalk on exterior exposure, film integrity is not adversely effected. Do not use for potable water. For Professional Use Only; Not Intended for Household Use.

SAFETY

Proper safety procedures should be followed at all times while handling this product. USE WITH ADEQUATE VENTILATION. KEEP OUT OF REACH OF CHILDREN. Explosion-proof equipment must be used when coating with these materials in confined areas. Keep containers closed and away from heat, sparks, and flames when in use. Spray equipment must be handled with due care and in accordance with manufacturer's recommendation. High-pressure injection of coatings into the skin by airless equipment may cause serious injury. Read all label and Material Safety Data Sheet for important health/safety information prior to use. MSDS are available through our website www.ppghpc.com or by calling 1-800-441-9695.

PPG Industries, Inc.
Architectural Coatings
One PPG Place
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Technical Services
1-800-441-9695
1-888-807-5123 fax

Architect/Specifier
1-800-PPG-IDEA

PPG Architectural Finishes, Inc.
400 S. 13th Street
Louisville, KY 40203

PPG Canada, Inc.
Architectural Coatings
4 Kenview Blvd
Brampton, ON L6T 5E4

I23 2/2009
Supersedes (3/2008)

3.8.7 PPG PITT-THERM® 97-724 Series Top Coating

PPG **PITT-THERM®** **97-724 Series**

HPC/Industrial Maintenance **PITT-THERM® High Heat & Stress Corrosion Coating**

Generic Type

Air Dry Silicone, One Component

General Description

This coating is intended for use on austenitic stainless and carbon steel to provide protection against chloride attack and stress corrosion cracking on both insulated and uninsulated surfaces. PITT-THERM® has excellent thermal shock and barrier properties, and may be used as a heat resistant coating for carbon steel.

Recommended Uses

Austenitic Stainless Steel
Carbon Steel

Features / Benefits

High heat and thermal stress resistance.
Protects stainless steel against chloride attack and stress corrosion cracking.

Limitations of Use

For Professional Use Only; Not Intended for Household Use. Apply only when air, product and surface temperatures are 40°F (4.4°C) and when surface temperature is at least 5°F (3°C) above the dew point. Avoid exterior painting late in the day when dew or condensation are likely to form, or when rain is threatening. Special attention should be given to insure that this product is not contaminated by moisture during the application process. Drying times listed may vary depending on temperature, humidity, color and air movement.

Tinting and Base Information

97-724	Black
UC59571	Gray

Product Data

Gloss: Matte
VOC*: 4.62 lbs/gal 554.00 g/L
Coverage: 279 to 372 sq ft/gal (26 to 35 sq. m/3.78L)
Note: Does not include loss due to varying application method, surface porosity, or mixing.
DFT: 1.5 minimum to 2.0 maximum
Weight/Gallon*: 9.6 lbs. (4.5 kg) +/- 0.2 lbs. (91 g)
Volume Solids*: 34.8% +/- 2%
Weight Solids*: 52.1% +/- 2%
Clean-up: 97-727 PPG Xylol Thinner
Results will vary by color, thinning and other additives.
***Product data calculated on full formula.**

Drying Time:

To Touch:	20 minutes
To Handle:	2 hours
To Recoat:	16 hours

Dry Time @77°F (25°C); 50% relative humidity

In Service Temperature:
Dry Heat (F): 850° **Dry Heat (C):** 454°

Flash Point: 62°F, (16.7°C)

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PITT-THERM® 97-724 Series

HPC/Industrial Maintenance **PITT-THERM® High Heat & Stress Corrosion Coating**

General Surface Preparation

Remove all loose paint, mill scale, and rust. The surface to be coated must be dimensionally stable, dry, clean, and free of oil, grease, and other foreign materials. Service life of coating is in direct proportion to surface preparation. WARNING! If you scrape, sand, or remove old paint, you may release lead dust or fumes. LEAD IS TOXIC. EXPOSURE TO LEAD DUST OR FUMES CAN CAUSE SERIOUS ILLNESS, SUCH AS BRAIN DAMAGE, ESPECIALLY IN CHILDREN. PREGNANT WOMEN SHOULD ALSO AVOID EXPOSURE. Wear a properly fitted NIOSH-approved respirator and prevent skin contact to control lead exposure. Clean up carefully with a HEPA vacuum and a wet mop. Before you start, find out how to protect yourself and your family by contacting the USEPA National Lead Information Hotline at 1-800-424-LEAD or log on to www.epa.gov/lead. In Canada contact a regional Health Canada office. Follow these instructions to control exposure to other hazardous substances that may be released during surface preparation.
For application to Austenitic Stainless Steel SSPC-SP1 Solvent Wash is the minimum surface preparation. For Carbon Steel applications, SSPC-SP10 Near White Metal Blast is required. Where appropriate bare areas should be primed with a suitable primer.
HPC Systems in Detail Brochure (H10788) COATING SYSTEMS: 225-HD, 226-HD, 227-HD

Recommended Primers

none	**Refer to HD Coating Systems.**
Steel	Self Priming, 97-673/674 or 675, 97-676 or 677

Directions for Use

Mix thoroughly to suspend all pigmentation before, and during use. Explosion-proof equipment must be used when coating with these materials in confined areas. Keep containers closed and away from heat, sparks, and flames when not in use. USE WITH ADEQUATE VENTILATION. KEEP OUT OF REACH OF CHILDREN. Read all label and Material Safety Data Sheet (MSDS) information prior to use. MSDS are available through our website or by calling 1-800-441-9695.

Permissible temperatures during application:

Material:	**40 to 90°F**	**4 to 32°C**
Ambient:	**40 to 100°F**	**4 to 38°C**
Substrate:	**40 to 130°F**	**4 to 54°C**

Application Information

Recommended Spread Rates:

Wet Mils :	**4.3**	minimum to	**5.7**	maximum
Wet Microns:	**109.2**	minimum to	**144.8**	maximum
Dry Mils :	**1.5**	minimum to	**2.0**	maximum
Dry Microns:	**38.1**	minimum to	**50.8**	maximum

Application Equipment: Changes in application equipment, pressures and/or tip sizes may be required depending on ambient temperatures and application conditions. Spray equipment must be handled with due care and in accordance with manufacturer's recommendation. High-pressure injection of coatings into the skin by airless equipment may cause serious injury.
Conventional Spray: Fluid Nozzle: DeVilbiss MBC gun, with 704 or 777 air cap with E or FF tip and needle, or comparable equipment. Atomization Pressure: 55 - 70 Fluid Pressure: Can not specify, dependent on numerous factors.
Airless Spray: Pressure 1500 psi, tip 0.011" - 0.015"
Brush: Not Recommended
Roller: Not Recommended

Thinning:
DO NOT THIN. Spray product as received.

Packaging: 1-Gallon (3.78L)

Not all products are available in all sizes. All containers are not full-filled.

PPGAF believes the technical data presented is currently accurate: however, no guarantee of accuracy, comprehensiveness, or performance is given or implied. Improvements in coatings technology may cause future technical data to vary from what is in this bulletin. For complete, up-to-date technical information, visit our web site or call 1-800-441-9695.

PPG Industries, Inc. Architectural Coatings One PPG Place Pittsburgh, PA 15272 www.ppghpc.com | Technical Services 1-800-441-9695 1-888-807-5123 fax | Architect/Specifier 1-888-PPG-IDEA | PPG Architectural Finishes 400 S. 13th Street Louisville, KY 40203 | PPG Canada, Inc. Architectural Coatings 4 Kenview Blvd Brampton, ON L6T 5E4 | I7 10/2006

3.8.8 PPG DIMETCOTE® 9 Primer

PRODUCT DATA SHEET

October 28, 2015 (Revision of June 25, 2015)

DIMETCOTE® 9 / SIGMAZINC™ 9

DESCRIPTION

Two-component, moisture-curing zinc (ethyl) silicate coating

PRINCIPAL CHARACTERISTICS

- Specified for structural joints according to ASTM A325 or A490 Bolts RCSC specification, Class B
- Complies with the compositional requirements of SSPC-Paint 20, Level 1
- Anticorrosive primer for structural steel
- Suitable as a system primer in various paint systems based on unsaponifiable binders
- Can withstand substrate temperatures from –90°C (–130°F) up to 400°C (750°F), under normal atmospheric exposure conditions
- When suitably topcoated provides excellent corrosion protection for steel substrates up to 540°C (1000°F)
- Good low-temperature curing
- Good impact and abrasion resistance
- Must not be exposed to alkaline (more than pH 9) or acidic (less than pH 5.5) liquids

COLOR AND GLOSS LEVEL

- Greenish gray
- Flat

BASIC DATA AT 20°C (68°F)

Data for mixed product	
Number of components	Two
Mass density	2.4 kg/l (20.0 lb/US gal)
Volume solids	63 ± 3%
VOC (Supplied)	Directive 1999/13/EC, SED: max. 221.0 g/kg UK PG 6/23(92) Appendix 3: max. 480.0 g/l (approx. 4.0 lb/US gal)
Recommended dry film thickness	50 - 100 µm (2.0 - 4.0 mils) depending on system
Theoretical spreading rate	8.4 m²/l for 75 µm (337 ft²/US gal for 3.0 mils)
Dry to touch	15 minutes
Overcoating Interval	Minimum: 24 hours Maximum: Unlimited
Full cure after	46 hours
Shelf life	Binder: at least 9 months when stored cool and dry Pigment: at least 24 months when stored pigment moisture free

Notes:
- See ADDITIONAL DATA – Spreading rate and film thickness
- See ADDITIONAL DATA – Overcoating intervals
- See ADDITIONAL DATA – Curing time

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DIMETCOTE® 9 / SIGMAZINC™ 9

RECOMMENDED SUBSTRATE CONDITIONS AND TEMPERATURES

Immersion exposure

- Steel; blast cleaned to ISO-Sa2½, blasting profile 40 – 70 µm (1.6 – 2.8 mils)
- Steel with approved zinc silicate shop primer; sweep blasted to SPSS-Ss, welds, rusty and damaged areas blast cleaned to ISO-Sa2½
- Existing pipelines may have to be cleaned first by scraper pigs and solvents

Atmospheric exposure conditions

- Steel; blast cleaned to ISO-Sa2½ or minimum SSPC SP-6, blasting profile 40 – 70 µm (1.6 – 2.8 mils)
- Steel with approved zinc silicate shop primer; pretreated to SPSS-Pt3

Substrate temperature and application conditions

- Substrate temperature during application and curing down to -18°C (0°F) is acceptable; provided the substrate is free from ice and dry
- Substrate temperature during application up to 55°C (131°F) is acceptable
- Substrate temperature during application and curing should be at least 3°C (5°F) above dew point
- Relative humidity during curing should be above 50%

INSTRUCTIONS FOR USE

Mixing ratio by volume: binder to zinc powder 77:23

- Many of PPG's zinc silicates are supplied as two-pack materials consisting of a container with pigmented binder and a drum containing a bag of zinc powder.
- To ensure proper mixing of both components, the instructions given below must be followed
- To avoid lumps in the paint do not add the binder to the zinc powder
- [1] Take the bag with zinc powder out of the drum
- [2] Shake the binder in the jerrycan a few times to reach a certain degree of homogenization
- [3] Pour about 2/3 of the binder into the empty drum
- [4] With the jerrycan now reduced in weight and containing more free space, shake it vigorously to obtain a homogeneous mix with no deposits left on the bottom, and add this to the drum
- [5] Add the zinc powder gradually to the pigmented binder in the drum and, at the same time, continuously stir the mixture by using a mechanical mixer (keep the speed low)
- [6] Stir the zinc dust powder thoroughly through the binder (high speed) and keep stirring until a homogeneous mixture is obtained
- [7] Strain mixture through a 30 – 60 mesh screen
- [8] Agitate continuously during application (low speed). The use of a dedicated pump with a constant agitation for a zinc silicate coating is recommended

Note: At application temperature above 30°C (86°F) addition of max 10% by volume of THINNER 90-53 may be necessary

Induction time

None

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DIMETCOTE® 9 / SIGMAZINC™ 9

Pot life
8 hours

Note: See ADDITIONAL DATA – Pot life

Air spray

Recommended thinner
THINNER 90-53, THINNER 21-06 (AMERCOAT 65), THINNER 21-25 (AMERCOAT 101) FOR > 60°F (15°C)

Volume of thinner
0 - 10%, depending on required thickness and application conditions

Nozzle orifice
2.0 mm (approx. 0.079 in)

Nozzle pressure
0.3 MPa (approx. 3 Bar; 44 p.s.i.)

Note: A dedicated pump for a zinc silicate coating with constant agitation must be used

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Airless spray

Recommended thinner
THINNER 90-53, THINNER 21-06 (AMERCOAT 65), THINNER 21-25 (AMERCOAT 101) FOR > 60°F (15°C)

Volume of thinner
0 - 10%, depending on required thickness and application conditions

Nozzle orifice
Approx. 0.48 – 0.64 mm (0.019 – 0.025 in)

Nozzle pressure
9.0 - 12.0 MPa (approx. 90 - 120 bar; 1306 - 1741 p.s.i.)

Note: A dedicated pump for a zinc silicate coating with constant agitation must be used

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PRODUCT DATA SHEET October 28, 2015 (Revision of June 25, 2015)

DIMETCOTE® 9 / SIGMAZINC™ 9

Brush/roller

- Only for touch-up and spot repair
- Roller application is not recommended

Recommended thinner

THINNER 90-53, THINNER 21-06 (AMERCOAT 65), THINNER 21-25 (AMERCOAT 101) FOR > 60°F (15°C)

Volume of thinner

5 – 15%

Note: Apply a visible wet coat with a max. dft of 25 µm (1.0 mils)|same for subsequent coats in order to obtain the required dft

Cleaning solvent

THNNER 90-53, THINNER 90-58 (AMERCOAT 12) OR THINNER 21-06 (AMERCOAT 65)

Upgrading

- This is only valid for spray application
- If the DFT is below specification and an extra coat of DIMETCOTE 9 / SIGMAZINC 9 has to be applied, it should be thinned down with 25 – 50% Thinner 90-53, in order to obtain a visible wet coat that remains wet for some time

ADDITIONAL DATA

Spreading rate and film thickness	
DFT	**Theoretical spreading rate**
75 µm (3.0 mils)	8.4 m^2/l (337 ft^2/US gal)
100 µm (4.0 mils)	6.3 m^2/l (253 ft^2/US gal)
125 µm (5.0 mils)	5.0 m^2/l (202 ft^2/US gal)

Notes:

- Maximum DFT when brushing: 35 µm (1.4 mils)
- Above 150 µm (6.0 mils) mudcracking can occur
- Highly pigmented zinc silicate primers produce dry films with void spaces in between the particles

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DIMETCOTE® 9 / SIGMAZINC™ 9

Overcoating interval for DFT up to 100 µm (4.0 mils)					
Overcoating with...	**Interval**	**0°C (32°F)**	**10°C (50°F)**	**20°C (68°F)**	**30°C (86°F)**
recommended topcoats	Minimum	48 hours	36 hours	24 hours	18 hours
	Maximum	Unlimited	Unlimited	Unlimited	Unlimited

Notes:
- For recoating with itself to take required dft, recommend to apply within 2 days before full cure. No minimum recoating interval limitation for itself.
- To confirm cure to topcoat, conduct a MEK rub test per ASTM D4752. A rating of 4 or higher is sufficient for topcoating
- For measuring of the curing, the MEK rub test according to ASTM 4752 is a suitable method: after 50 double rubs with a cloth soaked in MEK (or alternatively THINNER 90-53) no dissolving of the coating should be observed
- Curing/recoating time will be shortened by the increase of humidity, please contact regional technical service team for details
- A mist coat / full coating application technique is required when topcoating to prevent application bubbling. Ensure dry spray is removed from the surface
- DIMETCOTE 9 / SIGMAZINC 9 is a moisture curing zinc silicate, this means that it only cures after sufficient take up of water from the atmosphere during and after application; it is recommended that relative humidity and temperature are measured during the curing time
- When curing conditions are unfavorable or when reduced overcoat times are desired, curing can be accelerated 4 hours after application by: [1] Wetting or soaking with water, keeping the surface wet for the next 2 hours, followed by drying; [2] Wetting or soaking with a 0.5% ammonia solution, followed by drying
- Maximum interval is only unlimited when the surface is free from any contamination

Curing time for DFT up to 75 µm (3.0 mils)		
Substrate temperature	**Dry to handle**	**Full cure**
0°C (32°F)	2 hours	4 days
10°C (50°F)	1 hour	3 days
20°C (68°F)	30 minutes	46 hours
30°C (86°F)	20 minutes	36 hours

Notes:
- DIMETCOTE 9 / SIGMAZINC 9 is a moisture curing zinc silicate, this means that it only cures after sufficient take up of water from the atmosphere during and after application
- It is recommended that relative humidity and temperature are measured during the curing time
- Relative humidity during curing recommended to be above 50%
- Adequate ventilation must be maintained during application and curing (please refer to INFORMATION SHEETS 1433 and 1434)

Pot life (at application viscosity)	
Mixed product temperature	**Pot life**
20°C (68°F)	8 hours

PRODUCT DATA SHEET

October 28, 2015 (Revision of June 25, 2015)

DIMETCOTE® 9 / SIGMAZINC™ 9

SAFETY PRECAUTIONS

- For paint and recommended thinners see INFORMATION SHEETS 1430, 1431 and relevant Material Safety Data Sheets
- This is a solvent-borne paint and care should be taken to avoid inhalation of spray mist or vapor, as well as contact between the wet paint and exposed skin or eyes

WORLDWIDE AVAILABILITY

It is always the aim of PPG Protective and Marine Coatings to supply the same product on a worldwide basis. However, slight modification of the product is sometimes necessary to comply with local or national rules/circumstances. Under these circumstances an alternative product data sheet is used.

REFERENCES

• CONVERSION TABLES	INFORMATION SHEET	1410
• EXPLANATION TO PRODUCT DATA SHEETS	INFORMATION SHEET	1411
• SAFETY INDICATIONS	INFORMATION SHEET	1430
• SAFETY IN CONFINED SPACES AND HEALTH SAFETY, EXPLOSION HAZARD – TOXIC HAZARD	INFORMATION SHEET	1431
• SAFE WORKING IN CONFINED SPACES	INFORMATION SHEET	1433
• DIRECTIVES FOR VENTILATION PRACTICE	INFORMATION SHEET	1434
• CLEANING OF STEEL AND REMOVAL OF RUST	INFORMATION SHEET	1490
• SPECIFICATION FOR MINERAL ABRASIVES	INFORMATION SHEET	1491
• RELATIVE HUMIDITY – SUBSTRATE TEMPERATURE – AIR TEMPERATURE	INFORMATION SHEET	1650

WARRANTY

PPG warrants (i) its title to the product, (ii) that the quality of the product conforms to PPG's specifications for such product in effect at the time of manufacture and (iii) that the product shall be delivered free of the rightful claim of any third person for infringement of any U.S. patent covering the product. THESE ARE THE ONLY WARRANTIES THAT PPG MAKES AND ALL OTHER EXPRESS OR IMPLIED WARRANTIES, UNDER STATUTE OR ARISING OTHERWISE IN LAW, FROM A COURSE OF DEALING OR USAGE OF TRADE, INCLUDING WITHOUT LIMITATION, ANY OTHER WARRANTY OF FITNESS FOR A PARTICULAR PURPOSE OR USE, ARE DISCLAIMED BY PPG. Any claim under this warranty must be made by Buyer to PPG in writing within five (5) days of Buyer's discovery of the claimed defect, but in no event later than the expiration of the applicable shelf life of the product, or one year from the date of the delivery of the product to the Buyer, whichever is earlier. Buyer's failure to notify PPG of such non-conformance as required herein shall bar Buyer from recovery under this warranty.

LIMITATIONS OF LIABILITY

IN NO EVENT WILL PPG BE LIABLE UNDER ANY THEORY OF RECOVERY (WHETHER BASED ON NEGLIGENCE OF ANY KIND, STRICT LIABILITY OR TORT) FOR ANY INDIRECT, SPECIAL, INCIDENTAL, OR CONSEQUENTIAL DAMAGES IN ANY WAY RELATED TO, ARISING FROM, OR RESULTING FROM ANY USE MADE OF THE PRODUCT. The information in this sheet is intended for guidance only and is based upon laboratory tests that PPG believes to be reliable. PPG may modify the information contained herein at any time as a result of practical experience and continuous product development. All recommendations or suggestions relating to the use of the PPG product, whether in technical documentation, or in response to a specific inquiry, or otherwise, are based on data, which to the best of PPG's knowledge, is reliable. The product and related information is designed for users having the requisite knowledge and industrial skills in the industry and it is the end-user's responsibility to determine the suitability of the product for its own particular use and it shall be deemed that Buyer has done so, as its sole discretion and risk. PPG has no control over either the quality or condition of the substrate, or the many factors affecting the use and application of the product. Therefore, PPG does not accept any liability arising from any loss, injury or damage resulting from such use or the contents of this information (unless there are written agreements stating otherwise). Variations in the application environment, changes in procedures of use, or extrapolation of data may cause unsatisfactory results. This sheet supersedes all previous versions and it is the Buyer's responsibility to ensure that this information is current prior to using the product. Current sheets for all PPG Protective & Marine Coatings Products are maintained at www.ppgpmc.com. The English text of this sheet shall prevail over any translation thereof.

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4.0 THERMAL EVALUATION

This section presents the thermal design and analyses of the Universal Storage System for normal conditions of storage of spent nuclear fuel. The analyses include consideration of design basis PWR and BWR fuel. Results of the analyses demonstrate that with the design basis contents, the Universal Storage System meets the thermal performance requirements of 10 CFR 72 [1].

4.1 Discussion

The Universal Storage System consists of a Transportable Storage Canister, Vertical Concrete Cask, and a transfer cask. In long-term storage, the canister is installed in the concrete cask, which provides passive radiation shielding and natural convection cooling. The fuel is loaded in a basket structure positioned within the canister. The transfer cask is used for the handling of the canister. The thermal performance of the concrete cask containing the design basis fuel (during storage) and the performance of the transfer cask containing design basis fuel (during handling) are evaluated herein.

The significant thermal design feature of the Vertical Concrete Cask is the passive convective air flow up along the side of the canister. Cool (ambient) air enters at the bottom of the concrete cask through four inlet vents. Heated air exits through the four outlets at the top of the cask. Radiant heat transfer occurs from the canister shell to the concrete cask liner, which also transmits heat to the adjoining air flow. Conduction does not play a substantial role in heat removal from the canister surface. Natural circulation of air inside the Vertical Concrete Cask, in conjunction with radiation from the canister surface, maintains the fuel cladding temperature and all of the concrete cask component temperatures below their design limits.

The UMS® Storage System design basis heat load is 23.0 kW for up to 24 PWR (0.958 kW per assembly) or up to 56 BWR (0.411 kW per assembly) fuel assemblies, except in cases where preferential loading patterns are employed.

The thermal evaluation considers normal, off-normal, and accident conditions of storage. Each of these conditions can be described in terms of the environmental temperature, use of solar insolation, and the condition of the air inlets and outlets, as shown in Table 4.1-1. The design conditions for transfer are defined in Table 4.1-2. The transfer conditions consider the transient effect for PWR and BWR fuel, starting from the removal of the transfer cask/canister from the spent fuel pool. The canister is considered under normal operation to be inside the transfer cask and initially filled with water. The canister is vacuum dried, back-filled with helium and then transferred into the Vertical Concrete Cask. As shown in Section 4.4.3, the time duration of the spent fuel in the water and vacuum conditions is administratively controlled to prevent general boiling of the water and to ensure that the allowable temperatures of the limiting components (fuel cladding, structural disks and heat transfer disks) are not exceeded.

This evaluation applies different component temperature limits and different material stress limits for long-term conditions and short-term conditions. Normal storage is considered to be a long-term condition. Off-normal and accident events, as well as the transfer condition that temporarily occurs during the preparation of the canister while it is in the transfer cask, are considered as short-term conditions. Thermal evaluations are performed for the design basis PWR and BWR fuels for all design conditions. The maximum allowable material temperatures for long-term and short-term conditions are provided in Table 4.1-3.

During normal conditions of storage and hypothetical accident conditions, the concrete cask must reject the fuel decay heat to the environment without exceeding the operational temperature ranges of the components important to safety. In addition, to maintain fuel rod integrity for normal conditions of storage the fuel must be maintained at a sufficiently low temperature in an inert atmosphere to preclude thermally induced fuel rod cladding deterioration. To preclude fuel degradation, the maximum allowable cladding temperature under normal conditions of storage and transfer for PWR fuel and BWR fuel assemblies is 752°F (400°C) in accordance with ISG-11, Revision 3 [37]. Additionally, the maximum cladding temperature under off-normal and accident conditions must remain below 1,058°F (570°C). Canisters containing fuel assemblies with burnup greater than 45 GWd/MTU are limited to 10 or fewer thermal cycles where the fuel cladding temperature change is greater than 117°F (65°C) during system drying, loading and transfer operations. Cycles in excess of this limit could inadvertently enhance undesirable hydride reorientation to form radial hydrides. The basis for this limitation on thermal cycles is provided by research performed by Westinghouse [38]. The implementation of the thermal cycling limitation for higher burnup fuel (>45 GWd/MTU) is provided in NAC-UMS® Technical Specification LCO 3.1.1 and discussed in Appendix 12C Technical Specification Bases for the NAC-UMS® System. Finally, for the structural components of the storage system, the thermally induced stresses, in combination with pressure and mechanical load stresses, must be below material allowable stress levels.

Thermal evaluations for normal conditions of storage and transfer (canister handling) condition operations are presented in Section 4.4. The finite element method is used to calculate the temperatures for the various components of the concrete cask, canister, basket, fuel cladding and transfer cask. Thermal models used in evaluation of normal and transfer conditions are described in Section 4.4.1.

A summary of the thermal evaluation results for the Universal Storage System are provided in Tables 4.1-4 and 4.1-5 for the PWR and BWR cases, respectively. Evaluation results for accident conditions of "All air inlets and outlets blocked" and "Fire" are presented in Chapter 11. The results demonstrate that the calculated temperatures are below the allowable component temperatures for all normal (long-term) storage conditions and for short-term events. The thermally induced stresses, combined with pressure and mechanical load stresses, are also within the allowable levels, as demonstrated in Chapter 3.

Table 4.1-1 Summary of Thermal Design Conditions for Storage

Condition[1]		Environmental Temperature (°F)	Solar Insolation[2]	Condition of Concrete Cask Inlets and Outlets
Normal		76	Yes	All inlets and outlets open
Off-Normal - Half Air Inlets Blocked		76	Yes	Half inlets blocked and all outlets open
Off-Normal - Severe Heat		106	Yes	All inlets and outlets open
Off-Normal - Severe Cold		-40	No	All inlets and outlets open
Accident - Extreme Heat		133	Yes	All inlets and outlets open
Accident - All Air Inlets and Outlets Blocked[3]		76	Yes	All inlets and outlets blocked
Accident - Fire[4]	During Fire	1475	Yes	All inlets and outlets open
	Before and After Fire	76	Yes	All inlets and outlets open

1. Off-normal and accident condition analyses are presented in Chapter 11.
2. Solar Insolation per 10 CFR 71:
   Curved Surface: 400 g cal/cm^2 (1475 Btu/ft^2) for a 12-hour period.
   Flat Horizontal Surface: 800 g cal/cm^2 (2950 Btu/ft^2) for a 12-hour period.
3. This condition bounds the case in which all inlets are blocked, with all outlets open.
4. The evaluated fire accident is the described in Section 11.2.6.

Table 4.1-2 Summary of Thermal Design Conditions for Transfer

Condition[1,2]	Maximum Duration (Hours)[3]	
	PWR	BWR
Canister Filled with Water[4]	20	17
Vacuum Drying	27	25
Canister Filled with Helium	20	16

(1) The canister is inside the transfer cask, with an ambient temperature of 76°F.

(2) See Section 8.1 for description of limiting conditions.

(3) Maximum durations based on 23 kW heat load.

(4) The initial water temperature is considered to be 100°F.

Table 4.1-3 Maximum Allowable Material Temperatures

Material	Temperature Limits (°F)		Reference
	Long Term	Short Term	
Concrete	150(B)/200(L)[(1)]	350	ACI-349 [4]
Fuel Clad PWR Fuel (5-year cooled) BWR Fuel (5-year cooled)	 752 752	 752/1,058[(2)] 752/1,058[(2)]	 ISG-11 [37] and PNL-4835 [2]
Aluminum 6061-T651	650	750	MIL-HDBK-5G [7]
NS-4-FR	300	300	GESC [8]
Chemical Copper Lead	600	600	Baumeister [9]
SA693 17-4PH Type 630 Stainless Steel	650	800	ASME Code [13] ARMCO [11]
SA240 Type 304 Stainless Steel	800	800	ASME Code [13]
SA240 Type 304L Stainless Steel	800	800	ASME Code [13]
ASTM A533 Type B Carbon Steel	700	700	ASME Code [13]
ASME SA588 Carbon Steel	700	700	ASME Code Case N-71-17 [12]
ASTM A36 Carbon Steel	700	700	ASME Code Case N-71-17 [12]

(1) B and L refer to bulk temperatures and local temperatures, respectively. The local temperature allowable applies to a restricted region where the bulk temperature allowable may be exceeded.

(2) The temperature limit of the fuel cladding is 400°C (752°F) for storage (long-term) and transfer (short-term) conditions. The temperature limit of the fuel cladding is 570°C (1,058°F) for off-normal and accident (short-term) conditions.

Table 4.1-4 Summary of Thermal Evaluation Results for the Universal Storage System: PWR Fuel

Long-Term Condition:						
	Maximum Temperatures (°F)					
Design Condition	Concrete Bulk	Concrete Local	Heat Transfer Disks	Support Disks[1]	Canister[2]	Fuel Clad
Normal (76°F Ambient)	135	186	599	601	351	648
Allowable	150	200	650	650	800	752

Short-Term Condition:					
	Maximum Temperatures (°F)				
Design Condition	Concrete	Heat Transfer Disks	Support Disks [1]	Canister[2]	Fuel Clad
Off-Normal - Half Inlets Blocked (76°F Ambient)	191	600	603	350	649
Off-Normal - Severe Heat (106°F Ambient)	228	626	628	381	672
Off-Normal - Severe Cold (-40°F Ambient	17	502	505	226	561
Accident - Extreme Heat (133°F Ambient)	262	648	650	408	693
Accident - Fire	244	639	641	391	688
Allowable	350	750	800	800	1058
	Maximum Temperatures (°F)				
Transfer - Vacuum Drying	N/A	641	644	304	732
Transfer - Backfilled with Helium	N/A	680	683	455	732
Allowable	350	750	800	800	752

1. SA 693, 17-4PH Type 630 SS.
2. SA240, Type 304L SS (including canister shell, lid and bottom plate).

Table 4.1-5 Summary of Thermal Evaluation Results for the Universal Storage System: BWR Fuel

Long-Term Condition:							
	Maximum Temperatures (°F)						
Design Condition	Concrete Bulk	Concrete Local	Heat Transfer Disks	Support Disks[1]	Canister[2]	Fuel Clad	
Normal (76°F Ambient)	136	192	612	614	376	642	
Allowable	150	200	650	700	800	752	

Short-Term Condition:					
	Maximum Temperatures (°F)				
Design Condition	Concrete	Heat Transfer Disks	Support Disks[1]	Canister[2]	Fuel Clad
Off-Normal - Half Inlets Blocked (76°F Ambient)	195	612	614	373	642
Off-Normal - Severe Heat (106°F Ambient)	231	638	640	405	667
Off-Normal - Severe Cold (-40°F Ambient)	20	504	505	252	540
Accident - Extreme Heat (133°F Ambient)	266	662	664	432	690
Accident - Fire	244	652	654	416	682
Allowable	350	750	700	800	1058
	Maximum Temperatures (°F)				
Transfer - Vacuum Drying	N/A	653	659	267	733
Transfer - Backfilled with Helium	N/A	683	686	462	733
Allowable	350	750	700	800	752

1. SA 533, Type B, CS.
2. SA240, Type 304L SS (including canister shell, lid and bottom plate).

4.2 Summary of Thermal Properties of Materials

The material thermal properties used in the thermal analyses are shown in Tables 4.2-1 through 4.2-13. Derivation of effective conductivities is described in Section 4.4.1. Tables 4.2-1 through 4.2-13 include only the materials that form the heat transfer pathways employed in the thermal analysis models. Materials for small components, which are not directly modeled are not included in the property tabulation.

Table 4.2-1 Thermal Properties of Solid Neutron Shield (NS-4-FR and NS-3)

Property (units) [8]	**NS-4-FR**	**NS-3**
Conductivity (Btu/hr-in-°F)	0.0311	0.0407
Density (borated) (lbm/in^3)	0.0589	0.0621
Density (nonborated) (lbm/in^3)	0.0607	0.0640
Specific Heat (Btu/lbm-°F)	0.319	0.149

Table 4.2-2 Thermal Properties of Stainless Steel

Type 304 and 304L

Property (units)	**Value at Temperature**				
	100°F	**200°F**	**400°F**	**550°F**	**750°F**
Conductivity (Btu/hr-in-°F) [13]	0.7250	0.7750	0.8667	0.9250	1.0000
Density (lb/in^3) [14]	0.2896	0.2888	0.2872	0.2857	0.2839
Specific Heat (Btu/lbm-°F) [14]	0.1156	0.1202	0.1274	0.1314	0.1355
Emissivity [14]	← 0.36 →				

17-4PH, Type 630

Property (units)	**Value at Temperature**			
	70°F	**200°F**	**400°F**	**650°F**
Conductivity (Btu/hr-in-°F) [13]	0.824	0.883	0.975	1.083
Density (lb/in^3) [13]	← 0.291 →			
Specific Heat (Btu/lbm-°F) [11]	← 0.11 →			
Emissivity [15]	← 0.58 →			

Table 4.2-3 Thermal Properties of Carbon Steel

Material[1] Property (units)	Value at Temperature					
	100°F	200°F	400°F	500°F	700°F	800°F
Conductivity (Btu/hr-in-°F) [13]	1.992	2.033	2.017	1.975	1.867	1.808
Density (lb/in^3) [16]	← 0.284 →					
Specific Heat (Btu/lbm-°F) [17]	← 0.113 →					
Emissivity [9]	← 0.80 →					

1. A-36, SA-533, A-588 and SA-350.

Table 4.2-4 Thermal Properties of Chemical Copper Lead

Property (units)	Value at Temperature			
	209°F	400°F	581°F	630°F
Conductivity (Btu/hr-in-°F) [18]	1.6308	1.5260	1.2095	1.0079
Density (lb/in^3) [18]	← 0.411 →			
Specific Heat (Btu/lbm-°F) [18]	← 0.03 →			
Emissivity [9]	← 0.28 (75°F) →			

Table 4.2-5 Thermal Properties of Type 6061-T651 Aluminum Alloy

Property (units)	Value at Temperature					
	200°F	300°F	400°F	500°F	600°F	750°F
Conductivity (Btu/hr-in-°F) [7,13]	8.25	8.38	8.49	8.49	8.49	8.49
Specific Heat (Btu/hr-in-°F) [13]	← 0.23 →					
Emissivity [15]	← 0.22 →					

Table 4.2-6 Thermal Properties of Helium

Property (units)	Value at Temperature			
	80°F	260°F	440°F	800°F
Conductivity (Btu/hr-in-°F) [20]	0.00751	0.00915	0.01068	0.01355

Property (units)	Value at Temperature			
	200°F	400°F	600°F	800°F
Density (lb/in^3) [19]	4.83E-06	3.70E-06	3.01E-06	2.52E-06
Specific Heat (Btu/lbm-°F) [19]	← 1.24 →			

Table 4.2-7 Thermal Properties of Dry Air

Property (units)	Value at Temperature			
	100°F	300°F	500°F	700°F
Conductivity (Btu/hr-in-°F) [19]	0.00128	0.00161	0.00193	0.00223
Density (lb/in^3) [19]	4.11E-05	3.01E-05	2.38E-05	1.97E-05
Specific Heat (Btu/lbm-°F) [19]	0.240	0.244	0.247	0.253

Table 4.2-8 Thermal Properties of Zirconium Alloy Cladding

Property (units)	Value at Temperature			
	392°F	572°F	752°F	932°F
Conductivity (Btu/hr-in-°F) [22]	0.69	0.73	0.80	0.87
Density (lb/in^3) [23]	← 0.237 →			
Specific Heat (Btu/lbm-°F) [22]	0.072	0.074	0.076	0.079
Emissivity [22]	← 0.75 →			

Table 4.2-9 Thermal Properties of Fuel (UO_2)

Property (units)	Value at Temperature				
	100°F	257°F	482°F	707°F	932°F
Conductivity (Btu/hr-in-°F) [22]	0.38	0.347	0.277	0.236	0.212
Specific Heat (Btu/lbm-°F) [22]	0.057	0.062	0.067	0.071	0.073
Density (lbm/in^3) [23]	← 0.396 →				
Emissivity [22]	← 0.85 →				

Table 4.2-10 Thermal Properties of BORAL Composite Sheet

Property (units)	Value at Temperature 100°F	Value at Temperature 500°F
Conductivity (Btu/hr-in-°F)		
Aluminum Clad [24]	7.805	8.976
Core Matrix		
PWR (calculated)	3.45	3.05
BWR (calculated)	6.60	7.23
Emissivity(1) [25]	← 0.15 →	

(1) The emissivity of the aluminum clad of the BORAL sheet ranges from 0.10 to 0.19. An averaged value of 0.15 is used.

Table 4.2-11 Thermal Properties of Concrete

Property (units)	Value at Temperature 100°F	200°F	300°F
Conductivity (Btu/hr-in-°F) [26]	0.091	0.089	0.086
Density (lbm/in^3) [27]	← 140 →		
Specific Heat (Btu/lbm-°F) [17]	← 0.20 →		
Emissivity (1) [17,28]	← 0.90 →		
Absorptivity [29]	← 0.60 →		

(1) Emissivity = 0.93 for masonry, 0.94 for rough concrete; 0.9 is used.

Table 4.2-12 Thermal Properties of Water

Property (units)	Value at Temperature		
	70°F	**200°F**	**300°F**
Conductivity (Btu/hr-in-°F) [32]	0.029	0.033	0.033
Specific Heat (Btu/lbm-°F) [32]	0.998	1.00	1.03
Density (lbm/in^3) [32]	0.036	0.035	0.033

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4.3 Technical Specifications for Components

Five major components of the Universal Storage System must be maintained within their safe operating temperature ranges: the concrete, the lead gamma shield, the NS-4-FR solid neutron shield in the transfer cask, the aluminum heat transfer disks and steel (17-4PH and ASTM A533) support disks in the basket structure inside the canister. The safe operating ranges for these components are from a minimum temperature of -40°F to the maximum temperatures as shown in Table 4.1-3.

The criterion for the safe operating range of the lead gamma shield is the prevention of the lead from reaching its melting point of 620°F [9]. The maximum operating temperature limit of the NS-4-FR solid neutron shield material, determined by the manufacturer, is to ensure sufficient neutron shielding capacity.

The primary consideration in establishing the safe operating range of the aluminum heat transfer disks and steel support disks is maintaining the integrity of the aluminum and steel.

The temperature limit for the aluminum heat transfer disks is 650°F and 750°F for the long-term and short-term conditions, respectively, based on data from MIL-HDBK-5G. Note that the heat transfer disk is not a structural component. During the limiting condition (short term) of canister transfer, the heat transfer disk is subjected to a maximum loading of 1.1 g (normal handling). An evaluation is performed for the heat transfer disks for both PWR and BWR configurations to the stresses for this condition. Two quarter-symmetry models were generated using ANSYS SHELL63 elements for the evaluation, as shown in Figures 4.3-1 and 4.3-2. The disks are supported at the basket tie-rod locations in the canister axial direction. Symmetry boundary conditions are applied at the planes of symmetry. An inertia load of 1.1 g is applied to the disk in the out-of-plane direction.

The analysis results indicate that the stress is less than 0.13 ksi at the central region of the basket where maximum temperature occurs for both the PWR and BWR configuration. The corresponding margin of safety is + 9.8 based on the yield stress of 1.4 ksi at 750°F. Therefore, the aluminum heat transfer disk will maintain its integrity as long as it does not exceed the temperature limits.

Figure 4.3-1 PWR Heat Transfer Disk Model for Normal Handling Condition

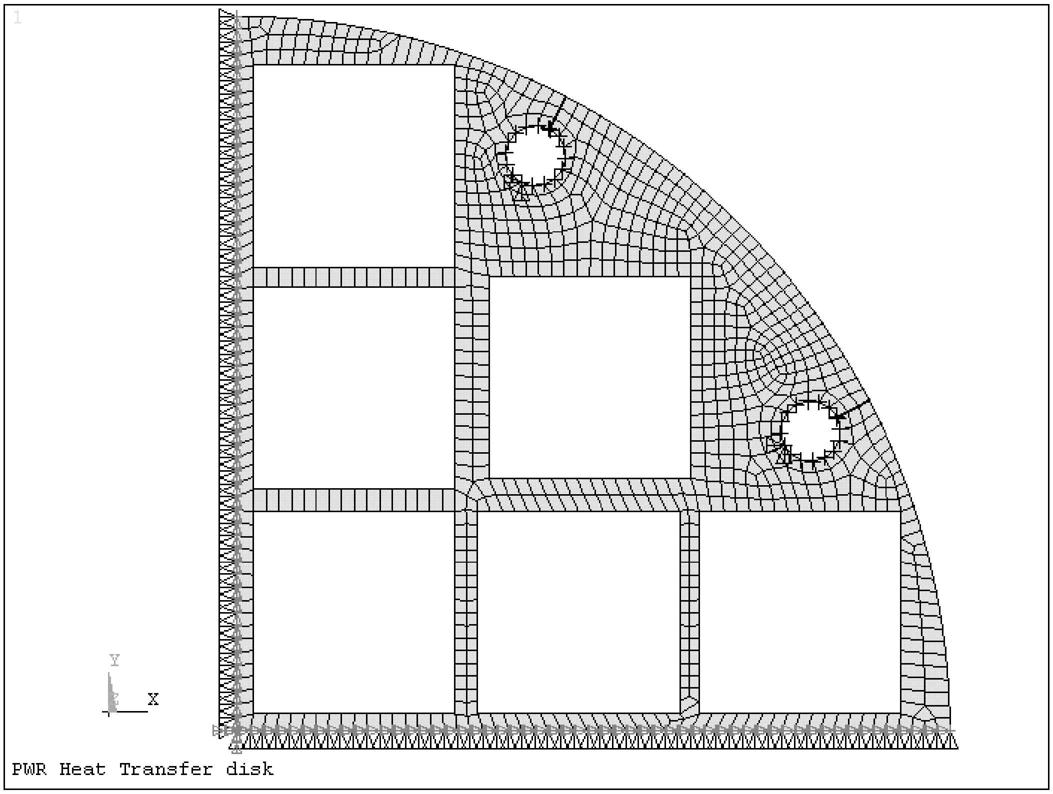
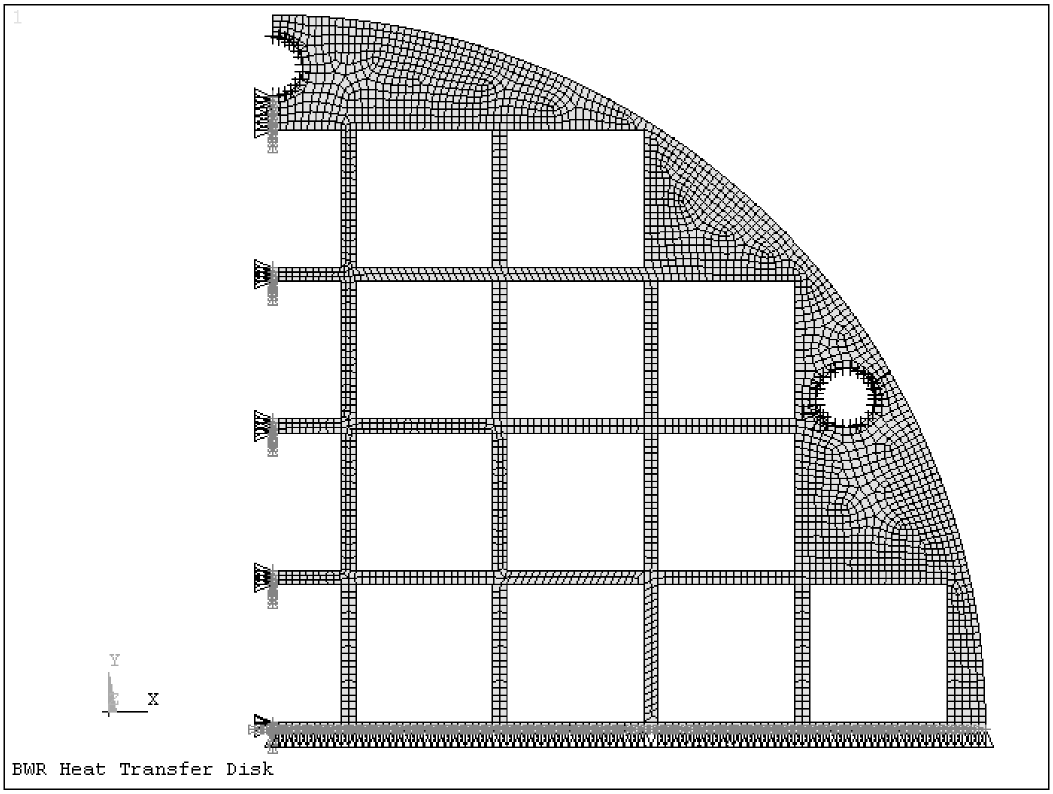


Figure 4.3-2 BWR Heat Transfer Disk Model for Normal Handling Condition



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4.4 Thermal Evaluation for Normal Conditions of Storage

The finite element method is used to evaluate the thermal performance of the Universal Storage System for normal conditions of storage. The general-purpose finite element analysis program ANSYS Revisions 5.2 and 5.5 [6] are used to perform the finite element evaluations.

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4.4.1 Thermal Models

Finite element models are utilized for the thermal evaluation of the Universal Storage System, as shown below. These models are used separately to evaluate the system for the storage of PWR or BWR fuel.

1. Two-Dimensional Axisymmetric Air Flow and Concrete Cask Models
2. Three-Dimensional Canister Models
3. Three-Dimensional Transfer Cask and Canister Models
4. Three-Dimensional Periodic Canister Internal Models
5. Two-Dimensional Fuel Models
6. Two-Dimensional Fuel Tube Models
7. Two-Dimensional Forced Air Flow Model for Transfer Cask Cooling

The two-dimensional axisymmetric air flow and concrete cask model includes the concrete cask, air in the air inlets, annulus and the air outlets, the canister and the canister internals, which are modeled as homogeneous regions with effective thermal conductivities. The effective thermal conductivities for the canister internals in the radial direction are determined using the three-dimensional periodic canister internal models. The effective conductivities in the canister axial direction are calculated using classical methods. The two-dimensional axisymmetric air flow and concrete cask model is used to perform computational fluid dynamic analyses to determine the mass flow rate, velocity and temperature of the air flow, as well as the temperature distribution of the concrete, concrete cask steel liner and the canister. Two models are generated for the evaluations of the PWR and the BWR systems, respectively. These models are essentially identical, but have slight differences in dimensions and the effective properties of the canister internals.

The three-dimensional canister model comprises the fuel assemblies, fuel tubes, stainless steel or carbon steel support disks, aluminum heat transfer disks, top and bottom weldments, the canister shell, lids and bottom plate. The canister model is employed to evaluate the temperature distribution of the fuel cladding and basket components. The fuel assemblies and the fuel tubes in the three-dimensional canister model are modeled using effective conductivities. The effective conductivities for the fuel assemblies are determined using the two-dimensional fuel models. The effective conductivities for the fuel tubes are determined using the two-dimensional fuel tube models. Two three-dimensional canister models are generated for the PWR and BWR canisters, respectively.

The three-dimensional transfer cask model includes the transfer cask and the canister with its internals. This model is used to perform transient and steady state analyses for the transfer condition, starting from removing the transfer cask/canister from the spent fuel pool, vacuum drying and finally back-filling the canister with helium. Separate transfer cask models are required for PWR and BWR systems.

The three-dimensional canister internal model consists of a periodic section of the canister internals. For the PWR canister, the model contains one support disk with two heat transfer disks (half thickness) on its top and bottom, fuel assemblies, fuel tubes and the media in the canister. For the BWR canister, two models are required. The first model, for the central region of the BWR canister, contains one heat transfer disk with two support disks (half thickness) on its top and bottom, fuel assemblies, fuel tubes and the media in the canister. The other model, for the region without heat transfer disks, contains two support disks (half thickness), fuel assemblies, fuel tubes and the media in the canister. The purpose of the three-dimensional periodic canister internal model is to determine the effective thermal conductivity of the canister internals in the canister radial direction. The effective conductivities are used in the two-dimensional axisymmetric air flow and concrete cask models. The media in the canister is considered to be helium. The fuel assemblies and fuel tubes in this model are modeled as homogeneous regions with effective thermal properties, which are determined by the two-dimensional fuel models and the two-dimensional fuel tube models.

The two-dimensional fuel model includes the fuel pellets, cladding and the media occupying the space between fuel rods. The media is considered to be helium for storage conditions and water, vacuum, helium or saturated steam for transfer conditions. The model is used to determine the effective thermal conductivities of the fuel assembly. In order to account for various types of fuel assemblies, a total of seven fuel models are generated: Four models for the 14×14, 15×15, 16×16 and 17×17 PWR fuel assemblies and three models for the 7×7, 8×8 and 9×9 BWR fuel assemblies. The effective properties are used in the three-dimensional canister models, the three-dimensional periodic canister internal models and the three-dimensional transfer cask and canister model.

The two-dimensional fuel tube model is used to determine the effective conductivities of the fuel tube wall and neutron absorber. BORAL effective conductivity is considered in the model for the neutron absorber. The effective conductivities are used in the three-dimensional canister models, the three-dimensional periodic canister internal models and the three-dimensional transfer cask and canister model.

The two-dimensional axisymmetric air flow model is used to determine the air flow rate needed for the forced air cooling of the canister inside the transfer cask.

Detailed description of the finite element models are presented in Sections 4.4.1.1 through 4.4.1.7.

4.4.1.1 Two-Dimensional Axisymmetric Air Flow and Concrete Cask Models

This section describes the finite element models used to evaluate the thermal performance of the vertical concrete cask for the PWR and BWR configurations. The model includes the concrete cask, the air in the air inlets, the annulus and the air outlets, the canister and the canister internals, which are modeled as homogeneous regions with effective thermal conductivities. Two separate two-dimensional axisymmetric models are used for the PWR and BWR configurations, respectively. The PWR model is shown in Figures 4.4.1.1-1 and 4.4.1.1-2. The BWR model is essentially identical to the PWR model, but it incorporates different effective thermal properties of the canister internals, and slight differences in dimensions.

The fuel canister is cooled by (1) natural/free convection of air through the lower vents (the air inlets), the vertical air annulus, and the upper vents (the air outlets); and (2) radiation heat transfer between the surfaces of the canister shell and the steel liner. The heat transferred to the liner is rejected by air convection in the annulus and by conduction through the concrete. The heat flow through the concrete is dissipated to the surroundings by natural convection and radiation heat transfer. The temperature in the concrete region is controlled by radiation heat transfer between the vertical annulus surfaces (the canister shell outer surface and the steel liner inner surface), natural convection of air in the annulus, and boundary conditions applicable to the concrete cask outer surfaces—e.g., natural convection and radiation heat transfer between the outer surfaces and the environment, including consideration of incident solar energy. These heat transfer modes are combined in the air flow and concrete cask model. The entire thermal system, including mass, momentum, and energy, is analyzed using the two-dimensional axisymmetric air flow and concrete cask models. The temperature distributions of the concrete cask, the air region and the canister are determined by these models. Detailed thermal evaluations for the canister internals (fuel cladding, basket, etc.) are performed using the three-dimensional canister models as described in Section 4.4.1.2.

The concrete cask has four air inlets at the bottom and four air outlets at the top that extend through the concrete. Since the configuration is symmetrical, it can be simplified into a two-dimensional axisymmetric model by using equivalent dimensions for the air inlets and outlets, which are assumed to extend around the concrete cask periphery. The canister internals are modeled as three homogeneous regions using effective thermal conductivities - the active fuel region and the regions above and below the active fuel region. The two-dimensional axisymmetric model is shown schematically in Figure 4.4.1.1-1. Determination of the effective properties is described in Section 4.4.1.4.

ANSYS FLOTRAN FLUID141 fluid thermal elements are used to construct the two-dimensional axisymmetric finite element models, as shown in Figure 4.4.1.1-2. In the air region (including the air inlet, outlet and annulus regions), only quadrilateral elements are used and the element sizes are nonuniform with much smaller element sizes close to the walls. In other regions, to simulate conduction, a mix of quadrilateral elements and triangular elements are used. Radiation heat transfer that occurs in the following regions is included in the model:

1. From the concrete outer surfaces to the ambient
2. Across the vertical air annulus (from the canister shell to the concrete cask liner)
3. From the top of the active fuel region to the bottom of the canister shield lid
4. From the bottom of the active fuel region to the top of the canister bottom plate
5. From the canister structural lid to the shield plug
6. From the shield plug to the concrete cask lid

Loads and Boundary Conditions

1. Heat generation in the active fuel region.

The distribution of the heat generation is based on the axial power distribution shown in Figures 4.4.1.1-3 and 4.4.1.1-4 for PWR and BWR fuels, respectively (see description in Chapter 5, Section 5.2.6, for the design-basis fuel).

2. Solar insolation to the outer surfaces of the concrete cask.

The solar insolation to the concrete cask outer surfaces is considered in the model. The incident solar energy is applied based on 24-hour averages as shown below.

Side surface: $$\frac{1475 \text{Btu/ft}^2}{24 \text{hrs}} = 61.46 \text{Btu/hr} \cdot \text{ft}^2$$

Top surface: $$\frac{2950 \text{Btu/ft}^2}{24 \text{hrs}} = 122.92 \text{Btu/hr} \cdot \text{ft}^2$$

3. Natural convection heat transfer at the outer surfaces of the concrete cask.

Natural convection heat transfer at the outer surfaces of the concrete cask is evaluated by using the heat transfer correlation for vertical and horizontal plates [17, 29]. This method assumes a surface temperature and then estimates Grashof (Gr) or Rayleigh (Ra) numbers to determine whether a heat transfer correlation for a laminar flow model or for a turbulent flow model should be used. Since Grashof or Rayleigh numbers are much higher than the critical values, correlation for the turbulent flow model is used as shown in the following.

Side surface [17]:

$$Nu = 0.13(Gr \cdot Pr)^{1/3}$$
$$h_c = Nu \cdot k_f / H_{VCC} \qquad \text{for } Gr > 10^9$$

Top surface [29]:

$$Nu = 0.15 Ra^{1/3}$$
$$h_c = Nu \cdot k_f / L \qquad \text{for } Ra > 10^7$$

where:

Gr	Grashof number
h_c	Average natural convection heat transfer coefficient
H_{vcc}	Height of the vertical concrete cask
k_f	Conductivity
L	Top surface characteristic length, L = area / perimeter
Nu	Average Nusselt number
Pr	Prandtl number
Ra	Rayleigh number

All material properties required in the above equations are evaluated based on the film temperature, that is, the average value of the surface temperature and the ambient temperature.

4. Radiation heat transfer at the concrete cask outer surfaces.

The radiation heat transfer between the outer surfaces and the ambient is evaluated in the model by calculating an equivalent radiation heat transfer coefficient.

$$h_{rad} = \frac{\sigma(T_1^2 + T_2^2)(T_1 + T_2)}{1/\varepsilon_1 + 1/\varepsilon_2 + 1/F_{12} - 2}$$

where:

h_{rad}	Equivalent radiation heat transfer coefficient
F_{12}	View factor
T_1 & T_2	Surface (T_1) and ambient (T_2) temperatures
ε_1 & ε_2	Surface (ε_1) and ambient (ε_2=1) emissivities
σ	Stefan-Boltzmann Constant

At the concrete cask side, an emissivity for a concrete surface of $\varepsilon_1 = 0.9$ is used and a calculated view factor (F_{12}) = 0.182 [29] is applied. The view factor is determined by conservatively assuming that the cask is surrounded by eight casks.

At the cask top, an emissivity, ε_1 , of 0.8 is conservatively used (emissivity for concrete is 0.9), and a view factor, F_{12} , of 1 is applied.

Accuracy Check of the Numerical Simulation

To ensure the accuracy of the numerical simulation of the air flow in the concrete cask, and to ensure reliable numerical results, the following checks and confirmations are performed.

1. Global convergence of the iteration process for the nonlinear system.

The system controlling air flow through the cask and, therefore, the temperature field is nonlinear and is solved iteratively.

The global iteration process is monitored by checking the variation of parameters with the global iteration—e.g., the maximum air temperature, the mass flow rate, and the net heat carried out of the concrete cask by air convection. All of the results presented are at the converged state.

2. Overall energy balance and mass balance.

This step validates the overall energy balance and mass balance. The mass balance is also shown in Figure 4.4.1.1-5. At the converged state, the mass flow rate at the air inlets matches the mass flow rate at the air outlets, showing that an excellent mass balance has been obtained.

The overall energy balance is checked by computing the total heat input (Q_{in}) and total heat output (Q_{out}). The total heat input includes the total heat from the fuel (Q_{fuel}) and the total absorbed solar energy (Q_{sun}) incident on the concrete cask outer surfaces. The total heat output is the sum of the net heat carried out of the cask by air (Q_{air}) and by convection and radiation heat loss at the concrete cask outer surfaces (Q_{con}).

For the normal storage condition with the PWR design heat load of 23.0 kW:

$$Q_{in} = Q_{fuel} + Q_{sun} = 23.0 \text{ kW} + 9.18 \text{ kW} = 32.18 \text{ kW}$$
$$Q_{out} = Q_{air} + Q_{con} = 20.97 \text{ kW} + 11.72 \text{ kW} = 32.69 \text{ kW}$$
$$Q_{out}/Q_{in} = 1.016$$

For the normal storage condition with the BWR design heat load of 23.0 kW:

$$Q_{in} = Q_{fuel} + Q_{sun} = 23.0 \text{ kW} + 9.52 \text{ kW} = 32.52 \text{ kW}$$
$$Q_{out} = Q_{air} + Q_{con} = 20.70 \text{ kW} + 12.12 \text{ kW} = 32.82 \text{ kW}$$
$$Q_{out}/Q_{in} = 1.009$$

The overall energy balance is demonstrated to be within 2 percent for all design conditions.

3. Finite Element Mesh Adequacy Study.

A sensitivity evaluation is performed to assess the effect of the number of elements used in the Two-dimensional Axisymmetric Air Flow and Concrete Cask Models. The sensitivity evaluation is performed with a reduced element model based on the model for the PWR fuel configuration. The total number of elements in the reduced-element model (13,371 elements) is 21% less than the number of elements used in the axisymmetric air flow and concrete cask model described above. The reduction in the number of elements occurs in the air flow region in the radial direction, which has the largest gradients in velocity and temperature. As shown below, the temperatures calculated by the reduced element model (Case ES1) are essentially the same as the temperatures calculated by the axisymmetric air flow and concrete cask model (Case ES2).

Case	Number of Elements in Model	Max. Air Temp. in Annular Region (Canister Surface)	Maximum Concrete Temp.	Average Air Temp. at the Outlet	Maximum Canister Shell Temp.
ES1	13,371	451 K	360 K	335 K	452 K
ES2	16,835	448 K	359 K	339 K	449 K
ES1/ES2	0.79	1.01	1.00	0.99	1.01

A comparison of the two models (Case ES1/ES2) shows that the maximum difference is 1%. Therefore, the number of elements used in the Two-dimensional Axisymmetric Air Flow and Concrete Cask Model (16,835) is adequate.

Supplemental Shielding Fixture Evaluation

The effect of the installation of an optional supplemental shielding fixture, shown in Drawing 790-613, installed in the air inlet is evaluated based on one-half of the air inlets blocked. The analysis results show that the maximum temperature increase is 5°F, which remains well below normal condition allowables. The pipes in the shielding fixture are offset to block (gamma) radiation, but allow air flow.

Off-Center Canister Evaluation

The analysis assumes that the canister is centered in the concrete cask. However, the potential exists for the canister to be placed off-center when it is installed in the storage cask. The support ring may be used to aid in centering the canister during the lowering of the canister into the concrete cask. The final placement of the canister shall not be closer than one inch to the concrete cask liner. This placement reduces the area of the air flow path in an arc established by the canister shell and concrete cask liner. An air flow analysis is performed to evaluate the effects of the off-center positioning of the canister. The analysis results show an increase in air mass flow rate occurs in the annulus, which results in a temperature reduction in the canister shell and concrete cask liner. Consequently, the off-center canister placement condition is bounded by the condition that the canister is at the center of the concrete cask, as considered in the two-dimensional axisymmetric finite element model described in this section.

Figure 4.4.1.1-1 Two-Dimensional Axisymmetric Air Flow and Concrete Cask Model: PWR

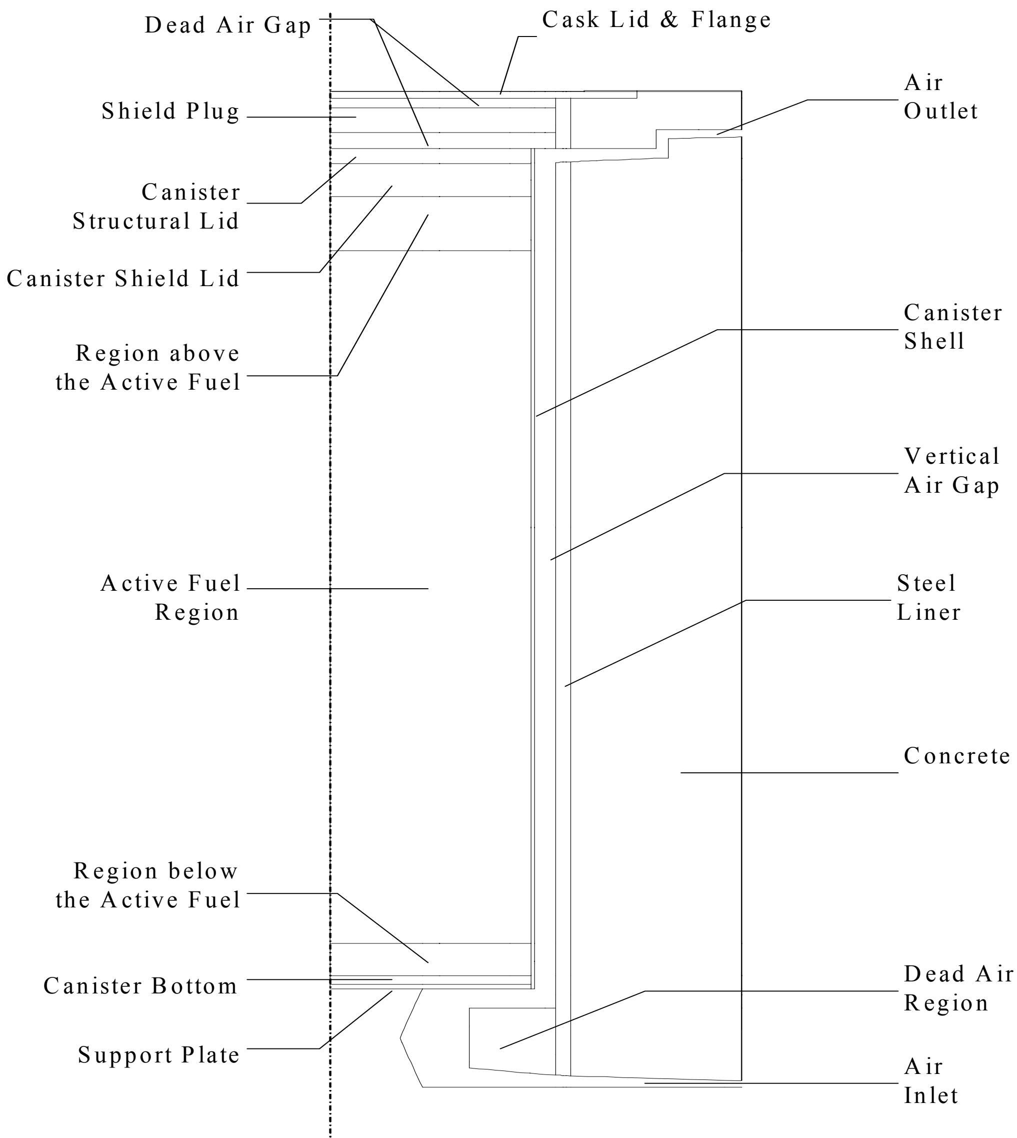


Figure 4.4.1.1-2 Two-Dimensional Axisymmetric Air Flow and Concrete Cask Finite Element Model: PWR

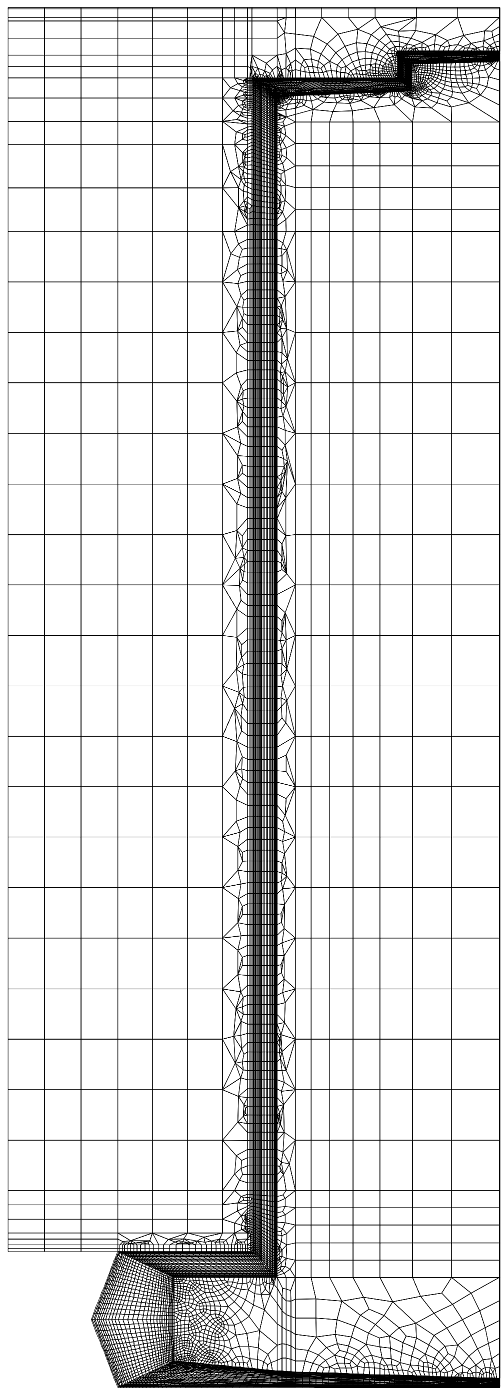


Figure 4.4.1.1-3 Axial Power Distribution for PWR Fuel

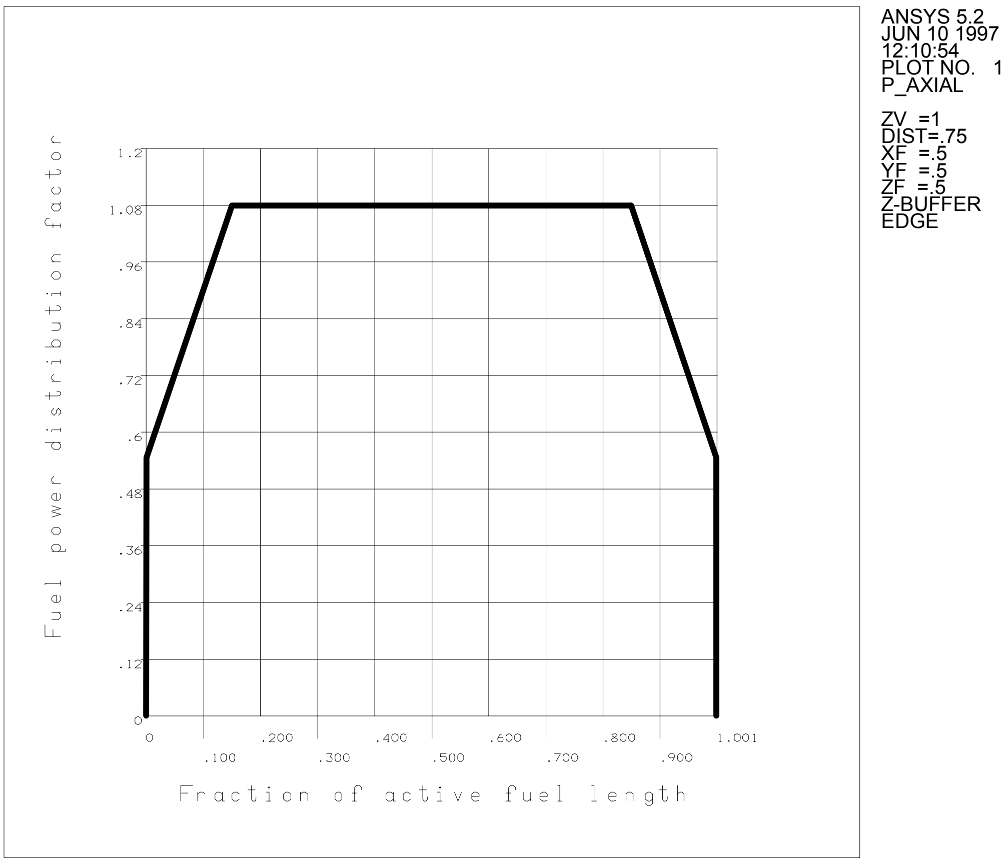
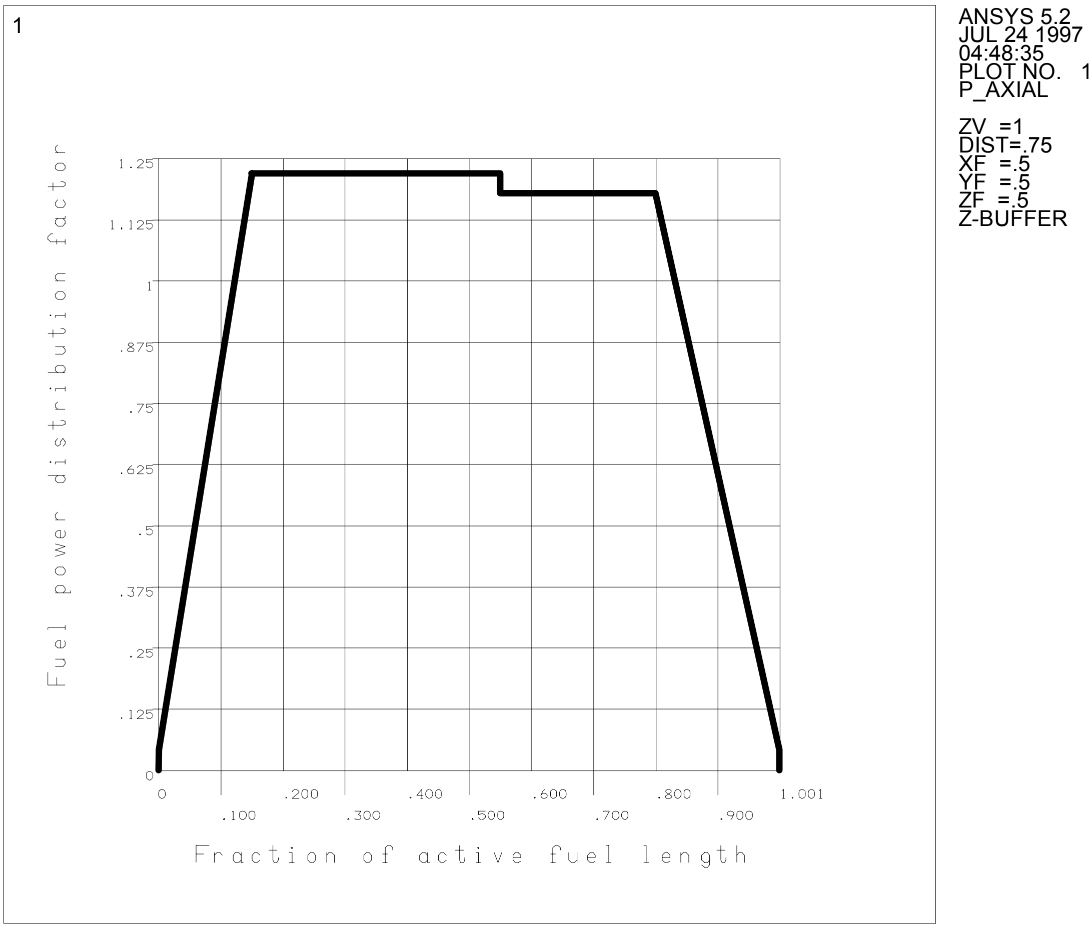


Figure 4.4.1.1-4 Axial Power Distribution for BWR Fuel



4.4.1.2 Three-Dimensional Canister Models

Two three-dimensional canister models are used to evaluate the temperature distribution of the fuel cladding and basket components inside the canister for the PWR and BWR configurations, respectively. The model for PWR fuel is shown in Figures 4.4.1.2-1 and 4.4.1.2-2. The model for BWR fuel is shown in Figures 4.4.1.2-3 and 4.4.1.2-4.

ANSYS SOLID70 three-dimensional conduction elements and LINK31 radiation elements are used to construct the model. The model includes the fuel assemblies, fuel tubes, support disks, heat transfer disks, top and bottom weldments, canister shell, lids, bottom plate and gas inside the canister (helium). Based on symmetry, only half of the canister is modeled. The plane of symmetry is considered to be adiabatic.

The canister outer surface temperatures obtained from the two-dimensional axisymmetric air flow and concrete cask model (Section 4.4.1.1) are applied at the canister surfaces in the model as boundary conditions. In the model, the fuel assemblies are considered to be centered in the fuel tubes. The fuel tubes are centered in the slots of the support disks and heat transfer disks. The basket is centered in the canister. These assumptions are conservative, since any contact between components will provide a more efficient path to reject the heat.

The gaps used in the three-dimensional canister model between the support disks and canister shell, as well as between the heat transfer disk and the canister shell, are shown in the following table.

		Nominal Gap At Room Temperature (inch)	Gap Used in the 3-D Thermal Model (inch)	
			At Room Temperature	At Elevated Temperature
PWR	Gap between Support Disk and Canister Shell	0.120	0.155	0.165
	Gap between Heat Transfer Disk and Canister Shell	0.245	0.280	0.195
BWR	Gap between Support Disk and Canister Shell	0.120	0.155	0.165
	Gap between Heat Transfer Disk and Canister Shell	0.280	0.315	0.232

The gaps at room temperature are first used in the model to calculate preliminary temperature distribution and to determine the differential thermal expansion of the disks and canister shell at the elevated temperatures. The gaps at elevated temperature are then established, based on the differential thermal expansions between components, and used in the model for final solution. As shown above, the room temperature gaps used in the thermal model bound the actual nominal gaps at room temperature.

These gap sizes are adjusted in the model to account for differential thermal expansion of the disks and canister shell based on thermal conditions. The gaps used in the model are shown to be larger than the actual gap size based on thermal expansion calculation using the thermal analysis results; therefore, the model is conservative.

A sensitivity study was performed to assess the effect of gap sizes on temperature results, with consideration of fabrication tolerance of the canister and basket. The ANSYS three-dimensional canister model for the PWR fuel is used for the study. The gaps between the disks and canister shell are increased to account for the worst case fabrication tolerance of the canister and basket. The gaps are also adjusted based on the differential thermal expansion of the canister and basket at elevated temperature. Compared to the gaps used in the original three-dimensional thermal model, the gap between the support disk and the canister shell is increased by 27% and the gap between the heat transfer disk and the canister shell is increased by 24%. The results of the sensitivity study indicate that the increase in the maximum fuel cladding and basket temperatures is less than 9°F, which is less than 3% of the temperature difference between the maximum temperature of the fuel cladding/basket and the canister shell. Therefore, the effect of the thermal model gap size on the maximum temperature of the basket and fuel cladding is not significant.

The structural lid and the shield lid are expected to be in full contact due to the weight of the structural lid. The thermal resistance across the contact surface is considered to be negligible and, therefore, no gap is modeled between the lids.

All material properties used in the model, except the effective properties discussed below, are shown in Tables 4.2-1 through 4.2-13.

The fuel assemblies and fuel tubes are modeled as homogenous regions with effective conductivities, determined by the two-dimensional fuel models (Section 4.4.1.5) and the two-dimensional fuel tube models (Section 4.4.1.6), respectively. The effective properties are listed in Tables 4.4.1.2-1 through 4.4.1.2-4. The properties corresponding to the PWR 14 × 14 assemblies are used for the PWR model, since the 14 × 14 assemblies have lower conductivities as compared to other PWR assemblies. For the same reason, the properties corresponding to the BWR 9 × 9 assemblies are used in the BWR model.

In the model, radiation heat transfer is taken into account in the following locations:

1. From the top of the fuel region to the bottom surface of the canister shield lid.
2. From the bottom of the fuel region to the top surface of the canister bottom plate.
3. From the exterior surfaces of the fuel tubes (surface between disks) to the inner surface of the canister shell.
4. From the edge of the PWR support disks to the inner surface of the canister shell.
5. From the edge of heat transfer disks to the inner surface of the canister shell.
6. Between disks in the PWR model in the canister axial direction.

The radiation heat transfer from the BWR support disk is conservatively neglected by using an emissivity value of 0.0001 for the BWR support disk in the model. An emissivity of 0.22 is used for the heat transfer disk, except the water-jet cut surfaces (the circumferential surfaces at the edges of the disks facing the canister shell and the inner surfaces of each slot). The surface condition of the water-jet cut surfaces is similar to that of the sandblasted surface and, therefore, an emissivity of 0.4 is used.

Radiation elements (LINK31) are used to model the radiation effect for the first three locations. Radiation across the gaps (Locations No. 4 through 6) is accounted for by establishing effective conductivities for the gas in the gap, as shown below. The gaps are small compared to the surfaces separated by the gaps.

Radiation heat transfer between two nodes i (hotter node) and j (colder node) is accounted for by the expression:

$$q_r = \sigma \varepsilon A F \left(T_i^4 - T_j^4\right)$$

where:

- σ = the Stefan-Boltzman constant
- ε = effective emissivity between two surfaces
- A = surface area

F = the gray body shape factor for the surfaces
T_i = temperature of the i th node
T_j = temperature of the j th node

The total heat transfer can be expressed as the sum of the radiation and the conduction processes:

$$Q_t = q_r + q_k$$

where q_r is specified above for the radiation heat transfer and q_k, which is the heat transfer by conduction is expressed as:

$$q_k = \frac{KA}{g}(T_i - T_j)$$

where:

T_i = temperature of the i th node
T_j = temperature of the j th node
g = gap distance (between the two surfaces defined by node i and node j)
K = conductivity of the gas in the gap
A = area of gap surface

By combining the two expressions (for q_k and q_r) and factoring out the term $A(T_i - T_j)/g$,

$$Q_t = [g\sigma\varepsilon F(T_i^2 + T_j^2)(T_i + T_j) + K][A(T_i - T_j)/g]$$

or

$$Q_t = K_{eff}A(T_i - T_j)/g$$

where:

$$K_{eff} = g\sigma\varepsilon F(T_i^2 + T_j^2)(T_i + T_j) + K$$

The material conductivity used in the analysis for the elements comprising the gap includes the heat transfer by both conduction and radiation.

Effective emissivities (ε) are used for all radiation calculations, based on the formula below [17]. The view factor is taken to be unity.

$\varepsilon = 1/ (1/\varepsilon_1 + 1/\varepsilon_2 -1)$ where ε_1 & ε_2 are the emissivities of two parallel plates

Radiation between the exterior surfaces of the fuel tubes is conservatively ignored in the model.

Volumetric heat generation (Btu/hr-in^3) is applied to the active fuel region based on design heat load, active fuel length of 144 inches and an axial power distribution as shown in Figures 4.4.1.1-3 and 4.4.1.1-4 for PWR and BWR fuel, respectively.

Figure 4.4.1.2-1 Three-Dimensional Canister Model for PWR Fuel

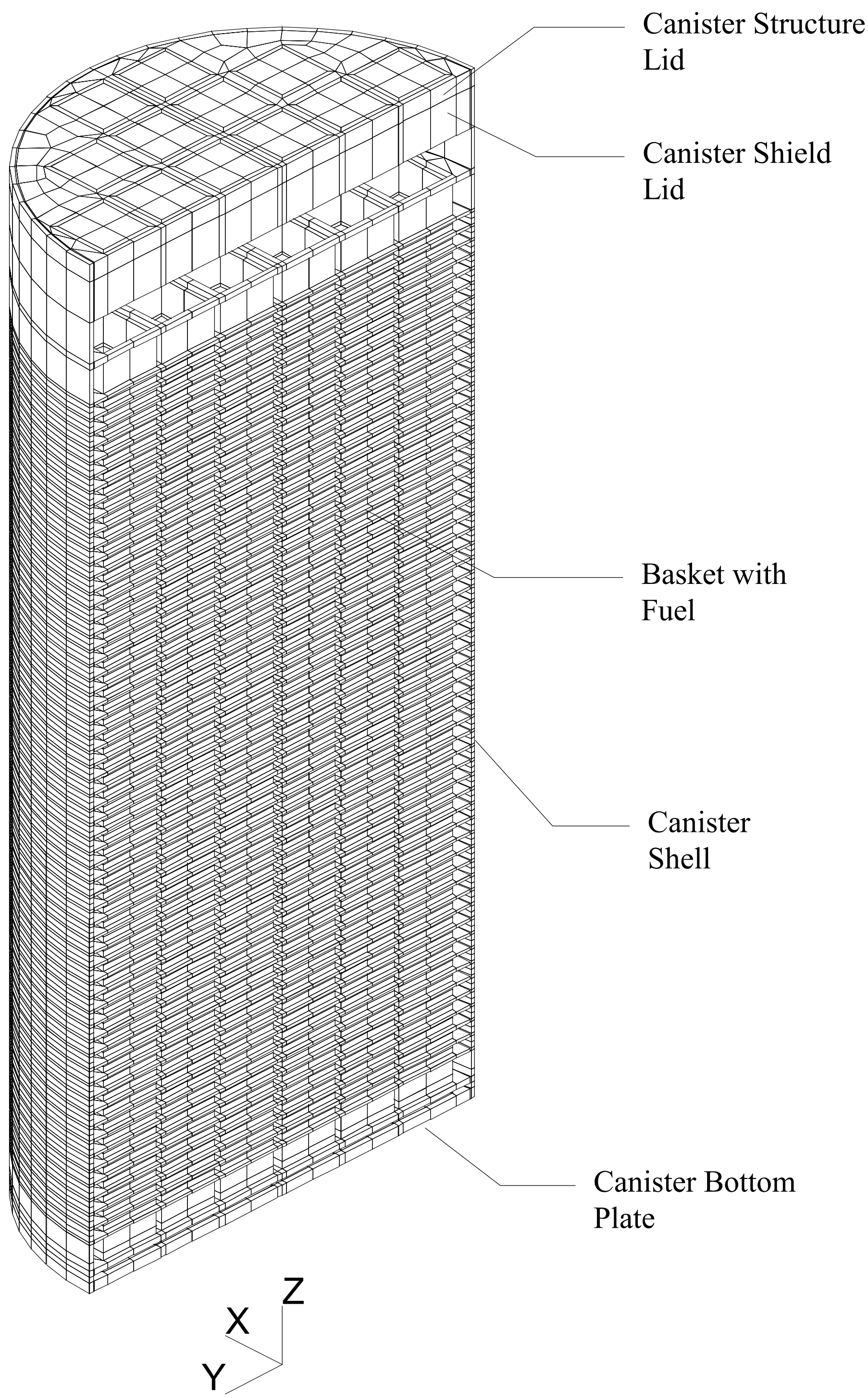


Figure 4.4.1.2-2 Three-Dimensional Canister Model for PWR Fuel – Cross Section

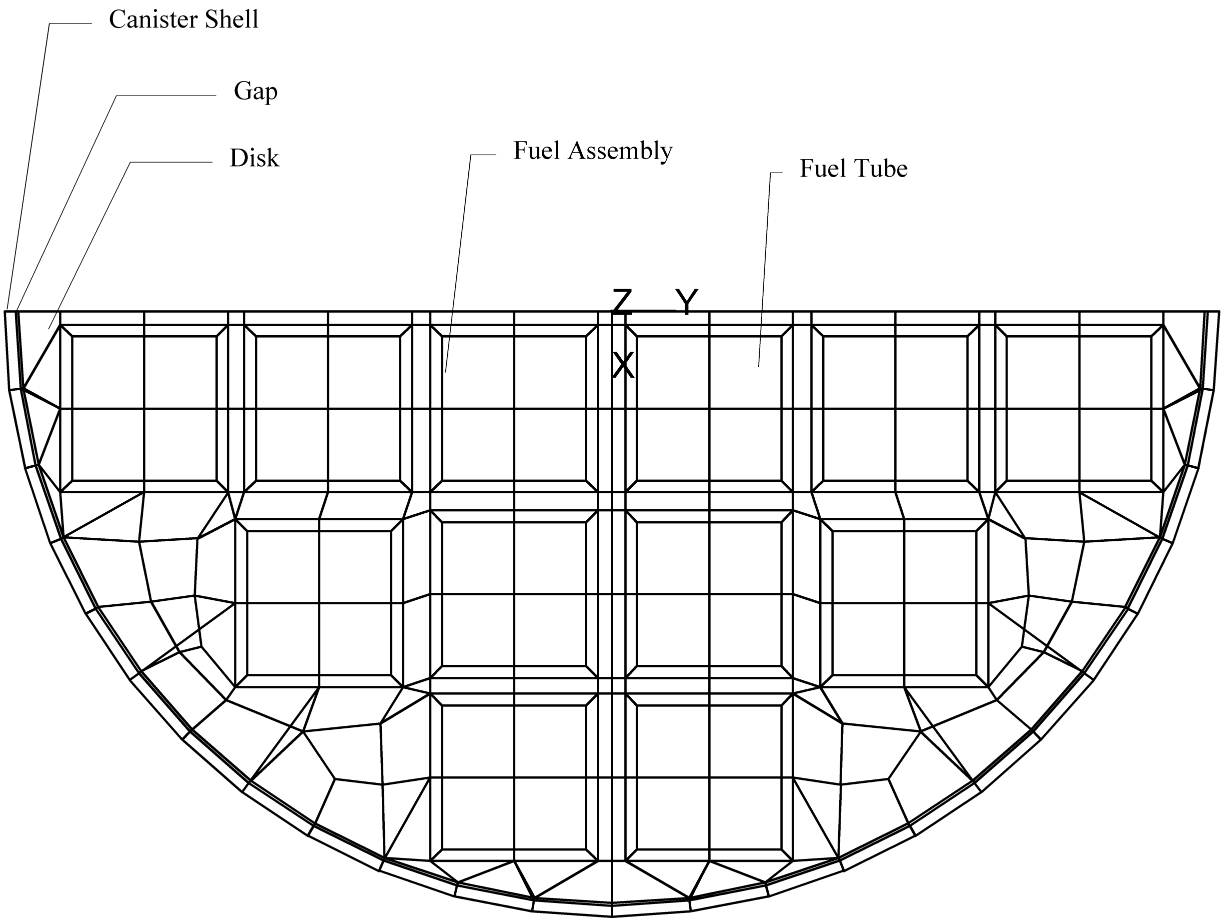


Figure 4.4.1.2-3 Three-Dimensional Canister Model for BWR Fuel

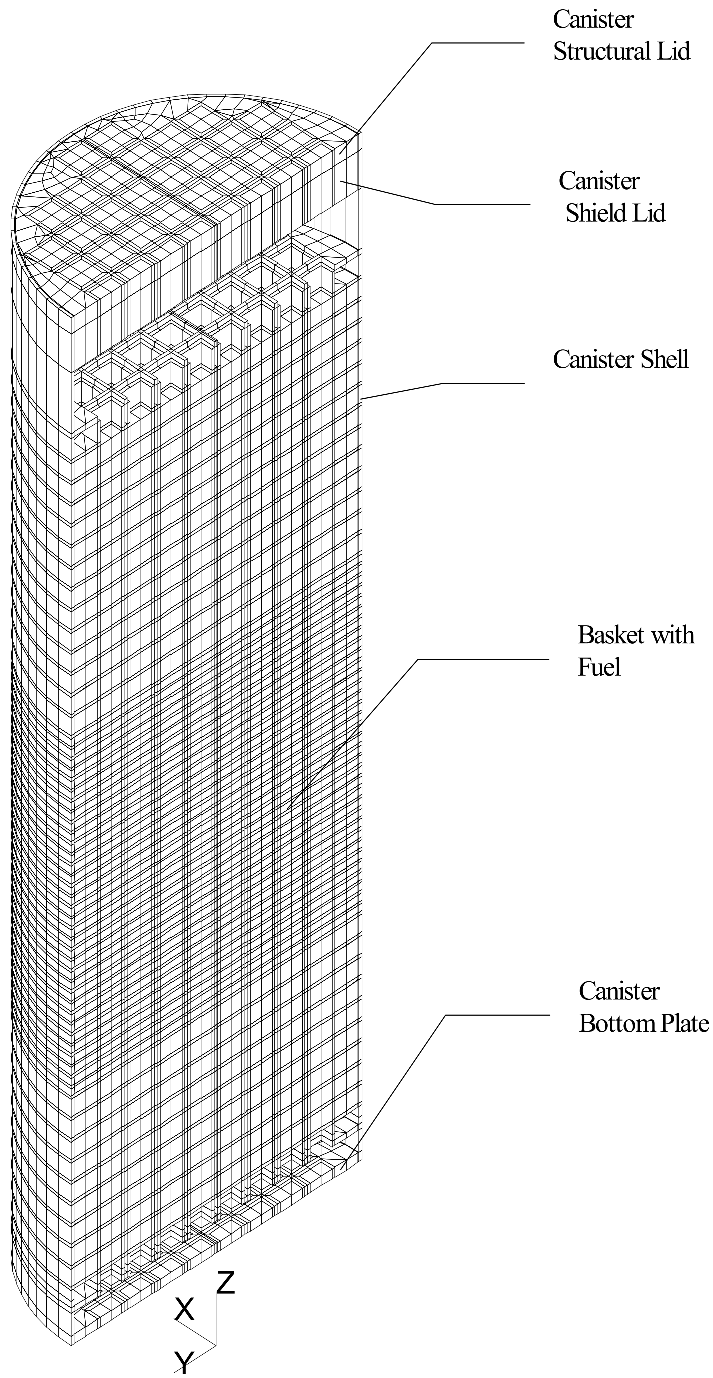


Figure 4.4.1.2-4 Three-Dimensional Canister Model for BWR Fuel – Cross Section

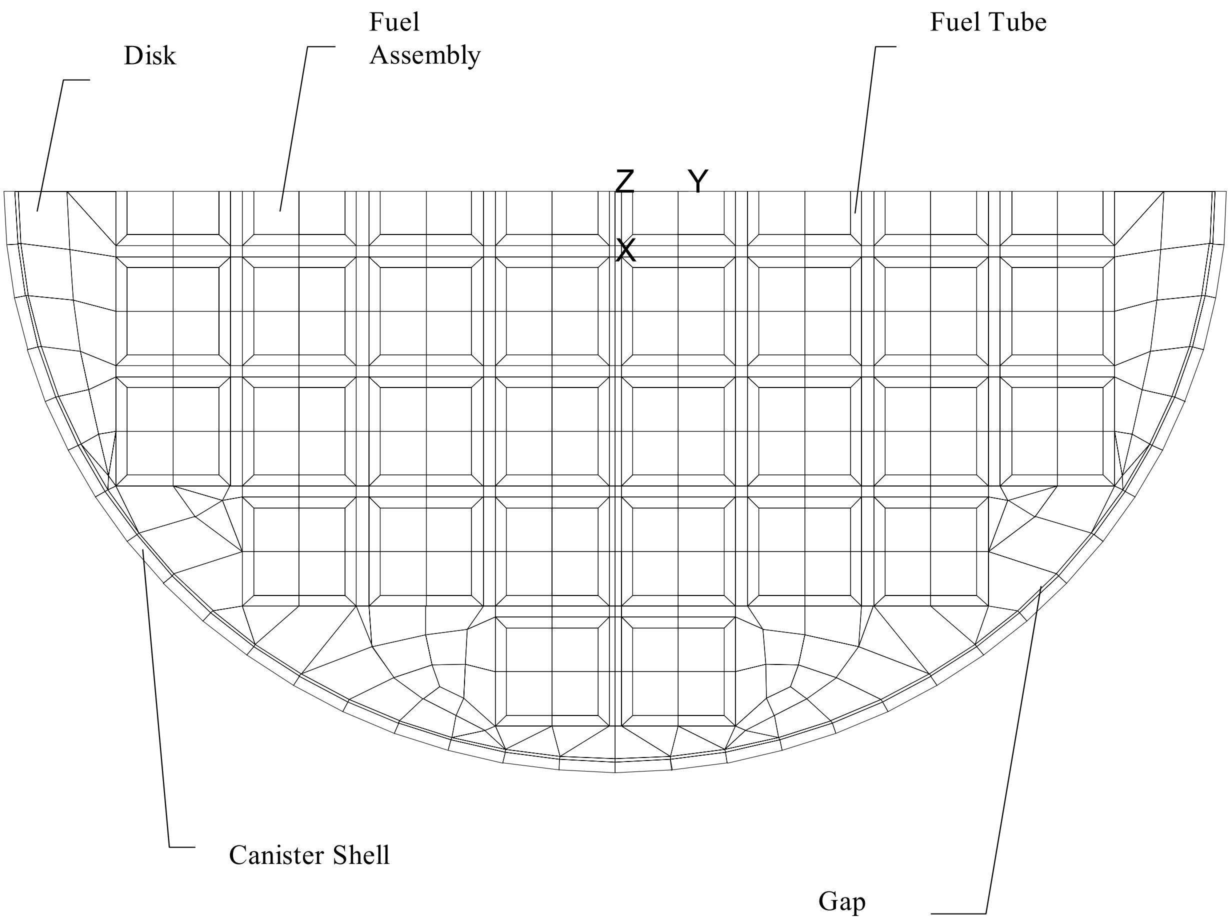


Table 4.4.1.2-1 Effective Thermal Conductivities for PWR Fuel Assemblies

Conductivity (Btu/hr-in-°F)	**Temperature (°F)**			
	220	**414**	**611**	**812**
Kxx	0.020	0.027	0.037	0.049
Kyy	0.020	0.027	0.037	0.049
Kzz	0.171	0.154	0.145	0.142

Note: x, y and z are in the coordinate system shown in Figure 4.4.1.2-1.

Table 4.4.1.2-2 Effective Thermal Conductivities for BWR Fuel Assemblies

Conductivity (Btu/hr-in-°F)	Temperature (°F)			
	186	**389**	**593**	**799**
Kxx	0.021	0.029	0.041	0.056
Kyy	0.021	0.029	0.041	0.056
Kzz	0.181	0.165	0.157	0.156

Note: x, y and z are in the coordinate system shown in Figure 4.4.1.2-3.

Table 4.4.1.2-3 Effective Thermal Conductivities for PWR Fuel Tubes

Fuel Assembly Group	Conductivity (Btu/hr-in-°F)	Temperature (°F)			
		206	405	604	803
In SS disk region					
	Kxx	0.022	0.028	0.033	0.040
	Kyy	1.54	1.57	1.59	1.61
	Kzz	1.54	1.57	1.59	1.61
In AL disk region					
	Kxx	0.022	0.027	0.032	0.038
	Kyy	1.54	1.57	1.59	1.61
	Kzz	1.54	1.57	1.59	1.61

Note: Kxx is in the direction across the thickness of the fuel tube wall.
Kyy is in the direction parallel to the fuel tube wall.
Kzz is in the canister axial direction.

Table 4.4.1.2-4 Effective Thermal Conductivities for BWR Fuel Tubes

	Conductivity	**Temperature (°F)**			
Tubes with Neutron Absorber	**(Btu/hr-in-°F)**	**200**	**400**	**600**	**800**
In CS disk region					
	Kxx	0.017	0.022	0.027	0.032
	Kyy	1.665	1.759	1.815	1.830
	Kzz	1.665	1.759	1.815	1.830
In AL disk region					
	Kxx	0.017	0.022	0.027	0.033
	Kyy	1.665	1.759	1.815	1.830
	Kzz	1.665	1.759	1.815	1.830
Tubes without Neutron Absorber		**200**	**400**	**600**	**800**
In CS disk region					
	Kxx	0.012	0.015	0.018	0.021
	Kyy	0.191	0.202	0.218	0.236
	Kzz	0.191	0.202	0.218	0.236
In AL disk region					
	Kxx	0.012	0.015	0.019	0.023
	Kyy	0.191	0.202	0.218	0.236
	Kzz	0.191	0.202	0.218	0.236

Note: Kxx is in the direction across the thickness of fuel tube wall.
Kyy is in the direction parallel to fuel tube wall.
Kzz is in the canister axial direction.

4.4.1.3 Three-Dimensional Transfer Cask and Canister Models

The three-dimensional quarter-symmetry transfer cask model is a representation of the PWR canister and transfer cask assembly. A half-symmetry model is used for the BWR canister and transfer cask. The model is used to perform a transient thermal analysis to determine the maximum water temperature in the canister for the period beginning immediately after removing the transfer cask and canister from the spent fuel pool. The model is also used to calculate the maximum temperature of the fuel cladding, the transfer cask and canister components during the vacuum drying condition and after the canister is backfilled with helium. The transfer cask is evaluated separately for PWR or BWR fuel using two models. For each fuel type, the class of fuel with the shortest associated canister and transfer cask is modeled in order to maximize the contents heat generation rate per unit volume and minimize the heat rejection from the external surfaces. The models for PWR and BWR fuel are shown in Figures 4.4.1.3-1 and 4.4.1.3-2, respectively. ANSYS SOLID70 three-dimensional conduction elements, LINK31 (PWR model) and MATRIX50 (BWR model) radiation elements are used. The model includes the transfer cask and the canister and its internals. The details of the canister and contents are modeled using the same methodology as that presented in Section 4.4.1.2 (Three-Dimensional Canister Models). Effective thermal properties for the fuel regions and the fuel tube regions are established using the fuel models and fuel tube models presented in Sections 4.4.1.5 and 4.4.1.6, respectively. The effective specific heat and density are calculated on the basis of material mass and volume ratio, respectively.

Radiation across the gaps was represented by the LINK31 elements or the MATRIX50 elements, which used the gray body emissivities for stainless and carbon steels. Convection is considered at the top of the canister lid, the exterior surfaces of the transfer cask, as well as at the annulus between the canister and the inner surface of the transfer cask. The combination of radiation and convection at the transfer cask exterior vertical surfaces and canister lid top surface is taken into account in the model using the same method described in Section 4.4.1.2 for the three-dimensional canister models. The bottom of the transfer cask is modeled as being in contact with the concrete floor. In the PWR configuration analysis, for the condition when the canister is filled with water at the start of the transfer operation, natural circulation of the water is taken into account by adjusting the effective conductivities in the fuel and water regions based on a classical energy balance calculation of the canister contents. Water circulation is not considered in the BWR configuration analysis. Volumetric heat generation (Btu/hr-in^3) is applied to the active fuel region based on a total heat load of 23 kW for both PWR and BWR fuel. The model considers the active fuel length of 144 inches and an axial power distribution, as shown in Figures 4.4.1.1-3 and 4.4.1.1-4 for PWR and BWR fuel, respectively.

An initial temperature of 100°F is considered in the entire model on the basis of the typical average water temperature in a spent fuel pool. For the design basis heat loads, the thermal transient analysis is performed for 20 hours (PWR) and 17 hours (BWR) with the water inside the canister, 27 hours (PWR) and 25 hours (BWR) for the vacuum condition, and 20 hours (PWR) and 16 hours (BWR) for the helium condition, followed by a steady-state analysis (in helium condition). Different time durations are used for the transient analyses for the reduced heat load cases, as specified in Section 4.4.3.1. The temperature history of the fuel cladding and the basket components, as well as the transfer cask components, is determined and compared with the short-term temperature limits presented in Tables 4.4.3-3 and 4.4.3-4.

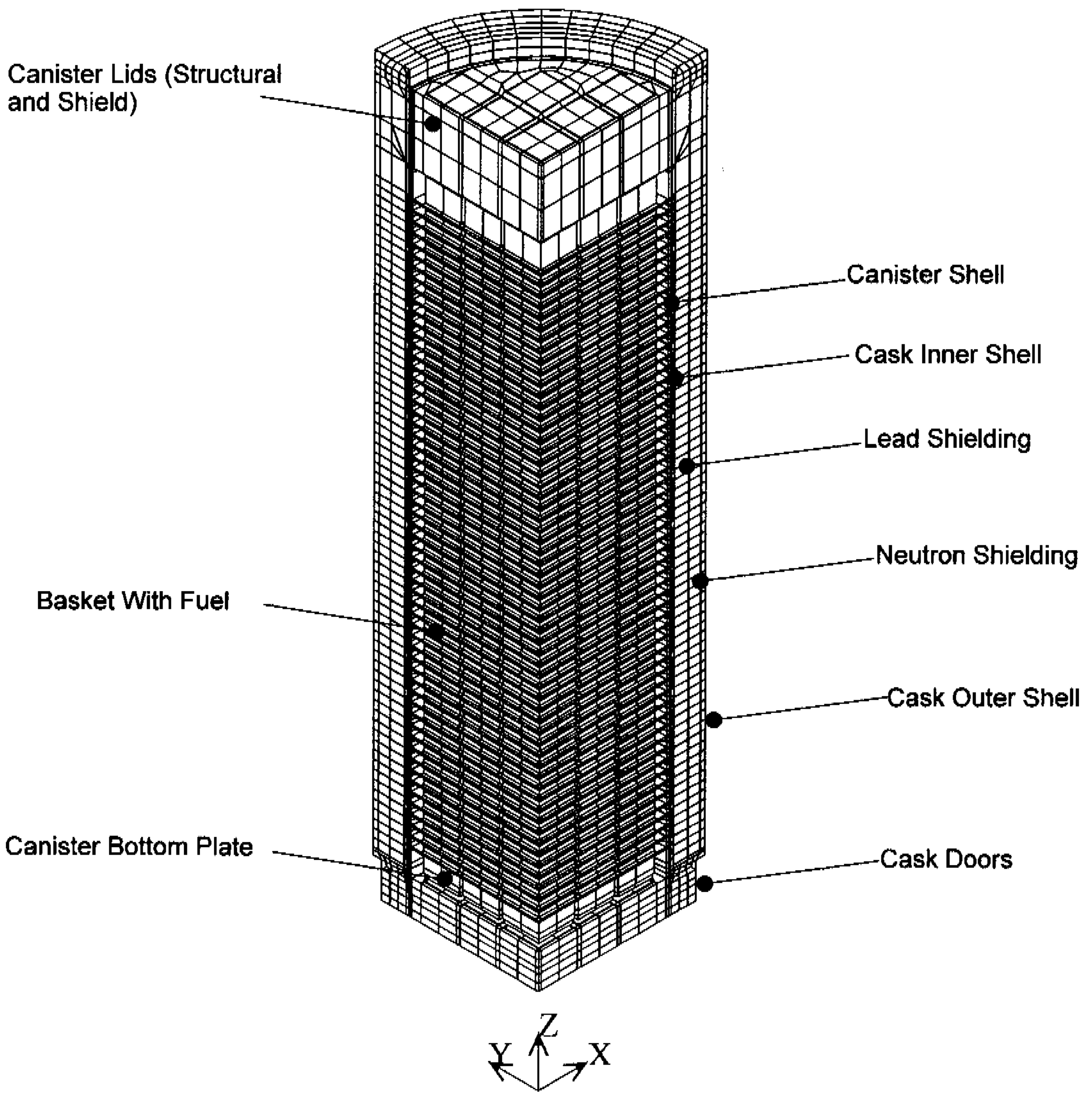
Note that the first phase of the thermal transient analysis considers that the canister is filled with water, including the period of canister draining as described in Step 12 of Section 8.1.1. A typical transportable storage canister drain-down process (performed by suction or by a blow-down gas pressure) ranges from 1 to 2 hours. The thermal analysis basis of assuming a water condition during drain-down is acceptable due to the following conservatisms in the thermal transient analysis for the transfer operation:

(1) The system as analyzed does not include the rejection of heat from the system due to the removal of water, which has significant thermal capacitance;

(2) The energy absorbed by the change in the state of residual water to steam, as the pressure is reduced during the vacuum drying phase of the transient, is ignored in the analysis; and

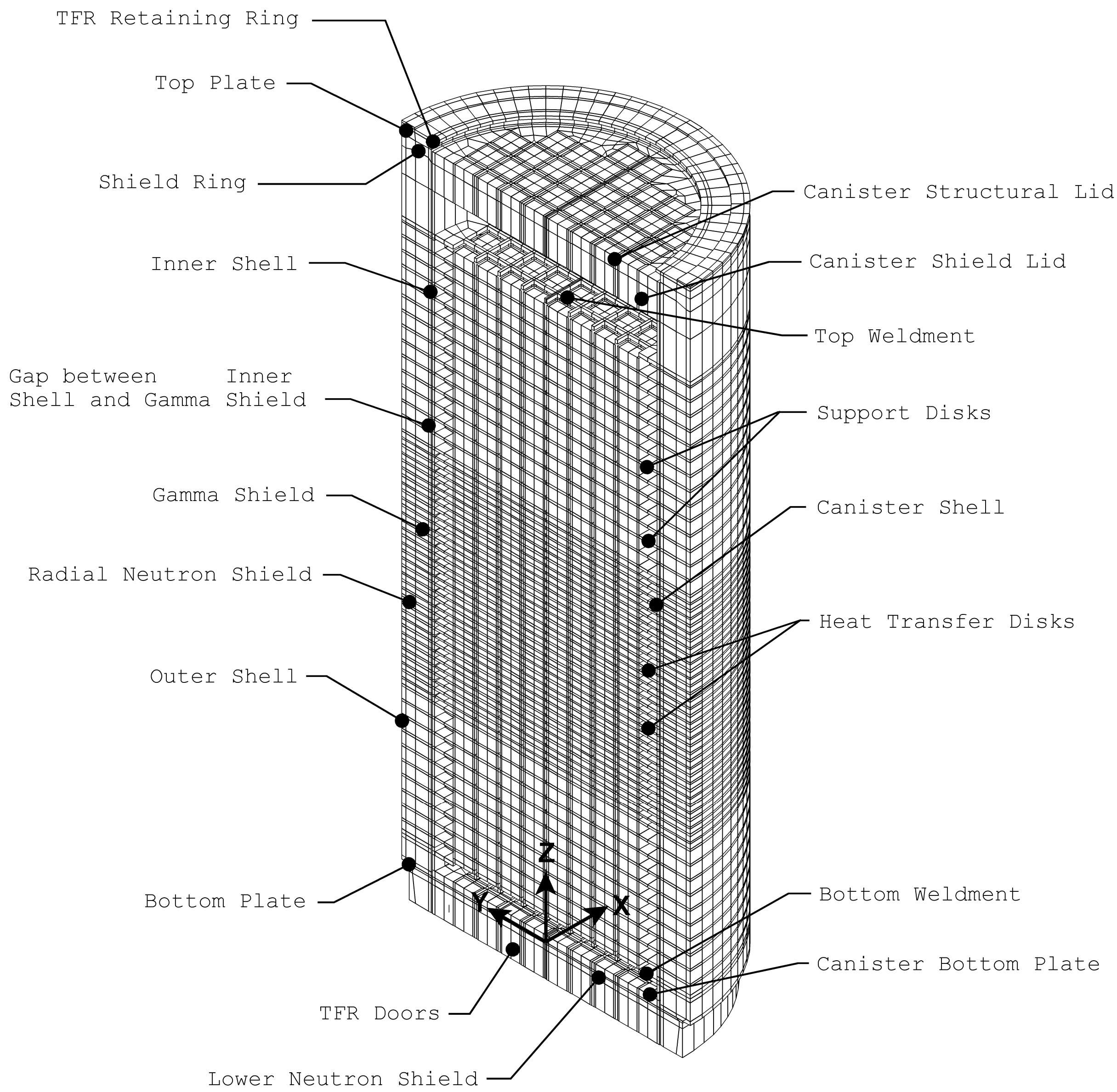
(3) No contact is considered between components in the transportable storage canister in the thermal model.

Figure 4.4.1.3-1 Three-Dimensional Transfer Cask and Canister Model - PWR



Note: Canister and transfer cask media not shown for clarity.

Figure 4.4.1.3-2 Three-Dimensional Transfer Cask and Canister Model - BWR



Canister and transfer cask media not shown for clarity.

4.4.1.4 Three-Dimensional Periodic Canister Internal Models

The three-dimensional periodic canister internal model consists of a periodic section of the canister internals. A total of three models are used: one for PWR fuel and two for BWR fuel. For the PWR canister, the model contains one support disk with two heat transfer disks (half thickness) on its top and bottom, the fuel assemblies, the fuel tubes and the media in the canister, as shown in Figure 4.4.1.4-1. The first BWR model, shown in Figure 4.4.1.4-2, represents the central region of the BWR canister, which contains one heat transfer disk with two support disks (half thickness) on its top and bottom, the fuel assemblies, the fuel tubes and the media in the canister. The second BWR model (not shown), for the region without heat transfer disks, contains two support disks (half thickness), the fuel assemblies, the fuel tubes and the media in the canister. The difference between the two BWR models is that the second model does not have the heat transfer disk. The purpose of these models is to determine the effective thermal conductivity of the canister internals in the canister radial direction. The effective conductivities are used in the two-dimensional axisymmetric air flow and concrete cask models. The media in the canister is considered to be helium. The fuel assemblies and fuel tubes in this model are represented by homogeneous regions with effective thermal properties. The effective conductivities for the fuel assemblies and the fuel tubes are determined by the two-dimensional fuel models (Section 4.4.1.5) and the two-dimensional fuel tube models (Section 4.4.1.6) respectively. The properties corresponding to the PWR 14 × 14 assemblies are used for the PWR model, since the 14 × 14 assemblies have the lowest conductivities as compared to other PWR assemblies. For the same reason, the properties corresponding to the BWR 9 × 9 assemblies are used for the BWR models.

The effective thermal conductivity (k_{eff}) of the fuel region in the radial direction is determined by considering the canister internals as a solid cylinder with heat generation. The temperature distribution in the cylinder may be expressed as [17]:

$$T - T_o = \frac{q''' R^2}{4k_{eff}}[1 - (\frac{r}{R})^2]$$

where:

T_o = the surface temperature of the cylinder

T = temperature at radius "r" of the cylinder

R = the outer radius of the cylinder,

r = radius

q''' = the heat generation rate = $\frac{Q}{\pi R^2 H}$

where: Q = total heat generated in the cylinder

H = length of the cylinder

Considering the temperature at the center of the canister to be T_{max}, the above equation can be simplified and used to compute the effective thermal conductivity (k_{eff}):

$$k_{eff} = \frac{Q}{4\pi H(T_{max} - T_o)} = \frac{Q}{4\pi H \Delta T}$$

where:

Q = total heat generated by the fuel

H = length of the active fuel region

T_o = temperature at outer surface internals (inside surface of the canister)

$\Delta T = T_{max} - T_o$

The value of ΔT is obtained from thermal analysis using the three-dimensional periodic canister internal model with the boundary temperature constrained to be T_o. The effective conductivity (k_{eff}) is then determined by using the above formula. Analysis is repeated by applying different boundary temperatures so that temperature-dependent conductivities can be determined.

Figure 4.4.1.4-1 Three-Dimensional Periodic Canister Internal Model - PWR

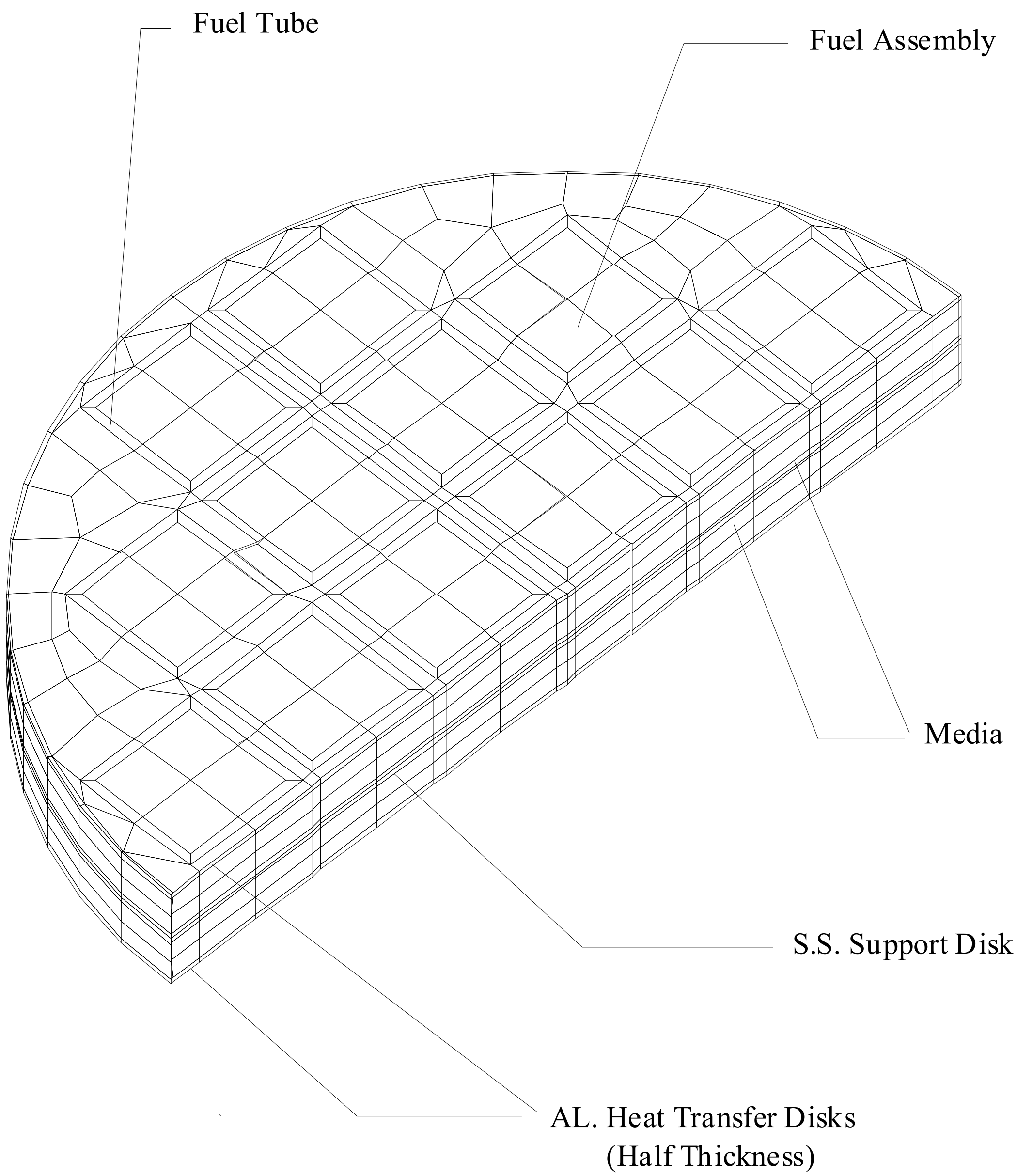
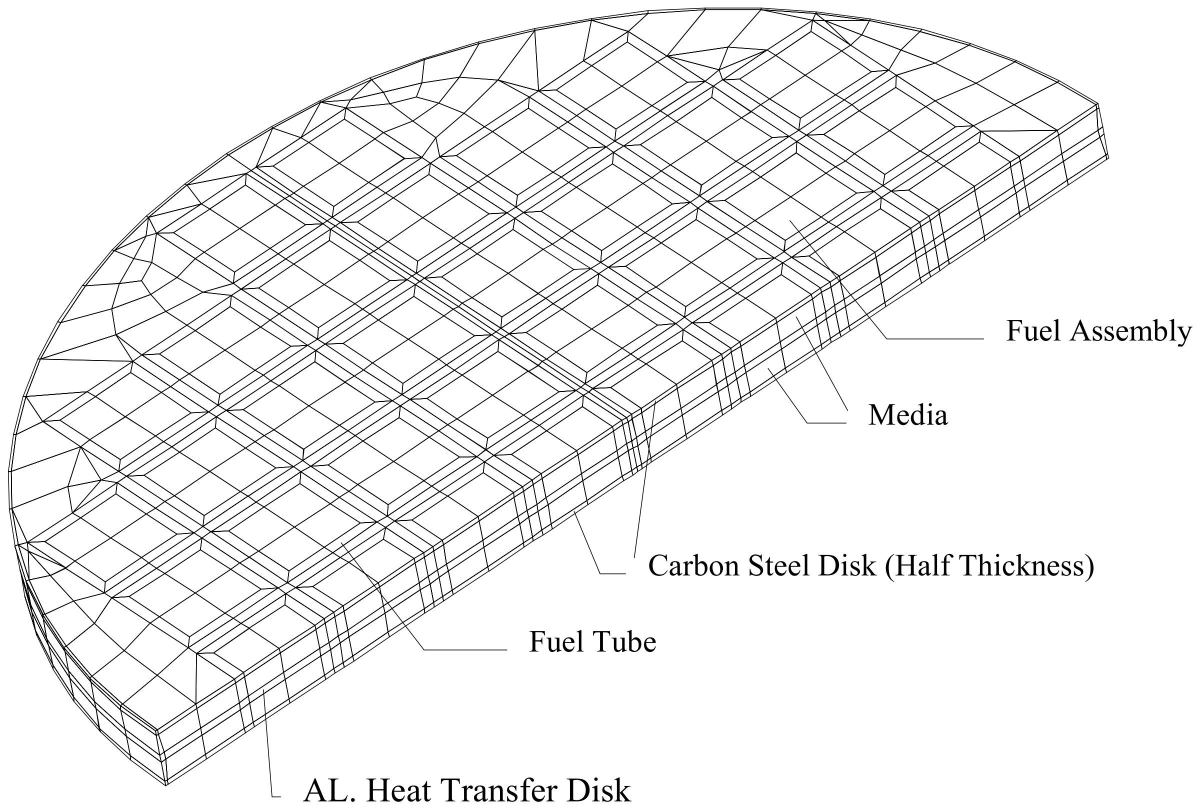


Figure 4.4.1.4-2 Three-Dimensional Periodic Canister Internal Model - BWR



4.4.1.5 Two-Dimensional Fuel Models

The effective conductivity of the fuel is determined by the two-dimensional finite element model of the fuel assembly. The effective conductivity is used in the three-dimensional canister models (Section 4.4.1.2) and the three-dimensional periodic canister internal models (Section 4.4.1.4). A total of seven models are required: four models for the 14×14, 15×15, 16×16 and 17×17 PWR fuels and three models for the 7×7, 8×8 and 9×9 BWR fuels. Because of similarity, only the figure for the PWR 17×17 model is shown in this section (Figure 4.4.1.5-1). All models contain a full cross-section of an assembly to accommodate the radiation elements.

The model includes the fuel pellets, cladding, media between fuel rods, media between the fuel rods and the inner surface of the fuel tube (PWR) or fuel channel (BWR), and helium at the gap between the fuel pellets and cladding. Four types of media are considered: helium, water, a vacuum and saturated steam. Modes of heat transfer modeled include conduction and radiation between individual fuel rods for the steady-state condition. ANSYS PLANE55 conduction elements and MATRIX50 radiation elements are used to model conduction and radiation. Radiation elements are defined between fuel rods and from rods to the wall. Radiation at the gap between the pellets and the cladding is conservatively ignored.

The effective conductivity for the fuel is determined by using an equation defined in a Sandia National Laboratory Report [30]. The equation is used to determine the maximum temperature of a square cross-section of an isotropic homogeneous fuel with a uniform volumetric heat generation. At the boundary of the square cross-section, the temperature is constrained to be uniform. The expression for the temperature at the center of the fuel is given by:

$$T_c = T_e + 0.29468\ (Qa^2 / K_{eff})$$

where: T_c = the temperature at the center of the fuel (°F)

T_e = the temperature applied to the exterior of the fuel (°F)

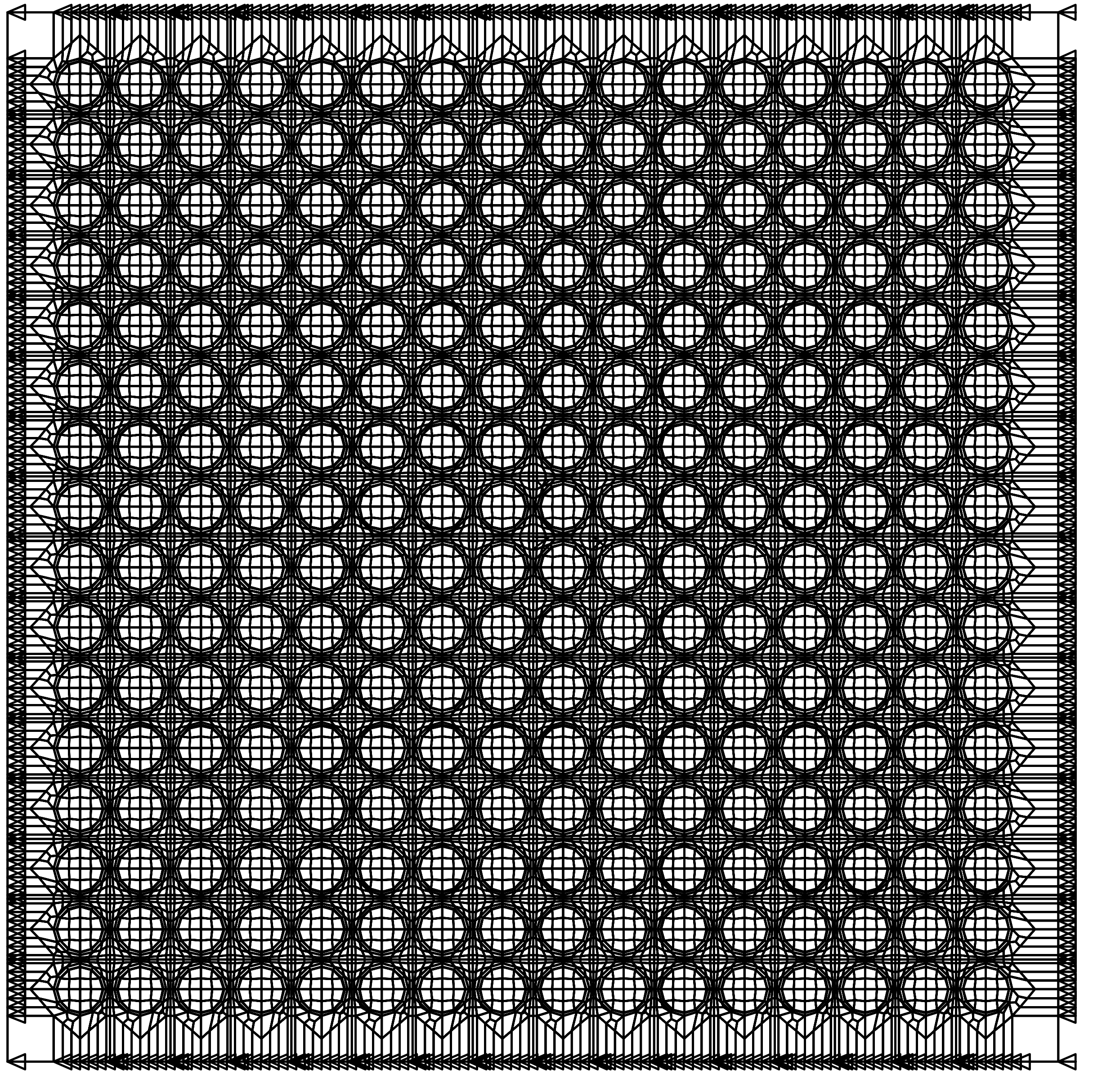
Q = volumetric heat generation rate (Btu/hr-in^3)

a = half length of the square cross-section of the fuel (inch)

K_{eff} = effective thermal conductivity for the isotropic homogeneous fuel material (Btu/hr-in-°F)

Volumetric heat generation (Btu/hr-in^3) based on the design heat load is applied to the pellets. The effective conductivity is determined based on the heat generated and the temperature difference from the center of the model to the edge of the model. Temperature-dependent effective properties are established by performing multiple analyses using different boundary temperatures. The effective conductivity in the axial direction of the fuel assembly is calculated on the basis of the material area ratio.

Figure 4.4.1.5-1 Two-Dimensional PWR (17×17) Fuel Model



4.4.1.6 Two-Dimensional Fuel Tube Models

The two-dimensional fuel tube model is used to calculate the effective conductivities of the fuel tube wall and BORAL plate. These effective conductivities are used in the three-dimensional canister models (Section 4.4.1.2), the three-dimensional transfer cask and canister models (Section 4.4.1.3) and the three-dimensional periodic canister internal models (Section 4.4.1.4). A total of three models is required: one PWR model and two BWR models (one with the neutron absorber plate, one without the neutron absorber plate), corresponding to the enveloping configurations of the 7×7, 8×8 and 9×9 BWR fuels.

In the neutron absorber evaluation, the configuration shown in the fuel tube models in Figures 4.4.1.6-1 and 4.4.1.6-2 (for PWR and BWR fuel, respectively) incorporates the BORAL core matrix sandwiched between two layers of aluminum cladding. The thermal properties of BORAL are presented in Table 4.2-10.

As shown in Figure 4.4.1.6-1, the PWR model includes the fuel tube, the BORAL plate (including the core matrix sandwiched by aluminum cladding), the stainless steel cladding and the gap between the stainless steel cladding and the support disk or heat transfer disk. Four types of media are considered in the gaps: helium, water, a vacuum and saturated steam.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used to construct the model. The model consists of six layers of conduction elements and two radiation elements (radiation elements are not used for water condition) that are defined at the gaps (two for each gap). The thickness of the model (x-direction) is the distance measured from the outside face of the fuel assembly to the inside face of the slot in the support disk (assuming the fuel tube is centered in the hole in the disk). The gap size between the neutron absorber plate and the stainless steel cladding is 0.003 inch. The height of the model is defined as equal to the width of the model.

The fuel tubes in the BWR fuel basket differ from those in the PWR fuel basket in that not all sides of the fuel tubes contain neutron absorber. In addition, the BWR fuel assembly is contained in a fuel channel. Therefore, two effective conductivity models are necessary, one fuel tube model with the neutron absorber plate (a total of eight layers of materials) and another fuel tube model with a gap replacing the neutron absorber plate (a total of four layers of materials).

As shown in Figure 4.4.1.6-2, the BWR fuel tube model with neutron absorber includes the fuel channel, the gap between the fuel channel and fuel tube, the fuel tube, the neutron absorber plate (including the core matrix sandwiched by aluminum claddings), and a gap between the stainless steel cladding for the neutron absorber plate and the support disk or heat transfer disk. The effective conductivity of the fuel tube without the neutron absorber plate is determined using the second BWR fuel tube model. As shown in Figure 4.4.1.6-3, this model includes the gap between fuel assembly and the fuel channel, the fuel channel, gap between the fuel channel and stainless steel fuel tube, the fuel tube, and a gap between the fuel tube and the support disk or heat transfer disk. An emissivity value of 0.0001 is conservatively used for the BWR support disk in the model.

Heat flux is applied at the left side of the model (fuel tube for PWR models and fuel channel for BWR models), and the temperature at the right boundary of the model is constrained. The heat flux is determined based on the design heat load. The maximum temperature of the model (at the left boundary) and the temperature difference (ΔT) across the model are calculated by the ANSYS model. The effective conductivity (K_{xx}) is determined using the following formula:

$$q = K_{xx} (A/L) \Delta T$$

or

$$K_{xx} = q \; L / (A \, \Delta T)$$

where:

K_{xx} = effective conductivity (Btu/hr-in-°F) in X direction in Figure 4.4.1.6-1.
q = heat rate (Btu/hr)
A= area (in^2)
L= length (thickness) of model (in)
ΔT = temperature difference across the model (°F)

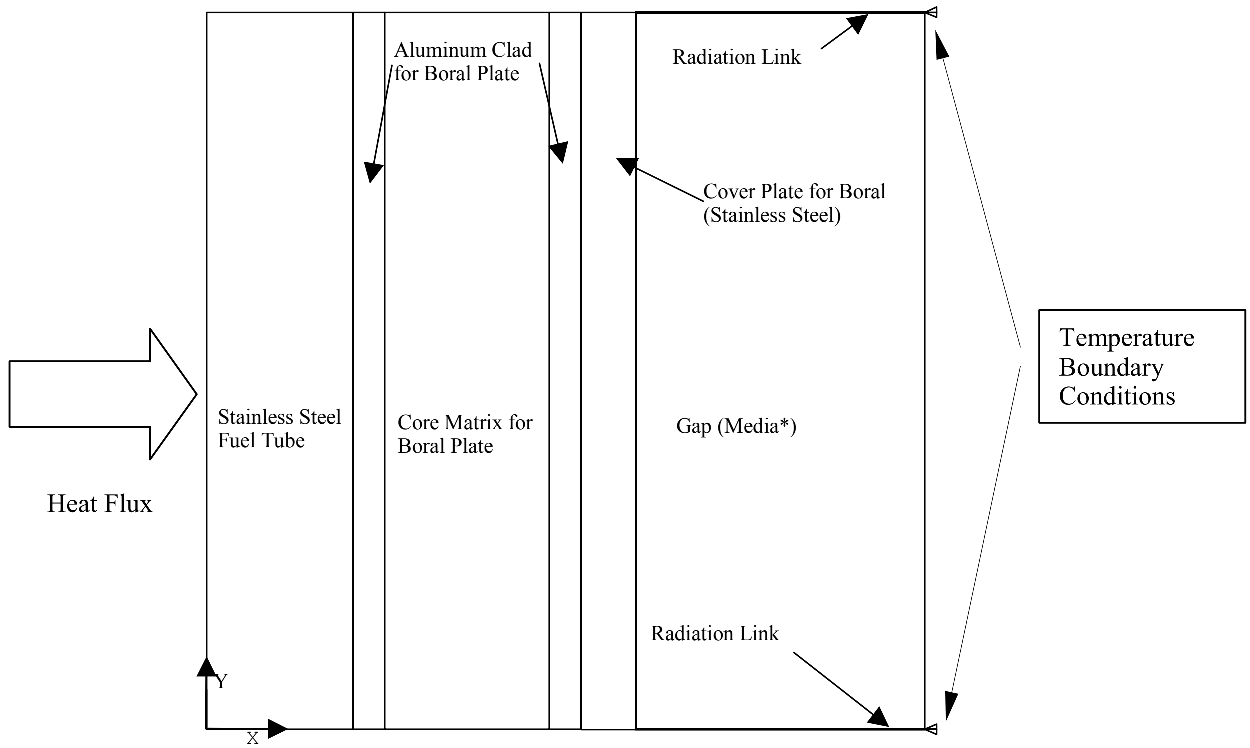
The temperature-dependent conductivity is determined by varying the temperature constraints at one boundary of the model and resolving for the heat rate (q) and temperature difference. The effective conductivity for the parallel path (the Y direction in Figure 4.4.1.6-1) is calculated by:

$$K_{yy} = \frac{\sum K_i \, t_i}{L}$$

where:

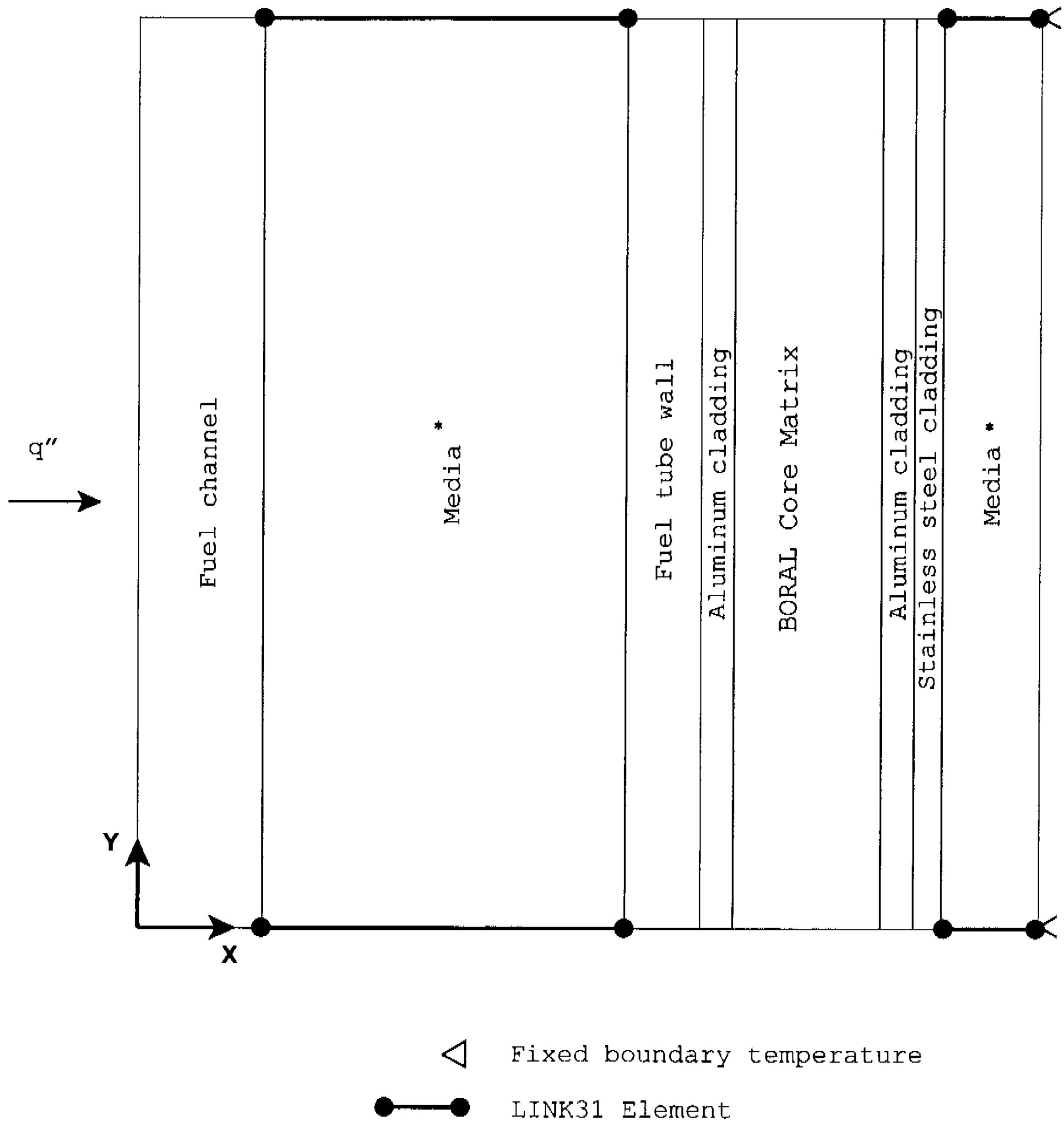
K_i = thermal conductivity of each layer
t_i = thickness of each layer
L = total length (thickness) of the model

Figure 4.4.1.6-1 Two-Dimensional Fuel Tube Model: PWR Fuel



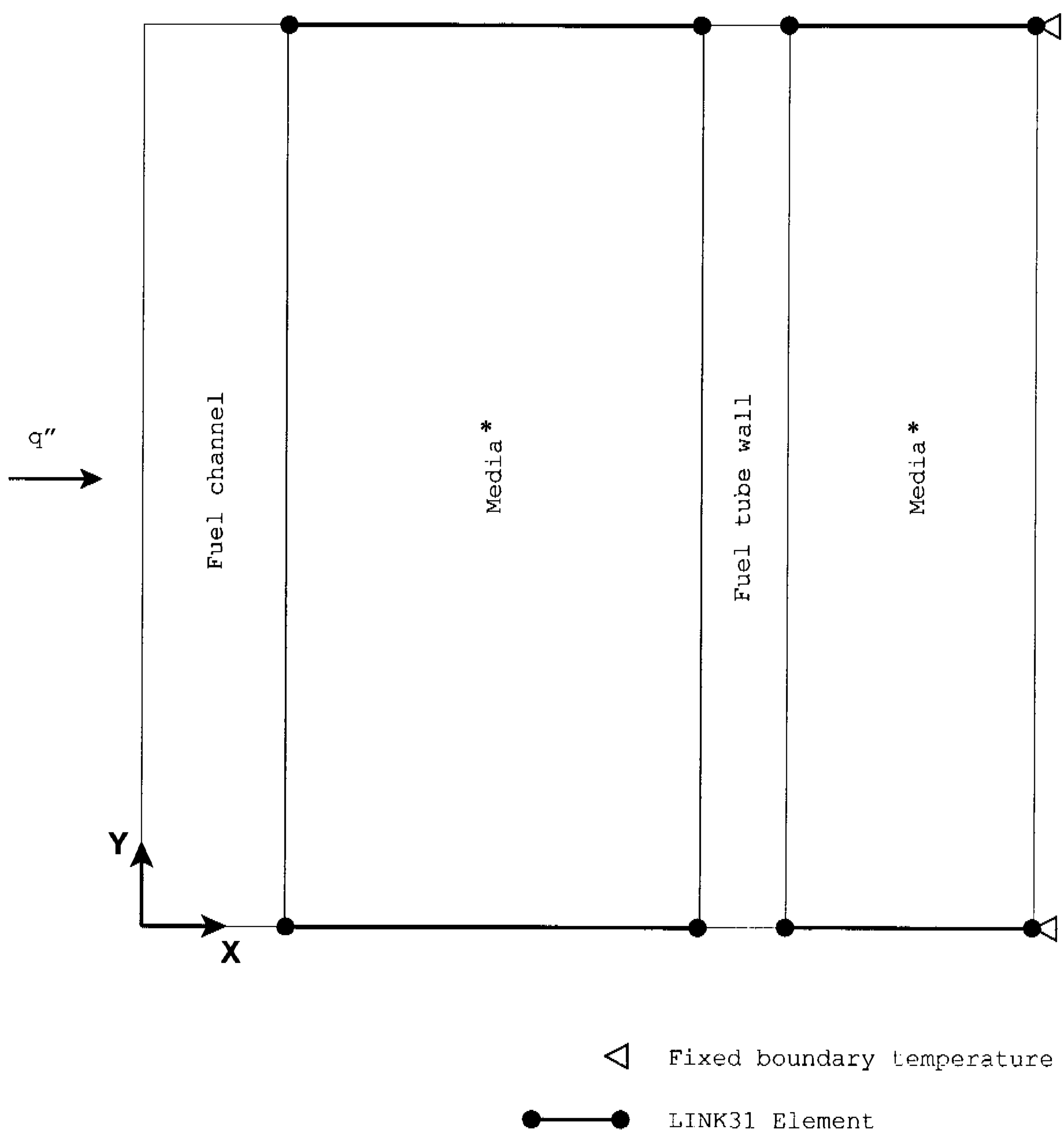
*Media can be water, vacuum, helium or saturated steam.

Figure 4.4.1.6-2 Two-Dimensional Fuel Tube Model: BWR Fuel Tube with Neutron Absorber



*Media can be water, vacuum, helium or saturated steam.

Figure 4.4.1.6-3 Two-Dimensional Fuel Tube Model: BWR Fuel Tube without Neutron Absorber



*Media can be water, vacuum, helium or saturated steam.

4.4.1.7 Two-Dimensional Forced Air Flow Model for Transfer Cask Cooling

A two-dimensional axisymmetric air flow model is used to determine the air flow rate needed to ensure that the maximum temperature of the canister shell and canister components inside the transfer cask do not exceed those presented in Tables 4.4.3-3 and 4.4.3-4 for the helium condition. This air flow model considers a 0.34-inch air annulus between the outer surface of the canister shell and the inner surface of the transfer cask, and has a total length of 191-inches. The fuel canister is cooled by forced convection in the air annulus resulting from air pumped in through fill/drain ports in the body of the transfer cask. The radiation heat transfer between the vertical annulus surfaces (the canister shell outer surface and the transfer cask inner surface) is conservatively neglected. All heat is considered to be removed by the air flow.

ANSYS FLOTRAN FLUID141 fluid thermal elements are used to construct the two-dimensional axisymmetric air flow finite element model for transfer cask cooling. The model and the boundary conditions applied to the model, are shown in Figures 4.4.1.7-1, 4.4.1.7-2 and 4.4.1.7-3.

As shown in Tables 4.4.3-3 and 4.4.3-4, the temperature margin of the governing component (the heat transfer disk) for the PWR fuel configuration is lower than the margin for the BWR fuel configuration; therefore, the thermal loading for the PWR configuration is used. The non-uniform heat generation applied in the model, shown in Figure 4.4.1.7-4, is based on the axial power distribution shown in Figure 4.4.1.1-3 for PWR fuel.

The inlet air velocity is specified based on the volume flow rate. Room temperature (76°F) is applied to the inlet nodes, while zero air velocity, in both the X and Y directions, is defined as the boundary condition for the vertical solid sides.

Results of the analyses of forced air cooling of the canister inside the transfer cask are shown in Figure 4.4.1.7-5. As shown in the figure, the maximum canister shell temperature is less than 416°F for a forced air flow rate of 275 ft^3/minute, or higher, where 416°F is the calculated maximum canister shell temperature for the typical transfer operation for the PWR configuration (Table 4.4.3-3). A forced air volume flow rate of 375 ft^3/minute is conservatively specified for cooling the canister in the event that forced air cooling is required. Evaluation of a forced air volume flow rate of 375 ft^3/minute, results in a maximum canister shell temperature of 321°F, which is significantly less than the design basis temperature of 416°F.

Figure 4.4.1.7-1 Two-Dimensional Axisymmetric Finite Element Model for Transfer Cask Forced Air Cooling

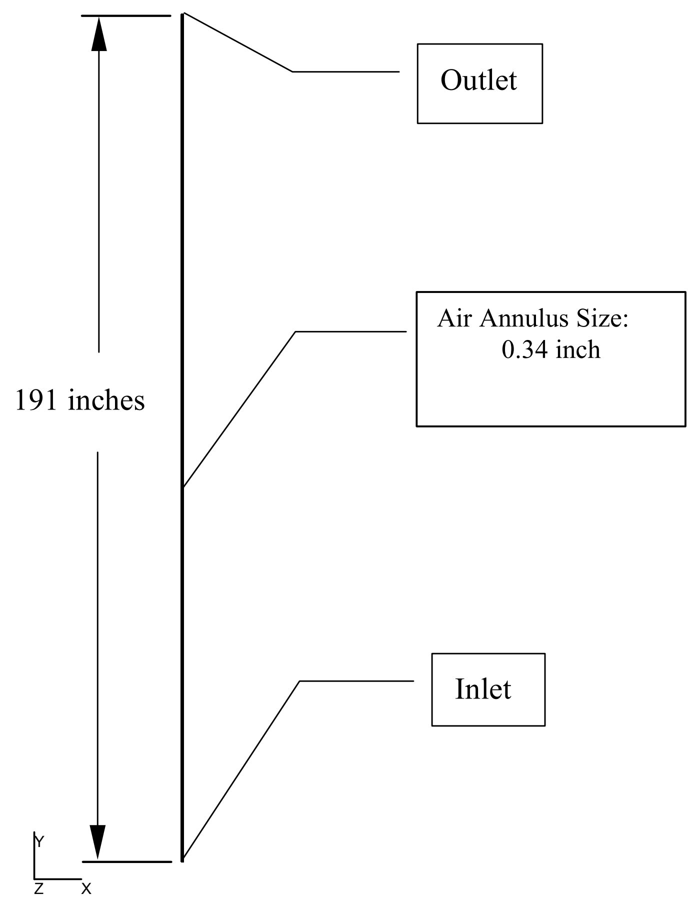


Figure 4.4.1.7-2 Two-Dimensional Axisymmetric Outlet Air Flow Model for Transfer Cask Cooling

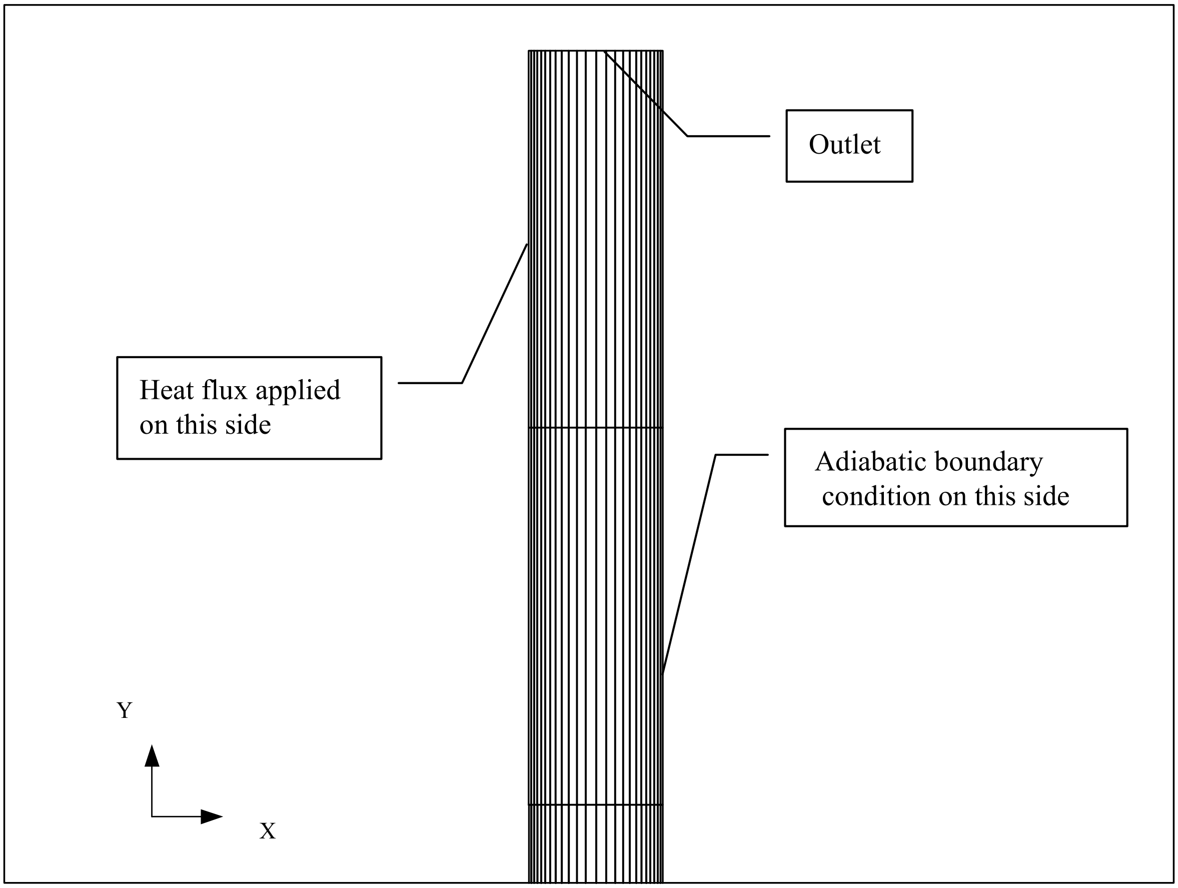


Figure 4.4.1.7-3 Two-Dimensional Axisymmetric Inlet Air Flow Model for Transfer Cask Cooling

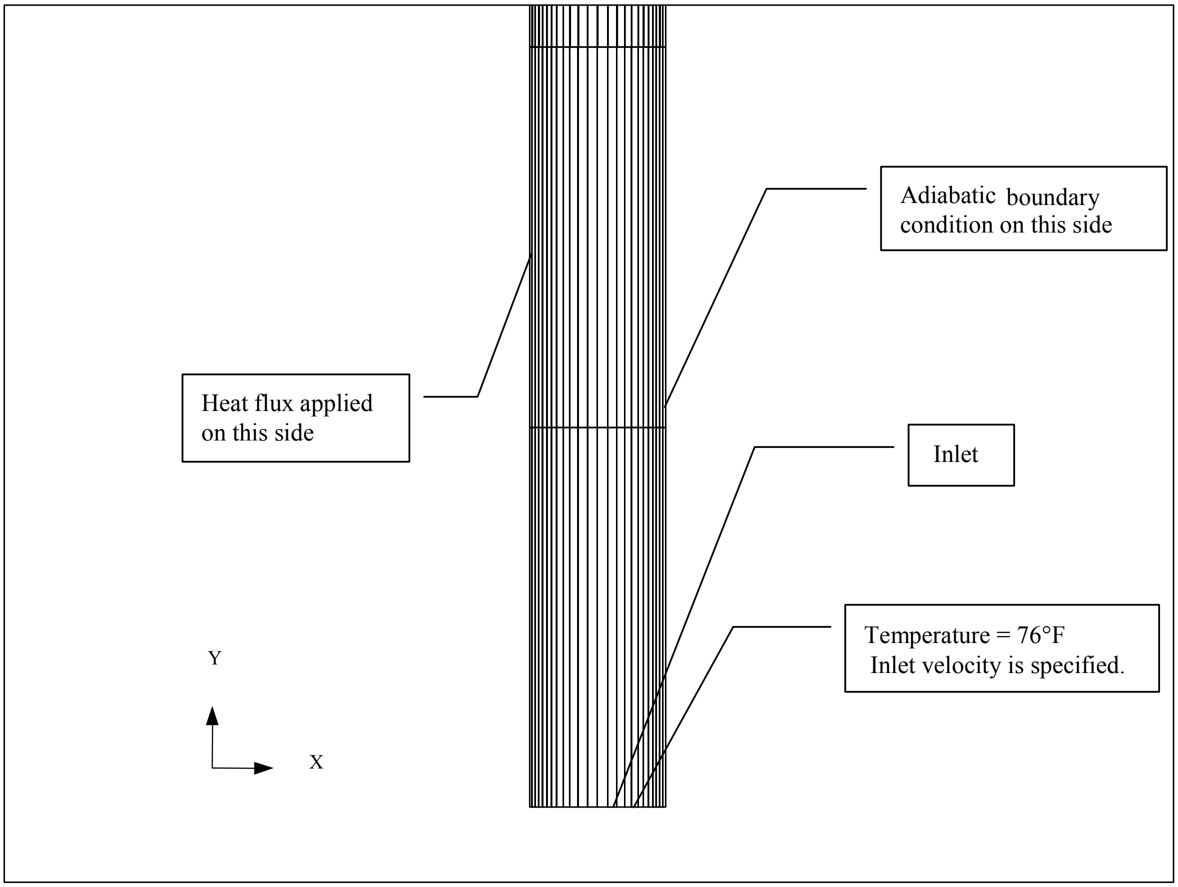


Figure 4.4.1.7-4 Non-Uniform Heat Load from Canister Contents

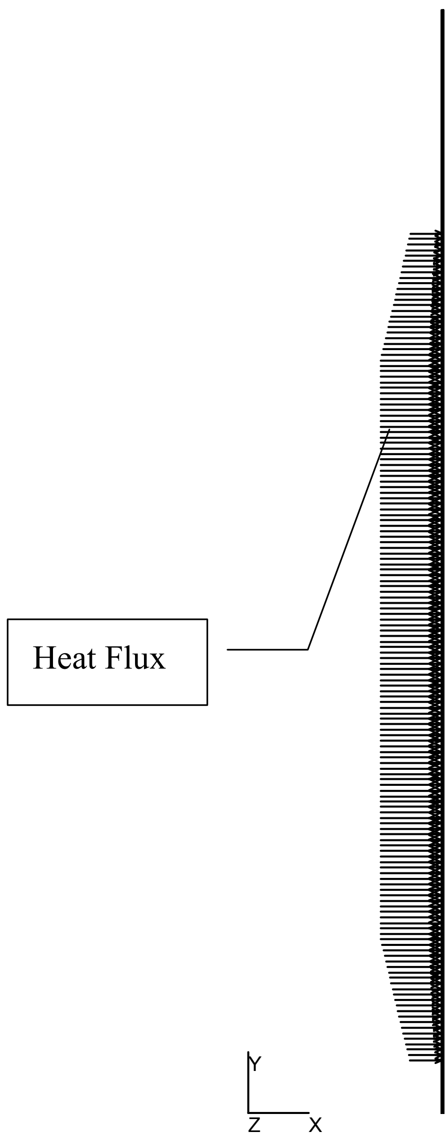
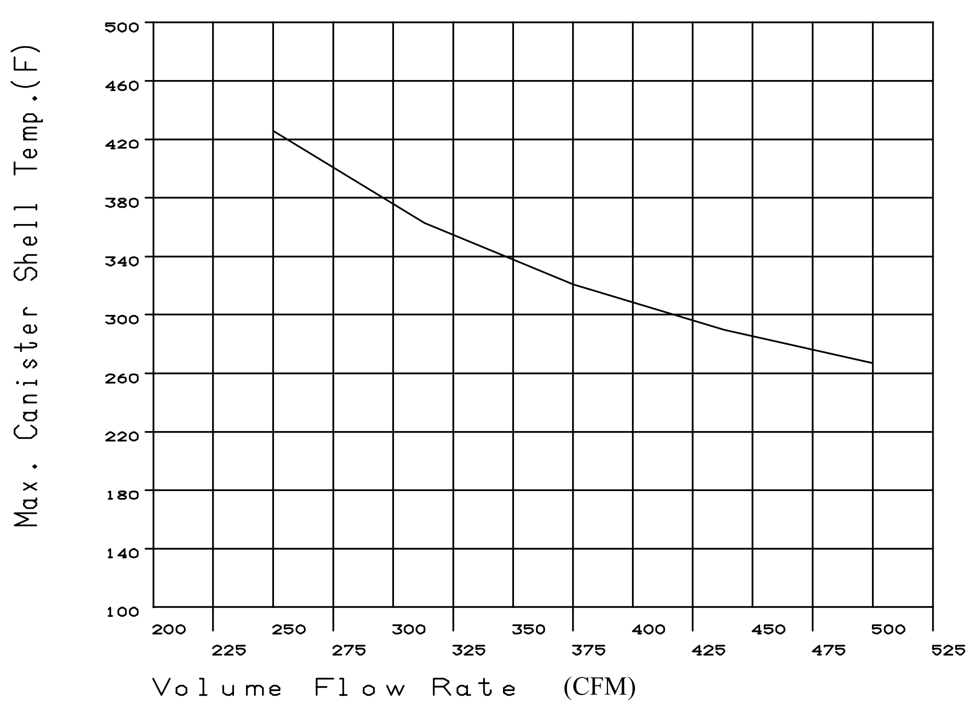


Figure 4.4.1.7-5 Maximum Canister Temperature Versus Air Volume Flow Rate



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4.4.2 Test Model

The Universal Storage System is conservatively designed by analysis. Therefore, no physical model is employed for thermal analysis.

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4.4.3 Maximum Temperatures for PWR and BWR Fuel

Temperature distribution and maximum component temperatures for the Universal Storage System under the normal conditions of storage and transfer, based on the use of the transfer cask, are provided in this section. Components of the Universal Storage System containing PWR and BWR fuels are addressed separately. Temperature distributions for the evaluated off-normal and accident conditions are presented in Sections 11.1 and 11.2.

Figure 4.4.3-1 shows the temperature distribution of the Vertical Concrete Cask and the canister containing the PWR design basis fuel for the normal, long-term storage condition. The air flow pattern and air temperatures in the annulus between the PWR canister and the concrete cask liner for the normal condition of storage are shown in Figures 4.4.3-2 and 4.4.3-3, respectively. The temperature distribution in the concrete portion of the concrete cask for the PWR assembly is shown in Figure 4.4.3-4. The temperature distribution for the BWR design basis fuel is similar to that of the PWR fuel and is, therefore, not presented. Table 4.4.3-1 shows the maximum component temperatures for the normal condition of storage for the PWR design basis fuel. The maximum component temperatures for the normal condition of storage for the BWR design basis fuel are shown in Table 4.4.3-2.

As shown in Figure 4.4.3-3, a high-temperature gradient exists near the wall of the canister and the liner of the concrete cask, while the air in the center of the annulus exhibits a much lower temperature gradient, indicating significant boundary layer features of the air flow. The temperatures at the concrete cask steel liner surface are higher than the air temperature, which indicates that salient radiation heat transfer occurs across the annulus. As shown in Figure 4.4.3-4, the local temperature in the concrete, directly affected by the radiation heat transfer across the annulus, can reach 186°F (less than the 200°F allowable temperature). The bulk temperature in the concrete, as determined using volume average of the temperatures in the concrete region, is 135°F, less than the allowable value of 150°F.

Under typical operations, the transient history of maximum component temperatures for the transfer conditions (canister, inside the transfer cask, containing water for 20 hours for PWR and 17 hours for BWR, vacuum for 27 hours for PWR and 25 hours for BWR, and in helium for 20 hours for PWR and 16 hours for BWR) is shown in Figures 4.4.3-5 and 4.4.3-6 for PWR and BWR fuels, respectively. The maximum component temperatures for the transfer conditions (vacuum and helium conditions) are shown in Tables 4.4.3-3 and 4.4.3-4, for PWR and BWR fuels, respectively. Note that the media inside the canister is considered to be saturated steam during the first four hours of the vacuum condition.

The maximum calculated water temperature is 203°F for both the PWR and BWR fuels at the end of 17 hours based on an initial water temperature of 100°F.

4.4.3.1 Maximum Temperatures at Reduced Total Heat Loads

This section provides the evaluation of component temperatures for fuel heat loads less than the design basis heat load of 23 kW. Transient thermal analyses are performed for PWR fuel heat loads of 20, 17.6, 14, 11 and 8 kW to establish the allowable time limits for the vacuum condition in the canister as described in the Technical Specifications for the Limiting Conditions of Operation (LCO), LCOs 3.1.1 and 3.1.4. The time limits ensure that the allowable temperatures of the limiting components — the heat transfer disks and the fuel cladding — are not exceeded. A steady-state evaluation is also performed for all the heat load cases in the vacuum condition and all the heat load cases in the helium condition. If the steady-state temperature calculated is less than the limiting component allowable temperature, then the allowable time duration in the vacuum or helium conditions is defined to be 600 hours (25 days) based on the 30 day time test for abnormal regimes as described in PNL-4835 [34].

The three-dimensional transfer cask and canister model for the PWR fuel configuration, described in Section 4.4.1.3, is used for the transient and steady-state thermal analysis for the reduced heat load cases. To obtain the bounding temperatures for all possible loading configurations, thermal analyses are performed for a total of 14 cases as tabulated in the following table. The basket locations are shown in Figure 4.4.3-7. Since the maximum temperature for the limiting components (fuel cladding and heat transfer disk) always occurs at the central region of the basket, hotter fuels (maximum allowable heat load for 5-year cooled fuel: 0.958 kW = 23 kW/24) are specified at the central basket locations. The bounding cases for each heat load condition are noted with an asterisk (*) in the tabulation which follows. Six cases (cases 3 through 8) are evaluated for the 17.6 kW heat load condition. The first four cases (cases 3 through 6) represent standard UMS® system fuel loadings. The remaining two cases (cases 7 and 8) account for the preferential loading configuration for Maine Yankee site-specific fuel (Section 4.5.1.2), with case 8 being the bounding case for the Maine Yankee fuel.

Canister Heat Load (kW)	Heat Load Case	Heat Load (kW) Evaluated in Each Basket Location (See Figure 4.4.3-7)					
		1	2	3	4	5	6
20	1	0.958	0.958	0.709	0.958	0.709	0.709
20*	2	0.958	0.958	0.958	0.958	0.958	0.210
17.6	3	0.958	0.958	0.509	0.958	0.509	0.509
17.6*	4	0.958	0.958	0.568	0.958	0.958	0.000
17.6	5	0.958	0.958	0.958	0.958	0.568	0.000
17.6	6	0.958	0.958	0.284	0.958	0.958	0.284
17.6	7	0.958	0.146	1.050	0.146	1.050	1.050
17.6	8	0.958	0.958	1.050	0.384	1.050	0.000
14	9	0.958	0.958	0.209	0.958	0.209	0.209
14*	10	0.958	0.958	0.000	0.958	0.626	0.000
11	11	0.958	0.896	0.000	0.896	0.000	0.000
11*	12	0.958	0.958	0.000	0.834	0.000	0.000
8	13	0.958	0.521	0.000	0.521	0.000	0.000
8*	14	0.958	0.958	0.000	0.084	0.000	0.000

The heat load (23 kW/24 Assemblies = 0.958 kW) at the four (4) central basket locations corresponds to the maximum allowable canister heat load for 5-year cooled fuel (Table 4.4.7-8). The non-uniform heat loads evaluated in this section bound the equivalent uniform heat loads, since they result in higher maximum temperatures of the fuel cladding and heat transfer disk.

Volumetric heat generation (Btu/hr-in^3) is applied to the active fuel region in each fuel assembly location of the model using the axial power distribution for PWR fuel (Figure 4.4.1.1-3) in the axial direction.

The thermal analysis results for the closure and transfer of a loaded PWR fuel canister in the transfer cask for the reduced heat load cases are shown in Table 4.4.3-5, with a comparison to the results for the design basis heat load case. The temperatures shown are the maximum temperatures for the limiting components (fuel cladding and heat transfer disk). The maximum temperatures of the fuel cladding and the heat transfer disk are less than the allowable temperatures (Table 4.1-3) of these components for the short-term conditions of vacuum drying and helium backfill. As shown in Table 4.4.3-5, a time limit of 600 hours is specified for moving the canister out of the transfer cask after the canister is filled with helium. This time limit is for the heat load cases where the maximum fuel cladding/heat transfer disk temperatures for the steady-state condition are below the short-term allowable temperatures. Based on the differences in the PWR and BWR models for the transient analysis of the “water period” (see Section 4.4.1.3), a different method is used in post-processing the analysis results to determine the maximum water temperature at the end of the “water period.” For the PWR configuration, the maximum water temperature is considered to be the maximum temperature of the fuel region in the model. For the

BWR configuration, the maximum water temperature is considered to be the volumetric average temperature of the calculated cladding temperatures in the active fuel region of the hottest fuel assembly. The maximum water temperature is below 212°F for all PWR and BWR cases evaluated.

The Technical Specifications specify the remedial actions, either in-pool or forced air cooling, required to ensure that the fuel cladding and basket component temperatures do not exceed their short-term allowable temperatures, if the time limits are not met. LCOs 3.1.1 and 3.1.4 incorporate the operating times for heat loads that are less than the design basis heat loads as evaluated in this section.

Using the same three-dimensional transfer cask/canister models, analysis is performed for the conditions of in-pool cooling and forced air cooling followed by the vacuum drying and helium backfill operation (LCO 3.1.1). The conditions at the end of the vacuum drying as shown in Tables 4.4.3-5 (PWR) and 4.4.3-8 (BWR) are used as the initial conditions of the analyses. The LCO 3.1.1 "Action" analysis results are shown in Tables 4.4.3-6 and 4.4.3-7 for the PWR configuration and Tables 4.4.3-9 and 4.4.3-10 for the BWR configuration. Note that the duration of the second vacuum (after completion of the in-pool or forced air cooling) is limited (calculated based on the heat-up rate of the first vacuum), so the maximum temperatures at the end of the second vacuum cycle will not exceed those at the end of the first vacuum cycle. The maximum temperatures at the end of the first vacuum (Table 4.4.3-5 for PWR and Table 4.4.3-8 for BWR) are conservatively presented as the maximum temperatures for the second vacuum condition. The maximum temperatures for the fuel cladding and the heat transfer disk are below the short-term allowable temperatures.

The in-pool cooling and the forced-air cooling operations (helium in canister) in LCO 3.1.4 are also evaluated for the PWR configuration for the 23 kW case and the BWR configuration for the 23 kW and 20 kW cases. The temperature profiles at the end of the helium condition, as shown in Table 4.4.3-5 for PWR and Table 4.4.3-8 for BWR, are used as the initial condition. The results for the BWR are shown in Tables 4.4.3-11 and 4.4.3-12 for the in-pool cooling and forced-air cooling, respectively. The results for the PWR are shown in Tables 4.4.3-13 and 4.4.3-14 for the in-pool cooling and forced-air cooling, respectively. Note that the time limit for the first helium backfill condition is used for the second helium backfill condition (after completion of the in-pool or forced-air cooling). Based on the heat-up rate of the first helium condition, the maximum component temperatures at the end of the second helium condition are well below the maximum temperatures at the end of the first helium condition. The maximum temperatures at the end of the first helium condition (Table 4.4.3-5 for PWR and Table 4.4.3-8 for BWR) are conservatively presented as the maximum temperatures for the second helium backfill condition, as shown in Tables 4.4.3-11 and 4.4.3-12 for the BWR configuration and Tables 4.4.3-13 and 4.4.3-14 for the PWR configuration.

Figure 4.4.3-1 Temperature Distribution (°F) for the Normal Storage Condition: PWR Fuel

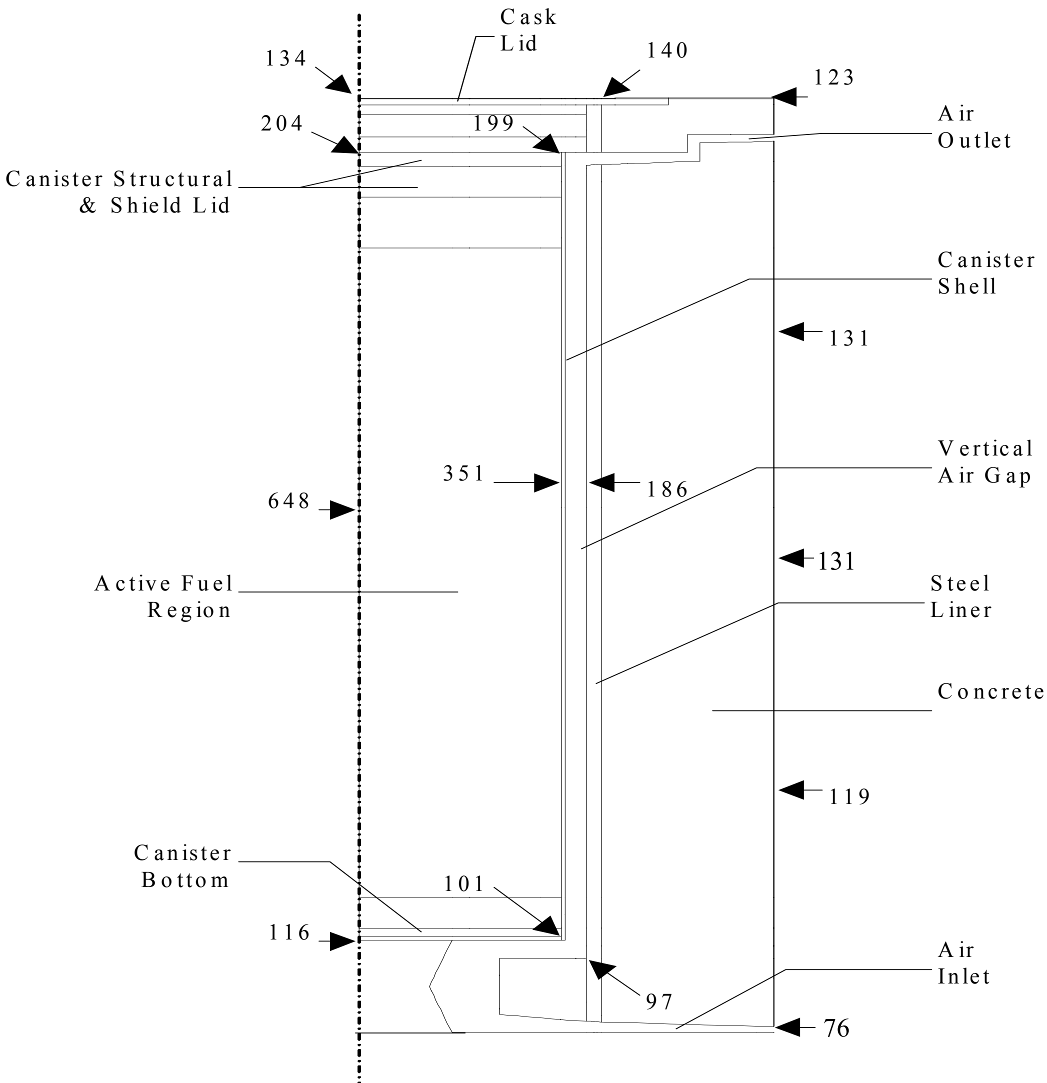


Figure 4.4.3-2 Air Flow Pattern in the Concrete Cask in the Normal Storage Condition: PWR Fuel

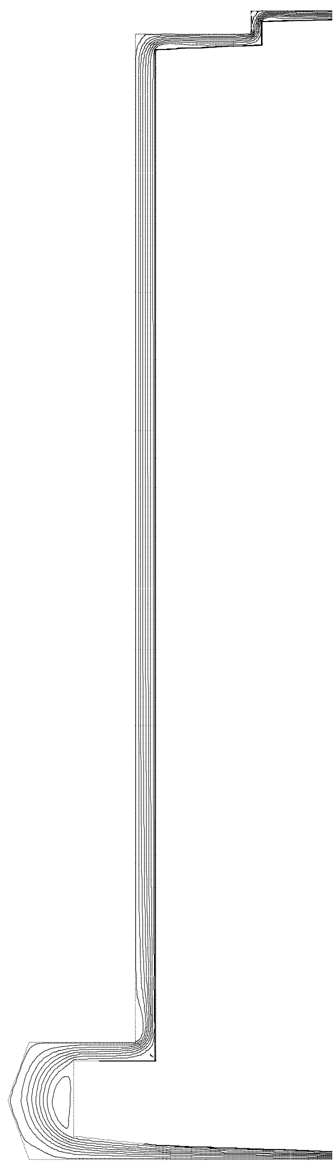


Figure 4.4.3-3 Air Temperature (°F) Distribution in the Concrete Cask During the Normal Storage Condition: PWR Fuel

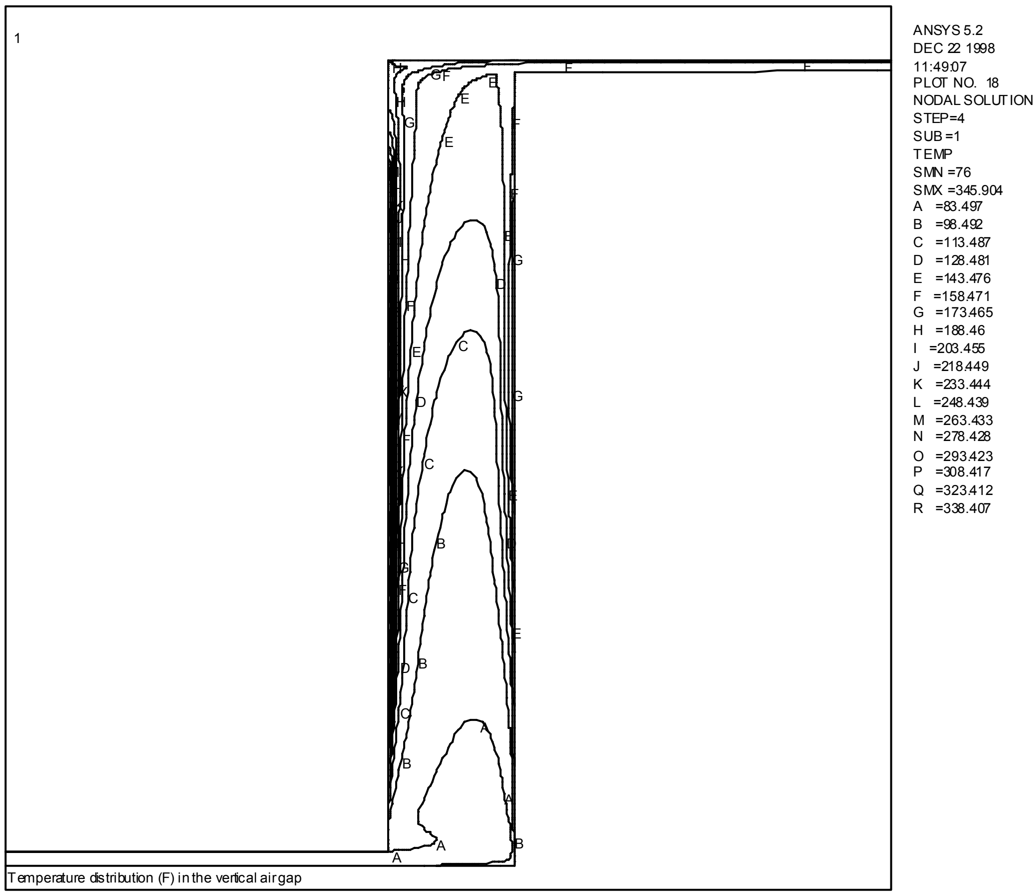


Figure 4.4.3-4 Concrete Temperature (°F) Distribution During the Normal Storage Condition: PWR Fuel

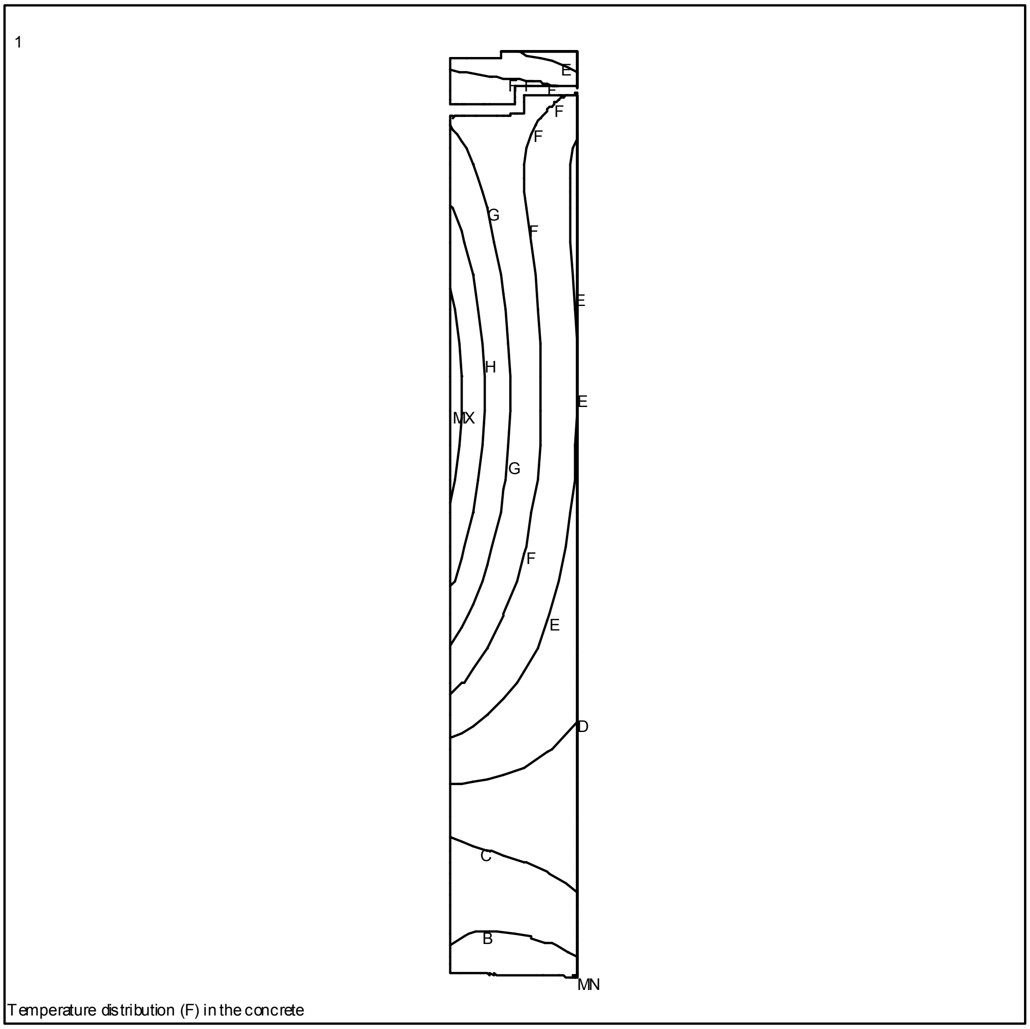
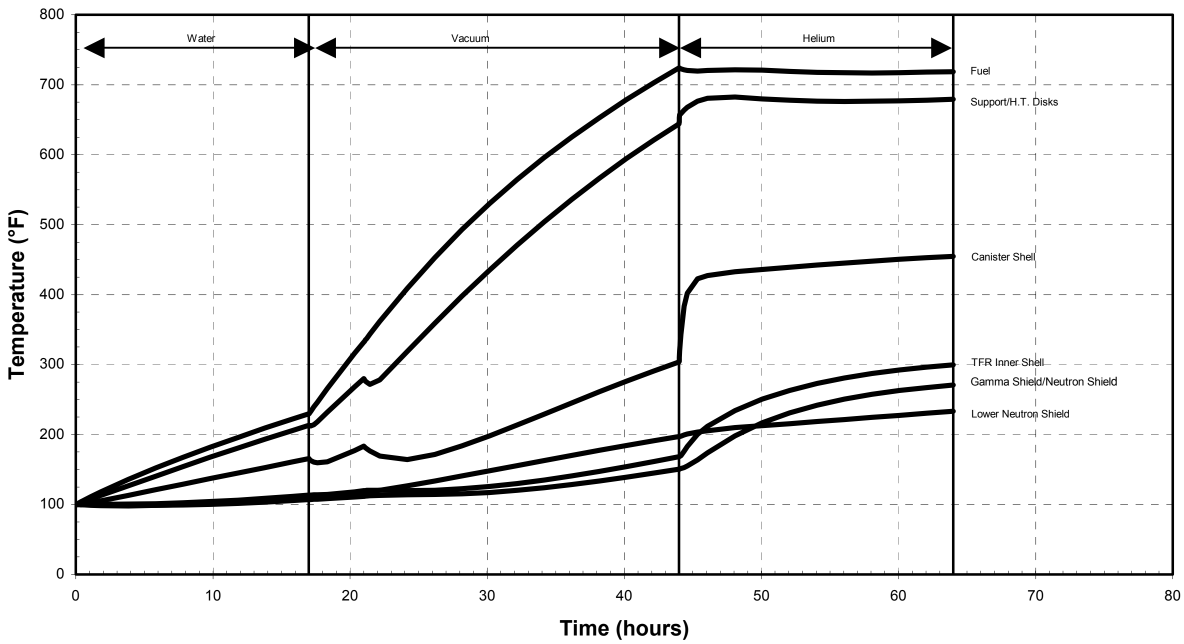


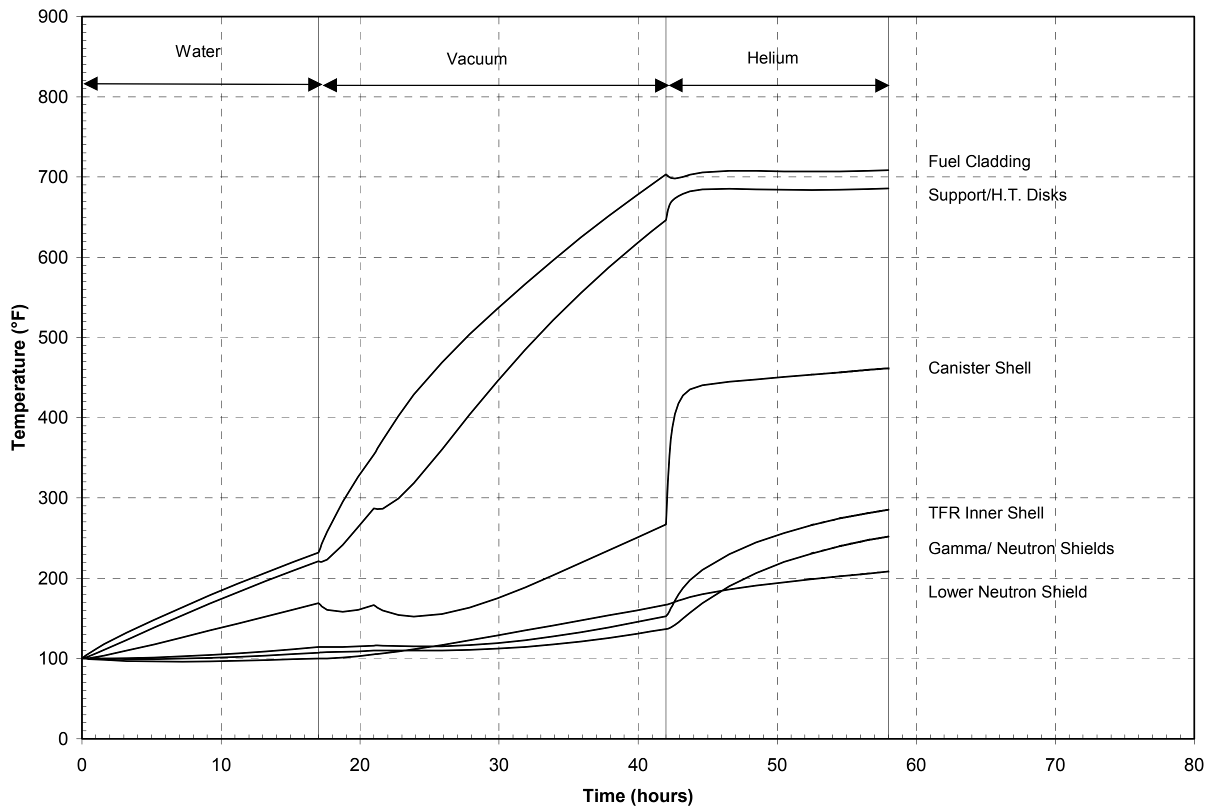
Figure 4.4.3-5 History of Maximum Component Temperature (°F) for Transfer Conditions for PWR Fuel with Design Basis 23 kW Uniformly Distributed Heat Load



Notes:

1. This graph corresponds to a canister containing water for 17 hours, vacuum for 27 hours and 20 hours in the helium condition. The results correspond to a uniformly distributed decay heat load of 23 kW.
2. "TFR" refers to the transfer cask.

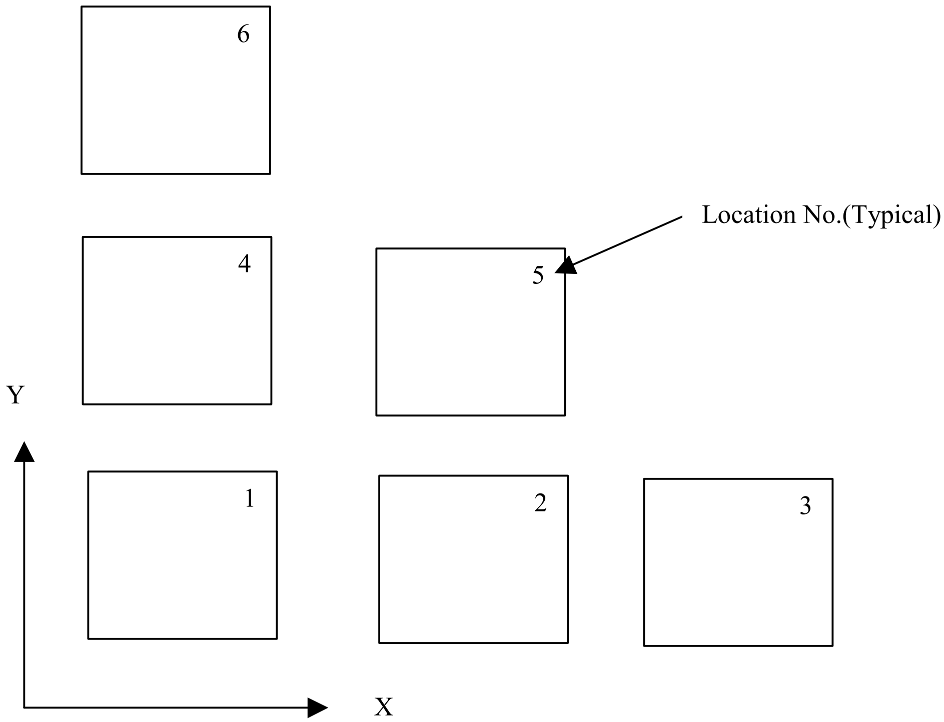
Figure 4.4.3-6 History of Maximum Component Temperature (°F) for Transfer Conditions for BWR Fuel with Design Basis 23 kW Uniformly Distributed Heat Load



Notes:

1. This graph corresponds to a canister containing water for 17 hours, vacuum for 25 hours and 16 hours in the helium condition. The results correspond to a uniformly distributed decay heat load of 23 kW.
2. "TFR" refers to the transfer cask.

Figure 4.4.3-7 Basket Location for the Thermal Analysis of PWR Reduced Heat Load Cases



A quarter symmetry configuration is considered. X and Y axes are at the centerlines of the basket.

Figure 4.4.3-8 BWR Fuel Basket Location Numbers

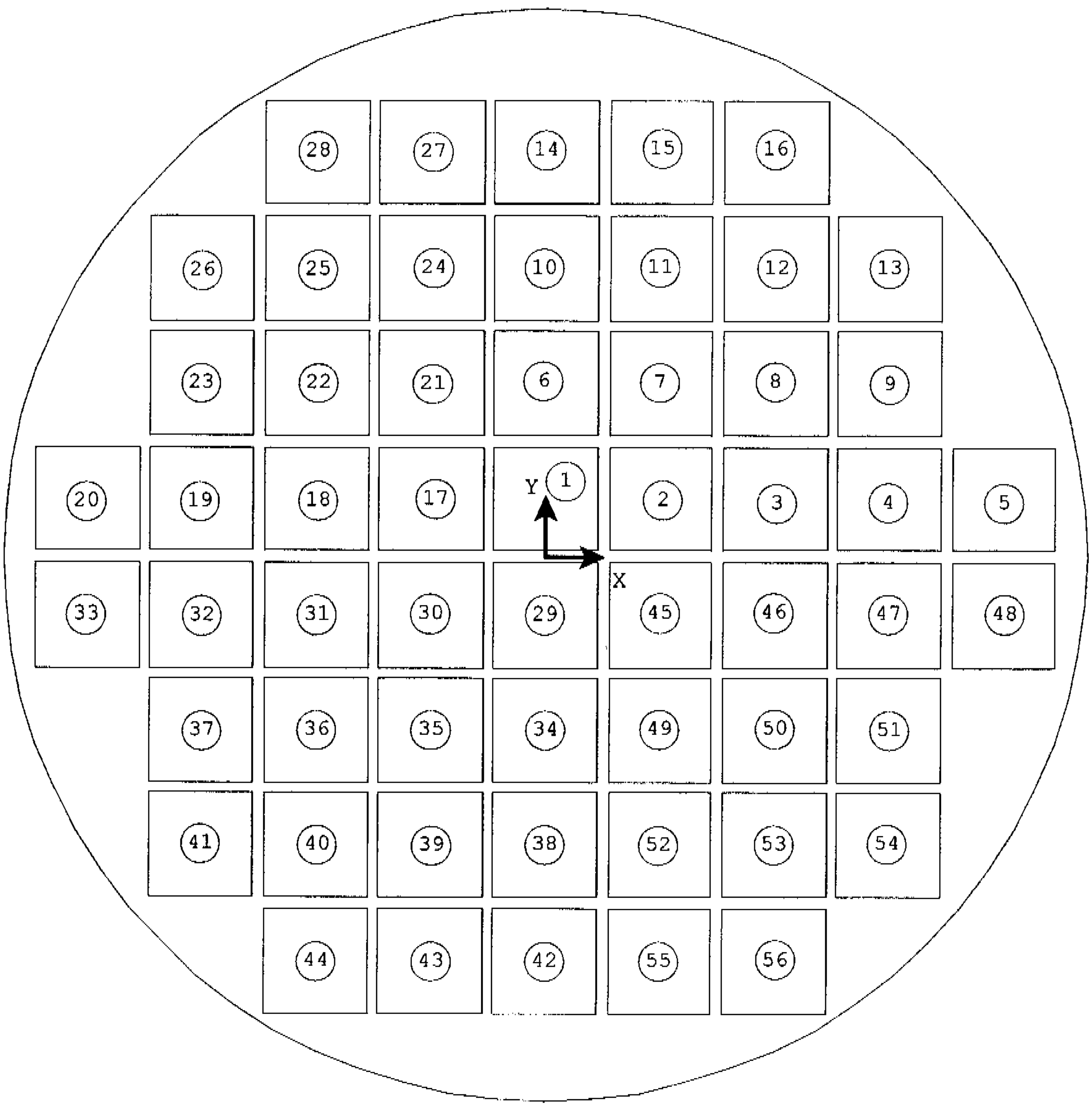


Table 4.4.3-1 Maximum Component Temperatures for the Normal Storage Condition - PWR

Component	Maximum Temperature (°F)	Allowable Temperatures (°F)
Fuel Cladding	648	752
Heat Transfer Disk	599	650
Support Disk	601	650
Top Weldment	399	800
Bottom Weldment	159	800
Canister Shell	351	800
Canister Structural Lid	204	800
Canister Shield Lid	212	800
Concrete	186 (local) 135 (bulk*)	200 (local) 150 (bulk)

* The volume average temperature of the concrete region is used as the bulk concrete temperature.

Table 4.4.3-2 Maximum Component Temperatures for the Normal Storage Condition - BWR

Component	Maximum Temperature (°F)	Allowable Temperatures (°F)
Fuel Cladding	642	752
Heat Transfer Disk	612	650
Support Disk	614	700
Top Weldment	361	800
Bottom Weldment	276	800
Canister Shell	376	800
Canister Structural Lid	180	800
Canister Shield Lid	185	800
Concrete	192 (local) 136 (bulk*)	200 (local) 150 (bulk)

*The volume average temperature of the concrete region is used as the bulk concrete temperature.

Table 4.4.3-3 Maximum Component Temperatures for the Transfer Condition – PWR Fuel with Design Basis 23 kW Uniformly Distributed Heat Load

Component	Maximum Temperature (°F)		Allowable Temperature (°F)
	Vacuum[1]	Helium[1]	
Fuel	724	724	752
Lead	151	271	600
Neutron Shield	149	267	300
Heat Transfer Disk	641	680	750
Support Disk	644	683	800
Canister	304	455	800
Transfer Cask Shells	168	300	700

1. See Figure 4.4.3-5 for history of maximum component temperatures.

Table 4.4.3-4 Maximum Component Temperatures for the Transfer Condition – BWR Fuel with Design Basis 23 kW Uniformly Distributed Heat Load

Component	Maximum Temperature (°F)		Allowable Temperature (°F)
	Vacuum[1]	Helium[1]	
Fuel	703	708	752
Lead	137	252	600
Neutron Shield	135	249	300
Heat Transfer Disk	645	683	750
Support Disk	646	686	700
Canister	267	462	800
Transfer Cask Shells	153	286	700

1. See Figure 4.4.3-6 for history of maximum component temperatures.

Table 4.4.3-5 Maximum Limiting Component Temperatures in Transient Operations for the Reduced Heat Load Cases for PWR Fuel

Heat Load (kW)	Water			Vacuum			Helium		
	Duration (hours)	Maximum Temperature (°F)		Duration (hours)	Maximum Temperature (°F)		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23.0	20	190	189	27	724	641	20	724[2]	680[2]
20.0	23	188	188	30	728	628	600[1]	728/708	664/664
17.6	27	188	187	33	731	617	600[1]	731/672	651/624
14.0	30	178	177	40	732	596	600[1]	732/613	630/559
11.0	35	169	168	52	730	575	600[1]	730/555	611/495
8.0	40	155	155	103	731	557	600[1]	731/483	595/412

1. Duration is defined based on a test time of 30 days for abnormal regimes, as described in PNL-4835 [34].
2. Since the time in helium is limited for the 23 kW configuration, only the maximum temperatures are listed.

Table 4.4.3-6 Maximum Limiting Component Temperatures in Transient Operations for the Reduced Heat Load Cases for PWR Fuel after In-Pool Cooling

Heat Load (kW)	In-Pool (helium)			Vacuum			Helium		
	Duration (hours)	End Temperature (°F)		Duration[1] (hours)	Maximum Temperature (°F)[2]		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23.0	24	491	415	14	724	641	20	724[4]	680[4]
20.0	24	477	397	17	728	628	600[3]	728/708	664/664
17.6	24	465	383	20	731	617	600[3]	731/672	651/624
14.0	24	445	360	26	732	596	600[3]	732/613	630/559
11	24	422	334	38	730	575	600[3]	730/555	611/495
8	24	390	293	89	731	557	600[3]	731/483	595/412

1. The maximum allowable time in the Technical Specification for this condition is equal to 2 hours less than the maximum allowable time shown in this table. This 2-hour reduction allows the handling time required to enter the next stage.
2. The maximum temperatures at the end of the first vacuum (Table 4.4.3-5) are conservatively presented.
3. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
4. Since the time in helium is limited for the 23 kW configuration, only the maximum temperatures are listed.

Table 4.4.3-7 Maximum Limiting Component Temperatures in Transient Operations for the Reduced Heat Load Cases for PWR Fuel after Forced-Air Cooling

Heat Load (kW)	Forced-Air (helium)			Vacuum			Helium		
	Duration (hours)	End Temperature (°F)		Duration[1] (hours)	Maximum Temperature (°F)[2]		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23.0	24	621	564	5	724	641	20	724[4]	680[4]
20.0	24	591	530	8	728	628	600[3]	728/708	664/664
17.6	24	567	502	11	731	617	600[3]	731/672	651/624
14.0	24	530	458	18	732	596	600[3]	732/613	630/559
11	24	493	415	29	730	575	600[3]	730/555	611/495
8	24	450	363	80	731	557	600[3]	731/483	595/412

1. The maximum allowable time in the Technical Specification for this condition is equal to 2 hours less than the maximum allowable time shown in this table. This 2-hour reduction allows the handling time required to enter the next stage.
2. The maximum temperatures at the end of the first vacuum (Table 4.4.3-5) are conservatively presented.
3. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
4. Since the time in helium is limited for the 23 kW configuration, only the maximum temperatures are listed.

Table 4.4.3-8 Maximum Limiting Component Temperatures in Transient Operations for BWR Fuel

Heat Load (kW)	Water			Vacuum			Helium		
	Duration (hours)	Maximum Temperature (°F)		Duration (hours)	Maximum Temperature (°F)		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	17	232	221	25	703	645	16	708[2]	683[2]
20	18	234	222	27	694	627	30	694[2]	661[2]
17	19	234	221	33	701	629	600[1]	701/660	659/631
14	20	232	219	45	719	643	600[1]	719/606	671/574
11	23	234	220	72	733	653	600[1]	733/543	679/508
8	31	236	220	600[1]	724	639	600[1]	724/467	639/427

1. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
2. Since the time in helium is limited for the 23 kW and 20 kW cases, only the maximum temperatures are listed.

Table 4.4.3-9 Maximum Limiting Component Temperatures in Transient Operations after Vacuum for BWR Fuel after In-Pool Cooling

Heat Load (kW)	In-Pool (helium)			Vacuum			Helium		
	Duration (hours)	End Temperature (°F)		Duration[1] (hours)	Maximum Temperature (°F)[2]		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	488	444	12	703	645	16	708[4]	683[4]
20	24	476	431	13	694	627	30	694[4]	661[4]
17	24	467	419	19	701	629	600[3]	701/660	659/631
14	24	455	404	28	719	643	600[3]	719/606	671/574
11	24	439	383	54	733	653	600[3]	733/543	679/508

1. The maximum allowable time in the Technical Specification for this condition is equal to 2 hours less than the maximum allowable time shown in this table. This 2-hour reduction allows the handling time required to enter the next stage.
2. The maximum temperatures at the end of the first vacuum (Table 4.4.3-8) are conservatively presented.
3. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
4. Since the time in helium is limited for the 23 kW and 20 kW cases, only the maximum temperatures are listed.

Table 4.4.3-10 Maximum Limiting Component Temperatures in Transient Operations after Vacuum for BWR Fuel after Forced-Air Cooling

Heat Load (kW)	Forced-Air (helium)			Vacuum			Helium		
	Duration (hours)	End Temperature (°F)		Duration[1] (hours)	Maximum Temperature (°F)[2]		Duration (hours)	Max. Temp. / Temp. at Steady-state (°F)	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	623	591	4	703	645	16	708[4]	683[4]
20	24	592	558	5	694	627	30	694[4]	661[4]
17	24	565	528	10	701	629	600[3]	701/660	659/631
14	24	541	503	20	719	643	600[3]	719/606	671/574
11	24	519	477	43	733	653	600[3]	733/543	679/508

1. The maximum allowable time in the Technical Specification for this condition is equal to 2 hours less than the maximum allowable time shown in this table. This 2-hour reduction allows the handling time required to enter the next stage.
2. The maximum temperatures at the end of the first vacuum (Table 4.4.3-8) are conservatively presented.
3. Duration is defined based on a test time of 30 days for abnormal regimes as described in PNL-4835.
4. Since the time in helium is limited for the 23 kW and 20 kW cases, only the maximum temperatures are listed.

Table 4.4.3-11 Maximum Limiting Component Temperatures in Transient Operations after Helium for BWR Fuel after In-Pool Cooling

Heat Load (kW)	In-Pool (helium)			Helium		
	Duration (hours)	End Temperature (°F)		Duration (hours)	Max. Temp. (°F)[1]	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	489	444	16	708	683
20	24	477	431	30	694	661

1. The maximum temperatures at the end of helium in Table 4.4.3-8 are conservatively used.

Table 4.4.3-12 Maximum Limiting Component Temperatures in Transient Operations after Helium for BWR Fuel after Forced-Air Cooling

Heat Load (kW)	Forced-Air (helium)			Helium		
	Duration (hours)	End Temperature (°F)		Duration (hours)	Max. Temp. (°F)[1]	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	630	598	16	708	683
20	24	601	566	30	694	661

1. The maximum temperatures at the end of helium in Table 4.4.3-8 are conservatively used.

Table 4.4.3-13 Maximum Limiting Component Temperatures in Transient Operations after Helium for PWR Fuel after In-Pool Cooling

Heat Load (kW)	In-Pool (helium)			Helium		
	Duration (hours)	End Temperature (°F)		Duration (hours)	Max. Temp. (°F)[1]	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	489	413	20	724	680

1. The maximum temperatures at the end of helium in Table 4.4.3-5 are conservatively used.

Table 4.4.3-14 Maximum Limiting Component Temperatures in Transient Operations after Helium for PWR Fuel after Forced-Air Cooling

Heat Load (kW)	Forced-Air (helium)			Helium		
	Duration (hours)	End Temperature (°F)		Duration (hours)	Max. Temp. (°F)[1]	
		Fuel	Heat Transfer Disk		Fuel	Heat Transfer Disk
23	24	626	569	20	724	680

1. The maximum temperatures at the end of helium in Table 4.4.3-5 are conservatively used.

4.4.4 Minimum Temperatures

The minimum temperatures of the Vertical Concrete Cask and components occur at -40°F with no heat load. The temperature distribution for this off-normal environmental condition is provided in Section 11.1. At this extreme condition, the component temperatures are above their minimum material limits.

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4.4.5 Maximum Internal Pressures

The maximum internal operating pressures for normal conditions of storage are calculated in the following sections for the PWR and BWR Transportable Storage Canisters.

4.4.5.1 Maximum Internal Pressure for PWR Fuel Canister

The internal pressures within the PWR fuel canister are a function of fuel type, fuel condition (failure fraction), burnup, UMS® canister type, and the backfill gases in the canister cavity. Gases included in the canister pressure evaluation include rod-fill, rod fission and rod backfill gases, canister backfill gases and burnable poison generated gases. Each of the fuel types expected to be loaded into the UMS® canister system is separately evaluated to arrive at a bounding canister pressure.

Fission gases include all fuel material generated gases including long-term actinide decay generated helium. Based on detailed SAS2H calculations of the maximum fissile material mass assemblies in each canister class, the quantity of gas generated by the fuel rods rises as burnup and cool time is increased and enrichment is decreased. To assure the maximum gas is available for release, the PWR inventories are extracted from 60,000 MWd/MTU burnup cases at an enrichment of 1.9 wt. % ^{235}U and a cool time of 40 years. Gases included are all krypton, iodine, and xenon isotopes in addition to helium and tritium (^{3}H). Molar quantities for each of the maximum fissile mass assemblies are summarized in Table 4.4.5-1. Fuel generated gases are scaled by fissile mass to arrive at molar contents of other UMS® fuel types.

Fuel rod backfill pressure varies significantly between the PWR fuel types. The maximum reported backfill pressure is listed for the Westinghouse 17×17 fuel assembly at 500 psig. With the exception of the B&W fuel assemblies, which are limited to 435 psig, all fuel assemblies evaluated are set to the maximum 500 psig backfill reported for the Westinghouse assembly. Backfill quantities are based on the free volume between the pellet and the clad and the plenum volume. The fuel rod backfill gas temperature is conservatively assumed to have an initial temperature of 68°F.

Burnable poison rod assemblies (BPRAs) placed within the UMS® storage canister may contribute additional molar gas quantities due to (n,alpha) reactions of fission generated neutrons with ^{10}B during in-core operation. ^{10}B forms the basis of a portion of the neutron poison population. Other neutron poisons, such as gadolinium and erbium, do not produce a significant amount of helium nuclides (alpha particles) as part of their activation chain. Primary BPRAs in existence include Westinghouse Pyrex (borosilicate glass) and WABA (wet annular burnable absorber) configurations, as well as B&W BPRAs and shim rods employed in CE cores. The CE shim rods replace standard fuel rods to form a complete assembly array. The quantity of helium available for release from the BPRAs is directly related to the initial boron content of the rods and the release fraction of gas from the matrix material in question. Release from either of the low temperature, solid matrix materials is likely to be limited, but no release fractions were available in open literature. As such, a 100% release fraction is assumed based on a boron content of 0.0063 g/cm ^{10}B per rod, with the maximum number of rods per assembly. The maximum number of rods is 16 for Westinghouse core 14×14 assemblies, 20 rods for Westinghouse and B&W 15×15 assemblies, and 24 rods for Westinghouse and B&W 17×17 assemblies. The length of the absorber is conservatively taken as the active fuel length. CE core shim rods are modeled at 0.0126 g/cm ^{10}B for 16, 12, and 12 rods applied to CE manufactured 14×14, 15×15 and 16×16 cores, respectively.

The canister backfill gases are conservatively assumed to be at 250°F, which is significantly below the canister shell maximum initial temperature of 304°F at the end of vacuum drying. The initial pressure of the canister backfill gas is 1 atm (0.0 psig). Free volume inside each PWR canister class is listed in Table 4.4.5-2. The listed free volumes do not include fuel assembly components since these components vary for each assembly type and fuel insert. Subtracting out the rod and guide tube volumes and all hardware components arrives at free volume of the canisters including fuel assemblies and a load of 24 BPRAs. For the Westinghouse BPRAs, the Pyrex volume is employed since it displaces more volume than the WABA rods.

The total pressure for each of the UMS® payloads is found by calculating the releasable molar quantity of each gas (30% of the fission gas and 100% of the rod backfill adjusted for the 1% fuel failure fraction), and summing the quantities directly. The quantity of gas is then employed in the ideal gas equation in conjunction with the average gas temperature at normal operating conditions to arrive at system pressures. The normal system pressure calculation for maximum system pressure limits assumes the average PWR gas temperature to be 420°F. The actual calculated gas temperature determined by the three-dimensional canister model is 421°F for the normal storage condition. The 1°F temperature difference has an insignificant effect on the system pressure calculation. Each of the UMS® PWR fuel types is individually evaluated for normal condition pressure, and sets the maximum normal condition pressure at 4.21 psig. A summary of the maximum pressure in each PWR canister class is shown in Table 4.4.5-3. The table also includes the fuel type producing the listed maximum pressures.

4.4.5.2 Maximum Internal Pressure for BWR Fuel Canister

BWR canister maximum pressures are determined in the same manner as those documented for the PWR canister cases. Primary differences between PWR and BWR analysis include a maximum normal condition average gas temperature of 410°F, rod backfill gas pressures of 132 psig, and limits pressurizing gases to fission gases (including helium actinide decay gas), rod backfill gases, and canister backfill gas. The 132 psig employed in this analysis is significantly higher than the 6 atmosphere maximum pressure reported in open literature. BWR assemblies do not contain an equivalent to the PWR BPRAs and, therefore, do not require ^{10}B helium generated gases to be added. Fissile gas inventories for the maximum fissile material assemblies in each of the three BWR lattices configurations (7×7, 8×8, and 9×9) are shown in Table 4.4.5-4. Free volumes, without fuel components, in UMS® canister classes 4 and 5 are shown in Table 4.4.5-5. Maximum pressures for each canister class are listed in Table 4.4.5-6. The maximum normal condition pressure of 3.97 psig is based on a GE 7×7 assembly, designed for a BWR/2-3 reactor, with gas inventories conservatively taken from a 60,000 MWD/MTU source term. The normal condition pressure for a UMS® storage canister containing the GE 9×9 fuel assembly with 79 fuel rods is 3.96 psig. Similar fuel masses and displaced volume account for similar canister pressures.

Table 4.4.5-1 PWR Per Assembly Fuel Generated Gas Inventory (Fission Gas Basis – 60 GWd/MTU, 1.9 wt % ^{235}U)

Array	Assy Type	MTU	Moles
14×14	WE Standard	0.4144	35.52
15×15	B&W	0.4807	41.32
16×16	CE (System 80)	0.4417	38.10
17×17	WE Standard	0.4671	40.18

Table 4.4.5-2 PWR Canister Free Volume (No Fuel or Inserts)

Canister Class	1	2	3
Basket Volume (in^3)	69800	74490	77460
Canister Height (inch)	175.05	184.15	191.75
Canister Free Volume w/o Fuel (liter)	7970	8400	8770

Table 4.4.5-3 PWR Maximum Normal Condition Pressure Summary

Canister Class	Fuel Type	Pressure (psig)
Class 1	WE 17×17 Standard	4.20
Class 2	B&W 17×17 Mark C	4.21
Class 3	CE 16×16 System 80	4.11

Table 4.4.5-4 BWR Per Assembly Fuel Generated Gas Inventory

Array	Assy Type	MTU	Moles
7×7	GE 7×7 (49 Rods)	0.1985	16.78
8×8	GE 8×8 (63 Rods)	0.1880	16.07
9×9	GE 9×9 (79 Rods)	0.1979	16.86

Table 4.4.5-5 BWR Canister Free Volume (No Fuel or Inserts)

Canister Class	4	5
Basket Volume (in^3)	73110	74680
Canister Height (inch)	185.55	190.35
Canister Free Volume w/o Fuel (liter)	8500	8740

Table 4.4.5-6 BWR Maximum Normal Condition Pressure Summary

Canister Class	Fuel Type	Pressure (psig)
Class 4	GE 7×7	3.97
Class 5	GE 9×9	3.96

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4.4.6 Maximum Thermal Stresses

The results of thermal stress calculations for normal conditions of storage are reported in Section 3.4.4.

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4.4.7 Evaluation of System Performance for Normal Conditions of Storage

Results of thermal analysis of the Universal Storage System containing PWR or BWR fuel under normal conditions of storage are summarized in Tables 4.4.3-1 through 4.4.3-4. The maximum PWR and BWR fuel rod cladding temperatures are below the allowable temperatures; temperatures of safety-related components during storage and transfer operations under normal conditions are maintained within their safe operating ranges; and thermally induced stresses in combination with pressure and mechanical load stresses are shown in the structural analysis of Chapter 3.0 to be less than the allowable stresses. Therefore, the Universal Storage System performance meets the requirements for the safe storage of design basis fuel under the normal operating conditions specified in 10 CFR 72.

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4.5 Thermal Evaluation for Site Specific Spent Fuel

This section presents the thermal evaluation of fuel assemblies or configurations, which are unique to specific reactor sites or which differ from the UMS® Storage System design basis fuel. These site specific configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel that is classified as damaged. Damaged fuel includes fuel rods with cladding that exhibit defects greater than pinhole leaks or hairline cracks.

Site-specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation.

4.5.1 Maine Yankee Site Specific Spent Fuel

The standard spent fuel assembly for the Maine Yankee site is the Combustion Engineering (CE) 14×14 fuel assembly. Fuel of the same design has also been supplied by Westinghouse and by Exxon. The standard 14×14 fuel assembly is included in the population of the design basis PWR fuel assemblies for the UMS® Storage System (See Table 2.1.1-1). The maximum decay heat for the standard Maine Yankee fuel is the design basis heat load for the PWR fuels (23 kW total, or 0.958 kW per assembly). This heat load is bounded by the thermal evaluations in Section 4.4 for the normal conditions of storage, Section 4.4.3.1 for less than design basis heat loads and Chapter 11 for off-normal and accident conditions.

Some Maine Yankee site specific fuel has a burnup greater than 45,000 MWD/MTU, but less than 50,000 MWD/MTU. As shown in Table B2-6 in Appendix B of the CoC Number 1015 Technical Specifications, loading of fuel assemblies in this burnup range is subject to preferential loading in designated basket positions in the Transportable Storage Canister. Certain fuel assemblies in this burnup range must be loaded in one of the two configurations of the Maine Yankee Fuel Can.

The site specific fuels included in this evaluation are:

1. Consolidated fuel rod lattices consisting of a 17 × 17 lattice fabricated with 17 × 17 grids, 4 stainless steel support rods and stainless steel end fittings. One of these

lattices contains 283 fuel rods and 2 rod position vacancies. The other contains 172 fuel rods, with the remaining rod position locations either empty or containing stainless steel dummy rods.

2. Standard fuel assemblies with a Control Element Assembly (CEA) inserted in each one.
3. Standard fuel assemblies that have been modified by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods, or 1.95 wt % enriched fuel rods.
4. Standard fuel assemblies that have had the burnable poison rods removed and replaced with hollow zirconium alloy tubes.
5. Standard fuel assemblies with in-core instrument thimbles stored in the center guide tube.
6. Standard fuel assemblies that are designed with variable enrichment (radial) and axial blankets.
7. Standard fuel assemblies that have some fuel rods removed.
8. Standard fuel assemblies that have damaged fuel rods.
9. Standard fuel assemblies that have some type of damage or physical alteration to the cage (fuel rods are not damaged).
10. Two (2) rod holders, designated CF1 and CA3. CF1 is a lattice having approximately the same dimensions as a standard fuel assembly. It is a 9×9 array of tubes, some of which contain damaged fuel rods. CA3 is a previously used fuel assembly lattice that has had all of the rods removed, and in which damaged fuel rods have been inserted.
11. Standard fuel assemblies that have damaged fuel rods stored in their guide tubes.
12. Standard fuel assemblies with inserted startup sources and other non-fuel items.

The Maine Yankee site specific fuels are also described in Section 1.3.2.1.

The thermal evaluations of these site specific fuels are provided in Section 4.5.1.1. Section 4.5.1.2 presents the evaluation of the Maine Yankee preferential loading of fuel exceeding the design basis heat load (0.958 kW) per assembly on the basket periphery.

4.5.1.1 Thermal Evaluation for Maine Yankee Site Specific Spent Fuel

The maximum heat load per assembly for site specific fuel considered in this section is limited to the design basis heat load (0.958 kW). The evaluation of fuel configurations having a greater heat load is presented in Section 4.5.1.2.

4.5.1.1.1 Consolidated Fuel

There are two (2) consolidated fuel lattices. One lattice contains 283 fuel rods and the other contains 172 fuel rods. Conservatively, only one consolidated fuel lattice is loaded in any Transportable Storage Canister.

The maximum decay heat of the consolidated fuel lattice having 283 fuel rods is 0.279 kW. This heat load is bounded by the design basis PWR fuel assembly, since it is less than one-third of the design basis heat load.

The second consolidated fuel lattice has 172 fuel rods with 76 stainless steel dummy rods at the outer periphery of the lattice. Due to the existence of the stainless steel rods, the effective thermal conductivities of this assembly may be slightly lower than those of the standard CE 14×14 fuel assembly. While the stainless steel rods provide better conductance in the axial direction, the radiation heat transfer is less effective at the surface of stainless steel rods, as compared to the standard fuel rods. The radiation is a function of surface emissivity and the emissivity for stainless steel (0.36) is less than one-half of that for zirconium alloy (0.75). A parametric study is performed to demonstrate that the thermal performance of the UMS PWR basket loading configuration consisting of 23 standard CE 14×14 fuel assemblies and the consolidated fuel lattice with stainless rods is bounded by that of the configuration consisting of 24 standard CE 14×14 fuel assemblies. Two finite element models are used in the study: a two-dimensional fuel assembly model and a three-dimensional periodic canister internal model.

The two-dimensional model is used to determine the effective thermal conductivities of the consolidated fuel lattice with stainless steel rods. Considering the symmetry of the consolidated fuel, the finite element model represents a one-quarter section as shown in Figure 4.5.1.1-1. The methodology used in Section 4.4.1.5 for the two-dimensional fuel model for PWR fuel is employed in this model. The model includes the fuel pellets, cladding, helium between the fuel rods, and helium occupying the gap between the fuel pellets and cladding. In addition, the rods at the two outer layers are modeled as solid stainless steel rods to represent the configuration of this consolidated fuel lattice. Modes of heat transfer modeled include conduction and radiation between individual rods for steady-state condition. ANSYS PLANE55 conduction elements and LINK31 radiation elements are used in the model. Radiation elements are defined between rods and from rods to the boundary of the model. The effective conductivity for the fuel is determined using the procedure described in Section 4.4.1.5.

The three-dimensional periodic canister internal model consists of a periodic section of the canister internals. The model contains one support disk with two heat transfer disks (half thickness) on its top and bottom, the fuel assemblies, the fuel tubes and the helium in the canister, as shown in Figure 4.5.1.1-2. The purpose of this model is to compare the maximum fuel cladding temperatures of the following cases:

1) Base Case: All 24 positions loaded with standard CE 14×14 fuel assemblies.
2) Case 2: 23 positions with standard fuel, with one consolidated fuel lattice in position 2.
3) Case 3: 23 positions with standard fuel, with one consolidated fuel lattice in position 3.
4) Case 4: 23 positions with standard fuel, with one consolidated fuel lattice in position 4.
5) Case 5: 23 positions with standard fuel, with one consolidated fuel lattice in position 5.

Positions 2, 3, 4, and 5 are shown in Figure 4.5.1.1-3. Based on symmetry, these locations represent all of the possible locations for consolidated fuel in the basket.

The fuel assemblies and fuel tubes are represented by homogeneous regions with effective thermal conductivities. The effective conductivities for the consolidated fuel are determined by the two-dimensional fuel assembly model discussed above. The effective conductivities for the CE 14×14 fuel assemblies are established based on the model described in Section 4.4.1.5. Effective properties for the fuel tubes are determined by the two-dimensional fuel tube model in Section 4.4.1.6. Volumetric heat generation corresponding to the design basis heat load of 0.958 kW per assembly is applied to the CE 14×14 fuel regions in the model. Similarly, a heat generation rate corresponding to 0.279 kW is applied to the consolidated fuel assembly region. The heat conduction in the axial direction is conservatively ignored by assuming that the top and bottom surfaces of the model are adiabatic. A constant temperature of 400°F is applied to the outer surface of the model as boundary conditions. Note that the maximum canister temperature is 351°F for PWR configurations for the normal condition of storage (Table 4.1-4). Steady state thermal analysis is performed for all five cases and the calculated maximum fuel cladding temperatures in the model are:

	Base Case	Case 2	Case 3	Case 4	Case 5
Maximum Fuel Cladding Temperature (°F)	755	733	738	740	740

As shown, the maximum temperatures for Cases 2 through 5 are less than those of the Base Case. It is concluded that the thermal performance of the configuration consisting of 23 standard CE 14×14 fuel assemblies and one consolidated fuel lattice is bounded by that of the configuration consisting of 24 standard CE 14×14 fuel assemblies. This study shows that a consolidated fuel lattice can be located in any basket position. However, as shown in Table B2-6 of Appendix B, the consolidated fuel assembly must be loaded in a corner position of the fuel basket (e.g., Position 5 shown in Figure 4.5.1.1-3).

4.5.1.1.2 Standard CE 14 × 14 Fuel Assemblies with Control Element Assemblies

A Control Element Assembly (CEA) consists of five solid B_4C rods encapsulated in stainless steel tubes. The B_4C material has a conductivity of 1.375 BTU/hr-in-°F. With the CEA inserted into the guide tubes of the CE 14×14 fuel assembly, the effective conductivity in the axial direction of the fuel assembly is increased because solid material replaces helium in the guide tubes. The change in the effective conductivity in the transverse direction of the fuel assembly is negligible since the CEA is inside of the guide tubes. Note that the total heat load, including the small amount of extra heat generated by the CEA, remains below the design basis heat load. Therefore, the thermal performance of the fuel assemblies with CEAs inserted is bounded by that of the standard fuel assemblies.

4.5.1.1.3 Modified Standard Fuel Assemblies

These assemblies include those standard fuel assemblies that have been modified by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods or 1.95 wt % enriched fuel rods.

The maximum number of fuel rods replaced by stainless steel rods is six (6) per assembly, which is about 3% of the total number of fuel rods in each assembly (176). The conductivity of the stainless steel is similar to that of zirconium alloy and better than that of the UO_2. The resultant increase in effective conductivity of the modified fuel assembly in the axial direction offsets the decrease in the effective conductivity in the transverse direction (due to slight reduction of radiation heat transfer at the surface of the stainless steel rods). The maximum number of fuel rods replaced by solid zirconium alloy rods is five (5) per assembly. Since the solid zirconium alloy rod has a higher conductivity than the fuel rod (UO_2 with zirconium alloy clad), the effective conductivity of the repaired fuel assembly is increased. The thermal properties for the enriched fuel rod remain the same as for standard fuel rods, so there is no change in effective conductivity of the fuel assembly results from the use of fuel rods enriched to 1.95 wt % ^{235}U. These rods replace other fuel rods in the assembly after the first or second burnup cycles were completed. Therefore, these replacement fuel rods have been burned a minimum of one cycle less than the remainder of the assembly, producing a proportionally lower per rod heat load. The heat load (on a per rod basis) of the fuel rods in a standard assembly, bounds the heat load of the 1.95 wt % ^{235}U enriched fuel rods. Consequently, the loading of modified fuel assemblies is bounded by the thermal evaluation of the standard fuel assembly.

4.5.1.1.4 Use of Hollow Zirconium Alloy Tubes

Certain standard fuel assemblies have had the burnable poison rods removed. These rods were replaced with hollow zirconium alloy tubes.

There are 16 locations where burnable poison rods were removed and hollow zirconium alloy tubes were installed in their place. Since the maximum heat load for these assemblies is 0.552 kW per assembly (less than two-thirds of the design basis heat load) and the number of hollow zirconium alloy tubes is only about one-tenth (16/176) of the total number of the fuel rods, the thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

4.5.1.1.5 Standard Fuel with In-core Instrument Thimbles

Certain fuel assemblies have in-core instrument thimbles stored within the center guide tube of each fuel assembly. Storing an in-core instrument thimble assembly in the center guide tube of a fuel assembly will slightly increase the axial conductance of the fuel assembly (helium replaced by solid material). Therefore, there is no negative impact on the thermal performance of the fuel assembly with this configuration. The thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

4.5.1.1.6 Standard Fuel Assemblies with Variable Enrichment and Axial Blankets

The Maine Yankee variably enriched fuel assemblies are limited to two batches of fuel, which have a maximum burnup less than 30,000 MWD/MTU. The variably enriched rods in the fuel assemblies have enrichments greater than 3.4 wt % ^{235}U, except that the axial blankets on one batch are enriched to 2.6 wt % ^{235}U. As shown in Table B2-8 of Appendix B, fuel at burnups less than or equal to 30,000 MWD/MTU with any enrichment greater than, or equal to, 1.9 wt % ^{235}U may be loaded with 5 years cool time.

The thermal conductivities of the fuel assemblies with variable enrichment (radial) and axial blankets are considered to be essentially the same as those of the standard fuel assemblies. Since the heat load per assembly is limited to the design basis heat load, there is no effect on the thermal performance of the system due to this loading configuration.

4.5.1.1.7 Standard Fuel Assemblies with Removed Fuel Rods

Except for assembly number EF0046, the maximum number of missing fuel rods from a standard fuel assembly is 14, or 8% (14/176) of the total number of rods in one fuel assembly. The maximum heat load for any one of these fuel assemblies is conservatively determined to be 0.63 kW. This heat load is 34% less than the design basis heat load of 0.958 kW. Fuel assembly EF0046 was used in the consolidated fuel demonstration program and has only 69 rods remaining in its lattice. This fuel assembly has a heat load of 70 watts, or 7% of the design basis heat load of 0.958 kW. Therefore, the thermal performance of fuel assemblies with removed fuel rods is bounded by that of the standard fuel assemblies.

4.5.1.1.8 Fuel Assemblies with Damaged Fuel Rods

Damaged fuel assemblies are standard fuel assemblies with fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. Fuel, classified as damaged, will be placed in one of the two configurations of the Maine Yankee Fuel Can. The primary function of the fuel can is to confine fuel material within the can and to facilitate handling and retrievability. The Maine Yankee fuel can is shown in Drawings 412-501 and 412-502. The placement of the loaded fuel cans is restricted by the operating procedures and/or Technical

Specifications to loading into the four corner positions at the periphery of the fuel basket as shown in Figure B2-1. The heat load for each damaged fuel assembly is considered to be the design basis heat load of 0.958 kW (23 kW/24).

A steady-state thermal analysis is performed using the three-dimensional canister model described in Section 4.4.1.2 simulating 100% failure of the fuel rods, fuel cladding, and guide tubes of the damaged fuel held in the Maine Yankee fuel can. The canister is assumed to contain twenty (20) design basis PWR fuel assemblies and damaged fuel assemblies in fuel cans in each of the four corner positions.

Two debris compaction levels are considered for the 100% failure condition: (Case 1) 100% compaction of the fuel rod, fuel cladding, and guide tube debris resulting in a 52-inch debris level in the bottom of each fuel can, and (Case 2) 50% compaction of the fuel rod, fuel cladding, and guide tube debris resulting in a 104-inch debris level in the bottom of each fuel can. The entire heat generation rate for a single fuel assembly (i.e., 0.958 kW) is concentrated in the debris region with the remainder of the active fuel region having no heat generation rate applied. To ensure the analysis is bounding, the debris region is located at the lower part of the active fuel region in lieu of the bottom of the fuel can. This location is closer to the center of the basket where the maximum fuel cladding temperature occurs. The effective thermal conductivities for the design basis PWR fuel assembly (Section 4.4.1.5) are used for the debris region. This is conservative since the debris (100% failed rods) is expected to have higher density (better conduction) and more surface area (better radiation) than an undamaged fuel assembly. In addition, the thermal conductivity of helium is used for the remainder of the active fuel length. Boundary conditions corresponding to the normal condition of storage are used at the outer surface of the canister model (see Section 4.4.1.2). A steady-state thermal analysis is performed. The results of the thermal analyses performed for 100% fuel rod, fuel cladding, and guide tube failure are:

Description	Maximum Temperature (°F)			
	Fuel Cladding	**Damaged Fuel**	**Support Disk**	**Heat Transfer Disk**
Case 1 (100% Compaction)	654	672	598	594
Case 2 (50% Compaction)	674	594	620	616
Design Basis PWR Fuel	670	N/A	615	612
Allowable	752	N/A	650	650

As demonstrated, the extreme case of 100% fuel rod, fuel cladding, and guide tube failure with 50% compaction of the debris results in temperatures that are less than 1% higher than those calculated for the design basis PWR fuel. The maximum temperatures for the fuel cladding, damaged fuel assembly, support disks, and heat transfer disks remain within the allowable temperature range for both 100% failure cases. Additionally, the temperatures used in the structural analyses of the fuel basket envelop those calculated for both 100% failure cases.

Additionally, the above analysis has been repeated to consider a maximum heat load of 1.05 kW/assembly (see Section 4.5.1.2) in the Maine Yankee fuel cans. To maintain the 23 kW total heat load per canister, the model considers a heat load of 1.05 kW/assembly in the four (4) Maine Yankee fuel cans and 0.94 kW/assembly in the rest of the twenty (20) basket locations. The analysis results indicate that the maximum temperatures for the fuel cladding and basket components are slightly lower than those for the case with a heat load of 0.958 kW in the damaged fuel can, as presented above. The maximum fuel cladding temperature is 650°F (< 654°F) and 672°F (< 674°F) for 100% and 50% compaction ratio cases, respectively. Therefore, the case with 1.05 kW/assembly in the Maine Yankee fuel can is bounded by the case with 0.958 kW/assembly in the fuel cans.

4.5.1.1.9 Standard Fuel Assemblies with Damaged Lattice

Certain standard fuel assemblies may have damage or physical alteration to the lattice or cage that holds the fuel rods, but not exhibit damage to the fuel rods. Fuel assemblies with lattice damage are evaluated in Section 11.2.15. The structural analysis demonstrates that these assemblies retain their configuration in the design basis accident events and loading conditions.

The effective thermal conductivity for the fuel assembly used in the thermal analyses in Section 4.4 is determined by the two-dimensional fuel model (Section 4.4.1.5). The model conservatively ignores the conductance of the steel cage of the fuel assembly. Therefore, damage or physical alteration to the cage has no effect on the thermal conductivity of the fuel assembly used in the thermal models. The thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

4.5.1.1.10 Damaged Fuel Rod Holders

The Maine Yankee site specific fuel inventory includes two (2) damaged fuel rod holders designated CF1 and CA3. CF1 is a 9×9 array of tubes having roughly the same dimensions as a fuel assembly. Some of the tubes hold damaged fuel rods. CA3 is a previously used fuel assembly cage (i.e., a fuel assembly with all of the fuel rods removed), into which damaged fuel rods have been inserted.

Similar to the fuel assemblies that have damaged fuel rods, the damaged fuel rod holders will be placed in one of the two configurations of the Maine Yankee Fuel Can and their location in the basket is restricted to one of the four corner fuel tube positions of the basket. The decay heat generated by the fuel in each of these rod holders is less than one-fourth of the design basis heat load of 0.958 kW. Therefore, the thermal performance of the damaged fuel rod holders is bounded by that of the standard fuel assemblies.

4.5.1.1.11 Assemblies with Damaged Fuel Rods Inserted in Guide Tubes

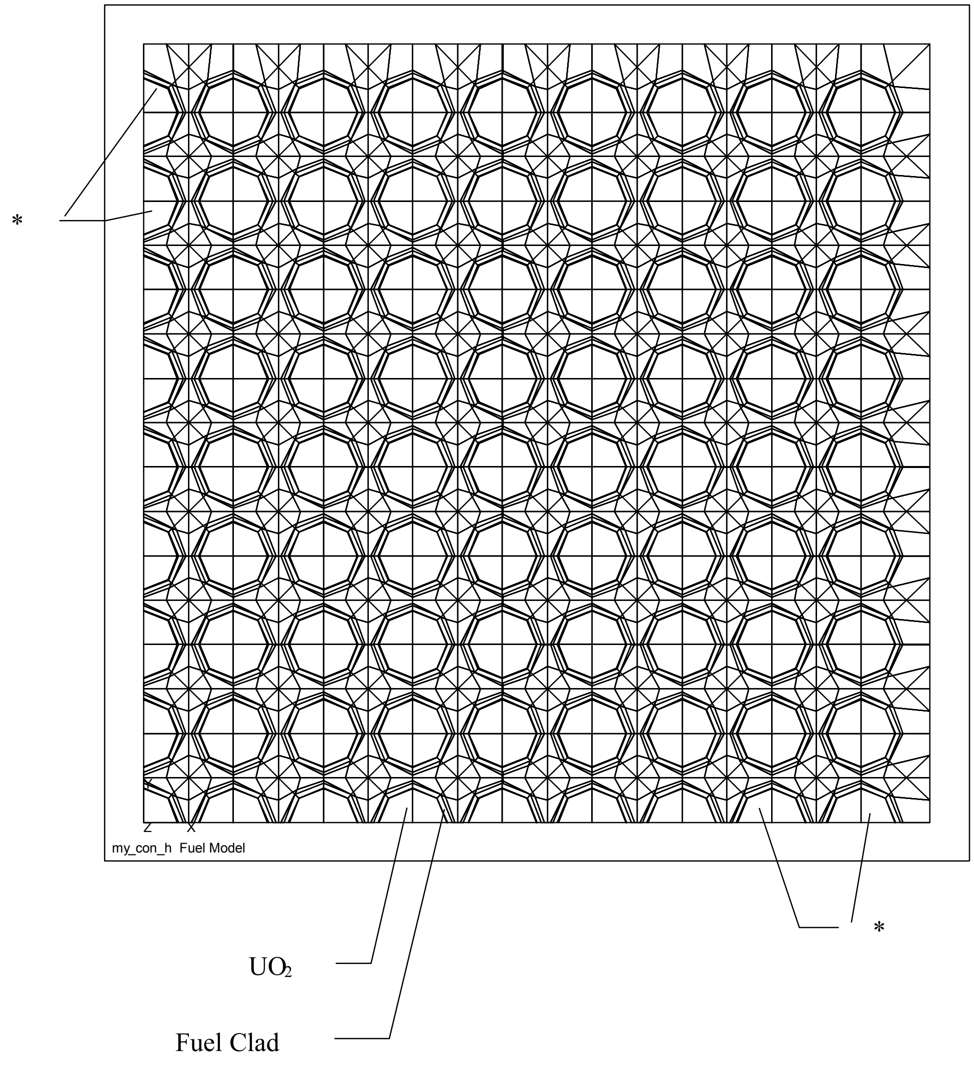
Similar to fuel assemblies that have damaged fuel rods, fuel assemblies that have up to two damaged fuel rods or poison rods stored in each guide tube are placed in one of the two configurations of the Maine Yankee Fuel Can and their loading positions are restricted to the four corner fuel tubes in the basket. The rods inserted in the guide tubes can not be from a different fuel assembly (i.e., any rod in a guide tube originally occupied a rod position in the same fuel assembly). Storing fuel rods in the guide tubes of a fuel assembly slightly increases the axial conductance of the fuel assembly (helium replaced by solid material). The design basis heat load bounds the heat load for these assemblies. Therefore, the thermal performance of fuel assemblies with rods inserted in the guide tubes is bounded by that of the standard fuel assemblies.

4.5.1.1.12 Standard Fuel Assemblies with Inserted Start-up Sources and Other Non-Fuel Items

Five Control Element Assembly (CEA) fingertips and a 24-inch ICI segment may be placed into the guide tubes of a fuel assembly. In addition, four irradiated start-up neutron sources and one unirradiated source, having a combined total heat load of 15.4 watts, will be loaded into separate fuel assemblies. With the CEA fingertips and the neutron sources inserted into the guide tubes of the fuel assemblies, the effective conductivity in the axial direction of the fuel assembly is increased because solid material replaces helium in the guide tubes. The change in the effective conductivity in the transverse direction of the fuel assembly is negligible, since the non-fuel items are inside of the guide tubes. In addition, the fuel assemblies that hold these non-fuel items are restricted to basket corner loading locations, which have an insignificant effect on the maximum fuel cladding and basket component temperatures at the center of the basket.

Note that the total heat load of the fuel assembly, including the small amount of extra heat generated by the CEA fingertips, ICI 24-inch segment, and the neutron sources, remains below the design basis heat load. Therefore, the thermal performance of the fuel assemblies with these non-fuel items inserted is bounded by that of the standard fuel assemblies.

Figure 4.5.1.1-1 Quarter Symmetry Model for Maine Yankee Consolidated Fuel



* Two outer layers (rows) of rods are modeled as stainless steel

Figure 4.5.1.1-2 Maine Yankee Three-Dimensional Periodic Canister Internal Model

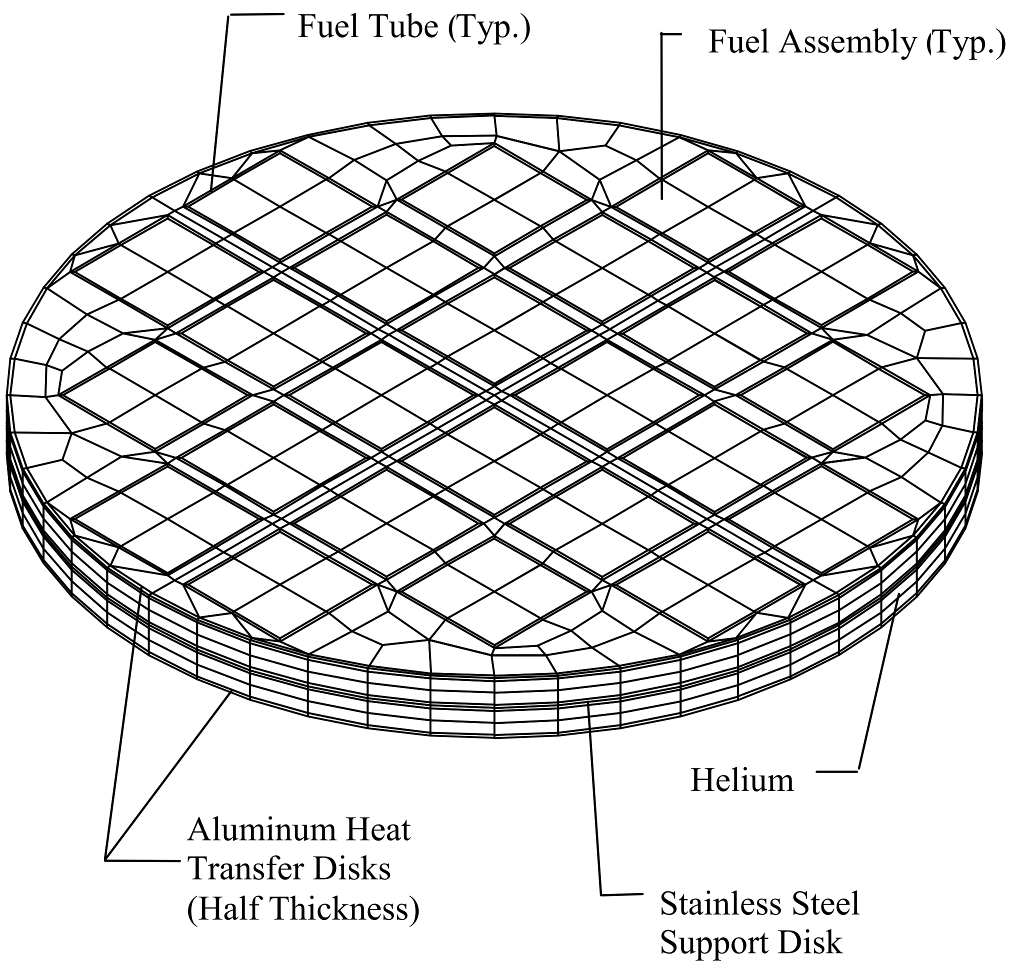


Figure 4.5.1.1-3 Evaluated Locations for the Maine Yankee Consolidated Fuel Lattice in the PWR Fuel Basket

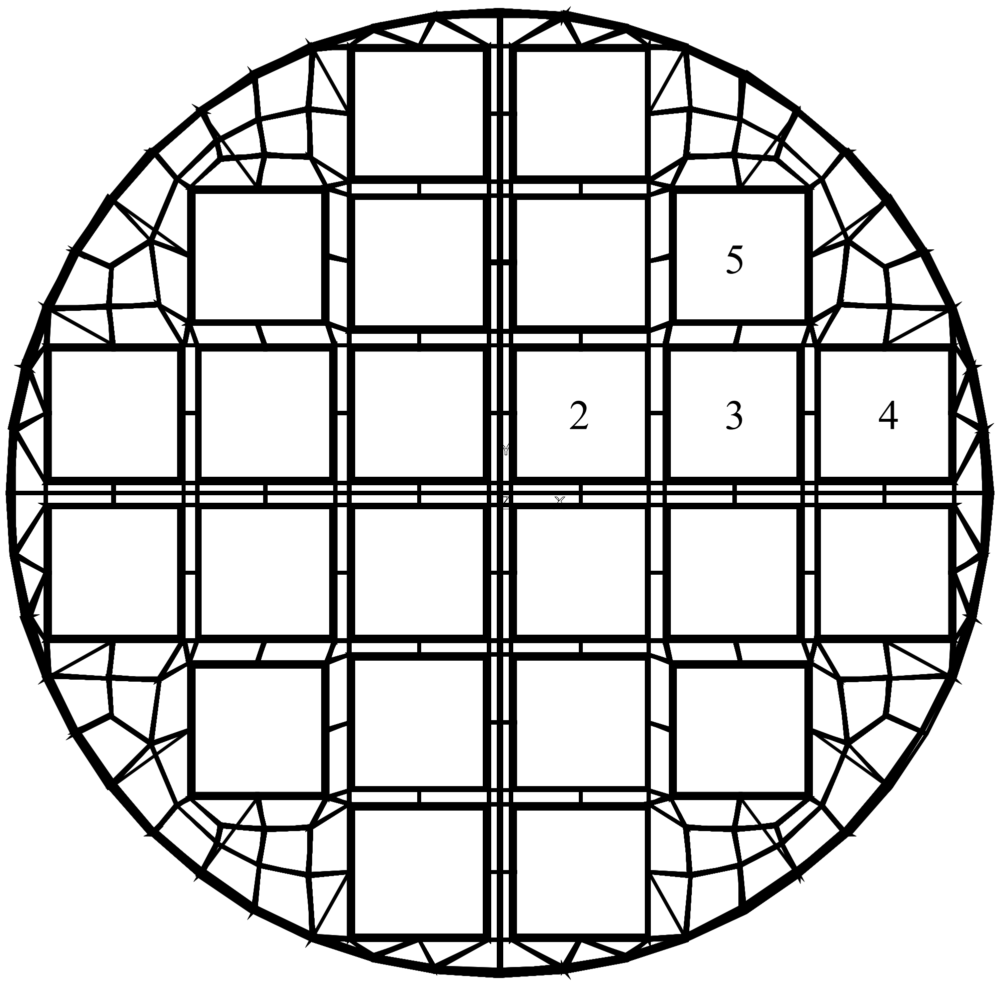
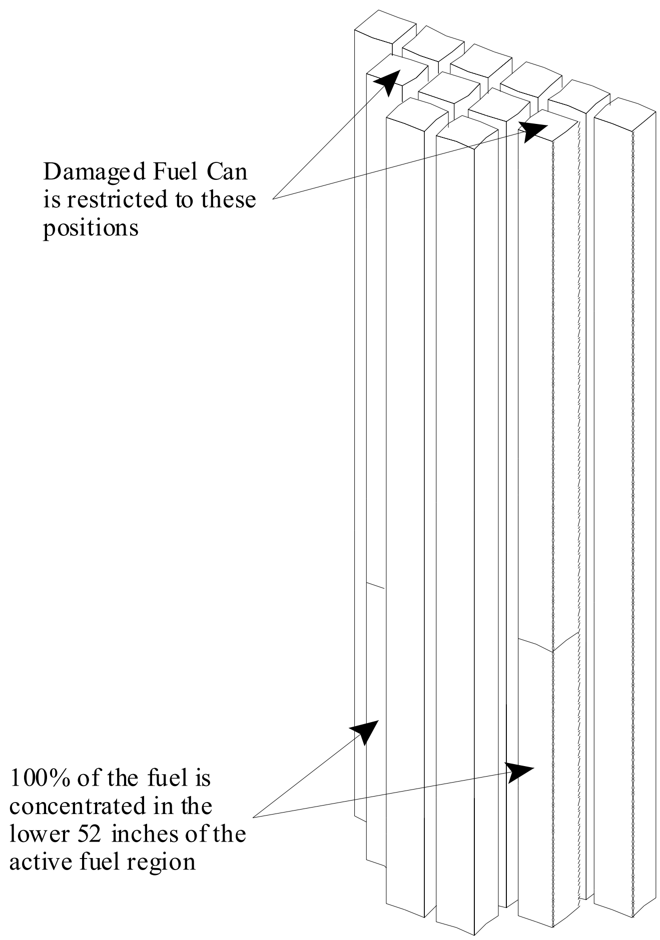
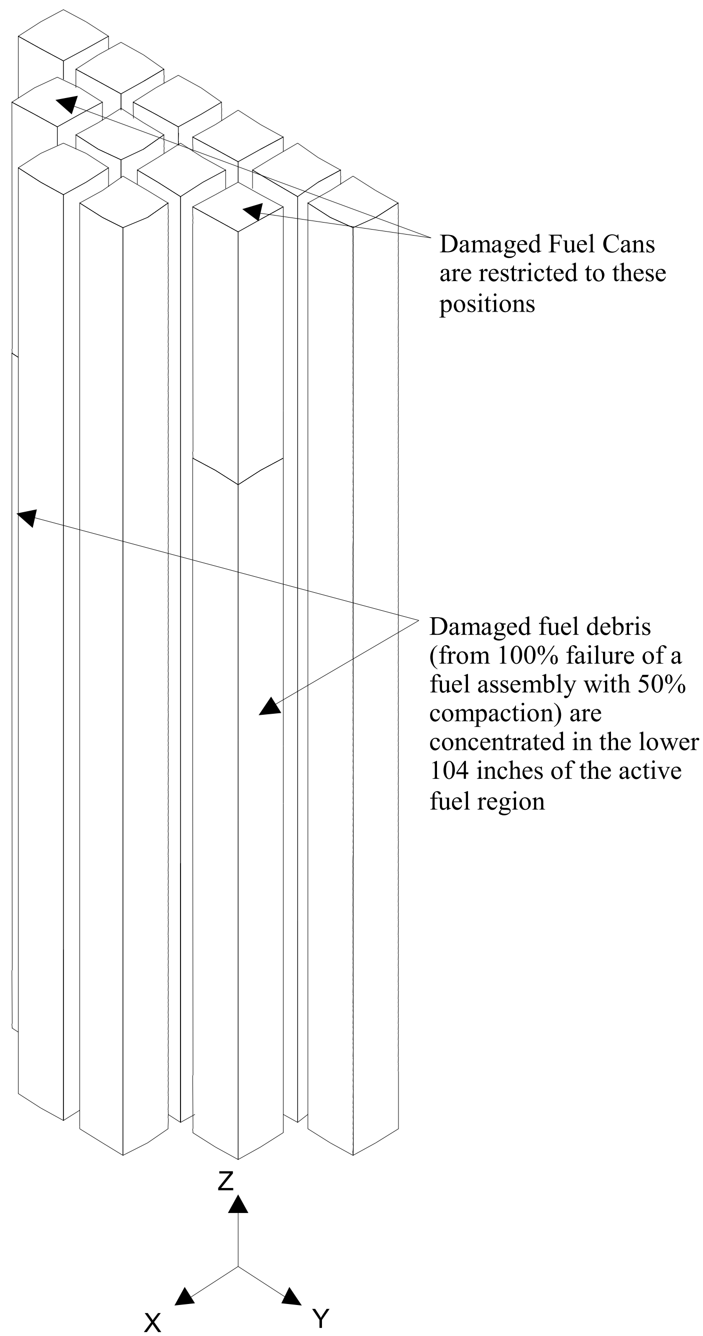


Figure 4.5.1.1-4 Active Fuel Region in the Three-Dimensional Canister Model



Note: Finite element mesh not shown for clarity.

Figure 4.5.1.1-5 Fuel Debris and Damaged Fuel Regions in the Three-Dimensional Canister Model



4.5.1.2 Preferential Loading with Higher Heat Load (1.05 kW) at the Basket Periphery

The Maine Yankee fuel inventory includes fuel assemblies that will exceed the initial per assembly heat load of 0.958 kW. To enable loading of these assemblies into the storage cask, a higher peripheral heat load is evaluated. The maximum heat load for peripheral assemblies is set at 1.05 kW. The maximum basket heat load for this configuration remains restricted to 23 kW.

To ensure that these fuel assemblies do not exceed their allowable cladding temperatures, a loading pattern is shown that places higher heat load assemblies at the periphery of the basket (Positions "A" in Figure 4.5.1.2-1) and compensates by placing lower heat load assemblies in the basket interior positions (Positions "B" in Figure 4.5.1.2-1). There are 12 interior basket locations and 12 peripheral basket locations in the UMS® PWR basket design. The maximum total basket heat load of 23 kW is maintained for these peripheral loading scenarios.

Given the higher than design basis heat load in peripheral basket locations, an evaluation is performed to assure that maximum cladding temperature does not exceed the allowable temperature of 400°C (752°F) per ISG-11, Revision 2 [37].

A parametric study is performed using the three-dimensional periodic model, as described in Section 4.5.1.1 (Figure 4.5.1.1-2), to demonstrate that placing a higher heat load in the peripheral locations does not result in heating of the fuel assemblies in the interior locations beyond that found in the uniform heat loading case. The side surface of the model is assumed to have a uniform temperature of 350°F.

Two cases are considered (total heat load per cask = 20 kW for both cases):

1. Uniform loading: Heat load = 0.833 (20/24) kW per assembly for all 24 assemblies
2. Non-uniform loading:
   Heat load = 0.958 (23/24) kW per assembly for 12 peripheral assemblies
   Heat load = 0.708 (17/24) kW per assembly for 12 interior assemblies

The analysis results (maximum temperatures) are:

	Case 1 Uniform Loading (°F)	Case 2 Non-Uniform Loading (°F)
Fuel (Location 1)	675	648
Fuel (Locations 2 & 4)	632	611
Fuel (Location 5)	577	588
Fuel (Locations 3 & 6)	563	576
Basket	611	592

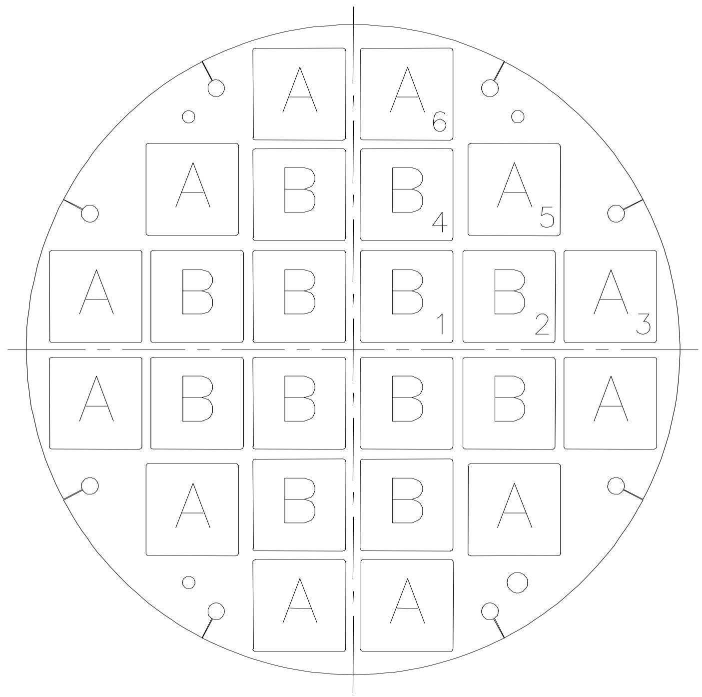
Locations are shown in Figure 4.5.1.2-1.

The maximum fuel cladding temperature for Case 2 (non-uniform loading pattern) is well below that for Case 1 (uniform loading pattern). The comparison shows that placing hotter fuel in the peripheral locations of the basket and cooler fuel in the interior locations (while maintaining the same total heat load per basket) reduces the maximum fuel cladding temperature (which occurs in the interior assembly), as well as the maximum basket temperature.

Based on the parametric study (uniform versus non-uniform analysis) of the 20 kW basket, a 15% redistribution of heat load resulted in a maximum increase of 13°F (576-563=13) in a peripheral basket location. Changing the basket peripheral location heat load from 0.958 kW maximum to 1.05 kW is a less than 10% redistribution for the 23 kW maximum basket heat load. The highest temperature of a peripheral basket location may, therefore, be estimated by adding 13°F to 566°F (maximum temperature in peripheral assemblies for the 23 kW basket with uniform heat load distribution). The 579°F (304°C) temperature is well below the allowable cladding temperature of 400°C .

Therefore, the maximum fuel cladding temperature for the preferential loading configuration with the higher heat load of 1.05 kW at the periphery basket locations will not exceed the allowable fuel cladding temperature.

Figure 4.5.1.2-1 Canister Basket Preferential Loading Plan



"A" indicates peripheral locations.

"B" indicates interior locations.

Numbered locations indicate positions where maximum fuel temperatures are presented.

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5.0 SHIELDING EVALUATION

Specific dose rate limits for individual casks in a storage array are not established by 10 CFR 72 [1]. Annual dose limit criteria for the independent spent fuel storage installation (ISFSI) controlled area boundary are established by 10 CFR 72.104 and 10 CFR 72.106 for normal conditions and for design basis accidents. These regulations require that, for an array of casks in an ISFSI, the annual dose to an individual outside the controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ during normal operations. For a design basis accident, the dose to an individual outside the controlled area boundary must not exceed 5 rem to the whole body. The ISFSI must be at least 100 meters from the owner controlled area boundary. In addition, the occupational dose limits and radiation dose limits established in 10 CFR Part 20 (Subparts C and D) [2] for individual members of the public must be met.

This chapter describes the shielding design and the analysis used to establish bounding radiological dose rates for the storage of various types of PWR and BWR fuel assemblies. The analysis shows that the Universal Storage System meets the requirements of 10 CFR 72.104 and 10 CFR 72.106 when the system is configured and used in accordance with the design basis established by this Safety Analysis Report.

The Universal Storage System compliance with the requirements of 10 CFR 72 with regard to annual and occupational doses at the owner controlled area boundary is demonstrated in Section 10.3 and 10.4.

5.1 Discussion and Results

The transfer cask is provided in either the Standard or Advanced configuration. Canister handling, fuel loading and canister closing are operationally identical for either transfer cask configuration.

The Standard and Advanced transfer casks have a radial shield comprised of 0.75 inch of low alloy steel, 4.00 inches of lead, 2.75 inches of solid borated polymer (NS-4-FR), and 1.25 inches of low alloy steel. An additional 0.625 inch of stainless steel shielding is provided, radially, by the canister shell. Gamma shielding is provided primarily by the steel and lead layers, and neutron shielding is provided primarily by the NS-4-FR. The transfer cask bottom shield design is a solid section of 7.5 inches of low alloy steel and 1.5 inches of NS-4-FR. The top shielding of the transfer cask is provided by the stainless steel canister shield and structural lids, which are 7 inches and 3 inches thick, respectively. In addition, 5 inches of steel is used as temporary shielding during welding, draining, drying, helium backfill, and other operations related to closing the canister. This temporary shielding is removed prior to storage.

The Advanced transfer cask incorporates a trunnion support plate that allows it to lift a heavier canister. The support plate has no significant shielding impact due to its location above the trunnion. The evaluations and results provided for the Standard transfer cask are, therefore, applicable to the Advanced transfer cask.

The vertical concrete cask radial shield design is comprised of a 2.5-inch thick carbon steel inner liner surrounded by 28.25 inches of concrete. Gamma shielding is provided by both the carbon steel and concrete, and neutron shielding is provided primarily by the concrete. As in the transfer cask, an additional 0.625 inch thickness of stainless steel radial gamma shielding is provided by the canister shell. The concrete cask top shielding design is comprised of 10 inches of stainless steel from the canister lids, a shield plug containing a 1-inch thickness of NS-4-FR or 1.5 inches of NS-3 and 4.1 inches of carbon steel, and a 1.5-inch thick carbon steel lid. Since the bottom of the concrete cask rests on a concrete pad, the cask bottom shielding is comprised of 1.75 inch of stainless steel from the canister bottom plate, 2 inches of carbon steel (pedestal plate) and 1 inch of carbon steel cask base plate. The base plate and pedestal base are structural components that position the canister above the air inlets. The cask base supports the concrete cask during lifting, and forms the cooling air inlet channels at the cask bottom. An optional carbon steel supplemental shielding fixture, shown in Drawing 790-613, may be installed to reduce the radiation dose rates at the air inlets.

The spent fuel that may be stored in the Universal Storage System is divided into five classes, three PWR and two BWR, depending on the length of the fuel assembly. The transportable storage canister, transfer cask, and vertical concrete cask are provided in five lengths, corresponding to the lengths of the fuel assemblies.

The shielding analysis is based on the use of bounding dose rates for the design basis PWR and BWR fuel assembly, and its associated canister, transfer cask, and concrete cask. Shielding evaluations are performed for the transfer cask with both wet and dry canister cavities. The wet canister cavity condition occurs during the welding of the shield lid. During the welding of the structural lid, the canister cavity is assumed to be completely dry. Note that in the wet canister condition, the modeled water level is the base of the upper end fitting. Shielding evaluations for the concrete cask assume a dry cavity.

Site-specific fuels, which may have configurations or parameters that are not considered in the design basis fuels, are described in Section 5.6. As described in Section 5.6, the site-specific fuels must either be shown to be bounded by the evaluation of the design basis fuel or be separately evaluated to establish limits that are maintained by administrative controls.

Shielding evaluations to determine maximum system dose rates for the range of fuel types and allowable enrichment and burnup combinations rely on a three-step approach. In the initial step, one-dimensional evaluations consider all assembly types intended for storage in the Universal Storage System at a fixed burnup, initial enrichment (referred to as shielding design basis) to determine a bounding, i.e., shielding design basis assembly design. In the second step, the design basis assembly design is evaluated using a three-dimensional Monte Carlo code to determine maximum licensing (design basis) dose rates. In the third analysis step, one-dimensional shielding evaluations are added to extend the burnup and enrichment range from the design basis values for each of the primary fuel types.

Shielding Evaluations to Determine Bounding Fuel Type at Fixed (Design Basis) Burnup, Initial Enrichment and Cool Time

The design basis PWR and BWR fuel assemblies are determined by considering all assembly types intended for storage in the Universal Storage System, and identifying those assemblies expected to have the highest source terms based on initial loading of fuel and other operating factors. The design basis depletion characteristics for PWR assemblies in this evaluation step are an average burnup of 40,000 MWd/MTU, an initial enrichment of 3.7 wt % ^{235}U, and a 5-year cooling time. The design basis BWR depletion characteristics are an average burnup of 40,000 MWd/MTU, an initial enrichment of 3.25 wt % ^{235}U, and a 5-year cooling time. Detailed source descriptions of these selected assemblies are developed by using the SCALE SAS2H code [5]. The resulting source descriptions for each assembly type are employed in one-dimensional shielding calculations in order to identify bounding design basis assembly descriptions for both PWR and BWR assemblies on the basis of computed dose rates.

The determination of design basis fuel descriptions on the basis of one-dimensional shielding analyses is a unique approach that captures the combined effects of fuel self-shielding, spectral differences between assembly source terms, the relative contributions from gamma and neutron sources, and the influence of cask shielding materials and geometry. The design basis is selected as the result of computed dose rates rather than from a single gross assembly characteristic such as source rate or initial heavy metal loading.

As discussed in Section 5.2.5, the resulting design basis PWR fuel assembly for the shielding evaluation of the standard transfer cask and vertical concrete cask is the Westinghouse 17×17 standard assembly with an average burnup of 40,000 MWd/MTU, an initial enrichment of 3.7 wt % ^{235}U, and a 5-year cooling time, modified by increasing its hardware source. The shielding design basis BWR fuel is a GE 9×9 assembly with a burnup of 40,000 MWd/MTU, an initial enrichment of 3.25 wt % ^{235}U, and a 5-year cooling time, modified by increasing its hardware source. The source term specification is provided in Section 5.2.

Maximum Licensing Dose Rates

Three-dimensional analyses of the hardware source modified Westinghouse 17×17 and GE 9×9 design basis assemblies are then conducted to establish licensing basis dose rates. The three-dimensional dose rates are calculated for the design basis 40,000 GWd/MTU burned, 5-year cooled source used in the one-dimensional fuel comparisons. Section 5.1.3 contains the resulting maximum dose rate discussion. Detailed discussions and dose rate profiles for the transfer cask and the vertical concrete cask are presented in Section 5.4. Maximum dose rates obtained from the three-dimensional analyses are generally higher than those obtained from the one-dimensional analysis, as explicit disk models versus one-dimensional homogenous smearing of the disks captures local mass details and resulting radiation shield performance. Three-dimensional evaluations are also capable of capturing dose peaks associated with radiation streaming paths, such as the air inlets and outlets of the vertical concrete cask.

Shielding and Source Term Extension to the Range of Assemblies, Burnup and Initial Enrichments

One-dimensional shielding evaluations in conjunction with heat load limits are employed in setting minimum cool times for the range of fuel types, including design basis assemblies, having different burnups and initial enrichments than the initial design basis burnup of 40 GWd/MTU with 5-year cooling and initial enrichments of 3.7 wt % ^{235}U PWR or 3.25 wt % ^{235}U BWR. Dose rate limits are set by determining one-dimensional dose rates for the 40 GWd/MTU, 5-year-cooled design basis Westinghouse 17×17 and GE 9×9 fuel assembly designs. The calculated dose rates at this depletion point are the design basis values, not to be exceeded by any other fuel type, burnup/initial enrichment/cool time combination. Not exceeding the one-dimensional limits provides assurance that the three-dimensional dose rates documented in Section 5.1.3 are not exceeded. Details on the minimum cool time evaluations are provided in Sections 5.4 and 5.5.

Shielding evaluations are performed for the transfer cask with both wet and dry canister cavities. The wet canister cavity condition occurs during the welding of the shield lid. During the welding of the structural lid, the canister cavity is assumed to be completely dry. Note that in the wet canister condition, the modeled water level is the base of the upper end fitting. Shielding evaluations for the concrete cask assume a dry cavity.

Dose rate profiles for the transfer cask and the vertical concrete cask are presented in Section 5.4.

Site-specific fuels, which may have configurations or parameters that are not considered in the design basis fuels, are described in Section 5.6. As described in Section 5.6, the site-specific fuels must either be shown to be bounded by the evaluation of the design basis fuel, or be separately evaluated to establish limits which are maintained by administrative controls.

5.1.1 Fuel Assembly Classification

5.1.1.1 PWR Fuel Assembly Classes

As discussed in Chapters 1.0 and 6.0 of this report, the PWR fuel assemblies to be stored in the vertical concrete cask are divided into three classes on the basis of similarity of their lengths. Of the PWR assemblies to be stored, the following four are selected for further analysis on the basis of their computed radiation source terms:

PWR Assembly Type	Class
Westinghouse 15×15 Std	Class 1
Westinghouse 17×17 Std	Class 1
Babcock & Wilcox 15×15 Mark B	Class 2
Combustion Engineering 16×16 System 80	Class 3

These assembly types represent candidate design basis assemblies. The design basis assembly is chosen by performing one-dimensional shielding calculations for each assembly type. The results of the one-dimensional analysis are used to identify the single limiting assembly type which is then used in subsequent detailed three-dimensional shielding calculations in order to determine bounding dose rates for the PWR case. Using this approach, the limiting assembly type is determined on the basis of actual computed dose rates, including factors such as fuel self-shielding and spectral effects, which would otherwise be ignored if the design basis were selected on the basis of source rates alone.

The candidate PWR fuel assemblies are analyzed on the basis of an assumed initial enrichment of 3.7 wt % ^{235}U, a burnup of 40,000 MWd/MTU, and a cooling time of 5 years. The initial enrichment assumed in the shielding analysis is significantly less than the criticality analysis design basis value of 4.2 wt % ^{235}U, so that the calculated neutron source rate bounds that of higher enrichment fuel, which may reach the design basis burnup of 40,000 MWd/MTU. This assumption produces a neutron source that is 30% higher than that calculated assuming a 4.2 wt % ^{235}U initial enrichment.

In addition, the source terms for each assembly type include bounding fuel and nonfuel hardware source terms associated with certain control components, including burnable poison clusters and power shaping elements specific to each fuel type. The source specifications for the design basis fuel are discussed in Section 5.2.

5.1.1.2 BWR Fuel Assembly Classes

On the basis of similarity of length, the BWR fuel assemblies to be stored in the vertical concrete cask are divided into two classes (Class 4 corresponds to BWR/2–3 assemblies and Class 5 corresponds to BWR/4–6 assemblies). In a manner similar to that employed in the PWR case, the following BWR assemblies are chosen as candidate design basis assemblies for the shielding analysis on the basis of their computed radiation source terms:

BWR Assembly Type	Class
GE 7×7 BWR/2–3 version GE-2b	Class 4
GE 8×8-2 BWR/2–3 version GE-5	Class 4
GE 8×8-4 BWR/2–3 version GE-8	Class 4
GE 7×7 BWR/4–6 version GE-2	Class 5
GE 8×8-2 BWR/4–6 version GE-5	Class 5
GE 8×8-4 BWR/4–6 version GE-10	Class 5
GE 9×9-2 BWR/4–6 version GE-11	Class 5

One-dimensional shielding calculations are performed for each assembly in order to identify a single assembly type as the design basis assembly for subsequent detailed three-dimensional shielding analysis. The candidate BWR fuel assemblies are analyzed on the basis of an initial enrichment of 3.25 wt % ^{235}U, a burnup of 40,000 MWd/MTU, and a cooling time of 5 years. The initial enrichment assumed in the shielding analysis is significantly less than the criticality analysis design basis value of 4.0 wt % ^{235}U, so that the calculated neutron source rate bounds that of higher enrichment fuel, which may reach the design basis burnup of 40,000 MWd/MTU. This assumption produces a neutron source that is 20% higher than that calculated assuming a 4.0 wt % ^{235}U initial enrichment.

5.1.2 Codes Employed

The SCALE 4.3PC [4] code system is used in the analysis of the vertical concrete cask and the transfer cask, with the MCBEND [23] code used to calculate dose rates at the concrete cask air inlets and outlets. Source terms are generated by using the SAS2H [5] sequence as described in Section 5.2. One-dimensional radial and axial SAS1 [6] analyses are performed in order to identify design basis PWR and BWR fuel types. With these design basis source descriptions, detailed three-dimensional analyses are performed by using the SAS4 [3] Monte Carlo shielding analysis sequence and the MCBEND Monte Carlo code. Modifications to SAS4 permit computation of dose rate profiles along surface detectors. These changes are further described in Section 5.4.1.

The 27-group neutron, 18 group gamma, coupled cross-section library (27N-18COUPLE) [7] derived from ENDF/B-IV data is used in the concrete cask and standard transfer cask shielding evaluations. The MCBEND shielding evaluations use the 28-group and 22-group gamma energy structures embedded in the code. Source terms include fuel neutron, fuel gamma, and gamma contributions from activated hardware. The effects of subcritical neutron multiplication and secondary gamma production due to neutron capture are included in the analysis. Dose rate evaluations include the effect of axial fuel burnup variation on fuel neutron and gamma source terms as described in Section 5.2.6.

5.1.3 Results of Analysis

This section summarizes the results of the three-dimensional shielding analysis. Reported values are rounded up to the indicated level of precision. Due to the statistical nature of Monte Carlo analysis, all dose rate results are shown with the relative standard deviation in the result, expressed as a percentage.

5.1.3.1 Dose Rates for Vertical Concrete Cask

Cask Containing PWR Fuel

A summary of the maximum calculated dose rates for the concrete cask under normal and accident conditions is shown in Table 5.1-1 for the design basis PWR fuel. These dose rates are based on three-dimensional Monte Carlo analysis. Uncertainty in Monte Carlo results is indicated in parentheses. Under normal conditions with design basis fuel and the Transportable Storage Cask centered in the Vertical Concrete Cask, the concrete cask maximum side wall surface dose rate is 49 (<1%) mrem/hr at the fuel midplane and 56 (6%) mrem/hr on the top surface at locations on the cask top directly above the outlet vents. Since the concrete cask is vertical during normal storage operation, the cask bottom is inaccessible. The maximum surface dose rate at the lower air inlet openings is 136 (1%) mrem/hr with supplemental shielding and 694 (<1%) mrem/hr without supplemental shielding. The maximum surface dose rate at the air outlet openings is 63 (1%) mrem/hr. The average maximum inlet plus outlet dose rate is 99.5 mrem/hr with supplemental shielding.

The overall cask side average surface dose rate is 38 (<1%) mrem/hr for the PWR design basis fuel. On the cask top, the PWR average surface dose rate is 27 (2%) mrem/hr.

The postulated accident condition involves a projectile impact resulting in localized loss of 6 inches of concrete. The accident is analyzed assuming that the outermost 3 inches of concrete is lost from the entire outer surface of the cask. In this case, the surface average dose rate increases to 89 (<1%) mrem/hr with design basis PWR fuel. The maximum dose rate, assuming a 3-inch concrete loss over the entire radial surface of the cask, is 143 (3%) mrem/hr. At the postulated missile impact area, the estimated localized dose rate is less than 250 mrem/hr. There are no design basis accidents that result in a tip-over of the concrete cask.

Cask Containing BWR Fuel

Table 5.1-2 provides the maximum calculated dose rates for the concrete cask under normal and accident conditions for the design basis BWR fuel. As in the PWR case, these dose rates are based on three-dimensional Monte Carlo analysis. Uncertainty in Monte Carlo results is indicated in parentheses. Under normal conditions with design basis BWR fuel, the concrete cask maximum side surface dose rate is 31 (1%) mrem/hr at the fuel midplane and 43 (5%) mrem/hr on the top surface at locations directly above the air outlet structures. The dose rate at the air inlet opening is 129 (1%) mrem/hr with supplemental shielding and 645 (<1%) mrem/hr without supplemental shielding. The maximum surface dose rate at the air outlet openings is 55 (1%) mrem/hr.

Under accident conditions involving a projectile impact and an assumed 3 inches of concrete removed from the entire radial surface of the cask, the surface dose rate maximum increases to 85 (4%) mrem/hr with design basis BWR fuel. The radial surface average dose rate increases to 54 (<1%) mrem/hr and the cask surface dose rate for the localized loss of 6 inches of concrete is estimated to be less than 250 mrem/hr.

The overall cask side average surface dose rates are 23 (<1%) mrem/hr for the BWR design basis fuel. On the cask top, the BWR average surface dose rate is 20 (1%) mrem/hr.

5.1.3.2 Dose Rates for Transfer Cask

Transfer Cask Containing PWR Fuel

Maximum dose rates for the standard or advanced transfer cask with a wet and dry canister cavity are shown in Table 5.1-3 for design basis PWR fuel. Under wet canister conditions, the maximum surface dose rates with design basis PWR fuel are 259 (<1%) mrem/hr on the cask side and 579 (<1%) mrem/hr on the cask bottom. The cask side average surface dose rate under wet conditions is 137 (<1%) mrem/hr, and the bottom average surface dose rate is 258 (<1%) mrem/hr. Under dry conditions, the maximum surface dose rates are 410 (<1%) mrem/hr on the cask side and 819 (<1%) mrem/hr on the cask bottom. Cask average surface dose rates are 306 (<1%) mrem/hr on the side and 374 (<1%) mrem/hr on the bottom. In normal operation, the bottom of the transfer cask is inaccessible during welding of the canister lids.

During the lid welding operation, localized maximum surface dose rates occur at the canister periphery. Under wet canister conditions with a 5-inch temporary shield in place atop the shield lid, the maximum contact dose rate is 2,092 (4%) mrem/hr. This dose rate is highly localized to the narrow gap between the temporary shielding and the cask inner wall. At 1 meter above the top of the cask, the maximum dose rate is 320 (6%) mrem/hr. The surface average dose rate at the cask top surface is 579 (3%) mrem/hr under these conditions.

Under dry conditions with the shield lid and structural lid in place, and with no additional temporary shielding, the maximum surface dose rate is 715 (<1%) mrem/hr. The cask top average surface dose rate is 369 (2%) mrem/hr under these conditions.

Transfer Cask Containing BWR Fuel

Maximum dose rates for the standard or advanced transfer cask with a wet and dry canister cavity are shown in Table 5.1-4 for design basis BWR fuel. Under wet canister conditions, the maximum surface dose rates with design basis BWR fuel are 189 (<1%) mrem/hr on the cask side and 539 (<1%) mrem/hr on the cask bottom. The cask side average surface dose rate under wet conditions is 79 (<1%) mrem/hr, and the bottom average surface dose rate is 254 (<1%) mrem/hr. Under dry conditions, the maximum surface dose rates are 325 (<1%) mrem/hr on the cask side and 786 (<1%) mrem/hr on the cask bottom. Cask average surface dose rates are 228 (<1%) mrem/hr on the side and 379 (<1%) mrem/hr on the bottom. In normal operation, the bottom of the transfer cask is inaccessible during welding of the canister lids.

During the lid welding operation, localized maximum dose rates occur at the canister periphery. Under wet canister conditions with a 5 inch temporary shield in place atop the shield lid, the maximum surface dose rate is 1803 (4%) mrem/hr. This dose rate is highly localized to the narrow gap between the temporary shielding and the cask inner wall. At 1 meter above the top of the cask, the maximum dose rate is 314 (7%) mrem/hr. The surface average dose rate at the cask top surface is 466 (3%) mrem/hr under these conditions.

Under dry conditions with the shield lid and structural lid in place, and no additional temporary shielding, the maximum surface dose rate is 396 (<1%) mrem/hr. The cask top average surface dose rate is 222 (3%) mrem/hr under these conditions.

Table 5.1-1 Summary of Maximum Dose Rates: Vertical Concrete Cask with PWR Fuel

Condition	Source	Cask Surface (mrem/hr with relative uncertainty)				1 Meter From Surface (mrem/hr with relative uncertainty)			
		Side		Top		Side		Top	
Normal	Neutron	0.1	1%	0.3	14%	<0.1	<1%	5.3	1%
	Gamma	48.6	<1%	55.1	6%	25.2	<1%	8.0	7%
	Total	49.[2]	<1%	56.	6%	26.	<1%	14.	5%
Design Basis Accident	Neutron	0.3	10%	N/A[1]		0.1	2%	N/A[1]	
	Gamma	141.9	3%	N/A[1]		62.5	<1%	N/A[1]	
	Total	143.[3]	3%	N/A[1]		63.	<1%	N/A[1]	

1. No design basis accident impacts top dose rates.
2. At the fuel midplane. Without supplemental shielding, the air inlet dose rate is 694 (<1%) mrem/hr.
3. At the missile impact area.

Table 5.1-2 Summary of Maximum Dose Rates: Vertical Concrete Cask with BWR Fuel

Condition	Source	Cask Surface (mrem/hr with relative uncertainty)				1 Meter From Surface (mrem/hr with relative uncertainty)			
		Side		Top		Side		Top	
Normal	Neutron	0.2	<1%	0.2	19%	<0.1	<1%	3.2	2%
	Gamma	30.6	1%	42.2	5%	15.3	<1%	5.3	4%
	Total	31.[2]	1%	43.	5%	16.	<1%	9.	2%
Design Basis Accident	Neutron	0.5	8%	N/A[1]		0.2	1%	N/A[1]	
	Gamma	83.8	4%	N/A[1]		38.3	1%	N/A[1]	
	Total	85.[3]	4%	N/A[1]		39.	1%	N/A[1]	

1. No design basis accident impacts top dose rates.
2. At the fuel midplane. Without supplemental shielding, the air inlet dose rate is 645 (<1%) mrem/hr.
3. At the missile impact area.

Table 5.1-3 Summary of Maximum Dose Rates: Standard or Advanced Transfer Cask with PWR Fuel

		Cask Surface (mrem/hr with relative uncertainty)						1 Meter From Surface (mrem/hr with relative uncertainty)					
Condition	**Source**	**Side**		**Top**		**Bottom**		**Side**		**Top**		**Bottom**	
Normal Wet[1]	Neutron	0.1	8%	0.2	3%	0.3	2%	1.3	<1%	<0.1	2%	0.1	2%
	Gamma	258.7	<1%	2091.	4%	578.2	<1%	65.3	<1%	319.8	6%	266.4	<1%
	Total	259.	<1%	2092.	4%	579.	<1%	67.	<1%	320.	6%	267.	<1%
Normal Dry[2]	Neutron	12.6	2%	111.5	<1%	37.8	<1%	29.5	<1%	28.7	1%	10.0	<1%
	Gamma	397.2	<1%	603.4	<1%	781.1	<1%	126.5	<1%	278.3	<1%	365.5	<1%
	Total	410.	<1%	715.	<1%	819.	<1%	156.	<1%	307.	<1%	376.	<1%

[1] 5 inches of carbon steel temporary shielding, shield lid in position.
[2] Shield lid and structural lid in position, no additional temporary shielding.

Table 5.1-4 Summary of Maximum Dose Rates: Standard or Advanced Transfer Cask with BWR Fuel

		Cask Surface (mrem/hr with relative uncertainty)						1 Meter From Surface (mrem/hr with relative uncertainty)					
Condition	**Source**	**Side**		**Top**		**Bottom**		**Side**		**Top**		**Bottom**	
Normal Wet[1]	Neutron	<0.1	31%	<0.1	17%	<0.1	7%	2.3	<1%	<0.1	12%	<0.1	6%
	Gamma	188.2	<1%	1803.	4%	538.1	<1%	34.8	<1%	313.2	7%	258.5	<1%
	Total	189.	<1%	1803.	4%	539.	<1%	38.	<1%	314.	7%	259.	<1%
Normal Dry[2]	Neutron	152.3	<1%	62.1	1%	34.7	1%	53.5	<1%	16.3	2%	9.3	1%
	Gamma	171.8	1%	333.6	<1%	750.7	<1%	67.3	<1%	156.4	1%	360.3	<1%
	Total	325.	<1%	396.	<1%	786.	<1%	121.	<1%	173.	1%	370.	<1%

[1] 5 inches of carbon steel temporary shielding, shield lid in position.
[2] Shield lid and structural lid in position, no additional temporary shielding.

5.2 Source Specification

The procedure used to identify a design basis fuel assembly for PWR and BWR fuel types is described in this section. Each of the candidate fuel assemblies described in Section 5.1.1 is represented in one-dimensional radial and axial models of the fully loaded cask. The results of this one-dimensional shielding analysis are then used to identify a limiting fuel design for PWR and BWR fuel types. The limiting fuel design is then used in the shielding evaluation of the standard transfer cask and vertical concrete cask.

The SAS2H code sequence [5] is used to generate source terms for the shielding analysis. This code sequence is part of the SCALE 4.3 code package [4] for the personal computer (PC). SAS2H includes an XSDRNPM [8] neutronics model of the fuel assembly and ORIGEN-S [9] fuel depletion and source term calculations. Source terms are generated for both UO_2 fuel and fuel assembly hardware. The hardware activation is calculated by light element transmutation using the incore neutron flux spectrum produced by the SAS2H neutronics model. The hardware is assumed to be Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity. The effects of axial flux spectrum and magnitude variation on hardware activation are estimated by flux ratios determined from empirical data [15].

References [24] through [28] contain extensive SAS2H validation for PWR burnups up to 47 GWd/MTU and BWR burnups up to 57 GWd/MTU. As indicated in the reference documentation, the SAS2H sequence is applicable to LWR fuel assembly source term generation for high burnup fuel. Open literature validations of the SCALE SAS2H sequence versus experimental data do not extend to the system allowable burnup of 62.5 GWd/MTU peak average rod. Studies performed in NUREG/CR-6701 (Appendix B) [29] indicate no analysis trends in system sensitivity for LWR evaluations up to a burnup of 75 GWd/MTU. As such, the SAS2H sequence is applicable to the high burnup fuel evaluated. Not all benchmarking referenced employed SAS2H as a component of SCALE 4.3 with the 27-group library as employed in the UMS® source term generation. References also employed SCALE 4.4 with the 44-group library. NAC comparisons indicate good agreement for PWR source terms generated using the two methods.

5.2.1 Design Basis Gamma Source

The fuel gamma source contains contributions from both fission products and actinides. The spectra are presented in the 18-group structure consistent with the SCALE 4.3 27N-18COUPLE cross-section library. The hardware gamma spectra contain contributions primarily from ^{60}Co due to the activation of Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity and with minor contributions from ^{59}Ni and ^{58}Fe. The hardware gamma spectral distribution is determined by the irradiation of 1 kg of stainless steel in the incore flux spectrum produced by the SAS2H neutronics calculation.

The activated fuel assembly hardware source term magnitudes are found by multiplying the source strength from 1 kg by the total mass of steel and inconel in the plenum, upper end fitting, or lower end fitting regions, and by then multiplying this result by a regional flux activation ratio. This regional flux ratio accounts for the effects of both magnitude and spectrum variation on hardware activation. These ratios are determined from empirical data [15]. A flux ratio of 0.2 is applied to hardware regions directly adjacent to the active core region, i.e., upper plenum and lower end fitting (or lower plenum, if present), and a flux ratio of 0.1 is applied to hardware regions once removed from the active core region, i.e., upper end fitting region.

In evaluations using the SCALE package, spectra are presented in the 18-group structure consistent with the SCALE 4.3 27N-18COUPLE cross-section library. MCBEND evaluations employ the same spectra rebinned onto the 22-group structure inherent to the code, shown in Table 5.2-29.

5.2.2 Design Basis Neutron Source

Light water reactor spent fuel neutron sources result from actinide spontaneous fission and from (α,n) reactions. The isotopes ^{242}Cm and ^{244}Cm characteristically produce all but a few percent of the spontaneous fission neutrons and (α,n) source in PWR and BWR fuel. The next largest contribution is from (α,n) reactions in ^{238}Pu. The neutron spectra for each emission type are included in the ORIGEN-S nuclear data libraries of the SCALE 4.3 code package. The spectra are collapsed from the energy group structure of the data library into that of the SCALE 27-group neutron cross-section library [7].

In evaluations using the SCALE package, the spectra are collapsed from the energy group structure of the data library into that of the SCALE 27-group neutron cross-section library [7]. MCBEND evaluations employ the same spectra rebinned onto the 28-group structure inherent to the code, shown in Table 5.2-28.

Neutron shielding evaluations for fissile material must account for subcritical multiplication (neutron production) inside the system being evaluated. This subcritical multiplication may be taken into account either by directly calculating the additional neutron source during the Monte Carlo simulation, as done in MORSE, or by adjusting the input neutron source term or output dose result by a scale factor. The code module in MCBEND responsible for accelerating result convergence by importance biasing does not efficiently account for subcritical multiplication. Code biasing is set to optimize the speed at which cask surface dose rates are obtained. Thermal energy neutrons within the fuel region are not likely to escape the shielded storage system and tend to be biased out of the evaluation. However, the thermal neutrons account for a significant portion of the subcritical multiplication. Removing the thermal neutrons from the system by biasing for cask surface dose, therefore, undersamples the subcritical multiplication. To account for undersampling, neutron source rates are scaled by a subcritical multiplication factor based on the system multiplication factor, k_{eff}:

$$\text{Scale Factor} = \frac{1}{1 - k_{eff}}$$

For dry cask conditions, the system k_{eff} is taken as 0.4, with a resulting scale factor of 1.67. The scale factor is applied in MCBEND input as a component to the source strength in Unit 15 (source strength).

5.2.3 PWR Fuel Assembly Descriptions

The radiation source in the Universal Storage System consists of 24 design basis PWR spent fuel assemblies. The design basis PWR fuel has an average burnup of 40,000 MWd/MTU, an initial enrichment of 4.2 wt % ^{235}U, and a post-irradiation cooling time of 5 years and includes source contributions from an activated burnable absorber assembly. However, to bound the neutron source produced by lower enrichment fuel which may achieve this burnup, the design basis PWR source terms are calculated with an initial enrichment of 3.7 wt % ^{235}U. This assumption produces a neutron source 30% higher than that obtained by assuming 4.2 wt % ^{235}U initial enrichment. Assembly power density and cycle parameters are selected such that the assembly is activated at a power level 5% greater than a typical PWR assembly to allow for assembly power peaking during core residence. This treatment results in conservatively higher source rates due to enhanced actinide production and a shorter activation period.

Source spectra and source region elevations are determined for the four major PWR fuel assembly types (see Table 5.2-1):

PWR Assembly Type	Class
Westinghouse 15×15 Std	Class 1
Westinghouse 17×17 Std	Class 1
Babcock & Wilcox 15×15 Mark B	Class 2
Combustion Engineering 16×16 System 80	Class 3

These assembly types are referred to by the abbreviated names given in Table 5.2-1. Based on their initial heavy metal loading, these assemblies produce the limiting source terms for the specified design basis burnup of 40,000 MWd/MTU. Fuel assembly physical characteristics are given in Table 5.2-2, and hardware masses are given in Table 5.2-3. The results of the source term analysis for the fuel types given here are summarized in Table 5.2-4. Fuel assembly activated hardware source terms are shown in Table 5.2-5. These non-fuel source terms are determined on the basis of the hardware source per kilogram given in Table 5.2-4 and the hardware masses given in Table 5.2-3. The hardware activation is based on a stainless steel Type 304 composition with an assumed ^{59}Co impurity level of 1.2 g/kg. A sketch of the WE 17×17 fuel assembly source region elevations is shown in Figure 5.2-5. Additional assembly detail is employed in the MCBEND assembly models. Three-dimensional parameters for the MCBEND fuel assembly model are shown in Table 5.2-27.

In order to account for spectral differences in the activating neutron flux, a flux ratio of 0.2 is applied to hardware regions directly adjacent to the active core region, i.e., the lower end-fitting and upper plenum. A flux ratio of 0.1 is applied to the upper end-fitting region, except for the CE 16×16 upper end-fitting for which a 0.05 flux ratio is used. The lower end fitting region in the BW 15×15 fuel assembly model uses a 0.1 flux ratio since the model explicitly includes a lower plenum region adjacent to the fuel region. This lower plenum region in the BW 15×15 assembly is activated with a 0.20 flux ratio. The ORIGEN-S code is used directly to calculate hardware activation spectra by activating the fuel assembly components in the SAS2H-calculated flux spectrum for each assembly type.

Sources for fuel assemblies at enrichment, burnup and initial cool times different from those discussed in this section are generated within the Section 5.5 minimum cool time matrix determinations. The 40 GWd/MTU sources described in this section produce system maximum dose rates, as cool times are adjusted for any other burnup limit to produce dose rates not to exceed the 40 GWd/MTU design basis values.

5.2.4 BWR Fuel Assembly Descriptions

The Universal Storage System can store up to 56 undamaged BWR fuel assemblies. BWR fuel is analyzed on the basis of 3.25 wt % ^{235}U initial enrichment, 40,000 MWd/MTU average burnup, and a post irradiation cooling time of 5 years. Assembly power density and cycle parameters are selected such that the assembly is activated at a power level 10% greater than a typical BWR assembly to allow for assembly power peaking during core residence. This treatment results in conservatively higher source rates due to enhanced actinide production and a shorter activation period.

Source term spectra and source region elevations are determined for the major BWR fuel assembly types (see Table 5.2-1):

BWR Assembly Type	Class
GE 7×7 BWR/2-3 version GE-2b	Class 4
GE 8×8 BWR/2-3 version GE-5, 2 water holes	Class 4
GE 8×8 BWR/2-3 version GE-10, 1 large water hole	Class 4
GE 7×7 BWR/4-6 version GE-2	Class 5
GE 8×8 BWR/4-6 version GE-5, 2 water holes	Class 5
GE 8×8 BWR/4-6 version GE-10, 1 large water hole	Class 5
GE 9×9 BWR/4-6 version GE-11, 2 water holes, 79 fuel rods	Class 5

These assembly types are referred to by the abbreviated names given in Table 5.2-1. The abbreviated name of each BWR class assembly includes a suffix designation indicating the reactor type. The "S" designation corresponds to BWR/2-3 class reactors, and the "L" designation corresponds to BWR/4-6 class reactors. The physical characteristics of the two classes of BWR fuel are given in Table 5.2-6 and Table 5.2-7. For fuel assemblies with an average burnup of 40,000 MWD/MTU, the fuel requires a minimum of 5 years of cooling after discharge to meet the radiation source rate values specified in Table 5.2-8 and Table 5.2-9. The GE BWR/2-3 8×8 fuel assembly designs are analyzed on the basis of a 144-inch active fuel length in order to provide a consistent basis for comparison with the other BWR/2-3 fuel assembly designs. The GE-2b version of the GE BWR/2-3 7×7 fuel assembly is selected over the older GE-2a design since it has been discharged more recently, although the GE-2a assembly has a marginally higher (0.4%) initial heavy metal loading. A sketch of the GE 9×9-2L fuel assembly source region elevations is shown in Figure 5.2-6. Additional assembly detail is employed in the MCBEND assembly models. Three-dimensional parameters for the MCBEND fuel assembly model are shown in Table 5.2-27.

Sources for fuel assemblies at enrichment, burnup and initial cool times different from those discussed in this section are generated within the Section 5.5 minimum cool time matrix determinations. The 40 GWd/MTU sources described in this section produce system maximum dose rates, as cool times are adjusted for any other burnup limit to produce dose rates not to exceed the 40 GWd/MTU design basis values.

5.2.5 Design Basis Fuel Assemblies

For the shielding analysis, the WE 17×17 and GE 9×9-2L fuel assembly types are selected as the design basis PWR and BWR fuel assemblies, respectively. These assembly designs are selected on the basis of the one-dimensional shielding analysis results for both the storage and standard transfer casks. Standard transfer cask results are presented in Table 5.2-10 through Table 5.2-15. Similar results are obtained for the storage cask one-dimensional analysis. To facilitate comparison, the results for each fuel assembly type are shown on a normalized basis relative to the design basis fuel assembly dose rates. With the exceptions discussed below, the computed dose rates vary over a narrow range.

In the PWR case, the inclusion of source terms from fuel assembly control components (i.e., burnable poison clusters) causes the WE 15×15 assembly to give slightly higher dose rates than the WE 17×17 assembly. However, the WE 17×17 is limiting with respect to dose rate delivered by fuel neutron and fuel gamma sources alone. Hence, in order to develop a single limiting fuel description, the WE 17×17 upper end-fitting and fuel hardware source rates are scaled to match the WE 15×15 values. This scaling results in a 35% increase in the WE 17×17 end-fitting source rate and a 17% increase in the fuel hardware source rate. Both of these source rate increases are considered in the SCALE and MCBEND evaluations. In the SCALE analysis, no corresponding adjustment is made to material smear densities; the MCBEND analysis takes credit for the additional self-shielding as shown by the activated hardware inventory in Table 5.2-30.

Five-year cooled source spectra for the PWR design basis WE 17×17 fuel assembly are shown in Table 5.2-16 through Table 5.2-18.

In the BWR case, the GE 7×7 BWR/2–3 and GE 7×7 BWR/4–6 fuel assembly types show the highest radial model dose rates. However, these assemblies are not considered as design basis assemblies because they are no longer in common use, and the U.S. spent fuel inventory does not contain a significant number of these assemblies with burnup, initial enrichment, and decay time combinations leading to source rates as high as those of the GE 9×9-2L [10].

In a manner similar to the PWR case, the GE 9×9-2L assembly represents the bounding case with respect to dose rate delivered by fuel neutron and fuel gamma sources, but the inclusion of additional non-fuel hardware sources leads to higher overall dose rates from the GE 8×8-4L assembly. Again, a bounding characterization is achieved by scaling the GE 9×9-2L upper end fitting, lower end fitting, and fuel hardware source terms to match the GE 8×8-4L values. As for the PWR models, both the SCALE and MCBEND analyses have scaled source terms, while the MCBEND analysis takes credit for the self-shielding of the additional mass as shown in Table 5.2-30. A summary of non-fuel hardware scale factors is provided in Table 5.2-24. These scale factors are included in the MCBEND analysis by increasing the activated mass as shown in Table 5.2-30.

Five-year cooled source spectra for the BWR design basis GE 9×9-2L fuel assembly are shown in Table 5.2-19 through Table 5.2-21.

5.2.6 Axial Profiles

5.2.6.1 Axial Burnup Profile

For PWR fuel with burnup exceeding 30 GWD/MTU, an enveloping axial burnup profile with a 1.08 uniform peaking factor can be justified on the basis of calculated PWR data from Seabrook Station and Maine Yankee and from measured Turkey Point gamma data [16,17,18,19,20]. This normalized enveloping shape is shown in Figure 5.2-1. A uniform burnup peaking factor of 1.08 is applied between 15% and 85% of core height. Above and below these elevations, the relative burnup/decay heat decreases linearly to 0.547 at the top and bottom of the active fuel region.

For BWR fuel with burnup exceeding 30 GWD/MTU, an enveloping burnup profile with a 1.22 maximum peaking factor can be justified on the basis of calculated BWR data from Washington Public Power BWR/4-6 data [21]. This normalized enveloping shape is shown in Figure 5.2-2. Uniform peaking factors of 1.22 and 1.18 are applied from 15% to 55% and from 55% to 80% of core height, respectively. Above and below these elevations, the burnup profile decreases linearly to 0.043 at the top and bottom of the active fuel region.

5.2.6.2 Axial Source Profile

In the three-dimensional analyses, axial radiation source rate profiles are related to the axial burnup profile described in the previous section. Source rates are assumed to vary with burnup according to:

$$S = aB^b$$

where "S" is the source rate for a particular radiation type, "B" is the burnup at a given axial elevation, "a" is a normalization factor, and "b" is the exponent given in Table 5.2-22 for each radiation type. The exponent "b" is determined from fits to SAS2H-computed source rates at various burnups for both PWR and BWR fuels. The numeric value of "a" is not computed explicitly.

Neutron source is not proportional to burnup. Therefore, the axially integrated source is not equal to the source at the average burnup. MCBEND directly applies the source profile as axial source scaling factors. By default, SAS4 normalizes the source profile, and the scaling factor is applied to the source magnitude. This scaling factor "r" relates the total source rate (SAS4 input parameter) to the source rate at the average burnup (as computed from SAS2H analyses).

$$\mathrm{r} = \frac{\overline{\mathrm{S}}}{\mathrm{S}(\overline{\mathrm{B}})} = \frac{\frac{\mathrm{a}}{\mathrm{H}}\int \mathrm{B}^{\mathrm{b}}\mathrm{dz}}{\mathrm{a}\overline{\mathrm{B}}^{\mathrm{b}}}$$

where "H" is the height of the fuel region. With the burnup profile normalized to unity, this becomes:

$$r = \frac{1}{H}\int B^{b}\,dz\,.$$

The integral is evaluated numerically by using the trapezoid rule, and the resulting scale factors are shown in Table 5.2-23 for PWR and BWR neutron source rates. The scale factor for gamma sources is 1.0 because the computed relation between gamma source rate and burnup is linear.

The fuel neutron and fuel gamma source rate profile for the design basis PWR fuel assembly is tabulated in Table 5.2-25 and shown graphically in Figure 5.2-3. Corresponding BWR profiles are given in Table 5.2-26 and Figure 5.2-4.

In the BWR case, the axial source profiles are asymmetric with respect to the fuel axial midplane. To ensure that the correct total source is modeled in each SAS4 half model, a scale factor is computed which relates the actual source rate in each half model to the total assembly source rate. These values are shown in Table 5.2-24. This scaling is necessary in order that cask features located near the top or bottom of the cask are modeled correctly.

The results of the one-dimensional dose rate calculations indicate that bounding, conservative PWR and BWR source descriptions are achieved by scaling the design basis WE 17×17 and GE 9×9-2L non-fuel hardware gamma source rates to match the corresponding WE 15×15 and

GE 8×8-4L values, respectively. The SCALE evaluations perform a source rate scaling without the corresponding increase in mass; the MCBEND evaluations both increase the source and the associated mass. This scaling is only performed in the analysis of the standard transfer cask and the storage cask.

As a final remark, the scale factors given in Table 5.2-24 are actually applied in a post-processing step to the computed dose rate associated with each source region, rather than to the source rate itself. In this manner, all SAS1 and SAS4 input files are developed using consistent source rates.

Figure 5.2-1 Enveloping Axial Burnup Profile for PWR Design Basis Fuel

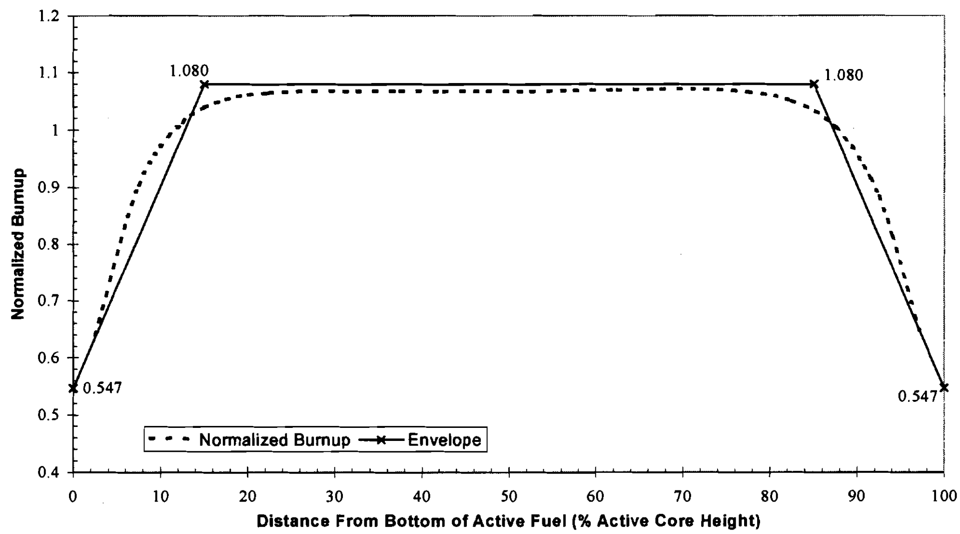


Figure 5.2-2 Enveloping Axial Burnup Profile for BWR Design Basis Fuel

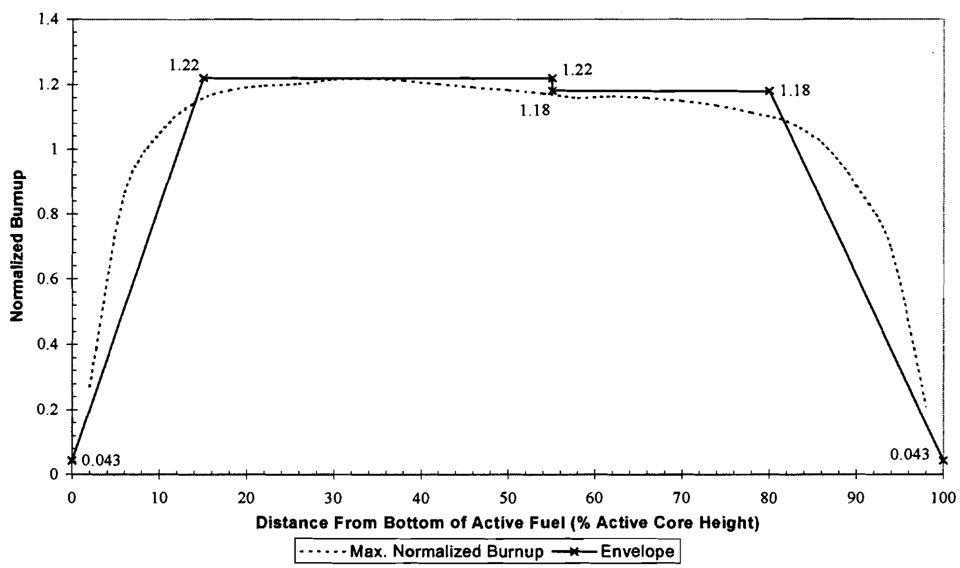


Figure 5.2-3 PWR Photon and Neutron Axial Source Profiles

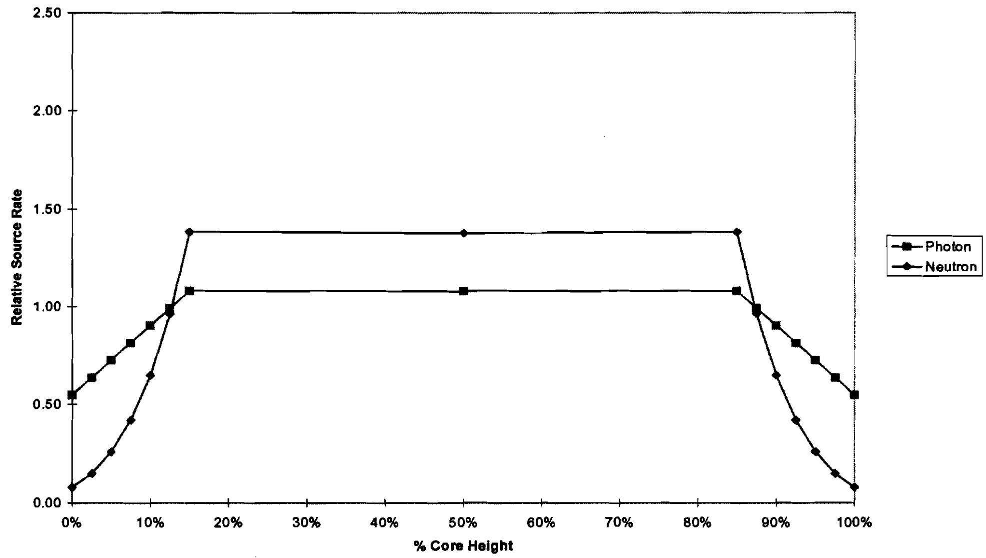


Figure 5.2-4 BWR Photon and Neutron Axial Source Profiles

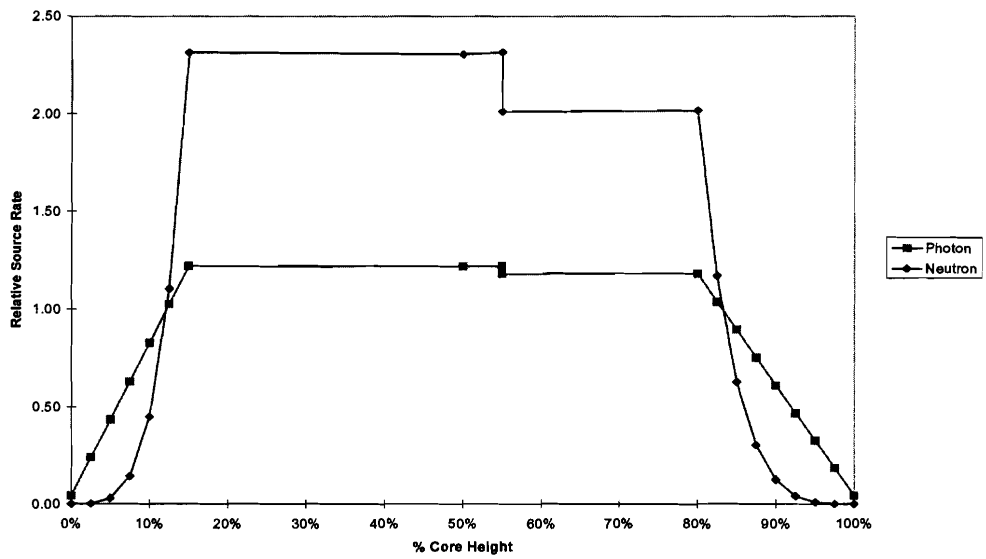
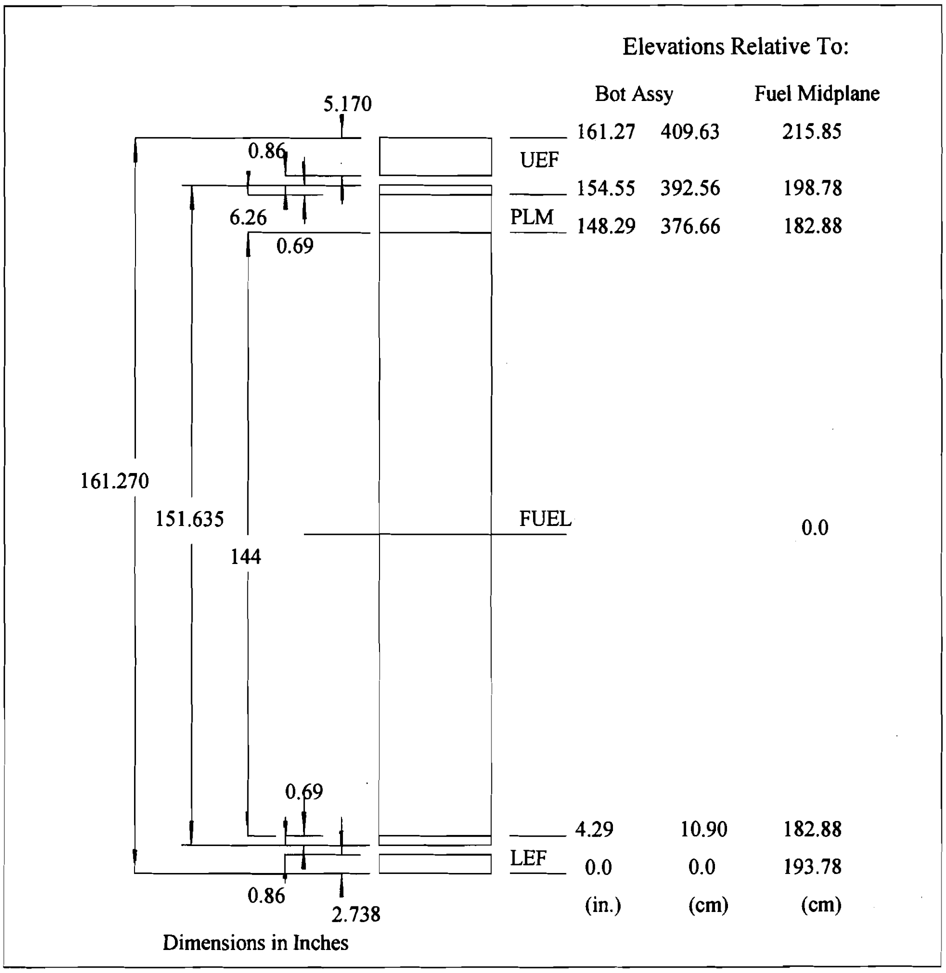


Figure 5.2-5 WE 17×17 Assembly Geometrical Parameters



UEF = Upper End Fitting Region
LEF = Lower End Fitting Region
PLM = Plenum Region

Figure 5.2-6 GE 9×9-2L Assembly Geometrical Parameters

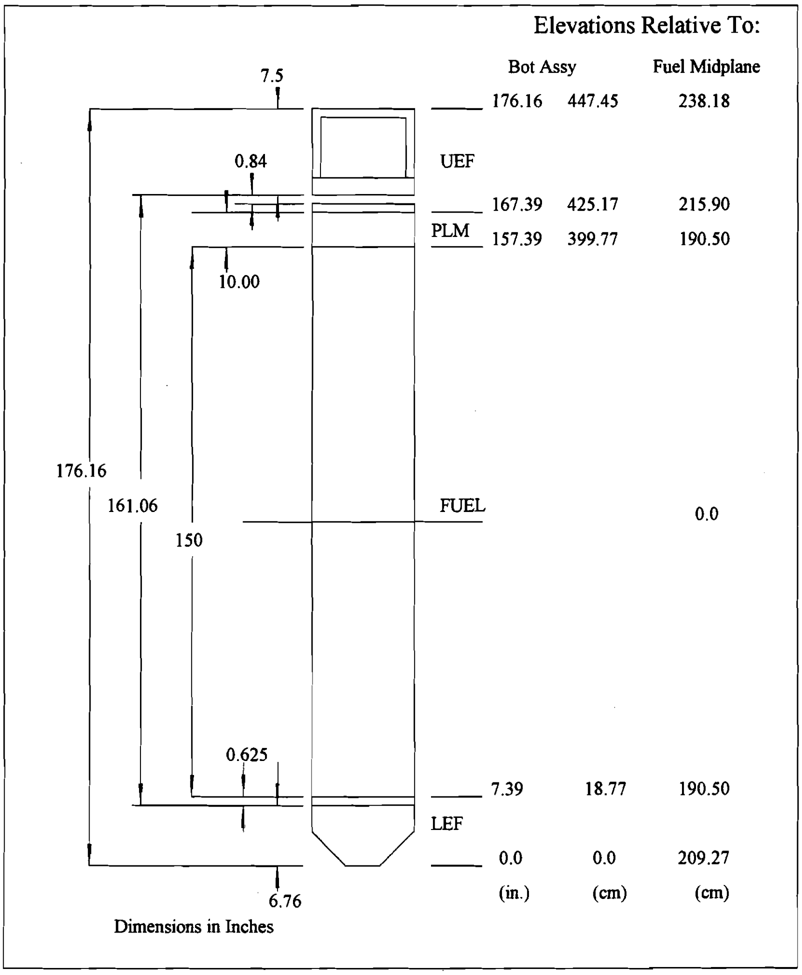


Table 5.2-1 Description of Design Basis Fuel Assembly Types

Fuel Type	Class	Description
WE 15×15	1	Westinghouse 15×15
WE 17×17	1	Westinghouse 17×17
BW 15×15	2	Babcock & Wilcox 15×15
CE 16×16	3	Combustion Engineering 16×16
GE 7×7S	4	General Electric 7×7 BWR/2-3 Reactor Type
GE 8×8-2S	4	General Electric 8×8 BWR/2-3 Reactor Type, 2 water holes
GE 8×8-4S	4	General Electric 8×8 BWR/2-3 Reactor Type, 1 water hole
GE 7×7L	5	General Electric 7×7 BWR/4-6 Reactor Type
GE 8×8-2L	5	General Electric 8×8 BWR/4-6 Reactor Type, 2 water holes
GE 8×8-4L	5	General Electric 8×8 BWR/4-6 Reactor Type, 1 water hole
GE 9×9-2L	5	General Electric 9×9 BWR/4-6 Reactor Type, 2 water holes, 79 fuel rods

Table 5.2-2 Representative Design Basis PWR Fuel Assembly Physical Characteristics

Fuel Parameter	WE 15×15 Std (Class 1)	WE 17×17 Std (Class 1)	BW 15×15 Mark B (Class 2)	CE 16×16 System 80 (Class 3)
Assembly Data				
Rod array	15×15	17×17	15×15	16×16
Assembly length, in(1)	161.27	161.27	170.75	178.25
Assembly width, in	8.43	8.43	8.54	8.10
Active fuel length, in	144	144	144	150
Max U loading, kg	464.6	467.1	480.7	441.7
Assy power level, MW	16.28	18.48	16.49	16.59
Fuel temperature, K	900	900	900	900
Clad temperature, K	620	620	620	620
Moderator temperature, K	580	580	580	580
Fuel Rod Data				
No. of fuel rods	204	264	208	236
Rod pitch, in	0.563	0.496	0.568	0.506
Rod diameter, in	0.422	0.374	0.430	0.382
Cladding material	Zirc–4	Zirc–4	Zirc–4	Zirc–4
Cladding thickness, in	0.0242	0.0225	0.0265	0.0250
Pellet diameter, in	0.3659	0.3225	0.3686	0.3250
Init. Enrich, wt %	3.7	3.7	3.7	3.7
Guide Tube Data				
No. tubes	16	24	16	5
Tube diameter, in.	0.545	0.482	0.530	0.98
Tube thickness, in.	0.015	0.016	0.016	0.035
Tube material	Zirc	Zirc	Zirc	Zirc
Instrument Tube Data				
No. tubes	1	1	1	0
Tube diameter, in.	0.545	0.482	0.493	–
Tube thickness, in.	0.015	0.016	0.026	–
Tube material	Zirc	Zirc	Zirc	–

1. Fuel assembly length including burnable absorber rods or thimble plugs.

Table 5.2-3 Representative Design Basis PWR Fuel Assembly Hardware Data Per Assembly

Assembly Region	WE 15×15 Std (Class 1)	WE 17×17 Std (Class 1)	BW 15×15 Mark B (Class 2)	CE 16×16 System 80 (Class 3)
	Material Mass [kg/assembly]			
Upper End Fitting	Inconel/SS 11.80	Inconel/SS 7.85	Inconel/SS 10.76	Inconel/SS 15.90
Lower End Fitting	Inconel/SS 5.44	Inconel/SS 5.90	Inconel/SS 8.31	Inconel/SS 7.30
Upper End Fitting BP / Thimble Plug	SS 2.47	SS 2.95	SS 3.64	–
Upper Plenum Springs	Inconel 4.07	Inconel 4.43	Inconel 1.98	Inconel 10.70
Upper Plenum Grid	Inconel 1.07	Inconel/SS 0.88	Zirc 1.04	Zirc 0.82
Upper Plenum BP / Thimble Plug	SS 3.16	SS 3.16	SS 3.41	–
Lower Plenum Springs	–	–	Inconel 1.98	–
Lower Plenum Grid	–	–	Zirc 1.3	–
Incore Grid	Inconel/SS 8.06	Inconel/SS 5.44	Inconel 4.9	Zirc 7.35
Guide Tubes	Zirc 9.39	Zirc 9.53	Zirc 8.64	Zirc 11.3
Incore Burnable Poison (BP)	SS 11.39	SS 11.00	–	–

Table 5.2-4 Nuclear Parameters of Design Basis PWR Fuel Assemblies with 3.7 wt % ^{235}U Enrichment, 40,000 MWD/MTU Burnup, 5-Year Cooling Time

Assembly	Neutron Source [n/s]	Gamma Source [γ/s]	Hardware Source [γ/kg/s]
WE 15×15	1.985E+08	5.870E+15	6.919E+12
WE 17×17	1.984E+08	5.946E+15	7.005E+12
BW 15×15	1.961E+08	5.825E+15	6.925E+12
CE 16×16	1.872E+08	5.603E+15	6.951E+12

Table 5.2-5 Design Basis PWR Fuel Assembly Activated Hardware Comparison [γ/s], 5-Year Cooling Time

Fuel Type	Lower End-Fitting	Lower Plenum	Fuel Hardware	Upper Plenum	Upper End-Fitting
WE 15×15	7.528E+12	–	1.346E+14	1.149E+13	9.874E+12
WE 17×17	8.266E+12	–	1.151E+14	1.187E+13	7.565E+12
BW 15×15	5.755E+12	2.742E+12	3.393E+13	7.785E+12	9.813E+12
CE 16×16	1.015E+13	–	0.000E+00	1.488E+13	5.839E+12

Table 5.2-6 Representative Design Basis BWR Fuel Physical Characteristics

Assembly	**GE 7×7**		**GE 8×8-2**		**GE 8×8-4**		**GE 9×9-2**
BWR Reactor	2–3	4–6	2–3	4–6	2–3	4–6	4–6
Canister Class	4	5	4	5	4	5	5
Assembly Version	GE-2b	GE-2	GE-5	GE-5	GE-10	GE-10	GE-11
Assembly Data							
Assembly length, in.(2)	171.3	176.2	171.3	176.2	171.3	176.2	176.2
Assembly width, in.	5.44	5.44	5.44	5.44	5.44	5.44	5.44
Active fuel length, in.	144	144	144(1)	150	144(1)	150	150
Max U loading, kg	197.7	197.7	177.3(1)	184.7	171.7(1)	178.7	197.9
Assembly power, MW	3.85	4.95	3.85	4.95	3.85	4.95	4.95
Fuel temperature, K	840	840	840	840	840	840	840
Clad temperature, K	620	620	620	620	620	620	620
Moderator void frac	0.4	0.4	0.4	0.4	0.4	0.4	0.4
Fuel Rod Data							
No. fuel rods	49	49	62	62	60	60	79
Rod pitch, in.	0.738	0.738	0.640	0.640	0.640	0.640	0.567
Rod diameter, in.	0.563	0.563	0.483	0.483	0.484	0.484	0.441
Cladding material	Zirc–2	Zirc–2	Zirc–2	Zirc–2	Zirc–4	Zirc–4	Zirc–4
Cladding thick, in.	0.032	0.032	0.032	0.032	0.032	0.032	0.028
Pellet diameter, in.	0.487	0.487	0.410	0.410	0.410	0.410	0.376

(1) Active fuel length normalized to 144 inches.

(2) Modeled assembly length standardized to 171.3 inches for BWR/2-3 fuel and 176.2 inches for BWR/4-6 fuel.

Table 5.2-7 Representative Design Basis BWR Fuel Assembly Hardware Data

Array	Reactor Type	Upper End Mass [kg]	Lower End Mass [kg]	Plenum Spring Mass [kg]	Incore Grid Mass [kg]	
					Zirc	Inconel
GE 7×7S	BWR/2–3	2.05	4.36	1.85	1.70	0.32
GE 7×7L	BWR/4–6	2.05	4.36	1.85	1.70	0.32
GE 8×8-2S	BWR/2–3	2.1	4.83	2.0	2.20	0.29
GE 8×8-2L	BWR/4–6	2.1	4.83	2.0	2.20	0.29
GE 8×8-4S	BWR/2–3	2.56	4.75	1.3	2.20	0.29
GE 8×8-4L	BWR/4–6	2.56	4.75	1.3	2.20	0.29
GE 9×9-2L	BWR/4–6	2.08	4.74	1.68	2.50	0.12

Table 5.2-8 Nuclear and Thermal Parameters of Design Basis BWR Fuel with 3.25 wt % ^{235}U Enrichment, 40,000 MWD/MTU Burnup, and 5-Year Cooling Time

Assembly	Reactor	Neutron Source [n/s]	Gamma Source [γ/s]	Hardware Source [γ/kg/s]
GE 7×7S	BWR/2–3	1.045E+08	2.227E+15	7.011E+12
GE 7×7L	BWR/4–6	1.055E+08	2.354E+15	7.518E+12
GE 8×8-2S	BWR/2–3	8.595E+07	2.029E+15	7.298E+12
GE 8×8-2L	BWR/4–6	9.016E+07	2.209E+15	7.698E+12
GE 8×8-4S	BWR/2–3	8.115E+07	1.974E+15	7.440E+12
GE 8×8-4L	BWR/4–6	8.474E+07	2.146E+15	7.824E+12
GE 9×9-2L	BWR/4–6	1.028E+08	2.347E+15	7.450E+12

Table 5.2-9 Design Basis BWR Fuel Assembly Activated Hardware Comparison [γ/s] at 40,000 MWD/MTU Burnup, 5-Year Cooling Time

Fuel Type	Reactor	Lower End-Fitting	Fuel Hardware	Upper Plenum	Upper End-Fitting
GE 7×7S	BWR/2–3	4.585E+12	2.243E+12	2.594E+12	1.437E+12
GE 7×7L	BWR/4–6	4.917E+12	2.406E+12	2.782E+12	1.541E+12
GE 8×8-2S	BWR/2–3	5.288E+12	2.117E+12	2.919E+12	1.533E+12
GE 8×8-2L	BWR/4–6	5.577E+12	2.232E+12	3.079E+12	1.616E+12
GE 8×8-4S	BWR/2–3	5.301E+12	2.158E+12	1.934E+12	1.905E+12
GE 8×8-4L	BWR/4–6	5.575E+12	2.269E+12	2.034E+12	2.003E+12
GE 9×9-2L	BWR/4–6	5.297E+12	8.940E+11	2.503E+12	1.550E+12

Table 5.2-10 Standard Transfer Cask One-Dimensional Top Axial Dose Rate Results Relative to PWR Design Basis

Fuel	Shield Lid Wet	Structural Lid Dry	Weld Shield Wet
WE 17×17	1.00	1.00	1.00
WE 15×15	0.85	0.89	0.85
CE 16×16	0.46	0.57	0.45
BW 15×15	0.74	0.78	0.78

Table 5.2-11 Standard Transfer Cask One-Dimensional Radial Dose Rate Results Relative to PWR Design Basis

Fuel	Dry Condition	Wet Condition
WE 17×17	1.00	1.00
WE 15×15	0.99	0.98
CE 16×16	0.69	0.59
BW 15×15	0.73	0.68

Table 5.2-12 Standard Transfer Cask One-Dimensional Bottom Axial Dose Rate Results Relative to PWR Design Basis

Fuel	Dry Condition	Wet Condition
WE 17×17	1.00	1.00
WE 15×15	0.96	0.95
CE 16×16	1.00	1.00
BW 15×15	0.63	0.59

Table 5.2-13 Standard Transfer Cask One-Dimensional Top Axial Dose Rate Results Relative to BWR Design Basis

Fuel	Shield Lid Wet	Structural Lid Dry	Weld Shield Wet
GE 9×9-2L	1.00	1.00	1.00
GE 8×8-4L	0.91	0.85	0.90
GE 8×8-4S	0.84	0.80	0.83
GE 8×8-2L	0.91	0.90	0.91
GE 8×8-2S	0.83	0.85	0.82
GE 7×7L	0.76	0.85	0.74
GE 7×7S	0.79	0.94	0.78

Table 5.2-14 Standard Transfer Cask One-Dimensional Radial Dose Rate Results Relative to BWR Design Basis

Fuel	Dry Condition	Wet Condition
GE 9×9-2L	1.00	1.00
GE 8×8-4L	0.90	0.94
GE 8×8-4S	0.85	0.86
GE 8×8-2L	0.95	0.97
GE 8×8-2S	0.90	0.89
GE 7×7L	1.04	1.03
GE 7×7S	0.98	0.94

Table 5.2-15 Standard Transfer Cask One-Dimensional Bottom Axial Dose Rate Results Relative to BWR Design Basis

Fuel	Dry Condition	Wet Condition
GE 9×9-2L	1.00	1.00
GE 8×8-4L	0.97	1.00
GE 8×8-4S	0.93	0.95
GE 8×8-2L	0.98	0.99
GE 8×8-2S	0.93	0.94
GE 7×7L	0.95	0.91
GE 7×7S	0.89	0.85

Table 5.2-16 Design Basis PWR 5-Year Fuel Neutron Source Spectrum

Group	E_{low} [MeV]	E_{high} [MeV]	Spectrum [n/sec/assy]
1	6.43E+00	2.00E+01	3.661E+06
2	3.00E+00	6.43E+00	4.163E+07
3	1.85E+00	3.00E+00	4.615E+07
4	1.40E+00	1.85E+00	2.598E+07
5	9.00E-01	1.40E+00	3.514E+07
6	4.00E-01	9.00E-01	3.832E+07
7	1.00E-01	4.00E-01	7.500E+06
8	1.70E-02	1.00E-01	0.000E+00
9	3.00E-03	1.70E-02	0.000E+00
10	5.50E-04	3.00E-03	0.000E+00
11	1.00E-04	5.50E-04	0.000E+00
12	3.00E-05	1.00E-04	0.000E+00
13	1.00E-05	3.00E-05	0.000E+00
14	3.05E-06	1.00E-05	0.000E+00
15	1.77E-06	3.05E-06	0.000E+00
16	1.30E-06	1.77E-06	0.000E+00
17	1.13E-06	1.30E-06	0.000E+00
18	1.00E-06	1.13E-06	0.000E+00
19	8.00E-07	1.00E-06	0.000E+00
20	4.00E-07	8.00E-07	0.000E+00
21	3.25E-07	4.00E-07	0.000E+00
22	2.25E-07	3.25E-07	0.000E+00
23	1.00E-07	2.25E-07	0.000E+00
24	5.00E-08	1.00E-07	0.000E+00
25	3.00E-08	5.00E-08	0.000E+00
26	1.00E-08	3.00E-08	0.000E+00
27	1.00E-11	1.00E-08	0.000E+00
Total			**1.984E+08**

Table 5.2-17 Design Basis PWR 5-Year Fuel Photon Spectrum

Group	E_{low} [MeV]	E_{high} [MeV]	Spectrum [γ/sec/assy]	Spectrum [MeV/sec/assy]
1	8.00E+00	1.00E+01	1.1222E+05	1.0100E+06
2	6.50E+00	8.00E+00	5.2856E+05	3.8321E+06
3	5.00E+00	6.50E+00	2.6947E+06	1.5495E+07
4	4.00E+00	5.00E+00	6.7148E+06	3.0217E+07
5	3.00E+00	4.00E+00	1.0245E+10	3.5858E+10
6	2.50E+00	3.00E+00	8.2534E+10	2.2697E+11
7	2.00E+00	2.50E+00	2.6257E+12	5.9078E+12
8	1.66E+00	2.00E+00	1.1070E+12	2.0258E+12
9	1.33E+00	1.66E+00	2.5755E+13	3.8504E+13
10	1.00E+00	1.33E+00	1.1513E+14	1.3413E+14
11	8.00E-01	1.00E+00	3.2879E+14	2.9591E+14
12	6.00E-01	8.00E-01	2.3388E+15	1.6372E+15
13	4.00E-01	6.00E-01	7.2421E+14	3.6211E+14
14	3.00E-01	4.00E-01	6.6148E+13	2.3152E+13
15	2.00E-01	3.00E-01	9.8414E+13	2.4604E+13
16	1.00E-01	2.00E-01	3.6058E+14	5.4087E+13
17	5.00E-02	1.00E-01	4.2971E+14	3.2228E+13
18	1.00E-02	5.00E-02	1.4544E+15	4.3632E+13
Total			**5.9458E+15**	**2.6537E+15**

Table 5.2-18 Design Basis PWR 5-Year Hardware Photon Spectrum

Group	E_{low} [MeV]	E_{high} [MeV]	Spectrum [γ/sec/kg]	Spectrum [MeV/sec/kg]
1	8.00E+00	1.00E+01	0.0000E+00	0.0000E+00
2	6.50E+00	8.00E+00	0.0000E+00	0.0000E+00
3	5.00E+00	6.50E+00	0.0000E+00	0.0000E+00
4	4.00E+00	5.00E+00	0.0000E+00	0.0000E+00
5	3.00E+00	4.00E+00	1.4997E-15	5.2490E-15
6	2.50E+00	3.00E+00	5.5445E+04	1.5247E+05
7	2.00E+00	2.50E+00	3.5757E+07	8.0453E+07
8	1.66E+00	2.00E+00	4.1887E+02	7.6653E+02
9	1.33E+00	1.66E+00	1.5067E+12	2.2525E+12
10	1.00E+00	1.33E+00	5.3355E+12	6.2159E+12
11	8.00E-01	1.00E+00	4.7463E+10	4.2717E+10
12	6.00E-01	8.00E-01	6.3039E+06	4.4127E+06
13	4.00E-01	6.00E-01	1.8179E+07	9.0895E+06
14	3.00E-01	4.00E-01	2.8721E+08	1.0052E+08
15	2.00E-01	3.00E-01	2.1890E+08	5.4725E+07
16	1.00E-01	2.00E-01	4.4085E+09	6.6128E+08
17	5.00E-02	1.00E-01	1.8273E+10	1.3705E+09
18	1.00E-02	5.00E-02	9.1992E+10	2.7598E+09
Total			**7.0049E+12**	**8.5161E+12**

Table 5.2-19 Design Basis BWR 5-Year Fuel Neutron Source Spectrum

Group	E_{low} [MeV]	E_{high} [MeV]	Spectrum [n/sec/assy]
1	6.43E+00	2.00E+01	1.902E+06
2	3.00E+00	6.43E+00	2.158E+07
3	1.85E+00	3.00E+00	2.384E+07
4	1.40E+00	1.85E+00	1.346E+07
5	9.00E-01	1.40E+00	1.823E+07
6	4.00E-01	9.00E-01	1.990E+07
7	1.00E-01	4.00E-01	3.895E+06
8	1.70E-02	1.00E-01	0.000E+00
9	3.00E-03	1.70E-02	0.000E+00
10	5.50E-04	3.00E-03	0.000E+00
11	1.00E-04	5.50E-04	0.000E+00
12	3.00E-05	1.00E-04	0.000E+00
13	1.00E-05	3.00E-05	0.000E+00
14	3.05E-06	1.00E-05	0.000E+00
15	1.77E-06	3.05E-06	0.000E+00
16	1.30E-06	1.77E-06	0.000E+00
17	1.13E-06	1.30E-06	0.000E+00
18	1.00E-06	1.13E-06	0.000E+00
19	8.00E-07	1.00E-06	0.000E+00
20	4.00E-07	8.00E-07	0.000E+00
21	3.25E-07	4.00E-07	0.000E+00
22	2.25E-07	3.25E-07	0.000E+00
23	1.00E-07	2.25E-07	0.000E+00
24	5.00E-08	1.00E-07	0.000E+00
25	3.00E-08	5.00E-08	0.000E+00
26	1.00E-08	3.00E-08	0.000E+00
27	1.00E-11	1.00E-08	0.000E+00
Total			**1.028E+08**

Table 5.2-20 Design Basis BWR 5-Year Fuel Photon Spectrum

Group	E_{low} [MeV]	E_{high} [MeV]	Spectrum [γ/sec/assy]	Spectrum [MeV/sec/assy]
1	8.00E+00	1.00E+01	5.8267E+04	5.2440E+05
2	6.50E+00	8.00E+00	2.7444E+05	1.9897E+06
3	5.00E+00	6.50E+00	1.3991E+06	8.0448E+06
4	4.00E+00	5.00E+00	3.4861E+06	1.5687E+07
5	3.00E+00	4.00E+00	3.5189E+09	1.2316E+10
6	2.50E+00	3.00E+00	2.8233E+10	7.7641E+10
7	2.00E+00	2.50E+00	7.8700E+11	1.7708E+12
8	1.66E+00	2.00E+00	3.7821E+11	6.9212E+11
9	1.33E+00	1.66E+00	9.9908E+12	1.4936E+13
10	1.00E+00	1.33E+00	4.8604E+13	5.6624E+13
11	8.00E-01	1.00E+00	1.2924E+14	1.1632E+14
12	6.00E-01	8.00E-01	9.5614E+14	6.6930E+14
13	4.00E-01	6.00E-01	2.7603E+14	1.3802E+14
14	3.00E-01	4.00E-01	2.4650E+13	8.6275E+12
15	2.00E-01	3.00E-01	3.7603E+13	9.4008E+12
16	1.00E-01	2.00E-01	1.3759E+14	2.0639E+13
17	5.00E-02	1.00E-01	1.6347E+14	1.2260E+13
18	1.00E-02	5.00E-02	5.6276E+14	1.6883E+13
Total			**2.3473E+15**	**1.0656E+15**

Table 5.2-21 Design Basis BWR 5-Year Hardware Photon Spectrum

Group	E_{low} [MeV]	E_{high} [MeV]	Spectrum [γ/sec/kg]	Spectrum [MeV/sec/kg]
1	8.00E+00	1.00E+01	0.0000E+00	0.0000E+00
2	6.50E+00	8.00E+00	0.0000E+00	0.0000E+00
3	5.00E+00	6.50E+00	0.0000E+00	0.0000E+00
4	4.00E+00	5.00E+00	0.0000E+00	0.0000E+00
5	3.00E+00	4.00E+00	1.2594E-15	4.4079E-15
6	2.50E+00	3.00E+00	5.9094E+04	1.6251E+05
7	2.00E+00	2.50E+00	3.8110E+07	8.5748E+07
8	1.66E+00	2.00E+00	3.2071E+02	5.8690E+02
9	1.33E+00	1.66E+00	1.6059E+12	2.4008E+12
10	1.00E+00	1.33E+00	5.6866E+12	6.6249E+12
11	8.00E-01	1.00E+00	3.4375E+10	3.0938E+10
12	6.00E-01	8.00E-01	6.7188E+06	4.7032E+06
13	4.00E-01	6.00E-01	1.9368E+07	9.6840E+06
14	3.00E-01	4.00E-01	3.0611E+08	1.0714E+08
15	2.00E-01	3.00E-01	2.3331E+08	5.8328E+07
16	1.00E-01	2.00E-01	4.6987E+09	7.0481E+08
17	5.00E-02	1.00E-01	1.9476E+10	1.4607E+09
18	1.00E-02	5.00E-02	9.8098E+10	2.9429E+09
Total			**7.4498E+12**	**9.0620E+12**

Table 5.2-22 Source Rate Versus Burnup Fit Parameters

Radiation Type	Exponent, *b*
Neutron	4.22
Photon	1.00

Table 5.2-23 SAS4 SCALE Factors Applied to Neutron Source Rate at Average Burnup

Fuel Type	Scale Factor
PWR	1.125
BWR	1.582

Table 5.2-24 Additional SCALE Factors Applied to Region Source Rates for SAS4 Analysis

Source Region	WE 17×17	GE 9x9-2L
Upper End Fitting	1.345	1.293
Upper Plenum	1.000	1.000
Top Fuel Neutron	1.000	0.887
Bot Fuel Neutron	1.000	1.113
Top Fuel Gamma	1.000	0.957
Bot Fuel Gamma	1.000	1.043
Fuel Hardware	1.169	2.538
Lower End Fitting	1.000	1.052

Table 5.2-25 PWR Axial Source Profile

% Core Height	Burnup Profile	Photon Source	Neutron Source
0.00%	0.5470	0.5470	7.840E-02
2.50%	0.6358	0.6358	1.479E-01
5.00%	0.7247	0.7247	2.569E-01
7.50%	0.8135	0.8135	4.185E-01
10.00%	0.9023	0.9023	6.481E-01
12.50%	0.9912	0.9912	9.633E-01
15.00%	1.0800	1.0800	1.384E+00
50.00%	1.0790	1.0790	1.378E+00
85.00%	1.0800	1.0800	1.384E+00
87.50%	0.9912	0.9912	9.633E-01
90.00%	0.9023	0.9023	6.481E-01
92.50%	0.8135	0.8135	4.185E-01
95.00%	0.7247	0.7247	2.569E-01
97.50%	0.6358	0.6358	1.479E-01
100.00%	0.5470	0.5470	7.840E-02

Table 5.2-26 BWR Axial Source Rate Profile

% Core Height	Burnup Profile	Photon Source	Neutron Source
0.00%	0.0430	0.0430	1.711E-06
2.50%	0.2392	0.2392	2.388E-03
5.00%	0.4353	0.4353	2.991E-02
7.50%	0.6315	0.6315	1.437E-01
10.00%	0.8277	0.8277	4.501E-01
12.50%	1.0238	1.0238	1.105E+00
15.00%	1.2200	1.2200	2.314E+00
50.00%	1.2190	1.2190	2.306E+00
55.00%	1.2200	1.2200	2.314E+00
55.01%	1.1800	1.1800	2.011E+00
80.00%	1.1810	1.1810	2.018E+00
82.50%	1.0379	1.0379	1.170E+00
85.00%	0.8958	0.8958	6.284E-01
87.50%	0.7536	0.7536	3.031E-01
90.00%	0.6115	0.6115	1.255E-01
92.50%	0.4694	0.4694	4.110E-02
95.00%	0.3272	0.3272	8.970E-03
97.50%	0.1851	0.1851	8.104E-04
100.00%	0.0430	0.0430	1.711E-06

Table 5.2-27 MCBEND Three-Dimensional Design Basis Fuel Assembly Descriptions

Parameter	WE17×17	GE9×9-2L
Fuel Rod Height [cm]	385.14	407.80
Top End-Cap Height [cm]	1.74	0.88
Bottom End-Cap Height [cm]	1.74	1.59
Active Fuel Region Height [cm]	365.76	381.00
Fuel Rod Diameter [cm]	0.95	1.12
Fuel Clad Thickness [cm]	0.06	0.07
Fuel Pellet Diameter [cm]	0.82	0.96
Fuel Rod Pitch [cm]	1.26	1.44
Number of Water Rods	----	2
Water Rod OD [cm]	----	1.12
Water Rod Thickness [cm]	----	0.07
Channel Inner Dimension [cm]	----	13.41
Channel Thickness [cm]	----	0.20
Number of Guide Tubes	24	----
Guide Tube OD [cm]	1.22	----
Guide Tube Thickness [cm]	0.04	----
Number of Instrument Tubes	1	----
Instrument Tube OD [cm]	1.22	----
Instrument Tube Thickness [cm]	0.04	----
Fuel Assembly Height [cm]	405.89	447.45
Fuel Assembly Width [cm]	21.40	14.02
Lower Nozzle Height [cm]	6.86	17.17
Upper Nozzle Height [cm]	9.32	19.05
Gap Fuel Rod to Top Nozzle [cm]	4.57	3.43
Upper Plenum Region Height [cm]	15.90	24.33
Number of Fuel Rods	264	79
Calculated MTU [MTU]	0.4671	0.1979

Table 5.2-28 MCBEND Standard 28 Group Neutron Boundaries

Group	E Lower [MeV]	E Upper [MeV]	E Average [MeV]
1	1.360E+01	1.460E+01	1.410E+01
2	1.250E+01	1.360E+01	1.305E+01
3	1.125E+01	1.250E+01	1.188E+01
4	1.000E+01	1.125E+01	1.063E+01
5	8.250E+00	1.000E+01	9.125E+00
6	7.000E+00	8.250E+00	7.625E+00
7	6.070E+00	7.000E+00	6.535E+00
8	4.720E+00	6.070E+00	5.395E+00
9	3.680E+00	4.720E+00	4.200E+00
10	2.870E+00	3.680E+00	3.275E+00
11	1.740E+00	2.870E+00	2.305E+00
12	6.400E-01	1.740E+00	1.190E+00
13	3.900E-01	6.400E-01	5.150E-01
14	1.100E-01	3.900E-01	2.500E-01
15	6.740E-02	1.100E-01	8.870E-02
16	2.480E-02	6.740E-02	4.610E-02
17	9.120E-03	2.480E-02	1.696E-02
18	2.950E-03	9.120E-03	6.035E-03
19	9.610E-04	2.950E-03	1.956E-03
20	3.540E-04	9.610E-04	6.575E-04
21	1.660E-04	3.540E-04	2.600E-04
22	4.810E-05	1.660E-04	1.071E-04
23	1.600E-05	4.810E-05	3.205E-05
24	4.000E-06	1.600E-05	1.000E-05
25	1.500E-06	4.000E-06	2.750E-06
26	5.500E-07	1.500E-06	1.025E-06
27	7.090E-08	5.500E-07	3.105E-07
28	1.000E-11	7.090E-08	3.546E-08

Table 5.2-29 MCBEND Standard 22 Group Gamma Boundaries

Group	E Lower [MeV]	E Upper [MeV]	E Average [MeV]
1	1.200E+01	1.400E+01	1.300E+01
2	1.000E+01	1.200E+01	1.100E+01
3	8.000E+00	1.000E+01	9.000E+00
4	6.500E+00	8.000E+00	7.250E+00
5	5.000E+00	6.500E+00	5.750E+00
6	4.000E+00	5.000E+00	4.500E+00
7	3.000E+00	4.000E+00	3.500E+00
8	2.500E+00	3.000E+00	2.750E+00
9	2.000E+00	2.500E+00	2.250E+00
10	1.660E+00	2.000E+00	1.830E+00
11	1.440E+00	1.660E+00	1.550E+00
12	1.220E+00	1.440E+00	1.330E+00
13	1.000E+00	1.220E+00	1.110E+00
14	8.000E-01	1.000E+00	9.000E-01
15	6.000E-01	8.000E-01	7.000E-01
16	4.000E-01	6.000E-01	5.000E-01
17	3.000E-01	4.000E-01	3.500E-01
18	2.000E-01	3.000E-01	2.500E-01
19	1.000E-01	2.000E-01	1.500E-01
20	5.000E-02	1.000E-01	7.500E-02
21	2.000E-02	5.000E-02	3.500E-02
22	1.000E-02	2.000E-02	1.500E-02

Table 5.2-30 MCBEND Fuel Assembly Hardware Mass and Flux Factors by Source Region

WE17×17		
Region	**Act. Mass**	**Flux**
	[kg/assy]	**Factor**
Lower Nozzle	5.90	0.20
Fuel	19.21	1.00
Upper Plenum	8.47	0.20
Upper Nozzle	14.53	0.10
GE9×9-2L		
Region	**Act. Mass**	**Flux**
	[kg/assy]	**Factor**
Lower Nozzle	4.99	0.15
Fuel	0.30	1.00
Upper Plenum	1.68	0.20
Upper Nozzle	2.69	0.10

5.3 Model Specification

The transfer cask and storage cask are evaluated using one-dimensional SAS1 and three-dimensional SAS4 models. The storage cask air inlets and outlets are evaluated using the three-dimensional MCBEND Monte Carlo transport code.

SCALE Package Model Specification

Both one-dimensional SAS1 and three-dimensional SAS4 models are used in the shielding evaluations of the Universal Storage System. The SAS1 radial and axial model results are used to determine the bounding design basis fuel assembly descriptions for subsequent use in detailed three-dimensional analyses. The one-dimensional models represent the casks as either semi-infinite cylinders or slabs. The method of solution uses the XSDRNPM [8] discrete ordinates code and the XSDOSE [14] flux-at-a-point estimation code. Bucklings are applied to the SAS1 models to account for transverse leakage. The one-dimensional analysis also serves as a cross-check to the more complex three-dimensional model results.

The SAS4 three-dimensional shielding models are used to estimate the dose profiles at the surfaces of the cask and at potential streaming paths such as the canister vent and drain ports. The method of solution is Monte Carlo [3] with an adjoint discrete ordinates biasing technique using the XSDRNPM and MORSE codes. The adjoint biasing performed by XSDRNPM is a one-dimensional solution, which may not generate optimal importance maps for geometries which differ from the user-supplied bias map. For example, an importance map optimized for particle acceleration at the fuel midplane elevation yields an unoptimized radial importance map at the concrete cask air inlets. This phenomenon has yielded non-converging dose rate results at the inlets, due to high weight, low probability particles passing the detector surface. In order to more effectively estimate air inlet dose rates, the MCBEND code is employed, which has as one of its features the option of a user-supplied three-dimensional importance map. In order to present a consistent set of results, the air outlets are also analyzed using MCBEND.

Since SAS4 requires model symmetry at the fuel midplane, two models are created for each cask, a top and a bottom model. Radial biasing is performed to estimate dose rates on the sides of the cask, and axial biasing is performed to estimate dose rates on the top and bottom surfaces of the cask. Modifications are made to SAS4 to tally dose rates along the radial, top, and bottom surfaces of the cask as well as any cylindrical surface surrounding the cask. Thus, detailed dose rate profiles are determined that explicitly show peaks due to the fuel burnup profile, activated hardware gamma emission, and streaming paths.

In both SAS1 and SAS4 models, the fuel and hardware source regions are homogenized within the volumes defined by the periphery of the basket tubes and the respective elevations of the active fuel region, the plenum, and the end fittings. Within these volumes, the material masses of the fuel assembly and basket are homogenized. The resulting material and nuclide densities for both the PWR and the BWR cases are shown in Table 5.3-1 through Table 5.3-5. In all models, the cask and canister shield thicknesses and axial extents are explicitly represented.

Furthermore, in the three-dimensional models, the homogenized fuel region is represented geometrically by a shape approximating the periphery of the fuel assembly bundle within the basket.

Both the SAS1 and SAS4 models utilize fuel midplane symmetry. Thus, all shielding models are developed with respect to the fuel midplane as the origin. This symmetry is required in the SAS4 models due to the automated biasing techniques employed.

The axial source profile is considered in establishing total fuel region source rates for the top and bottom models. In the BWR case, due to the asymmetry in source profile about the fuel midplane, a greater fraction of the total fuel neutron and gamma source is emitted in the bottom half of the fuel. The three-dimensional shielding model therefore represents a higher total source rate in the bottom model. The relative fractions of source in each model region are given in Table 5.2-24. Since the PWR source profile is symmetric about the fuel axial midplane, the corresponding relative source fractions are 1.0.

MCBEND Model Description

Three-dimensional MCBEND models are constructed to analyze the concrete cask air inlets and outlets. Detailed models are constructed of the fuel assemblies, basket, and cask shield configurations, including streaming paths.

The MCBEND three-dimensional shielding models are used to estimate the dose profiles at the concrete cask inlets and outlets. MCBEND employs an automated biasing technique for the Monte Carlo calculation based on a three-dimensional adjoint diffusion calculation. Mesh cells for the adjoint solution are selected based on half value thicknesses for each material. Radial biasing is performed to estimate dose rates at the storage cask inlets and outlets, with an additional angular biasing component used to capture the azimuthal variation in bulk shielding properties.

In the MCBEND fuel assembly model, the fuel and hardware source regions are homogenized within a volume defined by the fuel assembly width and height. This volume is subdivided axially into lower end fitting, active fuel, upper plenum, and upper end fitting source regions. Within these axial volumes, the material masses of the fuel assembly are homogenized. In all models, the cask and canister shield thicknesses and axial extents are explicitly represented.

The gamma and neutron axial source profiles from Section 5.2 are input directly into MCBEND and require none of the source scaling factors required by SAS4.

5.3.1 Description of Radial and Axial Shielding Configurations

The vertical concrete cask has an interior cavity with a radius of 37.25 inches. Radial shielding consists of a 2.5-inch carbon steel shell surrounded by 28.25 inches of concrete. Gamma shielding is provided by both the carbon steel and concrete, and neutron shielding is provided primarily by the concrete. An additional 0.625 inch of stainless steel is provided by the canister shell for radial gamma shielding. The concrete cask top shielding is comprised of 10 inches of stainless steel from the canister lids, 4.1 inches of carbon steel from the shield plug which encloses 1 inch of NS-4-FR or 1.5 inches of NS-3, and 1.5 inches of carbon steel from the concrete cask lid. The bottom of the cask rests on the concrete pad and is inaccessible. In the case of the concrete cask inlets, some shielding is provided by the cask structural components. These components include 2 inches of carbon steel from the pedestal plate and 1 inch of carbon steel from the cask base plate. There is also 1.75 inches of stainless steel from the canister bottom plate.

The Standard or Advanced transfer cask has an inside radius of 33.875 inches and has a multi-wall radial shield design consisting of 0.75 inch of low alloy steel, 4.00 inches of lead, 2.75 inches of a solid borated polymer (NS-4-FR), and 1.25 inches of low alloy steel in an outer shell. Gamma shielding is provided by the steel and lead layers, and neutron shielding is provided primarily by the NS-4-FR. An additional 0.625 inch of stainless steel gamma shielding is provided by the canister shell. The transfer cask bottom shield design is comprised of carbon steel doors 9 inches thick. The top shielding of the transfer cask is provided by the 7-inch stainless steel shield lid and the 3-inch stainless steel structural lid. In addition, a 5-inch carbon steel temporary shield is used during welding, draining, and drying operations. This temporary shielding is assumed to be removed prior to moving the cask.

5.3.2 SCALE One-Dimensional Radial and Axial Shielding Models

Since the fuel assembly and basket features are not explicitly modeled in one-dimensional analysis, the fuel/basket interior is modeled as a set of homogenized material volumes based on an equivalent cylindrical volume. This volume is defined by the cross-sectional area created by the periphery of the basket tubes and the respective elevations of the fuel, end-fitting, and plenum regions.

5.3.2.1 SCALE One-Dimensional Radial Model

In the one-dimensional model, the canister interior is divided into two homogenized radial regions: a fuel/basket region and a basket/disk region. The fuel region smear has an equivalent radius of 71.695 cm in the PWR model and 73.235 cm in the BWR model. Support disk and heat transfer disk materials which fall within this region are homogenized in the fuel region smear. Basket and support disk materials outside this equivalent radius are homogenized in the annular region outside the fuel. Note that this annular smear is employed in the one-dimensional analysis only. In the three-dimensional analysis, basket support and heat transfer disks in the annular region are modeled explicitly.

The fuel region smear consists of the relevant fuel assembly material and any basket material present within the axial extent of the active fuel region. Basket materials include the steel support disks, aluminum heat transfer disks, top and bottom weldments, fuel tubes, neutron absorber sheets, and neutron absorber cover sheets. Fuel assembly materials include: UO_2, cladding, and spacer grids. The resulting material and nuclide densities are described in Section 5.3.5.

Similarly, homogenized material descriptions of the end-fitting and plenum source regions are determined by considering the mass and composition of fuel assembly and basket materials, which lie within the axial extents of the source region.

The one-dimensional radial models of the concrete cask and transfer cask are based on the cylindrical representation of the fuel/basket source regions (previously described) surrounded by the explicit canister and cask radial shield dimensions. An axial buckling equal to the active fuel height is employed for all radial models.

5.3.2.2 SCALE One-Dimensional Axial Model

The one-dimensional top and bottom axial models of the storage cask and transfer casks are based on a buckled slab representation of the fuel/basket, canister, and concrete cask axial shield regions. As previously stated, the one-dimensional axial model elevations are specified with respect to the active fuel midplane, which is modeled as a reflecting boundary in SAS1. Two axial models are utilized for each cask: one from the active fuel midplane to the top of the cask; and one from the active fuel midplane to the bottom of the cask. For gamma calculations, the transverse buckling parameter is set equal to the fuel region equivalent diameter. For neutron calculations, the buckling parameter is set to the diameter of the cask.

5.3.3 SCALE Three-Dimensional Top and Bottom Shielding Models

SAS4 three-dimensional shielding analysis allows detailed modeling of the fuel assemblies, basket, and cask shield configuration including streaming paths. Some fuel assembly and basket detail is homogenized to simplify model input and improve computational efficiency. Thus, the three-dimensional models maintain the equivalent fuel/basket source volumes developed for the one-dimensional models, but explicitly model the axial extent of the source regions, the basket spacer plates outside the homogenized source region, and the cask body details. In addition, the source region axial cross-section is represented in a volume-conserving rectilinear shape, which approximates the periphery of the fuel assemblies in the basket.

As in the SAS1 models, the fuel and hardware source regions are homogenized within the volumes defined by the periphery of the basket tubes and the various source region elevations. Cask body details include the true axial extent of the cask shield as described by the license drawings in Chapter 1, as well as radiation streaming paths such as the storage cask inlets and outlets and the canister vent and drain ports.

SAS4 requires cask model symmetry at the fuel midplane due to the nature of the automated biasing techniques employed and because dose rate tallies from the symmetric halves of the model are averaged together for computational efficiency. Thus, two SAS4 models are created for each cask, a top and a bottom model. As in the SAS1 models, all three-dimensional shielding models are developed with respect to the fuel axial midplane.

The geometric description of a SAS4 model is based on the MARS combinatorial geometry system embedded in the MORSE code [12]. In this system, bodies such as cylinders and rectangular parallelepipeds, and their logical intersections and unions, are used to describe the extent of material zones.

SAS4 employs an automated biasing technique for the MORSE Monte Carlo calculations based on either a radial or an axial XSDRNPM adjoint calculation. In the case of radial biasing, the adjoint calculation is performed based on a one-dimensional description of the radial shields and corresponding fuel/basket regions. In the case of axial biasing, the adjoint calculation is performed for the top or bottom shields and corresponding axial fuel/basket regions. Radial biasing is employed to improve the Monte Carlo computational efficiency and dose rate statistics on the sides of the cask. Axial biasing is employed to improve Monte Carlo computational efficiency and dose rate statistics on the top or bottom surfaces of the cask. The dose rate profiles resulting from both radial and axial biasing calculations yield a complete dose profile of the entire cask.

MORSE Monte Carlo calculations are performed for each source type present in each source region. This approach entails six separate analyses for the top model, encompassing fuel neutron, fuel gamma, fuel n-gamma (secondary gammas arising from neutron interaction in the shield), fuel hardware, upper plenum, and upper end-fitting gamma sources. The bottom model requires a similar level of detail, although only five source cases are considered since no plenum is included in the lower fuel assembly region. Typically, a total of some 20 to 30 million histories are tracked to yield dose rate profiles for each model. These cases are analyzed for both radial and axial geometries, and for both PWR and BWR design basis fuel descriptions. Furthermore, the standard transfer cask top axial model is analyzed for a number of operational configurations of the top lids and temporary shielding, and for both wet and dry canisters. All told, the storage and transfer cask results presented here are based on a total of 220 distinct SAS4 analyses.

5.3.3.1 SCALE Canister and Basket Model

For a given fuel type, the SAS4 description of the canister and basket elements forms a common submodel employed in all storage cask and standard transfer cask analyses. The key features of the model are the accurate positioning of basket support and heat transfer disks and the inclusion of the vent and drain ports located in the canister shield lid.

The axial elevations of basket support and heat transfer disks are determined by placing the elements accurately with respect to the axial elevation of either the top or bottom end of the canister depending on which SAS4 half model is under consideration. In this way, basket disks are accurately located with respect to important cask features, such as trunnions and shield wall axial extents.

The vent ports in the canister shield lid are modeled as a series of three overlapping concentric cylinders, as shown in Figure 5.3-1 with port cover in place. The vent port cover is also modeled, but may or may not be in place depending on the particular operational condition specified. In the top axial analysis of the transfer cask, the vent port covers are assumed to be installed when the canister is in a dry condition, and removed when the canister is modeled in a wet condition. Port covers are in place in all storage cask top model analyses. The vent port cover is modeled as a solid piece of stainless steel of the dimensions indicated in Figure 5.3-1.

5.3.3.2 SCALE Vertical Concrete Cask Three-Dimensional Models

Three-Dimensional Top Model

The three-dimensional top model of the vertical concrete cask containing design basis fuel assemblies is based on the homogenized representation of the basket, and the following features of the storage cask upper region:

- Heat transfer annulus
- Carbon steel weldment with four cutouts for outlet vents
- Concrete shield with four cutouts for outlet vents
- Four outlet vents including carbon steel lining
- Carbon steel shield plug
- Shield plug neutron shield
- Carbon steel top lid

Detailed model parameters used in creating the three-dimensional top model are taken directly from the license drawings in Chapter 1. Elevations associated with the concrete cask three-dimensional features are established with respect to the active fuel midplane of the fuel assembly for the combinatorial model. The three-dimensional concrete cask top models for the design basis PWR are shown in Figure 5.3-2. The cask dimensions in the BWR model are identical, although the cask model is elongated slightly to allow for the longer active fuel region in the BWR design basis assembly.

A detailed sketch of a cross-section of the air outlet model is shown in Figure 5.3-3. The outlet channel walls are modeled as carbon steel.

Three-Dimensional Bottom Model

The three-dimensional bottom model of the concrete cask containing design basis fuel assemblies is based on the homogenized representation of the fuel/basket and the following bottom features of the concrete cask:

- Heat transfer annulus
- Carbon steel weldment with four cutouts for the air inlets
- Concrete shield with four cutouts for the air inlets
- Four inlets with carbon steel linings
- Carbon steel bottom base plate
- Carbon steel support stand with four cutouts for air flow
- Carbon steel shield ring
- Carbon steel storage cask bottom
- Concrete pad below base plate

The three-dimensional concrete cask bottom model is shown in Figure 5.3-4 for the design basis PWR fuel. An identical cask model is employed in the BWR analysis, except that it is elongated slightly to allow for the longer active fuel region in the design basis BWR assembly.

5.3.3.3 SCALE Transfer Cask Three-Dimensional Models

Three-Dimensional Top Model

In order to estimate occupational dose rates associated with the canister sealing operation, a number of operational configurations of the standard transfer cask are considered for the three-dimensional model of the upper cask region. These include wet and dry canister conditions and various shield lid, structural lid, and temporary shielding configurations. The temporary shield is modeled as a 5-inch thick cylindrical carbon steel plate with a radius two inches shorter than the canister inner radius. The temporary shield is assumed to have oversized cutouts to permit access to the canister vent ports and a 45 degree taper around the circumference to permit the automated welding machine access to the canister shield lid/canister wall interface. This configuration amounts to an assumed temporary shielding configuration. In reality, the temporary shield may be supplemented with additional shielding materials on site, although no credit for such material is taken here.

The top configuration of the transfer cask is evaluated in detail for the welding, draining, and drying operations. As with the concrete cask models, top models of the transfer cask containing design basis fuel assemblies are based on a homogenized representation of the basket. Model features include:

- Vent and drain port openings in the canister shield lid.
- Edge tapering and port cutouts in the temporary shielding.
- Upper trunnions cut through the radial shield and extending from the inner shell to the outer shell. No credit for the radial extent of the trunnions outside the cask outer shell is taken.
- Equivalent-volume model of the heat transfer fins embedded in the neutron shield.
- Lead and neutron shielding overlap at the top as per the transfer cask drawings.

Details of the elevations and radii used in creating the three-dimensional top model are taken directly from the license drawings in Chapter 1. As with the other three-dimensional models, elevations associated with the transfer cask three-dimensional features are established with respect to the active fuel midplane of the fuel assembly for the combinatorial geometry model.

The three-dimensional transfer cask top model including shield and structural lid installation is shown in Figure 5.3-5 for the cask containing design basis PWR fuel. The BWR model is identical, except the BWR fuel and basket homogenizations and elevations are employed and the model is reflected about the axial midplane of the design basis BWR fuel assembly.

Initial designs of the standard transfer cask called for the insertion of carbon steel heat transfer fins embedded in the radial neutron shield region and a lead thickness of 3.75 inches. In the SCALE model of the standard transfer cask, the fins are treated as a thin shell of carbon steel placed between the lead and NS-4-FR walls and modeled on an equivalent-volume basis. The resulting modeled thickness of the heat fin shell is 0.304 inch. The neutron shield material thickness is then modeled as 2.696 inches, so the combined thickness of both regions is 3.00 inches. The modeled radial configuration of the transfer cask shields is, thus, 0.75 inch of low alloy steel, 3.75 inches of lead, 0.304 inch of low alloy steel (heat fins), 2.696 inches of NS-4-FR, and 1.25 inches of low alloy steel. The model conservatively underestimates the amount of both neutron shielding, due to less NS-4-FR, and gamma shielding, due to the attenuation difference between the 4 inches of lead in the design and the 3.75 inches of lead and 0.304 inch of steel modeled.

Three-Dimensional Bottom Model

The three-dimensional bottom model of the transfer cask is based on the same homogenized representation of the fuel/basket as the top model. As with the top model of the transfer cask, evaluations of both a wet and dry canister are performed. The following bottom features of the transfer cask are considered:

- Termination of the radial shields at the bottom plate.
- An explicit model of the bottom door assembly including door rails and axial neutron shield configuration.

The transfer cask bottom model is shown in Figure 5.3-6 for the cask containing PWR design basis fuel. The BWR model is identical, except the BWR fuel and basket homogenizations and elevations are employed and the model is reflected about the axial midplane of the design basis BWR fuel assembly.

5.3.4 MCBEND Three-Dimensional Concrete Cask Models

MCBEND three-dimensional shielding analysis allows detailed modeling of fuel assemblies, basket, and cask shield configuration, including streaming paths. For fuel assembly sources, some fuel assembly detail is homogenized in the model to simplify model input and improve computational efficiency. Thus, the three-dimensional models represent the various fuel assembly source regions as homogenized zones within the basket, but explicitly model the axial extent of the source regions. The fuel and hardware source regions of each assembly are therefore homogenized within the volumes defined by the periphery of the fuel assembly and the source region axial extents. The basket plate details are explicitly modeled. Cask details include the axial extent of the cask shield as described by the License Drawings.

The geometric description of a MCBEND model is based on the combinatorial geometry system embedded in the code. In this system, bodies such as cylinders and rectangular parallelpipeds, and their logical intersections and unions, are used to describe the extent of material zones.

MCBEND employs an automated biasing technique for the Monte Carlo calculation based on a three-dimensional adjoint diffusion calculation. Mesh cells for the adjoint solution are selected based on half value thicknesses for each material.

MCBEND Monte Carlo calculations are performed for each source type present in each source region. This approach entails seven separate analyses, encompassing fuel neutron, fuel gamma, fuel n-gamma (secondary gammas arising from neutron interaction in the shield), fuel region hardware, upper plenum, and upper and lower end-fitting gamma sources. Typically, a total of 5 to 20 million histories are tracked to yield dose rate profiles for each model. These cases are analyzed for azimuthally divided radial detectors at the concrete cask air inlets and outlets.

5.3.4.1 MCBEND Fuel Assembly Model

Based on the fuel assembly physical parameters provided in Table 5.2-27 and the hardware masses in Table 5.2-30, homogenized treatments of fuel assembly source regions are developed. The homogenized fuel assembly is represented in the model as a stack of boxes with width equal to the fuel assembly width. The height of each box corresponds to the modeled height of the corresponding assembly region.

The active fuel region homogenizations for the two design basis assemblies are shown in Table 5.3-6. The non-fuel assembly material is void for dry storage conditions. The clad region is zirconium alloy (density 6.55 g/cm^3). The resulting fuel compositions on an atom/barn-cm basis are shown in Table 5.3-8.

Fuel assembly non-fuel regions are homogenized as shown in Table 5.3-7. Volume fractions of material are based on the modeled regional volume and the volume of stainless steel present. The stainless steel volume is computed from the modeled mass and density (7.92 g/cm^3).

5.3.4.2 MCBEND Basket Model

For a given fuel type, the MCBEND description of the basket elements forms a common sub-model employed in the PWR and BWR concrete cask analyses. The key feature of the model is the detailed representation of the geometry of the basket support and heat transfer disks.

5.3.4.3 MCBEND Concrete Cask Model

The three-dimensional model of the vertical concrete cask containing design basis fuel is based on the explicit modeling of the basket, and the following features of the storage cask:

- Heat transfer annulus.
- Carbon steel weldment with cutouts for inlets and outlets.
- Concrete shield with cutouts for inlets and outlets.

- Air outlet model including carbon steel channel walls.
- Air inlet model including baffle pipes and carbon steel channel walls.
- Carbon steel shield plug with 1.0-inch NS-4-FR and 68-inch outer diameter steel cap.
- Carbon steel top lid.
- Carbon steel bottom base plate.
- Carbon steel support stand with four cutouts for air flow.
- Carbon steel shield ring.
- Carbon steel storage cask bottom.
- Concrete pad below base plate.

Detailed model parameters used in creating the three-dimensional model are taken directly from the License drawings. Elevations associated with the concrete cask three-dimensional features are established with respect to the bottom plate of the canister for the global model. The three-dimensional concrete cask model is shown in Figures 5.3-7 and 5.3-8.

5.3.5 Shield Regional Densities

Shield regional densities for the SAS1 and SAS4 analysis of the transfer and concrete casks are discussed in Section 5.3.5.1. Shield regional densities for the MCBEND analysis of the storage cask air inlets and outlets are discussed in Section 5.3.5.2.

5.3.5.1 SCALE Shield Regional Densities

The SCALE 4.3 standard composition library [11] default compositions and isotopic distributions are used unless otherwise indicated. The composition densities before homogenization are:

Material	Density (g/cm^3)
UO_2	10.412
Zirconium Alloy	6.56
H_2O	0.9982
Type 304 Stainless Steel	7.92
Lead	11.344
Aluminum	2.702
Neutron Absorber (core)	2.623
NS-4-FR	1.68
Concrete	2.243
Carbon Steel	7.821

The regional homogenized densities and shield densities for the PWR and BWR fuel are provided in Table 5.3-1 through Table 5.3-5.

5.3.5.2 MCBEND Shield Regional Densities

Based on the homogenization described in Section 5.3.4.1, the resulting active fuel regional densities are shown in Table 5.3-8. Material compositions for remaining structural and shield materials are shown in Table 5.3-9. Compositions for fuel assembly non-fuel regions are equivalent to the stainless steel composition in Table 5.3-9 scaled by the material volume fractions shown in Table 5.3-7.

Figure 5.3-1 SCALE Vent Port Model with Port Cover in Place (Dimensions in cm)

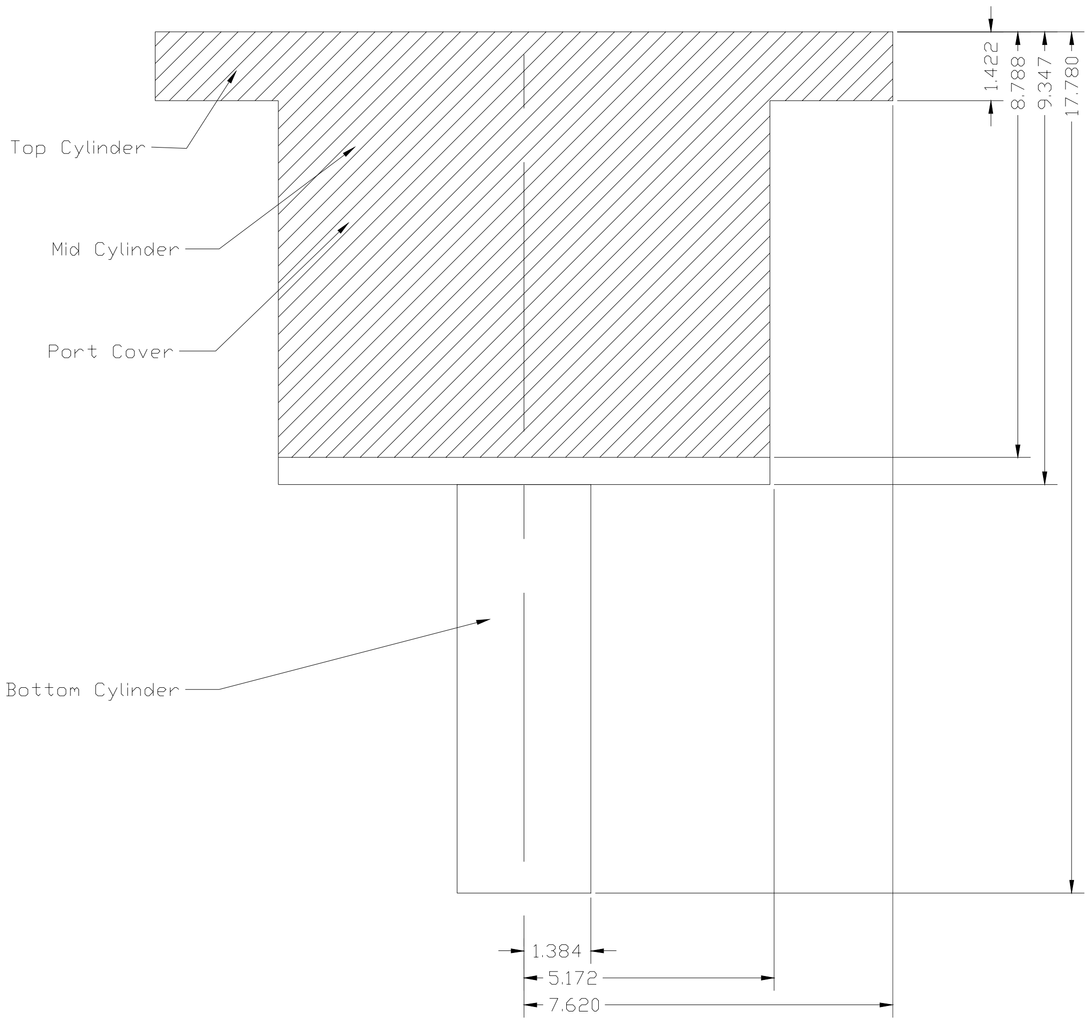


Figure 5.3-2 SCALE Vertical Concrete Cask Three-Dimensional Top Model PWR Design Basis

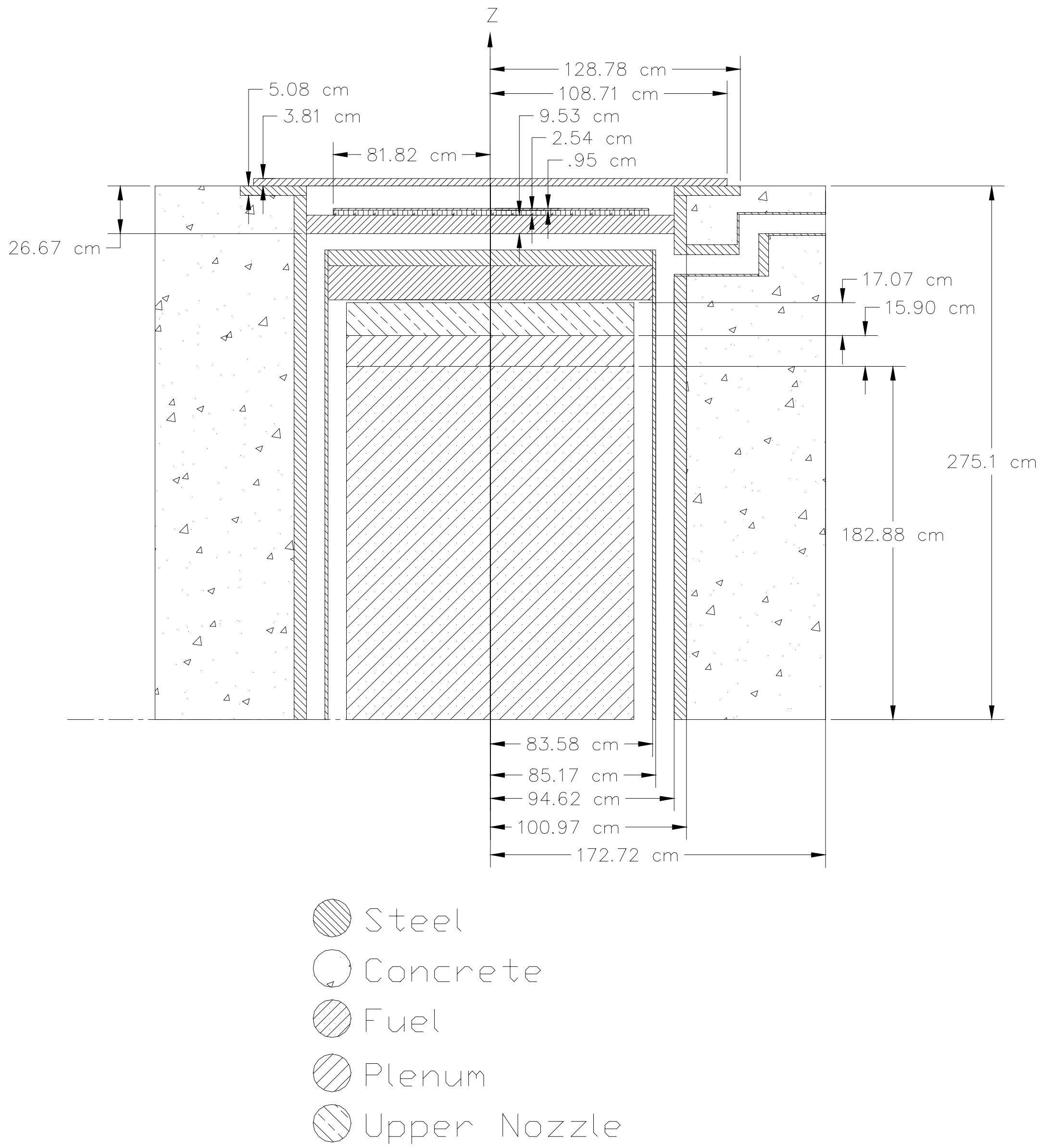
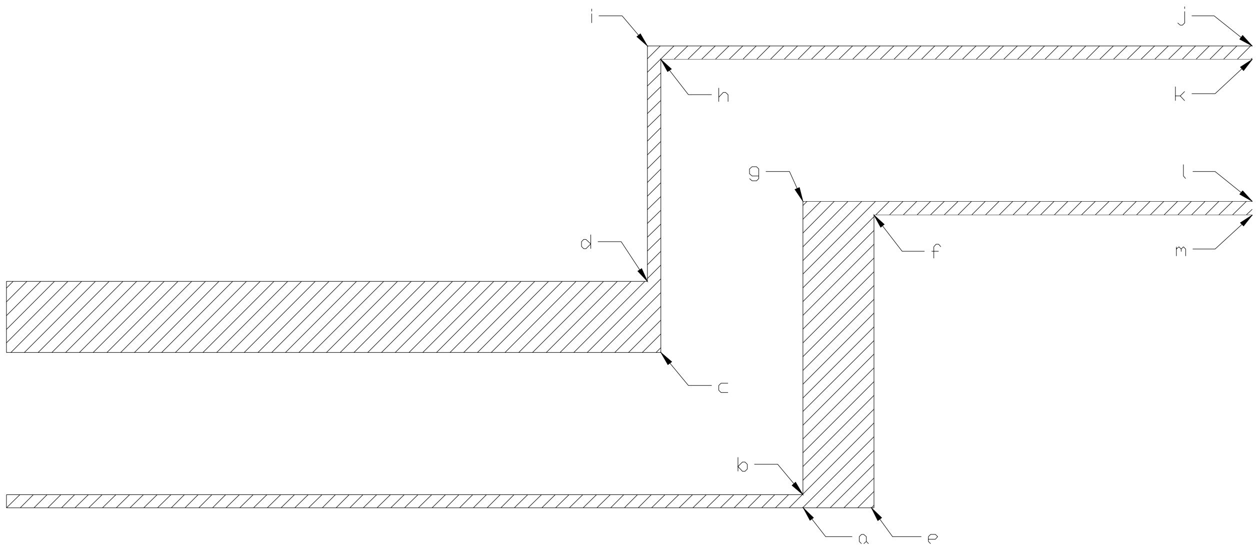


Figure 5.3-3 Schematic of SCALE Upper Vent Model Showing Key Points



Key Point	Radial Dim[1]		Axial Dim[2]	
	[in]	[cm]	[in]	[cm]
a	22.430	56.972	0.000	0.000
b	22.430	56.972	0.375	0.953
c	18.430	46.812	4.375	11.113
d	18.055	45.860	6.375	16.193
e	24.430	62.052	0.000	0.000
f	24.430	62.052	8.245	20.942
g	22.430	56.972	8.625	21.908
h	18.430	46.812	12.625	32.068
i	18.055	45.860	13.000	33.020
j	35.081	89.105	13.000	33.020
k	35.081	89.105	12.625	32.068
l	35.081	89.105	8.625	21.908
m	35.081	89.105	8.245	20.942

Notes:

(1) Dimension with respect to a reference point at a radial distance 83.615 cm along a radius extending through the outlet center.

(2) Dimension with respect to base of lower outlet channel.

Figure 5.3-4 SCALE Vertical Concrete Cask Three-Dimensional Bottom Model – PWR Design Basis

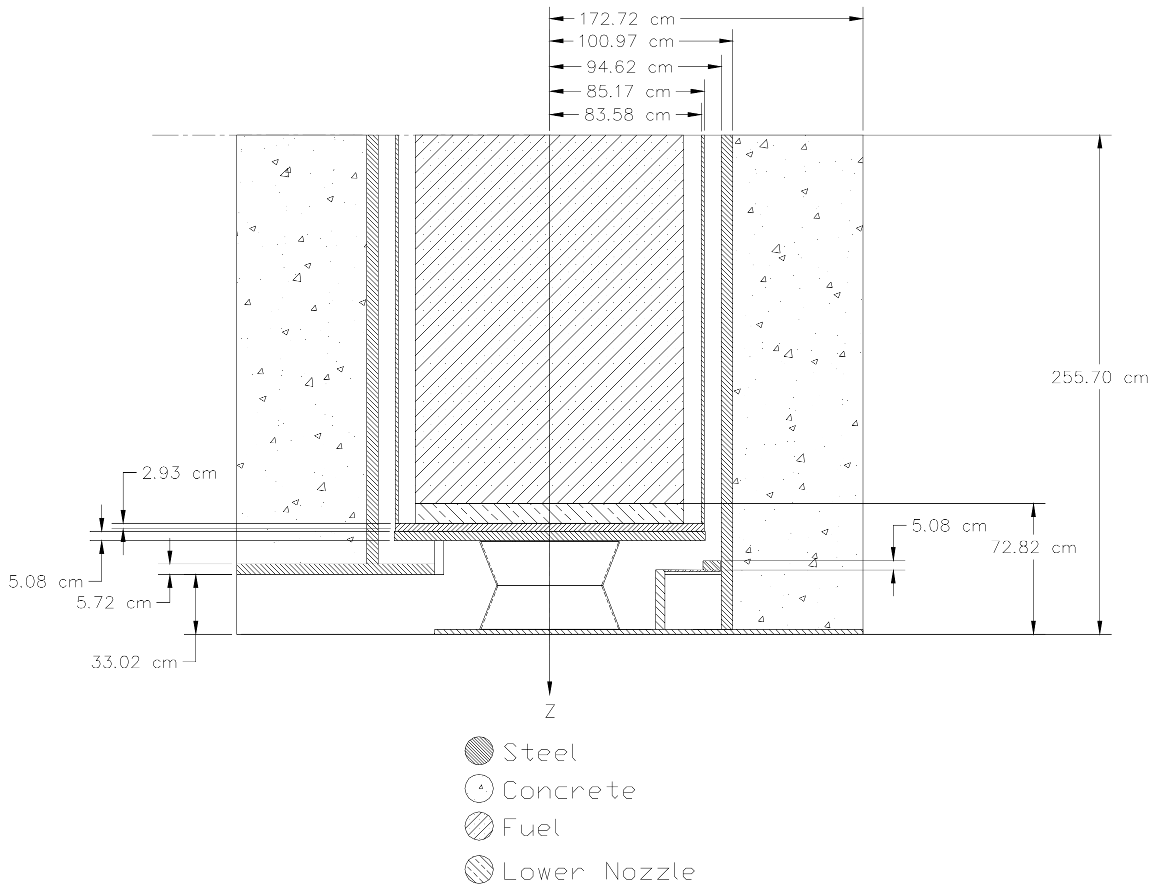


Figure 5.3-5 SCALE Standard Transfer Cask Three-Dimensional Top Model Including Shield and Structural Lid – PWR Design Basis

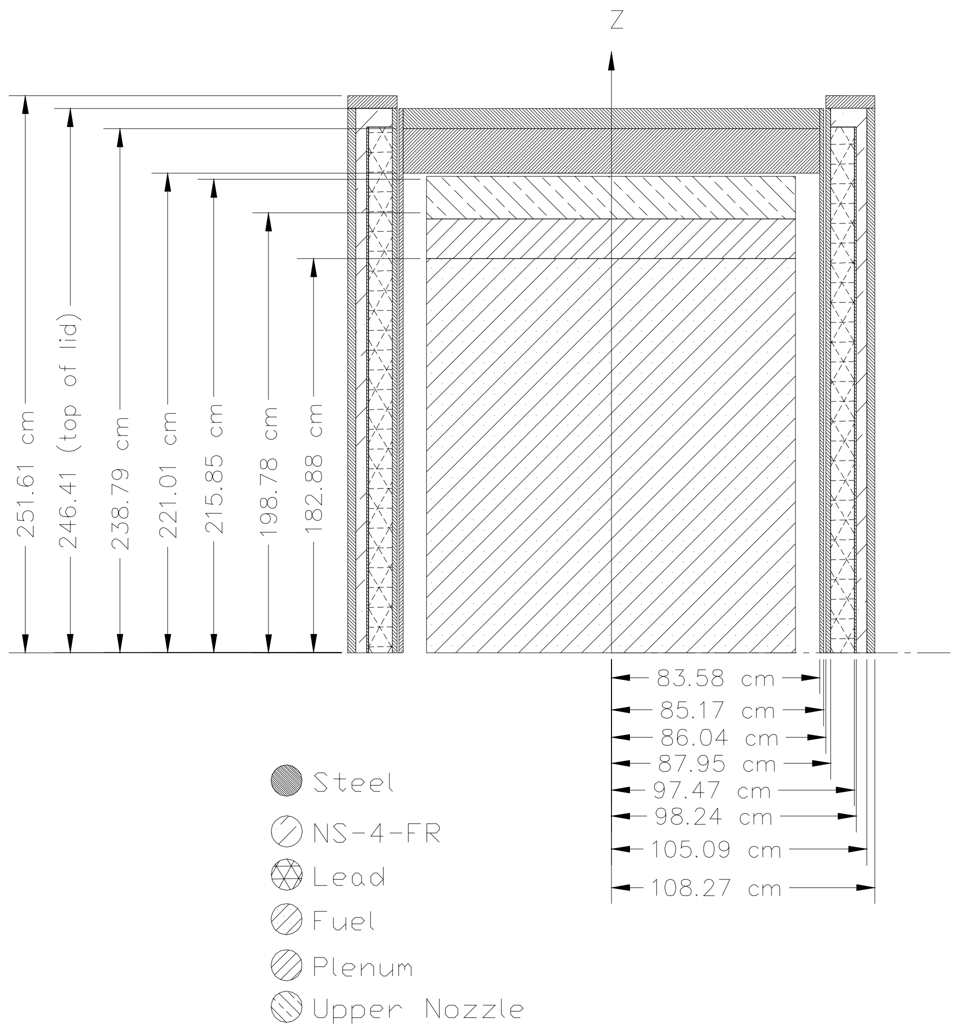


Figure 5.3-6 SCALE Standard Transfer Cask Three-Dimensional Bottom Model – PWR Design Basis

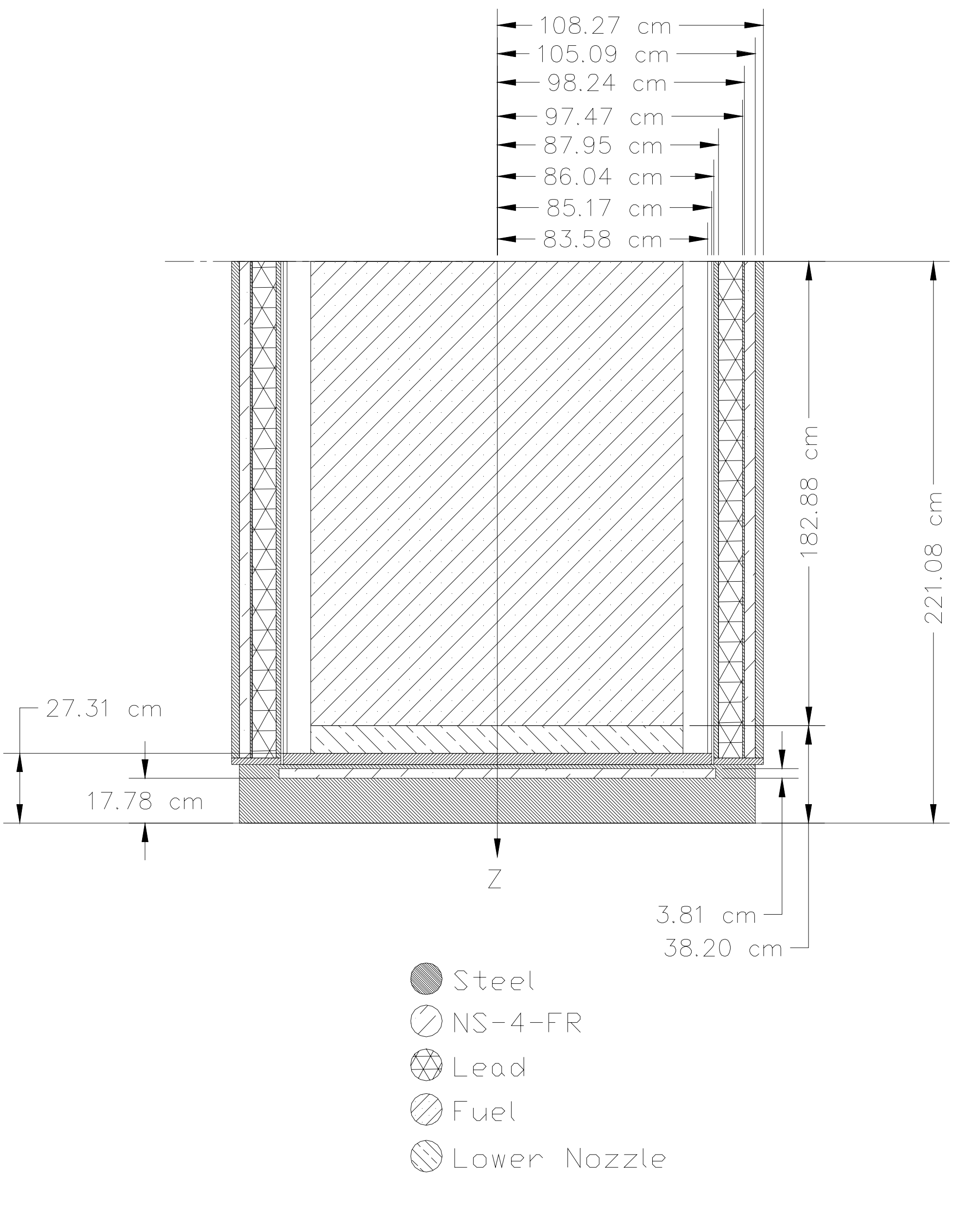


Figure 5.3-7 MCBEND Three-Dimensional Vertical Concrete Cask Model – Axial Dimensions

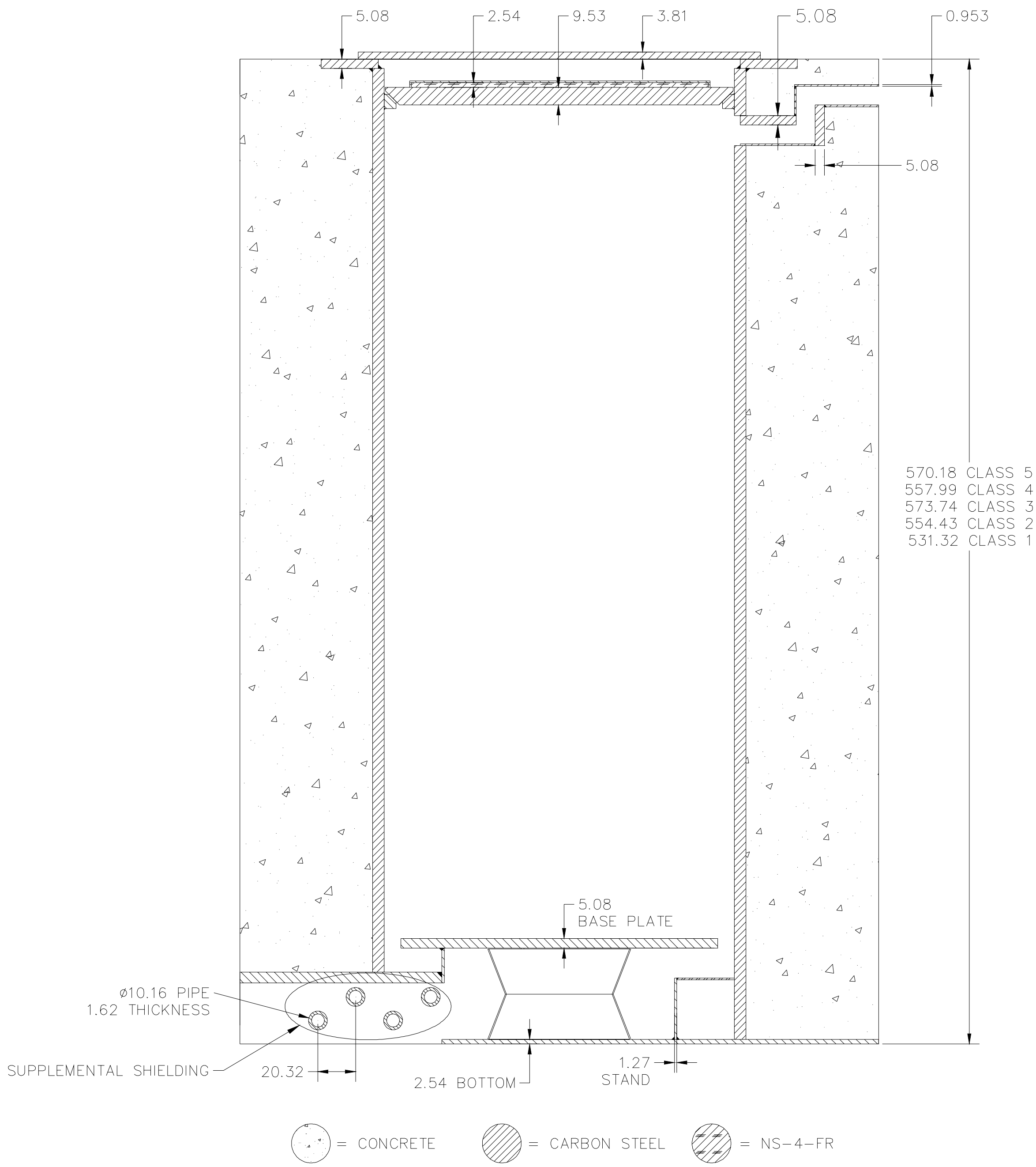


Figure 5.3-8 MCBEND Three-Dimensional Vertical Concrete Cask Model – Radial Dimensions

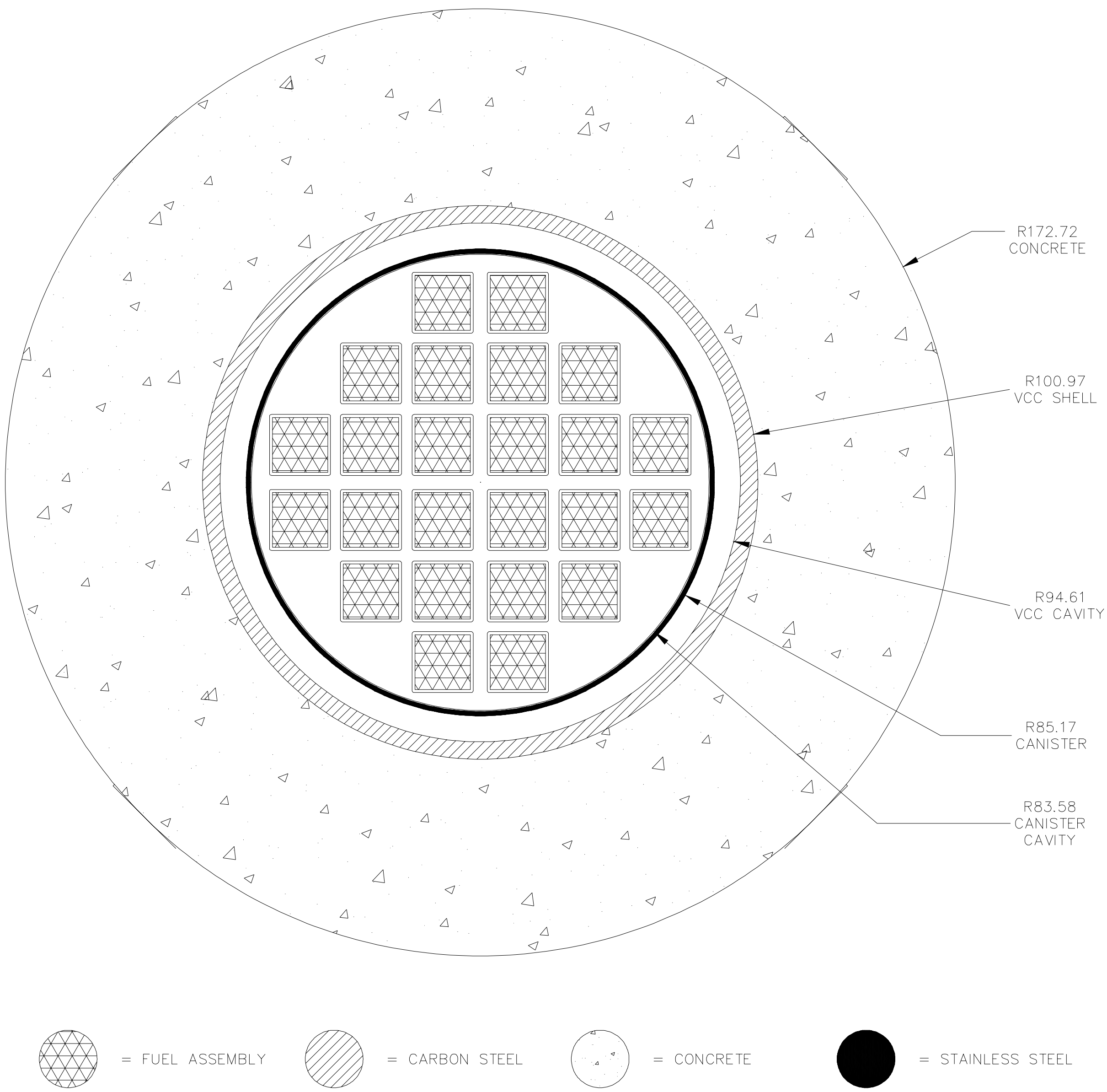


Table 5.3-1 SCALE PWR Dry Canister Material Densities

Material	Mixture ID	SCL Name	Density [g/cm³]	27N-18G Library Nuclide	Density [a/barn-cm]
Fuel Region	1	UO2	2.1530	BORON-10	1.9090E-04
		ZIRC. ALLOY	0.4494	BORON-11	7.6839E-04
		SS304	0.3807	CARBON-12	2.3982E-04
		AL	0.0894	OXYGEN-16	9.6038E-03
		B4C	0.0220	ALUMINUM	1.9953E-03
				CHROMIUM(SS304)	8.3776E-04
				MANGANESE	8.3462E-05
				IRON(SS304)	2.8532E-03
				NICKEL(SS304)	3.7112E-04
				ZIRC. ALLOY	2.9669E-03
				URANIUM-234	2.6411E-07
				URANIUM-235	3.4574E-05
				URANIUM-238	4.7671E-03
Fuel Region Annulus (One-D only)	2	SS304	0.7691	ALUMINUM	5.6780E-03
		AL	0.2544	CHROMIUM(SS304)	1.6925E-03
				MANGANESE	1.6861E-04
				IRON(SS304)	5.7642E-03
				NICKEL(SS304)	7.4975E-04
Upper Plenum	3	ZIRC. ALLOY	0.4494	CHROMIUM(SS304)	2.2706E-03
		SS304	1.0318	MANGANESE	2.2621E-04
				IRON(SS304)	7.7330E-03
				NICKEL(SS304)	1.0058E-03
				ZIRC. ALLOY	2.9669E-03
Upper Plenum Annulus (One-D only)	4	SS304	0.6101	CHROMIUM(SS304)	1.3426E-03
				MANGANESE	1.3375E-04
				IRON(SS304)	4.5725E-03
				NICKEL(SS304)	5.9475E-04
Upper End Fitting	5	SS304	1.2537	CHROMIUM(SS304)	2.7589E-03
				MANGANESE	2.7485E-04
				IRON(SS304)	9.3961E-03
				NICKEL(SS304)	1.2222E-03
Upper End Fitting Annulus (One-D only)	6	SS304	1.1366	CHROMIUM(SS304)	2.5012E-03
				MANGANESE	2.4918E-04
				IRON(SS304)	8.5185E-03
				NICKEL(SS304)	1.1080E-03
Lower End Fitting	9	SS304	1.4554	CHROMIUM(SS304)	3.2027E-03
				MANGANESE	3.1907E-04
				IRON(SS304)	1.0908E-02
				NICKEL(SS304)	1.4188E-03
Lower End Fitting Annulus (One-D only)	10	SS304	1.7805	CHROMIUM(SS304)	3.9181E-03
				MANGANESE	3.9035E-04
				IRON(SS304)	1.3344E-02
				NICKEL(SS304)	1.7357E-03

Table 5.3-2 SCALE PWR Wet Canister Material Densities

Material	Mixture ID	SCL Name	Density [g/cm^3]	27N-18G Library Nuclide	Density [a/barn-cm]
Fuel	1	UO2	2.1530	HYDROGEN	4.1824E-02
		ZIRC. ALLOY	0.4494	BORON-10	1.9090E-04
		SS304	0.3807	BORON-11	7.6839E-04
		AL	0.0894	CARBON-12	2.3982E-04
		B4C	0.0220	OXYGEN-16	3.0516E-02
		H2O	VF=0.6264	ALUMINUM	1.9953E-03
				CHROMIUM(SS304)	8.3776E-04
				MANGANESE	8.3462E-05
				IRON(SS304)	2.8532E-03
				NICKEL(SS304)	3.7112E-04
				ZIRC.ALLOY	2.9669E-03
				URANIUM-234	2.6411E-07
				URANIUM-235	3.4574E-05
				URANIUM-238	4.7671E-03
Fuel Region Annulus (One-D only)	2	SS304	0.7691	HYDROGEN	5.3996E-02
		AL	0.2544	OXYGEN-16	2.6998E-02
		H2O	VF=0.8087	ALUMINUM	5.6780E-03
				CHROMIUM(SS304)	1.6925E-03
				MANGANESE	1.6861E-04
				IRON(SS304)	5.7642E-03
				NICKEL(SS304)	7.4975E-04
Upper Plenum	3	ZIRC. ALLOY	0.4494	HYDROGEN	3.9127E-02
		SS304	1.0318	OXYGEN-16	1.9563E-02
		H2O	VF=0.5860	CHROMIUM(SS304)	2.2706E-03
				MANGANESE	2.2621E-04
				IRON(SS304)	7.7330E-03
				NICKEL(SS304)	1.0058E-03
				ZIRC. ALLOY	2.9669E-03
Upper Plenum Annulus (One-D only)	4	SS304	0.6101	HYDROGEN	6.1628E-02
		H2O	VF=0.9230	OXYGEN-16	3.0814E-02
				CHROMIUM(SS304)	1.3426E-03
				MANGANESE	1.3375E-04
				IRON(SS304)	4.5725E-03
				NICKEL(SS304)	5.9475E-04
Upper End Fitting	5	SS304	1.2537	CHROMIUM(SS304)	2.7589E-03
				MANGANESE	2.7485E-04
				IRON(SS304)	9.3961E-03
				NICKEL(SS304)	1.2222E-03
Upper End Fitting Annulus (One-D only)	6	SS304	1.1366	CHROMIUM(SS304)	2.5012E-03
				MANGANESE	2.4918E-04
				IRON(SS304)	8.5185E-03
				NICKEL(SS304)	1.1080E-03

Table 5.3-2 SCALE PWR Wet Canister Material Densities (continued)

Material	Mixture ID	SCL Name	Density [g/cm^3]	27N-18G Library Nuclide	Density [a/barn-cm]
Lower End Fitting	9	SS304 H2O	1.4554 VF=0.8162	HYDROGEN OXYGEN-16 CHROMIUM(SS304) MANGANESE IRON(SS304) NICKEL(SS304)	5.4497E-02 2.7249E-02 3.2027E-03 3.1907E-04 1.0908E-02 1.4188E-03
Lower End Fitting Annulus (One-D only)	10	SS304 H2O	1.7805 VF=0.7752	HYDROGEN OXYGEN-16 CHROMIUM(SS304) MANGANESE IRON(SS304) NICKEL(SS304)	5.1760E-02 2.5880E-02 3.9181E-03 3.9035E-04 1.3344E-02 1.7357E-03

Table 5.3-3 SCALE BWR Dry Canister Material Densities

Material	Mixture ID	SCL Name	Density [g/cm^3]	27N-18G Library Nuclide	Density [a/barn-cm]
Fuel	1	UO2	1.9583	BORON-10	5.1195E-05
		ZIRC.ALLOY	0.6769	BORON-11	2.0607E-04
		SS304	0.2228	CARBON-12	1.6127E-04
		CARBON STEEL	0.1932	OXYGEN-16	8.7353E-03
		AL	0.0874	ALUMINUM	1.9507E-03
		B4C	0.0059	CHROMIUM(SS304)	4.9029E-04
				MANGANESE	4.8845E-05
				IRON	2.0626E-03
				IRON(SS304)	1.6698E-03
				NICKEL(SS304)	2.1719E-04
				ZIRC.ALLOY	4.4688E-03
				URANIUM-234	2.4022E-07
				URANIUM-235	3.1447E-05
				URANIUM-238	4.3360E-03
Fuel Region Annulus	2	CARBON STEEL	1.2195	CARBON-12	6.1200E-04
		AL	0.1404	ALUMINUM	3.1336E-03
				IRON	1.3019E-02
Upper Plenum	3	ZIRC. ALLOY	0.6551	CARBON-12	7.4574E-05
		SS304	0.2198	CHROMIUM(SS304)	4.8369E-04
		CARBON STEEL	0.1486	MANGANESE	4.8188E-05
				IRON	1.5864E-03
				IRON(SS304)	1.6473E-03
				NICKEL(SS304)	2.1427E-04
				ZIRC. ALLOY	4.3248E-03
Upper Plenum Annulus	4	CARBON STEEL	0.9381	CARBON-12	4.7078E-04
				IRON	1.0015E-02
Upper End Fitting	5	SS304	0.5708	CHROMIUM(SS304)	1.2561E-03
				MANGANESE	1.2514E-04
				IRON(SS304)	4.2780E-03
				NICKEL(SS304)	5.5644E-04
Upper End Fitting Annulus	6	SS304	0.8665	CHROMIUM(SS304)	1.9068E-03
				MANGANESE	1.8997E-04
				IRON(SS304)	6.4942E-03
				NICKEL(SS304)	8.4470E-04
Lower End Fitting	9	SS304	1.4132	CHROMIUM(SS304)	3.1099E-03
				MANGANESE	3.0982E-04
				IRON(SS304)	1.0592E-02
				NICKEL(SS304)	1.3776E-03
Lower End Fitting Annulus	10	SS304	1.0283	CHROMIUM(SS304)	2.2629E-03
				MANGANESE	2.2544E-04
				IRON(SS304)	7.7068E-03
				NICKEL(SS304)	1.0024E-03

Table 5.3-4 SCALE BWR Wet Canister Material Densities

Material	Mixture ID	SCL Name	Density [g/cm^3]	27N-18G Library Nuclide	Density [a/barn-cm]
Fuel	1	UO2	1.9583	HYDROGEN	4.0869E-02
		ZIRC. ALLOY	0.6769	BORON-10	5.1195E-05
		SS304	0.2228	BORON-11	2.0607E-04
		CARBON STEEL	0.1932	CARBON-12	1.6127E-04
		AL	0.0874	OXYGEN-16	2.9170E-02
		B4C	0.0059	ALUMINUM	1.9507E-03
		H2O	0.6121	CHROMIUM(SS304)	4.9029E-04
				MANGANESE	4.8845E-05
				IRON	2.0626E-03
				IRON(SS304)	1.6698E-03
				NICKEL(SS304)	2.1719E-04
				ZIRC. ALLOY	4.4688E-03
				URANIUM-234	2.4022E-07
				URANIUM-235	3.1447E-05
				URANIUM-238	4.3360E-03
Fuel Region Annulus	2	CARBON STEEL	1.2195	HYDROGEN	5.2888E-02
		AL	0.1404	CARBON-12	6.1200E-04
		H2O	0.7921	OXYGEN-16	2.6444E-02
				ALUMINUM	3.1336E-03
				IRON	1.3019E-02
Upper Plenum	3	ZIRC.ALLOY	0.6551	HYDROGEN	4.3814E-02
		SS304	0.2198	CARBON-12	7.4574E-05
		CARBON STEEL	0.1486	OXYGEN-16	2.1907E-02
		H2O	0.6562	CHROMIUM(SS304)	4.8369E-04
				MANGANESE	4.8188E-05
				IRON	1.5864E-03
				IRON(SS304)	1.6473E-03
				NICKEL(SS304)	2.1427E-04
				ZIRC. ALLOY	4.3248E-03
Upper Plenum Annulus	4	CARBON STEEL	0.9381	HYDROGEN	5.8764E-02
		H2O	0.8801	CARBON-12	4.7078E-04
				OXYGEN-16	2.9382E-02
				IRON	1.0015E-02
Upper End Fitting	5	SS304	0.5708	CHROMIUM(SS304)	1.2561E-03
				MANGANESE	1.2514E-04
				IRON(SS304)	4.2780E-03
				NICKEL(SS304)	5.5644E-04
Upper End Fitting Annulus	6	SS304	0.8665	CHROMIUM(SS304)	1.9068E-03
				MANGANESE	1.8997E-04
				IRON(SS304)	6.4942E-03
				NICKEL(SS304)	8.4470E-04

Table 5.3-4 SCALE BWR Wet Canister Material Densities (continued)

Material	Mixture ID	SCL Name	Density [g/cm^3]	27N-18G Library Nuclide	Density [a/barn-cm]
Lower End Fitting	9	SS304 H2O	1.4132 0.8216	HYDROGEN OXYGEN-16 CHROMIUM(SS304) MANGANESE IRON(SS304) NICKEL(SS304)	5.4858E-02 2.7429E-02 3.1099E-03 3.0982E-04 1.0592E-02 1.3776E-03
Lower End Fitting Annulus	10	SS304 H2O	1.0283 0.8702	HYDROGEN OXYGEN-16 CHROMIUM(SS304) MANGANESE IRON(SS304) NICKEL(SS304)	5.8103E-02 2.9051E-02 2.2629E-03 2.2544E-04 7.7068E-03 1.0024E-03

Table 5.3-5 SCALE Standard Transfer Cask Material Densities

Material	Mixture ID	SCL Name	Density [g/cm^3]	27N-18G Library Nuclide	Density [a/barn-cm]
Carbon and Low-Alloy Steel	11	CARBONSTEEL	7.8212	CARBON-12 IRON	3.9250E-03 8.3498E-02
Stainless Steel	12	SS304	7.9200	CHROMIUM(SS304) MANGANESE IRON(SS304) NICKEL(SS304)	1.7429E-02 1.7363E-03 5.9358E-02 7.7207E-03
Lead	13	PB	11.3440	LEAD	3.2969E-02
NS-4-FR	14	H B-10 B-11 C N O AL	1.63	HYDROGEN BORON-10 BORON-11 CARBON-12 NITROGEN-14 OXYGEN-16 ALUMINUM	5.8540E-02 8.5530E-05 3.4220E-04 2.2640E-02 1.3940E-03 2.6090E-02 7.7630E-03
Aluminum	17	AL	2.7020	ALUMINUM	6.0307E-02
Concrete	18	REG-CONCRETE	2.2426	HYDROGEN OXYGEN-16 SODIUM-23 ALUMINUM SILICON CALCIUM IRON	1.3401E-02 4.4931E-02 1.7036E-03 1.7018E-03 1.6205E-02 1.4826E-03 3.3857E-04
Canister Void (Dry Conditions)	19	N	VF=1.0E-6	NITROGEN-14	4.3006E-08
Canister Water (Wet Conditions)	19	H2O	0.9982	HYDROGEN OXYGEN-16	6.6769E-02 3.3385E-02

Table 5.3-6 MCBEND Fuel Region Homogenization

WE 17×17						
Component	**Area [cm^2]**	**Area Fraction**	**Volume Fraction of Components**			
			UO_2	**Void**	**Clad**	**Interstitial**
Fuel	1.3913E+02	3.0375E-01	3.0375E-01			
Gap	5.6649E+00	1.2367E-02		1.2367E-02		
Clad	4.2318E+01	9.2389E-02			9.2389E-02	
Guide Tube	3.4075E+00	7.4392E-03			7.4392E-03	
Instrument Tube	1.4198E-01	3.0997E-04			3.0997E-04	
Inside Tubes	2.5881E+01	5.6502E-02				5.6502E-02
Interstitial	2.4150E+02	5.2725E-01				5.2725E-01
Total	4.5805E+02		3.0375E-01	1.2367E-02	1.0014E-01	5.8375E-01
GE 9×9-2L						
Component	**Area [cm^2]**	**Area Fraction**	**Volume Fraction of Components**			
			UO_2	**Void**	**Clad**	**Interstitial**
Fuel	5.6593E+01	2.8809E-01	2.8809E-01			
Gap	2.8033E+00	1.4271E-02		1.4271E-02		
Clad	1.8455E+01	9.3945E-02			9.3945E-02	
Water Rod	4.6792E-01	2.3820E-03			2.3820E-03	
Inside Tubes	1.5024E+00	7.6484E-03				7.6484E-03
Interstitial	1.1662E+02	5.9366E-01				5.9366E-01
Total	1.9644E+02		2.8809E-01	1.4271E-02	9.6327E-02	6.0131E-01

Table 5.3-7 MCBEND Fuel Assembly Hardware Region Homogenization

WE 17×17					
Region	**Mass SS [kg/assy]**	**SS Volume [cm^3/assy]**	**Height [cm]**	**Volume [cm^3/assy]**	**Volume Fraction**
Lower Nozzle	5.90	7.4495E+02	8.5979	3.9382E+03	1.8916E-01
Upper Plenum	8.47	1.0694E+03	22.2123	1.0174E+04	1.0511E-01
Upper Nozzle	14.53	1.8341E+03	9.3218	4.2698E+03	4.2955E-01
GE 9×9-2L					
Region	**Mass SS [kg/assy]**	**SS Volume [cm^3/assy]**	**Height [cm]**	**Volume [cm^3/assy]**	**Volume Fraction**
Lower Nozzle	4.99	6.2961E+02	18.7579	3.6848E+03	1.7087E-01
Upper Plenum	1.68	2.1212E+02	28.6421	5.6265E+03	3.7701E-02
Upper Nozzle	2.69	3.3958E+02	19.0500	3.7422E+03	9.0743E-02

Table 5.3-8 MCBEND Homogenized Fuel Regional Densities

Element	**Density [atom/b-cm]**	
	WE 17x17	**GE 9×9-2L**
CR	7.5967E-06	7.3075E-06
FE	1.4146E-05	1.3607E-05
HF	2.2130E-07	2.1288E-07
NI	6.7299E-07	6.4737E-07
O	1.4131E-02	1.3403E-02
SN	4.9912E-05	4.8011E-05
U	7.0560E-03	6.6919E-03
ZR	4.2469E-03	4.0852E-03

Table 5.3-9 MCBEND Regional Densities for Concrete Cask Structural and Shield Materials

Material	Element	Density [atom/b-cm]
Stainless Steel	CR	1.6511E-02
	FE	6.3199E-02
	NI	6.5009E-03
Carbon Steel	C	3.9250E-03
	FE	8.3498E-02
Aluminum	AL	6.0263E-02
NS-4-FR	AL	7.8000E-03
	B	4.2750E-04
	C	2.2600E-02
	H	5.8500E-02
	N	1.3900E-03
	O	2.6100E-02
Concrete	AL	1.7018E-03
	CA	1.4826E-03
	FE	3.3857E-04
	H	1.3401E-02
	NA	1.7036E-03
	O	4.4931E-02
	SI	1.6205E-02

5.4 Shielding Evaluation

This section evaluates the shielding design of the vertical concrete cask and the standard transfer cask. The calculational methods and the computer codes used in the evaluation are described. Shielding calculations are performed with design basis PWR and BWR fuel source terms at 40,000 MWD/MTU and 5-year cooling time. Dose rate profiles are reported as a function of distance from the sides and top of the concrete cask and from the sides, top, and bottom of the transfer cask containing PWR or BWR fuel. Top axial dose rates for operational configurations of the transfer cask during the canister sealing operation are also provided.

5.4.1 Calculational Methods

5.4.1.1 SCALE Package Calculational Methods

The shielding evaluations of the concrete cask and standard transfer cask are performed with SCALE 4.3 for the PC [4]. In particular, SCALE shielding analysis sequence SAS2H [5] is used to generate source terms for the design basis fuel. SAS1 [6] is used to perform one-dimensional radial and axial shielding analyses in order to identify bounding PWR and BWR fuel descriptions. A modified version of SAS4 [3] is used to perform three-dimensional shielding analysis. The coupled 27 group neutron, 18 group gamma ENDF/B-IV (27N-18COUPLE) cross-section library is used in all shielding evaluations. Source terms include fuel neutron, fuel gamma, and gamma contributions from activated hardware. Dose rate evaluations include the effect of fuel burnup peaking on fuel neutron and gamma source terms. The SCALE shielding analysis sequences and cross-section libraries have recently been benchmarked to measurements of light water reactor fuel source terms, shielding material dose rate attenuation, and spent fuel storage and transport cask dose rates [13].

As discussed in Section 5.2, the SAS2H code sequence [5] is used to generate source terms for the PWR and BWR design basis fuel. SAS2H includes an XSDRNPM [8] neutronics model of the fuel assembly and ORIGEN-S [9] fuel depletion/source terms calculations. Source terms are generated for both UO_2 fuel and fuel assembly hardware. The hardware activation is calculated by ORIGEN-S using the incore neutron flux spectrum produced by the SAS2H neutronics model. The hardware is assumed to be Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity. The effects of axial flux spectrum and magnitude variation on hardware activation are estimated by flux ratios based on empirical data [15].

The SAS4 shielding models are used to estimate the dose profiles along the surfaces of the transfer and concrete casks and to estimate doses in and around streaming paths such as the canister vent and drain ports. The SAS4 models represent the cask body and any streaming paths with combinatorial logic. The method of solution is adjoint discrete ordinates and Monte Carlo [3] using the XSDRNPM and MORSE codes, respectively. Since SAS4 requires model symmetry at the fuel midplane, two models are created for each cask, a top and a bottom model. Radial biasing is performed to estimate dose rates on the sides of the cask, and axial biasing is performed to estimate dose rates on the top and bottom surfaces of the cask. Modifications are made to SAS4 to determine dose rates all along the radial, top, and bottom surfaces of the cask as well as any cylindrical surface surrounding the cask. Thus, detailed dose profiles are determined that explicitly show peaks due to the fuel burnup profile, activated hardware gamma emission, and potential streaming paths.

In both the SAS1 and SAS4 models, the fuel and hardware source regions are homogenized within the volumes described by the periphery of the basket tubes, and defined by fuel assembly active fuel, plenum, and end fitting elevations. Within these volumes, the material masses of the fuel assembly and basket are preserved.

5.4.1.2 MCBEND Calculational Methods

The shielding evaluations of the storage cask air inlets and outlets are performed with MCBEND version 9E. Source terms include fuel neutron, fuel gamma, and gamma contributions from activated hardware. As described in Section 5.2, these evaluations include the effect of fuel burnup peaking on fuel neutron and gamma source terms.

The MCBEND shielding models described in Section 5.3 are utilized with the source terms described in Section 5.2 to estimate the azimuthal dose rate profiles at the surface of the concrete cask inlets and outlets. The method of solution is continuous energy Monte Carlo with an adjoint diffusion solution for generating importance meshes. Radial biasing is performed within the MCBEND code, with an additional azimuthal component added to the splitting mesh to account for the angular variations in the bulk shielding properties of the concrete cask at the inlets and outlets.

The MCBEND code has been validated against various classical shielding problems, including fast and thermal neutron sources penetrating through single material slab geometries of iron, graphite and water. The validation suite also includes fast neutron transmission through alternating slabs of iron and water. Of particular interest is a benchmark of MCBEND to gamma and neutron dose rates outside a metal transport cask, where agreement between measurement and calculation is within 20% for the majority of dose locations.

MCBEND results are calculated using the JEF2.2 neutron cross-section library and the ANSWERS gamma library.

5.4.2 Flux-to-Dose Rate Conversion Factors

The ANSI/ANS 6.1.1-1977 flux-to-dose rate conversion factors [22] are used in all Universal Storage System shielding evaluations. These factors are the defaults for SCALE 4.3 analyses. Tables 5.4-1 and 5.4-2 show the group flux-to-dose rate factors associated with the coupled 27 group neutron and 18 group gamma cross-section library used in the SCALE shielding evaluations. Tables 5.4-3 and 5.4-4 show the group flux-to-dose rate factors in the 28-group neutron and 22-group gamma energy structure employed in the MCBEND evaluations.

5.4.3 Dose Rate Results

This section provides detailed dose rate profiles for the vertical concrete cask and the standard transfer cask based on the source terms presented in Section 5.2. Design basis fuel source terms include contributions from fuel neutron, fuel gamma, and activated hardware gamma. The fuel assembly activated hardware gamma source terms include: steel and inconel in the upper and lower fuel assembly end fittings, upper fuel rod plenum hardware, and activated non-fuel material in the active fuel region. The three-dimensional model dose rates include the effects of axial profiles for neutron and gamma source distributions shown in Figure 5.2-3 and Figure 5.2-4 for PWR and BWR fuel assemblies, respectively.

Three-dimensional dose rates for the concrete cask side and top are calculated using SAS4, with detailed air inlet and outlet results calculated using MCBEND. Three-dimensional dose rates for the transfer cask are calculated using SAS4 exclusively.

5.4.3.1 Vertical Concrete Cask Dose Rates

One-Dimensional Dose Rates

One-dimensional radial dose rates with design basis PWR or BWR fuel are found to be in good agreement with the corresponding three-dimensional models at the radial midplane. Generally, the homogenization of canister annulus basket material employed in the one-dimensional analysis leads to a slight under-prediction of radial gamma dose rates. The three-dimensional models more accurately characterize the shielding effectiveness of the basket support disks. One-dimensional analysis is found to support the results of the more sophisticated three-dimensional models.

Three-Dimensional Dose Rates for Concrete Cask Containing PWR Fuel

The three-dimensional model dose rates for the concrete cask containing PWR fuel are presented in Figures 5.4-1 through 5.4-5. Figure 5.4-1 shows the axial dose rate profile along the cask surface broken down by contributing radiation type. Dose rates along the cask axial surface are dominated by gamma contributions due to the relatively high neutron shielding effectiveness of the concrete. Figure 5.4-2 shows the total dose rate profile at various radial distances from the cask surface.

In the axial profile plots, each datum represents the circumferentially averaged dose rate at the corresponding elevation. Negative elevations indicate axial locations below the fuel axial midplane, and correspond to results obtained from the three-dimensional bottom half model. In the vertical dose profile, peaking is observed at the upper and lower end fitting locations as well as at the locations of the lower intake and upper outlet vents.

At locations away from the air inlets and outlets, the maximum axial dose rates occur at the fuel midplane, where a peak dose rate of 49 (<1%) mrem/hr is computed. At the air outlets, an azimuthal maximum of 63 mrem/hr (1%) is computed. Figure 5.4-3 illustrates the azimuthal variation of total dose rate at the air outlet elevation. Dose rates at the inlets are considerably higher than at the outlets. The dose rate at the air inlet openings is 136 (1%) mrem/hr with supplemental shielding and 694 (<1%) mrem/hr without supplemental shielding. The azimuthal variation of dose rate at the air inlets is shown in Figure 5.4-4 with supplemental shielding in the inlets.

In Figure 5.4-5, the radial dose rate profile at the top surface of the cask is shown. Two peaks occur in the radial profile. Above the canister/weldment annulus, a peak is formed from approximately equal contributions of end-fitting and plenum gamma and fuel neutron. At radial locations above the upper vents, another peak is observed due primarily to end-fitting gammas.

Three-Dimensional Dose Rates for Concrete Cask Containing BWR Fuel

Figures 5.4-6 through 5.4-10 present the three-dimensional model dose rates for the concrete cask containing BWR fuel. Figure 5.4-6 shows the axial dose rate profile along the cask surface broken down by contributing radiation type. Dose rates along the cask axial surface are dominated by gamma contributions due to the relatively high neutron shielding effectiveness of the concrete. Figure 5.4-7 shows the total dose rate profile at various radial distances from the cask surface.

In the axial profile plots, each datum represents the circumferentially averaged dose rate at the corresponding elevation. Negative elevations indicate axial locations below the fuel axial midplane, and correspond to results obtained from the three-dimensional bottom half model. In the vertical dose profile, peaking is observed at the upper and lower end fitting locations as well as at the locations of the lower intake and upper outlet vents.

At locations away from the air inlets and outlets, the maximum axial dose rates occurs at the fuel midplane, where a peak dose rate of 31 (1%) mrem/hr is computed. At the air outlets, an azimuthal maximum of 55 mrem/hr (1%) is computed. Figure 5.4-8 illustrates the azimuthal variation of total dose rate at the air outlet elevation. Dose rates at the air inlets are considerably higher than at the air outlets. The dose rate at the air inlet opening is 129 (1%) mrem/hr with supplemental shielding and 645 (<1%) mrem/hr without supplemental shielding. The azimuthal variation of dose rate at the air inlet is shown in Figure 5.4-9.

In Figure 5.4-10, the radial dose rate profile at the top surface of the cask is shown. Two peaks occur in the radial profile. Above the canister/weldment annulus, a peak is formed from approximately equal contributions of end-fitting and plenum gamma and fuel neutron. At radial locations above the upper vents, another peak is observed due primarily to end-fitting gammas.

5.4.3.2 Standard Transfer Cask Dose Rates

One-Dimensional Dose Rates

One-dimensional radial dose rates for the standard transfer cask with design basis PWR or BWR fuel are in good agreement with the corresponding three-dimensional models at the radial midplane. As with the concrete cask one-dimensional radial model, the peaks in the radial dose rates due to activated end fittings cannot be captured by one-dimensional analysis. One-dimensional analysis supports the results of the more sophisticated three-dimensional models.

Three-Dimensional Dose Rates for the Standard Transfer Cask Containing PWR Fuel

The three-dimensional model dose rates for the standard transfer cask containing PWR fuel are presented in Figures 5.4-11 through 5.4-19. For the top and bottom axial cases, the SAS4 surface detectors are subdivided in a manner which gives the centermost subdetector a relatively large radius. This detector partitioning more closely balances subdetector areas and avoids poor Monte Carlo statistics on the central subdetector.

The transfer cask side dose rate profiles with a dry cavity are shown in Figure 5.4-11 for the constituent source components and in Figure 5.4-13 at various distances from the cask surface. In this condition, the majority of the dose rate is from fuel neutron and gamma source, but significant peaks are shown from the activated end fittings. In this condition, the peak dose rate on the side of the transfer cask is 410 (<1%) mrem/hr.

The transfer cask side dose rate profiles with a wet canister are shown in Figure 5.4-12 for the constituent source components and in Figure 5.4-14 at various distances from the cask surface. In the wet case, the majority of the dose rate is from fuel gamma sources and activated non-fuel hardware gamma. Note that in the wet condition, it is assumed in the model that the water level in the canister is lowered to the base of the upper end-fitting in order to facilitate the lid welding operations. Thus, the top end fitting is uncovered and causes a peak in dose rate at the top of the transfer cask due to the gamma source from the activated top end fitting. In this condition, the peak dose rate on the side of the transfer cask is 259 (<1%) mrem/hr.

When configured for the shield lid welding operation, the standard transfer cask, with wet canister and temporary shielding in place, has a peak surface dose rate of 2,092 (4%) in the narrow gap between the temporary shield and the cask inner shell. This dose rate is highly localized; the average dose rate on the top of the cask under these conditions is 579 (3%) mrem/hr. Refer to Figure 5.4-15 for a plot of the radial dose profile.

After draining the canister cavity and in preparation for the vent port cover welding operation, the shield lid, temporary shield, and vent port covers are in place. Under these conditions, the surface dose rate radial profile is shown in Figure 5.4-16. The peak surface dose rate is 1147 (2%) mrem/hr, and the surface average value is 382 (2%) mrem/hr.

After completion of the lid welding operation, the transfer cask will have a dry canister cavity, and both shield lid and structural lids in place with no temporary shielding. In this condition, the transfer cask top dose rate profile is shown in Figure 5.4-17 for each source component. In this condition, the majority of the dose rate is from end fitting gamma. The peak and average dose rates on the top of the transfer cask containing PWR fuel are 715 (<1%) mrem/hr and 369 (2%) mrem/hr, respectively.

The standard transfer cask bottom dose rate radial profiles with dry and wet canisters are shown in Figures 5.4-18 and 5.4-19, respectively. In the dry canister condition, the peak and average dose rates on the bottom of the transfer cask are 819 (<1%) mrem/hr and 374 (<1%) mrem/hr, respectively. In the wet condition, the peak and average dose rates on the bottom of the transfer cask are 579 (<1%) mrem/hr and 258 (<1%) mrem/hr, respectively.

Three-Dimensional Dose Rates for Standard Transfer Cask Containing BWR Fuel

The three-dimensional model dose rates for the standard transfer cask containing BWR fuel are presented in Figures 5.4-20 through 5.4-28.

The transfer cask side dose rate profiles with a dry cavity are shown in Figure 5.4-20 for the constituent source components and in Figure 5.4-22 at various distances from the cask surface. In this condition, the majority of the dose rate is from fuel neutron and gamma source, but significant peaks are shown from the activated end fittings. In this condition, the peak dose rate on the side of the transfer cask is 325 (<1%) mrem/hr.

The transfer cask side dose rate profiles with a wet canister are shown in Figure 5.4-21 for the constituent source components and in Figure 5.4-23 at various distances from the cask surface.

In the wet case, the majority of the dose rate is from fuel gamma sources and activated non-fuel hardware gamma. Note that in the wet condition, it is assumed in the model that the water level in the canister is lowered to the base of the upper end-fitting in order to facilitate the lid welding operations. Thus, the top end fitting is uncovered and causes a peak in dose rate at the top of the transfer cask due to the gamma source from the activated top end fitting. In this condition, the peak dose rates on the side of the transfer cask is 189 (<1%) mrem/hr.

When configured for the shield lid welding operation, the standard transfer cask, with wet canister and temporary shielding in place, has a peak surface dose rate of 1803 (4%) in the narrow gap between the temporary shield and the cask inner shell. This dose rate is highly localized; the average dose rate on the top of the cask under these conditions is 466 (3%) mrem/hr. Refer to Figure 5.4-24 for a plot of the radial dose profile.

After draining the canister cavity and in preparation for the vent port cover welding operation, the shield lid, temporary shield, and vent port covers are in place. Under these conditions, the radial surface dose rate profile is shown in Figure 5.4-25. The localized peak surface dose rate is 846 (3%) mrem/hr, and the surface average value is 264 (2%) mrem/hr.

After completion of the lid welding operation, the transfer cask will have a dry canister cavity, and both shield lid and structural lids in place with no temporary shielding. In this condition, the transfer cask top dose rate profile is shown in Figure 5.4-26 for each source component. In this condition, the majority of the dose rate is from end fitting gamma. The peak and average dose rates on the top of the transfer cask containing BWR fuel are 396 (<1%) mrem/hr and 222 (2%) mrem/hr, respectively.

The standard transfer cask bottom dose rate radial profiles with dry and wet canisters are shown in Figures 5.4-27 and 5.4-28, respectively. In the dry canister condition, the peak and average dose rates on the bottom of the transfer cask are 786 (<1%) mrem/hr and 379 (<1%) mrem/hr, respectively. In the wet condition, the peak and average dose rates on the bottom of the transfer cask are 539 (<1%) mrem/hr and 254 (<1%) mrem/hr, respectively.

<u>Transfer Cask Extension</u>

The transfer cask may be lengthened using a steel transfer cask extension. The extension is used when loading canisters containing fuel assemblies with control element assemblies inserted, which generally requires a longer canister than the canister used if fuel does not contain control elements. The transfer cask extension does not require neutron shielding since it is located axially above the active fuel region. As shown in the axial dose rate plots of Figure 5.4-11 and 5.4-12, the neutron dose decreases rapidly above the active fuel region. Since the top of the control element is located well outside the active core during reactor operations, activation of the top of the control element is minimal. Therefore, the solid steel extension is sufficient to attenuate the gamma sources in this region of the transfer cask.

To accommodate the use of the transfer cask extension, the transfer cask design is modified to replace the axial three inches of neutron shielding (NS-4-FR) by an annular steel ring equal in radii to the lead shield. The removed NS-4-FR is an annular ring modeled between the lead shield and the transfer cask top plate. Replacing the NS-4-FR with steel minimizes the gamma dose rate peaking at the radial cask surface below the interface between the cask extension and the transfer cask top plate, when a longer canister is used.

The annular steel ring serves to decrease cask surface dose rates at the ring elevation to a value lower than the calculated maximum radial dose rate for the cask without extension. Without the steel replacement, the longer canister, in the otherwise shorter transfer cask body, results in the canister lids shifting axially above the elevation of the lead shield, thereby providing a gamma ray streaming path.

Figure 5.4-1 Vertical Concrete Cask Axial Surface Dose Rate Profile by Source Component – Azimuthal Average – PWR Fuel

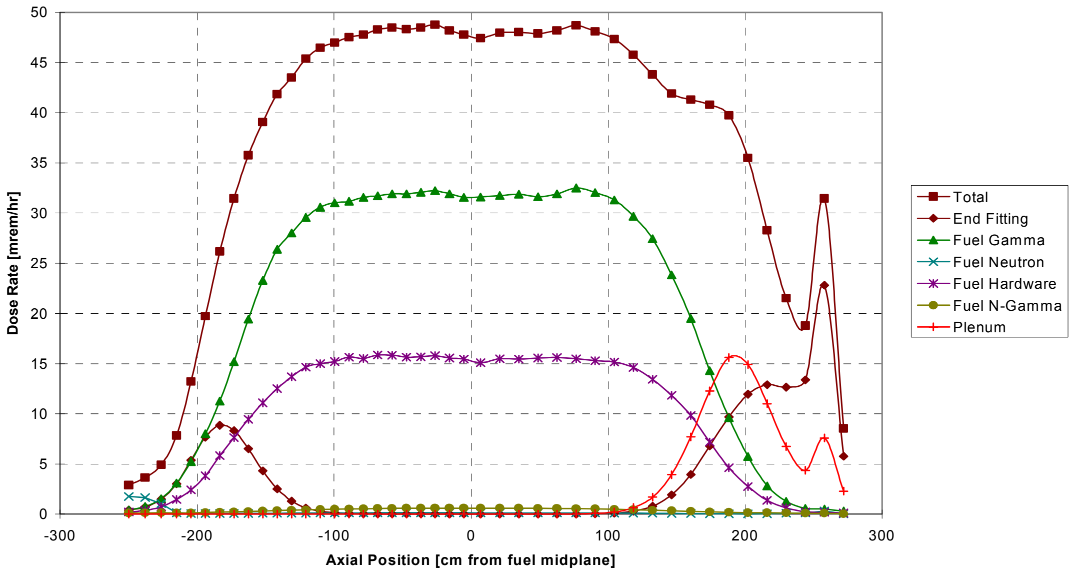


Figure 5.4-2 Vertical Concrete Cask Axial Surface Dose Rate Profile at Various Distances from Cask – Azimuthal Average – PWR Fuel

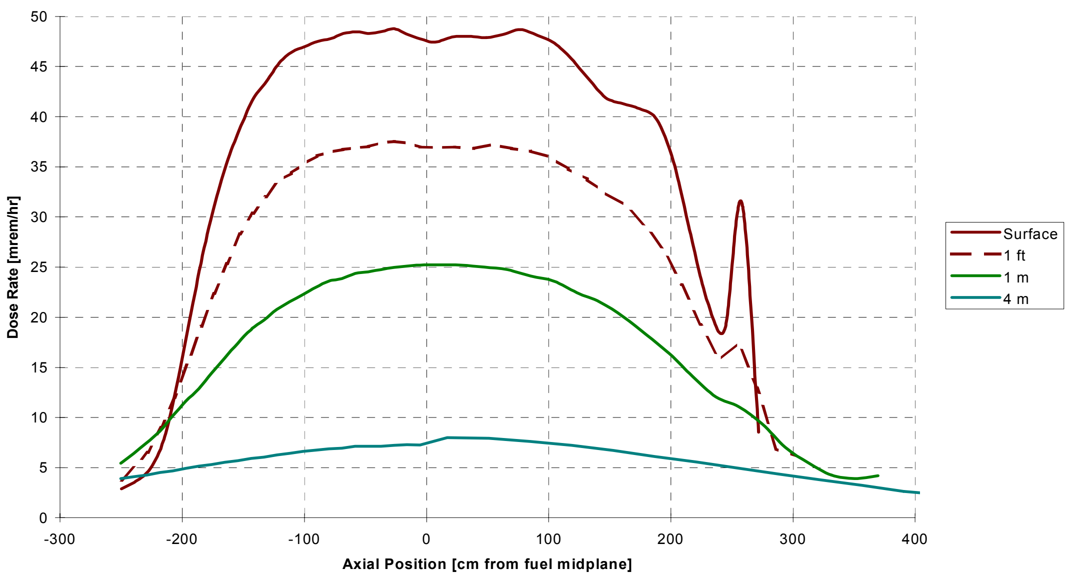


Figure 5.4-3 Vertical Concrete Cask Top Air Outlet Elevation Azimuthal Surface Dose Rate Profile – PWR Fuel

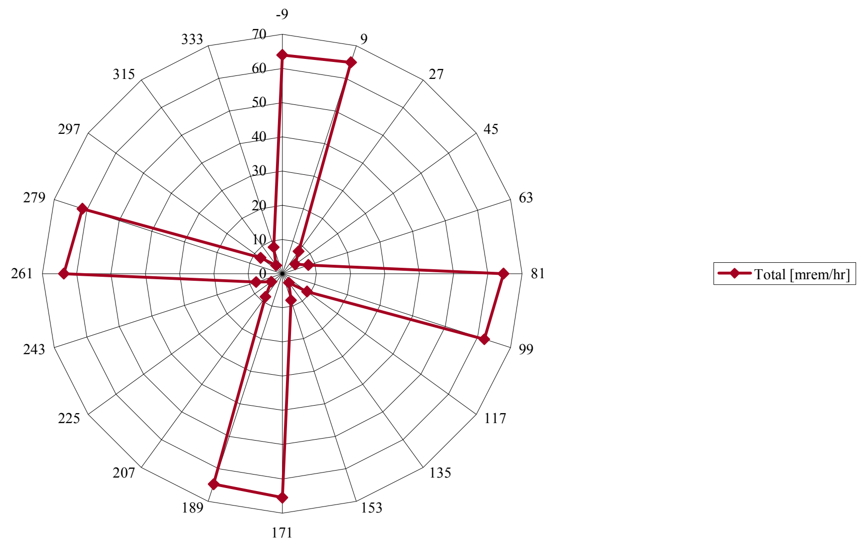


Figure 5.4-4 Vertical Concrete Cask Bottom Air Inlet Elevation Azimuthal Dose Rate Profile – PWR Fuel

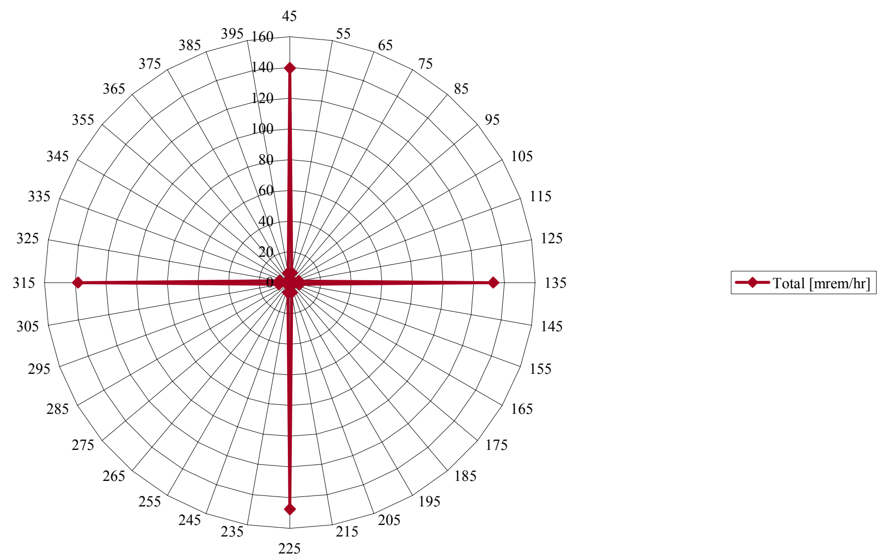


Figure 5.4-5 Vertical Concrete Cask Top Radial Surface Dose Rate Profile – Azimuthal Maximum – PWR Fuel

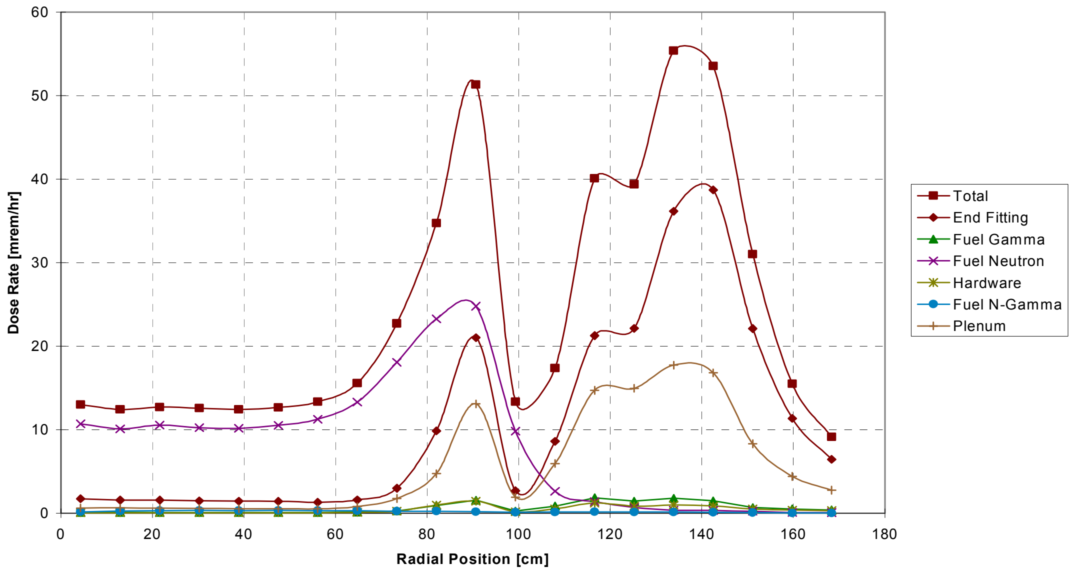


Figure 5.4-6 Vertical Concrete Cask Surface Dose Rate Profile by Source Component – Azimuthal Average – BWR Fuel

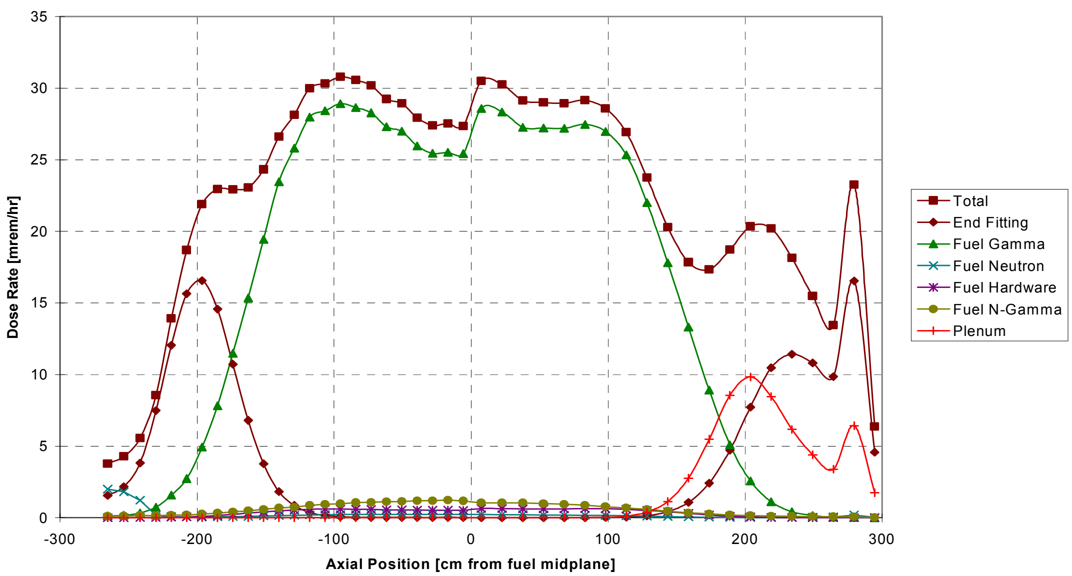


Figure 5.4-7 Vertical Concrete Cask Surface Dose Rate Profile at Various Distances from Cask – Azimuthal Average – BWR Fuel

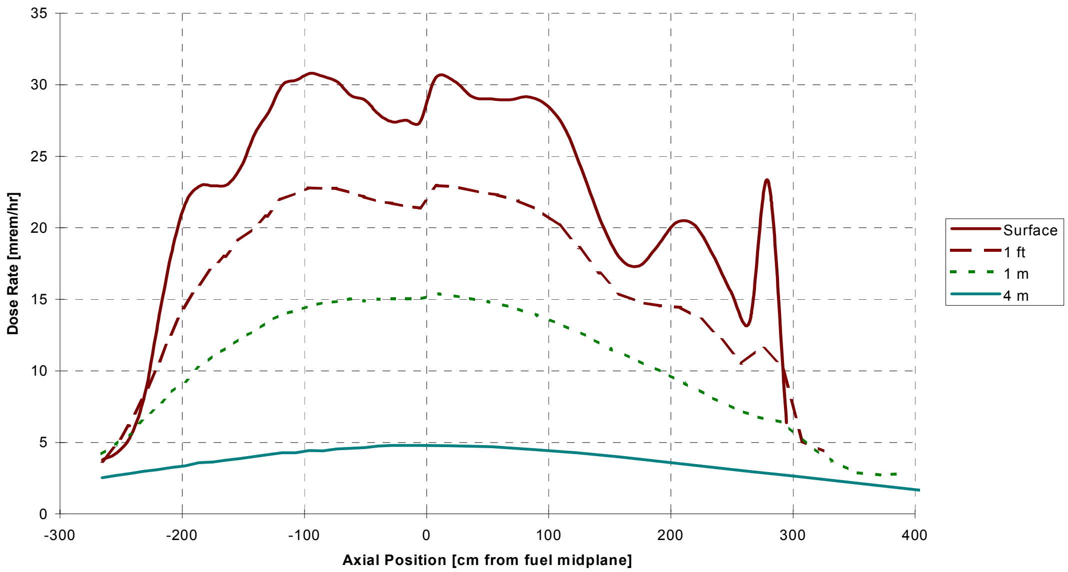


Figure 5.4-8 Vertical Concrete Cask Top Air Outlet Elevation Azimuthal Surface Dose Rate Profile – BWR Fuel

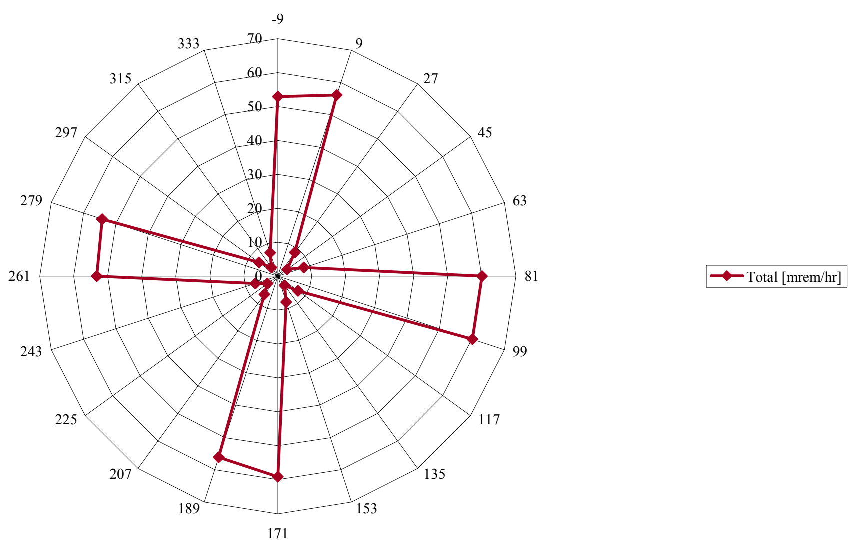


Figure 5.4-9 Vertical Concrete Cask Bottom Air Inlet Elevation Azimuthal Dose Rate Profile – BWR Fuel

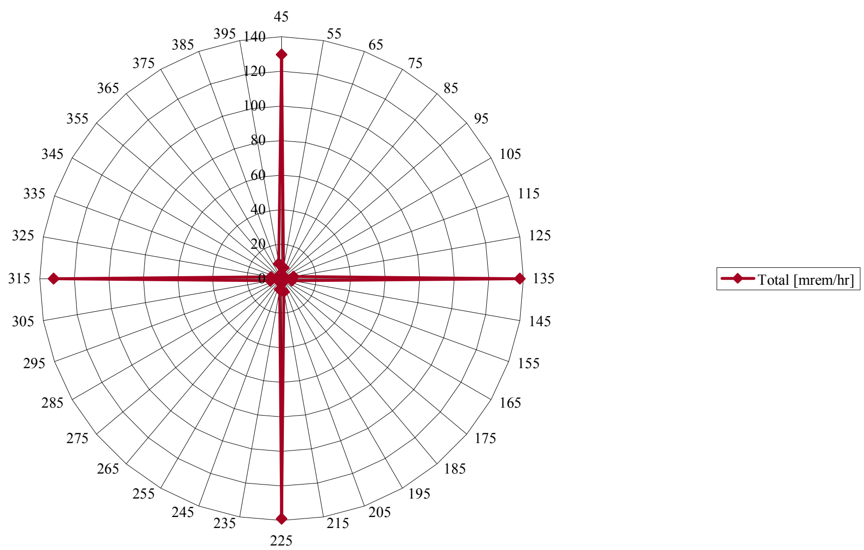


Figure 5.4-10 Vertical Concrete Cask Top Radial Surface Dose Rate Profile – Azimuthal Maximum – BWR Fuel

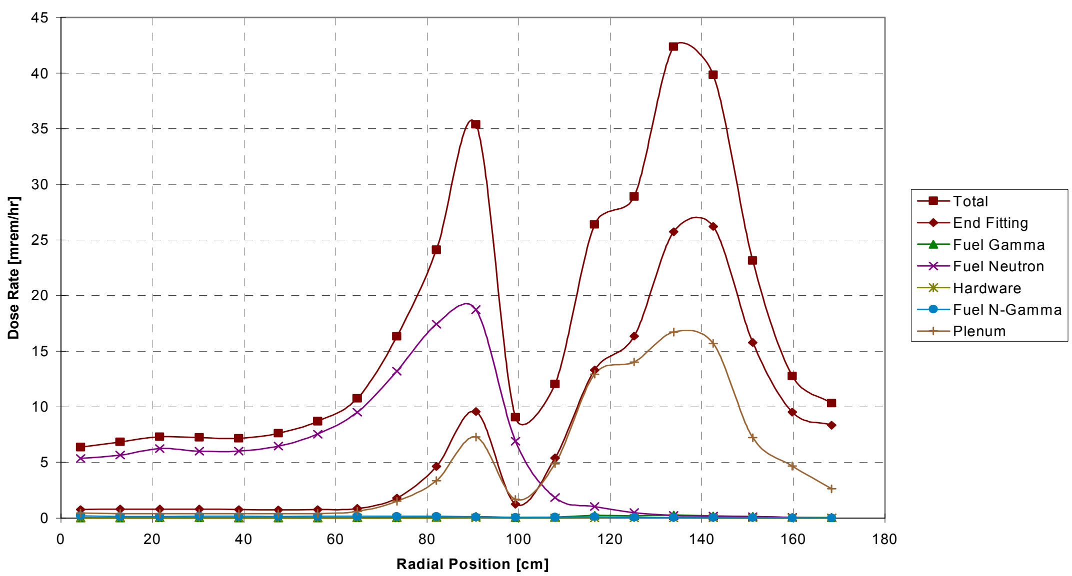


Figure 5.4-11 Standard Transfer Cask Axial Surface Dose Rate Profile – Dry Canister – PWR Fuel

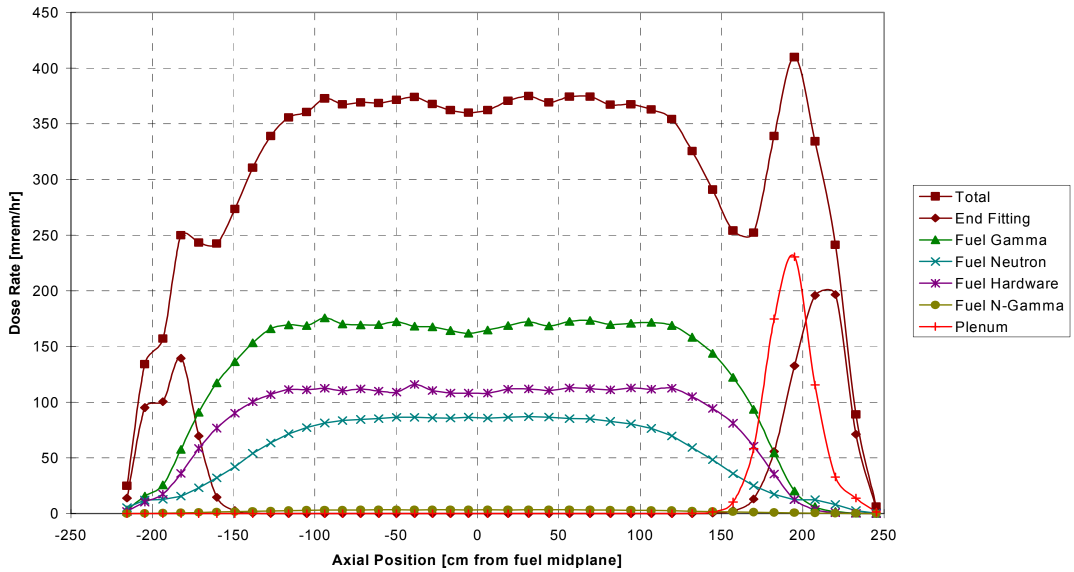


Figure 5.4-12 Standard Transfer Cask Axial Surface Dose Rate Profile – Wet Canister – PWR Fuel

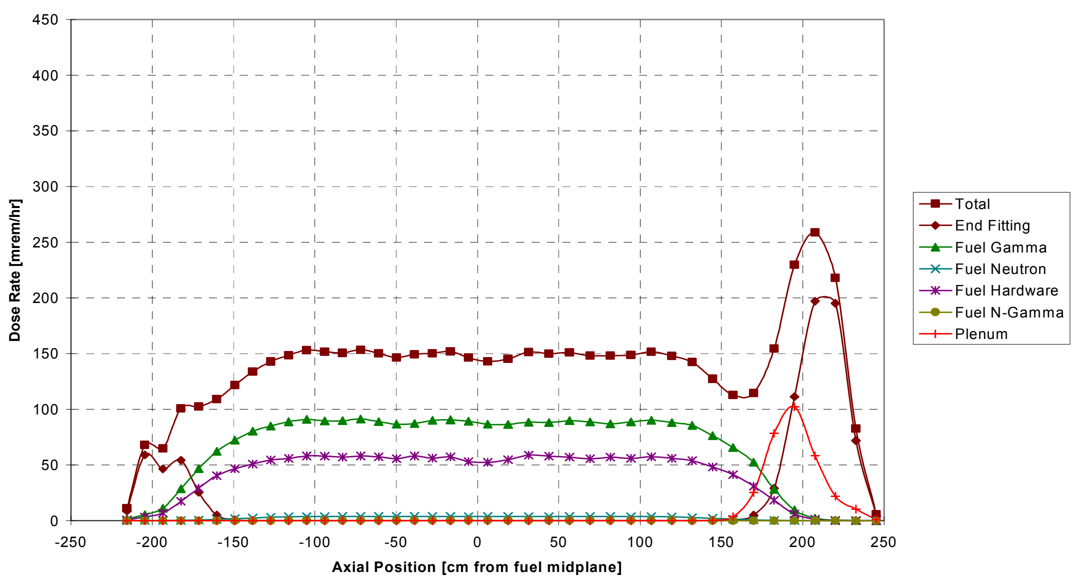


Figure 5.4-13 Standard Transfer Cask Axial Dose Rate Profile at Various Distances from Cask – Dry Canister – PWR Fuel

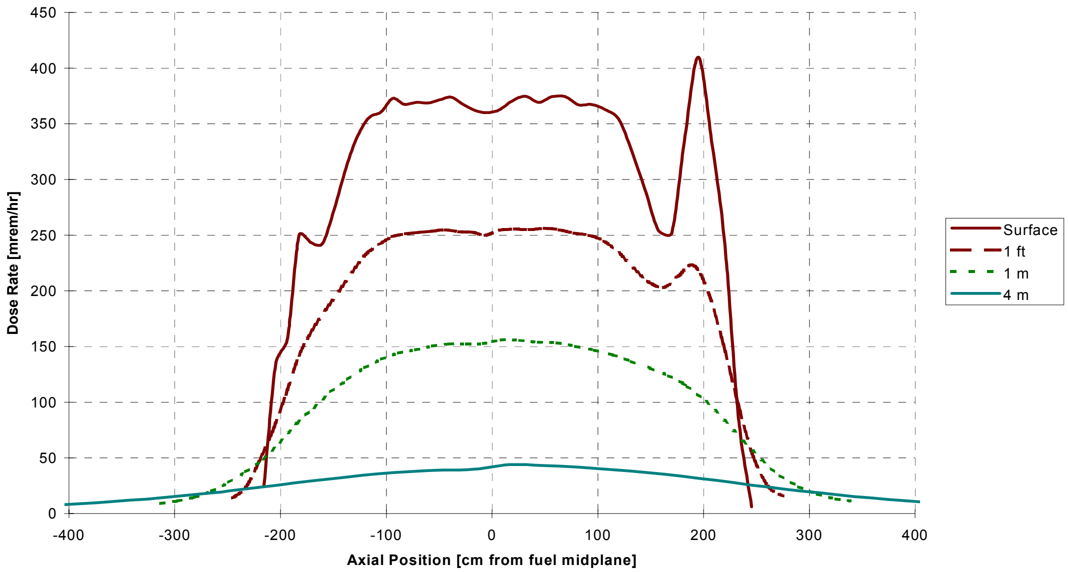


Figure 5.4-14 Standard Transfer Cask Axial Dose Rate Profile at Various Distances from Cask – Wet Canister – PWR Fuel

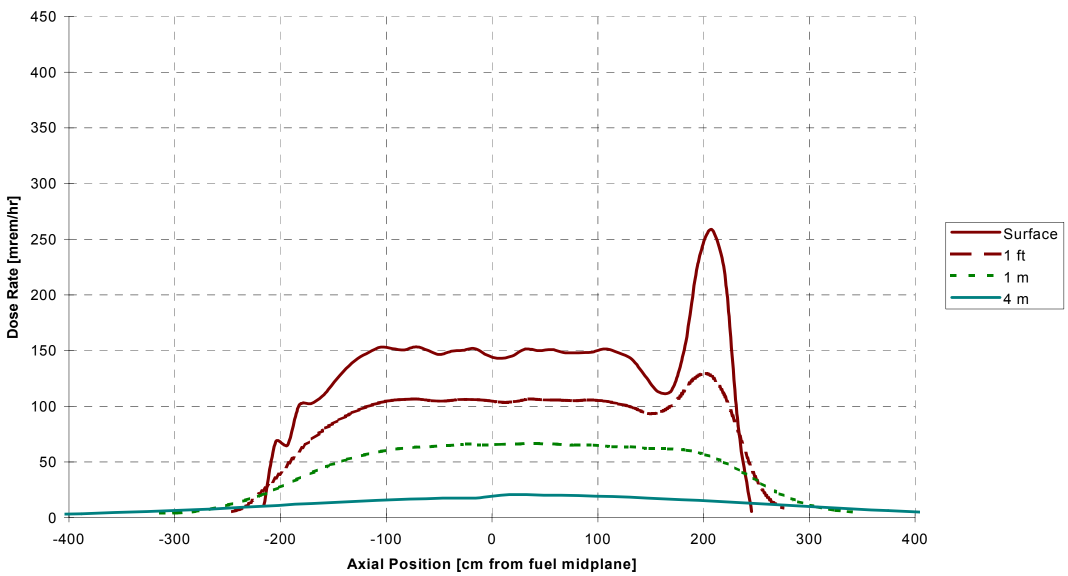


Figure 5.4-15 Standard Transfer Cask Top Radial Surface Dose Rate Profile – Shield Lid and Temporary Shield – Vent Port Covers Off – Wet Canister – PWR Fuel

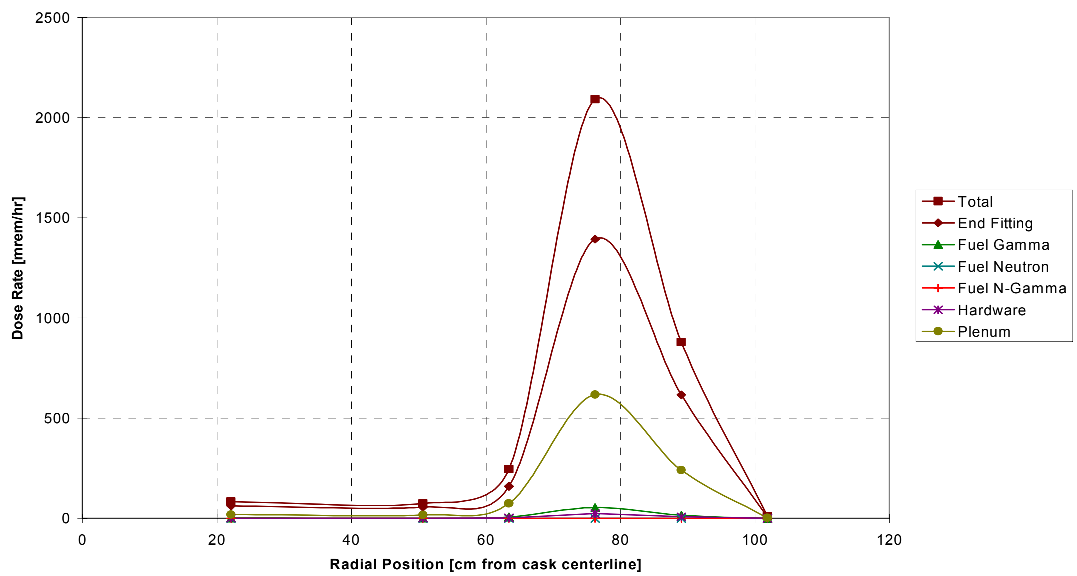


Figure 5.4-16 Standard Transfer Cask Top Radial Surface Dose Rate Profile – Shield Lid and Temporary Shield – Vent Port Covers On – Dry Canister – PWR Fuel

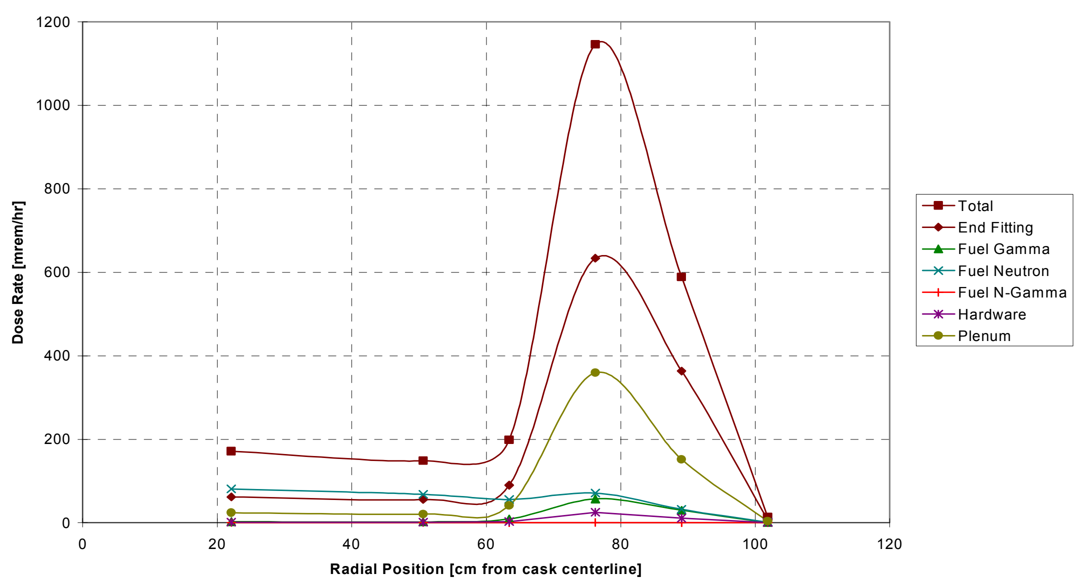


Figure 5.4-17 Standard Transfer Cask Top Radial Surface Dose Rate Profile – Shield Lid and Structural Lid – Dry Canister – PWR Fuel

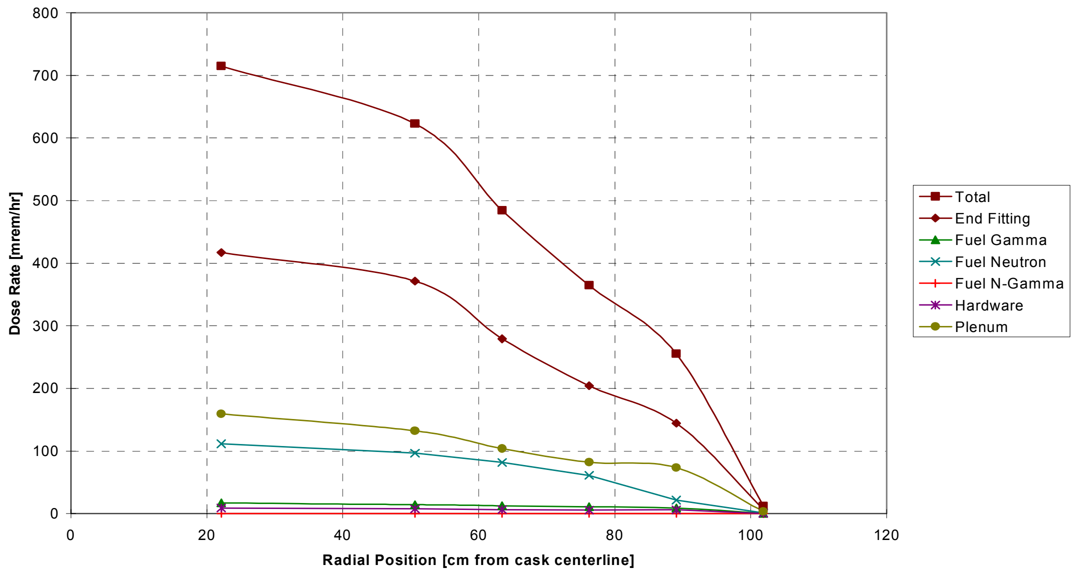


Figure 5.4-18 Standard Transfer Cask Bottom Radial Surface Dose Rate Profile – Dry Canister – PWR Fuel

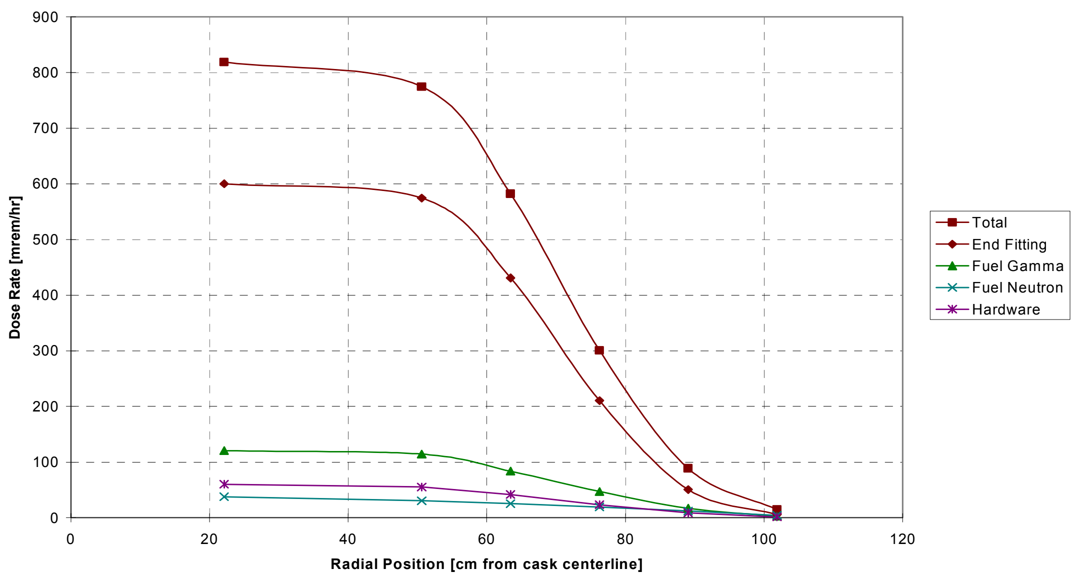


Figure 5.4-19 Standard Transfer Cask Bottom Radial Surface Dose Rate Profile – Wet Canister – PWR Fuel

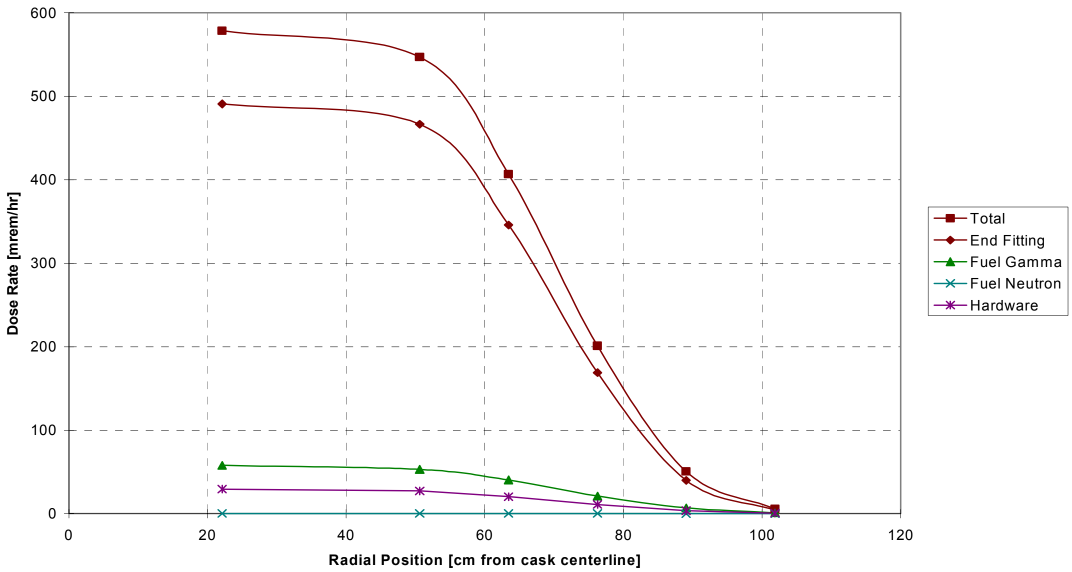


Figure 5.4-20 Standard Transfer Cask Axial Surface Dose Rate Profile – Dry Canister – BWR Fuel

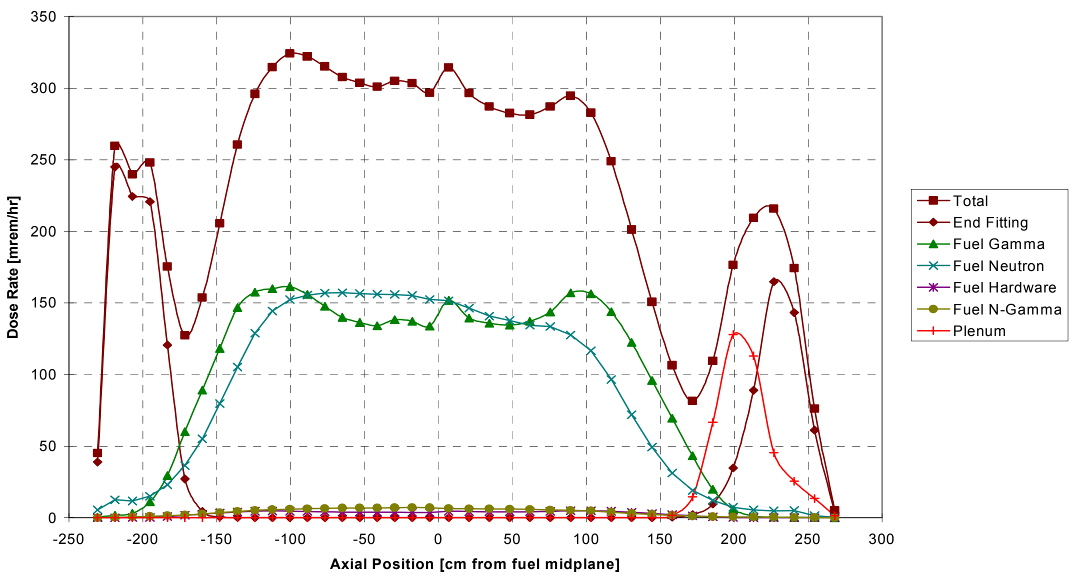


Figure 5.4-21 Standard Transfer Cask Axial Surface Dose Rate Profile – Wet Canister – BWR Fuel

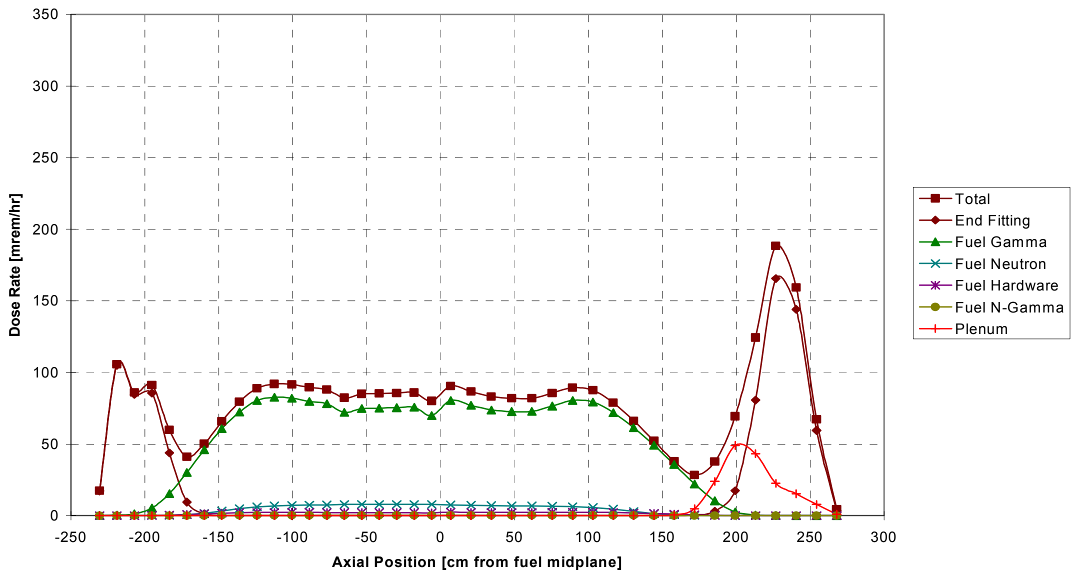


Figure 5.4-22 Standard Transfer Cask Axial Surface Dose Rate Profile at Various Distances From Cask – Dry Canister – BWR Fuel

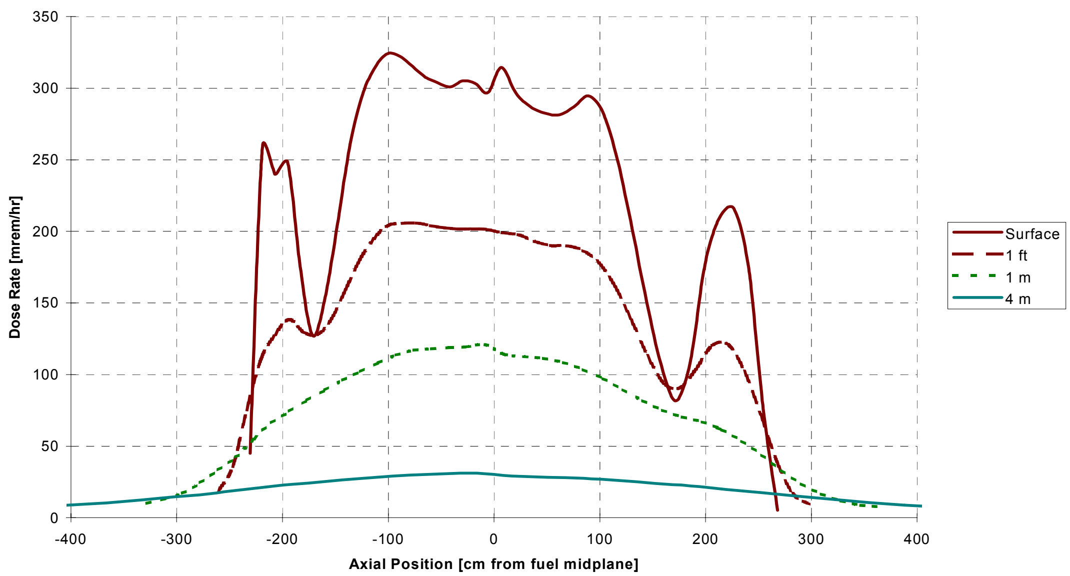


Figure 5.4-23 Standard Transfer Cask Axial Surface Dose Rate Profile at Various Distances From Cask – Wet Canister – BWR Fuel

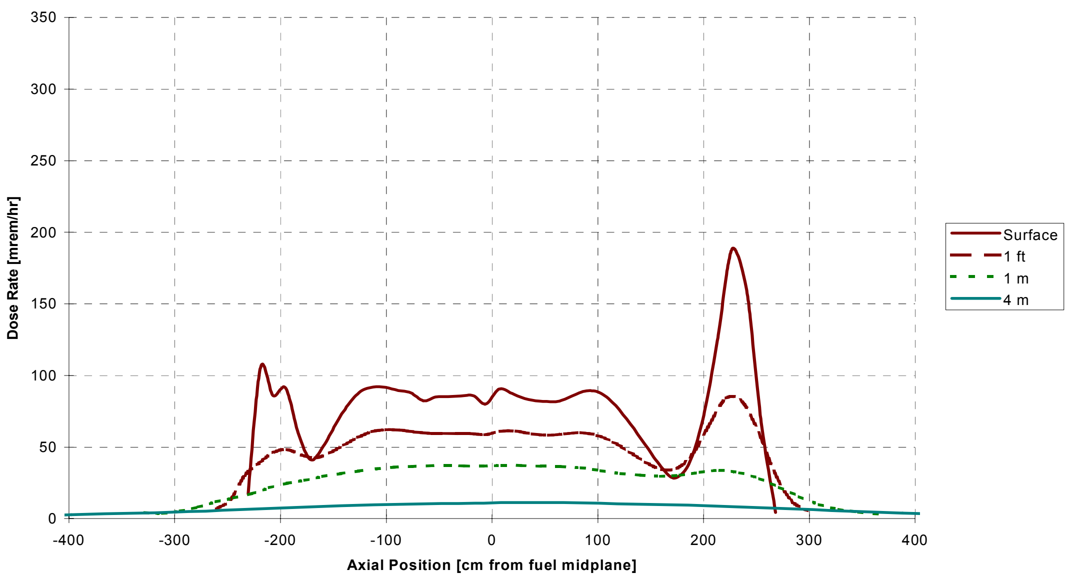


Figure 5.4-24 Standard Transfer Cask Top Radial Surface Dose Rate Profile – Shield Lid and Temporary Shield – Vent Port Covers Off – Wet Canister – BWR Fuel

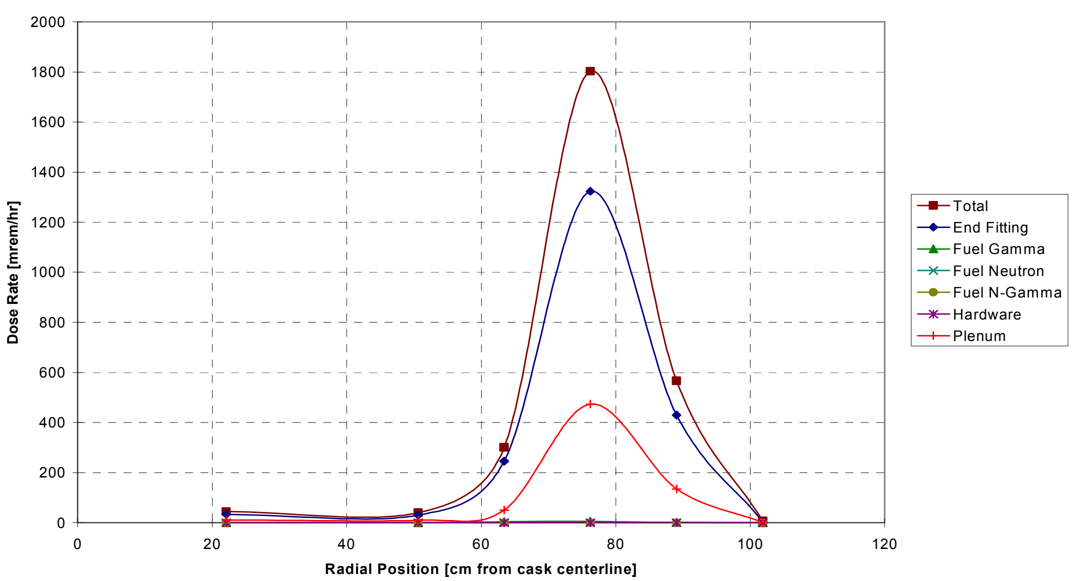


Figure 5.4-25 Standard Transfer Cask Top Radial Surface Dose Rate Profile – Shield Lid and Temporary Shield – Vent Port Covers On – Dry Canister – BWR Fuel

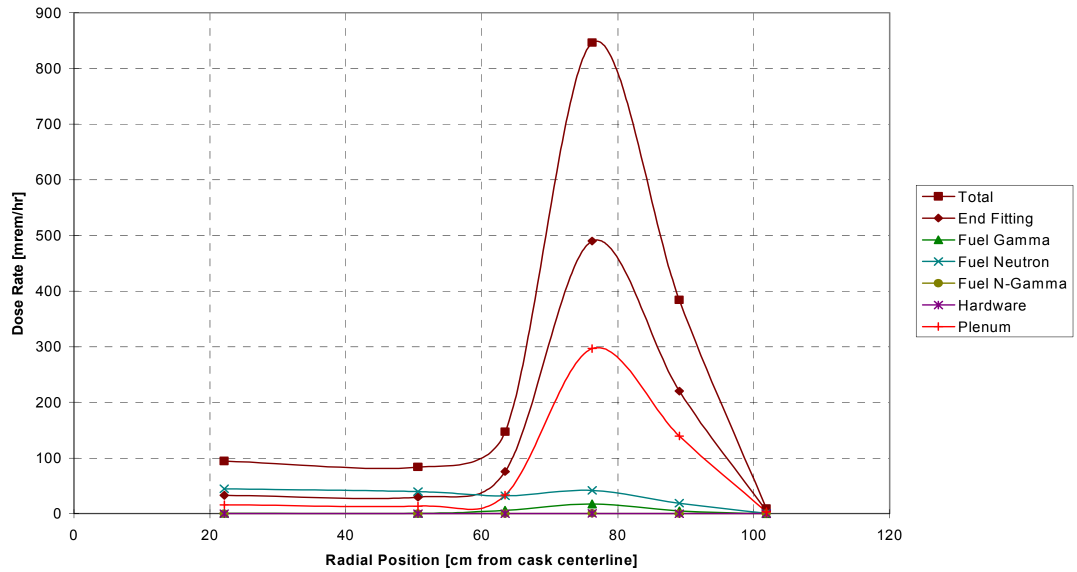


Figure 5.4-26 Standard Transfer Cask Top Radial Surface Dose Rate Profile – Shield Lid and Structural Lid – Dry Canister – BWR Fuel

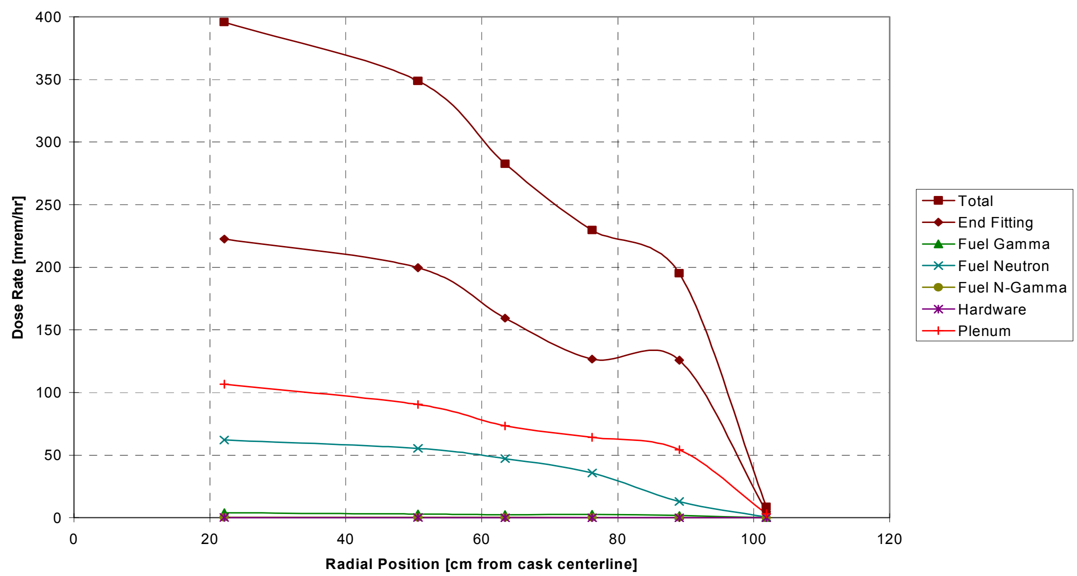


Figure 5.4-27 Standard Transfer Cask Bottom Radial Surface Dose Rate Profile – Dry Canister – BWR Fuel

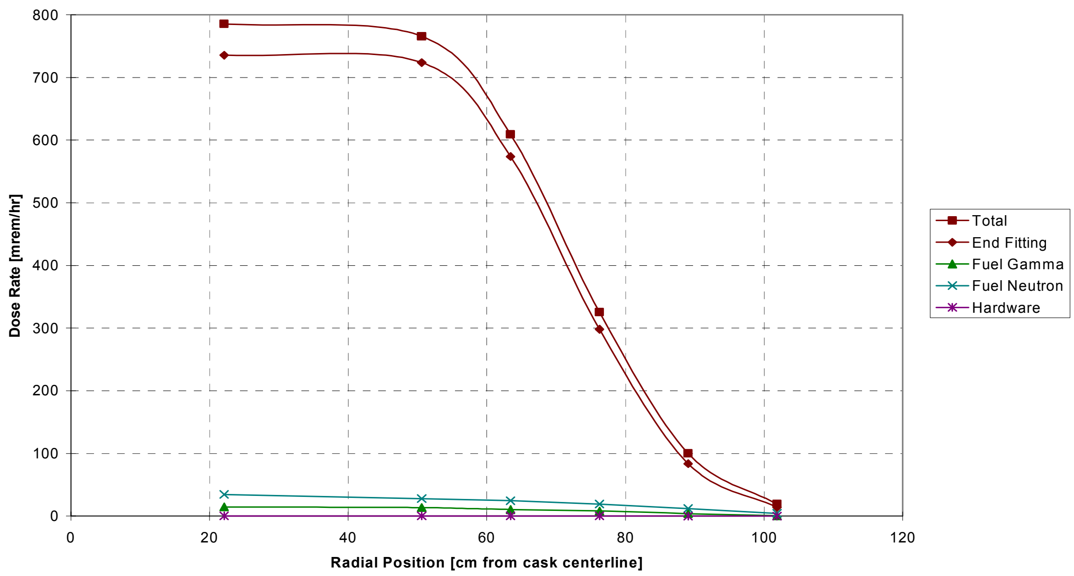


Figure 5.4-28 Standard Transfer Cask Bottom Radial Surface Dose Rate Profile – Wet Canister – BWR Fuel

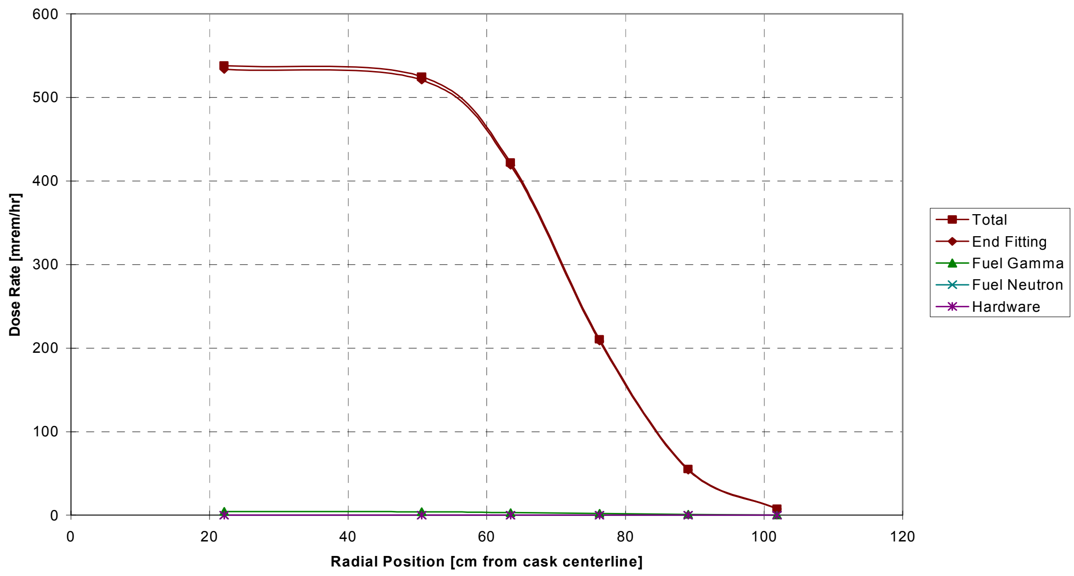


Table 5.4-1 ANSI Standard Neutron Flux-To-Dose Rate Factors

Group	(rem/hr)/(n/cm^2/sec)
1	1.49160E-04
2	1.44640E-04
3	1.27010E-04
4	1.28110E-04
5	1.29770E-04
6	1.02810E-04
7	5.11830E-05
8	1.23189E-05
9	3.83650E-06
10	3.72469E-06
11	4.01500E-06
12	4.29259E-06
13	4.47439E-06
14	4.56760E-06
15	4.55809E-06
16	4.51850E-06
17	4.48790E-06
18	4.46649E-06
19	4.43450E-06
20	4.32709E-06
21	4.19750E-06
22	4.09759E-06
23	3.83900E-06
24	3.67480E-06
25	3.67480E-06
26	3.67480E-06
27	3.67480E-06

Table 5.4-2 ANSI Standard Gamma Flux-To-Dose Rate Factors

Group	(rem/hr)/(γ/cm^2/sec)
1	8.77160E-06
2	7.47849E-06
3	6.37479E-06
4	5.41360E-06
5	4.62209E-06
6	3.95960E-06
7	3.46860E-06
8	3.01920E-06
9	2.62759E-06
10	2.20510E-06
11	1.83260E-06
12	1.52280E-06
13	1.17250E-06
14	8.75940E-07
15	6.30610E-07
16	3.83380E-07
17	2.66930E-07
18	9.34720E-07

Table 5.4-3 ANSI Standard Neutron Flux-To-Dose Rate Factors in MCBEND Group Structure

Group	Upper E [MeV]	Lower E [MeV]	Response [(mrem/hr)/(n/cm^2/sec)]
1	1.46E+01	1.36E+01	2.0533E-01
2	1.36E+01	1.25E+01	1.8999E-01
3	1.25E+01	1.13E+01	1.7250E-01
4	1.13E+01	1.00E+01	1.5399E-01
5	1.00E+01	8.25E+00	1.4700E-01
6	8.25E+00	7.00E+00	1.4700E-01
7	7.00E+00	6.07E+00	1.4929E-01
8	6.07E+00	4.72E+00	1.5348E-01
9	4.72E+00	3.68E+00	1.4580E-01
10	3.68E+00	2.87E+00	1.3478E-01
11	2.87E+00	1.74E+00	1.2657E-01
12	1.74E+00	6.40E-01	1.2570E-01
13	6.40E-01	3.90E-01	8.8205E-02
14	3.90E-01	1.10E-01	4.6004E-02
15	1.10E-01	6.74E-02	1.8108E-02
16	6.74E-02	2.48E-02	1.0774E-02
17	2.48E-02	9.12E-03	4.9057E-03
18	9.12E-03	2.95E-03	3.6168E-03
19	2.95E-03	9.61E-04	3.7152E-03
20	9.61E-04	3.54E-04	3.8611E-03
21	3.54E-04	1.66E-04	4.0252E-03
22	1.66E-04	4.81E-05	4.1919E-03
23	4.81E-05	1.60E-05	4.3795E-03
24	1.60E-05	4.00E-06	4.5200E-03
25	4.00E-06	1.50E-06	4.4895E-03
26	1.50E-06	5.50E-07	4.3924E-03
27	5.50E-07	7.09E-08	3.9685E-03
28	7.09E-08	0.00E+00	2.3759E-03

Table 5.4-4 ANSI Standard Gamma Flux-To-Dose Rate Factors in MCBEND Group Structure

Group	Upper E [MeV]	Lower E [MeV]	Response [(mrem/hr)/(γ/cm²/sec)]
1	1.40E+01	1.20E+01	1.1728E-02
2	1.20E+01	1.00E+01	1.0225E-02
3	1.00E+01	8.00E+00	8.7164E-03
4	8.00E+00	6.50E+00	7.4457E-03
5	6.50E+00	5.00E+00	6.3551E-03
6	5.00E+00	4.00E+00	5.3991E-03
7	4.00E+00	3.00E+00	4.5984E-03
8	3.00E+00	2.50E+00	3.9449E-03
9	2.50E+00	2.00E+00	3.4485E-03
10	2.00E+00	1.66E+00	2.9982E-03
11	1.66E+00	1.44E+00	2.6706E-03
12	1.44E+00	1.22E+00	2.3929E-03
13	1.22E+00	1.00E+00	2.1055E-03
14	1.00E+00	8.00E-01	1.8164E-03
15	8.00E-01	6.00E-01	1.5143E-03
16	6.00E-01	4.00E-01	1.1686E-03
17	4.00E-01	3.00E-01	8.6947E-04
18	3.00E-01	2.00E-01	6.2398E-04
19	2.00E-01	1.00E-01	3.8050E-04
20	1.00E-01	5.00E-02	2.7163E-04
21	5.00E-02	2.00E-02	5.8620E-04
22	2.00E-02	1.00E-02	2.3540E-03

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5.5 Minimum Allowable Cooling Time Evaluation for PWR and BWR Fuel

Sections 5.1 through 5.4 include the source term and shielding analyses for the design basis UMS® PWR and BWR assemblies with a burnup of 40 GWD/MTU and a 5-year cool time. The shielding evaluation design basis fuel assemblies source term are based on an initial minimum enrichment of 3.7 wt % ^{235}U for PWR and 3.25 wt % ^{235}U for BWR fuel assemblies. The source terms for the design basis assemblies represent a maximum heat load of 25.2 kW for the PWR cask and 24 kW for the BWR cask. The maximum allowable heat load for the UMS® storage system is 23 kW.

This section determines minimum cooling times for assembly average burnups ranging from 30 to 60 GWd/MTU for PWR assemblies and 30 to 45 GWd/MTU for BWR assemblies, with corresponding minimum initial enrichments from 1.9 wt % ^{235}U to 5.0 wt % ^{235}U. For each combination of initial enrichment and burnup, the minimum cooling times necessary to meet the maximum allowable decay heat, maximum transfer cask dose rate and maximum storage cask dose rate are determined. The listed minimum cooling times are the most limiting time required to meet either the canister maximum allowable heat load, the transfer cask design basis radial dose rates or storage cask design basis radial dose rates.

To address differences in the fissile material loading between assemblies, the assemblies are grouped by fuel pin array size. The BWR fuel types evaluated are 7×7, 8×8 and 9×9 assemblies and the PWR fuel types evaluated are 14×14, 15×15, 16×16 and 17×17 fuel assemblies.

5.5.1 Selection of Limiting PWR and BWR Fuel Types for Minimum Cooling Time Determination

The bounding PWR and BWR fuel assemblies are listed in Table 5.5-1. The selection of the limiting PWR and BWR assemblies is made based upon bounding maximum initial uranium loadings at 95% theoretical fuel density. Detailed PWR and BWR fuel characteristics, including the maximum MTU loadings, are documented in Table 6.2-1 and Table 6.2-2 for a wide range of PWR and BWR fuel assemblies.

Bounding uranium loadings produce the maximum heat loads and fuel radiation source terms. To ensure that fuel hardware such as grid spacers and burnable poison rods are fully considered in selecting the shielding limited fuel types the Westinghouse 15×15, GE 8×8-62 fuel rod and GE 8×8 60 fuel rod fuel assembly types are also evaluated.

5.5.2 Decay Heat Limit

The maximum allowable heat load, or decay heat limit as used in the context of this chapter, is based on the overall maximum decay heat limit of 23 kW. The maximum allowable heat load on a per assembly basis is 0.958 kW.

As documented in Section 5.4.1, the SAS2H sequence of SCALE 4.3 is used to determine source term magnitudes for each fuel assembly type, initial enrichment and burnup combination. Source term in this context implies both heat load and radiation sources for both fuel and activated hardware.

5.5.3 Storage Cask and Standard Transfer Cask Dose Rate Limits and Dose Calculation Method

Storage cask and standard transfer cask radial surface dose rates for the design basis assemblies are presented in Tables 5.1-1 through 5.1-4. The design basis radial storage and transfer cask (dry cavity) dose rates are used as an upper bound dose rate limit for any other fuel assembly type, burnup, and enrichment combination.

To avoid the significant effort required to prepare and execute hundreds of one-dimensional cases for all fuel configurations and burnups under consideration, a unique device is employed which permits the ready calculation of dose rates at a given location using a dose rate response function. The dose rate response function for a given source type at a given detector location is a collection of values, one for each energy group, each of which gives the contribution to the dose rate at the detector location from a unit source strength in that energy group. With this response function, the dose rate, d, at the corresponding detector location is determined for any given fuel type by vector multiplying the unnormalized source spectrum, f, by the response function, r:

$$d = r \cdot f$$

The dose rate response function is computed by solving a series of one-dimensional cases, one for each energy group, with a unit source strength in each energy group. In practice, the source strength is normalized to some large value (here, 10^{10}) in order to avoid numeric underflow in the calculation.

Sample response functions for the PWR and BWR storage casks and the standard transfer cask are listed in Table 5.5-3 and Table 5.5-4 for neutron and gamma sources, respectively. Only seven energy groups are presented for the fuel neutron source since the complete SAS2H neutron source is located in these energy groups.

With the dose rate response method a convenient and simple method for determining storage and transfer cask surface dose rates is available.

5.5.4 Minimum Allowable Cooling Time Determination

The following strategy is used to determine limiting cooling times for each combination of fuel type, initial enrichment, and burnup:

a) Determine decay heat and dose rate values at each cooling time step.
b) Interpolate in the resulting collection of data to find minimum cooling time required to meet each limiting value, decay heat and transfer and storage cask dose rate, individually.
c) Select the maximum of this collection of minimum required cooling times, rounded up to the next whole year, as the minimum required cooling time for this combination of burnup, enrichment and cooling time.

5.5.4.1 PWR and BWR Assembly Minimum Cooling Times

Minimum allowable cooling times are established for each of the fuel type, burnup, and enrichment combinations based on the cask decay heat limit of 23 kW. Listed in Table 5.5-2 is a comparison of one-dimensional limits to the corresponding three-dimensional dose rates, demonstrating that the dose rate limits applied in the one-dimensional analysis comply with the three-dimensional analysis results in Sections 5.1 and 5.4. A sample of the calculated cooling times required to reach each of the limits for Westinghouse 17×17 and GE 9×9 fuel assemblies at 40 GWD/MTU are shown in Tables 5.5-5 and 5.5-6, respectively. The identical calculation sequence is repeated for all the assembly types and burnups indicated in Section 5.5-1. The limiting cooling times are then collapsed to array size specific limiting values as listed in Table 5.5-7 and Table 5.5-8.

Table 5.5-1 Limiting PWR and BWR Fuel Types Based on Uranium Loading

Reactor	Array	Fuel Assembly
PWR	17×17	WE 17×17 Standard
PWR	16×16	CE 16×16 System 80
PWR	15×15	BW 15×15
PWR	14×14	WE 14×14
BWR	9×9	GE 9×9-79 Fuel Rods (GE 9×9-2L)
BWR	8×8	GE 8×8-63 Fuel Rods
BWR	7×7	GE 7×7

Table 5.5-2 Design Basis Assembly Dose Rate Limit (mrem/hr)

Configuration (Radial Dose Rates)	Neutron	Gamma	Hardware Gamma	1-D Total (mrem/hr)	3-D Total (mrem/hr)
PWR Storage	0.6	22.5	11.1	34.2	49
BWR Storage	0.9	16.5	0.1	17.6	31
PWR Transfer (dry)	68.1	127.4	82.4	277.8	~375
BWR Transfer (dry)	108.0	92.6	1.1	201.6	~320

Table 5.5-3 Radial Surface Response to Neutrons

Group	E_{avg} (MeV)	Storage Cask[1]		Standard Transfer Cask[1]	
		WE17×17	GE9×9-2L	WE17×17	GE9×9-2L
1	1.32E+01	1.4090E+07	1.4092E+07	1.6120E+09	1.5776E+09
2	4.72E+00	8.0067E+06	8.3652E+06	1.0621E+09	1.0579E+09
3	2.43E+00	6.9413E+06	7.4255E+06	9.9278E+08	1.0147E+09
4	1.63E+00	5.4135E+06	5.9542E+06	6.9518E+08	7.2993E+08
5	1.15E+00	4.7208E+06	5.2315E+06	5.1436E+08	5.4559E+08
6	6.50E-01	4.4771E+06	5.0154E+06	3.6579E+08	4.0159E+08
7	2.50E-01	3.3522E+06	3.8567E+06	1.0275E+08	1.1960E+08

1. mrem/hr per 10^{10} neutrons/second.

Table 5.5-4 Radial Surface Response to Gammas

Group	E_{avg} (MeV)	Storage Cask[1]		Standard Transfer Cask[1]	
		WE17×17	GE9×9-2L	WE17×17	GE9×9-2L
1	9.00E+00	2.1642E+05	2.0388E+05	8.8476E+04	8.4060E+04
2	7.25E+00	1.6963E+05	1.6008E+05	1.0748E+05	1.0204E+05
3	5.75E+00	1.1013E+05	1.0387E+05	1.1130E+05	1.0542E+05
4	4.50E+00	6.0237E+04	5.6658E+04	1.0110E+05	9.5390E+04
5	3.50E+00	2.8958E+04	2.7132E+04	8.0031E+04	7.5091E+04
6	2.75E+00	1.1556E+04	1.0761E+04	5.2075E+04	4.8507E+04
7	2.25E+00	4.8220E+03	4.4662E+03	2.9488E+04	2.7289E+04
8	1.83E+00	1.6449E+03	1.5139E+03	1.2645E+04	1.1617E+04
9	1.50E+00	5.2977E+02	4.8508E+02	4.3386E+03	3.9611E+03
10	1.17E+00	1.1402E+02	1.0395E+02	7.2955E+02	6.6298E+02
11	9.00E-01	1.6902E+01	1.5317E+01	4.6940E+01	4.2324E+01
12	7.00E-01	2.8772E+00	2.6068E+00	1.5996E+00	1.4442E+00
13	5.00E-01	2.5056E-01	2.2771E-01	8.5258E-04	7.7577E-04
14	3.50E-01	6.0028E-03	5.3421E-03	1.8977E-11	1.6941E-11
15	2.50E-01	2.1429E-04	1.8519E-04	1.1956E-32	1.0398E-32
16	1.50E-01	4.3116E-08	2.8984E-08	0.0000E+00	0.0000E+00
17	7.50E-02	2.5041E-35	2.2809E-36	0.0000E+00	0.0000E+00
18	3.00E-02	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

1. mrem/hr per 10^{10} γ/second.

Table 5.5-5 Westinghouse 17×17 Minimum Cooling Time Evaluation

Enrichment (wt % ^{235}U)	Minimum Cooling Time (Years)[1]				Active Constraint
	Decay Heat	Storage Dose	Transfer Dose	Limiting	
1.9	6.3	6.0	9.5	10	Transfer Dose
2.1	6.2	5.8	8.5	9	Transfer Dose
2.3	6.1	5.7	7.7	8	Transfer Dose
2.5	6	5.6	7	7	Transfer Dose
2.7	5.9	5.5	6.5	7	Transfer Dose
2.9	5.9	5.4	6.1	7	Transfer Dose
3.1	5.8	5.3	5.8	6	Decay Heat
3.3	5.8	5.2	5.5	6	Decay Heat
3.5	5.7	5.1	5.2	6	Decay Heat
3.7	5.7	5	5	6	Decay Heat
3.9	5.6	5	5	6	Decay Heat
4.1	5.6	5	5	6	Decay Heat
4.3	5.5	5	5	6	Decay Heat
4.5	5.5	5	5	6	Decay Heat
4.7	5.4	5	5	6	Decay Heat
4.9	5.4	5	5	6	Decay Heat

1. 40,000 MWD/MTU burnup.

Table 5.5-6 GE 9×9-2L Minimum Cooling Time Evaluation

Enrichment	Minimum Cooling Time (Years)[1]				Active
(wt % ^{235}U)	Decay Heat	Storage Dose	Transfer Dose	Limiting	Constraint
1.9	5.8	5.6	14.3	15	Transfer Dose
2.1	5.7	5.5	11.7	12	Transfer Dose
2.3	5.6	5.4	9.5	10	Transfer Dose
2.5	5.5	5.3	7.9	8	Transfer Dose
2.7	5.4	5.2	6.7	7	Transfer Dose
2.9	5.3	5.2	5.9	6	Transfer Dose
3.1	5.2	5.1	5.4	6	Transfer Dose
3.3	5.2	5	5	6	Decay Heat
3.5	5.1	5	5	6	Decay Heat
3.7	5	5	5	5	Decay Heat
3.9	5	5	5	5	Decay Heat
4.1	5	5	5	5	Decay Heat
4.3	5	5	5	5	Decay Heat
4.5	5	5	5	5	Decay Heat
4.7	5	5	5	5	Decay Heat
4.9	5	5	5	5	Decay Heat

1. 40,000 MWD/MTU burnup.

Table 5.5-7 Minimum Cooling Time Versus Burnup/Initial Enrichment for PWR Fuel

Minimum Initial Enrichment	**Assembly Average Burnup ≤30 GWd/MTU Minimum Cooling Time [years]**				**30< Assembly Average Burnup ≤35 GWd/MTU Minimum Cooling Time [years]**			
wt % ^{235}U (E)	**14×14**	**15×15**	**16×16**	**17×17**	**14×14**	**15×15**	**16×16**	**17×17**
1.9 ≤ E < 2.1	5	5	5	5	7	7	5	7
2.1 ≤ E < 2.3	5	5	5	5	7	6	5	6
2.3 ≤ E < 2.5	5	5	5	5	6	6	5	6
2.5 ≤ E < 2.7	5	5	5	5	6	6	5	6
2.7 ≤ E < 2.9	5	5	5	5	6	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5	5	5	5
3.7 ≤ E < 3.9	5	5	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5	5	5
E ≥ 4.9	5	5	5	5	5	5	5	5
Minimum Initial Enrichment	**35< Assembly Average Burnup ≤40 GWd/MTU Minimum Cooling Time [years]**				**40< Assembly Average Burnup ≤45 GWd/MTU Minimum Cooling Time [years]**			
wt % ^{235}U (E)	**14×14**	**15×15**	**16×16**	**17×17**	**14×14**	**15×15**	**16×16**	**17×17**
1.9 ≤ E < 2.1	10	10	7	10	15	15	11	15
2.1 ≤ E < 2.3	9	9	6	9	14	13	9	13
2.3 ≤ E < 2.5	8	8	6	8	12	12	8	12
2.5 ≤ E < 2.7	8	7	6	7	11	11	7	11
2.7 ≤ E < 2.9	7	7	6	7	10	10	7	10
2.9 ≤ E < 3.1	7	6	6	7	9	9	7	9
3.1 ≤ E < 3.3	6	6	6	6	9	8	7	8
3.3 ≤ E < 3.5	6	6	6	6	8	8	7	8
3.5 ≤ E < 3.7	6	6	6	6	7	8	7	7
3.7 ≤ E < 3.9	6	6	6	6	7	8	7	7
3.9 ≤ E < 4.1	6	6	6	6	7	7	7	7
4.1 ≤ E < 4.3	5	6	6	6	6	7	7	7
4.3 ≤ E < 4.5	5	6	6	6	6	7	7	7
4.5 ≤ E < 4.7	5	6	5	6	6	7	6	7
4.7 ≤ E < 4.9	5	6	5	6	6	7	6	7
E ≥ 4.9	5	6	5	6	6	7	6	7

Table 5.5-7 Minimum Cooling Time Versus Burnup/Initial Enrichment for PWR Fuel (continued)

Minimum Initial Enrichment	45< Assembly Average Burnup ≤50 GWd/MTU Minimum Cooling Time [years]				50< Assembly Average Burnup ≤55 GWd/MTU Minimum Cooling Time [years]			
wt % ^{235}U (E)	14×14	15×15	16×16	17×17	14×14	15×15	16×16	17×17
1.9 ≤ E < 2.1	21	21	18	21	27	27	25	27
2.1 ≤ E < 2.3	19	19	16	19	25	25	23	25
2.3 ≤ E < 2.5	17	17	14	17	23	24	21	24
2.5 ≤ E < 2.7	16	16	12	16	21	22	19	22
2.7 ≤ E < 2.9	14	14	11	14	20	20	17	20
2.9 ≤ E < 3.1	13	13	9	13	18	18	15	18
3.1 ≤ E < 3.3	12	12	9	12	17	17	13	17
3.3 ≤ E < 3.5	11	11	9	11	15	15	12	15
3.5 ≤ E < 3.7	10	10	8	10	14	14	11	14
3.7 ≤ E < 3.9	9	10	8	9	13	13	11	13
3.9 ≤ E < 4.1	9	10	8	9	12	13	11	12
4.1 ≤ E < 4.3	8	10	8	9	11	13	10	12
4.3 ≤ E < 4.5	8	9	8	9	10	13	10	12
4.5 ≤ E < 4.7	7	9	8	9	10	12	10	12
4.7 ≤ E < 4.9	7	9	8	9	9	12	10	12
E ≥ 4.9	7	9	8	9	9	12	10	11

Minimum Initial Enrichment	55< Assembly Average Burnup ≤60 GWd/MTU Minimum Cooling Time [years]							
wt % ^{235}U (E)	14×14	15×15	16×16	17×17				
1.9 ≤ E < 2.1	33	34	32	34				
2.1 ≤ E < 2.3	31	32	30	32				
2.3 ≤ E < 2.5	29	30	28	30				
2.5 ≤ E < 2.7	28	28	26	28				
2.7 ≤ E < 2.9	26	26	24	26				
2.9 ≤ E < 3.1	24	24	22	24				
3.1 ≤ E < 3.3	22	23	20	23				
3.3 ≤ E < 3.5	21	21	18	21				
3.5 ≤ E < 3.7	19	19	17	20				
3.7 ≤ E < 3.9	18	18	15	18				
3.9 ≤ E < 4.1	17	18	14	17				
4.1 ≤ E < 4.3	15	17	14	16				
4.3 ≤ E < 4.5	14	17	14	16				
4.5 ≤ E < 4.7	13	17	14	16				
4.7 ≤ E < 4.9	12	17	13	16				
E ≥ 4.9	12	16	13	15				

Table 5.5-8 Minimum Cooling Time Versus Burnup/Initial Enrichment for BWR Fuel

Minimum Initial Enrichment	Assembly Average Burnup ≤30 GWd/MTU Minimum Cooling Time [years]			30< Assembly Average Burnup ≤35 GWd/MTU Minimum Cooling Time [years]		
wt % ^{235}U (E)	7×7	8×8	9×9	7×7	8×8	9×9
1.9 ≤ E < 2.1	5	5	5	8	7	7
2.1 ≤ E < 2.3	5	5	5	6	6	6
2.3 ≤ E < 2.5	5	5	5	6	5	6
2.5 ≤ E < 2.7	5	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5	5
3.7 ≤ E < 3.9	5	5	5	5	5	5
3.9 ≤ E < 4.1	5	5	5	5	5	5
4.1 ≤ E < 4.3	5	5	5	5	5	5
4.3 ≤ E < 4.5	5	5	5	5	5	5
4.5 ≤ E < 4.7	5	5	5	5	5	5
4.7 ≤ E < 4.9	5	5	5	5	5	5
E ≥ 4.9	5	5	5	5	5	5

Minimum Initial Enrichment	35< Assembly Average Burnup ≤40 GWd/MTU Minimum Cooling Time [years]			40< Assembly Average Burnup ≤45 GWd/MTU Minimum Cooling Time [years]		
wt % ^{235}U (E)	7×7	8×8	9×9	7×7	8×8	9×9
1.9 ≤ E < 2.1	16	14	15	26	24	25
2.1 ≤ E < 2.3	13	12	12	23	21	22
2.3 ≤ E < 2.5	11	9	10	20	18	19
2.5 ≤ E < 2.7	9	8	8	18	16	17
2.7 ≤ E < 2.9	8	7	7	15	13	14
2.9 ≤ E < 3.1	7	6	6	13	11	12
3.1 ≤ E < 3.3	6	6	6	11	10	10
3.3 ≤ E < 3.5	6	5	6	9	8	9
3.5 ≤ E < 3.7	6	5	6	8	7	7
3.7 ≤ E < 3.9	6	5	5	7	6	7
3.9 ≤ E < 4.1	5	5	5	7	6	7
4.1 ≤ E < 4.3	5	5	5	7	6	6
4.3 ≤ E < 4.5	5	5	5	6	6	6
4.5 ≤ E < 4.7	5	5	5	6	6	6
4.7 ≤ E < 4.9	5	5	5	6	6	6
E ≥ 4.9	5	5	5	6	6	6

5.6 Shielding Evaluation for Site Specific Spent Fuel

This section presents the shielding evaluation for spent fuel configurations that are unique to specific reactor sites. These site specific fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, and testing programs intended to improve reactor operations. Site specific fuel configurations include standard fuel with inserted non fuel-bearing components, fuel assemblies with missing or replaced fuel rods or poison rods, fuel assemblies unique to the reactor design, fuel with a parameter that exceeds the design basis parameter, such as enrichment or burnup, consolidated fuel and fuel that is classified as damaged.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation of the configuration.

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5.6.1 Shielding Evaluation for Maine Yankee Site Specific Spent Fuel

This analysis considers both assembly fuel sources and sources from activated non-fuel material such as control element assemblies (CEA), in-core instrument (ICI) segments, and fuel assemblies containing activated stainless steel replacement (SSR) rods and other non-fuel material, including neutron sources. It considers the consolidated fuel, damaged fuel, and fuel debris present in the Maine Yankee spent fuel inventory, in addition to those fuel assemblies having a burnup between 45,000 and 50,000 MWD/MTU.

The Maine Yankee spent fuel inventory also contains fuel assemblies with hollow zirconium alloy tubes, removed fuel rods, axial blankets, poison rods, variable radial enrichment, and low enriched substitute rods. These components do not result in additional sources to be considered in shielding evaluations and are, therefore, enveloped by the standard fuel assembly evaluation. For shielding considerations of the variable radial enrichment assemblies, the planar-average enrichment is employed in determining minimum cool times. As described in Section 6.6.1.2.2, fuel assemblies with variable radial enrichment incorporate fuel rods that are enriched to one of two levels of enrichment. Fuel assemblies that also incorporate axial blankets are described in Section 6.6.1.2.3. Axial blankets consist of annular fuel pellets enriched to 2.6 wt % ^{235}U, used in the top and bottom 5% (≈ 7 inches) of the active fuel length. The remaining active fuel length of the fuel rod is enriched to one of two levels of enrichment incorporated in the fuel design.

5.6.1.1 Fuel Source Term Description

Maine Yankee utilized 14×14 array size fuel based on designs provided by Combustion Engineering, Westinghouse, and Exxon Nuclear. The previously analyzed Combustion Engineering CE 14×14 standard fuel design is selected as the design basis for this analysis because its uranium loading is the highest of the three vendor fuel types, based on a 0.3765-inch nominal fuel pellet diameter, a 137-inch active fuel length, and a 95% theoretical fuel density. This results in a fuel mass of 0.4037 MTU. This exceeds the maximum reported Maine Yankee fuel mass of 0.397 MTU and, therefore, produces bounding source terms. The SAS2H model of the CE 14×14 assembly (shown in Figure 5.6.1-1) at a nominal burnup of 40,000 MWD/MTU and initial enrichment of 3.7 wt % ^{235}U, is based on data provided in Table 2.1.1-1.

Source terms for various combinations of burnup and initial enrichment are computed by adjusting the SAS2H BURN parameter to model the desired burnup and specifying the initial enrichment in the Material Information Processor input for UO_2.

5.6.1.1.1 Control Element Assemblies (CEA)

For the CEA evaluation, the assumptions are:

1. The irradiated portion of the CEA assembly is limited to the CEA tips since during normal operation the elements are retracted from the core and only the tips are subject to significant neutron flux.
2. The CEA tips are defined as that portion present in the "Gas Plenum" neutron source region in the Characteristics Database (CDB) [10].
3. Material subject to activation in the CEA tips is limited to stainless steel, Inconel and Ag-In-Cd in the tip of the CEA absorber rods. Stainless steel and Inconel is assumed to have a concentration of 1.2 g/kg ^{59}Co. The CDB indicates that a total of 2.495 kg/CEA of this material is present in the Gas Plenum region of the core during operation. The Ag-In-Cd alloy present in the gas plenum region during core operation is approximately 80% silver and weighs 2.767 kg/CEA.
4. The irradiated CEA material is assumed to be present in the lower 8 inches of the active fuel region when inserted in the assembly. The location of the CEA source is based on the relative length of the fuel assembly and CEA rods and the insertion depth of the CEA spider into the top end-fitting.
5. The decay heat generated in the most limiting CEA at 5 years cool time is 2.16 W/kg of activated steel and inconel, and 3.11 W/kg of activated Ag-In-Cd. Although longer cool times are considered in this analysis for the fuel source term, this decay heat generation rate is conservatively used for all longer CEA cool times. For a cask fully loaded with fuel assemblies containing design basis CEAs, the additional heat generation due to the CEAs amounts to (2.16 W/kg × 2.495 kg/CEA + 3.11 × 2.767 kg/CEA)(24 CEA/cask) = 336 W/cask, which is conservatively rounded to 350 W/cask.

Since the activated portion of the CEA is present only in the lower 8 inches of the active fuel, an adjustment to the one-dimensional dose rate limit is derived based on detailed three-dimensional results obtained for the CE 14×14 fuel with and without a CEA present.

Table 5.6.1-1 shows the activation history for CEAs employed at Maine Yankee. Based on this data, individual source term calculations are performed for each CEA group, and a single bounding CEA description is determined based on the maximum computed source rate as of January 1, 2001. The bounding CEA description is based on CEA group "A1-A8," and the resulting CEA spectra at 5, 10 15, and 20 years cool time are shown in Table 5.6.1-2.

5.6.1.1.2 In-Core Instrument (ICI) Thimbles

Activation of ICI thimble material is determined by accumulating the hardware activation incurred during each cycle the ICI thimble is present in the reactor core. The ICI thimbles are first grouped according to exposure history as shown in Table 5.6.1-3. The cycle exposure data for each Maine Yankee cycle is shown in Table 5.6.1-4. With these data, the accumulated hardware source is obtained by summing the contributions made from each cycle of exposure. It is assumed that:

1. The average cycle exposure is sufficient to represent the ICI thimble exposure during each cycle.
2. Spectral differences between hardware source terms are insignificant.
3. The ICI thimble activated hardware source rate does not decrease after January 1, 2001.
4. The ICI thimble activated hardware spectrum is assumed to be identical to the fuel activated hardware spectrum in distribution, but not total source strength, i.e., the majority of the source is the result of ^{60}Co at a fixed spectrum.

The portion of the ICI thimble present in the active fuel region during reactor operation is composed entirely of zirconium alloy and receives no significant activation. The activated components of the ICI thimble are present in the upper end fitting region of the core, and the material is assumed to be irradiated at a flux factor of 0.1 consistent with the activation ratio used for upper end fitting hardware. A total mass of 0.664 kg/ICI thimble of activated material (assumed to be stainless steel with an initial ^{59}Co concentration of 1.2 g/kg) is modeled in the upper end fitting region.

The resulting total source rate as of January 1, 2001, for the activated components of each ICI thimble group are shown in Table 5.6.1-3. ICI Thimble Group J has the highest source rate (1.4940E+13 γ/sec), and this value is selected as the design basis for the loading table analysis. Note that for the purposes of determining the required cool time for a fuel assembly containing a ICI thimble, no further decay of the ICI thimble is considered after January 1, 2001.

5.6.1.1.3 Stainless Steel Replacement Rods

Maine Yankee fuel assemblies containing stainless steel replacement (SSR) rods are listed in Table 5.6.1-5. Note that for "N" and "R" numbered fuel assemblies, the SSR rods are only subject to exposure after the first fuel assembly cycle of irradiation. For "U" numbered assemblies, the assemblies saw no additional exposure after the rods were inserted. Hence, these "U" numbered assemblies are not further considered since their SSR rods received no activation.

The SSR rod is assumed to be solid stainless steel with the same dimensions as a fuel rod and with an initial ^{59}Co concentration of 1.2 g/kg. The SSR rod mass is 2.91 kg/SSR. Hardware gamma source terms are generated for each of the SSR rods in Table 5.6.1-5 based on the one or two cycle exposure seen by the stainless steel rods in question. This additional hardware source is then used to increase the existing hardware source of the assembly.

5.6.1.1.4 Consolidated Fuel

There are two consolidated fuel lattices. The lattices house fuel rods taken from assemblies as shown in Table 5.6.1-6. Each lattice presents a 17×17 array, with top and bottom end fittings connected by solid steel connector rods. No explicit source term analysis is conducted for the consolidated fuel lattices themselves, instead, an analysis is presented based on the source term computed for the fuel assemblies from which the contents are derived.

5.6.1.2 Model Specification

The one- and three-dimensional models described in Section 5.3 are employed in this analysis. No modifications are required to the models except for the substitution of CE 14×14 homogenized source descriptions. These homogenizations are shown in Tables 5.6.1-7 through 5.6.1-9.

5.6.1.3 Shielding Evaluation

The shielding evaluation consists of a loading table analysis of the CE 14×14 fuel following the methodology developed in Section 5.5 (Minimum Allowable Cooling Time Evaluation for PWR and BWR fuel). Fuel assemblies which include non-fuel hardware are addressed explicitly. The results of the analysis are loading tables which give the required cool time for a particular fuel configuration.

No restrictions are placed on the loading locations for any of the non-fuel assembly hardware components. This implies that a canister may contain up to 24 CEAs, 24 ICI thimbles, or 24 steel substitute rod assemblies or any combination thereof as long as the most limiting cool time is selected for any of the components in the canister. Neither CEAs or ICI thimbles may be placed into an assembly containing steel substitute rods that have received core exposure. ICI thimbles and CEAs may be inserted in fuel assemblies that also have hollow zirconium alloy tubes replacing burnable poison rods, solid steel rods replacing fuel rods provided there has been no reactor core exposure of the steel rods, fuel assemblies with fuel rods removed from the lattice, fuel assemblies with variable enrichment or low enrichment replacement fuel rods, or axial blanket fuel assemblies. Due to physical constraints, ICI thimbles and CEAs cannot be located in the same assembly.

5.6.1.4 Standard Fuel Source Term

Results are obtained, for CE 14×14 fuel with no additional non-fuel material included, by following the minimum allowable cooling time evaluation (loading table analysis) methodology developed in Section 5.5. CE 14×14 source terms at various combinations of initial enrichment and burnup are computed using the CE 14×14 SAS2H model described in Section 5.6.1.1.

Following the methodology developed in Section 5.5, one-dimensional shielding calculations are performed for CE 14×14 fuel region sources at various combinations of initial enrichment, burnup, and cool time. The resulting dose rate and source term data is interpolated to determine the cool time required for each combination of enrichment and burnup to decay below the design basis limiting values of dose and heat generation rate.

The resulting loading table for CE 14×14 fuel with no additional non-fuel material is shown in Table 5.6.1-10.

In addition to the standard fuel evaluation, a preferential loading strategy is analyzed. The preferential loading configuration relies on placing higher heat load fuel assemblies on the periphery of the basket than would be allowed with a uniform loading strategy. Peripheral loadings are evaluated with decay heats of up to 1.05 kW per peripheral assembly. To maintain the maximum allowable heat load per basket of 23 kW, the maximum allowable per assembly heat load in the interior location of the basket is reduced to compensate for the higher heat load peripheral elements. Burnup and cool time combinations for peripheral and interior assemblies are listed in Table 5.6.1-10 as a function of initial enrichment. The cool time column for peripheral element and interior assembly loading is indicated by the "P" and "I" indicators in the column headings.

5.6.1.4.1 Control Element Assemblies (CEA)

The result of the analysis is a set of loading tables for Maine Yankee fuel giving the cool time required for a fuel assembly with a specified burnup and enrichment combination to contain a design basis CEA with a cool time of 5, 10, 15, or 20 years. Fuel assemblies containing CEAs will be loaded into Class 2 canisters, which are slightly longer than the Class 1 canisters used for bare fuel assemblies. The additional length is required to accommodate the CEA, which is inserted in the top of the fuel assembly.

The approach taken is to compute downward adjustments to the design basis one-dimensional dose rate limiting value for the storage cask (as specified in Table 5.5-3) which ensures that the fuel sources have decayed adequately to cover the effect of the additional source added as a result of CEA containment. The adjustment is determined on the basis of a conservative comparison of three-dimensional shielding analysis results for the original Class 1 canister containing CE 14×14 fuel assemblies and the Class 2 canister containing either no CEA or CEAs cooled to 5, 10, 15, or 20 years. Results for CEA cool times longer than 20 years are bounded by the 20 year results.

Assuming design basis CE 14×14 fuel with a burnup of 40,000 MWD/MTU, 3.7 wt % ^{235}U enrichment and a 5-year cool time, the additional CEA source results in a localized peak near the bottom of the transfer cask that results in a surface dose rate that is less than 500 mrem/hr. Since this is comparable to the no-CEA case, it is not necessary to extend cool time of fuel assemblies with CEAs inserted to account for an increased transfer cask surface dose.

5.6.1.4.1.1 Establishment of Limiting Values

Since the additional activated material in the CEA analysis is assumed present in the lower 8 inches of the active fuel source region, the one-dimensional dose methodology is not appropriate to address the additional source term due to its small axial extent. The one-dimensional analysis is based on the response from the full-length fuel region source. To account for the additional source, the one-dimensional normal conditions dose rate limit is adjusted by an amount that ensures that the contribution from the additional activated material is bounded.

By adjusting the one-dimensional dose rate limit, we require the fuel to cool to a point where the decrease in fuel region dose rate matches the increased dose rate due to the additional CEA material. Hence, it is necessary to determine the amount by which the dose rate increases as a result of the added material. A one-dimensional calculation of this additional dose rate is not reasonable due to the small axial extent of the CEA source. One-dimensional buckling corrections are inaccurate for a cylindrical source where the ratio of height to diameter of the source is less than unity, as is the case here.

Instead, the additional contribution to dose rate due to the activated material is computed by a detailed three-dimensional shielding model. The model is based on the three-dimensional models described in Section 5.3. However, the fuel is modeled in a Class 2 canister since that canister will be used to store/transfer CEA-bearing assemblies.

The three-dimensional shielding evaluation is conducted for the CE 14×14 fuel at a burnup of 40,000 MWD/MTU and initial enrichment of 3.7 wt % ^{235}U. According to the cool time analysis conducted for PWR fuels in Table 5.6.1-10, this fuel will require a 5-year cool time before it is acceptable for transfer or storage in the UMS® vertical concrete cask. Hence, the 5-year cooled CE 14×14 at 40,000 MWD/MTU and 3.7 wt % ^{235}U initial enrichment provides the base case for the dose rate limit adjustment calculation.

Additional three-dimensional models are defined based on the base case fuel configuration in a Class 2 canister and either containing a design basis CEA assumed to be cooled for 5, 10, 15, or 20 years or containing no CEA at all (no CEA case below).

5.6.1.4.1.2 Three-Dimensional Model Results

Table 5.6.1-11 gives the three-dimensional UMS® Vertical Concrete Cask and transfer cask bottom model results for each case. Only the bottom model is considered because the top model is not sensitive to changes in the CEA description. The parameter Delta shown in the table is the difference between the base case maximum (from Table 5.5-2 for the storage cask) dose rate and the value computed for each remaining case. This quantity is directly applied to the one-dimensional design basis normal conditions dose rate limit, as specified in Table 5.6.1-11 for the storage cask to determine a modified limiting value applicable to each CEA decay case. The resulting dose rate limits are shown in the "Limit" column of the table.

Note that direct application of the "Delta" to the one-dimensional dose rate limit is somewhat conservative. The three-dimensional maximum dose rate results are significantly higher than the one-dimensional results, hence a given difference between three-dimensional results represents a larger percentage of the corresponding one-dimensional results.

Also note that the dose rate delta for the "No-CEA" case in Table 5.6.1-11 is zero. Unlike the UMS® transport cask, where a spacer positions the canister in the cask, the UMS® standard transfer and storage casks are extended to accommodate the longer Class 2 canister. These cask extensions maintain the spacing of the fuel assembly with respect to the points of minimum shielding in the bottom cask model, and thereby result in identical cask bottom half dose rates for fuel assemblies in Class 1 and Class 2 canisters.

5.6.1.4.1.3 Decay Heat Limits

As discussed in Section 5.6.1.1.1, the additional decay heat associated with a full cask of CEAs is conservatively taken as 0.35 kW/cask. This additional heat load is accounted for by reducing the fuel assembly decay heat limit to 22.65 kW/cask.

5.6.1.4.1.4 Loading Table Analysis

With the adjusted one-dimensional dose and heat generation rate limits established above, the loading table analysis proceeds following the methodology developed in Section 5.5. Each combination of initial enrichment and burnup is analyzed to determine the minimum required cool time in order for an assembly to either 1) contain a design basis CEA cooled 5, 10, 15, or 20 years or 2) to be present in a Class 2 canister with no CEA inserted. The resulting cool times are shown in Table 5.6.1-12.

5.6.1.4.2 In-Core Instrument (ICI) Thimbles

The loading table analysis of the in-core instrument thimble follows the same methodology as that developed above for activated CEA hardware. The activated portion of the ICI thimbles is present in the upper end fitting region when loaded into a host fuel assembly. Since the source region is outside the fuel region, direct application of the one-dimensional loading table analysis is not possible. Instead, as in the CEA case, the approach is to identify a conservative adjustment to the one-dimensional dose rate limit, thereby forcing the fuel to cool a longer time in order to offset the additional dose from the ICI thimble.

5.6.1.4.2.1 Establishment of Limiting Values

Decay heat from the activated ICI thimble is insignificant (< 0.05 kW/cask), so no adjustment to the decay heat limit is employed. The one-dimensional normal conditions dose rate limit is adjusted in a manner identical to that employed in the CEA analysis. Two configurations of the CE 14×14 Class 1 canister are analyzed in full three-dimensional detail. The first configuration is the base case CE 14×14 fuel at 40,000 MWD/MTU, 3.7 wt % enrichment, and 5-year cool time. The second configuration is identical to the base case with the addition of the source term for 24 ICI thimbles to the upper end fitting source region. No credit is taken for the self-shielding effectiveness of the added material. The base case upper end fitting total source strength is 2.031E+14 γ/sec. The design basis ICI thimble source strength is determined in Section 5.6.1.1.2 to be 2.988E+13 γ/sec. The ICI thimble activated hardware spectrum is assumed to be identical to the fuel activated hardware spectrum.

The results of the two three-dimensional cases for the storage cask are shown in Table 5.6.1-13. The addition of ICI thimble sources to the top end-fitting source region has no discernable impact on the storage cask or transfer cask surface average dose rate. A 2 mrem/hr increase in the storage cask air outlet dose was calculated due to the additional source (against a 70 mrem/hr dose rate for the CE 14×14 40,000 MWD/MTU burned, 5-year cooled base case). Due to the location and size of the air outlets in relation to the storage cask total surface the small increase in dose will have no impact on the on-site and off-site exposures. The presence of the ICI thimble source in the transfer cask also has no discernable effect on the computed cask maximum surface dose rate. A slight increase in surface dose rate is observed in the vicinity of the added source, but the computed dose rate at this location is less than the maximum surface dose rate.

5.6.1.4.2.2 Loading Table Analysis

Since no significant dose rate changes occurred due to the addition of the ICI thimble source no revised loading tables are provided. The standard and preferential assembly loading table (Table 5.6.1-10) may be used for determining minimum assembly cool time for loading with or without an ICI thimble assembly.

5.6.1.4.3 Stainless Steel Replacement Rods

Maine Yankee fuel assemblies containing stainless steel replacement (SSR) rods are listed in Table 5.6.1-5. Note that for "N" and "R" numbered fuel assemblies, the SSR rods are only subject to exposure after the first cycle of irradiation of the fuel assembly. For "U" numbered assemblies, the assemblies saw no additional exposure after the rods were inserted. Hence, these "U" numbered assemblies are not further considered since the SSR rods received no activation.

The SSR rod is assumed to be solid stainless steel with the same dimensions as a fuel rod and a mass of 2.91 kg/SSR.

Based on the exposure data provided, SAS2H source calculations are performed explicitly for each SSR-bearing fuel assembly, which received additional exposure. Each fuel assembly is modeled at its initial enrichment (rounded down to the nearest enrichment level equal to a modeled enrichment value) and cycle length parameters are computed to achieve the required burnups as indicated in Table 5.6.1-5. The resulting SSR source strengths as of January 1, 2001, are shown in Table 5.6.1-14.

A cool time analysis is conducted for each assembly containing irradiated SSR rods. The activated SSR material is treated explicitly by adding the source directly to the fuel hardware source term. Hence, no adjustment to the one-dimensional dose rate limits is required as in previous analyses involving added non-fuel sources. The results of the cool time analysis for each assembly are shown in Table 5.6.1-14.

The desired final fuel loading time for the Maine Yankee spent fuel inventory is August 2002. As such two assemblies fall outside the standard loading curve. Employing the preferential loading pattern, permitting 1.05 kW per peripheral assembly, reduces the minimum cool time based on thermal constraints to 6 years. The storage cask dose rate constraint is satisfied for the preferentially loaded assembles after 5 years cooling. Recognizing that only two of the assemblies in the Maine Yankee spent fuel inventory, R439 and R444, require peripheral loading, the transfer cask dose rate limit is not applied for these two assemblies. Since the dose rate comparisons are made on the basis of an assumed fuel cask of assemblies, the transfer cask dose rate limit is unnecessarily restrictive.

5.6.1.4.4 Consolidated Fuel

There are two consolidated fuel lattices intended for storage (and transfer) in the Universal Storage Cask. The lattices house fuel rods taken from assemblies as shown in Table 5.6.1-6. This fuel has decayed for over twenty years and does not represent a significant shielding issue.

A limiting cool time analysis is conducted by identifying a fuel assembly description analyzed in the loading table analysis that bounds the parameters of the fuel rods in the consolidated fuel lattices. The parameters of those fuel rods are shown in Table 5.6.1-15. The CE 14×14 fuel at 30,000 MWd/MTU and 1.9 wt % ^{235}U enrichment represents a bounding assembly type, since it has a significantly higher burnup and a lower enrichment than the original assemblies. This fuel requires 6-year cool time before it can be loaded in the storage or transfer cask as shown in Table 5.6.1-10. The consolidated fuel has been cooled for at least 24 years. For container CN-1 lattice, one can immediately conclude that dose rates are bounded by the limiting fuel.

However, the CN-10 lattice contains significantly more fuel rods than an undamaged assembly. Neglecting the mitigating effects of additional self-shielding, this situation is addressed by comparing the radiation source strength of the limiting fuel at six- and 24-year cool time. Conservatively assuming that all fuel rods present in CN-10 are at the limiting conditions of 30,000 MWd/MTU and 1.9 wt % ^{235}U, the ratio of the source rate in the CN-10 to the source rate in the limiting fuel assembly is shown to be less than one for each source type in Table 5.6.1-16. For each source type, the ratio is computed as:

Ratio = (Num Rods in CN-10)(Source Rate at 24 Yr) / (Num Rods in F/A)(Source Rate at 6 Yr)

Hence, CN-10 is also bounded by the limiting case as of January 1, 2001.

5.6.1.4.5 Damaged Fuel and Fuel Debris

The Maine Yankee spent fuel inventory includes fuel assemblies containing damaged fuel rods and fuel debris. Damaged fuel rods and fuel debris will be placed into one of the two configurations of the screened Maine Yankee Fuel Can prior to loading in the UMS® basket. Maine Yankee fuel cans are restricted to loading into one of the four corner basket locations. The damaged fuel mass cannot exceed the fuel mass of 100% of an undamaged fuel assembly. Damaged fuel rods may be loaded in the can with undamaged rods.

To approximate the effect of collapsed fuel inside the Maine Yankee fuel can, a three-dimensional shielding analysis was performed doubling the source magnitude and material density in the four corner basket locations. Conservatively, the screened can itself is not included in the shielding model. As expected, the increased self-shielding of the collapsed fuel material minimizes the dose rate increase resulting from the source term density doubling. Based on a cask average surface dose rate of less than 40 mrem/hr under normal operating conditions, no significant increases in personnel exposures are expected as a result of the collapsed fuel material.

Where no collapse of the fuel rods occurs, the analysis presented for the undamaged fuel assemblies bounds that of the damaged fuel rods. Since the additional shielding provided by the screened canister is not being credited by this approach, the actual expected dose rates will be lower for the transportable storage canisters loaded with damaged fuel. For cases in which the Maine Yankee fuel can holds fuel rods from multiple assemblies, the minimum cool time for the rods containing the most restrictive enrichment and burnup combination is applied to the contents of the entire can.

Fuel debris must be placed into a rod structure prior to loading into the screened canister. Once the fuel debris is configured in a rod structure it can be treated from a shielding perspective identical to the damaged fuel rods.

5.6.1.4.6 Additional Nonfuel and Neutron Source Material

The additional nonfuel material consists of:

1. Three plutonium-beryllium (Pu-Be) neutron sources, two irradiated and one unirradiated.
2. Two antimony-beryllium (Sb-Be) neutron sources, both irradiated.

3. Control element assembly (CEA) fingertips.
4. ICI string segment.

The five neutron sources will be inserted into the center guide tubes of five different assemblies and loaded into Class 1 canisters. These five assemblies will be loaded in five different canisters. This requirement is conservative since the shielding evaluation shows that only the irradiated Pu-Be sources must be placed in different canisters and that the remaining sources may be loaded in any remaining corner positions of the canister. The CEA fingertips and ICI string segment may be inserted into one or more assemblies and loaded into a Class 2 canister to accommodate a CEA flow plug to close the guide tubes with the added hardware. These fuel assemblies must be loaded in corner positions in the fuel basket.

The characterization of the additional non-fuel hardware is provided in Tables 5.6.1-17 and 5.6.1-18. The data is divided into two separate categories:

1. Non-neutron producing radiation sources – this category includes the CEA fingertips, ICI string, and the Sb-Be neutron sources (the neutron production rate of these is negligible).
2. Neutron producing radiation sources – this category includes the two irradiated and one unirradiated Pu-Be neutron sources.

The masses of ^{238}Pu and ^{239}Pu given for the unirradiated Pu-Be source are used in conjunction with the delivery date of May 1972 to generate source terms.

The neutron sources have an additional source component due to the irradiation of the stainless steel rod encasing the source. The quantity of irradiated steel is taken as 10 lbs. (4.54 kg) for this evaluation.

From the waste characterization, it is apparent that the Sb-Be sources already include the contribution of irradiated stainless steel. Therefore, only the Pu-Be irradiated stainless steel requires activation. The hardware source spectra for the irradiated Pu-Be sources are based on the Maine Yankee exposure history shown in Table 5.6.1-4. The combined Pu-Be assembly hardware irradiation for Cycles 1-13 is shown in Table 5.6.1-19 at a cool time of five years from 1/1/1997.

The waste characterization sources given in Tables 5.6.1-17 and 5.6.1-18 are used to generate source terms using ORIGEN-S [9]. For the non-neutron producing sources, the total curie content is assigned to ^{60}Co to provide bounding source terms. Also, only one Sb-Be spectrum is produced, based on the higher curie content source. For the neutron producing sources, the given curie contents are used for irradiated sources, whereas the plutonium masses are used for the unirradiated Pu-Be source.

Based on the loading plan, there are two areas of application of both spectra and dose rates. The CEA fingertips and the ICI string segment will be loaded into one assembly. Therefore, the gamma spectra of these items are summed and only one gamma spectrum is used to calculate the dose rates due to this loaded assembly. If these items are loaded into separate fuel assemblies, the source term is lower. Each of the five neutron sources will be loaded into a separate assembly, and the spectra are presented accordingly. The single assembly spectra for the inserted hardware items are presented in Table 5.6.1-20. The startup source spectra are presented in Table 5.6.1-21.

Dose rates are calculated by simply groupwise multiplying the spectra and CE 14×14 dose rate response functions and adjusting by a factor of 24/(10E+10 × 5.6193E+06) to remove the volume component and the calculation scaling factor. Dose rates are presented in Tables 5.6.1-22 through 5.6.1-24 and show the minimal dose rate contribution due to the inclusion of the additional non-fuel material.

Figure 5.6.1-1 SAS2H Model Input File – CE 14 × 14

```
=SAS2H      PARM=(HALT03,SKIPSHIPDATA)
CE 14 x 14 3.7 W/O U235, 45000 MWD/MTU 12.0-22.0 YEAR COOLING
27GROUPNDF4 LATTICECELL
UO2        1 0.950 900 92235 3.7 92238 96.3 END
ZIRCALLOY  2 1.0 620 END
H2O        3 DEN=0.725 1.0 580 END
ARBM-BORMOD 0.725 1 1 0 0 5000 100 3 550.0E-6 580 END
END COMP
SQUAREPITCH 1.4732 0.9563 1 3 1.1176 2 0.9754 0 END
NPIN=176 FUEL=347.98 NCYC=3 NLIB=1 PRIN=6 LIGH=5
INPL=1 NUMH=20 NUMI=0 ORTU=0.5588 SRTU=0.49285 END
POWER=13.065 BURN=463.5350 DOWN=60.0 END
POWER=13.065 BURN=463.5350 DOWN=60.0 END
POWER=13.065 BURN=463.5350 DOWN=1461.00 END
```

Table 5.6.1-1 Maine Yankee CEA Exposure History by Group

CEA Group	First Cycle	Last Cycle	Maximum Exposure (MWD/MTU)	Number of Cycles	Exposure Per Cycle (MWD/MTU)	Cool Time as of 1/1/2001 (y)
A1-A8	7	15	60239	9	6693	4
B1-B5	9	15	48909	7	6987	4
C1-C11, C13-C15	10	15	44315	6	7386	4
D1-D15	11	15	35283	5	7057	4
E1-E17, GN, *78, 101, 102, 138-153	12	15	29367	4	7342	4
F1,F2	13	15	18663	3	6221	4
4A	12	12	9786	1	9786	8
C12	10	12	24309	3	8103	8
NA	1	11	75444	11	6859	10
1-69	1	8	53258	8	6657	15

Note: The asterisk is added to CEA 78* to distinguish it from the original CEA 78.

Table 5.6.1-2 Maine Yankee CEA Hardware Spectra - 5, 10, 15 and 20 Years Cool Time

Energy Group	**5 yr (γ / sec)**	**10 yr (γ / sec)**	**15 yr (γ / sec)**	**20 yr (γ / sec)**
1	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
3	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
4	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
5	1.3479E-04	4.4697E-06	1.4822E-07	4.9154E-09
6	7.1467E+06	2.6384E+06	1.3598E+06	7.0431E+05
7	4.0337E+09	1.6979E+09	8.7691E+08	4.5422E+08
8	3.7246E+10	2.3434E+08	1.4804E+06	1.5188E+04
9	1.8642E+14	7.1649E+13	3.6955E+13	1.9142E+13
10	4.8840E+14	2.5265E+14	1.3086E+14	6.7790E+13
11	1.3804E+14	9.4554E+11	4.7779E+10	3.7897E+10
12	1.1469E+15	9.3808E+14	9.1172E+14	8.8714E+14
13	4.3885E+14	4.2316E+14	4.1174E+14	4.0065E+14
14	9.1526E+11	5.5505E+11	5.2913E+11	5.0949E+11
15	1.2039E+12	8.4093E+11	8.0140E+11	7.6939E+11
16	3.8479E+12	2.9855E+12	2.7489E+12	2.5803E+12
17	5.1828E+13	4.4134E+13	4.2118E+13	4.0659E+13
18	3.4899E+14	2.7741E+14	2.6393E+14	2.5520E+14
Steel/Inc Source Rate	6.3886E+14	3.2951E+14	1.7066E+14	8.8413E+13
Ag-In-Cd Source Rate	2.1666E+15	1.6829E+15	1.6308E+15	1.5861E+15
Total Source Rate	2.8055E+15	2.0124E+15	1.8014E+15	1.6745E+15
SFA	5.6110E+15	4.0249E+15	3.6029E+15	3.3490E+15

Table 5.6.1-3 Maine Yankee ICI Thimble Exposure History and Source Rate by Group

Group	Quantity	Cycles Exposed	Number of Cycles	Total Source [γ/sec]
A	41	1, 1A, 2	3	9.1881E+11
B	1	1	1	2.3775E+11
C	2	1, 1A	2	3.6244E+11
D	1	1A, 2	2	6.8106E+11
E	3	2	1	5.5637E+11
F	15	3 thru 11, 13	10	1.1695E+13
G	12	3 thru 11, 14	10	1.2126E+13
H	12	3 thru 11, 15	10	1.1454E+13
I	3	3 thru 9,14,15	9	1.1309E+13
J	2	10 thru 15	6	1.4940E+13
K	1	10 thru 12	3	6.1296E+12
L	25	12 thru 15	4	1.1491E+13
M	17	12	1	2.6801E+12
N	3	13 thru 15	3	8.8105E+12

Table 5.6.1-4 Maine Yankee Core Exposure History by Cycle of Operation

Cycle	Discharge Date	Cycle Burnup [MWD/MTU]	Core Average Enrichment [wt %]
1	6/29/74	10367	2.44
1A	5/2/75	4492	2.30
2	4/9/77	17365	2.45
3	7/14/78	11105	2.59
4	1/11/80	10500	2.84
5	5/8/81	10799	2.98
6	9/24/82	11585	3.01
7	3/31/84	12483	3.10
8	8/17/85	12504	3.20
9	3/28/87	14424	3.29
10	10/15/88	12675	3.36
11	4/7/90	13786	3.50
12	2/14/92	15364	3.62
13	7/30/93	13668	3.68
14	1/14/95	13075	3.75
15	12/6/96	7859	3.76

Table 5.6.1-5 Burnup of Maine Yankee Fuel Assemblies with Stainless Steel Replacement Rods

Assembly Number	1st Cycle	2nd Cycle	3rd Cycle	1st Cycle Burnup[1]	2nd Cycle Burnup[1]	3rd Cycle Burnup[1]	Number of SSR Rods
N420	9	10	11	16,428	13,467	11,893	3
N842	9	10	-	18,420	13,885	0	1
N868	9	10	11	18,622	13,386	4,919	1
R032	12	13	14	16,464	15,386	12,168	1
R439	12	13	14	20,371	14,779	11,685	1
R444	12	13	14	20,371	14,779	11,685	4
U01	15	-	-	7,339	0	0	1
U05	15	-	-	7,339	0	0	1
U16	15	-	-	10,598	0	0	1
U37	15	-	-	9,005	0	0	1
U51	15	-	-	8,288	0	0	1
U60	15	-	-	8,288	0	0	6

1. MWD/MTU.

Table 5.6.1-6 Contents of Maine Yankee Consolidated Fuel Lattices CN-1 and CN-10

Consolidated Fuel Lattice	Original Fuel Assembly	Number of Rods	Actual Burnup [MWD/MTU]	Initial Enrichment [wt %]
CN-1	EF0039	172	5150	1.929
CN-10	EF0045	176	17150	1.953
	EF0046	107	17150	1.953

Table 5.6.1-7 Maine Yankee CE 14 × 14 Homogenized Fuel Region Isotopic Composition

Isotope	CE 14 × 14 [atom/b-cm]
ALUMINUM	2.05114E-03
BORON-10	1.90898E-04
BORON-11	7.68387E-04
CARBON-12	2.39821E-04
CHROMIUM(SS304)	7.19369E-04
IRON(SS304)	2.4501E-03
MANGANESE	7.16674E-05
NICKEL(SS304)	3.18674E-04
OXYGEN-16	8.72597E-03
URANIUM-234	2.39964E-07
URANIUM-235	3.14135E-05
URANIUM-238	4.33133E-03
ZIRCALLOY	3.06324E-03

Table 5.6.1-8 Isotopic Compositions of Maine Yankee CE 14 × 14 Fuel Assembly Non-Fuel Source Regions

Isotope	Upper Plenum [atom/b-cm]	Upper End Fit [atom/b-cm]	Lower End Fit [atom/b-cm]
CHROMIUM(SS304)	1.59190E-03	1.89910E-03	3.08125E-03
MANGANESE	1.58594E-04	1.89199E-04	3.06971E-04
IRON(SS304)	5.42166E-03	6.46791E-03	1.04941E-02
NICKEL(SS304)	7.05196E-04	8.41284E-04	1.36497E-03
ZIRCALLOY	3.22036E-03	–	–

Table 5.6.1-9 Isotopic Compositions of Maine Yankee CE 14 × 14 Canister Annular Region Materials (One-Dimensional Analysis Only)

Isotope	Fuel Annulus [atom/b-cm]	Upper Plenum Annulus [atom/b-cm]	Upper End Fit Annulus [atom/b-cm]	Lower End Fit Annulus [atom/b-cm]
ALUMINUM	5.96817E-03	–	–	–
CHROMIUM(SS304)	1.77895E-03	9.31065E-04	2.53529E-03	4.13797E-03
MANGANESE	1.77228E-04	9.27577E-05	2.52579E-04	4.12247E-04
IRON(SS304)	6.05870E-03	3.1710E-03	8.63463E-03	1.40930E-02
NICKEL(SS304)	7.88057E-04	4.12453E-04	1.12311E-03	1.83308E-03

Table 5.6.1-10 Loading Table for Maine Yankee CE 14 × 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable

Enrichment	Burnup ≤ 30 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	**Preferential (I)[2]**	**Preferential (P)[3]**
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	**30 < Burnup ≤ 35 GWD/MTU – Minimum Cool Time [years] for**		
	Standard[1]	**Preferential (I)[2]**	**Preferential (P)[3]**
1.9 ≤ E < 2.1	5	5	5
2.1 ≤ E < 2.3	5	5	5
2.3 ≤ E < 2.5	5	5	5
2.5 ≤ E < 2.7	5	5	5
2.7 ≤ E < 2.9	5	5	5
2.9 ≤ E < 3.1	5	5	5
3.1 ≤ E < 3.3	5	5	5
3.3 ≤ E < 3.5	5	5	5
3.5 ≤ E < 3.7	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5
Enrichment	**35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time [years] for**		
	Standard[1]	**Preferential (I)[2]**	**Preferential (P)[3]**
1.9 ≤ E < 2.1	7	7	5
2.1 ≤ E < 2.3	6	6	5
2.3 ≤ E < 2.5	6	6	5
2.5 ≤ E < 2.7	5	6	5
2.7 ≤ E < 2.9	5	6	5
2.9 ≤ E < 3.1	5	6	5
3.1 ≤ E < 3.3	5	6	5
3.3 ≤ E < 3.5	5	6	5
3.5 ≤ E < 3.7	5	6	5
3.7 ≤ E ≤ 4.2	5	6	5

1. “Standard” loading pattern: allowable decay heat = 0.958 kW per assembly
2. “Preferential” loading pattern, interior basket locations: allowable heat decay = 0.867 kW per assembly
3. “Preferential” loading pattern, periphery basket locations: allowable heat decay = 1.05 kW per assembly

Table 5.6.1-10 Loading Table for Maine Yankee CE 14 × 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable (Continued)

Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time [years] for		
	Standard[1]	**Preferential (I)**[2]	**Preferential (P)**[3]
1.9 ≤ E < 2.1	11	11	6
2.1 ≤ E < 2.3	9	9	6
2.3 ≤ E < 2.5	8	8	6
2.5 ≤ E < 2.7	7	7	6
2.7 ≤ E < 2.9	7	7	6
2.9 ≤ E < 3.1	6	7	6
3.1 ≤ E < 3.3	6	7	5
3.3 ≤ E < 3.5	6	7	5
3.5 ≤ E < 3.7	6	7	5
3.7 ≤ E ≤ 4.2	6	7	5
Enrichment	**45 < Burnup ≤ 50 GWD/MTU - Minimum Cool Time [years] for**		
	Standard[1]	**Preferential (I)**[2]	**Preferential (P)**[3]
1.9 ≤ E < 2.1	Not allowed	Not allowed	7
2.1 ≤ E < 2.3	Not allowed	Not allowed	7
2.3 ≤ E < 2.5	Not allowed	Not allowed	7
2.5 ≤ E < 2.7	Not allowed	Not allowed	7
2.7 ≤ E < 2.9	Not allowed	Not allowed	7
2.9 ≤ E < 3.1	Not allowed	Not allowed	7
3.1 ≤ E < 3.3	Not allowed	Not allowed	7
3.3 ≤ E < 3.5	Not allowed	Not allowed	6
3.5 ≤ E < 3.7	Not allowed	Not allowed	6
3.7 ≤ E ≤ 4.2	Not allowed	Not allowed	6

1. "Standard" loading pattern: allowable decay heat = 0.958 kW per assembly
2. "Preferential" loading pattern, interior basket locations: allowable heat decay = .0.867 kW per assembly
3. "Preferential" loading pattern, periphery basket locations: allowable heat decay = 1.05 kW per assembly

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Table 5.6.1-11 Three-Dimensional Shielding Analysis Results for Various Maine Yankee CEA Configurations Establishing One-Dimensional Dose Rate Limits for Loading Table Analysis

CEA Cool Time [years]	**Dose Rate [mrem/hr]**	**FSD**	**Delta [mrem/hr]**	**Limit [mrem/hr]**
Class 1 Result	32.0	0.85%	-	34.2
No CEA	32.0	0.85%	-0.0	34.2
05y	43.8	0.59%	-11.8	22.4
10y	33.1	0.69%	-1.1	33.1
15y	32.0	0.85%	-0.0	34.2
20y	32.0	0.85%	-0.0	34.2

Table 5.6.1-12 Loading Table for Maine Yankee CE 14 × 14 Fuel Containing CEA Cooled to Indicated Time

Enrichment	≤ 30 GWD/MTU Burnup - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	7	7	7	7	7
2.1 ≤ E < 2.3	6	6	6	6	6
2.3 ≤ E < 2.5	6	6	6	6	6
2.5 ≤ E < 2.7	5	6	5	5	5
2.7 ≤ E < 2.9	5	6	5	5	5
2.9 ≤ E < 3.1	5	6	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 2)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	11	11	11	11	11
2.1 ≤ E < 2.3	9	9	9	9	9
2.3 ≤ E < 2.5	8	8	8	8	8
2.5 ≤ E < 2.7	7	7	7	7	7
2.7 ≤ E < 2.9	7	7	7	7	7
2.9 ≤ E < 3.1	6	6	6	6	6
3.1 ≤ E < 3.3	6	6	6	6	6
3.3 ≤ E < 3.5	6	6	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6
3.7 ≤ E ≤ 4.2	6	6	6	6	6

Note: The NoCEA (Class 2) column is provided for comparison. Fuel assemblies without a CEA insert may not be loaded in a Class 2 canister.

Table 5.6.1-13 Establishment of Dose Rate Limit for Maine Yankee ICI Thimble Analysis

	Top Model	
Case	Rate (mrem/hr)	FSD
No ICI Thimble	33.3	1.4%
4 Year Cooled ICI Thimble	33.3	1.4%
Delta	0.0	

Table 5.6.1-14 Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods

Assembly Number	Burnup [MWD/MTU]	Enrichment [wt %]	SSR Source [g/s/assy]	Cool Time [years]	Earliest Loadable	Loading Configuration
N420	45,000	3.3	2.1602E+13	6	Jan 2001	Standard
N842	35,000	3.3	3.1396E+12	5	Jan 2001	Standard
N868	40,000	3.3	5.2444E+12	5	Jan 2001	Standard
R032	45,000	3.5	1.4550E+13	6	Jan 2002	Standard
R439	50,000	3.5	1.3998E+13	7	Jan 2003	Standard
R444	50,000	3.5	5.5993E+13	8	Jan 2004	Standard
R439	50,000	3.5	1.3998E+13	6	Jan 2002	Pref(1.050)
R444	50,000	3.5	5.5993E+13	6	Jan 2002	Pref(1.050)

Table 5.6.1-15 Maine Yankee Consolidated Fuel Model Parameters

			Actual		Modeled		Required	Cool Time
Lattice	Assy	Num Rods	Burnup [MWD/MTU]	Enrichment [wt %]	Burnup [MWD/MTU]	Enrichment [wt %]	Cool Time [y]	1/1/01 [y]
CN-1	EF0039	172	5150	1.929	30000	1.9	6	26
CN-10	EF0045	176	17150	1.953	30000	1.9	6	24
	EF0046	107	17150	1.953	30000	1.9	6	24

Table 5.6.1-16 Maine Yankee Source Rate Analysis for CN-10 Consolidated Fuel Lattice

Cool Time [years]	Num Rods Present	Decay Heat [kW/cask]	Fuel Neutron [n/s/assy]	Fuel Gamma [g/sec/assy]	Fuel Hardware [g/sec/assy]
6	176	13.9	1.63E+08	3.16E+15	9.28E+12
24	283	7.42	8.41E+07	1.28E+15	8.67E+11
Src Ratio 24/6		0.86	0.83	0.65	0.15

Table 5.6.1-17 Additional Maine Yankee Non-Fuel Hardware Characterization – Non-Neutron Sources

Non Fuel Material	Waste Volume [ft^3]	Total Curies	Co-60 Curies
Sb-Be Source 1H1	0.020	4.15E+02	2.22E+02
Sb-Be Source 6H4	0.020	4.32E+02	2.31E+02
CEA Tips	0.100	1.06E+02	8.90E+01
ICI	0.007	2.82E+01	1.76E+01

Table 5.6.1-18 Additional Maine Yankee Non-Fuel Hardware Characterization – Neutron Sources

Non Fuel Material	Pu-238 grams	Pu-238 Curies	Pu-239 grams	Pu-239 Curies
Pu-Be Unirradiated Source	1.16	-	0.24	-
Pu-Be Irradiated Sources	1.16	5.10E-02	0.24	5.88E-05

Table 5.6.1-19 Pu-Be Assembly Hardware Spectra (Cycles 1-13) – 5 Year Cool Time from 1/1/1997

Group	Pu-Be SS Hardware [g/sec]
1	0.0000E+00
2	0.0000E+00
3	0.0000E+00
4	0.0000E+00
5	1.8059E-15
6	3.5714E+05
7	2.3032E+08
8	8.9078E-03
9	9.7053E+12
10	3.4367E+13
11	1.2604E+10
12	4.0605E+07
13	1.1692E+08
14	1.8500E+09
15	1.4100E+09
16	2.8397E+10
17	1.1771E+11
18	5.9808E+11
TOTAL	4.4833E+13

Table 5.6.1-20 Additional Maine Yankee Non-Fuel Hardware – HW Assembly Spectra (Class 2 Canister) – 5 Year Cool Time from 1/1/1997

Group	ICI Segment [g/sec]	CEA Tips [g/sec]	Total Gamma [g/sec]
1	0.0000E+00	0.0000E+00	0.00E+00
2	0.0000E+00	0.0000E+00	0.00E+00
3	0.0000E+00	0.0000E+00	0.00E+00
4	0.0000E+00	0.0000E+00	0.00E+00
5	0.0000E+00	0.0000E+00	0.00E+00
6	5.6364E+04	1.4995E+04	7.14E+04
7	3.6350E+07	9.6704E+06	4.60E+07
8	0.0000E+00	0.0000E+00	0.00E+00
9	1.5317E+12	4.0749E+11	1.94E+12
10	5.4239E+12	1.4430E+12	6.87E+12
11	2.4164E+08	6.4285E+07	3.06E+08
12	6.4084E+06	1.7049E+06	8.11E+06
13	1.8453E+07	4.9092E+06	2.34E+07
14	2.9197E+08	7.7675E+07	3.70E+08
15	2.2253E+08	5.9201E+07	2.82E+08
16	4.4816E+09	1.1923E+09	5.67E+09
17	1.8576E+10	4.9418E+09	2.35E+10
18	9.3171E+10	2.4787E+10	1.18E+11
Total	7.0726E+12	1.8816E+12	8.95E+12

Table 5.6.1-21 Additional Maine Yankee Non-Fuel Hardware – Source Assembly Spectra – 5 Year Cool Time from 1/1/1997

	Sb-Be Source	Pu-Be Unirradiated Source		Pu-Be Irradiated Source			
Group	Gamma [g/sec]	Gamma [g/sec]	Neutron [n/sec]	Gamma [g/sec]	Hw Gamma [g/sec]	Total Gamma [g/sec]	Neutron [n/sec]
1	0.0000E+00	1.8438E+00	4.7620E+01	5.9037E-03	0.0000E+00	5.9037E-03	1.5250E-01
2	0.0000E+00	9.0379E+00	3.1850E+03	2.8938E-02	0.0000E+00	2.8938E-02	1.0200E+01
3	0.0000E+00	4.8704E+01	8.0950E+03	1.5595E-01	0.0000E+00	1.5595E-01	2.5920E+01
4	0.0000E+00	1.2868E+02	2.3510E+03	4.1204E-01	0.0000E+00	4.1204E-01	7.5290E+00
5	0.0000E+00	4.0697E+02	1.5900E+03	1.3030E+00	1.8059E-15	1.3030E+00	5.0900E+00
6	2.2971E+05	4.7836E+02	8.2740E+02	1.5315E+00	3.5714E+05	3.5714E+05	2.6490E+00
7	1.4814E+08	8.6530E+02	1.4900E+02	2.7621E+00	2.3032E+08	2.3032E+08	4.7700E-01
8	0.0000E+00	1.5016E+03	-	4.7854E+00	8.9078E-03	4.7943E+00	-
9	6.2425E+12	4.2159E+00	-	4.6985E-07	9.7053E+12	9.7053E+12	-
10	2.2105E+13	8.9859E+03	-	2.8745E+01	3.4367E+13	3.4367E+13	-
11	9.8479E+08	3.9420E+04	-	1.2621E+02	1.2604E+10	1.2604E+10	-
12	2.6117E+07	3.0176E+05	-	9.6649E+02	4.0605E+07	4.0606E+07	-
13	7.5204E+07	8.7531E+03	-	3.4464E+01	1.1692E+08	1.1692E+08	-
14	1.1899E+09	2.6915E+04	-	1.0614E+02	1.8500E+09	1.8500E+09	-
15	9.0690E+08	2.5370E+04	-	8.3993E+01	1.4100E+09	1.4100E+09	-
16	1.8265E+10	2.0487E+07	-	6.5574E+04	2.8397E+10	2.8397E+10	-
17	7.5705E+10	2.8935E+07	-	9.2577E+04	1.1771E+11	1.1771E+11	-
18	3.7972E+11	3.1017E+10	-	9.9310E+07	5.9808E+11	5.9818E+11	-
Total	2.8825E+13	3.1067E+10	1.625E+04	9.9470E+07	4.4833E+13	4.4833E+13	5.202E+01

Table 5.6.1-22 Additional Maine Yankee Non-Fuel Hardware – Hardware Assembly Dose Rates (Class 2) – 5 Years Cooled from 1/1/1997

Group	Storage - Surface Gamma Dose [mrem/hr]	Transfer - Surface Gamma Dose [mrem/hr]
1	3.66E-10	1.51E-10
2	1.41E-09	8.97E-10
3	4.92E-09	5.00E-09
4	7.10E-09	1.20E-08
5	1.08E-08	2.99E-08
6	4.21E-08	1.91E-07
7	9.96E-06	6.12E-05
8	2.24E-09	1.72E-08
9	4.59E-02	3.77E-01
10	3.49E-02	2.24E-01
11	2.31E-07	6.42E-07
12	1.82E-09	1.02E-09
13	2.68E-10	9.13E-13
14	9.84E-11	3.12E-19
15	2.65E-12	1.49E-40
16	1.11E-14	0.00E+00
17	1.91E-41	0.00E+00
18	0.00E+00	0.00E+00
Total	8.09E-02	6.01E-01

Table 5.6.1-23 Additional Maine Yankee Non-Fuel Hardware – Storage Cask Source Assembly Surface Dose Rates – 5 Years Cooled from 1/1/1997

Group	Sb-Be Source Dose	Pu-Be Unirradiated Source Dose		Pu-Be Irradiated Source Dose	
	Gamma [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]
1	0.00E+00	1.81E-11	2.94E-08	5.78E-14	9.41E-11
2	0.00E+00	6.93E-11	1.11E-06	2.22E-13	3.57E-09
3	0.00E+00	2.42E-10	2.45E-06	7.76E-13	7.85E-09
4	0.00E+00	3.50E-10	5.57E-07	1.12E-12	1.78E-09
5	0.00E+00	5.31E-10	3.29E-07	1.70E-12	1.05E-09
6	1.19E-07	2.49E-10	1.62E-07	1.86E-07	5.19E-10
7	3.21E-05	1.87E-10	2.19E-08	4.99E-05	7.02E-11
8	0.00E+00	1.11E-10	-	3.53E-13	-
9	1.48E-01	9.99E-14	-	2.30E-01	-
10	1.12E-01	4.57E-11	-	1.75E-01	-
11	7.41E-07	2.97E-11	-	9.48E-06	-
12	3.34E-09	3.86E-11	-	5.19E-09	-
13	8.37E-10	9.74E-14	-	1.30E-09	-
14	3.15E-10	7.13E-15	-	4.90E-10	-
15	8.52E-12	2.38E-16	-	1.32E-11	-
16	3.34E-14	3.74E-17	-	5.19E-14	-
17	5.99E-41	2.29E-44	-	9.31E-41	-
18	0.00E+00	0.00E+00	-	0.00E+00	-
Total	2.60E-01	1.87E-09	4.67E-06	4.05E-01	1.49E-08

Table 5.6.1-24 Additional Maine Yankee Non-Fuel Hardware – Transfer Cask Source Assembly Surface Dose Rates – 5 Years Cooled from 1/1/1997

Group	Sb-Be Source Dose	Pu-Be Unirradiated Source Dose		Pu-Be Irradiated Source Dose	
	Gamma [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]
1	0.00E+00	7.43E-12	3.40E-06	2.38E-14	1.09E-08
2	0.00E+00	4.42E-11	1.50E-04	1.42E-13	4.81E-07
3	0.00E+00	2.46E-10	3.57E-04	7.89E-13	1.14E-06
4	0.00E+00	5.90E-10	7.29E-05	1.89E-12	2.33E-07
5	0.00E+00	1.47E-09	3.65E-05	4.72E-12	1.17E-07
6	5.40E-07	1.12E-09	1.34E-05	8.40E-07	4.30E-08
7	1.97E-04	1.15E-09	6.69E-07	3.06E-04	2.14E-09
8	0.00E+00	8.53E-10	-	2.72E-12	-
9	1.21E+00	8.20E-13	-	1.89E+00	-
10	7.21E-01	2.93E-10	-	1.12E+00	-
11	2.06E-06	8.25E-11	-	2.64E-05	-
12	1.86E-09	2.15E-11	-	2.89E-09	-
13	2.85E-12	3.32E-16	-	4.44E-12	-
14	9.99E-19	2.26E-23	-	1.55E-18	-
15	4.77E-40	1.33E-44	-	7.42E-40	-
16	0.00E+00	0.00E+00	-	0.00E+00	-
17	0.00E+00	0.00E+00	-	0.00E+00	-
18	0.00E+00	0.00E+00	-	0.00E+00	-
Total	1.94E+00	5.89E-09	6.34E-04	3.01E+00	2.03E-06

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6.0 CRITICALITY EVALUATION

This chapter documents the criticality evaluation of the Universal Storage System with either PWR or BWR contents. The results demonstrate that the effective neutron multiplication factor, k_{eff}, of the Universal Storage System under normal, off-normal, and accident conditions, is less than 0.95 including biases and uncertainties. The system design therefore meets the criticality requirements of 10 CFR 72.124(a) [1], 10 CFR 72.236(c), and Chapter 6 of NUREG-1536 [2].

6.1 Discussion and Results

The Universal Storage System consists of a Transportable Storage Canister, a transfer cask and a Vertical Concrete Cask. The system is designed to safely store up to 24 undamaged PWR fuel assemblies or 56 undamaged BWR fuel assemblies. Maximum initial enrichment for each PWR and BWR fuel assembly grouping, as a function of the assemblies' key parameters, is shown in Tables 6.1-1 and 6.1-2. For PWR fuel assemblies, the maximum allowable enrichment ranges from 4.2 wt. % to 5.0 wt. % without any soluble boron. With at least 1000 ppm of soluble boron, the maximum allowable enrichment is 5.0 wt. % for all PWR assemblies. Maximum initial enrichment is defined as peak rod enrichment for PWR assemblies and the maximum initial peak planar-average enrichment for BWR assemblies. The maximum initial peak planar-average enrichment is the maximum planar-average enrichment at any height along the axis of the fuel assembly. For BWR fuel assemblies, the maximum enrichment allowed ranges from 4.4 wt. % to 4.8 wt. %.

Primarily on the basis of their lengths and cross-sections, the fuel assemblies are categorized into classes. Three classes of PWR fuel assemblies and two classes of BWR fuel assemblies are evaluated for storage. Five Transportable Storage Canister assemblies of different lengths and configuration are designed to store the three classes of PWR fuel assemblies and the two classes of BWR fuel assemblies. The canister is comprised of a stainless steel canister and a fuel basket within which fuel is loaded. The canister is loaded into the Vertical Concrete Cask for storage. The length of the Vertical Concrete Cask also varies depending upon the type of the canister it is designed to store.

A transfer cask is used for handling the canister during loading of spent fuel. Fuel is loaded into the canister contained within the transfer cask underwater in the spent fuel pool. Once loaded with fuel, the canister is drained, dried, inerted, and welded shut. The transfer cask is then used to transfer the canister into and out of the concrete cask or shipping cask. The transfer cask provides shielding during the canister loading and transfer operations.

The PWR transfer cask is designed in two configurations, standard and advanced. The advanced design is identical to the standard design with the exception of a trunnion support plate. This plate has no impact on system reactivity. Therefore, all analysis of the standard transfer cask applies to the advanced transfer cask.

Under normal conditions, such as loading in a spent fuel pool, moderator (water) is present in the canister while it is in the transfer cask. Also, during draining and drying operations, moderator with varying density is present. Thus, the criticality evaluation of the transfer cask includes a variation in moderator density and a determination of optimum moderator density. Off-normal and accident conditions are bounded by assuming the most reactive mechanical basket configuration as well as moderator intrusion into the fuel cladding (i.e., 100% fuel failure).

Under normal and accident conditions, moderator is not present in the canister while it is in the concrete cask. However, access to the environment is possible via the air inlets in the concrete cask and the convective heat transfer annulus between the canister and the cask steel liner. This access provides paths for moderator intrusion during a flood. Under off-normal conditions, moderator intrusion into the convective heat transfer annulus is evaluated. For the initial evaluation without soluble boron credit, under hypothetical accident conditions, it is assumed that the canister confinement fails, and moderator intrusion into the canister and into the fuel cladding (100% fuel failure) is evaluated. This is a conservative assumption, since normal, off-normal and design basis accident analysis shows that the confinement boundary remains undamaged. Therefore, there are no circumstances under which there would be water in the canister. In the PWR soluble boron evaluation, credit is taken for the dry canister. For this configuration, a wet transfer cask containing a canister filled with a water/soluble boron mixture and a dry canister in a concrete cask are assumed.

Criticality control in the PWR basket is achieved by using a flux trap, or a combination flux trap and soluble neutron absorber (boron). Individual fuel assemblies are held in place by fuel tubes surrounded by four neutron absorber sheets. The neutron absorber modeled is a borated aluminum neutron absorber. Any similar material meeting the ^{10}B areal density and physical dimension requirement will produce similar reactivity results. A stainless steel cover holds the neutron absorber sheets in place. The fuel tubes are separated by a gap that is filled with water when the canister is flooded. Fast neutrons escaping one fuel assembly are moderated in the water gap and are absorbed by the neutron absorber between the assemblies before they can cause a fission in the adjacent assembly. The flux trap gap spacing is maintained by the basket's stainless steel support disks, which separate individual fuel assembly tubes. Alternating stainless steel disks and aluminum heat transfer disks are placed axially at intervals determined by thermal and structural constraints. The PWR basket design includes 30, 32, or 34 support disks and 29, 31, or 33 heat transfer disks, respectively. The minimum loading of the neutron absorber sheets in the PWR fuel tubes is 0.025 g ^{10}B/cm^2. To reach higher initial enrichments than those allowed by using only the flux trap for criticality control, a separate evaluation, including soluble boron at 1000 ppm in the moderator, is performed. The soluble boron absorbs thermal neutrons inside the assembly, as well as in the flux traps. In combination with the flux traps and fixed neutron poison, the soluble boron allows loading of PWR fuel assemblies with an initial enrichment up to 5.0 wt. % ^{235}U.

Criticality control in the BWR basket is achieved by a single neutron absorber sheet between each fuel assembly. The neutron absorber modeled is a borated aluminum neutron absorber. Any similar material meeting the ^{10}B areal density and physical dimension requirement will produce similar reactivity results. Individual fuel assemblies in the BWR basket are held in place by fuel tubes. The fuel tubes are of three types: tubes with neutron absorber on two sides; tubes with neutron absorber on one side; and tubes with no neutron absorber. The fuel tube types are arranged such that there is at least one sheet of neutron absorber between adjacent assemblies. As in the PWR basket, a stainless steel cover holds the neutron absorber sheets in place, and the fuel tubes are separated by a gap that is filled with water when the canister is flooded. In the case of BWR fuel, this arrangement is sufficient to moderate and absorb thermal neutrons before they can cause a fission in the adjacent assembly. The use of flux traps between BWR assemblies is not necessary because of the smaller size and amount of fissile material in BWR assemblies compared with PWR assemblies. Of the total 56 fuel tubes in each BWR basket, 42 tubes contain neutron absorber sheets on two sides of the tubes; 11 tubes contain neutron absorber sheets on one side; and the remaining 3 tubes contain no neutron absorber sheets. The engineered placement of the neutron absorber sheets assures sufficient absorption of thermal neutrons to achieve a neutron multiplication factor (k_s) below 0.95. The minimum loading of the neutron absorber sheets in the BWR tubes is 0.011 g ^{10}B/cm^2. The BWR Class 4 and 5 basket designs include 40 and 41 carbon steel support disks, respectively. The BWR basket design also includes 17 aluminum heat transfer disks.

The SCALE 4.3 Criticality Safety Analysis Sequence (CSAS) [3, 4] and ANSWERS MONK module [20] are used to perform the Universal Storage System criticality analysis. This sequence includes KENO-Va [5] Monte Carlo analysis to determine k_{eff} under normal and accident conditions. The 27-group ENDF/B-IV neutron cross-section library [6] is used in all calculations. CSAS with the 27-group library is benchmarked by comparison to 63 critical experiments relevant to light water reactor fuel in storage and transport casks. The MONK8A Monte Carlo Program for Nuclear Criticality Safety Analysis (SERCO Assurance [20]) employs the Monte Carlo technique in combination with JEF 2.2-based point energy neutron libraries to determine the effective neutron multiplication factor (k_{eff}). The specific libraries are dice96j2v5 for general neutron cross-section information and therm96j2v2 for thermal scatter data in the water moderator. MONK8a, with the JEF 2.2 neutron cross-section libraries, is benchmarked by comparison to critical experiments relevant to light-water reactor fuel in storage and transport casks shown in Section 6.5.

The most reactive PWR assembly is the Westinghouse 17×17 OFA and the most reactive BWR fuel assembly is the Exxon/ANF/Siemens Power Corp. (Ex/ANF) 9×9 with 79 fuel rods (see Section 6.4.1.2 for detailed discussion). These assemblies, respectively, bound all PWR (Classes 1-3) and BWR (Classes 4-5) fuel assemblies to be stored (see Tables 6.2-1 and 6.2-2), as demonstrated in Section 6.4.1.2. The most reactive PWR and BWR fuel assemblies, evaluated as fresh fuel in their respective basket configuration, are used in the criticality calculations for the transfer cask and the concrete cask.

The maximum multiplication factors with uncertainties and code bias are calculated, using conservative assumptions, for the transfer cask and the Vertical Concrete Cask containing PWR (4.2 wt. % ^{235}U) or BWR (4.0 wt. % ^{235}U) fuel. The calculations for the transfer cask are performed for normal and accident conditions, and those for the concrete cask are performed for normal, accident, and off-normal conditions. The results of the analyses are presented in detail in Section 6.4.3 and are summarized as:

Condition	Maximum Multiplication Factors with Uncertainties (k_s)			
	PWR Fuel		BWR Fuel	
	Transfer Cask	Concrete Cask	Transfer Cask	Concrete Cask
Normal	0.93921	0.38329	0.91919	0.38168
Accident	0.94749	0.94704	0.92235	0.92332
Off-Normal	--	0.37420	--	0.38586

Analysis of simultaneous moderator density variation inside and outside either the transfer or concrete casks shows a monotonic decrease in reactivity with decreasing moderator density. Thus, the full moderator density condition bounds any off-normal or accident condition. Analysis of moderator intrusion into the concrete cask heat transfer annulus with the dry canister shows a slight decrease in reactivity from the completely dry condition.

The fixed maximum enrichment evaluation is augmented by assembly-specific analyses. Fuel types identified in Section 6.2 are grouped based on key fuel lattice characteristics. Each of the groups is then evaluated to determine the maximum enrichment for which cask reactivity (k_{eff}) plus two sigma (2σ) remains below the upper safety limit (USL) of 0.9426. The maximum allowed enrichment with the key lattice parameters is shown in Tables 6.1-1 and 6.1-2 for PWR and BWR fuel assemblies, respectively. Table 6.1-2 enrichments do not take credit for any soluble boron. At 1000 ppm soluble boron, the maximum allowed initial enrichment for all PWR fuel assembly types is 5.0 wt. % ^{235}U.

Table 6.1-1 PWR Fuel Assembly Maximum Allowed Enrichment

ID	No. of Fuel Rods	Max MTU	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)	No. Guide/ Instr. Tubes	Min Tube Thick (in)	Max Enrich. (wt.% ^{235}U)
ce14a	176	0.404	0.580	0.440	0.0280	0.3765	137.0	5	N/A	5.0
we14d	176	0.411	0.580	0.440	0.0260	0.3805	136.7	5	N/A	5.0
ce14my	176	0.411	0.590	0.4375	0.0240	0.3800	137.0	5	N/A	4.7
ex14a	179	0.369	0.556	0.424	0.0300	0.3505	142.0	17	0.034	5.0
we14a	179	0.414	0.556	0.422	0.0225	0.3674	145.2	17	0.034	5.0
we14b	179	0.361	0.556	0.400	0.0243	0.3444	144.0	17	0.034	5.0
ex15a	204	0.441	0.563	0.424	0.0300	0.3565	144.0	21	0.017	4.6
we15a	204	0.465	0.563	0.422	0.0242	0.3659	144.0	21	0.015	4.3
bw15a	208	0.481	0.568	0.430	0.0265	0.3686	144.0	17	0.016	4.4
ce16e	236	0.443	0.506	0.382	0.0230	0.3255	150.0	5	N/A	4.8
ex17a	264	0.412	0.496	0.360	0.0250	0.3030	144.0	25	0.016	4.4
we17a	264	0.467	0.496	0.374	0.0225	0.3225	144.0	25	0.015	4.5
we17b	264	0.428	0.496	0.360	0.0225	0.3088	144.0	25	0.015	4.3
bw17a	264	0.466	0.502	0.379	0.0240	0.3232	143.0	25	0.0175	4.4
Palisades[1]	216	0.432	0.550	0.418	0.0260	0.3580	132.0	N/A	N/A	4.2[1]
Palisades[1]	179	0.374	0.556	0.417	0.0300	0.3505	144.0	5	N/A	4.2[1]
Palisades[1]	216	0.431	0.550	0.417	0.0300	0.3580	131.8	N/A	N/A	4.2[1]

Note: Site specific.

1. Palisades 15×15 fuel assemblies and Prairie Island 14×14 assemblies are not re-evaluated and remain at the 4.2 wt% original design basis enrichment.

Table 6.1-2 BWR Fuel Assembly Maximum Allowed Enrichment – No Soluble Boron

ID	No. of Fuel Rods	Max MTU	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)	No. Water Rods	Min Rod Thick (in)	Max Enrich. (wt.% ^{235}U)
ex07a	48	0.196	0.738	0.570	0.036	0.4900	144.0	0	N/A	4.5
ge07a	49	0.198	0.738	0.570	0.036	0.4880	144.0	0	N/A	4.5
ge07f	49	0.198	0.738	0.563	0.032	0.4870	144.0	0	N/A	4.5
ge07h	49	0.192	0.738	0.563	0.037	0.4770	146.0	0	N/A	4.7
ge08i	60	0.179	0.640	0.484	0.032	0.4100	150.0	1	N/A	4.5
ge08k	62	0.185	0.640	0.483	0.032	0.4100	150.0	2	0.0300	4.5
ex08b	62	0.180	0.641	0.484	0.036	0.4045	150.0	2	0.0360	4.7
ge08n	63	0.188	0.640	0.493	0.034	0.4160	146.0	1	0.0340	4.8
ex08a	63	0.177	0.641	0.484	0.036	0.4045	145.2	0	N/A	4.7
ex09b	74	0.167	0.572	0.424	0.030	0.3565	150.0	2	N/A	4.4
ge09a	74	0.185	0.566	0.441	0.028	0.3760	150.0	2	N/A	4.5
ex09c	79	0.178	0.572	0.424	0.030	0.3565	150.0	2	0.0300	4.4
ge09b	79	0.198	0.566	0.441	0.028	0.3760	150.0	2	0.0280	4.6

6.2 Spent Fuel Loading

The Universal Storage System is designed to store Transportable Storage Canisters containing spent nuclear fuel. Canisters of five different lengths are designed, each to accommodate one of three classes of PWR fuel assemblies or one of two classes of BWR fuel assemblies. The classification of the fuel assemblies is based primarily on fuel assembly length and cross-section. The classes of major fuel assemblies to be stored in the Universal Storage System and their characteristics are shown in Tables 6.2-1 (PWR) and 6.2-2 (BWR). Sections 6.4.5 and 6.4.6 extend the evaluation of the single PWR (4.2 wt. % ^{235}U) and BWR (4.0 wt. % ^{235}U) maximum initial enrichments to an assembly-specific maximum initial enrichment. The enrichments represent maximum planar average enrichment for BWR assemblies and peak fuel rod enrichments for PWR assemblies. Tables 6.2-1 and 6.2-2 include a column containing an identifier linking each of the listed assembly types to the allowable maximum initial enrichment searches in Sections 6.4.5 and 6.4.6.

Class 1 Westinghouse fuel assemblies and Class 2 B&W fuel assemblies include inserts. Fuel assembly inserts are nonfuel-bearing components, such as flow mixers, in-core instrument thimbles, burnable poison rods or solid stainless steel rods. These components are inserted into the fuel assembly guide tubes. The criticality analyses do not take credit for displacement of moderator by the inserts. For the unborated moderator analyses, insertion of an in-core instrument thimble, a burnable poison rod assembly or a solid stainless steel rod reduces reactivity by further decreasing the (unborated) moderator to fuel ratio in the fuel assembly lattice. For the analyses that take credit for soluble boron in the moderator, insertion of an in-core instrument thimble, a burnable poison rod assembly or a solid stainless steel rod would displace boron for which credit is taken. Therefore, a burnable poison rod assembly, an in-core instrument thimble or a solid stainless steel rod insert shall only be loaded into an assembly that does not require credit to be taken for soluble boron in the moderator in order to meet the assembly enrichment limit. Insertion of a flow mixer is not restricted, as this component does not displace moderator in the active fuel region.

To preclude a potential increase in reactivity as a result of empty fuel rod positions in the assembly, any empty fuel rod position is to be filled with a solid filler rod. Filler rods may be fabricated from either solid zirconium alloy or solid Type 304 stainless steel, or may be solid neutron absorber rods inserted for in-core reactivity control prior to reactor operations.

Table 6.2-1 PWR Fuel Assembly Characteristics (Zirc-4 Clad)

Fuel Class	Vendor	Array	Version	Max MTU	No of Fuel Rods	Pitch (in)	Rod Dia. (in)	Clad Thick (in)	Pellet Dia (in)	Active Length (in)	ID
1	CE	14 × 14	Std.	0.4037	176	0.5800	0.440	0.0280	0.3765	137.0	ce14a
1	CE	14 × 14	Ft Cal.	0.3772	176	0.5800	0.440	0.0280	0.3765	128.0	ce14a
1	CE	15 × 15	Palis.	0.4317	216	0.5500	0.418	0.0260	0.3580	132.0	—[1]
1	CE	16 × 16	Lucie 2	0.4025	236	0.5060	0.382	0.0250	0.3250	136.7	ce16d
1	Ex/ANF	14 × 14	WE	0.3689	179	0.5560	0.424	0.0300	0.3505	142.0	ex14a
1	Ex/ANF	14 × 14	CE	0.3814	176	0.5800	0.440	0.0310	0.3700	134.0	ce14a
1	Ex/ANF	14 × 14	Praire Isl.	0.3741	179	0.5560	0.417	0.0300	0.3505	144.0	—[1]
1	Ex/ANF	15 × 15	WE	0.4410	204	0.5630	0.424	0.0300	0.3565	144.0	ex15a
1	Ex/ANF	15 × 15	Palis	0.4310	216	0.5500	0.417	0.0300	0.3580	131.8	—[1]
1	Ex/ANF	17 × 17	WE	0.4123	264	0.4960	0.360	0.0250	0.3030	144.0	ex17a
1	WE	14 × 14	Std/ZCA	0.4144	179	0.5560	0.422	0.0225	0.3674	145.2	we14a
1	WE	14 × 14	OFA	0.3612	179	0.5560	0.400	0.0243	0.3444	144.0	we14b
1	WE	14 × 14	Std/ZCB	0.4144	179	0.5560	0.422	0.0225	0.3674	145.2	we14a
1	WE	14 × 14	CE Model	0.4115	176	0.5800	0.440	0.0260	0.3805	136.7	we14d
1	WE	15 × 15	Std	0.4646	204	0.5630	0.422	0.0242	0.3659	144.0	we15a
1	WE	15 × 15	Std/ZC	0.4646	204	0.5630	0.422	0.0242	0.3659	144.0	we15a
1	WE	15 × 15	OFA	0.4646	204	0.5630	0.422	0.0242	0.3659	144.0	we15a
1	WE	17 × 17	Std	0.4671	264	0.4960	0.374	0.0225	0.3225	144.0	we17a
1	WE	17 × 17	OFA	0.4282	264	0.4960	0.360	0.0225	0.3088	144.0	we17b
1	WE	17 × 17	Vant 5	0.4282	264	0.4960	0.360	0.0225	0.3088	144.0	we17b
2	B&W	15 × 15	Mark B	0.4807	208	0.5680	0.430	0.0265	0.3686	144.0	bw15a
2	B&W	15 × 15	Mark BZ	0.4807	208	0.5680	0.430	0.0265	0.3686	144.0	bw15a
2	B&W	17 × 17	Mark C	0.4658	264	0.5020	0.379	0.0240	0.3232	143.0	bw17a
3	CE	16 × 16	Sono 2&3	0.4417	236	0.5060	0.382	0.0230	0.3255	150.0	ce16e
3	CE	16 × 16	ANO2	0.4417	236	0.5060	0.382	0.0230	0.3255	150.0	ce16e
3	CE	16 × 16	SYS80	0.4417	236	0.5060	0.382	0.0230	0.3255	150.0	ce16e

1. These site specific fuels were not re-evaluated and remain at a maximum initial enrichment of 4.2 wt% ^{235}U.

Table 6.2-2 BWR Fuel Assembly Characteristics (Zirc-2 Clad)

Fuel Class	Vendor	Array	Version	Max MTU	No of Fuel Rods	Pitch (in)	Rod Dia (in)	Clad Thick (in)	Pellet Dia (in)	Active Length (in)	ID
4[5]	Ex/ANF	7 × 7	GE	0.1960	48	0.738	0.570	0.036	0.490	144	ex07a
4	Ex/ANF	8 × 8	JP-3	0.1764	63	0.641	0.484	0.036	0.4045	145.2	ex08a
4	Ex/ANF	9 × 9	JP-3	0.1722	79	0.572	0.424	0.03	0.3565	145.2	ex09c
4	GE	7 × 7	GE-2a	0.1985	49	0.738	0.570	0.036	0.488	144	ge07a
4	GE	7 × 7	GE-2b	0.1977	49	0.738	0.563	0.032	0.487	144	ge07f
4	GE	7 × 7	GE-3	0.1896	49	0.738	0.563	0.037	0.477	144	ge08h
4	GE	8 × 8	GE-4	0.1855	63	0.640	0.493	0.034	0.416	144	ge08n
4	GE	8 × 8	GE-5	0.1788	62	0.640	0.483	0.032	0.410	145.2	ge08k
4	GE	8 × 8	GE-6 (prep)	0.1788	62	0.640	0.483	0.032	0.410	145.2	ge08k
4	GE	8 × 8	GE-7 (barr)	0.1788	62	0.640	0.483	0.032	0.410	145.2	ge08k
4	GE	8 × 8	GE-8	0.1730	60	0.640	0.484	0.032	0.410	145.2[1]	ge08i
4	GE	8 × 8	GE-10	0.1730	60	0.640	0.484	0.032	0.410	145.2[1,2]	ge08i
5[6]	Ex/ANF	8 × 8	JP-4,5	0.1793	62	0.641	0.484	0.036	0.4045	150	ex08b
5	Ex/ANF	9 × 9	JP-4,5	0.1779	79	0.572	0.424	0.03	0.3565	150	ex09c
5	Ex/ANF	9 × 9	JP-4,5	0.1666	74	0.572	0.424	0.03	0.3565	150	ex09b
5	GE	7 × 7	GE-2	0.1977	49	0.738	0.563	0.032	0.487	144	ge07f
5	GE	7 × 7	GE-3a	0.1896	49	0.738	0.563	0.037	0.477	144	ge07h
5	GE	7 × 7	GE-3b	0.1923	49	0.738	0.563	0.037	0.477	146	ge07h
5	GE	8 × 8	GE-4a	0.1855	63	0.640	0.493	0.034	0.416	144	ge08n
5	GE	8 × 8	GE-4b	0.1880	63	0.640	0.493	0.034	0.416	146	ge08n
5	GE	8 × 8	GE-5	0.1847	62	0.640	0.483	0.032	0.410	150[1]	ge08k
5	GE	8 × 8	GE-6 (prep)	0.1847	62	0.640	0.483	0.032	0.410	150[1]	ge08k
5	GE	8 × 8	GE-7 (barr)	0.1847	62	0.640	0.483	0.032	0.410	150[1]	ge08k
5	GE	8 × 8	GE-10	0.1787	60	0.640	0.484	0.032	0.410	150[1,2]	ge08i
5	GE	9 × 9	GE-11	0.1854	74	0.566	0.441	0.028	0.376	150[1,3,4]	ge09a
5	GE	9 × 9	GE-11	0.1979	79	0.566	0.441	0.028	0.376	150[1,3,4]	ge09b

Notes
1. 6-in, natural uranium blankets on top and bottom.
2. 1 large water hole - 3.2 cm ID, 0.1 cm thickness.
3. 2 large water holes occupying 7 fuel rod locations - 2.5 cm ID, 0.07 cm thickness.
4. Shortened active fuel length in some rods.
5. Class of fuel for BWR/2-3.
6. Class of fuel for BWR/4-6.

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6.3 Criticality Model Specification

6.3.1 Calculational Methodology

Evaluations determining the maximum reactivity configuration of the Universal Storage System for PWR and BWR fuel at design basis enrichment levels are performed with the SCALE 4.3 PC CSAS sequence [3, 4]. Assembly specific maximum enrichment level determinations, with and without soluble boron, are performed with the ANSWERS MONK8A code [20].

The SCALE 4.3 PC CSAS25 [3, 4] sequence and the SCALE 27-group neutron library are used to perform the criticality analysis of the Universal Storage System. This sequence includes the SCALE Material Information Processor [7], BONAMI-S [8], NITAWL-S [9], and KENO-Va [5]. The Material Information Processor generates number densities for standard compositions, prepares geometry data for resonance self-shielding, and creates data input files for the cross-section processing codes. The BONAMI-S and NITAWL-S codes are used to prepare a resonance-corrected cross-section library in AMPX working format. The KENO-Va code uses Monte Carlo techniques to calculate k_{eff}. The 27-group ENDF/B-IV group neutron library is used in all cask criticality calculations.

The CSAS criticality analysis sequence is validated through a series of calculations based on critical experiments performed by Babcock and Wilcox [13], Pacific Northwest Laboratory [14, 15, 16, and 17], and Valduc Critical Mass Laboratory [18]. The 27-group ENDF/B-IV neutron cross-section library is used in the validation, which includes statistical analysis of results. Validation of the CSAS and the method statistics are addressed in Section 6.5.

The MONK8A (AEA Technology) Monte Carlo Program for Nuclear Criticality Safety Analysis employs the Monte Carlo technique in combination with JEF 2.2-based point energy neutron libraries to determine the effective neutron multiplication factor (k_{eff}). The specific libraries are dice96j2v5 for general neutron cross-section information and therm96j2v2 for thermal scatter data in the water moderator. MONK8A, with the JEF 2.2 neutron cross-section libraries, is benchmarked by comparison to critical experiments relevant to light water reactor fuel in storage and transport casks as shown in Section 6.5.

The criticality analysis of the Universal Storage System is performed in several steps.

- The PWR and BWR fuel assembly designs described in Tables 6.2-1 and 6.2-2 are screened to identify sets of standard PWR and BWR arrays.

- The identified sets of arrays are analyzed to determine the most reactive PWR and BWR fuel assemblies for the initial design basis limiting condition.

- The criticality impact of mechanical perturbations and geometric tolerances is evaluated using a fuel tube-in-basket model (PWR) and a basket in-cask-model (BWR) based on the most reactive assembly of each type. These models are described in Section 6.4.1.3.

- A canister-in-cask model is prepared to evaluate the reactivity variation between normal and worst-case configurations of the cask contents under normal and hypothetical accident conditions.

- Key fuel parameters are evaluated to determine a bounding description set. This set of parameters maximizes system reactivity based on the number of fuel rods, a minimum rod outer diameter, maximum pellet diameter, minimum clad thickness, maximum active fuel length, and minimum guide/instrument tube or water rod thickness.

- The fuel data set is, again, reviewed based on the maximum/minimum criteria and a set of bounding fuel assemblies is determined. This set is evaluated at various enrichment levels to set the maximum initial enrichment levels producing a reactivity lower than the upper safety limit (USL).

- For the PWR fuel assemblies, the maximum allowed initial enrichment search is repeated based on a 1000 ppm soluble boron level.

The results of criticality calculations for PWR and BWR assembly loaded casks are provided in Sections 6.4.3.2 and 6.4.3.3, respectively.

6.3.2 Model Assumptions

Assumptions for the basket model are as follows.

- The fuel assembly is modeled at a fuel density of 95% theoretical (0.95×10.96 gm/cm^3 = 10.412 g/cm^3).

- Baseline enrichment for the PWR fuel assembly is 4.2 wt % ^{235}U. The PWR fuel assembly included in this model is the Westinghouse 17×17 OFA fuel assembly which is determined to be the most reactive assembly in the PWR basket (see Section 6.4.1.2.1). The most reactive BWR fuel assembly included in this model is the Ex/ANF 9×9 fuel assembly with an enrichment of 4.00 wt % ^{235}U (see Section 6.4.1.2.2). BWR analysis of heterogeneous versus homogeneous pin enrichment shows that assuming a homogeneous enrichment produces conservative k_{eff} values in the BWR canister (see section 6.4.1.3.2). Homogeneous enrichment is defined to be a planar-average enrichment.

- With the exception of the fuel assembly channels in the BWR case, no fuel assembly structural materials (e.g., spacer grids, thimble plugs, burnable poison rod inserts or solid stainless steel rod inserts as applicable to PWR/BWR fuel types) are included in the active fuel region. Eliminating the structural materials simplifies model construction significantly. Removing parasitic absorbers and increasing the effective H/U ratio in the normally under-moderated assembly increases reactivity. Evaluation of the reactivity impact for a variety of channel dimensions in the BWR most reactive assembly analysis demonstrates that the impact of the channel material on cask criticality is not statistically significant. Removal of the channel on the most reactive assembly (Ex/ANF 9×9) results in k_{eff} decrease of 0.001 from 0.872 to 0.871 with a Monte Carlo uncertainty of 0.001.

- Fuel assembly neutron poisons, e.g., gadolinium rods (BWR), are excluded from the analysis, thereby substantially increasing assembly reactivity of the unburned assembly.

- Fuel assembly cladding is intact. For normal operating conditions, no water is present in the gap between fuel pellet and clad. For hypothetical accident conditions, water is assumed to be present in the pellet-to-clad gap. Because the canister is shown not to fail structurally under normal or accident conditions and the presence of water in the pellet-to-clad gap requires failure of the sealed canister and the fuel, the assumption of water in the pellet-to-clad gap for accident analysis is extremely conservative.

- The moderator is assumed to be pure water (no soluble boron) at standard temperature and pressure (293K and 0.9982 gm/cm^3) or water containing soluble boron at 1000 ppm. The density of 0.9982 gm/cm^3 corresponds to a relative density in SCALE's Material Information Processor of 1.0. The fuel, cladding and other structural materials are assumed to be at 293K.

- The models for all analyses are axially infinite, i.e., no axial leakage. The BWR basket design contains fuel elevations with and without heat transfer disks. The axially infinite length basket model relies on the basket elevation containing the aluminum heat transfer disk. Criticality control in both PWR and BWR baskets is by neutron absorber plate. The neutron absorber plates contain ^{10}B as a neutron absorber, which requires thermalization of the neutrons prior to capture. Modeling the basket elevation containing the heat transfer disk displaces water required for neutrons to be thermalized prior to reaching the neutron absorber plate and, therefore, increases the reactivity of the system.

- ^{10}B density is reduced to 75% in accordance with 10 CFR 71 [10] licensing guidance and requirements provided in the "Standard Review Plan for Dry Cask Storage Systems" (NUREG-1536) [2].

- Geometric tolerances and mechanical perturbations (fuel movement in tube, tube movement in the disk opening, and combined fuel and tube movement) are analyzed to arrive at the highest reactivity basket configuration. PWR system geometric tolerances and mechanical perturbations are initially evaluated by using an "infinite array" of tubes in the basket model. An "infinite array" of tubes is produced by modeling mirrored boundary conditions in the x-y plane and a single fuel tube surrounded by the basket structure out to one half the web width. A basket-in-canister model taking into account any positive biases determined from the single-tube-in-basket model is the "worst case," highest reactivity, concrete cask configuration. BWR geometric tolerances and mechanical perturbations are directly evaluated by a basket-in-cask model.

- Fuel assembly and basket will retain their structure and will not show any significant permanent deformation during normal or accident conditions.

- The canister support disks are modeled as stainless steel 304 instead of stainless steel 17-4PH. The SCALE Material Composition Library and ANSWERS standard mixture library stainless steel definitions are used for all types of stainless steel in the criticality analysis.

- The A-588 Low Alloy Steel used in the transfer cask shell is modeled using the carbon steel properties resident in the SCALE4.3 Standard Composition Library.

- All carbon steel rebar in the concrete is ignored in the concrete cask model.

- The concrete cask center-to-center spacing in the SCALE 4.3 models is 15 feet. The ANSWERS models are directly reflected on the cask surface. The concrete shield reduces the neutron flux to negligible levels. No significant neutron interaction occurs between the storage casks.

- No fuel assembly inserts (in particular poison rods) are modeled.

6.3.3 Description of Calculational Models

The PWR and BWR KENO-Va basket cell models are shown in Figure 6.3-1 and Figure 6.3-2, respectively. The PWR KENO-Va models for the transfer cask and the concrete cask are shown in Figure 6.3-3 and Figure 6.3-4, respectively. Figures 6.3-5 and 6.3-6 show the BWR KENO-Va models for the standard transfer cask and concrete cask. Criticality control provisions in the PWR and BWR basket designs are illustrated in Figures 6.3-7 and 6.3-8, respectively. Sketches of the three-dimensional ANSWERS transfer cask PWR and vertical concrete cask BWR models are shown in Figures 6.3-9 and 6.3-10, respectively. Cross-sections of the ANSWERS model are similar to those of the SCALE models, with the difference being a discrete modeling of the BWR basket with aluminum heat transfer disks restricted to the central fuel area.

The PWR KENO-Va models are derived from a cylindrical segment of either the transfer or storage cask at the active fuel region. Each model is a stack of four slices: one at the steel disk elevation and thickness, one at the aluminum disk elevation and thickness, and two composed of the water space between disks. The basket is modeled in each slice and contains 24 design basis PWR fuel assemblies at 4.2 wt % ^{235}U enrichment and a fuel density corresponding to UO_2 at 95% of theoretical. Each fuel assembly array is explicitly modeled in each of the 24 basket locations. Each basket slice is surrounded by the cask body shielding regions of either the transfer or the storage cask. Each cask slice is surrounded by a KENO-Va cuboid. The four slices are stacked into the KENO global unit.

The BWR KENO-Va models are also derived from a cylindrical segment of either the transfer or storage cask at the center of the active fuel region. As with the PWR models, the BWR models are a stack of four slices, one at the carbon steel disk elevation and thickness, one at the aluminum disk elevation and thickness, and two composed of the water space between disks. The basket is modeled in each slice and contains 56 design basis BWR fuel assemblies at 4.0 wt % ^{235}U enrichment and fuel density corresponding to a 95% theoretical fuel density. Each fuel assembly array is explicitly modeled in each of the 56 basket locations. Each basket slice is surrounded by the cask body shielding regions of either the transfer or storage cask. Each cask slice is surrounded by a KENO-Va cuboid. The four slices are stacked into the KENO global unit.

In both the PWR and BWR KENO-Va models, periodic boundary conditions are imposed on the top and bottom of the global KENO-Va unit to simulate an infinite cylinder, and reflecting boundary conditions are imposed on the sides, thereby simulating an infinite number of casks in the x-y plane. The reflecting boundary condition on the exterior cuboid's x-y faces forms a square pitch array. As shown in Section 6.4, due to the size of the transfer and storage casks, the baskets are neutronically isolated from one another. Moderator density is varied both in the cask cavity and in the exterior cuboid.

Similar to the SCALE 4.3 models, the ANSWERS code is used to model a UMS® storage and transfer cask containing a PWR or BWR canister and basket, with either 24 PWR assemblies or 56 BWR fuel assemblies. The ANSWERS geometry package uses fractal geometry, which allows the model to be divided into self-contained parts. The self-contained parts can be used to separate canister, cask, and fuel into individual components that can be easily modified and checked. Fractal geometry is the result of combining structured geometry and combinatorial geometry (CG). The basic component of the fractal geometry package is a set of simple bodies, such as spheres, boxes, and rods (cylinders). Models are constructed by combining geometry components (bodies) into PARTS. PARTS may be included within other PARTS to any depth of nesting, and a given PART may be included in different positions within the geometry. An additional feature referred to as a HOLE can be used as special contents in different material zones. The advantage to using HOLES is converting a complex geometric description into a simple one. Finite cask/canister/basket/fuel models (termed cask model henceforth) are constructed for the UMS® storage and transfer system containing PWR and BWR canisters. The cask models are constructed in a set of distinct phases. The first four phases are repeated for the

PWR and BWR canisters. The fifth phase represents the UMS® storage and transfer cask model, which is the same for both canisters. In the first phase, a fuel assembly is constructed from the basic components of the fuel assembly, i.e., fuel rod, guide tube, instrument tube (water rods for the BWR assemblies) and nozzles. An array feature is used to form the rod arrangements. To minimize the complexity of these arrays, a check is made on all water rod or guide/instrument tubes to verify that they only occupy one lattice location. If the rod or tube exceeds one lattice location (such as the CE guide tubes), the tube or rod material is neglected from the model. Next the fuel assembly is placed into a fuel tube and surrounded by neutron absorber sheets. These fuel assemblies, with the fuel tube and attached neutron absorber, are then placed in a planar (x-y) configuration. The tubes are placed in the basket stack composed of bottom weldment, stainless steel or carbon steel support disks, aluminum heat transfer disks, and the top weldment. After completing the canister cavity model, a canister shell is placed around the basket with a structural and shield lid stacked on top of the basket. The appropriate cask shields then surround the canister.

6.3.4 Cask Regional Densities

The densities used in the criticality analyses are listed in the following table. Slight differences in the default densities employed by the SCALE and ANSWERS codes exist. These differences do not significantly impact the results of the criticality analysis. For the neutron absorber, densities for the BORAL core material are provided.

Material	ANSWERS Model Density (g/cc)	SCALE Model Density (g/cc)
UO_2	10.412 (95% theoretical)	10.412 (95% theoretical)
Zirconium alloy	6.55	6.56
H_2O	0.9982	0.9982
Stainless steel	7.93	7.92
Carbon steel	7.82	7.82
Lead	11.04	11.35
Aluminum	2.70	2.70
BORAL (core) PWR	2.60	2.60
BORAL (core) BWR	2.68	2.68
NS-4-FR	1.63	1.63
NS-3	1.65	1.65
Concrete	2.24	2.24
H_2O + H_3BO_3 (borated water) – Full Density – 1000 ppm Boron	1.0015	---

6.3.4.1 Active Fuel Region

Fuel rod densities for normal operations conditions are shown below.

Material	Element	Density (atoms/barn-cm)
UO_2 (4.2 wt % ^{235}U)	^{235}U	9.877×10^{-4}
	^{238}U	2.224×10^{-2}
	O	4.646×10^{-2}
UO_2 (4.0 wt % ^{235}U)	^{235}U	9.406×10^{-4}
	^{238}U	2.229×10^{-2}
	O	4.646×10^{-2}
Zirconium Alloy	Zr	4.331×10^{-2}
H_2O	H	6.677×10^{-2}
	O	3.338×10^{-2}
H_2O+ H_3BO_3	H	6.675×10^{-2}
	O	3.346×10^{-2}
	B	5.581×10^{-5}
	O	3.338×10^{-2}

6.3.4.2 Cask Material

SCALE 4.3 model cask material densities used in the criticality evaluation are listed in the following table. With the exception of the slightly higher stainless steel and lower lead, default densities employed by the ANSWERS code, the material composition is identical between SCALE and ANSWERS models.

Material	Element	Density (atoms/barn-cm)
Neutron Absorber core	^{10}B	8.880×10^{-3} (75% of Nominal)
(0.025 g $^{10}B/cm^2$)	^{11}B	4.906×10^{-2}
	C	1.522×10^{-3}
	Al	2.694×10^{-2}
Neutron Absorber core	^{10}B	2.212×10^{-3} (75% of Nominal)
(0.011 g $^{10}B/cm^2$)	^{11}B	1.219×10^{-2}
	C	3.786×10^{-3}
	Al	5.217×10^{-2}

Aluminum	Al	6.031×10^{-2}
Steel 304	Cr	1.743×10^{-2}
	Fe	5.936×10^{-2}
	Ni	7.721×10^{-3}
	Mn	1.736×10^{-3}
Carbon steel	C	3.925×10^{-3}
	Fe	8.350×10^{-2}
Lead	Pb	3.297×10^{-2}
NS-4-FR	H	5.854×10^{-2}
	O	2.609×10^{-2}
	C	2.264×10^{-2}
	N	1.394×10^{-3}
	Al	7.763×10^{-3}
	^{11}B	3.422×10^{-4}
	^{10}B	8.553×10^{-5}
Concrete		
	O	4.494×10^{-2}
	Si	1.621×10^{-2}
	H	1.340×10^{-2}
	Na	1.704×10^{-3}
	Ca	1.483×10^{-3}
	Fe	3.386×10^{-4}
	Al	1.702×10^{-3}

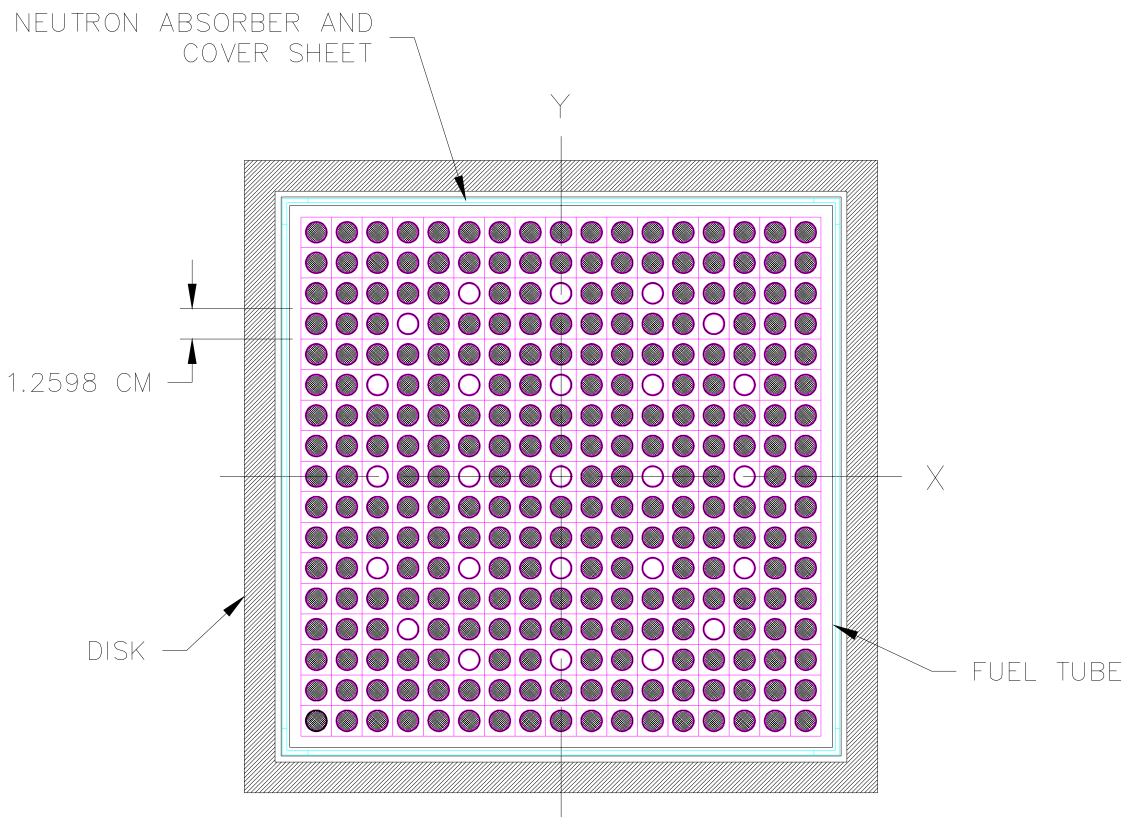
6.3.4.3 Water Reflector Densities

The material densities for the water reflector outside the cask are:

Material	Element	Density (atoms/barn-cm)
H_2O	H	6.677×10^{-2}
	O	3.338×10^{-2}

Water density is varied using the VF (volume fraction) parameter on the SCALE 4.3 material information processor card. This acts as a simple multiplier on the previously listed densities. ANSWERS models are directly reflected on the cask surface and, therefore, do not employ an exterior material.

Figure 6.3-1 KENO-Va PWR Basket Cell Model



Neutron Absorber on Four Sides

Figure 6.3-2 KENO-Va BWR Basket Cell Model

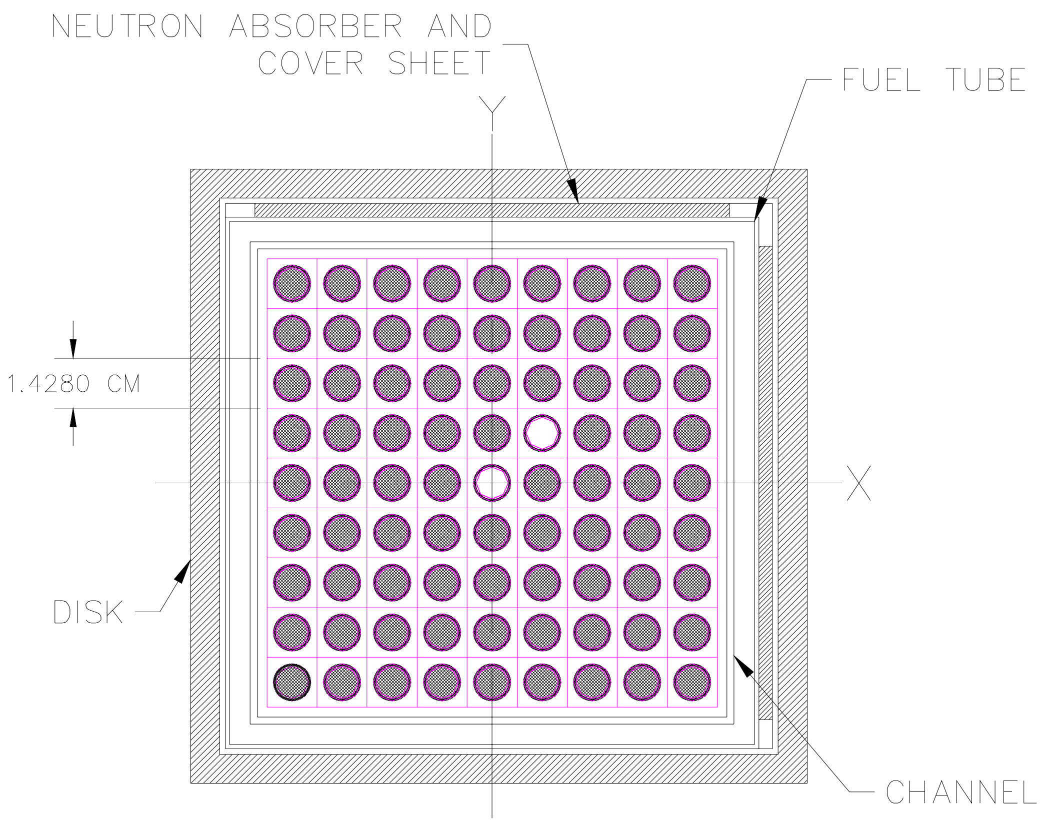


Figure 6.3-3 PWR KENO-Va Transfer Cask Model

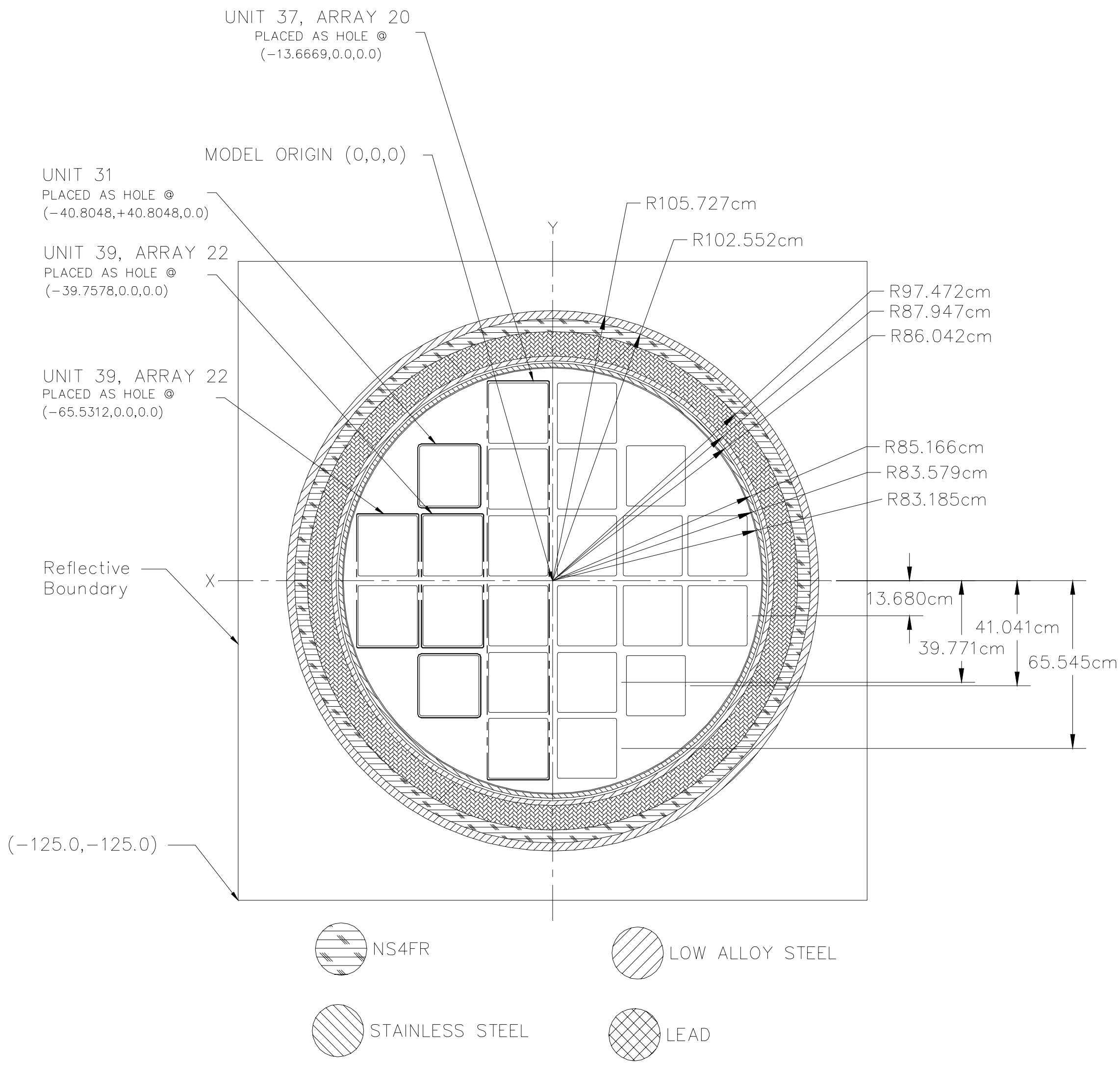


Figure 6.3-4 PWR KENO-Va Vertical Concrete Cask Model

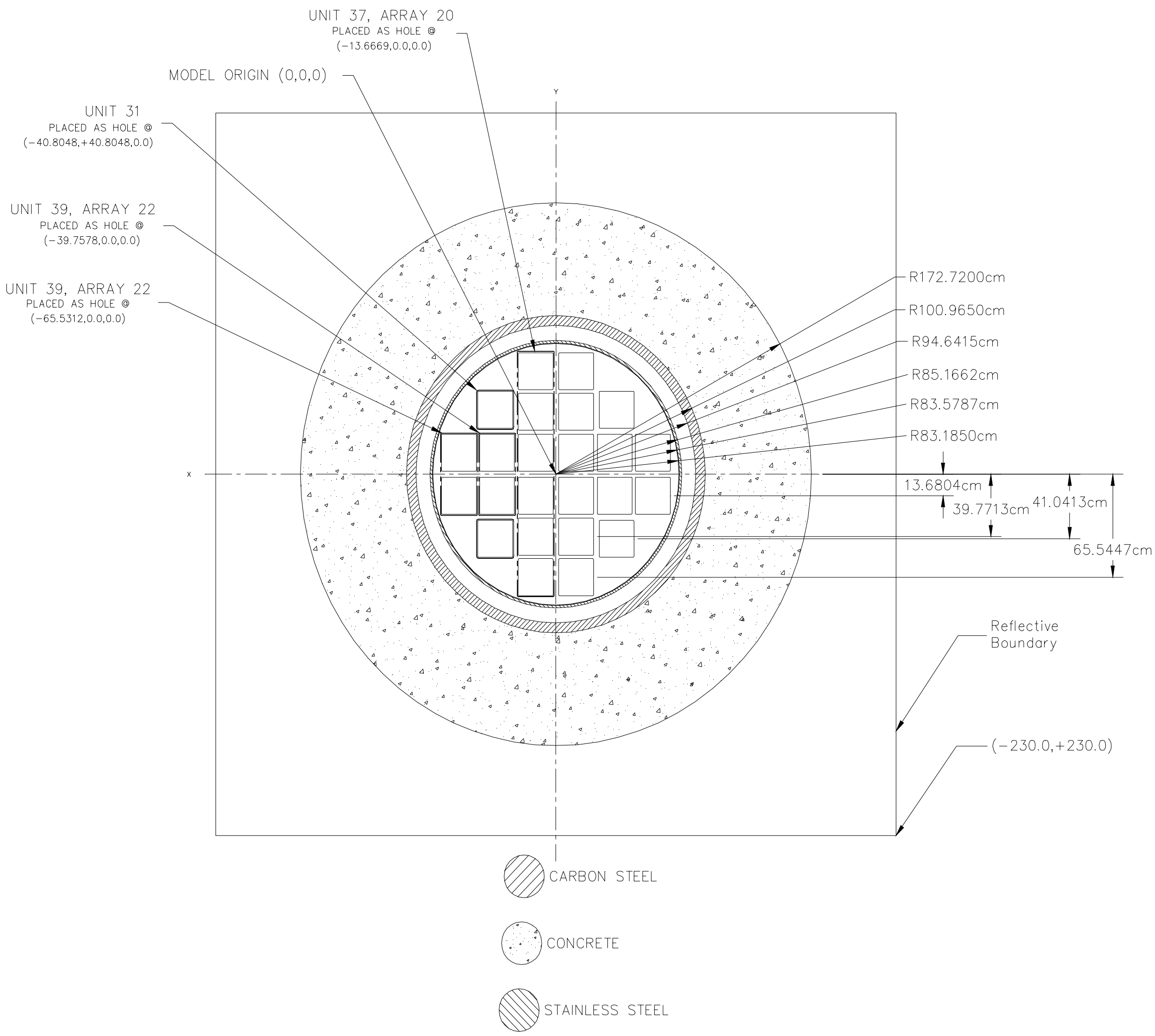


Figure 6.3-5 BWR KENO-Va Transfer Cask Model

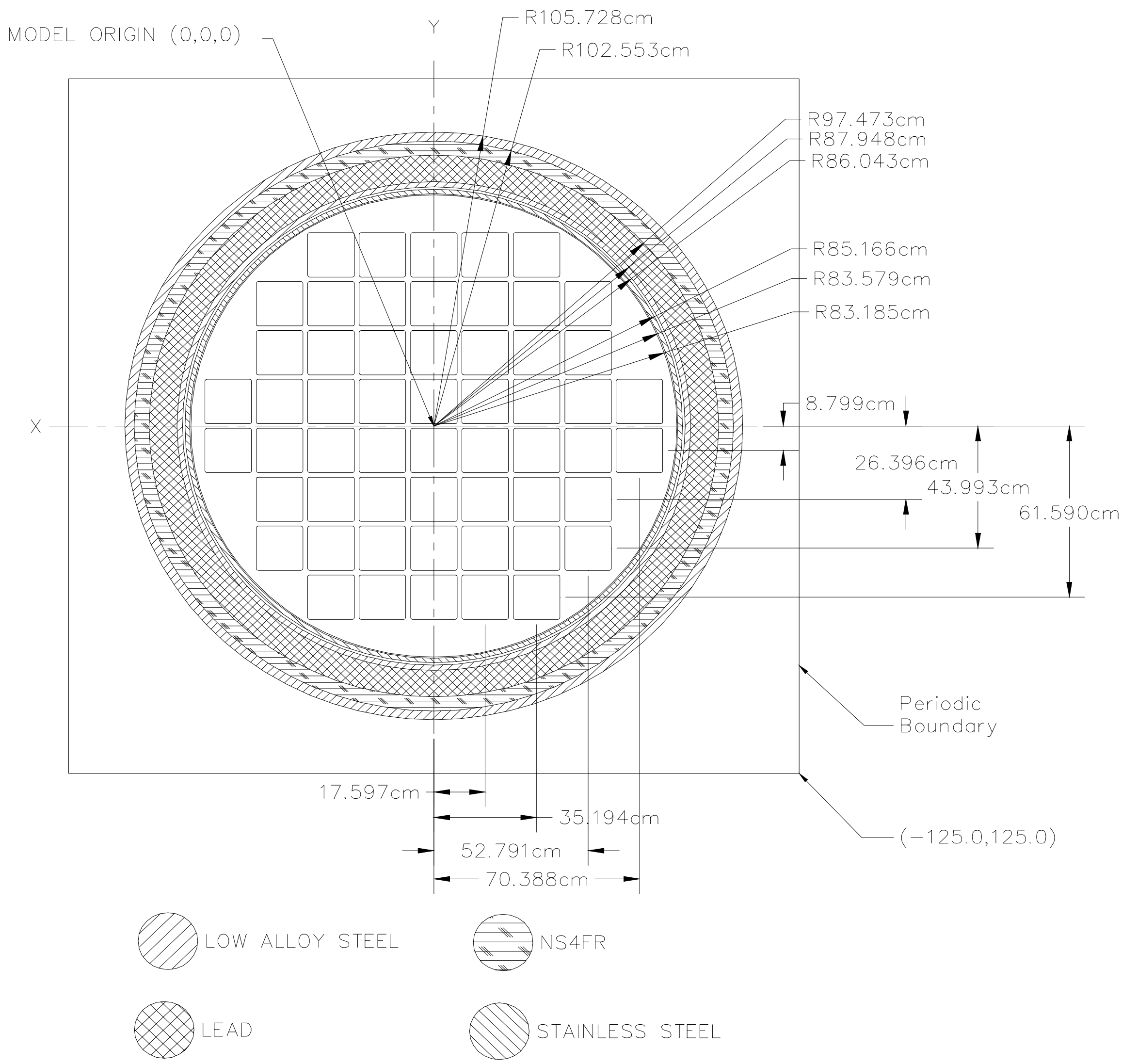


Figure 6.3-6 BWR KENO-Va Vertical Concrete Cask Model

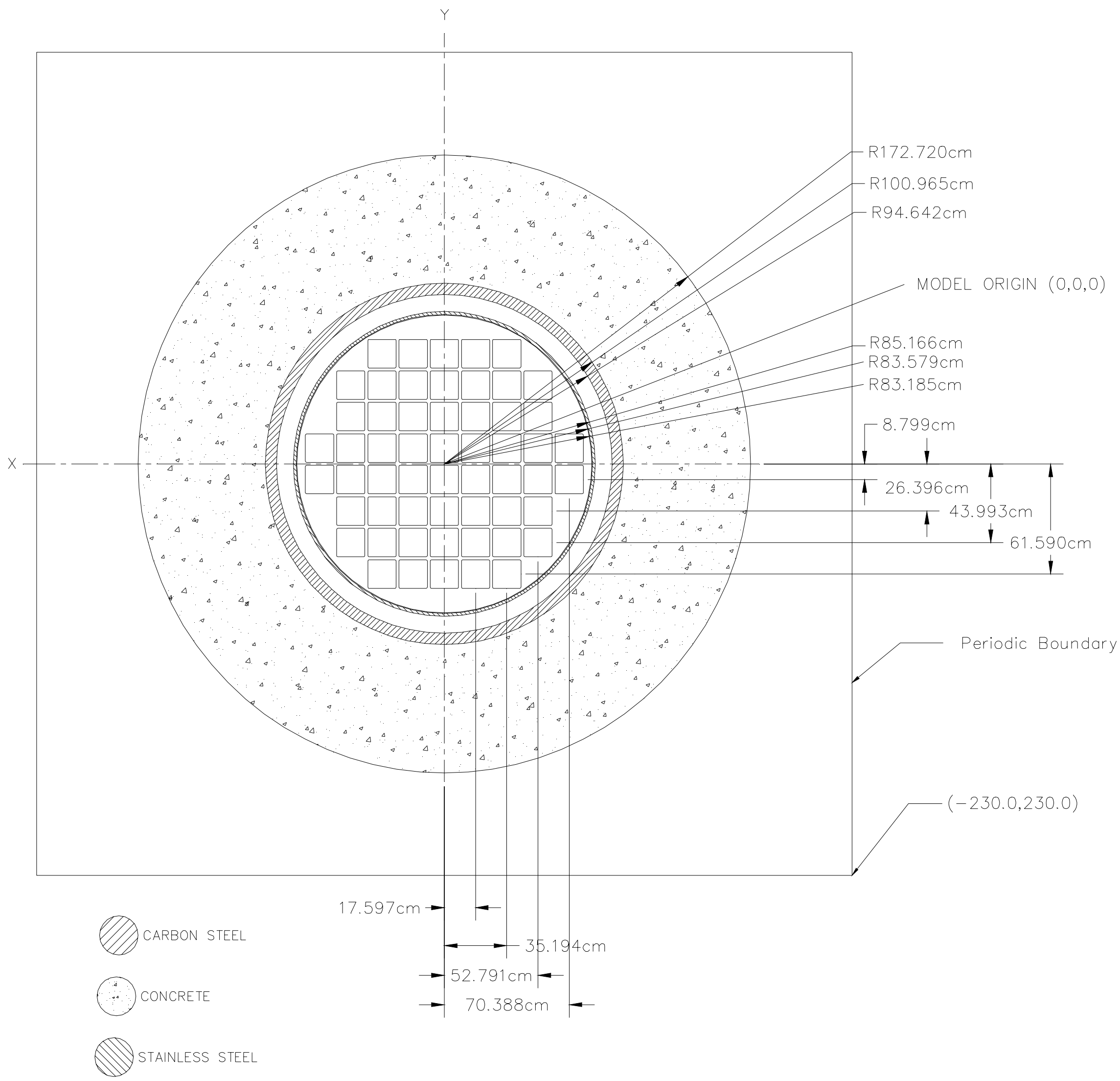


Figure 6.3-7 PWR Basket Criticality Control Design

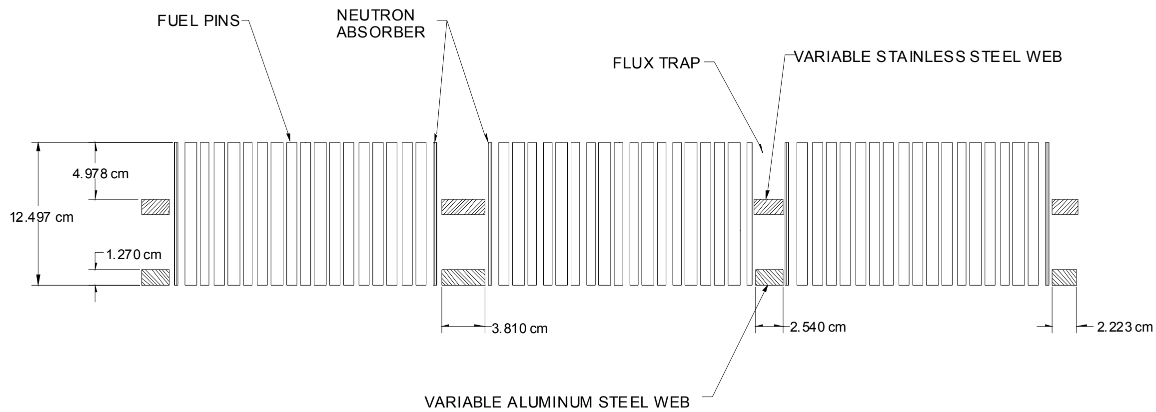


Figure 6.3-8 BWR Basket Criticality Control Design

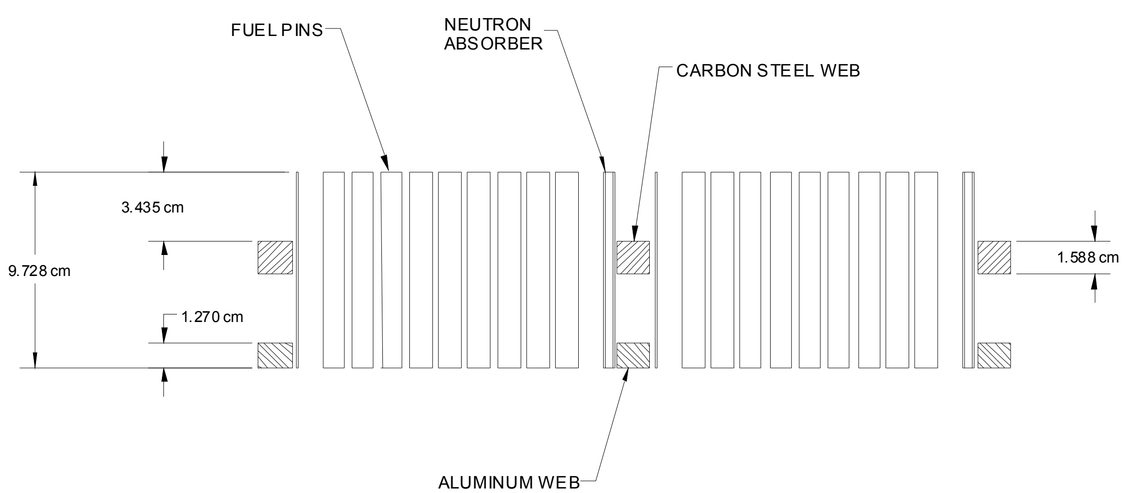


Figure 6.3-9 Standard Transfer Cask Containing a PWR Basket and Canister

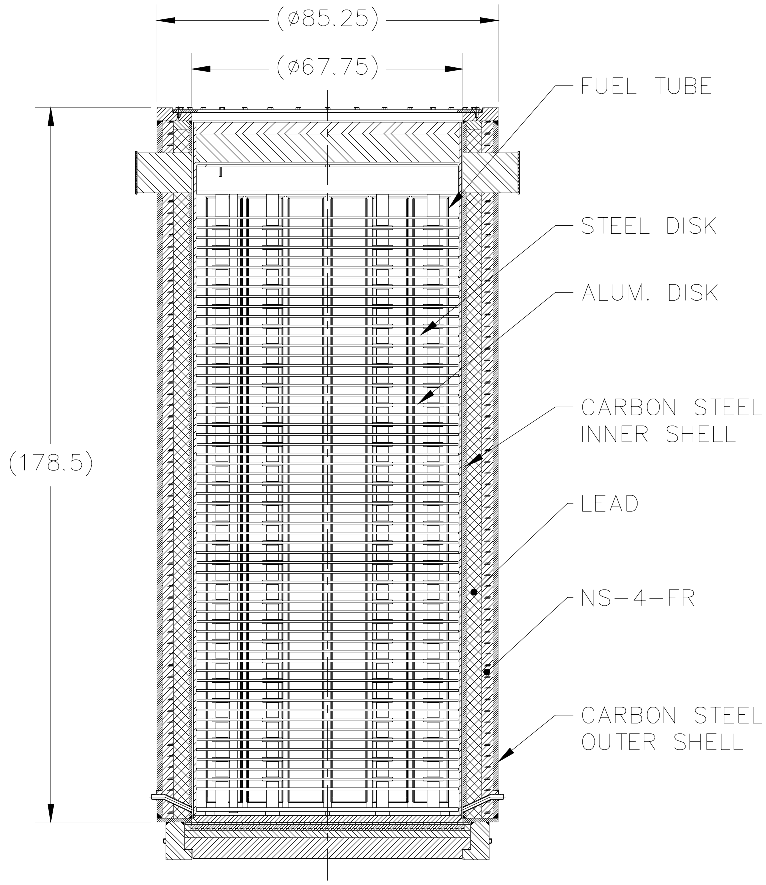
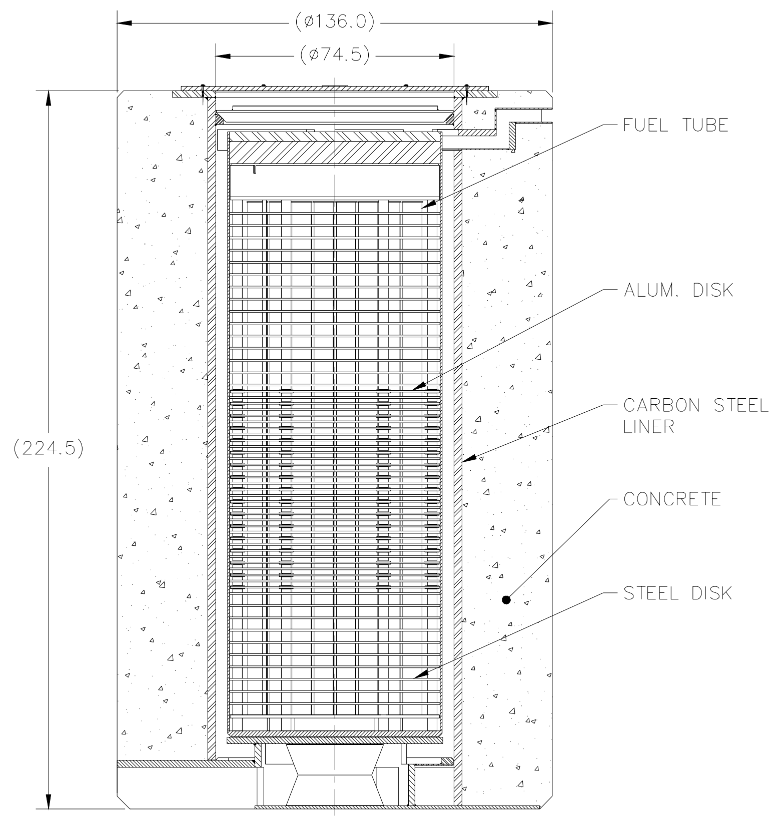


Figure 6.3-10 Vertical Concrete Cask Containing a BWR Basket and Canister



6.4 Criticality Calculation

6.4.1 Calculational or Experimental Method

As discussed earlier, criticality analysis of the Universal Storage System involves identification of fuel arrays for analysis, determination of most reactive PWR and BWR assemblies, and cask criticality analysis. Section 6.4.5 augments the evaluation of the most reactive PWR and BWR assemblies by determining assembly specific maximum initial enrichments.

6.4.1.1 Determination of Fuel Arrays for Criticality Analysis

As shown previously, the maximum values for physical dimensions, cross-sections, and weights vary among the fuel assemblies. Therefore, qualitatively determining one enveloping assembly for the criticality analysis is difficult. Thus, a set of standard fuel arrays in the basket configuration are selected and modeled with KENO-Va. Since the assembly is considered to be axially infinite in length, the selected standard PWR and BWR arrays that bound other assemblies in their sub classes and are as follows.

PWR Fuel Assemblies

- B&W 15×15 Mark B
- B&W 17×17 Mark C
- CE 14×14
- CE 16×16 System 80
- Westinghouse 14×14
- Westinghouse 14×14 OFA
- Westinghouse 15×15
- Westinghouse 17×17
- Westinghouse 17×17 OFA
- Ex/ANF 14×14 (CE)
- Ex/ANF 14×14 (WE)
- Ex/ANF 15×15 (WE)
- Ex/ANF 17×17 (WE)

BWR Fuel Assemblies

- Ex/ANF 7×7
- Ex/ANF 8×8 (63)*
- Ex/ANF 8×8 (62)*
- Ex/ANF 9×9 (79)*
- Ex/ANF 9×9 (74)*
- GE 7×7
- GE 8×8 (63)*
- GE 8×8 (62)*
- GE 8×8 (60)*
- GE 9×9 (79)*
- GE 9×9 (74)*

*Number of Fuel Rods Shown in Parentheses

For the BWR arrays, variation in zirconium alloy channel thickness is also evaluated. Section 6.4.4 augments the assembly characteristics definition by evaluating the reactivity impact of variations in fuel rod pitch, pellet diameter, clad thickness and guide tube thickness.

6.4.1.2 Most Reactive Fuel Assembly Determination

To determine the most reactive assembly within each type of fuel, a KENO-Va calculation is performed for the PWR and BWR fuel assemblies identified in Section 6.4.1.1. The calculated k_{eff} values for the various classes of fuel are given in Tables 6.4-1 through 6.4-4. The model for the PWR and the BWR fuel assembly types is discussed in the following paragraphs. On the basis of this analysis, the Westinghouse 17×17 OFA fuel assembly is determined to be the most reactive PWR fuel assembly. The Ex/ANF 9 × 9 fuel assembly with 79 fuel rods is determined to be the most reactive BWR fuel assembly.

6.4.1.2.1 Most Reactive PWR Assembly Analysis

The most reactive assembly analysis is based on an infinite array of basket cells, Figure 6.3-1. The assembly is in the PWR basket surrounded by the steel tube, four neutron absorber sheets, neutron absorber cover sheets, water to disk gap and steel, aluminum or water disk material. For the most reactive assembly analysis, the assembly is centered in the tube and the tube centered in the disk opening. Web thickness of 1.5, 1.0 and 0.875 in. is present in the PWR basket. Web thickness is assumed to have minimal impact on the most reactive assembly analysis. Therefore, the analysis is performed for a web thickness of 1.0 inch.

The basket cell model requires four basket slices at the active fuel elevation: one at the stainless steel disk elevation and thickness, one at the aluminum disk elevations and thickness, and two of the water space between disks. By stacking four of the slices (water, steel, water, and aluminum) on top of one another and periodically reflecting the disk stack, an axially infinite fuel-assembly-in-basket model is created. By imposing reflective boundary conditions on the sides of the basket cell model an infinite x-y array is also created.

With the exception of the axial (z) length, identical KENO-Va units are constructed for fuel pins, guide/instrument tubes, and neutron absorber sheets in the water and disk slice. Neutron absorber sheet KENO-Va units are required, one sheet running parallel to the x-plane, and one for the y plane for disk and water elevations. Axial dimensions for these units are made equal to either the water gap between disks or the disk heights (stainless steel disk and aluminum disk). In this analysis, all unit cells, except for the global unit, are centered on themselves, which implies symmetric upper and lower z elevation bounds.

After establishing fuel pin, guide tube, instrument tubes and neutron absorber sheet KENO-Va units, the fuel assembly arrays are constructed. The fuel assembly array, composed of fuel pins and guide/instrument tubes, is surrounded by a water gap, the fuel tube, and a water gap equal in x, y dimensions to the exterior of the neutron absorber sheet. The neutron absorber sheets are placed as holes into the water cuboid surrounding the tube. The cuboid containing the neutron absorber sheets is then surrounded by a thin encapsulating shell and a water cuboid out to the disk opening. Surrounding the disk opening cuboid is either water or disk material out to one half the web thickness (in this case 0.5 in. of material). The fuel tube is centered in the disk opening and the assembly is centered in the tube.

Calculated values of k_{eff} for the PWR assemblies selected for most reactive assembly analysis are listed in Table 6.4-1. The table includes data for assemblies with water in the fuel-pellet-to-cladding gap and for assemblies with no water in the gap. Also included is a Δk between the dry and wet cases. Note, the k_{eff} values in Table 6.4-1 are for a representative 1.0 inch flux trap, a ^{10}B areal density of 0.02 g/cm^2 and represent an infinite array basket cells. Therefore, k_{eff} exceeds 0.95 for a number of the assemblies analyzed. The purpose of this table is to justify the most reactive assembly. The k_{eff} values of the transfer and storage casks with the most reactive assembly are below 0.95 with bias and uncertainty included.

Table 6.4-1 results are based on a web width of 1.0 inch. The basket centerline web thickness is 1.5 inch. To assure that the most reactive assembly calculation applies to the whole basket and to verify that web spacing does not impact results, Table 6.4-2 is generated to include reactivity data for the highest reactivity assemblies in a 1.5-inch web.

From the 1.0-inch web, dry gap analysis, the Westinghouse 15×15 fuel assembly has a 0.0005 higher k_{eff} than the Westinghouse 17×17 OFA assembly. However, given the 0.001 Monte Carlo uncertainty associated with the k_{eff} values calculated, no statistically significant difference exists between the k_{eff} values. The 1.5-inch web analysis results in a statistically significantly higher k_{eff} for the Westinghouse 17×17 OFA assembly than for the Westinghouse 15×15 assembly, a Δk_{eff} of +0.005. Therefore, the Westinghouse 17 × 17 OFA fuel assembly is selected as the most reactive design basis PWR fuel for criticality analysis.

6.4.1.2.2 <u>Most Reactive BWR Assembly Analysis</u>

The most reactive assembly analysis is based on the full cask (transfer or concrete cask) model. Assemblies in the BWR basket are surrounded by the assembly channel, channel-to-tube gap, steel fuel tube, neutron absorber sheet and neutron absorber cover sheet on applicable sides of the tube, water-to-disk gap, and steel and aluminum disk material. For the most reactive assembly analysis, the assembly is centered in the tube and the tube centered in the disk opening.

The full cask model requires four basket slices to be made at the active fuel elevation: one at the carbon steel disk elevation and thickness, one at the aluminum disk elevation and thickness, and two at the water space between disk elevation and thickness. Each of the disks containing the fuel tubes is surrounded by the canister shell and the cask radial shields. By stacking the three cask slices on top of one another and periodically reflecting the stack, an axially infinite cask model is built. Building an axially infinite model eliminates axial leakage. Into each of the basket slices, the 56 disk openings are inserted as KENO-Va HOLE's. Each of the disk openings contains a KENO-Va HOLE representing the fuel tube, which in turn has the fuel assembly, including channel, inserted as a HOLE. This modeling approach facilitates component movement, fuel tube or fuel assembly, by simply modifying the HOLE origin coordinate.

Calculated values of k_{eff} for the BWR assemblies selected for analysis of the most reactive assembly are provided in Tables 6.4-3 and 6.4-4. The table includes data for no water in the pellet-to-clad gap. As can be seen from the table, the most reactive is the Ex/ANF 9×9 fuel assembly with 79 fuel pins and 2 water rods. It is statistically significantly more reactive than any of the other BWR assemblies analyzed; therefore, no "wet" gap cases were analyzed. In addition, the BWR fuel assembly is analyzed with and without the channel. The channel is shown to have little effect on the criticality results.

6.4.1.3 Transfer Cask and Vertical Concrete Cask Criticality Analysis

The KENO-Va models employed in the criticality analysis of the transfer cask and the Vertical Concrete Cask are built on those developed in the most reactive assembly calculations (See Section 6.4.1.2). The criticality analysis for the transfer and concrete casks is performed in three steps.

1. Resolution of the criticality impact of mechanical perturbations and geometric tolerances on the basis of a fuel tube-in-basket model (PWR) and basket-in-cask-model (BWR) using the most reactive assembly.
2. Preparation of a basket-in-cask model (PWR) to evaluate the reactivity variation between normal and worst-case configuration (a BWR basket-in-cask model having been constructed in step 1 for the most reactive assembly analysis).

3. Evaluation of k_{eff} and k_s for a single transfer cask, a single concrete cask, and for an array of casks on the basis of the worst-case configured cask basket under normal and accident conditions.

Construction of the cask criticality models for normal and accident conditions involves modifications to moderator compositions, cask spacing, material in the gap between fuel pellet and clad, and cask neutron shield material description.

This section presents the evaluation of the standard transfer cask configuration in significant detail. The evaluation identifies the most reactive standard transfer cask conditions.

6.4.1.3.1 Standard Transfer Cask and Vertical Concrete Cask Containing PWR Fuel

Mechanical Perturbations and Geometric Tolerance: Fuel Tube in PWR Basket Unit Cell Model

Because of the gaps between the fuel assembly and the fuel tube, and between the fuel tube and disk opening, a certain amount of mechanical perturbation in the configuration is possible. In addition, manufacturing tolerances in the basket may cause variation in the gaps and basket disk fuel tube hole positions. The criticality impact of such mechanical variations is evaluated with a KENO-Va model of the PWR basket unit cell. The following mechanical and geometric perturbations are evaluated:

a. Fuel assembly movement in the fuel tube,
b. Fuel tube movement in the disk opening,
c. Variation in the basket fuel tube opening,
d. Variation in the disk opening, and
e. Variation in positioning of the disk opening,

Fuel assembly movement in the tube is based on the physical limits of the inside envelope of the tube and the width of the fuel assembly array. For the design basis fuel, the maximum movement within the tube is ± 0.184 in. (0.468 cm). As a result of PWR basket tube symmetry, only one movement direction requires analysis. Fuel assembly movement is bounded by shifting the fuel assembly to the upper right-hand corner of the basket tube. This corner movement maximizes the reactivity impact of movement in one direction.

Similarly, movement of the fuel tube is maximized by shifting to the upper right hand corner of the basket disk opening. The maximum tube movement in the basket disk opening is ± 0.095 in. (0.242 cm). The tube outer neutron absorber sheet, and neutron absorber cover sheet dimensions are moved based on the inner tube dimension plus the relevant material thickness.

Both the fuel assembly movement and the fuel tube movement are analyzed with periodic and mirrored boundary conditions. The periodic boundary condition approximates a shift of all assemblies/fuel tubes in the basket to one side (i.e., the upper right hand corner). The mirrored boundary approximates clusters of four assemblies or fuel tubes moved towards a central location.

Variation in the fuel tube opening is evaluated by adding or subtracting a tolerance of ± 0.030 in. (0.076 cm) to the nominal dimensions and adjusting the neutron absorber sheet and cover sheet positions accordingly. Variation in basket disk opening is modeled by adding or subtracting a tolerance of ± 0.015 in. (0.038 cm) to the nominal dimension of the opening. The tolerance on the opening size modifies the web thickness but does not impact tube positioning.

Variation in basket disk opening position is limited by the positional tolerance, within the diameter, of 0.015 in. (0.038 cm). As with the fuel assembly and tube movements, the reactivity effect of the opening position is maximized by shifting the opening to the upper right hand corner by 0.0053 in. $(0.0075^2/2)^{1/2}$ in both +x and +y directions. This minimizes the webbing and corresponding flux trap gap effectiveness.

The results of the PWR basket unit cell perturbation evaluations are shown in Table 6.4-5.

Mechanical Perturbations and Geometric Tolerance: PWR Basket in Cask

To establish the maximum credible k_{eff} for the PWR basket with design basis fuel, the mechanical perturbations and basket geometric tolerances, shown in previous sections to produce positive reactivity relative to the nominal configuration, are included in the full transfer cask model and the full concrete cask model. The mechanical variations which produce positive reactivity effects are as follows.

a. Maximum tube size,
b. Fuel assembly centered in tube,
c. Fuel tube with assembly centered moved towards the basket center, and
d. Disk opening coordinates moved toward the basket center.

The above conditions define the worst-case PWR basket configuration. The results are shown in Tables 6.4-6 and 6.4-7 for the transfer cask and the concrete cask, respectively. Side and corner shifts are included in the tables to provide a k_{eff} comparison to different orientation of the components in the casks.

An additional evaluation is made addressing tolerances associated with the neutron absorber sheet. The minimum neutron absorber sheet widths are included in the most reactive cask configuration in order to evaluate the potential reactivity effects from both manufacturing tolerances and shifting of the neutron absorber sheets beneath the cover plates. For this model, neutron absorber sheet widths are reduced by a total of 0.10 inches to 8.10 inches and all assemblies are shifted radially in towards the center of the cask. This results in a combined Δk_{eff} of +0.00246. However, incorporating this increase in reactivity, as derived from the worst case accident scenario, results in a k_s = 0.94749 which is below the NRC criticality safety limit of 0.95. This Δk_{eff} of +0.00246 is, therefore, added to the results of all bounding PWR fuel conditions of the storage cask array and the transfer cask array reported in Section 6.4.3.1.

PWR Criticality Calculations for Single Standard Transfer Cask and Array of Concrete Casks

Values of k_{eff} and k_s (the bias adjusted k_{eff}) are evaluated for a single transfer cask, a single concrete cask, and for an array of casks containing PWR fuel. The evaluation is based on the worst-case configured cask basket under normal operating (dry interior) and accident (wet interior, no neutron shield) conditions of storage. The k_{eff} produced by KENO-Va is adjusted according to the following equations to account for code bias and Monte Carlo uncertainty. KENO-Va bias is calculated to be 0.0052 with a one-sided 95/95 uncertainty factor of 0.0087 (See Section 6.5). Base model for the KENO-Va interior and exterior moderator variation is the "worst configuration, highest reactivity" basket inputs.

$$k_s = k_{eff} + \Delta k_{Bias} + \sqrt{\sigma_{Bias}{}^2 + (2 * \sigma_{mc})^2} \le 0.95$$

$$k_s = k_{eff} + 0.0052 + \sqrt{0.0087^2 + (2\sigma_{mc})^2} \le 0.95$$

where:

k_s = the calculated allowable maximum multiplication factor, k_{eff}, of system being evaluated for all normal or credible abnormal conditions or events.

k_{eff} = the KENO - Va calculated k_{eff}

σ_{mc} = KENO - Va calculated Monte Carlo error.

Results of the criticality calculations are provided in Section 6.4.3.2.

6.4.1.3.2 Standard Transfer Cask and Vertical Concrete Cask Containing BWR Fuel

Mechanical Perturbations and Geometric Tolerance: BWR Basket in Cask

The BWR basket is subject to the same types of mechanical perturbations and geometric tolerances, which have an impact on the criticality evaluation, as is considered for the PWR basket. However, due to the asymmetry of the BWR basket and the engineered placement of neutron absorber among the fuel tubes, a full basket surrounded by the cask shield regions is used in the evaluation of mechanical and geometric tolerances. As with the PWR basket, the following mechanical and geometric tolerances are evaluated:

a. Fuel assembly (with channel) movement in the tube,
b. Fuel tube movement in the disk opening,
c. Variation in the basket tube opening,
d. Variation in disk opening, and
e. Variation in positioning of the disk opening,

For the design basis fuel, the maximum fuel movement within the tube is ± 0.231 in. (0.587 cm). The maximum movement of the tube in the disk opening is ± 0.064 in. (0.165 cm). For the movement analysis, the components, fuel tube or assembly, are shifted radially inward, radially outward, left, right, top, bottom and to the four basket corner locations. Due to the asymmetric neutron absorber sheet pattern of the BWR basket, all ten movement directions are evaluated.

Variations in the tube opening are evaluated by adding or subtracting a tolerance of ± 0.02 in. (0.051 cm) to the nominal tube inner width. Tube outer, neutron absorber sheet, and neutron absorber cover sheet dimensions are adjusted accordingly. Variations in disk opening are also evaluated by adding or subtracting a tolerance of ± 0.015 in. to the nominal disk opening.

Variation in basket disk opening position is limited by the positional tolerance within a diameter of 0.015 in. As with the fuel assembly and tube movements, the reactivity effect of the opening position is maximized by shifting the opening to the upper right hand corner by 0.0053 in. $(0.0075^2/2)^{1/2}$ in both +x and +y directions. This minimizes the webbing and neutron absorber effectiveness.

The results are shown in Tables 6.4-8 and 6.4-9 for the transfer cask and the concrete cask, respectively. The mechanical perturbations that produce a significant positive reactivity are included in a full cask model to establish the maximum credible k_{eff} for the transfer cask and the Vertical Concrete Cask loaded with 4.00 wt % ^{235}U Ex/ANF 9×9 fuel assembly. The combination of the radial movement of the fuel assembly and the fuel tube towards the basket center results in the maximum positive reactivity. This configuration is defined to be the worst-case for the BWR basket.

An additional evaluation is made addressing tolerances associated with the neutron absorber sheet. The minimum neutron absorber sheet widths are included in an analysis of the most reactive cask configuration in order to evaluate the potential reactivity effects from both manufacturing tolerances and shifting of the neutron absorber sheets beneath the cover plates.

For this model, neutron absorber sheet widths are reduced by a total of 0.08 inches to 6.22 inches. The resulting change in reactivity is within the statistics of the Monte Carlo code. Therefore, it is appropriate to neglect these tolerances in the maximum reactivity BWR model.

In addition to the neutron absorber sheet width evaluation, an analysis modeling the four oversized fuel tubes is included. The oversized fuel tubes are 0.15 inch larger to allow space for assemblies with channels that are bowed or twisted. However, the spacer grids of the fuel assembly maintain the pitch of the fuel rod array. Therefore, the fuel rod lattice and rod dimensions are not changed by the minor distortions that occur in the channel. An additional BWR criticality analysis is added which conservatively models the four ‘oversized’ fuel tubes (with nominal (straight) fuel assemblies) shifted further in towards the center of the cask as far as physically possible. This geometry minimizes the distance between the absorber sheets of the neighboring fuel tubes. This results in a k_{eff} of 0.91032. The change in reactivity, a Δk_{eff} of +0.00105, is within 2σ of the base case. Therefore, no statistically significant conclusion can be made as to the actual impact of the model change, and the existing most reactive configuration is left unchanged.

BWR Criticality Calculations for Single Standard Transfer Cask and Array of Concrete Casks

Values of k_{eff} and k_s (the bias adjusted k_{eff}) are evaluated for a single transfer cask, a single Vertical Concrete Cask, and for arrays of casks containing BWR fuel. The evaluation is based on the worst-case configured cask basket under normal operating (dry interior) and accident (wet interior, no neutron shield) conditions. The k_{eff} produced by KENO-Va is adjusted according to the following equation (the same equation used for the PWR fuel criticality calculations - see Section 6.4.1.3.1). A KENO-Va bias of 0.0052 and a one-sided 95/95 uncertainty factor of 0.0087 are used in the BWR fuel criticality calculations.

$$k_s = k_{eff} + \Delta k_{Bias} + \sqrt{\sigma_{Bias}{}^2 + (2 * \sigma_{mc})^2} \leq 0.95$$

$$k_s = k_{eff} + 0.0052 + \sqrt{0.0087^2 + (2\sigma_{mc})^2} \leq 0.95$$

where:

k_s = calculated allowable maximum multiplication factor, k_{eff}, of the system being evaluated for all normal or credible abnormal conditions or events

k_{eff} = KENO - Va calculated k_{eff}

σ_{mc} = KENO - Va calculated Monte Carlo error.

The results of the criticality analysis for a single cask (transfer cask and concrete cask) and for arrays of casks under normal, off-normal (concrete cask only), and accident conditions are provided in Section 6.4.3.3.

Homogeneous versus Heterogeneous Assembly Enrichment Evaluation

BWR fuel assemblies are typically loaded with a heterogeneous enrichment scheme of multiple fuel pin enrichments in one assembly. For the criticality analysis presented previously, a initial peak planar-average enrichment is used. The initial peak planar-average enrichment is the maximum planar-average enrichment at any height along the axis of the fuel assembly. This section demonstrates that the use of a planar-average enrichment provides a conservative eigenvalue compared to the heterogeneous fuel assembly. Three fuel assembly loading patterns are evaluated using both homogeneous and heterogeneous enrichment schemes and the resulting eigenvalues are compared. No gadolinium poisons are included in any of the models.

Fuel assembly types studied are the GE 8 × 8 60 and 62 fuel rod assembly types, the GE 9 × 9 74 fuel rod and the Ex/ANF 74 fuel rod assembly type. Each of the fuel assemblies is evaluated at a planar-average homogeneous enrichment and the actual documented enrichment pattern. In addition to actual documented enrichment patterns, BWR assemblies are analyzed at a planar-average enrichment of 3.75 and 4.0 wt % ^{235}U (4.0 wt % being the UMS® BWR design basis enrichment). Also evaluated is the impact of rotating water holes inside the assembly and the generation of a hypothetical enrichment pattern with 5.0 wt % enriched fuel surrounding the central water holes. Results of the heterogeneous versus homogeneous analyses, listed in Table 6.4-10, shows that for all cases, the heterogeneous enrichment produces a lower k_{eff} than the homogeneous bundle average enrichment case. This demonstrates that applying the bundle average enrichment provides a conservative estimate of the cask k_s. The maximum and minimum pin enrichments in each of the assemblies evaluated are listed in Table 6.4-10.

In addition to the homogeneous versus heterogeneous eigenvalue comparison, an in-core k_∞ for the GE 8×8-62 fuel rod assembly is calculated. The in-core k_∞ of the design basis BWR fuel assembly is 1.41. This fuel assembly design basis reactivity is much higher than is typically allowed for BWR fuel in the core.

6.4.2 Fuel Loading Optimization

The fuel loading is optimized in the Universal Storage System criticality models by using: 1) fresh fuel; 2) the most reactive PWR or BWR fuel assembly type; 3) the highest possible fuel stack density (95% of theoretical); and 4) the most reactive basket configuration. The cask models represent fully loaded baskets with 24 PWR or 56 BWR design basis fuel assemblies. The models use reflective boundary conditions on the sides and periodic boundary conditions on the top and bottom. These boundary conditions simulate an infinite array of casks of infinite axial extent.

6.4.3 Criticality Results

6.4.3.1 Summary of Maximum Criticality Values

The effective neutron multiplication factor, k_s, for the standard transfer cask and the Vertical Concrete Cask containing the most reactive PWR or BWR fuel assemblies in the most reactive configuration is below the 0.95 NRC criticality safety limit, including all biases and uncertainties, under normal, off normal and accident conditions.

Criticality Values for the Standard Transfer Cask

The maximum neutron multiplication factor with uncertainties for the standard transfer cask containing PWR fuel assemblies is 0.93921 under normal transfer conditions and 0.94749 under accident conditions. For the standard transfer cask containing BWR fuel, the multiplication factor is 0.91919 under normal transfer conditions and 0.92235 under accident conditions. These values reflect the following conditions:

- A method bias and uncertainty associated with KENO-Va and the 27 group ENDF/B-IV library
- An infinite cask array (even though there will only be one built)
- Full interior, exterior and fuel clad gap moderator (water) density
- 24 Westinghouse 17×17 OFA fuel assemblies at 4.2 wt % ^{235}U (most reactive PWR fuel assembly type) or 56 Ex/ANF 9×9-79 rod fuel assemblies at 4.00 wt % ^{235}U (most reactive BWR fuel assembly type)
- No fuel burnup
- 75% of nominal ^{10}B loading in the neutron absorber
- Most reactive mechanical configuration for PWR: (Assemblies and fuel tubes moved toward the center of the basket; maximum fuel tube openings; minimum neutron absorber sheet widths and closely packed disk openings)
- Most reactive mechanical configuration for BWR (Assemblies and fuel tubes moved toward the center of the basket)

Analysis of moderator density variation inside the transfer cask basket shows a monotonic decrease in reactivity with decreasing moderator density. Thus, the full moderator density situation bounds draining and drying operations in the transfer cask. As shown in Sections 6.4.3.2 and 6.4.3.3, the change in reactivity between the two transfer cask configurations is within the statistics of the Monte Carlo code (2σ) for the most reactive conditions.

Criticality Values for the Vertical Concrete Storage Cask

The maximum multiplication factor with uncertainties for the Vertical Concrete Cask containing PWR fuel assemblies is 0.38329 under normal storage conditions, 0.37420 under off-normal conditions and 0.94704 under accident conditions involving full moderator intrusion.

Corresponding values for the cask containing BWR fuel assemblies are 0.38168 under normal storage conditions, 0.38586 under off-normal conditions and 0.92332 under accident conditions involving full moderator intrusion. These values reflect the following conditions:

- A method bias and uncertainty associated with KENO-Va and the 27 group ENDF/B-IV library
- An infinite cask array
- Normal condition is defined to be a dry basket, dry heat transfer annulus and dry exterior
- Accident condition is defined to be full interior, exterior and fuel clad gap moderator (water) intrusion
- Westinghouse 17×17 OFA fuel assemblies at 4.2 wt % ^{235}U (most reactive PWR fuel assembly type) or 56 Ex/ANF 9×9-79 rod fuel assemblies at 4.0 wt % ^{235}U (most reactive BWR fuel assembly type)
- No fuel burnup
- 75% of nominal ^{10}B loading in the neutron absorber
- Most reactive mechanical configuration for PWR (assemblies and fuel tubes moved toward the center of the basket; maximum fuel tube openings; minimum neutron absorber sheet widths and closely packed disk openings)
- Most reactive mechanical configuration for BWR (assemblies and fuel tubes moved toward the center of the basket)

Analysis of simultaneous moderator density variation inside and outside the concrete cask shows a monotonic decrease in reactivity with decreasing moderator density. Thus, the full moderator density situation bounds any off normal or accident condition. Analysis of moderator intrusion into the cask heat transfer annulus with a dry canister shows a slight decrease in reactivity from the completely dry situation. This is due to better neutron reflection from the concrete cask steel shell and concrete shielding with no moderator present.

Analysis of the BWR cask reactivity of the fuel assemblies in the axial region above the top of partial length rods shows this region to be less reactive than the region with all of the fuel rods present. Therefore, it is appropriate to represent partial length rods as full length rods in the BWR fuel models.

6.4.3.2 Criticality Results for PWR Fuel

Transfer Cask

Results of the calculations for the standard transfer cask containing PWR fuel are provided in Tables 6.4-11 through 6.4-13. The tables list k_s without the Δk penalty associated with neutron absorber plates. A Δk of 0.00246 is added in the k_s listed below. CSAS input for the normal conditions analysis for the standard transfer cask is provided in Figure 6.8-1. Figure 6.8-2 provides CSAS input for the standard transfer cask analysis under hypothetical accident conditions.

Under normal conditions involving loading, draining and drying, the maximum k_{eff} including bias and uncertainties (k_s) is 0.93921 for the standard transfer cask. In the accident situation involving fuel failure and moderator intrusion, the maximum k_{eff} including biases and uncertainties (k_s) is 0.94749. Thus, the multiplication factor for the standard transfer cask containing 24 design basis PWR fuel assemblies of the most reactive type in the most reactive configuration is below the NRC criticality safety limit of 0.95 including all biases and uncertainties under normal, and accident conditions.

Vertical Concrete Cask

Results of the calculations for the Vertical Concrete Cask containing PWR fuel are provided in Tables 6.4-14 through 6.4-16. Figure 6.8-3 provides CSAS input for the analysis of the cask under normal conditions. Figure 6.8-4 provides CSAS input for the concrete cask analysis for hypothetical accident conditions.

Under normal dry conditions, maximum k_{eff} including biases and uncertainty (k_s) is 0.38329 for the concrete cask. Under off-normal conditions involving flooding of the heat transfer annulus, the k_s of the cask is even less (0.37420). Under accident conditions involving full moderator intrusion into the canister and fuel clad gap, the maximum k_s of the concrete cask is 0.94704. Thus, the multiplication factor for the concrete cask containing 24 design basis PWR fuel assemblies of the most reactive type in the most reactive configuration is below the NRC criticality safety limit of 0.95 including all biases and uncertainties under normal, off-normal, and accident conditions.

6.4.3.3 Criticality Results for BWR Fuel

Transfer Cask

Results of the criticality calculations for the standard transfer cask containing BWR fuel are provided in Tables 6.4-17 through 6.4-19. CSAS input for the normal conditions analysis for the standard transfer cask are provided in Figure 6.8-5. Figure 6.8-6 provides CSAS input for the analysis for the standard transfer cask hypothetical accident conditions.

As the tables show, under normal conditions involving loading, draining and drying, the maximum k_{eff} including bias and uncertainties is 0.91919 for the standard transfer cask. In the accident condition involving fuel failure and moderator intrusion, the maximum k_{eff} including biases and uncertainties is 0.92235. Thus, the multiplication factor for the transfer cask containing 56 design basis BWR fuel assemblies of the most reactive type in the most reactive configuration is below the NRC criticality safety limit of 0.95 including all biases and uncertainties under normal, and accident conditions.

Vertical Concrete Cask

Tables 6.4-20 through 6.4-22 provide results of the criticality calculations for the Vertical Concrete Cask containing BWR fuel assemblies. CSAS input for the normal condition analysis for the concrete cask are provided in Figure 6.8-7. Figure 6.8-8 provides CSAS input under hypothetical accident conditions.

For the concrete cask containing BWR fuel, under normal dry conditions, maximum k_{eff} including biases and uncertainty is calculated to be 0.38168. Under off-normal conditions involving flooding of the heat transfer annulus, the k_{eff} of the cask is 0.38586. Under accident conditions involving full moderator intrusion into the canister and fuel clad gap, the maximum k_{eff} of the concrete cask is 0.92332. Thus, the multiplication factor for the concrete cask containing 56 design basis BWR fuel assemblies of the most reactive type in the most reactive configuration is below the NRC criticality safety limit of 0.95 including all biases and uncertainties under normal, off-normal, and accident conditions.

6.4.4 Fuel Assembly Lattice Dimension Variations

The nominal lattice dimensions for the most reactive PWR and BWR fuel under the most reactive accident conditions are varied to determine if dimensional perturbations significantly affect the reactivity of the system. Accident conditions are defined to be full interior, exterior and fuel-clad gap moderator (water) intrusion at a density of 1 g/cc and a temperature of 70 ^{o}F. Flooding the fuel-clad gap magnifies the effect on reactivity from lattice dimensional variations by adding or removing moderator from the undermoderated fuel lattice. The conclusions drawn are then used to establish fuel dimension limits for the PWR and BWR fuel assemblies previously evaluated as UMS® contents nominal fuel assembly dimensions.

The PWR analysis is performed modeling a Westinghouse 17×17 OFA fuel assembly in an infinite array of infinitely tall fuel tube cells. This prevents any leakage of neutrons from the system. The BWR analysis is performed modeling an infinite array of infinitely tall Vertical Concrete Casks filled with Exxon/ANF 9×9 fuel assemblies. The following fuel assembly nominal lattice dimensions are modified to determine if these perturbations significantly affect the reactivity of the system:

a) Pellet Radius
b) Clad Inner Radius
c) Clad Outer Radius
d) Water Rod Inner Radius
e) Water Rod Outer Radius

As shown in Tables 6.4-23 and 6.4-24, the following dimensional perturbations were determined to significantly decrease the reactivity of both the PWR and the BWR systems: decreasing the clad inner radius and increasing the clad outer radius. Decreasing the pellet radius of the BWR fuel assembly was also determined to significantly decrease the reactivity. The results are as expected as these perturbations decrease the H/U ratio in the undermoderated fuel lattice. Additionally, varying the BWR water rod dimensions was determined to have an insignificant effect on the reactivity of the system. Therefore, these nominal dimension variations are of no concern with regards to the criticality safety of the system.

The following perturbations were determined to significantly increase the reactivity of both the PWR and BWR systems: increasing the clad inner radius and decreasing the clad outer radius, increasing the guide tube inner radius, decreasing the guide tube outer radius. The increase in reactivity is due to the fact that these perturbations increase the H/U ratio in the undermoderated fuel lattice.

An increase in reactivity was also seen in the PWR system when decreasing the pellet diameter. This slight increase in reactivity, 0.004 Δk, is due to flooding of the pellet-to-clad gap in the accident model, which provides additional moderator to the lattice. Since 100% of clad failure is not expected during normal or accident operating conditions, no lower bound limit is placed on the fuel pellet diameter.

The effect on reactivity from perturbations in the nominal fuel dimensions requires the following limits on the fuel assembly lattice parameters in order to retain the maximum reactivity of the UMS system below existing design basis results:

PWR

a) Fuel Rod Diameter ≥ Nominal Dimension
b) Clad Thickness ≥ Nominal Dimension
c) Fuel Rod Pitch ≤ Nominal Dimension
d) Guide Tube (Instrument Tube) Thickness ≥ Nominal Dimension
e) Pellet Diameter ≤ Nominal Dimension

BWR

a) Fuel Rod Diameter ≥ Nominal Dimension
b) Clad Thickness ≥ Nominal Dimension
c) Fuel Rod Pitch ≤ Nominal Dimension
d) Pellet Diameter ≤ Nominal Dimension

6.4.5 PWR and BWR Fuel Assembly Specific Maximum Initial Enrichments

After grouping the assemblies listed in Tables 6.2-1 and 6.2-2, according to the criteria presented in Section 6.4.4, each assembly group is evaluated at enrichments ranging up to 5.0 wt. % ^{235}U. Maximum initial enrichments are set by comparing the resulting reactivity from each of the runs to the upper safety limit (USL) of 0.9426.

6.4.5.1 PWR Maximum Initial Enrichment – No Soluble Boron

The various UMS® design basis fuel assembly groups are evaluated at enrichments ranging from 4.2 to 5.0 wt. % ^{235}U. For each of the cases, the most reactive configuration determined in Section 6.4.1 is employed. Rather than adding reactivity offsets for the shifted neutron absorber sheet, each of these cases contains a shifted, minimum width, neutron absorber sheet. The resulting $k_{eff} + 2\sigma$ is compared to the USL of 0.9426. The reactivity of each of the bounding fuel groupings is listed in Table 6.4-25. A summary maximum enrichment table for all standard PWR fuel types, including the critical fuel dimensions, is shown in Table 6.1-1. To simplify model construction, guide tubes larger than one lattice location are conservatively neglected from the model. This results in N/A (not applicable) entries in Table 6.1-1.

The maximum enrichment for the Maine Yankee fuel data set was determined to be 4.7 wt. % ^{235}U at a $k_{eff} + 2\sigma$ of 0.9404.

6.4.5.2 PWR Storage Cask Result Verification

To verify that the reactivity of the canister evaluated in the transfer configuration is not significantly different in reactivity to that of the storage configuration, a simple comparison for the Westinghouse 17×17 OFA (See we17b in Table 6.1-1) assembly is made at an enrichment of 4.2 wt. % ^{235}U inside the storage cask. Cases are executed with and without soluble boron in the moderator.

Executing the cases results in a k_{eff} of 0.9346 for the unborated water case and 0.8175 for the borated water case. The storage case is 0.0001 Δk higher than that of the transfer cask, while the difference in the borated case is 0.0016 Δk. Both runs validate the use of the transfer cask results for both transfer and storage operations.

6.4.5.3 BWR Maximum Initial Enrichment – No Soluble Boron

Each of the BWR fuel assembly groups is evaluated at enrichments ranging from 4.0 wt. % ^{235}U (UMS® design basis) to 5.0 wt. % ^{235}U. The resulting k_{eff} + 2σ is compared to the USL of 0.9426. The reactivity of each of the bounding fuel groupings is listed in Table 6.4-26. A summary maximum enrichment table for all standard BWR fuel types, including the critical fuel dimensions, is shown in Table 6.1-2. Similar to the PWR cask evaluations, a comparison analysis to the storage cask is made, demonstrating a slightly lower reactivity for the canister inside the concrete cask body.

6.4.6 PWR Soluble Boron Credit Evaluation

The maximum reactivity configuration employed in the previously described analysis results from the particular basket geometry that separates the fuel assemblies by borated aluminum sheets and water "flux traps." Filling the space with a water/soluble boron solution may result in a modified most reactive basket/fuel configuration. For the soluble boron analysis, the maximum reactivity configuration study is, therefore, repeated prior to the enrichment study. Also verified is the assumption that the maximum reactivity is achieved at full density water plus soluble boron. All analyses are based on 1000 ppm by weight of boron being present in the water spaces of the canister cavity. Water spaces include the flux traps, tube to assembly gap, lattice space between the rods and the pellet to clad gaps.

6.4.6.1 Maximum Reactivity Geometry

A limited evaluation of component tolerances and shifting is performed to verify the most reactive configuration for the PWR basket containing borated water. The assembly chosen for this evaluation is the Westinghouse 17×17 OFA (we17b) fuel assembly at 4.2 wt. % ^{235}U.

The key fabrication tolerance impacted variables evaluated are the size of the tube and disk opening and the location of the disk opening within the disk. Similar to the unborated evaluation, the maximum fuel tube opening increases reactivity in the shifted radial in configuration. While the maximum disk opening did not statistically impact the results of the evaluation, it is modeled at its maximum size for the enrichment search. These configuration changes make the soluble boron model consistent with that of the unborated cases.

Component movements evaluated are the fuel tube shifting within the disk opening and the assembly shifting within the tube. As shown in Table 6.4-27, the most reactive configuration is a shifted radial in fuel tube and assembly.

Also included in the evaluations is the shifted minimum neutron absorber width, since it will increase neutron interaction between assemblies. The result of the evaluation containing the maximum reactivity combination of parameters is included in Table 6.4-27.

6.4.6.2 Soluble Boron and Moderator Density Study

A moderator density study is performed to confirm that maximum reactivity occurs at full water density. Reducing water density in the borated cases not only reduces the moderating medium but also removes poison. As seen in Table 6.4-28, the maximum reactivity occurs at full density water.

6.4.6.3 Maximum Allowed Initial Enrichment Search

Similar to the unborated water configuration, the various UMS® design basis fuel assembly groups are evaluated at enrichments ranging from 4.2 to 5.0 wt. % ^{235}U. For each of the cases, the most reactive configuration determined in Section 6.4.6.1 is employed. The resulting $k_{eff} + 2\sigma$ is compared to the USL of 0.9426. The reactivity of each of the bounding fuel groupings is listed in Table 6.4-29. A summary maximum enrichment table for all standard PWR fuel types, including the critical fuel dimensions, is shown in Table 6.1-1.

To verify that a dry gap would not result in a more reactive configuration, the enrichment study is repeated with a dry gap. A dry gap has the potential for increasing reactivity due to the removal of the soluble boron. For all cases evaluated, reactivity decreased when the gap material was changed to “dry.”

Table 6.4-1 k_{eff} for Most Reactive PWR Fuel Assembly Determination

Assembly Type	Dry Pellet Clad Gap		Wet Pellet Clad Gap		Δk_{eff} [1]
	k_{eff}	σ	k_{eff}	σ	Wet - Dry
B&W 15×15 Mark B	0.9613	0.0011	0.9692	0.0012	0.0079
B&W 17×17 Mark C	0.9621	0.0012	0.9705	0.0011	0.0084
CE 14×14	0.9295	0.0013	0.9381	0.0011	0.0085
CE 16×16 SYS 80	0.9348	0.0012	0.9442	0.0012	0.0095
West 14×14	0.9177	0.0013	0.9264	0.0012	0.0086
West 14×14 OFA	0.9238	0.0012	0.9326	0.0012	0.0088
West 15×15	0.9662	0.0011	0.9712	0.0012	0.0050
West 17×17	0.9596	0.0012	0.9673	0.0012	0.0077
West 17×17 OFA	0.9656	0.0013	0.9727	0.0012	0.0070
Ex/ANF 14×14 CE	0.9309	0.0012	0.9362	0.0011	0.0053
Ex/ANF 14×14 WE	0.9065	0.0012	0.9176	0.0011	0.0111
Ex/ANF 15×15 WE	0.9559	0.0012	0.9634	0.0013	0.0074
Ex/ANF 17×17 WE	0.9631	0.0012	0.9704	0.0012	0.0073

1. Infinite Array of Basket Cells with a 1.0-inch Web.

Table 6.4-2 k_{eff} for Highest Reactivity PWR Fuel Assemblies

Assembly Type	k_{eff} [1]	σ
B&W 15×15 Mark B4	0.9119	0.0011
B&W 17×17 Mark C	0.9141	0.0011
West 15×15	0.9147	0.0013
West 17×17	0.9116	0.0012
West 17×17 OFA	0.9196	0.0012
Ex/ANF 17×17 WE	0.9172	0.0011

1. Infinite Array of Basket Cells with a 1.5-inch Web.

Table 6.4-3 k_{eff} for Most Reactive BWR Fuel Assembly Determination (Standard Transfer Cask)

Assembly Type	Number of Rods		Channel Thickness	Dry Gap	
	Fuel	Water		k_{eff}	σ
GE 7×7	49	0	80Mils	0.88240	0.00113
GE 8×8	63	1	80Mils	0.87868	0.00114
GE 8×8	63	1	100 Mils	0.87803	0.00116
GE 8×8	63	1	120 Mils	0.87709	0.00108
GE 8×8	62	2	80Mils	0.88130	0.00118
GE 8×8	62	2	100 Mils	0.88388	0.00110
GE 8×8	60	4	2mm	0.87917	0.00122
GE 9×9	79	2	2mm	0.87746	0.00115
GE 9×9	74	2(1)	2mm	0.87874	0.00114
GE 9×9	74	2(1)	80 Mils	0.88232	0.00114
Ex 7×7	49	0	80Mils	0.88070	0.00117
Ex 8×8-1	63	1	80Mils	0.87477	0.00111
Ex 8×8-2	62	2	80Mils	0.87778	0.00119
Ex 9×9	79	2	2mm	0.88498	0.00082
Ex 9×9	79	2	80Mils	0.88669	0.00081
Ex 9×9	74	2(1)	2mm	0.88594	0.00108

Note: (1) Two large water rods occupying the space of seven fuel rods.

Table 6.4-4 k_{eff} for Most Reactive BWR Fuel Assembly Determination (Vertical Concrete Cask)

Assembly Type	Number of Rods		Channel Thickness	Dry Gap	
	Fuel	Water		k_{eff}	σ
GE 7×7	49	0	80Mils	0.87876	0.00120
GE 8×8	63	1	80Mils	0.87850	0.00118
GE 8×8	63	1	100 Mils	0.87586	0.00111
GE 8×8	63	1	120 Mils	0.87612	0.00114
GE 8×8	62	2	80Mils	0.87917	0.00120
GE 8×8	62	2	100 Mils	0.88278	0.00119
GE 8×8	60	4	2mm	0.88093	0.00112
GE 9×9	79	2	2mm	0.87682	0.00115
GE 9×9	74	2	2mm	0.87645	0.00121
GE 9×9	74	2	80 Mils	0.88104	0.00113
Ex 7×7	49	0	80Mils	0.87910	0.00120
Ex 8×8-1	63	1	80Mils	0.87823	0.00111
Ex 8×8-2	62	2	80Mils	0.87640	0.00126
Ex 9×9	79	2	2mm	0.88794	0.00087
Ex 9×9	79	2	80Mils	0.88560	0.00077
Ex 9×9	74	2(2)	2mm	0.88571	0.00120

Table 6.4-5 PWR Fuel Tube in Basket Model KENO-Va Results for Geometric Tolerances and Mechanical Perturbations

	k_{eff}	σ	Δk_{eff}	$\Delta k_{eff}/\sigma$
Reference case	0.9582	0.0006		
Dimensions Tolerance on Disk Opening Center Location				
Minimum web	0.9598	0.0006	0.0015	2.6
Maximum web	0.9575	0.0006	-0.0008	-1.3
Dimensions tolerance on tube opening				
Minimum tube	0.9546	0.0006	-0.0036	-6.2
Maximum tube	0.9627	0.0006	0.0045	7.6
Dimension tolerance on disk opening				
Minimum opening	0.9594	0.0006	0.0012	2.0
Maximum opening	0.9591	0.0006	0.0008	1.4
Fuel movement in tube - tube centered in disk opening				
Mirrored boundary	0.9572	0.0006	-0.0011	-1.8
Periodic boundary	0.9566	0.0006	-0.0016	-2.8
Tube movement in disk opening - fuel assembly centered in tube				
Mirrored boundary	0.9606	0.0006	0.0024	4.0
Periodic boundary	0.9591	0.0006	0.0009	1.5
Move fuel tube in opening and assembly in tube				
Mirrored boundary	0.9595	0.0006	0.0012	2.1
Periodic boundary	0.9567	0.0006	-0.0015	-2.5

Table 6.4-6 PWR Basket in Transfer Cask KENO-Va Results for Geometric Tolerances and Tube Movement

Analysis	k_{eff}	σ	Δk_{eff}	$\Delta k_{eff}/\sigma$
Nominal	0.91306	0.00088	N/A	N/A
Nominal Wet Gap	0.92212	0.00085	0.00906	10.7
Geometric Tolerance	0.92278	0.00088	0.00972	11.0
Geo. Tol.+Tube In	0.93096	0.00084	0.01790	21.3
Geo. Tol.+Tube Out	0.91716	0.00086	0.00410	4.8
Geo. Tol.+Tube Side	0.92506	0.00083	0.01200	14.5
Geo. Tol.+Tube Corner	0.92275	0.00084	0.00969	11.5

Table 6.4-7 PWR Basket in Vertical Concrete Cask KENO-Va Results for Geometric Tolerances and Tube Movement

Analysis	k_{eff}	σ	Δk_{eff}	$\Delta k_{eff}/\sigma$
Nominal	0.91486	0.00087	N/A	N/A
Nominal Wet Gap	0.92266	0.00082	0.00780	9.5
Geometric Tolerance	0.92545	0.00086	0.01059	12.3
Geo. Tol.+Tube In	0.93052	0.00084	0.01566	18.6
Geo. Tol.+Tube Out	0.91659	0.00085	0.00173	2.0
Geo. Tol.+Tube Side	0.92415	0.00088	0.00929	10.6
Geo. Tol.+Tube Corner	0.92477	0.00082	0.00991	12.1

Table 6.4-8 BWR Basket in Transfer Cask KENO-Va Results for Geometric Tolerances and Mechanical Perturbations

Analysis	k_{eff}	σ	Δk_{eff}	$\Delta k_{eff}/\sigma$
Nominal Basket	0.88696	0.00082	N/A	N/A
Geometric Tolerances				
Min Tube	0.88401	0.00081	-0.00295	-3.642
Max Tube	0.88913	0.00084	0.00217	2.583
Min Disk Opening	0.88549	0.00083	-0.00147	-1.771
Max Disk Opening	0.88663	0.00081	-0.00033	-0.407
Shift Openings In	0.88659	0.00084	-0.00037	-0.440
Shift Openings Out	0.88434	0.00084	-0.00262	-3.119
Mechanical Perturbations				
Assembly Shift Top Right	0.86659	0.00086	-0.02037	-23.686
Assembly Shift Top	0.87661	0.00082	-0.01035	-12.622
Assembly Shift Top Left	0.88278	0.00087	-0.00418	-4.805
Assembly Shift Left	0.89037	0.00082	0.00341	4.159
Assembly Shift Bottom Left	0.89539	0.00081	0.00843	10.407
Assembly Shift Bottom	0.89270	0.00080	0.00574	7.175
Assembly Shift Bottom Right	0.88264	0.00083	-0.00432	-5.205
Assembly Shift Right	0.87691	0.00082	-0.01005	-12.256
Assembly Shift Radial In	0.89991	0.00080	0.01295	16.188
Assembly Shift Radial Out	0.87083	0.00082	-0.01613	-19.671
Fuel Tube Shift Top Right	0.88792	0.00084	0.00096	1.143
Fuel Tube Shift Top	0.88668	0.00085	-0.00028	-0.329
Fuel Tube Shift Top Left	0.88682	0.00086	-0.00014	-0.163
Fuel Tube Shift Left	0.88707	0.00083	0.00011	0.133
Fuel Tube Shift Bottom Left	0.88601	0.00081	-0.00095	-1.173
Fuel Tube Shift Bottom	0.88553	0.00086	-0.00143	-1.663
Fuel Tube Shift Bottom Right	0.88561	0.00082	-0.00135	-1.646
Fuel Tube Shift Right	0.88589	0.00083	-0.00107	-1.289
Fuel Tube Shift Radial In	0.89236	0.00081	0.00540	6.667
Fuel Tube Shift Radial Out	0.88287	0.00083	-0.00409	-4.928
Combined Analysis				
Tube + Assembly Radial In	0.90434	0.00082	0.01738	21.195

Table 6.4-9 BWR Basket in Vertical Concrete Cask KENO-Va Results for Geometric Tolerances and Mechanical Perturbations

Analysis	k_{eff}	σ	Δk_{eff}	$\Delta k_{eff}/\sigma$
Nominal Basket	0.88524	0.00078	N/A	N/A
Geometric Tolerances				
Min Tube	0.88476	0.00083	-0.00048	-0.578
Max Tube	0.88835	0.00082	0.00311	3.793
Min Disk Opening	0.88685	0.00081	0.00161	1.988
Max Disk Opening	0.88734	0.00082	0.00210	2.561
Shift Openings In	0.88740	0.00084	0.00216	2.571
Shift Openings Out	0.88627	0.00082	0.00103	1.256
Mechanical Perturbations				
Assembly Shift Top Right	0.86663	0.00087	-0.01861	-21.391
Assembly Shift Top	0.87675	0.00081	-0.00849	-10.481
Assembly Shift Top Left	0.88012	0.00084	-0.00512	-6.095
Assembly Shift Left	0.89115	0.00083	0.00591	7.120
Assembly Shift Bottom Left	0.89484	0.00083	0.00960	11.566
Assembly Shift Bottom	0.89129	0.00080	0.00605	7.563
Assembly Shift Bottom Right	0.88037	0.00081	-0.00487	-6.012
Assembly Shift Right	0.87643	0.00080	-0.00881	-11.013
Assembly Shift Radial In	0.89903	0.00081	0.01379	17.025
Assembly Shift Radial Out	0.86978	0.00086	-0.01546	-17.977
Fuel Tube Shift Top Right	0.88733	0.00084	0.00209	2.488
Fuel Tube Shift Top	0.88752	0.00084	0.00228	2.714
Fuel Tube Shift Top Left	0.88611	0.00086	0.00087	1.012
Fuel Tube Shift Left	0.88649	0.00084	0.00125	1.488
Fuel Tube Shift Bottom Left	0.88560	0.00083	0.00036	0.434
Fuel Tube Shift Bottom	0.88406	0.00082	-0.00118	-1.439
Fuel Tube Shift Bottom Right	0.88633	0.00084	0.00109	1.298
Fuel Tube Shift Right	0.88571	0.00084	0.00047	0.560
Fuel Tube Shift Radial In	0.89183	0.00083	0.00659	7.940
Fuel Tube Shift Radial Out	0.88298	0.00079	-0.00226	-2.861
Combined Analysis				
Tube + Assembly Radial In	0.90454	0.00083	0.01930	23.253

Table 6.4-10 Heterogeneous vs. Homogeneous Enrichment Analysis Results

Case		Enrichment (% ^{235}U)			Loading Pattern				
Array	Fuel Rods	Average	Min	Max	Heterog.	Homog.	k_{eff}	σ	Δk/σ
8×8	62	2.824	N/A	N/A		X	0.8024	0.0011	---
8×8	62	2.824	1.30	3.80	X		0.7894	0.0011	-12.28
8×8	62	3.750	N/A	N/A		X	0.8683	0.0011	---
8×8	62	3.750	1.73	3.98	X		0.8501	0.0011	-15.93
8×8	60	3.404	N/A	N/A		X	0.8418	0.0012	---
8×8	60	3.404	1.60	3.90	X		0.8364	0.0011	-4.53
8×8	60	3.750	N/A	N/A		X	0.8648	0.0012	---
8×8	60	3.750	1.76	4.35	X		0.8547	0.0011	-8.22
9×9	74	4.085	N/A	N/A		X	0.8884	0.0012	---
9×9	74	4.085	2.00	4.90	X		0.8785	0.0012	-8.37
9×9(1)	74	4.085	2.00	4.90	X		0.8809	0.0012	-6.31
9×9	74	3.750	N/A	N/A		X	0.8707	0.0011	---
9×9	74	3.750	1.84	4.50	X		0.8608	0.0011	-8.84
9×9(1)	74	3.750	1.84	4.50	X		0.8672	0.0011	-7.13
9×9	74	4.000	N/A	N/A		X	0.8839	0.0011	N/A
9×9	74	4.000	1.96	4.80	X		0.8759	0.0012	-7.06
9×9(2)	74	4.000	N/A	N/A		X	0.8890	0.0012	N/A
9×9(2)	74	4.000	1.96	4.80	X		0.8805	0.0012	-7.08
9×9(3)	74	4.000	3.68	5.00	X		0.8821	0.0012	-5.77

Notes:

(1) Rotated water holes.

(2) Exxon Assembly.

(3) Eighteen 5 wt% ^{235}U enriched rods near center of assembly.

Table 6.4-11 PWR Single Standard Transfer Cask Analysis Criticality Results

Water Density (g/cm^3)		Water in Gap	^{10}B	k_{eff}	σ	k_s [1]
Inside Canister	Outside Canister					
1.0	1.0	No	100%	0.91385	0.00088	0.92793
1.0	1.0	No	75%	0.92319	0.00086	0.93726
0.0001	1.0	No	75%	0.33461	0.00061	0.34860
1.0	1.0	Yes	100%	0.92116	0.00091	0.93525
1.0	1.0	Yes	75%	0.93052	0.00084	0.94458
0.0001	1.0	Yes	75%	0.33471	0.00064	0.34870

1. Does not include Δk of 0.00246 from neutron absorber plate evaluation.

Table 6.4-12 PWR Standard Transfer Cask Array Analysis Criticality Results - Normal Conditions

Water Density (g/cm³)		Water in Gap	^{10}B	k_{eff}	σ	k_s[1]
Inside Canister	Outside Canister					
1.0	1.0	No	75%	0.92225	0.00164	0.93675
0.9	0.9	No	75%	0.89374	0.00189	0.90843
0.8	0.8	No	75%	0.85650	0.00172	0.87106
0.6	0.6	No	75%	0.77522	0.00148	0.78961
0.4	0.4	No	75%	0.67495	0.00151	0.68936
0.2	0.2	No	75%	0.54140	0.00117	0.55561
0.1	0.1	No	75%	0.46986	0.00091	0.48395
1.0	0.0001	No	75%	0.91984	0.00171	0.93439
0.9	0.0001	No	75%	0.89131	0.00184	0.90596
0.8	0.0001	No	75%	0.85741	0.00171	0.87196
0.6	0.0001	No	75%	0.77648	0.00160	0.79095
0.4	0.0001	No	75%	0.67475	0.00153	0.68917
0.2	0.0001	No	75%	0.54275	0.00128	0.55702
0.1	0.0001	No	75%	0.47271	0.00088	0.48679

[1] Does not include Δk of 0.00246 from the neutron absorber plate evaluation.

Table 6.4-13 PWR Standard Transfer Cask Array Analysis Criticality Results - Accident Conditions

Water Density (g/cm³)		Water in Gap	^{10}B	k_{eff}	σ	k_s[1]
Inside Canister	Outside Canister					
1.0	1.0	Yes	75%	0.93096	0.00084	0.94502
0.9	0.9	Yes	75%	0.89908	0.00084	0.91314
0.8	0.8	Yes	75%	0.86363	0.00084	0.87769
0.6	0.6	Yes	75%	0.78291	0.00108	0.79707
0.4	0.4	Yes	75%	0.68031	0.00105	0.69446
0.2	0.2	Yes	75%	0.54830	0.00114	0.56249
0.1	0.1	Yes	75%	0.47334	0.00094	0.48744

[1] Does not include Δk of 0.00246 from the neutron absorber plate evaluation.

Table 6.4-14 PWR Single Vertical Concrete Cask Analysis Criticality Results

Water Density (g/cm³)		**Water in**				
Inside	**Outside**	**Gap**	**^{10}B**	**k_{eff}**	**σ**	**k_s[1]**
1.0	1.0	No	100%	0.91385	0.00088	0.92793
1.0	1.0	No	75%	0.92319	0.00086	0.93726
0.0001	1.0	No	75%	0.33461	0.00061	0.34860
1.0	1.0	Yes	100%	0.92116	0.00091	0.93525
1.0	1.0	Yes	75%	0.93052	0.00084	0.94458
0.0001	1.0	Yes	75%	0.33471	0.00064	0.34870

[1] Does not include Δk of 0.00246 from the neutron absorber plate evaluation.

Table 6.4-15 PWR Vertical Concrete Cask Array Analysis Criticality Results - Normal and Off-Normal Conditions

Water Density (g/cm³)						
Inside Canister	**Outside Canister**	**Water in Gap**	**^{10}B**	**k_{eff}**	**σ**	**k_s[1]**
0.0001	1.0	No	75%	0.33461	0.00061	0.34860
0.0001	0.9	No	75%	0.33453	0.00065	0.34853
0.0001	0.8	No	75%	0.33383	0.00061	0.34782
0.0001	0.6	No	75%	0.33542	0.00062	0.34941
0.0001	0.4	No	75%	0.33844	0.00064	0.35243
0.0001	0.2	No	75%	0.34600	0.00065	0.36000
0.0001	0.1	No	75%	0.35777	0.00057	0.37174
0.0001	0.0001	No	75%	0.36684	0.00064	0.38083

[1] Does not include Δk of 0.00246 from the neutron absorber plate evaluation.

Table 6.4-16 PWR Vertical Concrete Cask Array Analysis Criticality Results - Accident Conditions

Water Density (g/cm^3)						
Inside Canister	Outside Canister	Water in Gap	^{10}B	k_{eff}	σ	k_s[1]
1.0	1.0	Yes	75%	0.93052	0.00084	0.94458
0.9	0.9	Yes	75%	0.89707	0.00084	0.91113
0.8	0.8	Yes	75%	0.86351	0.00087	0.87758
0.6	0.6	Yes	75%	0.78276	0.00117	0.79697
0.4	0.4	Yes	75%	0.67967	0.00101	0.69380
0.2	0.2	Yes	75%	0.54104	0.00118	0.55525
0.1	0.1	Yes	75%	0.46245	0.00078	0.47649

[1] Does not include Δk of 0.00246 from the neutron absorber plate evaluation.

Table 6.4-17 BWR Single Standard Transfer Cask Analysis Criticality Results

Water Density (g/cm^3)						
Inside Canister	Outside Canister	Water in Gap	^{10}B	k_{eff}	σ	k_s
1.0	1.0	No	100%	0.88987	0.00083	0.90393
1.0	1.0	No	75%	0.90369	0.00082	0.91774
0.0001	1.0	No	75%	0.38112	0.00065	0.39512
1.0	1.0	Yes	100%	0.89298	0.00084	0.90704
1.0	1.0	Yes	75%	0.90710	0.00080	0.92115
0.0001	1.0	Yes	75%	0.38145	0.00067	0.39545

Table 6.4-18 BWR Standard Transfer Cask Array Analysis Criticality Results - Normal Conditions

Water Density (g/cm^3)						
Inside Canister	Outside Canister	Water in Gap	^{10}B	k_{eff}	σ	k_s
1.0	1.0	No	75%	0.90446	0.00081	0.91851
0.9	0.9	No	75%	0.88965	0.00081	0.90370
0.8	0.8	No	75%	0.87509	0.00081	0.88914
0.6	0.6	No	75%	0.83357	0.00079	0.84761
0.4	0.4	No	75%	0.76643	0.00075	0.78046
0.2	0.2	No	75%	0.64878	0.00115	0.66298
0.1	0.1	No	75%	0.55967	0.00105	0.57382
1.0	0.0001	No	75%	0.90513	0.00083	0.91919
0.9	0.0001	No	75%	0.88954	0.00080	0.90359
0.8	0.0001	No	75%	0.87540	0.00078	0.88944
0.6	0.0001	No	75%	0.83281	0.00111	0.84699
0.4	0.0001	No	75%	0.76682	0.00149	0.78122
0.2	0.0001	No	75%	0.65055	0.00122	0.66479
0.1	0.0001	No	75%	0.56286	0.00106	0.57701

Table 6.4-19 BWR Standard Transfer Cask Array Analysis Criticality Results - Accident Conditions

Water Density (gm/cm^3)						
Inside Canister	Outside Canister	Water in Gap	^{10}B	k_{eff}	σ	k_s
1.0	1.0	Yes	75%	0.90831	0.00079	0.92235
0.9	0.9	Yes	75%	0.89634	0.00086	0.91041
0.8	0.8	Yes	75%	0.87974	0.00080	0.89379
0.6	0.6	Yes	75%	0.83966	0.00082	0.85371
0.4	0.4	Yes	75%	0.76946	0.00071	0.78348
0.2	0.2	Yes	75%	0.65287	0.00062	0.66686
0.1	0.1	Yes	75%	0.55975	0.00101	0.57388

Table 6.4-20 BWR Single Vertical Concrete Cask Analysis Criticality Results

Water Density (g/cm^3)		Water in				
Inside	Outside	Gap	^{10}B	k_{eff}	σ	k_s
1.0	1.0	No	100%	0.88991	0.00077	0.90395
1.0	1.0	No	75%	0.90327	0.00078	0.91731
0.0001	1.0	No	75%	0.35531	0.00062	0.36930
1.0	1.0	Yes	100%	0.89567	0.00081	0.90972
1.0	1.0	Yes	75%	0.90842	0.00085	0.92248
0.0001	1.0	Yes	75%	0.35418	0.00070	0.36819

Table 6.4-21 BWR Vertical Concrete Cask Array Analysis Criticality Results - Normal and Off-Normal Conditions

Water Density (gm/cm^3)						
Inside Canister	Outside Canister	Water in Gap	^{10}B	k_{eff}	σ	k_s
0.0001	1.0	No	75%	0.35565	0.00069	0.36966
0.0001	0.9	No	75%	0.35586	0.00077	0.36990
0.0001	0.8	No	75%	0.35506	0.00074	0.36908
0.0001	0.6	No	75%	0.35674	0.00071	0.37076
0.0001	0.4	No	75%	0.35783	0.00072	0.37185
0.0001	0.2	No	75%	0.36488	0.00072	0.37890
0.0001	0.1	No	75%	0.37186	0.00065	0.38586
0.0001	0.0001	No	75%	0.36769	0.00061	0.38168

Table 6.4-22 BWR Vertical Concrete Cask Array Analysis Criticality Results - Accident Conditions

Water Density (gm/cm^3)						
Inside Canister	Outside Canister	Water in Gap	^{10}B	k_{eff}	σ	k_s
1.0	1.0	Yes	75%	0.90927	0.00081	0.92332
0.9	0.9	Yes	75%	0.89683	0.00083	0.91089
0.8	0.8	Yes	75%	0.87840	0.00077	0.89244
0.6	0.6	Yes	75%	0.83670	0.00078	0.85074
0.4	0.4	Yes	75%	0.76572	0.00069	0.77973
0.2	0.2	Yes	75%	0.64324	0.00062	0.65723
0.1	0.1	Yes	75%	0.54855	0.00049	0.56251

Table 6.4-23 PWR Lattice Parameter Study Criticality Analysis Results

Description	k_{eff}	σ	Δ k	2 σ	Δ k / 2 σ
base case	**0.9732**	**0.0008**	**----**	**0.0016**	**----**
decreases clad inner radius by 0.005 cm	0.9697	0.0008	-0.0035	----	-2.1875
increases clad inner radius by 0.005 cm	0.9784	0.0008	0.0052	----	3.2500
decreases clad outer radius by 0.005 cm	0.9782	0.0009	0.0050	----	3.1250
increases clad outer radius by 0.005 cm	0.9702	0.0009	-0.0030	----	-1.8750
decreases pellet radius by 0.005 cm	0.9744	0.0008	0.0012	----	0.7500
decreases pellet radius by 0.010 cm	0.9742	0.0008	0.0010	----	0.6250
decreases pellet radius by 0.015 cm	0.9773	0.0008	0.0041	----	2.5625
decreases pellet radius by 0.020 cm	0.9758	0.0008	0.0026	----	1.6250
decreases pellet radius by 0.025 cm	0.9761	0.0008	0.0029	----	1.8125
decreases pellet radius by 0.030 cm	0.9754	0.0008	0.0022	----	1.3750
decreases pellet radius by 0.035 cm	0.9750	0.0008	0.0018	----	1.1250
decreases pellet radius by 0.040 cm	0.9750	0.0008	0.0018	----	1.1250
increases pellet radius by 0.005 cm	0.9714	0.0009	-0.0018	----	-1.1250
decreases pellet & clad inner radii by 0.015 cm	0.9637	0.0008	-0.0095	----	-5.9375
decreases guide tube inner radius by 0.010 cm	0.9710	0.0008	-0.0022	----	-1.3750
increases guide tube inner radius by 0.015 cm	0.9753	0.0008	0.0021	----	1.3125
increases guide tube inner radius by 0.010 cm	0.9740	0.0009	0.0008	----	0.5000
decreases guide tube outer radius by 0.010 cm	0.9755	0.0008	0.0023	----	1.4375
increases guide tube outer radius by 0.015 cm	0.9712	0.0008	-0.0020	----	-1.2500
increases guide tube outer radius by 0.010 cm	0.9720	0.0008	-0.0012	----	-0.7500

Table 6.4-24 BWR Lattice Parameter Study Criticality Analysis Results

Description	k_{eff}	σ	Δ k	2 σ	Δ k / 2 σ
base case	**0.8904**	**0.0008**	**----**	**0.0016**	**----**
decreases clad inner radius by 0.005 cm	0.8889	0.0008	-0.0015	----	-0.9375
decreases clad inner radius by 0.008 cm	0.8874	0.0008	-0.0030	----	-1.8750
increases clad inner radius by 0.005 cm	0.8930	0.0008	0.0026	----	1.6250
decreases clad outer radius by 0.005 cm	0.8919	0.0008	0.0015	----	0.9375
decreases clad outer radius by 0.010 cm	0.8957	0.0008	0.0053	----	3.3125
increases clad outer radius by 0.005 cm	0.8885	0.0009	-0.0019	----	-1.1875
increases clad outer radius by 0.010 cm	0.8830	0.0009	-0.0074	----	-4.6250
decreases pellet radius by 0.005 cm	0.8896	0.0008	-0.0008	----	-0.5000
decreases pellet radius by 0.010 cm	0.8909	0.0008	0.0005	----	0.3125
decreases pellet radius by 0.015 cm	0.8881	0.0008	-0.0023	----	-1.4375
decreases pellet radius by 0.020 cm	0.8832	0.0008	-0.0072	----	-4.5000
decreases pellet radius by 0.025 cm	0.8867	0.0008	-0.0037	----	-2.3125
decreases pellet radius by 0.030 cm	0.8835	0.0008	-0.0069	----	-4.3125
decreases pellet radius by 0.035 cm	0.8837	0.0008	-0.0067	----	-4.1875
decreases pellet radius by 0.040 cm	0.8807	0.0008	-0.0097	----	-6.0625
increases pellet radius by 0.005 cm	0.8908	0.0008	0.0004	----	0.2500
increases pellet radius by 0.008 cm	0.8907	0.0009	0.0003	----	0.1875
decreases water rod inner radius by 0.010 cm	0.8908	0.0008	0.0004	----	0.2500
decreases water rod inner radius by 0.015 cm	0.8916	0.0008	0.0012	----	0.7500
increases water rod inner radius by 0.010 cm	0.8919	0.0008	0.0015	----	0.9375
increases water rod inner radius by 0.015 cm	0.8911	0.0008	0.0007	----	0.4375
decreases water rod outer radius by 0.010 cm	0.8901	0.0008	-0.0003	----	-0.1875
decreases water rod outer radius by 0.015 cm	0.8913	0.0008	0.0009	----	0.5625
increases water rod outer radius by 0.010 cm	0.8916	0.0008	0.0012	----	0.7500
increases water rod outer radius by 0.015 cm	0.8892	0.0009	-0.0012	----	-0.7500
replaces water rod with water	0.8926	0.0008	0.0022	----	1.3750

Table 6.4-25 PWR Maximum Allowable Enrichment – No Soluble Boron

Fuel Type	Enrichment (^{235}U wt%)	k_{eff}	σ	$k_{eff} + 2\sigma$
ce14a	5.0	0.9369	0.0008	0.9385
we14d	5.0	0.9359	0.0008	0.9375
ex14a	5.0	0.9184	0.0008	0.9200
we14a	5.0	0.9258	0.0008	0.9274
we14b	5.0	0.9340	0.0008	0.9356
ex15a	4.5	0.9363	0.0008	0.9379
we15a	4.4	0.9397	0.0008	0.9413
bw15a	4.4	0.9379	0.0008	0.9395
ce16e	4.8	0.9374	0.0008	0.9390
ex17a	4.4	0.9399	0.0008	0.9415
we17a	4.5	0.9385	0.0008	0.9401
we17b	4.3	0.9388	0.0008	0.9404
bw17a	4.4	0.9383	0.0008	0.9399
ce14MY	4.7	0.9404	0.0008	0.9420

Table 6.4-26 BWR Maximum Allowable Enrichment – No Soluble Boron

Fuel Type	Enrichment (^{235}U wt%)	k_{eff}	σ	$k_{eff} + 2\sigma$
ex07a	4.5	0.9403	0.0008	0.9419
ge07a	4.5	0.9375	0.0008	0.9391
ge07f	4.5	0.9381	0.0008	0.9397
ge07h	4.7	0.9402	0.0008	0.9418
ge08a	4.6	0.9379	0.0008	0.9395
ge08b	4.5	0.9322	0.0008	0.9338
ge08i	4.5	0.9391	0.0008	0.9407
ge08k	4.5	0.9369	0.0008	0.9385
ge08n	4.7	0.9368	0.0008	0.9384
ex08a	4.7	0.9394	0.0008	0.9410
ex08b	4.6	0.9355	0.0008	0.9371
ex09b	4.4	0.9361	0.0008	0.9377
ge09a	4.5	0.9391	0.0008	0.9407
ex09c	4.5	0.9404	0.0008	0.9420
ge09b	4.6	0.9390	0.0008	0.9406

Table 6.4-27 Most Reactive Geometry for a Borated Water PWR Canister

Tube Outer Width	Tube Thick.	Neutron Absorber Width	Disk Op. Width	Disk Op. Location	Rad Fuel Shift	Rad Tube Shift	Neutron Absorber Shift	$k_{eff} + 2\sigma$	Δk
Nom	Nom	Nom	Nom	Nom	Center	Center	No	0.8045	0.0000
Nom	Nom	Nom	Nom	Nom	In	In	No	0.8119	0.0074
Nom	Nom	Nom	Nom	Nom	In	In	Yes	0.8152	0.0107
Nom	Nom	Nom	Min	Nom	In	In	Yes	0.8142	-0.0010
Nom	Nom	Nom	Max	Nom	In	In	Yes	0.8151	-0.0001
Nom	Nom	Nom	Nom	Min	In	In	Yes	0.8158	0.0006
Nom	Nom	Nom	Nom	Max	In	In	Yes	0.8145	-0.0007
Min	Nom	Nom	Nom	Nom	In	In	Yes	0.8125	-0.0027
Max	Nom	Nom	Nom	Nom	In	In	Yes	0.8166	0.0014
Nom	Min	Nom	Nom	Nom	In	In	Yes	0.8140	-0.0012
Nom	Max	Nom	Nom	Nom	In	In	Yes	0.8149	-0.0003
Max	Nom	Min	Max	Nom	In	In	Yes	0.8175	0.0023

Table 6.4-28 Moderator Density versus Reactivity for the Borated Water Cases

Water Density[1] (g/cc)	k_{eff}	σ	$k_{eff} + 2\sigma$	Δk
0.9998	0.8159	0.0008	0.8175	0.0000
0.95	0.8065	0.0008	0.8081	-0.0094
0.9	0.7985	0.0008	0.8001	-0.0174
0.85	0.7900	0.0008	0.7916	-0.0259
0.8	0.7789	0.0008	0.7805	-0.0370
0.75	0.7689	0.0008	0.7705	-0.0470
0.7	0.7565	0.0008	0.7581	-0.0594
0.65	0.7437	0.0008	0.7453	-0.0722
0.6	0.7280	0.0008	0.7296	-0.0879
0.55	0.7125	0.0008	0.7141	-0.1034
0.5	0.6941	0.0008	0.6957	-0.1218
0.45	0.6732	0.0008	0.6748	-0.1427
0.4	0.6518	0.0008	0.6534	-0.1641
0.35	0.6257	0.0008	0.6273	-0.1902
0.3	0.5963	0.0008	0.5979	-0.2196
0.25	0.5658	0.0008	0.5674	-0.2501
0.2	0.5345	0.0008	0.5361	-0.2814
0.15	0.4985	0.0008	0.5001	-0.3174
0.1	0.4648	0.0008	0.4664	-0.3511
0.05	0.4332	0.0008	0.4348	-0.3827
0.0001	0.3605	0.0008	0.3621	-0.4554

Notes:

[1] Indicates water density prior to insertion of 1000 ppm boron.

Table 6.4-29 PWR Maximum Allowable Enrichment – Soluble Boron

Fuel Type	Enrichment (^{235}U wt%)	k_{eff}	σ	$k_{eff} + 2\sigma$
ce14a	5.0	0.8237	0.0008	0.8253
we14d	5.0	0.8231	0.0008	0.8247
ex14a	5.0	0.8034	0.0008	0.8050
we14a	5.0	0.8187	0.0008	0.8203
we14b	5.0	0.8097	0.0008	0.8113
ex15a	5.0	0.8460	0.0008	0.8476
we15a	5.0	0.8613	0.0008	0.8629
bw15a	5.0	0.8604	0.0008	0.8620
ce16e	5.0	0.8344	0.0008	0.8360
ex17a	5.0	0.8486	0.0008	0.8502
we17a	5.0	0.8606	0.0008	0.8622
we17b	5.0	0.8556	0.0008	0.8572
bw17a	5.0	0.8594	0.0008	0.8610

6.5 Critical Benchmark Experiments

Criticality code validation is performed for the CSAS analysis sequence in the SCALE 4.3 package in Section 6.5.1 and for the MONK8A code of the ANSWERS software package in Section 6.5.2.

6.5.1 SCALE 4.3 Benchmark Experiments and Applicability

This section provides the validation of the CSAS25 criticality analysis sequence contained in Version 4.3 of the SCALE package. CSAS includes the SCALE Material Information Processor, BONAMI-S, NITAWL-S, and KENO-Va. The Material Information Processor generates number densities for standard compositions, prepares geometry data for resonance self-shielding, and creates data input files for the cross-section processing codes. The BONAMI-S and NITAWL- S codes are used to prepare a resonance-corrected cross-section library in AMPX working format. The KENO-Va code uses Monte Carlo techniques to calculate the model k_{eff}. The 27-group ENDF/B-IV neutron cross-section library is used in this validation. The CSAS validation is required by the criticality safety standards ANSI/ANS-8.1 [11]. The section describes the method, computer program and cross-section libraries used, experimental data, areas of applicability, and bias and margins of safety.

ANSI/ANS-8.17 [12] prescribes the criterion to establish subcriticality safety margins. This criterion is as follows:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad (1)$$

where:

k_s = calculated allowable maximum multiplication factor, k_{eff}, of system being evaluated for all normal or credible abnormal conditions or events.

k_c = mean k_{eff} that results from calculation of benchmark criticality experiments using particular calculational method. If calculated k_{eff} values for criticality experiments exhibit trend with parameter, then k_c shall be determined by extrapolation based on best fit to calculated values. Criticality experiments used as benchmarks in computing k_c should have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of system being evaluated.

Δk_s = allowance for

a. statistical or convergence uncertainties, or both, in computation of k_s,
b. material and fabrication tolerances, and
c. geometric or material representations used in computational method.

Δk_c = margin for uncertainty in k_c which includes allowance for

a. uncertainties in critical experiments,
b. statistical or convergence uncertainties, or both, in computation of k_c,
c. uncertainties resulting from extrapolation of k_c outside range of experimental data, and
d. uncertainties resulting from limitations in geometrical or material representations used in computational method.

Δk_m = arbitrary margin to ensure subcriticality of k_s.

The various uncertainties are combined statistically if they are independent. Correlated uncertainties are combined by addition.

Equation 1 can be rewritten as:

$$k_s \leq 1 - \Delta k_m - \Delta k_s - (1-k_c) - \Delta k_c \qquad (2)$$

Noting that the NRC requires a 5% subcriticality margin ($\Delta k_m = 0.05$) and the definition of the bias ($\beta = 1-k_c$), the equation 2 can then be written as:

$$k_s \leq 0.95 - \Delta k_s - \beta - \Delta\beta \qquad (3)$$

where $\Delta\beta = \Delta k_c$. Thus, the k_s (the maximum allowable value for k_{eff}) must be below 0.95 minus the bias, uncertainties in the bias, and uncertainties in the system being analyzed (i.e., Monte Carlo, mechanical, and modeling). This is an upper safety limit criteria often used in the DOE criticality safety community.

Alternatively, equation 3 can be rewritten applying the bias and uncertainties to the k_{eff} of the system being analyzed as:

$$k_s \equiv k_{eff} + \Delta k_s + \beta + \Delta\beta \leq 0.95 \qquad (4)$$

In Equation 4, k_{eff} replaces k_s, and k_s has been redefined as the effective multiplication factor of the system being analyzed, including the method bias and all uncertainties. This is a maximum calculated k_{eff} criteria often used in light water reactor spent fuel storage and transport analyses.

Both β and $\Delta\beta$ are evaluated below for KENO-Va with the 27-group ENDF/B-IV library for use in criticality evaluations of light water reactor fuel in storage and transport casks.

6.5.1.1 Description of Experiments

The 63 critical experiments selected are as follows: nine B&W 2.46 wt % ^{235}U fuel storage [13], ten PNL 4.31 wt % ^{235}U lattice [14], twenty-one PNL 2.35 and 4.31 wt % ^{235}U with metal reflectors (Bierman, April 1979 and August 1981) [15, 16], twelve PNL flux trap [14, 17] and eleven VCML 4.74 wt % ^{235}U experiments, some involving moderator density variations [18]. These experiments span a range of fuel enrichments, fuel rod pitches, neutron absorber sheet characteristics, shielding materials and geometries that are typical of light water reactor fuel in a cask.

To achieve accurate results, three-dimensional models, as close to the actual experiment as possible, are used to evaluate the experiments. Stochastic Monte Carlo error is kept within ±0.1% by executing at least 1,000 neutrons/generation for more than 400 generations.

6.5.1.2 Applicability of Experiments

All of the experiments chosen in this validation are applicable to either PWR or BWR fuel. Fuel enrichments have covered a range from 2.35 up to 4.74 wt % ^{235}U, typical of light water reactor fuel presently used. The experiment fuel rod and pitch characteristics are within the range of standard PWR or BWR fuel rods (i.e., pellet OD from 0.78 to 1.2 cm, rod OD from 0.95 to 1.88 cm, and pitch from 1.26 to 1.87 cm). This is particularly true of the VCML (PWR rod type) and B&W experiments (BWR rod type). The H/U volume ratios of the experimental fuel arrays are within the range of PWR fuel assemblies (1.6 to 2.32) and BWR fuel assemblies (1.6 to 1.9). Experiments covered the geometry and neutron absorber sheet arrangements typical of NAC basket designs. Flux trap gap spacings of 3.81 cm such as those in the NAC-STC and UMS® PWR baskets and gap spacings as low 1.91 cm as in the NAC-MPC were included. ^{10}B neutron absorber loadings, also typical of NAC basket designs (0.005 to 0.025 $^{10}B/cm^2$), were included as well. The experiments addressed the influence of water and metal reflector regions, including steel and lead, that would be present in storage and transport cask shielding.

Confidence in predicting criticality, including bias and uncertainty, has been demonstrated for light water reactor fuel with enrichments up to 4.74 wt % ^{235}U and results indicate confidence well above 5 wt % ^{235}U. Confidence in predicting criticality has been demonstrated for storage and transport arrays in which critical controls consist of flux trap or single neutron absorber sheets or simple spacing. Confidence in predicting criticality has been demonstrated for light water reactor fuel storage and transport arrays next to water and metal reflector regions.

6.5.1.3 Results of Benchmark Calculations

The k-effective results for the experiments are shown in Table 6.5.1-1 and a frequency plot is provided in Figure 6.5.1-1. Five sets of cases are presented: Set 1, B&W; Set 2, PNL lattice; Set 3, PNL reflector; Set 4, PNL flux trap, and Set 5,VCML critical experiments. Sixty-three results are reported.

The overall average and standard deviation of the 63 cases is 0.9948±0.0044. The average Monte Carlo error (statistical convergence) is ±0.0012 for the 63 cases. This uncertainty component is statistically subtracted from the uncertainties, because it is previously included in the standard deviation. The KENO-Va models are three-dimensional, fully explicit representations (no homogenization) of the experimental geometry. Therefore, the uncertainty resulting from limitations of geometrical modeling is taken to be 0.0. The experiments modeled cover the range of fuel types, enrichments, neutron absorber configurations, neutron absorber B^{10} loading, and metal reflector effects so that no extrapolations are necessary outside the range of data, and the uncertainty resulting from extrapolation is also taken to be 0.0.

On the basis of the reported experimental error for the B&W cases, the reported error of the critical size number of rods for the PNL cases and the reported error for the critical height in the VCML cases, the experimental error is conservatively taken to be ±0.001. Criticality can then be represented as 1.000±0.001. This uncertainty component is statistically added to the sum of the other uncertainties, because the bias is the difference between two random variates (i.e., criticality and code prediction, and the uncertainty in the difference between two random variables is the statistical sum [(rms)] of their individual uncertainties).

Thus, the bias or average difference between code calculated and the critical condition is β=1-0.9948 = 0.0052. The uncertainty in the bias, accounting for the statistical convergence (Monte Carlo error) and the uncertainty in criticality is $(0.0044^2 - 0.0012^2 + 0.0010^2)^{1/2} = 0.0043$. For 63 samples of criticality, the 95/95 one-side tolerance factor is 2.012 [19]. The result is a 95/95 one-sided uncertainty in the bias of $\Delta\beta = 2.012 \times 0.0043 = 0.0087$. Equation 3 now becomes:

$$k_{eff} + \Delta k_s + 0.0052 + 0.0087 \leq 0.95 \qquad (5)$$

where Δk_s becomes the uncertainty in k_s resulting from Monte Carlo error, mechanical and material tolerances, and geometric or material representations. If the nominal representation of the system is evaluated for k_s, then the mechanical and material perturbations can be evaluated independently and can be combined statistically as the root sum of squares. If the worst-case mechanical and material tolerances are used to calculate k_s (e. g., 75% of boron loading and most reactive positioning of fuel or basket components), then Δk_s becomes 0.0 and the Monte Carlo error, σ_{mc}, can be combined statistically, because it is independent, with the uncertainty in the bias as:

$$k_{eff} + 0.0052 + \sqrt{0.0087^2 + (2\sigma)^2} \leq 0.95 \qquad (6)$$

6.5.1.4 Trends

Scatter plots of k_{eff} versus wt % ^{235}U, rod pitch, H/U volume ratio, average neutron group causing fission, ^{10}B loading for flux trap cases, and flux trap gap thickness are shown in Figures 6.5.1-2 through 6.5.1-7. Included in these scatter plots are linear regression lines with a corresponding correlation coefficient (r) to statistically indicate any trend or lack thereof. In particular, the correlation coefficient is a measure of the linear relationship between k_{eff} and a critical experiment parameter. If r is +1, a perfect linear relationship with a positive slope is indicated, and if r is -1, a perfect linear relationship with a negative slope is indicated. When r is 0, no linear relationship is indicated.

The largest correlation coefficient indicated in the plots is 0.3608 (k_{eff} versus enrichment) and the lowest is 0.0693 (k_{eff} versus ^{10}B loading in flux trap experiments). On the basis of the correlation coefficients, no statistically significant trends exist over the range of variables studied. Most importantly, no trend is shown with flux trap gap spacing and/or ^{10}B loading. This is the major criticality control feature of the UMS® Storage System basket.

6.5.1.5 Comparison of NAC Method to NUREG/CR-6361 – SCALE 4.3

NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages" (NUREG), provides a guide to LWR criticality benchmark calculations and the determination of bias and subcritical limits in critical safety evaluations. In Section 2 of the NUREG, a series of LWR critical experiments are described in sufficient detail for independent modeling. In Section 3, the critical experiments are modeled, and the results (k_{eff} values) are presented. The method utilized in the NUREG is KENO-Va with the 44 group ENDF/B-V cross-section library embedded in SCALE 4.3. Inputs are provided in Appendix A of the NUREG. In Section 4, a guide for the determination of bias and subcritical safety limits is provided based on ANSI/ANS-8.17 and statistical analysis of the trending in the bias. Finally, guidelines for experiment selection and applicability are presented in Section 5. The approach outlined in Section 4 of the NUREG is described in detail below and is compared to the NAC approach presented in Sections 6.5.1, 6.5.1.1 and 6.5.1.2.

NAC has performed an extensive LWR critical benchmarking as documented in Sections 6.5.1.1 and 6.5.1.2. The method used in NAC benchmarking/validation included the CSAS25 (KENO-Va) criticality analysis sequence, with the 27 group ENDF/B-IV library, contained in SCALE 4.3. Trending in k_{eff} was evaluated for the following independent variables: wt % ^{235}U, rod pitch, H/U volume ratio, average neutron group causing fission, ^{10}B loading for flux trap cases, and flux trap gap thickness. No statistically significant trends were found, and a constant bias with associated uncertainty was determined for criticality evaluation.

Both the NUREG/CR-6361 and the NAC approach to criticality evaluation start with ANSI/ANS-8.17 criticality safety criterion. This criterion is as follows:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad (1)$$

where:

k_s = calculated allowable maximum multiplication factor, k_{eff}, of the system being evaluated for all normal or credible abnormal conditions or events.

k_c = mean k_{eff} that results from a calculation of benchmark criticality experiments using a particular calculation method. If the calculated k_{eff} values for the criticality experiments exhibit a trend with an independent parameter, then k_c shall be determined by extrapolation based on best fit to calculated values. Criticality experiments used as benchmarks in computing k_c should have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of the system being evaluated.

Δk_s = allowance for:

a) statistical or convergence uncertainties, or both, in computation of k_s,
b) material and fabrication tolerances, and
c) geometric or material representations used in computational method.

Δk_c = margin for uncertainty in k_c which includes allowance for:

a) uncertainties in critical experiments,
b) statistical or convergence uncertainties, or both, in computation of k_c,
c) uncertainties resulting from extrapolation of k_c outside range of experimental data, and
d) uncertainties resulting from limitations in geometrical or material representations used in the computational method.

Δk_m= arbitrary administrative margin to ensure subcriticality of k_s

The various uncertainties are combined statistically if they are independent. Correlated uncertainties are combined by addition.

Equation 1 can be rewritten as:

$$k_s \leq 1 - \Delta k_m - \Delta k_s - (1 - k_c) - \Delta k_c \qquad (2)$$

Noting that the definition of the bias is $\beta = 1 - k_c$, Equation 2 can be written as:

$$k_s + \Delta k_s \leq 1 - \Delta k_m - \beta - \Delta\beta \qquad (3)$$

where $\Delta\beta = \Delta k_c$. Thus, the maximum allowable value for k_{eff} plus uncertainties in the system being analyzed must be below 1 minus an administrative margin (typically 0.05), which includes the bias and the uncertainty in the bias. This can also be written as:

$$k_s + \Delta k_s \leq \text{Upper Subcritical Limit (USL)} \qquad (4)$$

where:

$$USL \equiv 1 - \Delta k_m - \beta - \Delta\beta \qquad (5)$$

This is the Upper Subcritical Limit criterion as described in Section 4 of NUREG/CR-6361. Two methods are prescribed for the statistical determination of the USL: Confidence Band with

Administrative Margin (USL-1) and Single Sided Uniform with Close Approach (USL-2). In the first method, $\Delta k_m = 0.05$ and a lower confidence band (usually 95%) is specified based on a linear regression of k_{eff} as a function of some system parameter. In the second method, the arbitrary administrative margin is set to zero and a uniform lower tolerance band is determined based on a linear regression. The second method provides a criticality safety margin that is generally less than 0.05. In cases where there are a limited number of data points, this method may indicate the need for a larger administrative margin. In both cases, all of the significant system parameters need to be studied to determine the strongest correlation.

In the analyses presented in Section 6.5.1.2, the bias and uncertainties are applied directly to the estimate of the system k_{eff}. Noting that the NRC requires a 5% subcriticality margin ($\Delta k_m = 0.05$), Equation 3 can be rewritten applying the bias and uncertainty in the bias to the k_{eff} of the system being analyzed as:

$$k_s + \Delta k_s + \beta + \Delta\beta \leq 0.95 \qquad (6)$$

In Equation 6, the method bias and all uncertainties are added to k_s. This is the maximum k_{eff} criterion defined in Section 6.5.1.2.

To this point, both the USL criterion and maximum k_{eff} criterion are equivalent. The effects of trending in the bias or the uncertainty in the bias can be directly incorporated into either Equation 5 or Equation 6. Trending is established by performing a regression analysis of k_{eff} as a function of the principle system variables such as: enrichment, rod pitch, H to U ratio, average group of fission, ^{10}B absorber loading and flux trap gap spacing. Usually, simple linear regression is performed, and the line with the greatest correlation is used to functionalize β. This approach is recommended in NUREG/CR-6361. However, if no strong correlation can be determined, then a constant bias adjustment can be made. This is typically done with a one-side tolerance factor that guarantees 95% confidence in the uncertainty in the bias. This is the approach taken in the UMS® criticality analysis.

Both NUREG/CR-6361 and the NAC evaluation perform regression analysis on key system parameters. For all of the major system parameters, the evaluation found no strong correlation. This is based on the observation that the correlation coefficients are all much less than ± 1. Thus a constant bias with a 95/95 confidence factor is applied to the system k_{eff}. NAC's statistical analysis of the k_{eff} results produced a bias of 0.0052 and a 95/95 uncertainty of 0.0087. Adding the two together and subtracting from 0.95 yields an effective constant USL of 0.9361.

To assure compliance with NUREG/CR-6361, an upper safety limit is generated using USLSTATS and is compared to the constant NAC bias and bias uncertainty used in Section 6.5.1.2.

To evaluate the relative importance of the trend analysis to the upper subcritical limits, correlation coefficients are required for all independent parameters. Table 6.5.1-2 contains the correlation coefficient, R, for each linear fit of k_{eff} versus experimental parameter (data is extracted from Figures 6.5.1-2 through 6.5.1-7 by taking the square root of the R^2 value). Based on the highest correlation coefficient and the method presented in NUREG/CR-6361, a USL is established based on the variation of k_{eff} with enrichment. Note that even the enrichment function shows a low statistical correlation coefficient (an |R| equal or near 1 would indicate a good fit). The output generated by USLSTATS is shown in Figure 6.5.1-8.

The NAC applied USL of 0.9361 bounds the calculated upper subcritical limits for all enrichment values above 3.0 wt % ^{235}U. Since the maximum reactivities in the UMS® are calculated at enrichments well above this level, the existing bias bounds the NUREG calculated USL. The parameters of the most reactive cask configuration are presented in Table 6.5.1-3. The most reactive UMS® configuration is the PWR basket configuration with Westinghouse 17×17 OFA fuel assemblies.

Figure 6.5.1-1 KENO-Va Validation—27-Group Library Results: Frequency Distribution of k_{eff} Values

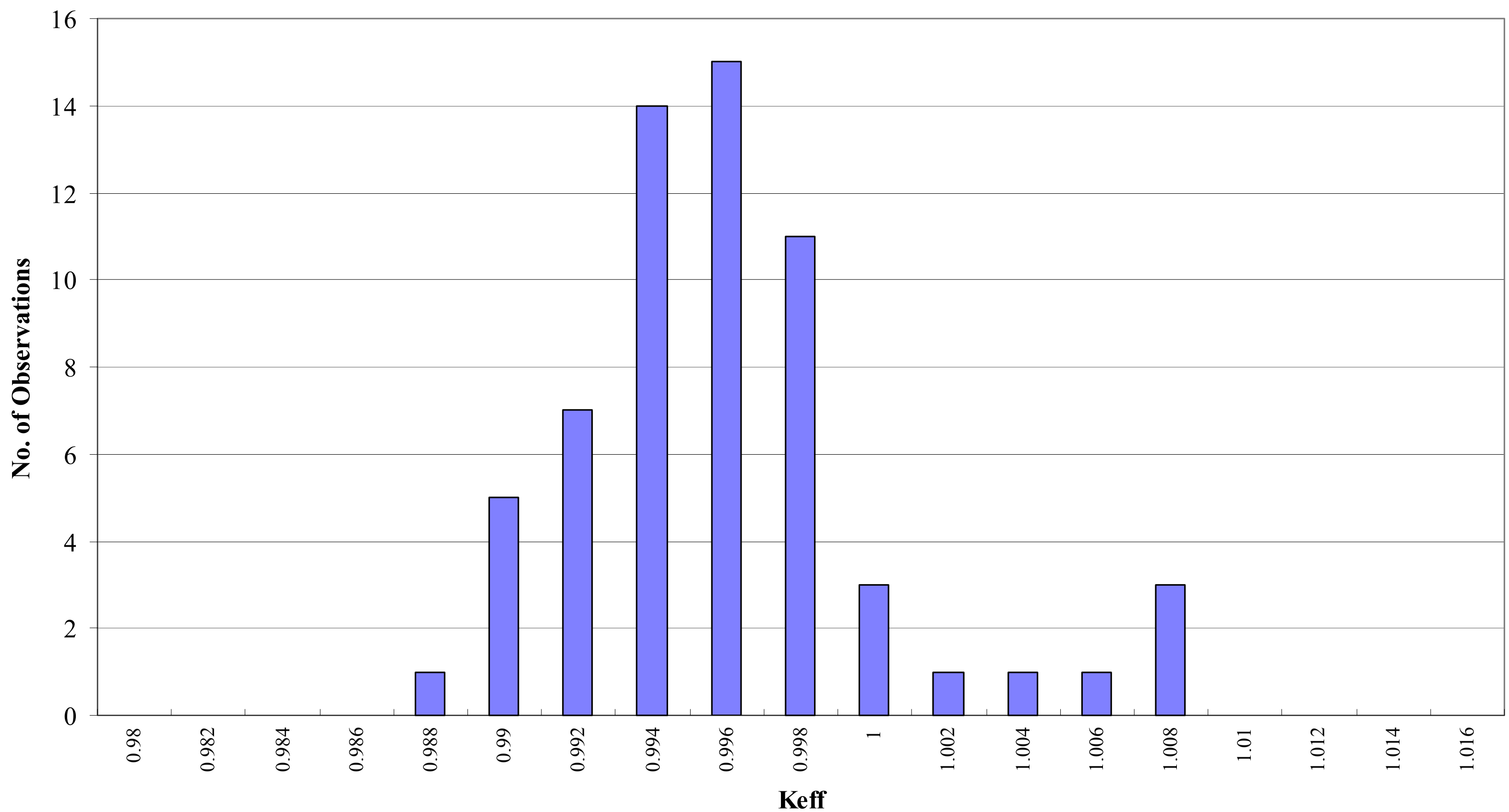


Figure 6.5.1-2 KENO-Va Validation—27-Group Library Results: k_{eff} versus Enrichment

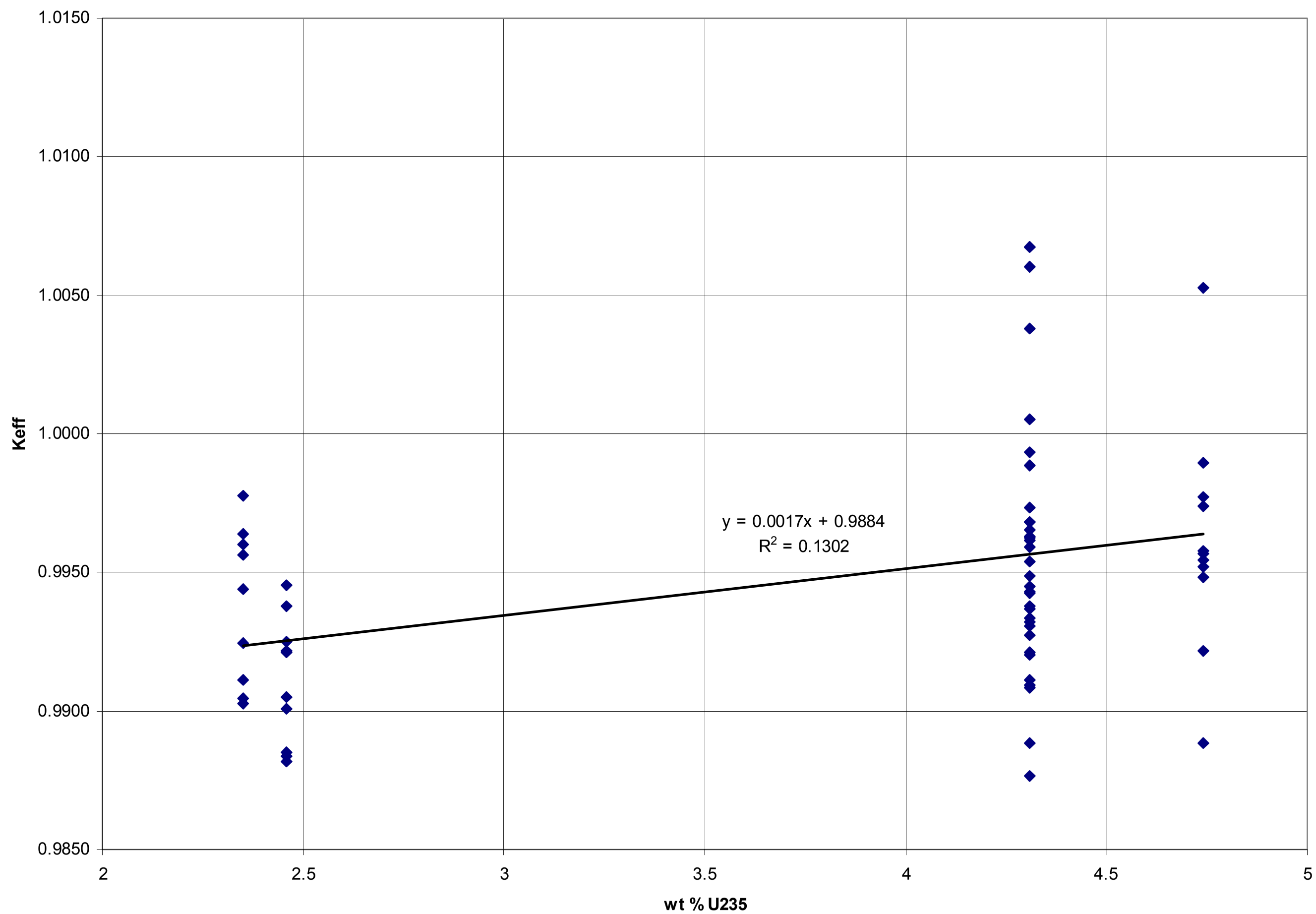


Figure 6.5.1-3 KENO-Va Validation—27-Group Library Results: k_{eff} versus Rod Pitch

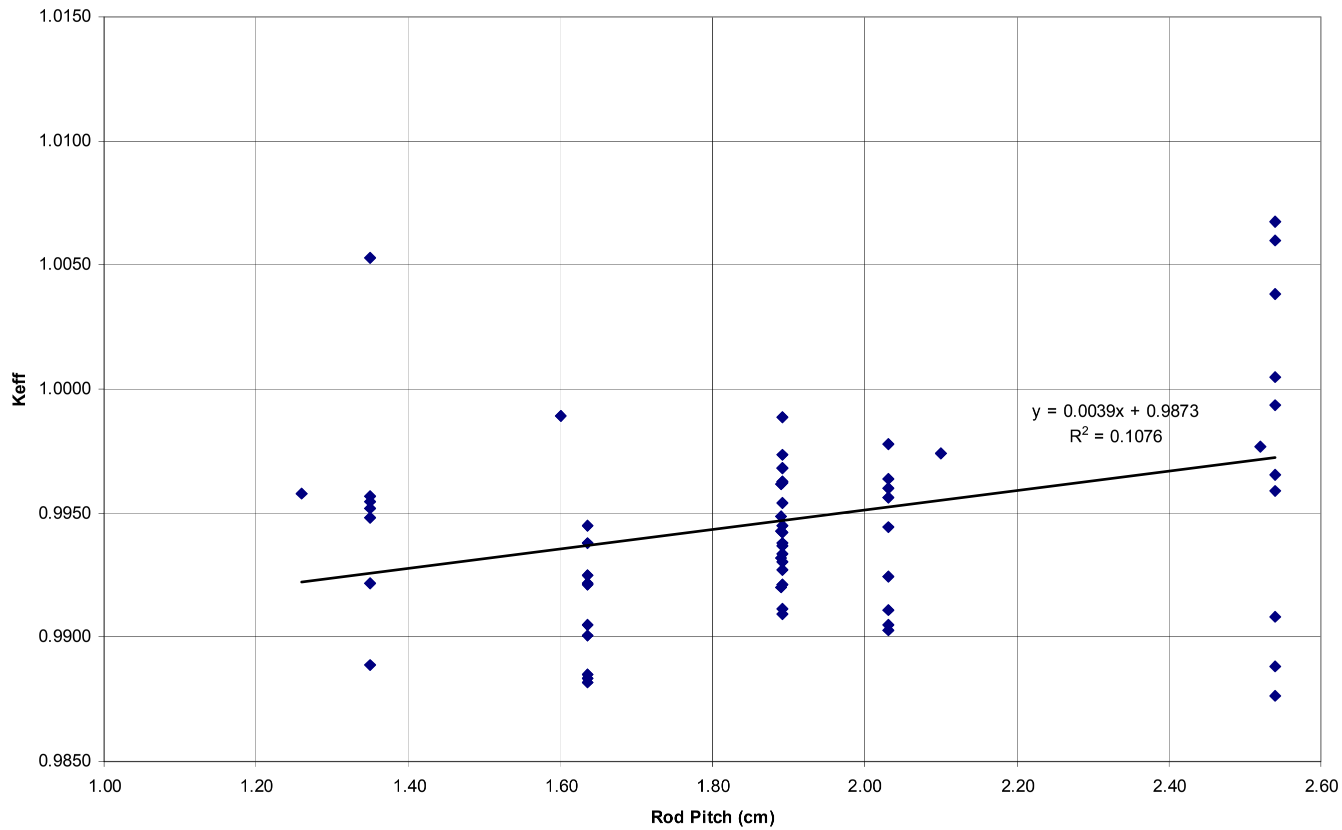


Figure 6.5.1-4 KENO-Va Validation—27-Group Library Results: k_{eff} versus H/U Volume Ratio

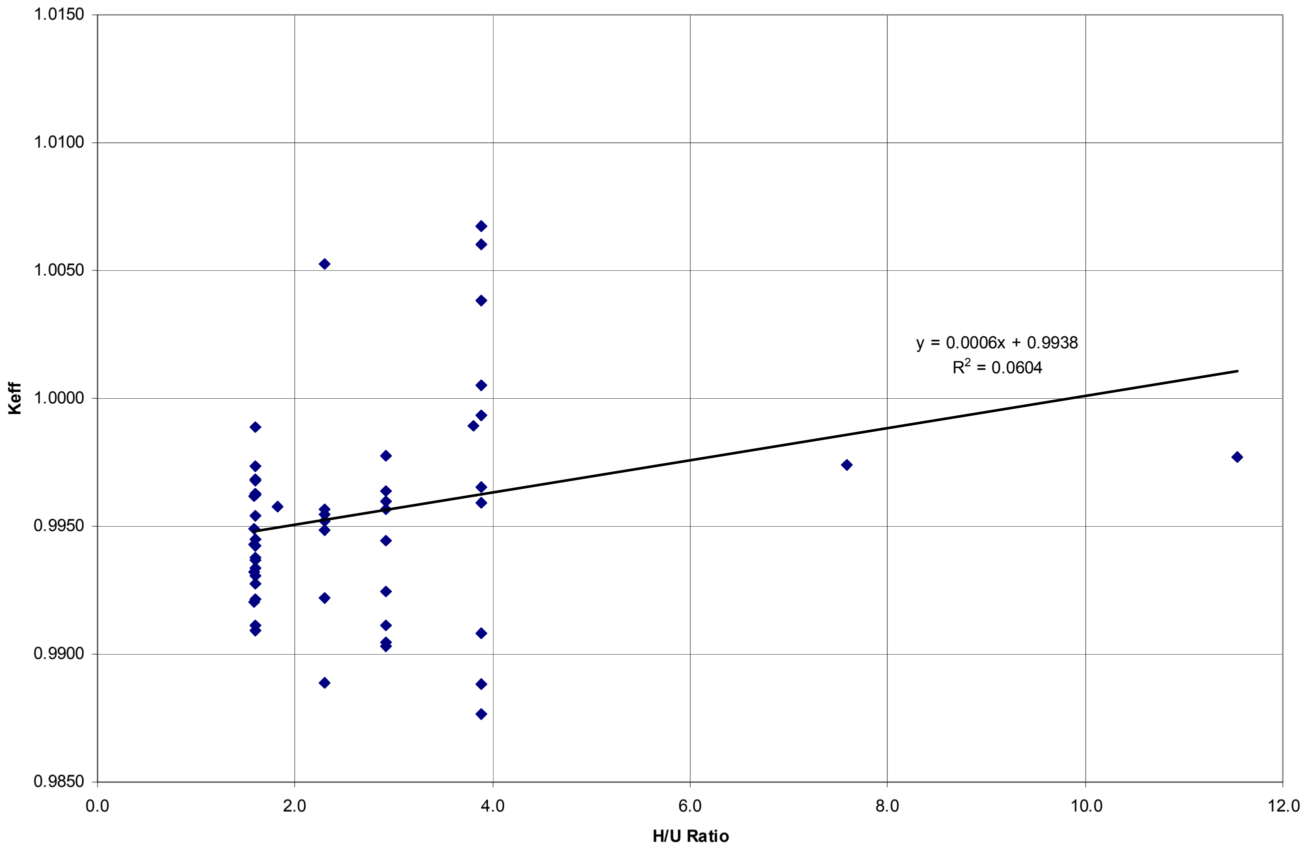


Figure 6.5.1-5 KENO-Va Validation—27-Group Library Results: k_{eff} versus Average Group of Fission

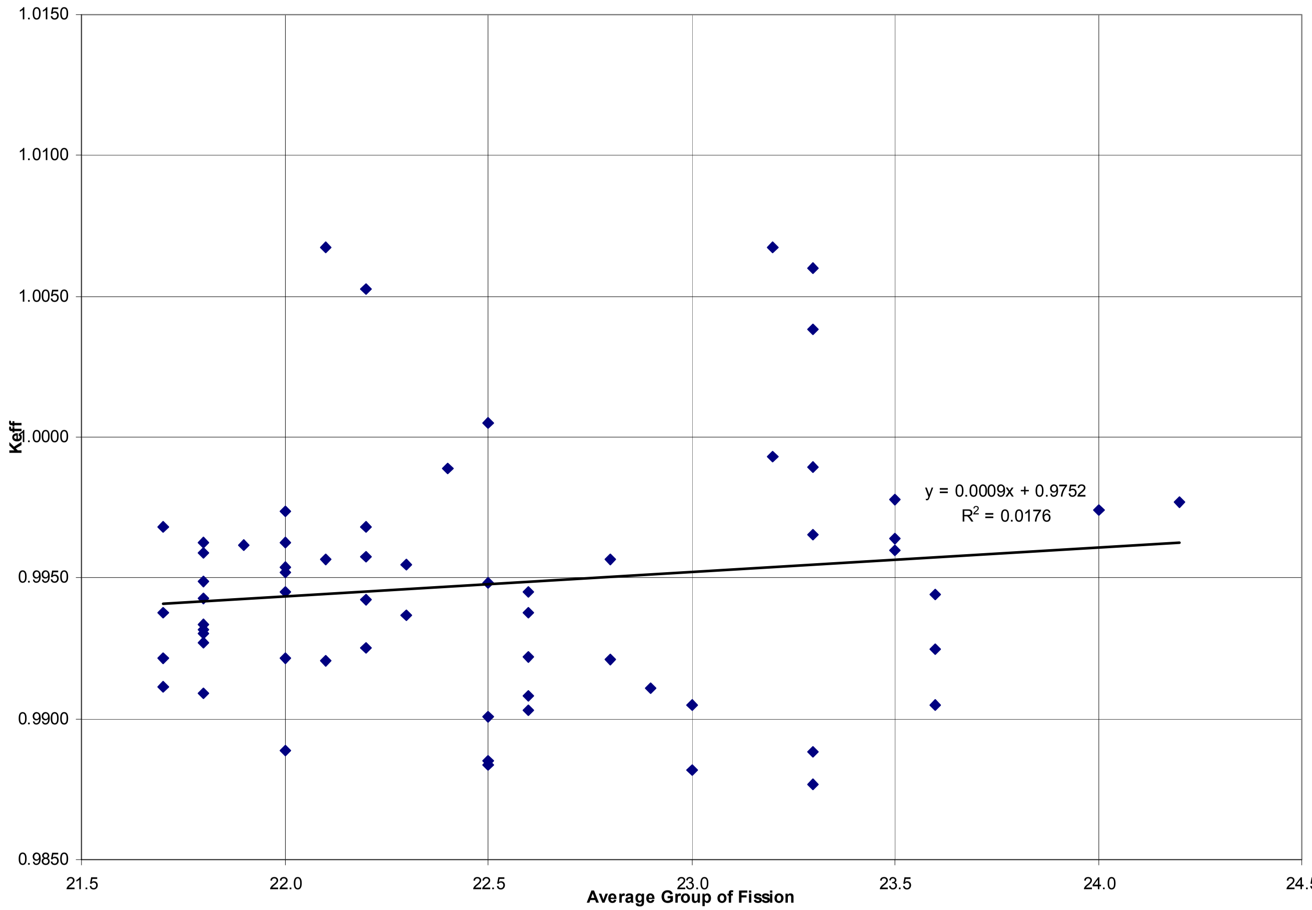


Figure 6.5.1-6 KENO-Va Validation—27-Group Library Results: k_{eff} versus ^{10}B Loading for Flux Trap Criticals

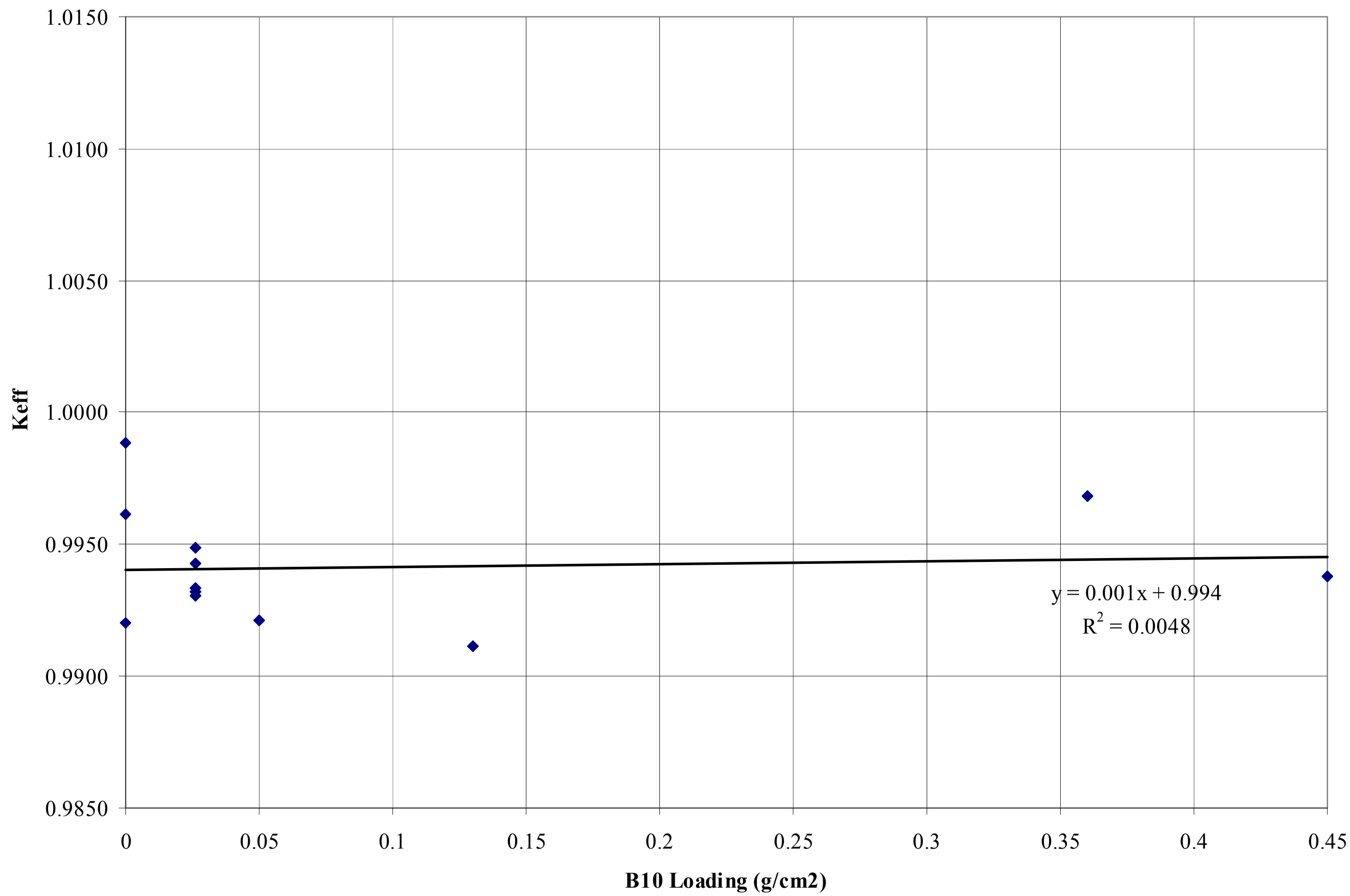


Figure 6.5.1-7 KENO-Va Validation—27-Group Library Results: k_{eff} versus Flux Trap Critical Gap Thickness

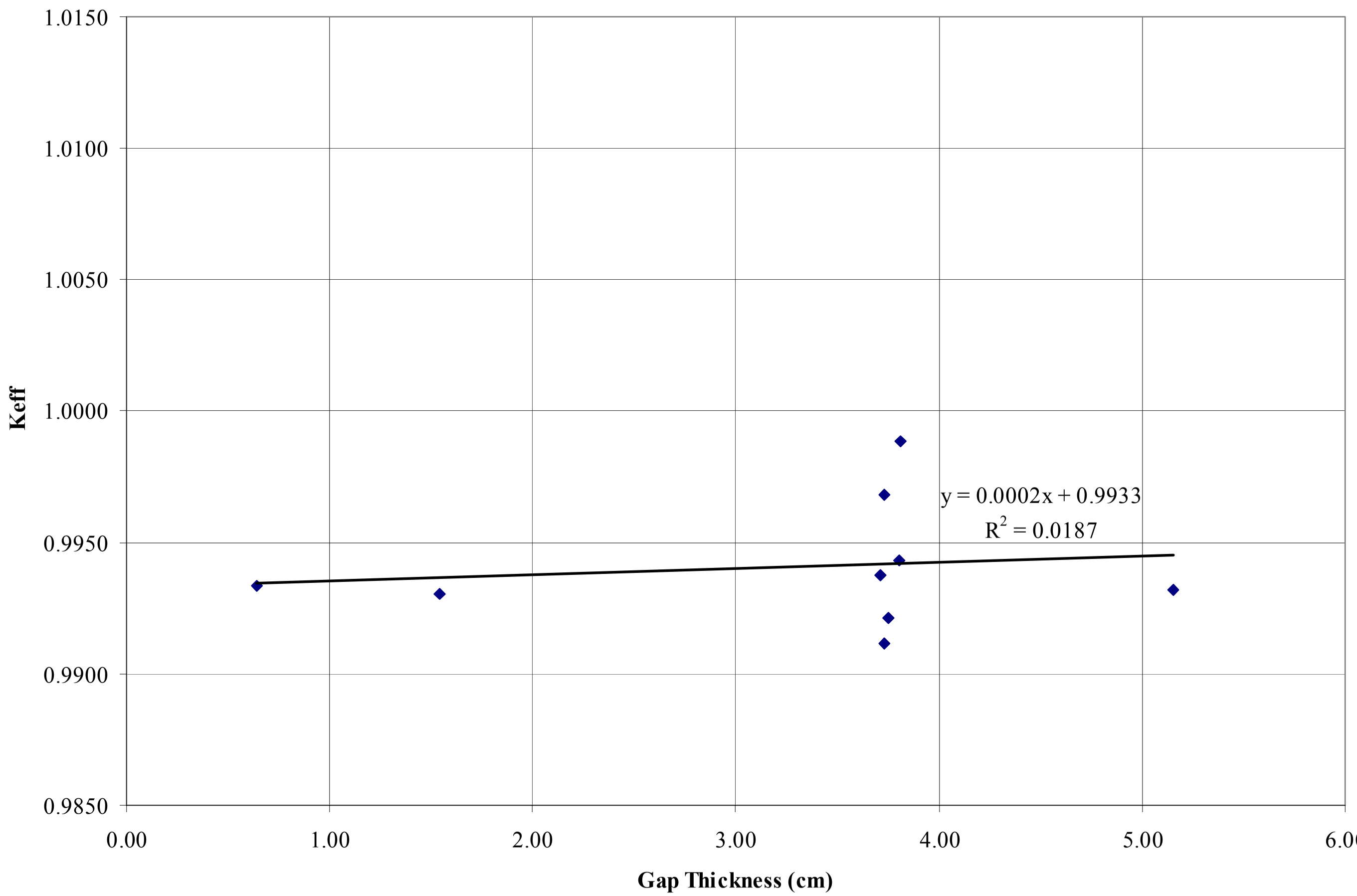


Figure 6.5.1-8 USLSTATS Output for Fuel Enrichment Study

```
uslstats: a utility to calculate upper subcritical
limits for criticality safety applications

**************************************************************************
Version 1.3.4, February 12, 1998
Oak Ridge National Laboratory
**************************************************************************

Input to statistical treatment from file:EN_KEFF.TXT

Title: 63 LWR CRITICAL EXPERIMENT KEFF VS ENRICHMENT

Proportion of the population = .995
Confidence of fit            = .950
Confidence on proportion     = .950
Number of observations       =   63
Minimum value of closed band =    0.00
Maximum value of closed band =    0.00
Administrative margin        =    0.05
```

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
2.35000E+00	9.96400E-01	1.00000E-03	4.31000E+00	9.96500E-01	1.10000E-03
2.35000E+00	9.94400E-01	1.00000E-03	4.31000E+00	1.00680E+00	2.10000E-03
2.35000E+00	9.90500E-01	1.00000E-03	4.31000E+00	1.00380E+00	1.20000E-03
2.35000E+00	9.96000E-01	1.10000E-03	4.31000E+00	9.88900E-01	1.10000E-03
2.35000E+00	9.97800E-01	1.00000E-03	4.31000E+00	9.95900E-01	1.10000E-03
2.35000E+00	9.92500E-01	1.00000E-03	4.31000E+00	1.00670E+00	1.00000E-03
2.35000E+00	9.90300E-01	9.00000E-04	4.31000E+00	1.00050E+00	1.10000E-03
2.35000E+00	9.95700E-01	1.00000E-03	4.31000E+00	9.90800E-01	1.10000E-03
2.35000E+00	9.91100E-01	1.00000E-03	4.31000E+00	9.98900E-01	1.20000E-03
2.46000E+00	9.92100E-01	1.10000E-03	4.31000E+00	9.92100E-01	1.20000E-03
2.46000E+00	9.92500E-01	9.00000E-04	4.31000E+00	9.91100E-01	1.20000E-03
2.46000E+00	9.93800E-01	9.00000E-04	4.31000E+00	9.96800E-01	1.30000E-03
2.46000E+00	9.90500E-01	1.00000E-03	4.31000E+00	9.93800E-01	1.20000E-03
2.46000E+00	9.88200E-01	1.00000E-03	4.31000E+00	9.93400E-01	1.00000E-03
2.46000E+00	9.94500E-01	1.00000E-03	4.31000E+00	9.93100E-01	1.00000E-03
2.46000E+00	9.92200E-01	1.00000E-03	4.31000E+00	9.94300E-01	1.00000E-03
2.46000E+00	9.88500E-01	1.00000E-03	4.31000E+00	9.93200E-01	1.00000E-03
2.46000E+00	9.88400E-01	1.00000E-03	4.31000E+00	9.94900E-01	1.00000E-03
2.46000E+00	9.90100E-01	9.00000E-04	4.31000E+00	9.92000E-01	1.00000E-03
4.31000E+00	9.95400E-01	1.40000E-03	4.31000E+00	9.96200E-01	1.00000E-03
4.31000E+00	9.94500E-01	1.30000E-03	4.74000E+00	9.92200E-01	1.30000E-03
4.31000E+00	9.97400E-01	1.30000E-03	4.74000E+00	9.88900E-01	1.30000E-03
4.31000E+00	9.96300E-01	1.30000E-03	4.74000E+00	9.95700E-01	1.30000E-03
4.31000E+00	9.92700E-01	1.20000E-03	4.74000E+00	1.00530E+00	1.10000E-03
4.31000E+00	9.90900E-01	1.20000E-03	4.74000E+00	9.95500E-01	1.20000E-03
4.31000E+00	9.96200E-01	1.20000E-03	4.74000E+00	9.94800E-01	1.30000E-03
4.31000E+00	9.93700E-01	1.30000E-03	4.74000E+00	9.95800E-01	1.20000E-03
4.31000E+00	9.94200E-01	1.20000E-03	4.74000E+00	9.95200E-01	1.20000E-03
4.31000E+00	9.96800E-01	1.20000E-03	4.74000E+00	9.98900E-01	1.30000E-03
4.31000E+00	9.87700E-01	2.30000E-03	4.74000E+00	9.97400E-01	1.20000E-03
4.31000E+00	9.99300E-01	1.20000E-03	4.74000E+00	9.97700E-01	1.10000E-03
4.31000E+00	1.00600E+00	2.20000E-03			

```
chi =  2.1587 (upper bound = 9.49). The data tests normal.


Output from statistical treatment

63 LWR CRITICAL EXPERIMENT KEFF VS ENRICHMENT

Number of data points (n)                         63
Linear regression, k(X)                         0.9884 + ( 1.6748E-03)*X
Confidence on fit (1-gamma) [input]               95.0%
Confidence on proportion (alpha) [input]          95.0%
Proportion of population falling above
lower tolerance interval (rho) [input]            99.5%
Minimum value of X                                   2.3500
Maximum value of X                                   4.7400
Average value of X                                   3.81143
```

Figure 6.5.1-8 USLSTATS Output for Fuel Enrichment Study (Continued)

```
Average value of k                                  0.99482
Minimum value of k                                  0.98770
Variance of fit, s(k,X)^2                          1.6973E-05
Within variance, s(w)^2                            1.4306E-06
Pooled variance, s(p)^2                            1.8404E-05
Pooled std. deviation, s(p)                        4.2900E-03
C(alpha,rho)*s(p)                                  1.5488E-02
student-t @ (n-2,1-gamma)                          1.67078E+00
Confidence band width, W                           7.3606E-03
Minimum margin of subcriticality, C*s(p)-W         8.1273E-03

Upper subcritical limits: (  2.35000 <= X <=   4.74000)
***** *********** *******

USL Method 1 (Confidence Band with
Administrative Margin)            USL1 = 0.9311 + ( 1.6748E-03)*X

USL Method 2 (Single-Sided Uniform
Width Closed Interval Approach)    USL2 = 0.9729 + ( 1.6748E-03)*X



USLs Evaluated Over Range of Parameter X:
**** ********* **** ***** ** ********* **

X:   2.35   2.69   3.03   3.37   3.72   4.06   4.40   4.74
---------------------------------------------------------
USL-1:  0.9350 0.9356 0.9362 0.9367 0.9373 0.9379 0.9384 0.9390
USL-2:  0.9769 0.9775 0.9780 0.9786 0.9792 0.9797 0.9803 0.9809
---------------------------------------------------------

****************************************************************************
Thus spake USLSTATS
Finis.
```

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics

Criticals	Configura-tion	wt % ^{235}U	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
Set 1										**Gap**				
B&W-I	Cylindrical	2.46	1.636	1.03	1.206	1.6	0	na	na	0	na	22.8	0.9921	0.0011
B&W-II	3×3-14×14	2.46	1.636	1.03	1.206	1.6	1037	na	na	0	na	22.2	0.9925	0.0009
B&W-III	3×3-14×14	2.46	1.636	1.03	1.206	1.6	764	na	na	1.636	0.9982	22.6	0.9938	0.0009
B&W-IX	3×3-14×14	2.46	1.636	1.03	1.206	1.6	0	na	na	6.543	0.9982	23	0.9905	0.0010
B&W-X	3×3-14×14	2.46	1.636	1.03	1.206	1.6	143	na	na	4.907	0.9982	23	0.9882	0.0010
B&W-XI	3×3-14×14	2.46	1.636	1.03	1.206	1.6	514	Steel	0	1.636	0.9982	22.6	0.9945	0.0010
B&W-XIII	3×3-14×14	2.46	1.636	1.03	1.206	1.6	15	B-Al	0.0052	1.636	0.9982	22.6	0.9922	0.0010
B&W-XIV	3×3-14×14	2.46	1.636	1.03	1.206	1.6	92	B-Al	0.0040	1.636	0.9982	22.5	0.9885	0.0010
B&W-XVII	3×3-14×14	2.46	1.636	1.03	1.206	1.6	487	B-Al	0.0008	1.636	0.9982	22.5	0.9884	0.0010
B&W-XIX	3×3-14×14	2.46	1.636	1.03	1.206	1.6	634	B-Al	0.0003	1.636	0.9982	22.5	0.9901	0.0009
												Average	0.9911	0.0023

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (Continued)

Criticals Set 2	Configuration	wt % ^{235}U	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
PNL-043	17×13 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9954	0.0014
PNL-044	16×14 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9945	0.0013
PNL-045	14×16 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9974	0.0013
PNL-046	12×19 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9963	0.0013
PNL-087	4 11×14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.066	2.83	0.9982	21.8	0.9927	0.0012
PNL-079	4 11×14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.030	2.83	0.9982	21.8	0.9909	0.0012
PNL-093	4 11×14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.026	2.83	0.9982	21.8	0.9962	0.0012
PNL-115	4 9×12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Aluminum	0	2.83	0.9982	22.3	0.9937	0.0013
PNL-064	4 9×12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Steel (.302)	0	2.83	0.9982	22.2	0.9942	0.0012
PNL-071	4 9×12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Steel (.485)	0	2.83	0.9982	22.2	0.9968	0.0012
												Average	0.9948	0.0020

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (Continued)

Criticals Set 3	Configura-tion	wt % ^{235}U	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap Cluster (cm)	Gap Wall/ Cluster (cm)	Ave. Group Fission	k_{eff}	σ
PNL-STA	3×1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	10.65	0.00	23.5	0.9964	0.0010
PNL-STB	3×1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.20	1.32	23.6	0.9944	0.0010
PNL-STC	3×1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	10.36	2.62	23.6	0.9905	0.0010
PNL-PBA	3×1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.84	0.00	23.5	0.9960	0.0011
PNL-PBB	3×1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.72	0.66	23.5	0.9978	0.0010
PNL_PBC	3×1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.25	2.62	23.6	0.9925	0.0010
PNL-DUA	3×1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.83	0.00	22.6	0.9903	0.0009
PNL-DUB	3×1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	14.11	1.96	22.8	0.9957	0.0010
PNL-DUC	3×1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.70	2.62	22.9	0.9911	0.0010

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (Continued)

Criticals	Configura-tion	wt % ^{235}U	itch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap (cm)	Ave. Group Fission	k_{eff}	σ
Set 3 (Contd.)										**Cluster**	**Wall/ Cluster**			
PNL-H20	3×1 H2O Refl	4.31	2.54	1.265	1.415	3.9	0	na	na	8.24	inf	23.3	0.9877	0.0023
PNL-ST0	3×1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	12.89	0	23.2	0.9993	0.0012
PNL-ST1	3×1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	14.12	1.32	23.3	1.0060	0.0022
PNL-ST26	3×1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	12.44	2.62	23.3	0.9965	0.0011
PNL-PB0	3×1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	20.62	0	23.2	1.0068	0.0021
PNL-PB13	3×1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	19.04	1.32	23.3	1.0038	0.0012
PNL-PB5	3×1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	10.3	5.41	23.3	0.9889	0.0011
PNL-DU0	3×1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	15.38	0	21.8	0.9959	0.0011
PNL-DU13	3×1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	19.04	1.32	22.1	1.0067	0.0010
PNL-DU39	3×1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	18.05	3.91	22.5	1.0005	0.0011
PNL-DU54	3×1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	13.49	5.41	22.6	0.9908	0.0011
												Average	0.9964	0.0060

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (Continued)

Criticals	Configura-tion	wt % U^{235}	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
Set 4														
PNL-229	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Aluminum	0	3.81	0.9982	22.4	0.9989	0.0012
PNL-230	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.05	3.75	0.9982	21.7	0.9921	0.0012
PNL-228	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.13	3.73	0.9982	21.7	0.9911	0.0012
PNL-214	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.36	3.73	0.9982	21.7	0.9968	0.0013
PNL-231	2×2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.45	3.71	0.9982	21.7	0.9938	0.0012
PNL-127	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.026	0.64	0.9982	21.8	0.9934	0.0010
PNL-126	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.026	1.54	0.9982	21.8	0.9931	0.0010
PNL-123	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.026	3.80	0.9982	21.8	0.9943	0.0010
PNL-125	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.026	5.16	0.9982	21.8	0.9932	0.0010
PNL-124	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Neutron Absorber	0.026	INF	0.9982	21.8	0.9949	0.0010
PNL-123-S	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Steel	0	3.80	0.9982	22.1	0.9920	0.0010
PNL-124-S	2×1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Steel	0	INF	0.9982	21.9	0.9962	0.0010
												Average	0.9941	0.0022

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (Continued)

Criticals	Configuration	wt % U^{235}	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
Set 5														
VCML	2×2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0	22.0	0.9922	0.0013
VCML	2×2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.0323	22.0	0.9889	0.0013
VCML	2×2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.2879	22.1	0.9957	0.0013
VCML	2×2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.5540	22.2	1.0053	0.0011
VCML	2×2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	2.50	0.9982	22.3	0.9955	0.0012
VCML	2×2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	5.00	0.9982	22.5	0.9948	0.0013
VCML	Square Lattice	4.74	1.26	0.79	0.94	1.8	0	na	na	na	na	22.2	0.9958	0.0012
VCML	Square Lattice	4.74	1.35	0.79	0.94	2.3	0	na	na	na	na	22.0	0.9952	0.0012
VCML	Square Lattice	4.74	1.60	0.79	0.94	3.8	0	na	na	na	na	23.3	0.9989	0.0013
VCML	Square Lattice	4.74	2.10	0.79	0.94	7.6	0	na	na	na	na	24.0	0.9974	0.0012
VCML	Square Lattice	4.74	2.52	0.79	0.94	11.5	0	na	na	na	na	24.2	0.9977	0.0011
												Average	0.9961	0.0041

Table 6.5.1-2 SCALE 4.3 Correlation Coefficient for Linear Curve-Fit of Critical Benchmarks

Correlation Studied	Correlation Coefficient (R)
k_{eff} versus enrichment	0.361
k_{eff} versus rod pitch	0.328
k_{eff} versus H/U volume ratio	0.246
k_{eff} versus ^{10}B loading	0.069
k_{eff} versus average group causing fission	0.133
k_{eff} versus flux gap thickness	0.137

Table 6.5.1-3 SCALE 4.3 Range of Correlated Parameters of Most Reactive Configurations

Parameter	Benchmark Minimum Value	Benchmark Maximum Value	UMS® Design Basis PWR Fuel Most Reactive Configuration	Maine Yankee Fuel Most Reactive Configuration
Enrichment (wt. % ^{235}U)	2.35	4.74	4.2	4.2
Rod pitch (cm)	1.26	2.54	1.26	1.50
H/U volume ratio	1.6	11.5	1.9	2.6
^{10}B areal density (g/cm^2)	0.00	0.45	0.025	0.025
Average energy group causing fission	21.7	24.2	22.3	22.5
Flux gap thickness (cm)	0.64	5.16	2.2 to 3.8	2.22 to 3.8

6.5.2 MONK Validation in Accordance with NUREG/CR-6361

NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages" (NUREG), provides a guide to LWR criticality benchmark calculations and the determination of bias and subcritical limits in critical safety evaluations. Section 6.5.1.5 presents the implementation of the NUREG in subcritical limit evaluations for the UMS® storage and transfer casks. This section implements the USLSTATS method of the NUREG for MONK8A application with JEF 2.2 point energy libraries in LWR transport and storage applications.

SERCO Assurance has performed an extensive benchmarking of MONK8A. The cross-section set and key geometry features employed in the critical benchmark models are reflected in the UMS® cask evaluation models. Consequently, the SERCO produced critical benchmark models are applicable to the evaluation of the UMS® system. The critical benchmarks relevant to LWR fuel evaluations were extracted from the total benchmark set and listed in Table 6.5.2-3. The range of the parameters to be benchmarked is summarized in Table 6.5.2-1. Trending in k_{eff} was evaluated for the following independent variables: enrichment, rod pitch, fuel pellet diameter, fuel rod diameter, H/U ratio, average neutron group causing fission, ^{10}B plate loading for flux trap cases, flux trap gap thickness, and soluble boron concentration in the moderator. The data is plotted in Figures 6.5.2-1 through 6.5.2-9.

To evaluate the relative importance of the trend analysis to the upper safety limits, correlation coefficients are required for all independent parameters. Table 6.5.2-2 contains the correlation coefficient, R, for each linear fit of k_{eff} versus experimental parameter (data is extracted from Figure 6.5.2-1 through Figure 6.5.2-9 by taking the square root of the R^2 value). The k_{eff} versus soluble boron concentration in the moderator displays the most statistically significant correlation to system reactivity. The k_{eff} versus cluster (assembly) gap thickness displays the second most statistically significant correlation to system reactivity. Not all NAC criticality safety evaluations take credit for the soluble boron within the spent fuel pool water at PWR reactors. Based on NUREG/CR-6361 guidance, k_{eff} versus soluble boron concentration in the moderator and k_{eff} versus cluster gap thickness are, therefore, chosen to calculate the USL (Upper Safety Limit). Note that even the flux trap function shows a low statistical correlation coefficient (an |R| equal or near 1 would indicate a good fit). The output generated by USLSTATS is shown in Figure 6.5.2-10. If no credit is taken for boron in the moderator and the maximum gap thickness is 3.5 inches, then the appropriate USL to use is 0.9426. However, if credit is taken for a boron concentration in the moderator of more than 298.2 ppm (by mass), then it is acceptable to apply a USL of 0.9441.

The NAC-applied USL is 0.9426, and bounds the calculated upper safety limits for the typical flux trap spacing found in multi-purpose casks and typical soluble boron concentrations within the spent fuel pool water at PWR reactors. The range of the correlated parameters of the most reactive design basis fuel is included in Table 6.5.2-1 to show that the most reactive configuration is within the range of applicability of the validation.

Figure 6.5.2-1 MONK8A – JEF 2.2 Library Validation Statistics – k_{eff} versus Fuel Enrichment

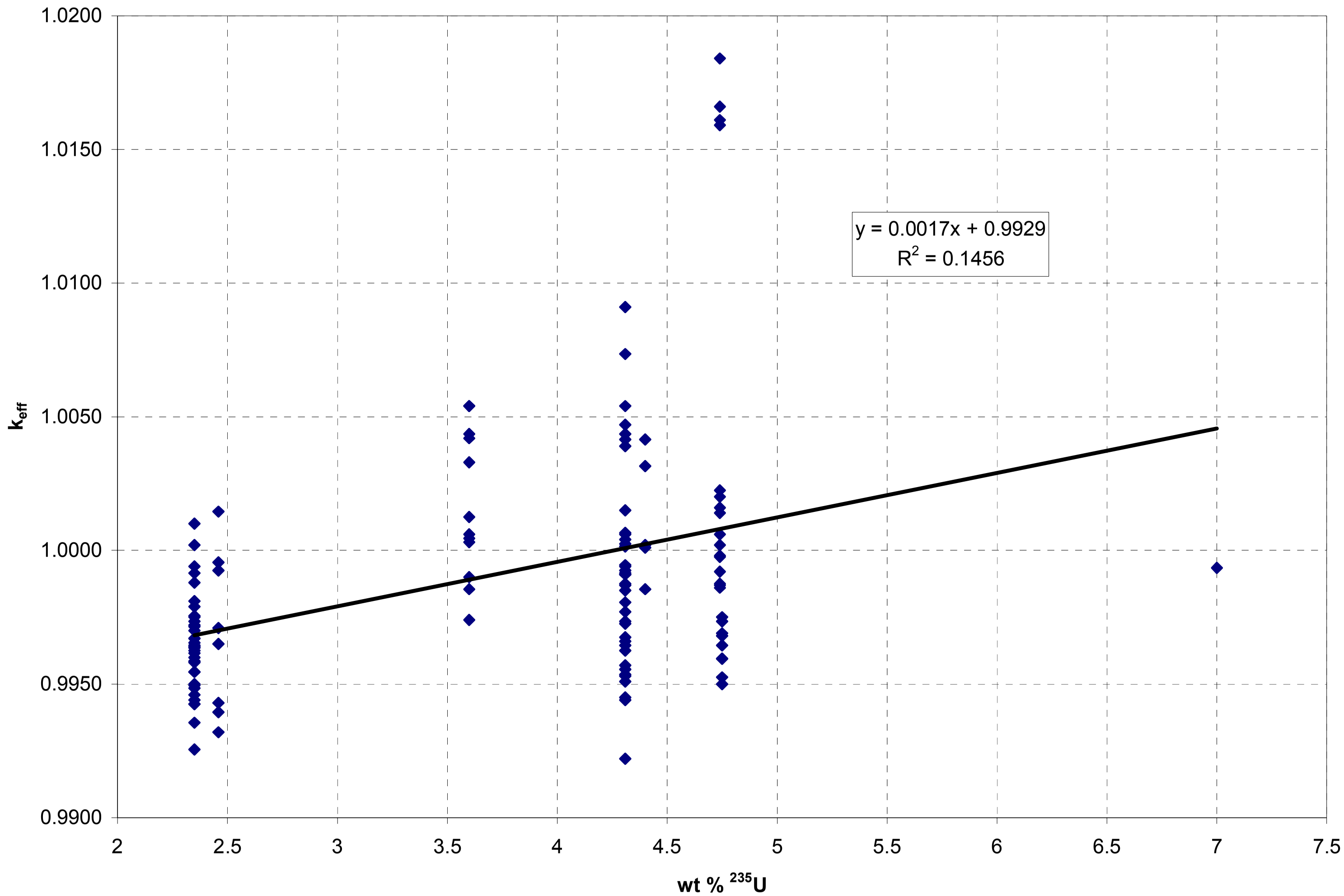


Figure 6.5.2-2 MONK8A – JEF 2.2 Library – k_{eff} versus Rod Pitch

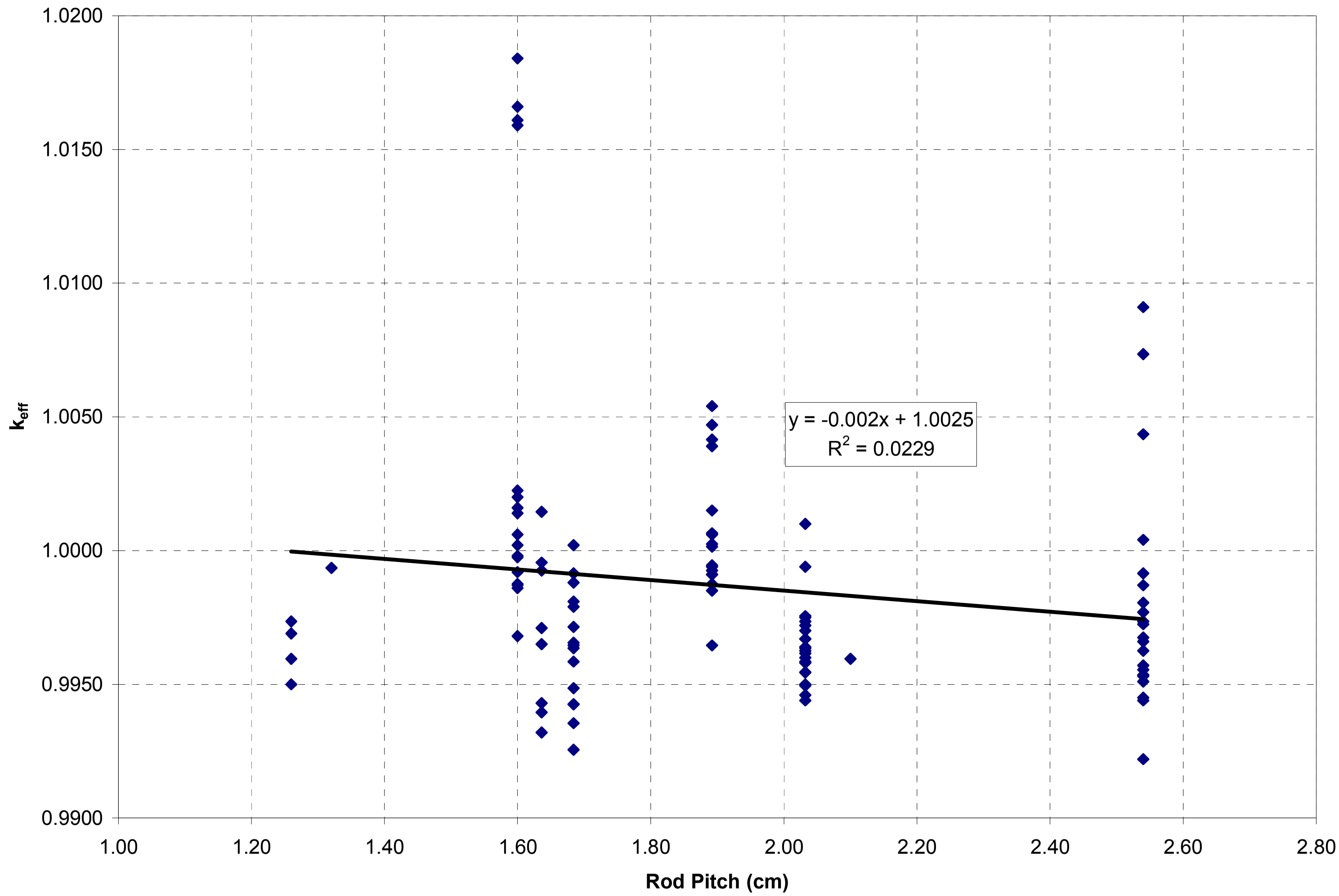


Figure 6.5.2-3 MONK8A – JEF 2.2 Library – k_{eff} versus H/U (fissile) Atom Ratio

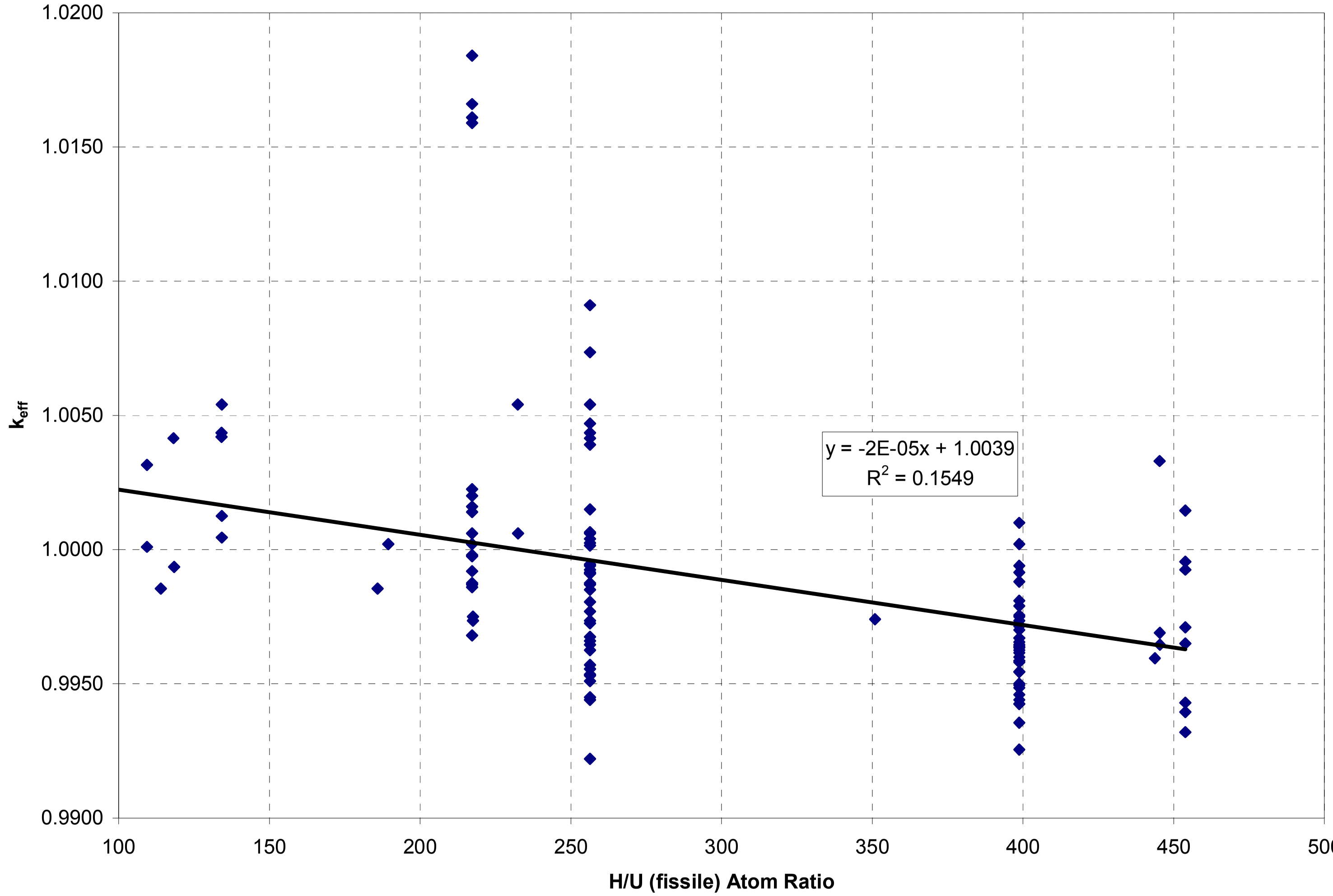


Figure 6.5.2-4 MONK8A – JEF 2.2 Library – k_{eff} versus ^{10}B Plate Loading

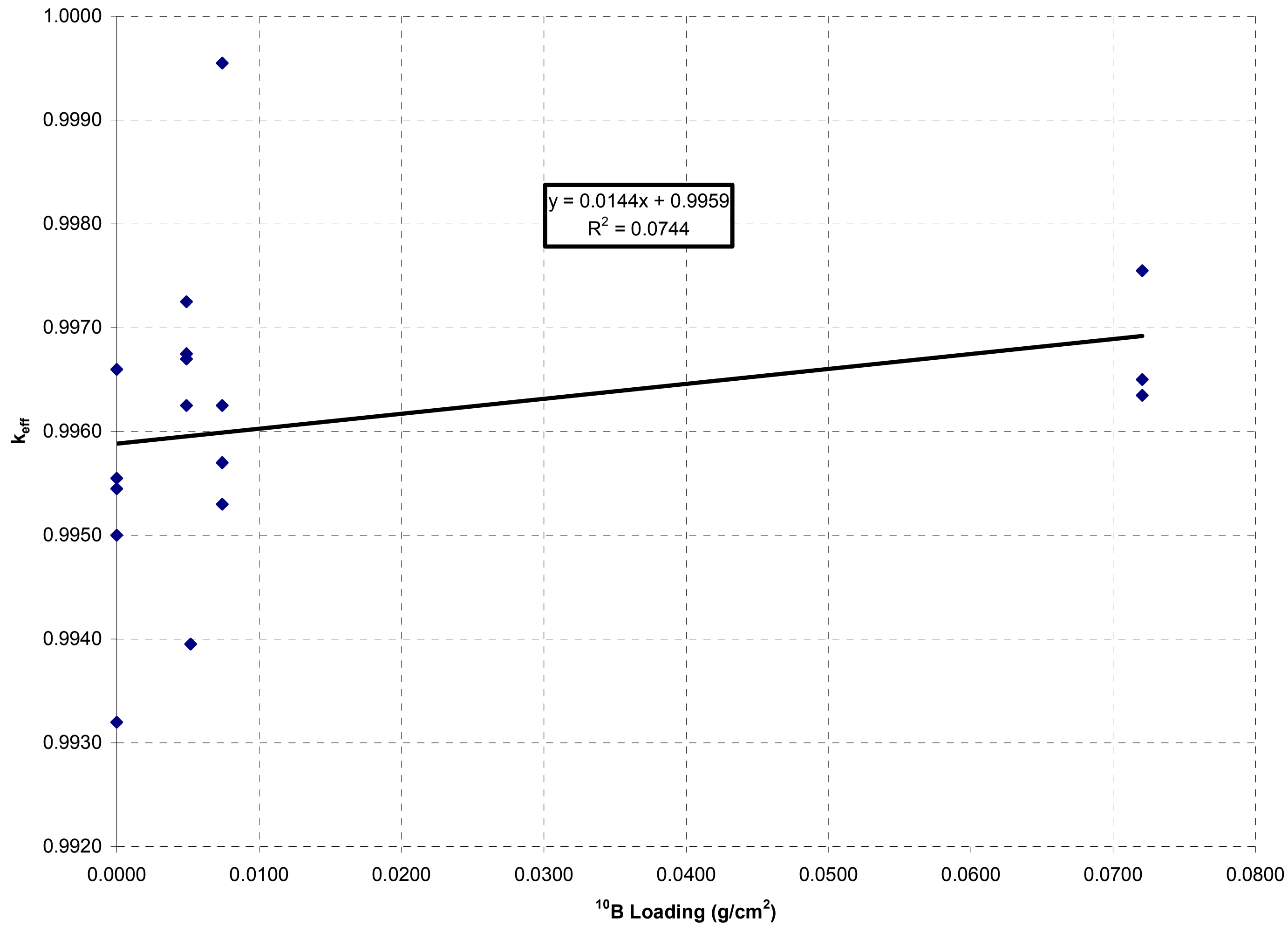


Figure 6.5.2-5 MONK8A – JEF 2.2 Library – k_{eff} versus Mean Neutron Log(E) Causing Fission

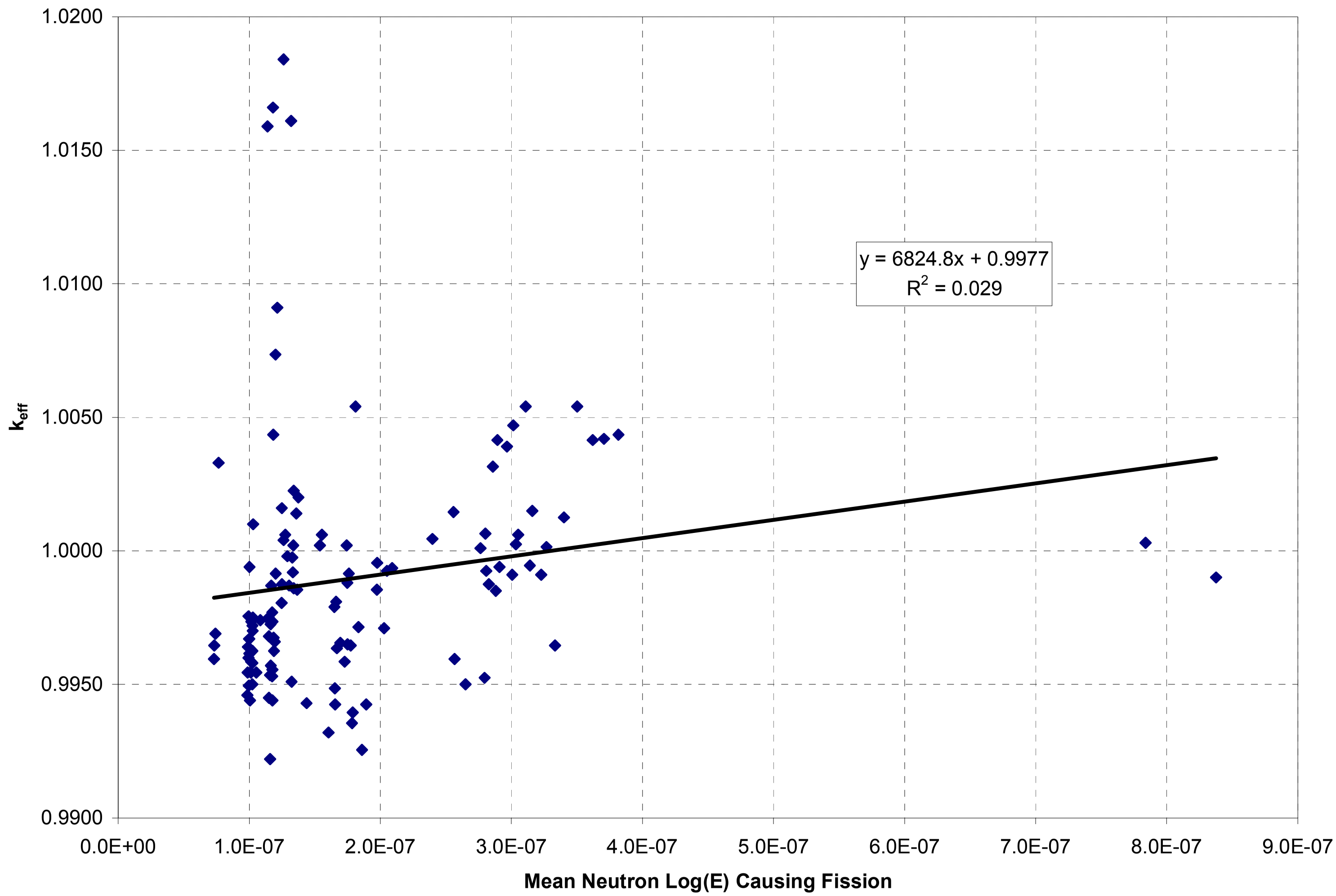


Figure 6.5.2-6 MONK8A – JEF 2.2 Library – k_{eff} versus Cluster Gap Thickness

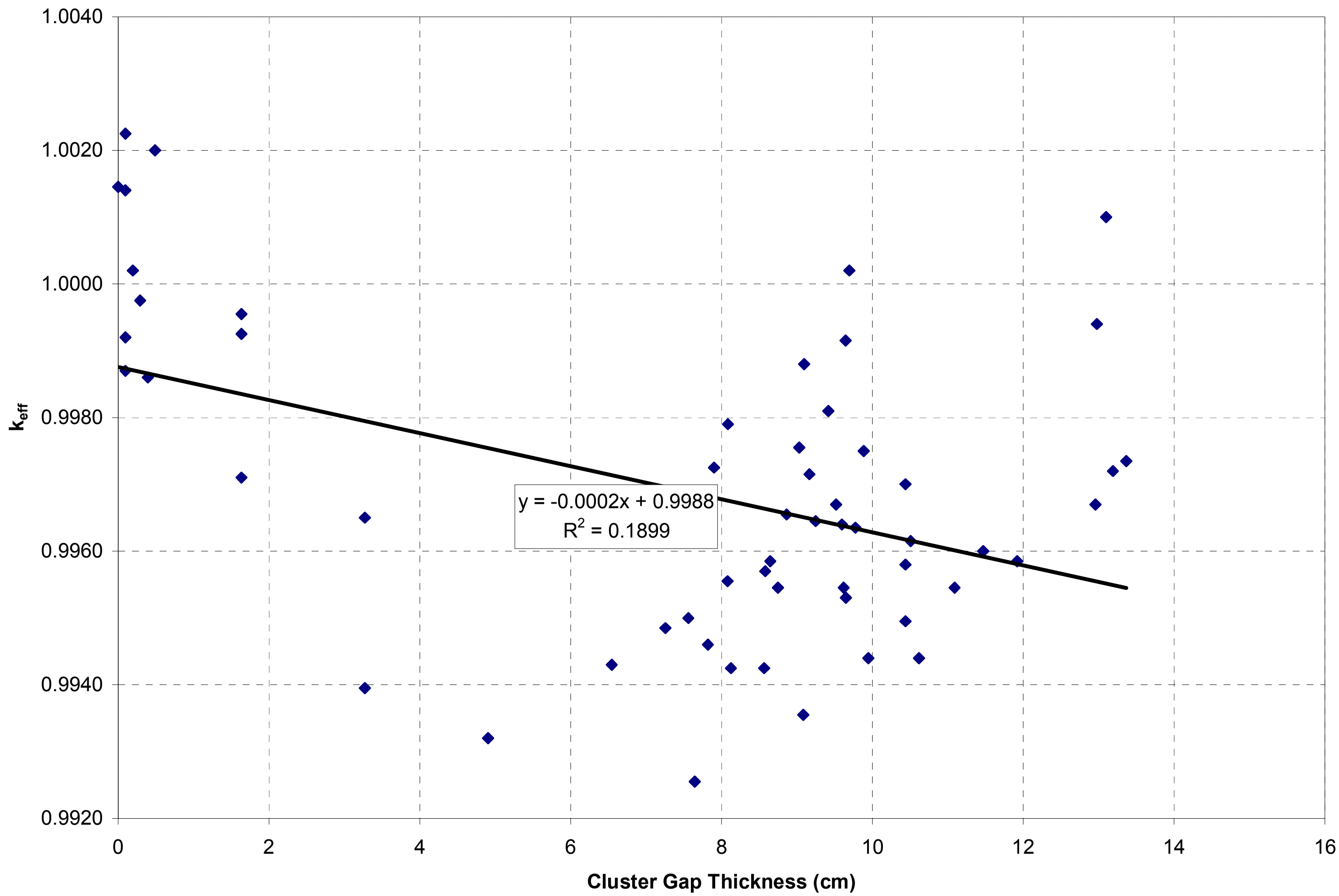


Figure 6.5.2-7 MONK8A – JEF 2.2 Library – k_{eff} versus Fuel Pellet Outside Diameter

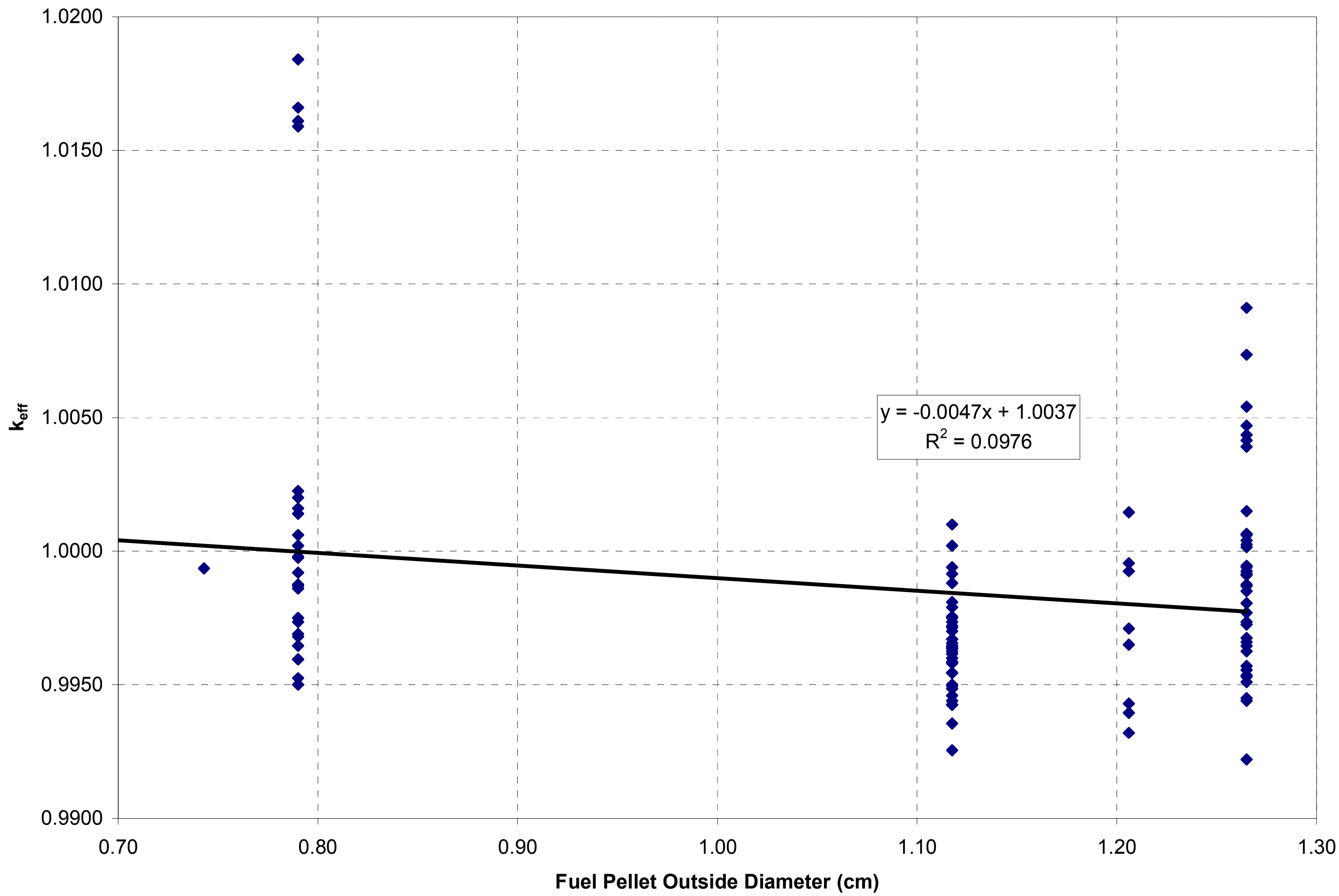


Figure 6.5.2-8 MONK8A – JEF 2.2 Library – k_{eff} versus Fuel Rod Outside Diameter

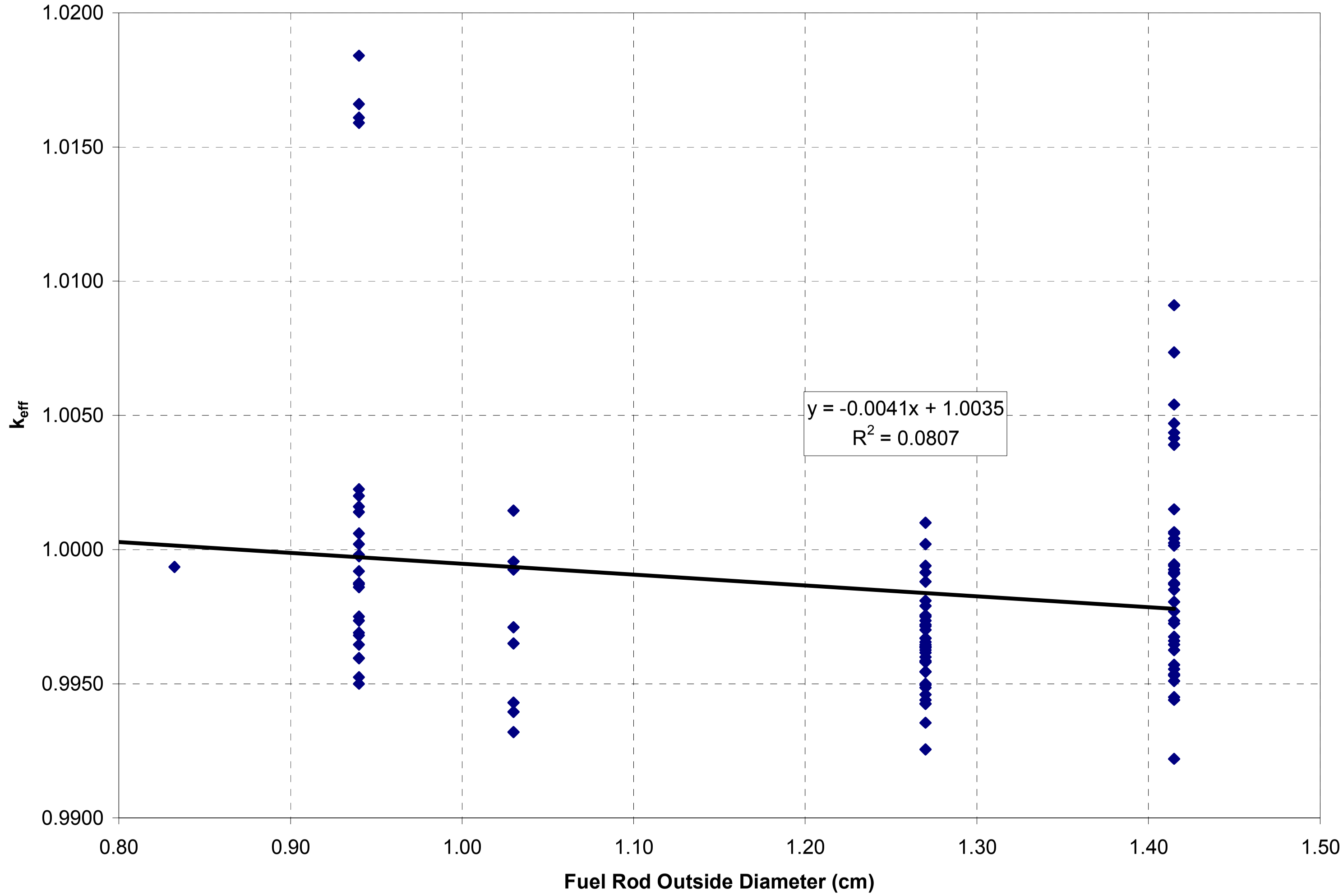


Figure 6.5.2-9 MONK8A – JEF 2.2 Library – k_{eff} versus Soluble Boron PPM in Moderator

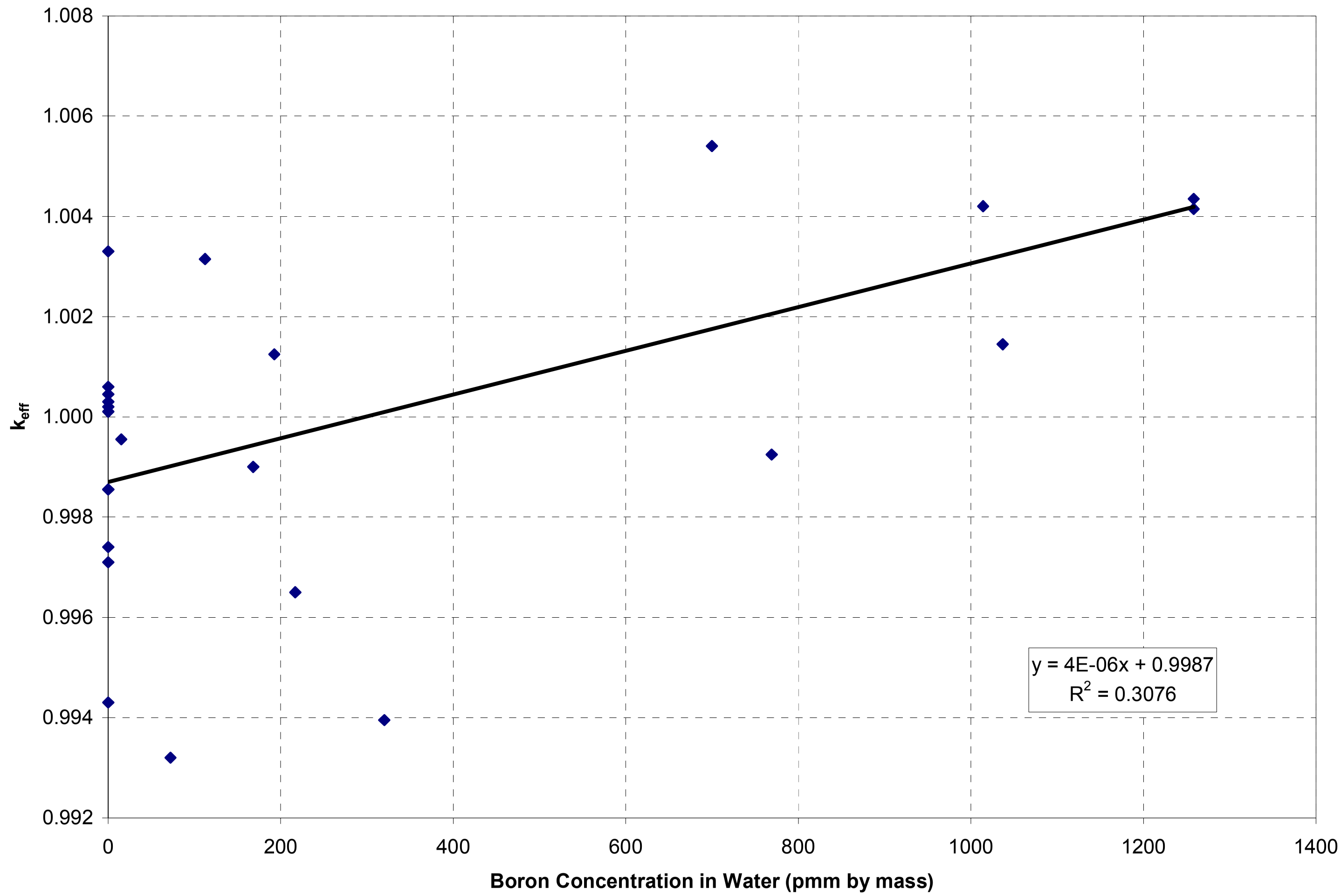


Figure 6.5.2-10 USLSTATS Output – k_{eff} versus Gap Thickness

```
uslstats: a utility to calculate upper subcritical
                        limits for criticality safety applications

  ****************************************************************************
                         Version 1.3.4, February 12, 1998
                        Oak Ridge National Laboratory
  ****************************************************************************

   Input to statistical treatment from file:Gap_keff.txt

Title: 62 Critical Experiment KEFFs VS Gap Thickness - Experiments 1, 3, 7, 17, & 40

   Proportion of the population = .995
   Confidence of fit            = .950
   Confidence on proportion     = .950
   Number of observations       =   62
   Minimum value of closed band =    0.00
   Maximum value of closed band =    0.00
   Administrative margin        =    0.05
```

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
6.33000E+00	9.96350E-01	1.00000E-03	1.29600E+01	9.96700E-01	1.00000E-03
9.03000E+00	9.97550E-01	1.00000E-03	9.95000E+00	9.94400E-01	1.00000E-03
1.04400E+01	9.94950E-01	1.00000E-03	7.82000E+00	9.94600E-01	1.00000E-03
1.14700E+01	9.96000E-01	1.00000E-03	9.89000E+00	9.97500E-01	1.00000E-03
7.56000E+00	9.95000E-01	1.00000E-03	1.04400E+01	9.97000E-01	1.00000E-03
9.62000E+00	9.95450E-01	1.00000E-03	1.04400E+01	9.95800E-01	1.00000E-03
7.36000E+00	9.96250E-01	1.00000E-03	9.60000E+00	9.96400E-01	1.00000E-03
9.52000E+00	9.96700E-01	1.00000E-03	8.75000E+00	9.95450E-01	1.00000E-03
1.19200E+01	9.95850E-01	1.00000E-03	8.57000E+00	9.94250E-01	1.00000E-03
1.06200E+01	9.94400E-01	1.00000E-03	9.17000E+00	9.97150E-01	1.00000E-03
8.58000E+00	9.95700E-01	1.00000E-03	9.10000E+00	9.98800E-01	1.00000E-03
9.65000E+00	9.95300E-01	1.00000E-03	9.25000E+00	9.96450E-01	1.00000E-03
6.10000E+00	9.96600E-01	1.00000E-03	8.87000E+00	9.96550E-01	1.00000E-03
8.08000E+00	9.95550E-01	1.00000E-03	8.65000E+00	9.95850E-01	1.00000E-03
5.76000E+00	9.96750E-01	1.00000E-03	8.13000E+00	9.94250E-01	1.00000E-03
7.90000E+00	9.97250E-01	1.00000E-03	7.26000E+00	9.94850E-01	1.00000E-03
6.72000E+00	9.96250E-01	1.00000E-03	9.65000E+00	9.99150E-01	1.00000E-03
0.00000E+00	1.00145E+00	1.00000E-03	9.70000E+00	1.00020E+00	1.00000E-03
1.64000E+00	9.99250E-01	1.00000E-03	8.09000E+00	9.97900E-01	1.00000E-03
1.64000E+00	9.97100E-01	1.00000E-03	7.65000E+00	9.92550E-01	1.00000E-03
1.64000E+00	9.99550E-01	1.00000E-03	9.09000E+00	9.93550E-01	1.00000E-03
3.27000E+00	9.96500E-01	1.00000E-03	9.42000E+00	9.98100E-01	1.00000E-03
3.27000E+00	9.93950E-01	1.00000E-03	9.78000E+00	9.96350E-01	1.00000E-03
4.91000E+00	9.93200E-01	1.00000E-03	1.00000E-01	9.99200E-01	1.00000E-03
6.54000E+00	9.94300E-01	1.00000E-03	2.00000E-01	1.00020E+00	1.00000E-03
1.31000E+01	1.00100E+00	1.00000E-03	2.90000E-01	9.99750E-01	1.00000E-03
1.29800E+01	9.99400E-01	1.00000E-03	3.90000E-01	9.98600E-01	1.00000E-03
1.05100E+01	9.96150E-01	1.00000E-03	4.90000E-01	1.00200E+00	1.00000E-03
1.10900E+01	9.95450E-01	1.00000E-03	1.00000E-01	1.00140E+00	1.00000E-03
1.31900E+01	9.97200E-01	1.00000E-03	1.00000E-01	1.00225E+00	1.00000E-03
1.33700E+01	9.97350E-01	1.00000E-03	1.00000E-01	9.98700E-01	1.00000E-03

```
   chi =  3.1613 (upper bound = 9.49). The data tests normal.
```

Figure 6.5.2-10 USLSTATS Output - k_{eff} versus Gap Thickness (continued)

```
        Output from statistical treatment

   62 Critical Experiment KEFFs VS Gap Thickness - Experiments 1, 3, 7, 17, & 40

    Number of data points (n)                           62
    Linear regression, k(X)                           0.9988 + (-2.4725E-04)*X
    Confidence on fit (1-gamma) [input]                 95.0%
    Confidence on proportion (alpha) [input]            95.0%
    Proportion of population falling above
    lower tolerance interval (rho) [input]              99.5%
    Minimum value of X                                     0.0000
    Maximum value of X                                    13.3700
    Average value of X                                     7.38403
    Average value of k                                   0.99693
    Minimum value of k                                   0.99255
    Variance of fit, s(k,X)^2                           4.1441E-06
    Within variance, s(w)^2                             1.0000E-06
    Pooled variance, s(p)^2                             5.1441E-06
    Pooled std. deviation, s(p)                         2.2681E-03
    C(alpha,rho)*s(p)                                   8.4077E-03
    student-t @ (n-2,1-gamma)                           1.67100E+00
    Confidence band width, W                            3.9264E-03
    Minimum margin of subcriticality, C*s(p)-W          4.4812E-03

    Upper subcritical limits: (  0.00000 <= X <=  13.37000)
    ***** *********** *******

    USL Method 1 (Confidence Band with
    Administrative Margin)            USL1 = 0.9448 + (-2.4725E-04)*X

    USL Method 2 (Single-Sided Uniform
    Width Closed Interval Approach)    USL2 = 0.9903 + (-2.4725E-04)*X



    USLs Evaluated Over Range of Parameter X:
    **** ********* **** ***** ** ********* **

    X:   0.00   1.91   3.82   5.73   7.64   9.55  11.46  13.37
       --------------------------------------------------------
USL-1:  0.9448 0.9444 0.9439 0.9434 0.9429 0.9425 0.9420 0.9415
USL-2:  0.9903 0.9899 0.9894 0.9889 0.9885 0.9880 0.9875 0.9870
       --------------------------------------------------------

  ****************************************************************************
                             Thus spake USLSTATS
                                    Finis.
```

Table 6.5.2-1 MONK8A Range of Correlated Parameters for Design Basis Fuel

Parameter	Benchmark Minimum Value	Benchmark Maximum Value	Design Basis (WE 17×17 OFA)
Enrichment (wt % ^{235}U)	2.35	7.00	5.00
Rod pitch (cm)	1.26	2.54	1.26
H/U (fissile) atomic ratio	72.1	453.84	111.31
^{10}B plate loading (g/cm^2)	0.000	0.072	0.025
Log energy causing fission	7.31E-08	3.33E-07	2.39E-07
Cluster gap thickness (cm)	0.0	13.37	2.22-3.81
Fuel diameter (cm)	0.743	1.265	0.7844
Clad diameter (cm)	0.8324	1.4150	0.9144
Soluble boron ppm	0	1258	1000

Table 6.5.2-2 MONK8A – Correlation Coefficient for Linear Curve-Fit of Critical Benchmarks

Correlation Studied	Correlation Coefficient (R)
k_{eff} versus enrichment	0.382
k_{eff} versus rod pitch	0.151
k_{eff} versus H/U (fissile) atomic ratio	0.394
k_{eff} versus ^{10}B plate loading	0.273
k_{eff} versus log energy causing fission	0.170
k_{eff} versus cluster gap thickness	0.436
k_{eff} versus fuel diameter	0.312
k_{eff} versus clad diameter	0.284
k_{eff} versus soluble boron ppm	0.555

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
1.01	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Neutron Absorber	0.0720	6.33	Inf	Water	1.00E-07	0.9964	0.0010
1.02	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Neutron Absorber	0.0720	9.03	Inf	Water	9.95E-08	0.9976	0.0010
1.03	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	304L Steel (no boron)	0	10.44	Inf	Water	9.97E-08	0.9950	0.0010
1.04	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	304L Steel (no boron)	0	11.47	Inf	Water	9.95E-08	0.9960	0.0010
1.05	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	304L Steel (1.05% boron)	0.0049	7.56	Inf	Water	1.02E-07	0.9950	0.0010
1.06	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	304L Steel (1.05% boron)	0.0049	9.62	Inf	Water	1.01E-07	0.9955	0.0010
1.07	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	304L Steel (1.62% boron)	0.0074	7.36	Inf	Water	1.02E-07	0.9963	0.0010
1.08	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	304L Steel (1.62% boron)	0.0074	9.52	Inf	Water	9.99E-08	0.9967	0.0010
1.09	3 clusters; 20×17 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	None	Na	11.92	Inf	Water	1.01E-07	0.9959	0.0010
2.01	1.26 (square)	4.75	1.26	0.79	0.94	Al	98.21	0	Na	Na	Na	Na	Water	2.57E-07	0.9960	0.0010
2.02	1.60 (square)	4.75	1.60	0.79	0.94	Al	217.26	0	Na	Na	Na	Na	Water	1.15E-07	0.9968	0.0010
2.03	2.10 (square)	4.75	2.10	0.79	0.94	Al	443.75	0	Na	Na	Na	Na	Water	7.31E-08	0.9960	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
2.04	1.35 (triangular)	4.75	1.35	0.79	0.94	Al	97.08	0	Na	Na	Na	Na	Water	2.80E-07	0.9953	0.0010
2.05	1.72 (triangular)	4.75	1.72	0.79	0.94	Al	217.51	0	Na	Na	Na	Na	Water	1.15E-07	0.9975	0.0010
2.06	2.26 (triangular)	4.75	2.26	0.79	0.94	Al	445.38	0	Na	Na	Na	Na	Water	7.34E-08	0.9965	0.0010
2.07	1.26 (square-1 in 5 missing)	4.75	1.26	0.79	0.94	Al	97.08	0	Na	Na	Na	Na	Water	2.65E-07	0.9950	0.0010
2.08	1.26 (square-1 in 2 missing)	4.75	1.26	0.79	0.94	Al	217.51	0	Na	Na	Na	Na	Water	1.16E-07	0.9974	0.0010
2.09	1.26 (square-1 in 3 missing)	4.75	1.26	0.79	0.94	Al	445.38	0	Na	Na	Na	Na	Water	7.42E-08	0.9969	0.0010
3.01	3 clusters; 8×15 pins	4.31	2.54	1.265	1.415	Al	256.38	0	None	Na	10.62	Inf	Water	1.18E-07	0.9944	0.0010
3.02	3 clusters; 8×15 pins	4.31	2.54	1.265	1.415	Al	256.38	0	304L Steel (no boron)	0	8.58	Inf	Water	1.17E-07	0.9957	0.0010
3.03	3 clusters; 8×15 pins	4.31	2.54	1.265	1.415	Al	256.38	0	304L Steel (no boron)	0	9.65	Inf	Water	1.18E-07	0.9953	0.0010
3.04	3 clusters; 8×15 pins	4.31	2.54	1.265	1.415	Al	256.38	0	304L Steel (1.05% boron)	0.0049	6.10	Inf	Water	1.19E-07	0.9966	0.0010
3.05	3 clusters; 8×15 pins	4.31	2.54	1.265	1.415	Al	256.38	0	304L Steel (1.05% boron)	0.0049	8.08	Inf	Water	1.18E-07	0.9956	0.0010
3.06	3 clusters; 8×15 pins	4.31	2.54	1.265	1.415	Al	256.38	0	304L Steel (1.62% boron)	0.0074	5.76	Inf	Water	1.18E-07	0.9968	0.0010
3.07	3 clusters; 8×15 pins	4.31	2.54	1.265	1.415	Al	256.38	0	304L Steel (1.62% boron)	0.0074	7.90	Inf	Water	1.16E-07	0.9973	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
3.08	3 clusters; 8×15 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Neutron Absorber	0.0720	6.72	Inf	Water	1.19E-07	0.9963	0.0010
7.01	3×3 clusters; 14×14 pins	2.46	1.6358	1.206	1.03	Al	453.84	1037	None	Na	0	Inf	Water	2.56E-07	1.0015	0.0010
7.02	3×3 clusters; 14×14 pins	2.46	1.6358	1.206	1.03	Al	453.84	769	None	Na	1.64	Inf	Water	2.05E-07	0.9993	0.0010
7.03	3×3 clusters; 14×14 pins	2.46	1.6358	1.206	1.03	Al	453.84	0	B_4C Pins	Na	1.64	Inf	Water	2.03E-07	0.9971	0.0010
7.04	3×3 clusters; 14×14 pins	2.46	1.6358	1.206	1.03	Al	453.84	15	B/Al (1.61wt% B)	0.0052	1.64	Inf	Water	1.98E-07	0.9996	0.0010
7.05	3×3 clusters; 14×14 pins	2.46	1.6358	1.206	1.03	Al	453.84	217	Stainless Steel	0	3.27	Inf	Water	1.75E-07	0.9965	0.0010
7.06	3×3 clusters; 14×14 pins	2.46	1.6358	1.206	1.03	Al	453.84	320	B/Al (0.1wt% B)	0.0003	3.27	Inf	Water	1.79E-07	0.9940	0.0010
7.07	3×3 clusters; 14×14 pins	2.46	1.6358	1.206	1.03	Al	453.84	72	B/Al (0.1wt% B)	0.0003	4.91	Inf	Water	1.61E-07	0.9932	0.0010
7.08	3×3 clusters; 14×14 pins	2.46	1.6358	1.206	1.03	Al	453.84	0	None	Na	6.54	Inf	Water	1.44E-07	0.9943	0.0010
27.01	Cylindrical	7.00	1.32	0.743	0.8324	SS	118.39	0	Na	Na	Na	Na	Water	2.09E-07	0.9994	0.0010
32.01	14×14 array	4.74	1.60	0.79	0.94	Al	217.31	0	Na	Na	Na	0.0	Lead and light water	1.32E-07	1.0161	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
32.02	14×14 array	4.74	1.60	0.79	0.94	Al	217.31	0	Na	Na	Na	0.5	Lead and light water	1.26E-07	1.0184	0.0010
32.03	14×14 array	4.74	1.60	0.79	0.94	Al	217.31	0	Na	Na	Na	1.0	Lead and light water	1.18E-07	1.0166	0.0010
32.04	14×14 array	4.74	1.60	0.79	0.94	Al	217.31	0	Na	Na	Na	1.5	Lead and light water	1.14E-07	1.0159	0.0010
40.01	22×22	4.74	1.60	0.79	0.94	Al	217.31	0	Hafnium plate	Na	0.0978	Na	Water	1.33E-07	0.9992	0.0010
40.02	22×22	4.74	1.60	0.79	0.94	Al	217.31	0	Hafnium plate	Na	0.1956	Na	Water	1.34E-07	1.0002	0.0010
40.03	22×22	4.74	1.60	0.79	0.94	Al	217.31	0	Hafnium plate	Na	0.2934	Na	Water	1.33E-07	0.9998	0.0010
40.04	22×22	4.74	1.60	0.79	0.94	Al	217.31	0	Hafnium plate	Na	0.3912	Na	Water	1.34E-07	0.9986	0.0010
40.05	22×22	4.74	1.60	0.79	0.94	Al	217.31	0	Hafnium plate	Na	0.489	Na	Water	1.37E-07	1.0020	0.0010
40.06	21×21	4.74	1.60	0.79	0.94	Al	217.31	0	Hafnium plate	Na	0.0978	Na	Water	1.36E-07	1.0014	0.0010
40.07	20×21	4.74	1.60	0.79	0.94	Al	217.31	0	Hafnium plate	Na	0.0978	Na	Water	1.34E-07	1.0023	0.0010
40.08	20×20	4.74	1.60	0.79	0.94	Al	217.31	0	Hafnium plate	Na	0.0978	Na	Water	1.30E-07	0.9987	0.0010
40.09	22×22	4.74	1.60	0.79	0.94	Al	217.31	0	None	Na	-	Na	Water	1.29E-07	0.9998	0.0010
40.10	21×21	4.74	1.60	0.79	0.94	Al	217.31	0	None	Na	-	Na	Water	1.27E-07	1.0006	0.0010
40.11	21×20	4.74	1.60	0.79	0.94	Al	217.31	0	None	Na	-	Na	Water	1.25E-07	1.0016	0.0010
40.12	20×20	4.74	1.60	0.79	0.94	Al	217.31	0	None	Na	-	Na	Water	1.25E-07	0.9988	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
17.01	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	13.100	0.000	Lead	1.03E-07	1.0010	0.0010
17.02	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	12.980	0.660	Lead	1.00E-07	0.9994	0.0010
17.03	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	10.510	2.616	Lead	1.00E-07	0.9962	0.0010
17.04	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	11.090	0.000	Uranium	1.05E-07	0.9955	0.0010
17.05	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	13.190	1.321	Uranium	1.02E-07	0.9972	0.0010
17.06	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	13.370	1.956	Uranium	1.02E-07	0.9974	0.0010
17.07	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	12.960	2.616	Uranium	1.00E-07	0.9967	0.0010
17.08	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	9.950	5.405	Uranium	1.01E-07	0.9944	0.0010
17.09	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	7.820	10.676	Uranium	9.86E-08	0.9946	0.0010
17.10	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	9.888	0.000	Steel	1.03E-07	0.9975	0.0010
17.11	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	10.438	0.660	Steel	1.03E-07	0.9970	0.0010
17.12	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	10.438	1.321	Steel	1.02E-07	0.9958	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
17.13	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	9.598	2.616	Steel	9.91E-08	0.9964	0.0010
17.14	3 clusters; 16×19 pins	2.35	2.032	1.1176	1.27	Al	398.80	0	Na	Na	8.748	3.912	Steel	9.88E-08	0.9955	0.0010
17.15	18×25(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	8.566	0.000	Steel	1.89E-07	0.9943	0.0010
17.16	18×25(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	9.166	0.660	Steel	1.83E-07	0.9972	0.0010
17.17	18×25(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	9.096	1.321	Steel	1.75E-07	0.9988	0.0010
17.18	18×25(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	9.246	1.684	Steel	1.77E-07	0.9965	0.0010
17.19	18×25(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	8.866	2.344	Steel	1.69E-07	0.9966	0.0010
17.20	18×25(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	8.646	3.005	Steel	1.73E-07	0.9959	0.0010
17.21	18×25(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	8.126	3.912	Steel	1.66E-07	0.9943	0.0010
17.22	18×25(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	7.256	6.726	Steel	1.65E-07	0.9949	0.0010
17.23	18×23(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	9.646	0.000	Lead	1.76E-07	0.9992	0.0010
17.24	18×23(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	9.696	0.660	Lead	1.74E-07	1.0002	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
17.25	18×23(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	8.086	3.276	Lead	1.65E-07	0.9979	0.0010
17.26	18×23(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	7.646	0.000	Uranium	1.86E-07	0.9926	0.0010
17.27	18×23(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	9.086	1.321	Uranium	1.78E-07	0.9936	0.0010
17.28	18×23(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	9.416	2.616	Uranium	1.66E-07	0.9981	0.0010
17.29	18×23(center), 18×20(two outer)	2.35	1.684	1.1176	1.27	Al	398.80	0	Na	Na	9.776	3.912	Uranium	1.67E-07	0.9964	0.0010
10.01	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	19.495	0.000	Lead	1.22E-07	1.0091	0.0010
10.02	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	19.655	0.660	Lead	1.20E-07	1.0074	0.0010
10.03	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	17.915	1.321	Lead	1.18E-07	1.0044	0.0010
10.04	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	9.175	5.405	Lead	1.15E-07	0.9945	0.0010
10.05	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	14.255	0.000	Uranium	1.32E-07	0.9951	0.0010
10.06	3 clusters; 8×12 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	14.195	1.956	Uranium	1.18E-07	0.9974	0.0010
10.07	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	16.925	3.912	Uranium	1.18E-07	0.9977	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
10.08	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	12.365	5.405	Uranium	1.16E-07	0.9922	0.0010
10.09	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	11.765	0.000	Steel	1.26E-07	1.0004	0.0010
10.10	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	13.125	0.660	Steel	1.25E-07	0.9981	0.0010
10.11	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	12.995	1.321	Steel	1.20E-07	0.9992	0.0010
10.12	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	11.315	2.616	Steel	1.17E-07	0.9987	0.0010
10.13	3 clusters; 8×13 pins	4.31	2.54	1.265	1.415	Al	256.38	0	Na	Na	8.675	5.405	Steel	1.16E-07	0.9954	0.0010
10.14	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	14.393	0.000	Steel	3.27E-07	1.0002	0.0010
10.15	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	15.263	0.660	Steel	3.16E-07	1.0015	0.0010
10.16	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	15.393	1.321	Steel	3.04E-07	1.0003	0.0010
10.17	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	15.363	1.956	Steel	2.97E-07	1.0039	0.0010
10.18	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	14.973	2.616	Steel	2.91E-07	0.9994	0.0010
10.19	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	13.343	5.405	Steel	2.80E-07	1.0007	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
10.20	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	17.263	0.000	Lead	3.11E-07	1.0054	0.0010
10.21	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	17.703	0.660	Lead	3.01E-07	1.0047	0.0010
10.22	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	16.953	1.956	Lead	2.89E-07	1.0042	0.0010
10.23	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	13.873	5.001	Lead	2.81E-07	0.9993	0.0010
10.24	3 clusters; 12x16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	14.853	0.000	Uranium	3.33E-07	0.9965	0.0010
10.25	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	16.233	0.660	Uranium	3.23E-07	0.9991	0.0010
10.26	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	17.793	1.321	Uranium	3.14E-07	0.9995	0.0010
10.27	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	18.763	1.956	Uranium	3.05E-07	1.0006	0.0010
10.28	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	18.893	2.616	Uranium	3.01E-07	0.9991	0.0010
10.29	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	18.303	3.276	Uranium	2.88E-07	0.9985	0.0010
10.30	3 clusters; 12×16 pins	4.31	1.892	1.265	1.415	Al	256.38	0	Na	Na	15.923	5.405	Uranium	2.83E-07	0.9988	0.0010

Table 6.5.2-3 MONK8A – JEF 2.2 Library Validation Statistics (continued)

Case	Configuration	wt % ^{235}U	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Clad Mat'l.	H/U (fissile)	Sol. B (ppm)	Poison Type/Absorber	G $^{10}B/cm^2$	Cluster Gap (cm)	Wall/ Cluster (cm)	Reflector	Mean Log(E) Neutrons Causing Fission	k_{eff} (JEF2.2)	σ
50.01	triangular pitch	3.6	1.27	0.76	0.905	Zr	134.24	0	None	na	na	na	Water	2.40E-07	1.0005	0.0010
50.02	triangular pitch	3.6	1.27	0.76	0.905	Zr	134.23	193	None	na	na	na	Water	3.40E-07	1.0013	0.0010
50.03	triangular pitch	3.6	1.27	0.76	0.905	Zr	134.20	700	None	na	na	na	Water	3.50E-07	1.0054	0.0010
50.04	triangular pitch	3.6	1.27	0.76	0.905	Zr	134.18	1014	None	na	na	na	Water	3.71E-07	1.0042	0.0010
50.05	triangular pitch	3.6	1.27	0.76	0.905	Zr	134.16	1258	None	na	na	na	Water	3.82E-07	1.0044	0.0010
50.06	triangular pitch	3.6	1.1	0.76	0.905	Zr	72.08	0	None	na	na		Water	7.84E-07	1.0003	0.0010
50.07	triangular pitch	3.6	1.1	0.76	0.905	Zr	72.08	168	None	na	na		Water	8.38E-07	0.9990	0.0010
50.08	triangular pitch	3.6	1.5	0.76	0.905	Zr	232.53	0	None	na	na		Water	1.55E-07	1.0006	0.0010
50.09	triangular pitch	3.6	1.5	0.76	0.905	Zr	232.45	700	None	na	na		Water	1.81E-07	1.0054	0.0010
50.10	triangular pitch	4.4	1.5	0.76	0.905	Zr	189.50	0	None	na	na		Water	1.54E-07	1.0002	0.0010
50.11	triangular pitch	3.6	1.905	0.76	0.905	Zr	445.28	0	None	na	na		Water	7.67E-08	1.0033	0.0010
50.12	triangular pitch	3.6	1.27	0.76	0.905	Zr	185.93	0	None	na	na		Water	1.97E-07	0.9986	0.0010
50.13	triangular pitch	4.4	1.27	0.76	0.905	Zr	109.40	0	None	na	na		Water	2.76E-07	1.0001	0.0010
50.14	triangular pitch	4.4	1.27	0.76	0.905	Zr	109.39	112	None	na	na		Water	2.86E-07	1.0032	0.0010
50.15	triangular pitch	4.4	1.27	0.76	0.905	Zr	114.01	0	None	na	na		Water	1.37E-07	0.9986	0.0010
50.16	triangular pitch	4.4	1.27	0.76	0.905	Zr	118.14	1258	None	na	na		Water	3.62E-07	1.0042	0.0010
50.17	triangular pitch	3.6	1.5	0.76	0.905	Zr	350.93	0	None	na	na		Water	1.09E-07	0.9974	0.0010

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6.6 Criticality Evaluation for Site Specific Spent Fuel

This section presents the criticality evaluation for fuel assembly types or configurations, which are unique to specific reactor sites. Site specific spent fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, testing programs intended to improve reactor operations and from decommissioning activities. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable by specific evaluation of the configuration.

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6.6.1 Criticality Evaluation for Maine Yankee Site Specific Spent Fuel

In Section 6.4, loading the storage cask with the standard CE 14 × 14 fuel assembly is shown to be less reactive than loading the cask with the most reactive Westinghouse 17 × 17 OFA design basis spent fuel. This analysis addresses variations in fuel assembly dimensions, variable enrichment axial zoning patterns, annular axial fuel blankets, removed fuel rods or empty rod positions, fuel rods placed in guide tubes, fuel assemblies with a start-up source or other components in a guide tube, consolidated fuel assemblies, and damaged fuel and fuel debris. These configurations are not included in the standard fuel analysis, but are present in the site fuel inventory that must be stored.

6.6.1.1 Maine Yankee Fuel Criticality Model

The criticality evaluations of the Maine Yankee fuel inventory require the basket cell and basket in cask models described in Section 6.3 and 6.4. The basket cell model is principally employed in the most reactive dimension evaluation for the Maine Yankee undamaged fuel types. The basket cell model represents an infinite array of fuel tubes separated by one-inch flux traps and neglects the radial neutron leakage of the basket. This will result in k_{eff} values greater than 0.95. The basket cell model is, therefore, only used to determine relative reactivities of the various physical dimensions of the Maine Yankee fuel inventory, not to establish maximum k_s values for the basket loaded with Maine Yankee fuel assemblies. The basket-in-cask model is used for the evaluation of the remaining fuel configurations. The basket criticality model uses the nominal basket configuration with full moderation under accident conditions, where accident conditions implying the loss of fuel cladding integrity and flooding of the pellet to cladding gap in all fuel rods. The analyses presented are performed using the UMS® transport cask shield geometry. Based on the evaluation presented in Section 6.4 and the licensing analysis of the transport overpack, the most reactive transportable storage canister configuration is independent of the canister outer shell geometry (i.e., different casks – transport, transfer, or storage). Since the criticality evaluation is not sensitive to the shielding geometry outside of the canister, this result is applicable to the concrete storage cask and the transfer cask. The transport cask criticality model is identical to the transfer cask and storage cask models with the exception that the radial shielding outside of the canister is comprised of a total of 4.75 inches of steel, 2.75 inches of NS-4-FR neutron shielding and 2.75 inches of lead. The $k_{eff} + 2\sigma$ of this configuration is 0.9210, which is slightly lower than the wet gap $k_{eff} + 2\sigma$ values of 0.9238 and 0.9234 reported in Tables 6.4-6 and 6.4-7 for the transfer cask and storage cask, respectively.

6.6.1.2 Maine Yankee Undamaged Spent Fuel

The evaluation of the undamaged Maine Yankee spent fuel inventory demonstrates that, under all conditions, the maximum reactivity of the UMS® basket loaded with Maine Yankee fuel assemblies is bounded by the Westinghouse 17 × 17 OFA evaluation presented in Section 6.4. The undamaged fuel assembly evaluation includes the determination of maximum reactivity dimensions of the Maine Yankee fuel assemblies, and the reactivity effects of variably enriched assemblies, annular axial end blankets, removed rods, fuel in guide tubes, and consolidated fuel assemblies. Where necessary, loading restrictions are applied to limit the number and location of the basket payload evaluated.

6.6.1.2.1 Fuel Assembly Lattice Dimensional Variations

Maine Yankee 14 × 14 PWR fuel has been provided by Combustion Engineering, Exxon/ANF, and Westinghouse. The range of fuel assembly dimensions evaluated for Maine Yankee is shown in Table 6.6.1-1. Bounding fuel assembly dimensions are determined using the guidelines presented in Section 6.4.4 and are reported in Table 6.6.1-2. The dimensional perturbations that can increase the reactivity of an undermoderated array of fuel assemblies in a flooded system (including flooding the fuel-cladding gap) are:

- Decreasing the cladding outside diameter (OD)
- Increasing the cladding inside diameter (ID) (i.e., increasing the gap)
- Decreasing the pellet diameter
- Decreasing the guide tube thickness

To conservatively model the cladding thickness of the Maine Yankee standard fuel, the outside diameter of the cladding is decreased until the cladding thickness reaches the minimum. The pellet diameter is studied separately to determine which diameter maximizes the reactivity of the assembly. This study is performed using an infinite array of hybrid 14 × 14 fuel assemblies. These hybrid assemblies have the combination of the most reactive dimensions listed in Table 6.6.1-2 and are used in the evaluation of site specific fuel configurations as described in the following sections. The pellet diameter is modeled first at the maximum diameter; then it is iteratively decreased until a peak reactivity (H/U ratio) is reached. The results of this study are reported in Table 6.6.1-3. The maximum reactivity occurs at a pellet diameter of 0.3527 inches. This pellet diameter is conservatively used in the analyses of an assembly with 176 fuel rods.

The reactivity of an infinite array of basket unit cells containing infinitely tall, hybrid 14 × 14 fuel assemblies and a flooded fuel-cladding gap is $k_{eff} + 2\sigma = 0.96268$. This is less reactive than the same array of Westinghouse 17 × 17 OFA assemblies ($k_{eff} + 2\sigma = 0.9751$ from Table 6.4-1). Therefore, the design basis Westinghouse 17 × 17 OFA fuel criticality evaluation is bounding. The conservatism obtained by decreasing the pellet diameter below that of the reported Maine Yankee fuel pellet diameter is equivalent to a Δk_{eff} of 0.00247.

The most reactive lattice dimensions determined by the basket cell model are incorporated into the basket in cask model. Evaluating 24 hybrid 14 × 14 fuel assemblies with the most reactive pellet diameter for the accident condition produces a $k_{eff} + 2\sigma$ of 0.91014. This is less reactive than the accident condition for the transport cask loaded with the Westinghouse 17 × 17 OFA assemblies ($k_{eff} + 2\sigma$ of 0.9210). Therefore, the Westinghouse 17 × 17 OFA fuel criticality evaluation is bounding.

6.6.1.2.2 Variably Enriched Fuel Assemblies

Two batches of fuel used at Maine Yankee contain variably enriched fuel rods. Fuel rod enrichments of one batch are 4.21 wt % ^{235}U and 3.5 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.99 wt %. In the other batch, the fuel rod enrichments are 4.0 wt % and 3.4 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.92 wt %. Loading 24 variably enriched fuel assemblies having both a maximum fuel rod enrichment of 4.21 wt % and a maximum planar average enrichment of 3.99 wt % results in a $k_{eff} + 2\sigma$ of 0.89940. Using a planar fuel rod enrichment of 4.2 wt % results in a $k_{eff} + 2\sigma$ of 0.91014. Therefore, all of the fuel rods are conservatively modeled as if enriched to 4.2 wt % ^{235}U for the remaining Maine Yankee analyses.

6.6.1.2.3 Assemblies with Annular Axial End Blankets

One batch of variably enriched fuel also incorporates 2.6 wt % ^{235}U axial end blankets with annular fuel pellets. The top and bottom 5% of the active fuel length of each fuel rod in this batch contains annular fuel pellets having an inner diameter of 0.183 inches.

This geometry is discretely modeled as approximately 5% annular fuel, 90% solid fuel and then 5% annular fuel, with all fuel materials enriched to 4.2 wt % ^{235}U. The diameter of all pellets is initially modeled as the most reactive pellet diameter. The accident case model, which includes flooding of the fuel cladding annulus, is used in this evaluation. Axial periodic boundary conditions are placed on the model, retaining the conservatism of the infinite fuel length. Use of a smaller pellet diameter is not considered to be conservative when evaluating the annular fuel pellets. The smaller pellet diameter is the most reactive diameter under the assumption that it is solid and not an annulus. Flooding the axial end blanket annulus provides additional moderator to the fuel lattice. Therefore, the diameter of the annular pellets is also modeled as the maximum pellet diameter of 0.380 inch. The 0.380-inch diameter is applied to the annular pellets, while the smaller diameter is applied to the solid pellets. The results of both evaluations are reported in Table 6.6.1-4.

The most reactive annular fuel model for the annular axial end blankets results in a slightly more reactive system than the hybrid fuel accident evaluation, the annular condition is less reactive than the evaluation including Westinghouse 17 × 17 OFA assemblies. Therefore, the Westinghouse 17 × 17 OFA fuel criticality evaluation is bounding.

6.6.1.2.4 Assemblies with Removed Fuel Rods

Some of the Maine Yankee fuel assemblies have had fuel rods removed from the 14 × 14 lattice or have had poison rods replaced by hollow zirconium alloy tubes. The exact number and location of removed rods and hollow tubes differs from one assembly to another. To determine a bounding reactivity for these assemblies, an analysis changing the location and the number of removed rods is performed. The removed rod analysis bounds that of the hollow tube analysis, since the zirconium alloy tubes displace moderator in the under moderated assembly lattice. For each case, all 24 assemblies are centered in the fuel tubes and have the same number and location of removed fuel rods. Various patterns of removed fuel rod locations are analyzed when the number of removed fuel rods is small enough to allow a different and possibly more reactive geometry. As the number of removed fuel rods increases, the number of possible highly reactive locations for these removed rods decreases. The fuel pellet diameter is modeled first at the most reactive diameter (0.3527 inches as determined in Section 6.6.1.2.1), and then at the maximum diameter of 0.380 inches.

The results of these analyses, which determine the most reactive number and geometry of removed rods for any Maine Yankee assembly, are presented in Tables 6.6.1-5 and 6.6.1-6. Table 6.6.1-5 contains the results based on a 0.3527-inch fuel pellet. All of the removed fuel rod cases using the smaller pellet diameter show cask reactivity levels lower than those of Westinghouse 17 × 17 OFA fuel. Table 6.6.1-6 contains the results of the evaluation using the maximum pellet diameter of 0.380 inch. Using the maximum pellet diameter provides for a more reactive system, since moderator is added (at the removed rod locations), to an assembly that contains more fuel. The most reactive removed fuel rod case occurs when 24 fuel rods are removed in the diamond shaped geometry shown in Figure 6.6.1-1, from the model containing the largest allowed pellet diameter.

This case represents the bounding number and geometry of removed fuel rods for the Maine Yankee fuel assemblies. It results in a more reactive system than either the Maine Yankee hybrid 14 × 14 fuel accident case or the Westinghouse 17 × 17 OFA accident case assuming unrestricted loading. However, as shown in Table 6.6.1-6, when the loading of any assembly with less than 176 fuel rods or filler rods is restricted to the four corner fuel tubes, the reactivity of the worse case drops well below that of the Westinghouse 17 × 17 OFA fuel assemblies. Therefore, loading of Maine Yankee fuel assemblies with removed fuel rods, or with hollow zirconium alloy tubes, is restricted to the four corner fuel tube positions of the basket. With this loading restriction, the Westinghouse 17 × 17 OFA criticality evaluation remains bounding.

6.6.1.2.5 Assemblies with Fuel Rods in the Guide Tubes

A few of the Maine Yankee undamaged assemblies may contain up to two undamaged fuel rods in some of the guide tubes (i.e., allowing for the potential storage of individual undamaged fuel rods in an undamaged fuel assembly). To evaluate loading of these assemblies into the canister, an analysis adding 1 and then 2 undamaged fuel rods into 1, 2, 3 and then 5 guide tubes is made. This approach considers a fuel assembly with up to 186 fuel rods. The results of the evaluation of these configurations are shown in Table 6.6.1-7. While higher in reactivity than the Maine Yankee hybrid base case, any fuel configuration with up to 2 fuel rods per guide tube is less reactive than the accident case for the Westinghouse 17 × 17 OFA fuel assemblies. Therefore, the Westinghouse 17 × 17 OFA fuel criticality evaluation is bounding.

Fuel rods may also be inserted in the guide tubes of fuel assemblies from which the fuel rods were removed (i.e., fuel rods removed from a fuel assembly and re-installed in the guide tubes of the same fuel assembly). These fuel rods may be undamaged or damaged. The maximum number of fuel rods in these assemblies, including fuel rods in the guide tubes remains 176. These configurations are restricted to loading in one of the two configurations of the Maine Yankee Fuel Can in a corner fuel position in the basket. As shown in Section 6.6.1.2.4 for the removed fuel rods, and Section 6.6.1.3 for the damaged fuel, the maximum reactivity of Maine Yankee assemblies containing 176 fuel rods in various configurations is bounded by the Westinghouse 17 × 17 OFA evaluation. These non-standard Maine Yankee assemblies are restricted to the corner fuel positions.

In addition to the fuel rods, some Maine Yankee assemblies may contain poison shim rods in guide tubes. These solid fill rods will serve as parasitic absorber and displace moderator and are, therefore, not included in the criticality model but are bounded by the evaluation performed.

6.6.1.2.6 Consolidated Fuel

The consolidated fuel is a 17 × 17 array of undamaged fuel rods with a pitch of 0.492 inches. Some of the locations in the array contain solid fill rods and some are empty. To determine the reactivity of the consolidated fuel lattice with empty fuel rod positions, an analysis changing the location and the number of empty positions is performed. This analysis considers 24 consolidated fuel lattices in the basket. All 24 consolidated fuel lattices are centered in the fuel tubes and have the same number and location of empty fuel rod positions.

As shown in Section 6.6.1.2.4, the removed fuel rod configuration with a 0.380-inch pellet diameter provides a more reactive system than a system using the optimum pellet diameter from Section 6.6.1.2.1. The larger pellet cases are more reactive, since moderator is added at the empty fuel rod positions to an assembly that contains more fuel. Therefore, the consolidated assembly empty rod position evaluation is performed with the 0.380-inch pellet diameter.

The results of this evaluation are shown in Table 6.6.1-8. Configurations having more than 73 empty positions result in a more reactive system than the Westinghouse 17 × 17 OFA model. The most reactive consolidated assembly case occurs with 113 empty rod positions in the geometry shown in Figure 6.6.1-2. However, when the loading of the consolidated fuel is restricted to the four corner fuel tubes, the reactivity of the system is lower than the accident condition of the basket loaded with Westinghouse 17 × 17 OFA assemblies. Therefore, loading of the consolidated fuel is restricted to the four corner fuel tube positions of the basket. With this loading restriction, the Westinghouse 17 × 17 OFA fuel criticality evaluation is bounding.

6.6.1.2.7 Conclusions

The criticality analyses for the Maine Yankee site specific fuel demonstrate that the UMS® basket loaded with these fuel assemblies results in a system that is less reactive than loading the basket with the Westinghouse 17 × 17 OFA fuel assemblies, provided that loading is restricted to the four corner fuel tube positions in the basket for:

- All 14 × 14 fuel assemblies with less than 176 fuel rods or solid filler rods
- All 14 × 14 fuel assemblies with hollow zirconium alloy tubes
- All 17 × 17 consolidated fuel lattices
- All 14 × 14 fuel assemblies with fuel rods in the guide tubes and a maximum of 176 fuel rods or solid rods and fuel rods.

The following Maine Yankee fuels are not restricted as to loading position within the basket:

- All 14 × 14 fuel assemblies with 176 fuel rods or solid filler rods at a maximum enrichment of 4.2 wt % ^{235}U.
- Variably enriched fuel with a maximum fuel rod enrichment of 4.21 wt % ^{235}U with a maximum planar average enrichment of 3.99 wt % ^{235}U.
- Fuel with solid stainless steel filler rods, solid zirconium alloy filler rods or solid poison shim rods in any location.
- Fuel with annular axial end blankets of up to 4.2 wt % ^{235}U.
- Fuel with a maximum of 2 undamaged fuel rods in each guide tube for a total of 186 fuel rods.

Assemblies defined as unrestricted may be loaded into the basket in any basket location and may be mixed in the same basket. While not analyzed in detail, CEAs and ICI thimble assemblies may be loaded into any undamaged assemblies. These components displace a significant amount of water in the fuel lattice while adding parasitic absorber, thereby reducing system reactivity.

Since the storage cask and the transfer cask loaded with the Westinghouse 17 × 17 OFA fuel assemblies is criticality safe, it is inherent that the same cask loaded with the less reactive fuel assemblies employed at Maine Yankee, using the fuel assembly loading restrictions presented above, is also criticality safe.

6.6.1.3 Maine Yankee Damaged Spent Fuel and Fuel Debris

Damaged fuel assemblies are placed in one of the two configurations of the Maine Yankee Fuel Can prior to loading in the basket (see Drawings 412-501 and 412-502). The Maine Yankee Fuel Can has screened openings in the baseplate and the lid to permit drainage, vacuum drying, and inerting of the can. This evaluation conservatively considers 100% of the fuel rods in the fuel can as damaged.

Fuel debris can be loaded in a rod or tube structure that is subsequently loaded into a Maine Yankee fuel can. The mass of fuel debris placed in the rod or tube is restricted to the mass equivalent of a fuel rod of an undamaged fuel assembly.

The Maine Yankee spent fuel inventory includes fuel assemblies with fuel rods inserted in the guide tubes of the assembly. If the integrity of the cladding of the fuel rods in the guide tubes cannot be ascertained, then those fuel rods are assumed to be damaged.

6.6.1.3.1 Damaged Fuel Rods

All of the spent fuel classified as damaged, and all of the spent fuel not in its original lattice, are stored in a Maine Yankee fuel can. This fuel is analyzed using a 100% fuel rod failure assumption. The screened fuel can is designed to preclude the release of pellets and gross particulate to the canister cavity. Evaluation of the canister with four (4) Maine Yankee fuel cans containing CE 14 × 14 fuel assemblies that have up to 176 damaged fuel rods, or consolidated fuel consisting of up to 289 fuel rods, considers 100% dispersal of the fuel from these rods within the fuel can. The Maine Yankee fuel can is restricted to loading in the four corner positions of the basket.

All loose fuel in each analysis is modeled as a homogeneous mixture of fuel and water of which the volume fractions of the fuel versus the water are varied from 0 - 100. By varying the fuel fraction up to 100%, this evaluation addresses fuel masses significantly larger than those available in a standard or consolidated fuel assembly. First, loose fuel from damaged fuel rods within a fuel assembly is evaluated between the remaining rods of the most reactive missing rod array. The results of this analysis, provided in Table 6.6.1-9, show a slight decrease in the reactivity of the system. This results from adding fuel to the already optimized H/U ratio of the bounding missing rod array. This effectively returns the system to an undermoderated state. Second, loose fuel is considered above and below the active fuel region of this most reactive missing rod array. This analysis is performed within a finite cask model. The results of this study, provided in Table 6.6.1-10, show that any possible mixture combination of fuel and water above and below the active fuel region, and hence, above and below the neutron absorber sheet coverage, will not significantly increase the reactivity of the system beyond that of the missing rod array. Loose fuel is also considered to replace all contents of the Maine Yankee fuel can in each four corner fuel tube location. The results of this study, provided in Table 6.6.1-11, show that any mixture of fuel and water within this cavity will not significantly increase the reactivity of the system beyond that of the missing rod array.

Damaged fuel within the fuel can may also result from a loss of integrity of a consolidated fuel assembly. As described in Section 6.6.1.2.6, the consolidated assembly missing rod study shows that a potentially higher reactivity heterogeneous configuration does not increase the overall reactivity of the system beyond that of loading 24 Westinghouse 17 × 17 OFA assemblies when this configuration is restricted to the four corner locations. The homogeneous mixture study of loose fuel and water replacing the contents of the Maine Yankee fuel can (in each of the four corner fuel tube locations) considers more fuel than is present in the 289 fuel rod consolidated assembly. This study shows that a homogeneous mixture at an optimal H/U ratio within the fuel can also does not affect the reactivity of the system.

The transfer and the storage casks loaded with the Westinghouse 17 × 17 OFA fuel assemblies remain subcritical. Therefore, it is inherent that a statistically equivalent, or less reactive, canister loading of 4 Maine Yankee fuel cans containing assemblies with up to 176 damaged rods, or consolidated assemblies with up to 289 rods and 20 of the most reactive Maine Yankee fuel assemblies, will remain subcritical. Consequently, assemblies with up to 176 damaged rods and consolidated assemblies with up to 289 rods are allowed contents as long as they are loaded into Maine Yankee fuel cans.

6.6.1.3.2 Fuel Debris

Prior to loading fuel debris into the screened Maine Yankee fuel can, fuel debris must be placed into a rod type structure. Placing the debris into rods confines the spent nuclear material to a known volume and allows the fuel debris to be treated identically to the damaged fuel for criticality analysis.

Based on the arguments presented in Section 6.6.1.3.1, the maximum k_s of the UMS® canister with fuel debris will be less than 0.95, including associated uncertainty and bias.

6.6.1.4 Fuel Assemblies with a Source or Other Component in Guide Tubes

The effect on reactivity from loading Maine Yankee fuel assemblies with components inserted in the center or corner guide tube positions is also evaluated. These components include start-up sources, Control Element Assembly (CEA) fingertips, and a 24-inch ICI segment. Start-up sources must be inserted in the center guide tube. The CEA fingertips and ICI segment must be inserted in a corner guide tube that is closed at the bottom end of the assembly and closed at the top using a CEA flow plug.

6.6.1.4.1 Assemblies with Start-up Sources

Maine Yankee has three Pu-Be sources and two Sb-Be sources that will be installed in the center guide tubes of 14 × 14 assemblies that subsequently must be loaded in one of the four corner fuel positions of the basket. Each source is designed to fit in the center guide tube of an assembly. All five of these start-up sources contain Sb-Be pellets, which are 50% beryllium (Be) by volume. The moderation potential of the Be is evaluated to ensure that this material will not increase the reactivity of the system beyond that reported for the accident condition. The antimony (Sb) content is ignored. The start-up source is assumed to remain within the center guide tube for all conditions. The base case infinite height model used for comparison is the bounding Maine Yankee geometry with fuel assemblies that have 24 empty rod positions in the most reactive geometry, in the four corner locations of the basket, i.e., Case "24 (Four Corners)" reported in Table 6.6.1-6. The center guide tube of this model is filled with 50% water and 50% Be. The analysis assumes that assemblies with start-up sources are loaded in all four of the basket corner fuel positions. This configuration, resulting in a system reactivity of $k_{eff} \pm \sigma$, or 0.91085 ± 0.00087, shows that loading Sb-Be sources or the used Pu-Be sources into the center guide tubes of the assemblies in the four corner locations of the basket does not significantly impact the reactivity of the system.

One of the three Pu-Be sources was never irradiated. Analysis of this source is equivalent to assuming that the spent Pu-Be sources are fresh. The unused source has 1.4 grams of plutonium in two capsules. All of this material is conservatively assumed to be in one capsule and is modeled as ^{239}Pu. The diameter of the capsule cavity is 0.270 inch and its length is 9.75 inches. This corresponds to a capsule volume of approximately 9.148 cubic centimeters. Thus, the 1.4 grams of ^{239}Pu occupies ~0.77% of the volume at a density of 19.84 g/cc. This material composition is then conservatively assumed to fill the entire center guide tube, which models considerably more ^{239}Pu than is actually present within the Pu-Be source. The remaining volume of the guide tube is analyzed at various fractions of Be, water and/or void to ensure that any combination of these materials is considered. The results of these analyses, provided in Table 6.6.1-12, show that loading a fresh Pu-Be start-up source into the center guide tube of each of the four corner assemblies does not significantly impact the reactivity of the system. Both heterogeneous and homogeneous analyses are performed.

6.6.1.4.2 Fuel Assemblies with Inserted CEA Fingertips or ICI String Segment

Maine Yankee fuel assemblies may have CEA finger ends (fingertips) or an ICI segment inserted in one of the four corner guide tubes of the same 14 × 14 assembly. The ICI segment is approximately 24 inches long. These components do not contain fissile or moderating material. Therefore, it is conservative to ignore these components, as they displace moderator when the basket is flooded, thereby reducing reactivity.

6.6.1.4.3 Maine Yankee Miscellaneous Component Loading Restrictions

Based on the evaluation of Maine Yankee fuel assemblies with start-up sources, CEA fingertips, or an ICI segment inserted in guide tubes, the following loading restrictions apply:

1) Any Maine Yankee fuel assembly having a component evaluated in this section inserted in a corner or center guide tube must be loaded in one of the four corner fuel loading positions of the UMS® basket. Basket corner positions are also peripheral positions and are marked "P/C" in Figure 2.1.3.1-1.
2) Start-up sources shall be restricted to loading in the center guide tubes of fuel assemblies classified as undamaged and must be loaded in a Class 1 canister.
3) Only one start-up source may be loaded into any undamaged fuel assembly.
4) The CEA finger tips and ICI segment must be loaded in a guide tube location that is closed at the bottom end (corner guide tubes) of an undamaged fuel assembly. The guide tube must be closed at the top end using a CEA flow plug.
5) Fuel assemblies having a CEA flow plug installed must be loaded in a Class 2 canister.
6) Up to four undamaged fuel assemblies with inserted start-up sources may be loaded in any canister (using the four corner positions of the basket).

When loaded in accordance with these restrictions, the evaluated components do not significantly impact the reactivity of the system.

6.6.1.5 Maine Yankee Fuel Comparison to Criticality Benchmarks

The most reactive system configuration parameters for Maine Yankee fuel have been compared to the range of applicability of the critical benchmarks evaluated using the KENO-Va code of the SCALE 4.3 CSAS sequence. As shown in the following table, all of the Maine Yankee fuel parameters fall within the benchmark range.

Parameter	**Benchmark Minimum Value**	**Benchmark Maximum Value**	**Maine Yankee Fuel Most Reactive Configuration**
Enrichment (wt. % ^{235}U)	2.35	4.74	4.2
Rod pitch (cm)	1.26	2.54	1.50
H/U volume ratio	1.6	11.5	2.6
^{10}B areal density (g/cm^2)	0.00	0.45	0.025
Average energy group causing fission	21.7	24.2	22.5
Flux gap thickness (cm)	0.64	5.16	2.22 to 3.81
Fuel diameter (cm)	0.790	1.265	0.896
Clad diameter (cm)	0.940	1.415	1.111

The H/U volume ratio for the assembly is shown. The lattice H/U volume ratio is 2.2 for the clad gap flooded scenario.

The results of the NAC-UMS® Storage System benchmark calculations are provided in Section 6.5.1.

Figure 6.6.1-1 24 Removed Fuel Rods - Diamond Shaped Geometry, Maine Yankee Site Specific Fuel

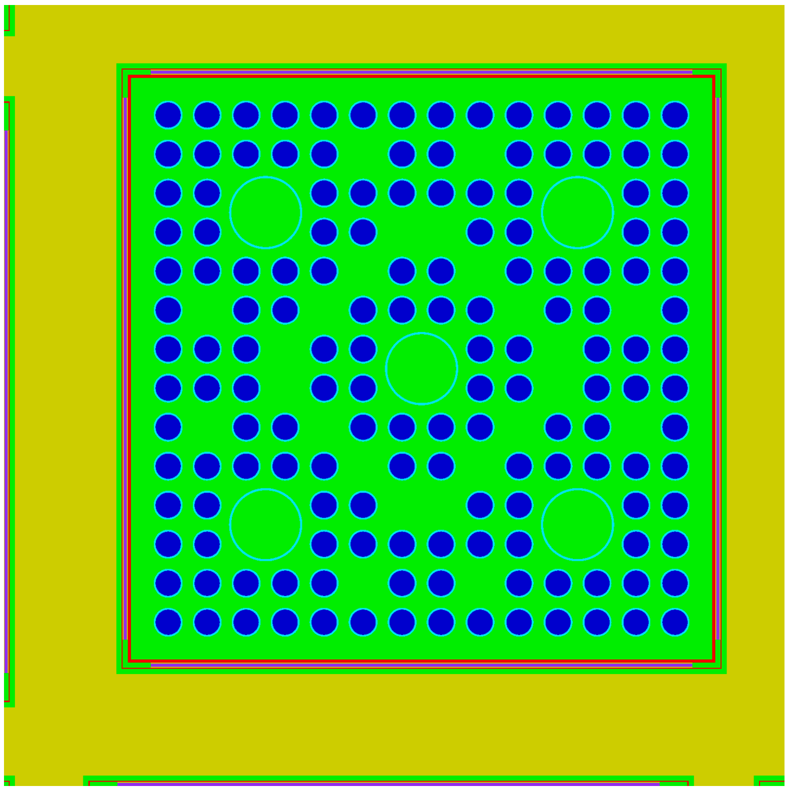

Figure 6.6.1-2 Consolidated Fuel Geometry, 113 Empty Fuel Rod Positions, Maine Yankee Site Specific Fuel

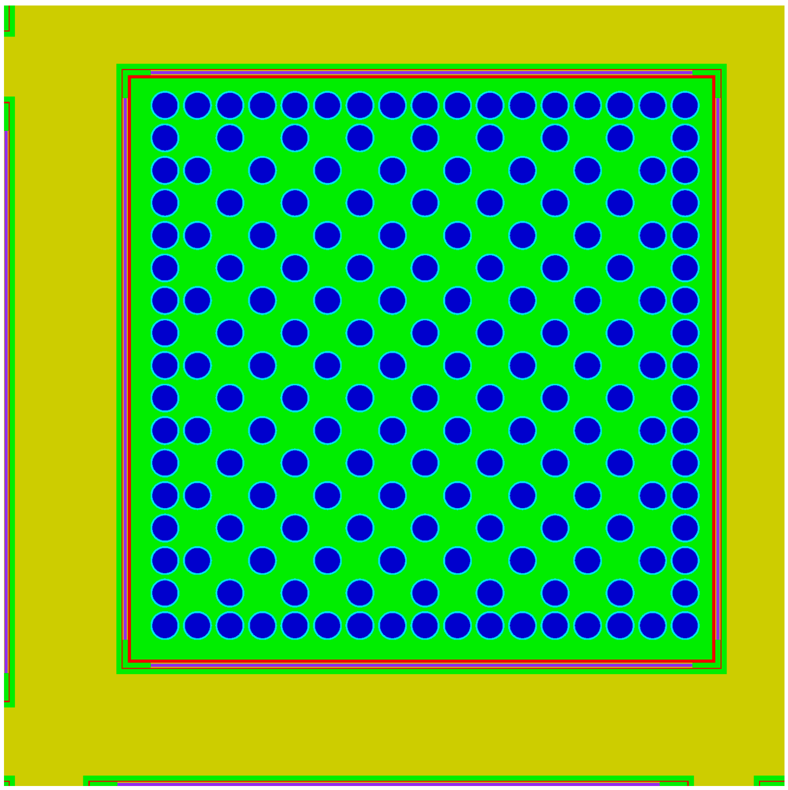

Table 6.6.1-1 Maine Yankee Standard Fuel Characteristics

Fuel Class[1]	Vendor	Array	Version	Number of Fuel Rods	Pitch (in.)	Rod Diameter (in.)	Clad ID (in.)	Clad Thickness (in.)	Pellet Diameter (in.)	GT[2] Thickness (in.)
1	CE	14×14	Std.	160[3]-176	0.570-0.590	0.438-0.442	0.3825-0.3895	0.024-0.028	0.376-0.380	0.036-0.040
1	Ex/ANF	14×14	CE	164[4]-176	0.580	0.438-0.442	0.3715-0.3795	0.0294-0.031	0.3695-0.3705	0.036-0.040
1	WE	14×14	CE	176	0.575-0.585	0.438-0.442	0.3825-0.3855	0.0262-0.028	0.376-0.377	0.034-0.038

1. All fuel rods are zirconium alloy clad.
2. Guide Tube thickness.
3. Up to 16 fuel rod positions may have solid filler rods or burnable poison rods.
4. Up to 12 fuel rod positions may have solid filler rods or burnable poison rods.

Table 6.6.1-2 Maine Yankee Most Reactive Fuel Dimensions

Parameter	Bounding Dimensional Value
Maximum Rod Enrichment[1]	4.2 wt % ^{235}U
Maximum Number of Fuel Rods[2]	176
Maximum Pitch (in.)	0.590
Maximum Active Length (in.)	N/A – Infinite Model
Minimum Clad OD (in.)	0.4375
Maximum Clad ID (in.)	0.3895
Minimum Clad Thickness (in.)	0.024
Maximum Pellet Diameter (in.)	0.3800 - Study
Minimum Guide Tube OD (in.)	1.108
Maximum Guide Tube ID (in.)	1.040
Minimum Guide Tube Thickness (in.)	0.034

1. Variably enriched fuel assemblies may have a maximum fuel rod enrichment of 4.21 wt % ^{235}U with a maximum planar average enrichment of 3.99 wt % ^{235}U.
2. Assemblies with less than 176 fuel rods or solid dummy rods are addressed after the determination of the most reactive dimensions.

Table 6.6.1-3 Maine Yankee Pellet Diameter Study

Diameter (inches)	k_{eff}	σ	k_{eff} +2σ
0.3800	0.95585	0.00085	0.95755
0.3779	0.95784	0.00080	0.95944
0.3758	0.95714	0.00085	0.95884
0.3737	0.95863	0.00082	0.96027
0.3716	0.95862	0.00084	0.96030
0.3695	0.95855	0.00083	0.96021
0.3674	0.95863	0.00085	0.96033
0.3653	0.95982	0.00084	0.96150
0.3632	0.95854	0.00088	0.96030
0.3611	0.95966	0.00083	0.96132
0.3590	0.95990	0.00084	0.96158
0.3569	0.96082	0.00082	0.96246
0.3548	0.96053	0.00083	0.96219
0.3527	0.96104	0.00082	0.96268
0.3506	0.95964	0.00087	0.96138
0.3485	0.95993	0.00086	0.96165
0.3464	0.95916	0.00084	0.96084
0.3443	0.95847	0.00083	0.96013
0.3422	0.95876	0.00083	0.96042
0.3401	0.95865	0.00081	0.96027
0.3380	0.95734	0.00084	0.95902

Table 6.6.1-4 Maine Yankee Annular Fuel Results

Case Description	k_{eff}	σ	k_{eff} + 2σ
All pellets with a diameter of 0.3527 inches	0.90896	0.00083	0.91061
Annular pellet diameter changed to 0.3800 inches	0.91013	0.00087	0.91187

Table 6.6.1-5 Maine Yankee Removed Rod Results with Small Pellet Diameter

Number of Removed Rods	Number of Fuel Rods	k_{eff}	σ	$k_{eff} + 2\sigma$
4	172	0.91171	0.00088	0.91347
4	172	0.91292	0.00086	0.91464
4	172	0.91479	0.00081	0.91640
4	172	0.91125	0.00087	0.91299
6	170	0.91418	0.00087	0.91592
6	170	0.91264	0.00085	0.91435
6	170	0.91314	0.00086	0.91487
6	170	0.90322	0.00086	0.90493
8	168	0.91555	0.00087	0.91729
8	168	0.91490	0.00093	0.91676
8	168	0.91457	0.00088	0.91633
8	168	0.91590	0.00087	0.91764
8	168	0.89729	0.00088	0.89905
12	164	0.91654	0.00086	0.91827
12	164	0.91469	0.00085	0.91639
12	164	0.91149	0.00083	0.91315
16	160	0.91725	0.00084	0.91893
16	160	0.91567	0.00084	0.91735
16	160	0.90986	0.00088	0.91162
16	160	0.90849	0.00083	0.91015
16	160	0.90704	0.00086	0.90876
24	152	0.91572	0.00083	0.91739
32	144	0.91037	0.00088	0.91213
48	128	0.89385	0.00085	0.89554
48	128	0.84727	0.00079	0.84886
64	112	0.79602	0.00083	0.79768
96	80	0.69249	0.00077	0.69402
Westinghouse 17 × 17 OFA		0.9192	0.0009	0.9210

Table 6.6.1-6 Maine Yankee Removed Fuel Rod Results with Maximum Pellet Diameter

Number of Removed Rods	**Number of Fuel Rods**	$\mathbf{k_{eff}}$	σ	$\mathbf{k_{eff} + 2\sigma}$
4	172	0.91078	0.00086	0.91250
4	172	0.90916	0.00085	0.91085
4	172	0.91164	0.00087	0.91338
4	172	0.90809	0.00085	0.90979
6	170	0.91223	0.00085	0.91393
6	170	0.91223	0.00080	0.91384
6	170	0.91270	0.00086	0.91442
6	170	0.90245	0.00086	0.90416
6	170	0.89801	0.00086	0.89972
8	168	0.91567	0.00085	0.91736
8	168	0.91448	0.00085	0.91618
8	168	0.91355	0.00086	0.91526
8	168	0.91293	0.00085	0.91463
12	164	0.91639	0.00090	0.91818
12	164	0.91803	0.00086	0.91974
12	164	0.91235	0.00083	0.91401
16	160	0.91665	0.00091	0.91847
16	160	0.92136	0.00087	0.92310
16	160	0.91231	0.00084	0.91400
16	160	0.90883	0.00087	0.91057
24	152	0.92227	0.00087	0.92400
32	144	0.92164	0.00088	0.92340
48	128	0.91212	0.00081	0.91373
48	128	0.86308	0.00082	0.86472
64	112	0.81978	0.00080	0.82138
88	88	0.72087	0.00083	0.72247
24 (Four Corners)	152	0.91153	0.00085	0.91323
Westinghouse 17 × 17 OFA		0.9192	0.0009	0.9210

Table 6.6.1-7 Maine Yankee Fuel Rods in Guide Tube Results

Number of Guide Tubes with Rods	Number of Rods in Each	k_{eff}	σ	$k_{eff} + 2\sigma$
1	1	0.91102	0.00089	0.91280
2	1	0.91059	0.00088	0.91234
3	1	0.91172	0.00087	0.91346
5	1	0.91411	0.00086	0.91583
1	2	0.91169	0.00090	0.91349
2	2	0.91201	0.00087	0.91375
3	2	0.91173	0.00086	0.91344
5	2	0.91357	0.00086	0.91529
Design Basis Westinghouse 17 × 17 OFA		0.9192	0.0009	0.9210

Table 6.6.1-8 Maine Yankee Consolidated Fuel Empty Fuel Rod Position Results

Number of Empty Positions	Number of Fuel Rods	k_{eff}	σ	$k_{eff} + 2\sigma$
4	285	0.79684	0.00082	0.79848
9	280	0.80455	0.00081	0.80616
9	280	0.80812	0.00079	0.80970
13	276	0.81573	0.00083	0.81739
24	265	0.84187	0.00080	0.84347
25	264	0.84017	0.00083	0.84182
25	264	0.84634	0.00081	0.84795
25	264	0.84583	0.00083	0.84750
25	264	0.85524	0.00083	0.85690
25	264	0.83396	0.00081	0.83558
25	264	0.84625	0.00083	0.84790
27	262	0.85438	0.00083	0.85604
29	260	0.85179	0.00081	0.85340
31	258	0.85930	0.00084	0.86098
33	256	0.86407	0.00082	0.86571
35	254	0.86740	0.00082	0.86904
37	252	0.87372	0.00084	0.87541
45	244	0.88630	0.00081	0.88793
45	244	0.87687	0.00079	0.87844
52	237	0.90062	0.00083	0.90228
57	232	0.87975	0.00087	0.88149
61	258	0.89055	0.00083	0.89221
73	216	0.90967	0.00082	0.91131
84	205	0.93261	0.00091	0.93443
85	204	0.94326	0.00086	0.94499
113	176	0.95626	0.00084	0.95794
117	172	0.95373	0.00088	0.95549
119	170	0.95315	0.00085	0.95485
125	164	0.95020	0.00086	0.95192
141	148	0.94348	0.00086	0.94521
145	144	0.93868	0.00089	0.94047
113 (Four Corners)	176	0.91292	0.00087	0.91466
Design Basis Westinghouse 17 × 17 OFA		0.9192	0.0009	0.9210

Table 6.6.1-9 Fuel Can Infinite Height Model Results of Fuel-Water Mixture Between Rods

Volume Fraction of UO_2 in Water	k_{eff}	Δk_{eff} to 24 (Four Corners)[1]
0.000	0.91090	-0.00063
0.001	0.91138	-0.00015
0.002	0.91120	-0.00033
0.003	0.91177	0.00024
0.004	0.91285	0.00132
0.005	0.90908	-0.00245
0.006	0.91001	-0.00152
0.007	0.90895	-0.00258
0.008	0.91005	-0.00148
0.009	0.90986	-0.00167
0.010	0.90864	-0.00289
0.020	0.91003	-0.00150
0.030	0.90963	-0.00190
0.040	0.91063	-0.00090
0.050	0.90931	-0.00222
0.060	0.90765	-0.00388
0.070	0.90753	-0.00400
0.080	0.91088	-0.00065
0.090	0.91122	-0.00031
0.100	0.90879	-0.00274
0.150	0.90968	-0.00185
0.200	0.90952	-0.00201
0.250	0.90815	-0.00338
0.300	0.90748	-0.00405
0.350	0.90581	-0.00572
0.400	0.90963	-0.00190
0.450	0.90547	-0.00606
0.500	0.90603	-0.00550
0.550	0.90753	-0.00400
0.600	0.90674	-0.00479
0.650	0.90589	-0.00564
0.700	0.90594	-0.00559
0.750	0.90568	-0.00585
0.800	0.90532	-0.00621
0.850	0.90693	-0.00460
0.900	0.90639	-0.00514
0.950	0.90684	-0.00469
1.000	0.90677	-0.00476

Table 6.6.1-10 Fuel Can Finite Model Results of Fuel-Water Mixture Outside Neutron Absorber Coverage

Volume Fraction of UO_2 in Water	k_{eff}	Δk_{eff} to 0.00 UO_2 in Water	Δk_{eff} to 24 (Four Corners)[1]
0.00	0.91045[2]	NA	-0.00108
0.05	0.90781	-0.00264	-0.00372
0.10	0.90978	-0.00067	-0.00175
0.15	0.91048	0.00003	-0.00105
0.20	0.90916	-0.00129	-0.00237
0.25	0.90834	-0.00211	-0.00319
0.30	0.90935	-0.00110	-0.00218
0.35	0.90786	-0.00259	-0.00367
0.40	0.90892	-0.00153	-0.00261
0.45	0.91015	-0.00030	-0.00138
0.50	0.91011	-0.00034	-0.00142
0.55	0.91003	-0.00042	-0.00150
0.60	0.90874	-0.00171	-0.00279
0.65	0.91165	0.00120	0.00012
0.70	0.90977	-0.00068	-0.00176
0.75	0.90813	-0.00232	-0.00340
0.80	0.90909	-0.00136	-0.00244
0.85	0.91028	-0.00017	-0.00125
0.90	0.91061	0.00016	-0.00092
0.95	0.91129	0.00084	-0.00024
1.00	0.91076	0.00031	-0.00077

1. See Table 6.6.1-6.
2. $\sigma = 0.00084$.

Table 6.6.1-11 Fuel Can Finite Model Results of Replacing All Rods with Fuel-Water Mixture

Volume Fraction of UO_2 in Water	k_{eff}	Δk_{eff} to 24 (Four Corners) Finite Height Model[1]	Δk_{eff} to 24 (Four Corners) Infinite Height Model[2]
0	0.90071	-0.00974	-0.01082
5	0.90194	-0.00851	-0.00959
10	0.90584	-0.00461	-0.00569
15	0.90837	-0.00208	-0.00316
20	0.91008	-0.00037	-0.00145
25	0.91086	0.00041	-0.00067
30	0.90964	-0.00081	-0.00189
35	0.90828	-0.00217	-0.00325
40	0.90805	-0.00240	-0.00348
45	0.90730	-0.00315	-0.00423
50	0.90637	-0.00408	-0.00516
55	0.90672	-0.00373	-0.00481
60	0.90649	-0.00396	-0.00504
65	0.90632	-0.00413	-0.00521
70	0.90435	-0.00610	-0.00718
75	0.90792	-0.00253	-0.00361
80	0.90376	-0.00669	-0.00777
85	0.90528	-0.00517	-0.00625
90	0.90454	-0.00591	-0.00699
95	0.90360	-0.00685	-0.00793
100	0.90416	-0.00629	-0.00737

1. The k_{eff} comparison basis for this column is the finite height model with the four corner locations of the basket loaded with Maine Yankee assemblies in the most reactive missing rod geometry. This case is the first case presented in Table 6.6.1-10 with 0% UO_2 in the water above and below the active fuel of the missing rod array.
2. The k_{eff} comparison basis for this column is the infinite height model with the four corner locations of the basket loaded with Maine Yankee assemblies in the most reactive missing rod geometry, the case presented in Table 6.6.1-6 labeled “24 (Four Corners)”, k_{eff} = 0.91153.

Table 6.6.1-12 Infinite Height Analysis of Maine Yankee Start-up Sources

Pu Vf	Be Vf	H_2O Vf	Void Vf	k_{eff}	sd	k_{eff}+2sd	Delta K*
0	0.5	0.5	0	0.91085	0.00087	0.91259	-0.00068
0.008	0.992	0	0	0.91034	0.00089	0.91212	-0.00119
0.008	0.9	0.092	0	0.91151	0.00087	0.91325	-0.00002
0.008	0.8	0.192	0	0.91138	0.00087	0.91312	-0.00015
0.008	0.7	0.292	0	0.91042	0.00085	0.91212	-0.00111
0.008	0.6	0.392	0	0.91231	0.00086	0.91403	0.00078
0.008	0.5	0.492	0	0.90922	0.00083	0.91088	-0.00231
0.008	0.4	0.592	0	0.91197	0.00087	0.91371	0.00044
0.008	0.3	0.692	0	0.91203	0.00086	0.91375	0.00050
0.008	0.2	0.792	0	0.90922	0.00084	0.91090	-0.00231
0.008	0.1	0.892	0	0.91140	0.00085	0.91310	-0.00013
0.008	0	0.992	0	0.91149	0.00086	0.91321	-0.00004
0.008	0.9	0	0.092	0.91075	0.00087	0.91249	-0.00078
0.008	0.8	0	0.192	0.91143	0.00091	0.91325	-0.00010
0.008	0.7	0	0.292	0.91182	0.00086	0.91354	0.00029
0.008	0.6	0	0.392	0.91072	0.00082	0.91236	-0.00081
0.008	0.5	0	0.492	0.90984	0.00085	0.91154	-0.00169
0.008	0.4	0	0.592	0.90982	0.00091	0.91164	-0.00171
0.008	0.3	0	0.692	0.91055	0.00087	0.91229	-0.00098
0.008	0.2	0	0.792	0.91054	0.00085	0.91224	-0.00099
0.008	0.1	0	0.892	0.91006	0.00088	0.91182	-0.00147
0.008	0	0	0.992	0.90957	0.00086	0.91129	-0.00196

*Change in reactivity from case “24 (Four Corners)” in Table 6.6.1-6.

6.7 References

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19. Owen, D. B., “Factors for One-Sided Tolerance Limits and for Variables Sampling Plans,” SCR-607, 1963.

20. SERCO Assurance, “MONK, A Monte Carlo Program for Nuclear Criticality Safety and Reactor Physics Analysis, Version 8A.”

6.8 CSAS Inputs

The CSAS25 input files for the criticality analyses of the Universal Storage System standard transfer and concrete casks containing PWR or BWR fuel, under normal and accident conditions, are provided in Figures 6.8-1 through 6.8-8. A standard transfer cask PWR Westinghouse 17×17 OFA (we17b) input file containing soluble boron at 1000 ppm, with a fuel initial enrichment of 5.0 wt. % ^{235}U, is shown in Figure 6.8-9. A BWR standard transfer cask model input containing 56 Exxon/ANF 9×9 79-fuel rod assemblies (ex09c) at 4.4 wt. % ^{235}U is shown in Figure 6.8-10.

Figure 6.8-1 CSAS Input for Normal Conditions - Transfer Cask Containing PWR Fuel

```
=CSAS25
UMS PWR TFR; NORMAL OP; ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 250 CM PITCH
27GROUPNDF4 LATTICECELL
UO2  1  0.95 293.0 92235 4.20 92238 95.80 END
ZIRCALLOY   2  1.0  293.0  END
H2O  3  1.0  293.0  END
AL   4  1.0  293.0  END
SS304 5 1.0  293.0  END
AL    6 DEN=2.6000 0.4627 293.0 END
B-10  6 DEN=2.6000 0.0568 293.0 END
B-11  6 DEN=2.6000 0.3449 293.0 END
C     6 DEN=2.6000 0.1167 293.0 END
PB    7 1.0 293.O  END
B-10  8 0.0 8.553-5 293.0 END
B-11  8 0.0 3.422-4 293.0 END
AL    8 0.0 7.763-3 293.0 END
H     8 0.0 5.854-2 293.0 END
O     8 0.0 2.609-2 293.0 END
C     8 0.0 2.264-2 293.0 END
N     8 0.0 1.394-3 293.0 END
H2O   9 1.0 293.0 END
H2O  10 1.0 293.0 END
CARBONSTEEL 11 1.0 293.0 END
END COMP
SQUAREPITCH  1.2598 0.7844 1 3 0.9144 2 0.8001 0 END
UMS PWR TFR; NORMAL OP; ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 250 CM PITCH
READ PARAM RUN=YES PLT=NO TME=5000 GEN=203 NPG=1000 END PARAM
READ GEOM
UNIT  1
COM='FUEL PIN CELL - BETWEEN DISKS'
CYLINDER   1 1  0.3922   2P2.4892
CYLINDER   0 1  0.4001   2P2.4892
CYLINDER   2 1  0.4572   2P2.4892
CUBOID     3 1  4P0.6299 2P2.4892
UNIT  2
COM='WATER ROD CELL - BETWEEN DISKS'
CYLINDER  3 1  0.5715   2P2.4892
CYLINDER  2 1  0.6121   2P2.4892
CUBOID    3 1  4P0.6299 2P2.4892
UNIT  3
COM='FUEL PIN CELL - FOR DISK SLICE OF CASK'
CYLINDER   1 1  0.3922   2P0.6350
CYLINDER   0 1  0.4001   2P0.6350
CYLINDER   2 1  0.4572   2P0.6350
CUBOID     3 1  4P0.6299 2P0.6350
UNIT  4
COM='WATER ROD CELL - FOR DISK SLICE OF CASK'
CYLINDER  3 1  0.5715   2P0.6350
CYLINDER  2 1  0.6121   2P0.6350
CUBOID    3 1  4P0.6299 2P0.6350
UNIT 5
COM='X-X BORAL SHEET BETWEEN DISKS'
CUBOID 6 1 2P10.4140 2P0.0635 2P2.4892
CUBOID 4 1 2P10.4140 2P0.0951 2P2.4892
UNIT 6
COM='Y-Y BORAL SHEET BETWEEN DISKS'
CUBOID 6 1 2P0.0635 2P10.4140 2P2.4892
CUBOID 4 1 2P0.0951 2P10.4140 2P2.4892
UNIT 7
COM='X-X BORAL SHEET WITH DISKS'
CUBOID 6 1 2P10.4140 2P0.0635 2P0.6350
CUBOID 4 1 2P10.4140 2P0.0951 2P0.6350
UNIT 8
COM='Y-Y BORAL SHEET WITH DISKS'
CUBOID 6 1 2P0.0635 2P10.4140 2P0.6350
CUBOID 4 1 2P0.0951 2P10.4140 2P0.6350
UNIT 10
COM='TUBE CELL IN H2O BETWEEN DISKS (A)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P2.4892
UNIT 11
COM='TUBE CELL IN H2O BETWEEN DISKS (B)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
```

Figure 6.8-1 (continued)

```
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P2.4892
UNIT 12
COM='TUBE CELL IN H2O BETWEEN DISKS (C)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P2.4892
UNIT 13
COM='TUBE CELL IN H2O BETWEEN DISKS (D)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P2.4892
UNIT 14
COM='WEB UNIT (1.5" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.8725 2P2.4892
UNIT 15
COM='WEB UNIT (1.0" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.2510 2P2.4892
UNIT 16
COM='WEB UNIT (0.875" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.0923 2P2.4892
UNIT 17
COM='6X1 FUEL TUBE STACK BETWEEN DISKS (-X)'
ARRAY 10 -11.7946 -77.3262 -2.4892
UNIT 18
COM='6X1 FUEL TUBE STACK BETWEEN DISKS (+X)'
ARRAY 11 -11.7946 -77.3262 -2.4892
UNIT 19
COM='2X1 FUEL TUBE STACK OF TUBES BETWEEN DISKS (-X)'
ARRAY 12 -11.7946 -25.4616 -2.4892
UNIT 20
COM='2X1 FUEL TUBE STACK OF TUBES BETWEEN DISKS (+X)'
ARRAY 13 -11.7946 -25.4616 -2.4892
UNIT 30
COM='TUBE CELL IN ST DISK (A)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P0.6350
UNIT 31
COM='TUBE CELL IN ST DISK (B)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P0.6350
UNIT 32
COM='TUBE CELL IN ST DISK (C)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P0.6350
UNIT 33
COM='TUBE CELL IN ST DISK (D)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P0.6350
UNIT 34
COM='WEB UNIT (1.5" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.8725 2P0.6350
```

Figure 6.8-1 (continued)

```
UNIT 35
COM='WEB UNIT (1.0" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.2510 2P0.6350
UNIT 36
COM='WEB UNIT (0.875" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.0923 2P0.6350
UNIT 37
COM='6x1 FUEL TUBE STACK ST DISK (-X)'
ARRAY 20 -11.7946 -77.3262 -0.6350
UNIT 38
COM='6x1 FUEL TUBE STACK ST DISK (+X)'
ARRAY 21 -11.7946 -77.3262 -0.6350
UNIT 39
COM='2X1 FUEL TUB STACK OF TUBES ST DISK (-X)'
ARRAY 22 -11.7946 -25.4616 -0.6350
UNIT 40
COM='2X1 FUEL TUB STACK OF TUBES ST DISK (+X)'
ARRAY 23 -11.7946 -25.4616 -0.6350
UNIT 50
COM='TUBE CELL IN AL DISK (A)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P0.6350
UNIT 51
COM='TUBE CELL IN AL DISK (B)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P0.6350
UNIT 52
COM='TUBE CELL IN AL DISK (C)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P0.6350
UNIT 53
COM='TUBE CELL IN AL DISK (D)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P0.6350
UNIT 54
COM='WEB UNIT (1.5" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.8725 2P0.6350
UNIT 55
COM='WEB UNIT (1.0" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.2510 2P0.6350
UNIT 56
COM='WEB UNIT (0.875" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.0923 2P0.6350
UNIT 57
COM='6X1 FUEL TUBE STACK AL DISK'
ARRAY 30 -11.7946 -77.3262 -0.6350
UNIT 58
COM='6X1 FUEL TUBE STACK AL DISK'
ARRAY 31 -11.7946 -77.3262 -0.6350
UNIT 59
COM='2X1 FUEL TUBE STACK OF TUBES AL DISK'
ARRAY 32 -11.7946 -25.4616 -0.6350
UNIT 60
COM='2X1 FUEL TUBE STACK OF TUBES AL DISK'
ARRAY 33 -11.7946 -25.4616 -0.6350
UNIT 70
COM='BASKET STRUCTURE IN TRANSFER CASK - WATER DISK'
CYLINDER 3 1 +83.5787 2P2.4892
HOLE 17  -13.6669 0.0 0.0
HOLE 18  +13.6669 0.0 0.0
HOLE 19  -39.7578 0.0 0.0
HOLE 20  +39.7578 0.0 0.0
HOLE 19  -65.5312 0.0 0.0
HOLE 20  +65.5312 0.0 0.0
HOLE 10  +40.8048 +40.8048 0.0
HOLE 11  -40.8048 +40.8048 0.0
```

Figure 6.8-1 (continued)

```
HOLE 12  -40.8048 -40.8048 0.0
HOLE 13  +40.8048 -40.8048 0.0
CYLINDER  5 1  +85.1662 2P2.4892
CYLINDER   9 1  +86.0425 2P2.4892
CYLINDER  11 1  +87.9475 2P2.4892
CYLINDER   7 1  +97.4725 2P2.4892
CYLINDER   8 1 +102.5525 2P2.4892
CYLINDER  11 1 +105.7275 2P2.4892
CUBOID     9 1 4P125.0   2P2.4892
UNIT 71
COM='BASKET STRUCTURE IN TRANSFER CASK - ST DISK'
CYLINDER  5 1 +83.1850 2P0.6350
HOLE 37  -13.6669 0.0 0.0
HOLE 38  +13.6669 0.0 0.0
HOLE 39  -39.7578 0.0 0.0
HOLE 40  +39.7578 0.0 0.0
HOLE 39  -65.5312 0.0 0.0
HOLE 40  +65.5312 0.0 0.0
HOLE 30  +40.8048 +40.8048 0.0
HOLE 31  -40.8048 +40.8048 0.0
HOLE 32  -40.8048 -40.8048 0.0
HOLE 33  +40.8048 -40.8048 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER   9 1  +86.0425 2P0.6350
CYLINDER  11 1  +87.9475 2P0.6350
CYLINDER   7 1  +97.4725 2P0.6350
CYLINDER   8 1 +102.5525 2P0.6350
CYLINDER  11 1 +105.7275 2P0.6350
CUBOID     9 1 4P125.0   2P0.6350
UNIT 72
COM='BASKET STRUCTURE IN TRANSFER CASK - AL DISK'
CYLINDER  4 1 +82.8675 2P0.6350
HOLE 57  -13.6669 0.0 0.0
HOLE 58  +13.6669 0.0 0.0
HOLE 59  -39.7578 0.0 0.0
HOLE 60  +39.7578 0.0 0.0
HOLE 59  -65.5312 0.0 0.0
HOLE 60  +65.5312 0.0 0.0
HOLE 50  +40.8048 +40.8048 0.0
HOLE 51  -40.8048 +40.8048 0.0
HOLE 52  -40.8048 -40.8048 0.0
HOLE 53  +40.8048 -40.8048 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER   9 1  +86.0425 2P0.6350
CYLINDER  11 1  +87.9475 2P0.6350
CYLINDER   7 1  +97.4725 2P0.6350
CYLINDER   8 1 +102.5525 2P0.6350
CYLINDER  11 1 +105.7275 2P0.6350
CUBOID     9 1 4P125.0   2P0.6350
GLOBAL UNIT 73
COM='DISK SLICE STACK'
ARRAY  40 -125.0 -125.0 0.0
END GEOM
READ ARRAY
ARA=1   NUX=17 NUY=17 NUZ=1  FILL
                        34R1
          5R1   2   2R1   2   2R1   2   5R1
               3R1   2   9R1   2   3R1
                        17R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        34R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        34R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        17R1
               3R1   2   9R1   2   3R1
          5R1   2   2R1   2   2R1   2   5R1
                        34R1
END FILL
ARA=2  NUX=17 NUY=17 NUZ=1  FILL
                        34R3
          5R3   4   2R3   4   2R3   4   5R3
               3R3   4   9R3   4   3R3
                        17R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        34R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        34R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        17R3
               3R3   4   9R3   4   3R3
          5R3   4   2R3   4   2R3   4   5R3
                        34R3
END FILL
ARA=10 NUX=1 NUY=11 NUZ=1 FILL 12 16 12 15 12 14 11 15 11 16 11 END FILL
ARA=11 NUX=1 NUY=11 NUZ=1 FILL 13 16 13 15 13 14 10 15 10 16 10 END FILL
ARA=12 NUX=1 NUY=3  NUZ=1 FILL 12 14 11 END FILL
ARA=13 NUX=1 NUY=3  NUZ=1 FILL 13 14 10 END FILL
ARA=20 NUX=1 NUY=11 NUZ=1 FILL 32 36 32 35 32 34 31 35 31 36 31 END FILL
ARA=21 NUX=1 NUY=11 NUZ=1 FILL 33 36 33 35 33 34 30 35 30 36 30 END FILL
ARA=22 NUX=1 NUY=3  NUZ=1 FILL 32 34 31 END FILL
ARA=23 NUX=1 NUY=3  NUZ=1 FILL 33 34 30 END FILL
ARA=30 NUX=1 NUY=11 NUZ=1 FILL 52 56 52 55 52 54 51 55 51 56 51 END FILL
ARA=31 NUX=1 NUY=11 NUZ=1 FILL 53 56 53 55 53 54 50 55 50 56 50 END FILL
ARA=32 NUX=1 NUY=3  NUZ=1 FILL 52 54 51 END FILL
ARA=33 NUX=1 NUY=3  NUZ=1 FILL 53 54 50 END FILL
```

Figure 6.8-1 (continued)

```
  ARA=40 NUX=1 NUY=1 NUZ=4 FILL 70 71 70 72 END FILL
  END ARRAY
  READ BOUNDS ZFC=PER YXF=MIRROR END BOUNDS
  END DATA
  END

SECONDARY MODULE 000008   HAS BEEN CALLED.
```

Figure 6.8-2 CSAS Input for Accident Conditions– Transfer Cask Containing PWR Fuel

```
=CSAS25
UMS PWR TFR; ACCIDENT; ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 250 CM PITCH
27GROUPNDF4 LATTICECELL
UO2  1  0.95 293.0 92235 4.20 92238 95.80 END
ZIRCALLOY   2  1.0  293.0  END
H2O  3  1.0  293.0  END
AL   4  1.0  293.0  END
SS304 5 1.0  293.0  END
AL    6 DEN=2.6000 0.4627 293.0 END
B-10  6 DEN=2.6000 0.0568 293.0 END
B-11  6 DEN=2.6000 0.3449 293.0 END
C     6 DEN=2.6000 0.1167 293.0 END
PB    7 1.0 293.O  END
B-10  8 0.0 8.553-5 293.0 END
B-11  8 0.0 3.422-4 293.0 END
AL    8 0.0 7.763-3 293.0 END
H     8 0.0 5.854-2 293.0 END
O     8 0.0 2.609-2 293.0 END
C     8 0.0 2.264-2 293.0 END
N     8 0.0 1.394-3 293.0 END
H2O   9 1.0 293.0 END
H2O  10 1.0 293.0 END
CARBONSTEEL 11 1.0 293.0 END
END COMP
SQUAREPITCH  1.2598 0.7844 1 3 0.9144 2 0.8001 10 END
UMS PWR TFR; ACCIDENT; ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 250 CM PITCH
READ PARAM RUN=YES PLT=NO TME=5000 GEN=803 NPG=1000 END PARAM
READ GEOM
UNIT  1
COM='FUEL PIN CELL - BETWEEN DISKS'
CYLINDER   1 1  0.3922   2P2.4892
CYLINDER  10 1  0.4001   2P2.4892
CYLINDER   2 1  0.4572   2P2.4892
CUBOID     3 1  4P0.6299 2P2.4892
UNIT  2
COM='WATER ROD CELL - BETWEEN DISKS'
CYLINDER  3 1  0.5715   2P2.4892
CYLINDER  2 1  0.6121   2P2.4892
CUBOID    3 1  4P0.6299 2P2.4892
UNIT  3
COM='FUEL PIN CELL - FOR DISK SLICE OF CASK'
CYLINDER   1 1  0.3922   2P0.6350
CYLINDER  10 1  0.4001   2P0.6350
CYLINDER   2 1  0.4572   2P0.6350
CUBOID     3 1  4P0.6299 2P0.6350
UNIT  4
COM='WATER ROD CELL - FOR DISK SLICE OF CASK'
CYLINDER  3 1  0.5715   2P0.6350
CYLINDER  2 1  0.6121   2P0.6350
CUBOID    3 1  4P0.6299 2P0.6350
UNIT 5
COM='X-X BORAL SHEET BETWEEN DISKS'
CUBOID 6 1 2P10.4140 2P0.0635 2P2.4892
CUBOID 4 1 2P10.4140 2P0.0951 2P2.4892
UNIT 6
COM='Y-Y BORAL SHEET BETWEEN DISKS'
CUBOID 6 1 2P0.0635 2P10.4140 2P2.4892
CUBOID 4 1 2P0.0951 2P10.4140 2P2.4892
UNIT 7
COM='X-X BORAL SHEET WITH DISKS'
CUBOID 6 1 2P10.4140 2P0.0635 2P0.6350
CUBOID 4 1 2P10.4140 2P0.0951 2P0.6350
UNIT 8
COM='Y-Y BORAL SHEET WITH DISKS'
CUBOID 6 1 2P0.0635 2P10.4140 2P0.6350
CUBOID 4 1 2P0.0951 2P10.4140 2P0.6350
UNIT 10
COM='TUBE CELL IN H2O BETWEEN DISKS (A)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P2.4892
UNIT 11
COM='TUBE CELL IN H2O BETWEEN DISKS (B)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
```

Figure 6.8-2 (continued)

```
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P2.4892
UNIT 12
COM='TUBE CELL IN H2O BETWEEN DISKS (C)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P2.4892
UNIT 13
COM='TUBE CELL IN H2O BETWEEN DISKS (D)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P2.4892
UNIT 14
COM='WEB UNIT (1.5" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.8725 2P2.4892
UNIT 15
COM='WEB UNIT (1.0" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.2510 2P2.4892
UNIT 16
COM='WEB UNIT (0.875" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.0923 2P2.4892
UNIT 17
COM='6X1 FUEL TUBE STACK BETWEEN DISKS (-X)'
ARRAY 10 -11.7946 -77.3262 -2.4892
UNIT 18
COM='6X1 FUEL TUBE STACK BETWEEN DISKS (+X)'
ARRAY 11 -11.7946 -77.3262 -2.4892
UNIT 19
COM='2X1 FUEL TUBE STACK OF TUBES BETWEEN DISKS (-X)'
ARRAY 12 -11.7946 -25.4616 -2.4892
UNIT 20
COM='2X1 FUEL TUBE STACK OF TUBES BETWEEN DISKS (+X)'
ARRAY 13 -11.7946 -25.4616 -2.4892
UNIT 30
COM='TUBE CELL IN ST DISK (A)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P0.6350
UNIT 31
COM='TUBE CELL IN ST DISK (B)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P0.6350
UNIT 32
COM='TUBE CELL IN ST DISK (C)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P0.6350
UNIT 33
COM='TUBE CELL IN ST DISK (D)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P0.6350
UNIT 34
COM='WEB UNIT (1.5" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.8725 2P0.6350
UNIT 35
```

Figure 6.8-2 (continued)

```
COM='WEB UNIT (1.0" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.2510 2P0.6350
UNIT 36
COM='WEB UNIT (0.875" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.0923 2P0.6350
UNIT 37
COM='6x1 FUEL TUBE STACK ST DISK (-X)'
ARRAY 20 -11.7946 -77.3262 -0.6350
UNIT 38
COM='6x1 FUEL TUBE STACK ST DISK (+X)'
ARRAY 21 -11.7946 -77.3262 -0.6350
UNIT 39
COM='2X1 FUEL TUB STACK OF TUBES ST DISK (-X)'
ARRAY 22 -11.7946 -25.4616 -0.6350
UNIT 40
COM='2X1 FUEL TUB STACK OF TUBES ST DISK (+X)'
ARRAY 23 -11.7946 -25.4616 -0.6350
UNIT 50
COM='TUBE CELL IN AL DISK (A)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P0.6350
UNIT 51
COM='TUBE CELL IN AL DISK (B)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P0.6350
UNIT 52
COM='TUBE CELL IN AL DISK (C)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P0.6350
UNIT 53
COM='TUBE CELL IN AL DISK (D)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P0.6350
UNIT 54
COM='WEB UNIT (1.5" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.8725 2P0.6350
UNIT 55
COM='WEB UNIT (1.0" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.2510 2P0.6350
UNIT 56
COM='WEB UNIT (0.875" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.0923 2P0.6350
UNIT 57
COM='6X1 FUEL TUBE STACK AL DISK'
ARRAY 30 -11.7946 -77.3262 -0.6350
UNIT 58
COM='6X1 FUEL TUBE STACK AL DISK'
ARRAY 31 -11.7946 -77.3262 -0.6350
UNIT 59
COM='2X1 FUEL TUBE STACK OF TUBES AL DISK'
ARRAY 32 -11.7946 -25.4616 -0.6350
UNIT 60
COM='2X1 FUEL TUBE STACK OF TUBES AL DISK'
ARRAY 33 -11.7946 -25.4616 -0.6350
UNIT 70
COM='BASKET STRUCTURE IN TRANSFER CASK - WATER DISK'
CYLINDER 3 1 +83.5787 2P2.4892
HOLE 17  -13.6669 0.0 0.0
HOLE 18  +13.6669 0.0 0.0
HOLE 19  -39.7578 0.0 0.0
HOLE 20  +39.7578 0.0 0.0
HOLE 19  -65.5312 0.0 0.0
HOLE 20  +65.5312 0.0 0.0
HOLE 10  +40.8048 +40.8048 0.0
HOLE 11  -40.8048 +40.8048 0.0
HOLE 12  -40.8048 -40.8048 0.0
```

Figure 6.8-2 (continued)

```
HOLE 13  +40.8048 -40.8048 0.0
CYLINDER  5 1  +85.1662 2P2.4892
CYLINDER   9 1  +86.0425 2P2.4892
CYLINDER  11 1  +87.9475 2P2.4892
CYLINDER   7 1  +97.4725 2P2.4892
CYLINDER   8 1 +102.5525 2P2.4892
CYLINDER  11 1 +105.7275 2P2.4892
CUBOID     9 1 4P125.0   2P2.4892
UNIT 71
COM='BASKET STRUCTURE IN TRANSFER CASK - ST DISK'
CYLINDER  5 1 +83.1850 2P0.6350
HOLE 37  -13.6669 0.0 0.0
HOLE 38  +13.6669 0.0 0.0
HOLE 39  -39.7578 0.0 0.0
HOLE 40  +39.7578 0.0 0.0
HOLE 39  -65.5312 0.0 0.0
HOLE 40  +65.5312 0.0 0.0
HOLE 30  +40.8048 +40.8048 0.0
HOLE 31  -40.8048 +40.8048 0.0
HOLE 32  -40.8048 -40.8048 0.0
HOLE 33  +40.8048 -40.8048 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER   9 1  +86.0425 2P0.6350
CYLINDER  11 1  +87.9475 2P0.6350
CYLINDER   7 1  +97.4725 2P0.6350
CYLINDER   8 1 +102.5525 2P0.6350
CYLINDER  11 1 +105.7275 2P0.6350
CUBOID     9 1 4P125.0   2P0.6350
UNIT 72
COM='BASKET STRUCTURE IN TRANSFER CASK - AL DISK'
CYLINDER  4 1 +82.8675 2P0.6350
HOLE 57  -13.6669 0.0 0.0
HOLE 58  +13.6669 0.0 0.0
HOLE 59  -39.7578 0.0 0.0
HOLE 60  +39.7578 0.0 0.0
HOLE 59  -65.5312 0.0 0.0
HOLE 60  +65.5312 0.0 0.0
HOLE 50  +40.8048 +40.8048 0.0
HOLE 51  -40.8048 +40.8048 0.0
HOLE 52  -40.8048 -40.8048 0.0
HOLE 53  +40.8048 -40.8048 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER   9 1  +86.0425 2P0.6350
CYLINDER  11 1  +87.9475 2P0.6350
CYLINDER   7 1  +97.4725 2P0.6350
CYLINDER   8 1 +102.5525 2P0.6350
CYLINDER  11 1 +105.7275 2P0.6350
CUBOID     9 1 4P125.0   2P0.6350
GLOBAL UNIT 73
COM='DISK SLICE STACK'
ARRAY  40 -125.0 -125.0 0.0
END GEOM
READ ARRAY
ARA=1   NUX=17 NUY=17 NUZ=1  FILL
                        34R1
          5R1   2   2R1   2   2R1   2   5R1
               3R1   2   9R1   2   3R1
                        17R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        34R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        34R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        17R1
               3R1   2   9R1   2   3R1
          5R1   2   2R1   2   2R1   2   5R1
                        34R1
END FILL
ARA=2  NUX=17 NUY=17 NUZ=1  FILL
                        34R3
          5R3   4   2R3   4   2R3   4   5R3
               3R3   4   9R3   4   3R3
                        17R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        34R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        34R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        17R3
               3R3   4   9R3   4   3R3
          5R3   4   2R3   4   2R3   4   5R3
                        34R3
END FILL
ARA=10 NUX=1 NUY=11 NUZ=1 FILL 12 16 12 15 12 14 11 15 11 16 11 END FILL
ARA=11 NUX=1 NUY=11 NUZ=1 FILL 13 16 13 15 13 14 10 15 10 16 10 END FILL
ARA=12 NUX=1 NUY=3  NUZ=1 FILL 12 14 11 END FILL
ARA=13 NUX=1 NUY=3  NUZ=1 FILL 13 14 10 END FILL
ARA=20 NUX=1 NUY=11 NUZ=1 FILL 32 36 32 35 32 34 31 35 31 36 31 END FILL
ARA=21 NUX=1 NUY=11 NUZ=1 FILL 33 36 33 35 33 34 30 35 30 36 30 END FILL
ARA=22 NUX=1 NUY=3  NUZ=1 FILL 32 34 31 END FILL
ARA=23 NUX=1 NUY=3  NUZ=1 FILL 33 34 30 END FILL
ARA=30 NUX=1 NUY=11 NUZ=1 FILL 52 56 52 55 52 54 51 55 51 56 51 END FILL
ARA=31 NUX=1 NUY=11 NUZ=1 FILL 53 56 53 55 53 54 50 55 50 56 50 END FILL
ARA=32 NUX=1 NUY=3  NUZ=1 FILL 52 54 51 END FILL
ARA=33 NUX=1 NUY=3  NUZ=1 FILL 53 54 50 END FILL
ARA=40 NUX=1 NUY=1 NUZ=4 FILL 70 71 70 72 END FILL
```

Figure 6.8-2 (continued)

```
END ARRAY
READ BOUNDS ZFC=PER YXF=MIRROR END BOUNDS
END DATA
END
```

Figure 6.8-3 CSAS Input for Normal Conditions–Vertical Concrete Cask Containing PWR Fuel

```
=CSAS25
UMS PWR SC; NORMAL OP; ARRAY; 0.0001 GM/CC IN - 0.0001 GM/CC EX; 460 CM PITCH
27GROUPNDF4 LATTICECELL
UO2  1  0.95 293.0 92235 4.20 92238 95.80 END
ZIRCALLOY   2  1.0  293.0  END
H2O  3  0.0001  293.0  END
AL   4  1.0  293.0  END
SS304 5 1.0  293.0  END
AL    6 DEN=2.6000 0.4627 293.0 END
B-10  6 DEN=2.6000 0.0568 293.0 END
B-11  6 DEN=2.6000 0.3449 293.0 END
C     6 DEN=2.6000 0.1167 293.0 END
CARBONSTEEL  7 1.0    293.O  END
REG-CONCRETE 8 0.9750 293.0 END
H2O   9 0.0001 293.0 END
H2O  10 0.0001 293.0 END
END COMP
SQUAREPITCH  1.2598 0.7844 1 3 0.9144 2 0.8001 0 END
UMS PWR SC; NORMAL OP; ARRAY; 0.0001 GM/CC IN - 0.0001 GM/CC EX; 460 CM PITCH
READ PARAM RUN=YES PLT=NO TME=5000 GEN=203 NPG=1000 END PARAM
READ GEOM
UNIT  1
COM='FUEL PIN CELL - BETWEEN DISKS'
CYLINDER   1 1  0.3922   2P2.4892
CYLINDER   0 1  0.4001   2P2.4892
CYLINDER   2 1  0.4572   2P2.4892
CUBOID     3 1  4P0.6299 2P2.4892
UNIT  2
COM='WATER ROD CELL - BETWEEN DISKS'
CYLINDER  3 1  0.5715   2P2.4892
CYLINDER  2 1  0.6121   2P2.4892
CUBOID    3 1  4P0.6299 2P2.4892
UNIT  3
COM='FUEL PIN CELL - FOR DISK SLICE OF CASK'
CYLINDER   1 1  0.3922   2P0.6350
CYLINDER   0 1  0.4001   2P0.6350
CYLINDER   2 1  0.4572   2P0.6350
CUBOID     3 1  4P0.6299 2P0.6350
UNIT  4
COM='WATER ROD CELL - FOR DISK SLICE OF CASK'
CYLINDER  3 1  0.5715   2P0.6350
CYLINDER  2 1  0.6121   2P0.6350
CUBOID    3 1  4P0.6299 2P0.6350
UNIT 5
COM='X-X BORAL SHEET BETWEEN DISKS'
CUBOID 6 1 2P10.4140 2P0.0635 2P2.4892
CUBOID 4 1 2P10.4140 2P0.0951 2P2.4892
UNIT 6
COM='Y-Y BORAL SHEET BETWEEN DISKS'
CUBOID 6 1 2P0.0635 2P10.4140 2P2.4892
CUBOID 4 1 2P0.0951 2P10.4140 2P2.4892
UNIT 7
COM='X-X BORAL SHEET WITH DISKS'
CUBOID 6 1 2P10.4140 2P0.0635 2P0.6350
CUBOID 4 1 2P10.4140 2P0.0951 2P0.6350
UNIT 8
COM='Y-Y BORAL SHEET WITH DISKS'
CUBOID 6 1 2P0.0635 2P10.4140 2P0.6350
CUBOID 4 1 2P0.0951 2P10.4140 2P0.6350
UNIT 10
COM='TUBE CELL IN H2O BETWEEN DISKS (A)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P2.4892
UNIT 11
COM='TUBE CELL IN H2O BETWEEN DISKS (B)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P2.4892
UNIT 12
COM='TUBE CELL IN H2O BETWEEN DISKS (C)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P2.4892
```

Figure 6.8-3 (continued)

```
UNIT 13
COM='TUBE CELL IN H2O BETWEEN DISKS (D)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P2.4892
UNIT 14
COM='WEB UNIT (1.5" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.8725 2P2.4892
UNIT 15
COM='WEB UNIT (1.0" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.2510 2P2.4892
UNIT 16
COM='WEB UNIT (0.875" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.0923 2P2.4892
UNIT 17
COM='6X1 FUEL TUBE STACK BETWEEN DISKS (-X)'
ARRAY 10 -11.7946 -77.3262 -2.4892
UNIT 18
COM='6X1 FUEL TUBE STACK BETWEEN DISKS (+X)'
ARRAY 11 -11.7946 -77.3262 -2.4892
UNIT 19
COM='2X1 FUEL TUBE STACK OF TUBES BETWEEN DISKS (-X)'
ARRAY 12 -11.7946 -25.4616 -2.4892
UNIT 20
COM='2X1 FUEL TUBE STACK OF TUBES BETWEEN DISKS (+X)'
ARRAY 13 -11.7946 -25.4616 -2.4892
UNIT 30
COM='TUBE CELL IN ST DISK (A)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P0.6350
UNIT 31
COM='TUBE CELL IN ST DISK (B)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P0.6350
UNIT 32
COM='TUBE CELL IN ST DISK (C)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P0.6350
UNIT 33
COM='TUBE CELL IN ST DISK (D)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P0.6350
UNIT 34
COM='WEB UNIT (1.5" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.8725 2P0.6350
UNIT 35
COM='WEB UNIT (1.0" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.2510 2P0.6350
UNIT 36
COM='WEB UNIT (0.875" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.0923 2P0.6350
UNIT 37
COM='6x1 FUEL TUBE STACK ST DISK (-X)'
ARRAY 20 -11.7946 -77.3262 -0.6350
UNIT 38
COM='6x1 FUEL TUBE STACK ST DISK (+X)'
ARRAY 21 -11.7946 -77.3262 -0.6350
UNIT 39
```

Figure 6.8-3 (continued)

```
COM='2X1 FUEL TUB STACK OF TUBES ST DISK (-X)'
ARRAY 22 -11.7946 -25.4616 -0.6350
UNIT 40
COM='2X1 FUEL TUB STACK OF TUBES ST DISK (+X)'
ARRAY 23 -11.7946 -25.4616 -0.6350
UNIT 50
COM='TUBE CELL IN AL DISK (A)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P0.6350
UNIT 51
COM='TUBE CELL IN AL DISK (B)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P0.6350
UNIT 52
COM='TUBE CELL IN AL DISK (C)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P0.6350
UNIT 53
COM='TUBE CELL IN AL DISK (D)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P0.6350
UNIT 54
COM='WEB UNIT (1.5" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.8725 2P0.6350
UNIT 55
COM='WEB UNIT (1.0" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.2510 2P0.6350
UNIT 56
COM='WEB UNIT (0.875" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.0923 2P0.6350
UNIT 57
COM='6X1 FUEL TUBE STACK AL DISK'
ARRAY 30 -11.7946 -77.3262 -0.6350
UNIT 58
COM='6X1 FUEL TUBE STACK AL DISK'
ARRAY 31 -11.7946 -77.3262 -0.6350
UNIT 59
COM='2X1 FUEL TUBE STACK OF TUBES AL DISK'
ARRAY 32 -11.7946 -25.4616 -0.6350
UNIT 60
COM='2X1 FUEL TUBE STACK OF TUBES AL DISK'
ARRAY 33 -11.7946 -25.4616 -0.6350
UNIT 70
COM='BASKET STRUCTURE IN STORAGE CASK - WATER DISK'
CYLINDER 3 1 +83.5787 2P2.4892
HOLE 17  -13.6669 0.0 0.0
HOLE 18  +13.6669 0.0 0.0
HOLE 19  -39.7578 0.0 0.0
HOLE 20  +39.7578 0.0 0.0
HOLE 19  -65.5312 0.0 0.0
HOLE 20  +65.5312 0.0 0.0
HOLE 10  +40.8048 +40.8048 0.0
HOLE 11  -40.8048 +40.8048 0.0
HOLE 12  -40.8048 -40.8048 0.0
HOLE 13  +40.8048 -40.8048 0.0
CYLINDER  5 1  +85.1662 2P2.4892
CYLINDER  9 1  +94.615 2P2.4892
CYLINDER  7 1 +100.965 2P2.4892
CYLINDER  8 1 +172.72 2P2.4892
CUBOID    9 1 4P230.0 2P2.4892
UNIT 71
COM='BASKET STRUCTURE IN STORAGE CASK - ST DISK'
CYLINDER  5 1 +83.1850 2P0.6350
HOLE 37  -13.6669 0.0 0.0
HOLE 38  +13.6669 0.0 0.0
HOLE 39  -39.7578 0.0 0.0
```

Figure 6.8-3 (continued)

```
HOLE 40  +39.7578 0.0 0.0
HOLE 39  -65.5312 0.0 0.0
HOLE 40  +65.5312 0.0 0.0
HOLE 30  +40.8048 +40.8048 0.0
HOLE 31  -40.8048 +40.8048 0.0
HOLE 32  -40.8048 -40.8048 0.0
HOLE 33  +40.8048 -40.8048 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER  9 1  +94.615 2P0.6350
CYLINDER  7 1 +100.965 2P0.6350
CYLINDER  8 1 +172.72 2P0.6350
CUBOID    9 1 4P230.0 2P0.6350
UNIT 72
COM='BASKET STRUCTURE IN STORAGE CASK - AL DISK'
CYLINDER  4 1 +82.8675 2P0.6350
HOLE 57  -13.6669 0.0 0.0
HOLE 58  +13.6669 0.0 0.0
HOLE 59  -39.7578 0.0 0.0
HOLE 60  +39.7578 0.0 0.0
HOLE 59  -65.5312 0.0 0.0
HOLE 60  +65.5312 0.0 0.0
HOLE 50  +40.8048 +40.8048 0.0
HOLE 51  -40.8048 +40.8048 0.0
HOLE 52  -40.8048 -40.8048 0.0
HOLE 53  +40.8048 -40.8048 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER  9 1  +94.615 2P0.6350
CYLINDER  7 1 +100.965 2P0.6350
CYLINDER  8 1 +172.72 2P0.6350
CUBOID    9 1 4P230.0 2P0.6350
GLOBAL UNIT 73
COM='DISK SLICE STACK'
ARRAY  40 -230.0 -230.0 0.0
END GEOM
READ ARRAY
ARA=1   NUX=17 NUY=17 NUZ=1  FILL
                        34R1
          5R1   2   2R1   2   2R1   2   5R1
               3R1   2   9R1   2   3R1
                        17R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        34R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        34R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        17R1
               3R1   2   9R1   2   3R1
          5R1   2   2R1   2   2R1   2   5R1
                        34R1
END FILL
ARA=2  NUX=17 NUY=17 NUZ=1  FILL
                        34R3
          5R3   4   2R3   4   2R3   4   5R3
               3R3   4   9R3   4   3R3
                        17R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        34R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        34R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        17R3
               3R3   4   9R3   4   3R3
          5R3   4   2R3   4   2R3   4   5R3
                        34R3
END FILL
ARA=10 NUX=1 NUY=11 NUZ=1 FILL 12 16 12 15 12 14 11 15 11 16 11 END FILL
ARA=11 NUX=1 NUY=11 NUZ=1 FILL 13 16 13 15 13 14 10 15 10 16 10 END FILL
ARA=12 NUX=1 NUY=3  NUZ=1 FILL 12 14 11 END FILL
ARA=13 NUX=1 NUY=3  NUZ=1 FILL 13 14 10 END FILL
ARA=20 NUX=1 NUY=11 NUZ=1 FILL 32 36 32 35 32 34 31 35 31 36 31 END FILL
ARA=21 NUX=1 NUY=11 NUZ=1 FILL 33 36 33 35 33 34 30 35 30 36 30 END FILL
ARA=22 NUX=1 NUY=3  NUZ=1 FILL 32 34 31 END FILL
ARA=23 NUX=1 NUY=3  NUZ=1 FILL 33 34 30 END FILL
ARA=30 NUX=1 NUY=11 NUZ=1 FILL 52 56 52 55 52 54 51 55 51 56 51 END FILL
ARA=31 NUX=1 NUY=11 NUZ=1 FILL 53 56 53 55 53 54 50 55 50 56 50 END FILL
ARA=32 NUX=1 NUY=3  NUZ=1 FILL 52 54 51 END FILL
ARA=33 NUX=1 NUY=3  NUZ=1 FILL 53 54 50 END FILL
ARA=40 NUX=1 NUY=1 NUZ=4 FILL 70 71 70 72 END FILL
END ARRAY
READ BOUNDS ZFC=PER YXF=MIRROR END BOUNDS
END DATA
END
```

Figure 6.8-4 CSAS Input for Accident Conditions– Vertical Concrete Cask Containing PWR Fuel

```
=CSAS25
UMS PWR SC; ACCIDENT; ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 460 CM PITCH
27GROUPNDF4 LATTICECELL
UO2  1  0.95 293.0 92235 4.20 92238 95.80 END
ZIRCALLOY   2  1.0  293.0  END
H2O  3  1.0  293.0  END
AL   4  1.0  293.0  END
SS304 5 1.0  293.0  END
AL    6 DEN=2.6000 0.4627 293.0 END
B-10  6 DEN=2.6000 0.0568 293.0 END
B-11  6 DEN=2.6000 0.3449 293.0 END
C     6 DEN=2.6000 0.1167 293.0 END
CARBONSTEEL  7 1.0    293.O  END
REG-CONCRETE 8 0.9750 293.0 END
H2O   9 1.0 293.0 END
H2O  10 1.0 293.0 END
END COMP
SQUAREPITCH  1.2598 0.7844 1 3 0.9144 2 0.8001 10 END
UMS PWR SC; ACCIDENT; ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 460 CM PITCH
READ PARAM RUN=YES PLT=NO TME=5000 GEN=803 NPG=1000 END PARAM
READ GEOM
UNIT  1
COM='FUEL PIN CELL - BETWEEN DISKS'
CYLINDER   1 1  0.3922   2P2.4892
CYLINDER  10 1  0.4001   2P2.4892
CYLINDER   2 1  0.4572   2P2.4892
CUBOID     3 1  4P0.6299 2P2.4892
UNIT  2
COM='WATER ROD CELL - BETWEEN DISKS'
CYLINDER  3 1  0.5715   2P2.4892
CYLINDER  2 1  0.6121   2P2.4892
CUBOID    3 1  4P0.6299 2P2.4892
UNIT  3
COM='FUEL PIN CELL - FOR DISK SLICE OF CASK'
CYLINDER   1 1  0.3922   2P0.6350
CYLINDER  10 1  0.4001   2P0.6350
CYLINDER   2 1  0.4572   2P0.6350
CUBOID     3 1  4P0.6299 2P0.6350
UNIT  4
COM='WATER ROD CELL - FOR DISK SLICE OF CASK'
CYLINDER  3 1  0.5715   2P0.6350
CYLINDER  2 1  0.6121   2P0.6350
CUBOID    3 1  4P0.6299 2P0.6350
UNIT 5
COM='X-X BORAL SHEET BETWEEN DISKS'
CUBOID 6 1 2P10.4140 2P0.0635 2P2.4892
CUBOID 4 1 2P10.4140 2P0.0951 2P2.4892
UNIT 6
COM='Y-Y BORAL SHEET BETWEEN DISKS'
CUBOID 6 1 2P0.0635 2P10.4140 2P2.4892
CUBOID 4 1 2P0.0951 2P10.4140 2P2.4892
UNIT 7
COM='X-X BORAL SHEET WITH DISKS'
CUBOID 6 1 2P10.4140 2P0.0635 2P0.6350
CUBOID 4 1 2P10.4140 2P0.0951 2P0.6350
UNIT 8
COM='Y-Y BORAL SHEET WITH DISKS'
CUBOID 6 1 2P0.0635 2P10.4140 2P0.6350
CUBOID 4 1 2P0.0951 2P10.4140 2P0.6350
UNIT 10
COM='TUBE CELL IN H2O BETWEEN DISKS (A)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P2.4892
UNIT 11
COM='TUBE CELL IN H2O BETWEEN DISKS (B)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P2.4892
UNIT 12
COM='TUBE CELL IN H2O BETWEEN DISKS (C)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
```

Figure 6.8-4 (continued)

```
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P2.4892
UNIT 13
COM='TUBE CELL IN H2O BETWEEN DISKS (D)'
ARRAY  1  -10.7083 -10.7083 -2.4892
CUBOID 3 1 4P11.2141 2P2.4892
CUBOID 5 1 4P11.3355 2P2.4892
CUBOID 3 1 4P11.5260 2P2.4892
HOLE  5 0.0 +11.4308 0.0
HOLE  5 0.0 -11.4308 0.0
HOLE  6 +11.4308 0.0 0.0
HOLE  6 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P2.4892
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P2.4892
UNIT 14
COM='WEB UNIT (1.5" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.8725 2P2.4892
UNIT 15
COM='WEB UNIT (1.0" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.2510 2P2.4892
UNIT 16
COM='WEB UNIT (0.875" WEB) - BETWEEN DISKS'
CUBOID 3 1  2P11.7946 2P1.0923 2P2.4892
UNIT 17
COM='6X1 FUEL TUBE STACK BETWEEN DISKS (-X)'
ARRAY 10 -11.7946 -77.3262 -2.4892
UNIT 18
COM='6X1 FUEL TUBE STACK BETWEEN DISKS (+X)'
ARRAY 11 -11.7946 -77.3262 -2.4892
UNIT 19
COM='2X1 FUEL TUBE STACK OF TUBES BETWEEN DISKS (-X)'
ARRAY 12 -11.7946 -25.4616 -2.4892
UNIT 20
COM='2X1 FUEL TUBE STACK OF TUBES BETWEEN DISKS (+X)'
ARRAY 13 -11.7946 -25.4616 -2.4892
UNIT 30
COM='TUBE CELL IN ST DISK (A)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P0.6350
UNIT 31
COM='TUBE CELL IN ST DISK (B)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P0.6350
UNIT 32
COM='TUBE CELL IN ST DISK (C)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P0.6350
UNIT 33
COM='TUBE CELL IN ST DISK (D)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P0.6350
UNIT 34
COM='WEB UNIT (1.5" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.8725 2P0.6350
UNIT 35
COM='WEB UNIT (1.0" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.2510 2P0.6350
UNIT 36
COM='WEB UNIT (0.875" WEB) - ST DISKS'
CUBOID 5 1  2P11.7946 2P1.0923 2P0.6350
UNIT 37
COM='6x1 FUEL TUBE STACK ST DISK (-X)'
ARRAY 20 -11.7946 -77.3262 -0.6350
```

Figure 6.8-4 (continued)

```
UNIT 38
COM='6x1 FUEL TUBE STACK ST DISK (+X)'
ARRAY 21 -11.7946 -77.3262 -0.6350
UNIT 39
COM='2X1 FUEL TUB STACK OF TUBES ST DISK (-X)'
ARRAY 22 -11.7946 -25.4616 -0.6350
UNIT 40
COM='2X1 FUEL TUB STACK OF TUBES ST DISK (+X)'
ARRAY 23 -11.7946 -25.4616 -0.6350
UNIT 50
COM='TUBE CELL IN AL DISK (A)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +12.0176 -11.5715 2P0.6350
UNIT 51
COM='TUBE CELL IN AL DISK (B)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +12.0176 -11.5715 2P0.6350
UNIT 52
COM='TUBE CELL IN AL DISK (C)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +11.5715 -12.0176 +11.5715 -12.0176 2P0.6350
UNIT 53
COM='TUBE CELL IN AL DISK (D)'
ARRAY  2 -10.7083 -10.7083 -0.6350
CUBOID 3 1 4P11.2141 2P0.6350
CUBOID 5 1 4P11.3355 2P0.6350
CUBOID 3 1 4P11.5260 2P0.6350
HOLE  7 0.0 +11.4308 0.0
HOLE  7 0.0 -11.4308 0.0
HOLE  8 +11.4308 0.0 0.0
HOLE  8 -11.4308 0.0 0.0
CUBOID 5 1 4P11.5715 2P0.6350
CUBOID 3 1 +12.0176 -11.5715 +11.5715 -12.0176 2P0.6350
UNIT 54
COM='WEB UNIT (1.5" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.8725 2P0.6350
UNIT 55
COM='WEB UNIT (1.0" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.2510 2P0.6350
UNIT 56
COM='WEB UNIT (0.875" WEB) - AL DISKS'
CUBOID 4 1  2P11.7946 2P1.0923 2P0.6350
UNIT 57
COM='6X1 FUEL TUBE STACK AL DISK'
ARRAY 30 -11.7946 -77.3262 -0.6350
UNIT 58
COM='6X1 FUEL TUBE STACK AL DISK'
ARRAY 31 -11.7946 -77.3262 -0.6350
UNIT 59
COM='2X1 FUEL TUBE STACK OF TUBES AL DISK'
ARRAY 32 -11.7946 -25.4616 -0.6350
UNIT 60
COM='2X1 FUEL TUBE STACK OF TUBES AL DISK'
ARRAY 33 -11.7946 -25.4616 -0.6350
UNIT 70
COM='BASKET STRUCTURE IN STORAGE CASK - WATER DISK'
CYLINDER 3 1 +83.5787 2P2.4892
HOLE 17  -13.6669 0.0 0.0
HOLE 18  +13.6669 0.0 0.0
HOLE 19  -39.7578 0.0 0.0
HOLE 20  +39.7578 0.0 0.0
HOLE 19  -65.5312 0.0 0.0
HOLE 20  +65.5312 0.0 0.0
HOLE 10  +40.8048 +40.8048 0.0
HOLE 11  -40.8048 +40.8048 0.0
HOLE 12  -40.8048 -40.8048 0.0
HOLE 13  +40.8048 -40.8048 0.0
CYLINDER  5 1  +85.1662 2P2.4892
CYLINDER  9 1  +94.615 2P2.4892
CYLINDER  7 1 +100.965 2P2.4892
CYLINDER  8 1 +172.72 2P2.4892
CUBOID    9 1 4P230.0 2P2.4892
UNIT 71
COM='BASKET STRUCTURE IN STORAGE CASK - ST DISK'
CYLINDER  5 1 +83.1850 2P0.6350
```

Figure 6.8-4 (continued)

```
HOLE 37  -13.6669 0.0 0.0
HOLE 38  +13.6669 0.0 0.0
HOLE 39  -39.7578 0.0 0.0
HOLE 40  +39.7578 0.0 0.0
HOLE 39  -65.5312 0.0 0.0
HOLE 40  +65.5312 0.0 0.0
HOLE 30  +40.8048 +40.8048 0.0
HOLE 31  -40.8048 +40.8048 0.0
HOLE 32  -40.8048 -40.8048 0.0
HOLE 33  +40.8048 -40.8048 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER  9 1  +94.615 2P0.6350
CYLINDER  7 1 +100.965 2P0.6350
CYLINDER  8 1 +172.72 2P0.6350
CUBOID    9 1 4P230.0 2P0.6350
UNIT 72
COM='BASKET STRUCTURE IN STORAGE CASK - AL DISK'
CYLINDER  4 1 +82.8675 2P0.6350
HOLE 57  -13.6669 0.0 0.0
HOLE 58  +13.6669 0.0 0.0
HOLE 59  -39.7578 0.0 0.0
HOLE 60  +39.7578 0.0 0.0
HOLE 59  -65.5312 0.0 0.0
HOLE 60  +65.5312 0.0 0.0
HOLE 50  +40.8048 +40.8048 0.0
HOLE 51  -40.8048 +40.8048 0.0
HOLE 52  -40.8048 -40.8048 0.0
HOLE 53  +40.8048 -40.8048 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER  9 1  +94.615 2P0.6350
CYLINDER  7 1 +100.965 2P0.6350
CYLINDER  8 1 +172.72 2P0.6350
CUBOID    9 1 4P230.0 2P0.6350
GLOBAL UNIT 73
COM='DISK SLICE STACK'
ARRAY  40 -230.0 -230.0 0.0
END GEOM
READ ARRAY
ARA=1   NUX=17 NUY=17 NUZ=1  FILL
                        34R1
          5R1   2   2R1   2   2R1   2   5R1
               3R1   2   9R1   2   3R1
                        17R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        34R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        34R1
2R1   2   2R1   2   2R1   2   2R1   2   2R1   2   2R1
                        17R1
               3R1   2   9R1   2   3R1
          5R1   2   2R1   2   2R1   2   5R1
                        34R1
END FILL
ARA=2  NUX=17 NUY=17 NUZ=1  FILL
                        34R3
          5R3   4   2R3   4   2R3   4   5R3
               3R3   4   9R3   4   3R3
                        17R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        34R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        34R3
2R3   4   2R3   4   2R3   4   2R3   4   2R3   4   2R3
                        17R3
               3R3   4   9R3   4   3R3
          5R3   4   2R3   4   2R3   4   5R3
                        34R3
END FILL
ARA=10 NUX=1 NUY=11 NUZ=1 FILL 12 16 12 15 12 14 11 15 11 16 11 END FILL
ARA=11 NUX=1 NUY=11 NUZ=1 FILL 13 16 13 15 13 14 10 15 10 16 10 END FILL
ARA=12 NUX=1 NUY=3  NUZ=1 FILL 12 14 11 END FILL
ARA=13 NUX=1 NUY=3  NUZ=1 FILL 13 14 10 END FILL
ARA=20 NUX=1 NUY=11 NUZ=1 FILL 32 36 32 35 32 34 31 35 31 36 31 END FILL
ARA=21 NUX=1 NUY=11 NUZ=1 FILL 33 36 33 35 33 34 30 35 30 36 30 END FILL
ARA=22 NUX=1 NUY=3  NUZ=1 FILL 32 34 31 END FILL
ARA=23 NUX=1 NUY=3  NUZ=1 FILL 33 34 30 END FILL
ARA=30 NUX=1 NUY=11 NUZ=1 FILL 52 56 52 55 52 54 51 55 51 56 51 END FILL
ARA=31 NUX=1 NUY=11 NUZ=1 FILL 53 56 53 55 53 54 50 55 50 56 50 END FILL
ARA=32 NUX=1 NUY=3  NUZ=1 FILL 52 54 51 END FILL
ARA=33 NUX=1 NUY=3  NUZ=1 FILL 53 54 50 END FILL
ARA=40 NUX=1 NUY=1 NUZ=4 FILL 70 71 70 72 END FILL
END ARRAY
READ BOUNDS ZFC=PER YXF=MIRROR END BOUNDS
END DATA
END
```

Figure 6.8-5 CSAS Input for Normal Conditions – Transfer Cask Containing BWR Fuel

```
=CSAS25
UMS BWR TFR; NORMAL OP; CASK ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 75%B10
27GROUPNDF4 LATTICECELL
UO2   1 0.95 293.0 92235 4.00 92238 96.00 END
ZIRCALLOY  2 1.0  293.0  END
H2O   3 1.0  293.0  END
AL    4 1.0  293.0  END
SS304 5 1.0  293.0  END
AL    6 DEN=2.6849 0.8706 293.O END
B-10  6 DEN=2.6849 0.0137 293.O END
B-11  6 DEN=2.6849 0.0830 293.O END
C     6 DEN=2.6849 0.0281 293.O END
CARBONSTEEL 7 1.0 293.0 END
PB    8 1.0 293.O  END
B-10  9 0.0 8.553-5 END
B-11  9 0.0 3.422-4 END
AL    9 0.0 7.763-3 END
H     9 0.0 5.854-2 END
O     9 0.0 2.609-2 END
C     9 0.0 2.264-2 END
N     9 0.0 1.394-3 END
H2O  10 1.0 293.0 END
END COMP
SQUAREPITCH  1.4529 0.9055 1 3 1.0770 2 0.9246 0 END
UMS BWR TFR; NORMAL OP; CASK ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 75%B10
READ PARAM RUN=YES PLT=NO TME=5000 GEN=803 NPG=1000 END PARAM
READ GEOM
UNIT  1
COM='FUEL PIN CELL - WITH H2O'
CYLINDER  1 1  0.4528   2P1.7145
CYLINDER  0 1  0.4623   2P1.7145
CYLINDER  2 1  0.5385   2P1.7145
CUBOID    3 1  4P0.7264 2P1.7145
UNIT  2
COM='WATER ROD CELL - WITH H2O'
CYLINDER  3 1  0.4623   2P1.7145
CYLINDER  2 1  0.5385   2P1.7145
CUBOID    3 1  4P0.7264 2P1.7145
UNIT  3
COM='FUEL PIN CELL - WITH ST DISK'
CYLINDER  1 1  0.4528   2P0.7938
CYLINDER  0 1  0.4623   2P0.7938
CYLINDER  2 1  0.5385   2P0.7938
CUBOID    3 1  4P0.7264 2P0.7938
UNIT  4
COM='WATER ROD CELL - WITH ST DISK'
CYLINDER  3 1  0.4623   2P0.7938
CYLINDER  2 1  0.5385   2P0.7938
CUBOID    3 1  4P0.7264 2P0.7938
UNIT  5
COM='FUEL PIN CELL - WITH AL DISK'
CYLINDER  1 1  0.4528   2P0.6350
CYLINDER  0 1  0.4623   2P0.6350
CYLINDER  2 1  0.5385   2P0.6350
CUBOID    3 1  4P0.7264 2P0.6350
UNIT  6
COM='WATER ROD CELL - WITH AL DISK'
CYLINDER  3 1  0.4623   2P0.6350
CYLINDER  2 1  0.5385   2P0.6350
CUBOID    3 1  4P0.7264 2P0.6350
UNIT  7
COM='FUEL PIN ARRAY + CHANNEL - BETWEEN DISKS'
ARRAY  1   -6.5376 -6.5376 -1.7145
CUBOID 3 1 4P6.7031 2P1.7145
CUBOID 2 1 4P6.9063 2P1.7145
UNIT  8
COM='FUEL PIN ARRAY + CHANNEL - ST DISKS'
ARRAY  2   -6.5376 -6.5376 -0.7938
CUBOID 3 1 4P6.7031 2P0.7938
CUBOID 2 1 4P6.9063 2P0.7938
UNIT  9
COM='FUEL PIN ARRAY + CHANNEL - AL DISKS'
ARRAY  3   -6.5376 -6.5376 -0.6350
CUBOID 3 1 4P6.7031 2P0.6350
CUBOID 2 1 4P6.9063 2P0.6350
UNIT 10
COM='X-X BORAL + COVER SHEET BETWEEN DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P1.7145
CUBOID 4 1 2P6.7310 2P0.1714 2P1.7145
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P1.7145
UNIT 11
COM='Y-Y BORAL + COVER SHEET BETWEEN DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P1.7145
CUBOID 4 1 2P0.1714 2P6.7310 2P1.7145
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P1.7145
UNIT 12
COM='X-X BORAL + COVER SHEET WITH ST DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P0.7938
CUBOID 4 1 2P6.7310 2P0.1714 2P0.7938
```

Figure 6.8-5 (continued)

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CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P0.7938
UNIT 13
COM='Y-Y BORAL + COVER SHEET WITH ST DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P0.7938
CUBOID 4 1 2P0.1714 2P6.7310 2P0.7938
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P0.7938
UNIT 14
COM='X-X BORAL + COVER SHEET WITH AL DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P0.6350
CUBOID 4 1 2P6.7310 2P0.1714 2P0.6350
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P0.6350
UNIT 15
COM='Y-Y BORAL + COVER SHEET WITH AL DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P0.6350
CUBOID 4 1 2P0.1714 2P6.7310 2P0.6350
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P0.6350
UNIT 20
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (TR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 21
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 22
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 23
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 24
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (T)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 25
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 26
COM='FUEL TUBE CELL TOP BORAL SHEETS - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
UNIT 27
COM='FUEL TUBE CELL TOP BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
UNIT 28
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 29
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 30
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
```

Figure 6.8-5 (continued)

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UNIT 31
COM='FUEL TUBE CELL NO BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
UNIT 40
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (TR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 41
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 42
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 43
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 44
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (T)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 45
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 46
COM='FUEL TUBE CELL TOP BORAL SHEETS - STEEL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
UNIT 47
COM='FUEL TUBE CELL TOP BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
UNIT 48
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 49
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 50
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 51
COM='FUEL TUBE CELL NO BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
UNIT 60
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (TR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
```

Figure 6.8-5 (continued)

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HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 61
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 62
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 63
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 64
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (T)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 65
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 66
COM='FUEL TUBE CELL TOP BORAL SHEETS - AL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
UNIT 67
COM='FUEL TUBE CELL TOP BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
UNIT 68
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 69
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 70
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 71
COM='FUEL TUBE CELL NO BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
UNIT 80
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (TR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   20  -0.0297 -0.0297 0.0
UNIT 81
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   21  -0.3586 -0.0297 0.0
UNIT 82
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   22  -0.3586 -0.3586 0.0
UNIT 83
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   23  -0.0297 -0.3586 0.0
UNIT 84
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Figure 6.8-5 (continued)

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COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (T)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   24  -0.1942 -0.0297 0.0
UNIT 85
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   25  -0.1942 -0.3586 0.0
UNIT 86
COM='DISK OPENING TOP BORAL SHEET TUBE - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   26  -0.3586 -0.0297 0.0
UNIT 87
COM='DISK OPENING TOP BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   27  -0.3586 -0.3586 0.0
UNIT 88
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   28  -0.3586 -0.3586 0.0
UNIT 89
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   29  -0.1942 -0.3586 0.0
UNIT 90
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   30  -0.0297 -0.3586 0.0
UNIT 91
COM='DISK OPENING NO BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   31  -0.3586 -0.3586 0.0
UNIT 100
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (TR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   40  -0.0297 -0.0297 0.0
UNIT 101
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   41  -0.3586 -0.0297 0.0
UNIT 102
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   42  -0.3586 -0.3586 0.0
UNIT 103
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   43  -0.0297 -0.3586 0.0
UNIT 104
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (T)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   44  -0.1942 -0.0297 0.0
UNIT 105
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   45  -0.1942 -0.3586 0.0
UNIT 106
COM='DISK OPENING TOP BORAL SHEET TUBE - STEEL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   46  -0.3586 -0.0297 0.0
UNIT 107
COM='DISK OPENING TOP BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   47  -0.3586 -0.3586 0.0
UNIT 108
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   48  -0.3586 -0.3586 0.0
UNIT 109
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   49  -0.1942 -0.3586 0.0
UNIT 110
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   50  -0.0297 -0.3586 0.0
UNIT 111
COM='DISK OPENING NO BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   51  -0.3586 -0.3586 0.0
UNIT 120
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (TR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   60  -0.0297 -0.0297 0.0
UNIT 121
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   61  -0.3586 -0.0297 0.0
UNIT 122
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   62  -0.3586 -0.3586 0.0
UNIT 123
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   63  -0.0297 -0.3586 0.0
UNIT 124
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (T)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   64  -0.1942 -0.0297 0.0
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Figure 6.8-5 (continued)

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UNIT 125
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   65  -0.1942 -0.3586 0.0
UNIT 126
COM='DISK OPENING TOP BORAL SHEET TUBE - AL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   66  -0.3586 -0.0297 0.0
UNIT 127
COM='DISK OPENING TOP BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   67  -0.3586 -0.3586 0.0
UNIT 128
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   68  -0.3586 -0.3586 0.0
UNIT 129
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   69  -0.1942 -0.3586 0.0
UNIT 130
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   70  -0.0297 -0.3586 0.0
UNIT 131
COM='DISK OPENING NO BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   71  -0.3586 -0.3586 0.0
UNIT 140
COM='BASKET STRUCTURE IN TRANPORT CASK - WATER DISK'
CYLINDER 3 1 +83.5787 2P1.7145
HOLE 90  -70.3885  +8.7986 0.0
HOLE 83  -52.7914  +8.7986 0.0
HOLE 83  -52.7914 +26.3957 0.0
HOLE 90  -52.7914 +43.9928 0.0
HOLE 83  -35.1942  +8.7986 0.0
HOLE 83  -35.1942 +26.3957 0.0
HOLE 83  -35.1942 +43.9928 0.0
HOLE 90  -35.1942 +61.5899 0.0
HOLE 83  -17.5971  +8.7986 0.0
HOLE 83  -17.5971 +26.3957 0.0
HOLE 83  -17.5971 +43.9928 0.0
HOLE 90  -17.5971 +61.5899 0.0
HOLE 85   0.0      +8.7986 0.0
HOLE 85   0.0     +26.3957 0.0
HOLE 85   0.0     +43.9928 0.0
HOLE 89   0.0     +61.5899 0.0
HOLE 82  +17.5971  +8.7986 0.0
HOLE 82  +17.5971 +26.3957 0.0
HOLE 82  +17.5971 +43.9928 0.0
HOLE 88  +17.5971 +61.5899 0.0
HOLE 82  +35.1942  +8.7986 0.0
HOLE 82  +35.1942 +26.3957 0.0
HOLE 82  +35.1942 +43.9928 0.0
HOLE 91  +35.1942 +61.5899 0.0
HOLE 82  +52.7914  +8.7986 0.0
HOLE 87  +52.7914 +26.3957 0.0
HOLE 91  +52.7914 +43.9928 0.0
HOLE 91  +70.3885  +8.7986 0.0
HOLE 80  -70.3885  -8.7986 0.0
HOLE 80  -52.7914  -8.7986 0.0
HOLE 80  -52.7914 -26.3957 0.0
HOLE 80  -52.7914 -43.9928 0.0
HOLE 80  -35.1942  -8.7986 0.0
HOLE 80  -35.1942 -26.3957 0.0
HOLE 80  -35.1942 -43.9928 0.0
HOLE 80  -35.1942 -61.5899 0.0
HOLE 80  -17.5971  -8.7986 0.0
HOLE 80  -17.5971 -26.3957 0.0
HOLE 80  -17.5971 -43.9928 0.0
HOLE 80  -17.5971 -61.5899 0.0
HOLE 84   0.0      -8.7986 0.0
HOLE 84   0.0     -26.3957 0.0
HOLE 84   0.0     -43.9928 0.0
HOLE 84   0.0     -61.5899 0.0
HOLE 81  +17.5971  -8.7986 0.0
HOLE 81  +17.5971 -26.3957 0.0
HOLE 81  +17.5971 -43.9928 0.0
HOLE 81  +17.5971 -61.5899 0.0
HOLE 81  +35.1942  -8.7986 0.0
HOLE 81  +35.1942 -26.3957 0.0
HOLE 81  +35.1942 -43.9928 0.0
HOLE 86  +35.1942 -61.5899 0.0
HOLE 81  +52.7914  -8.7986 0.0
HOLE 86  +52.7914 -26.3957 0.0
HOLE 86  +52.7914 -43.9928 0.0
HOLE 86  +70.3885  -8.7986 0.0
CYLINDER   5 1  +85.1662 2P1.7145
CYLINDER  10 1  +86.0425 2P1.7145
CYLINDER   7 1  +87.9475 2P1.7145
CYLINDER   8 1  +97.4725 2P1.7145
CYLINDER   9 1 +102.5525 2P1.7145
CYLINDER   7 1 +105.7275 2P1.7145
CUBOID    10 1 4P125.0   2P1.7145
UNIT 141
COM='BASKET STRUCTURE IN TRANPORT CASK - SS DISK'
CYLINDER 7 1  +83.1850 2P0.7938
HOLE 110  -70.3885  +8.7986 0.0
HOLE 103  -52.7914  +8.7986 0.0
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Figure 6.8-5 (continued)

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HOLE 103  -52.7914 +26.3957 0.0
HOLE 110  -52.7914 +43.9928 0.0
HOLE 103  -35.1942  +8.7986 0.0
HOLE 103  -35.1942 +26.3957 0.0
HOLE 103  -35.1942 +43.9928 0.0
HOLE 110  -35.1942 +61.5899 0.0
HOLE 103  -17.5971  +8.7986 0.0
HOLE 103  -17.5971 +26.3957 0.0
HOLE 103  -17.5971 +43.9928 0.0
HOLE 110  -17.5971 +61.5899 0.0
HOLE 105   0.0      +8.7986 0.0
HOLE 105   0.0     +26.3957 0.0
HOLE 105   0.0     +43.9928 0.0
HOLE 109   0.0     +61.5899 0.0
HOLE 102  +17.5971  +8.7986 0.0
HOLE 102  +17.5971 +26.3957 0.0
HOLE 102  +17.5971 +43.9928 0.0
HOLE 108  +17.5971 +61.5899 0.0
HOLE 102  +35.1942  +8.7986 0.0
HOLE 102  +35.1942 +26.3957 0.0
HOLE 102  +35.1942 +43.9928 0.0
HOLE 111  +35.1942 +61.5899 0.0
HOLE 102  +52.7914  +8.7986 0.0
HOLE 107  +52.7914 +26.3957 0.0
HOLE 111  +52.7914 +43.9928 0.0
HOLE 111  +70.3885  +8.7986 0.0
HOLE 100  -70.3885  -8.7986 0.0
HOLE 100  -52.7914  -8.7986 0.0
HOLE 100  -52.7914 -26.3957 0.0
HOLE 100  -52.7914 -43.9928 0.0
HOLE 100  -35.1942  -8.7986 0.0
HOLE 100  -35.1942 -26.3957 0.0
HOLE 100  -35.1942 -43.9928 0.0
HOLE 100  -35.1942 -61.5899 0.0
HOLE 100  -17.5971  -8.7986 0.0
HOLE 100  -17.5971 -26.3957 0.0
HOLE 100  -17.5971 -43.9928 0.0
HOLE 100  -17.5971 -61.5899 0.0
HOLE 104   0.0      -8.7986 0.0
HOLE 104   0.0     -26.3957 0.0
HOLE 104   0.0     -43.9928 0.0
HOLE 104   0.0     -61.5899 0.0
HOLE 101  +17.5971  -8.7986 0.0
HOLE 101  +17.5971 -26.3957 0.0
HOLE 101  +17.5971 -43.9928 0.0
HOLE 101  +17.5971 -61.5899 0.0
HOLE 101  +35.1942  -8.7986 0.0
HOLE 101  +35.1942 -26.3957 0.0
HOLE 101  +35.1942 -43.9928 0.0
HOLE 106  +35.1942 -61.5899 0.0
HOLE 101  +52.7914  -8.7986 0.0
HOLE 106  +52.7914 -26.3957 0.0
HOLE 106  +52.7914 -43.9928 0.0
HOLE 106  +70.3885  -8.7986 0.0
CYLINDER   3 1  +83.5787 2P0.7938
CYLINDER   5 1  +85.1662 2P0.7938
CYLINDER  10 1  +86.0425 2P0.7938
CYLINDER   7 1  +87.9475 2P0.7938
CYLINDER   8 1  +97.4725 2P0.7938
CYLINDER   9 1 +102.5525 2P0.7938
CYLINDER   7 1 +105.7275 2P0.7938
CUBOID    10 1 4P125.0   2P0.7938
UNIT 142
COM='BASKET STRUCTURE IN TRANPORT CASK - AL DISK'
CYLINDER 4 1  +82.8675 2P0.6350
HOLE 130  -70.3885  +8.7986 0.0
HOLE 123  -52.7914  +8.7986 0.0
HOLE 123  -52.7914 +26.3957 0.0
HOLE 130  -52.7914 +43.9928 0.0
HOLE 123  -35.1942  +8.7986 0.0
HOLE 123  -35.1942 +26.3957 0.0
HOLE 123  -35.1942 +43.9928 0.0
HOLE 130  -35.1942 +61.5899 0.0
HOLE 123  -17.5971  +8.7986 0.0
HOLE 123  -17.5971 +26.3957 0.0
HOLE 123  -17.5971 +43.9928 0.0
HOLE 130  -17.5971 +61.5899 0.0
HOLE 125   0.0      +8.7986 0.0
HOLE 125   0.0     +26.3957 0.0
HOLE 125   0.0     +43.9928 0.0
HOLE 129   0.0     +61.5899 0.0
HOLE 122  +17.5971  +8.7986 0.0
HOLE 122  +17.5971 +26.3957 0.0
HOLE 122  +17.5971 +43.9928 0.0
HOLE 128  +17.5971 +61.5899 0.0
HOLE 122  +35.1942  +8.7986 0.0
HOLE 122  +35.1942 +26.3957 0.0
HOLE 122  +35.1942 +43.9928 0.0
HOLE 131  +35.1942 +61.5899 0.0
HOLE 122  +52.7914  +8.7986 0.0
HOLE 127  +52.7914 +26.3957 0.0
HOLE 131  +52.7914 +43.9928 0.0
HOLE 131  +70.3885  +8.7986 0.0
HOLE 120  -70.3885  -8.7986 0.0
HOLE 120  -52.7914  -8.7986 0.0
HOLE 120  -52.7914 -26.3957 0.0
HOLE 120  -52.7914 -43.9928 0.0
HOLE 120  -35.1942  -8.7986 0.0
HOLE 120  -35.1942 -26.3957 0.0
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Figure 6.8-5 (continued)

```
HOLE 120  -35.1942 -43.9928 0.0
HOLE 120  -35.1942 -61.5899 0.0
HOLE 120  -17.5971  -8.7986 0.0
HOLE 120  -17.5971 -26.3957 0.0
HOLE 120  -17.5971 -43.9928 0.0
HOLE 120  -17.5971 -61.5899 0.0
HOLE 124   0.0      -8.7986 0.0
HOLE 124   0.0     -26.3957 0.0
HOLE 124   0.0     -43.9928 0.0
HOLE 124   0.0     -61.5899 0.0
HOLE 121  +17.5971  -8.7986 0.0
HOLE 121  +17.5971 -26.3957 0.0
HOLE 121  +17.5971 -43.9928 0.0
HOLE 121  +17.5971 -61.5899 0.0
HOLE 121  +35.1942  -8.7986 0.0
HOLE 121  +35.1942 -26.3957 0.0
HOLE 121  +35.1942 -43.9928 0.0
HOLE 126  +35.1942 -61.5899 0.0
HOLE 121  +52.7914  -8.7986 0.0
HOLE 126  +52.7914 -26.3957 0.0
HOLE 126  +52.7914 -43.9928 0.0
HOLE 126  +70.3885  -8.7986 0.0
CYLINDER   3 1  +83.5787 2P0.6350
CYLINDER   5 1  +85.1662 2P0.6350
CYLINDER  10 1  +86.0425 2P0.6350
CYLINDER   7 1  +87.9475 2P0.6350
CYLINDER   8 1  +97.4725 2P0.6350
CYLINDER   9 1 +102.5525 2P0.6350
CYLINDER   7 1 +105.7275 2P0.6350
CUBOID    10 1 4P125.0   2P0.6350
GLOBAL UNIT 143
COM='CASK SLICES TOGETHER'
ARRAY  4 -125.00 -125.00 0.0
END GEOM
READ ARRAY
ARA=1   NUX=9 NUY=9 NUZ=1  FILL
    36R1
 4R1  2   4R1
 5R1  2   3R1
    27R1
END FILL
ARA=2   NUX=9 NUY=9 NUZ=1  FILL
    36R3
 4R3  4   4R3
 5R3  4   3R3
    27R3
END FILL
ARA=3   NUX=9 NUY=9 NUZ=1  FILL
    36R5
 4R5  6   4R5
 5R5  6   3R5
    27R5
END FILL
ARA=4 NUX=1 NUY=1 NUZ=4 FILL 140 141 140 142 END FILL
END ARRAY
READ BOUNDS ZFC=PER YXF=PER END BOUNDS
END DATA
END
```

Figure 6.8-6 CSAS Input for Accident Conditions - Transfer Cask Containing BWR Fuel

```
=CSAS25
UMS BWR TFR; ACCIDENT OP; CASK ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 75%B10
27GROUPNDF4 LATTICECELL
UO2   1 0.95 293.0 92235 4.00 92238 96.00 END
ZIRCALLOY  2 1.0  293.0  END
H2O   3 1.0  293.0  END
AL    4 1.0  293.0  END
SS304 5 1.0  293.0  END
AL    6 DEN=2.6849 0.8706 293.O END
B-10  6 DEN=2.6849 0.0137 293.O END
B-11  6 DEN=2.6849 0.0830 293.O END
C     6 DEN=2.6849 0.0281 293.O END
CARBONSTEEL 7 1.0 293.0 END
PB    8 1.0 293.O  END
B-10  9 0.0 8.553-5 END
B-11  9 0.0 3.422-4 END
AL    9 0.0 7.763-3 END
H     9 0.0 5.854-2 END
O     9 0.0 2.609-2 END
C     9 0.0 2.264-2 END
N     9 0.0 1.394-3 END
H2O  10 1.0 293.0 END
H2O  11 1.0 293.0 END
END COMP
SQUAREPITCH  1.4529 0.9055 1 3 1.0770 2 0.9246 11 END
UMS BWR TFR; ACCIDENT OP; CASK ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 75%B10
READ PARAM RUN=YES PLT=NO TME=5000 GEN=803 NPG=1000 END PARAM
READ GEOM
UNIT  1
COM='FUEL PIN CELL - WITH H2O'
CYLINDER  1 1  0.4528   2P1.7145
CYLINDER 11 1  0.4623   2P1.7145
CYLINDER  2 1  0.5385   2P1.7145
CUBOID    3 1  4P0.7264 2P1.7145
UNIT  2
COM='WATER ROD CELL - WITH H2O'
CYLINDER  3 1  0.4623   2P1.7145
CYLINDER  2 1  0.5385   2P1.7145
CUBOID    3 1  4P0.7264 2P1.7145
UNIT  3
COM='FUEL PIN CELL - WITH ST DISK'
CYLINDER  1 1  0.4528   2P0.7938
CYLINDER 11 1  0.4623   2P0.7938
CYLINDER  2 1  0.5385   2P0.7938
CUBOID    3 1  4P0.7264 2P0.7938
UNIT  4
COM='WATER ROD CELL - WITH ST DISK'
CYLINDER  3 1  0.4623   2P0.7938
CYLINDER  2 1  0.5385   2P0.7938
CUBOID    3 1  4P0.7264 2P0.7938
UNIT  5
COM='FUEL PIN CELL - WITH AL DISK'
CYLINDER  1 1  0.4528   2P0.6350
CYLINDER 11 1  0.4623   2P0.6350
CYLINDER  2 1  0.5385   2P0.6350
CUBOID    3 1  4P0.7264 2P0.6350
UNIT  6
COM='WATER ROD CELL - WITH AL DISK'
CYLINDER  3 1  0.4623   2P0.6350
CYLINDER  2 1  0.5385   2P0.6350
CUBOID    3 1  4P0.7264 2P0.6350
UNIT  7
COM='FUEL PIN ARRAY + CHANNEL - BETWEEN DISKS'
ARRAY  1   -6.5376 -6.5376 -1.7145
CUBOID 3 1 4P6.7031 2P1.7145
CUBOID 2 1 4P6.9063 2P1.7145
UNIT  8
COM='FUEL PIN ARRAY + CHANNEL - ST DISKS'
ARRAY  2   -6.5376 -6.5376 -0.7938
CUBOID 3 1 4P6.7031 2P0.7938
CUBOID 2 1 4P6.9063 2P0.7938
UNIT  9
COM='FUEL PIN ARRAY + CHANNEL - AL DISKS'
ARRAY  3   -6.5376 -6.5376 -0.6350
CUBOID 3 1 4P6.7031 2P0.6350
CUBOID 2 1 4P6.9063 2P0.6350
UNIT 10
COM='X-X BORAL + COVER SHEET BETWEEN DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P1.7145
CUBOID 4 1 2P6.7310 2P0.1714 2P1.7145
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P1.7145
UNIT 11
COM='Y-Y BORAL + COVER SHEET BETWEEN DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P1.7145
CUBOID 4 1 2P0.1714 2P6.7310 2P1.7145
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P1.7145
UNIT 12
COM='X-X BORAL + COVER SHEET WITH ST DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P0.7938
CUBOID 4 1 2P6.7310 2P0.1714 2P0.7938
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P0.7938
UNIT 13
COM='Y-Y BORAL + COVER SHEET WITH ST DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P0.7938
CUBOID 4 1 2P0.1714 2P6.7310 2P0.7938
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P0.7938
```

Figure 6.8-6 (continued)

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UNIT 14
COM='X-X BORAL + COVER SHEET WITH AL DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P0.6350
CUBOID 4 1 2P6.7310 2P0.1714 2P0.6350
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P0.6350
UNIT 15
COM='Y-Y BORAL + COVER SHEET WITH AL DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P0.6350
CUBOID 4 1 2P0.1714 2P6.7310 2P0.6350
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P0.6350
UNIT 20
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (TR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 21
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 22
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 23
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 24
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (T)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 25
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 26
COM='FUEL TUBE CELL TOP BORAL SHEETS - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
UNIT 27
COM='FUEL TUBE CELL TOP BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
UNIT 28
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 29
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 30
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 31
COM='FUEL TUBE CELL NO BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
```

Figure 6.8-6 (continued)

```
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
UNIT 40
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (TR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 41
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 42
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 43
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 44
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (T)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 45
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 46
COM='FUEL TUBE CELL TOP BORAL SHEETS - STEEL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
UNIT 47
COM='FUEL TUBE CELL TOP BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
UNIT 48
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 49
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 50
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 51
COM='FUEL TUBE CELL NO BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
UNIT 60
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (TR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
```

Figure 6.8-6 (continued)

```
HOLE   15 +7.7859 0.0 0.0
UNIT 61
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 62
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 63
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 64
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (T)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 65
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 66
COM='FUEL TUBE CELL TOP BORAL SHEETS - AL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
UNIT 67
COM='FUEL TUBE CELL TOP BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
UNIT 68
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 69
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 70
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 71
COM='FUEL TUBE CELL NO BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
UNIT 80
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (TR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   20  -0.0297 -0.0297 0.0
UNIT 81
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   21  -0.3586 -0.0297 0.0
UNIT 82
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   22  -0.3586 -0.3586 0.0
UNIT 83
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   23  -0.0297 -0.3586 0.0
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Figure 6.8-6 (continued)

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UNIT 84
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (T)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   24  -0.1942 -0.0297 0.0
UNIT 85
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   25  -0.1942 -0.3586 0.0
UNIT 86
COM='DISK OPENING TOP BORAL SHEET TUBE - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   26  -0.3586 -0.0297 0.0
UNIT 87
COM='DISK OPENING TOP BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   27  -0.3586 -0.3586 0.0
UNIT 88
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   28  -0.3586 -0.3586 0.0
UNIT 89
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   29  -0.1942 -0.3586 0.0
UNIT 90
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   30  -0.0297 -0.3586 0.0
UNIT 91
COM='DISK OPENING NO BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   31  -0.3586 -0.3586 0.0
UNIT 100
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (TR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   40  -0.0297 -0.0297 0.0
UNIT 101
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   41  -0.3586 -0.0297 0.0
UNIT 102
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   42  -0.3586 -0.3586 0.0
UNIT 103
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   43  -0.0297 -0.3586 0.0
UNIT 104
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (T)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   44  -0.1942 -0.0297 0.0
UNIT 105
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   45  -0.1942 -0.3586 0.0
UNIT 106
COM='DISK OPENING TOP BORAL SHEET TUBE - STEEL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   46  -0.3586 -0.0297 0.0
UNIT 107
COM='DISK OPENING TOP BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   47  -0.3586 -0.3586 0.0
UNIT 108
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   48  -0.3586 -0.3586 0.0
UNIT 109
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   49  -0.1942 -0.3586 0.0
UNIT 110
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   50  -0.0297 -0.3586 0.0
UNIT 111
COM='DISK OPENING NO BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   51  -0.3586 -0.3586 0.0
UNIT 120
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (TR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   60  -0.0297 -0.0297 0.0
UNIT 121
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   61  -0.3586 -0.0297 0.0
UNIT 122
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   62  -0.3586 -0.3586 0.0
UNIT 123
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   63  -0.0297 -0.3586 0.0
UNIT 124
```

Figure 6.8-6 (continued)

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COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (T)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   64  -0.1942 -0.0297 0.0
UNIT 125
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   65  -0.1942 -0.3586 0.0
UNIT 126
COM='DISK OPENING TOP BORAL SHEET TUBE - AL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   66  -0.3586 -0.0297 0.0
UNIT 127
COM='DISK OPENING TOP BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   67  -0.3586 -0.3586 0.0
UNIT 128
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   68  -0.3586 -0.3586 0.0
UNIT 129
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   69  -0.1942 -0.3586 0.0
UNIT 130
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   70  -0.0297 -0.3586 0.0
UNIT 131
COM='DISK OPENING NO BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   71  -0.3586 -0.3586 0.0
UNIT 140
COM='BASKET STRUCTURE IN TRANPORT CASK - WATER DISK'
CYLINDER 3 1 +83.5787 2P1.7145
HOLE 90  -70.3885  +8.7986 0.0
HOLE 83  -52.7914  +8.7986 0.0
HOLE 83  -52.7914 +26.3957 0.0
HOLE 90  -52.7914 +43.9928 0.0
HOLE 83  -35.1942  +8.7986 0.0
HOLE 83  -35.1942 +26.3957 0.0
HOLE 83  -35.1942 +43.9928 0.0
HOLE 90  -35.1942 +61.5899 0.0
HOLE 83  -17.5971  +8.7986 0.0
HOLE 83  -17.5971 +26.3957 0.0
HOLE 83  -17.5971 +43.9928 0.0
HOLE 90  -17.5971 +61.5899 0.0
HOLE 85   0.0      +8.7986 0.0
HOLE 85   0.0     +26.3957 0.0
HOLE 85   0.0     +43.9928 0.0
HOLE 89   0.0     +61.5899 0.0
HOLE 82  +17.5971  +8.7986 0.0
HOLE 82  +17.5971 +26.3957 0.0
HOLE 82  +17.5971 +43.9928 0.0
HOLE 88  +17.5971 +61.5899 0.0
HOLE 82  +35.1942  +8.7986 0.0
HOLE 82  +35.1942 +26.3957 0.0
HOLE 82  +35.1942 +43.9928 0.0
HOLE 91  +35.1942 +61.5899 0.0
HOLE 82  +52.7914  +8.7986 0.0
HOLE 87  +52.7914 +26.3957 0.0
HOLE 91  +52.7914 +43.9928 0.0
HOLE 91  +70.3885  +8.7986 0.0
HOLE 80  -70.3885  -8.7986 0.0
HOLE 80  -52.7914  -8.7986 0.0
HOLE 80  -52.7914 -26.3957 0.0
HOLE 80  -52.7914 -43.9928 0.0
HOLE 80  -35.1942  -8.7986 0.0
HOLE 80  -35.1942 -26.3957 0.0
HOLE 80  -35.1942 -43.9928 0.0
HOLE 80  -35.1942 -61.5899 0.0
HOLE 80  -17.5971  -8.7986 0.0
HOLE 80  -17.5971 -26.3957 0.0
HOLE 80  -17.5971 -43.9928 0.0
HOLE 80  -17.5971 -61.5899 0.0
HOLE 84   0.0      -8.7986 0.0
HOLE 84   0.0     -26.3957 0.0
HOLE 84   0.0     -43.9928 0.0
HOLE 84   0.0     -61.5899 0.0
HOLE 81  +17.5971  -8.7986 0.0
HOLE 81  +17.5971 -26.3957 0.0
HOLE 81  +17.5971 -43.9928 0.0
HOLE 81  +17.5971 -61.5899 0.0
HOLE 81  +35.1942  -8.7986 0.0
HOLE 81  +35.1942 -26.3957 0.0
HOLE 81  +35.1942 -43.9928 0.0
HOLE 86  +35.1942 -61.5899 0.0
HOLE 81  +52.7914  -8.7986 0.0
HOLE 86  +52.7914 -26.3957 0.0
HOLE 86  +52.7914 -43.9928 0.0
HOLE 86  +70.3885  -8.7986 0.0
CYLINDER   5 1  +85.1662 2P1.7145
CYLINDER  10 1  +86.0425 2P1.7145
CYLINDER   7 1  +87.9475 2P1.7145
CYLINDER   8 1  +97.4725 2P1.7145
CYLINDER   9 1 +102.5525 2P1.7145
CYLINDER   7 1 +105.7275 2P1.7145
CUBOID    10 1 4P125.0   2P1.7145
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Figure 6.8-6 (continued)

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UNIT 141
COM='BASKET STRUCTURE IN TRANPORT CASK - SS DISK'
CYLINDER 7 1  +83.1850 2P0.7938
HOLE 110  -70.3885  +8.7986 0.0
HOLE 103  -52.7914  +8.7986 0.0
HOLE 103  -52.7914 +26.3957 0.0
HOLE 110  -52.7914 +43.9928 0.0
HOLE 103  -35.1942  +8.7986 0.0
HOLE 103  -35.1942 +26.3957 0.0
HOLE 103  -35.1942 +43.9928 0.0
HOLE 110  -35.1942 +61.5899 0.0
HOLE 103  -17.5971  +8.7986 0.0
HOLE 103  -17.5971 +26.3957 0.0
HOLE 103  -17.5971 +43.9928 0.0
HOLE 110  -17.5971 +61.5899 0.0
HOLE 105   0.0      +8.7986 0.0
HOLE 105   0.0     +26.3957 0.0
HOLE 105   0.0     +43.9928 0.0
HOLE 109   0.0     +61.5899 0.0
HOLE 102  +17.5971  +8.7986 0.0
HOLE 102  +17.5971 +26.3957 0.0
HOLE 102  +17.5971 +43.9928 0.0
HOLE 108  +17.5971 +61.5899 0.0
HOLE 102  +35.1942  +8.7986 0.0
HOLE 102  +35.1942 +26.3957 0.0
HOLE 102  +35.1942 +43.9928 0.0
HOLE 111  +35.1942 +61.5899 0.0
HOLE 102  +52.7914  +8.7986 0.0
HOLE 107  +52.7914 +26.3957 0.0
HOLE 111  +52.7914 +43.9928 0.0
HOLE 111  +70.3885  +8.7986 0.0
HOLE 100  -70.3885  -8.7986 0.0
HOLE 100  -52.7914  -8.7986 0.0
HOLE 100  -52.7914 -26.3957 0.0
HOLE 100  -52.7914 -43.9928 0.0
HOLE 100  -35.1942  -8.7986 0.0
HOLE 100  -35.1942 -26.3957 0.0
HOLE 100  -35.1942 -43.9928 0.0
HOLE 100  -35.1942 -61.5899 0.0
HOLE 100  -17.5971  -8.7986 0.0
HOLE 100  -17.5971 -26.3957 0.0
HOLE 100  -17.5971 -43.9928 0.0
HOLE 100  -17.5971 -61.5899 0.0
HOLE 104   0.0      -8.7986 0.0
HOLE 104   0.0     -26.3957 0.0
HOLE 104   0.0     -43.9928 0.0
HOLE 104   0.0     -61.5899 0.0
HOLE 101  +17.5971  -8.7986 0.0
HOLE 101  +17.5971 -26.3957 0.0
HOLE 101  +17.5971 -43.9928 0.0
HOLE 101  +17.5971 -61.5899 0.0
HOLE 101  +35.1942  -8.7986 0.0
HOLE 101  +35.1942 -26.3957 0.0
HOLE 101  +35.1942 -43.9928 0.0
HOLE 106  +35.1942 -61.5899 0.0
HOLE 101  +52.7914  -8.7986 0.0
HOLE 106  +52.7914 -26.3957 0.0
HOLE 106  +52.7914 -43.9928 0.0
HOLE 106  +70.3885  -8.7986 0.0
CYLINDER   3 1  +83.5787 2P0.7938
CYLINDER   5 1  +85.1662 2P0.7938
CYLINDER  10 1  +86.0425 2P0.7938
CYLINDER   7 1  +87.9475 2P0.7938
CYLINDER   8 1  +97.4725 2P0.7938
CYLINDER   9 1 +102.5525 2P0.7938
CYLINDER   7 1 +105.7275 2P0.7938
CUBOID    10 1 4P125.0   2P0.7938
UNIT 142
COM='BASKET STRUCTURE IN TRANPORT CASK - AL DISK'
CYLINDER 4 1  +82.8675 2P0.6350
HOLE 130  -70.3885  +8.7986 0.0
HOLE 123  -52.7914  +8.7986 0.0
HOLE 123  -52.7914 +26.3957 0.0
HOLE 130  -52.7914 +43.9928 0.0
HOLE 123  -35.1942  +8.7986 0.0
HOLE 123  -35.1942 +26.3957 0.0
HOLE 123  -35.1942 +43.9928 0.0
HOLE 130  -35.1942 +61.5899 0.0
HOLE 123  -17.5971  +8.7986 0.0
HOLE 123  -17.5971 +26.3957 0.0
HOLE 123  -17.5971 +43.9928 0.0
HOLE 130  -17.5971 +61.5899 0.0
HOLE 125   0.0      +8.7986 0.0
HOLE 125   0.0     +26.3957 0.0
HOLE 125   0.0     +43.9928 0.0
HOLE 129   0.0     +61.5899 0.0
HOLE 122  +17.5971  +8.7986 0.0
HOLE 122  +17.5971 +26.3957 0.0
HOLE 122  +17.5971 +43.9928 0.0
HOLE 128  +17.5971 +61.5899 0.0
HOLE 122  +35.1942  +8.7986 0.0
HOLE 122  +35.1942 +26.3957 0.0
HOLE 122  +35.1942 +43.9928 0.0
HOLE 131  +35.1942 +61.5899 0.0
HOLE 122  +52.7914  +8.7986 0.0
HOLE 127  +52.7914 +26.3957 0.0
HOLE 131  +52.7914 +43.9928 0.0
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Figure 6.8-6 (continued)

```
HOLE 131  +70.3885  +8.7986 0.0
HOLE 120  -70.3885  -8.7986 0.0
HOLE 120  -52.7914  -8.7986 0.0
HOLE 120  -52.7914 -26.3957 0.0
HOLE 120  -52.7914 -43.9928 0.0
HOLE 120  -35.1942  -8.7986 0.0
HOLE 120  -35.1942 -26.3957 0.0
HOLE 120  -35.1942 -43.9928 0.0
HOLE 120  -35.1942 -61.5899 0.0
HOLE 120  -17.5971  -8.7986 0.0
HOLE 120  -17.5971 -26.3957 0.0
HOLE 120  -17.5971 -43.9928 0.0
HOLE 120  -17.5971 -61.5899 0.0
HOLE 124   0.0      -8.7986 0.0
HOLE 124   0.0     -26.3957 0.0
HOLE 124   0.0     -43.9928 0.0
HOLE 124   0.0     -61.5899 0.0
HOLE 121  +17.5971  -8.7986 0.0
HOLE 121  +17.5971 -26.3957 0.0
HOLE 121  +17.5971 -43.9928 0.0
HOLE 121  +17.5971 -61.5899 0.0
HOLE 121  +35.1942  -8.7986 0.0
HOLE 121  +35.1942 -26.3957 0.0
HOLE 121  +35.1942 -43.9928 0.0
HOLE 126  +35.1942 -61.5899 0.0
HOLE 121  +52.7914  -8.7986 0.0
HOLE 126  +52.7914 -26.3957 0.0
HOLE 126  +52.7914 -43.9928 0.0
HOLE 126  +70.3885  -8.7986 0.0
CYLINDER   3 1  +83.5787 2P0.6350
CYLINDER   5 1  +85.1662 2P0.6350
CYLINDER  10 1  +86.0425 2P0.6350
CYLINDER   7 1  +87.9475 2P0.6350
CYLINDER   8 1  +97.4725 2P0.6350
CYLINDER   9 1 +102.5525 2P0.6350
CYLINDER   7 1 +105.7275 2P0.6350
CUBOID    10 1 4P125.0   2P0.6350
GLOBAL UNIT 143
COM='CASK SLICES TOGETHER'
ARRAY  4 -125.00 -125.00 0.0
END GEOM
READ ARRAY
ARA=1   NUX=9 NUY=9 NUZ=1  FILL
    36R1
 4R1  2   4R1
 5R1  2   3R1
    27R1
END FILL
ARA=2   NUX=9 NUY=9 NUZ=1  FILL
    36R3
 4R3  4   4R3
 5R3  4   3R3
    27R3
END FILL
ARA=3   NUX=9 NUY=9 NUZ=1  FILL
    36R5
 4R5  6   4R5
 5R5  6   3R5
    27R5
END FILL
ARA=4 NUX=1 NUY=1 NUZ=4 FILL 140 141 140 142 END FILL
END ARRAY
READ BOUNDS ZFC=PER YXF=PER END BOUNDS
END DATA
END
```

Figure 6.8-7 CSAS Input for Normal Conditions–Vertical Concrete Cask Containing BWR Fuel

```
=CSAS25
UMS BWR VCC; NORMAL OP; CASK ARRAY; 0.0001 GM/CC IN - 0.0001 GM/CC EX; 75%B10
27GROUPNDF4 LATTICECELL
UO2   1 0.95 293.0 92235 4.00 92238 96.00 END
ZIRCALLOY  2 1.0  293.0  END
H2O   3 0.0001  293.0  END
AL    4 1.0  293.0  END
SS304 5 1.0  293.0  END
AL    6 DEN=2.6849 0.8706 293.O END
B-10  6 DEN=2.6849 0.0137 293.O END
B-11  6 DEN=2.6849 0.0830 293.O END
C     6 DEN=2.6849 0.0281 293.O END
CARBONSTEEL 7 1.0 293.0 END
REG-CONCRETE 8 0.9750 293.0 END
H2O   9 0.0001 293.0 END
END COMP
SQUAREPITCH  1.4529 0.9055 1 3 1.0770 2 0.9246 0 END
UMS BWR VCC; NORMAL OP; CASK ARRAY; 0.0001 GM/CC IN - 0.0001 GM/CC EX; 75%B10
READ PARAM RUN=YES PLT=NO TME=5000 GEN=203 NPG=1000 END PARAM
READ GEOM
UNIT  1
COM='FUEL PIN CELL - WITH H2O'
CYLINDER  1 1  0.4528   2P1.7145
CYLINDER  0 1  0.4623   2P1.7145
CYLINDER  2 1  0.5385   2P1.7145
CUBOID    3 1  4P0.7264 2P1.7145
UNIT  2
COM='WATER ROD CELL - WITH H2O'
CYLINDER  3 1  0.4623   2P1.7145
CYLINDER  2 1  0.5385   2P1.7145
CUBOID    3 1  4P0.7264 2P1.7145
UNIT  3
COM='FUEL PIN CELL - WITH ST DISK'
CYLINDER  1 1  0.4528   2P0.7938
CYLINDER  0 1  0.4623   2P0.7938
CYLINDER  2 1  0.5385   2P0.7938
CUBOID    3 1  4P0.7264 2P0.7938
UNIT  4
COM='WATER ROD CELL - WITH ST DISK'
CYLINDER  3 1  0.4623   2P0.7938
CYLINDER  2 1  0.5385   2P0.7938
CUBOID    3 1  4P0.7264 2P0.7938
UNIT  5
COM='FUEL PIN CELL - WITH AL DISK'
CYLINDER  1 1  0.4528   2P0.6350
CYLINDER  0 1  0.4623   2P0.6350
CYLINDER  2 1  0.5385   2P0.6350
CUBOID    3 1  4P0.7264 2P0.6350
UNIT  6
COM='WATER ROD CELL - WITH AL DISK'
CYLINDER  3 1  0.4623   2P0.6350
CYLINDER  2 1  0.5385   2P0.6350
CUBOID    3 1  4P0.7264 2P0.6350
UNIT  7
COM='FUEL PIN ARRAY + CHANNEL - BETWEEN DISKS'
ARRAY  1   -6.5376 -6.5376 -1.7145
CUBOID 3 1 4P6.7031 2P1.7145
CUBOID 2 1 4P6.9063 2P1.7145
UNIT  8
COM='FUEL PIN ARRAY + CHANNEL - ST DISKS'
ARRAY  2   -6.5376 -6.5376 -0.7938
CUBOID 3 1 4P6.7031 2P0.7938
CUBOID 2 1 4P6.9063 2P0.7938
UNIT  9
COM='FUEL PIN ARRAY + CHANNEL - AL DISKS'
ARRAY  3   -6.5376 -6.5376 -0.6350
CUBOID 3 1 4P6.7031 2P0.6350
CUBOID 2 1 4P6.9063 2P0.6350
UNIT 10
COM='X-X BORAL + COVER SHEET BETWEEN DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P1.7145
CUBOID 4 1 2P6.7310 2P0.1714 2P1.7145
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P1.7145
UNIT 11
COM='Y-Y BORAL + COVER SHEET BETWEEN DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P1.7145
CUBOID 4 1 2P0.1714 2P6.7310 2P1.7145
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P1.7145
UNIT 12
COM='X-X BORAL + COVER SHEET WITH ST DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P0.7938
CUBOID 4 1 2P6.7310 2P0.1714 2P0.7938
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P0.7938
UNIT 13
COM='Y-Y BORAL + COVER SHEET WITH ST DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P0.7938
CUBOID 4 1 2P0.1714 2P6.7310 2P0.7938
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P0.7938
```

Figure 6.8-7 (continued)

```
UNIT 14
COM='X-X BORAL + COVER SHEET WITH AL DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P0.6350
CUBOID 4 1 2P6.7310 2P0.1714 2P0.6350
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P0.6350
UNIT 15
COM='Y-Y BORAL + COVER SHEET WITH AL DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P0.6350
CUBOID 4 1 2P0.1714 2P6.7310 2P0.6350
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P0.6350
UNIT 20
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (TR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 21
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 22
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 23
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 24
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (T)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 25
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 26
COM='FUEL TUBE CELL TOP BORAL SHEETS - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
UNIT 27
COM='FUEL TUBE CELL TOP BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
UNIT 28
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 29
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 30
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 31
COM='FUEL TUBE CELL NO BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
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Figure 6.8-7 (continued)

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UNIT 40
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (TR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 41
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 42
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 43
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 44
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (T)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 45
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 46
COM='FUEL TUBE CELL TOP BORAL SHEETS - STEEL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
UNIT 47
COM='FUEL TUBE CELL TOP BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
UNIT 48
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 49
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 50
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 51
COM='FUEL TUBE CELL NO BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
UNIT 60
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (TR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 61
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (TL)'
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Figure 6.8-7 (continued)

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CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 62
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 63
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 64
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (T)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 65
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 66
COM='FUEL TUBE CELL TOP BORAL SHEETS - AL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
UNIT 67
COM='FUEL TUBE CELL TOP BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
UNIT 68
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 69
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 70
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 71
COM='FUEL TUBE CELL NO BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
UNIT 80
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (TR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   20  -0.0297 -0.0297 0.0
UNIT 81
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   21  -0.3586 -0.0297 0.0
UNIT 82
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   22  -0.3586 -0.3586 0.0
UNIT 83
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   23  -0.0297 -0.3586 0.0
UNIT 84
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (T)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   24  -0.1942 -0.0297 0.0
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Figure 6.8-7 (continued)

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UNIT 85
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   25  -0.1942 -0.3586 0.0
UNIT 86
COM='DISK OPENING TOP BORAL SHEET TUBE - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   26  -0.3586 -0.0297 0.0
UNIT 87
COM='DISK OPENING TOP BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   27  -0.3586 -0.3586 0.0
UNIT 88
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   28  -0.3586 -0.3586 0.0
UNIT 89
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   29  -0.1942 -0.3586 0.0
UNIT 90
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   30  -0.0297 -0.3586 0.0
UNIT 91
COM='DISK OPENING NO BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   31  -0.3586 -0.3586 0.0
UNIT 100
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (TR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   40  -0.0297 -0.0297 0.0
UNIT 101
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   41  -0.3586 -0.0297 0.0
UNIT 102
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   42  -0.3586 -0.3586 0.0
UNIT 103
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   43  -0.0297 -0.3586 0.0
UNIT 104
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (T)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   44  -0.1942 -0.0297 0.0
UNIT 105
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   45  -0.1942 -0.3586 0.0
UNIT 106
COM='DISK OPENING TOP BORAL SHEET TUBE - STEEL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   46  -0.3586 -0.0297 0.0
UNIT 107
COM='DISK OPENING TOP BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   47  -0.3586 -0.3586 0.0
UNIT 108
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   48  -0.3586 -0.3586 0.0
UNIT 109
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   49  -0.1942 -0.3586 0.0
UNIT 110
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   50  -0.0297 -0.3586 0.0
UNIT 111
COM='DISK OPENING NO BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   51  -0.3586 -0.3586 0.0
UNIT 120
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (TR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   60  -0.0297 -0.0297 0.0
UNIT 121
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   61  -0.3586 -0.0297 0.0
UNIT 122
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   62  -0.3586 -0.3586 0.0
UNIT 123
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   63  -0.0297 -0.3586 0.0
UNIT 124
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (T)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   64  -0.1942 -0.0297 0.0
UNIT 125
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (B)'
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Figure 6.8-7 (continued)

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CUBOID 3 1 4P7.9731 2P0.6350
HOLE   65  -0.1942 -0.3586 0.0
UNIT 126
COM='DISK OPENING TOP BORAL SHEET TUBE - AL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   66  -0.3586 -0.0297 0.0
UNIT 127
COM='DISK OPENING TOP BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   67  -0.3586 -0.3586 0.0
UNIT 128
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   68  -0.3586 -0.3586 0.0
UNIT 129
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   69  -0.1942 -0.3586 0.0
UNIT 130
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   70  -0.0297 -0.3586 0.0
UNIT 131
COM='DISK OPENING NO BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   71  -0.3586 -0.3586 0.0
UNIT 140
COM='BASKET STRUCTURE IN TRANPORT CASK - WATER DISK'
CYLINDER 3 1 +83.5787 2P1.7145
HOLE 90  -70.3885  +8.7986 0.0
HOLE 83  -52.7914  +8.7986 0.0
HOLE 83  -52.7914 +26.3957 0.0
HOLE 90  -52.7914 +43.9928 0.0
HOLE 83  -35.1942  +8.7986 0.0
HOLE 83  -35.1942 +26.3957 0.0
HOLE 83  -35.1942 +43.9928 0.0
HOLE 90  -35.1942 +61.5899 0.0
HOLE 83  -17.5971  +8.7986 0.0
HOLE 83  -17.5971 +26.3957 0.0
HOLE 83  -17.5971 +43.9928 0.0
HOLE 90  -17.5971 +61.5899 0.0
HOLE 85   0.0      +8.7986 0.0
HOLE 85   0.0     +26.3957 0.0
HOLE 85   0.0     +43.9928 0.0
HOLE 89   0.0     +61.5899 0.0
HOLE 82  +17.5971  +8.7986 0.0
HOLE 82  +17.5971 +26.3957 0.0
HOLE 82  +17.5971 +43.9928 0.0
HOLE 88  +17.5971 +61.5899 0.0
HOLE 82  +35.1942  +8.7986 0.0
HOLE 82  +35.1942 +26.3957 0.0
HOLE 82  +35.1942 +43.9928 0.0
HOLE 91  +35.1942 +61.5899 0.0
HOLE 82  +52.7914  +8.7986 0.0
HOLE 87  +52.7914 +26.3957 0.0
HOLE 91  +52.7914 +43.9928 0.0
HOLE 91  +70.3885  +8.7986 0.0
HOLE 80  -70.3885  -8.7986 0.0
HOLE 80  -52.7914  -8.7986 0.0
HOLE 80  -52.7914 -26.3957 0.0
HOLE 80  -52.7914 -43.9928 0.0
HOLE 80  -35.1942  -8.7986 0.0
HOLE 80  -35.1942 -26.3957 0.0
HOLE 80  -35.1942 -43.9928 0.0
HOLE 80  -35.1942 -61.5899 0.0
HOLE 80  -17.5971  -8.7986 0.0
HOLE 80  -17.5971 -26.3957 0.0
HOLE 80  -17.5971 -43.9928 0.0
HOLE 80  -17.5971 -61.5899 0.0
HOLE 84   0.0      -8.7986 0.0
HOLE 84   0.0     -26.3957 0.0
HOLE 84   0.0     -43.9928 0.0
HOLE 84   0.0     -61.5899 0.0
HOLE 81  +17.5971  -8.7986 0.0
HOLE 81  +17.5971 -26.3957 0.0
HOLE 81  +17.5971 -43.9928 0.0
HOLE 81  +17.5971 -61.5899 0.0
HOLE 81  +35.1942  -8.7986 0.0
HOLE 81  +35.1942 -26.3957 0.0
HOLE 81  +35.1942 -43.9928 0.0
HOLE 86  +35.1942 -61.5899 0.0
HOLE 81  +52.7914  -8.7986 0.0
HOLE 86  +52.7914 -26.3957 0.0
HOLE 86  +52.7914 -43.9928 0.0
HOLE 86  +70.3885  -8.7986 0.0
CYLINDER  5 1  +85.1662 2P1.7145
CYLINDER  9 1  +94.615 2P1.7145
CYLINDER  7 1 +100.965 2P1.7145
CYLINDER  8 1 +172.72  2P1.7145
CUBOID    9 1 4P230.0  2P1.7145
UNIT 141
COM='BASKET STRUCTURE IN TRANPORT CASK - SS DISK'
CYLINDER 7 1  +83.1850 2P0.7938
HOLE 110  -70.3885  +8.7986 0.0
HOLE 103  -52.7914  +8.7986 0.0
HOLE 103  -52.7914 +26.3957 0.0
HOLE 110  -52.7914 +43.9928 0.0
HOLE 103  -35.1942  +8.7986 0.0
```

Figure 6.8-7 (continued)

```
HOLE 103  -35.1942 +26.3957 0.0
HOLE 103  -35.1942 +43.9928 0.0
HOLE 110  -35.1942 +61.5899 0.0
HOLE 103  -17.5971  +8.7986 0.0
HOLE 103  -17.5971 +26.3957 0.0
HOLE 103  -17.5971 +43.9928 0.0
HOLE 110  -17.5971 +61.5899 0.0
HOLE 105   0.0      +8.7986 0.0
HOLE 105   0.0     +26.3957 0.0
HOLE 105   0.0     +43.9928 0.0
HOLE 109   0.0     +61.5899 0.0
HOLE 102  +17.5971  +8.7986 0.0
HOLE 102  +17.5971 +26.3957 0.0
HOLE 102  +17.5971 +43.9928 0.0
HOLE 108  +17.5971 +61.5899 0.0
HOLE 102  +35.1942  +8.7986 0.0
HOLE 102  +35.1942 +26.3957 0.0
HOLE 102  +35.1942 +43.9928 0.0
HOLE 111  +35.1942 +61.5899 0.0
HOLE 102  +52.7914  +8.7986 0.0
HOLE 107  +52.7914 +26.3957 0.0
HOLE 111  +52.7914 +43.9928 0.0
HOLE 111  +70.3885  +8.7986 0.0
HOLE 100  -70.3885  -8.7986 0.0
HOLE 100  -52.7914  -8.7986 0.0
HOLE 100  -52.7914 -26.3957 0.0
HOLE 100  -52.7914 -43.9928 0.0
HOLE 100  -35.1942  -8.7986 0.0
HOLE 100  -35.1942 -26.3957 0.0
HOLE 100  -35.1942 -43.9928 0.0
HOLE 100  -35.1942 -61.5899 0.0
HOLE 100  -17.5971  -8.7986 0.0
HOLE 100  -17.5971 -26.3957 0.0
HOLE 100  -17.5971 -43.9928 0.0
HOLE 100  -17.5971 -61.5899 0.0
HOLE 104   0.0      -8.7986 0.0
HOLE 104   0.0     -26.3957 0.0
HOLE 104   0.0     -43.9928 0.0
HOLE 104   0.0     -61.5899 0.0
HOLE 101  +17.5971  -8.7986 0.0
HOLE 101  +17.5971 -26.3957 0.0
HOLE 101  +17.5971 -43.9928 0.0
HOLE 101  +17.5971 -61.5899 0.0
HOLE 101  +35.1942  -8.7986 0.0
HOLE 101  +35.1942 -26.3957 0.0
HOLE 101  +35.1942 -43.9928 0.0
HOLE 106  +35.1942 -61.5899 0.0
HOLE 101  +52.7914  -8.7986 0.0
HOLE 106  +52.7914 -26.3957 0.0
HOLE 106  +52.7914 -43.9928 0.0
HOLE 106  +70.3885  -8.7986 0.0
CYLINDER  3 1  +83.5787 2P0.7938
CYLINDER  5 1  +85.1662 2P0.7938
CYLINDER  9 1  +94.615 2P0.7938
CYLINDER  7 1 +100.965 2P0.7938
CYLINDER  8 1 +172.72  2P0.7938
CUBOID    9 1 4P230.0  2P0.7938
UNIT 142
COM='BASKET STRUCTURE IN TRANPORT CASK - AL DISK'
CYLINDER 4 1  +82.8675 2P0.6350
HOLE 130  -70.3885  +8.7986 0.0
HOLE 123  -52.7914  +8.7986 0.0
HOLE 123  -52.7914 +26.3957 0.0
HOLE 130  -52.7914 +43.9928 0.0
HOLE 123  -35.1942  +8.7986 0.0
HOLE 123  -35.1942 +26.3957 0.0
HOLE 123  -35.1942 +43.9928 0.0
HOLE 130  -35.1942 +61.5899 0.0
HOLE 123  -17.5971  +8.7986 0.0
HOLE 123  -17.5971 +26.3957 0.0
HOLE 123  -17.5971 +43.9928 0.0
HOLE 130  -17.5971 +61.5899 0.0
HOLE 125   0.0      +8.7986 0.0
HOLE 125   0.0     +26.3957 0.0
HOLE 125   0.0     +43.9928 0.0
HOLE 129   0.0     +61.5899 0.0
HOLE 122  +17.5971  +8.7986 0.0
HOLE 122  +17.5971 +26.3957 0.0
HOLE 122  +17.5971 +43.9928 0.0
HOLE 128  +17.5971 +61.5899 0.0
HOLE 122  +35.1942  +8.7986 0.0
HOLE 122  +35.1942 +26.3957 0.0
HOLE 122  +35.1942 +43.9928 0.0
HOLE 131  +35.1942 +61.5899 0.0
HOLE 122  +52.7914  +8.7986 0.0
HOLE 127  +52.7914 +26.3957 0.0
HOLE 131  +52.7914 +43.9928 0.0
HOLE 131  +70.3885  +8.7986 0.0
HOLE 120  -70.3885  -8.7986 0.0
HOLE 120  -52.7914  -8.7986 0.0
HOLE 120  -52.7914 -26.3957 0.0
HOLE 120  -52.7914 -43.9928 0.0
HOLE 120  -35.1942  -8.7986 0.0
HOLE 120  -35.1942 -26.3957 0.0
HOLE 120  -35.1942 -43.9928 0.0
HOLE 120  -35.1942 -61.5899 0.0
HOLE 120  -17.5971  -8.7986 0.0
HOLE 120  -17.5971 -26.3957 0.0
```

Figure 6.8-7 (continued)

```
HOLE 120  -17.5971 -43.9928 0.0
HOLE 120  -17.5971 -61.5899 0.0
HOLE 124   0.0      -8.7986 0.0
HOLE 124   0.0     -26.3957 0.0
HOLE 124   0.0     -43.9928 0.0
HOLE 124   0.0     -61.5899 0.0
HOLE 121  +17.5971  -8.7986 0.0
HOLE 121  +17.5971 -26.3957 0.0
HOLE 121  +17.5971 -43.9928 0.0
HOLE 121  +17.5971 -61.5899 0.0
HOLE 121  +35.1942  -8.7986 0.0
HOLE 121  +35.1942 -26.3957 0.0
HOLE 121  +35.1942 -43.9928 0.0
HOLE 126  +35.1942 -61.5899 0.0
HOLE 121  +52.7914  -8.7986 0.0
HOLE 126  +52.7914 -26.3957 0.0
HOLE 126  +52.7914 -43.9928 0.0
HOLE 126  +70.3885  -8.7986 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER  9 1  +94.615 2P0.6350
CYLINDER  7 1 +100.965 2P0.6350
CYLINDER  8 1 +172.72  2P0.6350
CUBOID    9 1 4P230.0  2P0.6350
GLOBAL UNIT 143
COM='CASK SLICES TOGETHER'
ARRAY  4 -230.00 -230.00 0.0
END GEOM
READ ARRAY
ARA=1   NUX=9 NUY=9 NUZ=1  FILL
    36R1
 4R1  2   4R1
 5R1  2   3R1
    27R1
END FILL
ARA=2   NUX=9 NUY=9 NUZ=1  FILL
    36R3
 4R3  4   4R3
 5R3  4   3R3
    27R3
END FILL
ARA=3   NUX=9 NUY=9 NUZ=1  FILL
    36R5
 4R5  6   4R5
 5R5  6   3R5
    27R5
END FILL
ARA=4 NUX=1 NUY=1 NUZ=4 FILL 140 141 140 142 END FILL
END ARRAY
READ BOUNDS ZFC=PER YXF=PER END BOUNDS
END DATA
END
```

Figure 6.8-8 CSAS Input for Accident Conditions–Vertical Concrete Cask Containing BWR Fuel

```
=CSAS25
UMS BWR VCC; ACCIDENT; CASK ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 75%B10
27GROUPNDF4 LATTICECELL
UO2   1 0.95 293.0 92235 4.00 92238 96.00 END
ZIRCALLOY  2 1.0  293.0  END
H2O   3 1.0  293.0  END
AL    4 1.0  293.0  END
SS304 5 1.0  293.0  END
AL    6 DEN=2.6849 0.8706 293.O END
B-10  6 DEN=2.6849 0.0137 293.O END
B-11  6 DEN=2.6849 0.0830 293.O END
C     6 DEN=2.6849 0.0281 293.O END
CARBONSTEEL 7 1.0 293.0 END
REG-CONCRETE 8 0.9750 293.0 END
H2O   9 1.0 293.0 END
H2O  10 1.0 293.0 END
END COMP
SQUAREPITCH  1.4529 0.9055 1 3 1.0770 2 0.9246 10 END
UMS BWR VCC; ACCIDENT; CASK ARRAY; 1.0 GM/CC IN - 1.0 GM/CC EX; 75%B10
READ PARAM RUN=YES PLT=NO TME=5000 GEN=803 NPG=1000 END PARAM
READ GEOM
UNIT  1
COM='FUEL PIN CELL - WITH H2O'
CYLINDER  1 1  0.4528   2P1.7145
CYLINDER 10 1  0.4623   2P1.7145
CYLINDER  2 1  0.5385   2P1.7145
CUBOID    3 1  4P0.7264 2P1.7145
UNIT  2
COM='WATER ROD CELL - WITH H2O'
CYLINDER  3 1  0.4623   2P1.7145
CYLINDER  2 1  0.5385   2P1.7145
CUBOID    3 1  4P0.7264 2P1.7145
UNIT  3
COM='FUEL PIN CELL - WITH ST DISK'
CYLINDER  1 1  0.4528   2P0.7938
CYLINDER 10 1  0.4623   2P0.7938
CYLINDER  2 1  0.5385   2P0.7938
CUBOID    3 1  4P0.7264 2P0.7938
UNIT  4
COM='WATER ROD CELL - WITH ST DISK'
CYLINDER  3 1  0.4623   2P0.7938
CYLINDER  2 1  0.5385   2P0.7938
CUBOID    3 1  4P0.7264 2P0.7938
UNIT  5
COM='FUEL PIN CELL - WITH AL DISK'
CYLINDER  1 1  0.4528   2P0.6350
CYLINDER 10 1  0.4623   2P0.6350
CYLINDER  2 1  0.5385   2P0.6350
CUBOID    3 1  4P0.7264 2P0.6350
UNIT  6
COM='WATER ROD CELL - WITH AL DISK'
CYLINDER  3 1  0.4623   2P0.6350
CYLINDER  2 1  0.5385   2P0.6350
CUBOID    3 1  4P0.7264 2P0.6350
UNIT  7
COM='FUEL PIN ARRAY + CHANNEL - BETWEEN DISKS'
ARRAY  1   -6.5376 -6.5376 -1.7145
CUBOID 3 1 4P6.7031 2P1.7145
CUBOID 2 1 4P6.9063 2P1.7145
UNIT  8
COM='FUEL PIN ARRAY + CHANNEL - ST DISKS'
ARRAY  2   -6.5376 -6.5376 -0.7938
CUBOID 3 1 4P6.7031 2P0.7938
CUBOID 2 1 4P6.9063 2P0.7938
UNIT  9
COM='FUEL PIN ARRAY + CHANNEL - AL DISKS'
ARRAY  3   -6.5376 -6.5376 -0.6350
CUBOID 3 1 4P6.7031 2P0.6350
CUBOID 2 1 4P6.9063 2P0.6350
UNIT 10
COM='X-X BORAL + COVER SHEET BETWEEN DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P1.7145
CUBOID 4 1 2P6.7310 2P0.1714 2P1.7145
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P1.7145
UNIT 11
COM='Y-Y BORAL + COVER SHEET BETWEEN DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P1.7145
CUBOID 4 1 2P0.1714 2P6.7310 2P1.7145
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P1.7145
UNIT 12
COM='X-X BORAL + COVER SHEET WITH ST DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P0.7938
CUBOID 4 1 2P6.7310 2P0.1714 2P0.7938
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P0.7938
UNIT 13
COM='Y-Y BORAL + COVER SHEET WITH ST DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P0.7938
CUBOID 4 1 2P0.1714 2P6.7310 2P0.7938
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P0.7938
```

Figure 6.8-8 (continued)

```
UNIT 14
COM='X-X BORAL + COVER SHEET WITH AL DISKS'
CUBOID 6 1 2P6.7310 2P0.1124 2P0.6350
CUBOID 4 1 2P6.7310 2P0.1714 2P0.6350
CUBOID 5 1 2P6.7765 +0.2168 -0.1714 2P0.6350
UNIT 15
COM='Y-Y BORAL + COVER SHEET WITH AL DISKS'
CUBOID 6 1 2P0.1124 2P6.7310 2P0.6350
CUBOID 4 1 2P0.1714 2P6.7310 2P0.6350
CUBOID 5 1 +0.2168 -0.1714 2P6.7765 2P0.6350
UNIT 20
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (TR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 21
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 22
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 23
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 24
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (T)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 25
COM='FUEL TUBE CELL 2 BORAL SHEETS - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
HOLE   11 +7.7859 0.0 0.0
UNIT 26
COM='FUEL TUBE CELL TOP BORAL SHEETS - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
UNIT 27
COM='FUEL TUBE CELL TOP BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P1.7145
HOLE   10 0.0 +7.7859 0.0
UNIT 28
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 29
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 30
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P1.7145
HOLE   11 +7.7859 0.0 0.0
UNIT 31
COM='FUEL TUBE CELL NO BORAL SHEETS - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.4930 2P1.7145
HOLE   7 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P1.7145
UNIT 40
```

Figure 6.8-8 (continued)

```
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (TR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 41
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 42
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 43
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 44
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (T)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 45
COM='FUEL TUBE CELL 2 BORAL SHEETS - STEEL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
HOLE   13 +7.7859 0.0 0.0
UNIT 46
COM='FUEL TUBE CELL TOP BORAL SHEETS - STEEL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
UNIT 47
COM='FUEL TUBE CELL TOP BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.7938
HOLE   12 0.0 +7.7859 0.0
UNIT 48
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 49
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 50
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - STEEL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.7938
HOLE   13 +7.7859 0.0 0.0
UNIT 51
COM='FUEL TUBE CELL NO BORAL SHEETS - STEEL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.7938
HOLE   8 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.7938
UNIT 60
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (TR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 61
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 +0.5867 0.0
```

Figure 6.8-8 (continued)

```
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 62
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 63
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 64
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (T)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 65
COM='FUEL TUBE CELL 2 BORAL SHEETS - AL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
HOLE   15 +7.7859 0.0 0.0
UNIT 66
COM='FUEL TUBE CELL TOP BORAL SHEETS - AL DISKS (TL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 +0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
UNIT 67
COM='FUEL TUBE CELL TOP BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +7.6144 -7.6144 +8.0028 -7.6144 +2P0.6350
HOLE   14 0.0 +7.7859 0.0
UNIT 68
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 69
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (B)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 0.0 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 70
COM='FUEL TUBE CELL RIGHT BORAL SHEETS - AL DISKS (BR)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 +0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
CUBOID 3 1 +8.0028 -7.6144 +7.6144 -7.6144 +2P0.6350
HOLE   15 +7.7859 0.0 0.0
UNIT 71
COM='FUEL TUBE CELL NO BORAL SHEETS - AL DISKS (BL)'
CUBOID 3 1 4P7.4930 2P0.6350
HOLE   9 -0.5867 -0.5867 0.0
CUBOID 5 1 4P7.6144 2P0.6350
UNIT 80
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (TR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   20  -0.0297 -0.0297 0.0
UNIT 81
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   21  -0.3586 -0.0297 0.0
UNIT 82
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   22  -0.3586 -0.3586 0.0
UNIT 83
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   23  -0.0297 -0.3586 0.0
UNIT 84
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (T)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   24  -0.1942 -0.0297 0.0
UNIT 85
COM='DISK OPENING 2 BORAL SHEET TUBE - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.9731 2P1.7145
```

Figure 6.8-8 (continued)

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HOLE   25  -0.1942 -0.3586 0.0
UNIT 86
COM='DISK OPENING TOP BORAL SHEET TUBE - BETWEEN DISKS (TL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   26  -0.3586 -0.0297 0.0
UNIT 87
COM='DISK OPENING TOP BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   27  -0.3586 -0.3586 0.0
UNIT 88
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   28  -0.3586 -0.3586 0.0
UNIT 89
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (B)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   29  -0.1942 -0.3586 0.0
UNIT 90
COM='DISK OPENING RIGHT BORAL SHEET TUBE - BETWEEN DISKS (BR)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   30  -0.0297 -0.3586 0.0
UNIT 91
COM='DISK OPENING NO BORAL SHEET TUBE - BETWEEN DISKS (BL)'
CUBOID 3 1 4P7.9731 2P1.7145
HOLE   31  -0.3586 -0.3586 0.0
UNIT 100
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (TR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   40  -0.0297 -0.0297 0.0
UNIT 101
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   41  -0.3586 -0.0297 0.0
UNIT 102
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   42  -0.3586 -0.3586 0.0
UNIT 103
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   43  -0.0297 -0.3586 0.0
UNIT 104
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (T)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   44  -0.1942 -0.0297 0.0
UNIT 105
COM='DISK OPENING 2 BORAL SHEET TUBE - STEEL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   45  -0.1942 -0.3586 0.0
UNIT 106
COM='DISK OPENING TOP BORAL SHEET TUBE - STEEL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   46  -0.3586 -0.0297 0.0
UNIT 107
COM='DISK OPENING TOP BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   47  -0.3586 -0.3586 0.0
UNIT 108
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   48  -0.3586 -0.3586 0.0
UNIT 109
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   49  -0.1942 -0.3586 0.0
UNIT 110
COM='DISK OPENING RIGHT BORAL SHEET TUBE - STEEL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   50  -0.0297 -0.3586 0.0
UNIT 111
COM='DISK OPENING NO BORAL SHEET TUBE - STEEL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.7938
HOLE   51  -0.3586 -0.3586 0.0
UNIT 120
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (TR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   60  -0.0297 -0.0297 0.0
UNIT 121
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (TL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   61  -0.3586 -0.0297 0.0
UNIT 122
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   62  -0.3586 -0.3586 0.0
UNIT 123
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   63  -0.0297 -0.3586 0.0
UNIT 124
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (T)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   64  -0.1942 -0.0297 0.0
UNIT 125
COM='DISK OPENING 2 BORAL SHEET TUBE - AL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   65  -0.1942 -0.3586 0.0
UNIT 126
COM='DISK OPENING TOP BORAL SHEET TUBE - AL DISKS (TL)'
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Figure 6.8-8 (continued)

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CUBOID 3 1 4P7.9731 2P0.6350
HOLE   66  -0.3586 -0.0297 0.0
UNIT 127
COM='DISK OPENING TOP BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   67  -0.3586 -0.3586 0.0
UNIT 128
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   68  -0.3586 -0.3586 0.0
UNIT 129
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (B)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   69  -0.1942 -0.3586 0.0
UNIT 130
COM='DISK OPENING RIGHT BORAL SHEET TUBE - AL DISKS (BR)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   70  -0.0297 -0.3586 0.0
UNIT 131
COM='DISK OPENING NO BORAL SHEET TUBE - AL DISKS (BL)'
CUBOID 3 1 4P7.9731 2P0.6350
HOLE   71  -0.3586 -0.3586 0.0
UNIT 140
COM='BASKET STRUCTURE IN TRANPORT CASK - WATER DISK'
CYLINDER 3 1 +83.5787 2P1.7145
HOLE 90  -70.3885  +8.7986 0.0
HOLE 83  -52.7914  +8.7986 0.0
HOLE 83  -52.7914 +26.3957 0.0
HOLE 90  -52.7914 +43.9928 0.0
HOLE 83  -35.1942  +8.7986 0.0
HOLE 83  -35.1942 +26.3957 0.0
HOLE 83  -35.1942 +43.9928 0.0
HOLE 90  -35.1942 +61.5899 0.0
HOLE 83  -17.5971  +8.7986 0.0
HOLE 83  -17.5971 +26.3957 0.0
HOLE 83  -17.5971 +43.9928 0.0
HOLE 90  -17.5971 +61.5899 0.0
HOLE 85   0.0      +8.7986 0.0
HOLE 85   0.0     +26.3957 0.0
HOLE 85   0.0     +43.9928 0.0
HOLE 89   0.0     +61.5899 0.0
HOLE 82  +17.5971  +8.7986 0.0
HOLE 82  +17.5971 +26.3957 0.0
HOLE 82  +17.5971 +43.9928 0.0
HOLE 88  +17.5971 +61.5899 0.0
HOLE 82  +35.1942  +8.7986 0.0
HOLE 82  +35.1942 +26.3957 0.0
HOLE 82  +35.1942 +43.9928 0.0
HOLE 91  +35.1942 +61.5899 0.0
HOLE 82  +52.7914  +8.7986 0.0
HOLE 87  +52.7914 +26.3957 0.0
HOLE 91  +52.7914 +43.9928 0.0
HOLE 91  +70.3885  +8.7986 0.0
HOLE 80  -70.3885  -8.7986 0.0
HOLE 80  -52.7914  -8.7986 0.0
HOLE 80  -52.7914 -26.3957 0.0
HOLE 80  -52.7914 -43.9928 0.0
HOLE 80  -35.1942  -8.7986 0.0
HOLE 80  -35.1942 -26.3957 0.0
HOLE 80  -35.1942 -43.9928 0.0
HOLE 80  -35.1942 -61.5899 0.0
HOLE 80  -17.5971  -8.7986 0.0
HOLE 80  -17.5971 -26.3957 0.0
HOLE 80  -17.5971 -43.9928 0.0
HOLE 80  -17.5971 -61.5899 0.0
HOLE 84   0.0      -8.7986 0.0
HOLE 84   0.0     -26.3957 0.0
HOLE 84   0.0     -43.9928 0.0
HOLE 84   0.0     -61.5899 0.0
HOLE 81  +17.5971  -8.7986 0.0
HOLE 81  +17.5971 -26.3957 0.0
HOLE 81  +17.5971 -43.9928 0.0
HOLE 81  +17.5971 -61.5899 0.0
HOLE 81  +35.1942  -8.7986 0.0
HOLE 81  +35.1942 -26.3957 0.0
HOLE 81  +35.1942 -43.9928 0.0
HOLE 86  +35.1942 -61.5899 0.0
HOLE 81  +52.7914  -8.7986 0.0
HOLE 86  +52.7914 -26.3957 0.0
HOLE 86  +52.7914 -43.9928 0.0
HOLE 86  +70.3885  -8.7986 0.0
CYLINDER  5 1  +85.1662 2P1.7145
CYLINDER  9 1  +94.615 2P1.7145
CYLINDER  7 1 +100.965 2P1.7145
CYLINDER  8 1 +172.72  2P1.7145
CUBOID    9 1 4P230.0  2P1.7145
UNIT 141
COM='BASKET STRUCTURE IN TRANPORT CASK - SS DISK'
CYLINDER 7 1  +83.1850 2P0.7938
HOLE 110  -70.3885  +8.7986 0.0
HOLE 103  -52.7914  +8.7986 0.0
HOLE 103  -52.7914 +26.3957 0.0
HOLE 110  -52.7914 +43.9928 0.0
HOLE 103  -35.1942  +8.7986 0.0
HOLE 103  -35.1942 +26.3957 0.0
HOLE 103  -35.1942 +43.9928 0.0
HOLE 110  -35.1942 +61.5899 0.0
HOLE 103  -17.5971  +8.7986 0.0
HOLE 103  -17.5971 +26.3957 0.0
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Figure 6.8-8 (continued)

```
HOLE 103  -17.5971 +43.9928 0.0
HOLE 110  -17.5971 +61.5899 0.0
HOLE 105   0.0      +8.7986 0.0
HOLE 105   0.0     +26.3957 0.0
HOLE 105   0.0     +43.9928 0.0
HOLE 109   0.0     +61.5899 0.0
HOLE 102  +17.5971  +8.7986 0.0
HOLE 102  +17.5971 +26.3957 0.0
HOLE 102  +17.5971 +43.9928 0.0
HOLE 108  +17.5971 +61.5899 0.0
HOLE 102  +35.1942  +8.7986 0.0
HOLE 102  +35.1942 +26.3957 0.0
HOLE 102  +35.1942 +43.9928 0.0
HOLE 111  +35.1942 +61.5899 0.0
HOLE 102  +52.7914  +8.7986 0.0
HOLE 107  +52.7914 +26.3957 0.0
HOLE 111  +52.7914 +43.9928 0.0
HOLE 111  +70.3885  +8.7986 0.0
HOLE 100  -70.3885  -8.7986 0.0
HOLE 100  -52.7914  -8.7986 0.0
HOLE 100  -52.7914 -26.3957 0.0
HOLE 100  -52.7914 -43.9928 0.0
HOLE 100  -35.1942  -8.7986 0.0
HOLE 100  -35.1942 -26.3957 0.0
HOLE 100  -35.1942 -43.9928 0.0
HOLE 100  -35.1942 -61.5899 0.0
HOLE 100  -17.5971  -8.7986 0.0
HOLE 100  -17.5971 -26.3957 0.0
HOLE 100  -17.5971 -43.9928 0.0
HOLE 100  -17.5971 -61.5899 0.0
HOLE 104   0.0      -8.7986 0.0
HOLE 104   0.0     -26.3957 0.0
HOLE 104   0.0     -43.9928 0.0
HOLE 104   0.0     -61.5899 0.0
HOLE 101  +17.5971  -8.7986 0.0
HOLE 101  +17.5971 -26.3957 0.0
HOLE 101  +17.5971 -43.9928 0.0
HOLE 101  +17.5971 -61.5899 0.0
HOLE 101  +35.1942  -8.7986 0.0
HOLE 101  +35.1942 -26.3957 0.0
HOLE 101  +35.1942 -43.9928 0.0
HOLE 106  +35.1942 -61.5899 0.0
HOLE 101  +52.7914  -8.7986 0.0
HOLE 106  +52.7914 -26.3957 0.0
HOLE 106  +52.7914 -43.9928 0.0
HOLE 106  +70.3885  -8.7986 0.0
CYLINDER  3 1  +83.5787 2P0.7938
CYLINDER  5 1  +85.1662 2P0.7938
CYLINDER  9 1  +94.615 2P0.7938
CYLINDER  7 1 +100.965 2P0.7938
CYLINDER  8 1 +172.72  2P0.7938
CUBOID    9 1 4P230.0  2P0.7938
UNIT 142
COM='BASKET STRUCTURE IN TRANPORT CASK - AL DISK'
CYLINDER 4 1  +82.8675 2P0.6350
HOLE 130  -70.3885  +8.7986 0.0
HOLE 123  -52.7914  +8.7986 0.0
HOLE 123  -52.7914 +26.3957 0.0
HOLE 130  -52.7914 +43.9928 0.0
HOLE 123  -35.1942  +8.7986 0.0
HOLE 123  -35.1942 +26.3957 0.0
HOLE 123  -35.1942 +43.9928 0.0
HOLE 130  -35.1942 +61.5899 0.0
HOLE 123  -17.5971  +8.7986 0.0
HOLE 123  -17.5971 +26.3957 0.0
HOLE 123  -17.5971 +43.9928 0.0
HOLE 130  -17.5971 +61.5899 0.0
HOLE 125   0.0      +8.7986 0.0
HOLE 125   0.0     +26.3957 0.0
HOLE 125   0.0     +43.9928 0.0
HOLE 129   0.0     +61.5899 0.0
HOLE 122  +17.5971  +8.7986 0.0
HOLE 122  +17.5971 +26.3957 0.0
HOLE 122  +17.5971 +43.9928 0.0
HOLE 128  +17.5971 +61.5899 0.0
HOLE 122  +35.1942  +8.7986 0.0
HOLE 122  +35.1942 +26.3957 0.0
HOLE 122  +35.1942 +43.9928 0.0
HOLE 131  +35.1942 +61.5899 0.0
HOLE 122  +52.7914  +8.7986 0.0
HOLE 127  +52.7914 +26.3957 0.0
HOLE 131  +52.7914 +43.9928 0.0
HOLE 131  +70.3885  +8.7986 0.0
HOLE 120  -70.3885  -8.7986 0.0
HOLE 120  -52.7914  -8.7986 0.0
HOLE 120  -52.7914 -26.3957 0.0
HOLE 120  -52.7914 -43.9928 0.0
HOLE 120  -35.1942  -8.7986 0.0
HOLE 120  -35.1942 -26.3957 0.0
HOLE 120  -35.1942 -43.9928 0.0
HOLE 120  -35.1942 -61.5899 0.0
HOLE 120  -17.5971  -8.7986 0.0
HOLE 120  -17.5971 -26.3957 0.0
HOLE 120  -17.5971 -43.9928 0.0
HOLE 120  -17.5971 -61.5899 0.0
HOLE 124   0.0      -8.7986 0.0
HOLE 124   0.0     -26.3957 0.0
HOLE 124   0.0     -43.9928 0.0
HOLE 124   0.0     -61.5899 0.0
```

Figure 6.8-8 (continued)

```
HOLE 121  +17.5971  -8.7986 0.0
HOLE 121  +17.5971 -26.3957 0.0
HOLE 121  +17.5971 -43.9928 0.0
HOLE 121  +17.5971 -61.5899 0.0
HOLE 121  +35.1942  -8.7986 0.0
HOLE 121  +35.1942 -26.3957 0.0
HOLE 121  +35.1942 -43.9928 0.0
HOLE 126  +35.1942 -61.5899 0.0
HOLE 121  +52.7914  -8.7986 0.0
HOLE 126  +52.7914 -26.3957 0.0
HOLE 126  +52.7914 -43.9928 0.0
HOLE 126  +70.3885  -8.7986 0.0
CYLINDER  3 1  +83.5787 2P0.6350
CYLINDER  5 1  +85.1662 2P0.6350
CYLINDER  9 1  +94.615 2P0.6350
CYLINDER  7 1 +100.965 2P0.6350
CYLINDER  8 1 +172.72  2P0.6350
CUBOID    9 1 4P230.0  2P0.6350
GLOBAL UNIT 143
COM='CASK SLICES TOGETHER'
ARRAY  4 -230.00 -230.00 0.0
END GEOM
READ ARRAY
ARA=1   NUX=9 NUY=9 NUZ=1  FILL
    36R1
 4R1  2   4R1
 5R1  2   3R1
    27R1
END FILL
ARA=2   NUX=9 NUY=9 NUZ=1  FILL
    36R3
 4R3  4   4R3
 5R3  4   3R3
    27R3
END FILL
ARA=3   NUX=9 NUY=9 NUZ=1  FILL
    36R5
 4R5  6   4R5
 5R5  6   3R5
    27R5
END FILL
ARA=4 NUX=1 NUY=1 NUZ=4 FILL 140 141 140 142 END FILL
END ARRAY
READ BOUNDS ZFC=PER YXF=PER END BOUNDS
END DATA
END
```

Figure 6.8-9 MONK8A Input for PWR Transfer Cask with Soluble Boron

```
columns 1 200
*
*   UMS Transfer Cask - we17b Standard
*
*   Cask Lid Configurations
*           Shield Lid - No Ports
*           Structural Lid - No Weld Shield
*
*   Neutron Poison Loading -  75 %
*   Exterior Water Density 0.0001
*   Cavity Water Density 0.9998
*   Fuel to Clad Gap Water Density 0.9998
*
*   Boron Content in Water - 1000 ppm
*
*   Model Revision v3.0
*
* Parameters
*
@randseed =  12345
*
*
* Unit 1 Control Data
*
begin control data
*READ   ! read and check each independently
*SEEK MULTIPLE DEFINITIONS

SEEDS  @randseed  @randseed
STAGES -15 810 4000 STDV 0.0008

end
*
* Unit 9 Material Specification
*
begin material specification
normalise
nmixtures  7
weight  mixture  1
      u235  4.4072E-02
      u238  8.3737E-01
      o16  1.1856E-01
atoms  mixture  2
      h  6.6667E-01
      o16  3.3333E-01
atoms  mixture  3
      h  6.6667E-01
      o16  3.3333E-01
atoms  mixture  4
      h  4.2857E-01
      b  1.4286E-01
      o16  4.2857E-01
weight  mixture  5
      al  4.6148E-01
      b10  7.5880E-02
      b11  3.4567E-01
      c  1.1697E-01
atoms  mixture  6
      c  2.8571E-01
      h  4.7619E-01
      o16  2.3810E-01
weight  mixture  7
      h  4.2152E-02
      o16  5.4785E-01
      fe  4.7900E-02
      c  9.3500E-02
      si  3.3600E-02
      ca  5.6100E-02
      al  1.7890E-01
*
* Materials List - v1.2 - Class 1 - we17b - WE17 (OFA) Fuel
*
nmaterials  23
volume              ! UO2 at 5%
material  1
  mixture  1  density  10.4120  prop  1.00000
volume              ! Fuel pin cladding
material  2
  zircalloy    density  6.5500  prop  1.00000
volume              ! Water In Lattice and Tube
material  3
  mixture  4  density  1.0015  prop  0.00572  ! mixBoricAcid
  mixture  2  density  1.0015  prop  0.99428  ! mixH2O
volume              ! Water In Fuel Rod Clad Gap
material  4
  mixture  4  density  1.0015  prop  0.00572  ! mixBoricAcid
  mixture  2  density  1.0015  prop  0.99428  ! mixH2O
volume              ! Lower Nozzle Material
material  5
  stainless 304l steel    density  7.9200  prop  0.23669
  mixture  4  density  1.0015  prop  0.00437  ! mixBoricAcid
  mixture  2  density  1.0015  prop  0.75894  ! mixH2O
volume              ! Upper Nozzle Material
material  6
  stainless 304l steel    density  7.9200  prop  0.23180
  mixture  4  density  1.0015  prop  0.00439  ! mixBoricAcid
  mixture  2  density  1.0015  prop  0.76381  ! mixH2O
```

Figure 6.8-9 (continued)

```
*
* Materials List - Common Materials - v2.0
*
volume              ! Tube wall and cover sheet
material  7
  stainless 304l steel    density  7.9300  prop  1.0000
volume              ! BORAL core
material  8
  mixture  5  density  1.9457  prop  1.0000  ! mixBORAL
volume              ! BORAL alumnimum clad
material  9
  aluminium        prop  1.0000
volume              ! Structural Disk Material
material  10
  stainless 304l steel    density  7.9300  prop  1.0000
volume              ! Weldment Material
material  11
  stainless 304l steel    density  7.9300  prop  1.0000
volume              ! Heat Transfer Disk Material
material  12
  aluminium        prop  1.0000
volume              ! Canister Material
material  13
  stainless 304l steel    density  7.9300  prop  1.0000
atoms              ! Transfer steel
material  14    density  0      !  (SCALE carbon steel)
    fe      prop  8.3498E-02
    c      prop  3.9250E-03
volume              ! Lead
material  15
  pb    density  11.0400  prop  1.0000
atoms              ! NS-4-FR
material  16    density  0      ! 0 means atom/b-cm
    b10      prop  8.5500E-05
    b11      prop  3.4200E-04
    al      prop  7.8000E-03
    h      prop  5.8500E-02
    o16      prop  2.6100E-02
    c      prop  2.2600E-02
    n      prop  1.3900E-03
volume              ! Stainless Steel 304
material  17
  stainless 304l steel    density  7.9300  prop  1.0000
volume              ! Vent port middle cylinder
material  18
  stainless 304l steel    density  7.9300  prop  0.5000
  void        prop  0.5000
atoms              ! SCALE Concrete
material  19    density  0
    h      prop  1.3401E-02
    o16      prop  4.4931E-02
    na      prop  1.7036E-03
    al      prop  1.7018E-03
    si      prop  1.6205E-02
    ca      prop  1.4826E-03
    fe      prop  3.3857E-04
volume              ! Heat fins for transport cask
material  20
  cu    density  8.9200  prop  0.4286
  stainless 304l steel    density  7.9300  prop  0.5714
volume              ! Balsa
material  21
  mixture  6  density  0.1250  prop  1.0000
volume              ! Redwood
material  22
  mixture  6  density  0.3870  prop  1.0000
volume              ! NS3
material  23
  mixture  7  density  1.6507  prop  1.0000  ! Weight loss @ 200F of 2.90%
end

*
* Unit 2 Material Geometry
*
begin material geometry
* Fuel Rod - Class 1 - we17b - WE17 (OFA)
PART  1
ZROD  1    0.0000  0.0000  1.7399  0.3922  365.7600    ! Fuel pellet stack
ZROD  2    0.0000  0.0000  1.7399  0.4001  381.6604    ! Annulus + Plenum
ZROD  3    0.0000  0.0000  0.0000  0.4572  385.1402    ! Clad
ZROD  4    0.0000  0.0000  385.1402  0.0000  4.5720    ! Fuel rod to top nozzle
BOX  5    -0.6299  -0.6299  0.0000  1.2597  1.2597  389.7122  ! Pitch box
ZONES
/Fuel/    M1  +1
/Fuel to Clad Gap/    M4  +2  -1
/Clad & End Plugs/    M2  +3  -2
/Rod to Top Nozzle/    M2  +4
/Rod in Pitch/    M3  +5  -4  -3
* PWR Guide Tube - Class 1 - we17b - WE17 (OFA)
PART  2  NEST
ZROD  M3    0.0000  0.0000  0.0000  0.5740  365.7600    ! Guide tube interior
ZROD  M2    0.0000  0.0000  0.0000  0.6121  365.7600    ! Clad
BOX  M3    -0.6299  -0.6299  0.0000  1.2597  1.2597  389.7122  ! Pitch box
* PWR Instrument Tube - Class 1 - we17b - WE17 (OFA)
PART  3  NEST
ZROD  M3    0.0000  0.0000  0.0000  0.5740  365.7600    ! Inst. tube interior
ZROD  M2    0.0000  0.0000  0.0000  0.6121  365.7600    ! Clad
BOX  M3    -0.6299  -0.6299  0.0000  1.2597  1.2597  389.7122  ! Pitch box
* Array_17x17_264
```

Figure 6.8-9 (continued)

```
PART  4  ARRAY
17  17  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  2  1  1  2  1  1  2  1  1  1  1  1
1  1  1  2  1  1  1  1  1  1  1  1  1  2  1  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  2  1  1  2  1  1  2  1  1  2  1  1  2  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  2  1  1  2  1  1  3  1  1  2  1  1  2  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  2  1  1  2  1  1  2  1  1  2  1  1  2  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  2  1  1  1  1  1  1  1  1  1  2  1  1  1
1  1  1  1  1  2  1  1  2  1  1  2  1  1  1  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
* Fuel Assembly Array Inserted Into Assembly - Class 1 - we17b - WE17 (OFA)
PART  5  NEST
BOX  P4  -10.7075  -10.7075  6.8580  21.4149  21.4149  389.7122    ! Array
BOX  M3  -10.7086  -10.7086  6.8580  21.4173  21.4173  389.7122    ! Fuel Width Envelope
BOX  M5  -10.7086  -10.7086  0.0000  21.4173  21.4173  396.5702    ! Lower Nozzle
BOX  M6  -10.7086  -10.7086  0.0000  21.4173  21.4173  405.8920    ! Upper Nozzle - Envelope
* PWR Neutron Poison and Cover Sheet            Configuration  R
PART  6
BOX  1  -9.9009  0.0318  0.0508  20.7467  0.1270  382.2700  ! BORAL Core
BOX  2  -9.9009  0.0000  0.0508  20.7467  0.1905  382.2700  ! BORAL Clad
BOX  3  -10.8458  0.0000  0.0508  21.6916  0.1905  384.2004  ! Space under Cover Sheet
BOX  4  -10.8915  0.0000  0.0051  21.7830  0.2362  384.2918  ! Cover Sheet (top/side)
BOX  5  -10.8966  0.0000  0.0000  21.7932  0.0457  384.3020  ! Remaining Cover Sheet
BOX  6  -10.8966  0.0000  0.0000  21.7932  0.2362  384.3020  ! Container
ZONES
/BORAL Core/    M8  +1
/BORAL Clad/    M9  +2  -1
/Space Under Cover/    H5  +3  -2
/Enclosing Cover/    M7  +4  -3
/Remaining Cover/    M7  +5  -4
/Container/    H5  +6  -5  -4
VOLUMES UNITY
* PWR Neutron Poison and Cover Sheet            Configuration  L
PART  7
BOX  1  -10.8458  0.0318  0.0508  20.7467  0.1270  382.2700  ! BORAL Core
BOX  2  -10.8458  0.0000  0.0508  20.7467  0.1905  382.2700  ! BORAL Clad
BOX  3  -10.8458  0.0000  0.0508  21.6916  0.1905  384.2004  ! Space under Cover Sheet
BOX  4  -10.8915  0.0000  0.0051  21.7830  0.2362  384.2918  ! Cover Sheet (top/side)
BOX  5  -10.8966  0.0000  0.0000  21.7932  0.0457  384.3020  ! Remaining Cover Sheet
BOX  6  -10.8966  0.0000  0.0000  21.7932  0.2362  384.3020  ! Container
ZONES
/BORAL Core/    M8  +1
/BORAL Clad/    M9  +2  -1
/Space Under Cover/    H5  +3  -2
/Enclosing Cover/    M7  +4  -3
/Remaining Cover/    M7  +5  -4
/Container/    H5  +6  -5  -4
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q1_4B
PART  8
BOX  1  -11.1684  -11.1684  0.0000  21.4173  21.4173  405.8920    ! Fuel assembly
BOX  2  -11.1684  -11.1684  0.0000  22.3368  22.3368  414.7820    ! Space inside tube from can lid to bottom
BOX  3  -11.2903  -11.2903  5.0800  22.5806  22.5806  388.1120    ! Fuel tube
BOX  4  -10.6807  11.2903  7.1120  21.7932  0.2362  384.3020    ! Boral plus cover sheet - Top (+Y)
BOX  5  11.1125  -11.2903  7.1120  21.7932  0.2362  384.3020  ZROT 180  ! Boral plus cover sheet - Bottom (-Y)
BOX  6  11.2903  11.1125  7.1120  21.7932  0.2362  384.3020  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  7  -11.2903  -10.6807  7.1120  21.7932  0.2362  384.3020  ZROT 270  ! Boral plus cover sheet - Left (-X)
BOX  8  -11.5265  -11.5265  0.0000  23.0530  23.0530  414.7820    ! Complete tube with poison
BOX  9  -11.5265  -11.5265  0.0000  23.5890  23.5890  414.7820    ! Disk Opening
ZONES
/Fuel Assembly/  P5  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P6  +4
/Boral plus Cover/  P7  +5
/Boral plus Cover/  P7  +6
/Boral plus Cover/  P6  +7
/Fuel Tube+Poison/  H5  +8  -3  -2  -4  -6  -5  -7
/Disk Opening/  H5  +9  -8
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q2_4B
PART  9
BOX  1  -10.2489  -11.1684  0.0000  21.4173  21.4173  405.8920    ! Fuel assembly
BOX  2  -11.1684  -11.1684  0.0000  22.3368  22.3368  414.7820    ! Space inside tube from can lid to bottom
BOX  3  -11.2903  -11.2903  5.0800  22.5806  22.5806  388.1120    ! Fuel tube
BOX  4  -11.1125  11.2903  7.1120  21.7932  0.2362  384.3020    ! Boral plus cover sheet - Top (+Y)
BOX  5  10.6807  -11.2903  7.1120  21.7932  0.2362  384.3020  ZROT 180  ! Boral plus cover sheet - Bottom (-Y)
BOX  6  11.2903  11.1125  7.1120  21.7932  0.2362  384.3020  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  7  -11.2903  -10.6807  7.1120  21.7932  0.2362  384.3020  ZROT 270  ! Boral plus cover sheet - Left (-X)
BOX  8  -11.5265  -11.5265  0.0000  23.0530  23.0530  414.7820    ! Complete tube with poison
BOX  9  -12.0625  -11.5265  0.0000  23.5890  23.5890  414.7820    ! Disk Opening
ZONES
/Fuel Assembly/  P5  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P7  +4
/Boral plus Cover/  P6  +5
/Boral plus Cover/  P7  +6
/Boral plus Cover/  P6  +7
/Fuel Tube+Poison/  H5  +8  -3  -2  -4  -6  -5  -7
```

Figure 6.8-9 (continued)

```
/Disk Opening/  H5  +9  -8
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q3_4B
PART  10
BOX  1  -10.2489  -10.2489  0.0000  21.4173  21.4173  405.8920    ! Fuel assembly
BOX  2  -11.1684  -11.1684  0.0000  22.3368  22.3368  414.7820    ! Space inside tube from can lid to bottom
BOX  3  -11.2903  -11.2903  5.0800  22.5806  22.5806  388.1120    ! Fuel tube
BOX  4  -11.1125  11.2903  7.1120  21.7932  0.2362  384.3020    ! Boral plus cover sheet - Top (+Y)
BOX  5  10.6807  -11.2903  7.1120  21.7932  0.2362  384.3020  ZROT 180  ! Boral plus cover sheet - Bottom (-Y)
BOX  6  11.2903  10.6807  7.1120  21.7932  0.2362  384.3020  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  7  -11.2903  -11.1125  7.1120  21.7932  0.2362  384.3020  ZROT 270  ! Boral plus cover sheet - Left (-X)
BOX  8  -11.5265  -11.5265  0.0000  23.0530  23.0530  414.7820    ! Complete tube with poison
BOX  9  -12.0625  -12.0625  0.0000  23.5890  23.5890  414.7820    ! Disk Opening
ZONES
/Fuel Assembly/  P5  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P7  +4
/Boral plus Cover/  P6  +5
/Boral plus Cover/  P6  +6
/Boral plus Cover/  P7  +7
/Fuel Tube+Poison/  H5  +8  -3  -2  -4  -6  -5  -7
/Disk Opening/  H5  +9  -8
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q4_4B
PART  11
BOX  1  -11.1684  -10.2489  0.0000  21.4173  21.4173  405.8920    ! Fuel assembly
BOX  2  -11.1684  -11.1684  0.0000  22.3368  22.3368  414.7820    ! Space inside tube from can lid to bottom
BOX  3  -11.2903  -11.2903  5.0800  22.5806  22.5806  388.1120    ! Fuel tube
BOX  4  -10.6807  11.2903  7.1120  21.7932  0.2362  384.3020    ! Boral plus cover sheet - Top (+Y)
BOX  5  11.1125  -11.2903  7.1120  21.7932  0.2362  384.3020  ZROT 180  ! Boral plus cover sheet - Bottom (-Y)
BOX  6  11.2903  10.6807  7.1120  21.7932  0.2362  384.3020  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  7  -11.2903  -11.1125  7.1120  21.7932  0.2362  384.3020  ZROT 270  ! Boral plus cover sheet - Left (-X)
BOX  8  -11.5265  -11.5265  0.0000  23.0530  23.0530  414.7820    ! Complete tube with poison
BOX  9  -11.5265  -12.0625  0.0000  23.5890  23.5890  414.7820    ! Disk Opening
ZONES
/Fuel Assembly/  P5  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P6  +4
/Boral plus Cover/  P7  +5
/Boral plus Cover/  P6  +6
/Boral plus Cover/  P7  +7
/Fuel Tube+Poison/  H5  +8  -3  -2  -4  -6  -5  -7
/Disk Opening/  H5  +9  -8
VOLUMES UNITY
* PWR Canister Cavity - Basket Radius v2.0
PART  12
BOX  1  -77.3392  -25.4749  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 1
BOX  2  -77.3392  1.8860  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 2
BOX  3  -52.8358  -52.8358  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 3
BOX  4  -51.5658  -25.4749  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 4
BOX  5  -51.5658  1.8860  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 5
BOX  6  -52.8358  29.2468  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 6
BOX  7  -25.4749  -77.3392  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 7
BOX  8  -25.4749  -51.5658  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 8
BOX  9  -25.4749  -25.4749  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 9
BOX  10  -25.4749  1.8860  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 10
BOX  11  -25.4749  27.9768  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 11
BOX  12  -25.4749  53.7502  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 12
BOX  13  1.8860  -77.3392  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 13
BOX  14  1.8860  -51.5658  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 14
BOX  15  1.8860  -25.4749  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 15
BOX  16  1.8860  1.8860  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 16
BOX  17  1.8860  27.9768  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 17
BOX  18  1.8860  53.7502  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 18
BOX  19  29.2468  -52.8358  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 19
BOX  20  27.9768  -25.4749  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 20
BOX  21  27.9768  1.8860  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 21
BOX  22  29.2468  29.2468  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 22
BOX  23  53.7502  -25.4749  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 23
BOX  24  53.7502  1.8860  0.0000  23.5890  23.5890  414.7820  ! Basket Opening 24
CONTAINER
ZROD  25  0.0000  0.0000  0.0000  82.9564  414.7820    ! Basket stack to cavity height
ZONES
/Opening01/  P10  +1
/Opening02/  P9  +2
/Opening03/  P10  +3           ! Corner position
/Opening04/  P10  +4
/Opening05/  P9  +5
/Opening06/  P9  +6           ! Corner position
/Opening07/  P10  +7
/Opening08/  P10  +8
/Opening09/  P10  +9
/Opening10/  P9  +10
/Opening11/  P9  +11
/Opening12/  P9  +12
/Opening13/  P11  +13
/Opening14/  P11  +14
/Opening15/  P11  +15
/Opening16/  P8  +16
/Opening17/  P8  +17
/Opening18/  P8  +18
/Opening19/  P11  +19           ! Corner position
/Opening20/  P11  +20
/Opening21/  P8  +21
/Opening22/  P8  +22           ! Corner position
/Opening23/  P11  +23
/Opening24/  P8  +24
```

Figure 6.8-9 (continued)

```
/Basket/  H1  +25  -1  -2  -3  -4  -5
    -6  -7  -8  -9  -10  -11
    -12  -13  -14  -15  -16  -17
    -18  -19  -20  -21  -22  -23
    -24

VOLUMES  UNITY
* Basket in Canister Cavity v2.0
PART  13  NEST
ZROD  P12  0.0000  0.0000  0.0000  82.9564  414.7820    ! Basket inserted - Includes gap to lid
ZROD  H5  0.0000  0.0000  0.0000  83.5787  414.7820    ! Inserts flood matl to canister shell
* Canister - Structural Lid - No Weld Shield v2.0
PART  14
ZROD  1  0.0000  0.0000  0.0000  83.5787  414.7820    ! Canister cavity contents
ZROD  2  0.0000  0.0000  -4.4450  85.1662  4.4450    ! Canister Bottom Plate
ZROD  3  0.0000  0.0000  414.7820  83.5787  17.7800    ! Shield Lid
ZROD  4  0.0000  0.0000  432.5620  83.5787  7.6200    ! Structural Lid
ZROD  5  0.0000  0.0000  0.0000  83.5787  440.1820    ! Canister Shell Inner
ZROD  6  0.0000  0.0000  0.0000  85.1662  440.1820    ! Canister Shell Outer
ZROD  7  0.0000  0.0000  -4.4450  85.1662  444.6270    ! Inner Detector Surface
ZONES
/Cavity/  P13  +1
/BottomPlate/  M13  +2
/ShieldLid/  P15  +3
/StructLid/  M13  +4
/Shell/  M13  +6  -5
/Canister/  M0  +7  -6  -4  -2
VOLUMES  UNITY
* Shield Lid - With Ports v2.0
PART  15  CLUSTER
ZROD  P16  -41.8271  59.7354  0.0000  7.6200  17.7800    ! Vent port
ZROD  P16  41.8271  -59.7354  0.0000  7.6200  17.7800    ! Drain port
ZROD  M13  0.0000  0.0000  0.0000  83.5787  17.7800    ! Shield Lid
* Vent Port Model - No Port v2.0
PART  16  CLUSTER
ZROD  M13  0.0000  0.0000  0.0000  1.3843  8.4328    ! Bottom Cylinder
ZROD  M13  0.0000  0.0000  8.4328  5.0800  7.9248    ! Middle Cyclinder
ZROD  M13  0.0000  0.0000  16.3576  7.6200  1.4224    ! Top Cylinder
ZROD  M13  0.0000  0.0000  0.0000  7.6200  17.7800    ! Shield lid material
* Transfer Cask Geometry - No Weld Shield - v2.0
PART  17
ZROD  1  0.0000  0.0000  0.0000  85.1662  444.6270    ! TSC
ZROD  2  0.0000  0.0000  0.0000  86.0425  450.3420    ! Cask cavity
ZROD  3  0.0000  0.0000  0.0000  108.2675  2.5400    ! Bottom plate
ZROD  4  0.0000  0.0000  2.5400  87.9475  442.7220    ! Inner shell
ZROD  5  0.0000  0.0000  2.5400  97.8535  436.6260    ! Lead shell
ZROD  6  0.0000  0.0000  2.5400  105.0925  442.7220    ! NS-4-FR shell
ZROD  7  0.0000  0.0000  2.5400  108.2675  442.7220    ! Outer shell
ZROD  8  0.0000  0.0000  445.2620  108.2675  5.0800    ! Top plate
ZROD  9  0.0000  0.0000  450.3420  82.2325  1.9050    ! Area inside retaining ring
ZROD  10  0.0000  0.0000  450.3420  97.8535  1.9050    ! Retaining ring
ZROD  11  0.0000  0.0000  -22.8600  108.2675  22.8600    ! Shield doors and rails
YP  12  102.5525            ! Y plane for shield door rail cutoff
YP  13  -102.5525            ! Y plane for shield door rail cutoff
XROD  14  -118.2675  0.0000  412.2420  12.7000  236.5350    ! Trunions (extended in x)
YROD  15  0.0000  -118.2675  412.2420  12.7000  236.5350    ! Trunions (extended in y)
ZROD  16  0.0000  0.0000  439.1660  97.8535  6.0960    ! Shielding ring
BOX  17  -2.5400  -86.8045  -5.0800  64.9732  173.6090  3.8100  ! Shield door B NS box
YXPRISM  18  62.4332  -86.8045  -5.0800        ! Shield door B NS trapezoid
    173.6090  39.4984  3.8100  36.9157  36.9157
BOX  19  -62.4332  -86.8045  -5.0800  54.8132  173.6090  3.8100  ! Shield door A NS box
YXPRISM  20  -101.9316  -34.2265  -5.0800        ! Shield door A NS trapezoid
    68.4530  39.4984  3.8100  143.0843  143.0843
YXPRISM  21  64.2620  -90.6780  -22.8600        ! Shield door B cut prism
    181.3560  41.4020  22.8600  36.9157  36.9157
XP  22  64.2620            ! Cut plane for NS boundary B
YXPRISM  23  -105.6640  -35.5600  -22.8600        ! Shield door A cut prism
    71.1200  41.4020  22.8600  143.0843  143.0843
XP  24  -64.2620            ! Cut plane for NS boundary A
ZROD  25  0.0000  0.0000  -22.8600  108.2675  475.1070    ! Container
ZONES
/TSC/  P14  +1            ! TSC
/CaskCavity/  M0  +2  -1          ! Cask cavity
/BottomPlate/  M14  +3  -2          ! Bottom plate
/InnerShell/  M14  +4  -2  -14  -15      ! Inner shell
/LeadShell/  M15  +5  -4  -14  -15      ! Lead shell
/NS-4-FRShell/  M16  +6  -5  -14  -15  -16    ! NS-4-FR shell
/ShieldRing/  M14  +16  -4  -14  -15      ! Shielding ring
/OuterShell/  M14  +7  -6  -14  -15      ! Outer shell
/TopPlate/  M14  +8  -2          ! Top plate
/RetRingInner/  M0  +9            ! Area inside retaining ring (null)
/RetRing/  M0  +10  -9          ! Retaining ring
/ShieldDoor1/  M14  +11  +13  -12  -17  -19    ! Shield doors and rails
    -18  -20  -22  +24
/ShieldDoor2/  M14  +11  +13  -12  -17  -19    ! Shield doors and rails
    -18  -20  +22  +21
/ShieldDoor3/  M14  +11  +13  -12  -17  -19    ! Shield doors and rails
    -18  -20  -24  +23
/ShieldDoorOuter1/  M0  +11  +12          ! Space outside of shield door
/ShieldDoorOuter2/  M0  +11  -13          ! Space outside of shield door
/ShieldDoorOuter3/  M0  +11  +22  -21        ! Space outside of shield door B
/ShieldDoorOuter4/  M0  +11  -24  -23        ! Space outside of shield door A
/XTrunions/  M14  +14  +7  -2        ! X Trunions
/YTrunions/  M14  +15  +7  -2        ! Y Trunions
/ShldDrBNSBox/  M16  +17            ! Shield door B NS box
/ShldDrANSBox/  M16  +19            ! Shield door A NS box
/ShldDrBNSTrap/  M16  +18            ! Shield door B NS trapezoid
/ShldDrANSTrap/  M16  +20            ! Shield door A NS trapezoid
/Container/  M0  +25  -11  -3  -7  -8  -10  ! Container
```

Figure 6.8-9 (continued)

```
VOLUMES UNITY
end

*
* Unit 5 - Source Geometry for
*
begin source geometry
ZONEMAT
ALL / MATERIAL    1
end

*
* Unit 3 Hole Data
*
begin hole data
* PWR Canister Hole Description v2.0
* Hole  1  General Basket Structure
PLATE
0  0  1
5
413.0040  0      ! Top of Basket
379.9840  -2      ! Top of Highest Support Disk
16.3068  -4      ! Bottom of Lowest Support Disk
0.0000  -3      ! Bottom of Basket
0.0000  3      ! Basket Offset
3

* Hole  2  Top Weldment Disk - no structure above the weldment disk
RZMESH
2        ! number of radial points
82.2198
83.1850
5        ! number of axial intervals
379.9840        ! Top of diskstack
394.4620        ! Bottom of weldment
397.6370        ! Top of weldment plate
406.1241        ! Ullage
411.7340        ! Flange
413.0040        ! Void to top of basket
3  3      ! Material below weldment
11  11      ! Plate Material
3  11      ! Ullage
3  11      ! Flange
3  3      ! Void to top of basket
3        ! Outside material

* Hole  3  Bottom Weldment Disk - no structure in the weldment disk support
RZMESH
1        ! number of radial points
83.1850
1        ! number of axial intervals
2.5400
5.0800        ! Coordinates inherited from PLATE Hole
11        ! Plate Material
3        ! Outside material

* Hole  4  Support disk and heat transfer disk stack
PLATE
origin  0  0  16.3068  ! Origin
0  0  1
4
cell  12.4968      ! Sets up a repeating lattice of cells
12.4968  3      ! flood matl
7.5184  3      ! water gap
6.2484  12      ! aluminium disk
1.2700  3      ! water gap
10        ! steel disk

* Hole  5  Flood material model
PLATE
0  0  1
1
406.1241  3      ! Above flooded region
3        ! Flooded region

end
```

Figure 6.8-10 MONK8A Input for BWR Transfer Cask

```
columns 1 200
*
*   UMS Transfer Cask - ex09c Standard
*
*   Cask Lid Configurations
*           Shield Lid - No Ports
*           Structural Lid - No Weld Shield
*
*   Neutron Poison Loading -  75 %
*   Exterior Water Density 0.0001
*   Cavity Water Density 0.9998
*   Fuel to Clad Gap Water Density 0.9998
*
*   Boron Content in Water - 0 ppm
*
*   Model Revision v3.0
*
* Parameters
*
@randseed =  12345
*
*
* Unit 1 Control Data
*
begin control data
*READ   ! read and check each independently
*SEEK MULTIPLE DEFINITIONS

SEEDS  @randseed  @randseed
STAGES -15 810 4000 STDV 0.0008

end
*
* Unit 9 Material Specification
*
begin material specification
normalise
nmixtures  7
weight  mixture  1
      u235  3.8784E-02
      u238  8.4267E-01
      o16  1.1855E-01
atoms  mixture  2
      h  6.6667E-01
      o16  3.3333E-01
atoms  mixture  3
      h  6.6667E-01
      o16  3.3333E-01
atoms  mixture  4
      h  4.2857E-01
      b  1.4286E-01
      o16  4.2857E-01
weight  mixture  5
      al  7.6834E-01
      b10  3.2642E-02
      b11  1.4870E-01
      c  5.0317E-02
atoms  mixture  6
      c  2.8571E-01
      h  4.7619E-01
      o16  2.3810E-01
weight  mixture  7
      h  4.2152E-02
      o16  5.4785E-01
      fe  4.7900E-02
      c  9.3500E-02
      si  3.3600E-02
      ca  5.6100E-02
      al  1.7890E-01
*
* Materials List - v1.2 - Class 5 - ex09c - Ex/ANF9 (JP-4,5) Fuel
*
nmaterials  23
volume              ! UO2 at 4.4%
material  1
  mixture  1  density  10.4120  prop  1.00000
volume              ! Fuel pin cladding
material  2
  zircalloy    density  6.5500  prop  1.00000
volume              ! Water In Lattice and Tube
material  3
  mixture  2  density  0.9998  prop  1.00000  ! mixH2O
volume              ! Water In Fuel Rod Clad Gap
material  4
  mixture  2  density  0.9998  prop  1.00000  ! mixH2O
volume              ! Lower Nozzle Material
material  5
  stainless 304l steel    density  7.9200  prop  0.17007
  mixture  2  density  0.9998  prop  0.82993  ! mixH2O
volume              ! Upper Nozzle Material
material  6
  stainless 304l steel    density  7.9200  prop  0.06774
  mixture  2  density  0.9998  prop  0.93226  ! mixH2O
```

Figure 6.8-10 (continued)

```
*
* Materials List - Common Materials - v2.0
*
volume              ! Tube wall and cover sheet
material  7
  stainless 304l steel    density  7.9300  prop  1.0000
volume              ! BORAL core
material  8
  mixture  5  density  1.9901  prop  1.0000  ! mixBORAL
volume              ! BORAL alumnimum clad
material  9
  aluminium        prop  1.0000
volume              ! Structural Disk Material
material  10
  stainless 304l steel    density  7.9300  prop  1.0000
volume              ! Weldment Material
material  11
  stainless 304l steel    density  7.9300  prop  1.0000
volume              ! Heat Transfer Disk Material
material  12
  aluminium        prop  1.0000
volume              ! Canister Material
material  13
  stainless 304l steel    density  7.9300  prop  1.0000
atoms              ! Transfer steel
material  14    density  0      !  (SCALE carbon steel)
    fe      prop  8.3498E-02
    c      prop  3.9250E-03
volume              ! Lead
material  15
  pb    density  11.0400  prop  1.0000
atoms              ! NS-4-FR
material  16    density  0      ! 0 means atom/b-cm
    b10      prop  8.5500E-05
    b11      prop  3.4200E-04
    al      prop  7.8000E-03
    h      prop  5.8500E-02
    o16      prop  2.6100E-02
    c      prop  2.2600E-02
    n      prop  1.3900E-03
volume              ! Stainless Steel 304
material  17
  stainless 304l steel    density  7.9300  prop  1.0000
volume              ! Vent port middle cylinder
material  18
  stainless 304l steel    density  7.9300  prop  0.5000
  void        prop  0.5000
atoms              ! SCALE Concrete
material  19    density  0
    h      prop  1.3401E-02
    o16      prop  4.4931E-02
    na      prop  1.7036E-03
    al      prop  1.7018E-03
    si      prop  1.6205E-02
    ca      prop  1.4826E-03
    fe      prop  3.3857E-04
volume              ! Heat fins for transport cask
material  20
  cu    density  8.9200  prop  0.4286
  stainless 304l steel    density  7.9300  prop  0.5714
volume              ! Balsa
material  21
  mixture  6  density  0.1250  prop  1.0000
volume              ! Redwood
material  22
  mixture  6  density  0.3870  prop  1.0000
volume              ! NS3
material  23
  mixture  7  density  1.6507  prop  1.0000  ! Weight loss @ 200F of 2.90%
end

*
* Unit 2 Material Geometry
*
begin material geometry
* Fuel Rod - Class 5 - ex09c - Ex/ANF9 (JP-4,5)
PART  1
ZROD  1    0.0000  0.0000  0.9017  0.4528  381.0000    ! Fuel pellet stack
ZROD  2    0.0000  0.0000  0.9017  0.4623  405.3281    ! Annulus + Plenum
ZROD  3    0.0000  0.0000  0.0000  0.5385  407.1315    ! Clad
ZROD  4    0.0000  0.0000  407.1315  0.2692  3.3782    ! Fuel rod to top nozzle
BOX  5    -0.7264  -0.7264  0.0000  1.4528  1.4528  410.5097  ! Pitch box
ZONES
/Fuel/    M1  +1
/Fuel to Clad Gap/    M4  +2  -1
/Clad & End Plugs/    M2  +3  -2
/Rod to Top Nozzle/    M2  +4
/Rod in Pitch/    M3  +5  -4  -3
* BWR Water Rod - Class 5 - ex09c - Ex/ANF9 (JP-4,5)
PART  2  NEST
ZROD  M3    0.0000  0.0000  0.0000  0.4623  381.0000    ! Water Rod Interior
ZROD  M2    0.0000  0.0000  0.0000  0.5385  381.0000    ! Clad
BOX  M3    -0.7264  -0.7264  0.0000  1.4528  1.4528  410.5097  ! Pitch box
* Array_9x9_79
PART  3  ARRAY
9  9  1
1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1  1  1
```

Figure 6.8-10 (continued)

```
1  1  1  1  1  1  1  1  1
1  1  1  1  2  1  1  1  1
1  1  1  1  1  2  1  1  1
1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1  1  1
* Fuel Assembly Array Inserted Into Assembly - Class 5 - ex09c - Ex/ANF9 (JP-4,5)
PART  4  NEST
BOX  P3  -6.5376  -6.5376  17.6276  13.0752  13.0752  410.5097    ! Array
BOX  M3  -6.7031  -6.7031  17.6276  13.4061  13.4061  410.5097    ! BWR Channel Interior
BOX  M2  -6.9063  -6.9063  17.6276  13.8125  13.8125  410.5097    ! BWR Channel
BOX  M3  -6.9063  -6.9063  17.6276  13.8125  13.8125  410.5097    ! Fuel Width Envelope
BOX  M5  -6.9063  -6.9063  0.0000  13.8125  13.8125  428.1373    ! Lower Nozzle
BOX  M6  -6.9063  -6.9063  0.0000  13.8125  13.8125  447.1873    ! Upper Nozzle - Envelope
* BWR Neutron Poison and Cover Sheet            Configuration  C
PART  5
BOX  1  -6.7031  0.1080  0.0508  13.4061  0.1270  396.4940  ! BORAL Core
BOX  2  -6.7031  0.0000  0.0508  13.4061  0.3429  396.4940  ! BORAL Clad
BOX  3  -7.1755  0.0000  0.0508  14.3510  0.3429  398.4244  ! Space under Cover Sheet
BOX  4  -7.2212  0.0000  0.0051  14.4424  0.3886  398.5158  ! Cover Sheet (top/side)
BOX  5  -7.2263  0.0000  0.0000  14.4526  0.0457  398.5260  ! Remaining Cover Sheet
BOX  6  -7.2263  0.0000  0.0000  14.4526  0.3886  398.5260  ! Container
ZONES
/BORAL Core/    M8  +1
/BORAL Clad/    M9  +2  -1
/Space Under Cover/    H5  +3  -2
/Enclosing Cover/    M7  +4  -3
/Remaining Cover/    M7  +5  -4
/Container/    H5  +6  -5  -4
VOLUMES UNITY
* BWR Neutron Poison and Cover Sheet            Configuration  R
PART  6
BOX  1  -6.2306  0.1080  0.0508  13.4061  0.1270  396.4940  ! BORAL Core
BOX  2  -6.2306  0.0000  0.0508  13.4061  0.3429  396.4940  ! BORAL Clad
BOX  3  -7.1755  0.0000  0.0508  14.3510  0.3429  398.4244  ! Space under Cover Sheet
BOX  4  -7.2212  0.0000  0.0051  14.4424  0.3886  398.5158  ! Cover Sheet (top/side)
BOX  5  -7.2263  0.0000  0.0000  14.4526  0.0457  398.5260  ! Remaining Cover Sheet
BOX  6  -7.2263  0.0000  0.0000  14.4526  0.3886  398.5260  ! Container
ZONES
/BORAL Core/    M8  +1
/BORAL Clad/    M9  +2  -1
/Space Under Cover/    H5  +3  -2
/Enclosing Cover/    M7  +4  -3
/Remaining Cover/    M7  +5  -4
/Container/    H5  +6  -5  -4
VOLUMES UNITY
* BWR Neutron Poison and Cover Sheet            Configuration  L
PART  7
BOX  1  -7.1755  0.1080  0.0508  13.4061  0.1270  396.4940  ! BORAL Core
BOX  2  -7.1755  0.0000  0.0508  13.4061  0.3429  396.4940  ! BORAL Clad
BOX  3  -7.1755  0.0000  0.0508  14.3510  0.3429  398.4244  ! Space under Cover Sheet
BOX  4  -7.2212  0.0000  0.0051  14.4424  0.3886  398.5158  ! Cover Sheet (top/side)
BOX  5  -7.2263  0.0000  0.0000  14.4526  0.0457  398.5260  ! Remaining Cover Sheet
BOX  6  -7.2263  0.0000  0.0000  14.4526  0.3886  398.5260  ! Container
ZONES
/BORAL Core/    M8  +1
/BORAL Clad/    M9  +2  -1
/Space Under Cover/    H5  +3  -2
/Enclosing Cover/    M7  +4  -3
/Remaining Cover/    M7  +5  -4
/Container/    H5  +6  -5  -4
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q1_2B
PART  8
BOX  1  -7.4981  -7.4981  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.1120  7.6200  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  7.6200  7.3406  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  6  -7.6200  -7.6200  0.0000  15.6286  15.6286  453.6440    ! Complete tube with poison
BOX  7  -7.6200  -7.6200  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P6  +4
/Boral plus Cover/  P7  +5
/Fuel Tube+Poison/  H5  +6  -3  -2  -4  -5
/Disk Opening/  H5  +7  -6
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q2_2B
PART  9
BOX  1  -6.3144  -7.4981  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.3406  7.6200  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  7.6200  7.3406  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  6  -7.6200  -7.6200  0.0000  15.6286  15.6286  453.6440    ! Complete tube with poison
BOX  7  -7.9375  -7.6200  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P7  +4
/Boral plus Cover/  P7  +5
/Fuel Tube+Poison/  H5  +6  -3  -2  -4  -5
/Disk Opening/  H5  +7  -6
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q3_2B
```

Figure 6.8-10 (continued)

```
PART  10
BOX  1  -6.3144  -6.3144  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.3406  7.6200  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  7.6200  7.1120  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  6  -7.6200  -7.6200  0.0000  15.6286  15.6286  453.6440    ! Complete tube with poison
BOX  7  -7.9375  -7.9375  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P7  +4
/Boral plus Cover/  P6  +5
/Fuel Tube+Poison/  H5  +6  -3  -2  -4  -5
/Disk Opening/  H5  +7  -6
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q4_2B
PART  11
BOX  1  -7.4981  -6.3144  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.1120  7.6200  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  7.6200  7.1120  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  6  -7.6200  -7.6200  0.0000  15.6286  15.6286  453.6440    ! Complete tube with poison
BOX  7  -7.6200  -7.9375  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P6  +4
/Boral plus Cover/  P6  +5
/Fuel Tube+Poison/  H5  +6  -3  -2  -4  -5
/Disk Opening/  H5  +7  -6
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  CT_2B
PART  12
BOX  1  -6.9063  -7.4981  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.2263  7.6200  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  7.6200  7.3406  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  6  -7.6200  -7.6200  0.0000  15.6286  15.6286  453.6440    ! Complete tube with poison
BOX  7  -7.7788  -7.6200  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P5  +4
/Boral plus Cover/  P7  +5
/Fuel Tube+Poison/  H5  +6  -3  -2  -4  -5
/Disk Opening/  H5  +7  -6
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  CB_2B
PART  13
BOX  1  -6.9063  -6.3144  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.2263  7.6200  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  7.6200  7.1120  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  6  -7.6200  -7.6200  0.0000  15.6286  15.6286  453.6440    ! Complete tube with poison
BOX  7  -7.7788  -7.9375  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P5  +4
/Boral plus Cover/  P6  +5
/Fuel Tube+Poison/  H5  +6  -3  -2  -4  -5
/Disk Opening/  H5  +7  -6
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q1_RB
PART  14
BOX  1  -7.4981  -7.4981  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  7.6200  7.3406  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  5  -7.6200  -7.6200  0.0000  15.6286  15.2400  453.6440    ! Complete tube with poison
BOX  6  -7.6200  -7.6200  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P7  +4
/Fuel Tube+Poison/  H5  +5  -3  -2    -4
/Disk Opening/  H5  +6  -5
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q2_RB
PART  15
BOX  1  -6.3144  -7.4981  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  7.6200  7.3406  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  5  -7.6200  -7.6200  0.0000  15.6286  15.2400  453.6440    ! Complete tube with poison
BOX  6  -7.9375  -7.6200  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
```

Figure 6.8-10 (continued)

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/Boral plus Cover/  P7  +4
/Fuel Tube+Poison/  H5  +5  -3  -2    -4
/Disk Opening/  H5  +6  -5
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  CT_RB
PART  16
BOX  1  -6.9063  -7.4981  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  7.6200  7.3406  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  5  -7.6200  -7.6200  0.0000  15.6286  15.2400  453.6440    ! Complete tube with poison
BOX  6  -7.7788  -7.6200  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P7  +4
/Fuel Tube+Poison/  H5  +5  -3  -2    -4
/Disk Opening/  H5  +6  -5
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q1_TB
PART  17
BOX  1  -7.4981  -7.4981  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.1120  7.6200  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  -7.6200  -7.6200  0.0000  15.2400  15.6286  453.6440    ! Complete tube with poison
BOX  6  -7.6200  -7.6200  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P6  +4
/Fuel Tube+Poison/  H5  +5  -3  -2  -4
/Disk Opening/  H5  +6  -5
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q4_TB
PART  18
BOX  1  -7.4981  -6.3144  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.1120  7.6200  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  -7.6200  -7.6200  0.0000  15.2400  15.6286  453.6440    ! Complete tube with poison
BOX  6  -7.6200  -7.9375  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P6  +4
/Fuel Tube+Poison/  H5  +5  -3  -2  -4
/Disk Opening/  H5  +6  -5
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q1_NB
PART  19
BOX  1  -7.4981  -7.4981  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.4981  -7.4981  0.0000  14.9962  14.9962  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.6200  -7.6200  12.7000  15.2400  15.2400  409.4480    ! Fuel tube
BOX  4  -7.6200  -7.6200  0.0000  15.2400  15.2400  453.6440    ! Complete tube with poison
BOX  5  -7.6200  -7.6200  0.0000  15.9461  15.9461  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Fuel Tube+Poison/  H5  +4  -3  -2
/Disk Opening/  H5  +5  -4
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q1O_NB
PART  20
BOX  1  -7.6759  -7.6759  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.6759  -7.6759  0.0000  15.3518  15.3518  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.7978  -7.7978  12.7000  15.5956  15.5956  409.4480    ! Fuel tube
BOX  4  -7.7978  -7.7978  0.0000  15.5956  15.5956  453.6440    ! Complete tube with poison
BOX  5  -7.7978  -7.7978  0.0000  16.3271  16.3271  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Fuel Tube+Poison/  H5  +4  -3  -2
/Disk Opening/  H5  +5  -4
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q2O_RB
PART  21
BOX  1  -6.1366  -7.6759  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.6759  -7.6759  0.0000  15.3518  15.3518  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.7978  -7.7978  12.7000  15.5956  15.5956  409.4480    ! Fuel tube
BOX  4  7.7978  7.5184  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  5  -7.7978  -7.7978  0.0000  15.9842  15.5956  453.6440    ! Complete tube with poison
BOX  6  -8.1407  -7.7978  0.0000  16.3271  16.3271  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P7  +4
/Fuel Tube+Poison/  H5  +5  -3  -2    -4
/Disk Opening/  H5  +6  -5
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q3O_2B
PART  22
BOX  1  -6.1366  -6.1366  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
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Figure 6.8-10 (continued)

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BOX  2  -7.6759  -7.6759  0.0000  15.3518  15.3518  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.7978  -7.7978  12.7000  15.5956  15.5956  409.4480    ! Fuel tube
BOX  4  -7.5184  7.7978  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  7.7978  6.9342  14.7320  14.4526  0.3886  398.5260  ZROT 90  ! Boral plus cover sheet - Right (+X)
BOX  6  -7.7978  -7.7978  0.0000  15.9842  15.9842  453.6440    ! Complete tube with poison
BOX  7  -8.1407  -8.1407  0.0000  16.3271  16.3271  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P7  +4
/Boral plus Cover/  P6  +5
/Fuel Tube+Poison/  H5  +6  -3  -2  -4  -5
/Disk Opening/  H5  +7  -6
VOLUMES UNITY
* Fuel Assembly in Tube v2.0          Configuration  Q4O_TB
PART  23
BOX  1  -7.6759  -6.1366  0.0000  13.8125  13.8125  447.1873    ! Fuel assembly
BOX  2  -7.6759  -7.6759  0.0000  15.3518  15.3518  453.6440    ! Space inside tube from can lid to bottom
BOX  3  -7.7978  -7.7978  12.7000  15.5956  15.5956  409.4480    ! Fuel tube
BOX  4  -6.9342  7.7978  14.7320  14.4526  0.3886  398.5260    ! Boral plus cover sheet - Top (+Y)
BOX  5  -7.7978  -7.7978  0.0000  15.5956  15.9842  453.6440    ! Complete tube with poison
BOX  6  -7.7978  -8.1407  0.0000  16.3271  16.3271  453.6440    ! Disk Opening
ZONES
/Fuel Assembly/  P4  +1
/Space in Tube/  H5  +2  -1
/Fuel Tube/  M7  +3  -2
/Boral plus Cover/  P6  +4
/Fuel Tube+Poison/  H5  +5  -3  -2  -4
/Disk Opening/  H5  +6  -5
VOLUMES UNITY
* BWR Canister Cavity - Basket Radius v2.0
PART  24
BOX  1  -78.3615  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 1
BOX  2  -78.3615  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 2
BOX  3  -60.9549  -52.1564  0.0000  16.3271  16.3271  453.6440  ! Basket Opening 3 - Oversize
BOX  4  -60.7644  -34.3687  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 4
BOX  5  -60.7644  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 5
BOX  6  -60.7644  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 6
BOX  7  -60.7644  18.4226  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 7
BOX  8  -60.9549  35.8292  0.0000  16.3271  16.3271  453.6440  ! Basket Opening 8 - Oversize
BOX  9  -43.1673  -69.5630  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 9
BOX  10  -43.1673  -51.9659  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 10
BOX  11  -43.1673  -34.3687  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 11
BOX  12  -43.1673  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 12
BOX  13  -43.1673  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 13
BOX  14  -43.1673  18.4226  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 14
BOX  15  -43.1673  36.0197  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 15
BOX  16  -43.1673  53.6169  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 16
BOX  17  -25.5702  -69.5630  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 17
BOX  18  -25.5702  -51.9659  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 18
BOX  19  -25.5702  -34.3687  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 19
BOX  20  -25.5702  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 20
BOX  21  -25.5702  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 21
BOX  22  -25.5702  18.4226  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 22
BOX  23  -25.5702  36.0197  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 23
BOX  24  -25.5702  53.6169  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 24
BOX  25  -7.9731  -69.5630  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 25
BOX  26  -7.9731  -51.9659  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 26
BOX  27  -7.9731  -34.3687  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 27
BOX  28  -7.9731  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 28
BOX  29  -7.9731  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 29
BOX  30  -7.9731  18.4226  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 30
BOX  31  -7.9731  36.0197  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 31
BOX  32  -7.9731  53.6169  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 32
BOX  33  9.6241  -69.5630  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 33
BOX  34  9.6241  -51.9659  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 34
BOX  35  9.6241  -34.3687  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 35
BOX  36  9.6241  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 36
BOX  37  9.6241  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 37
BOX  38  9.6241  18.4226  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 38
BOX  39  9.6241  36.0197  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 39
BOX  40  9.6241  53.6169  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 40
BOX  41  27.2212  -69.5630  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 41
BOX  42  27.2212  -51.9659  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 42
BOX  43  27.2212  -34.3687  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 43
BOX  44  27.2212  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 44
BOX  45  27.2212  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 45
BOX  46  27.2212  18.4226  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 46
BOX  47  27.2212  36.0197  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 47
BOX  48  27.2212  53.6169  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 48
BOX  49  44.6278  -52.1564  0.0000  16.3271  16.3271  453.6440  ! Basket Opening 49 - Oversize
BOX  50  44.8183  -34.3687  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 50
BOX  51  44.8183  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 51
BOX  52  44.8183  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 52
BOX  53  44.8183  18.4226  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 53
BOX  54  44.6278  35.8292  0.0000  16.3271  16.3271  453.6440  ! Basket Opening 54 - Oversize
BOX  55  62.4154  -16.7716  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 55
BOX  56  62.4154  0.8255  0.0000  15.9461  15.9461  453.6440  ! Basket Opening 56
CONTAINER
ZROD  57  0.0000  0.0000  0.0000  82.8675  453.6440    ! Basket stack to cavity height
ZONES
/Opening1/  P10  +1
/Opening2/  P15  +2
/Opening3/  P22  +3            ! Oversized opening
/Opening4/  P10  +4
/Opening5/  P10  +5
/Opening6/  P9  +6
/Opening7/  P9  +7
```

Figure 6.8-10 (continued)

```
/Opening8/  P21  +8            ! Oversized opening
/Opening9/  P10  +9
/Opening10/  P10  +10
/Opening11/  P10  +11
/Opening12/  P10  +12
/Opening13/  P9  +13
/Opening14/  P9  +14
/Opening15/  P9  +15
/Opening16/  P15  +16
/Opening17/  P10  +17
/Opening18/  P10  +18
/Opening19/  P10  +19
/Opening20/  P10  +20
/Opening21/  P9  +21
/Opening22/  P9  +22
/Opening23/  P9  +23
/Opening24/  P15  +24
/Opening25/  P13  +25
/Opening26/  P13  +26
/Opening27/  P13  +27
/Opening28/  P13  +28
/Opening29/  P12  +29
/Opening30/  P12  +30
/Opening31/  P12  +31
/Opening32/  P16  +32
/Opening33/  P11  +33
/Opening34/  P11  +34
/Opening35/  P11  +35
/Opening36/  P11  +36
/Opening37/  P8  +37
/Opening38/  P8  +38
/Opening39/  P8  +39
/Opening40/  P14  +40
/Opening41/  P18  +41
/Opening42/  P11  +42
/Opening43/  P11  +43
/Opening44/  P11  +44
/Opening45/  P8  +45
/Opening46/  P8  +46
/Opening47/  P8  +47
/Opening48/  P19  +48
/Opening49/  P23  +49            ! Oversized opening
/Opening50/  P18  +50
/Opening51/  P11  +51
/Opening52/  P8  +52
/Opening53/  P17  +53
/Opening54/  P20  +54            ! Oversized opening
/Opening55/  P18  +55
/Opening56/  P19  +56
/Basket/  H1  +57  -1  -2  -3  -4  -5
    -6  -7  -8  -9  -10  -11
    -12  -13  -14  -15  -16  -17
    -18  -19  -20  -21  -22  -23
    -24  -25  -26  -27  -28  -29
    -30  -31  -32  -33  -34  -35
    -36  -37  -38  -39  -40  -41
    -42  -43  -44  -45  -46  -47
    -48  -49  -50  -51  -52  -53
    -54  -55  -56

VOLUMES  UNITY
* Basket in Canister Cavity v2.0
PART  25  NEST
ZROD  P24  0.0000  0.0000  0.0000  82.8675  453.6440    ! Basket inserted - Includes gap to lid
ZROD  H5  0.0000  0.0000  0.0000  83.5787  453.6440    ! Inserts flood matl to canister shell
* Canister - Structural Lid - No Weld Shield v2.0
PART  26
ZROD  1  0.0000  0.0000  0.0000  83.5787  453.6440    ! Canister cavity contents
ZROD  2  0.0000  0.0000  -4.4450  85.1662  4.4450    ! Canister Bottom Plate
ZROD  3  0.0000  0.0000  453.6440  83.5787  17.7800    ! Shield Lid
ZROD  4  0.0000  0.0000  471.4240  83.5787  7.6200    ! Structural Lid
ZROD  5  0.0000  0.0000  0.0000  83.5787  479.0440    ! Canister Shell Inner
ZROD  6  0.0000  0.0000  0.0000  85.1662  479.0440    ! Canister Shell Outer
ZROD  7  0.0000  0.0000  -4.4450  85.1662  483.4890    ! Inner Detector Surface
ZONES
/Cavity/  P25  +1
/BottomPlate/  M13  +2
/ShieldLid/  P27  +3
/StructLid/  M13  +4
/Shell/  M13  +6  -5
/Canister/  M0  +7  -6  -4  -2
VOLUMES  UNITY
* Shield Lid - With Ports v2.0
PART  27  CLUSTER
ZROD  P28  -46.8743  55.8626  0.0000  7.6200  17.7800    ! Vent port
ZROD  P28  46.8743  -55.8626  0.0000  7.6200  17.7800    ! Drain port
ZROD  M13  0.0000  0.0000  0.0000  83.5787  17.7800    ! Shield Lid
* Vent Port Model - No Port v2.0
PART  28  CLUSTER
ZROD  M13  0.0000  0.0000  0.0000  1.3843  8.4328    ! Bottom Cylinder
ZROD  M13  0.0000  0.0000  8.4328  5.0800  7.9248    ! Middle Cyclinder
ZROD  M13  0.0000  0.0000  16.3576  7.6200  1.4224    ! Top Cylinder
ZROD  M13  0.0000  0.0000  0.0000  7.6200  17.7800    ! Shield lid material
* Transfer Cask Geometry - No Weld Shield - v2.0
PART  29
ZROD  1  0.0000  0.0000  0.0000  85.1662  483.4890    ! TSC
ZROD  2  0.0000  0.0000  0.0000  86.0425  489.2040    ! Cask cavity
ZROD  3  0.0000  0.0000  0.0000  108.2675  2.5400    ! Bottom plate
ZROD  4  0.0000  0.0000  2.5400  87.9475  481.5840    ! Inner shell
```

Figure 6.8-10 (continued)

```
ZROD  5  0.0000  0.0000  2.5400  97.8535  475.4880    ! Lead shell
ZROD  6  0.0000  0.0000  2.5400  105.0925  481.5840    ! NS-4-FR shell
ZROD  7  0.0000  0.0000  2.5400  108.2675  481.5840    ! Outer shell
ZROD  8  0.0000  0.0000  484.1240  108.2675  5.0800    ! Top plate
ZROD  9  0.0000  0.0000  489.2040  82.2325  1.9050    ! Area inside retaining ring
ZROD  10  0.0000  0.0000  489.2040  97.8535  1.9050    ! Retaining ring
ZROD  11  0.0000  0.0000  -22.8600  108.2675  22.8600    ! Shield doors and rails
YP  12  102.5525            ! Y plane for shield door rail cutoff
YP  13  -102.5525            ! Y plane for shield door rail cutoff
XROD  14  -118.2675  0.0000  451.1040  12.7000  236.5350    ! Trunions (extended in x)
YROD  15  0.0000  -118.2675  451.1040  12.7000  236.5350    ! Trunions (extended in y)
ZROD  16  0.0000  0.0000  478.0280  97.8535  6.0960    ! Shielding ring
BOX  17  -2.5400  -86.8045  -5.0800  64.9732  173.6090  3.8100  ! Shield door B NS box
YXPRISM  18  62.4332  -86.8045  -5.0800        ! Shield door B NS trapezoid
    173.6090  39.4984  3.8100  36.9157  36.9157
BOX  19  -62.4332  -86.8045  -5.0800  54.8132  173.6090  3.8100  ! Shield door A NS box
YXPRISM  20  -101.9316  -34.2265  -5.0800        ! Shield door A NS trapezoid
    68.4530  39.4984  3.8100  143.0843  143.0843
YXPRISM  21  64.2620  -90.6780  -22.8600        ! Shield door B cut prism
    181.3560  41.4020  22.8600  36.9157  36.9157
XP  22  64.2620            ! Cut plane for NS boundary B
YXPRISM  23  -105.6640  -35.5600  -22.8600        ! Shield door A cut prism
    71.1200  41.4020  22.8600  143.0843  143.0843
XP  24  -64.2620            ! Cut plane for NS boundary A
ZROD  25  0.0000  0.0000  -22.8600  108.2675  513.9690    ! Container
ZONES
/TSC/  P26  +1            ! TSC
/CaskCavity/  M0  +2  -1          ! Cask cavity
/BottomPlate/  M14  +3  -2          ! Bottom plate
/InnerShell/  M14  +4  -2  -14  -15      ! Inner shell
/LeadShell/  M15  +5  -4  -14  -15      ! Lead shell
/NS-4-FRShell/  M16  +6  -5  -14  -15  -16    ! NS-4-FR shell
/ShieldRing/  M14  +16  -4  -14  -15      ! Shielding ring
/OuterShell/  M14  +7  -6  -14  -15      ! Outer shell
/TopPlate/  M14  +8  -2          ! Top plate
/RetRingInner/  M0  +9            ! Area inside retaining ring (null)
/RetRing/  M0  +10  -9          ! Retaining ring
/ShieldDoor1/  M14  +11  +13  -12  -17  -19    ! Shield doors and rails
    -18  -20  -22  +24
/ShieldDoor2/  M14  +11  +13  -12  -17  -19    ! Shield doors and rails
    -18  -20  +22  +21
/ShieldDoor3/  M14  +11  +13  -12  -17  -19    ! Shield doors and rails
    -18  -20  -24  +23
/ShieldDoorOuter1/  M0  +11  +12          ! Space outside of shield door
/ShieldDoorOuter2/  M0  +11  -13          ! Space outside of shield door
/ShieldDoorOuter3/  M0  +11  +22  -21        ! Space outside of shield door B
/ShieldDoorOuter4/  M0  +11  -24  -23        ! Space outside of shield door A
/XTrunions/  M14  +14  +7  -2        ! X Trunions
/YTrunions/  M14  +15  +7  -2        ! Y Trunions
/ShldDrBNSBox/  M16  +17            ! Shield door B NS box
/ShldDrANSBox/  M16  +19            ! Shield door A NS box
/ShldDrBNSTrap/  M16  +18            ! Shield door B NS trapezoid
/ShldDrANSTrap/  M16  +20            ! Shield door A NS trapezoid
/Container/  M0  +25  -11  -3  -7  -8  -10  ! Container
VOLUMES UNITY
end

*
* Unit 5 - Source Geometry for
*
begin source geometry
ZONEMAT
ALL / MATERIAL    1
end

*
* Unit 3 Hole Data
*
begin hole data
* BWR Canister Hole Description v2.0
* Hole  1  General Basket Structure
PLATE
0  0  1
7
451.8660  0      ! Top of Basket
413.1056  -2      ! Top of Highest Support Disk
275.3233  -7      ! Resume support disk only
110.1598  -4      ! Start of support+heat disk region
22.6060  -6      ! Bottom of Lowest Support Disk
0.0000  -3      ! Bottom of Basket
0.0000  3      ! Basket Offset
3

* Hole  2  Top Weldment Disk - no structure above the weldment disk
RZMESH
2        ! number of radial points
82.2198
83.1850
5        ! number of axial intervals
413.1056        ! Top of diskstack
423.1640        ! Bottom of weldment
425.7040        ! Top of weldment plate
444.9861        ! Ullage
450.4690        ! Flange
451.8660        ! Void to top of basket
3  3      ! Material below weldment
11  11      ! Plate Material
3  11      ! Ullage
3  11      ! Flange
```

Figure 6.8-10 (continued)

```
3  3      ! Void to top of basket
3        ! Outside material

* Hole  3  Bottom Weldment Disk - no structure in the weldment disk support
RZMESH
1        ! number of radial points
83.1850
1        ! number of axial intervals
10.1600
12.7000        ! Coordinates inherited from PLATE Hole
11        ! Plate Material
3        ! Outside material

* Hole  4  Support disk and heat transfer disk stack
PLATE
origin  0  0  110.1598  ! Origin
0  0  1
4
cell  9.7155      ! Sets up a repeating lattice of cells
9.7155  3      ! flood matl
6.2865  3      ! water gap
5.0165  12      ! aluminium disk
1.5875  3      ! water gap
10        ! steel disk

* Hole  5  Flood material model
PLATE
0  0  1
1
444.9861  3      ! Above flooded region
3        ! Flooded region

* Hole  6  Support disk stack lower
PLATE
origin  0  0  22.6060  ! Origin
0  0  1
2
cell  9.7282      ! Sets up a repeating lattice of cells
9.7282  3      ! flood matl
1.5875  3      ! water gap
10        ! steel disk

* Hole  7  Support disk stack upper
PLATE
origin  0  0  275.3233  ! Origin
0  0  1
2
cell  9.7282      ! Sets up a repeating lattice of cells
9.7282  3      ! flood matl
1.5875  3      ! water gap
10        ! steel disk

end
```

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7.0 CONFINEMENT

The Universal Storage System Transportable Storage Canister provides confinement for its radioactive contents in long-term storage. The confinement boundary is closed by welding, creating a solid barrier to the release of contents in all of the design basis normal, off-normal and accident conditions. The welds are visually inspected and nondestructively examined to verify integrity.

The sealed canister contains an inert gas (helium). The confinement boundary retains the helium and also prevents the entry of outside air into the canister in long term storage. The exclusion of air precludes degradation of the fuel rod cladding, over time, due to cladding oxidation failures.

The Universal Storage System canister confinement system is designed, fabricated and tested to assure there will be no credible leakage from the confinement boundary and, therefore, the NAC-UMS® meets the requirements of 10 CFR 72.24 for protection of the public from release of radioactive material [2]. It also meets the requirements of 10 CFR 72.122 for protection of the spent fuel contents in long-term storage such that future handling of the contents would not pose an operational safety concern.

7.1 Confinement Boundary

The transportable storage canister provides confinement of the PWR or BWR contents in long-term storage. The welded canister forms the confinement vessel.

The primary confinement boundary of the canister consists of the canister shell, bottom plate, shield lid, the two port covers, and the welds that join these components. A secondary confinement boundary consists of the canister shell, the structural lid, and the welds that join the structural lid and canister shell. The confinement boundaries are shown in Figures 7.1-1 and 7.1-2. There are no bolted closures or mechanical seals in the primary or secondary confinement boundary. The confinement boundary welds are described in Table 7.1-1.

7.1.1 Confinement Vessel

The canister consists of three principal components: the canister shell, the shield lid, and the structural lid. The canister shell is a right circular cylinder constructed of 0.625-inch thick rolled Type 304L stainless steel plate. The edges of the rolled plate are joined using full penetration welds. It is closed at the bottom end by a 1.75-inch thick circular plate joined to the shell by a full penetration weld. The inside and outside diameters of the canister are 65.81 inches and 67.06 inches, respectively. The canister has a length that is variable, depending on the canister class.

The canister is fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [3], except for approved Code exceptions as listed in Table B3-1 of Appendix B of the CoC.

After loading, the canister is closed at the top by a shield lid and a structural lid. The shield lid is a 7-inch-thick Type 304 stainless steel plate. It is joined to the canister shell using a field installed bevel weld. The shield lid contains the drain and vent penetrations and provides gamma radiation shielding for the operators during the welding, draining, drying and inerting operations. After the shield lid is welded in place, the canister is pressure tested. Following draining, drying and inerting operations, the vent and drain penetrations are closed with Type 304 stainless steel port covers that are welded in place with bevel welds. The shield lid is then helium leakage tested to ensure no credible leakage from the confinement boundary using the evacuated envelope test method in accordance with ANSI N14.5 and ASME Code, Section V. The operating procedures, describing the handling steps to close the canister, are presented in Section 8.1.1. The pressure and leak test procedures are described in Section 9.1.

A secondary, or redundant, confinement boundary is formed at the top of the canister by the structural lid, which is placed over the shield lid. The structural lid is a 3-inch thick Type 304L stainless steel plate. The structural lid provides the attachment points for lifting the loaded canister. The structural lid is welded to the shell using a field installed bevel weld.

The weld specifications and the weld examination and acceptance criteria for the shield lid and stuctural lid welds are presented in Section 7.1.3.2 and Section 9.1.

The confinement boundaries are shown in Figures 7.1-1 and 7.1-2. As illustrated in Figure 7.1-2, the secondary confinement boundary includes the structural lid, the upper 3.2 inches of the canister shell and the joining weld. This boundary provides additional assurance of no credible leakage from the canister during its service life.

7.1.1.1 Design Documents, Codes and Standards

The canister is constructed in accordance with the license drawings presented in Section 1.8. The principal Codes and Standards that apply to the canister design, fabrication and assembly are described in Sections 7.1.1 and 7.1.3, and are shown on the licensing drawings.

7.1.1.2 Technical Requirements for the Canister

The canister confines up to 24 PWR, or 56 BWR, fuel assemblies. Over its 50-year design life, the canister precludes the release of radioactive contents and the entry of air that could potentially damage the cladding of the stored spent fuel. The design, fabrication and testing of the canister to the requirements of the ASME Code Section III, Subsection NB, with approved exceptions as listed in Table B3-1 of Appendix B of the CoC, ensures that the canister maintains confinement in all of the evaluated normal, off-normal, and accident conditions.

The canister has no exposed penetrations, no mechanical closures, and does not employ seals to maintain confinement. There is no requirement for continuous monitoring of the welded closures. The design of the canister allows the recovery of stored spent fuel should it become necessary.

The minimum helium purity level of 99.9% specified in Section 8.1.1 of the Operating Procedures maintains the quantity of oxidizing contaminants to less than one mole per canister for all loading conditions. Based on the calculations presented in Section 4.4.5, the free gas volume of the empty canister yields an inventory of less than 300 moles. Conservatively assuming that all of the impurities in 99.9% pure helium are oxidents, a maximum of 0.3 moles of oxidants could exist in the largest NAC-UMS® canister during storage. By limiting the amount of oxidants to less than one mole, the recommended limits for preventing cladding degradation found in the Pacific Northwest Laboratory, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365 [6] are satisfied.

The design criteria that apply to the canister, as an element of the NAC-UMS® dry storage system, are presented in Table 1.2-1. The design basis parameters of the PWR and BWR spent fuel contents are presented in Section 1.3.

7.1.1.3 Release Rate

The primary confinement boundary is formed by joining the canister confinement boundary stainless steel components by welding. The canister shell longitudinal and girth welds are visually inspected, ultrasonically examined and pressure tested as described in Section 9.1 to confirm integrity. The shield lid welds are liquid penetrant examined following the root and the final weld passes. The shield lid to canister shell weld is pressure tested as described in Section 9.1.2. The structural lid to canister shell multi-pass weld is either: 1) progressively liquid penetrant examined; or 2) ultrasonically examined in conjunction with a liquid penetrant examination of the final weld surface.

To demonstrate confinement integrity of the shield lid to canister shell weld, the leak rate criteria of 1×10^{-7} ref cm^3/sec, or 2×10^{-7} cm^3/sec (helium) at standard conditions, as defined in Section 2.1 of ANSI N14.5-1997, is applied. "Standard" conditions are defined as the leak rate at 298K (25°C) with a one atmosphere pressure differential in the test condition. Since helium at approximately 25°C (77°F) is injected into the canister, at the point of the procedure (Section 8.1.1, Step 32) that the leak test is performed, the actual temperature of the helium is always equal to, or higher than, 25°C due to the decay heat of the contents. This results in a pressure within the canister that is higher than the 0 psig (helium) that is initially established. To ensure that the leak test is conservatively performed, a leak rate of 2×10^{-7} cm^3/sec is used. The higher temperature and higher pressure differential that actually exist in the canister, are conservatively ignored. The sensitivity of the leak test is 1×10^{-7} cm^3/sec (helium). Using this criterion, there is no credible leakage from the canister, and the calculation of the radionuclide inventory and canister leakage is not required. The leak test is described in Section 8.1.1 (Step 47) and in Section 9.1.3.

These steps provide reasonable assurance that the canister confinement boundary does not allow any credible leakage and does not provide a path for the release of any of the content particulates, fission gases, volatiles, corrosion products or fill gases.

7.1.2 Confinement Penetrations

Two penetrations (with quick disconnect fittings) are provided in the canister shield lid for operator use. One penetration is used for draining residual water from the canister. It connects to a drain tube that extends to the bottom of the canister. The other penetration extends only to the underside of the shield lid. It is used to introduce air, or inert gas, into the top of the canister.

Once draining is completed, either penetration may be used for vacuum drying and backfilling with helium. After backfilling, both penetrations are closed with port covers that are welded to the shield lid. When the port covers are in place, the penetrations are not accessible. These port covers are enclosed and covered by the structural lid, which is also welded in place to form the secondary confinement boundary. The structural lid and the remainder of the canister have no penetrations.

7.1.3 Seals and Welds

This section describes the process used to properly assemble the confinement vessel (canister). Weld processes and inspection and acceptance criteria are described in Section 7.1.3.2 and Section 9.1.

No elastomer or metallic seals are used in the confinement boundary of the canister.

7.1.3.1 Fabrication

All cutting, machining, welding, and forming are performed in accordance with Section III, Article NB-4000 of the ASME Code, unless otherwise specified in the approved fabrication drawings and specifications. License drawings are provided in Section 1.8. Code exceptions are listed in Table B3-1 of Appendix B.

7.1.3.2 Welding Specifications

The canister body is assembled using longitudinal and, if required, circumferential shell welds and a circumferential weld to join the bottom plate to the shell.

Weld procedures and qualifications are in accordance with ASME Code Section IX [4]. The welds joining the canister shell are radiographed in accordance with ASME Code Section V, Article 2. The weld joining the bottom plate to the canister shell is ultrasonically examined in accordance with ASME Code Section V, Article 5 [5]. The acceptance criteria for these welds is as specified in ASME Code Section III, NB-5320 (radiographic) and NB-5330 (ultrasonic). The finished surfaces of these welds are liquid penetrant examined in accordance with ASME Code, Section III, NB-5350.

After loading, the canister is closed by the shield lid and the structural lid using field installed groove welds.

After the shield lid is welded in place, the canister is pneumatically (air/nitrogen/helium over water) pressure tested. Following draining, drying and inerting operations, the vent and drain ports are closed with port covers that are welded in place. The root and final surfaces of the shield lid to port cover welds are liquid penetrant examined in accordance with ASME Code Section V, Article 6 for welds requiring multiple passes. For port cover welds completed in a single pass, the final surface is liquid penetrant examined in accordance with the Section V, Article 6 criteria. Acceptance is in accordance with ASME Code Section III, NB-5350. The shield lid-to-canister shell weld is liquid penetrant examined at the root and final surfaces in accordance with ASME Code Section V, Article 6, with acceptance in accordance with ASME Code Section III, NB-5350, and is pressure and leak tested. The operating procedures, describing the handling steps to seal the canister are presented in Section 8.1.1. The pressure and leak test procedures are described in Sections 8.1.1 and 9.1.3.

A redundant confinement boundary is provided at the top of the canister by the structural lid, which is placed over the shield lid. The structural lid is welded to the canister shell using a field-installed groove weld. The structural lid to canister shell weld is either: 1) ultrasonically examined (UT) in accordance with ASME Code Section V, Article 5, with the final weld surface liquid penetrant (PT) examined in accordance with ASME Code Section V, Article 6; or, 2) progressive liquid penetrant examined in accordance with ASME Code Section V, Article 6. Acceptance criteria are specified in ASME Code Section III, NB-5330 (UT) and NB-5350 (PT).

All welding procedures are written and qualified in accordance with Section IX of the ASME Code. Each welder and welding operator must be qualified in accordance with Section IX of the ASME Code.

7.1.3.3 Testing, Inspection, and Examination

The detailed inspection, nondestructive examination and test programs for the confinement vessel and components are described in Chapter 9.

7.1.4 Closure

The primary closure of the transportable storage canister consists of the welded shield lid and the two welded port covers. There are no bolted closures or mechanical seals in the primary closure. A secondary closure is provided at the top end of the canister by the structural lid. The structural lid, when welded to the canister shell, fully encloses the shield lid and the port covers.

Figure 7.1-1 Transportable Storage Canister Primary and Secondary Confinement Boundaries

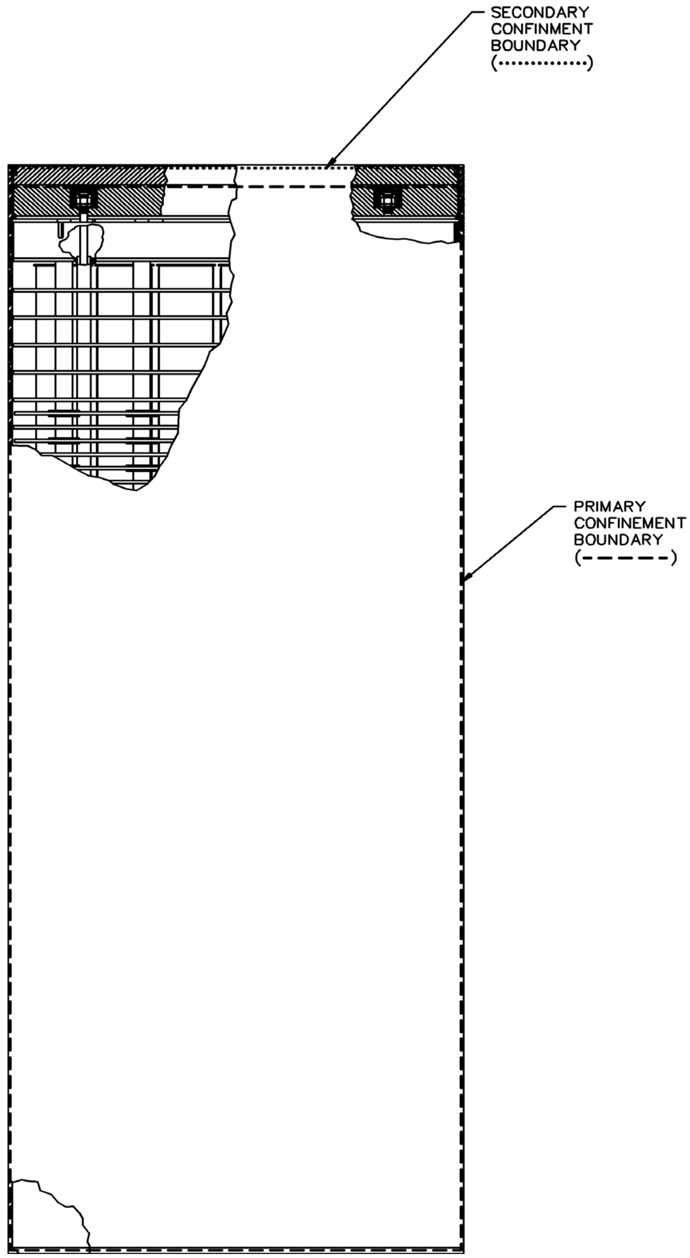

Figure 7.1-2 Confinement Boundary Detail at Shield Lid Penetration

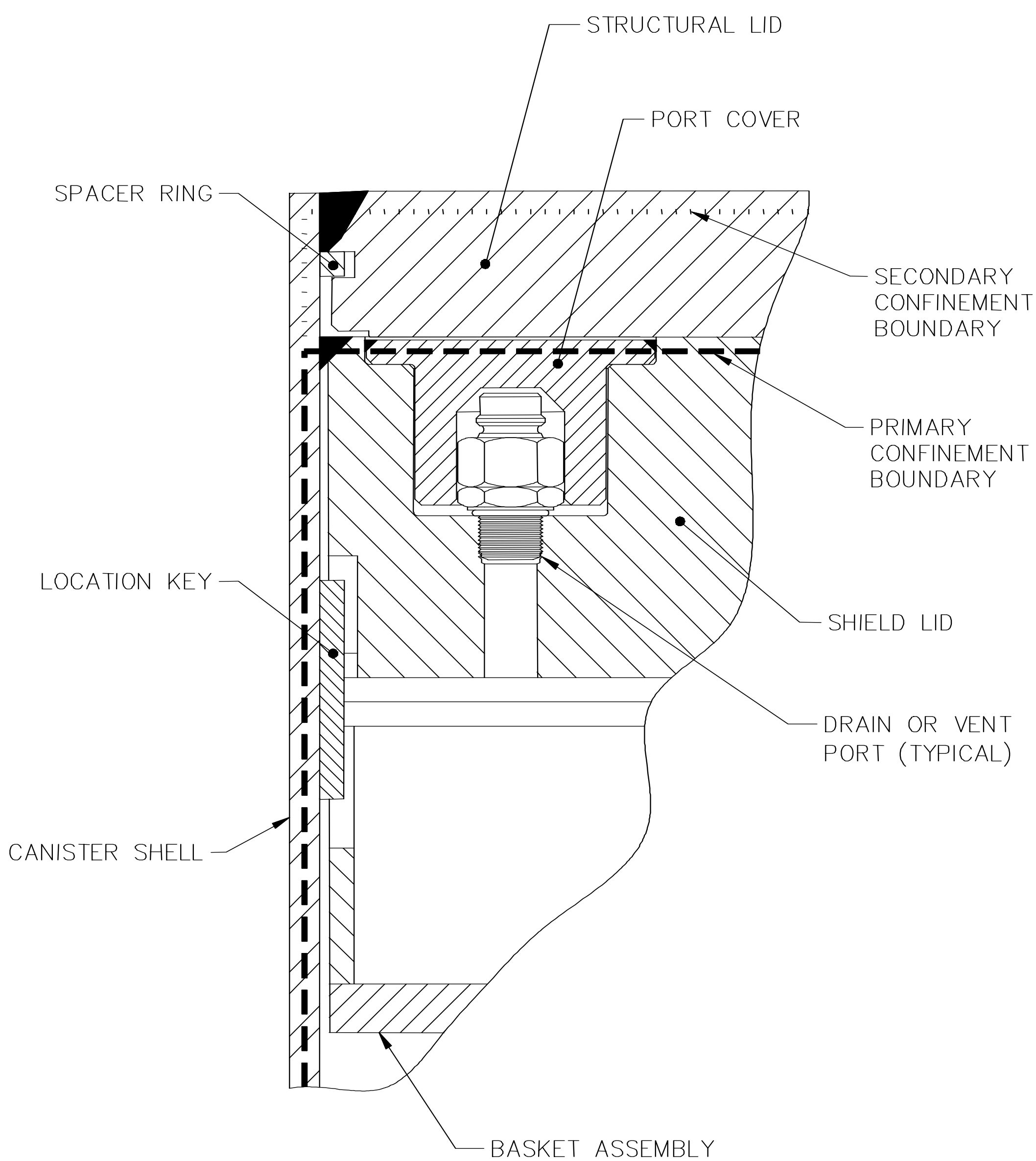

Table 7.1-1 Canister Confinement Boundary Welds

Confinement Boundary Welds		
Weld Location	**Weld Type**	**ASME Code Category (Section III, Subsection NB)**
Shell longitudinal	Full penetration groove (shop weld)	A
Shell circumferential (if used)	Full penetration groove (shop weld)	B
Bottom plate to shell	Full penetration groove (shop weld)	C
Shield lid to shell	Bevel (field weld)	C
Structural lid to shell	Bevel (field weld)	C
Vent and drain port covers to shield lid	Bevel (field weld)	C

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7.2 Requirements for Normal Conditions of Storage

The canister is transferred to a vertical concrete cask using a transfer cask. During this transfer, the canister is subject to handling loads. The evaluation of the canister for normal handling loads is provided in Section 3.4.4. The principal design criteria for the Universal Storage System are provided in Table 2-1.

Once the canister is placed inside of the vertical concrete storage cask, it is effectively protected from direct loading due to natural phenomena, such as wind, snow and ice loading. The principal direct loading for normal operating conditions arises from increased internal pressure caused by decay heat, solar insolation, and ambient temperature. The effect of the normal operating internal pressure is evaluated in Section 3.4.4.

7.2.1 Release of Radioactive Material

The structural analysis of the canister for normal conditions of storage presented in Section 3.4.4 shows that the canister is not breached in any of the normal operating events. Consequently, there is no release of radioactive material during normal conditions of storage.

7.2.2 Pressurization of the Confinement Vessel

The canister is vacuum dried and backfilled with helium at one atmosphere absolute prior to installing and welding the penetration port covers. In normal service, the internal pressure increases due to an increase in temperature of the helium and due to the postulated failure of fuel rod cladding of 3% of the fuel rods, which releases 30% of the available fission gases in those rods.

The canister, shield lid, fittings, and the canister basket are fabricated from materials that do not react with ordinary or borated spent fuel pool water to generate gases. The aluminum heat transfer disks are protected by an oxide film that forms shortly after fabrication. This oxide layer effectively precludes further oxidation of the aluminum components or other reaction with water in the canister at temperatures less than 200°F, which is higher than the typical spent fuel pool water temperature. The neutron absorber criticality control poison plates in the fuel baskets are retained by a welded stainless steel cover. No steels requiring protective coatings or paints are used in the PWR configuration canister, shield lid, fittings, or basket, or in the BWR configuration canister, shield lid, or fittings. Carbon steel support disks are used in the BWR configuration basket. These disks are completely coated to protect the disks in immersion in the spent fuel pool, as defined on Drawing 790-573. The consequence of the use of a coating in BWR spent fuel pools is evaluated in Sections 3.4.1.2.3 and 3.4.1.2.4. That evaluation shows that no adverse interactions result from the use of the coating. The coating does not contain Zinc, and no gases are formed as a result of the exposure of this coating to the neutrally buffered water used in BWR spent fuel pools.

Since the canister is vacuum dried and backfilled with helium prior to sealing, no significant moisture or gases, such as air, remain in the canister. Consequently, there is no potential that radiolytic decomposition could cause an increase in canister internal pressure or result in a build up of explosive gases in the canister.

The calculation of the canister pressure increase based on these conditions is less than the pressure evaluated in Section 3.4.4 for the maximum normal operating pressure. As shown in Section 3.4.4, there are no adverse consequences due to the internal pressure resulting from normal storage conditions.

Since the containment boundary is closed by welding and contains no seals or O-rings, and since the boundary is not ruptured or otherwise compromised in normal handling events, no leakage of contents occurs in normal conditions.

7.3 Confinement Requirements for Hypothetical Accident Conditions

The evaluation of the canister for off-normal and accident condition loading is provided in Sections 11.1 and 11.2, respectively.

Once the canister is placed inside the vertical concrete cask, it is effectively protected from direct loading due to natural phenomena, such as seismic events, flooding and tornado (wind driven) missiles. Accident conditions assume the cladding failure of all the fuel rods stored in the canister. Consequently, there is an increase in canister internal pressure due to the release of a fraction of the fission product and charge gases. The accident conditions internal pressure for the PWR and BWR configurations is calculated in Section 11.2.1.

For evaluation purposes, a class of events identified as off-normal is also considered in Section 11.1. The off-normal class of events is not considered here, since off-normal conditions are bounded by the hypothetical accident conditions.

The structural analysis of the canister for off-normal and accident conditions of storage, presented in Chapter 11, show that the canister is not breached in any of the evaluated events. Consequently, there is no credible leakage of radioactive material from the confinement boundary during off-normal or accident conditions of storage.

The resulting site boundary dose due to a hypothetical accident is, therefore, less than the 5 rem whole body or organ (including skin) dose at 100 meter minimum boundary required by 10 CFR 72.106 (b) for accident exposures.

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7.4 Confinement Evaluation for Site Specific Spent Fuel

This section presents the confinement evaluation for fuel assembly types or configurations, which are unique to specific reactor sites. Site specific spent fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel rod or assemblies that are classified as damaged.

The Transportable Storage Canister is designed, fabricated and tested to assure there will be no credible leakage from the confinement boundary, as described in Section 7.1. Consequently, site-specific fuel configurations need be evaluated only if the configuration results in a modification of the confinement boundary of the canister that is intended for use or when the configuration could result in a higher internal pressure or temperature than is used in the design basis analysis.

7.4.1 Confinement Evaluation for Maine Yankee Site Specific Spent Fuel

Maine Yankee site specific spent fuel is to be stored in either the Class 1 or Class 2 Transportable Storage Canister, depending on the overall length of the fuel assembly, including inserted non fuel-bearing components. These canisters are closed by welding and are inspected and tested to assure no credible leakage from the confinement boundary.

Site specific fuel includes fuel having variable enrichment radial zoning patterns and annular axial fuel blankets, removed fuel rods or empty rod positions, fuel rods placed in guide tubes, consolidated fuel, damaged fuel, and high burnup fuel (fuel with a burnup between 45,000 MWd/MTU and 50,000 MWd/MTU). These configurations are not included in the standard fuel analysis, but are present in the site fuel inventory that must be stored. As discussed in Section 4.5.1, the site specific fuel configurations do not result in a canister pressure or temperature that exceeds the canister design basis. Therefore, there is no credible leakage from a canister containing Maine Yankee high burnup fuel rods site-specific spent fuel.

Undamaged site specific fuel is loaded directly into the fuel tubes in the PWR basket. Damaged fuel is inserted into one of the two configurations of the Maine Yankee Fuel Can, shown in Drawings 412-501 and 412-502, which precludes the release of gross particulate material from the fuel can. The fuel can is sized to allow its insertion into a fuel position in the PWR basket.

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7.5 References

1. ANSI N14.5-1997, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," American National Standards Institute, 1997.

2. Title 10 of the Code of Federal Regulations, Part 72 (10 CFR 72), "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation," April 1996 Edition.

3. ASME Boiler and Pressure Vessel Code, Section III, Division I, "Rules for Construction of Nuclear Power Plant Components," 1995 Edition with 1995 Addenda.

4. ASME Boiler and Pressure Vessel Code, Section IX, "Qualification Standard for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators," 1995 Edition with 1995 Addenda.

5. ASME Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination," 1995 Edition with 1995 Addenda.

6. PNL-6365, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," Pacific Northwest Laboratory, Richland, Washington, November, 1987.

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8.0 OPERATING PROCEDURES

This chapter provides general guidance for operating the Universal Storage System. Three operating conditions are addressed. The first is loading the transportable storage canister, installing it in the vertical concrete cask, and transferring it to the storage (Independent Spent Fuel Storage Installation (ISFSI)) pad. The second is the removal of the loaded canister from the concrete storage cask. The third is opening the canister to remove spent fuel in the unlikely event that this should be necessary.

The operating procedure for transferring a loaded canister from a storage cask to the Universal Transport Cask, is described in Section 7.2.2 of the UMS® Universal Transport Cask Safety Analysis Report. [1]

Users shall develop written and approved site-specific procedures that implement the operational sequences presented in the procedures in this chapter. These procedures present the general guidance for operations and the establishment of the process in which Technical Specification limits and requirements presented in Appendix A of Certificate of Compliance No. 72-1015 are met. The procedures provide the guidance and basis for the development and implementation of more detailed site-specific operating and test procedures required of the NAC-UMS® Storage System user. A departure from the specific way in which a given operational activity is performed may result from variations in specific site equipment or operational philosophy. Site-specific procedures shall also incorporate site-specific Technical Specifications, surveillance requirements, administrative controls, and other limits appropriate to the use of the NAC-UMS® Storage System to ensure that system/component design function is maintained. The user's site-specific procedures shall incorporate spent fuel assembly selection and verification requirements to ensure that the spent fuel assemblies loaded into the UMS® Storage System are as authorized by the Approved Contents and Design Features presented in Appendix B of the CoC Number 1015 Technical Specifications and the Certificate of Compliance.

Operation of the Universal Storage System requires the use of ancillary equipment items. An example listing of ancillary equipment normally required for system operation is shown in Table 8.1.1-1. Alternative ancillary equipment such as heavy-haul trailer and canister lifting devices may be utilized based on a site-specific evaluation. When a specific ancillary equipment item is referred to in the procedure, alternative ancillary equipment is allowable (i.e., vertical cask transporter, canister lifting systems, etc.). The system does not rely on the use of bolted closures, but bolts are used to secure retaining rings and lids. The hoist rings used for lifting the shield lid and canister have threaded fittings. Table 8.1.1-2 provides the torque values for installed bolts and hoist rings. Supplemental shielding may be employed to reduce radiation exposure for certain of the tasks specified by these procedures. Use of supplemental shielding is at the discretion of the User.

The design of the Universal Storage System is such that the potential for spread of contamination during handling and future transport of the canister is minimized. The transportable storage canister is loaded in the spent fuel pool but is protected from gross contact with pool water by a jacket of clean or filtered pool water while it is in the transfer cask. Clean water is processed or filtered pool water, or any water external to the spent fuel pool that has a water chemistry that is compatible with use in the pool. Only the top of the open canister is exposed to contaminated pool water. The top of the canister is closed by the structural lid, which is not contaminated when it is installed. Consequently, the canister external surface is expected to be essentially free of contamination. There are no radioactive effluents from the canister or the concrete cask in routine operations or in the design basis accident events.

The guidance procedures described in this chapter allow the cask user to develop site-specific procedures that minimize the dose to the operators in accordance with As Low As Reasonably Achievable (ALARA) principles.

A training program is described in Section A 5.0 of Appendix A of the CoC Number 1015 Technical Specifications, that is intended to assist the User in complying with the training and dry run requirements of 10 CFR 72. This program addresses the controls and limits applicable to the UMS® Storage System. It also addresses the system operational features and requirements.

8.1 Procedures For Loading the Universal Storage System

The Universal Storage System consists of three principal components: the transportable storage canister (canister), the transfer cask, and the vertical concrete cask. The transfer cask is used to hold the canister during loading and while the canister is being closed and sealed. The transfer cask is also used to transfer the canister to the concrete cask and to load the canister into the transport cask. The principal handling operations involve closing and sealing the canister by welding, and placing the loaded canister in the vertical concrete cask. The typical vent and drain port locations are shown in Figure 8.1.1-1.

The transfer cask is provided in either the Standard or Advanced configuration that weigh approximately 121,500 pounds each, depending on Class. Canister handling, fuel loading and canister closing are operationally identical for either transfer cask configuration. Either transfer cask can accommodate an extension fixture to allow the use of the next longer length canister. The user shall verify that the appropriate extension is installed and torqued prior to initiating the canister-loading process.

This procedure assumes that the canister with an empty basket is installed in the transfer cask, that the transfer cask is positioned in the decontamination area or other suitable work station, and that the vertical concrete cask is positioned in the plant cask receiving area or other suitable staging area. The transfer cask extension must be installed on the transfer cask if its use is required. To facilitate movement of the transfer cask to the concrete cask, the staging area should be within the operational "footprint" of the cask handling crane. The concrete cask may be positioned on a heavy-haul transporter, or on the floor of the work area.

The User must ensure that the fuel assemblies selected for loading conform to the Approved Contents provisions of Section B2.0 of Appendix B of the CoC Number 1015 Technical Specifications. Fuel assembly loading may also be administratively controlled to ensure that fuel assemblies with site-specific characteristics are preferentially loaded in specified positions in the canister. Preferential loading requirements for site-specific fuel are described in Section B2.1.2 of Appendix B of the CoC Number 1015 Technical Specifications.

Certain steps of the procedures in this section may be completed out of sequence to allow for operational efficiency. Changing the order of these steps, within the intent of the procedures, has no effect on the safety of the canister loading process and does not violate any requirements stated in the Technical Specifications. These steps include the placement and installation of air pads and the sequence and use of an annulus fill system, including optional seals and/or foreign material exclusion devices.

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8.1.1 Loading and Closing the Transportable Storage Canister

1. Visually inspect the basket fuel tubes to ensure that they are unobstructed and free of debris. Ensure that the welding zones on the canister, shield, and structural lids, and the port covers are prepared for welding. Ensure transfer cask door lock bolts/lock pins are installed and secure.
2. Fill the canister with clean water until the water is about 4 inches from the top of the canister.

   Note: Do not fill the canister completely in order to avoid spilling water during the transfer to the spent fuel pool.

   Note: If fuel loading requires boron credit, the minimum boron concentration of the water in the canister must be at least 1,000 ppm (boron), in accordance with LCO 3.3.1.
3. Install the annulus fill system to transfer cask, including the clean water lines.
4. If it is not already attached, attach the transfer cask lifting yoke to the cask handling crane, and engage the transfer cask lifting trunnions.

   Note: The minimum temperature of the transfer cask (i.e., surrounding air temperature) must be verified to be higher than 0°F prior to lifting, in accordance with Section B3.4.1 (8) of Appendix B of the CoC Number 1015 Technical Specifications.
5. Raise the transfer cask and move it over the pool, following the prescribed travel path.
6. Lower the transfer cask to the pool surface and turn on the clean water line to fill the canister and the annulus between the transfer cask and canister.
7. Lower the transfer cask as the annulus fills with clean water until the trunnions are at the surface, and hold that position until the clean water overflows through the upper fill lines or annulus of the transfer cask. Then lower the transfer cask to the bottom of the pool cask loading area.

   Note: If an intermediate shelf is used to avoid wetting the cask handling crane hook, follow the plant procedure for use of the crane lift extension piece.
8. Disengage the transfer cask lifting yoke to provide clear access to the canister.
9. Load the previously designated fuel assemblies into the canister.

   Note: Contents must be in accordance with the Approved Contents provisions of Section B2.0 of Appendix B of the CoC Number 1015 Technical Specifications.

   Note: Contents shall be administratively controlled to ensure that fuel assemblies with certain site-specific characteristics are preferentially loaded in specified positions in the basket. Preferential loading requirements for site-specific fuel are presented in Section B2.1.2 of Appendix B of the CoC Number 1015 Technical Specifications.

10. Attach a three-legged sling to the shield lid using the swivel hoist rings. Torque hoist rings in accordance with Table 8.1-2. Attach the suction pump fitting to the vent port.

    Caution: Verify that the hoist rings are fully seated against the shield lid.

    Note: Ensure that the shield lid key slot aligns with the key welded to the canister shell.

11. Using the cask handling crane, or auxiliary hook, lower the shield lid until it rests in the top of the canister.

12. Raise the transfer cask until its top just clears the pool surface. Hold at that position, and using a suction pump, drain the pool water from above the shield lid. After the water is removed, continue to raise the cask. Note the time that the bottom of the transfer cask clears the spent fuel pool water. Operations through Step 28 must be completed in accordance with the time limits presented in Table 8.1.1-3. The "time in water" clock is to be initiated if the lifting of the transfer cask from the pool is interrupted with the cask partially removed from the pool.

    Note: For the PWR configuration, in the event that the drain time limit is not met, either forced air or in-pool cooling, or monitoring the water temperature (see following note) is required. Forced air cooling is implemented by supplying 375 CFM air with a maximum temperature of 76°F to the 8 transfer cask lower inlets. Forced air or in-pool cooling of the canister shall be maintained for a minimum of 24 hours. After 24 hours, the cooling may be discontinued based on heat load as follows:

    Time Periods for Discontinued Cooling after 24 Hours

Heat Load (kW)	For Forced Air Cooling (hrs)	For In-Pool Cooling (hrs)
$20 < L \leq 23$	4	15
$17.6 < L \leq 20$	7	18
$14 < L \leq 17.6$	11	22
$11 < L \leq 14$	14	24
$8 < L \leq 11$	20	29
$L \leq 8$	28	34

    Note: Alternately, the temperature of the water in the canister may be used to establish the time for completion through Step 28 for the PWR configuration. Those operations must be completed within 2 hours of the time that the canister water reaches the temperatures shown in the following table. For this alternative, the water temperature must be determined every 2 hours beginning at the time shown in the following table after the time the transfer cask is removed from the pool.

Heat Load (kW)	Canister Water Temperature (ºF)	Time to Start Temperature Measurement (hrs)
20 < L ≤ 23	180	18
17.6 < L ≤ 20	180	21
14 < L ≤ 17.6	180	25
11 < L ≤ 14	170	28
8 < L ≤ 11	160	33
L ≤ 8	150	38

Note: As an alternative, some sites may choose to perform welding operations for closure of the canister in a cask loading pit with water around the canister (below the trunnions) and in the annulus. This alternative provides additional shielding during the closure operation. If this alternative is implemented, the start time for compliance with Table 8.1.1-3 limits, as defined in Step 12, begins when the top of the canister is above the pool water surface (i.e., no longer fully submerged).

13. As the cask is raised, spray the transfer cask outer surface with clean water to wash off any gross contamination.
14. When the transfer cask is clear of the pool surface, but still over the pool, turn off the clean water flow to the annulus, remove hoses and allow the annulus water to drain to the pool. Move the transfer cask to the decontamination area or other suitable work station.
    Note: Access to the top of the transfer cask is required. A suitable work platform may need to be erected.
15. Verify that the shield lid is level and centered.
16. Attach the suction pump to the suction pump fitting on the vent port. Operate the suction pump to remove free water from the shield lid surface. Disconnect the suction pump and suction pump fitting. Remove any free standing water from the shield lid surface and from the vent and drain ports.
17. Decontaminate the top of the transfer cask and shield lid as required to allow welding and inspection activities.
    Note: Supplemental shielding may be used for activities around the shield lid.

`18. Insert the drain tube assembly with a female quick-disconnect attached through the drain port of the shield lid into the basket drain tube sleeve. Remove the female quick-disconnect. Torque the drain tube assembly by hand until metal-to-metal contact is achieved; then torque to 135 ± 15 ft-lbs for Furon metal seals or 115 ± 5 ft-lbs for elastomer seals (EPDM or Viton). Install a quick-disconnect in the vent port.

19. Connect the suction pump to the drain port. Verify that the vent port is open. Remove approximately 70 gallons of water from the canister. Disconnect and remove the pump. Caution: Radiation level may increase as water is removed from the canister.
20. Install the automatic welding equipment, including the supplemental shield plate.
21. Attach the hydrogen gas detector to the vent port. Verify that the concentration of any detectable hydrogen gas in the free volume beneath the shield lid is less than 2.4%. Continue monitoring for hydrogen gas during completion of the shield lid root pass weld.

    Note: If, at any time, the hydrogen gas concentration exceeds 2.4%, stop welding operations and connect and operate the vacuum system, or use a gas purge through the vent port to remove the gases from beneath the shield lid. Reverify that the hydrogen gas concentration beneath the shield lid is less than 2.4%. Disconnect and remove the vacuum or purging system.
22. Operate the welding equipment to complete the root weld joining the shield lid to the canister shell following approved procedures. Remove the hydrogen detector from the vent tube. Leave the connector and vent tube installed to vent the canister.
23. Examine the root weld using liquid penetrant and record the results.
24. Complete welding of the shield lid to the canister shell.
25. Liquid penetrant examine the final weld surface and record the results.
26. Attach a regulated air, nitrogen or helium supply line to the vent port. Install a fitting on the drain port. Pressurize the canister to 35 psia and hold the pressure. There must be no loss of pressure for a minimum of 10 minutes.
27. Release the pressure.

    Note: As an option, an informational helium leak test may be conducted at this point using the following steps (the record leak test is performed at Step 47).

    27a. Evacuate and backfill the canister with helium having a minimum purity of 99.9% to a pressure of 18.0 psia.

    27b. Using a helium leak detector ("sniffer" detector) with a test sensitivity of 5×10^{-5} cm^3/sec (helium), survey the weld joining the shield lid and canister shell.

    27c. At the completion of the survey, vent the canister helium pressure to one atmosphere (0 psig).
28. Drain the canister.

    Drain the remaining water from the canister cavity (typically, the process ranges from 1 to 2 hours). Draining of the canister may be performed by suction, by a blow-down gas pressure of 15-18 psig, or by a combination of suction and a blow-down gas pressure of 15-18 psig. After removal of the water from the canister, disconnect the equipment from the canister. Note the time that the last free water is removed from the canister cavity. If not already installed, install a quick-disconnect to the open vent port.

    Caution: Radiation levels at the top and sides of the transfer cask will rise as water is removed.

Note: If the canister draining operation is interrupted or only partially completed, the canister shall be refilled with water prior to start of the auxiliary cooling operations (i.e., forced air or in-pool cooling), per the Note following Step 12.

Note: The total time duration from the completion of draining the water from the canister (Step 28), or from completion of either in-pool or forced air cooling, through completion of dryness verification testing per LCO 3.1.2 (Step 31) and the completion of the helium backfill process per LCO 3.1.3 (Step 32) shall be controlled and monitored in accordance with the surveillance requirements and actions of LCO 3.1.1.

29. Attach the vacuum equipment to the vent and drain ports. Dry any free standing water in the vent and drain port recesses.
30. Operate the vacuum equipment until a vacuum of ≤10 mm of mercury exists in the canister and isolate the vacuum pump and turn the pump off.
31. Verify that no water remains in the canister by holding the vacuum of ≤10 mm of mercury for a minimum of 10 minutes. If water is present in the cavity, the pressure will rise as the water vaporizes. Continue the vacuum/hold cycle until the conditions of LCO 3.1.2 are met.

    Precaution: If the spent fuel pool water temperature for canisters vacuum dried in the pool, or the cask preparation area ambient temperature for canisters vacuum dried outside the pool is below 65°F, the vacuum drying of the canister shall be extended below the standard pressure value of ≤ 10 mm Hg until a cavity pressure of ≤5 mm Hg is achieved. The dryness verification shall be performed and meet the acceptance criteria as specified in LCO 3.1.2, but limiting any pressure rise during the 10-minute hold period to ≤5 mm Hg.
32. Evacuate the cavity until a vacuum of ≤3 mm of mercury exists and backfill the canister cavity with helium having a minimum purity of 99.9% to a pressure of one atmosphere (0 psig).

    Note: Canister helium backfill pressure must conform to the requirements of LCO 3.1.3.

    Note: Monitor the time from this step (completion of helium backfill) until completion of canister transfer into and closure of the concrete cask in accordance with LCO 3.1.4.
33. Disconnect the vacuum and helium supply lines from the vent and drain ports. Dry any residual water that may be present in the vent and drain port cavities.
34. Install the vent and drain port covers.
35. Complete the root pass weld of the drain port cover to the shield lid.

    Note: If the drain port cover weld is completed in a single pass, the weld final surface is liquid penetrant inspected in accordance with Step 38.
36. Prepare the weld and perform a liquid penetrant examination of the root pass. Record the results.

37. Complete welding of the drain port cover to the shield lid.
38. Prepare the weld and perform a liquid penetrant examination of the drain port cover weld final pass. Record the results.
39. Complete the root pass weld of the vent port cover to the shield lid.

    Note: If the vent port cover weld is completed in a single pass, the weld final surface is liquid penetrant inspected in accordance with Step 42.
40. Prepare the weld and perform a liquid penetrant examination of the root pass. Record the results.
41. Complete welding of the vent port cover to the shield lid.
42. Prepare the weld and perform a liquid penetrant examination of the weld final surface. Record the results.
43. Remove the welding machine and any supplemental shielding used during shield lid closure activities.
44. Install the helium leak test fixture.
45. Attach the vacuum line and leak detector to the leak test fixture fitting.
46. Operate the vacuum system to establish a vacuum in the leak test fixture.
47. Operate the helium leak detector to verify that there is no indication of a helium leak exceeding 2×10^{-7} cm^3/second, at a minimum test sensitivity of $\leq 1 \times 10^{-7}$ cm^3/second helium, in accordance with the requirements of LCO 3.1.5.
48. Release the vacuum and disconnect the vacuum and leak detector lines from the fixture.
49. Remove the leak test fixture.
50. Attach a three-legged sling to the structural lid using the swivel hoist rings.

    Caution: Ensure that the hoist rings are fully seated against the structural lid. Torque the hoist rings in accordance with Table 8.1.1-2. Verify that the spacer ring is in place on the structural lid.

    Note: Verify that the structural lid is stamped or otherwise marked to provide traceability of the canister contents.
51. Using the cask handling crane or the auxiliary hook, install the structural lid in the top of the canister. Verify that the structural lid is flush with, or protrudes slightly above, the canister shell. Verify that the gap in the spacer ring is not aligned with the shield lid alignment key. Remove the hoist rings.
52. Install the automatic welding equipment on the structural lid including the supplemental shield plate.
53. Operate the welding equipment to complete the root weld joining the structural lid to the canister shell.

54. Prepare the weld and perform a liquid penetrant examination of the weld root pass. Record the results.
55. Continue with the welding procedure, examining the weld at 3/8-inch intervals using liquid penetrant. Record the results of each intermediate and the final examination.

    Note: If ultrasonic testing of the weld is used, testing is performed after the weld is completed.
56. Remove the weld equipment and supplemental shielding.
57. Install the transfer cask retaining ring. Torque bolts to 155 ± 10 ft-lbs. (Table 8.1.1-2).
58. Decontaminate the external surface of the transfer cask to the limits established for the site.

Figure 8.1.1-1 Typical Vent and Drain Port Locations

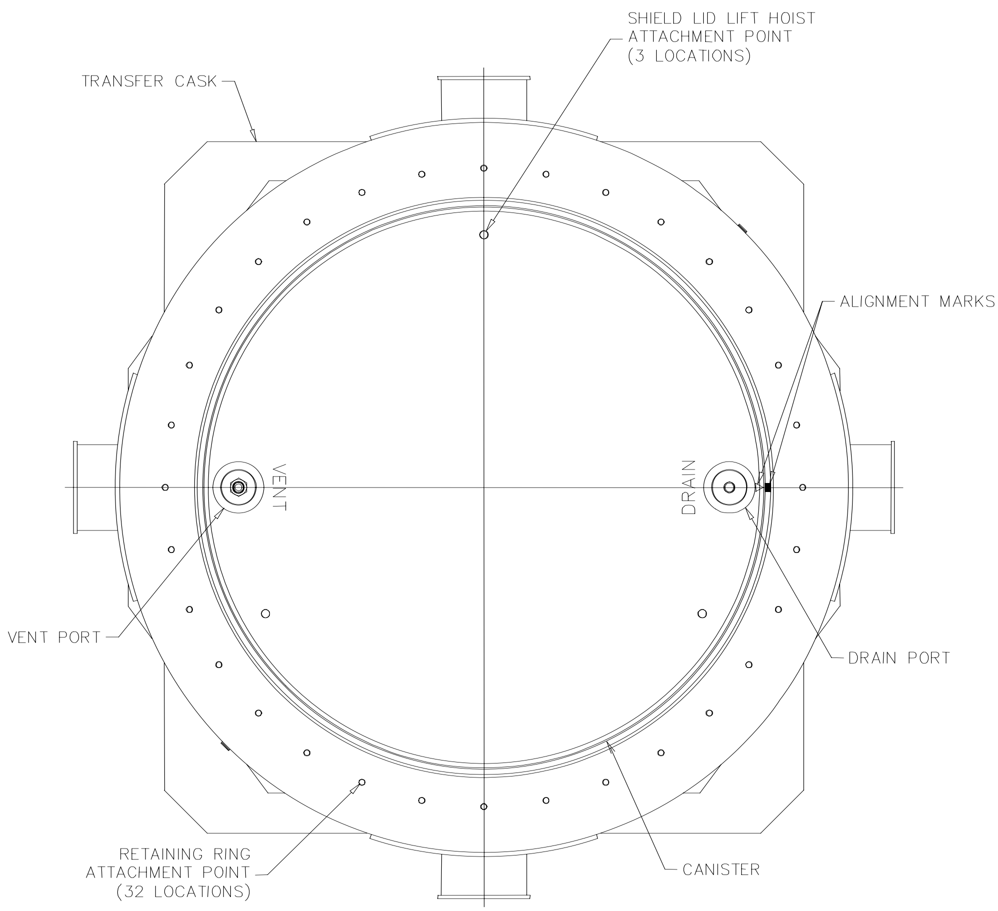

Table 8.1.1-1 List of Principal Ancillary Equipment

Item	Description
Transfer Cask Lifting Yoke	Required for lifting and moving the transfer cask.
Heavy-Haul Transporter (Optional)	Heavy-haul (double drop frame) trailer required for moving the loaded and empty vertical concrete cask to and from the ISFSI pad.
Mobile Lifting Frame (Optional)	A self-propelled or towed A-frame lifting device for the concrete cask. Mobile Lifting Frame is used to lift the cask and move it using two lifting lugs in the top of the concrete cask.
Helium Supply System	Supplies helium to the canister for helium backfill and purging operations.
Vacuum Drying System	Used for evacuating the canister. Used to remove residual water, air and initial helium backfill.
Automated Welding System	Used for welding the shield lid and structural lid to the canister shell.
Self-Priming Pump	Used to remove water from the canister.
Shield Lid Sling	A three-legged sling used for lifting the shield lid. It is also used to lift the concrete cask shield plug and lid.
Redundant Canister Lifting Sling System (1)	A set of 2 three-legged slings used for lifting the structural lid by itself, or for lifting the canister when the structural lid is welded to it. The slings are configured to provide for simultaneous loading during the canister lift.
Transfer Adapter	Used to align the transfer cask to the vertical concrete cask or the Universal Transport Cask. Provides the platform for the operation of the transfer cask shield doors.
Transfer Cask Extension	A carbon steel ring used to extend the height of the transfer cask when using the next longer size canister.
Hydraulic Unit	Operates the shield doors of the transfer cask.
Lift Pump Unit	Jacking system for raising and lowering the concrete cask.
Air Pad Rig Set	Air cushion system used for moving the concrete cask.
Supplemental Shielding Fixture	An optional carbon steel fixture inserted in the Vertical Concrete Cask air inlets to reduce radiation dose rates at the inlets.

(1) Note: Alternative canister lifting systems may be utilized based on a site-specific analysis and evaluation.

Table 8.1.1-2 Torque Values

Fastener	**Torque Value (ft-lbs)**	**Torque Pattern**
Transfer Adapter Bolts (Optional)	40 ± 5	None
Transfer Cask Retaining Ring	155 ±10	0º, 180º, 270º and 90º in two passes
Transfer Cask Extension	155 ±10	None
Vertical Concrete Cask Lid	40 ± 5	None
Lifting Hoist Rings – Canister Structural Lid		None
Lid Only	Hand Tight	
Loaded Canister	800 +80, -0	
Canister Lid Plug Bolts Shield Lid Plug Bolts	Hand Tight Hand Tight	None None
Transfer Cask Door Lock Bolts	Hand Tight	None
Canister Drain Tube	135 ± 15 (Furon metal seals) or 115 ± 5 (elastomer seals, EPDM or Viton)	None

Table 8.1.1-3 Handling Time Limits Based on Decay Heat Load with Canister Full of Water

Total Heat Load (L) (kW)	PWR Time Limit (Hours)	BWR Time Limit (Hours)
$20.0 < L \leq 23.0$	20	17
$17.6 < L \leq 20.0$	23	17
$14.0 < L \leq 17.6$	27	17
$11.0 < L \leq 14.0$	30	17
$8.0 < L \leq 11.0$	35	17
$L \leq 8.0$	40	17

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8.1.2 Loading the Vertical Concrete Cask

This section of the loading procedure assumes that the vertical concrete cask is located on the bed of a heavy-haul transporter, or on the floor of the work area, under a crane suitable for lifting the loaded transfer cask. The vertical concrete cask shield plug and lid are not in place, and the bottom pedestal plate cover is installed.

1. Using a suitable crane, place the transfer adapter on the top of the concrete cask.
2. If using the transfer adapter bolt hole pattern for alignment, align the adapter to the concrete cask. Bolt the adapter to the cask using four (4) socket head cap screws. (Note: Bolting of the transfer adapter to the cask is optional.)
3. Verify that the shield door connectors on the adapter plate are in the fully extended position.
   Note: Steps 4 through 6 may be performed in any order, as long as all items are completed.
4. If not already done, attach the transfer cask lifting yoke to the cask handling crane. Verify that the transfer cask retaining ring is installed.
5. Install six (6) swivel hoist rings in the structural lid of the canister and torque to the value specified in Table 8.1.1-2. Attach two (2) three-legged slings to the hoist rings.
   Caution: Ensure that the hoist rings are fully seated against the structural lid.
6. Stack the slings on the top of the canister so they are available for use in lowering the canister into the storage cask.
7. Engage the transfer cask trunnions with the transfer cask lifting yoke. Ensure that all lines are disconnected from the transfer cask.
   Note: The minimum temperature of the transfer cask (i.e., temperature of the surrounding air) must be verified to be higher than 0°F prior to lifting, in accordance with Section B 3.4.1(7) of Appendix B of the CoC Number 1015 Technical Specifications.
8. Raise the transfer cask and move it over the concrete cask. Lower the transfer cask, ensuring that the transfer cask shield door rails and connector tees align with the adapter plate rails and door connectors. Prior to final set down, remove transfer cask shield door lock bolts/lock pins (there is a minimum of one per door), or the door stop, as appropriate.
9. Ensure that the shield door connector tees are engaged with the adapter plate door connectors.
10 Disengage the transfer cask yoke from the transfer cask and from the cask handling crane hook.

11. Return the cask handling crane hook to the top of the transfer cask and engage the two (2) three-legged slings attached to the canister.

    Caution: The top connection of the three-legged slings must be at least 75 inches above the top of the canister.

12. Lift the canister slightly (about ½ inch) to take the canister weight off of the transfer cask shield doors.

    Note: A load cell may be used to determine when the canister is supported by the crane.

    Caution: Avoid raising the canister to the point that the canister top engages the transfer cask retaining ring, as this could result in lifting the transfer cask.

13. Using the hydraulic system, open the shield doors to access the concrete cask cavity.

14. Lower the canister into the concrete cask, using a slow crane speed as the canister nears the pedestal at the base of the concrete cask. The support ring may be used to aid in centering the canister during the lowering of the canister into the concrete cask.

15. When the canister is properly seated, disconnect the slings from the canister at the crane hook, and close the transfer cask shield doors.

16. Retrieve the transfer cask lifting yoke and attach the yoke to the transfer cask.

17. Lift the transfer cask off of the vertical concrete cask and return it to the decontamination area or designated work station.

    Note: The canister is intended to be centered in the concrete cask, but the final position of the canister may result in the canister shell being as close as one inch from the concrete cask liner due to system component alignment.

    Note: Perform removable contamination surveys on the canister exterior and/or transfer cask interior surfaces as required to confirm canister surface contamination is less than the limits specified in Technical Specification LCO 3.2.1.

18. Using the auxiliary crane, remove the adapter plate from the top of the concrete cask.

19. Remove the swivel hoist rings from the structural lid. At the option of the user, install threaded plugs.

20. Install three swivel hoist rings in the shield plug and torque in accordance with Table 8.1.1-2.

21. Using the auxiliary crane, retrieve the shield plug and install the shield plug in the top of the concrete cask. Remove swivel hoist rings.

22. At the option of the user, seal tape may be installed around the diameter of the lid bolting pattern on the concrete cask flange.

23. Using the auxiliary crane, retrieve the concrete cask lid and install the lid in the top of the concrete cask. Secure the lid using six stainless steel bolts. Torque bolts in accordance with Table 8.1.1-2.

24. Ensure that there is no foreign material left at the top of the concrete cask. At the option of the user, a tamper-indicating seal may be installed.

25. If used, install a supplemental shielding fixture in each of the four inlets. Note: The supplemental shielding fixtures may also be shop installed.

8.1.3 Transport and Placement of the Vertical Concrete Cask

This procedure assumes that the loaded vertical concrete cask is positioned on a heavy-haul transporter and is to be positioned on the ISFSI pad using the air pad set. Alternately, the concrete cask may be lifted and moved using a mobile lifting frame. The mobile lifting frame lifts the cask using four lifting lugs at the top of the concrete cask. The lifting frame may be self-propelled or towed, and does not use the air pad set. Caution shall be observed when lifting the concrete cask using the two pairs of lifting lugs to minimize possible uneven loading on the base of the concrete cask. For lifting devices provided with load measuring equipment, the load on each lug set should be evenly maintained, but in no case shall an uneven load exceed 25,000 pounds between lug sets.

The vertical concrete cask lift height limit is 24 inches when the cask is moved using the air pad set or the mobile lifting frame in accordance with the requirements of Section A5.6(c) and Table A5-1 of Appendix A of the CoC Number 1015 Technical Specifications. Because of lift fixture configuration, the maximum lift height of the concrete cask using the jacking arrangement is approximately 4 inches.

The concrete cask surface dose rates must be verified in accordance with the requirements of LCO 3.2.2. These measurements may be made prior to movement of the cask, at a location along the transport path, or at the ISFSI. An optional supplemental shielding fixture, shown in Drawing 790-613, may be installed in the concrete cask air inlets to reduce the radiation dose rate at the inlets.

1. Using a suitable towing vehicle, tow the heavy-haul transporter to the dry storage pad (ISFSI). Verify that the bed of the transporter is approximately at the same height as the pad surface. Install four (4) hydraulic jacks at the four (4) designated jacking points at the air inlets in the bottom of the vertical concrete cask.
2. Raise the concrete cask approximately 4 inches using the hydraulic jacks.
   Caution: Do not exceed a maximum lift height of 24 inches, in accordance with the requirements of Administrative Control A5.6(a).
3. Move the air-bearing rig set under the cask.
4. Inflate the air-bearing rig set. Remove the four (4) hydraulic jacks.
5. Using a suitable towing vehicle, move the concrete cask from the bed of the transporter to the designated location on the storage pad.
   Note: Spacing between concrete casks must not be less than 15 feet (center-to-center).
6. Turn off the air-bearing rig set, allowing it to deflate.

7. Reinstall the four (4) hydraulic jacks and raise the concrete cask approximately 4 inches.
   Caution: Do not exceed a maximum lift height of 24 inches, in accordance with the requirements of Administrative Control A5.6(a).
8. Remove the air-bearing rig set pads. Ensure that the surface of the dry storage pad under the concrete cask is free of foreign objects.
9. Lower the concrete cask to the surface and remove the four (4) hydraulic jacks.
10. Install the screens in the inlets and outlets.
11. Scribe/stamp concrete cask nameplate to indicate loading information.
12. Verify concrete cask operability in accordance with SR 3.1.6.2 of LCO A 3.1.6.
13. Verify continued concrete cask thermal operability in accordance with SR 3.1.6.1 of LCO A 3.1.6.

8.2 Removal of the Loaded Transportable Storage Canister from the Vertical Concrete Cask

Removal of the loaded canister from the vertical concrete cask is expected to occur at the time of shipment of the canistered fuel off site. Alternately, removal could be required in the unlikely event of an accident condition that rendered the concrete cask or canister unsuitable for continued long-term storage or for transport. This procedure assumes that the concrete cask is being returned to the reactor cask receiving area. However, the cask may be moved to another facility or area using the same operations. It identifies the general steps to return the loaded canister to the transfer cask and return the transfer cask to the decontamination station, or other designated work. area or facility. Since these steps are the reverse of those undertaken to place the canister in the concrete cask, as described in Section 8.1.2, they are only summarized here.

The concrete cask may be moved using the air pad set or a mobile lifting frame. This procedure assumes the use of the air pad set. If a lifting frame is used, the concrete cask is lifted using four lifting lugs in the top of the cask, and the air pad set and heavy haul transporter are not required. The mobile lifting frame may be self-powered or towed. Caution shall be observed when lifting the concrete cask using the two pairs of lifting lugs to minimize possible uneven loading on the base of the concrete cask. For lifting devices provided with load measuring equipment, the load on each lug set should be evenly maintained, but in no case shall an uneven load exceed 25,000 pounds between lug sets.

At the option of the user, the canister may be removed from the concrete cask and transferred to another concrete cask or to the Universal Transport Cask at the ISFSI site. This transfer is done using the transfer cask, which provides shielding for the canister contents during the transfer.

Certain steps of the procedures in this section may be completed out of sequence to allow for operational efficiency. Changing the order of these steps, within the intent of the procedures, has no effect on the safety of the canister loading process and does not violate any requirements stated in the Technical Specifications or the NAC-UMS® FSAR. This includes the placement and installation of the air pads.

1. Remove the screens and temperature-monitoring instrumentation, if installed.
2. Using the hydraulic jacking system and the air pad set, move the concrete cask from the ISFSI pad to the heavy-haul transporter. The bed of the transporter must be approximately level with the surface of the pad and sheet metal plates are placed across the gap between the pad and the transporter bed.
   Caution: Do not exceed a maximum lift height of 24 inches when raising the concrete cask.

3. Tow the transporter to the cask receiving area or other designated work area or facility.
4. Remove the concrete cask lid and shield plug. Install the hoist rings in the canister structural lid and torque to the value specified in Table 8.1.1-2. Verify that the hoist rings are fully seated against the structural lid and attach the lift slings. Install the transfer adapter on the top of the concrete cask.
5. Retrieve the transfer cask with the retaining ring installed, and position it on the transfer adapter. Attach the shield door hydraulic cylinders.

   Note: The surrounding air temperature for cask unloading operations shall be ≥ 0°F.
6. Open the shield doors. Attach the canister lift slings to the cask handling crane hook.

   Caution: The attachment point of the two three-legged slings must be at least 75 inches above the top of the canister.
7. Raise the canister into the transfer cask.

   Caution: Avoid raising the canister to the point that the canister top engages the transfer cask retaining ring, as this could result in lifting the transfer cask.
8. Close the shield doors. Lower the canister to rest on the shield doors. Disconnect the canister slings from the crane hook. Install and secure door lock bolts/lock pins.

   Note: Monitor the time from this step (closing of shield doors) until initiation of canister cooldown operations, or completion of transfer to a concrete cask or Universal Transport Cask in accordance with LCO 3.1.4.
9. Retrieve the transfer cask lifting yoke. Engage the transfer cask trunnions and move the transfer cask to the decontamination area or designated work station.

After the transfer cask containing the canister is in the decontamination area or other suitable work station, additional operations may be performed on the canister. It may be opened, transferred to another storage cask, or placed in the Universal Transport Cask.

8.3 Unloading the Transportable Storage Canister

This section describes the basic operations required to open the sealed canister if circumstances arise that dictate the opening of a previously loaded canister and the removal of the stored spent fuel. It is assumed that the canister is positioned in the transfer cask and that the transfer cask is in the decontamination station or other suitable work station in the facility. The principal mechanical operations are the cutting of the closure welds, filling the canister with water, cooling the fuel contents, and removing the spent fuel. Supplemental shielding is used as required. The canister cooling water temperature, flow rate and pressure must be limited in accordance with this procedure.

Certain steps of the procedures in this section may be completed out of sequence to allow for operational efficiency. Changing the order of these steps, within the intent of the procedures, has no effect on the safety of the canister loading process and does not violate any requirements stated in the Technical Specifications of the NAC-UMS® Storage FSAR. This includes the sequence and use of an annulus fill system including optional seals and/or foreign material exclusion devices.

1. Remove the transfer cask retaining ring.
2. Survey the top of the canister to establish the radiation level and contamination level at the structural lid.
3. Set up the weld cutting equipment to cut the structural lid weld (Abrasive grinding, hydrolaser, or similar cutting equipment).
4. Enclose the top of the transfer cask in a radioactive material retention tent, as required.
   Caution: Monitor for any out-gassing. Wear respiratory protection as required.
5. Operate the cutting equipment to cut the structural lid weld.
6. After proper monitoring, remove the retention tent. Remove the cutting equipment and attach a three-legged sling to the structural lid.
7. Using the auxiliary crane, lift the structural lid from the canister and out of the transfer cask.
8. Survey the top of the shield lid to determine radiation and contamination levels. Use supplemental shielding as necessary. Decontaminate the top of the shield lid, if necessary.
9. Reinstall the retention tent. Using an abrasive grinder or hydrolaser, or other appropriate cutting equipment excluding open flame, and wearing suitable respiratory protection if required, cut the welds joining the vent and drain port covers to the shield lid.
   Caution: The canister could be pressurized.

10. Remove the port covers. Monitor for any out-gassing and survey the radiation level at the quick-disconnect fittings.
11. Attach a nitrogen gas line to the drain port quick-disconnect and a discharge line from the vent port quick-disconnect to an off-gas handling system in accordance with the schematic shown in Figure 8.3-1. Set up the vent line with appropriate instruments so that the pressure in the discharge line and the temperature of the discharge gas are indicated. Continuously monitor the radiation level of the discharge line.

    Caution: The discharge gas temperature could initially be above 400°F. The discharge line and fittings may be very hot.

    Note: Any significant radiation level in the discharge gas indicates the presence of fission gas products. The temperature of the gas indicates the thermal conditions in the canister.
12. Start the flow of nitrogen through the line until there is no evidence of fission gas activity in the discharge line. Continue to monitor the gas discharge temperature. When there is no additional evidence of fission gas, stop the nitrogen flow and disconnect the drain and vent port line connections. The nitrogen gas flush must be maintained for at least 10 minutes.

    Note: See Figure 8.3-1 for Canister Reflood Piping and Control Schematic.
13. Ensure the vent port quick-disconnect has new Viton seals by replacing the seals in the existing quick-disconnect or installing a new quick-disconnect. Ensure the drain port quick-disconnect has new Viton seals by replacing the seals in the existing quick-disconnect, installing a new quick-disconnect or installing a new drain tube assembly. Ensure the quick-disconnect assemblies are torqued to the value specified in Table 8.1.1-2.
14. Perform canister refill and fuel cooldown operations. Attach a source of clean water with a minimum temperature of 70ºF and a maximum supply pressure of 25 (+10, -0) psig to the drain port quick-disconnect. Attach a steam rated discharge line to the vent port quick-disconnect and route it to the spent fuel pool, an in-pool cooler, or an in-pool steam condensing unit. Slowly start the flow of clean or filtered pool water to establish a flow rate at 5 (+3, -0) gpm. Monitor the discharge line pressure gauge during canister flooding. Stop filling the canister if the canister vent line pressure exceeds 45 psig. Re-establish water flow when the canister pressure is below 35 psig. The discharge line will initially discharge hot gas, but after the canister fills, it will discharge hot water.

    Caution: Relatively cool water may flash to steam as it encounters hot surfaces within the canister.

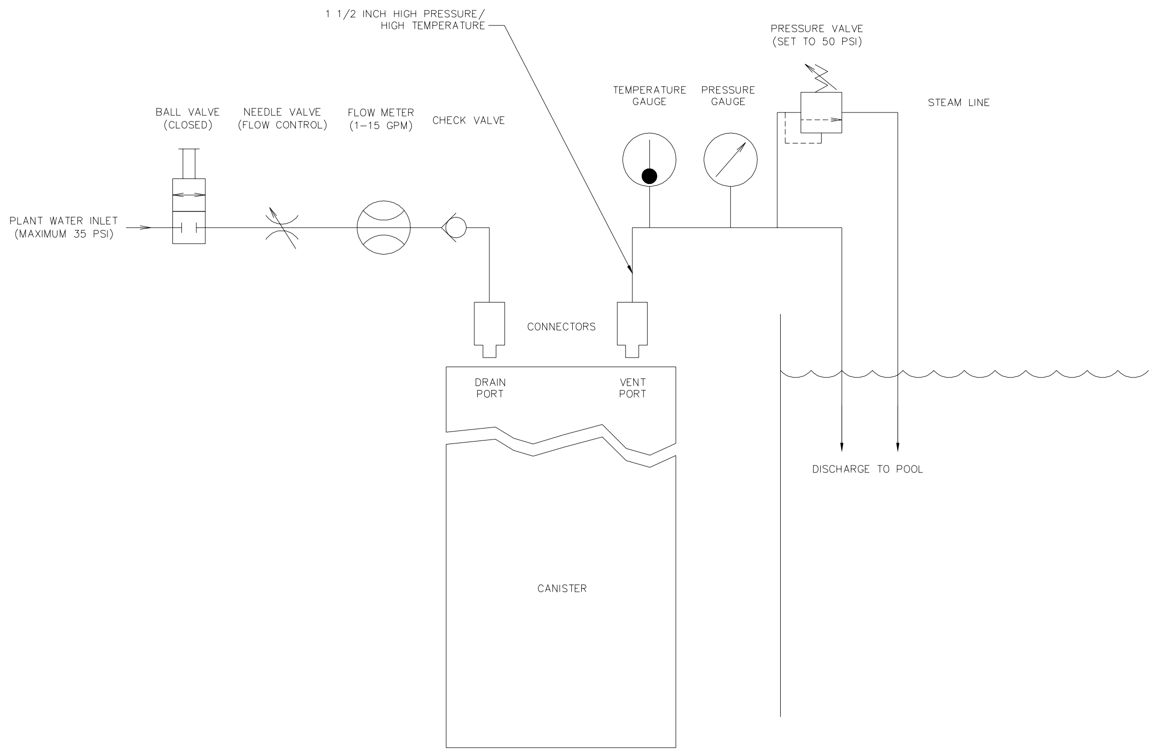
    Caution: If there are grossly failed or ruptured fuel rods within the canister, very high levels of radiation could rapidly appear at the discharge line. The radiation level of the discharge gas or water should be continuously monitored.

    Caution: Reflooding requires the use of borated water in accordance with LCO 3.3.1 if borated water was required for the initial fuel loading.

15. Monitor water flow through the canister until the water discharge temperature is below 200°F. Stop the flow of water and remove the connection to the drain line.
    Note: Monitor canister water temperature and reinitiate cooldown operations if temperature exceeds 200°F.
16. Connect a suction pump to the drain port and a vent line to the vent port. Operate the pump and remove approximately 70 gallons of water. Disconnect and remove the pump.
17. Set up the weld cutting equipment to cut the shield lid weld (Abrasive grinding, hydrolaser, or similar cutting equipment.). Route the vent line to avoid interference with the weld cutting operation.
18. Tent the top of the transfer cask and wear respiratory protection equipment as required. Attach a hydrogen gas detector to the vent port line. Verify that the concentration of hydrogen gas is less than 2.4%.
19. Operate the cutting equipment to cut the shield lid weld.
    Note: Stop the cutting operation if the hydrogen gas detector indicates a concentration of hydrogen gas above 2.4%. Connect the vacuum drying system and evacuate gas before proceeding with the cutting operation.
20. Remove the cutting equipment. Remove all loose shims. Remove supplemental shielding if used. Install the shield lid lifting hoist rings, verifying that the hoist rings are fully seated against the shield lid, and attach a three-legged sling. Attach a tag line to the sling set to aid in attaching the sling to the crane hook (at Step 25).
21. Install the annulus fill system to the transfer cask, including the clean water lines.
22. Retrieve the transfer cask lifting yoke and engage the transfer cask lifting trunnions.
23. Move the transfer cask over the pool and lower the bottom of the transfer cask to the surface. Start the flow of clean water to the transfer cask annulus. Continue to lower the transfer cask, as the annulus fills with water, until the top of the transfer cask is about 4 inches above the pool surface. Hold this position until clean water fills to the top of the transfer cask.
24. Lower the transfer cask to the bottom of the cask loading area and remove the lifting yoke.
25. Attach the shield lid lifting sling to the crane hook.
    Caution: The drain line tube is suspended from the under side of the shield lid. The lid should be raised as straight as possible until the drain tube clears the canister basket. The under side of the shield lid could be highly contaminated.
26. Slowly lift the shield lid. Move the shield lid to one side after it is raised clear of the transfer cask.
27. Visually inspect the fuel for damage.

At this point, the spent fuel could be transferred from the canister to the fuel racks. If the fuel is damaged, special handling equipment may be required to remove the fuel. In addition, the bottom of the canister could be highly contaminated. Care must be exercised in the handling of the transfer cask when it is removed from the pool.

Figure 8.3-1 Canister Reflood Piping and Controls Schematic



8.4 References

1. "Safety Analysis Report for the UMS® Universal Transport Cask," Docket Number 71-9270, NAC International, April 1997.

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9.0 ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

This chapter specifies the acceptance criteria and the maintenance program for the Universal Storage System primary components - the Vertical Concrete Cask and Transportable Storage Canister. The system components, such as the concrete cask liner, base and air outlets, and the canister shell with the bottom plate, the shield and structural lids, and the basket that holds the spent fuel, are shop fabricated. The concrete cask consists of reinforced concrete placed around the steel liner and base that are integral to its performance. The liner forms the central cavity of the vertical concrete cask, which is mounted on the base. The liner/base interface forms air inlet passageways to the central cavity. The inlets allow cool ambient air to be drawn in and passed by the canister that contains the fuel. Air outlets at the top of the concrete cask allow the air heated by the canister wall and concrete cask liner to be discharged. The base of the concrete cask acts as a pedestal to support the canister during storage.

The concrete reinforcing steel (rebar) is bent in the shop and delivered to the concrete cask construction site. Concrete cask construction begins with the erection of the cask liner onto the steel base. Reinforcing steel is placed around the liner, followed by a temporary outer form which encircles the cask liner and reinforcing steel. The temporary form creates an annulus region between the liner and the form into which the concrete is placed.

As described in Section 8.1.3, the vertical concrete cask may be lifted by: (1) hydraulic jacks and moved by using air pads underneath the base; or (2) lifting lugs and moved by a mobile lifting frame.

9.1 Acceptance Criteria

The acceptance criteria specified below ensure that the concrete cask, including the liner, base, and canister are fabricated, assembled, inspected and tested in accordance with the requirements of this SAR and the license drawings presented in Section 1.8.

9.1.1 Visual and Nondestructive Examination

The acceptance test program establishes the visual inspections and nondestructive examinations to be performed to verify the acceptability of the shop fabricated and field constructed UMS® components.

All components shall be visually examined for conformance to the license drawings. Fit-up tests of canister components will be performed during canister acceptance to demonstrate that the canister, basket, port covers and lids can be properly assembled and the fuel tubes will accommodate the applicable design bases fuel assembly.

Materials of construction and subcomponents shall be receipt inspected for visual, dimensional and material certification acceptability to specification requirements.

Welding of the canister and basket assembly shall be performed in accordance with the requirements of ASME Code, Section IX [5]. Visual examinations of the canister and basket assembly welds shall be performed in accordance with the ASME Code, Section V, Article 9 [2]. The acceptance criteria for canister visual inspections are ASME Code, Section III, Subsection NB [1], Articles NB-4424 and NB-4427, and Section VIII, Division 1 [3], Articles UW-35 and UW-36. The acceptance criterion for basket assembly visual inspections is ASME Code, Section III, Subsection NG [6], Article NG-5360. Unacceptable canister welds shall be repaired per NB-4450 or NG-4450, as applicable, and reinspected in accordance with the original acceptance criteria.

Welding of the steel components of the concrete cask shall be performed in accordance with ANSI/AWS D1.1-96 [4] or ASME Code, Section VIII, Division 1, Part UW. Visual inspection of concrete cask steel components shall use the acceptance criteria of ANSI/AWS D1.1, Section 8.15.1, or ASME Code, Section VIII, Division 1, UW-35 and UW-36.

A final inspection of the critical dimensions of fabricated components shall be performed to confirm as-built dimensions. All components shall be inspected for appropriate cleanliness, including surfaces free from foreign material, oil, grease and solvents. Fabricated components shall be appropriately packaged for shipment.

9.1.1.1 Nondestructive Weld Examination

The canister shall be fabricated in accordance with the ASME Code, Section III, Subsection NB requirements, except for approved Code exceptions as listed in Table B3-1 of Appendix B of the Certificate of Compliance (CoC). The final surface of canister welds shall be examined by dye penetrant examination in accordance with ASME Code, Section V, Article 6, with acceptance per Section III, Subsection NB-5350. Canister longitudinal (and circumferential, if required) shell welds shall be examined by radiographic examination in accordance with the requirements of the

ASME Code, Section V, Article 2, with acceptance criteria per Section III, Subsection NB-5320. The canister shell to base plate weld shall be examined by ultrasonic examination in accordance with ASME Code, Section V, Article 5, with acceptance per Section III, Subsection NB-5330. The field installed shield lid and structural lid welds shall be inspected by ultrasonic or dye penetrant examination methods. The shield lid to shell and port cover to shield lid welds shall be dye penetrant examined at the root and final pass in accordance with ASME Code, Section V, Article 6, with acceptance per Section III, Subsection NB-5350. Should the root and final pass be one and the same (i.e., single pass weld), then only one dye penetrant examination is required. The structural lid to shell weld shall be examined by either ultrasonic or dye penetrant examination in accordance with ASME Code, Section V, Articles 5 or 6, respectively.

Ultrasonic examinations acceptance criteria shall be in accordance with ASME Code, Section III, Subsection NB-5330. The acceptance criteria for the dye penetrant examination of the structural lid root, every 3/8-inch layer and final surface shall be in accordance with ASME Code, Section III, Subsection NB-5350. The results of the structural lid dye penetrant examination final interpretation, as described in ASME Code, Section V, Article 6, T-676, including all relevant indications, shall be recorded by video, photographic or other means to provide retrievable records of weld integrity.

The basket shall be fabricated in accordance with the ASME Code, Section III, Subsection NG requirements, except for approved Code exceptions as listed in Table B3-1 of Appendix B of the CoC. The final surface of identified basket welds shall be examined by the dye penetrant examination in accordance with ASME Code, Section V, Article 6, with acceptance per Section III, Subsection NG-5350.

Personnel performing nondestructive examinations shall be qualified in accordance with SNT-TC-1A [11]. A written report shall be prepared for each weld examined and shall include, at a minimum, the identification of the part, material, name and level of examiner, NDE procedure used, and the findings or dispositions, if any.

9.1.1.2 Construction Inspections

Concrete mixing slump, air entrainment, strength and density are field verified using either the American Concrete Institute (ACI) or the American Society for Testing and Materials (ASTM) standard testing methods and acceptance criteria, as appropriate, to ensure adequacy. Reinforcing steel is installed per specification requirements based on ACI-318 [7].

9.1.2 Structural and Pressure Test

The transportable storage canister is pressure tested at the time of use. After loading of the canister basket with spent fuel, the shield lid is welded in place after approximately 70 gallons of water are removed from the canister. Removal of the water ensures that the water level in the canister is below the bottom of the shield lid during welding of the shield lid to the canister shell. Prior to removing the remaining spent fuel pool water from the canister, the canister is pressure tested at 35 psia. This pressure is held for a minimum 10 minutes. Any loss of pressure during the test period is unacceptable. The leak must be located and repaired. The pressure test procedure is described in Section 8.1.1.

If the canister is to be ASME Code N-stamped, the canister shall be hydrostatically tested in accordance with the requirements of ASME Code Subsection NB-6220 and Code Case N-595-4 [12].

9.1.2.1 Transfer Casks

The transfer cask is provided in the Standard or Advanced configuration. The Standard transfer cask is restricted to handling the Standard weight canister. The Advanced transfer cask incorporates a reinforced trunnion design that allows it to handle either the standard weight, or a heavier weight, canister.

For any configuration, the transfer cask lifting trunnions and the bottom shield doors shall be tested in accordance with the requirements of ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4,500 kg) or More for Nuclear Materials" [8].

Standard Transfer Cask

The Standard transfer cask lifting trunnion load test shall consist of applying a vertical load of 630,000 pounds, which is greater than 300% of the maximum service load for the transfer cask and loaded canister with the shield lid and full of water (208,400 lbs). The bottom shield door and rail load test shall consist of applying a vertical load of 265,200 pounds, which is over 300% of the maximum service load (88,400 lbs). These maximum service loads are selected based on the heaviest configuration and, thus, bound all of the other configurations.

Advanced Transfer Cask

The Advanced transfer cask lifting trunnion load test shall consist of applying a vertical load of 690,000 pounds, which is greater than 300% of the maximum service load (225,000 pounds) for the transfer cask and loaded canister with the shield lid and full of water. The bottom shield door and rail load test shall consist of applying a vertical load of 300,000 pounds, which is over 300% of the maximum service load (98,000 lbs). These maximum service loads are based on the heaviest configuration and, thus, bound all the other configurations.

The load tests shall be held for a minimum of 10 minutes and shall be performed in accordance with approved, written procedures.

Following completion of the lifting trunnion load tests, all trunnion welds and all load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking. Liquid penetrant examination (the magnetic particle method may be used on ferrous material) shall be performed on accessible trunnion and shield door rail load-bearing welds in accordance with ASME Code Section V, Articles 1, 6 and/or 7, with acceptance in accordance with ASME Code Section III, NF-5340 or NF-5350, as applicable. Similarly, following completion of the bottom shield door and rail load tests, all door rail welds and all load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking.

Any evidence of permanent deformation, cracking or galling of the load bearing surfaces or unacceptable liquid penetrant examination results, shall be cause for evaluation, rejection, or rework of the affected component. Liquid penetrant or magnetic particle examinations of all load bearing welds shall be performed in accordance with ASME Code Section V, Articles 1, 6 and/or 7, with acceptance in accordance with ASME Code Section III, NF-5350 or NF-5340, as applicable.

9.1.2.2 Concrete Cask

The concrete cask, at the option of the user/licensee, may be provided with lifting lugs to allow for the vertical handling and movement of the concrete cask. The lifting lugs are provided as two sets of two lugs each. The concrete cask lifting lugs shall be load tested by applying a vertical load, which is greater than 150 percent of the maximum concrete cask weight plus a 10 percent dynamic load factor, where the concrete cask weight is determined, based on the class, from Table 3.2-1 or 3.2-2.

The test load shall be applied for a minimum of 10 minutes in accordance with approved, written procedures. Following completion of the load test, all load bearing surfaces of the lifting lugs shall be visually inspected for permanent deformation, galling, or cracking. Liquid penetrant or magnetic particle examinations of load bearing surfaces shall be performed in accordance with ASME Code, Section V, Articles 1, 6 and/or 7, with acceptance criteria in accordance with ASME Code, Section III, Subsection NF, NF-5350 or NF-5340, as applicable.

Any evidence of permanent deformation, cracking, or galling, or unacceptable liquid penetrant or magnetic particle examination results for the load bearing surfaces of the lifting anchors shall be cause for evaluation, rejection, or rework and retesting.

9.1.2.3 Transportable Storage Canister

The transportable storage canister shell may be hydrostatically or pneumatically pressure tested during fabrication in accordance with Section NB-6200 or NB-6300 of the ASME Code, respectively. Hydrostatic testing will be performed in accordance with NB-6221 using 1.25 times the design pressure of 15 psig. The test pressure shall be held a minimum of 10 minutes in accordance with NB-6223. Examination after the pressure test shall be in accordance with NB-6224. Alternately, a pneumatic pressure test may be performed in accordance with NB-6321 using 1.2 times the design pressure of 15 psig. The test pressure shall be held a minimum of 10 minutes in accordance with NB-6323. Examination after the pressure test shall be in accordance with NB-6224.

The canister shell shall consist of the completed Shell Weldment as shown on Drawing 790-582.

If the pressure test is not performed during fabrication, a pressure test must be performed upon closure of the canister with the shield lid as described in Section 8.1.1 of the operating procedures.

9.1.3 Leak Tests

The canister is leak tested at the time of use. After the pressure test described in Section 9.1.2, the canister is drained of residual water, vacuum dried and backfilled with helium. The canister is pressurized with helium to 0 psig. The shield lid to canister shell weld is helium leak tested using a test fixture installed above the shield lid. The leaktight criteria of 2.0×10^{-7} cm^3/sec (helium) of ANSI N14.5[9] is applied. The leak test is performed at a sensitivity of 1.0×10^{-7} cm^3/sec (helium). Any indication of a leak of 2.0×10^{-7} cm^3/sec (helium) or greater is unacceptable and repair is required as appropriate.

9.1.4 Component Tests

The components of the Universal Storage System do not require any special tests in addition to the material receipt, dimensional, and form and fit tests described in this chapter.

9.1.4.1 Valves, Rupture Disks and Fluid Transport Devices

The transportable storage canister and the vertical concrete cask do not contain rupture disks or fluid transport devices. There are no valves that are part of the confinement boundary for transport or storage. Quick-disconnect valves are installed in the vent and drain ports of the shield lid. These valves are convenience items for the operator, as they provide a means of quickly connecting ancillary drain and vent lines to the canister. During storage and transport, these fittings are not accessible, as they are covered by port covers that are welded in place when the canister is closed. As presented for storage and transport, the canister has no accessible valves or fittings.

9.1.4.2 Gaskets

The transportable storage canister and the vertical concrete cask have no mechanical seals or gaskets that form an integral part of the system, and there are no mechanical seals or gaskets in the confinement boundary.

9.1.5 Shielding Tests

Based on the conservative design of the Universal Storage System for shielding criteria and the detailed construction requirements, no shielding tests of the vertical concrete cask are required.

9.1.6 Neutron Absorber Tests

A neutron absorbing material is used for criticality control in the PWR, BWR and oversize BWR fuel tubes. The placement and dimensions of the neutron absorber are as shown on the License

Drawings for these components. The neutron absorbing material is an aluminum matrix material formed from aluminum and boron-carbide. The mixing of the aluminum and boron-carbide powder forming the neutron absorber material is controlled to assure the required ^{10}B areal density, as specified on the component License Drawings. The constituents of the neutron absorber material shall be verified by chemical testing and/or spectroscopy and by physical property measurement to ensure the quality of the finished plate or sheet. The results of all neutron absorber material tests and inspections, including the results of wet chemistry coupon testing, are documented and become part of the quality records documentation package for the fuel tube and basket assembly.

Aluminum/boron carbide neutron absorbing material is available under the trade name BORAL®. BORAL is procured and qualified under a Quality Assurance/Quality Control program in conformance with the requirements of 10 CFR 72, Subpart G.

The manufacturing process of BORAL consists of several steps. The initial step is the mixing of the aluminum and boron carbide powders that form the core of the finished material. The amount of each powder is a function of the desired ^{10}B areal density. The methods used to control the weight and blend the powders are patented and proprietary processes of the manufacturer.

After manufacturing, test samples from each batch of neutron absorber sheets shall be tested using wet chemistry techniques to verify the presence and minimum weight percent of ^{10}B. The tests shall be performed in accordance with approved written procedures.

9.1.6.1 Neutron Absorber Material Sampling Plan

The neutron absorber sampling plan is selected to demonstrate a 95/95 statistical confidence level in the neutron absorber sheet material in compliance with the specification. In addition to the specified sampling plan, each sheet of material is visually and dimensionally inspected using at least 6 measurements on each sheet. No rejected neutron absorber sheet is used. The sampling plan is supported by written and approved procedures.

The sampling plan requires that a coupon sample be taken from each of the first 100 sheets of absorber material. Thereafter, coupon samples are taken from 20 randomly selected sheets from each set of 100 sheets. This 1 in 5 sampling plan continues until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder, aluminum powder, or aluminum extrusion) or a process change. The sheet samples are indelibly marked and recorded for identification. This identification is used to document neutron absorber test results, which become part of the quality record documentation package.

9.1.6.2 Neutron Absorber Wet Chemistry Testing

Wet chemistry testing of the test coupons obtained from the sampling plan is used to verify the ^{10}B content of the neutron absorber material. Wet chemistry testing is applied because it is considered to be the most accurate and practical direct measurement method for determining ^{10}B, boron and B_4C content of metal materials and is considered by the Electric Power Research Institute (EPRI) to be the method of choice for this determination.

An approved facility with chemical analysis capability, which could include the neutron absorber vendor's facility, shall be selected to perform the wet chemistry tests. Personnel performing the testing shall be trained and qualified in the process and in the test procedure.

Wet chemistry testing is performed by dissolving the aluminum in the matrix, including the powder and cladding, in a strong acid, leaving the B_4C material. A comparison of the amount of B_4C material remaining to the amount required to meet the ^{10}B content specification is made using a mass-balance calculation based on sample size.

A statistical conclusion about the neutron absorber sheet from which the sample was taken and that batch of neutron absorber sheets may then be drawn based on the test results and the controlled manufacturing processes.

The adequacy of the wet chemistry method is based on its use to qualify the standards employed in neutron blackness testing. The neutron absorption performance of a test material is validated based on its performance compared to a standard. The material properties of the standard are demonstrated by wet chemistry testing. Consequently, the specified test regimen provides adequate assurance that the neutron absorber sheet thus qualified is acceptable.

9.1.6.3 Acceptance Criteria

The wet chemistry test results shall be considered acceptable if the ^{10}B areal density is determined to be equal to, or greater than, that specified on the fuel tube License Drawings. Failure of any coupon wet chemistry test shall result in 100% sampling, as described in the sampling plan, until compliance with the acceptance criteria is demonstrated.

9.1.7 Thermal Tests

No thermal acceptance testing of the Universal Storage System is required during construction. Thermal performance of the system was confirmed in accordance with the procedure specified in Section 9.2.3 and documented in a report. In addition, initial temperature measurements are taken of the concrete cask(s) placed in service, in accordance with LCO A 3.1.6 to verify the operability of the cask.

9.1.8 Cask Identification

A stainless steel nameplate is permanently attached at eye level on the outer surface of the concrete cask as shown on Drawing No. 790-562.

Drawing No. 790-565 shows the information included on the nameplate.

9.2 Maintenance Program

This section presents the maintenance requirements for the UMS® Universal Storage System and for the transfer cask.

9.2.1 UMS® Storage System Maintenance

The UMS® Universal Storage System is a passive system. No active components or systems are incorporated in the design. Consequently, only a minimal amount of maintenance is required over its lifetime.

The UMS® Universal Storage System has no valves, gaskets, rupture discs, seals, or accessible penetrations. Consequently, there is no maintenance associated with these types of features.

The routine thermal performance surveillance requirements for a loaded UMS® System are described in the Technical Specifications of Appendix A, Limiting Condition for Operation (LCO) 3.1.6.

Per the LCO, an initial verification of the concrete cask's thermal performance is completed by taking temperature measurements, per Surveillance Requirement (SR) 3.1.6.2, between 5 and 30 days following the start of storage operations.

Following the initial temperature measurements, the continuing operability of the concrete cask is verified on a 24-hour frequency by completion of SR 3.1.6.1, which allows verification by visual inspection of the inlet and outlet vents for blockage, or verification by measurement of the air temperature difference between ambient and outlet average. If the operable status of the concrete cask is reduced, the concrete cask will be returned to an operable status or placed in a safe condition as specified in the LCO.

In the event of any off-normal, accident or natural phenomena event, which could lead to the blockage of the concrete cask's inlets and outlets, full vent blockage shall be removed within 24 hours, and any partial blockage shall be corrected to restore the cask to operable status in accordance with LCO 3.1.6.

Annually or on a frequency established by the User based on the environmental conditions at the ISFSI (i.e., higher inspection frequency may be appropriate at ISFSIs exposed to marine environments, lower frequency for sites located in dry environments, etc.), a program of visual inspections and maintenance of the loaded UMS® systems in service shall be implemented. The Vertical Concrete Cask(s) shall be inspected as described herein.

- Visually inspect exterior concrete surfaces for chipping, spalling or other defects. Minor surface defects (i.e., approximately one cubic inch) shall be repaired by cleaning and grouting of the area in accordance with the grout manufacturer's recommendations.
- Visually inspect accessible exterior coated carbon steel surfaces including lifting lug assemblies, if installed, for loss of coating, corrosion or other damage. The maintenance and repair of corroded surfaces, or surfaces missing coating materials, shall be done by cleaning the areas and reapplying corrosion-inhibiting coatings in accordance with the coating manufacturer's recommendations. The licensee shall identify, evaluate and select acceptable coatings for use in routine maintenance of concrete cask external carbon steel surfaces.
- Visually inspect lid bolts for presence of corrosion. Excessively corroded or missing bolting shall be replaced with approved spare parts.
- Visually inspect the attachment hardware and the integrity of the inlet and outlet screens. Damaged or missing components shall be repaired or replaced with approved spare parts.
- Significant damage or defects identified during the visual inspections that exceed routine maintenance shall be processed as nonconforming items.

The schedule, results and corrective actions taken during the UMS® system inspection and maintenance program shall be documented and retained as part of the system maintenance program.

9.2.2 Transfer Cask Maintenance

The transfer cask trunnions and shield door assemblies shall be visually inspected for gross damage and proper function prior to each use.

Annually (or a period not exceeding 14 months), an inspection and testing program shall be performed on the transfer cask in accordance with the requirements of ANSI N14.6 [8]. The following actions or alternatives shall be performed:

- Visually inspect the lifting trunnions, shield doors and shield door rails for permanent deformation and cracking. Carbon steel-coated surfaces will be inspected for chipped, cracked or missing areas of coating, and repaired by reapplication of the approved coating(s) in accordance with the coating manufacturer's recommendations.
- In addition, one of the following testing/inspection methods shall be completed.

- Perform a load test equal to or greater than 300% (or 150% for facilities not implementing single-failure-proof lifting) of the maximum service load and a post-test visual inspection of major load-bearing welds and critical components for defects, weld cracking, material displacement or permanent deformation; or
- If surface cleanliness and conditions permit, perform a dimensional and visual inspection of load-bearing components, and a nondestructive examination of major load-bearing welds and critical areas.

The annual examination and testing program may be deferred during periods of nonuse of the transfer cask, provided that the transfer cask examination or testing program is performed prior to the next use of the transfer cask. The inspection results and corrective actions taken as part of the maintenance program shall be documented and retained as part of the system maintenance program.

9.2.3 Required Surveillance of First Storage System Placed in Service

For the first Universal Storage System placed in service with a heat load equal to or greater than 10 kW, the canister is loaded with spent fuel assemblies and the decay heat load calculated for that canister. The canister is then loaded into the vertical concrete cask, and the cask's thermal performance is evaluated by measuring the ambient and air outlet temperatures for normal air flow. The purpose of the surveillance is to measure the heat removal performance of the Universal Storage System and to establish baseline data. In accordance with 10 CFR 72.4, a letter report summarizing the results of the surveillance and evaluation will be submitted to the NRC within 30 days of placing the loaded cask on the ISFSI pad. The report will include a comparison of the calculated temperatures of the NAC-UMS® system heat load to the measured temperatures. A report is not required to be submitted for the NAC-UMS® systems that are subsequently loaded, provided that the performance of the first system placed in service with a heat load ≥ 10 kW, is demonstrated by the comparison of the calculated and measured temperatures.

NAC's "Report on the Thermal Performance of the NAC-UMS® System at the Palo Verde Nuclear Generating Station (PVNGS) Independent Spent Fuel Storage Installation" [10] dated May 30, 2003, was transmitted to the NRC by Arizona Public Service on June 4, 2003, in accordance with the requirements of NAC-UMS® Technical Specification A 5.3, "Special Requirements for the First System Placed in Service," and in compliance with 10 CFR 72.4. The report concludes that the measured temperature data demonstrates that the thermal models and analysis results reported in the NAC-UMS® FSAR correctly represent the heat transfer characteristics of the storage system.

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9.3 References

1. ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NB, "Class 1 Components," 1995 Edition with 1995 Addenda.

2. ASME Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination," 1995 Edition with 1995 Addenda.

3. ASME Boiler and Pressure Vessel Code, Section VIII, Subsection B, Part UW, "Requirements for Pressure Vessels Fabricated by Welding," 1995 Edition with 1995 Addenda.

4. American Welding Society, Inc., "Structural Welding Code - Steel," ANSI/AWS D1.1, 1996.

5. ASME Boiler and Pressure Vessel Code, Section IX, "Welding and Brazing Qualifications," 1995 Edition with 1995 Addenda.

6. ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NG, "Core Support Structures," 1995 Edition with 1995 Addenda.

7. American Concrete Institute, "Building Code Requirements for Structural Concrete," ACI-318-95, October 1995.

8. American National Standards Institute, "Radioactive Materials - Special Lifting Devices for Shipping Containers Weighting 10,000 Pounds (4,500 kg) or More," ANSI N14.6-1993, 1993.

9. American National Standards Institute, "Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

10. "Report on the Thermal Performance of the NAC-UMS® System at the Palo Verde Nuclear Generating Station (PVNGS) Independent Spent Fuel Storage Installation," NAC International, May 2003.

11. Recommended Practice No. SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing," The American Society for Nondestructive Testing, Inc., edition as invoked by the applicable ASME Code.

12. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Code Case N-595-4, "Requirements for Spent Fuel Storage Canisters," May 2004.

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10.0 RADIATION PROTECTION

10.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)

The Universal Storage System provides radiation protection for all areas and systems that may expose personnel to radiation or radioactive materials. The components of the PWR and BWR configurations of the system that require operation, maintenance and inspection are designed, fabricated, located, and shielded so as to minimize radiation exposure to personnel.

10.1.1 Policy Considerations

It is the policy of NAC International (NAC) to ensure that the Universal Storage System is designed so that operation, inspection, repair and maintenance can be carried out while maintaining occupational exposure as low as is reasonably achievable (ALARA).

10.1.2 Design Considerations

The design of the Universal Storage System complies with the requirement of 10 CFR 72.3 [1] concerning ALARA and meets the requirements of 10 CFR 72.126(a) and 10 CFR 20.1101 [2] with regard to maintaining occupational radiation exposures ALARA. Specific design features that demonstrate the ALARA philosophy are:

- Material selection and surface preparation that facilitate decontamination.
- A basket configuration that allows spent fuel canister loading using accepted standard practice and current experience.
- Positive clean water flow in the transfer cask/canister annulus to minimize the potential for contamination of the canister surface during in-pool loading.
- Passive confinement, thermal, criticality, and shielding systems that require no maintenance.
- Thick steel and concrete walls to reduce the side surface dose rate of the concrete cask to less than 50 mrem/hr (average).

- Nonplanar cooling air pathways to minimize radiation streaming at the inlets and outlets of the vertical concrete cask.

- Optional use of remote, automated outlet air temperature measurement to reduce surveillance time.

10.1.3 Operational Considerations

The ALARA philosophy is incorporated into the procedural steps necessary to operate the Universal Storage System in accordance with its design. The following features or actions, which comprise a baseline radiological controls approach, are incorporated in the design or procedures to minimize occupational radiation exposure:

- Use of automatic equipment for welding the shield lid and structural lid to the canister shell.

- Use of automatic equipment for weld inspections.

- Decontamination of the exterior surface of the transfer cask, welding of the shield lid, and pressure testing of the canister while the canister remains filled with water.

- Use of quick disconnect fittings at penetrations to facilitate required service connections.

- Use of remote handling equipment, where practical, to reduce radiation exposure.

- Use of prefabricated, shaped temporary shielding, if necessary, during automated welding equipment set up and removal, during manual welding, during weld inspection of the shield lid, and during all other canister closing and sealing operations conducted at the shield lid.

The operational procedures at a particular facility are determined by the user's operational conditions and facilities.

10.2 Radiation Protection Design Features

The radiation shielding design description is provided in Section 5.3.1. The design criteria radiation exposure rates are summarized in Table 2-1. The principal radiation protection design features are the shielding necessary to meet the design objectives, the placement of penetrations near the edge of the canister shield lid to reduce operator exposure and handling time, and the use of shaped supplemental shielding for work on and around the shield lid, as necessary. This supplemental shielding reduces operator dose rates during the welding, inspection, draining, drying and backfilling operations that seal the canister. An optional supplemental shielding fixture, shown in Drawing 790-613, may be installed in the air inlets to reduce the radiation dose rate at the base of the vertical concrete cask.

Radiation exposure rates at various work locations are determined for the principal Universal Storage System operational steps using a combination of the SAS4 [3] and SKYSHINE III [4] computer codes. The use of SAS4 is described in Section 5.1.2. The SKYSHINE-III code is discussed in Section 10.4. The calculated dose rates decrease with time.

10.2.1 Design Basis for Normal Storage Conditions

The radiation protection design basis for the Universal Storage System vertical concrete cask is derived from 10 CFR 72 and the applicable ALARA guidelines. The design basis surface dose rates, and the calculated surface and 1-foot dose rates are:

Vertical Concrete Cask	Design Basis Surface Dose Rate (mrem/hr)	Surface Dose Rate (mrem/hr)		1-Foot Maximum Dose Rate (mrem/hr)	
		PWR	BWR	PWR	BWR
Side wall	50.0 (avg.)	37.3	22.7	42.3	24.5
Air inlet(1)	100.0(2)	136	129	47.8	44.9
Air outlet	100.0(2)	63	55	15.7	12.8
Top lid	50.0 (avg.)	26.1	19.7	22.6	15.7

(1) Air inlet dose rates are based on the use of the air inlet shields. Design basis source terms require the use of the inlet shields to remain below the technical specification limits outlined in Appendix A.

(2) An air inlet and outlet average dose rate of 100 mrem/hr.

The calculated dose rates at these, and at other dose points, are reported in Sections 5.1.3 and 5.4.3. The dose rates presented are for the design basis 40,000 MWd/MTU, 5-year cooled fuel. These dose rates bound those of the higher burnup, but longer cooled, fuel described in Section 2.1.

Activities associated with closing the canister, including welding of the shield and structural lids, draining, drying, backfilling and testing, may employ temporary shielding to minimize personnel dose in the performance of those tasks.

10.2.2 Design Basis for Accident Conditions

Damage to the vertical concrete cask after a design basis accident does not result in a radiation exposure at the controlled area boundary in excess of 5 rem to the whole body or any organ. The high energy missile impact is estimated to reduce the concrete shielding thickness, locally at the point of impact, by approximately 6 inches. Localized cask surface dose rates for the removal of 6 inches of concrete are estimated to be less than 250 mrem/hr for the PWR and BWR configurations.

A hypothetical accident event, tip-over of the vertical concrete cask, is considered in Section 11.2.12. There is no design basis event that would result in the tip-over of the vertical concrete cask.

10.3 Estimated On-Site Collective Dose Assessment

Occupational radiation exposures (person-mrem) resulting from the use of the Universal Storage System are calculated using the estimated exposure rates presented in Sections 5.1.3, 5.4.3 and 10.2.1. Exposure is evaluated by identifying the tasks and estimating the duration and number of personnel performing those tasks based on industry experience. The tasks identified are based on the design basis operating procedures, as presented in Chapter 8.

Dose rates for the standard transfer cask and the concrete storage cask are calculated using the shielding analysis design basis fuel assemblies. The shielding design basis PWR assembly is the Westinghouse 17×17 Standard fuel assembly, with an initial enrichment of 3.7 wt % ^{235}U. The design basis BWR assembly is the GE 9×9, with 79 fuel rods and an initial enrichment of 3.25 wt % ^{235}U. Both design basis fuel assemblies have an assumed burnup of 40,000 MWD/MTU, and a cool time of 5 years. The selection of these assemblies for the shielding design basis is described in Section 5.1. The principal parameters of these assemblies are presented in Table 2.1.1-1.

10.3.1 Estimated Collective Dose for Loading a Single Universal Storage System

This section estimates the collective dose due to the loading, sealing, transfer and placement on the independent spent fuel storage installation (ISFSI) pad, of the Universal Storage System. The analysis assumes that the exposure incurred by the operators is independent of background radiation, as background radiation varies from site to site. The number of persons allocated to task completion is a typical number required for the task. Working area exposure rates are assigned based on the orientation of the worker with respect to the source and take into account the use of temporary shielding.

Table 10.3-1 summarizes the estimated total exposure by task, attributable to the loading, transfer, sealing and placement of a design basis Universal Storage System based on the use of the standard transfer cask. As documented in Section 5.1, exposures from the advanced transfer cask are not going to differ substantially from exposures documented for the standard transfer cask.

Exposures associated with shield lid operations are based on the presence of a temporary 5-inch thick steel shield.

This estimated dose is considered to be conservative as it assumes the loading of a cask with design basis fuel, and does not account for efficiencies in the loading process that occur with experience.

10.3.2 Estimated Annual Dose Due to Routine Operations

Once in place, the ISFSI requires limited ongoing inspection and surveillance throughout its service life. The annual dose evaluations presented in Tables 10.3-4 through 10.3-7 estimate the exposure due to a combination of inspection and surveillance activities and other tasks that are anticipated to be representative of an operational facility. The visual inspection exposure, based on a daily inspection of the storage cask or storage cask array, is provided for information only since a daily inspection is not required as long as the temperature monitoring system is operational. Other than an inspection of the Vertical Concrete Cask surface, no annual maintenance of the storage system is required. Collective dose due to design basis off-normal conditions and accident events, such as clearing the blockage of air vents, is accounted for in Chapter 11.0, and is not included in this evaluation.

Routine operations are expected to include:

- The optional daily electronic measurement of ambient air and air outlet temperatures for each cask in service. The outlet temperature-monitoring station may be located away from the cask array. Remote temperature measurement is not assumed to contribute to operator dose.

- An optional daily inspection of the concrete cask inlet and outlet screens to verify they are intact and unobstructed. The time required to perform the inspection, and the expected dose, will be site specific due to ISFSI pad dimensions and configurations, concrete cask array, distance of the inspector from the cask, etc.

- A daily security inspection of the fence and equipment surrounding the storage area. The security inspection is assumed to make no significant additional contribution to operator dose.

- Grounds maintenance performed every other week by 1 maintenance technician. Grounds maintenance is assumed to require 0.5 hour.

- Quarterly radiological surveillance. The surveillance consists of a radiological survey comprised of a surface radiation measurement on each cask, the determination and/or verification of general area exposure rates and radiological postings. This surveillance is assumed to require 1 hour and 1 person.

- Annual inspection of the general condition of the casks. This inspection is estimated to require 15 minutes per cask and require 2 technicians.

Calculation of the dose due to annual operation and surveillance requirements is estimated based on a single cask containing design basis fuel, and on an ISFSI array of 20 casks that are assumed to be loaded at the rate of 2 casks per year over a ten-year period. Consequently, the casks in the array are assumed to have the cool times as shown in Table 10.3-2. To account for the reduction in source term with cool time, weighting factors are applied to the neutron and gamma radiation spectra as shown in Table 10.3-3.

The annual operation and surveillance requirements result in an estimated annual collective exposure of 26.4 person-mrem for a single PWR cask containing design basis fuel and 17.0 person-mrem for a single design basis BWR cask. The annual operation and surveillance requirements for the assumed single cask and total estimated dose are shown in Table 10.3-4 for the single PWR cask and in Table 10.3-6 for the BWR cask. The annual operation and surveillance requirements for the assumed 20-cask ISFSI are shown in Tables 10.3-5 and 10.3-7 for PWR and BWR configurations, respectively. These tables show an estimated annual collective exposure of 377.6 person-mrem for the PWR cask configuration and 239.4 person-mrem for the BWR cask configuration for operation and maintenance of a 20-cask array.

Figure 10.3-1 Typical ISFSI 20 Cask Array Layout

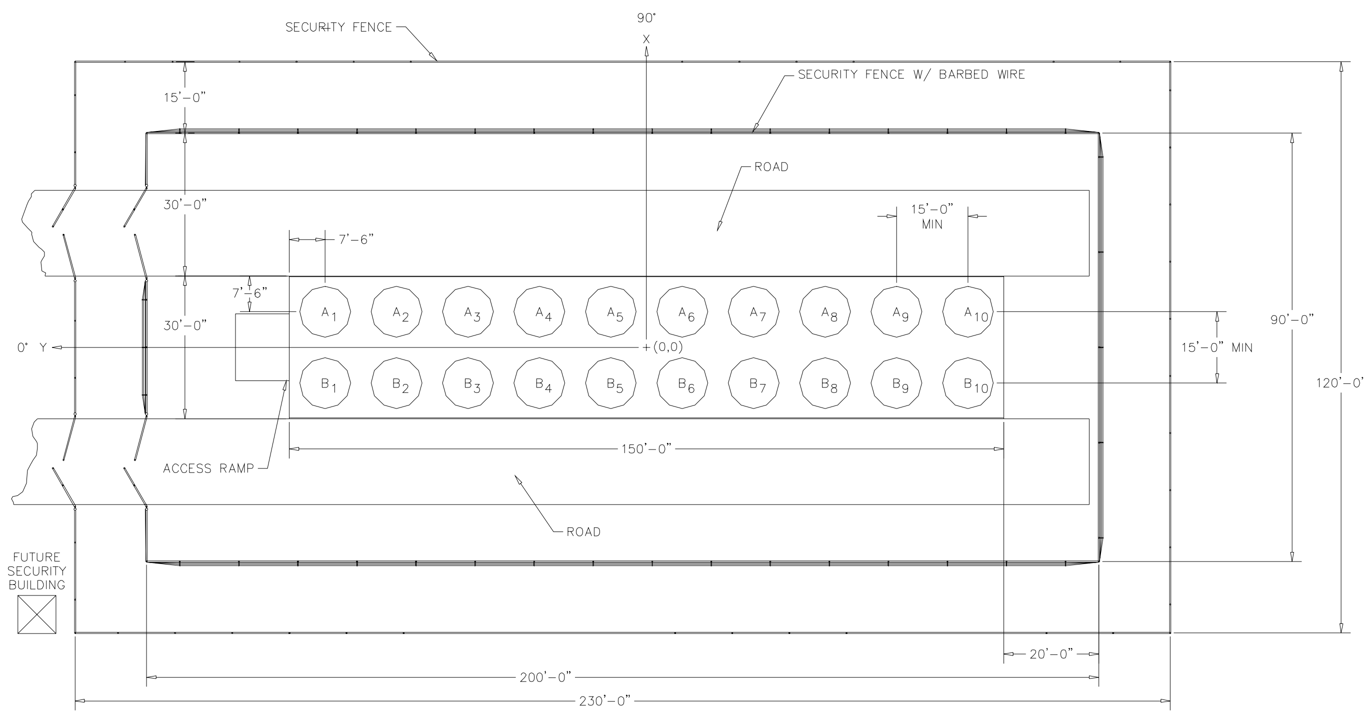

Table 10.3-1 Estimated Exposure for Operations Using the Standard Transfer Cask

Design Basis Fuel Assemblies Loading and Handling Activity	Estimated Number of Personnel[6]	Exposure Duration (hr)	Average Dose Rate (mrem/hr)		Exposure (person-mrem)	
			PWR	BWR	PWR	BWR
Load Canister[1]	2	9.9/21.9	2.1	2.0	42	88
Move to Decon Area/Prep for Weld	2	0.6	29.1	19.4	33	22
Setup Shield Lid Weld[3]	2	0.5	39.6	25.7	37	24
Welding Operation (Automated)	1	0.3	BDR[2]	BDR[2]	0	0
Weld Inspections[3,4]	1	7.5	10.4	6.6	78	50
Drain/ Vacuum Dry/Backfill and Leak Test[3,5]	2	0.4	30.0	20.4	25	17
Weld and Inspect Port Covers[3,4]	2	2.2	35.1	22.8	151	98
Setup Structural Lid Weld[3]	2	0.3	25.3	15.8	16	10
Welding Operation (Automated)	1	0.3	BDR[2]	BDR[2]	0	0
Weld Inspections[3,4]	1	7.7	6.8	4.0	52	31
Transfer to Vertical Concrete Cask	4	2.8	22.0	13.4	249	152
Position on ISFSI Pad	2	0.8	16.3	11.3	26	18
Total					709	510

1. Assumes 22.5 minutes for the loading of each PWR or BWR fuel assembly with additional time for installation of drain tube and shield lid prior to move to decontamination area.
2. Background Dose Rate (BDR). No exposure is estimated due to the canister contents.
3. Dose rates associated with the presence of a temporary shield on top of the shield lid.
4. Includes root, progressive, and final weld surface inspections.
5. Includes fixturing, connection and monitoring time. Operators not present during routine draining and drying process.
6. Number of personnel shown is a representative number. Personnel vary for the different operation stages, with total exposure divided over a larger number of personnel than the number shown.

Table 10.3-2 Assumed Contents Cooling Time of the Vertical Concrete Casks Depicted in the Typical ISFSI Array

Cask Number	**Cooling Time (yr)**		**Cask Number**	**Cooling Time (yr)**	
	PWR	**BWR**		**PWR**	**BWR**
A-1	14	14	B-1	14	14
A-2	13	13	B-2	13	13
A-3	12	12	B-3	12	12
A-4	11	11	B-4	11	11
A-5	10	10	B-5	10	10
A-6	9	9	B-6	9	9
A-7	8	8	B-7	8	8
A-8	7	7	B-8	7	7
A-9	6	6	B-9	6	6
A-10	5	5	B-10	5	5

Table 10.3-3 Vertical Concrete Cask Radiation Spectra Weighting Factors

Cask Numbers	Axial Neutron Weighting Factor		Axial Gamma Weighting Factor		Radial Neutron Weighting Factor		Radial Gamma Weighting Factor	
	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
A-1, B-1	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
A-2, B-2	0.96	0.96	0.83	0.84	0.96	0.96	0.83	0.83
A-3, B-3	0.93	0.93	0.72	0.74	0.93	0.93	0.72	0.74
A-4, B-4	0.89	0.89	0.65	0.67	0.89	0.89	0.65	0.67
A-5, B-5	0.86	0.86	0.59	0.62	0.86	0.86	0.59	0.62
A-6, B-6	0.83	0.83	0.55	0.58	0.83	0.83	0.55	0.58
A-7, B-7	0.80	0.80	0.52	0.55	0.80	0.80	0.52	0.55
A-8, B-8	0.77	0.77	0.50	0.52	0.77	0.77	0.50	0.52
A-9, B-9	0.74	0.74	0.47	0.50	0.74	0.74	0.48	0.50
A-10, B-10	0.72	0.72	0.45	0.48	0.72	0.72	0.46	0.48

Table 10.3-4 Estimate of Annual Exposure for the Operation and Surveillance of a Single PWR Cask

Activity	Dose Rate Distance (meters)	Frequency (days)	Time (min)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (Pers-mrem)
Radiological surveillance	4	4	15	7.40	1	7.4
Annual inspection						
Operations	1	1	15	25.30	1	6.3
Radiological Support	1	1	3	25.30	1	1.3
Grounds maintenance	10	26	15	1.76	1	11.4
Total Person-mrem						26.4

Table 10.3-5 Estimate of Annual Exposure for the Operation and Surveillance of a 20-Cask Array of PWR Casks

Activity	Dose Rate Distance (meters)	Frequency (days)	Time (min)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (Pers-mrem)
Radiological surveillance	4	4	60	5.96	1	23.8
Annual inspection						
Operations	1	1	15(1)	47.91	1	239.6
Radiological Support	1	1	3(1)	47.91	1	47.9
Grounds maintenance	10	26	60	2.55	1	66.3
Total Person-mrem for the 20-Cask Array						377.6
Total Person-mrem for a Single Cask in the Array						18.6

(1) Time listed is per cask; it is multiplied by 20 for the cask array.

Table 10.3-6 Estimate of Annual Exposure for the Operation and Surveillance of a Single BWR Cask

Activity	Dose Rate Distance (meters)	Frequency (days)	Time (min)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (mrem)
Radiological surveillance	4	4	15	4.9	1	4.9
Annual inspection						
Operations	1	1	15	15.2	1	3.8
Radiological Support	1	1	3	15.2	1	0.8
Grounds maintenance	10	26	15	1.16	1	7.5
Total Person - mrem						17.0

Table 10.3-7 Estimate of Annual Exposure for the Operation and Surveillance of a 20-Cask Array of BWR Casks

Activity	Dose Rate Distance (meters)	Frequency (days)	Time (min)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (mrem)
Radiological surveillance	4	4	60	4.2	1	16.8
Annual inspection						
Operations	1	1	15(1)	29.9	1	149.5
Radiological Support	1	1	3(1)	29.9	1	29.9
Grounds maintenance	10	26	60	1.7	1	43.2
Total Person - mrem for the 20-Cask Array						239.4
Total Person - mrem for a Single Cask in the Array						12.0

(1) Time listed is per cask; it is multiplied by 20 for the cask array.

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10.4 Exposure to the Public

The NAC Version 5.0.1 of the SKYSHINE-III code is used to evaluate the placement of the controlled area boundary for a single storage cask containing design basis fuel, and for a 20-cask array. For the 20-cask array, the storage casks are assumed to be loaded with design basis fuel at the rate of two casks per year. SKYSHINE III calculates dose rates for user defined detector locations for up to 100 point sources.

Version 5.0.1 of SKYSHINE-III explicitly calculates cask self-shielding based on the storage cask geometry and arrangement of the cask array. A ray tracing technique is utilized. Given the source position on the cask surface and the direction cosines for the source emission, geometric tests are made to see if any adjacent casks are in the path of the emission. If so, the emission history does not contribute to the air scatter dose. Also, given the source position on the cask surface and the direction cosines for the source to detector location, geometric tests are made to see if any adjacent casks are in the source path. If so, the emission position does not contribute to the uncollided dose at the detector location.

The code is benchmarked by modeling a set of Kansas State University ^{60}Co skyshine experiments and by modeling two Kansas State University neutron computational benchmarks. The code compares well with these benchmarks for both neutron and gamma doses versus distance.

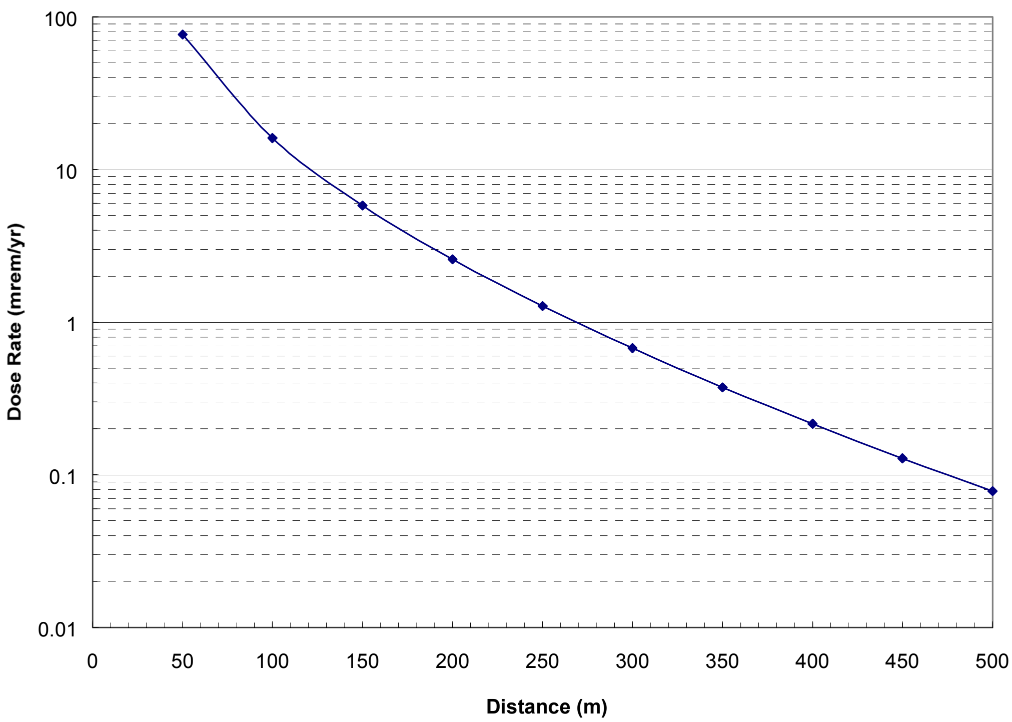
The storage cask array is explicitly modeled in the code, with the source term from each cask represented as top and side surface sources. Surface source emission fluxes are provided from one-dimensional SAS1 shielding evaluations. The top and side source energy distributions for both neutron and gamma radiation are taken from the design basis cask shielding evaluation. As stated in Section 10.3, the array cask source strengths are multiplied by weighting factors to correct for the differences in cooling times resulting from the assumption of a loading rate of 2 casks per year. The SKYSHINE cask surface fluxes (sources) are adjusted to reflect the higher cask surface fluxes calculated by the SAS4 three-dimensional shielding evaluation. Surface gamma-ray fluxes are also adjusted for dose peaks associated with fuel assembly end-fitting hardware and radiation streaming through the cask vents and canister-to-cask annulus. Air inlet and outlet dose rates have been recalculated in Section 5.4 based on the use of the MCBEND Monte Carlo code. The MCBEND generated air inlet dose rate results are significantly higher than those obtained from the SAS4 evaluation. Since the air inlets represent less than 0.6% of the total radial surface of the cask, and considering that the 100 mrem/hr air inlet and outlet dose rate limit is retained in the technical specification, an increase in the calculated air inlet dose rate (surface flux) will not significantly impact SKYSHINE results based on the SAS4 evaluation. The 2×10 ISFSI storage cask array layout is presented in Figure 10.3-1. For this analysis, the cask-to-cask pitch is conservatively taken at 16 feet, as opposed to the minimum 15 feet, to minimize cask-to-cask shadowing. These results are conservative for the minimum 15-foot cask center-to-center-spacing specified in Section 6.3.2.

Exposures are determined at distances ranging from 50 to 500 meters surrounding a single PWR and BWR storage cask containing design basis fuel. The results are presented graphically in Figures 10.4-1 and 10.4-2, for the PWR or BWR single cask, respectively. The storage casks in the 2×10 array are assumed to be loaded at the rate of 2 per year with design basis PWR and BWR spent fuel, with credit taken for the cool time that occurs during the 10-year period that the ISFSI array is completed. For both the single cask and 2×10 array calculations, the controlled area boundary is based on the 25 mrem/year limit. Occupancy at the controlled area boundary is assumed at 2,080 hours per year. While higher occupancy may be required at certain sites, the increased exposure time will likely be offset by increased cool time or decreased burnup.

Table 10.4-1 presents a summary of the dose rates versus distance for a single PWR and BWR storage cask containing design basis fuel. Linear interpolation of these results shows that minimum distances from a single cask to the site boundary of 93 meters and 84 meters for the design basis PWR and BWR fuels, respectively, are required for compliance with the requirements of 10 CFR 72.104(a), i.e., a dose rate of 25 mrem/year. Table 10.4-2 results show that a minimum site boundary of ≈195 meters is required for a 2×10 PWR cask array to meet the 10 CFR 72.104(a) 25 mrem/year requirement. The 2×10 BWR cask array requires a minimum site boundary of ≈186 meters to meet 10 CFR 72.104(a).

The distances used in Tables 10.4-1 and 10.4-2 are measured from the center of the 2×10 cask array along a line perpendicular to the center of the 10-cask face of the array.

Figure 10.4-1 SKYSHINE Exposures from a Single Cask Containing Design Basis PWR Fuel

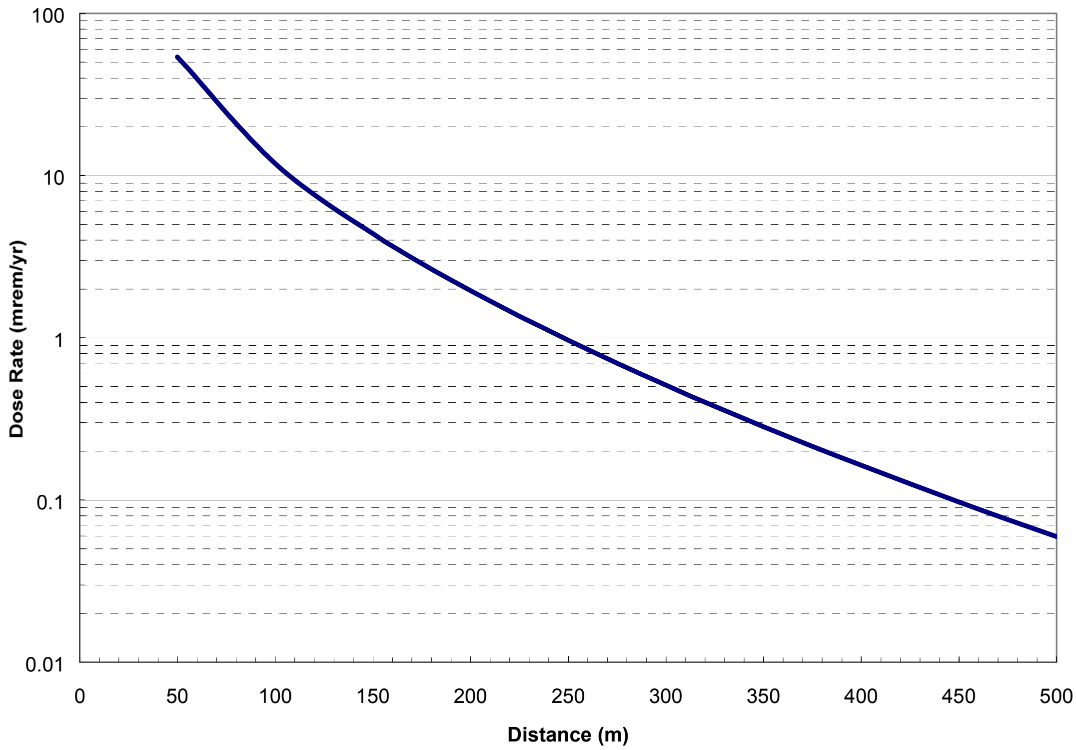


Distance from Center of Cask(m)	Dose Rate (mrem/year)			
	Gamma Dose	Neutron Dose	N-Gamma Dose	Total Dose
50	7.28E+01	3.85E+00	7.93E-04	77
100	1.47E+01	1.34E+00	8.07E-04	16
150	5.25E+00	5.56E-01	8.14E-04	5.8
200	2.32E+00	2.54E-01	7.86E-04	2.6
250	1.15E+00	1.24E-01	7.26E-04	1.3
300	6.12E-01	6.29E-02	6.43E-04	0.68
350	3.40E-01	3.34E-02	5.50E-04	0.37
400	1.97E-01	1.83E-02	4.58E-04	0.22
450	1.18E-01	1.03E-02	3.71E-04	0.13
500	7.19E-02	5.97E-03	2.95E-04	0.08

General Notes:

1. Based on a 2,080-hour exposure.
2. Axial gamma and radial neutron doses are negligible.

Figure 10.4-2 SKYSHINE Exposures from a Single Cask Containing Design Basis BWR Fuel



Distance from Center of Cask(m)	Dose Rate (mrem/year)			
	Gamma Dose	Neutron Dose	N-Gamma Dose	Total Dose
50	4.81E+01	5.80E+00	1.47E-03	54
100	9.86E+00	2.02E+00	1.27E-03	12
150	3.53E+00	8.40E-01	1.25E-03	4.4
200	1.57E+00	3.84E-01	1.20E-03	2.0
250	7.78E-01	1.86E-01	1.10E-03	0.97
300	4.15E-01	9.49E-02	9.78E-04	0.51
350	2.33E-01	5.03E-02	8.37E-04	0.28
400	1.35E-01	2.76E-02	6.96E-04	0.16
450	8.12E-02	1.56E-02	5.64E-04	0.10
500	5.00E-02	9.00E-03	4.48E-04	0.06

General Notes:

1. Based on a 2,080-hour exposure.
2. Axial gamma and radial doses are negligible.

Table 10.4-1 Dose Versus Distance For a Single Cask Containing Design Basis PWR or BWR Fuel

Distance from Center of Cask (m)	PWR Cask Total Dose Rate (mrem/y)[1]	BWR Cask Total Dose Rate (mrem/y)[1]
50	77	54
100	16	12
150	5.8	4.4
200	2.6	2.0
250	1.3	0.97
300	0.68	0.51
350	0.37	0.28
400	0.22	0.16
450	0.13	0.10
500	0.08	0.06

1. 2,080-hour exposure.

Table 10.4-2 Annual Exposures from a 2×10 Cask Array Containing Design Basis PWR or BWR Fuel

Distance from Center of Array (m)	PWR Cask Total Dose Rate (mrem/y)[1]	BWR Cask Total Dose Rate (mrem/y)[1]
50	600	466
100	135	111
150	49	41
200	22	19
250	11	9.2
300	5.8	4.9
350	3.2	2.7
400	1.9	1.5
450	1.1	0.90
500	0.67	0.55

1. 2,080-hour exposure.

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10.5 Radiation Protection Evaluation for Site Specific Spent Fuel

This section presents the radiation protection evaluation of fuel assemblies or configurations, which are unique to specific reactor sites. These site specific configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel that is classified as damaged.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation of the configuration.

10.5.1 Radiation Protection Evaluation for Maine Yankee Site Specific Spent Fuel

The shielding evaluation of Maine Yankee site specific fuel characteristics is presented in Section 5.6.1.1. In the shielding evaluation, the specific fuel assembly and non-fuel hardware sources are shown to be bounded by the design basis fuel assembly characteristics. To ensure that the Maine Yankee contents are bounded by the design basis fuel, specific evaluations are performed and minimum cooling time and loading restrictions are established.

Because the dose rates from the Maine Yankee contents are bounded by the design basis fuel, the radiological evaluations performed for the design basis fuel in Sections 10.3 and 10.4 are also bounding. Therefore, detailed radiological evaluations for the Maine Yankee site specific fuel configurations are not required and the evaluated on-site and off-site doses presented in Sections 10.3 and 10.4 can be used in site planning considerations.

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10.6 References

1. Title 10 of the Code of Federal Regulations, Part 72 (10 CFR 72), "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation," April 1996.

2. Title 10 of the Code of Federal Regulations, Part 20 (10 CFR 20), "Standards for Protection Against Radiation," April 1996.

3. ORNL/NUREG/CSD-2/V1/R5, Volume 1, Section S4, "SAS4: A Monte Carlo Cask Shielding Analysis Module Using an Automated Biasing Procedure," Tang, J. S., September 1995.

4. SKYSHINE III, "Calculation of the Effects of Structure Design on Neutron, Primary Gamma-Ray and Secondary Gamma-Ray Dose Rates in Air," RISC Code Package CCC-289, NAC International, Version 4.0.1, February 1997.

5. ORNL/NUREG/CSD-2/V3/R5, Volume 1, Section S1, "SAS1: A One-Dimensional Shielding Analysis Module," Knight, J.R. et al., September 1995.

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11.0 ACCIDENT ANALYSES

The analyses of the off-normal and accident design events, including those identified by ANSI/ANS 57.9-1992 [1], are presented in this chapter. Section 11.1 describes the off-normal events that could occur during the use of the Universal Storage System, possibly as often as once per calendar year. Section 11.2 addresses very low probability events that might occur once during the lifetime of the ISFSI or hypothetical events that are postulated because their consequences may result in the maximum potential impact on the surrounding environment.

The Universal Storage System includes Transportable Storage Canisters and Vertical Concrete Casks of five different lengths to accommodate three classes of PWR fuel or two classes of BWR fuel. In the analyses of this chapter, the bounding concrete cask parameters (such as weight and center of gravity) are conservatively used, as appropriate, to determine the cask's capability to withstand the effects of the analyzed events.

The load conditions imposed on the canisters and the baskets by the design basis normal, off-normal, and accident conditions of storage are less rigorous than those imposed by the transport conditions–including the 30-foot drop impacts and the fire accident (10 CFR 71) [2]. Consequently, the evaluation of the canisters and the baskets for transport conditions bounds those for storage conditions evaluated in this chapter. A complete evaluation of the normal and accident transport condition loading on the PWR and BWR canisters and the baskets is presented in the Safety Analysis Report for the Universal Transport Cask. [3]

This chapter demonstrates that the Universal Storage System satisfies the requirements of 10 CFR 72.24 and 10 CFR 72.122 [4] for off-normal and accident conditions. These analyses are based on conservative assumptions to ensure that the consequences of off-normal conditions and accident events are bounded by the reported results. If required for a site specific application, a more detailed evaluation could be used to extend the limits defined by the events evaluated in this chapter.

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11.1 Off-Normal Events

This section evaluates postulated events that might occur once during any calendar year of operations. The actual occurrence of any of these events is, therefore, infrequent.

11.1.1 Severe Ambient Temperature Conditions (106°F and -40°F)

This section evaluates the Universal Storage System for the steady state effects of severe ambient temperature conditions (106°F and -40°F).

11.1.1.1 Cause of Severe Ambient Temperature Event

Large geographical areas of the United States are subjected to sustained summer temperatures in the 90°F to 100°F range and winter temperatures that are significantly below zero. To bound the expected steady state temperatures of the canister and storage cask during these severe ambient conditions, analyses are performed to calculate the steady state storage cask, canister, and fuel cladding temperatures for a 106°F ambient temperature and solar loads (see Table 4.1-1). Similarly, winter weather analyses are performed for a -40°F ambient temperature with no solar load. Neither ambient temperature condition is expected to last more than several days.

11.1.1.2 Detection of Severe Ambient Temperature Event

Detection of off-normal ambient temperatures would occur during measurement of site ambient temperatures.

11.1.1.3 Analysis of Severe Ambient Temperature Event

Off-normal temperature conditions are evaluated by using the thermal models described in Section 4.4.1. The design basis heat load of 23.0 kW is used in the evaluation of PWR and BWR fuels. The concrete temperatures are determined using the two-dimensional axisymmetric air flow and concrete cask models (Section 4.4.1.1) and the canister, basket and fuel cladding temperatures are determined using the three-dimensional canister models (Section 4.4.1.2). A steady state condition is considered in all analyses. The temperature profiles for the concrete cask and for the air flow associated with a 106°F ambient condition are shown in Figure 11.1.1-1 and Figure 11.1.1-2, respectively. Temperature profiles for the -40°F ambient temperature condition for the PWR fuel are shown in Figure 11.1.1-3 and Figure 11.1.1-4. Temperature profiles for the BWR cask are similar.

The principal component temperatures for each of the ambient temperature conditions discussed above are summarized in the following table along with the allowable temperatures. As the table shows, the component temperatures are within the allowable values for the off-normal ambient conditions.

Component	106°F Ambient Max Temp. (°F)		-40°F Ambient Max Temp. (°F)		Allowable Temp. (°F)	
	PWR	BWR	PWR	BWR	PWR	BWR
Fuel Cladding	672	667	561	540	1058	1058
Support Disks	628	640	505	505	800	700
Heat Transfer Disks	626	638	502	504	750	750
Canister Shell	381	405	226	252	800	800
Concrete	228	231	17	20	350	350

The thermal stress evaluations for the concrete cask for these off-normal conditions are bounded by those for the accident condition of "Maximum Anticipated Heat Load (133°F ambient temperature)" as presented in Section 11.2.7. Thermal stress analyses for the canister and basket components are performed using the ANSYS finite element models as described in Section 3.4.4. Evaluations of the thermal stresses combined with the stresses due to other off-normal loads (e.g., canister internal pressure and handling) are shown in Section 11.1.3.

There are no adverse consequences for these off-normal conditions. The maximum component temperatures are within the allowable temperature values.

11.1.1.4 Corrective Actions

No corrective actions are required for this off-normal condition.

11.1.1.5 Radiological Impact

There is no radiological impact due to this off-normal event.

Figure 11.1.1-1 Concrete Temperature (ºF) for Off-Normal Storage Condition 106°F Ambient Temperature (PWR Fuel)

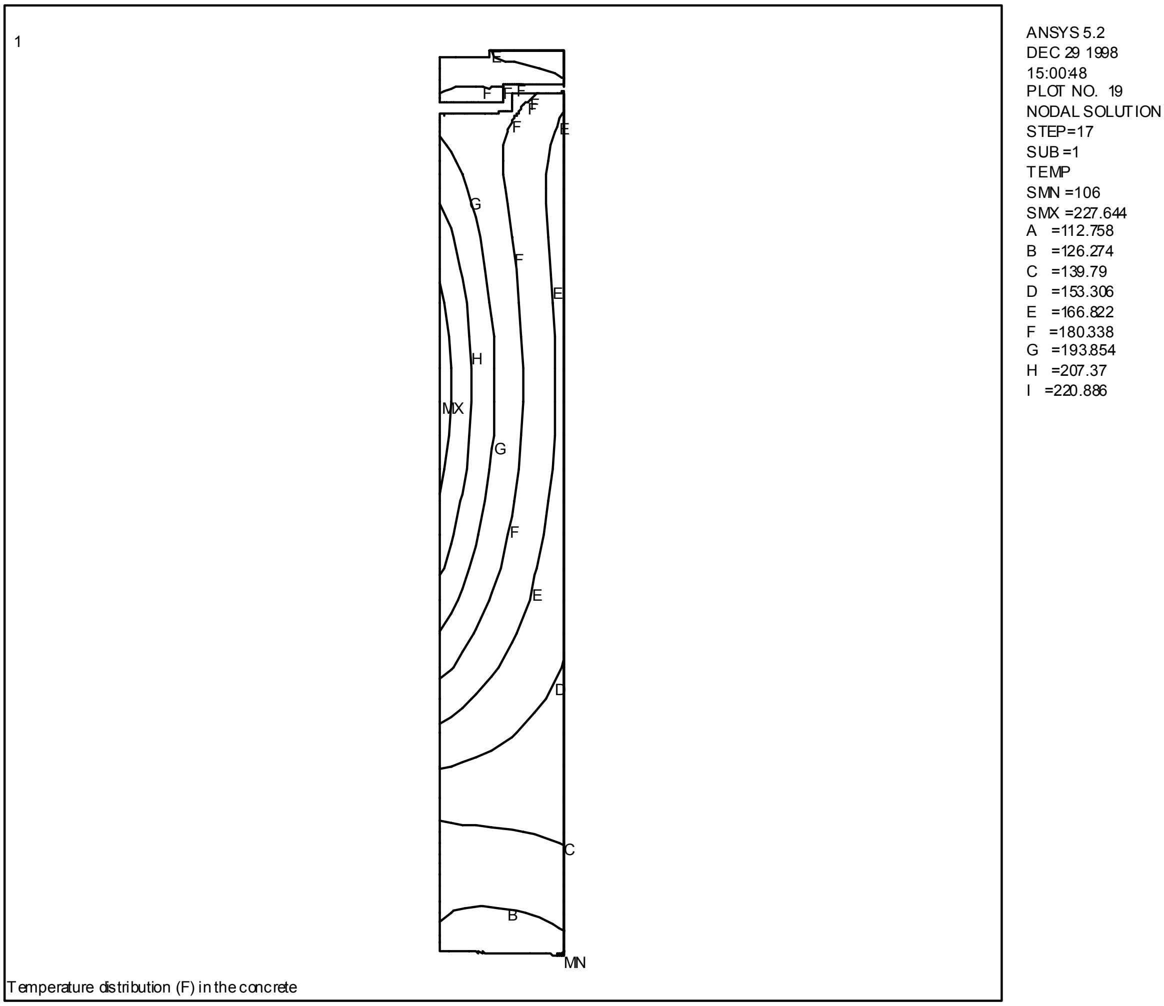
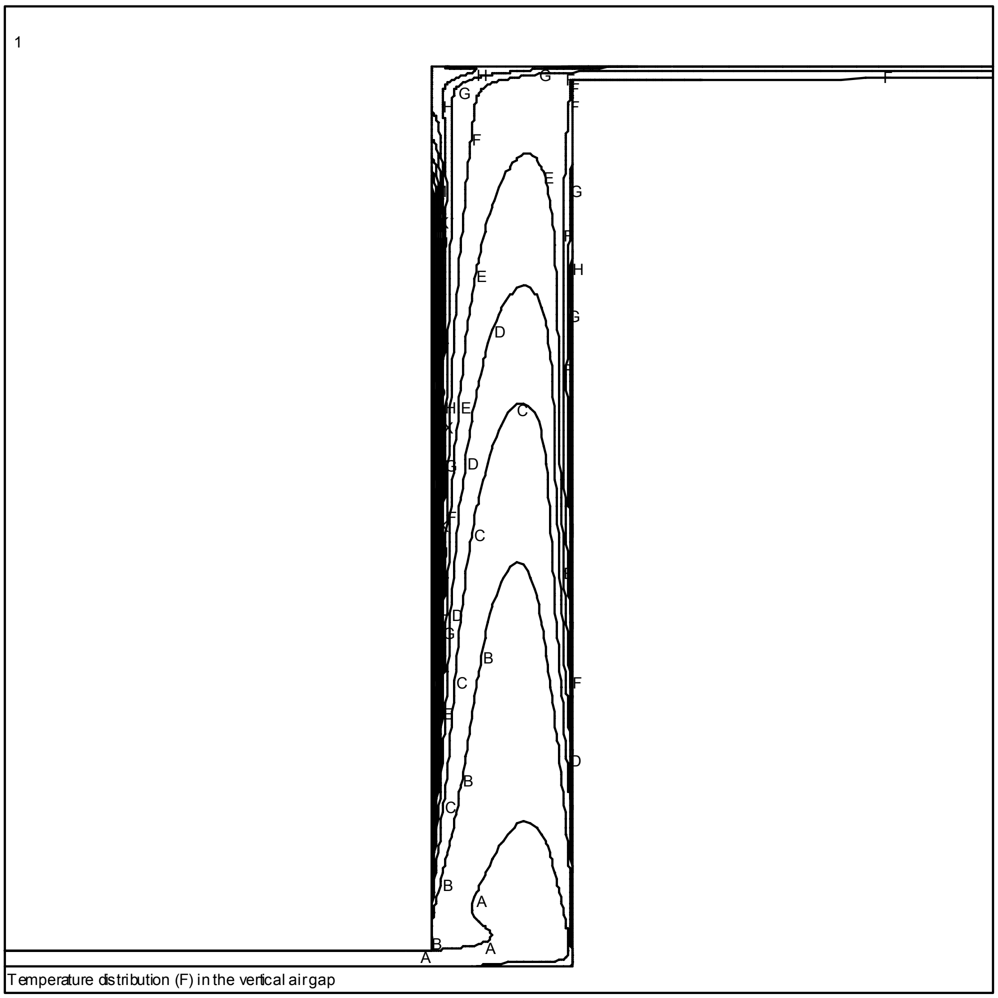


Figure 11.1.1-2 Vertical Concrete Cask Air Temperature (ºF) Profile for Off-Normal Storage Condition 106°F Ambient Temperature (PWR Fuel)



ANSYS 5.2
DEC 29 1998
15:00:44
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SUB =1
TEMP
SMN =106
SMX =376.253
A =113.507
B =128.521
C =143.535
D =158.549
E =173.563
F =188.577
G =203.591
H =218.605
I =233.619
J =248.634
K =263.648
L =278.662
M =293.676
N =308.69
O =323.704
P =338.718
Q =353.732
R =368.746

Figure 11.1.1-3 Concrete Temperature (ºF) for Off-Normal Storage Condition -40°F Ambient Temperature (PWR Fuel)

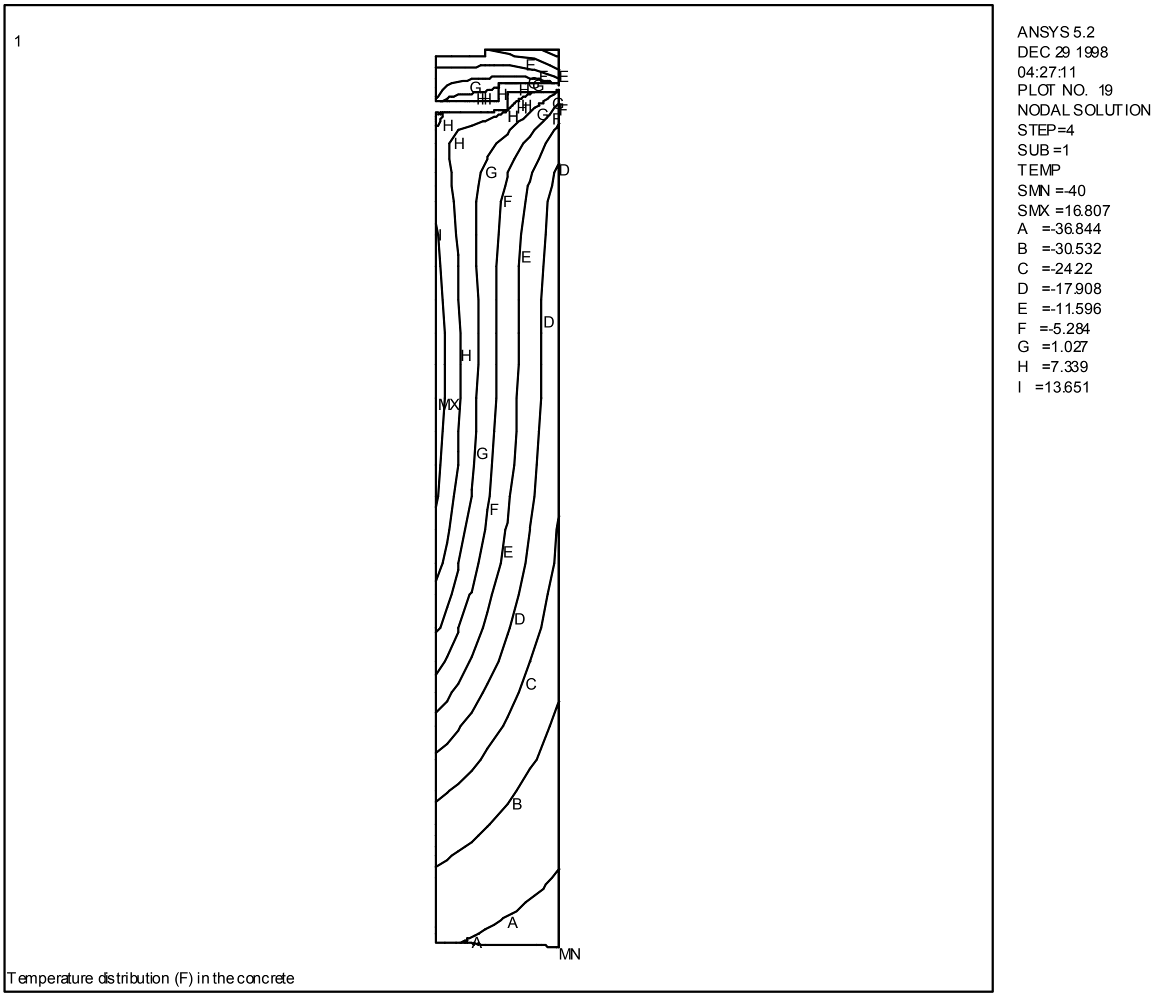
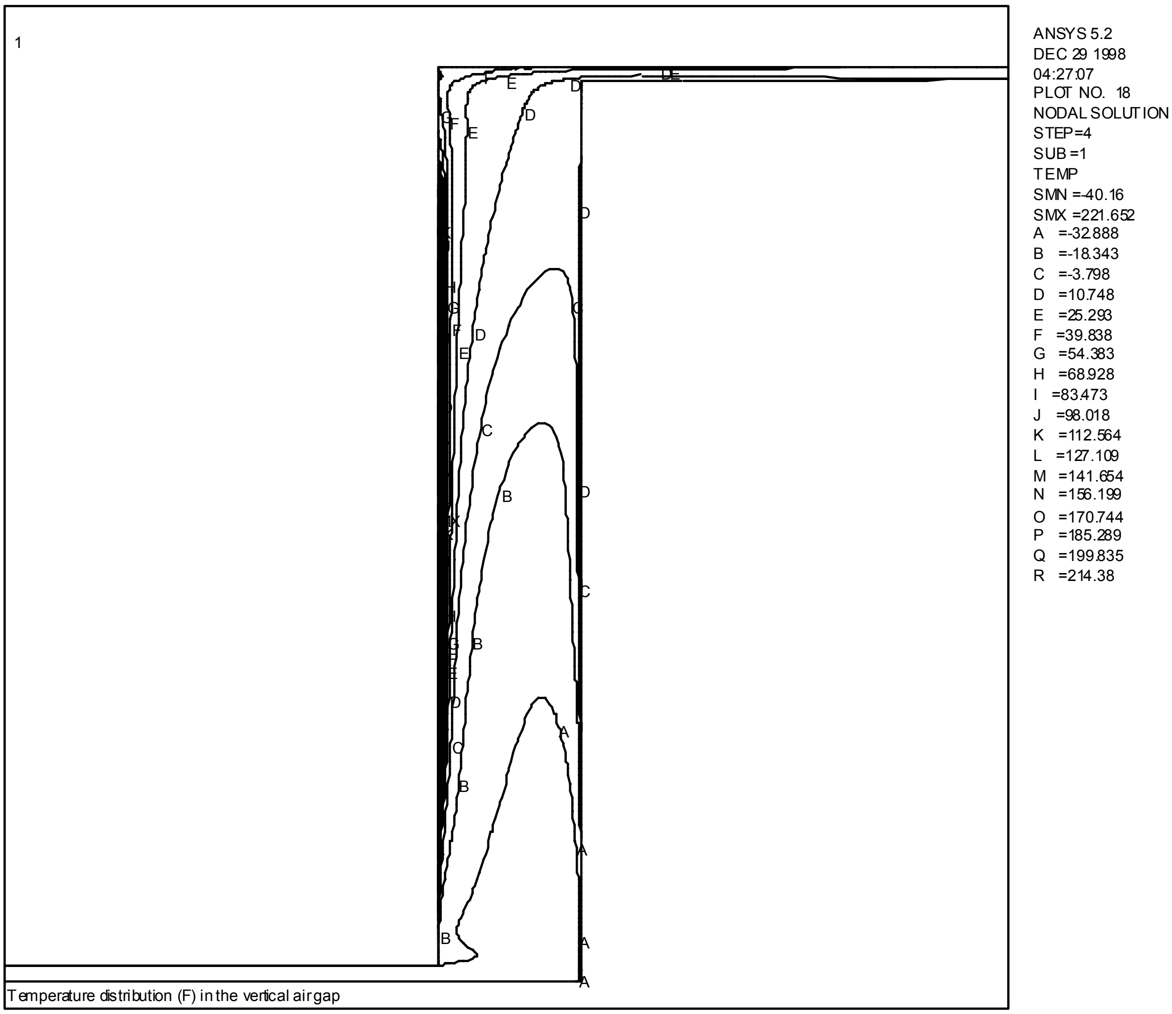


Figure 11.1.1-4 Vertical Concrete Cask Air Temperature (ºF) Profile for Off-Normal Storage Condition -40°F Ambient Temperature (PWR Fuel)



11.1.2 Blockage of Half of the Air Inlets

This section evaluates the Universal Storage System for the steady state effects of a blockage of one-half of the air inlets at the normal ambient temperature (76°F).

11.1.2.1 Cause of the Blockage Event

Although unlikely, blockage of half of the air inlets may occur due to blowing debris, snow, intrusion of a burrowing animal, etc. The screens over the inlets are expected to minimize any blockage of the inlet channels.

11.1.2.2 Detection of the Blockage Event

This event would be detected by the daily concrete cask operability inspection, which is performed either by outlet air temperature measurements or by visual inspection of the inlet and outlet screens for blockage and integrity. It could also be detected by security forces, or other operations personnel, engaged in other routine activities such as fence inspection, or grounds maintenance.

11.1.2.3 Analysis of the Blockage Event

Using the same methods and the same thermal models described in Section 11.1.1 for the off-normal conditions of severe ambient temperatures, thermal evaluations are performed for the concrete cask and the canister and its contents for this off-normal condition. The boundary condition of the two-dimensional axisymmetric air flow and concrete cask model is modified to allow only half of the air flow into the air inlet to simulate the half inlets blocked condition. The calculated maximum component temperatures due to this off-normal condition are compared to the allowable component temperatures. Table 11.1.2-1 summarizes the component temperatures for off-normal conditions. As the table demonstrates, the calculated temperatures are shown to be below the component allowable temperatures.

The thermal stress evaluations for the concrete cask for this off-normal condition are bounded by those for the accident condition of “Maximum Anticipated Heat Load (133°F ambient temperature)” as presented in Section 11.2.7. Thermal stress analyses for the canister and basket components are performed using the ANSYS finite element models described in Section 3.4.4. Evaluations of the thermal stresses combined with stresses due to other off-normal loads (e.g., canister internal pressure and handling) are shown in Section 11.1.3.

11.1.2.4 Corrective Actions

The debris blocking the affected air inlets must be manually removed. The nature of the debris may indicate that other actions are required to prevent recurrence of the blockage.

11.1.2.5 Radiological Impact

There are no significant radiological consequences for this event. Personnel will be subject to an estimated maximum contact dose rate of 66 mrem/hr when clearing the PWR cask inlets. If it is assumed that a worker kneeling with his hands on the inlets would require 15 minutes to clear the inlets, the estimated maximum extremity dose is 17 mrem. For clearing the BWR cask inlets, the maximum contact dose rate and the maximum extremity dose are estimated to be 51 mrem/hr and 13 mrem, respectively. The whole body dose in both PWR and BWR cases will be significantly less.

Table 11.1.2-1 Component Temperatures (°F) for Half of Inlets Blocked Off-Normal Event

Component	Half of Inlets Blocked Max Temperature (°F)		Allowable Temperature (°F)	
	PWR	BWR	PWR	BWR
Fuel Cladding	649	642	1058	1058
Support Disks	603	614	800	700
Heat Transfer Disks	600	612	750	750
Canister Shell	350	373	800	800
Concrete	191	195	350	350

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11.1.3 Off-Normal Canister Handling Load

This section evaluates the consequence of loads on the Transportable Storage Canister during the installation of the canister in the Vertical Concrete Cask, or removal of the canister from the concrete cask or from the transfer cask. The canister may be handled vertically in the Standard or Advanced transfer casks. The Standard and Advanced transfer casks are similar, except that the Advanced transfer cask incorporates a reinforcing gusset at the lifting trunnions allowing an increased canister weight.

11.1.3.1 Cause of Off-Normal Canister Handling Load Event

Unintended loads could be applied to the canister due to misalignment or faulty crane operation, or inattention of the operators.

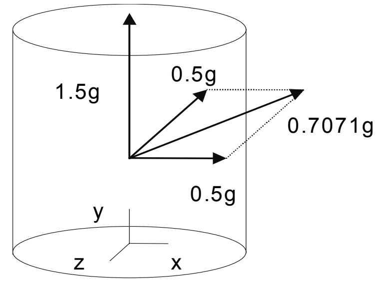
11.1.3.2 Detection of Off-Normal Canister Handling Load Event

The event can be detected visually during the handling of the canister, or banging or scraping noise associated with the canister movement. The event is expected to be obvious to the operators at the time of occurrence.

11.1.3.3 Analysis of Off-Normal Canister Handling Load Event

The canister off-normal handling analysis is performed using an ANSYS finite element model as shown in Figure 11.1.3.1-1. The model is based on the canister model presented in Section 3.4.4.1 with the elements for the fuel basket (support disks and top and bottom weldment disks) added. The disks are modeled with SHELL63 elements. These elements are included to transfer loads from the basket to the canister shell for loads in the canister transverse direction. The interface between the disks and the canister shell is simulated by CONTAC52 elements. For the transverse loads, uniform pressure loads representing the weight (including appropriate g-loading) of the fuel assemblies, fuel tubes, heat transfer disks, tie-rods, spacers, washers, and nuts are applied to the slots of the support/weldment disks. Interaction between the fuel basket and the canister during vertical load conditions is modeled by applying a uniform pressure representing the weight of the fuel assemblies and basket (including appropriate g-loading) to the canister bottom plate. The model is used to evaluate the canisters for both PWR and BWR fuel types by modeling the shortest canister with minimum lid-to-shell weld sizes (Class 1 PWR) with the heaviest fuel/fuel basket weight (Class 5 BWR). The material stress allowables used in the analysis consider the higher component temperatures that occur during transfer operations.

The off-normal canister handling loads are defined as 0.5g applied in all directions (i.e., in the global x, y, and z directions) in addition to a 1g lifting load applied in the finite element model. The resulting off-normal handling accelerations are 0.7071g in the lateral direction and 1.5g (0.5g + 1g) in the vertical direction.



The boundary conditions (restraints) for the canister model are the same as those described in Section 3.4.4.1.4 for the normal handling condition. In addition, for the lateral loading, the canister is assumed to be handled inside the vertical concrete cask. The interface between the canister shell and the concrete cask inner surface is represented using CONTAC52 elements.

The resulting maximum canister stresses for off-normal handling loads are summarized in Tables 11.1.3-1 and 11.1.3-2 for primary membrane and primary membrane plus bending stresses, respectively.

The resulting maximum canister stresses for combined off-normal handling, maximum off-normal internal pressure (15 psig), and thermal stress loads are summarized in Tables 11.1.3-3, 11.1.3-4, and 11.1.3-5 for primary membrane, primary membrane plus bending, and primary plus secondary stresses, respectively.

The sectional stresses shown in Tables 11.1.3-1 through 11.1.3-5 at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations of the stress sections are shown in Figure 3.4.4.1-4.

To determine the structural adequacy of the PWR and BWR fuel basket support disks and weldments for off-normal conditions, a structural analysis is performed by using ANSYS to evaluate off-normal handling loads. To simulate off-normal loading conditions, an inertial load of 1.5g is applied to the support disk and the weldments in the axial (canister axial) direction and 0.5g in two orthogonal disk in-plane directions (0.707g resultant), for the governing case (canister handled in the vertical orientation).

Stresses in the support disks and weldments are calculated by applying the off-normal loads to the ANSYS models described in Sections 3.4.4.1.8 and 3.4.4.1.9. An additional in-plane displacement constraint is applied to each model at one node (conservative) at the periphery of the disk or the weldment plate to simulate the side restraint of the canister shell for the lateral load (0.7071g). To evaluate the most critical regions of the support disks, a series of cross sections is considered. The locations of these sections on the PWR and BWR support disks are shown in Figures 3.4.4.1-7, 3.4.4.1-8, and Figures 3.4.4.1-13 through 3.4.4.1-16. (Note: stress allowables for support disks are taken at 800°F.) The stress evaluation for the support disk and weldment is performed according to ASME Code, Section III, Subsection NG. For off-normal conditions, Level C allowable stresses are used: the allowable stress is 1.2 S_m or S_y, 1.8 S_m or $1.5S_y$, and 3.0 S_m for the P_m, P_m+P_b, and P_m+P_b+Q stress categories, respectively. The stress evaluation results are presented in Tables 11.1.3-6 through 11.1.3-8 for the PWR support disks and in Tables 11.1.3-9 through 11.1.3-11 for the BWR support disks. The tables list the 40 sections with the highest P_m, P_m+P_b, and P_m+P_b+Q stress intensities. All of the support disk sections have large margins of safety. The stress results for the PWR and BWR weldments are shown in Table 11.1.3-12.

The canisters and fuel baskets maintain positive margins of safety for the off-normal handling condition. There is no deterioration of canister or fuel basket performance. The Universal Storage System is in compliance with all applicable regulatory criteria.

11.1.3.4 Corrective Actions

Operations should be halted until the cause of the misalignment, interference or faulty operation is identified and corrected. Since the radiation level of the canister sides and bottom is high, extreme caution should be exercised if inspection of these surfaces is required.

11.1.3.5 Radiological Impact

There are no radiological consequences associated with this off-normal event.

Figure 11.1.3.1-1 Canister and Basket Finite Element Model

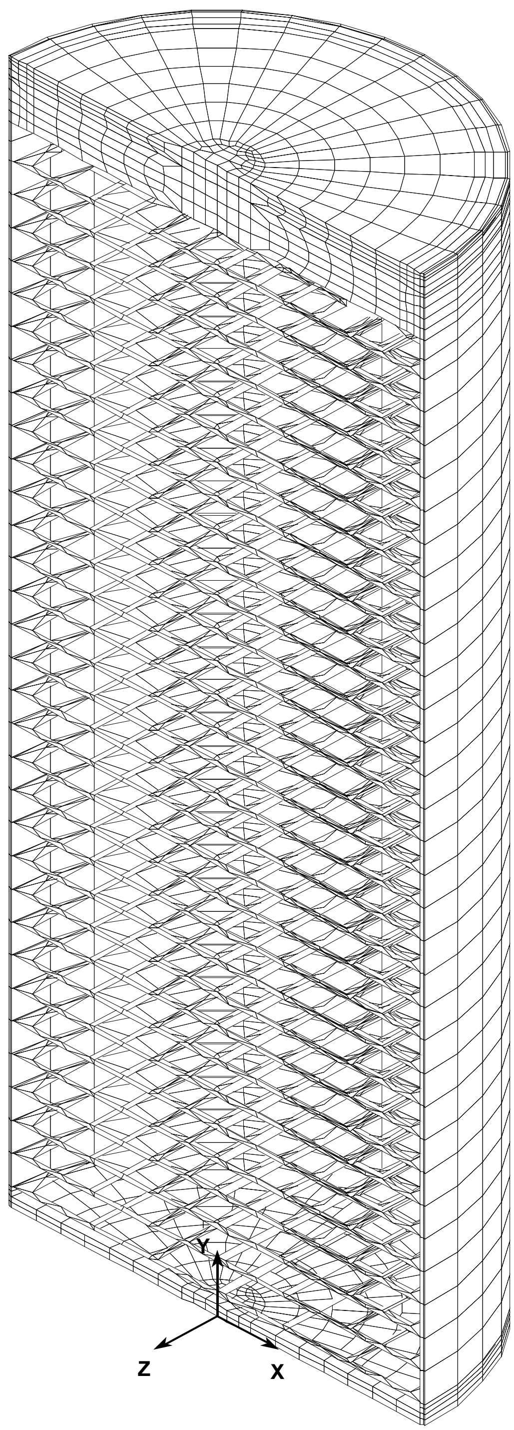


Table 11.1.3-1 Canister Off-Normal Handling (No Internal Pressure) Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	Angle (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0	-0.65	0.66	2.73	0.07	0.02	-0.03	3.39
2	0	2.02	-2.42	-1.40	0.36	0.07	-0.23	4.52
3	0	-0.32	-3.62	1.16	0.28	0.07	0.89	5.23
4	0	-0.04	0.00	0.83	0.00	0.01	-0.02	0.87
5	0	-0.09	0.00	0.76	0.00	0.00	0.00	0.86
6	0	-0.12	-0.01	0.79	0.00	0.00	0.00	0.91
7	0	-0.14	-0.04	0.93	0.01	-0.01	0.00	1.07
8	0	0.05	0.01	1.81	-0.03	-0.16	-0.02	1.85
9	0	0.05	0.55	2.77	-0.04	-0.29	0.10	2.77
10	0	-0.33	0.53	3.51	-0.12	-0.40	0.11	3.91
11	0	-0.62	1.28	2.39	-0.06	-0.31	-0.71	3.41
12	0	-0.14	0.76	3.53	-0.15	-0.21	0.30	3.75
13	0	-2.09	1.36	-0.52	-0.13	-0.05	-1.61	4.46
14	0	0.35	0.40	-0.01	0.00	0.19	-0.03	0.56
15	180	-0.04	-0.04	0.00	0.00	0.00	0.00	0.04
16	0	-0.02	0.03	0.00	0.00	-0.01	0.00	0.05

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

Table 11.1.3-2 Canister Off-Normal Handling (No Internal Pressure) Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	Angle (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0	3.64	0.54	7.08	0.13	-0.03	0.26	6.57
2	0	0.77	-5.92	-12.15	0.61	0.18	-0.84	13.09
3	0	-1.34	0.67	17.12	-0.15	-0.15	1.08	18.60
4	0	-0.04	-0.24	0.76	0.02	0.03	-0.02	1.01
5	0	-0.09	0.03	0.77	-0.01	0.00	0.00	0.86
6	0	-0.13	0.07	0.81	-0.01	0.00	0.00	0.94
7	0	-0.16	0.13	0.97	0.00	-0.01	0.00	1.13
8	0	0.06	0.14	1.96	-0.04	-0.13	-0.02	1.93
9	0	-0.15	0.50	3.08	0.00	-0.40	-0.06	3.29
10	0	-0.54	1.03	5.09	-0.21	-0.25	0.35	5.71
11	0	-1.12	1.25	1.58	-0.05	-0.28	-1.69	4.38
12	0	-0.58	0.92	4.68	-0.21	-0.24	0.34	5.35
13	0	-4.53	1.12	-1.97	-0.29	0.11	-1.38	6.29
14	180	8.93	8.96	0.25	0.00	0.17	-0.04	8.72
15	0	-0.25	-0.24	-0.01	0.00	0.00	0.00	0.24
16	0	1.02	1.08	0.03	0.01	-0.01	0.00	1.05

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

Table 11.1.3-3 Canister Off-Normal Handling plus Normal/Off-Normal Internal Pressure (15 psig) Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	Angle (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0	-0.63	1.20	4.20	0.04	0.01	-0.21	4.85	21.04	3.3
2	0	3.00	-3.67	-2.33	0.53	0.06	-0.44	6.79	21.03	2.1
3	0	-0.50	-5.51	1.61	0.44	0.12	1.32	7.80	20.99	1.7
4	0	-0.02	0.78	1.28	-0.06	0.02	-0.04	1.31	19.39	13.8
5	0	-0.09	0.78	1.18	-0.07	0.00	0.00	1.28	17.93	13.1
6	0	-0.12	0.77	1.20	-0.07	0.00	0.00	1.33	17.77	12.4
7	0	-0.16	0.74	1.33	-0.06	-0.01	0.00	1.49	19.12	11.8
8	0	0.01	0.47	2.24	-0.06	-0.18	-0.01	2.26	20.51	8.1
9	0	0.04	0.81	3.18	-0.08	-0.30	0.12	3.19	20.94	5.6
10	0	-0.43	0.74	3.78	-0.14	-0.41	0.04	4.27	20.95	3.9
11	0	-0.49	1.40	2.33	-0.08	-0.30	-0.71	3.23	21.06	5.5
12	0	-0.22	0.79	3.17	-0.16	-0.21	0.20	3.46	20.94	5.1
13	0	-1.83	1.53	-0.35	-0.17	-0.04	-1.56	4.36	21.07	3.8
14	0	0.59	0.65	-0.02	0.00	0.30	-0.05	0.90	20.08	21.4
15	180	-0.06	-0.06	-0.01	0.00	0.00	0.00	0.06	20.96	373.8
16	0	0.01	0.05	0.00	0.00	-0.01	0.00	0.06	21.08	367.5

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

(2) ASME Service Level C is used for material allowable stress.

Table 11.1.3-4 Canister Off-Normal Handling plus Normal/Off-Normal Internal Pressure (15 psig) Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	Angle (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0	4.89	0.68	10.67	0.18	-0.05	0.25	10.01	31.23	2.1
2	0	1.23	-9.06	-18.95	0.91	0.16	-1.32	20.43	31.21	0.53
3	0	-2.06	1.32	26.71	-0.24	-0.11	1.61	28.97	31.11	0.1
4	0	-0.02	1.10	1.36	-0.09	0.00	-0.04	1.39	27.25	18.7
5	0	-0.09	0.82	1.19	-0.07	0.00	0.00	1.28	24.83	18.4
6	0	-0.14	0.89	1.23	-0.08	0.01	0.00	1.38	24.62	16.9
7	0	-0.18	0.99	1.40	-0.08	-0.01	0.00	1.58	26.62	15.8
8	0	0.01	0.47	2.32	-0.06	-0.15	-0.01	2.33	29.94	11.9
9	0	-0.11	0.94	4.09	-0.06	-0.40	0.03	4.25	30.97	6.3
10	0	-0.63	1.00	4.54	-0.21	-0.26	0.22	5.23	31.01	4.9
11	0	-0.93	1.50	2.00	-0.08	-0.29	-1.72	4.56	31.28	5.9
12	0	-0.69	0.89	4.14	-0.21	-0.25	0.20	4.89	30.98	5.3
13	0	-4.11	1.30	-1.86	-0.34	0.12	-1.28	6.04	31.29	4.2
14	170	14.01	14.04	0.40	-0.01	0.27	-0.07	13.66	28.91	1.1
15	0	-0.20	-0.22	-0.02	0.00	0.00	0.00	0.20	31.03	150.7
16	0	1.04	1.11	0.03	0.01	-0.01	0.00	1.08	31.33	28.0

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

(2) ASME Service Level C is used for material allowable stress.

Table 11.1.3-5 Canister Off-Normal Handling plus Normal/Off-Normal Internal Pressure (15 psig) Primary plus Secondary (P + Q) Stresses (ksi)

Section No. [1]	Angle (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	60	4.78	3.21	14.22	0.17	0.20	0.19	11.03	50.10	3.5
2	0	1.64	-8.45	-22.75	0.88	-0.11	-1.36	24.62	50.10	1.03
3	50	-2.19	3.26	30.88	-0.05	0.48	1.50	33.21	50.10	0.5
4	0	-0.07	2.44	1.37	-0.18	0.56	0.00	2.78	48.46	16.5
5	0	-1.39	9.10	0.08	-0.90	0.79	-0.08	10.71	44.83	3.2
6	0	-1.60	9.78	0.13	-0.98	-0.87	0.10	11.62	44.44	2.8
7	0	-0.26	2.93	2.15	-0.20	-0.64	0.03	3.58	47.79	12.4
8	0	0.21	1.55	4.40	-0.11	-0.13	0.03	4.21	50.10	10.9
9	0	1.13	2.00	6.96	0.00	-0.12	1.36	6.44	50.10	6.8
10	0	-7.08	-1.89	2.43	-0.33	-0.11	-0.94	9.71	50.10	4.2
11	140	2.31	-2.03	-10.03	0.10	-0.09	0.99	12.50	50.10	3.01
12	0	-7.08	-1.89	2.43	-0.33	-0.11	-0.94	9.71	50.10	4.2
13	30	-5.47	-0.78	1.84	-0.39	0.07	0.65	7.46	50.10	5.7
14	180	-15.40	-15.03	-0.23	0.26	0.00	-0.11	15.31	50.10	2.27
15	180	-8.41	-7.57	-6.63	0.20	0.49	0.01	2.05	50.10	23.5
16	180	0.33	0.22	-0.56	0.03	-0.06	0.01	0.90	50.10	54.8

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

(2) ASME Service Level C is used for material allowable stress.

Table 11.1.3-6 P_m Stresses for PWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
120	0.8	-0.8	0.1	1.6	77.7	47.6
114	-0.5	1.0	-0.1	1.5	77.7	50.8
21	-0.3	-1.1	0.1	1.1	77.7	69.6
37	-1.1	-0.3	0.1	1.1	77.7	69.6
23	0.0	1.0	0.2	1.1	77.7	69.6
35	1.0	0.0	0.2	1.1	77.7	69.6
111	-0.3	0.5	0.2	0.9	77.7	85.3
112	0.5	-0.3	0.2	0.9	77.7	85.3
98	-0.5	-0.8	-0.2	0.9	77.7	85.3
40	0.1	-0.7	0.1	0.9	77.7	85.3
28	-0.8	0.1	0.1	0.9	77.7	85.3
51	0.8	0.1	0.1	0.8	77.7	96.1
7	0.1	0.8	0.1	0.8	77.7	96.1
110	-0.8	0.0	0.1	0.8	77.7	96.1
72	-0.8	-0.7	0.0	0.8	77.7	96.1
26	-0.8	-0.4	0.1	0.8	77.7	96.1
119	0.0	-0.8	0.1	0.8	77.7	96.1
42	-0.4	-0.8	0.1	0.8	77.7	96.1
95	0.0	-0.8	0.1	0.8	77.7	96.1
64	-0.8	0.0	0.1	0.8	77.7	96.1
49	-0.7	0.0	0.1	0.8	77.7	96.1
9	0.0	-0.7	0.1	0.8	77.7	96.1
94	-0.8	0.0	0.1	0.8	77.7	96.1
71	0.0	-0.7	0.1	0.8	77.7	96.1
46	-0.7	-0.2	0.1	0.7	77.7	110.0
123	-0.3	0.4	-0.1	0.7	77.7	110.0
124	0.4	-0.3	-0.1	0.7	77.7	110.0
96	-0.4	0.1	0.2	0.7	77.7	110.0
63	0.1	-0.4	0.2	0.7	77.7	110.0
92	0.2	-0.4	-0.2	0.7	77.7	110.0
91	-0.4	0.2	-0.2	0.7	77.7	110.0
99	-0.5	0.1	0.0	0.7	77.7	110.0
74	0.1	-0.5	0.0	0.7	77.7	110.0
104	-0.6	0.0	-0.2	0.6	77.7	128.5
106	0.1	-0.5	-0.1	0.6	77.7	128.5
117	-0.4	0.2	0.0	0.6	77.7	128.5
113	0.2	-0.3	0.0	0.6	77.7	128.5
67	-0.5	0.1	-0.1	0.6	77.7	128.5
88	0.5	0.2	-0.2	0.6	77.7	128.5
39	0.0	-0.5	0.1	0.6	77.7	128.5

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.
2. Stress allowables are taken at 800°F.

Table 11.1.3-7 $P_m + P_b$ Stresses for PWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
37	-2.5	-5.1	0.6	5.3	63.2	10.9
21	-5.1	-2.5	0.6	5.3	63.2	10.9
120	-0.4	-5.1	0.4	5.1	63.2	11.4
23	4.5	2.5	0.6	4.6	63.2	12.7
35	2.4	4.5	0.6	4.6	63.2	12.7
4	3.0	4.3	0.4	4.5	63.2	13.0
1	4.3	3.0	0.4	4.4	63.2	13.4
112	-1.1	-4.7	0.0	4.7	63.2	12.4
111	-4.7	-1.1	0.0	4.7	63.2	12.4
51	2.0	4.3	0.5	4.4	63.2	13.4
7	4.3	2.0	0.5	4.4	63.2	13.4
9	-3.9	-1.9	0.5	4.0	63.2	14.8
49	-1.9	-3.9	0.5	4.0	63.2	14.8
66	4.1	1.0	0.4	4.1	63.2	14.4
3	-3.7	-2.8	0.5	3.9	63.2	15.2
2	-2.8	-3.6	0.5	3.8	63.2	15.6
20	-2.9	-3.7	0.4	3.9	63.2	15.2
34	-3.7	-2.9	0.4	3.9	63.2	15.2
42	-0.9	-4.0	0.2	4.0	63.2	14.8
26	-4.0	-0.9	0.2	4.0	63.2	14.8
96	0.9	3.9	0.0	3.9	63.2	15.2
63	3.9	0.9	0.0	3.9	63.2	15.2
28	-3.6	-0.4	0.1	3.6	63.2	16.6
40	-0.4	-3.6	0.1	3.6	63.2	16.6
95	-3.3	-2.1	0.5	3.5	63.2	17.1
64	-2.1	-3.3	0.5	3.4	63.2	17.6
48	3.1	2.4	0.3	3.2	63.2	18.8
6	2.4	3.1	0.3	3.2	63.2	18.8
14	3.1	0.7	0.2	3.1	63.2	19.4
54	0.7	3.1	0.2	3.1	63.2	19.4
56	0.4	3.1	0.0	3.1	63.2	19.4
12	3.1	0.4	0.0	3.1	63.2	19.4
79	2.9	1.6	0.3	3.0	63.2	20.1
80	1.6	2.9	0.3	3.0	63.2	20.1
122	-2.8	-0.4	0.4	2.9	63.2	20.8
115	-0.4	-2.8	0.4	2.9	63.2	20.8
72	-1.5	-2.6	0.3	2.7	63.2	22.4
82	-2.4	-0.4	0.3	2.4	63.2	25.3
123	-1.9	0.2	-0.6	2.3	63.2	26.5
124	0.2	-1.9	-0.6	2.3	63.2	26.5

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.
2. Stress allowables are taken at 800°F.

Table 11.1.3-8 $P_m + P_b + Q$ Stresses for PWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
44	-9.2	-31.2	6.5	33.0	105.3	2.19
58	-9.0	-29.6	6.2	31.3	105.3	2.36
21	-25.3	-9.2	2.9	25.8	105.3	3.08
37	-9.1	-25.3	2.8	25.8	105.3	3.08
49	-8.5	-23.9	2.7	24.3	105.3	3.33
9	-23.8	-8.6	2.7	24.3	105.3	3.33
112	-8.8	-24.2	2.4	24.5	105.3	3.30
111	-24.1	-8.7	2.4	24.4	105.3	3.32
107	22.9	2.0	-4.2	23.7	105.3	3.44
123	21.9	2.6	5.8	23.5	105.3	3.48
124	2.5	21.9	5.7	23.4	105.3	3.50
76	1.9	22.7	-4.1	23.4	105.3	3.50
75	22.2	1.8	-4.1	22.9	105.3	3.60
80	-8.2	-22.1	2.3	22.5	105.3	3.68
79	-22.0	-8.1	2.3	22.4	105.3	3.70
92	2.1	21.3	5.4	22.7	105.3	3.64
91	21.2	2.3	5.6	22.7	105.3	3.64
108	1.6	21.9	-4.0	22.7	105.3	3.64
32	20.7	-0.4	-1.2	21.2	105.3	3.97
31	20.3	-0.5	1.6	21.1	105.3	3.99
45	-0.5	20.0	-1.5	20.7	105.3	4.09
17	19.9	-0.3	-1.2	20.4	105.3	4.16
18	19.5	-0.5	1.5	20.2	105.3	4.21
60	-0.4	19.2	-1.4	19.9	105.3	4.29
46	-2.3	17.2	0.3	19.5	105.3	4.40
20	-13.7	-13.8	4.9	18.6	105.3	4.66
34	-13.7	-13.7	4.9	18.5	105.3	4.69
59	-2.2	16.6	0.3	18.8	105.3	4.60
6	-13.0	-12.8	4.6	17.5	105.3	5.02
48	-12.7	-13.0	4.6	17.4	105.3	5.05
30	-11.4	-13.9	4.8	17.6	105.3	4.98
7	-16.2	-4.8	-1.9	16.5	105.3	5.38
120	-4.7	-17.0	1.4	17.2	105.3	5.12
42	-6.2	-16.7	1.5	16.9	105.3	5.23
95	-16.1	-7.2	-2.4	16.8	105.3	5.27
51	-4.7	-16.1	-1.9	16.4	105.3	5.42
26	-16.5	-6.1	1.4	16.7	105.3	5.31
64	-7.2	-16.0	-2.4	16.6	105.3	5.34
16	-10.8	-13.5	4.5	16.9	105.3	5.23
23	-16.0	-4.4	-1.8	16.3	105.3	5.46

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.
2. Stress allowables are taken at 800°F.

Table 11.1.3-9 P_m Stresses for BWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
265	-0.9	0.9	0.1	1.9	58.3	29.7
10	0.7	-0.4	-0.7	1.8	58.3	31.4
277	0.9	-0.9	0.1	1.8	58.3	31.4
262	-0.8	0.7	0.1	1.5	58.3	37.9
259	-0.7	0.6	0.1	1.4	58.3	40.6
77	0.6	-0.8	0.0	1.3	58.3	43.8
194	-0.6	0.6	0.1	1.2	58.3	47.6
197	-0.5	0.5	0.1	1.1	58.3	52.0
263	-0.9	-0.9	0.1	1.0	58.3	57.3
12	-0.4	0.0	-0.4	1.0	58.3	57.3
229	-0.8	0.2	0.1	1.0	58.3	57.3
264	-0.9	0.0	0.1	1.0	58.3	57.3
276	0.5	-0.4	0.1	0.9	58.3	63.8
76	0.6	-0.3	0.1	0.9	58.3	63.8
16	-0.3	0.4	-0.3	0.9	58.3	63.8
260	-0.8	-0.8	0.1	0.9	58.3	63.8
286	0.4	-0.5	0.1	0.9	58.3	63.8
85	-0.9	-0.8	0.0	0.9	58.3	63.8
269	-0.8	-0.9	0.0	0.9	58.3	63.8
273	0.0	-0.9	0.0	0.9	58.3	63.8
211	-0.6	0.3	0.1	0.9	58.3	63.8
261	-0.8	0.0	0.1	0.9	58.3	63.8
193	-0.7	-0.8	0.1	0.8	58.3	71.9
289	-0.8	-0.5	0.1	0.8	58.3	71.9
88	0.6	-0.2	0.1	0.8	58.3	71.9
103	-0.8	-0.1	0.1	0.8	58.3	71.9
9	0.0	-0.1	-0.4	0.8	58.3	71.9
14	-0.3	0.0	-0.3	0.8	58.3	71.9
81	0.0	-0.8	0.0	0.8	58.3	71.9
258	-0.7	0.0	0.1	0.8	58.3	71.9
268	-0.7	-0.4	0.1	0.7	58.3	82.3
97	0.6	-0.1	0.1	0.7	58.3	82.3
11	0.0	-0.1	-0.4	0.7	58.3	82.3
294	-0.7	-0.1	0.2	0.7	58.3	82.3
196	-0.6	-0.7	0.1	0.7	58.3	82.3
166	0.7	0.1	0.1	0.7	58.3	82.3
280	-0.7	-0.5	0.1	0.7	58.3	82.3
84	-0.7	-0.3	0.1	0.7	58.3	82.3
246	-0.1	-0.7	0.1	0.7	58.3	82.3
199	-0.5	-0.7	0.1	0.7	58.3	82.3

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.
2. Stress allowables are taken at 800°F.

Table 11.1.3-10 $P_m + P_b$ Stresses for BWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
265	-4.6	0.8	-0.2	5.3	48.6	8.2
295	-1.6	-5.0	0.5	5.1	48.6	8.5
294	-2.2	-4.9	0.5	5.0	48.6	8.7
254	-4.8	-2.2	0.5	4.9	48.6	8.9
257	-4.5	-1.6	0.6	4.6	48.6	9.6
293	-1.9	-4.4	0.4	4.5	48.6	9.8
289	-2.3	-4.3	0.6	4.5	48.6	9.8
243	-4.3	-1.5	0.2	4.3	48.6	10.3
24	-4.3	-1.4	0.1	4.3	48.6	10.3
263	-4.0	-2.4	0.7	4.3	48.6	10.3
275	1.7	4.3	0.3	4.3	48.6	10.3
252	4.2	1.7	0.3	4.3	48.6	10.3
246	-4.1	-1.7	0.5	4.2	48.6	10.6
274	1.7	4.1	0.3	4.2	48.6	10.6
10	-0.3	-2.2	-1.9	4.2	48.6	10.6
267	-1.6	-4.1	0.2	4.2	48.6	10.6
241	4.1	1.5	0.2	4.1	48.6	10.9
288	1.8	4.1	0.4	4.1	48.6	10.9
227	0.9	4.1	0.2	4.1	48.6	10.9
75	-1.7	-4.1	0.3	4.1	48.6	10.9
22	-4.1	-1.7	0.3	4.1	48.6	10.9
208	-1.6	-4.0	0.3	4.1	48.6	10.9
32	4.0	1.6	0.3	4.0	48.6	11.2
51	4.0	1.0	0.1	4.0	48.6	11.2
237	4.0	1.8	0.3	4.0	48.6	11.2
83	-1.6	-4.0	0.3	4.0	48.6	11.2
19	4.0	1.6	0.3	4.0	48.6	11.2
62	3.9	1.4	0.4	4.0	48.6	11.2
228	0.8	3.9	0.3	4.0	48.6	11.2
21	3.9	1.7	0.3	4.0	48.6	11.2
240	3.9	1.8	0.3	4.0	48.6	11.2
74	1.6	3.9	0.3	3.9	48.6	11.5
174	3.9	1.7	0.3	3.9	48.6	11.5
238	3.9	1.4	0.2	3.9	48.6	11.5
209	-1.4	-3.9	0.3	3.9	48.6	11.5
18	3.9	1.6	0.3	3.9	48.6	11.5
266	1.7	3.9	0.3	3.9	48.6	11.5
184	-3.8	-1.6	0.3	3.9	48.6	11.5
137	1.7	3.8	0.3	3.9	48.6	11.5
49	-3.8	-1.5	0.2	3.9	48.6	11.5

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.
2. Stress allowables are taken at 800°F.

Table 11.1.3-11 $P_m + P_b + Q$ Stresses for BWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
295	-2.0	-20.5	1.3	20.6	81.0	2.93
268	-9.2	-18.9	2.2	19.4	81.0	3.18
289	-6.6	-18.8	1.6	19.0	81.0	3.26
16	16.0	5.1	5.4	18.3	81.0	3.43
139	-8.7	-17.8	2.1	18.2	81.0	3.45
30	-9.1	-17.2	2.7	18.0	81.0	3.50
14	15.7	4.6	5.2	17.8	81.0	3.55
265	-17.5	-6.3	1.6	17.7	81.0	3.58
276	-6.3	-17.5	1.3	17.7	81.0	3.58
166	-0.3	-17.4	0.9	17.5	81.0	3.63
43	-9.3	-16.5	2.7	17.4	81.0	3.66
266	-9.7	-16.4	2.2	17.0	81.0	3.76
137	-9.6	-16.2	2.1	16.8	81.0	3.82
24	-15.6	-10.2	2.9	16.8	81.0	3.82
18	-16.0	-8.6	2.6	16.8	81.0	3.82
15	13.6	4.8	-6.2	16.8	81.0	3.82
160	-5.5	-16.4	1.4	16.6	81.0	3.88
31	-15.8	-8.6	2.6	16.6	81.0	3.88
21	-16.0	-7.8	2.4	16.6	81.0	3.88
269	-7.8	-15.9	1.9	16.3	81.0	3.97
263	-16.1	-6.6	1.5	16.3	81.0	3.97
147	-6.1	-16.1	1.3	16.3	81.0	3.97
34	-15.6	-7.5	2.4	16.3	81.0	3.97
2	-1.8	14.2	-1.0	16.1	81.0	4.03
1	-1.8	14.2	-1.0	16.1	81.0	4.03
274	-7.8	-15.7	1.9	16.1	81.0	4.03
246	-15.9	-5.2	1.6	16.1	81.0	4.03
13	13.0	4.4	-6.0	16.1	81.0	4.03
37	-14.5	-9.6	2.7	15.7	81.0	4.16
238	-15.3	-8.4	1.8	15.7	81.0	4.16
241	-15.5	-6.8	1.4	15.7	81.0	4.16
145	-7.7	-15.2	1.8	15.6	81.0	4.19
243	-15.4	-6.8	1.3	15.6	81.0	4.19
4	-1.8	13.6	-0.9	15.5	81.0	4.23
3	-1.8	13.6	-0.9	15.5	81.0	4.23
111	-15.0	-8.2	1.8	15.4	81.0	4.26
267	-9.2	-14.8	1.9	15.3	81.0	4.29
277	-3.8	-14.8	1.4	15.0	81.0	4.40
140	-7.4	-14.4	1.7	14.8	81.0	4.47
27	-13.9	-8.4	2.5	14.8	81.0	4.47

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.
2. Stress allowables are taken at 800°F.

Table 11.1.3-12 Summary of Maximum Stresses for PWR and BWR Fuel Basket Weldments – Off-Normal Condition (ksi)

Component	Stress Category	Maximum Stress Intensity[1]	Node Temperature (°F)	Allowable Stress[2,3]	Margin of Safety
PWR Top Weldment	$P_m + P_b$	0.7	297	20.7	+Large
	$P_m + P_b + Q$	52.1	292	56.1	+0.08
PWR Bottom Weldment	$P_m + P_b$	0.8	179	22.5	+Large
	$P_m + P_b + Q$	20.9	175	60.0	+1.87
BWR Top Weldment	$P_m + P_b$	1.2	226	19.4	+Large
	$P_m + P_b + Q$	14.6	383	52.5	+2.60
BWR Bottom Weldment	$P_m + P_b$	1.5	265	22.5	+Large
	$P_m + P_b + Q$	36.6	203	60.0	+0.64

1. Nodal stresses are from the finite element analysis.
2. Conservatively, stress allowables are taken at 400°F for the PWR top weldment, 300°F for the PWR bottom weldment, 500°F for the BWR top weldment, and 300°F for the BWR bottom weldment.
3. P_m stress allowables are conservatively used for the P_m+P_b evaluation.

11.1.4 Failure of Instrumentation

The Universal Storage System may use a temperature-sensing system to measure the outlet air temperature at each of the four air outlets on each concrete cask. The air temperatures at the outlets may be measured and reviewed daily.

11.1.4.1 Cause of Instrumentation Failure Event

The temperature instrumentation failure event could occur as a result of instrumentation component failure, or as a result of any event that interrupted power or altered temperature sensor output.

11.1.4.2 Detection of Instrumentation Failure Event

The temperature instrumentation failure event may be identified by the lack of, or an inappropriate, reading at the temperature reader terminal. The event could also be identified by disparities between outlet temperatures in a cask or between similar casks.

11.1.4.3 Analysis of Instrumentation Failure Event

For concrete casks incorporating daily temperature-monitoring programs, the maximum time period during which an increase in outlet air temperatures may go undetected is 24 hours. The principal condition that could cause an increase in temperature is the blockage of the air inlets and/or outlets. Section 11.2.13 shows that even if all of the inlets and outlets of a single cask are blocked immediately after a temperature measurement, it would take longer than 24 hours before any component approaches its allowable temperature limit. Therefore, there would be sufficient time to identify and correct temperature instrumentation failure events prior to critical system components reaching their temperature limits. During the period of loss of instrumentation, no significant change in canister temperature will occur under normal conditions. Therefore, instrument failure would be of no consequence when the affected storage cask continues to operate in a normal storage condition.

Because the canister and the concrete cask are a large heat sink, and because there are few conditions that could result in a cooling air temperature increase, the temporary loss of remote sensing and monitoring of the outlet air temperature is not a major concern. No applicable regulatory criteria are violated by the failure of the temperature instrumentation system.

11.1.4.4 Corrective Actions

This event requires that the temperature reporting equipment be replaced, repaired or otherwise returned to operable status, or that the concrete cask inlet and outlet screens be visually inspected for blockage.

11.1.4.5 Radiological Impact

There are no radiological consequences for this event.

11.1.5 Small Release of Radioactive Particulate From the Canister Exterior

The procedures for loading the canister provide for steps to minimize exterior surface contact with contaminated spent fuel pool water, and the exterior surface of the canister is surveyed by smear at the top end to verify canister surface conditions. Design features are also employed to ensure that the canister surface is generally free of surface contamination prior to its installation in the concrete cask. The surface of the canister is free of traps that could hold contamination. The presence of contamination on the external surface of the canister is unlikely, and, therefore, no particulate release from the canister exterior surface is expected to occur in normal use.

11.1.5.1 Cause of Radioactive Particulate Release Event

In spite of precautions taken to preclude contamination of the external surface of the canister, it is possible that a portion of the canister surface may become slightly contaminated during fuel loading by the spent fuel pool water and that the contamination may go undetected. Surface contamination could become airborne and be released as a result of the air flow over the canister surface.

11.1.5.2 Detection of Radioactive Particulate Release Event

The release of small amounts of radioactive particles over time is difficult to detect. Any release is likely to be too low to be detected by any of the normally employed long-term radiation dose monitoring methods (such as TLDs). It is possible that a suspected release could be verified by a smear survey of the air outlets.

11.1.5.3 Analysis of Radioactive Particulate Release Event

A calculation is made to determine the level of surface contamination that if released would result in a dose of one tenth of one (0.1) mrem at a minimum distance of 100 meters from a design basis storage cask. ISFSI-specific allowable dose rates and surface contamination limits will be calculated on a site specific basis to conform to 10 CFR 72. The method for determining the residual contamination limit is based on the plume dispersion calculations presented in U.S. NRC Regulatory Guides 1.109 [9] and 1.145 [13] and is highly conservative. The calculation shows that a residual contamination of approximately 1.57×10^5 dpm/100 cm^2 β-γ and 5.24×10^2 dpm/100 cm^2 α activity, on the surface of the design basis canister, is required to yield a dose of one tenth of one (0.1) mrem at the minimum distance of 100 meters. The canister surface area is inversely proportional to the allowable surface contamination. The design basis cask is, therefore, the Class 3 PWR cask, which has the largest canister surface area at 3.06×10^5 cm^2.

The above analysis demonstrates that the off-site radiological consequences from the release of canister surface contamination is negligible, and all applicable regulatory criteria can be met for an ISFSI array.

11.1.5.4 Corrective Actions

No corrective action is required since the radiological consequence is negligible.

11.1.5.5 Radiological Impact

As shown above, the potential off-site radiological impact due to the release of canister surface contamination is negligible.

11.1.6 Off-Normal Events Evaluation for Site Specific Spent Fuel

This section presents the off-normal events evaluation of spent fuel assemblies or configurations, which are unique to specific reactor sites. These site specific fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blankets and variable enrichment assemblies, fuel with burnup that exceeds the design basis, and fuel that is classified as damaged.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly of the same type (PWR or BWR), or are shown to be acceptable contents, by specific evaluation of the configuration.

11.1.6.1 Off-Normal Events Evaluation for Maine Yankee Site Specific Spent Fuel

Maine Yankee site specific fuels are described in Section 1.3.2.1. A thermal evaluation has been performed for Maine Yankee site specific fuels that exceed the design basis burnup as shown in Section 4.5.1.2. As shown in that section, loading of fuel with a burnup between 45,000 and 50,000 MWD/MTU is subject to preferential loading in designated basket positions in the Transportable Storage Canister.

With preferential loading, the design basis total heat load of the canister is not changed. Consequently, the thermal performance for the Maine Yankee site specific fuels is bounded by the design basis PWR fuels. Therefore, no further evaluation is required for the off-normal thermal events (severe ambient temperature conditions and blockage of half of the air inlets) as shown in Sections 11.1.1 and 11.1.2. In Section 3.6.1.1, the total weight of the canister contents for Maine Yankee site specific fuels is shown to be bounded by the PWR design basis fuels. Therefore, the evaluation for the off-normal canister handling load in Section 11.1.3 bounds the canister configuration loaded with Maine Yankee fuels.

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11.2 Accidents and Natural Phenomena

This section presents the results of analyses of the design basis and hypothetical accident conditions evaluated for the Universal Storage System. In addition to design basis accidents, this section addresses very low probability events, including natural phenomena, that might occur over the lifetime of the ISFSI, or hypothetical events that are postulated to occur because their consequences may result in the maximum potential impact on the immediate environment.

The Universal Storage System includes Transportable Storage Canisters and Vertical Concrete Casks of five different lengths to accommodate three classes of PWR fuel or two classes of BWR fuel. In the accident analyses of this section, the bounding cask parameters (such as weight and center of gravity) are conservatively used, as appropriate, to determine the cask's capability to withstand the effects of the accidents.

The results of analyses show that no credible potential accident exists that will result in a dose of ≥ 5 rem beyond the postulated controlled area. The Universal Storage System is demonstrated to have a substantial design margin of safety and to provide protection to the public and to occupational personnel during storage of spent nuclear fuel.

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11.2.1 Accident Pressurization

Accident pressurization is a hypothetical event that assumes the failure of all of the fuel rods contained within the Transportable Storage Canister (canister). No storage conditions are expected to lead to the rupture of all of the fuel rods.

Results of analysis of this event demonstrate that the canister is not significantly affected by the increase in internal pressure that results from the hypothetical rupture of all PWR or BWR fuel rods contained within the canister. Positive margins of safety exist throughout the canister.

11.2.1.1 Cause of Pressurization

The hypothetical failure of all of the fuel rods in a canister would release the fission and fill gases to the interior of the canister, resulting in the pressurization of the canister.

11.2.1.2 Detection of Accident Pressurization

The rupture of fuel rods within the canister is unlikely to be detected by any measurements or inspections that could be undertaken from the exterior of the canister or the concrete cask.

11.2.1.3 Analysis of Accident Pressurization

Analysis of this accident involves evaluation of the maximum canister internal pressure and the canister stress due to the maximum internal pressure. These evaluations are provided below.

Maximum Canister Accident Condition Internal Pressure

The analysis requires the calculation of the free volume of the canister, calculation of the releasable quantity of fill and fission gas in the fuel assemblies, BPRA gases, and the subsequent calculation of the pressure in the canister if these gases are added to the backfill helium pressure (initially at 1 atm) already present in the canister (Section 4.4.5). Canister pressures are determined for two accident scenarios, 100 percent fuel failure and a maximum temperature accident. The maximum temperature accident includes the fire accident and full vent blockage. While no design basis event results in a 100 percent fuel failure condition, the pressures from this condition are presented to form a complete licensing basis. The method employed in either of the accident analyses is identical to that employed in the normal condition evaluation of Section 4.4.5.

For the maximum temperature accident condition, the gas quantities are combined with the accident average gas temperatures of 505°F (PWR) and 465°F (BWR) to calculate conservative system pressures. Maximum pressures under the fire accident conditions are 6.14 psig (PWR) and 5.11 psig (BWR).

Canister pressures under the 100 percent fuel failure assumption are 59.2 psig (PWR) and 37.4 psig (BWR). Assemblies producing the maximum pressures are identical to those in the normal condition evaluation, i.e., B&W 17×17 Mark C in UMS® canister Class 2 for PWR assemblies and GE 7×7 (49 fuel rod) assembly in canister class 5 for BWR assemblies. Similar pressures result from the Westinghouse 17×17 standard fuel assembly in UMS® canister Class 1 and the GE 9×9 (79 fuel rod) assembly in canister Class 5.

Maximum Canister Stress Due to Internal Pressure

The stresses that result in the canister due to the internal pressure are evaluated using the ANSYS finite element model that envelops both PWR and BWR configurations as described in Section 3.4.4. The pressure used for the model is 65 psig, which bounds the results of 59.2 and 37.4 psig for the PWR and BWR configurations, respectively.

The resulting maximum canister stresses for accident pressure loads are summarized in Tables 11.2.1-1 and 11.2.1-2 for primary membrane and primary membrane plus bending stresses, respectively.

The resulting maximum canister stresses and margins of safety for combined normal handling (Tables 3.4.4.1-4 and 3.4.4.1-5) and maximum accident internal pressure (65 psig) are summarized in Tables 11.2.1-3 and 11.2.1-4 for primary membrane and primary membrane plus bending stresses, respectively.

The sectional stresses shown in Tables 11.2.1-1 through 11.2.1-4 at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations of the stress sections are shown in Figure 3.4.4.1-4.

All margins of safety are positive. Consequently, there is no adverse consequence to the canister as a result of the combined normal handling and maximum accident internal pressure (65 psig).

11.2.1.4 Corrective Actions

No recovery or corrective actions are required for this hypothetical accident.

11.2.1.5 Radiological Impact

There are no dose consequences due to this accident.

Table 11.2.1-1 Canister Accident Internal Pressure (65 psig) Only Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0.44	2.49	6.33	-0.17	-0.08	-0.91	6.19
2	4.24	-5.27	-4.12	0.71	-0.09	-0.90	9.71
3	-0.77	-8.07	1.82	0.68	0.16	1.82	10.91
4	-0.01	3.43	1.69	-0.30	0.00	0.00	3.49
5	-0.01	3.40	1.70	-0.30	0.00	0.00	3.45
6	0.00	3.40	1.70	-0.30	0.00	0.00	3.45
7	-0.01	3.40	1.70	-0.30	0.00	0.00	3.46
8	-0.01	2.28	1.69	-0.20	0.00	-0.04	2.33
9	0.16	0.90	1.25	-0.07	0.02	0.15	1.14
10	-0.55	0.60	0.84	-0.09	0.00	-0.15	1.42
11	0.71	0.41	-0.11	0.00	0.00	0.08	0.83
12	-0.29	0.17	-0.83	0.00	0.00	-0.27	1.11
13	-0.15	0.45	0.77	-0.06	0.03	0.06	0.94
14	1.05	1.05	-0.06	0.00	0.45	-0.07	1.44
15	-0.12	-0.12	-0.04	0.00	-0.01	0.00	0.08
16	0.09	0.09	0.00	0.00	-0.01	0.00	0.10

(1) See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 11.2.1-2 Canister Accident Internal Pressure (65 psig) Only Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	4.85	0.53	15.30	0.22	-0.09	-0.06	14.79
2	2.05	-13.36	-29.48	1.27	-0.15	-2.06	31.90
3	-3.08	2.85	41.20	-0.38	0.17	2.29	44.54
4	-0.02	3.45	1.64	-0.30	0.00	0.00	3.52
5	-0.02	3.44	1.70	-0.30	0.00	0.00	3.51
6	-0.02	3.44	1.70	-0.30	0.00	0.00	3.51
7	-0.02	3.44	1.70	-0.30	0.00	0.00	3.51
8	-0.03	2.31	1.89	-0.20	0.00	-0.04	2.37
9	0.18	1.32	2.67	-0.10	0.03	0.37	2.61
10	-0.41	1.34	3.21	-0.14	0.00	0.11	3.64
11	0.57	-0.13	-1.80	0.00	0.00	0.16	2.39
12	-0.78	-0.17	-1.52	0.00	0.00	-0.46	1.57
13	-1.11	0.07	0.32	-0.09	0.04	0.12	1.46
14	21.95	21.97	0.56	0.01	0.41	-0.09	21.43
15	-1.46	-1.46	-0.08	0.00	-0.01	0.00	1.38
16	0.75	0.75	0.02	0.00	-0.01	0.00	0.73

(1) See Figure 3.4.4.1-4 for definition of locations of stress sections.

Table 11.2.1-3 Canister Normal Handling plus Accident Internal Pressure (65 psig) Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.55	3.19	8.13	-0.22	-0.10	-1.17	7.95	40.08	4.0
2	5.41	-6.96	-5.28	0.92	-0.11	-1.17	12.63	40.08	2.2
3	-0.97	-10.70	2.35	0.90	0.20	2.30	14.34	40.08	1.8
4	-0.01	3.44	2.20	0.30	0.00	0.00	3.50	38.77	10.1
5	-0.01	3.40	2.18	0.30	0.00	0.00	3.46	35.86	9.4
6	-0.01	3.40	2.13	0.30	-0.01	0.00	3.46	35.55	9.3
7	-0.01	3.40	2.04	0.30	-0.01	0.00	3.46	38.23	10.0
8	0.01	2.24	2.79	-0.20	-0.07	-0.04	2.80	40.08	13.3
9	0.18	1.27	2.68	-0.10	-0.13	0.19	2.54	40.08	14.8
10	-0.78	0.94	2.52	-0.16	-0.22	-0.08	3.36	40.08	10.9
11	0.13	1.12	0.79	-0.09	-0.11	-0.44	1.26	40.08	30.8
12	-0.32	0.19	-1.12	-0.10	-0.23	-0.42	1.57	40.08	24.5
13	0.12	1.40	0.43	-0.22	0.00	-0.47	1.68	40.08	22.9
14	1.35	1.35	-0.03	0.00	0.60	-0.09	1.84	40.08	20.8
15	-0.13	-0.13	-0.06	0.00	-0.01	0.00	0.07	40.08	547.4
16	0.10	0.11	-0.02	0.00	-0.01	0.00	0.13	40.08	299.0

(1) See Figure 3.4.4.1-4 for definition of locations of stress sections.
(2) ASME Service Level D is used for material allowable stress.

Table 11.2.1-4 Canister Normal Handling plus Accident Internal Pressure (65 psig) Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	6.17	0.48	19.67	0.29	-0.11	-0.08	19.22	60.12	2.1
2	2.62	-17.34	-37.86	1.65	-0.19	-2.67	40.97	60.12	0.5
3	-3.93	3.38	53.13	-0.46	0.22	2.91	57.38	60.12	0.1
4	-0.03	3.52	2.14	0.31	0.00	-0.01	3.60	58.16	15.2
5	-0.03	3.57	2.23	0.32	-0.01	0.00	3.65	53.79	13.8
6	-0.03	3.62	2.20	0.32	-0.01	0.00	3.70	53.32	13.4
7	-0.02	3.60	2.11	0.31	0.00	0.00	3.67	57.35	14.6
8	-0.01	2.39	3.04	-0.22	-0.08	-0.04	3.07	60.12	18.6
9	0.12	1.68	4.32	-0.10	-0.18	0.37	4.28	60.12	13.0
10	-0.56	1.48	4.32	-0.16	-0.29	0.12	4.93	60.12	11.2
11	0.01	1.63	2.55	-0.13	-0.19	-0.91	3.16	60.12	18.0
12	-0.61	-0.09	-1.90	-0.11	-0.27	-0.61	2.13	60.12	27.2
13	0.09	1.30	1.27	-0.09	-0.12	-0.85	2.10	60.12	27.7
14	28.61	28.64	0.79	0.01	0.55	-0.12	27.88	60.12	1.2
15	-1.53	-1.53	-0.09	0.00	-0.01	0.00	1.44	60.12	40.7
16	0.75	0.74	0.00	0.00	-0.01	0.00	0.75	60.12	79.2

(1) See Figure 3.4.4.1-4 for definition of locations of stress sections.

(2) ASME Service Level D is used for material allowable stress.

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11.2.2 Failure of All Fuel Rods With a Ground Level Breach of the Canister

As no mechanistic failure of the canister occurs, there is no credible leakage of radioactive material from the canister. Therefore, this potential accident condition is not evaluated.

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11.2.3 Fresh Fuel Loading in the Canister

This section evaluates the effects of an inadvertent loading of up to 24 fresh, unburned PWR fuel assemblies or up to 56 fresh, unburned BWR fuel assemblies in a canister. There are no adverse effects on the canister due to this event since the criticality control features of the Universal Storage System ensure that the k_{eff} of the fuel is less than 0.95 for all loading conditions of fresh fuel.

11.2.3.1 Cause of Fresh Fuel Loading

The cause of this event is operator and/or procedural error. In-plant operational procedures and engineering and quality control programs are expected to preclude occurrence of this event. Nonetheless, it is evaluated here to demonstrate the adequacy of the canister design for accommodating fresh fuel without a resulting criticality event.

11.2.3.2 Detection of Fresh Fuel Loading

This accident is expected to be identified immediately by observation of the condition of the fuel installed in the canister or by a review of the fuel handling records.

11.2.3.3 Analysis of Fresh Fuel Loading

The criticality analysis presented in Chapter 6 assumes the loading of up to 24 design basis PWR or up to 56 design basis BWR fuel assemblies having no burnup. The maximum k_{eff} for the accident conditions remains below the upper safety limit.

The criticality control features of the Transportable Storage Canister and the basket ensure that the k_{eff} of the fuel is less than 0.95 for all loading conditions of fresh fuel. Therefore, there is no adverse impact on the Universal Storage System due to this event.

11.2.3.4 Corrective Actions

This event requires that the canister be unloaded when the incorrect fuel loading is identified. The cause for the error should be identified and procedural actions implemented to preclude recurrence.

11.2.3.5 Radiological Impact

There are no dose implications due to this event.

11.2.4 24-Inch Drop of Vertical Concrete Cask

This analysis evaluates a loaded Vertical Concrete Cask for a 24-inch drop onto a concrete storage pad. The cask containing the Transportable Storage Canister loaded with Class 5 BWR fuel is identified as the heaviest cask, and is conservatively used in the analysis as the bounding case. The results of the evaluation show that neither the concrete cask nor the Transportable Storage Canister experience significant adverse effects due to the 24-inch drop accident.

11.2.4.1 Cause of 24-Inch Cask Drop

The Vertical Concrete Cask may be lifted and moved using either an air pad system, which lifts the concrete cask from the bottom, or a mobile lifting frame, which lifts the concrete casks using lifting lugs in the top of the cask.

Using the air pad system, the concrete cask, containing a loaded canister, must be raised approximately 4 inches to enable installation of the inflatable air pads beneath it. The air pads use pressurized air to allow the cask to be moved across the surfaces of the transporter and the ISFSI pad to the designated position. The cask is raised using hydraulic jacks installed at jack-points in the cask's air inlets. The failure of one or more of the jacks or of the air pad system could result in a drop of the cask.

The concrete cask may be lifted and moved by a mobile lifting frame, which may be self-propelled or towed. The lifting frame uses hydraulic power to raise the cask approximately 24 inches using a lifting attachment that connects to the four cask lifting lugs. The failure of one or more of the lifting lugs, or the failure of the hydraulic pistons, could result in a drop of the cask.

11.2.4.2 Detection of 24-Inch Cask Drop

This event will be detected by the operators as it occurs.

11.2.4.3 Analysis of 24-Inch Cask Drop

A bottom end impact is assumed to occur normal to the concrete cask bottom surface, transmitting the maximum load to the concrete cask and the canister. The energy absorption is computed as the product of the compressive force acting on the concrete cask and its displacement. Conservatively assuming that the storage surface impacted is an infinitely rigid surface, the concrete cask body will crush until the impact energy is absorbed.

A compressive strength of 4,000 psi is used for the cask concrete. The evaluation conservatively ignores any energy absorption by the internal friction of the aggregate as crushing occurs.

The canister rests upon a base weldment designed to allow cooling of the canister. Following the initial impact, the inlet system will partially collapse, providing an energy absorption mechanism that somewhat reduces the deceleration force on the canister.

Evaluation of the Concrete Cask

In the 24-inch bottom drop of the concrete cask, the cylindrical portion of the concrete is in contact with the steel bottom plate that is a part of the base weldment. The plate is assumed to be part of an infinitely rigid storage pad. No credit is taken for the crush properties of the storage pad or the underlying soil layer. Therefore, energy absorbed by the crushing of the cylindrical concrete region of the concrete cask equals the product of the compressive strength of the concrete, the crush depth of the concrete, and the projected area of the concrete cylinder. Crushing of the concrete continues until the energy absorbed equals the potential energy of the cask at the initial drop height. The canister is not rigidly attached to the concrete cask, so it is not considered to contribute to the concrete crushing. The energy balance equation is:

$$w(h+\delta) = P_o A \delta,$$

where:

h = 24 in., the drop height,

δ = the crush depth of the concrete cask,

P_o = 4000 psi, the compressive strength of the concrete,

A = $\pi(R_1^2 - R_2^2)$ = 7,904 in^2, the projected area of the concrete shield wall,

w = 190,000 lbs (bounding concrete plus rebar)

It is assumed that the maximum force that can be exerted on the concrete cask is the compressive strength of the concrete multiplied by the area of the concrete being crushed. The concrete cask's steel shell will not experience any significant damage during a 24-inch drop. Therefore, its functionality will not be impaired due to the drop.

The crush distance computed from the energy balance equation is:

$$\delta = \frac{hw}{P_o A - w} = \frac{(24)(190{,}000)}{(4000)(7{,}904) - (190{,}000)} = 0.145 \text{ inch}$$

where, w = 190,000 lbs (the highest bounding weight is used to obtain the maximum deformation)

The resultant inlet deformation is 0.145 inch.

Evaluation of the Canister for a 24-inch Bottom End Drop

Upon a bottom end impact of the concrete cask, the canister produces a force on the base weldment located near the bottom of the cask (see Figure 11.2.4-1). The ring above the air inlets is expected to yield. To determine the resulting acceleration of the canister and deformation of the pedestal, a LS-DYNA analysis is used.

A half-symmetry model of the base weldment is built using the ANSYS preprocessor (see Figure 11.2.4-2). The model is constructed of 8-node brick and 4-node shell elements. Symmetry conditions are applied along the plane of symmetry (X-Z plane). Lumped mass elements located in the canister bottom plate represent the loaded canister. The impact plane is represented as a rigid plane, which is considered conservative, since the energy absorption due to the impact plane is neglected (infinitely rigid). To determine the maximum acceleration and deformations, impact analyses are solved using LS-DYNA program.

The weldment ring, weldment plate, and the inner cone (see Figure 11.2.4-1) materials are modeled using LS-DYNA's piece-wise linear plasticity model. This material model accepts stress–strain curves for different strain rates. These stress-strain curves were obtained from the Atlas of Stress-Strain Curves [44] and are shown in Figure 11.2.4-3. To ensure that maximum deformations and accelerations are determined, two analyses are performed. One analysis, which uses the static stress-strain curve, envelopes the maximum deformation of the pedestal. The second analysis employs the multiple stress-strain curves to account for different strain rates.

The maximum accelerations of the canister during the 24-inch bottom end impact are 45.0g and 44.5g for the variable strain rate material model and the static stress-strain curve, respectively. The resulting acceleration time histories of the bottom canister plate, which correspond to a filter frequency of 200 Hz, are shown in Figure 11.2.4-4 for the analysis using the static stress-strain curve and Figure 11.2.4-5 for the analysis corresponding to the series of stress-strain curves at different strain rates. These time histories indicate that the maximum accelerations do not occur at the beginning where the strain rate is maximum, but rather, at a time where the strain rate has a marginal effect on the accelerations. Therefore, the use of the multiple strain rate material model is considered to bound the accelerations imposed on the canister, since it considers the effect of strain rate on the stress-strain curves.

The filter frequency used in the LS-DYNA evaluation is determined by performing two modal analyses of a quarter symmetry model of the base weldment. Symmetry boundary conditions are applied on the planes of symmetry of the model for both analyses. The second analysis considers a boundary condition that is the center node of the base weldment bottom plate, restrained in the vertical direction. These analyses result in a modal frequency of 173 Hz and 188 Hz, respectively. Therefore, a filter frequency of 200 Hz is selected.

Results of the LS-DYNA analysis show that the maximum deformation of the base weldment is about 1 inch. This deformation is small when compared to the 12-inch height of the air inlet. Therefore, a 24-inch drop of the concrete cask does not result in a blockage of the air inlets.

The dynamic response of the canister and basket on impact is amplified by the most flexible components of the system. In the case of the canister and basket, the basket support disk bounds this response. To account for the transient response of the support disk, a dynamic load factor (DLF) for the support disk is computed for the inertia loading developed during the deceleration of the canister bottom plate. The DLF is determined using quarter symmetry models of the PWR and BWR disks as shown in Figures 11.2.4-6 and 11.2.4-7, respectively. These models are generated using ANSYS, Revision 5.5.

To support the disks in the models, restraints are applied at the basket tie-rod locations. For each tie-rod location, a single node is restrained in the vertical direction allowing the support disks to vibrate freely when the accelerations are applied at the tie rod locations. A transient analysis using ANSYS, Revision 5.5 is performed which uses the acceleration time histories computed from the LS-DYNA analyses. The time history corresponding to the stress–strain curves at different strain rates is used. This case is considered bounding since the maximum acceleration occurs when the rate dependent stress-strain curves are used.

The DLF is determined to be the maximum deflection of the disk (which occurs at the center of the disk) divided by the static displacement (The static analysis used the maximum acceleration determined from the LS-DYNA analysis). The DLF for the PWR and the BWR are determined to be 1.01 and 1.29, respectively.

Therefore, multiplying the calculated accelerations by the DLF's results in effective accelerations of 45.5g and 57.4g for the PWR and BWR canisters, respectively. These values are enveloped by the 60g acceleration employed in the stress evaluation of the end impact of the canister and support disks. These accelerations are considered to be bounding since they incorporate the effect of the strain rate on the plastic behavior of the pedestal and ignore any energy absorption by the impact plane.

Canister Stress Evaluation

The Transportable Storage Canister stress evaluation for the concrete cask 24-inch bottom end drop accident is performed using a load of 60g. This evaluation bounds the 57.4g load that is calculated for the 24-inch bottom end drop event determined above. This canister evaluation is performed using the ANSYS finite element program. The canister finite element model is shown in Figure 11.2.4-8. The construction and details of the finite element model are described in Section 3.4.4.1.1. Stress evaluations are performed with and without an internal pressure of 15 psig.

The principal components of the canister are the canister shell, including the bottom plate, the fuel basket, the shield lid, and the structural lid. The geometry and materials of construction of the canister, baskets, and lids are described in Section 1.2. The structural design criteria for the canister are contained in the ASME Code, Section III, Subsection NB. This analysis shows that the structural components of the canister (shell, bottom plate, and structural lid) satisfy the allowable stress intensity limits.

The results of the bounding canister analysis for the 60g bottom end impact loading are presented in Tables 11.2.4-1 through 11.2.4-4. These results are for the load case that includes a canister internal pressure of 15 psig, since that case results in the minimum margin of safety.

The minimum margin of safety at each section of the canister is presented by denoting the circumferential angle at which the minimum margin of safety occurs. A cross-section of the canister showing the section locations is presented in Figure 11.2.4-9. Stresses are evaluated at 9° increments around the circumference of the canister for each of the locations shown. The minimum margin of safety is denoted by an angular location at each section.

For the canister to structural lid weld (Section 13, Figure 11.2.4-9), base metal properties are used to define the allowable stress limits since the tensile properties of the weld filler metal are greater than those of the base metal. The allowable stress at Section 13 is multiplied by a stress reduction factor of 0.8 in accordance with NRC Interim Staff Guidance (ISG) No. 15.

The allowable stresses presented in Tables 11.2.4-1 through 11.2.4-4, and in Tables 11.2.4-6 and 11.2.4-7, are for Type 304L stainless steel. Because the shield lid is constructed of Type 304 stainless steel, which possesses higher allowable stresses, a conservative evaluation results. The allowable stresses are evaluated at 380°F. A review of the thermal analyses shows that the maximum temperature of the canister is 351°F (Table 4.1-4) for PWR fuel and 376°F (Table 4.1-5) for BWR fuel, which occurs in the center portion of the canister wall (Sections 5 and 6).

Canister Buckling Evaluation

Code Case N-284-1 of the ASME Boiler and Pressure Vessel Code is used to analyze the canister for the 60g bottom end impact. The evaluation requirements of Regulatory Guide 7.6, Paragraph C.5, are shown to be satisfied by the results of the buckling interaction equation calculations.

The internal stress field that controls the buckling of a cylindrical shell consists of the longitudinal (axial) membrane, circumferential (hoop) membrane, and in-plane shear stresses. These stresses may exist singly or in combination, depending on the applied loading. The buckling evaluation is performed without the internal 15 psig pressure, since this results in the minimum margin of safety.

The primary membrane stress results for the 60g bottom impact with no internal pressure are presented in Table 11.2.4-4.

The stress results from the ANSYS analyses are screened for the maximum values of the longitudinal compression, circumferential compression, and in-plane shear stresses for the 60g bottom end impact. For each loading case, the largest of each of the three stress components, regardless of location within the canister shell are combined.

The maximum stress components used in the evaluation and the resulting buckling interaction equation ratios are provided in Table 11.2.4-5. The results show that all interaction equation ratios are less than 1.0. Therefore, the buckling criteria of Code Case N-284-1 are satisfied, demonstrating that buckling of the canister does not occur.

Basket Stress Evaluation

Stresses in the support disks and weldments are calculated by applying the accident loads to the ANSYS models described in Sections 3.4.4.1.8 and 3.4.4.1.9. An inertial load of 60g is conservatively applied to the support disks and weldments in the axial (out of plane) direction. To evaluate the most critical regions of the support disks, a series of cross sections are considered. The locations of these sections on the PWR and BWR support disks are shown in Figures 3.4.4.1-7, 3.4.4.1-8 and Figures 3.4.4.1-13 through 3.4.4.1-16. The stress evaluations for the support disk and weldments are performed according to ASME Code, Section III, Subsection NG. For accident conditions, Level D allowable stresses are used: the allowable stress is $0.7S_u$ and S_u for P_m and P_m+P_b stress categories, respectively. The stress evaluation results are presented in Tables 11.2.4-6 and 11.2.4-7 for the PWR and BWR support disks, respectively. The tables list the 40 highest P_m+P_b stress intensities. The minimum margins of safety are +1.90 and +0.60 for PWR and BWR disks, respectively. The stress results for the PWR and BWR weldments are shown in Table 11.2.4-3. The minimum margin of safety is +1.31 and +0.26 for the PWR and BWR weldments, respectively. Note that the P_m stresses for the disks and weldments are essentially zero, since there are no loads in the plane of the support disk or weldment for a bottom end impact.

Fuel Basket Tie Rod Evaluation

The tie rods serve basket assembly purposes and are not part of the load path for the conditions evaluated. The tie rods are loaded during basket assembly by a 50 ± 10 ft-lbs torque applied to the tie rod end nut. The tensile pre-load on the tie rod, P_B, is [41]:

$$T = P_B\,(0.159\,L + 1.156\,\mu\,d)$$

where:

T = 60 ft-lb
L= 1/8
μ = 0.15
d = 1.625 in.

Solving for P_B:

$$P_B = 2{,}387 \text{ lbs. per rod}$$

The maximum tensile stress in the tie rod occurs while the basket is being lifted for installation in the canister. The BWR basket configuration is limiting because it has six tie rods, compared to eight tie rods in the PWR basket, and weighs more than the PWR basket. The load on each BWR basket tie rod is:

$$P = 2{,}387 + \frac{1.1 \times 17{,}551}{6} = 5{,}605 \text{ lbs. use } 6{,}000 \text{ lbs.}$$

where the weight of the BWR basket is 17,551 pounds.

The maximum tensile stress, S, at room temperature (70°F) is:

$$S = \frac{6{,}000}{\pi \times 0.25 \times 1.625^2} = 2{,}893 \text{ psi}$$

Therefore, the margin of safety is:

$$MS = \frac{20{,}000}{2{,}893} - 1 = +\text{Large}$$

This result bounds that for the PWR basket configuration. The tie rod is not loaded in drop events; therefore, no additional analysis of the tie rod is required.

PWR and BWR Tie Rod Spacer Analysis

The PWR and BWR basket support disks and heat transfer disks are connected by tie rods (8 for PWR and 6 for BWR) and located by spacers to maintain the disk spacing. The PWR and BWR spacers are constructed from ASME SA479 Type 304 stainless steel or ASME SA312 Type 304 stainless steel. The difference in using the two materials is the cross-sectional area of the spacers.

The geometry of the spacers is:

For SA479 stainless steel:

Spacer:	Outside Diameter	= 3.00 in.
	Inside Diameter	= 1.75 in.
Split Spacer:	Outside Diameter	= 2.50 in. (Machined down section)
	Inside Diameter	= 1.75 in.
	Outside Diameter	= 3.00 in.

For the full spacer, the cross-section area is 4.66 inches^2, and for the split spacer, the cross-section area is 2.5 inches^2.

For SA312 stainless steel:

Spacer:	Outside Diameter	= 2.875 in.
	Inside Diameter	= 1.771 in.
Split Spacer:	Outside Diameter	= 2.50 in. (Machined down section)
	Inside Diameter	= 1.771 in.
	Outside Diameter	= 2.875 in.

For the full spacer, the cross-section area is 4.03 inches^2, and for the split spacer, the cross-section area is 2.45 inches^2.

During a 24-inch drop, the weight of the support disks, top weldment, heat transfer disks, spacers, and end nuts are supported by the spacers on the tie rods. A conservative deceleration of 60g is applied to the spacers. The bounding spacer load occurs at the bottom weldment of the BWR basket. The bounding split-spacer load occurs at the 10th support disk (from bottom of the basket) of the BWR basket.

The applied load on the BWR bottom spacer is 126,000 lbs.

$$P = 60(P_S) + P_T = 125{,}147 \text{ lbs. use } 126{,}000 \text{ lbs.}$$

where:

P_T = 2387 lbs	torque pre-load
P_s = 2046 lbs	load on the spacer due to basket structure above the spacer location

$$P_s = \frac{17{,}551 - 623 - 4651}{6} = 2{,}046 \text{ lbs}$$

where:

17,551	lb.	BWR basket weight
623	lb.	BWR bottom weldment weight
4,651	lb.	BWR fuel tube weight

The applied load on the BWR split spacer is 102,000 lbs.

$$P = 60(P_S) + P_T = 101{,}747 \text{ lbs. use } 102{,}000 \text{ lbs.}$$

where:

P_T = 2387 lbs torque pre-load
P_s = 1656 lbs load on the spacer due to basket structure above the spacer location

$$P_s = \frac{17{,}551 - 623 - 4{,}651 - 10 \times 204 - 60 \times 5}{6} = 1{,}656 \text{ lbs}$$

17,551	lbs	BWR basket weight
623	lbs	BWR bottom weldment weight
4,651	lbs	BWR fuel tube weight
204	lbs	BWR support disk weight (Qty = 10)
5	lbs	BWR full spacer weight (Qty = 60)

The margins of safety for the spacers are:

	Applied Load (lbs)	Cross-sectional area (in^2)	Stress (psi)	Temperature (°F)	Allowable Stress (psi)	Margin of Safety
Spacer						
SA479	126,000	4.66	27,039	250	47,950	0.77
SA312	126,000	4.03	31,266	250	47,950	0.53
Split Spacer						
SA479	102,000	2.50	40,800	350	45,640	0.12
SA312	102,000	2.45	41,633	350	45,640	0.10

The temperatures used bound the analysis locations for all storage conditions. The actual temperatures at these locations for storage for the BWR spacer at the bottom weldment are 118°F (minimum bottom weldment temperature), and 329°F (minimum temperature of 10th support disk) for the split spacer. The 10th support disk is counted from bottom weldment.

Fuel Tube Analysis

During the postulated 24-inch end drop of the concrete cask, fuel assemblies are supported by the canister bottom plate. The fuel assembly weight is not carried by the fuel tubes in the end drop. Therefore, evaluation of the fuel tube is performed considering the weight of the fuel tube, the canister deceleration and the minimum fuel tube cross-section. The minimum cross-section is located at the contact point of the fuel tube with the basket bottom weldment. The PWR fuel tube analysis is bounding because its weight (153 pounds/tube) is approximately twice that of the BWR fuel tube (83 pounds/tube). The minimum cross-section area of the PWR fuel tube is:

$$A = \text{(thickness)(mean perimeter)}$$

$$A = (0.048\text{ in.})(8.80\text{ in.} + 0.048\text{ in.})(4) = 1.69\text{ in}^2$$

The maximum compressive and bearing stress in the fuel tube is:

$$S_b = \frac{(60g)(153\text{ lbs})}{1.69\text{ in}^2} = 5{,}432\text{ psi}$$

The Type 304 stainless steel yield strength is 17,300 psi at a conservatively high temperature of 750°F. The margin of safety is:

$$MS = \frac{S_y}{S_b} - 1 = \frac{17{,}300\text{ psi}}{5{,}432\text{ psi}} - 1 = +2.18\text{ at }750°F$$

Summary of Results

Evaluation of the UMS cask and canister during a 24-inch drop accident shows that the resulting maximum acceleration of the canister is 57.4g. The acceleration determined for the canister during the 24-inch drop is less than its design allowable g-load and, therefore, is considered bounded. This accident condition does not lead to a reduction in the cask's shielding effectiveness. The base weldment, which includes the air inlets, is crushed approximately 1-inch as the result of the 24-inch drop. The effect of the reduction of the inlet area by the drop is to reduce cooling airflow. This condition is bounded by the consequences of the loss of one-half of the air inlets evaluated in Section 11.1.2.

11.2.4.4 Corrective Actions

Following the accident event, perform the required Response Surveillance in accordance with Section A 5.4 of the Technical Specifications. Corrective actions shall be taken in accordance with the surveillance requirements to return the affected system to a safe operating condition, as applicable to the affected component(s).

11.2.4.5 Radiological Impact

There are no radiological consequences for this accident.

Figure 11.2.4-1 Concrete Cask Base Weldment

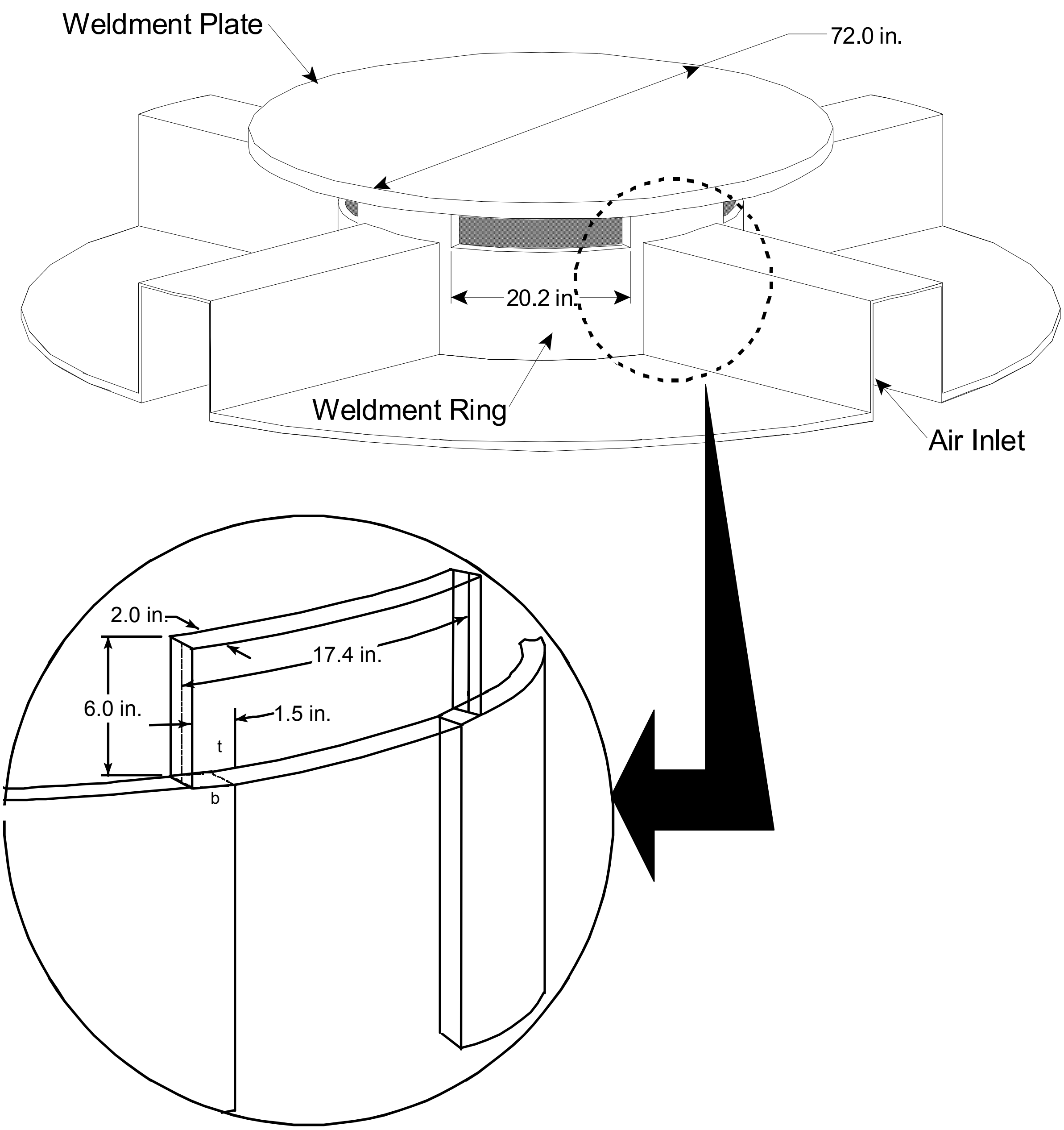


Figure 11.2.4-2 Concrete Cask Base Weldment Finite Element Model

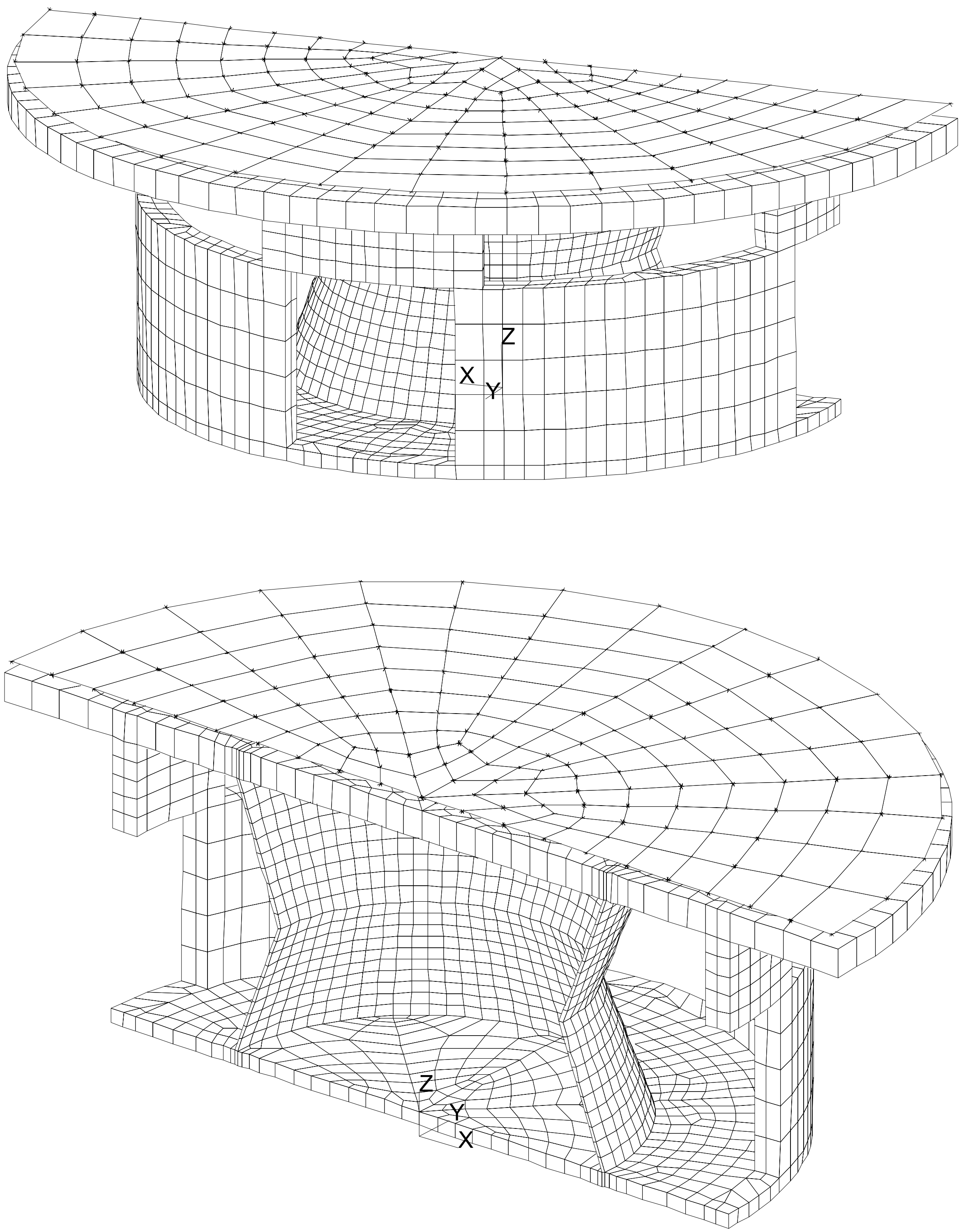


Figure 11.2.4-3 Strain Rate Dependent Stress-Strain Curves for Concrete Cask Base Weldment Structural Steel

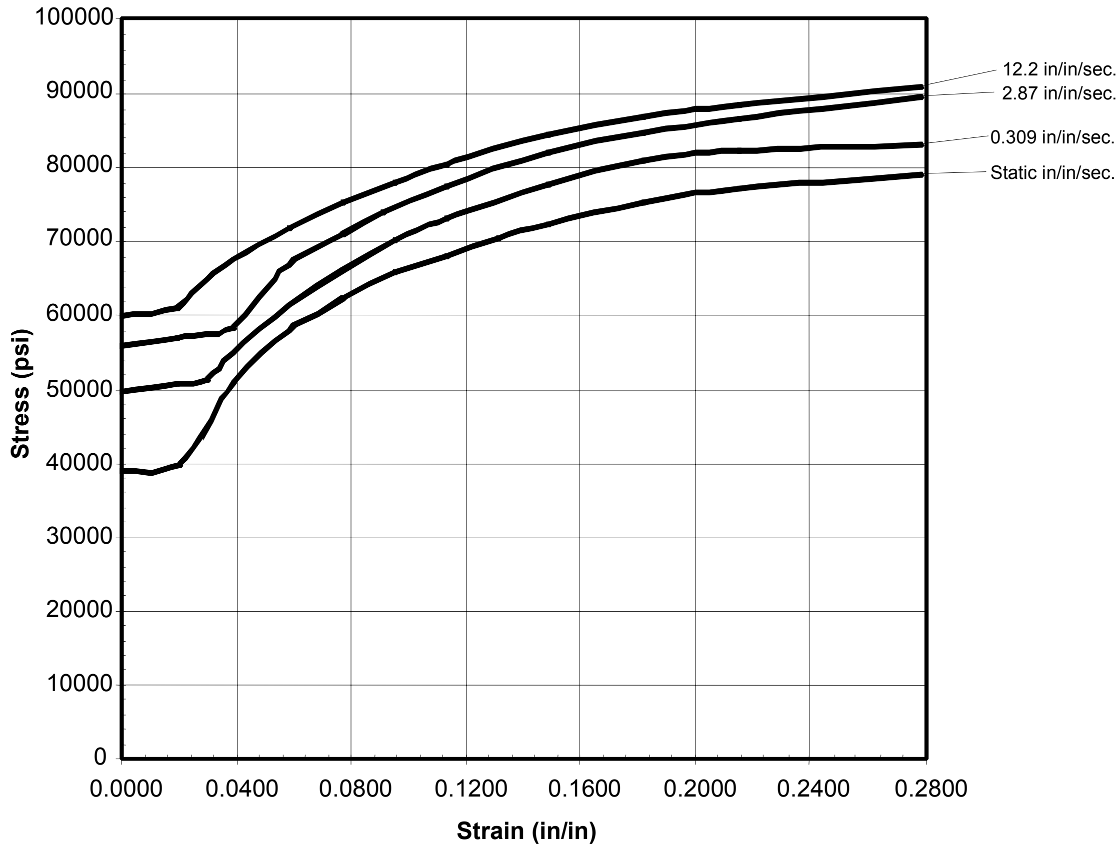


Figure 11.2.4-4 Acceleration Time-History of the Canister Bottom During the Concrete Cask 24-Inch Drop Accident With Static Strain Properties

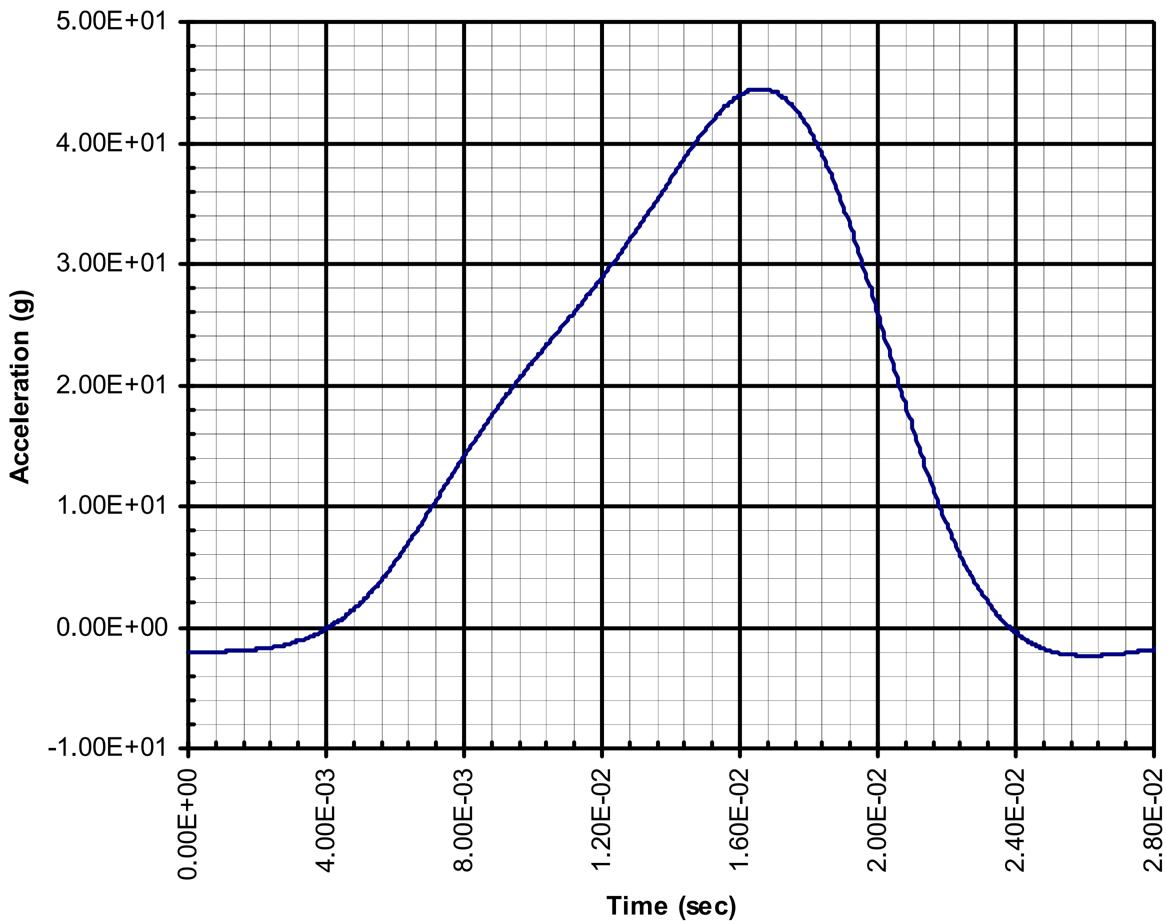


Figure 11.2.4-5 Acceleration Time-History of the Canister Bottom During the Concrete Cask 24-Inch Drop Accident With Strain Rate Dependent Properties

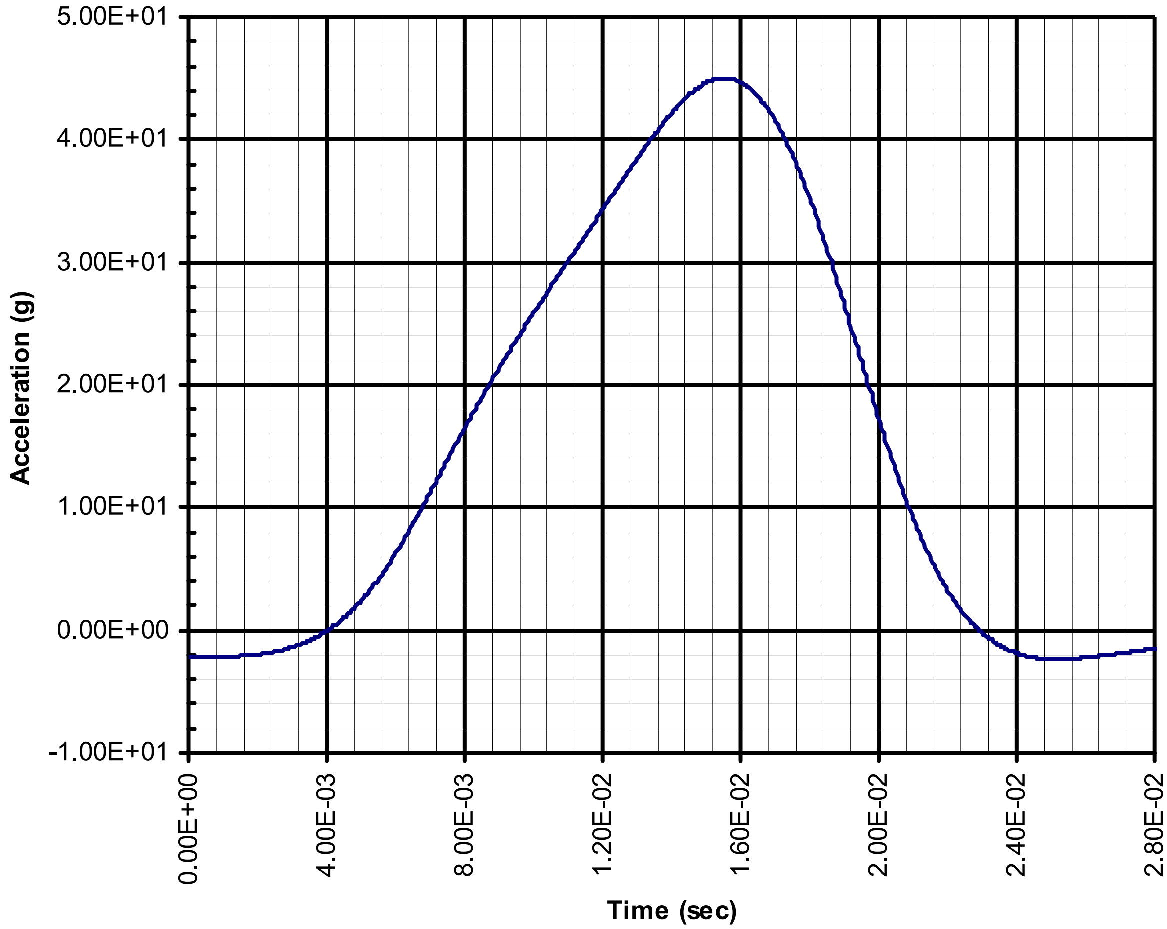


Figure 11.2.4-6 Quarter Model of the PWR Basket Support Disk

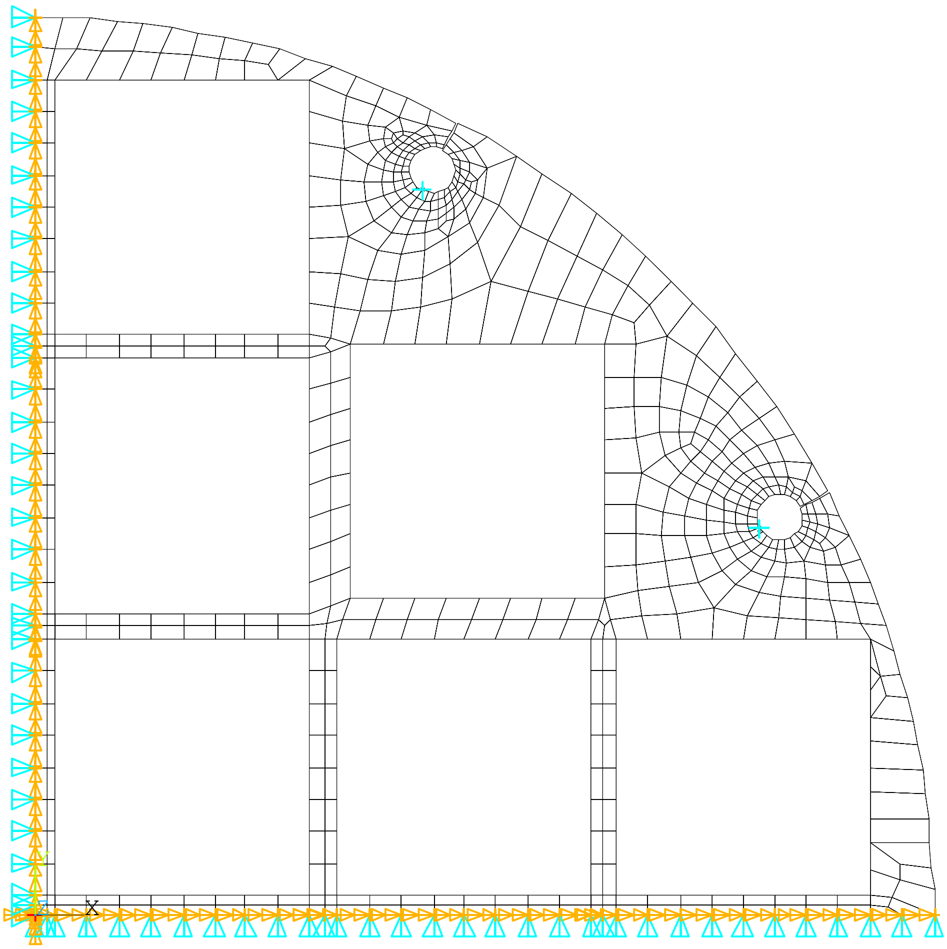


Figure 11.2.4-7 Quarter Model of the BWR Basket Support Disk

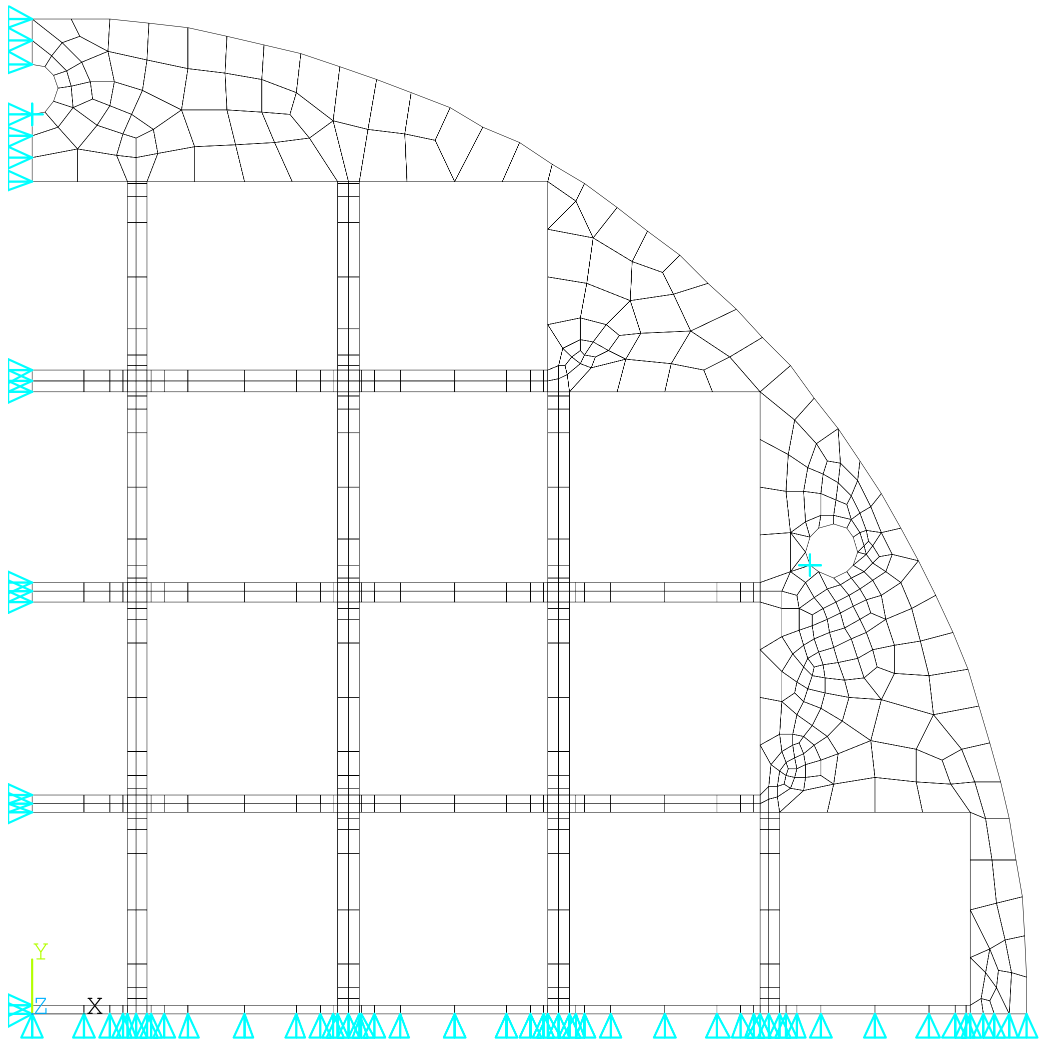


Figure 11.2.4-8 Canister Finite Element Model for 60g Bottom End Impact

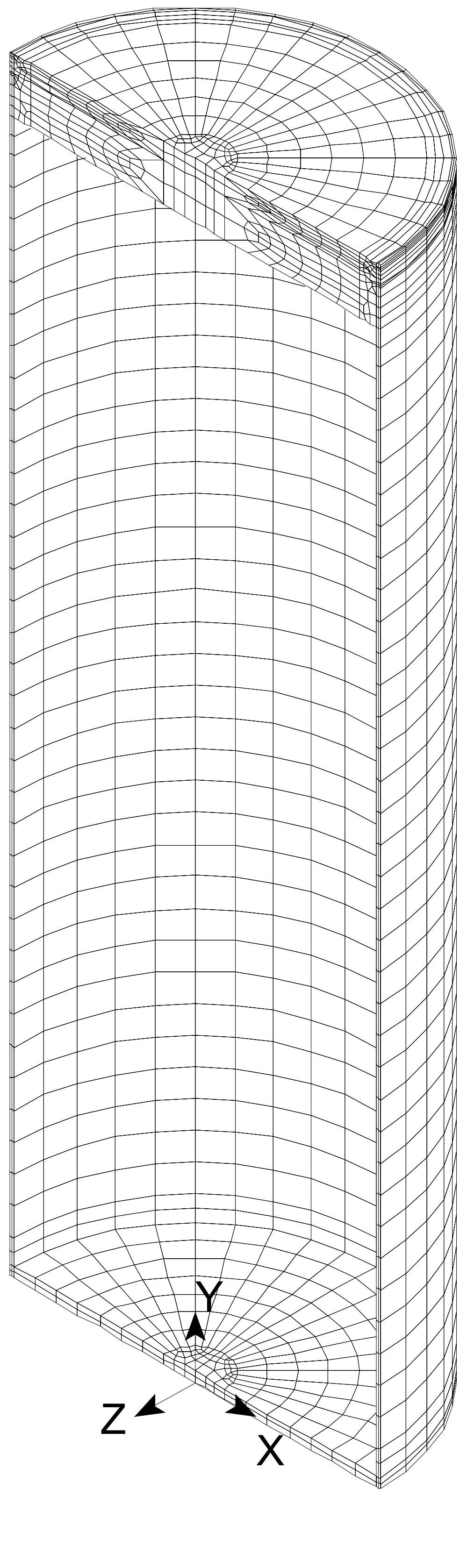
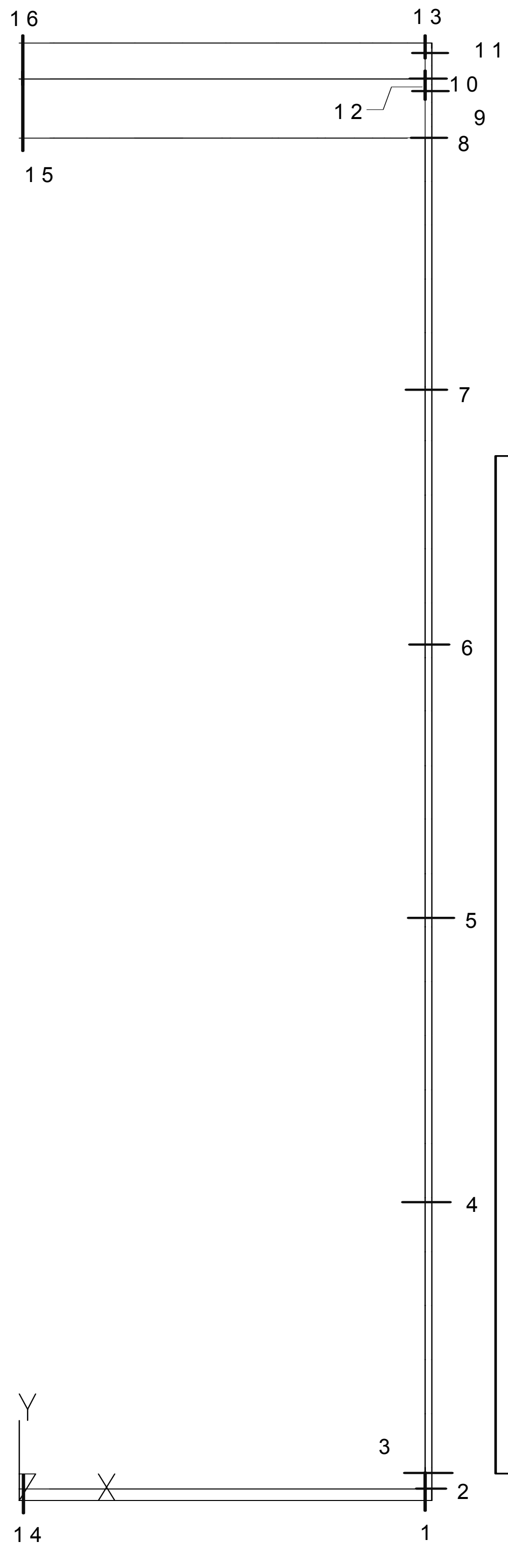
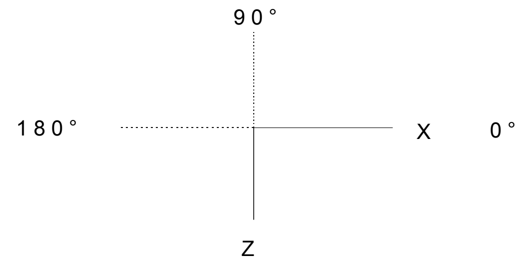


Figure 11.2.4-9 Identification of the Canister Sections for the Evaluation of Canister Stresses due to a 60g Bottom End Impact





Top View of Axis

PWR Section Coordinates at Z = 0

Axial Section	Node 1 x	Node 1 y	Node 2 x	Node 2 y
1	32.905	0.0	32.905	1.75
2	32.905	1.75	33.53	1.75
3	32.905	2.25	33.53	2.25
4	32.905	34.15	33.53	34.15
5	32.905	66.55	33.53	66.55
6	32.905	98.95	33.53	98.95
7	32.905	131.35	33.53	131.35
8	32.905	163.75	33.53	163.75
9	32.905	170.25	33.53	170.25
10	32.905	170.75	33.53	170.75
11	32.905	172.87	33.53	172.87
12	32.905	170.25	32.905	170.75
13	32.905	172.87	32.905	173.75
14	0.1	0.0	0.1	1.75
15	0.1	163.75	0.1	170.73
16	0.1	170.75	0.1	173.75

BWR Section Coordinates at Z = 0

Axial Section	Node 1 x	Node 1 y	Node 2 x	Node 2 y
1	32.905	0.0	32.905	1.75
2	32.905	1.75	33.53	1.75
3	32.905	2.25	33.53	2.25
4	32.905	37.79	33.53	37.79
5	32.905	73.83	33.53	73.83
6	32.905	109.87	33.53	109.87
7	32.905	145.91	33.53	145.91
8	32.905	181.95	33.53	181.95
9	32.905	188.45	33.53	188.45
10	32.905	188.95	33.53	188.95
11	32.905	191.07	33.53	191.07
12	32.905	188.45	32.905	188.95
13	32.905	191.07	32.905	191.95
14	0.1	0.0	0.1	1.75
15	0.1	181.95	0.1	188.73
16	0.1	188.97	0.1	191.95

Table 11.2.4-1 Canister P_m Stresses During a 60g Bottom Impact (15 psig Internal Pressure)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.0	-0.5	-2.9	0.0	-0.1	-0.3	3.0	40.1	12.4
2	0.8	-1.1	-6.2	0.2	0.0	-0.3	7.0	40.1	4.7
3	-0.2	-1.4	-7.2	0.1	0.0	0.2	7.0	40.1	4.7
4	0.0	0.8	-6.6	-0.1	0.0	0.0	7.4	38.8	4.2
5	0.0	0.8	-6.1	-0.1	0.0	0.0	6.9	35.9	4.2
6	0.0	0.8	-5.5	-0.1	0.0	0.0	6.3	35.6	4.7
7	0.0	0.8	-4.9	-0.1	0.0	0.0	5.7	38.2	5.7
8	0.1	0.8	-3.9	-0.1	0.0	0.1	4.7	40.1	7.5
9	-0.7	-2.0	-2.0	0.0	0.0	-0.5	1.6	40.1	24.9
10	1.5	-1.2	-1.3	0.2	0.0	0.2	2.8	40.1	13.2
11	-1.7	-0.9	0.5	0.0	0.0	-0.3	2.2	40.1	17.1
12	0.7	-0.6	1.6	0.1	-0.1	0.4	2.3	40.1	16.4
13	0.5	-1.0	-1.9	0.1	-0.1	-0.3	2.4	32.1[3]	12.4
14	0.1	0.1	-1.0	0.0	0.0	0.0	1.2	40.1	34.0
15	0.3	0.3	0.0	0.0	0.0	0.0	0.3	40.1	134.3
16	-0.2	-0.2	0.0	0.0	0.0	0.0	0.2	40.1	215.9

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level D is used for material allowable stresses.
3. Allowable stress includes a stress reduction factor for the weld: 0.8 × allowable stress.

Table 11.2.4-2 Canister $P_m + P_b$ Stresses During a 60g Bottom Impact (15 psig Internal Pressure)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.7	-0.3	-3.1	0.1	-0.1	-0.4	3.9	60.1	14.3
2	0.4	-2.0	-8.9	0.2	0.0	-0.2	9.4	60.1	5.4
3	-0.1	-1.7	-8.2	0.1	0.0	0.2	8.2	60.1	6.4
4	0.0	0.8	-6.6	-0.1	0.0	0.0	7.4	58.2	6.8
5	0.0	0.8	-6.1	-0.1	0.0	0.0	6.9	53.8	6.9
6	0.0	0.8	-5.5	-0.1	0.0	0.0	6.3	53.3	7.5
7	0.0	0.8	-4.9	-0.1	0.0	0.0	5.7	57.4	9.0
8	0.2	0.6	-4.9	0.0	0.0	0.2	5.5	60.1	9.9
9	-0.5	-2.8	-4.8	0.0	0.0	-0.8	4.6	60.1	12.2
10	0.8	-2.6	-5.6	0.0	0.0	-0.4	6.4	60.1	8.3
11	-1.3	0.4	4.5	-0.1	0.0	-0.4	5.8	60.1	9.3
12	2.5	0.3	2.8	0.2	0.0	0.9	3.2	60.1	17.6
13	2.9	-0.1	-0.8	0.2	-0.1	-0.3	3.8	60.1[3]	11.7
14	0.1	0.1	-1.0	0.0	0.0	0.0	1.2	60.1	51.5
15	3.6	3.6	0.0	0.0	0.0	0.0	3.6	60.1	15.8
16	-1.8	-1.8	-0.1	0.0	0.0	0.0	1.8	60.1	32.8

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level D is used for material allowable stresses.
3. Allowable stress includes a stress reduction factor for the weld: 0.8 × allowable stress.

Table 11.2.4-3 Summary of Maximum Stresses for PWR and BWR Basket Weldments During a 60g Bottom Impact

Case	Stress Category	Maximum Stress Intensity[1] (ksi)	Allowable Stress[2] (ksi)	Margin of Safety
PWR Top Weldment	$P_m + P_b$	27.5	63.5	1.31
PWR Bottom Weldment	$P_m + P_b$	12.0	68.5	+Large
BWR Top Weldment	$P_m + P_b$	34.1	64.0	0.88
BWR Bottom Weldment	$P_m + P_b$	51.9	65.2	0.26

1. Nodal stresses from the finite element analysis results are used.
2. Allowable stresses are conservatively determined at the maximum temperatures of the weldments.

Table 11.2.4-4 Canister P_m Stresses During a 60g Bottom Impact (No Internal Pressure)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.0	-0.7	-2.5	0.1	-0.1	-0.4	2.6	40.1	14.6
2	0.8	-1.4	-6.2	0.2	0.0	-0.4	7.1	40.1	4.7
3	-0.2	-1.8	-7.6	0.2	0.0	0.1	7.4	40.1	4.4
4	0.0	0.0	-7.0	0.0	0.0	0.0	7.0	38.8	4.5
5	0.0	0.0	-6.5	0.0	0.0	0.0	6.5	35.9	4.6
6	0.0	0.0	-5.9	0.0	0.0	0.0	5.9	35.6	5.0
7	0.0	0.0	-5.3	0.0	0.0	0.0	5.3	38.2	6.2
8	0.1	0.4	-4.2	0.0	0.0	0.1	4.6	40.1	7.8
9	-0.8	-2.2	-2.1	0.0	0.0	-0.5	1.7	40.1	23.3
10	1.7	-1.3	-1.4	0.2	0.0	0.2	3.1	40.1	12.1
11	-1.8	-0.9	0.5	0.0	0.0	-0.3	2.4	40.1	15.5
12	0.8	-0.6	1.7	0.1	-0.1	0.4	2.5	40.1	15.2
13	0.5	-1.1	-2.0	0.2	-0.1	-0.3	2.6	32.1[3]	11.3
14	0.1	0.1	0.0	0.0	0.0	0.0	0.1	40.1	351.2
15	0.3	0.3	0.0	0.0	0.0	0.0	0.3	40.1	126.8
16	-0.2	-0.2	0.0	0.0	0.0	0.0	0.2	40.1	197.0

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level D is used for material allowable stresses.
3. Allowable stress includes a stress reduction factor for the weld: 0.8 × allowable stress.

Table 11.2.4-5 Canister Buckling Evaluation Results for 60g Bottom End Impact

	Canister Shell
Longitudinal (Axial) Stress σ_ϕ (psi) [a, b]	9,000
Circumferential (Hoop) Stress σ_θ (psi) [a, b]	3,000
In-Plane Shear Stress $\sigma_{\phi\theta}$ (psi) [b]	500
Elastic Buckling Interaction Equations (ASME Code Case N-284-1, 1713.1.1)	
Axial Compression + Hoop Compression $(\sigma_\phi-0.5\sigma_{ha})/(\sigma_{xa}-0.5\sigma_{ha}) + (\sigma_\theta/\sigma_{ha})^2$	0.326
Axial Compression + Shear $(\sigma_\phi/\sigma_{xa}) + (\sigma_{\phi\theta}/\sigma_{\tau a})^2$	0.193
Hoop Compression + In-Plane Shear $(\sigma_\theta/\sigma_{ra}) + (\sigma_{\phi\theta}/\sigma_{\tau a})^2$	0.437
Axial Compression + Hoop Compression + In-Plane Shear $(\sigma_\phi-0.5K_s\sigma_{ha})/(K_s\sigma_{xa}-0.5K_s\sigma_{ha}) + (\sigma_\theta/K_s\sigma_{ha})^2$	0.326
Plastic Buckling Interaction Equations (ASME Code Case N-284-1, 1713.2.1)	
Axial Compression σ_ϕ/σ_{xc}	0.232
Hoop Compression $\sigma_\theta/\sigma_{rc}$	0.437
Axial Compression + Shear [c] $\sigma_\phi/\sigma_{xc} + (\sigma_{\phi\theta}/\sigma_{\tau c})^2$	0.232
Hoop Compression + Shear $\sigma_\theta/\sigma_{rc} + (\sigma_{\phi\theta}/\sigma_{\tau c})^2$	0.437

[a] Bounding compressive stresses.

[b] Component stresses include thermal stresses.

[c] σ_ϕ in this equation corresponds to the axial stress, which is misprinted as σ_θ in ASME Code Case N-284-1.

Table 11.2.4-6 $P_m + P_b$ Stresses for PWR Support Disk - 60g Concrete Cask Bottom End Impact (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress	Margin of Safety
66	37.2	18.9	15.6	46.2	135.0	1.9
72	18.1	37.2	15.3	45.7	135.0	2.0
120	17.7	37.3	-15.0	45.5	135.0	2.0
82	36.9	17.9	-15.0	45.1	135.0	2.0
12	-24.1	8.5	2.4	32.9	133.5	3.1
28	-24.1	8.5	2.4	32.9	133.5	3.1
26	-24.0	8.5	-2.3	32.8	133.5	3.1
54	8.5	-24.0	-2.3	32.8	133.5	3.1
14	-23.9	8.5	-2.3	32.8	133.5	3.1
42	8.4	-24.0	-2.3	32.7	133.5	3.1
56	8.5	-23.9	2.3	32.7	133.5	3.1
40	8.4	-24.0	2.3	32.7	133.5	3.1
90	24.5	4.1	-10.4	29.1	135.0	3.6
67	3.3	23.6	10.5	29.1	135.0	3.6
99	3.3	23.5	10.5	29.0	135.0	3.7
106	24.1	3.9	10.4	29.0	135.0	3.7
122	24.4	3.9	-10.3	29.0	135.0	3.7
74	24.1	3.9	10.4	29.0	135.0	3.7
83	3.6	23.7	-10.2	28.6	135.0	3.7
115	3.3	23.6	-10.1	28.6	135.0	3.7
88	12.4	9.5	-14.1	28.4	135.0	3.8
114	9.7	11.9	-14.1	28.4	135.0	3.8
104	11.5	10.4	13.5	27.1	135.0	4.0
98	11.7	11.0	13.1	26.2	135.0	4.2
4	-11.1	-19.7	-7.6	24.1	125.8	4.2
2	-11.1	-19.7	-7.7	24.1	125.8	4.2
3	-19.6	-11.0	-7.6	24.1	125.8	4.2
1	-19.6	-11.0	-7.6	24.0	125.8	4.2
35	-5.3	-22.4	-4.2	23.3	129.9	4.6
37	-5.4	-22.3	4.2	23.3	129.9	4.6
7	-22.3	-5.3	-4.2	23.3	129.9	4.6
51	-5.3	-22.3	-4.1	23.3	129.9	4.6
49	-5.3	-22.3	4.2	23.3	129.9	4.6
23	-22.3	-5.3	-4.2	23.3	129.9	4.6
21	-22.3	-5.3	4.2	23.2	129.9	4.6
9	-22.3	-5.3	4.1	23.2	129.9	4.6
11	-12.3	9.4	-4.3	23.4	133.5	4.7
25	-12.3	9.4	-4.2	23.3	133.5	4.7
53	9.4	-12.3	4.3	23.3	133.5	4.7
39	9.3	-12.3	4.3	23.2	133.5	4.8

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.

Table 11.2.4-7 $P_m + P_b$ Stresses for BWR Support Disk - 60g Concrete Cask Bottom End Impact (ksi)

Section[1]	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
129	53.2	18.4	10.7	56.2	90.0	0.60
54	52.1	11.4	10.9	54.8	90.0	0.64
171	9.1	52.8	7.7	54.1	90.0	0.66
300	9.1	52.8	7.6	54.1	90.0	0.66
65	50.3	16.0	-10.3	53.2	90.0	0.69
192	49.9	16.8	-10.9	53.1	90.0	0.69
257	45.6	23.2	-14.7	52.9	90.0	0.70
234	11.5	51.7	-6.6	52.8	90.0	0.71
108	9.9	51.6	-6.3	52.6	90.0	0.71
119	50.1	10.2	-9.9	52.5	90.0	0.72
246	49.4	9.1	-9.9	51.7	90.0	0.74
182	49.2	9.5	9.7	51.4	90.0	0.75
103	13.6	16.2	11.6	26.6	90.0	2.39
229	13.6	16.1	11.6	26.5	90.0	2.39
109	-5.3	20.1	2.5	25.9	90.0	2.47
77	10.6	-14.1	3.9	25.9	90.0	2.48
203	10.5	-14.1	3.9	25.7	90.0	2.50
140	10.5	-14.1	-3.8	25.7	90.0	2.50
295	13.4	15.1	-11.4	25.7	90.0	2.50
269	10.5	-14.1	-3.8	25.7	90.0	2.50
166	13.4	15.1	-11.4	25.7	90.0	2.51
301	-4.1	21.1	-2.1	25.6	90.0	2.51
172	-4.3	20.9	-2.2	25.6	90.0	2.52
134	1.7	11.8	-11.6	25.4	90.0	2.55
263	1.6	11.7	-11.6	25.3	90.0	2.55
197	1.6	11.8	11.6	25.3	90.0	2.55
71	1.7	11.8	11.6	25.3	90.0	2.55
235	-3.3	21.5	2.1	25.1	90.0	2.58
27	15.4	-8.9	-2.8	24.9	90.0	2.61
165	-12.3	-4.6	-11.8	24.9	90.0	2.61
228	-12.3	-4.5	11.8	24.9	90.0	2.62
294	-12.3	-4.6	-11.8	24.9	90.0	2.62
40	15.3	-8.9	2.9	24.8	90.0	2.62
102	-12.3	-4.5	11.8	24.8	90.0	2.62
73	4.2	14.1	11.3	24.6	90.0	2.65
199	4.1	14.2	11.2	24.6	90.0	2.66
124	-20.4	-6.4	-8.5	24.5	90.0	2.67
252	-20.4	-6.4	-8.5	24.4	90.0	2.68
60	-20.4	-6.5	8.6	24.4	90.0	2.69
187	-20.4	-6.4	8.5	24.4	90.0	2.69

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16

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11.2.5 Explosion

The analysis of a design basis flood presented in Section 11.2.9 shows that the flood exerts a pressure of 22 psig on the canister, and that the Universal Storage System experiences no adverse effects due to this pressure. The pressure of 22 psig is considered to bound any pressure due to an explosion occurring in the vicinity of the ISFSI.

11.2.5.1 Cause of Explosion

An explosion affecting the Universal Storage System may be caused by industrial accidents or the presence of explosive substances in the vicinity of the ISFSI. However, no flammable or explosive substances are stored or used at the storage facility. In addition, site administrative controls exclude explosive substances in the vicinity of the ISFSI. Therefore, an explosion affecting the site is extremely unlikely. This accident is evaluated in order to provide a bounding pressure that could be used in the event that the potential of an explosion must be considered at a given site.

11.2.5.2 Analysis of Explosion

Pressure due to an explosion event is bounded by the pressure effects of a flood having a depth of 50 feet. The Transportable Storage Canister shell is evaluated in Section 11.2.9 for the effects of the flood having a depth of 50 feet, and the results are summarized in Tables 11.2.9-1 and 11.2.9-2.

There is no adverse consequence to the canister as a result of the 22 psig pressure exerted by a design basis flood. This pressure conservatively bounds an explosion event.

11.2.5.3 Corrective Actions

In the unlikely event of a nearby explosion, inspection of the concrete casks is required to ensure that the air inlets and outlets are free of debris, and to ensure that the monitoring system and screens are intact. No further recovery or corrective actions are required for this accident.

11.2.5.4 Radiological Impact

There are no radiological consequences for this accident.

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11.2.6 Fire Accident

This section evaluates the effects of a bounding condition hypothetical fire accident, although a fire accident is a very unlikely occurrence in the lifetime of the Universal Storage System. The evaluation demonstrates that for the hypothetical thermal accident (fire) condition the cask meets its storage performance requirements.

11.2.6.1 Cause of Fire

A fire may be caused by flammable material or by a transport vehicle. While it is possible that a transport vehicle could cause a fire while transferring a loaded storage cask at the ISFSI, this fire will be confined to the vehicle and will be rapidly extinguished by the persons performing the transfer operations or by the site fire crew. The maximum permissible quantity of fuel in the combined fuel tanks of the transport vehicle and prime mover is the only means by which fuel (maximum 50 gallons) would be next to a cask, and potentially at, or above, the elevation of the surface on which the cask is supported.

The fuel carried by other on-site vehicles or by other equipment used for ISFSI operations and maintenance, such as air compressors or electrical generators, is considered not to be within the proximity of a loaded cask on the ISFSI pad. Site-specific analysis of fire hazards will evaluate the specific equipment used at the ISFSI and determine any additional controls required.

11.2.6.2 Detection of Fire

A fire in the vicinity of the Universal Storage System will be detected by observation of the fire or smoke.

11.2.6.3 Analysis of Fire

The vertical concrete cask with its internal contents, initially at the steady state normal storage condition, is subject to a hypothetical fire accident. The fire is due to the ignition of a flammable fluid, and operationally, the volume of flammable fluid that is permitted to be on the ISFSI pad (at, or above, the elevation of the surface on which a cask is supported and within approximately two feet of an individual cask) is limited to 50 gallons. The lowest burning rate (change of depth per unit time of flammable fluid for a pool of fluid) reported in the 18th Edition of the Fire Protection Handbook [37] is 5 inches/hour for kerosene. The flammable liquid is assumed to cover a 15-foot square area, corresponding to the center to center distance of the concrete casks less the footprint of the concrete cask, which is a 128-inch diameter circle. The depth (D) of the 50 gallons of flammable liquid is calculated as:

$$D = \frac{50\,(\text{gallons})\ \text{x}\ 231\,(\text{in}^3\ /(\text{gallon})}{15\ \text{x}\ 15\ \text{x}\ 144\,(\text{in}^2) - 3.14\ \text{x}\ 128^2\ /\ 4\,(\text{in}^2)}$$

D = 0.6 inches

With a burning rate of 5 inches/hour, the fire would continue for 7.2 minutes. The fire accident evaluation in this section conservatively considers an 8-minute fire. The temperature of the fire is taken to be 1475°F, which is specified for the fire accident condition in 10 CFR 71.73c(3).

The fire condition is an accident condition and is initiated with the concrete cask in a normal operating steady state condition. To determine the maximum temperatures of the concrete cask components, the two-dimensional axisymmetric finite element model for the BWR configuration described in Section 4.4.1.1 is used to perform a transient analysis. However, the effective properties for the canister content for specific heat, density and thermal conductivity for the PWR are used, to conservatively maximize the thermal diffusivity, which results in higher temperatures for the canister contents during the fire accident condition.

The initial condition of the fire accident transient analysis is based on the steady state analysis results for the normal condition of storage, which corresponds to an ambient temperature of 76°F in conjunction with solar insolation (as specified in Section 4.4.1.1). The fire condition is implemented by constraining the nodes at the inlet to be 1475°F for 8 minutes (see Figure 11.2.6-1). One of the nodes at the edge of the inlet is attached to an element in the concrete region. This temperature boundary condition is applied as a stepped boundary condition. During the 8-minute fire, solar insolation is also applied to the outer surface of the concrete cask. At the end of the 8 minutes, the temperature of the nodes at the inlet is reset to the ambient temperature of 76°F. The cool down phase is continued for an additional 10.7 hours to observe the maximum canister shell temperature and the average temperature of the canister contents.

The maximum temperatures of the fuel cladding and basket are obtained by adding the maximum temperature change due to the fire transient to the maximum component temperature for the normal operational condition. The maximum component temperatures are presented in Table 11.2.6-1, which shows that the component temperatures are below the allowable temperatures. The limited duration of the fire and the large thermal capacitance of the concrete cask restricted the temperatures above 244°F to a region less than 3 inches above the top surface of the air inlets. The maximum bulk concrete temperature is 138°F during and after the fire accident. This corresponds to an increase of less than 3°F compared to the bulk concrete temperature for normal condition of storage. These results confirm that the operation of the concrete cask is not adversely affected during and after the fire accident condition.

11.2.6.4 Corrective Actions

Immediately upon detection of the fire, appropriate actions should be taken by site personnel to extinguish the fire. The concrete cask should then be inspected for general deterioration of the concrete, loss of shielding (spalling of concrete), exposed reinforcing bar, and surface discoloration that could affect heat rejection. This inspection will be the basis for the determination of any repair activities necessary to return the concrete cask to its design basis configuration.

In addition, following the accident event, perform the required Response Surveillance in accordance with Section A 5.4 of the Technical Specifications. Corrective actions shall be taken in accordance with the surveillance requirements to return the affected system to a safe operating condition, as applicable to the affected component(s).

11.2.6.5 Radiological Impact

There are no significant radiological consequences for this accident. There may be local spalling of concrete during the fire event, which could lead to some minor reduction in shielding effectiveness. The principal effect would be local increases in radiation dose rate on the cask surface.

Figure 11.2.6-1 Temperature Boundary Condition Applied to the Nodes of the Inlet for the Fire Accident Condition

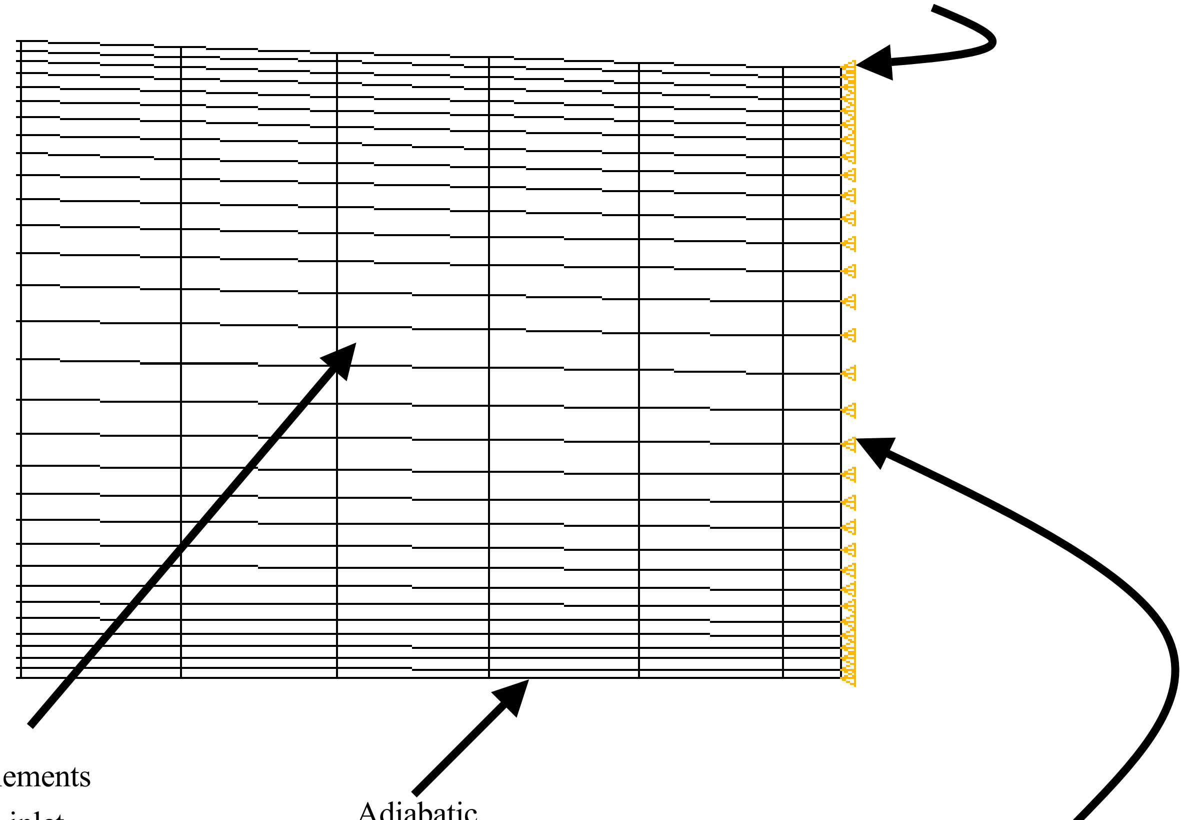

Table 11.2.6-1 Maximum Component Temperatures (°F) During and After the Fire Accident

Component	PWR Maximum temperature (°F)	PWR Allowable temperature (°F)	BWR Maximum temperature (°F)	BWR Allowable temperature (°F)
Fuel clad	688	1058	682	1058
Support disk	641	800	654	700
Heat transfer disk	639	750	652	750
Canister shell	391	800	416	800
Concrete*	244	350	244	350

* Temperatures of 244°F and greater are within 3 inches of the inlet, which does not affect the operation of the concrete cask.

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11.2.7 Maximum Anticipated Heat Load (133°F Ambient Temperature)

This section evaluates the Universal Storage System response to storage operation at an ambient temperature of 133°F. The condition is analyzed in accordance with the requirements of ANSI/ANS 57.9 to evaluate a credible worst-case thermal loading. A steady-state condition is considered in the thermal evaluation of the system for this accident condition.

11.2.7.1 Cause of Maximum Anticipated Heat Load

This condition results from a weather event that causes the concrete cask to be subject to a 133°F ambient temperature with full insolation.

11.2.7.2 Detection of Maximum Anticipated Heat Load

Detection of the high ambient temperature condition will be by observation of the site ambient temperature.

11.2.7.3 Analysis of Maximum Anticipated Heat Load

Using the same methods and thermal models described in Section 11.1.1 for the off-normal conditions of severe ambient temperatures (106°F and -40°F), thermal evaluations are performed for the concrete cask and the canister with its contents for this accident condition. The principal PWR and BWR cask component temperatures for this ambient condition are:

Component	133°F Ambient Max Temp. (°F)		Allowable Max Temp. (°F)	
	PWR	BWR	PWR	BWR
Fuel Cladding	693	690	1058	1058
Support Disks	650	664	800	700
Heat Transfer Disks	648	662	750	750
Canister Shell	408	432	800	800
Concrete	262	266	350	350

This evaluation shows that the component temperatures are within the allowable temperatures for the extreme ambient temperature conditions.

Thermal stress evaluations for the concrete cask are performed using the method and model presented in Section 3.4.4. The concrete temperature results obtained from the thermal analysis for this accident condition are applied to the structural model for stress calculation. The maximum stress, 7,869 psi in the reinforcing steel, occurs in the circumferential direction. The margin of safety is 54,000 psi/7,869 psi -1 = +5.9. The maximum compressive stress, 808 psi, in the concrete occurs in the vertical direction. The maximum circumferential compressive stress in the concrete is 116 psi. The margin of safety is [0.7(4,000 psi)/808 psi] -1 = +2.5. These stresses are used in the loading combination for the concrete cask shown in Section 3.4.4.2.

11.2.7.4 Corrective Actions

The high ambient temperature condition is a natural phenomenon, and no recovery or corrective actions are required.

11.2.7.5 Radiological Impact

There are no dose implications due to this event.

11.2.8 Earthquake Event

This section provides an evaluation of the response of the vertical concrete cask to an earthquake imparting a horizontal acceleration of 0.26g and 0.29g at the top surface of the concrete pad. This evaluation shows that the loaded or empty vertical concrete cask does not tip over or slide in the earthquake event. The vertical acceleration is defined as 2/3 of the horizontal acceleration in accordance with ASCE 4-86 [36].

11.2.8.1 Cause of the Earthquake Event

Earthquakes are natural phenomena to which the storage system might be subjected at any U.S. site. Earthquakes are detected by the ground motion and by seismic instrumentation on and off site.

11.2.8.2 Earthquake Event Analysis

In the event of earthquake, there exists a base shear force or overturning force due to the horizontal acceleration ground motion and a restoring force due to the vertical acceleration ground motion. This ground motion tends to rotate the concrete cask about the bottom corner at the point of rotation (at the chamfer). The horizontal moment arm extends from the center of gravity (C.G.) toward the outer radius of the concrete cask. The vertical moment arm reaches from the C.G. to the bottom of the cask. When the overturning moment is greater than or equal to the restoring moment, the cask will tip over. To maximize this overturning moment, the dimensions for the Class 3 PWR configuration, which has the highest C.G., are used in this evaluation. Based on the requirements presented in NUREG-0800 [22], the static analysis method is considered applicable if the natural frequency of the structure is greater than 33 cycles per second (Hz).

The combined effect of shear and flexure is computed as:

$$\frac{1}{f^2} = \frac{1}{f_f^{\,2}} + \frac{1}{f_s^{\,2}} = \frac{1}{348.6} + \frac{1}{150.7} \quad [19]$$

or

$$f = 105.2 \text{ Hz} > 33 \text{ Hz}$$

where:

f_f = frequency for the first free-free mode based on flexure deformation only (Hz),

f_s = frequency for the first free-free mode based on shear deformation only (Hz).

The frequency f_f is computed as:

$$F_f = \frac{\lambda^2}{2\pi L^2}\sqrt{\frac{EI}{M}} = \frac{4.730^2}{2\pi(226)^2}\sqrt{\frac{\left(3.38\times10^6\right)\times(1.4832\times10^7)}{2.005}} \quad [19]$$

$f_f = 348.6$ Hz

where:

$\lambda = 4.730$,

$L = 226$ in, length of concrete cask,

$E = 3.38 \times 10^6$ psi, modulus of elasticity for concrete at 200°F,

$$I = \text{moment of inertia} = \frac{\pi\left(D_o^4 - D_i^4\right)}{64} = \frac{\pi\left[(136\ \text{in})^4 - (79.5\ \text{in})^4\right]}{64} = 1.4832 \times 10^7\ \text{in}^4,$$

$$\rho = \frac{140}{1728 \times 386.4} = 2.096 \times 10^{-4}\ \text{lbm/in}^3\text{, mass density,}$$

$M = \pi(68^2 - 39.75^2) \times (2.096 \times 10^{-4}) = 2.005$ lbm/in

The frequency accounting for the shear deformation is:

$$f_s = \frac{\lambda_s}{2\pi L}\sqrt{\frac{KG}{\mu}} = \frac{3.141593}{2(3.141595)(226)}\sqrt{\left(\frac{(0.6947)\left(1.40\times10^6\right)}{2.096\times10^{-4}}\right)} \quad [19]$$

$f_s = 150.7$ Hz

where:

$\lambda_s = \pi$,

$L = 226$ in, length of concrete cask,

$$K = \frac{6(1+\nu)\left(1+m^2\right)^2}{(7+6\nu)\left(1+m^2\right)+(20+12\nu)m^2}\text{, shear coefficient,}$$

= 0.6947,

$$\mu = \frac{140}{1728 \times 386.4} = 2.096 \times 10^{-4} \text{ lbm/in}^3,$$ mass density of the material,

$$G = \frac{0.5E}{(1+\nu)} = \frac{0.5(3.38 \times 10^6)}{(1+0.2)} = 1.408 \times 10^6 \text{ psi},$$ modulus of rigidity,

and,

$m = R_i/R_o = 39.75/68 = 0.5846$,

$\nu = 0.2$, Poisson's ratio for concrete.

Since the fundamental mode frequency is greater than 33 Hz, static analysis is appropriate.

11.2.8.2.1 Tip-Over Evaluation of the Vertical Concrete Cask

To maintain the concrete cask in equilibrium, the restoring moment, M_R must be greater than, or equal to, the overturning moment, M_o (i.e. $M_R \geq M_o$). Based on this premise, the following derivation shows that 0.26g acceleration of the design basis earthquake at the surface of the concrete pad is well below the acceleration required to tip-over the cask.

The combination of horizontal and vertical acceleration components is based on the 100-40-40 approach of ASCE 4-86 [36], which considers that when the maximum response from one component occurs, the response from the other two components is 40% of the maximum. According to ASCE 4-86, the vertical component of acceleration shall be obtained by scaling the corresponding ordinates of the horizontal components by two-thirds. However, the vertical component of acceleration is conservatively considered to be the same as the horizontal component of acceleration in the evaluation in this section.

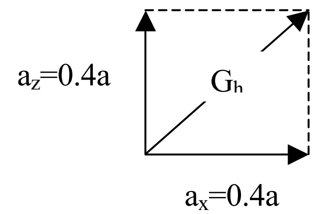
Let:

$a_x = a_z = a$ = horizontal acceleration components
$a_y = a$ = vertical acceleration component
G_h = Vector sum of two horizontal acceleration components
G_v = Vertical acceleration component

There are two cases that have to be analyzed:

Case 1) The vertical acceleration, a_y, is at its peak: (a_y = a, a_x = .4a, a_z = .4a)

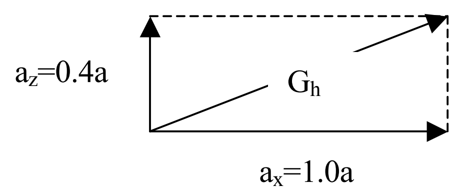
$$G_h = \sqrt{a_x^2 + a_z^2}$$
$$G_h = \sqrt{(0.4 \times a)^2 + (0.4 \times a)^2} = 0.566 \times a$$



$$G_v = 1.0 \times a_y = 1.0 \times a$$

Case 2) One horizonal acceleration, a_x, is at its peak: (a_y=.4a, a_x = a, a_z = .4a)

$$G_h = \sqrt{a_x^2 + a_z^2}$$
$$G_h = \sqrt{(1.0 \times a)^2 + (0.4 \times a)^2} = 1.077 \times a$$



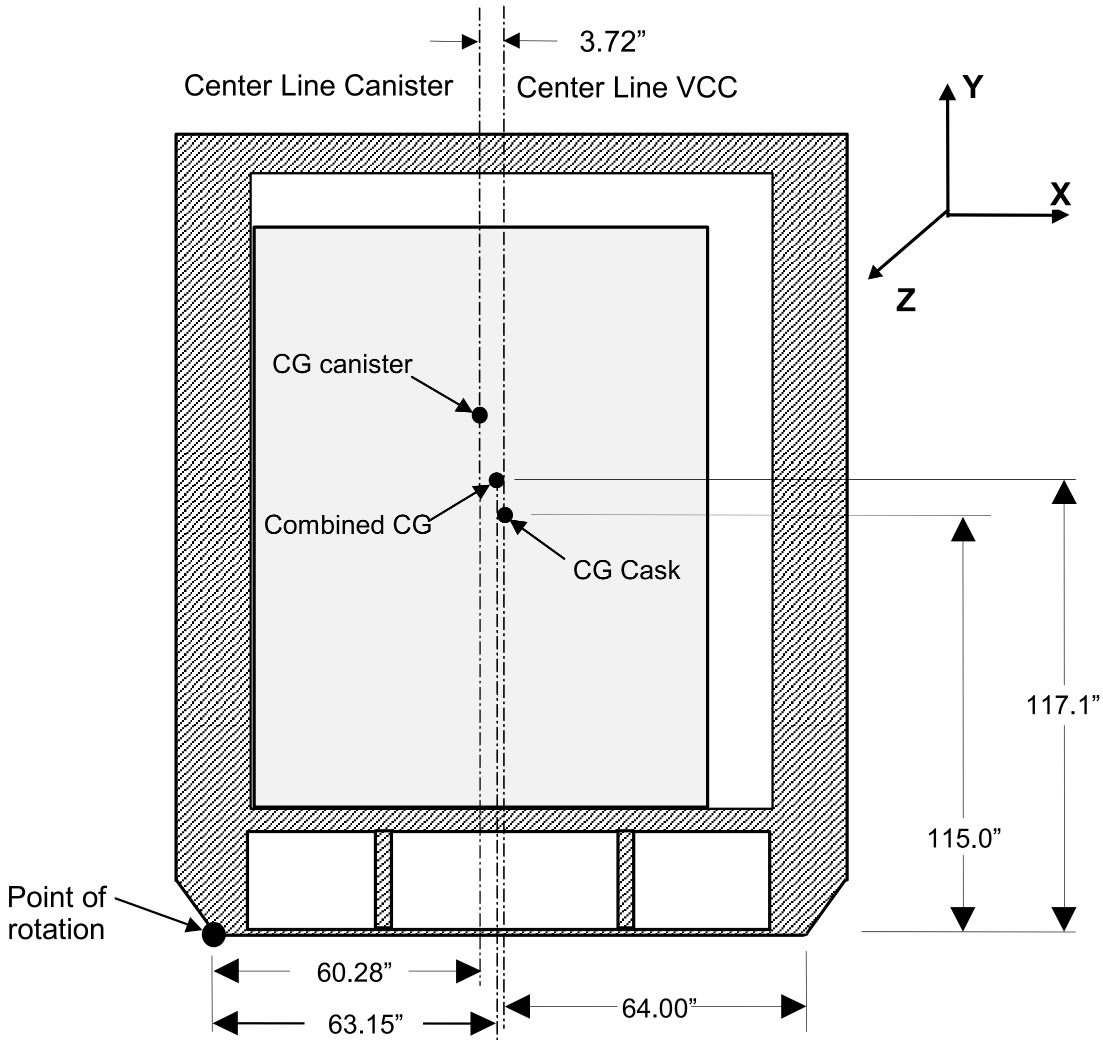
$$G_v = 0.4 \times a_y = 0.4 \times a$$

In order for the cask to resist overturning, the restoring moment, M_R, about the point of rotation, must be greater than the overturning moment, M_o, that:

$$M_R \geq M_o \text{, or}$$
$$F_r \times b \geq F_o \times d \Rightarrow (W \times 1 - W \times G_v) \times b \geq (W \times G_h) \times d$$

where:

- d = vertical distance measured from the base of the VCC to the center of gravity
- b = horizontal distance measured from the point of rotation to the C.G.
- W = the weight of the VCC
- F_o = overturning force
- F_r = restoring force



substituting for G_h and G_v gives:

Case 1

$$(1 - a)\frac{b}{d} \geq 0.566 \times a$$

$$a \leq \frac{b/d}{0.566 + 1.0\left(b/d\right)}$$

Case 2

$$\left(1 - 0.4a\right)\frac{b}{d} \geq 1.077a$$

$$a \leq \frac{b/d}{1.077 + 0.4\left(b/d\right)}$$

Because the canister is not attached to the concrete cask, the combined center of gravity for the concrete cask, with the canister in its maximum off-center position, must be calculated. The point of rotation is established at the outside lower edge of the concrete cask.

The inside diameter of the concrete cask is 74.5 inches and the outside diameter of the canister is 67.06 inches; therefore, the maximum eccentricity between the two is:

$$e = \frac{74.50\text{ in} - 67.06\text{ in}}{2} = 3.72\text{ in.}$$

The horizontal displacement, x, of the combined C.G. due to eccentric placement of the canister is:

$$x = \frac{70{,}701(3.72)}{310{,}345} = 0.85\text{ in.}$$

Therefore,

$$b = 64 - 0.85 = 63.15\text{ in.}$$

$$d = 117.1\text{ in.}$$

1) $$a \le \frac{63.15/117.1}{0.566 + 1.0 \times 63.15/117.1}$$

$a \le 0.49\,g$

2) $$a \le \frac{63.15/117.1}{1.077 + 0.4 \times 63.15/117.1}$$

$a \le 0.42\,g$

Therefore, the minimum ground acceleration that may cause a tip-over of a loaded concrete cask is 0.42g. Since the 0.26g design basis earthquake ground acceleration for the UMS® system is less than 0.42g, the storage cask will not tip over.

The factor of safety is 0.42 / 0.26 = 1.61, which is greater than the required factor of safety of 1.1 in accordance with ANSI/ANS-57.9.

Since an empty vertical concrete cask has a lower C.G. as compared to a loaded concrete cask, the tip-over evaluation for the empty concrete cask is bounded by that for the loaded concrete cask.

11.2.8.2.2 Sliding Evaluation of the Vertical Concrete Cask

For sites imposing the restriction that the Vertical Concrete Cask does not slide during a seismic event, the force holding the cask (F_s) has to be greater than or equal to the force trying to move the cask.

Based on the equation for static friction:

$$F_s = \mu N \geq G_h W$$
$$\mu\left(1 - G_v\right) W \geq G_h W$$

where:

μ = coefficient of friction
N = the normal force
W = the weight of the concrete cask
G_v = vertical acceleration component
G_h = resultant of horizontal acceleration component

Substituting G_h and G_v for the two cases:

Case 1	Case 2
μ (1 – 1.0a) ≥ 0.556a	μ (1 – 0.4a) ≥ 1.077a

For the coefficient of friction of 0.35 [21] between the steel bottom plate of the concrete cask and the concrete surface of the storage pad:

Case 1: $0.35\times(1\text{-a}) \geq 0.566\text{a}$
$\text{a} \leq 0.38\text{g}$

Case 2: $0.35\times(1\text{-}0.4\text{a}) \geq 1.077\text{a}$
$\text{a} \leq 0.29\text{g}$

For a design acceleration of 0.26g, the minimum factor of safety (FS) for acceleration is:

$$FS = \frac{0.29g}{0.26g} = 1.12$$

For a coefficient of friction of 0.4 between the steel bottom plate of the concrete cask and the concrete surface of the storage pad:

Case 1: $0.4\times(1\text{-a}) \geq 0.566\text{a}$
$\text{a} \leq 0.41$

Case 2: $0.4\times(1\text{-}0.4\text{a}) \geq 1.077\text{a}$
$\text{a} \leq 0.32$

For a design acceleration of 0.29g, the minimum factor of safety (FS) for acceleration is:

$$FS = \frac{0.32g}{0.29g} = 1.10$$

The analysis shows that the minimum safety factor against cask sliding for the design earthquake accelerations is 1.1 and meets the requirements of ANSI/ANS-57.9.

While the analyses presented in this section demonstrate that the minimum safety factors for sliding meet the requirements of ANSI/ANS 57.9, it should be noted that there is no safety concern with the sliding of a loaded concrete cask on the storage pad. The two possible outcomes of cask sliding are cask tip-over (see 11.2.12) and cask impact with another loaded cask. The stresses induced from the analyzed cask tip-over event far exceed those from the impact of two casks sliding into each other. Consequently, there is no safety concern with the impact of sliding casks. As a result, there is no safety concern if the designed pad coefficient of friction is reduced for any reason.

11.2.8.2.3 Stress Generated in the Vertical Concrete Cask During an Earthquake Event

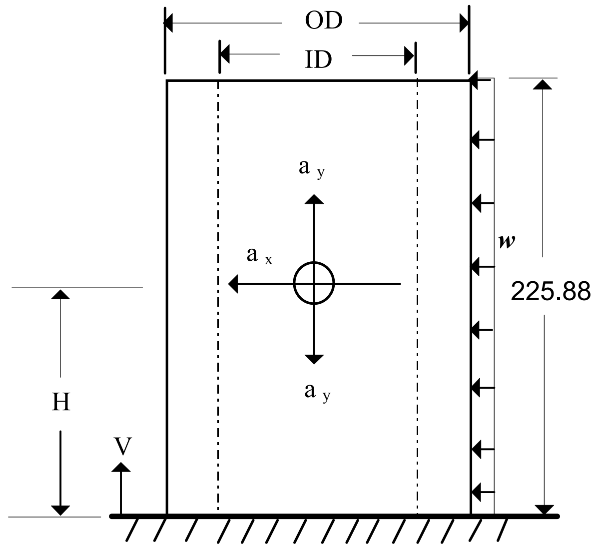
To demonstrate the ability of the concrete cask to withstand earthquake loading conditions, the fully loaded cask is conservatively evaluated for seismic loads of 0.5g in the horizontal direction and 0.5g in the vertical direction. These accelerations reflect a more rigorous seismic loading, and, therefore, bound the design basis earthquake event. No credit is taken for the steel inner liner of the concrete cask. The maximum compressive stress at the outer and inner surfaces of the concrete shell are conservatively calculated by assuming the vertical concrete cask to be a cantilever beam with its bottom end fixed. The maximum compressive stresses are:

$$\sigma_{v\ outer} = (M / S_{outer}) + ((1+a_y)(W_{vcc}) / A) = -84 - 51 = -135 \text{ psi},$$

$$\sigma_{v\ inner} = (M / S_{inner}) + ((1+a_y)(W_{vcc}) / A) = -49 - 51 = -100 \text{ psi},$$

where:

$a =$ 0.50 g, horizontal direction,
$a_y =$ 0.50 g, vertical direction,
H = 117.1 in., fully loaded C.G.,
W_{vcc}= 325,000 lbf, bounding cask weight
OD = 136 in., concrete exterior diameter,
ID = 79.50 in., concrete interior diameter,
$A = \pi (OD^2 - ID^{2)} / 4 = 9{,}562.8 \text{ in.}^2$,
$I = \pi (OD^4 - ID^{4)} / 64 = 14.83 \times 10^6 \text{ in.}^4$,
$S_{outer} = 2I / OD = 218{,}088.2 \text{ in.}^3$,
$S_{inner} = 2I / ID = 373{,}035.0 \text{ in.}^3$,
$w = a_x W_{vcc} / 225.88 \approx 720$ lbf / in.
$M = w (225.88)^2 / 2 = 1.84 \times 10^7$ in.-lbf ,
the maximum bending moment at the support.



The calculated compressive stresses are used in the load combinations for the vertical concrete cask as shown in Table 3.4.4.2-1.

11.2.8.2.4 <u>Vertical Concrete Cask Sliding</u>

For sites permitting the movement of the vertical concrete cask during the seismic event, it is possible that two vertical concrete casks may impact each other during the seismic event.

The bounding condition for the impact of the vertical concrete cask is for one cask to directly impact an adjacent cask with the direction of motion through the centerline of the casks. In this fashion, all the kinetic energy is absorbed in the crushing of the concrete or in the elastic deformation of the concrete. For an incremental thickness of crush (dy), the increment in the crush energy (dE_c) is:

$dE_c = L \times \sigma \times L_c \times dy$

where:

L = axial length of contact between the two vertical concrete casks (inch)
L_c = the width of the contact, (inch)
σ = crush strength (psi)

The width of the crush for a specific crush depth (y) varies as the crush increases and is expressed as:

$$L_c(y) = 2\left(R_o^2 - (R_o - y)^2\right)^{1/2}$$

where:

R_o = outer radius of the vertical concrete cask, (inch)

The crushing will continue until the energy absorbed by crush is equal to the initial kinetic energy, which is associated with the initial velocity V_o. The total sum of all the increments from the initiation of crush to the final value of D_f (depth of crush) is the integral of the above expression, and it is equated to the kinetic energy.

$$0.5 \times W\left(\frac{V_o^2}{g}\right) = \sigma \times L \int_0^{D_f} L_c \, dy$$

where:

W = weight of the vertical concrete cask (lb)
g = 386.3 in/sec^2

Evaluating this integral leads to:

$$0.5 \times W\left(\frac{V_o^2}{g}\right) = \sigma \times L \times R_o^2 \times F(\beta)$$

where:

$$F(\beta) = \left[\frac{\pi}{2} - (1-\beta)\left(1-(1-\beta)^2\right)^{1/2} - \sin^{-1}(1-\beta)\right]$$

and

$\beta = D_f/R_o$

The D_f is computed by incrementing β from zero until the kinetic energy equals the crush energy. For the PWR and the BWR, velocities of 68 in/sec and 50 in/sec are computed, respectively. These result in the following accelerations and crush depths using the weights and heights of the five classes of the vertical concrete cask.

Vertical Concrete Cask Acceleration/Crush Summary

	Class 1	Class 2	Class 3	Class 4	Class 5
VCC Side Impact Acceleration (g)	32.5	32.6	32.7	26.3	26.3
Design Basis Tip-over Acceleration (A_d) in (g)	40	40	40	30	30
Dynamic Load Factor (DLF) for the Tip-over Evaluation	1.19	1.11	1.2	1.05	1.04
A_d/DLF	33.6	36.0	33.3	28.6	28.8
Crush (in)	.3	.3	.3	.2	.2

As indicated in the preceding table, the accelerations resulting from the impact are less than the factored accelerations (A_d/DLF) of the basket used in the PWR and BWR basket and canister evaluations. Therefore, the stresses and displacements of the basket and canister resulting from the tip-over evaluation bound the stresses and displacements resulting from a side impact of two vertical concrete casks.

While the 15-foot center-to-center cask spacing was evaluated in the criticality analysis documented in Chapter 6, the calculations in Chapter 6, combined with minimal cask surface neutron fluxes shown in Chapter 5, clearly demonstrate that there is no neutronic interaction between casks in the array. Therefore, variations in cask spacing as a result of cask movement (including cask-to-cask contact) during abnormal/accident conditions will have no effect on system reactivity.

11.2.8.3 Corrective Actions

Following the natural phenomenon event, perform the required Response Surveillance in accordance with Section A 5.4 of the Technical Specifications. Corrective actions shall be taken in accordance with the surveillance requirements to return the affected system to a safe operating condition, as applicable to the affected component(s). Concrete casks shall be restored to operable status in accordance with LCO A 3.1.6 of the Technical Specifications. Optional temperature-monitoring equipment, if used, should be verified as operable, or repaired and returned to service. As sliding may occur, the positions of the concrete casks should be verified or the casks shall be repositioned to ensure they maintain the 15-foot center-to-center spacing on the ISFSI pad established in Section 8.1.3.

11.2.8.4 Radiological Impact

Minor radiological consequences may result if the concrete casks are required to be repositioned on the ISFSI pad.

11.2.9 Flood

This evaluation considers design basis flood conditions of a 50-foot depth of water having a velocity of 15 feet per second. This flood depth would fully submerge the Universal Storage System. Analysis demonstrates that the Vertical Concrete Cask does not slide or overturn during the design-basis flood. The hydrostatic pressure exerted by the 50-foot depth of water does not produce significant stress in the canister. The Universal Storage System is therefore not adversely impacted by the design basis flood.

Small floods may lead to a blockage of concrete cask air inlets. Full blockage of air inlets is evaluated in Section 11.2.13.

11.2.9.1 Cause of Flood

The probability of a flood event at a given ISFSI site is unlikely because geographical features, and environmental factors specific to that site are considered in the site approval and acceptance process. Some possible sources of a flood are: (1) overflow from a river or stream due to unusually heavy rain, snow-melt runoff, a dam or major water supply line break caused by a seismic event (earthquake); (2) high tides produced by a hurricane; and (3) a tsunami (tidal wave) caused by an underwater earthquake or volcanic eruption.

11.2.9.2 Analysis of Flood

The concrete cask is considered to be resting on a flat level concrete pad when subjected to a flood velocity pressure distributed uniformly over the projected area of the concrete cask. Because of the concrete cask geometry and rigidity, it is analyzed as a rigid body. Assuming full submersion of the concrete cask and steady-state flow conditions, the drag force, F_D, is calculated using classical fluid mechanics for turbulent flow conditions. A safety factor of 1.1 for stability against overturning and sliding is applied to ensure that the analyses bound design basis conditions. The coefficient of friction between carbon steel and concrete used in this analysis is 0.35 [23].

Analysis shows that the concrete cask configured for storing the Class 3 PWR spent fuel, because of its center of gravity, weight, and geometry has the least resistance of the five configurations to flood velocity pressure. Conservatively, the analysis is performed for a canister containing no fuel. The Class 3 PWR cask configuration analysis is as follows.

The buoyancy force, F_b, is calculated from the weight of water (62.4 lbs/ft^3) displaced by the fully submerged concrete cask. The displacement volume (vol) of the concrete cask containing the canister is 1,721 ft^3. The displacement volume is the volume occupied by the cask and the transport canister less the free space in the central annular cavity of the concrete cask.

$$F_b = \text{Vol} \times 62.4 \text{ lbs/ft}^3$$
$$= 107{,}383 \text{ lbs.}$$

Assuming the steady-state flow conditions for a rigid cylinder, the total drag force of the water on the concrete cask is given by the formula:

$$F_{D15} = (C_D)(\rho)(V^2)\left(\frac{A}{2}\right) \qquad [24]$$
$$= 32{,}831 \text{ lbs.}$$

where:

C_D = Drag coefficient, which is dependent upon the Reynolds Number (Re). For flow velocities greater than 6 ft/sec, the value of C_D approaches 0.7 [24].

ρ = mass density of water = 1.94 slugs/ft^3

D = Concrete cask outside diameter (136.0 in. / 12 = 11.33 ft)

V = velocity of water flow (15 ft/sec)

A = projected area of the cask normal to water flow (diameter 11.34 ft × overall height 18.95 ft = 214.9 ft^2)

The drag force required to overturn the concrete cask is determined by summing the moments of the drag force and the submerged weight (weight of the cask less the buoyant force) about a point on the bottom edge of the cask. This method assumes a pinned connection, i.e., the cask will rotate about the point on the edge rather than slide. When these moments are in equilibrium, the cask is at the point of overturning.

$$F_D \times \left(\frac{h}{2}\right) = \left(W_{cask} - F_b\right) \times r$$

F_D = 100,314 lbs

where:

h	=	concrete cask overall height (227.38 in.)
W_{CASK}	=	concrete cask weight = 275,000 lbs
		(Loaded concrete cask – fuel = 310,345 lbs – 35,520 lbs)
F_b	=	buoyant force = 107,383 lbs
r	=	concrete cask radius (5.67 ft)

Solving the drag force equation for the velocity, V, that is required to overturn the concrete cask:

$$V = \sqrt{\frac{2F_D}{C_D \rho A}}$$

= 25.0 ft/sec. (including safety factor of 1.1)

To prevent sliding, the minimum coefficient of friction (with a safety factor of 1.1) between the carbon steel bottom plate of the concrete cask and the concrete surface upon which it rests is,

$$\mu_{min} = \frac{(1.1)F_{D15}}{F_y} = \frac{(1.1)32,831 \text{ lb}}{(275,000 - 107,557)\text{lb}} = 0.22$$

where:

F_y = the submerged weight of the concrete cask.

The analysis shows that the minimum coefficient of friction, μ, required to prevent sliding of the concrete cask is 0.22. For a drag force of 57,160 pounds, the coefficient of friction to prevent sliding is 0.31. The coefficient of friction between the steel bottom plate of the concrete cask and the concrete surface of the storage pad (0.35) is greater than the minimum coefficient of friction required to prevent sliding of the concrete cask. Therefore, the concrete cask does not slide under design-basis flood conditions.

The water velocity required to overturn the concrete cask is greater than the design-basis velocity of 15 ft/sec. Therefore, the concrete cask is not overturned under design basis flood conditions.

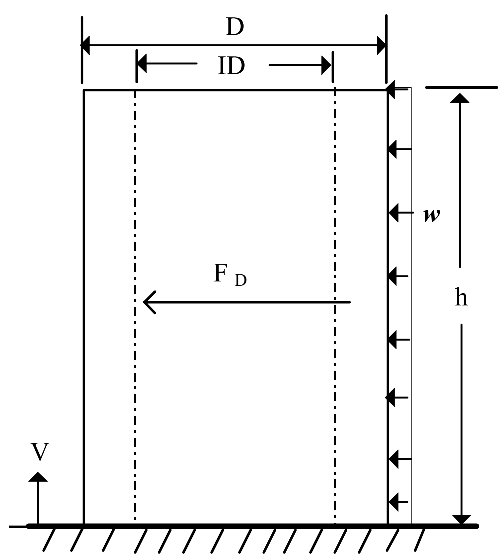
The flood depth of 50 feet exerts a hydrostatic pressure on the canister and the concrete cask. The water exerts a pressure of 22 psi (50 × 62.4/144) on the canister, which results in stresses in the canister shell. Canister internal pressure is conservatively taken as 0 psi. The canister structural analysis for the increased external pressure due to flood conditions is performed using an ANSYS finite element model as described in Section 3.4.4.1.

The resulting maximum canister stresses for flood loads are summarized in Tables 11.2.9-1 and 11.2.9-2 for primary membrane and primary membrane plus bending stresses, respectively.

The sectional stresses shown in Tables 11.2.9-1 and 11.2.9-2 at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations of the stress sections are shown in Figure 3.4.4.1-4. Consequently, there is no adverse consequence to the canister as a result of the hydrostatic pressure due to the flood condition.

The concrete cask is a thick monolithic structure and is not affected by the hydrostatic pressure due to design basis flood. Nonetheless, the stresses in the concrete due to the drag force (F_D) are conservatively calculated as shown below. The concrete cask is considered to be fixed at its base.

F_D = 32,831 lbs
D = 136.0 in. (concrete exterior diameter)
ID = 79.5 in. (concrete interior diameter)
h = 214.68 in. (cask overall height)
A = $\pi (D^2 - ID^2) / 4 = 9{,}563$ in.^{2}
(Cross-sectional area)
I = $\pi (D^4 - ID^4) / 64 = 14.83 \times 10^6$ in.^{4}
(Moment of Inertia)
S = $2I/D = 218{,}088$ in.^{3}
(Section Modulus for outer surface)
w = F_D/h = 155.0 lbf / in.
M = $w(h)^2 / 2 = 3.44 \times 10^6$ in.-lbs
(Bending Moment at the base)



Maximum stresses at the base surface:

$\sigma_v = M / S_{outer} \approx 20$ psi (tension or compression)

The compressive stresses are included in load combination No. 7 in Table 3.4.4.2–1. As shown in Table 3.4.4.2–1, the maximum combined stresses for the load combination due to dead, live, thermal and flood loading, are less than the allowable stress.

11.2.9.3 Corrective Actions

Following the natural phenomenon event, perform the required Response Surveillance in accordance with Section A 5.4 of the Technical Specifications. Corrective actions shall be taken in accordance with the surveillance requirements to return the affected system to a safe operating condition, as applicable to the affected component(s). Concrete casks shall be restored to operable status in accordance with LCO A 3.1.6 of the Technical Specifications. Optional temperature-monitoring equipment, if used, should be verified as operable, or repaired and returned to service.

11.2.9.4 Radiological Impact

There are no dose consequences associated with the design basis flood event.

Table 11.2.9-1 Canister Increased External Pressure (22 psi) with No Internal Pressure (0 psi) Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	-0.17	-0.86	-2.17	0.06	0.03	0.31	2.10	40.08	18.1
2	-1.46	1.76	1.37	-0.24	0.03	0.30	3.29	40.08	11.2
3	0.24	2.71	-0.64	-0.23	-0.05	-0.61	3.69	40.08	9.9
4	-0.02	-1.18	-0.60	0.10	0.00	0.00	1.18	38.77	31.8
5	-0.02	-1.17	-0.60	0.10	0.00	0.00	1.17	35.86	29.7
6	-0.02	-1.17	-0.60	0.10	0.00	0.00	1.17	35.55	29.4
7	-0.02	-1.17	-0.60	0.10	0.00	0.00	1.17	38.23	31.7
8	-0.01	-1.13	-0.54	0.10	0.00	0.00	1.13	40.08	34.3
9	-0.28	-0.34	-0.16	0.02	-0.01	-0.12	0.27	40.08	145.6
10	0.32	-0.13	-0.08	0.03	-0.01	-0.07	0.46	40.08	85.5
11	-0.27	-0.13	0.09	-0.01	-0.01	-0.06	0.37	40.08	106.1
12	0.07	-0.23	-0.17	0.03	-0.01	0.02	0.32	40.08	125.6
13	0.06	-0.16	-0.30	0.02	-0.01	-0.06	0.38	40.08	103.4
14	-0.38	-0.38	-0.01	0.00	-0.16	0.02	0.49	40.08	81.5
15	0.02	0.02	-0.01	0.00	0.00	0.00	0.03	40.08	1235.3
16	-0.03	-0.03	-0.02	0.00	0.00	0.00	0.02	40.08	2524.5

(1) See Figure 3.4.4.1-4 for definition of locations of stress sections.

(2) ASME Service Level D is used for material allowable stress.

Table 11.2.9-2 Canister Increased External Pressure (22 psi) with No Internal Pressure (0 psi) Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	-1.67	-0.20	-5.20	-0.07	0.03	0.02	5.01	60.12	11.0
2	-0.72	4.50	9.96	-0.43	0.05	0.70	10.80	60.12	4.6
3	1.02	-0.99	-13.97	0.13	-0.06	-0.78	15.08	60.12	3.0
4	-0.01	-1.19	-0.58	0.10	0.00	0.00	1.19	58.16	47.8
5	-0.01	-1.18	-0.60	0.10	0.00	0.00	1.19	53.79	44.3
6	-0.01	-1.19	-0.60	0.10	0.00	0.00	1.19	53.32	43.9
7	-0.01	-1.19	-0.60	0.10	0.00	0.00	1.19	57.35	47.3
8	-0.03	-1.16	-0.69	0.10	0.00	0.00	1.15	60.12	51.2
9	-0.19	-0.21	0.16	0.01	-0.01	-0.18	0.50	60.12	119.7
10	0.48	-0.05	0.01	0.04	-0.01	0.06	0.55	60.12	108.1
11	-0.19	0.07	0.69	-0.02	-0.01	-0.07	0.90	60.12	65.8
12	0.54	-0.02	0.07	0.04	-0.01	0.11	0.59	60.12	100.7
13	0.44	-0.01	-0.16	0.04	-0.02	-0.06	0.62	60.12	96.5
14	-7.47	-7.48	-0.23	0.00	-0.14	0.03	7.26	60.12	7.3
15	0.52	0.52	0.01	0.00	0.00	0.00	0.51	60.12	116.4
16	-0.28	-0.28	-0.03	0.00	0.00	0.00	0.25	60.12	240.5

(1) See Figure 3.4.4.1-4 for definition of locations of stress sections.

(2) ASME Service Level D is used for material allowable stress.

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11.2.10 Lightning Strike

This section evaluates the impact of a lightning strike on the Vertical Concrete Cask. The evaluation shows that the cask does not experience adverse effects due to a lightning strike.

11.2.10.1 Cause of Lightning Strike

A lightning strike is a random weather-related event. Because the Vertical Concrete Cask is located on an unsheltered pad, the cask may be subject to a lightning strike. The probability of a lightning strike is primarily dependent on the geographical location of the ISFSI site, as some geographical regions experience a higher frequency of storms containing lightning than others.

11.2.10.2 Detection of Lightning Strike

A lightning strike on a concrete cask may be visually detected at the time of the strike, or by visible surface discoloration at the point of entry or exit of the current flow. Most reactor sites in locations experiencing a frequency of lightning bearing storms have lightning detection systems as an aid to ensuring stability of site electric power.

11.2.10.3 Analysis of the Lightning Strike Event

The analysis of the lightning strike event assumes that the lightning strikes the upper-most metal surface and proceeds through the concrete cask liner to the ground. Therefore, the current path is from the lightning strike point on the outer radius of the top flange of the storage cask, down through the carbon steel inner shell and the bottom plate to the ground. The electrical current flow path results in current-induced Joulean heating along that path.

The integrated maximum current for a lightning strike is a peak current of 250 kiloamps over a period of 260 microseconds, and a continuing current of up to 2 kiloamps for 2 seconds in the case of severe lightning discharges [25].

From Joule's Law, the amount of thermal energy developed by the combined currents is given by the following expression [26]:

$$Q = 0.0009478R\left[I_1^2\left(dt_1\right) + I_2^2\left(dt_2\right)\right]$$
$$= (22.98 \times 10^3)\ R\ \text{Btu} \qquad \text{[Equation 11.2.10.1]}$$

where:

- Q = thermal energy (BTU)
- I_1 = peak current (amps)
- I_2 = continuing current (amps)
- dt_1 = duration of peak current (seconds)
- dt_2 = duration of continuing current (seconds)
- R = resistance (ohms)

The maximum lightning discharge is assumed to attach to the smallest current-carrying component, that is, the top flange connected to the cask lid.

The propagation of the lightning through the carbon steel cask liner, which is both permeable and conductive, is considered to be a transient. For static conditions, the current is distributed throughout the shell. In a transient condition the current will be near the surface of the conductor. Similar to a concentrated surface heat flux incident upon a small surface area, a concentrated current in a confined area of the steel shell will result in higher temperatures than if the current were spread over the entire area, which leads to a conservative result. This conservative assumption is used by constraining the current flow area to a 90 degree sector of the circular cross section of the steel liner as opposed to the entire cross section. The depth of the current penetration (δ in meters) is estimated [27] as:

$$\delta = \frac{1}{\sqrt{\pi \mu f \sigma}}$$

where:

- μ = permeability of the conductor = $100\mu_0$ ($\mu_0 = 4\pi \times 10^{-7}$ Henries/m)
- σ = electrical conductivity (seimens/meter) = $1/\rho$
  = 1/resistivity = $1/9.78 \times 10^{-8}$ (ohm-m)
- f = frequency of the field (Hz)

The pulse is represented conservatively as a half sine form, so that the equivalent $f = 1/2\tau$, where τ is the referenced pulse duration. Two skin depths, corresponding to different pulse duration, are computed. The larger effective frequency will result in a smaller effective area to conduct the current. The effective resistance is computed as:

$$R = \frac{\rho\ l}{a}$$

where:

R = resistance (ohms)
ρ = resistivity = 9.78×10^{-8} (ohm-m)
l = length of conductor path
a = area of conductor (m^2)

Using the current level of the pulse and the duration in conjunction with the carbon steel liner, the resulting energy into the shell is computed using Equation 11.2.10.1.

This thermal energy dissipation is conservatively assumed to occur in the localized volume of the carbon steel involved in the current flow path through the flange to the inner liner. Assuming no heat loss or thermal diffusion beyond the current flow boundary, the maximum temperature increase in the flange due to this thermal energy dissipation is calculated [28] as:

$$\Delta T = \frac{Q}{mc}$$

where:

ΔT = temperature change (°F)
Q = thermal energy (BTU)
C = 0.113 Btu/lbs °F
m = mass (lbm)

The ΔT_1 for the peak current (250KA, 260 μsec) is found to be 4.7°F.

The ΔT_2 for the continuous current (2 kA, 2 sec) is found to be negligible (0.0006°F).

The ΔT_1 corresponds to the increase in the maximum temperature of the steel within the current path. For the concrete to experience an increase in temperature, the heat must disperse from the steel surface throughout the steel. Using the total thickness of the steel, over the 90-degree section, the increase in temperature would be proportional to the volume of steel in this sector resulting in a temperature rise of less than 1°F.

Therefore the increase in concrete temperature attributed to Joulean heating is not significant.

11.2.10.4 Corrective Actions

Following the natural phenomenon event, perform the required Response Surveillance in accordance with Section A 5.4 of the Technical Specifications. Corrective actions shall be taken in accordance with the surveillance requirements to return the affected system to a safe operating condition, as applicable to the affected component(s).

11.2.10.5 Radiological Impact

There are no dose implications due to the lightning event.

11.2.11 Tornado and Tornado Driven Missiles

This section evaluates the strength and stability of the Vertical Concrete Cask for a maximum tornado wind loading and for the impacts of tornado generated missiles. The design basis tornado characteristics are selected in accordance with Regulatory Guide 1.76 [29].

The evaluation demonstrates that the concrete cask remains stable in tornado wind loading in conjunction with impact from a high energy tornado missile. The performance of the cask is not significantly affected by the tornado event.

11.2.11.1 Cause of Tornado and Tornado Driven Missiles

A tornado is a random weather event. Probability of its occurrence is dependent upon the time of the year and geographical areas. Wind loading and tornado driven missiles have the potential for causing damage from pressure differential loading and from impact loading.

11.2.11.2 Detection of Tornado and Tornado Driven Missiles

A tornado event is expected to be visually observed. Advance warning of a tornado and of tornado sightings may be received from the National Weather Service, local radio and television stations, local law enforcement personnel, and site personnel.

11.2.11.3 Analysis of Tornado and Tornado Driven Missiles

Classical techniques are used to evaluate the loading conditions. Cask stability analysis for the maximum tornado wind loading is based on NUREG-0800 [30], Section 3.3.1, “Wind Loadings,” and Section 3.3.2, “Tornado Loadings.” Loads due to tornado-generated missiles are based on NUREG-0800, Section 3.5.1.4, “Missiles Generated by Natural Phenomena.”

The concrete cask stability in a maximum tornado wind is evaluated based on the design wind pressure calculated in accordance with ANSI/ASCE 7-93 [31] and using classical free body stability analysis methods.

Local damage to the concrete shell is assessed using a formula developed for the National Defense Research Committee (NDRC) [32]. This formula is selected as the basis for predicting depth of missile penetration and minimum concrete thickness requirements to prevent scabbing of the concrete. Penetration depths calculated using this formula have been shown to provide reasonable correlation with test results (EPRI Report NP-440) [33].

The local shear strength of the concrete shell is evaluated on the basis of ACI 349-85 [34], Section 11.11.2.1, discounting the reinforcing and the steel internal shell. The concrete shell shear capacity is also evaluated for missile loading using ACI 349-85, Section 11.7.

The cask configuration used in this analysis combines the height of the tallest (Class 3 PWR) cask with the weight and center of gravity of the lightest (Class 1 PWR) cask. This configuration bounds all other configurations for cask stability. The cask properties considered in this evaluation are:

H = Cask Height = 225.88 in (Class 3 PWR)
D_o = Cask Outside Diameter = 136.0 in
D_i = Inside Diameter of concrete shell = 79.5 in
W_{VCC} = Weight of the cask with canister, basket and full fuel load = 285,000 lbs
(285,000 lbs is conservatively used [slightly lighter than the Class 1 PWR cask weight])
A_c = Cross section area of concrete shell = 9,563 in^2
I_c = Moment of inertia of concrete shell = 14.83×10^6 in^4
f_c' = Compressive strength of concrete shell = 4,000 psi

Tornado Wind Loading (Concrete Cask)

The tornado wind velocity is transformed into an effective pressure applied to the cask using procedures delineated in ANSI/ASCE 7-93 Building Code Requirements for Minimum Design Loads in Buildings and Other Structures. The maximum pressure, q, is determined from the maximum tornado wind velocity as follows:

$$q = (0.00256)\, V^2 \text{ psf}$$

where:

V = Maximum tornado wind speed = 360 mph

The velocity pressure exposure coefficient for local terrain effects K, Importance Factor I, and the Gust Factor G, may be taken as unity (1) for evaluating the effects of tornado wind velocity pressure. Then:

$$q = (0.00256)(360)^2 = 331.8 \text{ psf}$$

Considering that the cask is small with respect to the tornado radius, the velocity pressure is assumed uniform over the projected area of the cask. Because the cask is vented, the tornado-induced pressure drop is equalized from inside to outside and has no effect on the cask structure.

The total wind loading on the projected area of the cask, F_w is then computed as:

$$F_w = q \times G \times C_f \times A_p$$
$$= 36{,}100 \text{ lbs}$$

where:

q = Effective velocity pressure (psf) = 331.8 psf.

C_f = Force Coefficient = 0.51 (ASCE 7-93, Table 12 with D $q^{1/2}$ = 206.4 for a moderately smooth surface, h/D = 18.8 ft /11.3 ft = 1.7)

A_f = Projected area of cask = (225.88 in × 136.0 in)/144 = 213.3 ft^2

G = Constant = 1.0

The wind overturning moment, M_w , is computed as:

$$M_w = F_w \times H/2 = 36{,}100 \text{ lbs} \times 225.88 \text{ in}/12 \times 1/2 \cong 340{,}000 \text{ ft-lbs}$$

where H is the cask height.

The stability moment, M_s, of the cask (with the canister, basket and no fuel load) about an edge of the base, is:

$$M_s = W_{cask} \times D_o/2 = 1.52\times10^6 \text{ ft-lbs}$$

where:

D_o = Cask base plate diameter = 128.0 in

W_{cask} = Weight of the cask with canister

≅ 285,000 lbs

ASCE 7-93 requires that the overturning moment due to wind load shall not exceed two-thirds of the dead load stabilizing moment unless the structure is anchored. Therefore, the margin of safety, MS, against overturning is:

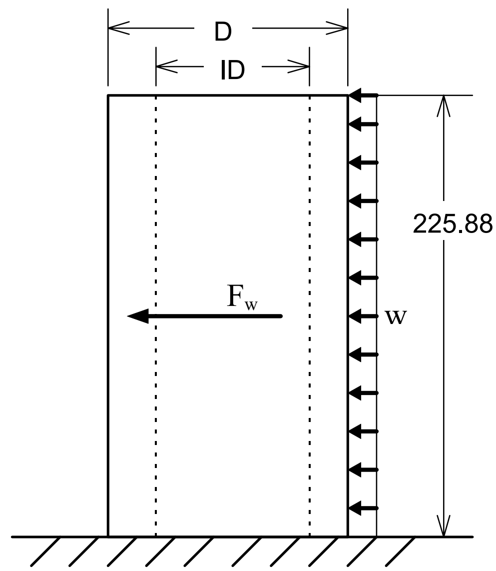
$$MS = \frac{M_S}{M_W} - 1 = \frac{(0.67)1.52\times10^6}{3.40\times10^5} - 1 = +2.00$$

A coefficient of friction of 0.13 (36,100/285,000) between the cask base and the concrete pad on which it rests will inhibit sliding.

Against a coefficient of friction of steel on concrete of approximately 0.35 [23], the margin of safety, MS, against sliding is:

$$MS = \frac{0.35}{0.13} - 1 = +1.69$$

The stresses in the concrete due to the tornado wind load are conservatively calculated below. The concrete cask is considered to be fixed at its base.



F_W = 36,100 lbs

D = 136.0 in. (concrete outside diameter)

ID = 79.5 in. (concrete inside diameter)

H = 225.8 in. /12 = 18.82 ft

A = $\pi (D^2 - ID^{2)} / 4 = 9{,}563\ in^2$

I = $\pi (D^4 - ID^{4)} / 64 = 14.83\times10^6\ in^4$ (Moment of Inertia)

M = $\frac{F_w \times H}{2} \cong 340{,}000$ lbs-ft

Maximum stresses:

$$\sigma = \frac{Mc}{I} = 18.7 \text{ psi} \qquad \text{(tension or compression)}$$

where:

c = D/2 = 68.0 in.

The compressive stresses are included in the load combination No. 3 in Table 3.4.4.2-1, since they are governing stresses for the load combination. As shown in Tables 3.4.4.2-1 and 3.4.4.2-2, the maximum combined stresses for the load combination of dead, live, thermal and tornado wind are less than the allowable stress.

Tornado Missile Loading (Concrete Cask)

The Vertical Concrete Cask is designed to withstand the effects of impacts associated with postulated tornado generated missiles identified in NUREG-0800, Section 3.5.1.4.III.4, Spectrum I missiles. These missiles consist of: 1) a massive high kinetic energy missile (4,000 lbs automobile, with a frontal area of 20 square feet that deforms on impact); 2) a 280 lbs, 8-inch-diameter armor piercing artillery shell; and 3) a small 1-inch diameter solid steel sphere. All of these missiles are assumed to impact in a manner that produces the maximum damage at a velocity of 126 mph (35% of the maximum tornado wind speed of 360 mph). The cask is evaluated for impact effects associated with each of the above missiles.

The principal dimensions and moment arms used in this evaluation are shown in Figure 11.2.11-1.

The concrete cask has no openings except for the four outlets at the top and four inlets at the bottom. The upper openings are configured such that a 1-inch diameter solid steel missile cannot directly enter the concrete cask interior. Additionally, the canister is protected by the canister structural and shield lids. The canister is protected from small missiles entering the lower inlets by a steel pedestal (bottom plate). Therefore, a detailed analysis of the impact of a 1-inch diameter steel missile is not required.

Concrete Shell Local Damage Prediction (Penetration Missile)

Local damage to the cask body is assessed by using the National Defense Research Committee (NDRC) formula [32]. This formula is selected as the basis for predicting depth of penetration and minimum concrete thickness requirements to prevent scabbing. Penetration depths calculated by using this formula have been shown to provide reasonable correlation with test results [33].

Concrete shell penetration depths are calculated as follows:

$x/2d \leq 2.0$

where:

d = Missile diameter = 8 in
x = Missile penetration depth = $[4KNWd^{-0.8}(V/1000)^{1.8}]^{0.5}$

where:

K= Coefficient depending on concrete strength
 = $180/(f_c')^{1/2} = 180/(4000)^{1/2} = 2.846$
N= 1.14 Shape factor for sharp nosed missiles
W= Missile weight = 280 lbs
V= Missile velocity = 126 mph = 185 ft/sec
$x = [(4)(2.846)(1.14)(280)(8^{-0.8})(185/1000)^{1.8}]^{0.5}$
 = 5.75 inches
$x/2d = 5.75/(2)(8) = 0.359 < 2.0$

The minimum concrete shell thickness required to prevent scabbing is three times the predicted penetration depth of 5.75 inches based on the NDRC formula, or 17.25 inches. The concrete cask wall thickness includes 28.25 inches of concrete, which is more than the thickness required to prevent damage due to the penetration missile. This analysis conservatively neglects the 2.5-inch steel shell at the inside face of the concrete shell.

Closure Plate Local Damage Prediction (Penetration Missile)

The concrete cask is closed with a 1.5-inch thick steel plate bolted in place. The following missile penetration analysis shows that the 1.5-inch steel closure plate is adequate to withstand the impact of the 280-lbs armor piercing missile, impacting at 126 mph.

The perforation thickness of the closure steel plate is calculated by the Ballistic Research Laboratories Formula with K = 1, formula number 2-7, in Section 2.2 of Topical Report BC-TOP-9A, Revision 2 [35].

$$T = [0.5 m_m V^2]^{2/3}/672d = 0.523 \text{ inch}$$

where:

T = Perforation thickness
m_m = Missile mass = W/g = 280 lbs/32.174 ft/sec^2 = 8.70 slugs
g = Acceleration of gravity = 32.174 ft/sec^2

BC-TOP-9A recommends that the plate thickness be 25% greater than the calculated perforation thickness, T, to prevent perforation. Therefore, the recommended plate thickness is:

$$T = 1.25 \times 0.523 \text{ in.} = 0.654 \text{ in.}$$

The closure plate is 1.5 inches thick; therefore the plate is adequate to withstand the local impingement damage due to the specified armor piercing missile.

Overall Damage Prediction for a Tornado Missile Impact (High Energy Missile)

The concrete cask is a free-standing structure. Therefore, the principal consideration in overall damage response is the potential of upsetting or overturning the cask as a result of the impact of a high energy missile. Based on the following analysis, it is concluded that the cask can sustain an impact from the defined massive high kinetic energy missile and does not overturn.

From the principle of conservation of momentum, the impulse of the force from the missile impact on the cask must equal the change in angular momentum of the cask. Also, the impulse force due to the impact of the missile must equal the change in linear momentum of the missile. These relationships may be expressed as follows:

Change in momentum of the missile, during the deformation phase

$$\int_{t_1}^{t_2} (F)(dt) = m_m(v_2 - v_1)$$

where:

F	=	Impact impulse force on missile
m_m	=	Mass of missile = 4000 lbs/g = 124 slugs/12 = 10.4 (lbs sec^2 /in)
t_1	=	Time at missile impact
t_2	=	Time at conclusion of deformation phase
v_1	=	Velocity of missile at impact = 126 mph = 185 ft/sec
v_2	=	Velocity of missile at time t_2

The change in angular momentum of the cask, about the bottom outside edge/rim, opposite the side of impact is:

$$\int_{t_1}^{t_2} M_c(dt) = \int_{t_1}^{t_2} (H)(F)(dt) = I_m(\omega_1 - \omega_2)$$

Substituting,

$$\int (F)(dt) = m_m(v_2 - v_1) = \frac{I_m(\omega_1 - \omega_2)}{H}$$

where:

M_c	=	Moment of the impact force on the cask
I_m	=	Concrete cask mass moment of inertia, about point of rotation on the bottom rim
ω_1	=	Angular velocity at time t_1
ω_2	=	Angular velocity at time t_2
m_c	=	Mass of concrete cask = W_c/g = 285,000/32.174
	=	8858.1 slugs/12 = 738.2 lbs sec^2 /in)
I_{mx}	=	Mass moment of inertia, VCC cask about x axis through its center of gravity
	≅	$1/12(m_c)(3r^2 + H^2)$ (Conservatively assuming a solid cylinder.)
	≅	$(1/12)(738.2)\ [(3)(68.0)^2 + (225.88)^2] = 3.99\times10^6$ lbs-sec^2-in
I_m	=	$I_{mx} + (m_c)(d_{CG})^2 = 3.99\times10^6 + (738.2)(126.23)^2 = 15.75\times10^6$ lbs-sec^2-in.
d_{CG}	=	The distance between the cask CG and a rotation point on base rim = 126.23 in.

(See Figure 11.2.11-1.)

Based on conservation of momentum, the impulse of the impact force on the missile is equated to the impulse of the force on the cask.

$$m_m(v_2 - v_1) = I_m(\omega_1 - \omega_2)/H$$

at time t_1, v_1 = 185 ft/sec and ω_1 = 0 rad/sec

at time t_2, v_2 = 0 ft/sec

During the restitution phase, the final velocity of the missile depends upon the coefficient of restitution of the missile, the geometry of the missile and target, the angle of incidence, and on the amount of energy dissipated in deforming the missile and target. On the basis of tests conducted by EPRI, the final velocity of the missile, v_f, following the impact is assumed to be zero. Assuming conservatively that all of the missile energy is transferred to the cask, and equating the impulse of the impact force on the missile to the impulse of the force on the cask,

$$(10.4)(v_2 - 185 \text{ ft/sec} \times 12 \text{ in/ ft}) = 15.75\times10^6 \text{ lbs-sec}^2\text{-in } (0 - \omega_2)/225.88$$

$$\omega_2 = 0.331 \text{ rad/sec (when } v_2 = 0)$$

Back solving for v_2

$$v_2 = 261.6 \times \omega_2 = (261.6)(0.331) = 86.6 \text{ in/sec}$$

where the distance from the point of missile impact to the point of cask rotation is $\sqrt{132.0^2 + 225.88^2} = 261.6$ in. (See Figure 11.2.11-1). The line of missile impact is conservatively assumed normal to this line.

Equating the impulse of the force on the missile during restitution to the impulse of the force on the cask yields:

$$-[m_m(v_f - v_2] = I_m (\omega_f - \omega_2)/H$$

$$-[10.4(0 - 86.6)] = 15.75\times10^6 \text{ lbs-sec}^2\text{-in } (\omega_f - 0.331)/225.88$$

$$\omega_f = 0.344 \text{ rad/sec}$$

where:

v_f = 0
v_2 = 86.6 in/sec
ω_2 = 0.331 rad/sec

Thus, the final energy of the cask following the impact , E_k , is:

$$E_k = (I_m)(\omega_f)^2 /(2) = (15.75\times10^6)(0.344)^2/(2) = 9.32\times10^5 \text{ in-lb}_f$$

The change in potential energy, E_p, of the cask due to rotating it until its center of gravity is above the point of rotation (the condition where the cask will begin to tip-over and the height of the center of gravity has increased by the distance, h_{PE}, see Figure 11.2.11-1) is:

$$E_p = (W_{cask})(h_{PE})$$
$$E_p = 285{,}000 \text{ lbs} \times 17.43 \text{ in}$$
$$E_p = 4.97\times10^6 \text{ in-lb}_f$$

The massive high kinetic energy tornado generated missile imparts less kinetic energy than the change in potential energy of the cask at the tip-over point. Therefore, cask overturning from missile impact is not postulated to occur. The margin of safety, MS, against overturning is:

$$MS = \frac{0.67\times4.97\times10^6}{9.32\times10^5} - 1 = +2.57$$

Combined Tornado Wind and Missile Loading (High Energy Missile)

The cask rotation due to the heavy missile impact is calculated as (See Figure 11.2.11-1 for dimensions):

$$h_{KE} = E_k / W_c = 9.32\times10^5 \text{ in-lb}_f / 285{,}000 \text{ lbs} = 3.27 \text{ in}$$

Then

$$\cos\beta = (h_{CG} + h_{KE}) / d_{CG}$$
$$\cos\beta = (108.8 + 3.27) / 126.23 = 0.8878$$
$$\beta = 27.4 \text{ deg}$$
$$\cos\alpha = 108.8 / 126.23 = 0.8619$$
$$\alpha = 30.5 \text{ deg}$$
$$e = d_{CG} \sin\beta$$
$$e = 126.23 \sin 27.4 = 58.1 \text{ in}$$

Therefore, cask rotation after impact $= \alpha - \beta = 30.5 - 27.4 = 3.1$ deg

The available gravity restoration moment after missile impact:

$= (W_c)(e)$
$= 285{,}000 \text{ lbs} \times 58.1 \text{ in}/12$
$= 1.38 \times 10^6$ ft-lbs >> Tornado Wind Moment $= 3.40 \times 10^5$ ft-lbs

Therefore, the combined effects of tornado wind loading and the high energy missile impact loading will not overturn the cask. Considering that the overturning moment should not exceed two-thirds of the restoring stability moment, the margin of safety, MS, is:

$$MS = \frac{0.67(1.38 \times 10^6)}{3.40 \times 10^5} - 1 = +1.72$$

Local Shear Strength Capacity of Concrete Shell (High Energy Missile)

This section evaluates the shear strength of the concrete at the top edge of the concrete shell due to a high energy missile impact based on ACI 349-85, Chapter 11, Section 11.11.2.1, on concrete punching shear strength.

The force developed by the massive high kinetic energy missile having a frontal area of 20 square feet, is evaluated using the methodology presented in Topical Report, BC-TOP-9A.

$F = 0.625(v)(W_M)$
$F = 0.625(185 \text{ ft/sec})(4{,}000 \text{ lbs}) = 462.5$ kips
$F_u = LF \times F = 1.1 \times 462.5 = 508.8$ kips

Based on a rectangular missile contact area, having proportions of 2 (horizontal) to 1 (vertical) and the top of the area flush with the top of the concrete cask, the required missile contact area based on the concrete punching shear strength (neglecting reinforcing) is calculated as follows.

$V_c = (2+4/\beta_c)\,(f_c')^{1/2} b_o\,d$, where $\beta_c = 2/1 = 2$

$V_c = 4\,(f_c')^{1/2} b_o\,d$

$d = 28.25 \text{ in} - 3.25 \text{ in} = 25 \text{ in}$

$(f_c')^{1/2} = 63.24$ psi, where $f_c' = 4{,}000$ psi

b_o = perimeter of punching shear area at d/2 from missile contact area

$b_o = (2b + 25) + 2(b + 12.5) = 4b + 50$

$V_u = \Phi(V_c + V_s)$, where $V_s = 0$, assuming no steel shear

$V_u = \Phi V_c = \Phi\, 4\,(f_c')^{1/2} b_o\,d = (0.85)(4)(63.24)(4b + 50)(25) = 21{,}501\,b + 268{,}770.$

Setting, V_u equal to F_u and solving for b

$508.8\times10^3 = 21{,}501\,b + 268{,}770$

$b = 11.12$ inches (say 1.0 ft)

The implied missile impact area required $= 2b \times b = 2 \times 1 \times 1 = 2.0$ sq ft < 20.0 sq ft

Thus, the concrete shell alone, based on the concrete conical punching strength and discounting the steel reinforcement and shell, has sufficient capacity to react to the high energy missile impact force.

The effects of tornado winds and missiles are considered both separately and combined in accordance with NUREG-800, Section 3.3.2 II.3.d. For the case of tornado wind plus missile loading, the stability of the cask is assessed and found to be acceptable. Equating the kinetic energy of the cask following missile impact to the potential energy yields a maximum postulated rotation of the cask, as a result of the impact, of 3.0 degrees. Applying the total tornado wind load to the cask in this configuration results in an available restoring moment considerably greater than the tornado wind overturning moment. Therefore, overturning of the cask under the combined effects of tornado winds, plus tornado-generated missiles, does not occur.

Tornado Effects on the Canister

The postulated tornado wind loading and missile impacts are not capable of overturning the cask, or penetrating the boundary established by the concrete cask. Consequently, there is no effect on the canister. Stresses resulting from the tornado-induced decreased external pressure are bounded by the stresses due to the accident internal pressure discussed in Section 11.2.1.

11.2.11.4 Corrective Actions

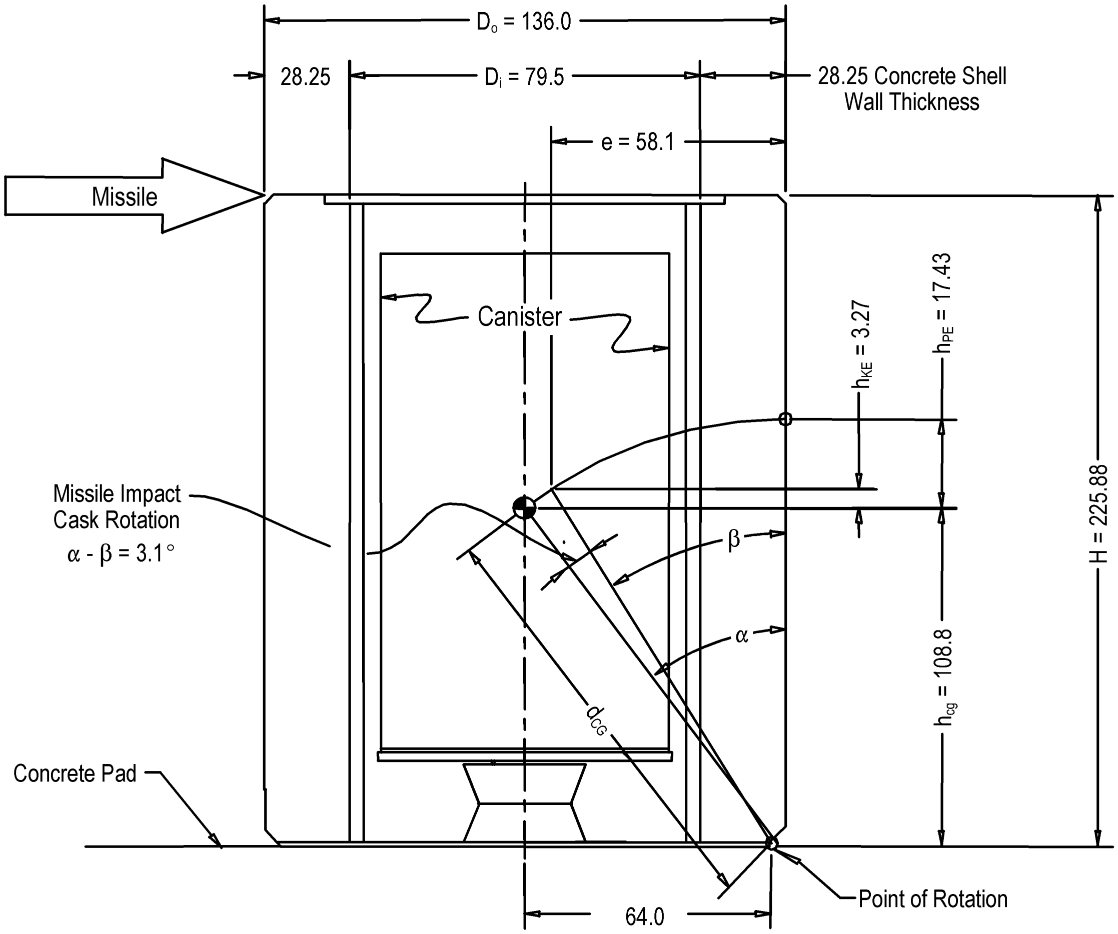
Following the natural phenomenon event, perform the required Response Surveillance in accordance with Section A 5.4 of the Technical Specifications. Corrective actions shall be taken in accordance with the surveillance requirements to return the affected system to a safe operating condition, as applicable to the affected component(s).

Concrete casks shall be restored to operable status in accordance with LCO A 3.1.6 of the Technical Specifications. Optional temperature-monitoring equipment, if used, should be verified as operable, or repaired and returned to service.

11.2.11.5 Radiological Impact

Damage to the vertical concrete cask after a design basis accident does not result in a radiation exposure at the controlled area boundary in excess of 5 rem to the whole body or any organ. The penetrating missile impact is estimated to reduce the concrete shielding thickness, locally at the point of impact, by approximately 6 inches. Localized cask surface dose rates for the removal of 6 inches of concrete are estimated to be less than 250 mrem/hr for the PWR and BWR configurations.

Figure 11.2.11-1 Principal Dimensions and Moment Arms Used in Tornado Evaluation



11.2.12 Tip-Over of Vertical Concrete Cask

Tip-over of the Vertical Concrete Cask (cask) is a non-mechanistic, hypothetical accident condition that presents a bounding case for evaluation. There are no design basis accidents that result in the tip-over of the cask.

Functionally, the cask does not suffer significant adverse consequences due to this event. The concrete cask, canister, and basket maintain design basis shielding, geometry control of contents, and contents confinement performance requirements.

Results of the evaluation show that supplemental shielding will be necessary, following the tip-over and until the cask can be righted, because the bottom ends of the concrete cask and the canister have significantly less shielding than the sides and tops of these components.

11.2.12.1 Cause of Cask Tip-Over

A tip-over of the cask is possible in an earthquake that significantly exceeds the design basis described in Section 11.2.8. No other events related to design bases are expected to result in a tip-over of the cask.

11.2.12.2 Detection of Cask Tip-Over

The tipped-over configuration of the concrete cask will be obvious during site inspection following the initiating event.

11.2.12.3 Analysis of Cask Tip-Over

For a tip-over event to occur, the center of gravity of the concrete cask and loaded canister must be displaced beyond its outer radius, i.e., the point of rotation. When the center of gravity passes beyond the point of rotation, the potential energy of the cask and canister is converted to kinetic energy as the cask and canister rotate toward a horizontal orientation on the ISFSI pad. The subsequent motion of the cask is governed by the structural characteristics of the cask, the ISFSI pad and the underlying soil.

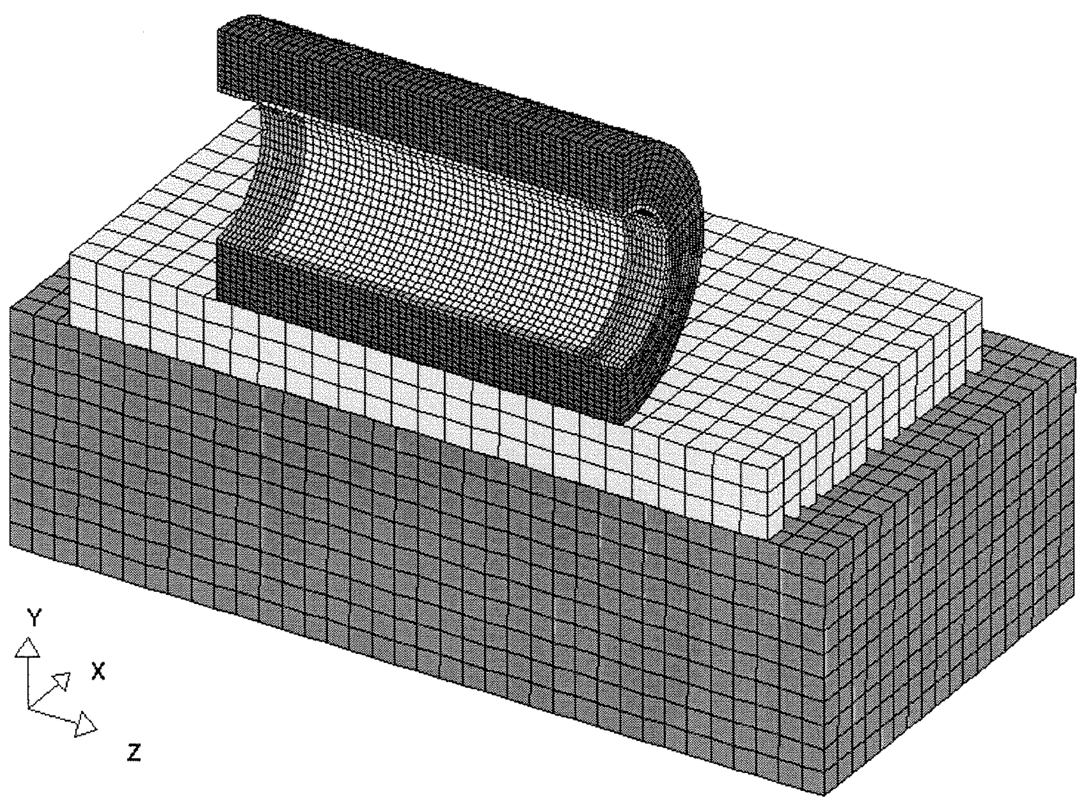
The objective of the evaluation of the response of the concrete cask in the tip-over event is to determine the maximum acceleration to be used in the structural evaluation of the loaded canister and basket (Section 11.2.12.4). The methodology to determine the concrete cask response follows the methodology contained in NUREG/CR-6608, "Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet Onto Concrete Pads" [38]. The LS-DYNA program is used in the evaluation. The validation of the analysis methodology is shown in Section 11.2.12.3.3.

The parameters of the ISFSI pad and foundation are:

Concrete thickness	36 inches maximum
Pad subsoil thickness	10 feet minimum
Specified concrete compressive strength	≤ 5,000 psi per ACI 318
Concrete dry density (ρ)	$125 \le \rho \le 160$ lbs/ft^3
Soil in place density (ρ)	$100 \le \rho \le 160$ lbs/ft^3
Soil Modulus of Elasticity	≤ 60,000 psi (PWR) or ≤ 30,000 psi (BWR)

11.2.12.3.1 Analysis of Cask Tip-Over for PWR Configurations

The finite element model includes a half section of the concrete cask, the concrete ISFSI pad and soil subgrade, as shown:



The concrete pad in the model corresponds to a pad 30-feet by 30-feet square and 3-feet thick, supporting one concrete cask in the center of the pad. The soil under the concrete pad is considered to be 35 feet by 35 feet square and 10 feet thick. Only one-half of the concrete cask, pad and soil configuration is modeled due to symmetry.

The concrete is represented as a homogeneous isotropic material. The concrete cask (outer shell) and the pad are modeled as material Type Number 16 in LS-DYNA. The values for concrete pad and soil properties provided below are typical values for the input to the LS-DYNA model. The material properties used in the model for the concrete ISFSI pad are:

Compressive Strength (f'_c) = 5,000 psi

Density (ρ_c) = 125 pcf

Poisson's Ratio (ν_c) = 0.22 (NUREG/CR-6608 [38])

Modulus of Elasticity (E_c) $= 33\ \rho_c^{1.5}\ \sqrt{f'_c} = 3.261E6$ psi (ACI 318-95)

Bulk Modulus (K_c) $= \dfrac{E_c}{3(1-2\nu_c)} = 1.941E6$ psi (Blevins [19])

The material properties used in the model for the soil below the ISFSI pad are:

Density = 160 pcf

Poisson's Ratio (ν_s) = 0.45 (NUREG/CR-6608)

Modulus of Elasticity = 60,000 psi

The concrete cask steel liner has the properties:

Density = 0.284 lbs/in^3

Poisson's ratio = 0.31

Modulus of elasticity = 2.9E7 psi

To account for the weight of the shield plug, the loaded canister, and the concrete cask pedestal, effective densities are used for the elements in the first row of the steel liner in the model adjacent to the impact plane of symmetry. These densities represent the regions (6° in the circumferential direction) of the steel liner subjected to the weight of the shield plug, the loaded canister and the pedestal, during the side impact (tip-over) condition. The contact angle (6°) is determined based on the canister/basket analysis for the tip-over condition (Section 11.2.12.4).

<u>Boundary Conditions and Initial Conditions</u>

A friction coefficient of 0.25 is used at the interface between the steel liner and the concrete shell, between the concrete cask and the pad, and between the pad and the soil. For all the embedded faces (three side surfaces and the bottom surface) of the soil in the model, the displacements in the direction normal to the surface are restrained. The symmetry boundary conditions are applied for all nodes at the plane of symmetry.

The initial condition corresponds to the concrete cask in a horizontal position with an initial vertical velocity into the concrete pad. The pad and soil are initially at rest.

The distribution of initial velocity of the concrete cask is simulated by applying an angular velocity (ω) to the entire cask. The point of rotation is taken to be the lower edge of the base of the concrete cask. The angular velocity value is computed by considering energy conservation at the cask "center of gravity over corner" tip condition versus the side impact condition.

From energy conservation:

$$mgh = \frac{I\omega^2}{2}$$

where:

mg = conservative, bounding weight of the loaded concrete cask
= 307,000 lbs (PWR Class 1*)
= 319,000 lbs (PWR Class 2*)
= 324,000 lbs (PWR Class 3*)

h = height change of the concrete cask center of gravity (L_{CG}) = $\sqrt{R^2 + \left(\frac{L_{CG}}{2}\right)^2} - R$

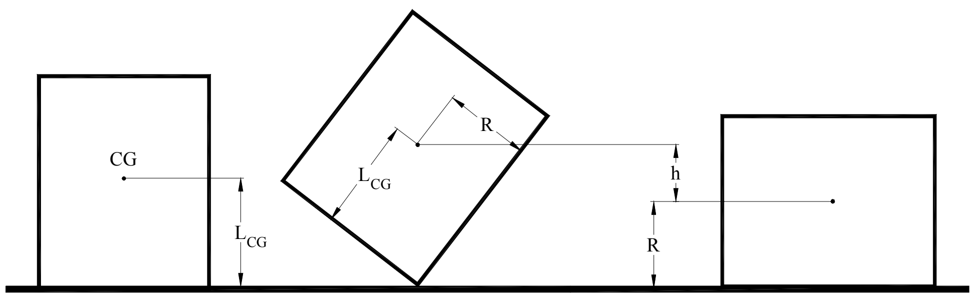
= 60.47 inches (PWR Class 1)
= 63.88 inches (PWR Class 2)
= 67.33 inches (PWR Class 3)

where:

L_{CG} = location of the center of gravity above the pad for the concrete cask
= 109.0 inches (PWR Class 1)
= 113.0 inches (PWR Class 2)
= 117.0 inches (PWR Class 3)

R = radius of the concrete cask = 68 inches

I = total mass moment of inertia of the concrete cask about the point of rotation
= 16,338,092 lbs-sec^2-inch (PWR Class 1)
= 18,091,985 lbs-sec^2-inch (PWR Class 2)
= 19,470,873 lbs-sec^2-inch (PWR Class 3)



The mass moment of inertia for the concrete shell and the steel liner is calculated using the formula for a hollow right circular cylinder (Blevins).

$$I = \frac{m}{12}(3R_1^2 + 3R_2^2 + 4L^2) + md^2$$

where:

m = mass (lbs-sec^2/in)

R_1 and R_2 = the outer and inner radius of the cylinder (inch)

L = height of the cylinder (inch)

d = distance between the center of gravity and the point of rotation (inch)

For the mass of the shield plug, loaded canister and the pedestal, the formula for the moment of inertia for a solid cylinder is used:

$$I = \frac{m}{12}(3R^2 + 4L^2) + md^2$$

where:

m = mass of the cylinder (lbs-sec^2/in)
R = radius of the cylinder (inch)
L = height of the cylinder (inch)
d = distance between the two pivot axes (inch)

The angular velocity is given by $\omega = \sqrt{\dfrac{2mgh}{I}}$

$= 1.51$ radians/sec (PWR Class 1)
$= 1.50$ radians/sec (PWR Class 2)
$= 1.50$ radians/sec (PWR Class 3)

Filter Frequency

The accelerations are evaluated at the inner surface of the cask liner, which physically corresponds to the interface of the liner and the loaded canister nearest the plane of impact. Following the methodology contained in NUREG/CR-6608, the Butterworth filter is applied to the nodal accelerations. The filter frequency is based on the fundamental mode of the cask.

The fundamental natural frequency of a beam in transverse vibration due to flexure only is given by Blevins as:

$$f = \frac{\lambda^2}{2\pi}\sqrt{\frac{EI}{\rho A L^4}}$$

where:

$\lambda = 3.92660231$ for a pin-free beam

The frequencies of the concrete (f_c) and the steel liner (f_s) are computed as:

Area of concrete cask = $\pi\ \{(68)^2 - (39.75)^2\} = 9562.8\ in^2$

Moment of inertia of concrete cask = $\frac{\pi}{4}\{(68)^4 - (39.75)^4\} = 14{,}832{,}070\ in^4$

$$f_c = 823{,}568\,\frac{\lambda^2}{L^2}$$

= 290 Hz (PWR Class 1)
= 267 Hz (PWR Class 2)
= 249 Hz (PWR Class 3)

Area of steel liner = $\pi\ \{(39.75)^2 - (37.25)^2\} = 604.8\ in^2$

Moment of inertia of steel liner = $\frac{\pi}{4}\{(39.75)^4 - (37.25)^4\} = 448{,}673\ in^4$

$$f_s = 861{,}707\,\frac{\lambda^2}{L^2}$$

= 304 Hz (PWR Class 1)
= 279 Hz (PWR Class 2)
= 260 Hz (PWR Class 3)

Since the concrete cask is short compared to its diameter, the contribution of the flexibility due to shear is also incorporated. This is accomplished by using Dunkerley's formula (Blevins). The system frequency is:

$$\frac{1}{f^2} = \frac{1}{f_c^2} + \frac{1}{f_s^2}$$

Thus, the system frequencies are 210 Hz (PWR Class 1), 193 Hz (PWR Class 2), and 180 Hz (PWR Class 3). Cut-off frequencies of 210 Hz (PWR Class 1), 195 Hz (PWR Class 2), and 180 Hz (PWR Class 3) are applied to filter the analysis results and measure the peak accelerations.

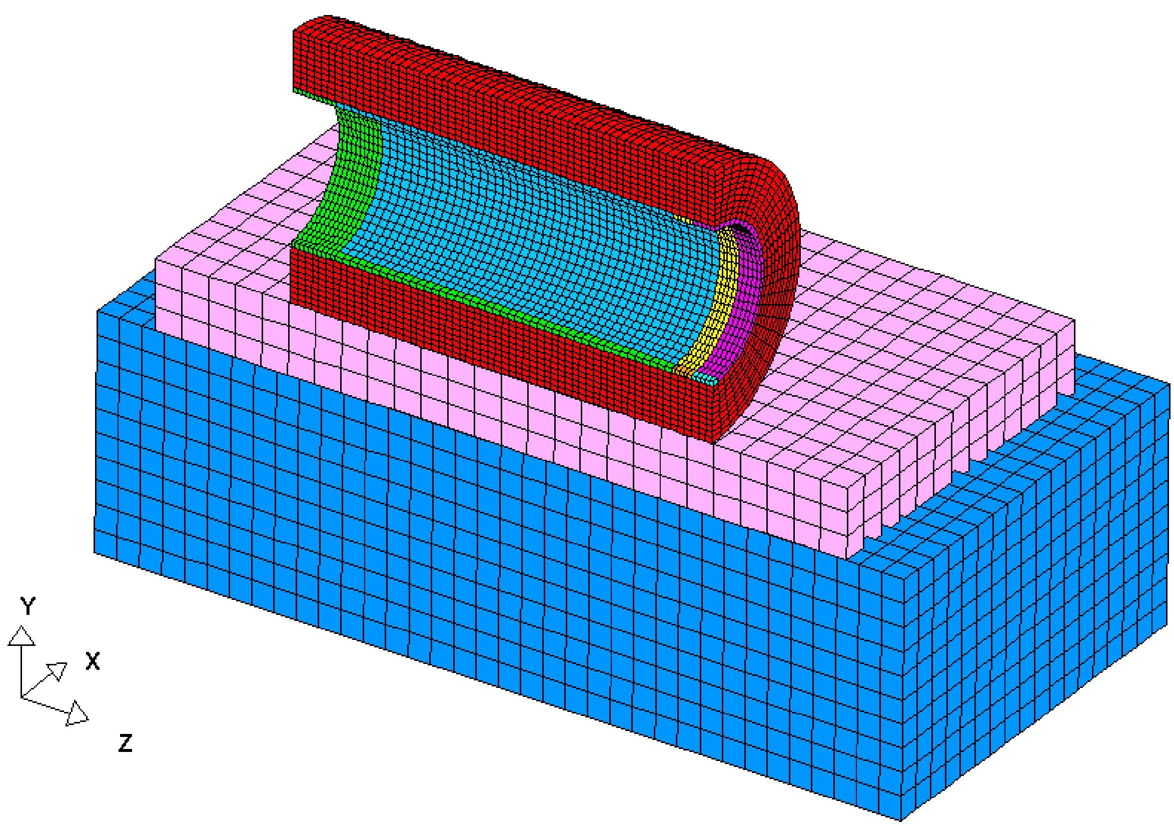
Results of the Transient Analysis

The maximum accelerations at key locations of the concrete cask liner that are required in the evaluation of the loaded canister/basket model (Section 11.2.12.4) are:

Location on Component	Position Measured from the Bottom of the Concrete Cask (inches)			Acceleration (g)		
	PWR Class 1	PWR Class 2	PWR Class 3	PWR Class 1	PWR Class 2	PWR Class 3
Top support disk	176.7	185.2	196.3	30.0	31.3	33.4
Top of the canister structural lid	197.9	207.0	214.6	32.8	34.2	35.7

11.2.12.3.2 <u>Analysis of Cask Tip-Over for BWR Configurations</u>

The BWR finite element model is similar to that for the PWR configuration. The concrete pad in this model corresponds to a pad 30-feet by 30-feet and 3-feet thick, supporting one concrete cask in the center of the pad. The soil under the concrete pad is considered to be 35-feet by 35-feet in area and 10-feet thick.



The material properties used in this model for the soil below the ISFSI pad are the same as those for the PWR model, except the modulus of elasticity of the soil is 30,000 psi.

Initial Conditions

The initial velocity for the BWRs was calculated in the same fashion as for the PWRs, but using the following data:

mg = total weight of the loaded concrete cask
= 322,000 lbs (BWR Class 4)
= 328,000 lbs (BWR Class 5)

h = height change of the concrete cask center of gravity (L_{CG}) = $\sqrt{R^2 + L_{CG}^2} - R$
= 64.74 inches (BWR Class 4)
= 66.46 inches (BWR Class 5)

where:

L_{CG} = location of the center of gravity above the pad for the concrete cask
= 114.0 inches (BWR Class 4)
= 116.0 inches (BWR Class 5)

I = total mass moment of inertia of the concrete cask about the point of rotation
= 18,437,994 lbs-sec^2-inch (BWR Class 4)
= 19,422,461 lbs-sec^2-inch (BWR Class 5)

The angular velocity is given by $\omega = \sqrt{\frac{2mgh}{I}}$
= 1.50 radians/sec (BWR Class 4)
= 1.50 radians/sec (BWR Class 5)

Conservatively, an angular velocity of 1.51 rad/sec is applied to the entire cask of each Class.

Filter Frequency

The filter frequency for the BWRs was calculated in the same fashion as for the PWRs but using the following data:

$$f_c = 823{,}568 \frac{\lambda^2}{L^2}$$

= 263 Hz (BWR Class 4)
= 252 Hz (BWR Class 5)

$$f_s = 861{,}707 \frac{\lambda^2}{L^2}$$
$$= 275 \text{ Hz (BWR Class 4)}$$
$$= 264 \text{ Hz (BWR Class 5)}$$

Thus, the system frequencies are 190 Hz (BWR Class 4), and 182 Hz (BWR Class 5). Cut-off frequencies of 190 Hz (BWR Class 4) and 185 Hz (BWR Class 5) are conservatively applied to filter the analysis results and measure the peak accelerations.

Results of the Transient Analysis

The maximum accelerations at key locations of the concrete cask liner that are required in the evaluation of the loaded canister/basket model (Section 11.2.12.4) are:

Location on Component	Position Measured from the bottom of the Concrete Cask (inches)		Acceleration (g)	
	BWR-4	BWR-5	BWR-4	BWR-5
Top support disk	178.7	182.9	24.2	24.2
Top of the canister structural lid	208.4	213.2	27.9	28.0

11.2.12.3.3 Validation of the Analysis Methodology

Tip-over tests of a steel billet onto a concrete pad were conducted and reported in NUREG/CR-6608. The purpose of the tests was to provide data, against which, analysis methodology could be validated. Using the geometry described in the benchmark along with the modeling methodology, these analyses were re-performed using the LS-DYNA program.

Using the filter frequency reported in the NUREG/CR-6608 benchmark, the following results are obtained:

Nodes / Gauge Location	Maximum Experiment (g)	NAC Analysis (g)
16115 / A1	237.5	237.1
17265 / A5	231.5	229.4

11.2.12.4 Analysis of Canister and Basket for Cask Tip-Over Event

Structural evaluations are performed for the transportable storage canister and fuel basket support disks for tip-over accident conditions for both PWR and BWR fuel configurations. ANSYS finite element models are used to evaluate this side impact loading condition.

Comparison of maximum stress results to the allowable stress intensities shows that the canister and support disks are structurally adequate for the concrete cask tip-over condition and satisfies the stress criteria in accordance with the ASME Code, Section III, Division I, Subsection NB and NG, respectively.

The structural response of the PWR and BWR canisters and fuel baskets to the tip-over condition is evaluated using ANSYS three-dimensional finite element models consisting of the top portion of the canister, the top five fuel basket support disks, and the fuel basket top weldment disk. The PWR with Fuel Class 1 configuration is used to evaluate the PWR canister and fuel basket, and the BWR with Fuel Class 4 configuration is used to evaluate the BWR canister and fuel basket. These two representative configurations are chosen because they bound the maximum load-per-support disk for the respective fuel configurations. For each fuel configuration analyzed, the structural analyses are performed for various fuel basket drop orientations in order to ensure that the maximum primary membrane (P_m) and primary membrane plus primary bending ($P_m + P_b$) stresses are evaluated. For the PWR fuel configuration, fuel basket drop orientations of 0°, 18.22°, 26.28°, and 45° are evaluated (see Figure 11.2.12.4.1-1). For the BWR fuel configuration, fuel basket drop orientations of 0°, 31.82°, 49.46°, 77.92°, and 90° are evaluated (see Figure 11.2.12.4.2-1).

11.2.12.4.1 Analysis of Canister and Basket for PWR Configurations

Four three-dimensional models of the PWR canister and fuel basket are evaluated for side loading conditions that conservatively simulate a tip-over event while inside the concrete cask. In each model, a different fuel basket drop orientation is used. Three-dimensional half-symmetry models are used for the basket orientation of 0° and 45°, since half-symmetry is applicable based on the support disk geometry and the drop orientation. Three-dimensional full-models are used for the basket drop orientations of 18.22° and 26.28°. Representative figures for the models are presented in this section (three-dimensional full-model with a basket orientation of 18.22°).

Model Description

The finite element model used to evaluate the PWR canister and fuel basket for the tip-over event is presented in Figure 11.2.12.4.1-2 through Figure 11.2.12.4.1-5. The figures presented are for the PWR canister and fuel basket model with a fuel basket drop orientation of 18.22° and are representative of the models for all drop orientations analyzed. Only half of the canister is shown in the figures to present the view of the fuel basket.

The canister shell, shield lid, and structural lids are constructed of SOLID45 elements, which have three degrees-of-freedom (UX, UY, and UZ) per node (see Figure 11.2.12.4.1-3). The interaction of the shield lid and structural lid with the canister shell (below the lid welds) is modeled using CONTAC52 elements with a gap size based on nominal dimensions. The interaction of the bottom edge of the shield lid with the support ring is modeled using COMBIN40 gap elements with a gap size of 1×10^{-8} inch. The interaction of the shield and structural lids is modeled using COMBIN40 gap elements with a conservative gap size of 0.08 inch, based on the flatness tolerance of the two lids. The interaction of the canister shell with the inner surface of the concrete cask is modeled using CONTAC52 elements with an initial gap size equal to the difference in the nominal radial dimensions of the outer surface of the canister and the inner surface of the concrete cask. A gap stiffness of 1×10^{6} lbs/inch is assigned to all CONTAC52 and COMBIN40 elements.

The top five fuel basket support disks and top weldment disk are modeled using SHELL63 elements, which have six degrees-of-freedom per node (UX, UY, UZ, ROTX, ROTY, and ROTZ). For the top (first) and fifth support disk, a refined mesh density is used (see Figure 11.2.12.4.1-4). The remaining support disks and the top weldment disk incorporate a course mesh density to account for the load applied to the canister shell. For the fine-meshed support disks, the tie-rod holes are modeled. CONTAC52 elements are included in the slits at the tie-rod holes. The interaction between the fuel basket support disks and top weldment disk and the canister shell is modeled using CONTAC52 elements with an initial gap size based on the nominal radial difference between the disks and canister shell. A gap stiffness of 1×10^{6} lbs/inch is assigned to all CONTAC52 elements.

The lower boundary of the canister shell (near the 5th support disk) is restrained in the axial (Y) direction. For the half-symmetry models (0° and 45° basket drop orientations), symmetry boundary conditions are applied at the plane of symmetry of the model. Since gap elements are used to represent the contact between the canister shell and the inner surface of the concrete cask, the nodes corresponding to the concrete cask are fixed in all degrees of freedom (UX, UY and UZ). In addition, the axial (UY) and in-plane rotational degrees of freedom (ROTX and ROTZ) of the basket nodes are fixed since there is no out-of-plane loading for the support disk for a side impact condition.

Loading of the model includes an internal pressure of 15 psig (design pressure for normal condition of storage) applied to the inner surfaces of the canister, pressure loads applied to the support disk slots, and the inertial loads. The pressure load applied to the support disk slots represents the weight of the fuel assemblies, fuel tubes, and aluminum heat transfer disks multiplied by the appropriate acceleration (see Figure 11.2.12.4.1-5). For the inertial loads, a maximum acceleration of 40g is conservatively applied to the entire model in the X-direction (see Figure 11.2.12.4.1-2) to simulate the side impact during the cask tip-over event.

As shown in Section 11.2.12.3.1, the maximum acceleration of the concrete cask steel liner at the locations of the top support disk and the top of the canister structural lid during the tip-over event is determined to be 33.4g and 35.7g, respectively. To determine the effect of the rapid application of the inertia loading for the support disk, a dynamic load factor (DLF) is computed using the mode shapes of a loaded support disk. The mode shapes corresponding to the in-plane motions of the disk are extracted using ANSYS. However, only the dominant modes with respect to modal mass participation factors are used in computing the DLF. The dominant resonance frequencies and corresponding modal mass participation factors from the finite element modal analyses of the PWR support disk are:

Frequency (Hz)	% Modal Mass Participation Factor
109.7	85.8
370.1	2.7
371.1	7.2

The mode shapes for these frequencies are shown in Figures 11.2.12.4.1-8 through 11.2.12.4.1-10. The displacement depicted in these figures is highly exaggerated by the ANSYS program in order to illustrate the modal shape. The stresses associated with the actual displacement are shown in Tables 11.2.12.4.1-4 through 11.2.12.4.1-8.

Using the acceleration time history of the concrete cask steel liner at the top support disk location developed from Section 11.2.12.3.1, the DLF is computed to be 1.18. Applying the DLF to the 33.4g results in a peak acceleration of 39.4g for the top support disk. The DLFs for the canister lids are considered to be unity since the lids have significant in-plane stiffness and are considered to be rigid (the structural lid is 3 inches thick and shield lid is 7 inches thick). Therefore, applying 40g to the entire canister/basket model is conservative.

A uniform temperature of 75°F is applied to the model to determine material properties during solution. During post processing for the support disk, temperature distribution with a maximum temperature of 700°F (at the center) and a minimum temperature of 400°F (at the outer edge) are conservatively used to determine the allowable stresses. A constant temperature of 500°F is used for the canister to determine the allowable stresses. These temperatures are the bounding temperatures for the normal, off-normal and accident conditions of storage.

Analysis Results for the Canister

The sectional stresses at 13 axial locations of the canister are obtained for each angular division of the model (a total of 80 angular locations for the full-models and 41 angular locations for the half-symmetry models). The locations for the stress sections are shown in Figure 11.2.12.4.1-6.

The stress evaluation for the canister is performed in accordance with the ASME Code, Section III, Subsection NB, by comparing the linearized sectional stresses against the allowable stresses. Allowable stresses are conservatively taken at a temperature of 500°F, except that 300°F and 250°F are used for the shield lid weld (Section 10) and the structural lid weld (Section 11). The calculated maximum temperatures for the shield lid and structural lid are 212°F and 204°F, respectively (Table 4.4.3-1). The allowable stresses for accident conditions are taken from Subsection NB as shown below. S_m and S_u are 14.8 ksi and 57.8 ksi, respectively, for Type 304L stainless steel (canister shell and structural lid). S_m and S_u are 17.5 ksi and 63.5 ksi, respectively, for Type 304 stainless steel (shield lid).

Stress Category	Accident (Level D) Allowable Stress
P_m	Lesser of 0.7 S_u or 2.4 S_m
P_m+P_b	Lesser of 1.0 S_u or 3.6 S_m

The primary membrane and primary membrane plus bending stresses for the PWR configuration for a 45° basket drop orientation are summarized in Table 11.2.12.4.1-1 and Table 11.2.12.4.1-2, respectively. The stress results for the canister are similar for all four basket drop orientation evaluations. The 45° basket orientation results are presented because this drop orientation results in the minimum margins of safety in the canister.

During the tip-over accident, the canister shell at the structural and shield lids is subjected to the inertial loads of the lids, which results in highly localized bearing stresses (Sections 7 through 9 at angular locations of approximately ± 4.5 degrees from the impact location). This stress is predominant because the weights of the structural and shield lids are transferred to the canister shell near these section locations. According to ASME Code Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions. Therefore, the stresses are not presented for the lid-bearing regions of the canister shell (Sections 7 through 9) in Tables 11.2.12.4.1-1 and 11.2.12.4.1-2. The stresses at the structural lid/canister shell weld region (Section 11) are determined by averaging the stresses over the impact region where the weld is in compression in the radial direction ($\sigma_x \leq 0.0$ psi). In accordance with ISG-15, Revision 0 [60], a 0.8 weld reduction factor is applied to the allowable stresses for the structural lid / canister shell weld. Use of the 0.8 factor is valid because the ultimate tensile strength of the weld material exceeds the base metal strength.

The stress evaluation results for the tip-over accident condition show that the minimum margin of safety in the canister for the PWR configuration is +0.29 for P_m stresses (Section 11). For P_m+P_b stresses, the margin of safety at is +0.64 (Section 11).

Analysis Results for the Support Disks

To evaluate the most critical regions of the support disk, a series of cross sections are considered. To aid in the identification of these sections, Figure 11.2.12.4.1-7 shows the locations on a support disk for the full-models. Table 11.2.12.4.1-3 lists the cross sections versus Point 1 and Point 2, which spans the cross section of the ligament in the plane of the support disk. Note that a local coordinate system (x and y parallel to the support disk ligaments) is used for the stress evaluation.

The stress evaluation for the support disk is performed according to ASME Code, Section III, Subsection NG. According to this subsection, linearized sectional stresses are to be compared against the allowable stresses. The allowable stresses for tip-over accident conditions are taken from Subsection NG as shown below, at the temperature of the Section. The temperature distribution of the disk is determined by a thermal conduction solution for a single disk with the maximum temperature of 700°F specified at the center and the minimum temperature of 400°F specified at the outer edge as boundary conditions.

Stress Category	Accident (Level D) Allowable Stresses
P_m	Lesser of 0.7 S_u or 2.4 S_m
P_m+P_b	Lesser of 1.0 S_u or 3.6 S_m

The shield lid and structural lid provide additional stiffness to the upper portion of the canister shell, which limits the shell and support disk deformations. Therefore, the maximum $P_m + P_b$ stress, and the minimum margin of safety, occur in the 5th support disk (from the top of the basket), where the stiffness effect of the shield and structural lids is not present.

The stress evaluation results for the 5th support disk for the tip-over condition are summarized in Table 11.2.12.4.1-4 for the four basket drop orientations evaluated. As shown in Table 11.2.12.4.1-4, the 26.28° drop orientation case generates the minimum margin of safety in the support disk; therefore, the P_m and $P_m + P_b$ stress intensities for the 26.28° basket drop orientation case are presented in Tables 11.2.12.4.1-6 and 11.2.12.4.1-7, respectively. These tables list stress results with the 30 lowest margins of safety for the 5th support disk. The highest P_m stress occurs at Section 18, with a margin of safety of +0.97 (See Table 11.2.12.4.1-6 for stresses and Figure 11.2.12.4.1-7 for section locations). The highest P_m+P_b stress occurs at Section 61, with a margin of safety of +0.05 (see Table 11.2.12.4.1-7 for stresses and Figure 11.2.12.4.1-7 for section locations).

Support Disk Buckling Evaluation

For the tip-over accident, the support disks experience in-plane loads. The in-plane loads apply compressive forces and in-plane bending moments on the support disk. Buckling of the support disk is evaluated in accordance with the methods and acceptance criteria of NUREG/CR-6322 [39]. Because the ASME Code identifies 17-4PH disk material as ferritic steel, the formulas for non-austenitic steel are used.

The buckling evaluation of the support disk ligaments is based on the Interaction Equations 31 and 32 in NUREG/CR-6322. These two equations adopt the "Limit Analysis Design" approach. Other equations applicable to the calculations are noted as they are applied. The maximum forces and moments for the tip-over accident are based on the finite element analysis stress results.

Symbols and Units

- P = applied axial compressive load, kip
- M = applied bending moment, kip-inch
- P_a = allowable axial compressive load, kip
- P_{cr} = critical axial compression load, kip
- P_e = Euler buckling loads, kip

P_y = average yield load, equal to profile area times specified minimum yield stress, kips (for normal operating condition)

C_c = column slenderness ratio separating elastic and inelastic buckling

C_m = coefficient applied to bending term in interaction equation

M_m = critical moment that can be resisted by a plastically designed member in the absence of axial load, kip-in.

M_p = plastic moment, kip-in.

F_a = axial compressive stress permitted in the absence of bending moment, ksi

F_e = Euler stress for a prismatic member divided by factor of safety, ksi

k = ratio of effective column length to actual unsupported length

l = unsupported length of member, in.

r = radius of gyration, in.

S_y = yield stress, ksi

A = cross sectional area of member, in^2

Z_x = plastic section modulus, in^3

λ = allowable reduction factor, dimensionless

From NUREG/CR-6322, the following equations are used to evaluate the support disk:

$$\frac{P}{P_{cr}} + \frac{C_m M}{M_m\left[1 - \frac{P}{P_e}\right]} \le 1.0 \qquad \text{(Equation 31)}$$

$$\frac{P}{P_y} + \frac{M}{1.18\,M_p} \le 1.0 \qquad \text{(Equation 32)}$$

where:

$$P_{cr} = 1.7 \times A \times F_a$$

$$F_a = \frac{P_a}{A} \quad \text{for } P_a = P_y\left[\frac{1 - \frac{\lambda^2}{4}}{1.11 + 0.5\lambda + 0.17\lambda^2 - 0.28\lambda^3}\right]$$

and $\lambda = \frac{1}{\pi}\left(\frac{kl}{r}\right)\sqrt{\frac{S_y}{E}}$ (accident conditions)

$$Pe = 1.92 \times A \times Fe$$

$$F_e = \frac{\pi^2 \cdot E}{1.3\left(\frac{k \cdot l}{r}\right)^2} \qquad \text{(Level D–Accident)}$$

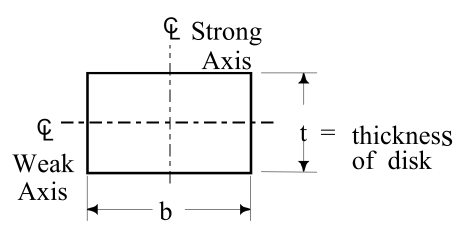
$$P_y = S_y \times A$$

$C_m =$ 0.85 for members with joint translation (sideways)

$$M_p = S_y \times Z_x$$

$$M_m = M_p \cdot \left(1.07 - \frac{\left(\frac{1}{r}\right) \cdot \sqrt{S_y}}{3160}\right) \le M_p$$

Buckling evaluation is performed in all sections in the disk ligaments defined in Figure 11.2.12.4.1-7. Using the cross-sectional stresses calculated at each section located in the ligament for each loading condition, the maximum corresponding compressive force (P) and bending moment (M) are determined as:



$$P = \sigma_m A$$

$$M = \sigma_b S$$

where, σ_m is the membrane stress, σ_b is the bending stress, A is the area (b × t), and S is the section modulus ($tb^2/6$). Note that the strong axis bending is considered in the buckling evaluation since the disk is only subjected to in-plane load during the tip-over event.

To determine the margin of safety:

$$P_1 = P/P_{cr} \qquad M_1 = \frac{C_m M}{(1 - P/P_e) M_m} \qquad (P_1 + M_1 \leq 1)$$

and

$$P_2 = P/P_y \qquad M_2 = \frac{M}{1.18\, M_p} \qquad (P_1 + M_1 \leq 1)$$

The margins of safety are:

$$MS1 = \frac{1}{P_1 + M_1} - 1$$

and

$$MS2 = \frac{1}{P_2 + M_2} - 1$$

The support disk buckling evaluation results for the 5th support disk (the 5th support disk experiences the highest stresses) for the tip-over impact condition are summarized in Table 11.2.12.4.1-5 for the four basket drop orientations evaluated. As shown in Table 11.2.12.4.1-5, the 26.28° case generates the minimum margin of safety for buckling; therefore, the results of the buckling analysis for the 26.28° basket drop orientation case are presented in Table 11.2.12.4.1-8. This table presents the 30 minimum margins of safety for this drop orientation. As the tables demonstrate, the support disks meet the requirements of NUREG/CR-6322.

Fuel Tube Analysis

The fuel tube provides structural support and a mounting location for neutron absorber plates. The fuel tube does not provide structural support for the fuel assembly. To ensure that the fuel tube remains functional during a tip-over accident, a structural evaluation of the tube is performed for a side impact assuming a deceleration of 60g. This g-load bounds the maximum g-load (40g) calculated to occur for the PWR basket in a vertical concrete cask tipover event.

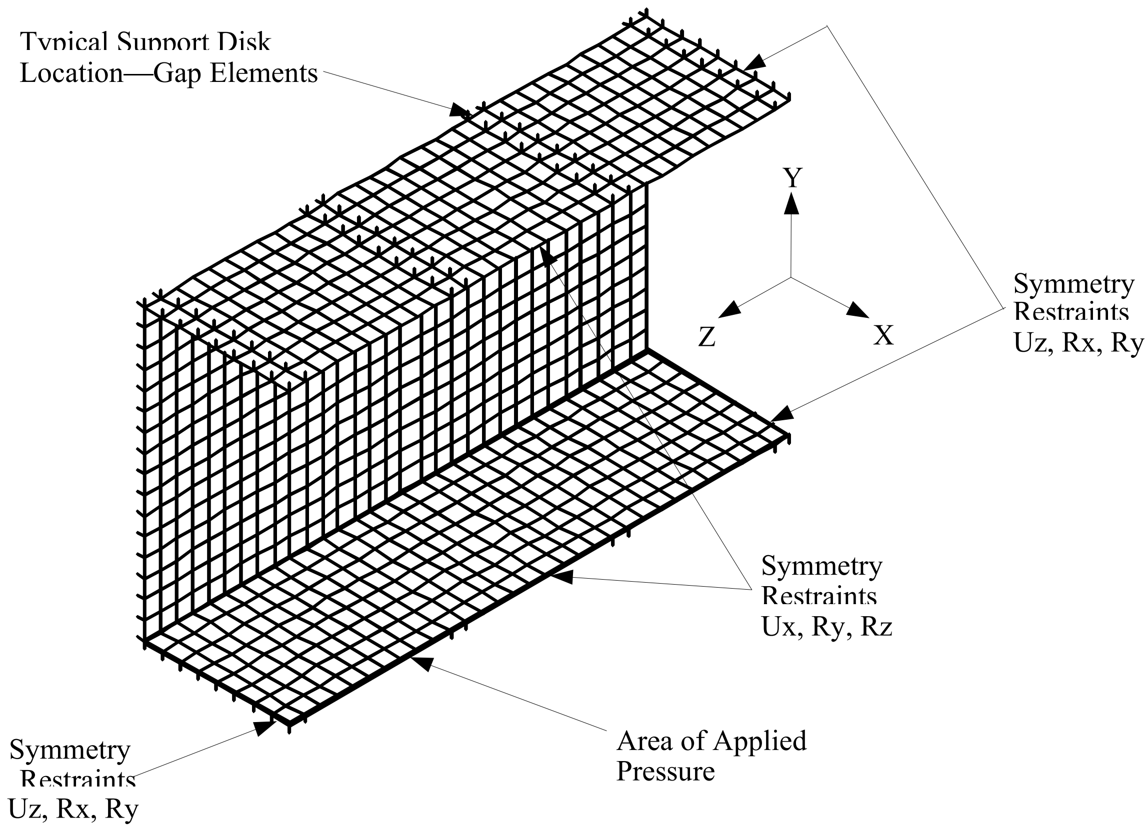
In the tip-over event, the stainless steel support disks in the fuel basket support the fuel tube. The fuel basket support disks, which support the full length of the fuel tube, are spaced 4.42-inches apart (which is less than one half of the fuel tube width of 8.8 inch). Considering the fuel tube subjected to a maximum PWR fuel assembly weight of 1,602 pounds with a 60g load factor and the 30 support locations provided by the basket support disks, the fuel tube shear stress is calculated as:

Shear load = (60g)(1,602)/30 = 3,204 lbs
Area = (0.048)(8.8)(2) = 0.845 in^2
Shear Stress = 3,204/0.845 = 3,792 psi

The yield strength of the tube material, Type 304 stainless steel, is 17,300 psi at 750°F. Conservatively, using the allowable shear stress as one-half the yield strength of the tube material (8,650 psi) results in a large positive margin of safety. Conservative evaluation of the tube loading resulting from its own mass during a side-impact shows that the tube structure maintains position and function.

The load transfer of the weight of the fuel assembly to the fuel basket support disk in the side impact is through direct bearing and compression of the distributed load of the fuel assembly through the fuel tube to the support disk web. Two load conditions are considered in the fuel tube evaluation. The first considers the fuel assembly load as a distributed pressure on the inside surface of the fuel tube. The second postulates that the fuel assembly grid is located at the center of the span between the support disks and produces a localized distributed load over the effective area of the grid.

Two different ANSYS finite element models of the tube are developed for these two load conditions since the fuel tube structural performance for either load is nonlinear. As shown below, the first model represents a fuel tube section with a length of three spans, i.e., the model is supported at four locations by support disks. The model conservatively considers the fuel tube wall thickness of 0.048 inch as the only material subjected to a distributed pressure load representative of the fuel assembly deceleration of 60g. Fuel assembly stiffness is not considered in the development of the imposed pressure load on the fuel tube.



The tube is modeled with the ANSYS plastic, quadrilateral shell element (SHELL43). The support disks are represented by gap elements (CONTAC52). The outer nodes of the gap elements are fully restrained in all three translational directions. Edge restraints were applied to the model to represent symmetry boundary conditions. The effective load on the fuel tube due to the 60g deceleration of the fuel assembly is applied as a pressure to the inside area of the fuel tube.

The finite element analysis results show that the maximum stress in the tube is 23.8 ksi, which is local to the sections of the tube resting on the support disks. At 750°F the ultimate strength for Type 304 stainless steel is 63.1 ksi. The margin of safety is

$$MS = \frac{63.1}{23.8} - 1 = +1.65$$

The analysis shows that the maximum total strain is 0.026 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F [42], the resulting margin of safety is:

$$MS = \frac{0.40/2}{0.026} - 1 = +\text{large}$$

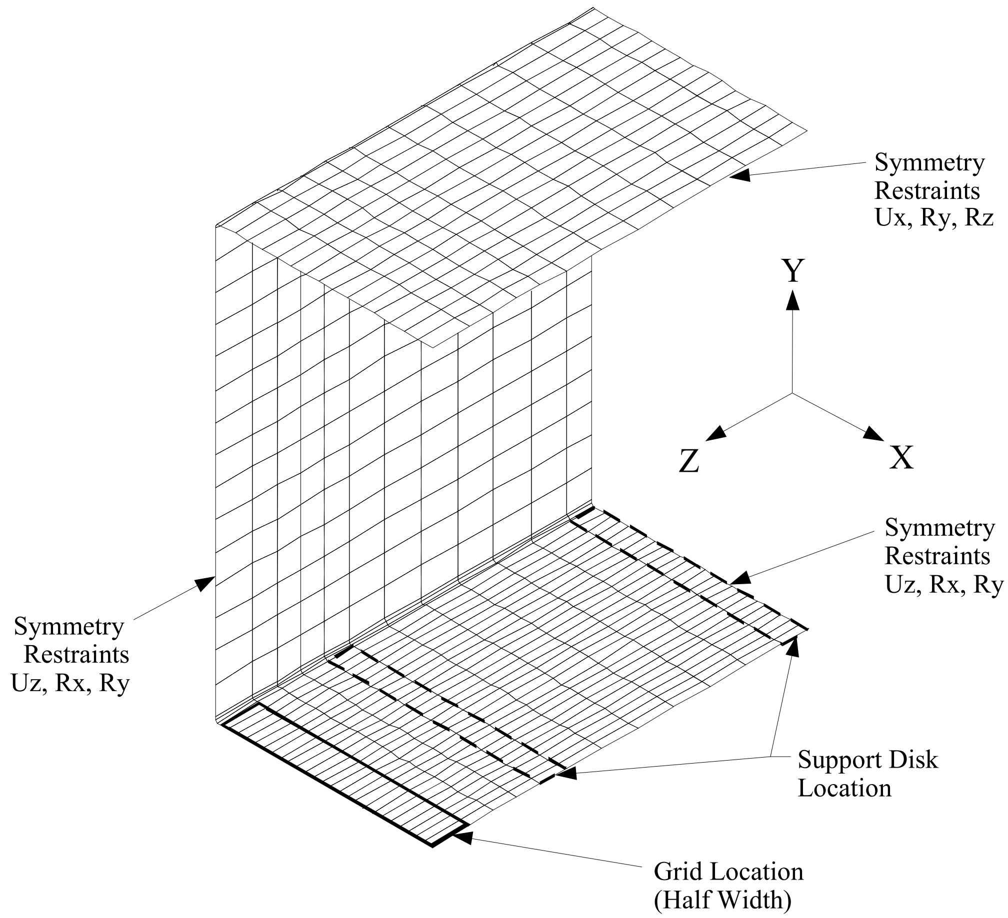
Similarly, the margin of safety for elastic-plastic stress becomes

$$MS = \frac{63.1 - 17.3}{23.8 - 17.3} - 1 = 6.05$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

The second finite element model is used to evaluate the load condition with the fuel assembly grid located at the center of the span between two support disks. The fuel tube is subjected to a localized distributed load over the effective area of the grid. As shown below, the model is a quarter-symmetry periodic section of the fuel tube. As in the finite element model used for the distributed pressure case, this model conservatively considers a fuel tube wall thickness of 0.048 inch. The neutron absorber plate (0.075 inch) and stainless steel cover plate (0.018 inch) are conservatively not included in the model. The tube wall is modeled with ANSYS SHELL43 elements. The support disks are modeled with CONTAC52 elements.

Based on the Lawrence Livermore evaluation of the fuel rods for a side impact (UCID-21246), the fuel rods and fuel assemblies maintain their structural integrity during the side impact resulting from a cask tip-over accident and the displacement of the fuel tube is limited. The maximum displacement of the fuel tube section between the support disks will not exceed the “thickness” of the grid spacer, which is the distance between the outer surface of the grid and the outer surface of the fuel rod array. When the displacement of the fuel tube reaches the “thickness” of the grid spacer, the fuel rods will be in contact with the inner surface of the fuel tube and the weight of the fuel rods will be transferred through the tube wall to the support disks. Therefore, a bounding load condition for this model is simulated by applying a constant displacement of 0.08 inch in the negative Y direction to the nodes corresponding to the grid location in the model. Note that 0.08 inch displacement bounds all PWR fuel assemblies. It is assumed that the fuel assembly grid spacer is rigid and therefore a constant displacement is conservatively applied.



The finite element analysis results show that the maximum stress in the tube is 38.4 ksi, which is local to the corner of the tube at the grid spacer location of the model close to the side wall of the tube. At 750°F the ultimate strength for Type 304 stainless steel is 63.1 ksi. The margin of safety is

$$MS = \frac{63.1}{38.4} - 1 = +0.64$$

The analysis shows that the maximum total strain is 0.11 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F [42], the resulting margin of safety is:

$$MS = \frac{0.40/2}{0.11} - 1 = 0.82$$

Similarly, the margin of safety for elastic-plastic stress becomes

$$MS = \frac{63.1 - 17.3}{38.4 - 17.3} - 1 = 1.17$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

Both the maximum total strain and the elastic-plastic stress analyses indicate that the tube position within the support basket is maintained.

Fuel Tube Yielding

Using the displacement of the fuel rod, a check of the fuel tube is performed to verify that the fuel tube remains elastic during a side-drop. The fuel rod displacement loading is a more realistic loading condition because the load is transmitted from the fuel rods to the fuel tube. The analysis is conservative as it assumes the cumulative displacement of 17 fuel rods (stacked on top of each other) in a 17×17 PWR fuel assembly.

The displacement of a single fuel rod assumed as a four-span continuous beam is calculated as:

$$\Delta_{max} = 0.0065 \frac{wL^4}{EI} = 2.2014 \times 10^{-5} \text{ in}$$

where:

w = mass/length = $\rho_{zirc} A_{zirc} + \rho_{UO2} A_{UO2}$ = 0.0404 lb/in × 17 rods = 0.6868 lb/in

Rod OD = 0.379 in

Rod ID = 0.379-2 × 0.024 = 0.331 in

Rod Density (Zirc-4) = ρ_{zirc} = 0.237 lb/in^3

Rod Area = $A_{zirc} = \frac{\pi}{4}(0.379^2 - 0.331^2)$ = 0.0268 in^2

UO_2 Density = ρ_{UO2} = 0.396 lb/in^3

UO_2 Area = $A_{UO2} = \frac{\pi}{4} x 0.331^2 = 0.086 \text{ in}^2$

L = Distance between support disks = 4.42 in

$E_{zirc} = 10.75 \times 10^6$ psi

$$I_{zirc} = \frac{\pi}{64}(0.379^4 - 0.331^4) = 4.236 \text{ x } 10^{-4} \text{ in}^4 \text{ x } 17 \text{ rods} = 0.0072 \text{ in}^4$$

Using the E_{zirc} and I_{zirc} as conservative assumptions, the maximum displacement is estimated as 2.2014×10^{-5} in. For 60g acceleration, this displacement becomes 1.321×10^{-3} inch.

Applying the displacement midway between support disks, the maximum stress intensity is 12,062 psi. The yield stress for the fuel tube (Type 304 stainless steel) is 17,300 psi at 750°F degrees; therefore, during a 60g side-drop, the fuel tube remains elastic.

Assurance that the neutron absorber remains attached to the fuel tube is evaluated by considering that loads produced by the neutron absorber plate and stainless steel attachment plate, assuming a 60g load, are carried by the attachment plate weld. Total load and resultant stress on the weld are calculated as:

$F_{b/ss} = (g)(\rho)(t)(w)(l)$ Load exerted by neutron absorber/stainless steel attachment plate

where:

g = acceleration (g)

ρ = density of material (lb/in^3) (The density of aluminum (0.098 lb/in^3) is conservatively used for the neutron absorber.

t = thickness of material (in.)

w = width of material (in.)

l = length of material section (in.)

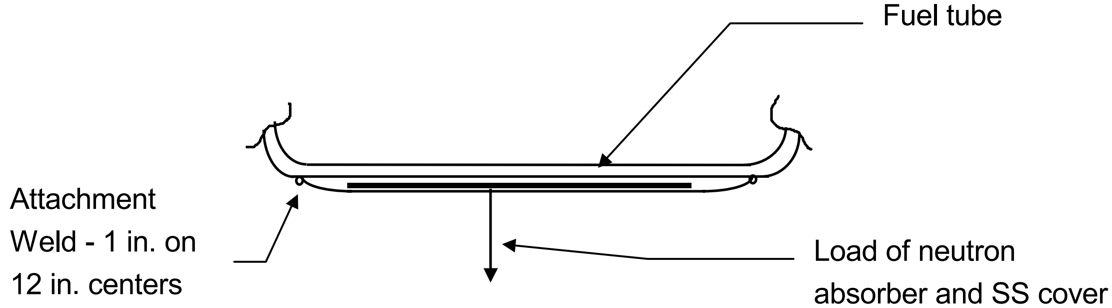
The forces on the weld due to a 12-inch section of neutron absorber (F_b) and a 12-inch section of stainless steel plate (F_{ss}) are:

F_b = (60g)(0.098 lb/in^3)(0.075 in.)(8.2 in.)(12 in.)
= 43.4 lbs

F_{ss} = (60g)(0.291 lb/in^3)(0.018 in.)(8.7 in.)(12 in.)
= 32.8 lbs

The total load (F_t) on a 1-inch attachment weld for a 12-inch section is:

$$F_t = 43.4 \text{ lbs} + 32.8 \text{ lbs} = 76.2 \text{ lbs}$$



The resulting weld stress is: $\sigma = P/A = (76.2 \text{ lb}/2) / (1 \text{ in.}) (0.018 \text{ in.}) = 2{,}117$ psi

Since the weld material is Type 304 stainless steel, the margin of safety (at 750°F) is:

$$MS = \frac{17{,}300}{2{,}117} - 1 = +7.2$$

Therefore, the neutron absorber remains enclosed on each outer surface of the fuel tube wall.

Figure 11.2.12.4.1-1 Basket Drop Orientations Analyzed for Tip-Over Conditions - PWR

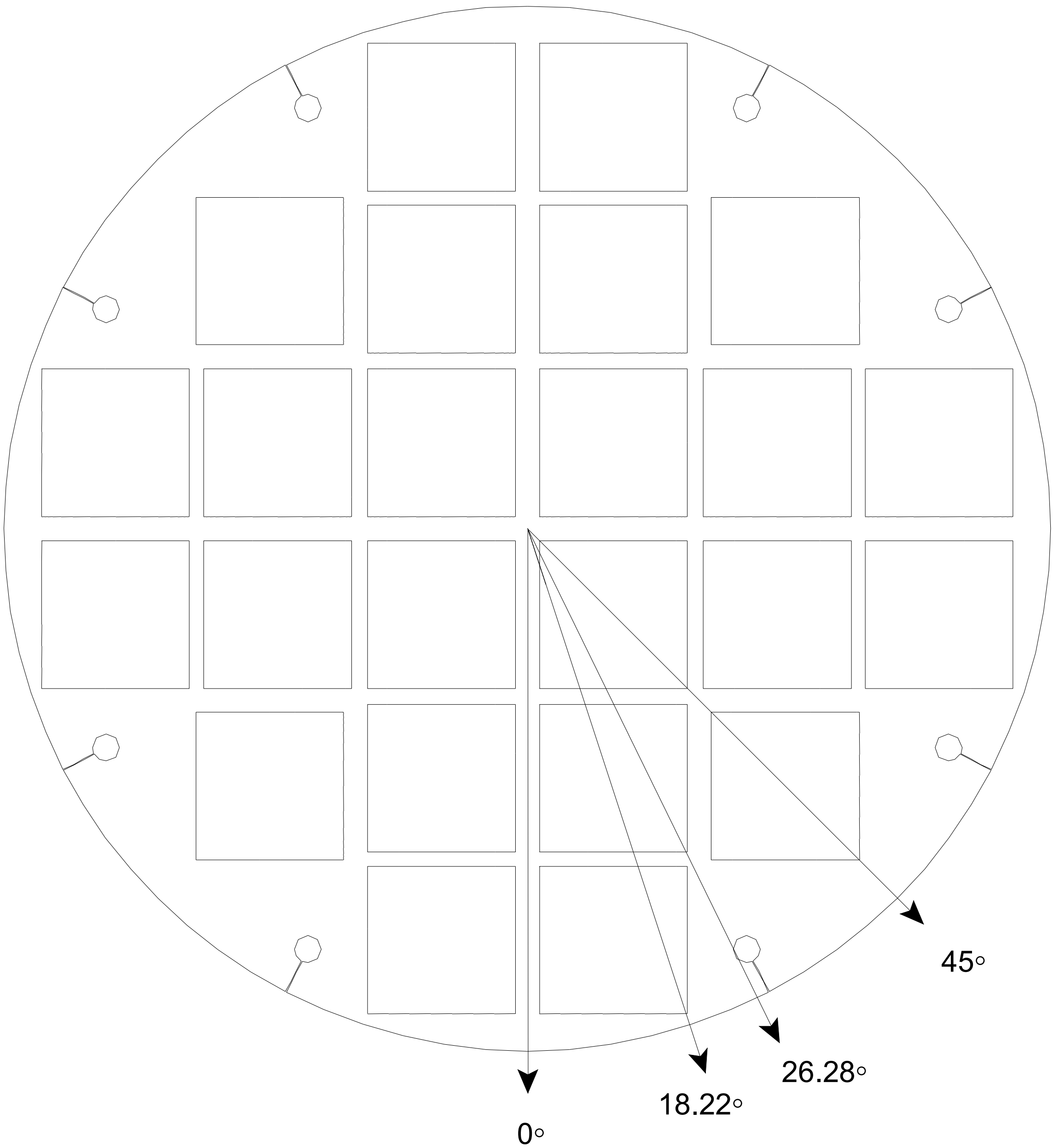
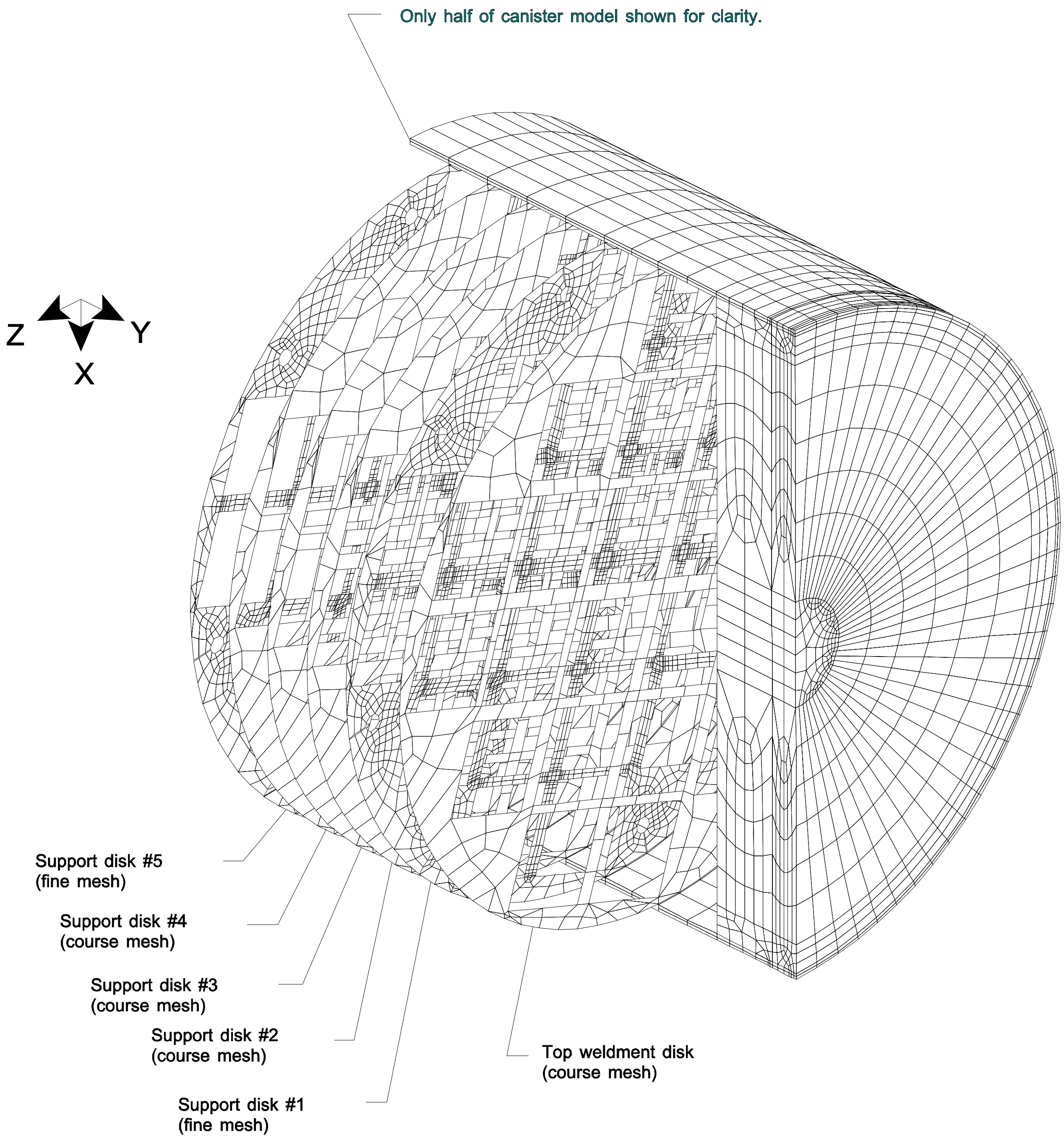
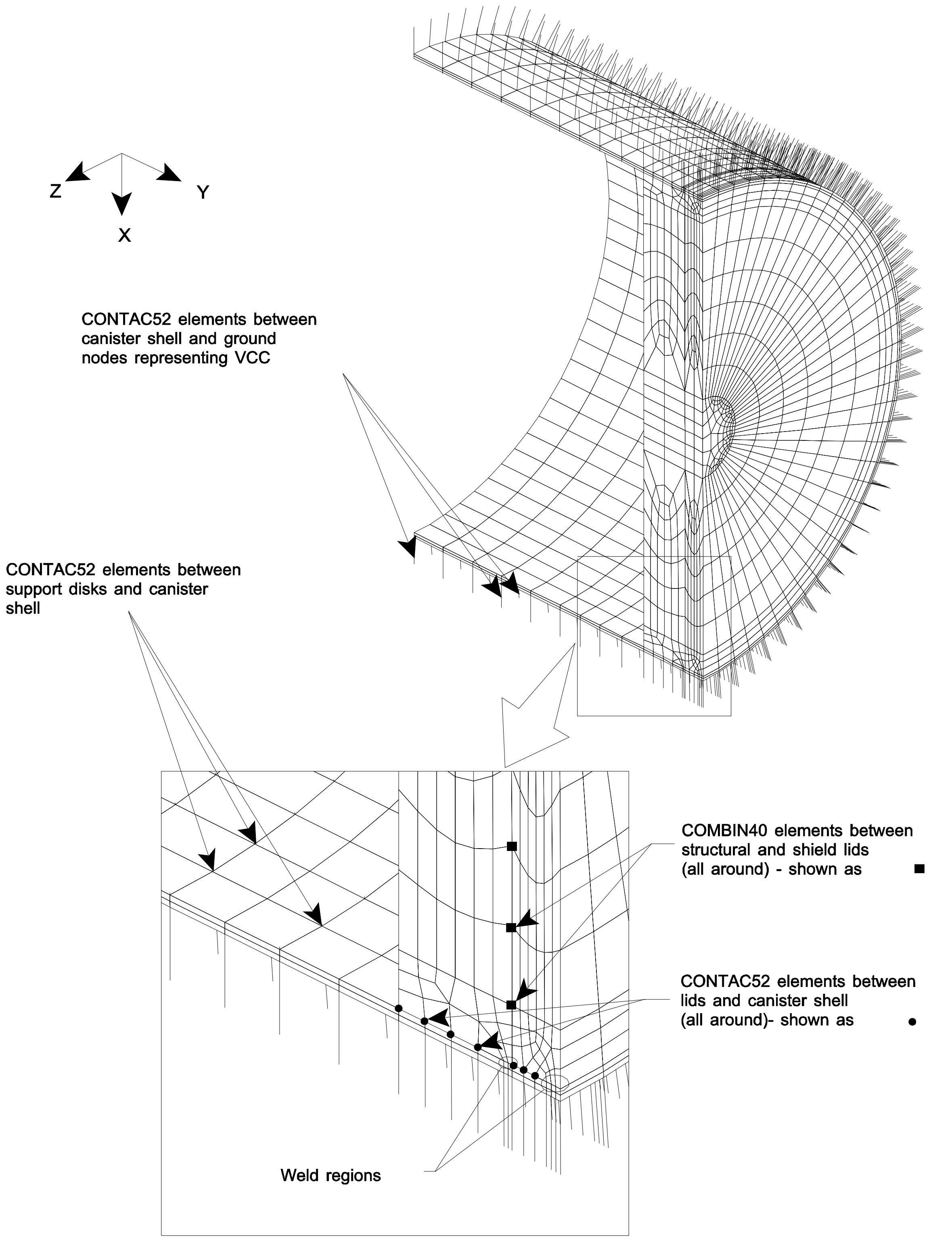


Figure 11.2.12.4.1-2 Fuel Basket/Canister Finite Element Model - PWR



18.22° Basket Drop Orientation

Figure 11.2.12.4.1-3 Fuel Basket/Canister Finite Element Model - Canister



Only Half of Canister Shown for Clarity

Figure 11.2.12.4.1-4 Fuel Basket/Canister Finite Element Model - Support Disk - PWR

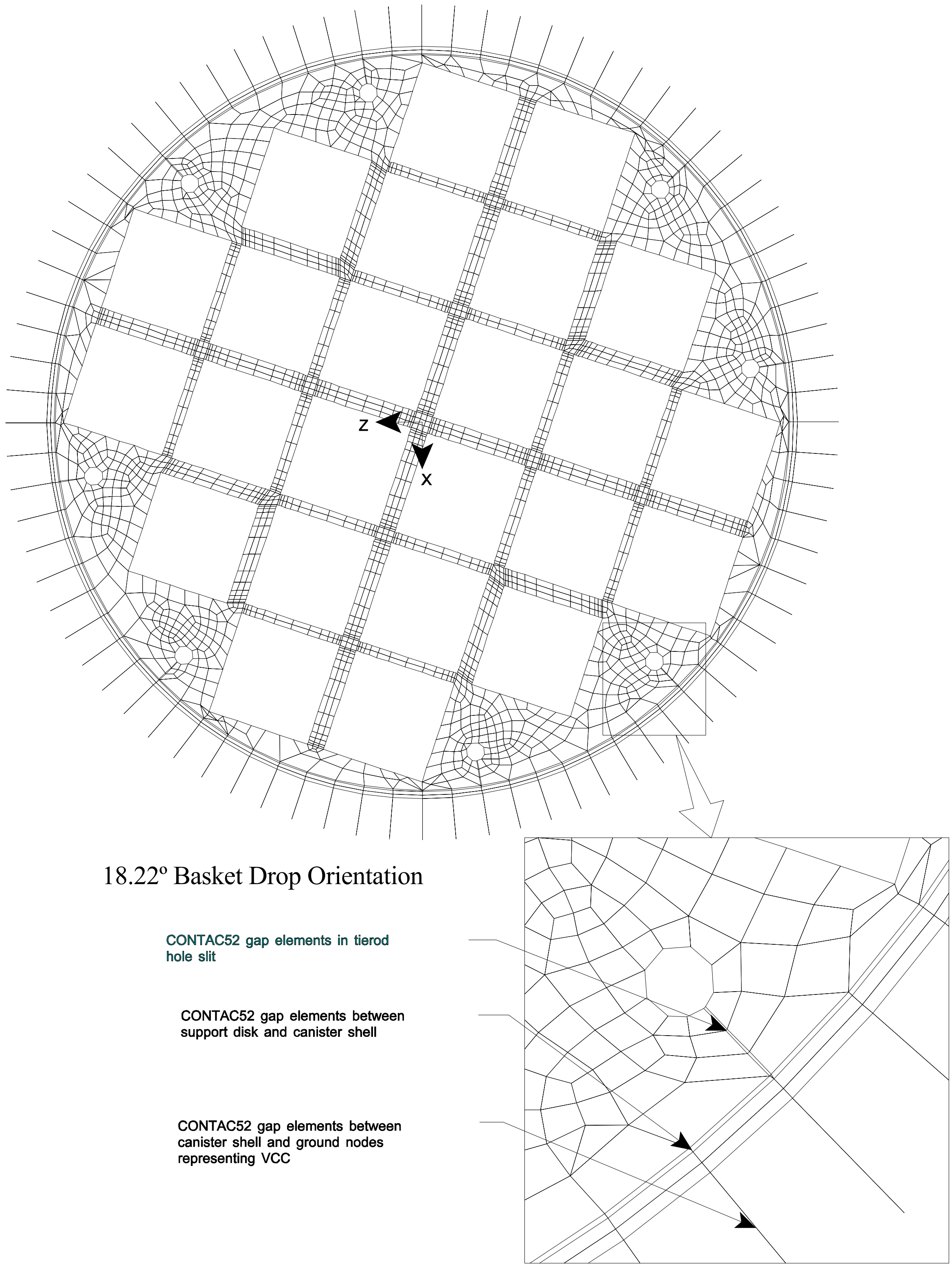
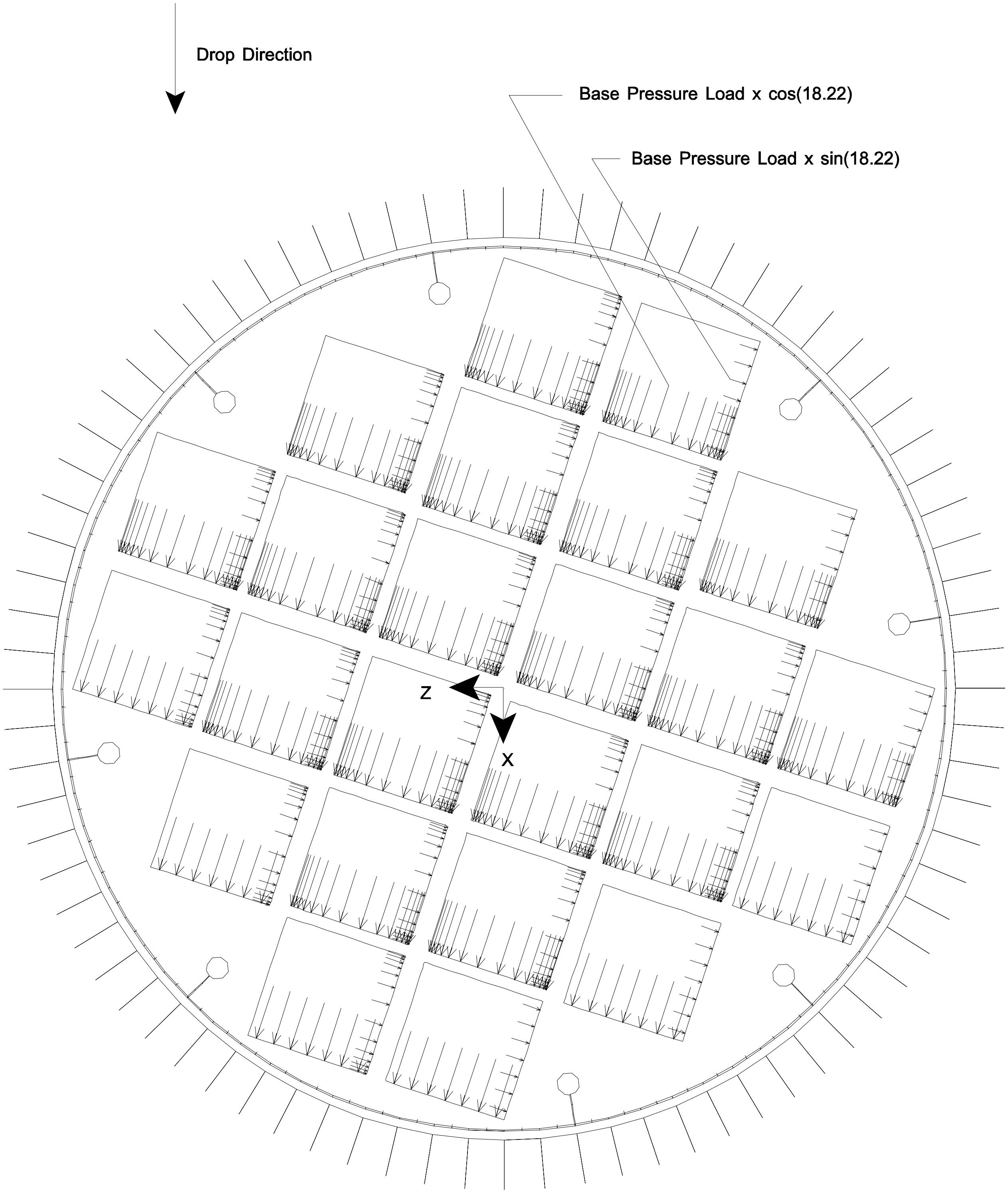
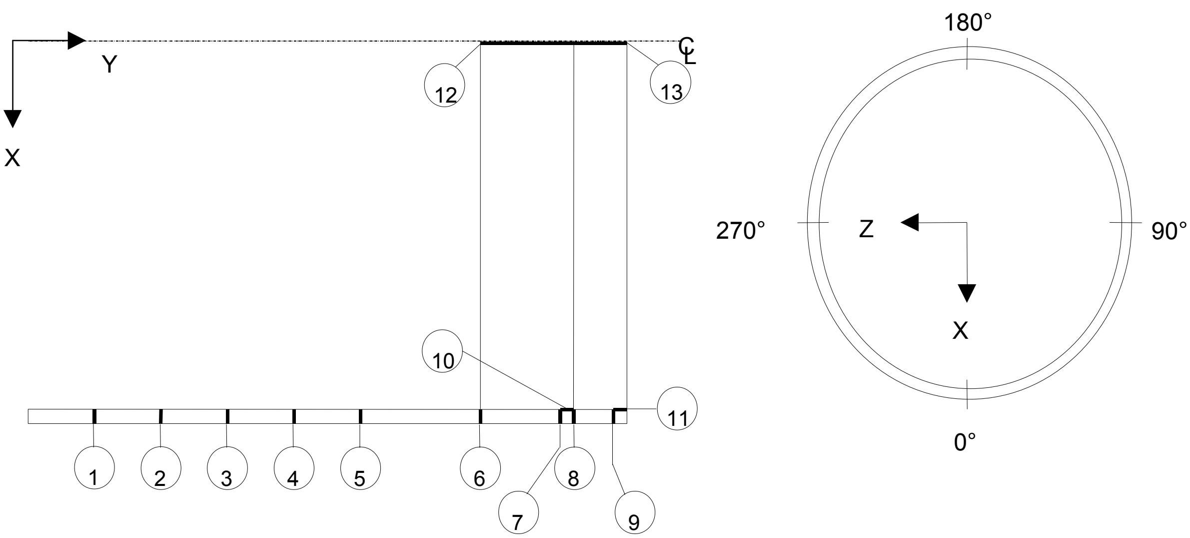


Figure 11.2.12.4.1-5 Fuel Basket/Canister Finite Element Model - Support Disk Loading - PWR



18.22º Basket Drop Orientation
Note: Finite Element Mesh Not Shown

Figure 11.2.12.4.1-6 Canister Section Stress Locations



Top View of Axis

PWR 1
Section Coordinates at Z = 0 and X > 0

	Point 1		Point 2	
Location	X	Y	X	Y
1	32.905	131.42	33.53	131.42
2	32.905	136.34	33.53	136.34
3	32.905	141.26	33.53	141.26
4	32.905	146.18	33.53	146.18
5	32.905	151.10	33.53	151.10
6	32.905	165.25	33.53	165.25
7	32.905	171.75	33.53	171.75
8	32.905	172.25	33.53	172.25
9	32.905	174.37	33.53	174.37
10	32.905	171.75	32.905	172.25
11	32.905	174.37	32.905	175.25
12	0.1	165.25	0.1	172.23
13	0.1	172.27	0.1	175.25

BWR 4
Section Coordinates at Z = 0 and X > 0

	Point 1		Point 2	
Location	X	Y	X	Y
1	32.905	144.32	33.53	144.32
2	32.905	148.15	33.53	148.15
3	32.905	151.98	33.53	151.98
4	32.905	155.81	33.53	155.81
5	32.905	159.64	33.53	159.64
6	32.905	175.25	33.53	175.25
7	32.905	182.25	33.53	182.25
8	32.905	182.75	33.53	182.75
9	32.905	184.87	33.53	184.87
10	32.905	182.25	32.905	182.75
11	32.905	184.87	32.905	185.75
12	0.1	175.75	0.1	182.73
13	0.1	182.77	0.1	185.75

General Notes:

1) Impact from the tipover condition is at 0° (in the circumferential direction).

2) For the full 360° models, there are 80 sections at each location for a total of 1040 sections. For the half 180° models, there are 41 sections at each location for a total of 533 sections.

3) Location 10 is through the length of the shield lid weld. Locations 8 and 7 are through the canister shell at top and bottom of the shield lid weld, respectively.

4) Location 13 is through the length of the structural lid weld. Location 9 is through the canister shell at the bottom of the structural lid weld.

Figure 11.2.12.4.1-7 Support Disk Section Stress Locations - PWR – Full Model

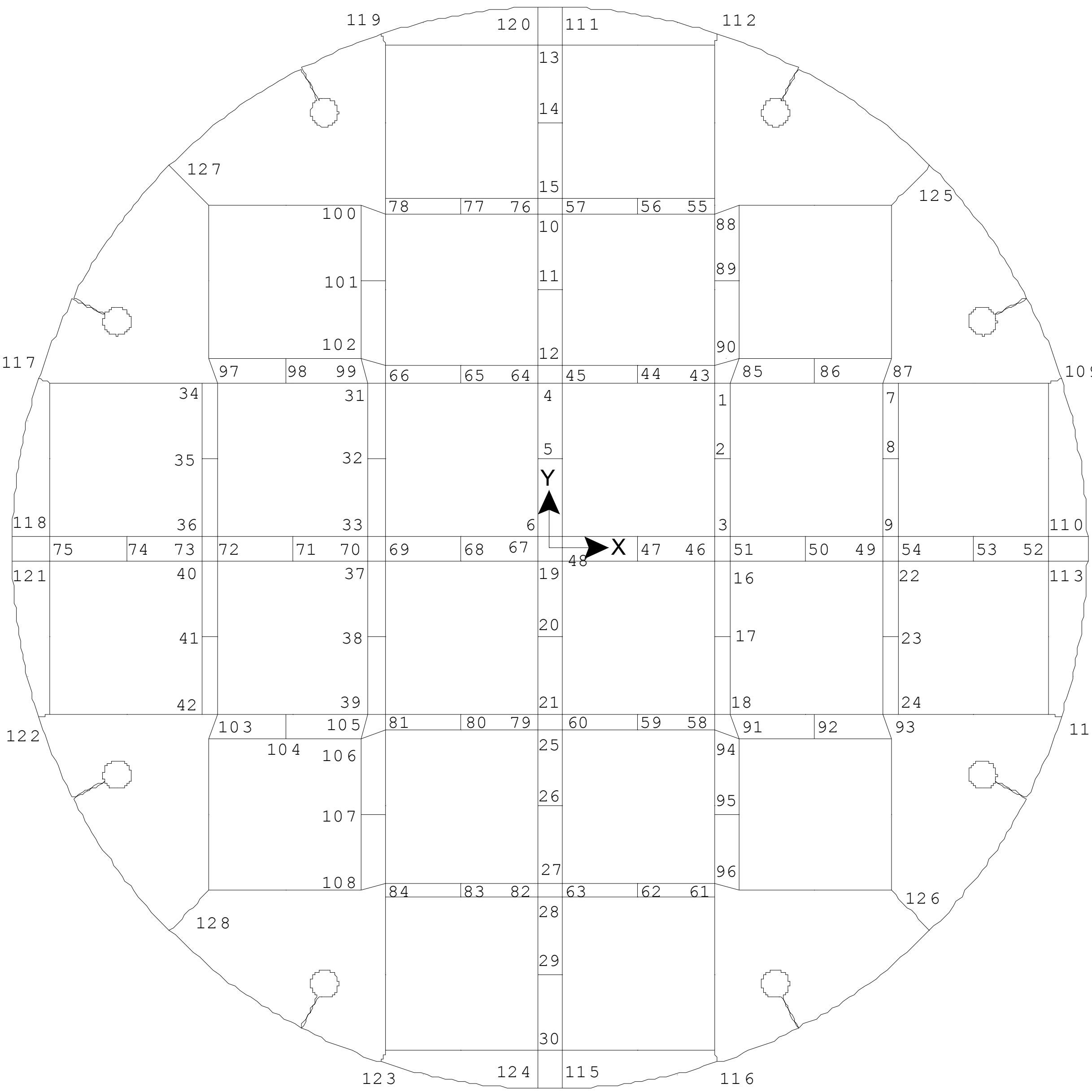
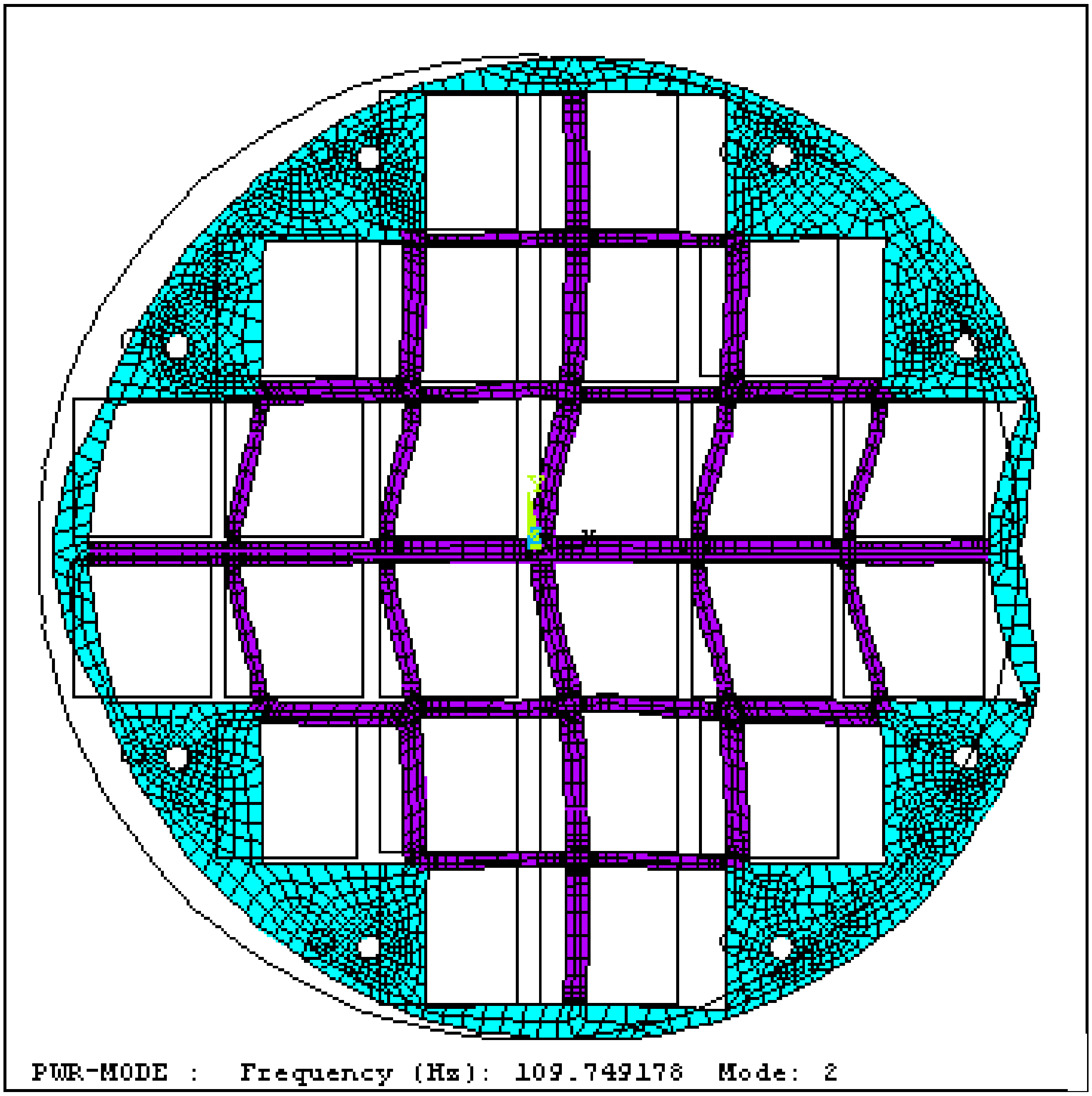
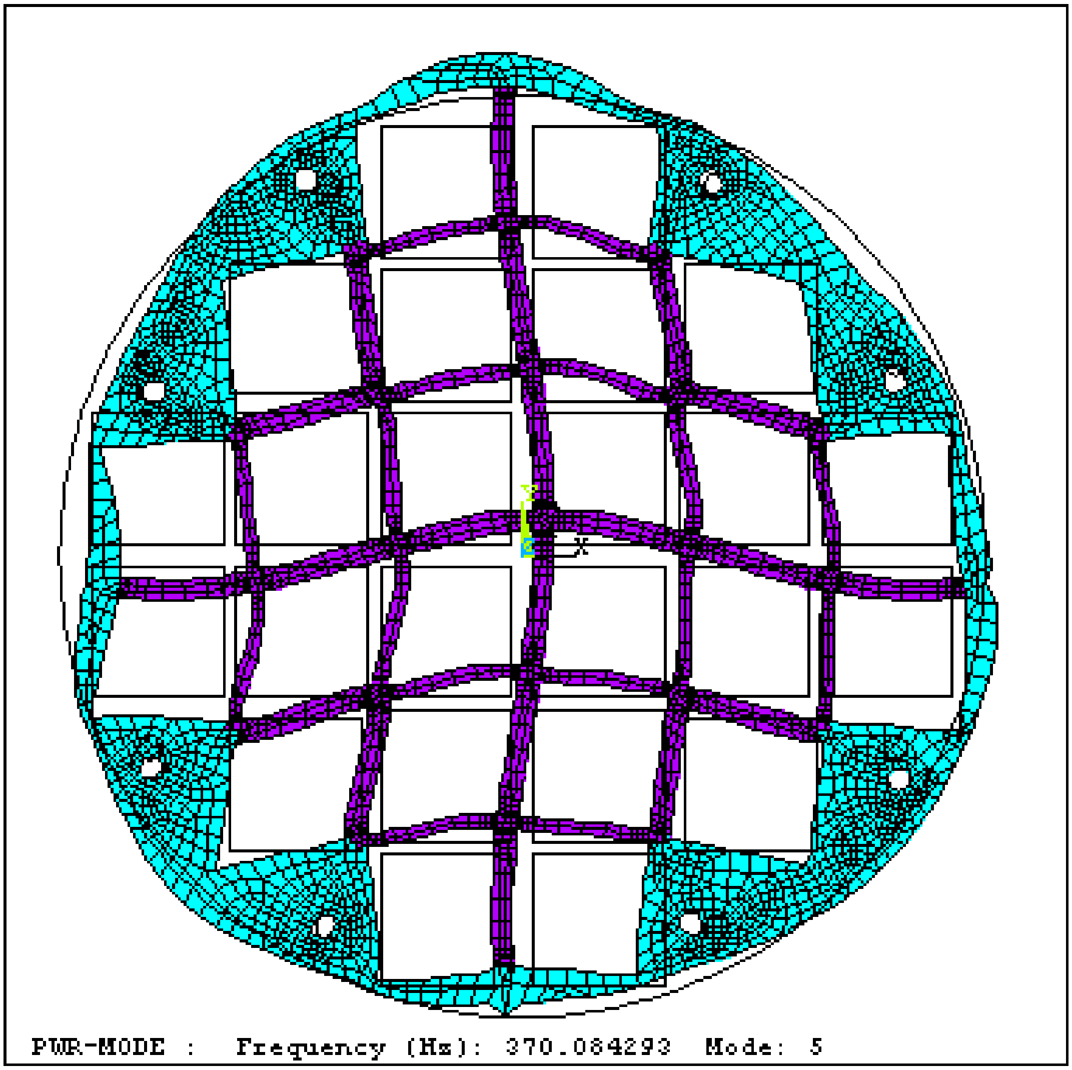


Figure 11.2.12.4.1-8 PWR - 109.7 Hz Mode Shape



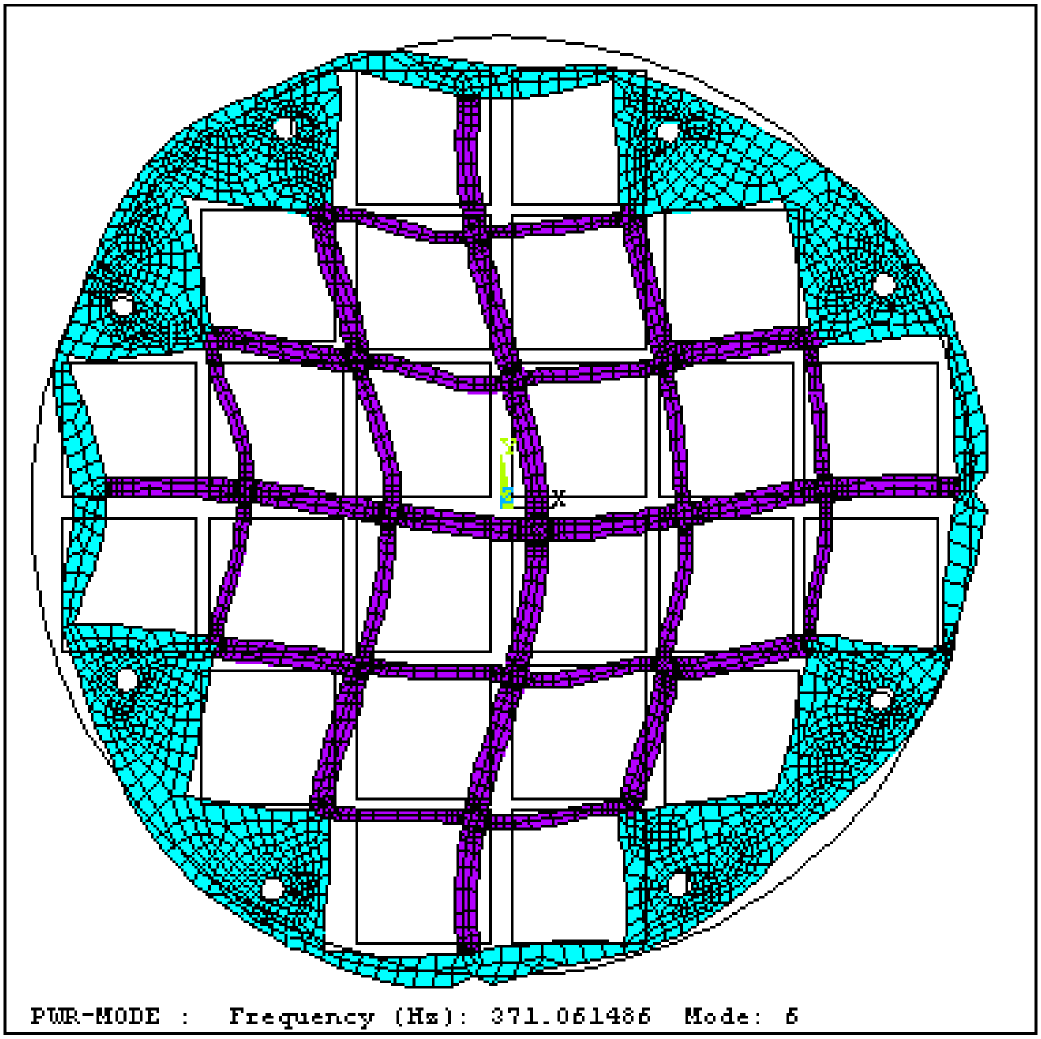
Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Figure 11.2.12.4.1-9 PWR – 370.1 Hz Mode Shape



Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Figure 11.2.12.4.1-10 PWR – 371.1 Hz Mode Shape



Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Table 11.2.12.4.1-1 Canister Primary Membrane (P_m) Stresses for Tip-Over Conditions – PWR - 45° Basket Drop Orientation (ksi)

Section(1) Location	Section Angle (deg)	S_x	S_y	S_z	S_{xy}	S_{yz}	S_{xz}	Stress Intensity	Allowable Stress	Margin of Safety
1	0	-1.5	6.4	1.4	-0.1	0	-0.2	7.98	35.52	3.45
2	0	-1.7	9.2	1.5	0.1	0	0.3	10.88	35.52	2.26
3	49.5	-0.2	9.3	6.3	-0.1	1.1	0	9.81	35.52	2.62
4	63	-0.3	8.9	5	0	3.4	0.4	11.22	35.52	2.17
5	90	0.1	2.8	-1	-0.3	6	0.1	12.6	35.52	1.82
6	85.5	0	0.3	0.1	-0.1	7.8	0	15.62	35.52	1.27
7(2)	9	1.0	0.6	7.0	2.7	-5.1	0.7	13.61	35.52	1.61
8(2)	9	6.8	0	6.9	0.6	-3.2	-1.0	10.09	35.52	2.52
9(2)	9	5.8	-3.4	1.0	2.4	-3.8	0	12.50	35.52	1.84
10(4)	0–9	-29.7	-15.7	-20.6	6.7	-0.8	-2.0	19.87	40.08(3)	1.02
11(4)	0–8.4	-30.0	-15.3	-8.8	7.1	-1.8	2.0	24.80	32.06(5)	0.29
12	0	-0.7	0.2	0	0	0	-0.1	0.93	35.52	37.05
13	0	-1.5	0.5	0	0	0	-0.1	1.98	35.52	16.92

Stresses are presented in the cylindrical coordinate system, x = radial, y = circumferential and z = axial directions.

1. Section locations are shown in Figure 11.2.12.4.1-6.
2. Stresses are not presented for the sections with localized bearing stress. In accordance with ASME Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions.
3. Allowable stress at 300°F.
4. Stresses are determined by averaging the stresses over the impact region.
5. A stress reduction factor of 0.8 is applied to the allowable stress at 250°F.

Table 11.2.12.4.1-2 Canister Primary Membrane + Primary Bending ($P_m + P_b$) Stresses for Tip-Over Conditions – PWR - 45° Basket Drop Orientation (ksi)

Section[(1)] Location	Section Angle (deg)	S_x	S_y	S_z	S_{xy}	S_{yz}	S_{xz}	Stress Intensity	Allowable Stress	Margin of Safety
1	0	-2.1	19.3	4.4	-0.6	-0.1	-0.1	21.38	53.28	1.49
2	0	-1.9	22.3	3	-0.3	0.1	0.2	24.26	53.28	1.2
3	0	-2.6	22.3	6.2	0.2	0	-0.1	24.92	53.28	1.14
4	0	-1.8	21	3.9	-0.8	-0.1	-0.3	22.88	53.28	1.33
5	72	-0.7	20.5	12.4	0.1	3.8	-0.9	22.8	53.28	1.34
6	0	0.6	-29.8	-7.6	2.3	-1.1	-0.9	30.93	53.28	0.72
7[(2)]	9	0.6	9.3	23.7	0.2	-4.0	1.6	24.32	53.28	1.19
8[(2)]	9	6.7	9.0	23.6	-0.8	-5.3	-3.7	21.08	53.28	1.53
9[(2)]	9	8.0	-5.9	4.8	4.4	-4.5	-0.3	18.42	53.28	1.89
10[(4)]	0-8.8	-42.5	-19.4	-24.1	7.1	0.4	-3.6	27.78	60.12[(3)]	1.16
11[(4)]	0-8.4	-26.6	-12.0	-1.2	8.0	-0.8	2.0	29.25	48.09[(5)]	0.64
12	0	-0.9	0	0	0	0	-0.1	0.95	53.28	54.84
13	0	-2.3	-0.7	0	0	0	-0.1	2.33	53.28	21.84

Stresses are presented in the cylindrical coordinate system, x = radial, y = circumferential and z = axial directions.

1. Section locations are shown in Figure 11.2.12.4.1-6.
2. Stresses are not presented for the sections with localized bearing stress. In accordance with ASME Code Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions.
3. Allowable stress at 300°F.
4. Stresses are determined by averaging the stresses over the impact region.
5. A stress reduction factor of 0.8 is applied to the allowable stress at 250°F.

Table 11.2.12.4.1-3 Support Disk Section Location for Stress Evaluation - PWR - Full Model

Sec. No.	Point 1		Point 2		Sec. No.	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
1	10.02	10.02	11.02	10.02	45	0.75	10.02	0.75	11.02
2	10.02	5.39	11.02	5.39	46	10.02	0.75	10.02	-0.75
3	10.02	0.75	11.02	0.75	47	5.39	0.75	5.39	-0.75
4	0.75	10.02	-0.75	10.02	48	0.75	0.75	0.75	-0.75
5	0.75	5.39	-0.75	5.39	49	20.29	0.75	20.29	-0.75
6	0.75	0.75	-0.75	0.75	50	15.66	0.75	15.66	-0.75
7	20.29	10.02	21.17	10.02	51	11.02	0.75	11.02	-0.75
8	20.29	5.39	21.17	5.39	52	30.44	0.75	30.44	-0.75
9	20.29	0.75	21.17	0.75	53	25.81	0.75	25.81	-0.75
10	0.75	20.29	-0.75	20.29	54	21.17	0.75	21.17	-0.75
11	0.75	15.66	-0.75	15.66	55	10.02	20.29	10.02	21.17
12	0.75	11.02	-0.75	11.02	56	5.39	20.29	5.39	21.17
13	0.75	30.44	-0.75	30.44	57	0.75	20.29	0.75	21.17
14	0.75	25.81	-0.75	25.81	58	10.02	-10.02	10.02	-11.02
15	0.75	21.17	-0.75	21.17	59	5.39	-10.02	5.39	-11.02
16	10.02	-0.75	11.02	-0.75	60	0.75	-10.02	0.75	-11.02
17	10.02	-5.39	11.02	-5.39	61	10.02	-20.29	10.02	-21.17
18	10.02	-10.02	11.02	-10.02	62	5.39	-20.29	5.39	-21.17
19	0.75	-0.75	-0.75	-0.75	63	0.75	-20.29	0.75	-21.17
20	0.75	-5.39	-0.75	-5.39	64	-0.75	10.02	-0.75	11.02
21	0.75	-10.02	-0.75	-10.02	65	-5.39	10.02	-5.39	11.02
22	20.29	-0.75	21.17	-0.75	66	-10.02	10.02	-10.02	11.02
23	20.29	-5.39	21.17	-5.39	67	-0.75	0.75	-0.75	-0.75
24	20.29	-10.02	21.17	-10.02	68	-5.39	0.75	-5.39	-0.75
25	0.75	-11.02	-0.75	-11.02	69	-10.02	0.75	-10.02	-0.75
26	0.75	-15.66	-0.75	-15.66	70	-11.02	0.75	-11.02	-0.75
27	0.75	-20.29	-0.75	-20.29	71	-15.66	0.75	-15.66	-0.75
28	0.75	-21.17	-0.75	-21.17	72	-20.29	0.75	-20.29	-0.75
29	0.75	-25.81	-0.75	-25.81	73	-21.17	0.75	-21.17	-0.75
30	0.75	-30.44	-0.75	-30.44	74	-25.81	0.75	-25.81	-0.75
31	-10.02	10.02	-11.02	10.02	75	-30.44	0.75	-30.44	-0.75
32	-10.02	5.39	-11.02	5.39	76	-0.75	20.29	-0.75	21.17
33	-10.02	0.75	-11.02	0.75	77	-5.39	20.29	-5.39	21.17
34	-20.29	10.02	-21.17	10.02	78	-10.02	20.29	-10.02	21.17
35	-20.29	5.39	-21.17	5.39	79	-0.75	-10.02	-0.75	-11.02
36	-20.29	0.75	-21.17	0.75	80	-5.39	-10.02	-5.39	-11.02
37	-10.02	-0.75	-11.02	-0.75	81	-10.02	-10.02	-10.02	-11.02
38	-10.02	-5.39	-11.02	-5.39	82	-0.75	-20.29	-0.75	-21.17
39	-10.02	-10.02	-11.02	-10.02	83	-5.39	-20.29	-5.39	-21.17
40	-20.29	-0.75	-21.17	-0.75	84	-10.02	-20.29	-10.02	-21.17
41	-20.29	-5.39	-21.17	-5.39	85	11.02	10.02	11.52	11.52
42	-20.29	-10.02	-21.17	-10.02	86	16.16	11.52	16.16	10.02
43	10.02	10.02	10.02	11.02	87	20.29	10.02	20.79	11.52
44	5.39	10.02	5.39	11.02	88	10.02	20.29	11.52	20.79

Note: See Figure 11.2.12.4.1-7 for section location.

Table 11.2.12.4.1-4 Summary of Maximum Stresses for PWR Support Disk for Tip-Over Condition

Drop Orientation	P_m			$P_m + P_b$		
	Stress Intensity (ksi)	**Allowable Stress (ksi)**	**Margin of Safety**	**Stress Intensity (ksi)**	**Allowable Stress (ksi)**	**Margin of Safety**
0°	58.2	90.8	+0.56	81.9	129.8	+0.58
18.22°	47.5	90.4	+0.91	111.6	130.8	+0.17
26.28°	46.0	90.4	+0.97	124.6	130.8	+0.05
45°	34.4	91.5	+1.66	101.4	129.1	+0.27

Note: See Figure 11.2.12.4.1-1 for Drop Orientation.

Table 11.2.12.4.1-5 Summary of Buckling Evaluation of PWR Support Disk for Tip-Over Condition

Drop Orientation	MS1	MS2
0°	+0.98	+0.96
18.22°	+0.31	+0.36
26.28°	+0.10	+0.15
45°	+0.31	+0.34

Note: See Figure 11.2.12.4.1-1 for Drop Orientation.

Table 11.2.12.4.1-6 Support Disk Primary Membrane (P_m) Stresses for Tip-Over Condition - PWR Disk No. 5 - 26.28° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
18	19.5	-26.1	3.1	46.0	90.4	0.97
3	27.1	-14.8	2.7	42.2	89.3	1.12
16	-38.3	-25.9	1	38.4	89.3	1.32
1	-33.5	-14.7	0.5	33.5	90.4	1.70
94	-28.3	-21.4	2.9	29.4	90.5	2.08
17	-0.1	-26	2	26.2	89.8	2.42
96	6.1	-16.4	-3.1	23.3	91.5	2.92
95	-0.1	-22.4	1.7	22.6	91.1	3.04
88	-18.4	-7	-7	21.7	91.5	3.21
84	-17.1	-20.7	-0.8	20.9	91.5	3.38
61	-17.8	-9.7	5.1	20.3	91.5	3.51
90	15	-5	0.6	20.1	90.5	3.51
60	-11.3	-18.4	1.1	18.6	89.3	3.80
30	-18	-10.1	3	19.0	91.9	3.83
82	-17.2	-7	4.1	18.7	90.8	3.87
62	-17.8	-0.2	2.6	18.4	91.2	3.97
58	-11.4	-13.8	5.4	18.2	90.4	3.97
91	-8.2	-17.5	-1.4	17.7	90.5	4.11
63	-17.8	-12.3	0.2	17.8	90.8	4.11
83	-17.2	-0.2	1.7	17.3	91.2	4.26
7	-16.5	-12.6	-0.8	16.7	91.5	4.49
24	-1.2	-15.8	2	16.1	91.5	4.69
28	-15.4	-10	1.6	15.8	90.9	4.74
23	-0.1	-15.8	0.8	15.8	91.2	4.78
22	-9.1	-15.7	-0.5	15.7	90.8	4.78
51	-3.6	-15.1	-2	15.4	89.4	4.79
37	11.1	-4.3	0.6	15.4	89.3	4.80
79	-6	6.5	4.5	15.4	89.3	4.82
2	-0.1	-14.7	1.6	15.0	89.8	5.00
85	-4.6	-11.2	-6.4	15.1	90.5	5.00

Note: See Figure 11.2.12.4.1-2 for disk location and Figure 11.2.12.4.1-7 for section locations.

Table 11.2.12.4.1-7 Support Disk Primary Membrane + Primary Bending ($P_m + P_b$) Stresses for Tip-Over Condition - PWR Disk No. 5 - 26.28° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
61	-123.4	-34.3	10.4	124.6	130.8	0.05
58	-115.3	-47.4	9.6	116.6	129.1	0.11
43	-95.4	-34.6	6.8	96.1	129.1	0.34
82	-92.1	-27.8	7.2	92.9	129.8	0.40
79	-86.9	-19.9	2.3	87.0	127.6	0.47
16	-54.3	-76.8	15.6	84.8	127.6	0.50
60	-82.9	-41	7.8	84.3	127.6	0.51
18	-4.1	-84.9	-2.5	85.0	129.1	0.52
46	-79.1	-52.5	10.4	82.7	127.6	0.54
55	-84.2	-31.4	5	84.7	130.8	0.54
3	9.1	-71.1	-5.7	81.0	127.6	0.57
64	-79.8	-32.4	7.2	80.9	127.6	0.58
30	-40.2	-74.7	11.7	78.3	131.3	0.68
63	-75.2	-27.9	4.9	75.7	129.8	0.71
76	72.6	21.9	5.2	73.1	129.8	0.77
48	-66.5	-43.2	3.9	67.1	125.7	0.87
19	-39.5	-66.4	2.9	66.7	125.7	0.88
6	-43.6	-63.2	5.2	64.5	125.7	0.95
94	-59.5	-44.7	11.1	65.5	129.3	0.97
21	-48.3	-59.4	5.2	61.5	127.6	1.08
45	-61.2	-14.4	-0.6	61.2	127.6	1.09
67	-56.6	-43.3	5.4	58.6	125.7	1.15
1	-49.4	-43.6	13.2	60.0	129.1	1.15
51	26.3	-30.4	4.7	57.5	127.7	1.22
33	-29.3	-54.9	7.1	56.7	127.6	1.25
39	-29.2	-52.9	6.2	54.5	129.1	1.37
24	-8.5	-52.1	4.1	52.5	130.8	1.49
81	-49.2	-30.8	5.5	50.7	129.1	1.55
4	-43.3	-43.7	5.8	49.3	127.6	1.59
28	-46.3	-28.1	9.2	50.1	129.9	1.59

Note: See Figure 11.2.12.4.1-2 for disk location and Figure 11.2.12.4.1-7 for section locations.

Table 11.2.12.4.1-8 Summary of Support Disk Buckling Evaluation for Tip-Over Condition - PWR Disk No. 5 - 26.28° Drop Orientation

Section Number	P (kip)	Pcr (kip)	Py (kip)	M (in-kip)	Mp (in-kip)	Mm (in-kip)	MS1	MS2
61	7.80	44.18	38.91	6.74	8.51	8.18	0.10	0.15
58	5.69	51.79	43.78	8.66	10.94	10.67	0.23	0.25
82	7.52	43.76	38.54	4.78	8.43	8.10	0.44	0.48
18	13.04	51.79	43.78	4.90	10.94	10.67	0.51	0.48
43	1.95	51.79	43.78	7.62	10.94	10.67	0.54	0.58
16	12.97	50.82	42.93	4.24	10.73	10.47	0.62	0.57
79	3.00	50.82	42.93	6.74	10.73	10.47	0.63	0.66
60	5.66	50.82	42.93	5.96	10.73	10.47	0.65	0.66
63	7.78	43.76	38.54	3.66	8.43	8.10	0.73	0.75
55	0.92	44.18	38.91	5.24	8.51	8.18	0.76	0.83
64	2.18	50.82	42.93	6.29	10.73	10.47	0.79	0.83
3	7.40	50.82	42.93	4.69	10.73	10.47	0.86	0.84
46	1.85	83.64	64.39	14.37	24.15	24.15	0.89	0.88
30	7.60	87.05	67.05	12.10	25.14	25.14	1.00	0.92
19	3.78	81.50	62.70	11.51	23.51	23.51	1.15	1.10
48	1.80	81.50	62.70	12.01	23.51	23.51	1.19	1.17
6	2.46	81.50	62.70	11.23	23.51	23.51	1.29	1.25
45	1.91	50.82	42.93	4.78	10.73	10.47	1.34	1.37
21	3.89	83.64	64.39	10.16	24.15	24.15	1.47	1.40
24	6.92	44.18	38.91	2.31	8.51	8.18	1.46	1.45
67	1.00	81.50	62.70	10.37	23.51	23.51	1.58	1.57
33	1.95	50.82	42.93	4.25	10.73	10.47	1.59	1.63
84	7.49	44.18	38.91	1.82	8.51	8.18	1.73	1.67
39	2.19	51.79	43.78	4.04	10.94	10.67	1.72	1.75
17	13.00	51.32	43.37	0.79	10.84	10.58	2.13	1.77
1	7.33	51.79	43.78	2.41	10.94	10.67	1.95	1.82
81	2.97	51.79	43.78	3.61	10.94	10.67	1.88	1.88
37	2.13	50.82	42.93	3.24	10.73	10.47	2.26	2.27
4	2.35	83.64	64.39	7.60	24.15	24.15	2.37	2.30
66	2.15	51.79	43.78	3.25	10.94	10.67	2.31	2.33

Note: See Figure 11.2.12.4.1-2 for disk location and Figure 11.2.12.4.1-7 for section locations.

11.2.12.4.2 Analysis of Canister and Basket for BWR Configurations

Five three-dimensional models of the BWR canister and fuel basket are evaluated for the cask tip-over event. Each model corresponds to a different fuel basket drop orientation. For the BWR fuel configuration, fuel basket drop orientations of 0°, 31.82°, 49.46°, 77.92°, and 90° are evaluated, as shown in Figure 11.2.12.4.2-1. Three-dimensional half-symmetry models are used for the basket drop orientations of 0° and 90°. Three-dimensional full-models are used for the basket orientations of 31.82°, 49.46° and 77.92°.

Model Description

The models used for the evaluation of the canister and basket for BWR configuration are similar to those used for the PWR (Section 11.2.12.4.1). The three-dimensional model used for the basket drop orientation of 31.82° is presented in Figure 11.2.12.4.2-2 and Figure 11.2.12.4.2-3.

The same modeling and analysis techniques described for the PWR model (see Section 11.2.12.4.1) are used for the BWR models. Loading of the BWR models includes an internal pressure of 15 psig (design pressure for normal condition of storage) applied to the inner surfaces of the canister, pressure loads applied to the support disk slots and the inertial loads. The pressure load applied to the support disk slots represents the combined weight of the BWR fuel assemblies, fuel tubes and aluminum heat transfer disks multiplied by 30g. Note that the BWR fuel assembly weight is 702 pounds.

For the inertial loads, a maximum acceleration of 30g is conservatively applied to the entire model. As shown in Section 11.2.12.3.2, the maximum acceleration of the concrete cask steel liner at the locations of the top support disk and the top of the canister structural lid during the tip-over event is determined to be 24.2g and 28.0g, respectively. Using the same method described in Section 11.2.12.4.1 for the PWR models, the DLF for the acceleration at the top support disk is computed to be 1.09. Applying the DLF to the 24.2g results in a peak acceleration of 26.4g for the top support disk.

The dominant resonance frequencies and corresponding modal mass participation factors from the finite element modal analyses of the BWR support disk are:

Frequency (Hz)	% Modal Mass Participation Factor
79.3	38.4
80.2	54.9
210.9	3.4

The mode shapes for these frequencies are shown in Figures 11.2.12.4.2-5 through 11.2.12.4.2-7. The displacement depicted in these figures is highly exaggerated by the ANSYS program in order to illustrate the modal shape. The stresses associated with the actual displacement are shown in Tables 11.2.12.4.2-4 through 11.2.12.4.2-8.

The DLFs for the canister lids are considered to be unity since the lids have significant in-plane stiffness and are considered to be rigid. Therefore, applying 30g to the entire canister/basket model is conservative.

A uniform temperature of 75°F is applied to the model to determine material properties during solution. During post processing for the support disk, temperature distribution with a maximum temperature of 700°F (at the center) and a minimum temperature of 400°F (at the outer edge) are conservatively used to determine the allowable stresses. A constant temperature of 500° is used for the canister to determine the allowable stresses. These temperatures are the bounding temperatures for the normal, off-normal and accident conditions of storage.

Analysis Results for Canister

The sectional stresses at 13 axial locations of the canister are obtained for each angular division of the model (a total of 80 angular locations for the full-models and a total of 41 angular locations for the half-symmetry models). The locations for the stress sections are shown in Figure 11.2.12.4.1-6.

The same stress allowables used in the evaluation of the PWR canister (see Section 11.2.12.4.1) are used in evaluating the BWR canister.

The primary membrane and primary membrane plus bending stresses for the BWR configuration for a 49.46° basket drop orientation are summarized in Table 11.2.12.4.2-1 and Table 11.2.12.4.2-2, respectively. The stress results of the canister are similar for all five models. Only the 49.46° basket drop orientation results are presented for the canister because this drop orientation generates the minimum margin of safety in the canister. The stress evaluation results for tip-over accident conditions show that the minimum margin of safety in the canister for BWR configurations is +0.35 for P_m (Section 10) and +0.46 for P_m+P_b (Section 10).

Analysis Results for Support Disks

To evaluate the most critical regions of the support disk, a series of cross sections are considered. To aid in the identification of these sections, Figure 11.2.12.4.2-4 shows the locations on a support disk for the full-models. Table 11.2.12.4.2-3 lists the cross-sections with their end point locations (Point 1 and Point 2), which spans the cross section of the ligament in the plane of the support disk. Note that a local coordinate system (x and y parallel to the support disk ligaments) is used for the stress evaluation.

The stress evaluation for the support disk is performed according to ASME Code, Section III, Subsection NG. The allowable stresses for each section are determined based on the temperature of the support disk at the section location. The temperature distribution of the disk is determined by a thermal conduction solution for a single disk with a temperature of 700°F specified at the center of the disk and a temperature of 400°F specified at the outer edge of the disk as boundary conditions. These temperatures are bounding temperatures for the normal, off-normal and accident conditions of storage.

The highest stress occurs at the 5th support disk. The stress evaluation results for the 5th support disk are summarized in Table 11.2.12.4.2-4 for the five basket drop orientations evaluated. As shown in Table 11.2.12.4.2-4, the 77.92° drop orientation case generates the minimum margin of safety in the support disk; therefore, the P_m and $P_m + P_b$ stress intensities for the 77.92° basket drop orientation case are presented in Table 11.2.12.4.2-6 and Table 11.2.12.4.2-7, respectively. These tables list the stresses with the 30 lowest margins of safety for the 5th support disk. The highest P_m stress occurs at Section 202, with a margin of safety of +0.33 (See Table 11.2.12.4.2-6 for stresses and Figure 11.2.12.4.2-4 for section locations). The highest P_m+P_b stress occurs at Section 169, with a margin of safety of +0.04 (see Table 11.2.12.4.2-7 for stresses and Figure 11.2.12.4.2-4 for section locations).

Support Disk Buckling Evaluation

The support disk buckling evaluation for the BWR support disks is performed using the same method as that presented for the PWR support disks (see Section 11.2.12.4.1). The support disk buckling evaluation results for the 5th support disk (the 5th support disk experiences the highest stresses) for the tip-over impact condition are summarized in Table 11.2.12.4.2-5 for the five basket drop orientations evaluated. As shown in Table 11.2.12.4.2-5, the 77.92° drop orientation case generates the minimum margin of safety for buckling; therefore, the results of the buckling analysis for the 77.92° basket drop orientation case are presented in Table 11.2.12.4.2-8. This table presents the results for 30 minimum margins of safety for this drop orientation. As the tables demonstrate, the support disks meet the requirements of NUREG/CR-6322.

Fuel Tube Analysis

The fuel tube provides structural support and a mounting location for neutron absorber plates. The fuel tube does not provide structural support for the fuel assembly. To ensure that the fuel tube remains functional during a tip-over accident, a structural evaluation of the tube is performed for a side impact assuming a deceleration of 60g. This g-load bounds the maximum g-load (30g) calculated to occur for the BWR basket in a vertical concrete cask tipover event.

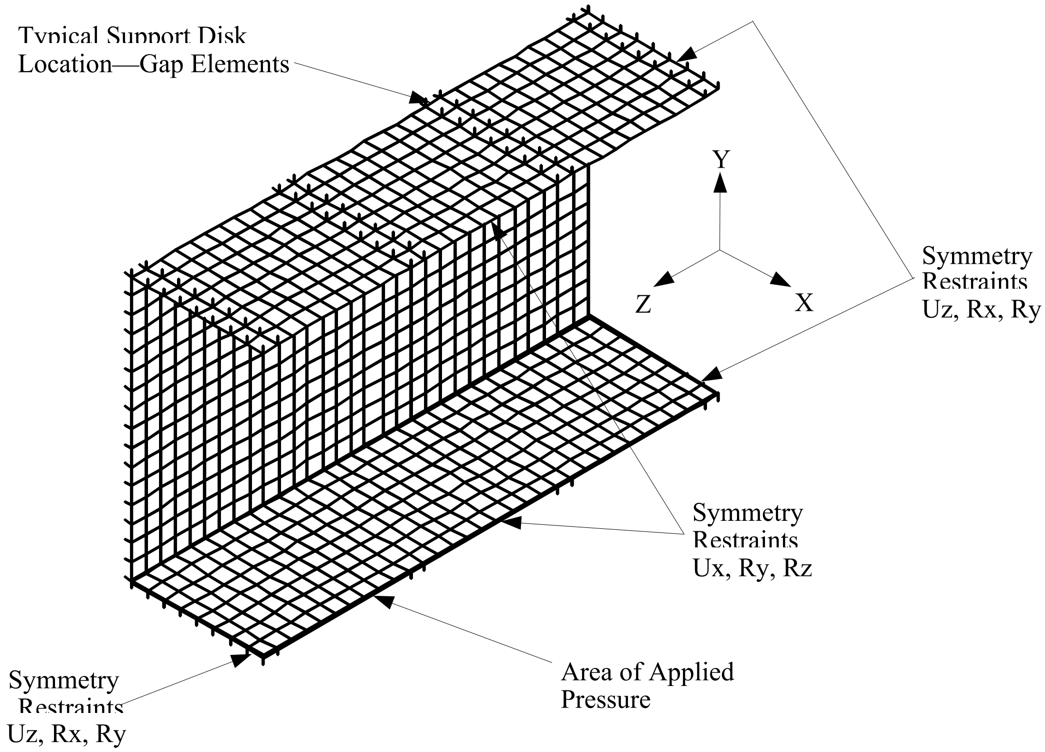
In the tipover event, the stainless steel support disks in the fuel basket support the fuel tube. The fuel basket support disks, which support the full length of the fuel tube, are spaced 3.205-inches apart (which is slightly more than one half of the fuel tube width of 5.9 inch). Considering the fuel tube subjected to a maximum BWR fuel assembly weight of 702 pounds with a 60g load factor and the 40 support locations provided by the basket support disks, the fuel tube shear stress is calculated as:

Shear load = (60g)(702)/40 = 1,053 lbs
Area = (0.048)(5.9)(2) = 0.566 in^2
Shear Stress = 1,053/0.566 = 1,860 psi

The yield strength of the tube material, Type 304 stainless steel, is 17,300 psi at 750°F. Conservatively using the allowable shear stress as one- half the yield strength of the tube material (8,650 psi) results in a large positive margin of safety. Conservative evaluation of the tube loading resulting from its own mass during a side impact shows that the tube structure maintains position and function.

The load transfer of the fuel assembly to the weight of the fuel basket support disk in the side impact is through direct bearing and compression of the distributed load of the fuel assembly through the fuel tube to the support disk web. Two load conditions are considered in the fuel tube evaluation. The first considers the fuel assembly load as a distributed pressure on the inside surface of the fuel tube. The second postulates that the fuel assembly grid is located at the center of the span between the support disks and produces a localized distributed load over the effective area of the grid.

Two different ANSYS finite element models of the tube are developed for these two load conditions since the fuel assembly structural performance for either load is nonlinear. As shown below, the first model represents a fuel tube section with a length of three spans, i.e., the model is



supported at four locations by support disks. The model conservatively considers the fuel tube wall thickness of 0.048 inch as the only material subjected to a distributed pressure load representative of the fuel assembly deceleration of 60g. Fuel assembly stiffness is not considered in the development of the imposed pressure load on the fuel tube.

The fuel tube is modeled with the ANSYS plastic, quadrilateral shell element (SHELL43). The support disks are represented as rigid gap elements (CONTAC52). The outer nodes of the gap elements are fully restrained in all three translational directions. Edge restraints were applied to the model to represent symmetry boundary conditions. The effective load on the fuel tube due to the 60g deceleration of the assembly is applied as a pressure to the inside area of the fuel tube.

The finite element analysis results show that the maximum stress in the tube is 19.5 ksi, which is local to the sections of the tube resting on the support disks. At 750°F the ultimate strength for Type 304 stainless steel is 63.1 ksi. The margin of safety is:

$$MS = \frac{63.1}{19.5} - 1 = +2.24$$

The analysis shows that the maximum total strain is 0.0078 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F [42], the resulting margin of safety is:

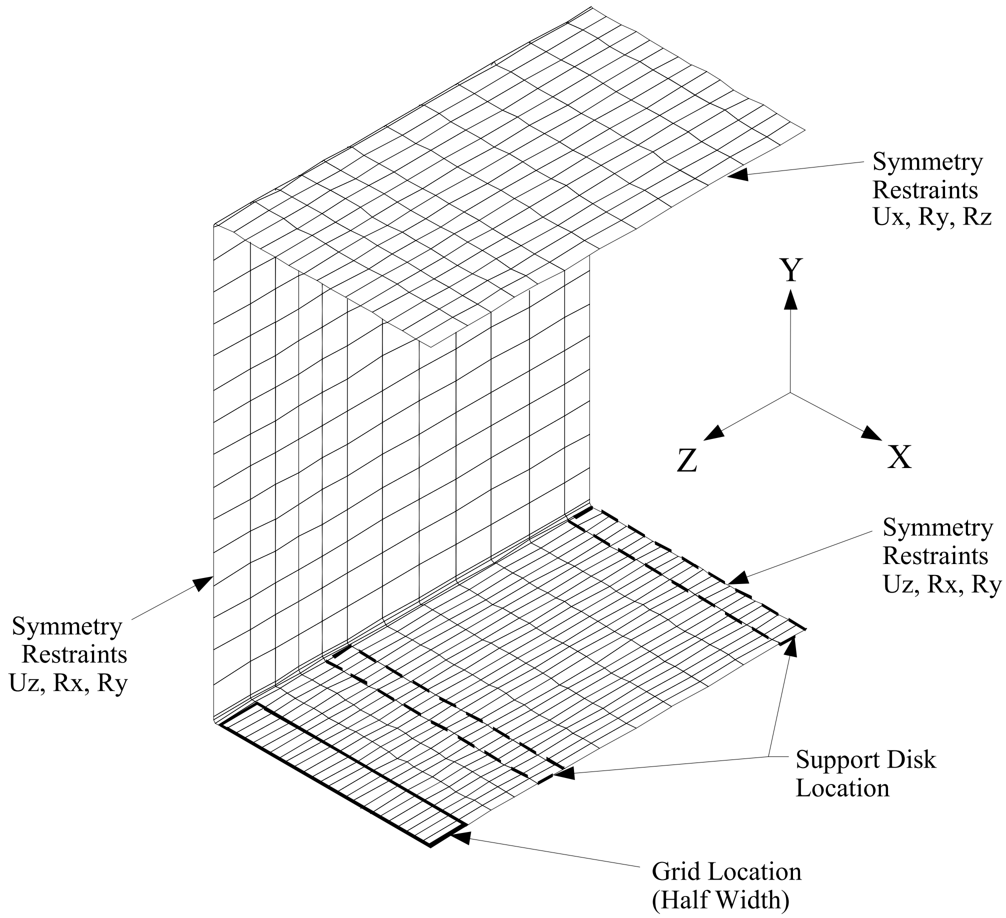
$$MS = \frac{0.40/2}{0.0078} - 1 = +\text{Large}$$

Similarly, the margin of safety for elastic-plastic stress becomes

$$MS = \frac{63.1 - 17.3}{19.5 - 17.3} - 1 = +\text{Large}$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

The second finite element model is used to evaluate the load condition with the fuel assembly grid located at the center of the span between two support disks. The fuel tube is subjected to a localized distributed load over the effective area of the grid. As shown below, the model is a quarter-symmetry periodic section of the fuel tube. As in the finite element model used for the distributed pressure case, this model conservatively considers a fuel tube wall thickness of 0.048 inch. The neutron absorber plate (0.135 inch) and stainless steel cover plate (0.018 inch) are conservatively not included in the model. The tube wall is modeled with ANSYS SHELL43 elements. The support disks are modeled with CONTAC52 elements. A uniform pressure corresponding to the fuel assembly weight with the 60g load is applied to the elements at the grid location of the model. The displacement in the Y-direction for the nodes at the grid location of the model are coupled to represent the structural rigidity of the spacer grid.



The finite element analysis results show that the maximum stress in the tube is 40.8 ksi. At 750°F, the ultimate strength for Type 304 stainless steel is 63.1 ksi. The margin of safety is

$$MS = \frac{63.1}{40.8} - 1 = +0.54$$

The analysis shows that the maximum total strain is 0.10 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F [42], the resulting margin of safety is:

$$MS = \frac{0.40/2}{0.127} - 1 = +0.57$$

Similarly, the margin of safety for elastic-plastic stress becomes

$$\mathrm{MS} = \frac{63.1 - 17.3}{40.8 - 17.3} - 1 = +\,0.94$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

Fuel Tube Yielding

Using the displacement of the fuel rod, a check of the fuel tube is performed to verify that the fuel tube remains elastic during a side-drop scenario. The fuel rod displacement loading is a more realistic loading condition because the load is transmitted from the fuel rods to the fuel tube. The analysis is conservative as it assumes the cumulative displacement of 9 fuel rods (stacked on top of each other) in a 9×9 PWR fuel assembly.

The displacement of a single fuel rod assumed as a four-span continuous beam is calculated as

$$\Delta_{\max} = 0.0065\frac{wL^4}{EI} = 4.415 \text{ x } 10^{-6} \text{ in}$$

where:

w = mass/length = $\rho_{zirc} A_{zirc} + \rho_{UO2} A_{UO2}$ = 0.05 lb/in × 9 rods = 0.4498 lb/in

Rod OD = 0.424 in

Rod ID = 0.424-2 × 0.03 = 0.364 in

Rod Density (Zirc-4) = ρ_{zirc} = 0.237 lb/in^3

Rod Area = $A_{zirc} = \frac{\pi}{4}(0.424^2 - 0.364^2)$ = 0.0371 in^2

UO_2 Density = ρ_{UO_2} = 0.396 lb/in^3

UO_2 Area = $A_{UO_2} = \frac{\pi}{4} x 0.364^2$ = 0.104 in^2

L = Distance between support disks = 3.205 in

E_{zirc} = 10.75 × 10^6 psi

$$I_{zirc} = \frac{\pi}{64}(0.424^4 - 0.364^4) = 7.247 \text{ x } 10^{-4} \text{ in}^4 \text{ x 9 rods} = 0.0065 \text{ in}^4$$

Using the E_{zirc} and I_{zirc} as conservative assumptions, the maximum displacement is estimated as 4.415×10^{-6} in. For 60g acceleration, this displacement becomes 0.0003 inch.

Applying the displacement midway between support disks, the maximum stress intensity is 5,812 psi. The yield stress for the fuel tube (Type 304 stainless steel) is 17,300 psi at 750°F degrees; therefore, during a 60g side-drop, the fuel tube remains elastic.

Both the maximum total strain and the elastic-plastic stress analyses indicate that the tube position within the support basket is maintained.

Assurance that the neutron absorber remains attached to the fuel tube is evaluated by considering that loads produced by the neutron absorber plate and stainless steel attachment plate, assuming a 60g load, are carried by the attachment plate weld. Total load and resultant stress on the weld are calculated as:

$F_{b/ss} = (g)(\rho)(t)(w)(l)$ Load exerted by neutron absorber/stainless steel attachment plate

where:

- g = acceleration (g)
- ρ = density of material (lb/in^3) (The density of aluminum ($0.098\ lb/in^3$) is conservatively used for the neutron absorber.
- t = thickness of material (in.)
- w = width of material (in.)
- l = length of material section (in.)

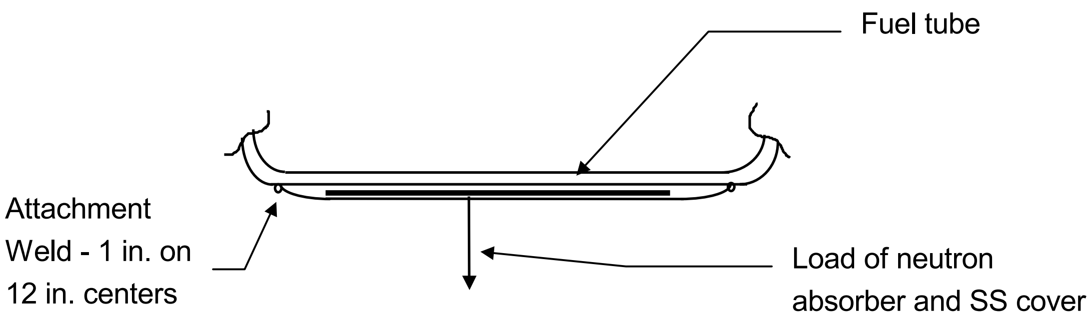
The forces on the weld due to a 12-inch section of neutron absorber (F_b) and a 12-inch section of stainless steel plate (F_{ss}) are:

$$F_b = (60g)(0.098\ lb/in^3)(0.135\ in)(5.45\ in)(12\ in) = 51.9\ lbs$$

$$F_{ss} = (60g)(0.291\ lb/in^3)(0.018\ in)(5.79\ in)(12\ in) = 21.8\ lbs$$

The total load (F_t) on a 1-inch attachment for a 12-inch section is:

$$F_t = 57.9\ lbs + 21.8\ lbs = 73.7\ lbs$$



The resulting weld stress is: σ = P/A = (73.7 lbs/2) / (1 in) (0.018 in) = 2,074 psi

Since the weld material is Type 304 stainless steel, the margin of safety (at 750°F) is:

$$MS = \frac{17,300}{2,047} - 1 = +\,7.5$$

Therefore, the neutron absorber remains enclosed on each outer surface of the fuel tube wall.

Figure 11.2.12.4.2-1 Fuel Basket Drop Orientations Analyzed for Tip-Over Condition - BWR

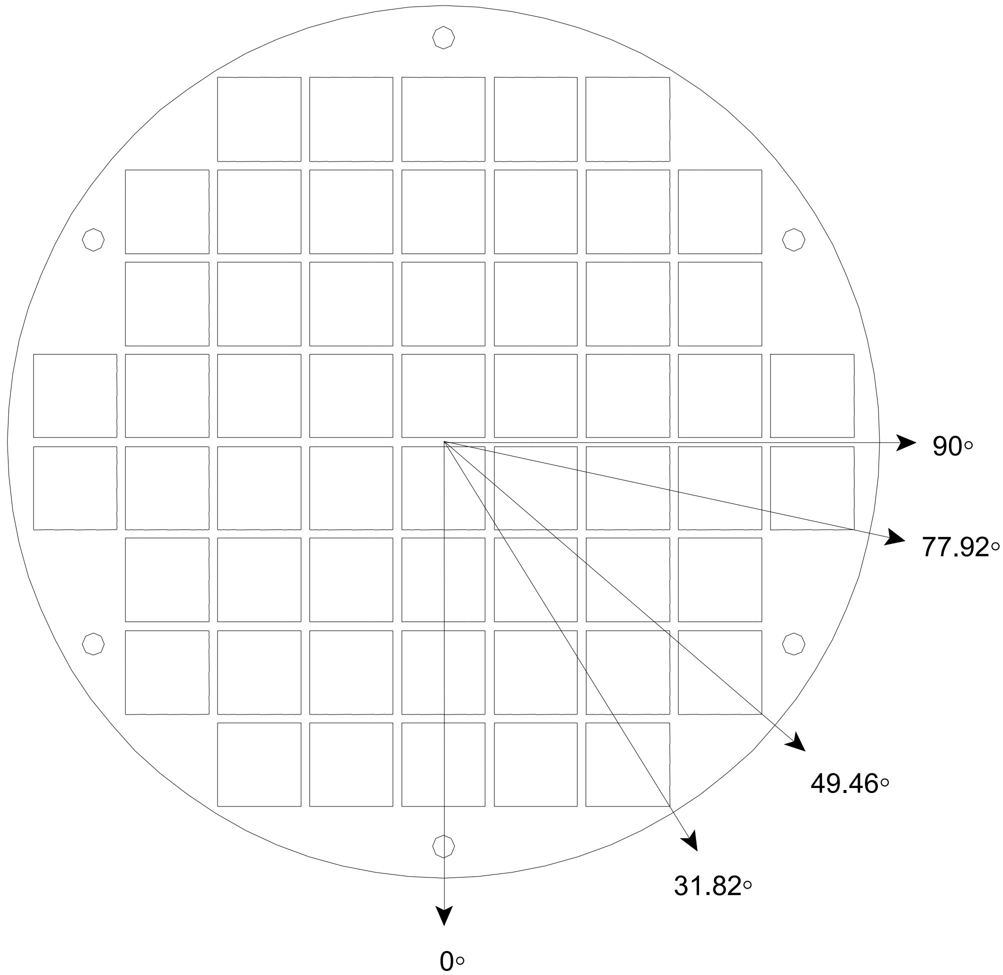
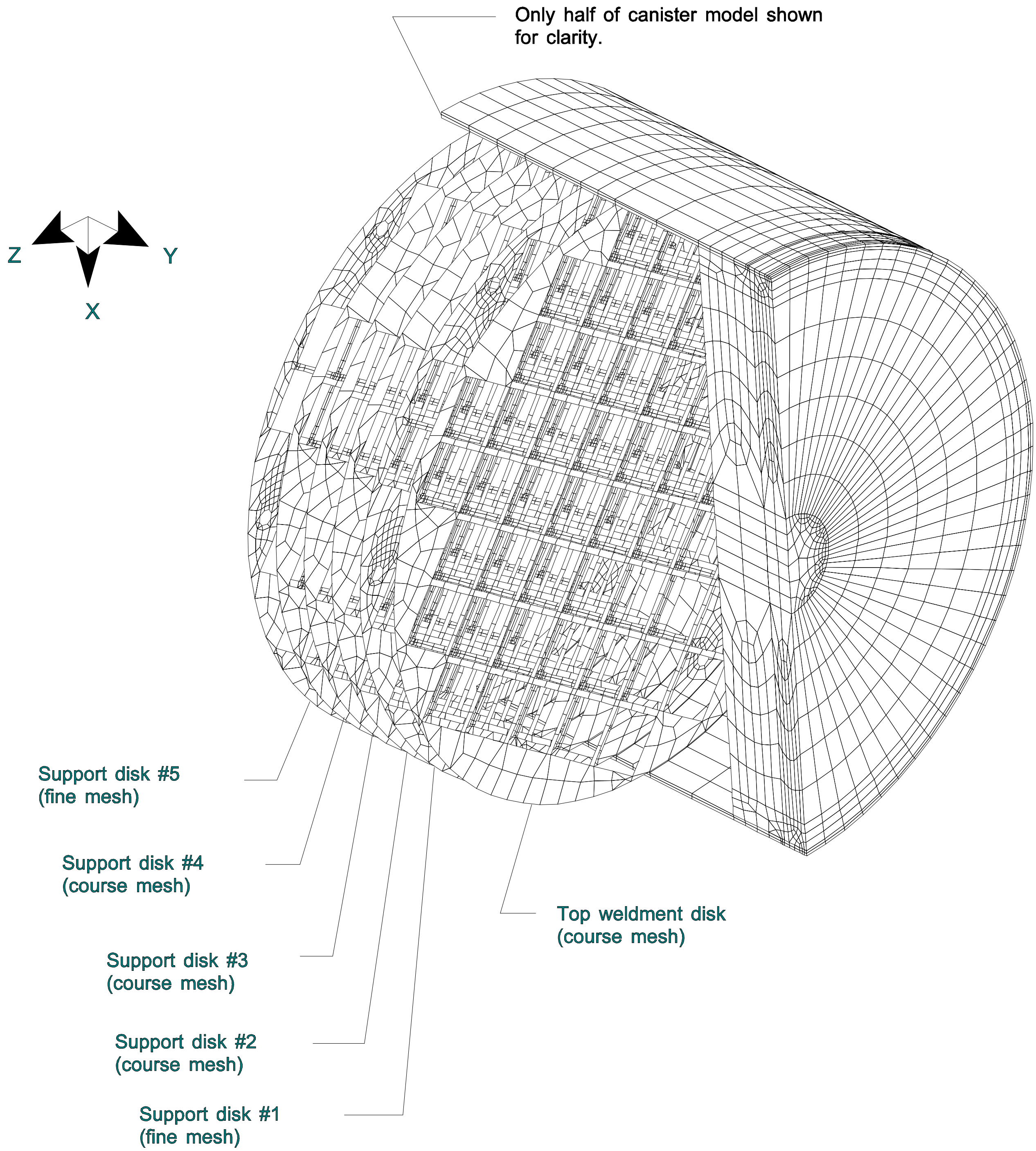


Figure 11.2.12.4.2-2 Fuel Basket/Canister Finite Element Model - BWR



31.82° Basket Drop Orientation

Figure 11.2.12.4.2-3 Fuel Basket/Canister Finite Element Model - Support Disk - BWR

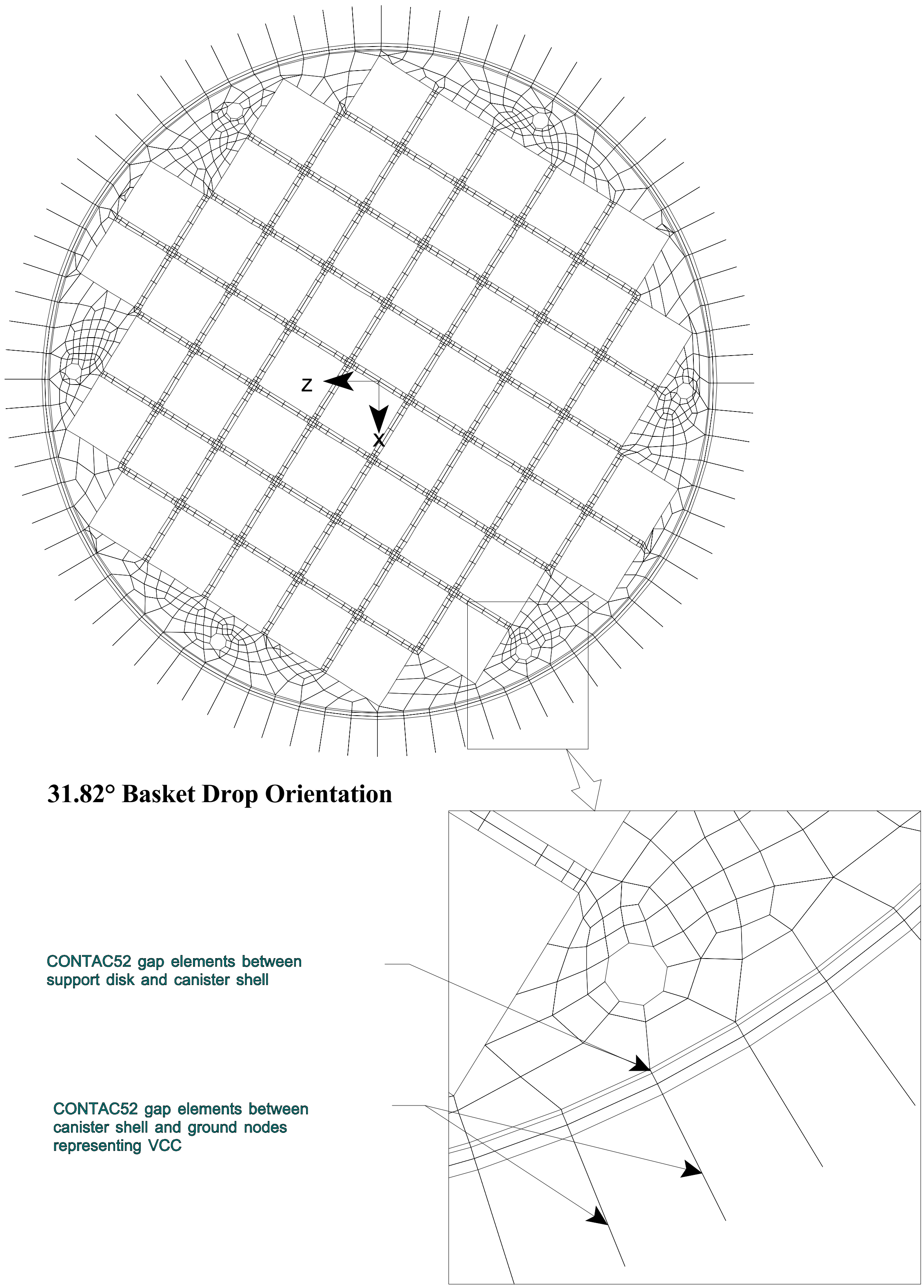


Figure 11.2.12.4.2-4 Support Disk Section Stress Locations - BWR - Full Model

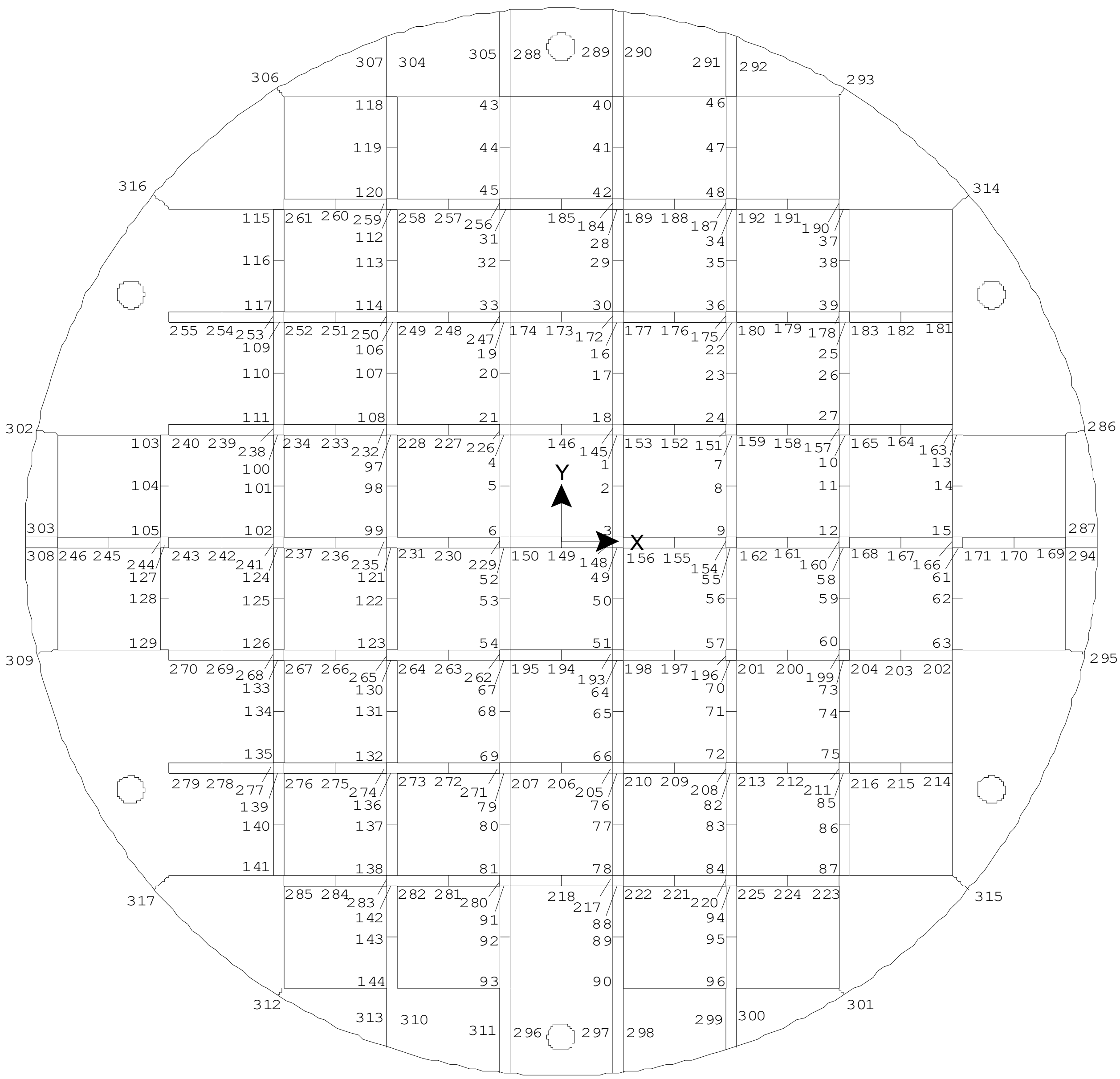
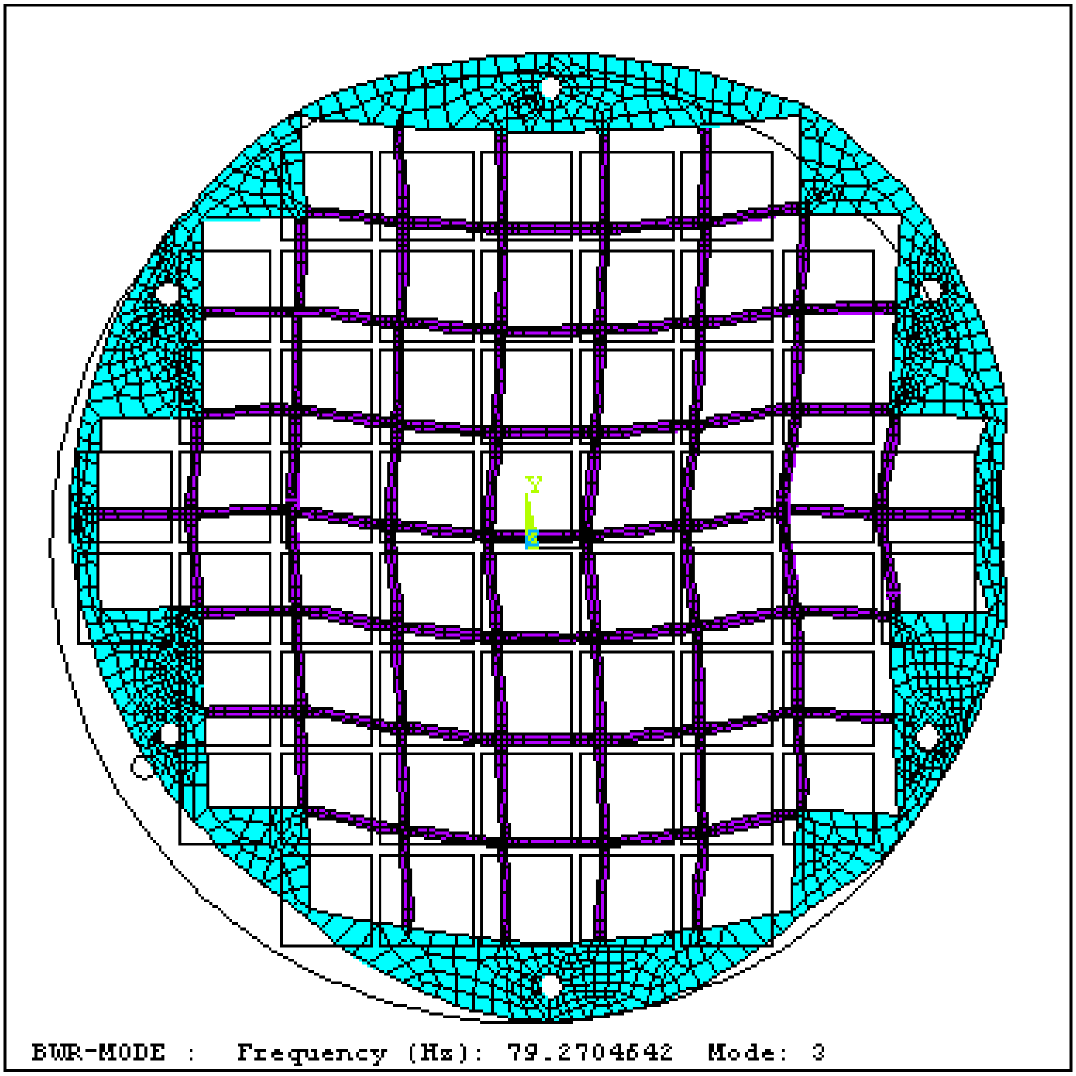
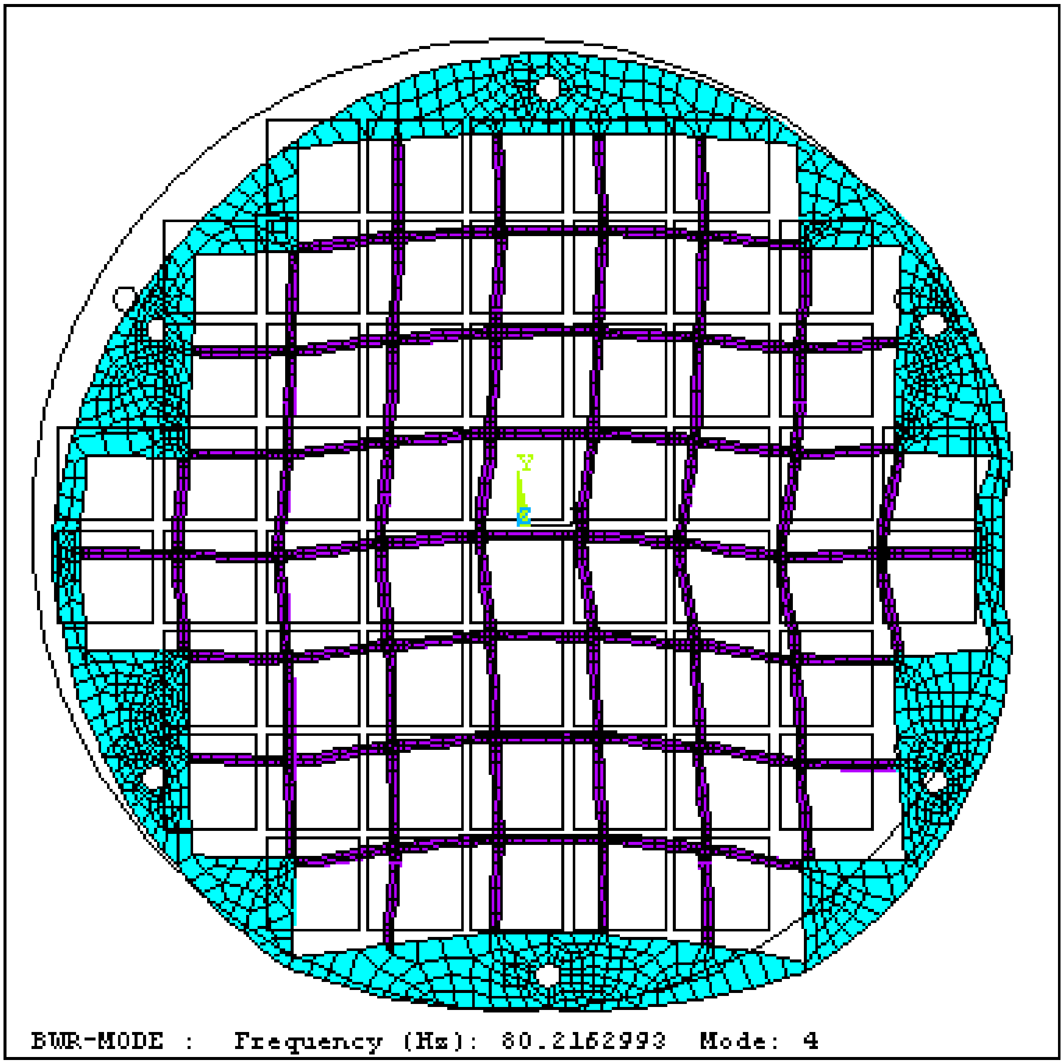


Figure 11.2.12.4.2-5 BWR – 79.3 Hz Mode Shape



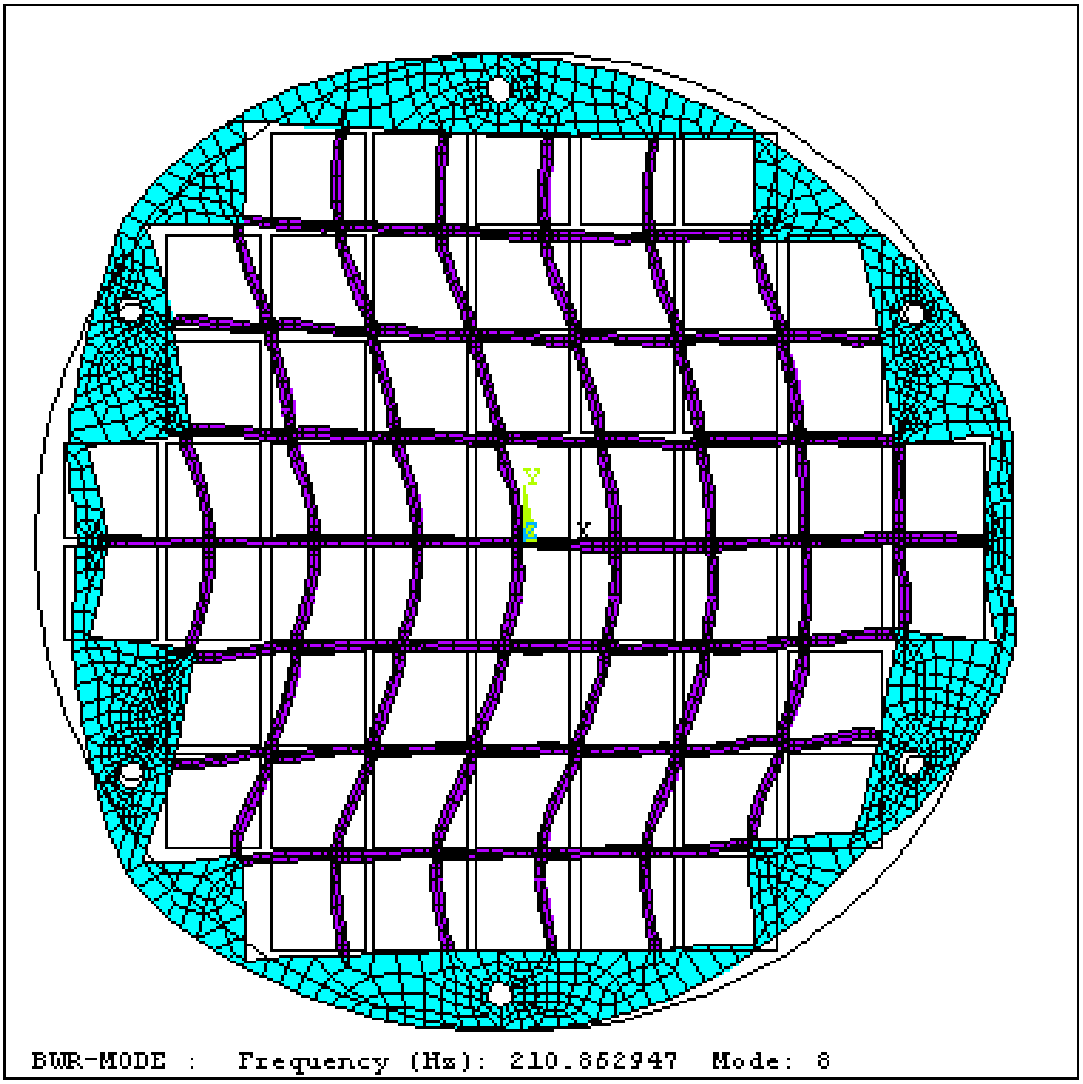
Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Figure 11.2.12.4.2-6 BWR – 80.2 Hz Mode Shape



Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Figure 11.2.12.4.2-7 BWR – 210.9 Hz Mode Shape



Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Table 11.2.12.4.2-1 Canister Primary Membrane (P_m) Stresses for Tip-Over Conditions - BWR - 49.46° Basket Drop Orientation (ksi)

Section Location[(1)]	Section Angle (deg)	Sx	Sy	Sz	Sxy	Syz	Sxz	Stress Intensity	Allowable Stress	Margin of Safety
1	0	-1.2	6.2	1.4	-0.1	-0.1	0.0	7.46	35.52	3.76
2	0	-1.6	8.2	1.4	0.0	-0.2	0.1	9.77	35.52	2.63
3	0	-1.5	7.9	1.4	0.0	-0.2	-0.1	9.41	35.52	2.78
4	90	-0.1	3.0	-2.1	-0.2	3.7	0.1	8.92	35.52	2.98
5	85.5	0.0	2.8	-1.0	-0.2	4.8	-0.1	10.29	35.52	2.45
6	76.5	0.0	0.3	-0.4	0.0	6.0	0.0	12.09	35.52	1.94
7[(2)]	9.0	0.6	0.3	4.8	1.6	-3.8	-0.2	9.60	35.52	2.70
8[(2)]	351.0	4.5	0.1	5.2	-0.1	2.3	-0.6	7.06	35.52	4.03
9[(2)]	351.0	4.5	-1.0	1.5	-1.6	2.8	-0.2	8.17	35.52	3.35
10	0	-38.6	-16.2	-30.4	0.5	0.0	-10.7	29.74	40.08[(3)]	0.35
11[(4)]	351.9 – 8.2	-22.1	-9.9	-6.7	-0.1	0.0	1.1	15.51	32.06[(4)]	1.07
12	0	-0.6	0.2	0.0	0.0	0.0	-0.3	0.92	35.52	37.66
13	0	-1.0	0.3	0.0	0.0	0.0	-0.4	1.46	35.52	23.31

Stresses are presented in the cylindrical coordinate system, x = radial, y = circumferential and z = axial directions.

1. Section locations are shown in Figure 11.2.12.4.1-6.
2. Stresses are not presented for the sections with localized bearing stress. In accordance with ASME Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions.
3. Allowable stress at 300°F.
4. Stresses are determined by averaging the stresses over the impact region. A stress reduction factor of 0.8 is applied to the allowable stress at 250°F.

Table 11.2.12.4.2-2 Canister Primary Membrane + Primary Bending ($P_m + P_b$) Stresses for Tip-Over Conditions - BWR - 49.46° Basket Drop Orientation (ksi)

Section Location[(1)]	Section Angle (deg)	Sx	Sy	Sz	Sxy	Syz	Sxz	Stress Intensity	Allowable Stress	Margin of Safety
1	0.0	-1.6	18.5	4.6	-0.2	-0.4	0.1	20.13	53.28	1.65
2	0.0	-1.8	20.2	2.7	0.0	-0.4	0.1	22.01	53.28	1.42
3	0.0	-2.3	20.6	4.8	-0.1	-0.3	-0.1	22.92	53.28	1.32
4	0.0	-1.8	20.2	3.9	-0.2	-0.4	-0.1	22.00	53.28	1.42
5	0.0	-2.2	19.7	6.4	-0.1	-0.6	0.1	21.94	53.28	1.43
6	0.0	0.0	-21.0	-3.8	0.0	-0.7	-0.7	21.21	53.28	1.51
7[(2)]	351.0	0.1	6.4	17.2	0.2	2.3	0.2	17.50	53.28	2.04
8[(2)]	351.0	3.3	5.2	13.5	0.7	3.6	-2.1	13.02	53.28	3.09
9[(2)]	351.0	5.9	-3.0	3.6	-3.0	3.2	-0.6	12.44	53.28	3.28
10	0.0	-42.9	-15.8	-27.8	0.4	0.3	-19.1	41.17	60.12[(3)]	0.46
11[(4)]	351.9 – 8.1	-18.8	-7.2	-1.7	-0.1	0.0	2.6	17.86	48.09[(4)]	1.69
12	0.0	-0.9	0.1	-0.1	0.0	0.0	-0.5	1.37	53.28	37.81
13	0.0	-1.1	0.4	0.0	0.0	0.0	-0.1	1.56	53.28	33.07

Stresses are presented in the cylindrical coordinate system, x = radial, y = circumferential and z = axial directions.

1. Section locations are shown in Figure 11.2.12.4.1-6.
2. Stresses are not presented for the sections with localized bearing stress. In accordance with ASME Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions.
3. Allowable stress at 300°F.
4. Stresses are determined by averaging the stresses over the impact region. A stress reduction factor of 0.8 is applied to the allowable stress at 250°F.

Table 11.2.12.4.2-3 Support Disk Section Locations for Stress Evaluation - BWR - Full Model

Section[1]	Point 1		Point 2	
	X	Y	X	Y
1	3.14	6.6	3.79	6.6
2	3.14	3.46	3.79	3.46
3	3.14	0.33	3.79	0.33
4	-3.14	6.6	-3.79	6.6
5	-3.14	3.46	-3.79	3.46
6	-3.14	0.33	-3.79	0.33
7	10.07	6.6	10.72	6.6
8	10.07	3.46	10.72	3.46
9	10.07	0.33	10.72	0.33
10	17	6.6	17.65	6.6
11	17	3.46	17.65	3.46
12	17	0.33	17.65	0.33
13	23.92	6.6	24.57	6.6
14	23.92	3.46	24.57	3.46
15	23.92	0.33	24.57	0.33
16	3.14	13.53	3.79	13.53
17	3.14	10.39	3.79	10.39
18	3.14	7.25	3.79	7.25
19	-3.14	13.53	-3.79	13.53
20	-3.14	10.39	-3.79	10.39
21	-3.14	7.25	-3.79	7.25
22	10.07	13.53	10.72	13.53
23	10.07	10.39	10.72	10.39
24	10.07	7.25	10.72	7.25
25	17	13.53	17.65	13.53
26	17	10.39	17.65	10.39
27	17	7.25	17.65	7.25
28	3.14	20.46	3.79	20.46
29	3.14	17.32	3.79	17.32
30	3.14	14.18	3.79	14.18
31	-3.14	20.46	-3.79	20.46
32	-3.14	17.32	-3.79	17.32
33	-3.14	14.18	-3.79	14.18
34	10.07	20.46	10.72	20.46
35	10.07	17.32	10.72	17.32
36	10.07	14.18	10.72	14.18
37	17	20.46	17.65	20.46
38	17	17.32	17.65	17.32
39	17	14.18	17.65	14.18
40	3.14	27.39	3.79	27.39
41	3.14	24.25	3.79	24.25
42	3.14	21.11	3.79	21.11
43	-3.14	27.39	-3.79	27.39
44	-3.14	24.25	-3.79	24.25
45	-3.14	21.11	-3.79	21.11
46	10.07	27.39	10.72	27.39
47	10.07	24.25	10.72	24.25
48	10.07	21.11	10.72	21.11
49	3.14	-0.33	3.79	-0.33
50	3.14	-3.46	3.79	-3.46
51	3.14	-6.6	3.79	-6.6
52	-3.14	-0.33	-3.79	-0.33
53	-3.14	-3.46	-3.79	-3.46
54	-3.14	-6.6	-3.79	-6.6
55	10.07	-0.33	10.72	-0.33
56	10.07	-3.46	10.72	-3.46
57	10.07	-6.6	10.72	-6.6
58	17	-0.33	17.65	-0.33
59	17	-3.46	17.65	-3.46
60	17	-6.6	17.65	-6.6
61	23.92	-0.33	24.57	-0.33
62	23.92	-3.46	24.57	-3.46
63	23.92	-6.6	24.57	-6.6
64	3.14	-7.25	3.79	-7.25
65	3.14	-10.39	3.79	-10.39
66	3.14	-13.53	3.79	-13.53
67	-3.14	-7.25	-3.79	-7.25
68	-3.14	-10.39	-3.79	-10.39
69	-3.14	-13.53	-3.79	-13.53
70	10.07	-7.25	10.72	-7.25
71	10.07	-10.39	10.72	-10.39
72	10.07	-13.53	10.72	-13.53
73	17	-7.25	17.65	-7.25
74	17	-10.39	17.65	-10.39
75	17	-13.53	17.65	-13.53
76	3.14	-14.18	3.79	-14.18
77	3.14	-17.32	3.79	-17.32
78	3.14	-20.46	3.79	-20.46
79	-3.14	-14.18	-3.79	-14.18
80	-3.14	-17.32	-3.79	-17.32
81	-3.14	-20.46	-3.79	-20.46
82	10.07	-14.18	10.72	-14.18
83	10.07	-17.32	10.72	-17.32
84	10.07	-20.46	10.72	-20.46
85	17	-14.18	17.65	-14.18
86	17	-17.32	17.65	-17.32

1. See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-3 Support Disk Section Locations for Stress Evaluation - BWR - Full Model (Continued)

Section[1]	Point 1		Point 2		Section[1]	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
87	17	-20.46	17.65	-20.46	130	-10.07	-7.25	-10.72	-7.25
88	3.14	-21.11	3.79	-21.11	131	-10.07	-10.39	-10.72	-10.39
89	3.14	-24.25	3.79	-24.25	132	-10.07	-13.53	-10.72	-13.53
90	3.14	-27.39	3.79	-27.39	133	-17	-7.25	-17.65	-7.25
91	-3.14	-21.11	-3.79	-21.11	134	-17	-10.39	-17.65	-10.39
92	-3.14	-24.25	-3.79	-24.25	135	-17	-13.53	-17.65	-13.53
93	-3.14	-27.39	-3.79	-27.39	136	-10.07	-14.18	-10.72	-14.18
94	10.07	-21.11	10.72	-21.11	137	-10.07	-17.32	-10.72	-17.32
95	10.07	-24.25	10.72	-24.25	138	-10.07	-20.46	-10.72	-20.46
96	10.07	-27.39	10.72	-27.39	139	-17	-14.18	-17.65	-14.18
97	-10.07	6.6	-10.72	6.6	140	-17	-17.32	-17.65	-17.32
98	-10.07	3.46	-10.72	3.46	141	-17	-20.46	-17.65	-20.46
99	-10.07	0.33	-10.72	0.33	142	-10.07	-21.11	-10.72	-21.11
100	-17	6.6	-17.65	6.6	143	-10.07	-24.25	-10.72	-24.25
101	-17	3.46	-17.65	3.46	144	-10.07	-27.39	-10.72	-27.39
102	-17	0.33	-17.65	0.33	145	3.14	6.6	3.14	7.25
103	-23.92	6.6	-24.57	6.6	146	0	6.6	0	7.25
104	-23.92	3.46	-24.57	3.46	147	-3.14	6.6	-3.14	7.25
105	-23.92	0.33	-24.57	0.33	148	3.14	0.33	3.14	-0.33
106	-10.07	13.53	-10.72	13.53	149	0	0.33	0	-0.33
107	-10.07	10.39	-10.72	10.39	150	-3.14	0.33	-3.14	-0.33
108	-10.07	7.25	-10.72	7.25	151	10.07	6.6	10.07	7.25
109	-17	13.53	-17.65	13.53	152	6.93	6.6	6.93	7.25
110	-17	10.39	-17.65	10.39	153	3.79	6.6	3.79	7.25
111	-17	7.25	-17.65	7.25	154	10.07	0.33	10.07	-0.33
112	-10.07	20.46	-10.72	20.46	155	6.93	0.33	6.93	-0.33
113	-10.07	17.32	-10.72	17.32	156	3.79	0.33	3.79	-0.33
114	-10.07	14.18	-10.72	14.18	157	17	6.6	17	7.25
115	-17	20.46	-17.65	20.46	158	13.86	6.6	13.86	7.25
116	-17	17.32	-17.65	17.32	159	10.72	6.6	10.72	7.25
117	-17	14.18	-17.65	14.18	160	17	0.33	17	-0.33
118	-10.07	27.39	-10.72	27.39	161	13.86	0.33	13.86	-0.33
119	-10.07	24.25	-10.72	24.25	162	10.72	0.33	10.72	-0.33
120	-10.07	21.11	-10.72	21.11	163	23.92	6.6	23.92	7.25
121	-10.07	-0.33	-10.72	-0.33	164	20.78	6.6	20.78	7.25
122	-10.07	-3.46	-10.72	-3.46	165	17.65	6.6	17.65	7.25
123	-10.07	-6.6	-10.72	-6.6	166	23.92	0.33	23.92	-0.33
124	-17	-0.33	-17.65	-0.33	167	20.78	0.33	20.78	-0.33
125	-17	-3.46	-17.65	-3.46	168	17.65	0.33	17.65	-0.33
126	-17	-6.6	-17.65	-6.6	169	30.85	0.33	30.85	-0.33
127	-23.92	-0.33	-24.57	-0.33	170	27.71	0.33	27.71	-0.33
128	-23.92	-3.46	-24.57	-3.46	171	24.57	0.33	24.57	-0.33
129	-23.92	-6.6	-24.57	-6.6	172	3.14	13.53	3.14	14.18

1. See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-3 Support Disk Section Locations for Stress Evaluation - BWR - Full Model (Continued)

Section[1]	Point 1		Point 2		Section[1]	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
173	0	13.53	0	14.18	216	17.65	-13.53	17.65	-14.18
174	-3.14	13.53	-3.14	14.18	217	3.14	-20.46	3.14	-21.11
175	10.07	13.53	10.07	14.18	218	0	-20.46	0	-21.11
176	6.93	13.53	6.93	14.18	219	-3.14	-20.46	-3.14	-21.11
177	3.79	13.53	3.79	14.18	220	10.07	-20.46	10.07	-21.11
178	17	13.53	17	14.18	221	6.93	-20.46	6.93	-21.11
179	13.86	13.53	13.86	14.18	222	3.79	-20.46	3.79	-21.11
180	10.72	13.53	10.72	14.18	223	17	-20.46	17	-21.11
181	23.92	13.53	23.92	14.18	224	13.86	-20.46	13.86	-21.11
182	20.78	13.53	20.78	14.18	225	10.72	-20.46	10.72	-21.11
183	17.65	13.53	17.65	14.18	226	-3.79	6.6	-3.79	7.25
184	3.14	20.46	3.14	21.11	227	-6.93	6.6	-6.93	7.25
185	0	20.46	0	21.11	228	-10.07	6.6	-10.07	7.25
186	-3.14	20.46	-3.14	21.11	229	-3.79	0.33	-3.79	-0.33
187	10.07	20.46	10.07	21.11	230	-6.93	0.33	-6.93	-0.33
188	6.93	20.46	6.93	21.11	231	-10.07	0.33	-10.07	-0.33
189	3.79	20.46	3.79	21.11	232	-10.72	6.6	-10.72	7.25
190	17	20.46	17	21.11	233	-13.86	6.6	-13.86	7.25
191	13.86	20.46	13.86	21.11	234	-17	6.6	-17	7.25
192	10.72	20.46	10.72	21.11	235	-10.72	0.33	-10.72	-0.33
193	3.14	-6.6	3.14	-7.25	236	-13.86	0.33	-13.86	-0.33
194	0	-6.6	0	-7.25	237	-17	0.33	-17	-0.33
195	-3.14	-6.6	-3.14	-7.25	238	-17.65	6.6	-17.65	7.25
196	10.07	-6.6	10.07	-7.25	239	-20.78	6.6	-20.78	7.25
197	6.93	-6.6	6.93	-7.25	240	-23.92	6.6	-23.92	7.25
198	3.79	-6.6	3.79	-7.25	241	-17.65	0.33	-17.65	-0.33
199	17	-6.6	17	-7.25	242	-20.78	0.33	-20.78	-0.33
200	13.86	-6.6	13.86	-7.25	243	-23.92	0.33	-23.92	-0.33
201	10.72	-6.6	10.72	-7.25	244	-24.57	0.33	-24.57	-0.33
202	23.92	-6.6	23.92	-7.25	245	-27.71	0.33	-27.71	-0.33
203	20.78	-6.6	20.78	-7.25	246	-30.85	0.33	-30.85	-0.33
204	17.65	-6.6	17.65	-7.25	247	-3.79	13.53	-3.79	14.18
205	3.14	-13.53	3.14	-14.18	248	-6.93	13.53	-6.93	14.18
206	0	-13.53	0	-14.18	249	-10.07	13.53	-10.07	14.18
207	-3.14	-13.53	-3.14	-14.18	250	-10.72	13.53	-10.72	14.18
208	10.07	-13.53	10.07	-14.18	251	-13.86	13.53	-13.86	14.18
209	6.93	-13.53	6.93	-14.18	252	-17	13.53	-17	14.18
210	3.79	-13.53	3.79	-14.18	253	-17.65	13.53	-17.65	14.18
211	17	-13.53	17	-14.18	254	-20.78	13.53	-20.78	14.18
212	13.86	-13.53	13.86	-14.18	255	-23.92	13.53	-23.92	14.18
213	10.72	-13.53	10.72	-14.18	256	-3.79	20.46	-3.79	21.11
214	23.92	-13.53	23.92	-14.18	257	-6.93	20.46	-6.93	21.11
215	20.78	-13.53	20.78	-14.18	258	-10.07	20.46	-10.07	21.11

1. See Figure 11.2.12.4.2-4 for section locations.

Table 11.12.12.4.2-3 Support Disk Section Locations for Stress Evaluation - BWR - Full Model (Continued)

Section[1]	Point 1		Point 2		Section[1]	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
259	-10.72	20.46	-10.72	21.11	289	3.14	27.39	3.14	32.63
260	-13.86	20.46	-13.86	21.11	290	3.79	27.39	3.79	32.56
261	-17	20.46	-17	21.11	291	10.07	27.39	10.07	31.2
262	-3.79	-6.6	-3.79	-7.25	292	10.72	27.39	10.72	30.98
263	-6.93	-6.6	-6.93	-7.25	293	17	27.39	17.29	27.86
264	-10.07	-6.6	-10.07	-7.25	294	30.85	-0.33	32.78	-0.33
265	-10.72	-6.6	-10.72	-7.25	295	30.85	-6.6	32.06	-6.86
266	-13.86	-6.6	-13.86	-7.25	296	-3.14	-27.39	-3.14	-32.63
267	-17	-6.6	-17	-7.25	297	3.14	-27.39	3.14	-32.63
268	-17.65	-6.6	-17.65	-7.25	298	3.79	-27.39	3.79	-32.56
269	-20.78	-6.6	-20.78	-7.25	299	10.07	-27.39	10.07	-31.2
270	-23.92	-6.6	-23.92	-7.25	300	10.72	-27.39	10.72	-30.98
271	-3.79	-13.53	-3.79	-14.18	301	17	-27.39	17.29	-27.86
272	-6.93	-13.53	-6.93	-14.18	302	-30.85	6.6	-32.06	6.86
273	-10.07	-13.53	-10.07	-14.18	303	-30.85	0.33	-32.78	0.33
274	-10.72	-13.53	-10.72	-14.18	304	-10.07	27.39	-10.07	31.2
275	-13.86	-13.53	-13.86	-14.18	305	-3.79	27.39	-3.79	32.56
276	-17	-13.53	-17	-14.18	306	-17	27.39	-17.29	27.86
277	-17.65	-13.53	-17.65	-14.18	307	-10.72	27.39	-10.72	30.98
278	-20.78	-13.53	-20.78	-14.18	308	-30.85	-0.33	-32.78	-0.33
279	-23.92	-13.53	-23.92	-14.18	309	-30.85	-6.6	-32.06	-6.86
280	-3.79	-20.46	-3.79	-21.11	310	-10.07	-27.39	-10.07	-31.2
281	-6.93	-20.46	-6.93	-21.11	311	-3.79	-27.39	-3.79	-32.56
282	-10.07	-20.46	-10.07	-21.11	312	-17	-27.39	-17.29	-27.86
283	-10.72	-20.46	-10.72	-21.11	313	-10.72	-27.39	-10.72	-30.98
284	-13.86	-20.46	-13.86	-21.11	314	23.92	20.46	24.92	21.31
285	-17	-20.46	-17	-21.11	315	23.92	-20.46	24.92	-21.31
286	30.85	6.6	32.06	6.86	316	-23.92	20.46	-24.92	21.31
287	30.85	0.33	32.78	0.33	317	-23.92	-20.46	-24.92	-21.31
288	-3.14	27.39	-3.14	32.63					

1. See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-4 Summary of Maximum Stresses for BWR Support Disk for Tip-Over Condition

Drop Orientation	P_m			$P_m + P_b$		
	Stress Intensity (ksi)	**Allowable Stress (ksi)**	**Margin of Safety**	**Stress Intensity (ksi)**	**Allowable Stress (ksi)**	**Margin of Safety**
0°	35.1	63.0	+0.80	46.1	90.0	+0.95
31.82°	25.8	63.0	+1.44	65.7	90.0	+0.37
49.46°	23.7	63.0	+1.65	55.5	90.0	+0.62
77.92°	47.5	63.0	+0.33	86.6	90.0	+0.04
90°	58.4	63.0	+0.08	69.6	90.0	+0.29

Note: See Figure 11.2.12.4.2-1 for Drop Orientation.

Table 11.2.12.4.2-5 Summary of Buckling Evaluation of BWR Support Disk for Tip-Over Condition

Drop orientation	MS1	MS2
0°	1.17	1.03
31.82°	0.56	0.53
49.46°	0.86	0.81
77.92°	0.18	0.16
90°	0.38	0.58

Table 11.2.12.4.2-6 Support Disk Primary Membrane (P_m) Stresses for Tip-Over Condition – BWR Disk No. 5 - 77.92° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
202	-24.9	22.5	1	47.5	63.0	0.33
199	-21.8	14.8	1.3	36.6	63.0	0.72
196	-18.8	12.5	1.3	31.4	63.0	1.01
193	-16	11.2	1.3	27.2	62.8	1.30
63	-18.3	8.5	2.4	27.2	63.0	1.32
203	-24.9	-0.1	0.8	24.9	63.0	1.53
204	-24.8	-16.1	0.7	24.9	63.0	1.53
262	-13.2	10.3	1.3	23.7	62.8	1.65
201	-21.7	-16	1	21.9	63.0	1.88
200	-21.7	0	1.1	21.8	63.0	1.89
73	-18.6	2.1	-0.6	20.8	63.0	2.03
265	-10.6	9.8	1.2	20.6	63.0	2.06
166	-12.3	7.9	1.6	20.4	63.0	2.09
169	-13.9	-19.2	2.3	20.0	63.0	2.15
198	-18.7	-15.1	1	19.0	62.8	2.31
197	-18.8	0	1.1	18.9	63.0	2.34
295	-6	-15.6	-6.3	18.7	63.0	2.37
15	-9.1	8.2	2.5	18.0	63.0	2.50
268	-8.1	9.7	0.9	17.8	63.0	2.53
195	-15.9	-14.2	1	16.3	62.8	2.85
194	-15.9	0	1.1	16.1	62.8	2.91
211	-12.2	3.6	0.6	15.8	63.0	2.98
60	-12.3	2.7	2.5	15.8	63.0	2.99
61	-6.8	8.5	1	15.5	63.0	3.06
160	-10.7	4.2	1.9	15.4	63.0	3.10
171	-13.8	0.8	2	15.2	63.0	3.15
70	-14.6	0.2	-0.3	14.9	63.0	3.24
170	-13.9	0	2.1	14.5	63.0	3.34
264	-13.2	-13.2	1	14.1	63.0	3.46
13	-5.7	8.2	1	14.1	63.0	3.48

See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-7 Support Disk Primary Membrane + Primary Bending (P_m+P_b) Stresses for Tip-Over Condition - BWR Disk No. 5 - 77.92° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
169	-85.6	-34.9	7.1	86.6	90.0	0.04
202	-50.9	15.4	-2.3	66.5	90.0	0.35
63	1.2	63.9	-1.5	63.9	90.0	0.41
160	-61.6	-14.9	1.5	61.7	90.0	0.46
171	-60	-17.6	3	60.2	90.0	0.49
60	3.8	59.5	0.4	59.5	90.0	0.51
57	4.8	59.1	0.1	59.1	90.0	0.52
15	10.2	58.9	1.1	59.0	90.0	0.53
51	-28.2	-57	4.7	57.7	89.5	0.55
154	-57.6	-16.5	1.6	57.7	89.8	0.56
199	-54.3	3	-1.4	57.3	90.0	0.57
162	-56.8	-22.8	3.4	57.1	89.9	0.57
54	-26	-55.3	4.3	55.9	89.5	0.60
156	-54.4	-22.8	3.3	54.8	87.8	0.60
148	-54.3	-16.2	1.5	54.4	87.6	0.61
9	14.6	54.1	1.5	54.1	89.8	0.66
166	-54.1	-9.7	0.5	54.1	90.0	0.66
3	-25.2	-52.1	3.5	52.6	87.6	0.67
13	3.7	53.7	1.1	53.7	90.0	0.68
12	15.2	53.5	2.1	53.6	90.0	0.68
123	-23.9	-52.9	3.9	53.4	90.0	0.69
150	-51.3	-22.4	3.2	51.7	87.6	0.69
6	-23.6	-51.1	3.3	51.5	87.6	0.70
229	-51.1	-15.6	1.3	51.2	87.8	0.71
201	-50.2	-27.9	6.7	52.0	90.0	0.73
196	-51.2	-0.2	-1	51.3	90.0	0.76
168	-50.4	-19.2	2.9	50.7	90.0	0.78
198	-48.4	-27.4	6.3	50.1	89.5	0.79
99	-22.1	-49.4	3.1	49.7	89.8	0.81
231	-48.5	-21.6	3	48.8	89.8	0.84

See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-8 Summary of Support Disk Buckling Evaluation for Tip-Over Condition - BWR Disk No. 5 - 77.92° Drop Orientation

Section Number	P (kip)	Pcr (kip)	Py (kip)	M (in-kip)	Mp (in-kip)	Mm (in-kip)	MS1	MS2
169	5.65	31.59	25.67	3.15	4.17	4.11	0.18	0.16
199	8.84	31.4	25.52	1.43	4.15	4.09	0.69	0.57
171	5.62	31.52	25.62	2.03	4.16	4.1	0.64	0.58
160	4.34	31.35	25.48	2.24	4.14	4.08	0.63	0.59
202	10.12	31.55	25.64	1.14	4.17	4.11	0.76	0.59
201	8.82	31.23	25.38	1.25	4.12	4.07	0.80	0.65
196	7.63	31.22	25.37	1.43	4.12	4.07	0.81	0.68
162	4.32	31.1	25.28	2.03	4.11	4.05	0.74	0.70
154	3.7	31.07	25.26	2.14	4.1	4.05	0.74	0.70
204	10.09	31.41	25.53	0.88	4.15	4.09	0.95	0.74
198	7.61	30.97	25.18	1.31	4.09	4.04	0.89	0.75
156	3.67	30.35	24.73	2	4.02	3.97	0.80	0.75
166	4.98	31.51	25.61	1.84	4.16	4.1	0.82	0.76
148	3.05	30.27	24.67	2.06	4.01	3.96	0.82	0.79
193	6.48	30.96	25.18	1.41	4.09	4.04	0.94	0.82
168	4.96	31.36	25.49	1.68	4.14	4.08	0.94	0.86
150	3.02	30.27	24.67	1.93	4.01	3.96	0.92	0.88
51	0.11	30.96	25.18	2.5	4.09	4.04	0.89	0.92
195	6.46	30.96	25.18	1.3	4.09	4.04	1.04	0.90
229	2.39	30.35	24.73	1.99	4.02	3.97	0.96	0.94
54	0.26	30.96	25.18	2.4	4.09	4.04	0.94	0.97
262	5.37	30.97	25.18	1.39	4.09	4.04	1.11	0.99
123	0.25	31.22	25.37	2.3	4.12	4.07	1.04	1.07
6	0.14	30.27	24.67	2.24	4.01	3.96	1.06	1.09
231	2.36	31.07	25.26	1.88	4.1	4.05	1.11	1.08
264	5.35	31.22	25.37	1.29	4.12	4.07	1.23	1.10
99	0.15	31.07	25.26	2.16	4.1	4.05	1.18	1.22
235	1.73	31.1	25.28	1.87	4.11	4.05	1.21	1.20
265	4.31	31.23	25.38	1.32	4.12	4.07	1.38	1.27
237	1.7	31.35	25.48	1.82	4.14	4.08	1.29	1.28

See Figure 11.2.12.4.2-4 for section locations.

11.2.12.5 Corrective Actions

Following the accident event, perform the required Response Surveillance in accordance with Section A 5.4 of the Technical Specifications. Corrective actions shall be taken in accordance with the surveillance requirements to return the affected system to a safe operating condition, as applicable to the affected component(s).

The most important recovery action required following a concrete cask tip-over is the uprighting of the cask to minimize the dose rate from the exposed bottom end. The uprighting operation will require a heavy lift capability and rigging expertise. The concrete cask must be returned to the vertical position by rotation around a convenient bottom edge, and by using a method and rigging that controls the rotation to the vertical position.

Surface and top and bottom edges of the concrete cask are expected to exhibit cracking and possibly loss of concrete down to the layer of reinforcing bar. If only minor damage occurs, the concrete may be repairable by using grout. Otherwise, it may be necessary to remove the canister for installation in a new concrete cask. If the canister remains in the cask, it should be returned to its centered storage position within the cask, in accordance with Section 8.1.2, Item 17, Note 1.

The storage pad, if damaged, must be repaired to preclude the intrusion of water that could cause further deterioration of the pad in freeze-thaw cycles.

11.2.12.6 Radiological Impact

There is an adverse radiological consequence in the hypothetical tip-over event since the bottom end of the concrete cask and the canister have significantly less shielding than the sides and tops of these same components. The dose rate at 1 meter is calculated, using a 1–D analysis, to be approximately 34 rem/hour, and the dose at 4 meters is estimated to be approximately 4 rem/hour. Consequently, following a tip-over event, supplemental shielding should be used until the concrete cask can be uprighted. Stringent access controls must be applied to ensure that personnel do not enter the area of radiation shine from the exposed bottom of the tipped-over concrete cask.

Damage to the edges or surface of the concrete cask may occur following a tip-over, which could result in marginally higher dose rates at the bottom edge or at surface cracks in the concrete. This increased dose rate is not expected to be significant, and would be dependent on the specific damage incurred.

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11.2.13 Full Blockage of Vertical Concrete Cask Air Inlets and Outlets

This section evaluates the Vertical Concrete Cask for the steady state effects of full blockage of the air inlets and outlets at the normal ambient temperature (76°F). It estimates the duration of the event that results in the fuel cladding, the fuel basket and the concrete reaching their design basis limiting temperatures (See Table 4.1-3 for the allowable temperatures for short-term conditions).

The evaluation demonstrates that there are no adverse consequences due to this accident, provided that the full blockage of the concrete cask inlets and outlets is cleared within 24 hours.

11.2.13.1 Cause of Full Blockage

The likely cause of complete cask air inlet and outlet blockage is the covering of the cask with earth in a catastrophic event that is significantly greater than the design basis earthquake or a landslide. This event is a bounding accident and is not credible.

11.2.13.2 Detection of Full Blockage

Blockage of the cask air inlets and outlets will be visually detected during the general site inspection following an earthquake, land slide, or other events with a potential for such blockage. In addition, a daily surveillance of the concrete cask to verify operability limits the potential for a full blockage event to go undetected.

11.2.13.3 Analysis of Full Blockage

The accident temperature conditions are evaluated using the thermal models described in Section 4.4.1. The analysis assumes initial normal storage conditions, with the sudden loss of convective cooling of the canister. Heat is then rejected from the canister to the Vertical Concrete Cask liner by radiation and conduction. The loss of convective cooling results in the fairly rapid and sustained heat-up of the canister and the concrete cask. To account for the loss of convective cooling in the ANSYS air flow model (Section 4.4.1.1), the elements in the model are replaced with thermal conduction elements. This model is used to evaluate the thermal transient resulting from the postulated boundary conditions. The analysis indicates that the maximum basket temperature (support disk and heat transfer disk) remain less than the allowable temperature for 24 hours after the initiation of the event. The maximum fuel cladding temperature and the maximum concrete bulk temperature remain less than the allowable temperatures for about 6 days (150 hours) after the initiation of the event. The heat-up of the fuel cladding, canister shell and concrete (bulk temperature) is shown in Figures 11.2.13-1 and 11.2.13-2, for the PWR and BWR configurations, respectively.

11.2.13.4 Corrective Actions

Following the natural phenomenon event, perform the required Response Surveillance in accordance with Section A 5.4 of the Technical Specifications. Corrective actions shall be taken in accordance with the surveillance requirements to return the affected system to a safe operating condition, as applicable to the affected component(s).

Following any event that could cause blockage of the concrete cask inlets and outlets, concrete casks shall be restored to operable status in accordance with LCO A 3.1.6 of the Technical Specifications. Optional temperature-monitoring equipment, if used, should be verified as operable, or repaired and returned to service.

11.2.13.5 Radiological Impact

There are no significant radiological consequences for this event, as the Vertical Concrete Cask retains its shielding performance. Dose is incurred as a consequence of uncovering the concrete cask and vent system. Since the dose rates at the air inlets and outlets are higher than the nominal rate (35 mrem/hr) at the cask wall, personnel will be subject to an estimated maximum dose rate of 100 mrem/hr when clearing the inlets and outlets. If it is assumed that a worker kneeling with his hands on the inlets or outlets requires 15 minutes to clear each inlet or outlet, the estimated extremity dose is 200 mrem for the 8 openings. The whole body dose will be slightly less. In addition, some dose is incurred clearing debris away from the cask body. This dose is estimated at 50 mrem, assuming 2 hours is spent near the cask exterior surface.

Figure 11.2.13-1 PWR Configuration Temperature History—All Vents Blocked

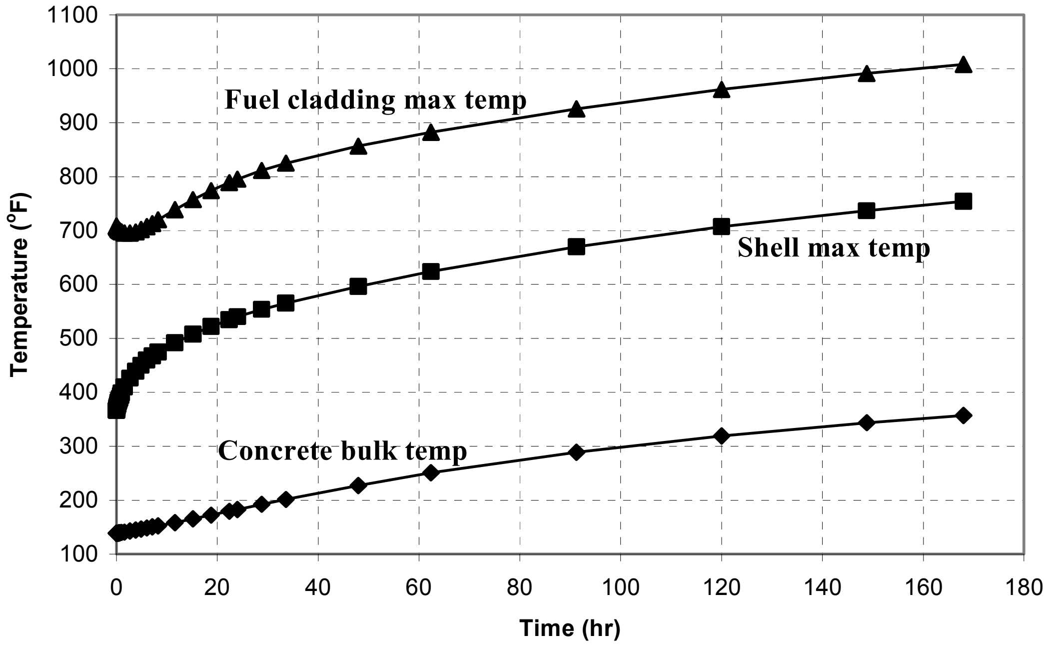
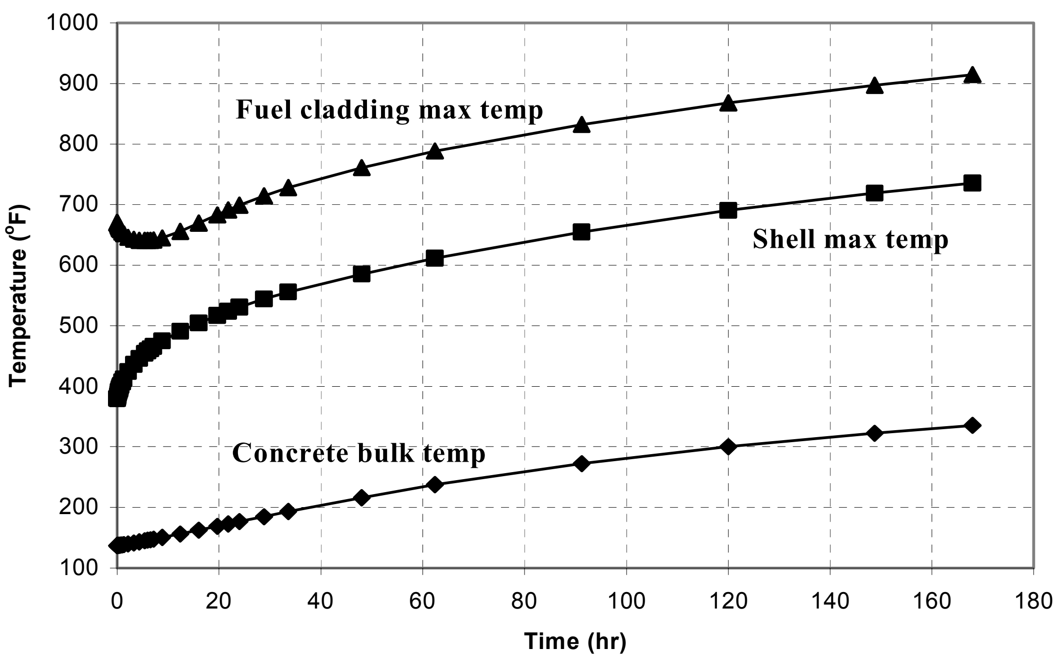


Figure 11.2.13-2 BWR Configuration Temperature History—All Vents Blocked



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11.2.14 Canister Closure Weld Evaluation

The closure weld for the canister is a groove weld with a thickness of 0.75 inches. The evaluation of this weld, in accordance with NRC guidance, is to incorporate a 0.8 stress reduction factor. Applying a factor of 0.8 to the weld stress allowable incorporates the stress reduction factor.

The stresses for the canister are evaluated using sectional stresses as permitted by Subsection NB of the ASME Code. Canister stresses resulting from the concrete cask tip-over accident (Section 11.2.12.4) are used for evaluation. The location of the section for the canister weld evaluation is shown in Figure 11.2.12.4.1-6 and corresponds to Section 11. The governing P_m and $P_m + P_b$ stress intensities for Section 11 and the associated allowables are listed in Tables 11.2.12.4.1-1 and Table 11.2.12.4.1-2, respectively. The factored allowables, incorporating a 0.8 stress reduction factor, and the resulting controlling Margins of Safety are:

Stress Category	Analysis Stress (ksi)	0.8 × Allowable Stress (ksi)	Margin of Safety
P_m	24.80	32.06	0.29
$P_m + P_b$	29.25	48.09	0.64

This confirms that the canister closure weld is acceptable for accident conditions.

Critical Flaw Size for the Canister Closure Weld

The closure weld for the canister is comprised of multiple weld beads using a compatible weld material for Type 304L stainless steel. An allowable (critical) flaw evaluation has been performed to determine the critical flaw size in the weld region. The result of the flaw evaluation is used to define the minimum flaw size, which must be identifiable in the nondestructive examination of the weld. Due to the inherent toughness associated with Type 304L stainless steel, a limit load analysis is used in conjunction with a J-integral/tearing modulus approach. The safety margins used in this evaluation correspond to the stress limits contained in Section XI of the ASME Code [63].

One of the stress components used in the evaluation for the critical flaw size is the radial stress component in the weld region of the structural lid. For an accident (Level D) event, in accordance with ASME Code Section XI, a safety factor of $\sqrt{2}$ is required. For the purpose of identifying the stress for the flaw evaluation, the weld region corresponds to Section 11 in Figure 11.2.12.4.1-6 is considered.

The maximum tensile radial stress at Section 11 is 6.9 ksi, based on the analysis results of the tip-over accident (Section 11.2.12.4). To perform the flaw evaluation, a 10 ksi stress is conservatively used, resulting in a significantly larger safety factor than the required safety factor of $\sqrt{2}$. Using 10 ksi as the basis for the evaluation, the minimum detectable flaw size is 0.44 inch for a flaw that extends 360 degrees around the circumference of the canister. Stress components for the circumferential and axial directions are also reported in the concrete cask tip-over analysis, which would be associated with flaws oriented in the radial or horizontal directions respectively. The maximum stress for these components is 4.0 ksi, which is also enveloped by the value of 10 ksi used in the critical flaw evaluation for stresses in the radial direction. The 360-degree flaw employed for the circumferential direction is considered to be bounding with respect to any partial flaw in the weld, which could occur in the radial and horizontal directions. Therefore, using a minimum detectable flaw size of 0.375 inch is acceptable, since it is less than the 0.44-inch critical flaw size.

11.2.15 Accident and Natural Phenomena Events Evaluation for Site Specific Spent Fuel

This section presents the accident and natural phenomena events evaluation of spent fuel assemblies or configurations, which are unique to specific reactor sites. These site specific fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blankets and variable enrichment assemblies, fuel with burnup that exceeds the design basis, and fuel that is classified as damaged. Damaged fuel includes fuel rods with cladding that exhibits defects greater than pinhole leaks or hairline cracks.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly of the same type (PWR or BWR), or are shown to be acceptable contents, by specific evaluation of the configuration.

11.2.15.1 Accident and Natural Phenomena Events Evaluation for Maine Yankee Site Specific Spent Fuel

Maine Yankee site specific fuels are described in Section 1.3.2.1. A thermal evaluation has been performed for Maine Yankee site specific fuels that exceed the design basis burnup, as shown in Section 4.5.1.2. As shown in that section, loading of fuel with a burnup between 45,000 and 50,000 MWD/MTU is subject to preferential loading in designated basket positions in the Transportable Storage Canister, and certain high burnup fuel may require loading in the Maine Yankee Fuel Can. The fuel can is provided in two configurations that differ only in the square cross-section of the can body. In both configurations, the walls of the body of the fuel can are 0.048-inch thick Type 304 stainless steel (18 gauge), have a length of 162.8 inches and have a bottom plate that is 0.63 inch thick. One configuration has a minimum square internal width of 8.52 inches; the second has a minimum square internal width of 8.32 inches.

With preferential loading, the design basis total heat load of the canister is not changed. Consequently, the thermal performance for the Maine Yankee site specific fuels is bounded by the design basis PWR fuels. Therefore, no further evaluation is required for the thermal accident events, as presented in Sections 11.2.6, 11.2.7, and 11.2.13.

As shown in Section 3.6.1.1, the total weight of the contents of the Transportable Storage Canister for Maine Yankee fuels is bounded by the total weight for the PWR design basis fuels. However, some design parameters for the Maine Yankee site ISFSI pad are different from those for the design basis ISFSI pad. Therefore, the hypothetical accident (non-mechanistic) tip-over event is evaluated to ensure that the maximum tip-over g-load remains below the bounding g-load (40g) used in the evaluation of the PWR canister and basket in Section 11.2.12.4. The evaluation of the UMS® Vertical Concrete Cask tip-over event on the Maine Yankee site ISFSI pad is presented in Section 11.2.15.1.1. The methodology used is similar to that used in Section 11.2.12.3.1.

Although the total weight, and the maximum g-load, for the Maine Yankee fuel is bounded by the PWR design basis fuels, the maximum weight of the consolidated fuel lattices (2,100 lbs) is larger than that of a single PWR Class 1 design basis fuel assembly (1,567 lbs). This additional weight need only be considered in the support disk evaluation for a side impact condition, similar to the analysis presented in Section 11.2.12.4.1. A parametric study is presented in Section 11.2.15.1.2 to demonstrate that the maximum stress in the support disk due to the consolidated fuel lattice remains bounded by the maximum stress for the support disk for the PWR design basis fuels for a side impact condition.

Section 11.2.15.1.3 provides the structural evaluation for the Maine Yankee fuel can for the 24-inch drop (Section 11.2.4) and the tip-over (Section 11.2.12) accident events.

A Maine Yankee site earthquake evaluation is presented in Section 11.2.15.1.4 to demonstrate the stability of the Vertical Concrete Cask on the Maine Yankee site ISFSI pad.

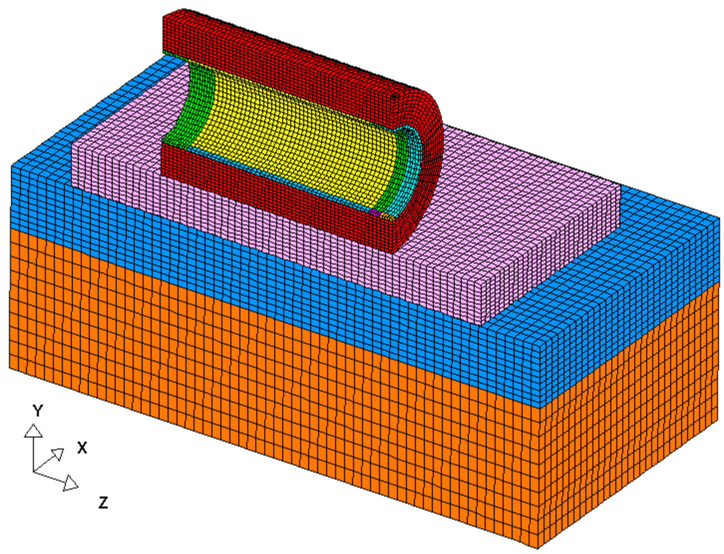
11.2.15.1.1 Maine Yankee Vertical Concrete Cask Tip-Over Analysis

This section evaluates the maximum acceleration of the Transportable Storage Canister and basket during the Vertical Concrete Cask tip-over event on the Maine Yankee site ISFSI pad. This evaluation applies the methodology of Section 11.2.12 for the design basis cask tip-over evaluation.

A finite element model is generated using the LS-DYNA program to determine the acceleration of the vertical concrete cask during the tip-over event.

The concrete pad in the model corresponds to a pad 31-feet by 31-feet square and 3-feet thick, supporting one concrete cask in the center of the pad. The soil under the concrete pad is considered to be 40-feet by 40-feet square and made up of two layers: a 4.5-foot thick upper layer and a 10-foot thick lower layer. Only one-half of the concrete cask, pad and soil configuration is modeled due to symmetry. Both the Class 1 and Class 2 UMS® configurations are evaluated.

The model includes a half section of the concrete cask, the concrete ISFSI pad and soil subgrade, as shown:



Concrete Pad Properties

Vertical concrete cask tip-over analyses are performed for ISFSI pad concrete compressive strengths of 3,000 and 4,000 psi. The Poisson's Ratio (ν_c) is 0.22. The concrete dry density is considered to be between 135 pcf and 145 pcf. To account for the weight of reinforcing bar in the pad, three values of Density (ρ) are used in the model:

ρ (lbs/ft^3)	E_c (psi)	K_c (psi)
140	2.994×10^6	1.782×10^6
145	3.156×10^6	1.879×10^6
152	3.387×10^6	2.016×10^6

The corresponding values of Modulus of Elasticity (E_c) and Bulk Modulus (K_c) are also provided, where:

Modulus of Elasticity (E_c) $= 33\rho_c^{1.5}\sqrt{f_c'}$ (ACI 318-95)

Bulk Modulus (K_c) $= \dfrac{E_c}{3(1-2\nu_c)}$ (Blevins [19])

Soil Properties

The soil properties used in the model are based on three soil sets. The vertical concrete cask tip-over analyses are performed for three different combinations of soil densities: (1) 4.5-foot thick upper layer density of 135 pcf (Modulus of Elasticity, E = 162,070 psi), with a 10-foot thick lower layer density of 127 pcf (E = 31,900 psi); (2) 4.5-foot thick upper layer density of 130 pcf, with a 10-foot thick lower layer density of 127 pcf; and (3) 15-foot depth with density of 145 pcf (E ≤ 60,000 psi). The Poisson's Ratio (ν_s) of the soil is 0.45.

Summary of Design Basis ISFSI Pad Parameters

The ISFSI pads and foundation shall include the following characteristics as applicable to the end drop and tip-over analyses:

Concrete thickness	36 inches maximum
Pad subsoil thickness	15 feet minimum
Specified concrete compressive strength	≤ 4,000 psi at 28 days
Soil in place density (ρ)	$\rho \le 145$ lbs/ft^3 (upper layer)
Concrete dry density (ρ)	$135 \le \rho \le 145$ lbs/ft^3
Soil Modulus of Elasticity	≤ 60,000 psi

The concrete pad maximum thickness excludes the ISFSI pad footer. The compressive strength of the concrete is determined in accordance with Section 5.6 of ACI-318 with concrete acceptance in accordance with the same section. Steel reinforcement is used in the pad and footer. The soil modulus of elasticity is determined according to the test method described in ASTM D4719.

Vertical Concrete Cask Properties

The material properties used in the model for the Vertical Concrete Cask are the same as the properties used in the PWR models in Section 11.2.12.3. The tip-over impact is simulated by applying an initial angular velocity of 1.485 rad/sec (PWR Class 1) and 1.483 rad/sec (PWR Class 2), respectively, to the entire cask. The angular velocity values are determined by the method used in Section 11.2.12 based on the weight of the loaded concrete cask with Maine Yankee fuel (285,513 pounds and 297,509 pounds for PWR Class 1 and PWR Class 2, respectively).

A cut-off frequency of 210 Hz (PWR Class 1) and 190 Hz (PWR Class 2) is applied to filter the analysis results from the LS-DYNA models and determine the peak accelerations. The resulting calculated accelerations on the canister at the location of the top support disk and of the top of the structural lid are tabulated for all of the analysis cases that were run. The maximum accelerations at the two key locations on the canister for the PWR Class 1 and Class 2 configurations are:

Component Location	Position Measured from the Bottom of the Concrete Cask (inches)		Acceleration (g)	
	Class 1	Class 2	Class 1	Class 2
Top Support Disk	176.7	185.2	32.3	34.2
Top of the Canister Structural Lid	197.9	207.0	35.3	37.6

The impact accelerations for the vertical concrete cask tip-over on the Maine Yankee ISFSI pad site are observed to be slightly higher than those reported in Section 11.2.12.3.1 for the design-basis ISFSI pad. Therefore, peak accelerations are calculated for the top support disk and are evaluated with respect to the analysis presented in Section 11.2.12.4.1.

To determine the effect of the rapid application of the inertia loading for the support disk, a dynamic load factor (DLF) is computed using the method presented in Section 11.2.12.4. The DLF is computed to be 1.07 and 1.02 for PWR Class 1 and Class 2, respectively. Applying the DLFs to the 32.3g and 35.4g results in peak accelerations of 34.6g and 36.1g for the top support disk PWR Class 1 and Class 2, respectively. The DLFs for the canister lids are considered to be unity since the lids have significant in-plane stiffness and are considered to be rigid. Additional sensitivity evaluations considering varying values of the ISFSI concrete pad density have been performed. The results of those evaluations demonstrate that the maximum acceleration for the canister and basket are below 40g. Therefore, the maximum acceleration for the canister and basket for the cask tipover accident on the Maine Yankee site ISFSI pad is bounded by the 40g used in Section 11.2.12.4.1 (analysis of canister and basket for PWR configurations for tip-over event).

11.2.15.1.2 Parametric Study of Support Disk Evaluation for Maine Yankee Consolidated Fuel

A parametric study is performed to show that the PWR basket loaded with a Maine Yankee consolidated fuel lattice is bounded by the PWR basket design basis loading for a side impact condition. Only one consolidated fuel lattice, in a Maine Yankee Fuel Can, will be loaded in any single Transportable Storage Canister. However, Maine Yankee Fuel Cans holding other undamaged or damaged fuel can be loaded in the other three corner positions of the basket. Maine Yankee Fuel Cans may be loaded only in the four corner positions of the basket. See Figure

11.2.15.1.2-2 for corner positions. Therefore, the bounding case for Maine Yankee is the basket configuration with twenty (20) Maine Yankee fuel assemblies, three (3) fuel cans containing spent fuel, and one (1) fuel can containing consolidated fuel.

A two-dimensional ANSYS model is employed for the parametric study as shown in Figure 11.2.15.1.2-1. The load from a PWR fuel assembly is modeled as a pressure load at the inner surface of each support disk slot opening. The design basis fuel pressure loading (1g) is 12.26 psi. Based on the same design parameters (slot size = 9.272 in., disk thickness = 0.5 inch, and the number of disks = 30), the pressure load corresponding to a Maine Yankee standard CE 14×14 fuel assembly is 10.3 psi. The pressure load is 11.3 psi for a Maine Yankee fuel can holding an undamaged or damaged fuel assembly. For a Maine Yankee fuel can holding consolidated fuel the pressure load is 17.0 psi.

This study considers a 60g side impact condition for four different basket orientations: 0°, 18.22°, 26.28° and 45°, as shown in Figure 11.2.15.1.2-2. The 60g bounds the g-load for the PWR support disks (40g) due to the Vertical Concrete Cask tip-over accident as shown in Section 11.2.12.

A total of five cases are considered in the study. Inertial loads are applied to the support disk in all cases. The base case considers that all 24 fuel positions hold design basis PWR fuel assemblies. The other four cases (Cases 1 through 4) represent four possible load combinations for the placement of four Maine Yankee fuel cans in the corner positions, one of which holds consolidated fuel. The remaining twenty basket positions hold Maine Yankee standard 14×14 fuel assemblies. The basket loading positions are shown in Figure 11.2.15.1.2-2. The load combinations evaluated in the four Maine Yankee fuel can loading cases are:

Case	Basket Position 1	Basket Position 2	Basket Position 3	Basket Position 4
1	Consolidated	Damaged	Damaged	Damaged
2	Damaged	Consolidated	Damaged	Damaged
3	Damaged	Damaged	Damaged	Consolidated
4	Damaged	Damaged	Consolidated	Damaged

Table 11.2.15.1.2-1 provides a parametric comparison between the Base Case and the four cases evaluated, based on the maximum sectional stress in the support disk. As shown in the table, the maximum stress in the PWR basket support disk loaded with 20 standard fuel assemblies and four Maine Yankee fuel cans, including one holding consolidated fuel, is bounded by that for the support disk loaded with the design basis PWR fuel.

Additionally, a three-dimensional analysis was performed for Case 4 with a 26.28° drop orientation using the three-dimensional canister/basket model presented in Section 11.2.12.4.1. Results of the analysis for the top support disk, where maximum stress occurs, are presented in Tables 11.2.15.1.2-2 and 11.2.15.1.2-3. The minimum margin of safety is +1.12 and +0.11 for P_m stresses and $P_m + P_b$ stresses, respectively. The minimum margin of safety for the corresponding analysis for the design basis PWR configuration is +0.97 and +0.05 for P_m and $P_m + P_b$ stresses, respectively (see Table 11.2.12.4.1-4). Therefore, it is further demonstrated that the maximum stress in the PWR support disk loaded with Maine Yankee fuel with consolidated fuel is bounded by the stress for the PWR support disk loaded with the design basis PWR fuel.

Since no credit is taken for the structural integrity of the consolidated fuel or damaged fuel inside the fuel can, it is assumed that 100% of the fuel rods fail during an accident. For a Maine Yankee standard 14×14 fuel assembly, the volume of 176 fuel rods (100%) and 5 guide tubes will fill up the lower 103.6 inches (about at the elevation of the 21st support disk) assuming a 50% volume compaction factor. For the consolidated fuel, the volume of 283 rods (100%) and 4 connector rods will fill up the lower 109.6 inches (about at the elevation of the 22nd support disk) assuming a 75% compaction factor. The compaction factor of 75% for the consolidated fuel considers that the number of rods in the consolidated fuel is approximately 1.5 times of the number of rods in the standard Maine Yankee fuel and these rods are initially more closely spaced.

During a tip-over accident of the vertical concrete cask, the maximum total load on the support disk (top/30th disk) for the design basis PWR basket is 54.6 kips (12.26 psi × 9.272-inch × 0.5-inch × 24 × 40g), considering the design deceleration of 40g (Section 11.2.12.4). With the assumption of 100% rod failure for the damaged fuel and consolidated fuel in the Maine Yankee fuel can, the 21st disk is subjected to the maximum total load (including weight from 20 standard fuel assemblies, 3 damaged fuel assemblies and the consolidated fuel). The pressure load (1g) on the support disk corner slot corresponding to 100% failed damaged fuel is 15.3 psi (load distributed to 21 support disks) and the pressure load corresponding to the 100% failed consolidated fuel is 22.6 psi (load distributed on 22 support disks). In the tip-over accident, the g-load at the 21st disk is 30g, based on the design deceleration of 40g at the top (30th) disk. The total load (W_{21}) on the 21st support disk is:

$$W_{21} = (10.3\times20+15.3\times3+22.6\times1) \times 9.272 \times 0.5 \times 30 = 38{,}200 \text{ pounds} = 38.2 \text{ kips}$$

The support disk load is only 70% (38.2/54.6 = 0.7) of the maximum total load on the support disk due to the design basis PWR fuel load. Consequently, the maximum stress in the support disk, assuming 100% rod failure of the damaged and consolidated fuel in Maine Yankee fuel cans, is bounded by the maximum stress in the support disk calculated for the design basis fuel.

Figure 11.2.15.1.2-1 Two-Dimensional Support Disk Model

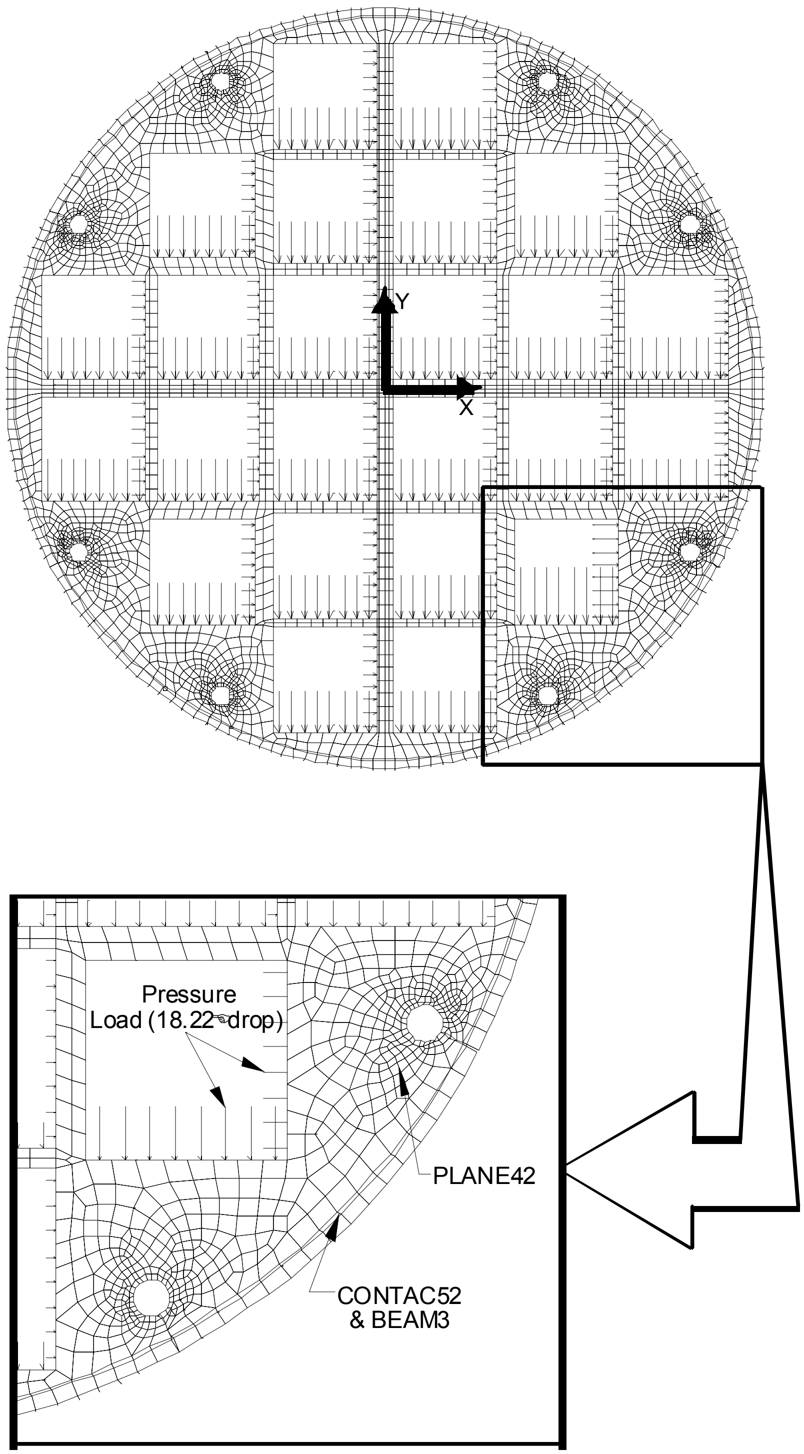


Figure 11.2.15.1.2-2 PWR Basket Impact Orientations and Case Study Loading Positions for Maine Yankee Consolidated Fuel

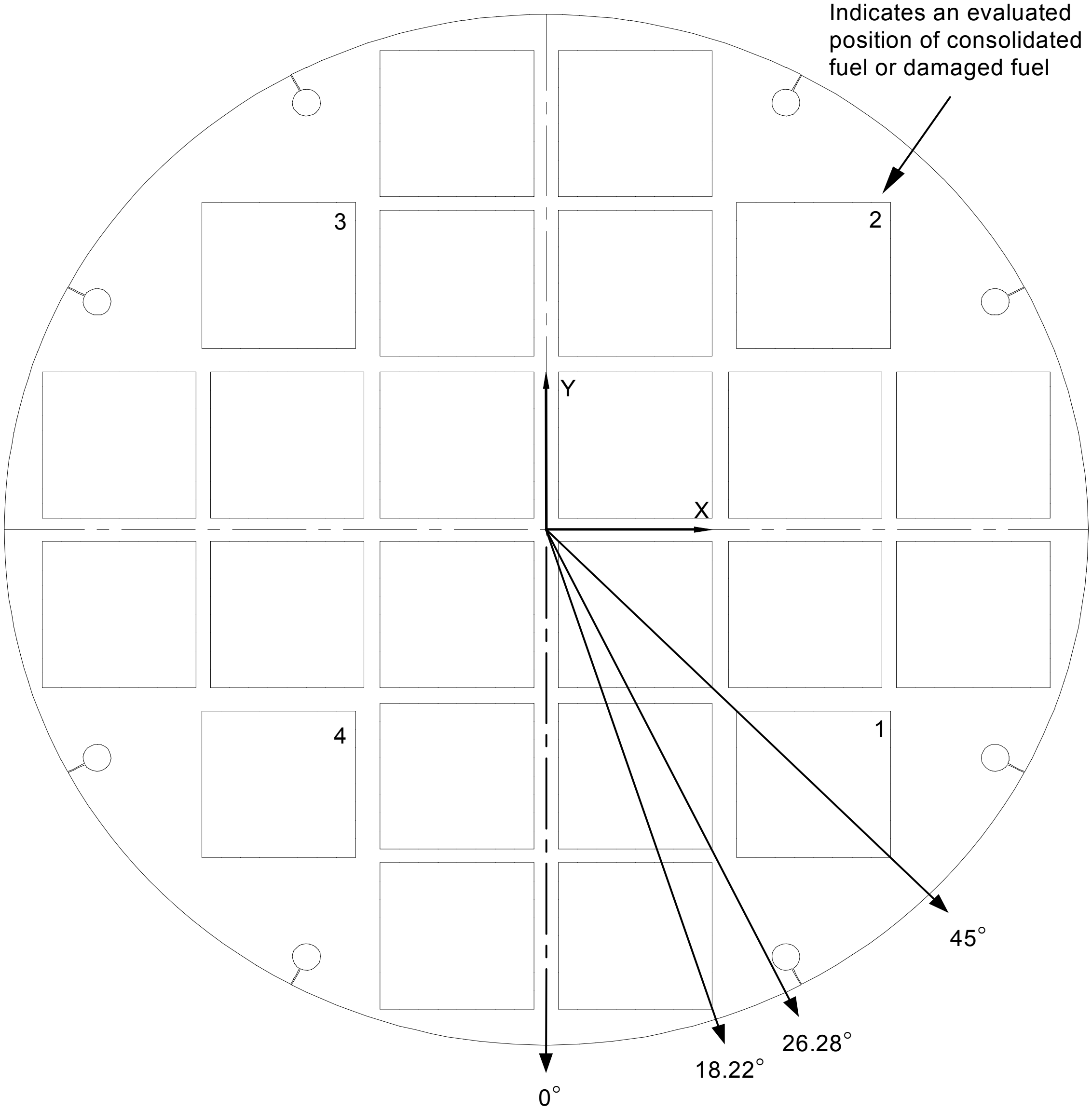


Table 11.2.15.1.2-1 Normalized Stress Ratios – PWR Basket Support Disk Maximum Stresses

Orientation[1]	Membrane Stress Ratio[2]				Membrane + Bending Stress Ratio[2]			
	0°	**18.22°**	**26.28°**	**45°**	**0°**	**18.22°**	**26.28°**	**45°**
Base Case	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
Case 1	0.91	0.94	0.94	0.94	0.96	0.94	0.94	0.94
Case 2	0.91	0.94	0.94	0.95	0.95	0.95	0.95	0.95
Case 3	0.91	0.95	0.95	0.95	0.96	0.95	0.95	0.95
Case 4	0.91	0.95	0.95	0.96	0.96	0.98	0.98	0.97

1. Orientations correspond to those shown in Figure 11.2.15.1.2-2.
2. Stress ratios are based on the maximum sectional stresses of the support disk.

Table 11.2.15.1.2-2 Support Disk Primary Membrane (P_m) Stresses for Case 4, 26.28° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
18	19.3	-22.9	2.8	42.6	90.4	1.12
3	27.1	-12.2	2.4	39.6	89.3	1.26
16	37.1	-22.8	1	37.2	89.3	1.4
1	32.3	-12.1	0.6	32.3	90.4	1.8
94	26.8	-19	2.7	27.6	90.5	2.28
17	-0.1	-22.8	1.9	23.1	89.8	2.9
88	18.3	-5.6	-7.3	21.6	91.5	3.23
96	6.7	-13.8	-3.2	21.4	91.5	3.27
95	-0.1	-19.9	1.5	20	91.1	3.55
90	15.3	-3.5	0.8	18.9	90.5	3.8
84	15.6	-18.5	-0.4	18.6	91.5	3.93
61	15.7	-10.5	4.7	18.5	91.5	3.96
60	10.2	-17.5	1.3	17.7	89.3	4.03
82	15.7	-7.8	3.8	17.2	90.8	4.27
37	11.9	-4.3	0.6	16.3	89.3	4.49
58	10.3	-12.1	5	16.3	90.4	4.54
62	15.7	-0.2	2.6	16.3	91.2	4.59
83	15.7	-0.2	1.7	15.8	91.2	4.75
91	-7.4	-15.4	-1.5	15.7	90.5	4.78
63	15.6	-9.9	0.5	15.7	90.8	4.8
30	14.1	-9.3	3.1	15.6	91.9	4.89
33	14.6	-4.7	2.3	15.1	89.3	4.93
108	13.5	-5.6	-3.9	15.1	91.5	5.07
24	-2	-14.3	1.7	14.5	91.5	5.31
79	-5.3	6.3	4.1	14.2	89.3	5.31
23	-0.1	-14.2	0.7	14.2	91.2	5.41
22	-7.3	-14.1	-0.4	14.2	90.8	5.42
28	13.2	-9.1	1.8	13.9	90.9	5.56
7	13.6	-11.9	-0.7	13.8	91.5	5.62
46	-2.4	-10.8	5.1	13.2	89.3	5.74

Note: See Figure 11.2.12.4.1-7 for Section locations.

Table 11.2.15.1.2-3 Support Disk Primary Membrane + Primary Bending (P_m + P_b) Stresses for Case 4, 26.28° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
61	-116.4	-39.3	10.1	117.7	130.8	0.11
58	-109.5	-43.9	8.7	110.6	129.1	0.17
43	-92.6	-32.4	6.2	93.2	129.1	0.39
82	-87.8	-27.9	7	88.6	129.8	0.46
60	-81.6	-39.9	7.7	83	127.6	0.54
79	-82	-18.9	2	82	127.6	0.56
55	-83.5	-29.3	4.6	83.9	130.8	0.56
16	-52.5	-71.9	15	80.1	127.6	0.59
46	-77.1	-49.3	9.5	80	127.6	0.59
64	-76.2	-31.8	7	77.2	127.6	0.65
30	-34.4	-75.2	13.1	79.1	131.3	0.66
18	-2.8	-77.6	-2.9	77.8	129.1	0.66
3	10.1	-65.4	-6	76.5	127.6	0.67
63	-75.4	-26	4.3	75.8	129.8	0.71
76	69	21	4.7	69.5	129.8	0.87
48	-66	-42.7	4	66.7	125.7	0.89
19	-38.2	-65.3	2.6	65.5	125.7	0.92
6	-43.2	-62	5.4	63.4	125.7	0.98
45	-63.2	-15.3	-0.2	63.2	127.6	1.02
94	-56.3	-40.8	10.4	61.5	129.3	1.1
21	-47.1	-57.5	5.3	59.7	127.6	1.14
67	-54.5	-42.3	5.3	56.5	125.7	1.22
1	-47.7	-40.7	12.7	57.3	129.1	1.25
33	-29.7	-52.9	7.4	55	127.6	1.32
51	26.7	-27.3	3.9	54.5	127.7	1.34
39	-29	-49.8	6.3	51.6	129.1	1.5
81	-49.9	-29.5	5.3	51.2	129.1	1.52
84	-48	-26.1	6.2	49.7	130.8	1.63
4	-41.7	-43.6	5.3	48	127.6	1.66
28	-44.6	-29.6	8.3	48.2	129.9	1.69

Note: See Figure 11.2.12.4.1-7 for Section locations.

11.2.15.1.3 Structural Evaluation for the Maine Yankee Fuel Can

Twenty-Four Inch Drop of the Vertical Concrete Cask

The 24-inch drop of the Vertical Concrete Cask onto an unyielding surface (Section 11.2.4) results in accelerations that are bounded by the 60g acceleration used in this structural evaluation for the Maine Yankee Fuel Can. The compressive load (P) on the tube is the combined weight of the lid, side plates and tube body. The Maine Yankee Fuel Can having the smallest internal cross-section (8.32 inches) is used in this analysis. This bounds the condition for the larger fuel can.

The compressive load (P) is:

$$P = (17.89 + 6.57 + 78.77) \times 60 = 6{,}193.8 \text{ lbs, use } 8{,}500 \text{ lbs.}$$

The compressive stress (S_c) in the tube body is:

$$S_c = \frac{P}{A} = \frac{8{,}500}{1.674} = 5{,}078 \text{ psi}$$

The margin of safety (MS) is determined based on the accident condition allowable primary membrane stress (0.7 S_u) at a bounding temperature of 600°F for Type 304 stainless steel:

$$MS = \frac{0.7 S_u}{S_c} - 1 = \frac{0.7(63{,}300)}{5{,}078} - 1 = +7.7$$

The potential buckling of the tube is evaluated, using the Euler formula, to determine the critical buckling load (P_{cr}):

$$P_{cr} = \frac{\pi^2 EI}{L_e^2} = \frac{\pi^2 (25.2 \times 10^6)(19.55)}{2(157.8)} = 48{,}817 \text{ lbs}$$

where:

$E = 25.2 \times 10^6$ psi

$$I = \frac{8.62^4 - 8.32^4}{12} = 19.55 \text{ in.}^4$$

L_e = 2L (worst case condition)

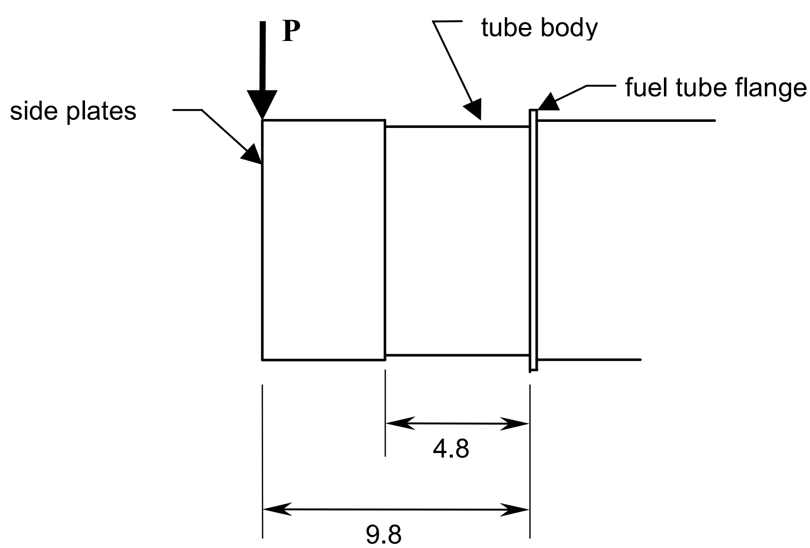
L = tube body length (157.8 in.)

Because the maximum compressive load (8,500 lbs under the accident condition) is much less than the critical buckling load (16.5×10^6 psi) the tube has adequate resistance to buckling.

Tip–Over of the Vertical Concrete Cask

The majority of the fuel can tube body is contained within the fuel tube in the basket assembly. Because both the tube body of the fuel can and the fuel tube have square cross sections, they are effectively in full contact (for 153.0 in. longitudinally) during a side impact and no significant bending stress is introduced into the tube body. The last 4.8 inches of the tube body and the 5.0 inches length of the side plates are unsupported past the fuel tube flange in the side impact orientation.

The tube body is evaluated as a cantilevered beam with the combined weight (P) of the overhanging tube body and side plates and conservatively, concentrated at the top end of the side plates multiplied by a deceleration factor of 60g. Note that the maximum g-load for the PWR basket is 40g for the tip-over accident (Section 11.2.12).



The maximum bending moment (M) is:

$$M = Pg \times L = 35(60)(9.8) = 20{,}581 \text{ lbs}\cdot\text{in.}$$

where:

P = 35 lbs (weight of the overhung tube and side plates)

g = 60 (conservative g-load that bounds the tip over condition)

L = 9.8 in. (the total overhung length of the tube body and side plates)

The maximum bending stress, f_b, is:

$$f_b = \frac{Mc}{I} = \frac{20{,}581(4.21)}{19.55} = 4{,}432 \text{ psi}$$

where:

c = half of the outer dimension of the tube

I = the moment of inertia

The shear stress (τ) is:

$$\tau = \frac{Pg}{A} = \frac{35(60)}{1.674} = 1{,}254 \text{ psi}$$

where:

A = the cross-sectional area of the tube = 1.674 in^2

The principal stresses are calculated to be 4,762 psi and -330 psi, and the corresponding stress intensity is determined to be 5,092 psi.

The margin of safety (MS) is calculated based on the allowable primary membrane plus bending stress (1.0 S_u) at a bounding temperature of 600°F for Type 304 stainless steel:

$$MS = \frac{1.0S_u}{\sigma_{max}} - 1 = \frac{63{,}300\ \text{psi}}{5{,}092\ \text{psi}} - 1 = +11.4$$

As discussed in Section 11.2.15.1.2, the Maine Yankee fuel can may hold a 100% failed damaged fuel lattice or consolidated fuel lattice. An evaluation is performed to demonstrate that the fuel can maintains its integrity during a tip-over accident for this condition. The fuel can is evaluated using the methodology presented in Section 11.2.12.4.1 for the PWR Fuel Tube Analysis for a 60-g side impact condition. This g-load bounds the maximum g-load (40g) for the PWR basket in the concrete cask tip-over event. Similar to the finite element model used for the PWR fuel tube analysis for the uniform pressure case (see Section 11.2.12.4.1), an ANSYS finite element model is generated to represent a section of the damaged fuel can with a length of three spans, i.e., the model is supported at four locations by the support disks. The fuel tube, the neutron absorber plate, and its stainless steel cover plate are conservatively ignored in the model. A bounding uniform pressure is applied to the lower inside surface of the fuel can wall. The pressure is determined based on the weight of the 100% failed consolidated fuel (2,100 lbs × 60g) occupying a length of 109.6 inches (see Section 11.2.15.1.2) as shown below. The inside dimension of the larger fuel can (8.52-inches) is conservatively used in the analysis, as it bounds the bending stress condition of the fuel can with the smaller cross-section.

$$P = \frac{2{,}100}{109.6(8.52)} \times 60 = 135\ \text{psi}$$

The finite element analysis results show that the maximum stress in the fuel can is 25.4 ksi, which is local to the sections of the tube resting on the support disks. At 750°F the ultimate strength for Type 304 stainless steel is 63.1 ksi. The Margin of Safety is:

$$MS = \frac{63.1}{25.4} - 1 = +1.48$$

The analysis shows that the maximum total strain is 0.05 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F, the resulting Margin of Safety is:

$$MS = \frac{0.40 \big/ 2}{0.05} - 1 = +3.0$$

Similarly, the Margin of Safety for elastic-plastic stress is:

$$MS = \frac{63.1 - 17.3}{25.4 - 17.3} - 1 = +4.65$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

Therefore the Maine Yankee fuel can maintains its integrity for the accident conditions.

11.2.15.1.4 Maine Yankee Site Specific Earthquake Evaluation of the Vertical Concrete Cask

This section provides an evaluation of the response of the vertical concrete cask to an earthquake imparting a horizontal acceleration of 0.38g at the top surface of the concrete pad. The evaluation shows that the loaded or empty vertical concrete cask does not tip over or slide in the earthquake event. The methodology used in this evaluation is identical to that presented in Section 11.2.8.

Tip-Over Evaluation of the Vertical Concrete Cask

To maintain the concrete cask in equilibrium, the restoring moment, M_R must be greater than, or equal to, the overturning moment, M_o (i.e. $M_R \geq M_o$). Based on this premise, the following derivation shows that a 0.38g acceleration of the design basis earthquake at the surface of the concrete pad is well below the acceleration required to tip-over the cask.

The combination of horizontal and vertical acceleration components is based on the 100-40-40 approach of ASCE 4-86 [36], which considers that when the maximum response from one component occurs, the response from the other two components are 40% of the maximum. The vertical component of acceleration is obtained by scaling the corresponding ordinates of the horizontal components by two-thirds.

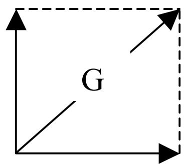
Using this method, two cases are evaluated where:

$a_x = a_z = a$ = horizontal acceleration components
$a_y = (2/3)\, a$ = vertical acceleration component
G_h = Vector sum of two horizontal acceleration components
G_v = Vertical acceleration component

In the first case, the horizontal acceleration is at its maximum. In the second, one horizontal acceleration is at its maximum.

Case 1) The vertical acceleration, $a_{y,}$ is at its peak: ($a_y = 2/3a$, $a_x = 0.4a$, $a_z = 0.4a$)

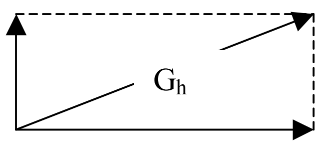
$$G_h = \sqrt{a_x^2 + a_z^2}$$
$$G_h = \sqrt{(0.4 \times a)^2 + (0.4 \times a)^2} = 0.566 \times a$$



$$G_v = 1.0 \times a_y = 1.0 \times \left(a \times \frac{2}{3} \right) = 0.667 \times a$$

Case 2) One horizonal acceleration, a_x, is at its peak: ($a_y = 0.4 \times 2/3a$, $a_x = a$, $a_z = 0.4a$)

$$G_h = \sqrt{a_x^2 + a_z^2}$$
$$G_h = \sqrt{(1.0 \times a)^2 + (0.4 \times a)^2} = 1.077 \times a$$



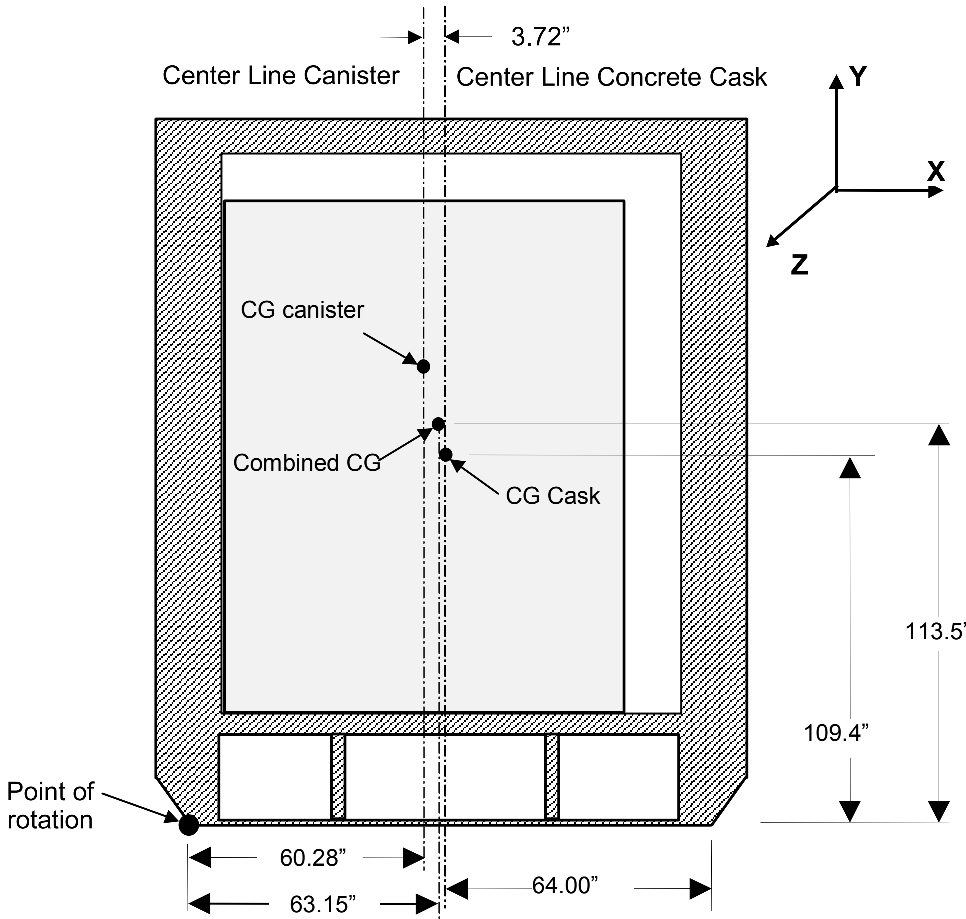
$$G_v = 0.4 \times a_y = 0.4 \times \left(a \times \frac{2}{3} \right) = 0.267 \times a$$

In order for the cask to resist overturning, the restoring moment, M_R, about the point of rotation, must be greater than the overturning moment, M_o, that:

$$M_R \geq M_o \text{, or}$$
$$F_r \times b \geq F_o \times d \Rightarrow (W \times 1 - W \times G_v) \times b \geq (W \times G_h) \times d$$

where:

- d = vertical distance measured from the base of the Vertical Concrete Cask to the center of gravity
- b = horizontal distance measured from the point of rotation to the C.G.
- W = the weight of the Vertical Concrete Cask
- F_o = overturning force
- F_r = restoring force



Substituting for G_h and G_v gives:

<u>Case 1</u>

$$(1 - 0.667a)\ \frac{b}{d} \geq 0.566 \times a$$

$$a \leq \frac{b/d}{0.566 + 0.667\left(b/d\right)}$$

<u>Case 2</u>

$$(1 - 0.267a)\frac{b}{d} \geq 1.077a$$

$$a \leq \frac{b/d}{1.077 + 0.267\left(b/d\right)}$$

Because the canister is not attached to the concrete cask, the combined center of gravity for the concrete cask, with the canister in its maximum off-center position, must be calculated. The point of rotation is established at the outside lower edge of the concrete cask.

The inside diameter of the concrete cask is 74.5 inches and the outside diameter of the canister is 67.06 inches; therefore, the maximum eccentricity between the two is:

$$e = \frac{74.50\text{ in} - 67.06\text{ in}}{2} = 3.72\text{ in.}$$

The horizontal displacement, x, of the combined C.G. due to eccentric placement of the canister is

$$x = \frac{70{,}783(3.72)}{308{,}432} = 0.85\text{ in}$$

Therefore,

$$b = 64 - 0.85 = 63.15\text{ in.}$$

and

$$d = 113.5\text{ in.}$$

The C.G. of the loaded Maine Yankee Vertical Concrete Cask is conservatively assumed to be 113.5 inches, which bounds all of the Maine Yankee UMS® Storage System configurations.

1) $$a \le \frac{63.15/113.5}{0.566 + 0.667 \times (63.15/113.5)}$$

$a \le 0.59g$

2) $$a \le \frac{63.15/113.5}{1.077 + 0.267 \times (63.15/113.5)}$$

$a \le 0.45g$

Therefore, the minimum ground acceleration that may cause a tip-over of a loaded concrete cask is 0.45g. Since the 0.38g design basis earthquake ground acceleration for the UMS® System at the Maine Yankee site is less than 0.45g, the storage cask will not tip-over.

The factor of safety is 0.45 / 0.38 = 1.18, which is greater than the required factor of safety of 1.1 in accordance with ANSI/ANS-57.9.

Since an empty vertical concrete cask has a lower C.G. as compared to a loaded concrete cask, the tip-over evaluation for the empty concrete cask is bounded by that for the loaded concrete cask.

Sliding Evaluation of the Vertical Concrete Cask

To keep the cask from sliding on the concrete pad, the force holding the cask (F_s) has to be greater than or equal to the force trying to move the cask.

Based on the equation for static friction:

$$F_s = \mu N \geq G_h W$$
$$\mu \left(1 - G_v\right) W \geq G_h W$$

Where:

μ = coefficient of friction
N = the normal force
W = the weight of the concrete cask
G_v = vertical acceleration component
G_h = resultant of horizontal acceleration component

Substituting G_h and G_v for the two cases:

Case 1) $\mu(1 - 0.667a) \geq 0.566\,a$

$$\mu \geq \frac{0.566a}{1 - 0.667a}$$

Case 2) $\mu(1 - 0.267a) \geq 1.077\,a$

$$\mu \geq \frac{1.077a}{1 - 0.267a}$$

For a = 0.38g

Case 1) $\mu \geq 0.29$

Case 2) $\mu \geq 0.45$

The analysis shows that the minimum coefficient of friction, μ, required to prevent sliding of the concrete cask is 0.45. The coefficient of friction between the steel bottom plate of the concrete cask and the concrete surface (broom finish) of the storage pad, 0.50, is greater than the coefficient of friction required to prevent sliding of the concrete cask [45,46]. Therefore, the concrete cask will not slide under design-basis earthquake conditions. The factor of safety is 0.50 / 0.45 =1.11 which is greater than the required factor of safety of 1.1 in accordance with ANSI/ANS-57.9 [1].

11.2.15.1.5 Buckling Evaluation for Maine Yankee High Burnup Fuel Rods

This section presents the buckling evaluation for Maine Yankee high burnup fuel (burnup between 45,000 and 50,000 MWD/MTU) having cladding oxide layers that are 80 and 120 microns thick. A similar evaluation is presented in Section 11.2.15.1.6 for Maine Yankee high burnup fuel with an oxide layer thickness of 80 microns that is also mechanically damaged. These analyses show that the high burnup fuel and the damaged high burnup fuel do not buckle in the design basis accident events. An end drop orientation is considered with an acceleration of 60 g, which subjects the fuel rod to axial loading. A reduced clad thickness is assumed, due to the cladding oxide layer.

In the end drop orientation, the fuel rods are laterally restrained by the grids and may come into contact with the fuel assembly base. The only vertical constraint for the fuel rod is the base of the assembly. The weight of the fuel pellets is included in this evaluation, as the pellets are considered to be vertically supported by the cladding. A two-dimensional model comprised of ANSYS BEAM3 elements, shown in Figure 11.2.15.1.5-1, is used for the evaluation. This evaluation is considered to be the bounding condition (as opposed to an evaluation, which considers the cladding only).

80 Micron Oxide Layer Thickness Evaluation

During the end drop, the fuel rod impacts the fuel assembly base. The fuel rod itself will respond as an elastic bar under a sudden compression load at its bottom end. The duration of this impact is bounded by the first extentional mode shape of the fuel rod. Contribution of higher frequency extentional modes of the rod would tend to shorten the duration of impact of the fuel rod with the fuel assembly base. The fuel rod, upon initiation of impact, corresponds to an undeformed state. In the process of the impact, the compression of the fuel rod will increase to a maximum and then return to a near uncompressed state, at which point the time of impact has been completed. This actually represents half of a cycle of the lowest frequency mode shape of the fuel rod. The shape of the time dependence of the deformation is sinusoidal. The single extentional mode shape can also be considered to be a single degree of freedom with a corresponding mass and stiffness. In viewing such an event as a spring mass system, the time variation of the deformation during the impact is expected to be sinusoidal.

The buckling mode for the fuel rod is governed by the boundary conditions. For this configuration, the grids provide a lateral support, but no vertical support. The only vertical restraint is considered to be at the point of contact of the fuel rod and the base of the assembly. The weight of the fuel rod pellets and cladding is assumed to be uniformly distributed along the length of the fuel rod. In the end drop, this results in the maximum compressive load occurring at the base of the fuel rod. The first buckling mode shape corresponding to these conditions is computed as shown in Figure 11.2.15.1.5-2.

Typically eigenvalue buckling is applied for static environments. For dynamic loading, it is assumed that the duration of the loading is sufficiently long to allow the system to experience the complete load, even as the deformation associated with the buckling is commenced. For dynamic loading, the lateral motion, which would correspond to the buckled shape, will correspond to the lowest mode shape. This lowest frequency mode shape is shown in Figure 11.2.15.1.5-2 and corresponds to a frequency of 25.9 Hz. The similarity of the two shapes shown in Figure 11.2.15.1.5-2 is expected, since both have the same displacement boundary conditions, the same stiffness matrix, and the same governing finite element equations, i.e.,

$$[K]\{\phi_i\} = \lambda_i \, [A]\{\phi_i\}$$

where:

$[K]$ = structure stiffness matrix
$\{\phi_i\}$ = eigenvector
λ_i = eigenvalue
$[A]$ = mass matrix for the mode shape calculation or stress stiffening matrix for the buckling evaluation

Based on the time duration of the impact and the inherent inability of the fuel rod to rapidly displace in the lateral direction, the effect of the actual lateral motion of buckling can be computed with a dynamic load factor (DLF) [47]. The expression for the DLF for a half-sine loading for a single degree of freedom is given by

$$\mathrm{DLF} = \frac{2\beta \cos(\pi/2\beta)}{1 - \beta^2}$$

where:

β = ratio of the first extentional mode frequency to the first lateral mode frequency

These values, computed in this section, are $\beta = 8.32$ and DLF = 0.244.

This DLF is applied to the end drop acceleration of 60g, which is the bounding load to potentially result in the buckling of the fuel rod. The product of 60g × DLF (= 14.6g) is well below the vertical acceleration corresponding to the first buckling mode shape, 37.9g as computed in this section. This indicates that the time duration of the impact of the fuel onto the fuel assembly base is of sufficiently short nature that buckling of the fuel rod cannot occur.

An effective cross-sectional property is used in the model to consider the properties of the fuel pellet and the fuel cladding. The modulus of elasticity (EX) for the fuel pellet has a nominal value of 26.0×10^6 psi [48]. To be conservative, only 50 percent of this value is used in the evaluation. The EX for the fuel pellet was, therefore, taken to be 13.0×10^6 psi. The value of EX (10.47×10^6 psi) was used for the irradiated zirconium alloy cladding (ISG-12). Reference information shows that there is no additional reduction of the ductility of the cladding due to extended burnup into the 45,000 – 50,000 MWD/MTU range [49].

The bounding dimensions and physical data (minimum clad thickness, maximum rod length and minimum number of support grids) for the Maine Yankee fuel rod used in the model are:

Outer diameter of cladding (inches)	0.434
Cladding thickness (inches)	0.023
Cladding density (lb/in^3)	0.237
Fuel pellet density (lb/in^3)	0.396

The cladding is reduced from its nominal value of 0.026 inches by the assumed 80 micron oxidation layer (0.003 inches) to 0.023 inches. Similarly, the fuel rod outer diameter is reduced from the nominal value of 0.44 inches to 0.434 inches.

The elevation of the grids, measured from the bottom of the fuel assembly are: 2.3, 33.0, 51.85, 70.7, 89.6, 108.4, 127.3 and 144.9 (inches).

The effective cross-sectional properties (EI_{eff}) for the beam are computed by adding the value of EI for the cladding and the pellet, where:

E = modulus of elasticity (lb/in^2)
I = cross-sectional moment of inertia (in^4)

The lowest frequency for the extentional mode shape was computed to be 219.0 Hz. The first mode shape corresponds to a frequency of 25.9 Hz. Using the expression for the DLF previously discussed, the DLF is computed to be 0.240 (β = 8.44).

120 Micron Oxide Layer Thickness Evaluation

The buckling calculation used the same model employed for the mode shape calculation. The load that would potentially buckle the fuel rod in the end drop is due to the deceleration of the rod. This loading was implemented by applying a 1g acceleration in the direction that would result in compressive loading of the fuel rod. The acceleration required to buckle the fuel rod is computed to be 37.3g.

Using the same fuel rod model, the acceleration required to buckle the fuel rods is found to be 37.3g, which is much higher than the calculated effective g-load (14.3g) due to the 60g end drop. Therefore, the fuel rods with a 120 micron cladding oxide layer do not buckle in the 60g end drop event.

Figure 11.2.15.1.5-1 Two-Dimensional Beam Finite Element Model for Maine Yankee Fuel Rod

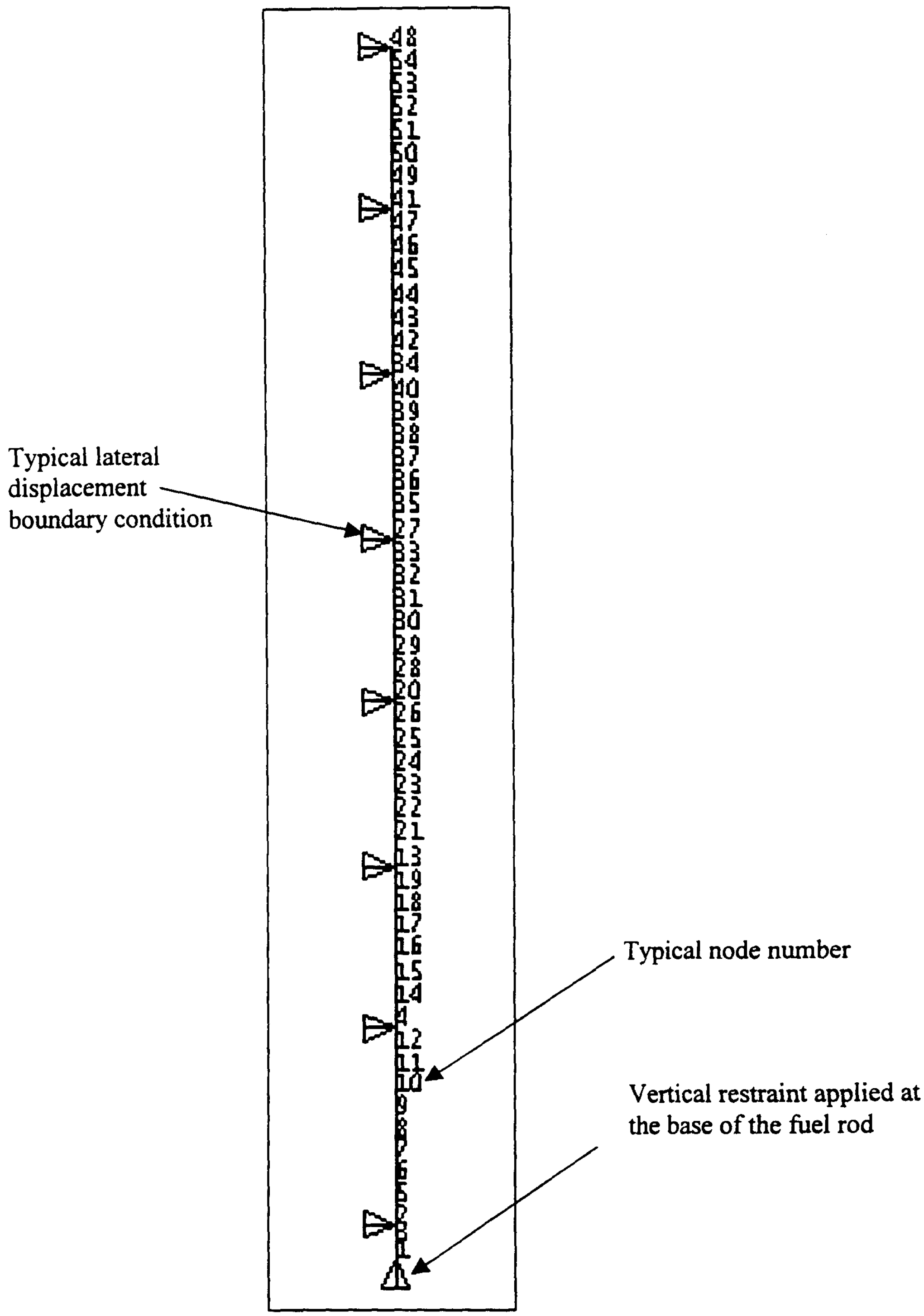
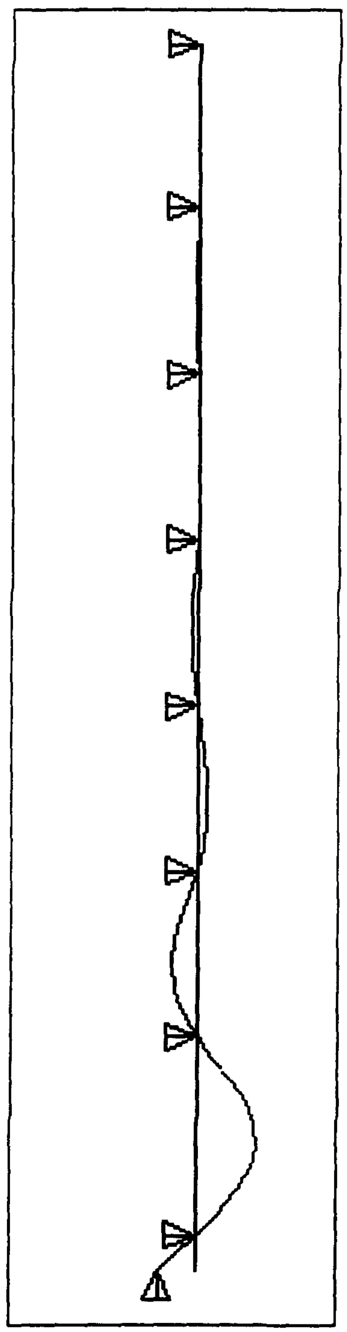
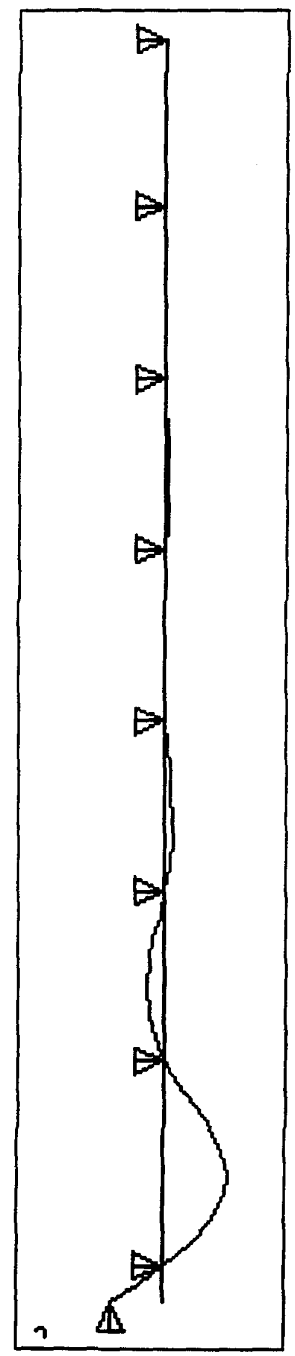


Figure 11.2.15.1.5-2 Mode Shape and First Buckling Shape for the Maine Yankee Fuel Rod

First Lateral Dynamic
Mode Shape at 25.9 Hz



First Buckling
Shape at 37.9g



11.2.15.1.6 Buckling Evaluation for High Burnup Fuel with Mechanical Damage

This section presents the buckling evaluation for high burnup fuel having an 80 micron cladding oxide layer thickness and with mechanical damage consisting of one or more missing support grids up to an unsupported fuel rod length of 60 inches.

End Drop Evaluation

The buckling load is maximized at the bottom of the fuel assembly. The bounding evaluation is the removal of the grid strap that maximizes the spacing at the lowest vertical elevation. The elevations of the grids in the model, measured from the bottom of the fuel assembly are: 2.3, 51.85, 70.7, 89.6, 108.4, 127.3 and 144.9 inches (Figure 11.2.15.1.6-1). The grid at the 33.0-inch elevation is removed, resulting in a grid spacing of approximately 50.0 inches. The grid located at 51.85 inches is conservatively assumed to be located at 62.3 inches, resulting in an unsupported rod length of 60.0 inches.

The case of the missing grid is evaluated using the methodology presented in Section 11.2.15.1.5 for the fuel assembly with the grids being present. The dimensions and physical data for the Maine Yankee fuel rod used in the model are:

Outer diameter of cladding (inches)	0.434
Cladding thickness (inches)	0.023
Cladding density (lb/in^3)	0.237
Fuel pellet density (lb/in^3)	0.396
Fuel pellet Modulus of Elasticity (psi)	13.0×10^6
Zirconium alloy cladding Modulus of Elasticity (psi)	10.47×10^6

The cladding is reduced from its nominal value of 0.026 inches by the assumed 80 micron oxidation layer thickness (0.003 inches) to 0.023 inches. Similarly, the fuel rod outer diameter is reduced from the nominal value of 0.44 inches to 0.434 inches. The fuel pellet modulus of elasticity is conservatively reduced 50%. The modulus of elasticity of the zirconium alloy is taken from ISG-12 [50].

With the grid missing, the frequency of the fundamental lateral mode shape is 7.8 Hz. The natural frequency of the fundamental extensional mode was determined to be 218.9 Hz. The DLF is computed to be 0.072, resulting in an effective acceleration of $0.072 \times 60 = 4.3$ g. Using the same method to compute the acceleration at which buckling occurs, the lowest buckling acceleration is 14.4 g, which is significantly greater than 4.3 g. Therefore, the fuel rod does not buckle during an end drop. Figures 11.2.15.1.6-1 and 11.2.15.1.6-2 show the finite element model and buckling results and mode shape.

Side Drop Evaluation

The Maine Yankee fuel rod is evaluated for a 60 g side drop with a missing support grid in the fuel assembly. Using the same assumptions as for the end drop evaluation, the span between support grids is assumed to be 60.0 inches.

For this analysis, the dimensions and physical data used are:

Fuel rod OD	0.434 in. (80 micron oxidation layer)
Clad ID	0.388 in.
E_{clad}	10.47E6 psi
E_{fuel}	13.0E6 psi
Clad density	0.237 lb/in^3
Fuel density	0.396 lb/in^3
A_{clad}	0.030 in^2 (cross-sectional area)
A_{fuel}	0.118 in^2 (cross-sectional area)

The mass of the fuel rod per unit length is:

$$m = \frac{0.396(0.122) + 0.237(0.030)}{386.4} = 0.000143\ \text{lb - s}^2/\text{in}^2$$

For the fuel rod, the product of the Modulus of Elasticity (E) and Moment of Inertia (I), is:

$$EI_{clad} = 10.47E6\frac{\pi(0.217^4 - 0.194^4)}{4} = 6{,}586\ \text{lb - in}^2$$

$$EI_{fuel} = 13.0E6\frac{\pi(0.194^4)}{4} = 14{,}462\ \text{lb - in}^2$$

$$EI = 6{,}586 + 14{,}462 = 21{,}048\ \text{lb - in}^2$$

During a side drop, the maximum deflection of a fuel rod is based on the fuel rod spacing of the fuel assembly. The pitch (center-to-center spacing) of fuel rods is 0.58 inches [51]. The maximum pitch is across the diagonal of the fuel assembly. The maximum pitch is:

$$dp = \frac{0.58}{\sin 45} = 0.82 \text{ in.}$$

The maximum deflection of a fuel rod is at the top of the fuel assembly and the minimum deflection is at the bottom of the fuel assembly.

Assuming a 17 × 17 array (which envelops the Maine Yankee 14 × 14 array), the maximum fuel rod deflection is:

$$(17\text{-}1) \times (0.82\text{-}0.43) = 6.18 \text{ in.}$$

The deflection of a simply supported beam with a distributed load is given by the equation:

$$\Delta = \frac{5\omega l^4}{384EI} = \frac{5(g\omega)l^4}{384\,(EI_{total})} \qquad [52]$$

$$g = \frac{384\Delta(EI_{total})}{5\omega l^4}$$

The cladding bending stress is given by the equation:

$$S = \frac{Mc}{I} = \frac{\left(\left(g\omega l^2\right)/8\right)c}{I_{clad}}\left(\frac{EI_{clad}}{EI_{total}}\right)$$

Inserting the equation for ‘g’:

$$S = \frac{384\Delta c E_{clad}}{40 \times L^2}$$

where:

c = 0.217 inch distance from center of fuel rod to extreme outer fiber

L = 60 inches (the unsupported fuel rod length)

Δ = 6.18 inches (the maximum deflection)

The bending stress in the fuel rod is:

$$S = \frac{384 \times 6.18 \times 0.217 \times 10.47E6}{40(60)^2} = 37.4 \text{ ksi}$$

The maximum hoop stress due to the fuel rod internal pressure is determined to be 19.1 ksi (131.4 MPa per Tables 4.4.7-3 and 4.5.1.2-1). Therefore, the maximum axial stress is 9.6 ksi (one half of the hoop stress [53]).

The bearing stress between two fuel rods under a 60 g load is:

$$S_{brg} = 0.591\sqrt{\frac{\omega E}{K_D}} = 0.591\sqrt{\frac{(0.000143 \times 386.4) \times 60 \times 10.47E6}{0.22}} = 7.4 \text{ ksi} \quad [53]$$

where:

$$K_D = \frac{D_1 D_2}{D_1 + D_2} = \frac{0.434 \times 0.434}{0.434 + 0.434} = 0.22$$

The total stress is:

$$S = 37.4 + 9.6 + 7.4 = 54.4 \text{ ksi}$$

The ultimate strength allowable for irradiated zirconium alloy is 83.4 ksi (Figure 3-2 [54]). Therefore, the margin of safety for ultimate strength is:

$$MS = \frac{83.4}{54.4} - 1 = 0.53$$

The yield strength allowable for irradiated zirconium alloy is 78.3 ksi (Figure 3-2 [54]). Therefore, the margin of safety for yield strength is:

$$MS = \frac{78.3}{54.4} - 1 = 0.44$$

The maximum bearing stress occurs between the bottom fuel rod and the fuel tube. The bearing stress is:

$$S_{brg} = 0.591\sqrt{\frac{17 \times 0.000143 \times 386.4 \times 60 \times 10.47E6}{0.44}} = 21.6\,\text{ksi}$$

The bending stress is negligible because the maximum deflection is equal to the spacing of the fuel rods established by the grid. Therefore, the top fuel rod is bounding.

Consequently, the fuel rods are demonstrated to be structurally adequate for the 60g side drop loading condition.

Figure 11.2.15.1.6-1 Two-Dimensional Beam Finite Element Model for a Fuel Rod with a Missing Grid

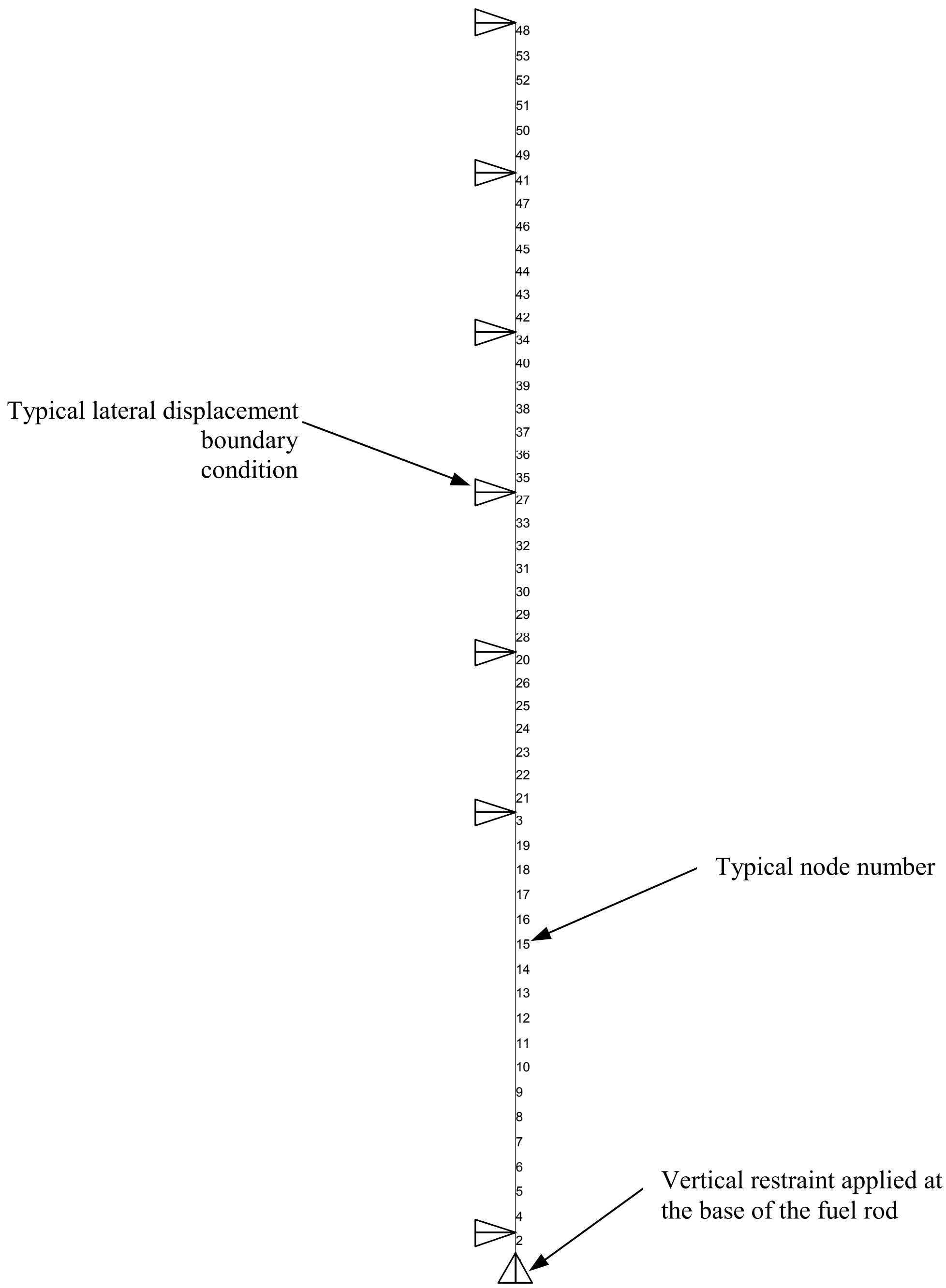
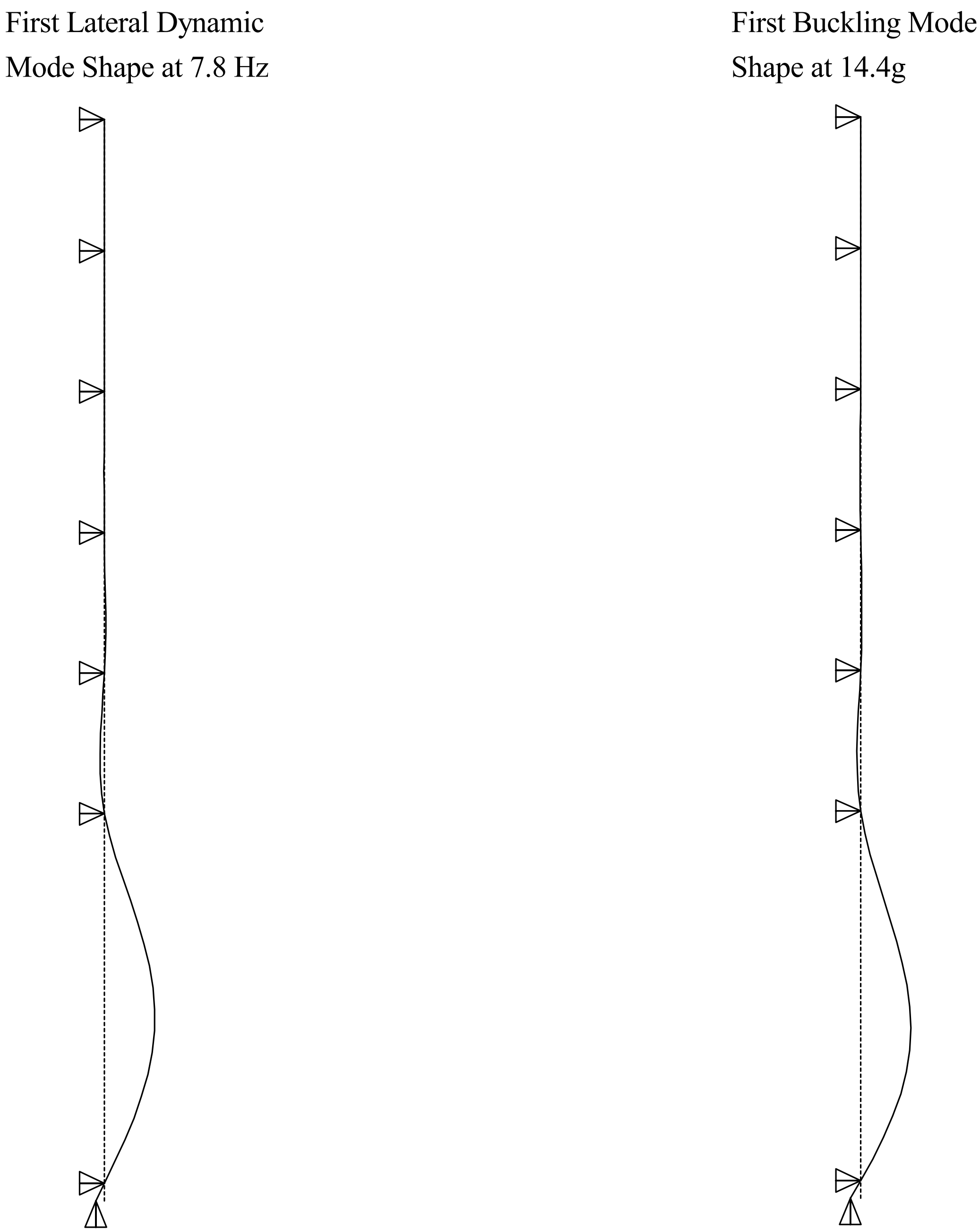


Figure 11.2.15.1.6-2 Modal Shape and First Buckling Mode Shape for a Fuel Rod with a Missing Grid



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11.2.16 Fuel Rods Structural Evaluation for Burnup to 60,000 MWd/MTU

This section presents a structural evaluation of PWR fuel rods with a maximum burnup of 60,000 MWd/MTU for normal and accident conditions of storage.

During normal and off-normal conditions for the fuel in the canister, the loads applied to the fuel assembly are minimal and do not require further evaluation. The only significant axial loading the fuel assembly will experience is the 24-inch drop of the vertical concrete cask. The bounding lateral loading on the fuel assembly occurs during the tip-over accident condition. The lateral loading and axial loading conditions are evaluated for PWR fuel rods considering a bounding configuration of a grid spacer missing for an unsupported fuel rod length up to 60 inches. Based on results of the drop analysis presented in the following sections, PWR fuel assemblies with one or more grid spacers missing or damaged for an unsupported fuel rod length up to 60 inches are, therefore, considered to be undamaged assemblies for loading within the UMS® Universal Storage System.

11.2.16.1 PWR Fuel Rod Evaluation

End Drop Evaluation

This section presents the buckling evaluation for the UMS® Universal Storage System high burnup PWR fuel rods (peak rod average burnup of 62.5 GWd/MTU). In order to account for the cladding oxide layer, a conservative 120-micron thick layer is assumed to be removed from the reference clad in the rod structural evaluation. The 120-micron clad removal is conservative, as this value represents double the maximum oxide layer thickness listed for end-of-life PWR fuel rods in PNL-4835[62]. Applying a time-dependent oxide layer growth approximation to the PNL-4835-reported maximum thickness of 60 microns to account for an increase in burnup from standard (45 GWd/MTU) to high (62.5 GWd/MTU) yields a maximum end-of-life oxide layer in the range of 90 microns. As high burnup claddings are designed to reduce oxidation, actual layers are expected to be significantly lower (reported as low as 20 microns for >70 GWd/MTU M5 zircaloy clad). Therefore, a significant margin exists to the evaluated oxide layer levels.

These analyses show that the maximum stresses in the high burnup PWR fuel remain below the yield strength in the design basis accident events and confirm that the fuel rods will return to their original configuration prior to the end drop event. An end drop orientation is considered with an acceleration of 48g, which subjects the fuel rods to axial loading. This 48g acceleration bounds the maximum end drop acceleration calculated for the 24-inch concrete cask end drop.

In the end drop orientation, the fuel rods are laterally restrained by the grids and come into contact with the fuel assembly base. The only vertical constraint for the fuel rod is the base of the assembly. As opposed to employing a straight fuel assembly in the evaluation with all the grids present, the fuel assembly is considered to be bowed, and a fuel assembly grid may be missing and still meet the acceptable configuration for undamaged fuel. The evaluation of the PWR fuel rods is based on the following representative samples.

Fuel Assembly	Cladding Diameter (in)	Cladding Thicknes s (in)	Fuel Rod Pitch (in)	Gap Between Fuel Assembly and Fuel Tube Wall (in)
We 17x17	0.360	0.021	0.496	0.504
We 15x15	0.417	0.024	0.563	0.501
We 14x14	0.400	0.022	0.556	1.172
CE16x16	0.382	0.025	0.506	0.828
CE14x14	0.440	0.031	0.580	0.820
BW17x17	0.377	0.022	0.502	0.391
BW15x15	0.414	0.022	0.568	0.434

Review of the design basis fuel inventory indicates that the largest gap between a straight fuel assembly and the basket fuel tube inner wall could be 1.17 inches, corresponding to a 14 ×14 rod array having a minimum rod pitch of 0.556 inch and a minimum rod diameter of 0.40 inch inside a basket fuel cell, with a maximum inside dimension of 8.8 inches. It is physically possible for a fuel rod to have a bow of 1.17 inches and still be able to fit into the larger basket cell. Actual fuel assembly bow is expected to be much less than this maximum value and on the order of 0.125 to 0.25 inch. A missing grid implies that the axial distance between two adjacent grids could be as large as 60 inches. The bounding axial loading is the 24-inch bottom end drop, which identifies the bounding condition to be the 60-inch distance from the end of the fuel rod to the first grid. To implement a conservative bow of 1.23 inches (an increase of 0.06 inch over 1.17 inches) into the fuel assembly, the half-symmetry ANSYS model shown in Figure 11.2.16-1 is used. This model contains 0.5-inch long individual fuel pellets modeled with brick elements, with a 0.002-inch gap between the fuel pellet and the clad. The clad is modeled with shell elements. Elastic properties are used for the fuel pellet and the clad as shown in the following table.

	Modulus of Elasticity (10^6 psi)	Density (lb/in^3)
Rod Clad	10.47 [61]	0.237
Fuel Pellet	13 [48]*	0.396

* To further reduce the strength of the pellet, this is 50% of the value reported in Reference 48.

CONTAC52 elements are used to maintain a gap between the individual fuel pellets, as well as between the fuel pellet and the clad. Each pellet is independent and cannot provide any contribution to the clad bending stiffness or axial stiffness. As shown in Figure 11.2.16-1, the fuel rod is simply supported at each end. A static inertial loading is applied to develop a 1.23-inch lateral displacement. The purpose of the ANSYS model and solution is to provide the coordinates of the clad and pellets for the LS-DYNA model. This is accomplished by obtaining a static solution with the ANSYS model, and then using the option to update the coordinates of the nodes with the displacements from the solution. Four LS-DYNA models are considered that incorporate the bow of 1.23 inches. These cases envelop the range of the cross-sectional moments for the PWR fuel rods and the grid spacing at the bottom of the fuel assembly as summarized in the following table.

Case	Lowest Grid Spacing (inches)	Cross-Sectional Moment of Inertia	Fuel Rod OD (inch)	Fuel Clad Thickness (w/o Oxide Effect) (inch)
1	60	Minimum	0.36	0.021
2	33	Minimum	0.36	0.021
3	25	Minimum	0.36	0.021
4	60	Maximum	0.44	0.031

In each case, the thickness of the clad was reduced by 120 microns (0.0047 inch). Each case requires a separate ANSYS model and LS-DYNA model to represent unique coordinates or boundary conditions. Figure 11.2.16-2 shows a typical LS-DYNA model. The effect of the grid was imposed by constraining a node on the clad in the lateral direction. While the grid closest to the bottom end fitting can be missing, the fuel rod, in order to experience an axial loading, must bear against the bottom end fitting. The contact of the fuel rod with the end fitting, which is a perforated component, is sufficient to prevent arbitrary lateral motion of the end of the rod that is in contact with the end fitting. The LS-DYNA model employs the same nodes and elements as the ANSYS model (with the incorporation of the 1.23-inch bow). The shell elements in LS-DYNA use additional integration points to ensure that the maximum shear stress at the surface of the shell elements is accurately computed. Elastic properties used in the LS-DYNA model are the same as those used in the ANSYS model. An initial downward velocity of 527 in/sec (corresponding to a 30-foot drop) is assigned to all nodes in the model. This significantly bounds the initial momentum of the rods as compared to the 24-inch cask bottom end drop. The deceleration applied to the base of the model has a duration of 0.05 second and a 48g maximum value, which provide bounding acceleration for the 24-inch end drop of the concrete cask.

The LS-DYNA analyses were performed for a duration of 0.15 second to capture the response of the fuel after the 0.05-second loading duration. Post-processing each analysis result identifies the maximum shear stress occurring at the shell surface. The maximum shear stress result from LS-DYNA is factored by two to determine the maximum stress intensity. The maximum stress intensity is shown for each case in the following table.

Case	Maximum Stress Intensity (ksi) at Midspan of Lowest Grid Spacing	Margin of Safety Against Yield Strength
1	22.8	+2.05
2	34.8	+1.00
3	17.0	+3.09
4	15.2	+3.58

The temperature of the fuel at the bottom end of the basket is bounded by 752°F (400°C); and from Reference 61, the static yield strength for irradiated ziracloy at 752°F is 69.6 ksi. This conservatively neglects any strengthening effect due to the dynamic loading for which yield strength values are reported in Reference 61. The case using the 33-inch spacing in conjunction with the minimal cross-section (Case 2) is identified as the bounding case. The lateral response for the 60-inch grid spacing is limited since the period of the first lateral mode shape is sufficiently large to allow the maximum lateral displacement to occur significantly after the loading has ceased. For this reason, the shorter spacing of 25 inches was also analyzed (Case 3) to confirm that the 33-inch spacing (the largest to occur without missing a grid in the fuel assembly) resulted in the maximum stress intensity. The load duration used in the evaluation bounds (is larger than) the duration of lower drop heights whose accelerations are less than 48g's. Comparing Case 4 to Cases 1 and 2, the effect of the maximum cross-sectional moment (the ratio of the maximum cross-sectional moment to the minimal cross-sectional moment is approximately 2.7) indicates that the cross-sectional moment has more influence than the grid spacing on the maximum stress.

These results confirm that high burnup PWR fuel with one missing grid will remain undamaged for design basis cask end drop load conditions.

Side Drop Evaluation

The analyzed bounding fuel rod length of 60.0 inches envelopes all fuel types and includes the condition with a missing support grid in the fuel assembly. This configuration is evaluated for a 60g side drop. During a side drop, the maximum deflection of a fuel rod is based on the fuel rod spacing of the fuel assembly. Assuming a 17×17 array (fuel assembly with the maximum number of rods in Table 6.1-1), the maximum fuel rod deflection, including the 120-micron oxide layer, is:

$$(17\text{-}1) \times (0.496\text{-}0.36+2\times120\times10^{-6}\times39.37) = 2.33 \text{ in.}$$

The side drop loading is evaluated for three fuel rods, which corresponds to the limits of the stress modulus Z (ratio of the cross-sectional moment of inertia to the maximum radius to relate the maximum fiber stress (S) to the bending moment (M), S=M/Z) and the maximum span, as shown in the table below.

Case	Rod diameter (inches)	Clad thickness (inches)	Z (in^3) (10^{-3})	Span (inches)
CE14×14	0.440	0.031	3.18	16.8
WE15×15	0.417	0.024	2.20	26.2
WE17×17	0.360	0.0205	1.33	20.6

ANSYS is used to perform a static analysis with a lateral loading of 60g. The model is shown in Figure 11.2.16-3. The fuel rod is modeled with beam elements having the properties for the fuel clad taking into account the reduction of the outer radius by 0.0047 inch (120 microns). The density of the beam element material was based on the zircaloy clad (0.237 lb/in^3) and the pellet density (0.396 lb/in^3). The lateral constraints in Figure 11.2.16-3 show the location of the grids used in the model and the distance from the end of the fuel rod to the first support is 60 inches. The analyses confirm that the rod lateral displacement is 2.33 inches (for the lateral restraint configuration shown in Figure 11.2.16-3), which results in the fuel rod being supported with the 60-inch distance between adjacent grids. Therefore, the location of the unsupported span along the fuel rod is not significant. The spacing for the adjacent grids is shown in the preceding table.

To represent the maximum gap of 2.33 inches, which the fuel rod can displace in the side drop, CONTAC52s were modeled at each node. The gap for each CONTAC52 was set to 2.33 inches to limit the lateral displacement of the fuel rod to 2.33 inches. The gap stiffness for each CONTAC52 was 10^6 lb/in, and the effect of this stiffness, whether larger or smaller, would not influence the maximum stress. The maximum stress in the fuel rods is shown in the following table and the allowable is the yield strength at 752°F (69.6 ksi).

Case	Maximum Stress (ksi)	Margin of Safety Against Yield Strength
CE14×14	37.1	+0.88
WE15×15	48.1	+0.45
WE17×17	46.3	+0.50

This confirms that the PWR fuel rod subject to high burnup will remain intact for a side drop condition, which bounds the tip-over accident condition.

Figure 11.2.16-1 Three-Dimensional ANSYS Finite Element Model for UMS® Fuel Rod

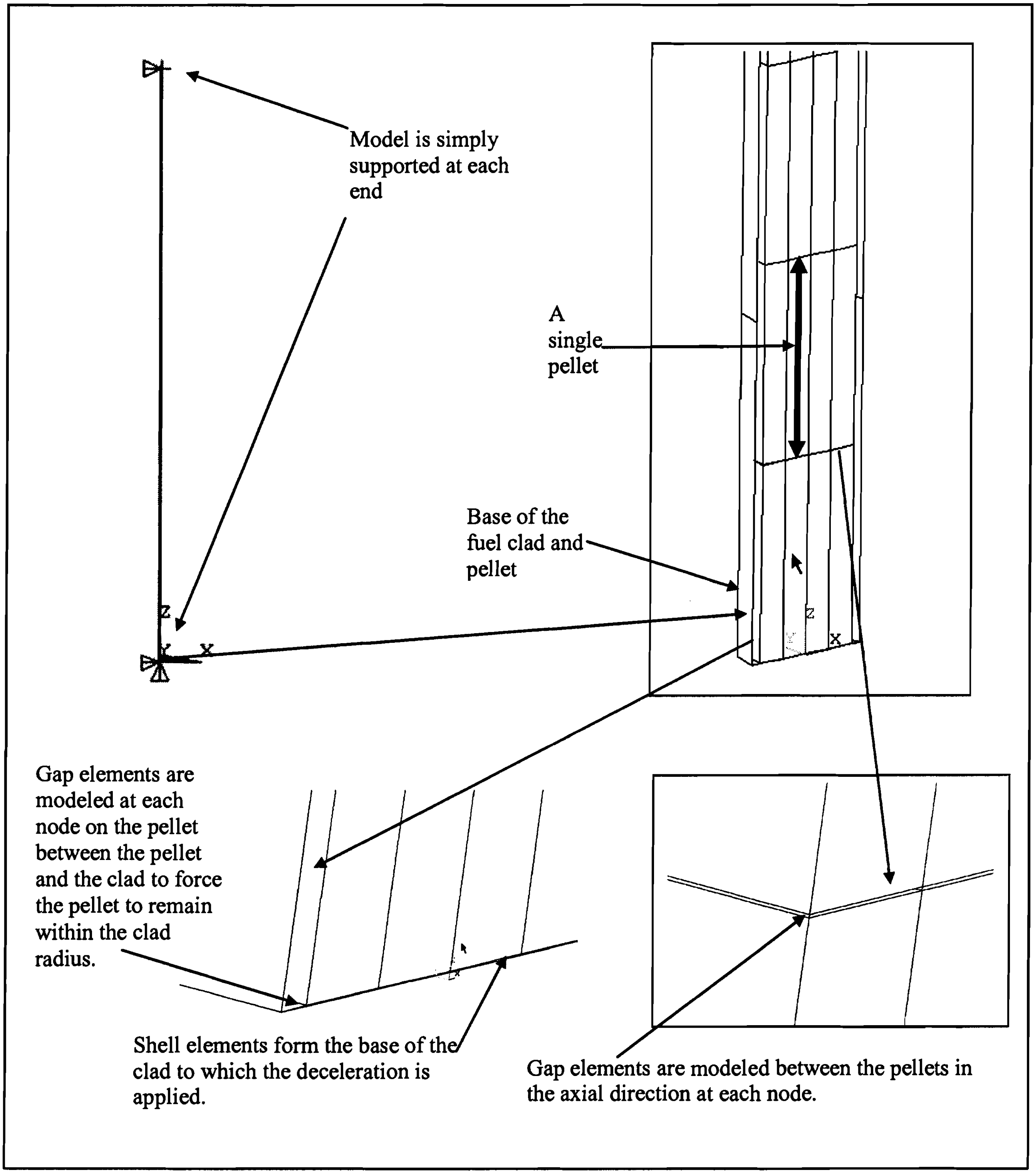
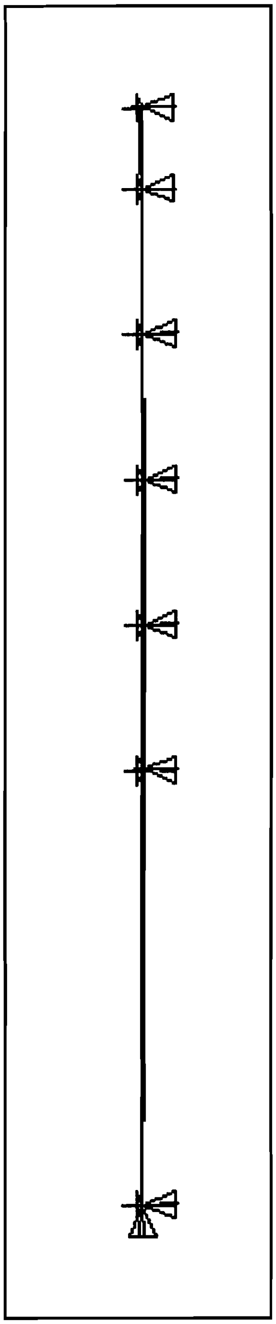


Figure 11.2.16-2 Typical Three-Dimensional LS-DYNA Model for UMS® Fuel with a 1.23-Inch Bow

Fuel rod with 1.23-inch bow with the missing grid at the lowest spacing (Case 1)



Fuel rod with 1.23-inch bow with all grids (Case 3)

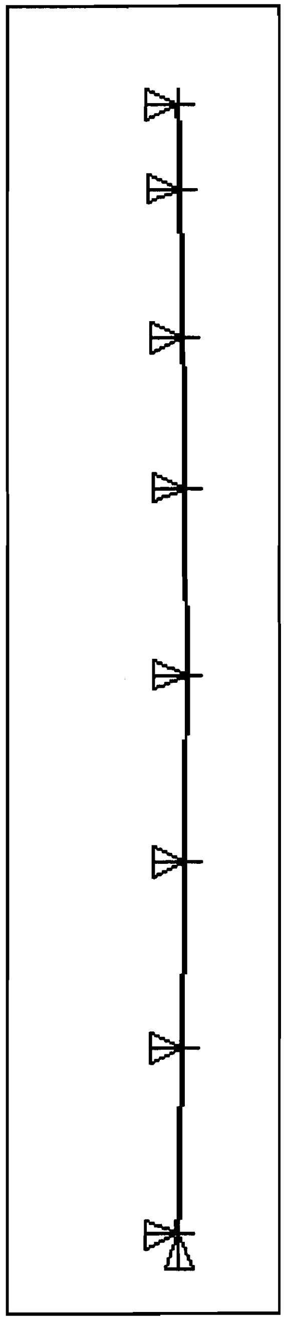
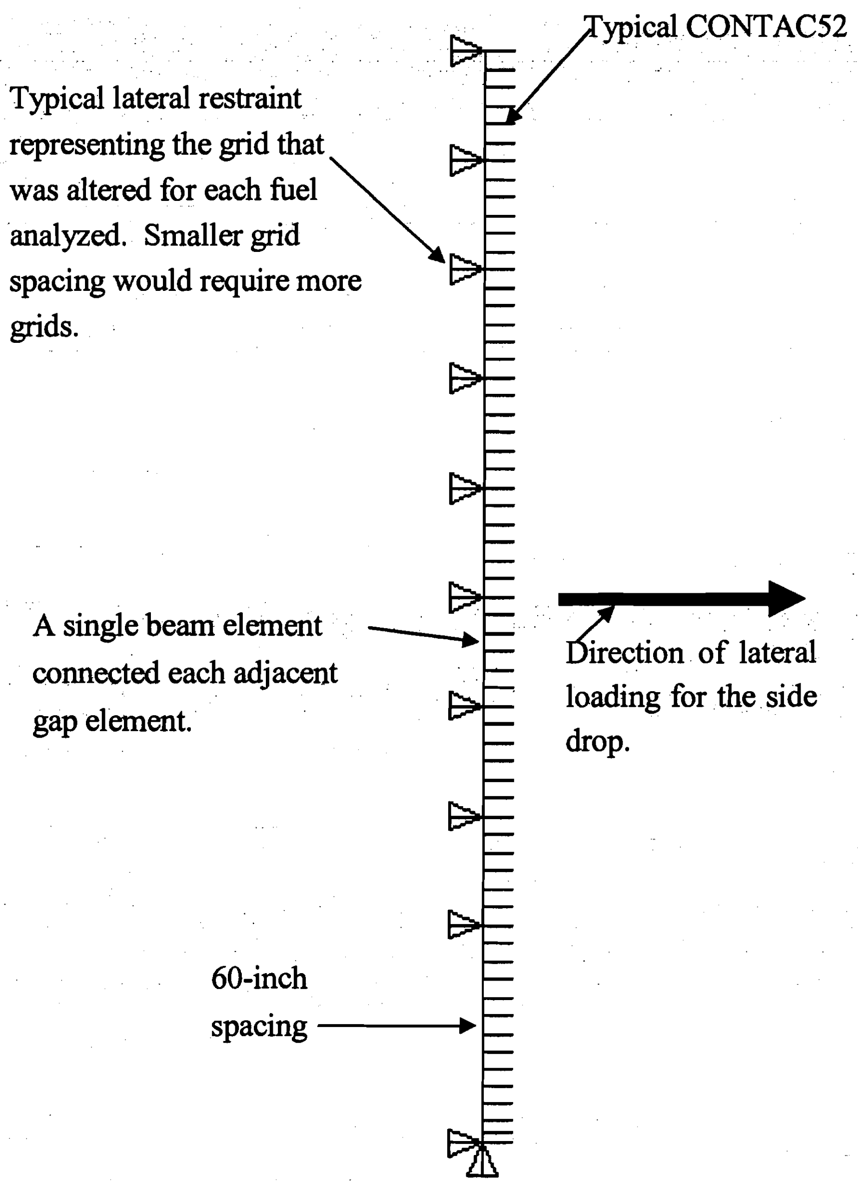


Figure 11.2.16-3 ANSYS Model for the PWR Fuel Rod High Burnup Condition



11.2.16.2 Thermal Evaluation of Fuel Rods

The UMS® system limits normal storage condition fuel cladding temperatures to levels below zirconium alloy or stainless steel cladding temperature limits; therefore, degradation is not expected to occur below this temperature in an inert gas environment.

As shown in Chapter 4, fuel cladding temperature limits for PWR fuel rods have been established at 400°C (752°F) for normal and off-normal conditions of storage, including transfer operations, and 570°C (1,058°F) for accident conditions. Chapter 4 demonstrates that the maximum fuel cladding temperatures are well below the temperature limits for all design conditions of storage.

11.3 References

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List of Tables

12.0 OPERATING CONTROLS AND LIMITS

This chapter identifies operating controls and limits, technical parameters and surveillance requirements imposed to ensure the safe operation of the NAC-UMS® System.

Controls used by NAC International (NAC) as part of the NAC-UMS® design and fabrication are provided in the NAC Quality Assurance Manual and Quality Procedure. The NAC Quality Assurance Program is discussed in Chapter 13.0. If procurement and fabrication of the NAC-UMS® System is performed by others, a Quality Assurance Program prepared in accordance with 10 CFR 72 Subpart G shall be implemented. Site specific controls for the organization, administrative system, procedures, record keeping, review, audit and reporting necessary to ensure that the NAC-UMS® storage system installation is operated in a safe manner, are the responsibility of the user of the system.

12.1 Administrative and Operating Controls and Limits for the NAC-UMS® System

The NAC-UMS® Storage System operating controls and limits are summarized in Table 12-1. Appendix A of the CoC Number 1015 Technical Specifications provides the proposed Limiting Conditions for Operations (LCO). The Approved Contents and Design Features for the NAC-UMS® System are presented in Appendix B of the CoC Number 1015 Technical Specifications. The Bases for the specified controls and limits are presented in Appendix 12C.

Section 3.0 of Appendix B presents Design Features that are important to the safe operation of the NAC-UMS® System, but that are not included as Technical Specifications. These include items which are singular events, those that cannot be readily determined or re-verified at the time of use of the system, or that are easily implemented, verified and corrected, if necessary, at the time the action is undertaken.

12.2 Administrative and Operating Controls and Limits for SITE-SPECIFIC FUEL

This section describes the administrative and operating controls and limits placed on the loading of fuel assemblies that are unique to specific reactor sites. SITE-SPECIFIC FUEL configurations result from conditions that occurred during reactor operations, participation in research and development programs, testing programs intended to improve reactor operations, from the placement of control components or other items within the fuel assembly and from the disposition of damaged fuel assemblies or fuel rods.

SITE-SPECIFIC FUEL assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation of the configuration. Separate evaluation may establish different limits, which are maintained by administrative controls for preferential loading. The preferential loading controls take advantage of design features of the UMS® Storage System to allow the loading of fuel configurations that are not specifically considered in the design basis fuel evaluation.

Unless specifically excepted, SITE SPECIFIC FUEL must meet all of the conditions specified for the design basis fuel presented in Table 12-1.

12.2.1 Operating Controls and Limits for Maine Yankee SITE-SPECIFIC FUEL

The fuel design used at Maine Yankee is the Combustion Engineering (CE) 14 × 14 fuel assembly. The CE 14 × 14 fuel assembly is one of those included in the design basis evaluation of the UMS® Storage System as shown in Table B2-2 of Appendix B of Certificate of Compliance No. 72-1015. The estimated Maine Yankee SITE-SPECIFIC FUEL inventory is shown in Table B2-6. Except as noted in this section, the spent fuel in this inventory meets the Fuel Assembly Limits provided in Table B2-1.

As shown in Table B2-6, certain of the Maine Yankee fuel has characteristics, such as fuel assembly lattice configurations, different from STANDARD FUEL, from PWR UNDAMAGED FUEL ASSEMBLIES - including CONSOLIDATED FUEL, DAMAGED FUEL and fuel with higher burnup or enrichment, that differs from the characteristics of the fuel considered in the design basis. As shown in Table B2-6, certain fuel configurations must be preferentially loaded in corner or peripheral fuel tube positions in the fuel basket based on the shielding, criticality or thermal evaluation of the fuel configuration.

The corner positions are used for the loading of fuel configurations with missing fuel rods, and for DAMAGED FUEL and CONSOLIDATED FUEL in the MAINE YANKEE FUEL CAN. Specification for placement in the corner fuel tube positions results primarily from shielding or criticality evaluations of the designated fuel configurations.

Spent fuel having a burnup from 45,000 to 50,000 MWd/MTU is assigned to peripheral locations, and may require loading in a Maine Yankee fuel can. The interior locations must be loaded with fuel that has lower burnup and/or longer cool times in order to maintain the design basis heat load and component temperature limits for the basket and canister.

The Fuel Assembly Limits for the Maine Yankee SITE SPECIFIC FUEL are shown in Table B2-7 of Appendix B of the CoC Number 1015 Technical Specifications. Part A of the table lists the STANDARD, UNDAMAGED FUEL ASSEMBLY and SITE SPECIFIC FUEL that does not require preferential loading.

Part B of the table lists the SITE SPECIFIC FUEL configurations that require preferential loading due to the criticality, shielding or thermal evaluation. The loading pattern for Maine Yankee SITE SPECIFIC FUEL that must be preferentially loaded is presented in Section B 2.1.2. The preferential loading controls take advantage of design features of the UMS® Storage System to allow the loading of fuel configurations that may have higher burnup or additional hardware or fuel source material that is not specifically considered in the design basis fuel evaluation.

Fuel assemblies with a Control Element Assembly (CEA) or a CEA plug inserted are loaded in a Class 2 canister and basket due to the increased length of the assembly with either of these components installed. However, these assemblies are not restricted as to loading position within the basket.

The Transportable Storage Canister loading procedures for Maine Yankee SITE SPECIFIC FUEL are administratively controlled in accordance with the requirements of Section B 2.1.2 for the loading of: (1) a fuel configuration with removed fuel or poison rods, (2) a MAINE YANKEE FUEL CAN, or (3) fuel with burnup between 45,000 MWd/MTU and 50,000 MWd/MTU.

Table 12-1 NAC-UMS® System Controls and Limits

Control or Limit	Applicable Technical Specification	Condition or Item Controlled
1. Fuel Characteristics	Table B2-1	Type and Condition
	Table B2-2	Class, Dimensions and Weight for PWR
	Table B2-3	Class, Dimensions and Weight for BWR
	Table B2-4	Minimum Cooling Time for PWR Fuel
	Table B2-5	Minimum Cooling Time for BWR Fuel
	Table B2-7	Maine Yankee Site Specific Fuel Limits
	Table B2-8	Minimum Cooling Time for Maine Yankee Fuel – No CEA
	Table B2-9	Minimum Cooling Time for Maine Yankee Fuel – With CEA
2. Canister	LCO 3.1.4	Time in Transfer Cask (fuel loading)
Fuel Loading	Table B2-1	Weight and Number of Assemblies
	Table B2-7	Maine Yankee Site Specific Fuel Limits
	Table B2-4	Minimum Cooling Time for PWR Fuel
	Table B2-5	Minimum Cooling Time for BWR Fuel
Drying	LCO 3.1.2	Vacuum Drying Pressure
Backfilling	LCO 3.1.3	Helium Backfill Pressure
Sealing	LCO 3.1.5	Helium Leak Rate
Vacuum	LCO 3.1.1	Time in Vacuum Drying
External Surface	LCO 3.2.1	Level of Contamination
Unloading	Note 1	Fuel Cooldown Requirement
3. Concrete Cask	LCO 3.2.2	Surface Dose Rates
	Note 1	Cask Spacing
	Note 2	Cask Handling Height
4. Surveillance	LCO 3.1.6	Heat Removal System
5. Transfer Cask	B 3.4(8)	Minimum Temperature
6. ISFSI Concrete Pad	B3.4.1(6) B3.4.2(7)	Seismic Event Performance

1. Procedure and/or limits are presented in the Operating Procedures of Chapter 8.
2. Lifting height and handling restrictions are provided in Section A5.6 of Appendix A.

APPENDIX 12A

TECHNICAL SPECIFICATIONS FOR THE NAC-UMS® SYSTEM

The Technical Specifications for the NAC-UMS® storage system, including the Limiting Conditions for Operation (LCOs), Surveillance Requirements (SRs) and the Administrative Controls and Programs, are incorporated in Appendix A of Certificate of Compliance No. 1015.

APPENDIX 12B

APPROVED CONTENTS AND DESIGN FEATURES FOR THE NAC-UMS® SYSTEM

The NAC-UMS® storage system Approved Contents and Design Features are incorporated in Appendix B of Certificate of Compliance No. 1015.

APPENDIX 12C

TECHNICAL SPECIFICATION BASES
FOR THE NAC-UMS® SYSTEM

Appendix 12C
Table of Contents

C 1.0 Introduction

This Appendix presents the design or operational condition, or regulatory requirement, which establishes the bases for the Technical Specifications provided in Appendix A of Certificate of Compliance No. 1015.

The section and paragraph numbering used in this Appendix is consistent to the numbering used in Appendix A, Technical Specifications for the NAC-UMS® SYSTEM, and Appendix B, Approved Contents and Design Features for the NAC-UMS® System, of Certificate of Compliance No. 1015.

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C 2.0 APPROVED CONTENTS

C 2.1 Fuel to be Stored in the NAC-UMS® SYSTEM

BASES

BACKGROUND	The NAC-UMS® SYSTEM design requires specifications for the spent fuel to be stored, such as the type of spent fuel, minimum and maximum allowable enrichment prior to irradiation, maximum burnup, minimum acceptable post-irradiation cooling time prior to storage, maximum decay heat, and condition of the spent fuel (i.e., UNDAMAGED FUEL). Other important limitations are the dimensions and weight of the fuel assemblies. The approved contents, which can be loaded into the NAC-UMS® SYSTEM are specified in Section B2.0 of Appendix B. Specific limitations for the NAC-UMS® SYSTEM are specified in Table B2-1 of Appendix B. These limitations support the assumptions and inputs used in the thermal, structural, shielding, and criticality evaluations performed for the NAC-UMS® SYSTEM.
APPLICABLE SAFETY ANALYSES	To ensure that the shield lid is not placed on a CANISTER containing an unauthorized fuel assembly, facility procedures require verification of the loaded fuel assemblies to ensure that the correct fuel assemblies have been loaded in the canister.
APPROVED CONTENTS	C 2.1.1 Approved Contents Section B2.0 refers to Table B2-1 in Appendix B for the specific fuel assembly characteristics for the PWR or BWR fuel assemblies authorized for loading into the NAC-UMS® SYSTEM. These fuel assembly characteristics include parameters such as cladding material, minimum and maximum enrichment, decay heat generation, post-irradiation cooling time, burnup, and fuel assembly length, width, and weight. Tables B2-2 through B2-5 are referenced from Table B2-1 and provide additional specific fuel characteristic limits for the fuel assemblies based on the fuel assembly class type, enrichment, burnup and cooling time.

(continued)

APPROVED CONTENTS (continued)

The fuel assembly characteristic limits of Tables B2-1 through B2-5 must be met to ensure that the thermal, structural, shielding, and criticality analyses supporting the NAC-UMS® SYSTEM Safety Analysis Report are bounding.

C 2.1.2

Approved Contents Section B2.0 in Appendix B requires preferential loading of Maine Yankee SITE SPECIFIC FUEL assemblies with significantly different post-irradiation cooling times. This preferential loading is required to prevent a cooler assembly from heating up due to being surrounded by hotter fuel assemblies. For the purposes of complying with this Approved Contents limit, only fuel assemblies with post-irradiation cooling times differing by one year or greater need to be loaded preferentially. This is based on the fact that the heat-up phenomenon can only occur with significant differences in decay heat generation characteristics between adjacent fuel assemblies having different post-irradiation cooling times.

APPROVED CONTENT LIMITS AND VIOLATIONS

C 2.2.1

If any Approved Contents limit of B2.1.1 or B2.1.2 in Appendix B is violated, the limitations on fuel assemblies to be loaded are not met. Action must be taken to place the affected fuel assembly(s) in a safe condition. This safe condition may be established by returning the affected fuel assembly(s) to the spent fuel pool. However, it is acceptable for the affected fuel assemblies to temporarily remain in the NAC-UMS® SYSTEM, in a wet or dry condition, if that is determined to be a safe condition.

C 2.2.2 and C 2.2.3

NRC notification of the Approved Contents limit violation is required within 24 hours. A written report on the violation must be submitted to the NRC within 30 days. This notification and written report are independent of any reports and notification that may be required by 10 CFR 72.216.

REFERENCES

1. FSAR, Sections 2.1, 4.4; Chapters 5 and 6.

C 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the NAC-UMS® SYSTEM is in the specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within the specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and,

b. Completion of the Required Actions is not required when an LCO is meet within the specified Completion Time, unless otherwise specified.

There are two basic Required Action types. The first Required Action type specifies a time limit, the Completion Time to restore a system or component or to restore variables to within specified limits, in which the LCO must be met. Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second Required Action type specifies the remedial measures that permit continued activities that are not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

(continued)

LCO 3.0.2 (continued)

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillance, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

LCO 3.0.3

This specification is not applicable to the NAC-UMS® SYSTEM because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the facility in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. NAC-UMS® SYSTEM conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in NAC-UMS® SYSTEM activities being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the NAC-UMS® SYSTEM. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

(continued)

LCO 3.0.4 (continued)

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of the NAC-UMS® SYSTEM.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with the ACTIONS. The sole purpose of the Specification is to provide an exception to LCO 3.0.2 (e.g. to not comply with the applicable Required Action[s]) to allow the performance of testing to demonstrate:

a. The equipment being returned to service meets the LCO; or

b. Other equipment meets the applicable LCOs.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

C 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

Surveillance Requirements (SRs)

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillance is performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to meet Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or,

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the NAC-UMS® SYSTEM is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including those invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance

(continued)

SR 3.0.1 (continued)	testing may not be possible in the current specified conditions in the Applicability, due to the necessary NAC-UMS® SYSTEM parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a specified condition where other necessary post-maintenance tests can be completed.
SR 3.0.2	SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per…" interval. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities). The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable." As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per…" basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.

(continued)

SR 3.0.2 (continued)	The provisions of SR 3.0.2 are not intended to be used repeatedly, merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.
SR 3.0.3	SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance. The basis for this delay period includes: consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency, based not on time intervals, but upon specified NAC-UMS® SYSTEM conditions, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance. SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions. Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility, which is not intended to be used as an operational convenience to extend Surveillance intervals.

(continued)

SR 3.0.3 (continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe operation of NAC-UMS® SYSTEM activities.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO.

(continued)

SR 3.0.4 (continued)

When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s), since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in a SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not in this situation, LCO 3.0.4 will govern any restrictions that may be (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of the NAC-UMS® SYSTEM.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances, when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO, prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering LCO Applicability, would have its Frequency specified such that is not “due” until the specific conditions needed are met.

Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or to be performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs’ annotation is found in Section 1.4, Frequency.

C 3.1 NAC-UMS® SYSTEM Integrity
C 3.1.1 CANISTER Maximum Time in Vacuum Drying
BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved to a preparation area, where dose rates are measured. The CANISTER shield lid is welded to the CANISTER shell, and the lid weld is examined and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium, and the CANISTER drain and vent port covers are installed, welded and examined. The shield lid welds are then helium leak tested using the evacuated envelope method, per ANSI N14.5. The structural lid is installed, welded and examined. Dose and contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.

Limiting the elapsed time from the end of CANISTER draining operations through dryness verification testing and subsequent backfilling of the CANISTER with helium ensures that the short-term temperature limits established in the Safety Analyses Report for the spent fuel cladding and CANISTER materials are not exceeded and that the test duration of 30 days (720 hours) considered in PNL-4835 for zirconium alloy clad fuel for storage in air is not exceeded.

A CANISTER containing a fuel assembly with burnup greater than 45 GWd/MTU is limited to nine (9) or fewer cooling/vacuum drying cycles performed in accordance with LCO 3.1.1.2. Each cooling/ vacuum drying cycle will exceed the cladding temperature change limit of 117°F (65°C). Excessive cladding temperature cycles (>10) of high burnup fuel could result in undesirable hydride reorientation as described in ISG-11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," and reported by F. Kammenzind, B. M. Berquist and R. Bajaj in "The Long Range Migration of Hydrogen Through Zircaloy in Response to Tensile and Compressive Stress Gradients."

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APPLICABLE SAFETY ANALYSIS

Limiting the total time for loaded CANISTER vacuum drying operations ensures that the short-term temperature limits for the fuel cladding and CANISTER materials are not exceeded. If vacuum drying operations are not completed in the required time period, the CANISTER is backfilled with helium and cooled for a minimum of 24 hours of in-pool cooling or forced air cooling.

Analyses reported in the Safety Analysis Report conclude that spent fuel cladding and CANISTER material short-term temperature limits will not be exceeded for total elapsed time in the vacuum drying operation and in the TRANSFER CASK with the CANISTER filled with helium. Since the rate of heat up is slower for lower total heat loads, the time required to reach component limits is longer than for the design basis heat load. Consequently, longer time limits are specified for heat loads below the design basis for the PWR and BWR fuel configurations as shown in LCO 3.1.1. As shown in the LCO, for total heat loads not specified, the time limit for the next higher specified heat load is conservatively applied. Analyses show that the fuel cladding and CANISTER component temperatures are below the allowable temperatures for the time durations specified from the completion of CANISTER draining, or from the end of in-pool cooling or forced air cooling, through the completion of vacuum drying, dryness verification testing per LCO 3.1.2, and the helium backfill process per LCO 3.1.3[1].

Following completion of helium backfill, the fuel cladding and CANISTER temperatures are also maintained within allowable limits for the time(s) specified in LCO 3.1.4 for the helium-filled CANISTER in the TRANSFER CASK through completion of the transfer of the CANISTER to the CONCRETE CASK.

(continued)

LCO	Limiting the length of time for vacuum drying operations through completion of the helium backfill operations for the CANISTER ensures that the spent fuel cladding and CANISTER material temperatures remain below the short-term temperature limits for the NAC-UMS® SYSTEM. Limiting a CANISTER containing a fuel assembly with burnup greater than 45 GWd/MTU to nine (9) or fewer cooling/vacuum drying cycles, per LCO 3.1.1.2, where the fuel cladding temperature change is greater than 117°F (65°C) controls hydride reorientation, maintains fuel rod cladding structural integrity and assures fuel retrievability.
APPLICABILITY	The elapsed time restrictions for vacuum drying operations on a loaded CANISTER apply during LOADING OPERATIONS from the completion of CANISTER draining operations through completion of dryness verification testing per LCO 3.1.2 and the completion of the helium backfill process per LCO 3.1.3[(1)]. LCO 3.1.1 is not applicable to TRANSPORT OPERATIONS or STORAGE OPERATIONS.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each NAC-UMS® SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each NAC-UMS® SYSTEM not meeting the LCO. Subsequent NAC-UMS® SYSTEMS that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the LCO time limit is exceeded, the CANISTER will be backfilled with helium to a pressure of 0 psig (+1,-0). <u>AND</u>

(continued)

ACTIONS (continued)

A.2.1.1
The TRANSFER CASK containing the loaded CANISTER shall be placed in the spent fuel pool. For in-pool cooling operations with the TRANSFER CASK and loaded CANISTER submerged, the annulus fill system is not required to be operating. If only the loaded CANISTER is submerged for in-pool cooling, the annulus fill system is required to be operating.

AND

A.2.1.2
The TRANSFER CASK and loaded CANISTER shall be maintained in the spent fuel pool with the water level above the top of the CANISTER, and a maximum water temperature of 100°F for a minimum of 24 hours prior to the restart of LOADING OPERATIONS.

OR

A.2.2.1
A cooling air flow of 375 CFM at a maximum temperature of 76°F shall be initiated. The airflow will be routed to the annulus fill/drain lines of the TRANSFER CASK and will flow through the annulus and cool the CANISTER.

AND

A.2.2.2
The cooling air flow shall be maintained for a minimum of 24 hours prior to restart of LOADING OPERATIONS.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The elapsed time shall be monitored from completion of CANISTER draining through completion of the vacuum dryness verification testing per LCO 3.1.2 and completion of the helium backfill process per LCO 3.1.3[(1)]. Monitoring the elapsed time ensures that if the drying process is not completed in the prescribed time, the CANISTER can be backfilled with helium and in-pool or forced air cooling operations initiated in a timely manner during LOADING OPERATIONS to prevent fuel cladding and CANISTER materials from exceeding short-term temperature limits.

SR 3.1.1.2

The elapsed time shall be monitored from the end of in-pool cooling or forced air cooling of the CANISTER through completion of vacuum dryness verification testing per LCO 3.1.2 and the completion of the helium backfill process per LCO 3.1.3[(1)]. Monitoring the elapsed time ensures that if the drying process is not completed in the prescribed time, the CANISTER can be backfilled with helium and in-pool or forced air cooling initiated in a timely manner during LOADING OPERATIONS to prevent the fuel cladding and CANISTER materials from exceeding short-term temperature limits.

REFERENCES

1. FSAR Sections 4.4 and 8.1.

Note:

[(1)] LCO 3.1.1, SR 3.1.1.1 and SR 3.1.1.2 specify time limitations and monitoring requirements for the allowable duration(s) from completion of draining of the CANISTER, or from the completion of in-pool or forced air cooling of the CANISTER, through completion of vacuum drying testing and the "introduction" of helium. Clarifications have been added to the Bases of LCO 3.1.1 to highlight that the introduction and start of helium backfill defines the system configuration that is established following completion of final helium pressure adjustment of the CANISTER as specified in LCO 3.1.3.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.2 CANISTER Vacuum Drying Pressure

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents Limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved to a preparation area, where dose rates are measured. The CANISTER shield lid is welded to the CANISTER shell, and the lid weld is examined and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium, and the CANISTER drain and vent port covers are installed, welded and examined. The shield lid weld is then helium leak tested using the evacuated envelope method, per ANSI N14.5. The structural lid is installed, welded and examined. Dose and contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad. CANISTER cavity vacuum drying is utilized to remove residual moisture from the CANISTER cavity after the water is drained from the CANISTER. Any water not drained from the CANISTER cavity evaporates due to the vacuum. This is aided by the temperature increase, due to the heat generation of the fuel.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of design basis spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on limiting the fuel cladding temperatures, the total number of thermal cycles (for high burnup fuel only), and establishing and maintaining an inert atmosphere in the CANISTER. This is accomplished by removing water from the CANISTER and backfilling the cavity with helium. The thermal analysis assumes that the CANISTER cavity is dried and filled with helium.

(continued)

APPLICABLE SAFETY ANALYSIS (continued)	The heat-up and thermal cycling of the CANISTER and contents will occur during CANISTER vacuum drying, but is controlled by LCO 3.1.1. Dryness of the CANISTER (e.g., no free water) is verified by holding a vacuum pressure below or equal to a selected pressure for a specified period of time. The vacuum pressure selected for this verification is related to the temperature of the environment the CANISTER is in while vacuum drying (i.e., either the spent fuel pool (SFP) water temperature for CANISTERS vacuum dried in the SFP, or the cask preparation area ambient air temperature for CANISTERS vacuum dried outside the SFP). The nominal vacuum pressure selected for the verification in the LCO is 10 mm of Hg, which corresponds to approximately one-half of the vapor pressure of water at 70°F. The temperature of the drying environment at facilities loading CANISTERS is expected to exceed this temperature under most circumstances. In the event that either SFP water temperature (for CANISTERS vacuum dried in the SFP) or the cask preparation area ambient air temperature (for CANISTERS vacuum dried outside the SFP) is below 65°F, a lower vacuum pressure of 5 mm of Hg shall be used as the test criterion. For either verification, a 10-minute hold period has been selected. Holding the vacuum pressure below 10 mm of Hg for 10 minutes (or under 5 mm at SFP or ambient temperatures <65°F), with the CANISTER isolated from the vacuum pump and the pump turned off, demonstrates that there is no free water in the CANISTER, since the presence of any significant free water would result in the vacuum pressure increasing in a short period of time to the vapor pressure corresponding to the average temperature of the CANISTER and contents, which is significantly greater than the selected vacuum pressure.
LCO	A vacuum pressure of ≤10 mm of mercury, as specified in this LCO, indicates that liquid water has evaporated and been removed from the CANISTER cavity. Removing water from the CANISTER cavity helps to ensure the long-term maintenance of fuel cladding integrity.
APPLICABILITY	Cavity vacuum drying is performed during LOADING OPERATIONS before the TRANSFER CASK holding the CANISTER is moved to transfer the CANISTER into the CONCRETE CASK. Therefore, the vacuum requirements do not apply after the CANISTER is backfilled with helium and leak tested prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

(continued)

ACTIONS (continued)

A.1

If the CANISTER cavity vacuum drying pressure limit cannot be met, actions must be taken to meet the LCO. Failure to successfully complete cavity vacuum drying could have many causes, such as failure of the vacuum drying system, inadequate draining, ice clogging of the drain lines, or leaking CANISTER welds. The Completion Time is sufficient to determine and correct most failure mechanisms. Excessive heat-up and thermal cycling of the CANISTER and contents is precluded by LCO 3.1.1.

B.1

If the CANISTER fuel cavity cannot be successfully vacuum dried, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met.

A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1. The Completion Time is reasonable, based on the time required to reflood the CANISTER, perform fuel cooldown operations, cut the shield lid weld, move the TRANSFER CASK into the spent fuel pool, and remove the CANISTER shield lid in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Cavity dryness is demonstrated by evacuating the cavity to a very low absolute pressure and verifying that the pressure remains below a specified vapor pressure for a specific period of time. A low vacuum pressure is an indication that the cavity is dry. The surveillance must be performed prior to TRANSPORT OPERATIONS, as the vacuum drying pressure must be achieved before the CANISTER is sealed. This allows sufficient time to backfill the CANISTER cavity with helium, while minimizing the time the fuel is in the CANISTER without water or the assumed inert atmosphere in the cavity.

REFERENCES

1. FSAR Sections 4.4, 7.1 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.3 CANISTER Helium Backfill Pressure

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved to a preparation area, where dose rates are measured. The CANISTER shield lid is welded to the CANISTER shell, and the lid weld is examined and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed and verified. The CANISTER cavity is then evacuated to ≤3 mm of mercury to remove any residual oxidizing gases and the cavity is backfilled with helium. The CANISTER drain and vent port covers are installed, welded and examined. The shield lid weld is then helium leak tested using the evacuated envelope method, per ANSI N14.5. The structural lid is installed, welded and examined. Dose and contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad. Evacuating and backfilling of the CANISTER cavity with helium removes residual oxidizing gases to ≤1 mole, promotes heat transfer from the spent fuel to the CANISTER structure and protects the fuel cladding. Providing a helium pressure equal to atmospheric pressure ensures that there will be no in-leakage of air over the life of the CANISTER, which might be harmful to the heat transfer features of the NAC-UMS® SYSTEM and harmful to the fuel.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on the ability of the NAC-UMS® SYSTEM to remove heat from the CANISTER and reject it to the

(continued)

APPLICABLE SAFETY ANALYSIS (continued)	environment. This is accomplished by removing water from the CANISTER cavity and backfilling the cavity with an inert gas. The heat-up of the CANISTER and contents will continue following backfilling with helium, but is controlled by LCO 3.1.4. The thermal analyses of the CANISTER assume that the CANISTER cavity is dried and filled with dry helium.
LCO	Backfilling the CANISTER cavity with helium at a pressure equal to atmospheric pressure ensures that there is no air in-leakage into the CANISTER, which could decrease the heat transfer properties and result in increased cladding temperatures and damage to the fuel cladding over the storage period. The helium backfill pressure of 0 psig specified in this LCO was selected based on a minimum helium purity of 99.9% to ensure that the CANISTER internal pressure and heat transfer from the CANISTER to the environment are maintained consistent with the design and analysis basis of the CANISTER.
APPLICABILITY	Helium backfill is performed during LOADING OPERATIONS, before the TRANSFER CASK and CANISTER are moved to the CONCRETE CASK for transfer of the CANISTER. Therefore, the backfill pressure requirements do not apply after the CANISTER is backfilled with helium and leak tested prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent condition entry and application of associated Required Actions. <u>A.1</u> If the backfill pressure cannot be established within limits, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which would prevent backfilling of the CANISTER cavity with helium. These actions include identification and repair of helium leak paths or replacement of the helium backfill equipment. In addition, the CANISTER can be maintained in a safe condition based on the use of forced air cooling or water cooling.

(continued)

ACTIONS (continued)

B.1

If the CANISTER cavity cannot be backfilled with helium to the specified pressure, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 cannot be extended by reperforming A.1. The Completion Time is reasonable based on the time required to re-flood the CANISTER, perform cooldown operations, cut the CANISTER shield lid weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the CANISTER shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert atmosphere and maintenance of adequate heat transfer mechanisms. Filling the CANISTER cavity with helium at a pressure within the range specified in this LCO will ensure that there will be no air in-leakage, which could potentially damage the fuel. This pressure of helium gas is sufficient to maintain fuel cladding temperatures within acceptable levels.

Backfilling of the CANISTER cavity must be performed successfully on each CANISTER before placing it in storage. The surveillance must verify that the CANISTER helium backfill pressure is within the limit specified prior to installation of the structural lid.

REFERENCES

1. FSAR Sections 4.4, 7.1 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.4 CANISTER Maximum Time in the TRANSFER CASK

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved to a preparation area, where dose rates are measured. The CANISTER shield lid is welded to the CANISTER shell, and the lid weld is examined and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium, and the CANISTER drain and vent port covers are installed, welded and examined. The shield lid weld is then helium leak tested using the evacuated envelope method, per ANSI N14.5. The structural lid is installed, welded and examined. Dose and contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.

Backfilling the CANISTER cavity with helium promotes heat transfer from the fuel and the inert atmosphere protects the fuel cladding. The cumulative time a loaded, helium backfilled CANISTER may remain in the TRANSFER CASK is limited to 600 hours. This limit ensures that the test duration of 30 days (720 hours) considered in PNL-4835 for zirconium alloy clad fuel for storage in air is not exceeded and ensures that the TRANSFER CASK is used as intended. The time limit is established to preclude long-term storage of a loaded CANISTER in the TRANSFER CASK.

Intermediate time limits are established for CANISTERS with heat loads above 20 kW (PWR) or 17 kW (BWR) if they are not in either forced air cooling or in-pool cooling. These intermediate limits assure that the short-term temperature limits established in the Safety Analysis Report for the spent fuel cladding and CANISTER materials are not exceeded. Placing the CANISTER in either forced air cooling or in-pool cooling for a minimum of 24 hours maintains temperatures within the short-term limits. For heat loads less than or equal to 20kW (PWR) or 17kW (BWR), neither forced air cooling nor in-pool cooling is required.

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APPLICABLE SAFETY ANALYSIS

Analyses reported in the Safety Analysis Report conclude that for heat loads greater than 20 kW (PWR) or greater than 17 kW (BWR), spent fuel cladding and CANISTER material short-term temperature limits will not be exceeded for the total elapsed times specified in LCO 3.1.4. As shown in the LCO, for total heat loads not specified, the time limit for the next higher specified heat load is conservatively applied. The thermal analysis shows that the fuel cladding and CANISTER component temperatures are below their allowable temperatures for the time durations specified, with the CANISTER in the TRANSFER CASK and backfilled with helium, after completion of 24 hours of in-pool cooling or forced air cooling. For lower heat loads, the steady state fuel cladding and component temperatures are below the allowable temperatures.

The basis for forced air cooling is an inlet maximum air temperature of 76°F which is the maximum normal ambient air temperature in the thermal analysis. The specified 375 CFM air flow rate exceeds the CONCRETE CASK natural convective cooling flow rate by a minimum of 10 percent. This comparative analysis conservatively excludes the higher flow velocity resulting from the smaller annulus between the TRANSFER CASK and CANISTER, which would result in improved heat transfer from the CANISTER.

From calculated temperatures reported in the Safety Analysis Report, it can be concluded that spent fuel cladding and CANISTER material short-term temperature limits will not be exceeded for a total elapsed time of greater than 20 hours for PWR fuel or 30 hours for BWR fuel for high heat loads, if the loaded CANISTER backfilled with helium is in the TRANSFER CASK. A 2 hour completion time is provided to establish in-pool or forced airflow cooling to ensure cooling of the CANISTER.

For heat loads of 20 kW or less (PWR), or 17 kW or less (BWR), and with the CANISTER backfilled with helium, the analysis shows that the fuel cladding and CANISTER components reach a steady-state temperature below the short-term allowable temperatures. Therefore, the time in the TRANSFER CASK is limited to 600 hours. For heat loads greater than 20 kW (PWR) or greater than 17 kW (BWR), and if the intermediate time is exceeded, the analysis shows that if in-pool cooling or forced air cooling at 375 CFM with air at 76°F is used, the temperatures of the fuel cladding and CANISTER components will not exceed short-term temperature limits.

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APPLICABLE SAFETY ANALYSIS (continued)

This limit ensures that the test duration of 30 days (720 hours) considered in PNL-4835 for zirconium alloy clad fuel for storage in air is not exceeded and ensures that the TRANSFER CASK is used as intended. Since the 600 hours is significantly less than the 720 hours considered in PNL-4835, operation in the TRANSFER CASK to this period is acceptable.

Since the cooling provided by the forced air is equivalent to the passive cooling provided by the CONCRETE CASK and TRANSPORT CASK, relocation of a loaded and helium-filled CANISTER to a CONCRETE CASK or TRANSPORT CASK ensures that the fuel cladding and CANISTER component short-term temperature limits are not exceeded.

LCO

For PWR heat loads less than or equal to 20 kW, and BWR heat loads less than or equal to 17 kW, the thermal analysis shows that the presence of helium in the CANISTER is sufficient to maintain the fuel cladding and CANISTER component temperatures below the short-term temperature limits. Therefore, forced air cooling or in-pool cooling is not required for these heat load conditions.

For higher heat loads of these fuels, as shown in the LCO, once forced air cooling or in-pool cooling is established, the amount of time the CANISTER resides in the TRANSFER CASK is not limited by the intermediate time limits, since the cooling provided by the forced air or water is equivalent to the passive cooling that is provided by the CONCRETE CASK or TRANSPORT CASK. If forced air flow or in-pool cooling is continuously maintained for a period of 24 hours, or longer, then the temperatures of the spent fuel cladding and CANISTER components are at, or below, the values calculated for the CONCRETE CASK normal conditions. Therefore, forced air cooling or in-pool cooling may be ended, allowing a new entry into Condition A of this LCO. This provides a new period in which continuation of LOADING OPERATIONS, TRANSFER OPERATIONS or UNLOADING OPERATIONS for high heat load PWR and BWR fuel may occur.

Similarly, in LOADING OPERATIONS, TRANSFER OPERATIONS or UNLOADING OPERATIONS for heat loads up to the design basis, continuous forced air cooling or in-pool cooling maintains the fuel cladding and CANISTER component temperatures below the short-term temperature limits. Therefore, the CANISTER may remain in the TRANSFER CASK for up to 600 hours, where the time limit is based on the test duration of 30 days (720 hours) considered in PNL-4835 for zirconium alloy clad fuel for storage in air rather than on temperature limits.

(continued)

APPLICABILITY	For LOADING OPERATIONS, the elapsed time restrictions on the loaded CANISTER apply from the completion point of the CANISTER helium backfilling through completion of the transfer from the TRANSFER CASK to the CONCRETE CASK and installing the CONCRETE CASK shield plug and cask lid. For TRANSFER OPERATIONS, the elapsed time restrictions on the loaded CANISTER apply from the completion point of the closing of the TRANSFER CASK shield doors through completion of the unloading of the CANISTER from the TRANSFER CASK. For UNLOADING OPERATIONS, the elapsed time restrictions on the loaded CANISTER apply from the completion point of the closing of the TRANSFER CASK shield doors through initiation of CANISTER cooldown.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each NAC-UMS® SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each NAC-UMS® SYSTEM not meeting the LCO. Subsequent NAC-UMS® SYSTEMS that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. A note has been added to Condition A that reminds users that all time spent in Condition A is included in the 600-hour cumulative limit. If LCO 3.1.4 intermediate time is exceeded: A.1.1 The TRANSFER CASK containing the loaded CANISTER shall be placed in the spent fuel pool. For in-pool cooling operations with the TRANSFER CASK and loaded CANISTER submerged, the annulus fill system is not required to be operating. If only the loaded CANISTER is submerged for in-pool cooling, the annulus fill system is required to be operating. AND A.1.2 The TRANSFER CASK and a loaded CANISTER shall be maintained in the spent fuel pool having a maximum water temperature of 100°F for a minimum of 24 hours prior to restart of LOADING OPERATIONS, TRANSFER OPERATIONS or UNLOADING OPERATIONS.

(continued)

ACTIONS (continued)	OR A.2.1 A cooling air flow of 375 CFM at a maximum temperature of 76° F shall be initiated. The airflow will be routed to the annulus fill/drain lines in the TRANSFER CASK and will flow through the annulus and cool the CANISTER. AND A.2.2 The cooling air flow shall be maintained for a minimum of 24 hours prior to restart of LOADING OPERATIONS, TRANSFER OPERATIONS or UNLOADING OPERATIONS. If the LCO 3.1.4. 600-hour cumulative time limit is exceeded: B.1 The CANISTER shall be placed in a CONCRETE CASK. OR B.2 The CANISTER shall be placed in a TRANSPORT CASK. OR B.3 The CANISTER shall be unloaded. The 5-day Completion Time for Required Actions B.1, B.2, and B.3 assures that the PNL-4835 30-day test duration used to establish the LCO limit will not be exceeded, taking into account the 600 hours allowed by the LCO.
SURVEILLANCE REQUIREMENTS	SR 3.1.4.1 The elapsed time from entry into the LCO conditions of Applicability until placement of the CANISTER in a CONCRETE CASK or TRANSPORT CASK, or until CANISTER cooldown is initiated for UNLOADING OPERATIONS shall be monitored. This SR ensures that the fuel cladding and CANISTER component temperature limits are not exceeded.
REFERENCES	1. FSAR Sections 4.4, 8.1 and 8.2.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.5 CANISTER Helium Leak Rate

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved to a preparation area, where dose rates are measured. The CANISTER shield lid is welded to the CANISTER shell, and the lid weld is examined and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium, and the CANISTER drain and vent port covers are installed, welded and examined. The shield lid weld is then helium leak tested using the evacuated envelope method, per ANSI N14.5. The structural lid is installed, welded and examined. Dose and contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad. Backfilling the CANISTER cavity with helium promotes heat transfer from the fuel to the CANISTER shell. The inert atmosphere protects the fuel cladding. Prior to transferring the CANISTER to the CONCRETE CASK, the CANISTER helium leak rate is verified to ensure that the fuel and helium backfill gas is confined and that there will be no credible leakage from the CANISTER.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on maintaining an inert atmosphere, and maintaining the cladding temperatures below established long-term limits. This is accomplished by removing water from the CANISTER and backfilling the cavity with helium. The heat-up of the CANISTER and contents will continue following backfilling the cavity and leak testing the shield lid-to-shell weld, but is controlled by LCO 3.1.4.

(continued)

LCO

Verifying that the CANISTER cavity helium leak rate is below the value specified in this LCO ensures that the CANISTER shield lid is sealed. Verifying the helium leak rate will also ensure that there will be no credible leakage from the CANISTER under off-normal or accident conditions.

APPLICABILITY

The helium leak rate verification is performed during LOADING OPERATIONS before the TRANSFER CASK and integral CANISTER are moved for transfer operations to the CONCRETE CASK. TRANSPORT OPERATIONS would not commence if the CANISTER helium leak rate was not below the test sensitivity. Therefore, CANISTER leak rate testing is not required during TRANSPORT OPERATIONS or STORAGE OPERATIONS.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the helium leak rate limit is not met, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which could cause a helium leak rate in excess of the limit. Actions to correct a failure to meet the helium leak rate limit would include, in ascending order of performance: 1) verification of helium leak test system performance; 2) inspection of weld surfaces to locate helium leakage paths using a helium sniffer probe; and 3) weld repairs, as required, to eliminate the helium leakage. Following corrective actions, the helium leak rate verification shall be reperformed.

(continued)

ACTIONS (continued)

B.1

If the CANISTER leak rate cannot be brought within the limit, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 cannot be extended by reperforming A.1. The Completion Time is reasonable based on the time required to reflood the CANISTER, perform fuel cooldown operations, cut the CANISTER shield lid weld, move the TRANSFER CASK into the spent fuel pool, remove the CANISTER shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

The primary design considerations of the CANISTER are that there will be no credible leakage and that the helium remains in the CANISTER during long-term storage. Long-term integrity of the stored fuel is dependent on storage in a dry, inert environment.

The helium leakage rate of each CANISTER shall be confirmed to meet the LCO prior to TRANSPORT OPERATIONS. The Surveillance Frequency allows sufficient time to backfill the CANISTER cavity with helium and to perform the leak test, while minimizing the time the fuel is in the CANISTER and loaded in the TRANSFER CASK.

REFERENCES

1. FSAR Sections 7.1 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.6 CONCRETE CASK Heat Removal System

BASES

BACKGROUND	The CONCRETE CASK Heat Removal System is a passive, air-cooled convective heat transfer system, which ensures that heat from the CANISTER is transferred to the environment by the upward flow of air through the CONCRETE CASK. Relatively cool air is drawn into the annulus between the CONCRETE CASK and the CANISTER through the four air inlets at the bottom of the CONCRETE CASK. The CANISTER transfers its heat from the CANISTER surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air flows back into the environment through the four air outlets at the top of the CONCRETE CASK.
APPLICABLE SAFETY ANALYSIS	The thermal analyses of the CONCRETE CASK take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the CONCRETE CASK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and CANISTER component temperatures do not exceed applicable limits. Under normal storage conditions, the four air inlets and four air outlets are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs. Analyses have been performed for the complete obstruction of all of the air inlets and outlets. The complete blockage of all air inlets and outlets stops air cooling of the CANISTER. The CANISTER will continue to radiate heat to the relatively cooler inner shell of the CONCRETE CASK. With the loss of air cooling, the CANISTER component temperatures will increase toward their respective short-term temperature limits. The limiting components are the CANISTER basket support and heat transfer disks, which, by analysis, approach their temperature limits in 24 hours, if no action is taken to restore air flow to the heat removal system. The maximum fuel clad temperatures remain below allowable accident limits for approximately six days (150 hours) with complete air flow blockage.
LCO	The CONCRETE CASK Heat Removal System must be verified to be OPERABLE to preserve the assumptions of the thermal analyses.

(continued)

LCO (continued)	Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environment at a sufficient rate to maintain fuel cladding and CANISTER component temperatures within design limits.
APPLICABILITY	The LCO is applicable during STORAGE OPERATIONS. Once a CONCRETE CASK containing a CANISTER loaded with spent fuel has been placed in storage, the heat removal system must be OPERABLE to ensure adequate heat transfer of the decay heat away from the fuel assemblies.
ACTIONS	A note has been added to ACTIONS that states for this LCO, separate Condition entry is allowed for each CONCRETE CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each CONCRETE CASK not meeting the LCO. Subsequent CONCRETE CASKs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the CONCRETE CASK heat removal system has been determined to not be OPERABLE, it must be restored to an analyzed safe status immediately, with adequate heat removal capability. Immediately, defined as the required action to be pursued without delay and in a controlled manner, provides a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path.

In order to meet A.1, adequate heat removal capability must be verified to exist, either by visual observation of at least two unobstructed air inlet and outlet screens or by physically clearing any blockage from two air inlet and outlet screens, to prevent exceeding the short-term temperature limits.

Thermal analysis of a fully blocked CONCRETE CASK shows that without adequate heat removal, the fuel cladding accident temperature limit could be exceeded over time. As a result, requiring immediate verification of adequate heat removal capability will ensure that the CONCRETE CASK and CANISTER components and the fuel cladding do not exceed their short-term temperature limits.

The thermal analysis also shows that complete blockage of two air inlet and outlet screens results in no potential for exceeding accident fuel cladding, CONCRETE CASK or CANISTER component temperature limits. As a result, verifying that there are at least two unobstructed

(continued)

ACTIONS (continued)

air inlet and outlet screens will ensure that the accident temperature limits are not exceeded during the time that the remainder of the air linlet and outlet screens are returned to OPERABLE status.

AND

A.2

In addition to Required Action A.1, the fuel loading per the Approved Contents condition of the CoC is verified.

The Completion Time for this Required Action of 7 days will ensure that the CANISTER remains in a safe, analyzed condition.

AND

A.3

In addition to Required Actions A.1 and A.2 that ensure the adequate heat removal capability and verify the fuel loading, restoring the CONCRETE CASK Heat Removal System to OPERABLE is not an immediate concern. Therefore, restoring it within 25 days is considered a reasonable period of time.

B.1

If the Required Actions A.1, A.2 or A.3 cannot be met, an engineering evaluation is performed to verify that the CONCRETE CASK heat removal system is OPERABLE.

The Completion Time for this Required Action of 5 days will ensure that the CANISTER remains in a safe, analyzed condition.

OR

B.2

Place the affected NAC-UMS SYSTEM in a safe condition.

The Completion Time for this Required Action is 5 days. Requiring B.2 action completion within 5 days will ensure that the NAC-UMS SYSTEM is maintained in a safe condition.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The long-term integrity of the stored fuel is dependent on the ability of the CONCRETE CASK to reject heat from the CANISTER to the environment. Visual observation that all four air inlet and outlet screens are unobstructed and intact ensures that air flow past the CANISTER is occurring and heat transfer is taking place. However, partial blockage of less than two air inlet or outlet screens or the equivalent effective screen area does not result in the heat removal system being unable to provide adequate heat removal. Corrective actions should be taken promptly to remove the obstruction and restore full flow through the affected air inlet and outlet screens. Alternatively, based on the analyses, if the air temperature rise is less than the limits stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long-term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility.

The Frequency of 24 hours is reasonable based on the time necessary for CONCRETE CASK and CANISTER components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of the blockage of the air inlet and outlet screens.

SR 3.1.6.2

The initial confirmation of the OPERABILITY of the CONCRETE CASK is established based on air temperature measurements at the CONCRETE CASK outlets and the ISFSI ambient, and verification that the air temperature rise is less than the limits stated in the SR. Following the initial confirmation, the continued OPERABILITY of the CONCRETE CASK shall be confirmed by one of the verification methods specified in SR 3.1.6.1.

The specified Frequency of once between 5 and 30 days after beginning STORAGE OPERATIONS is reasonable and ensures that the CONCRETE CASK has reached thermal equilibrium and, therefore, the outlet air temperature measurements will reflect expected temperatures under normal operations. Completion of the measurements within 30 days of placement of the CONCRETE CASK into STORAGE OPERATIONS ensures that corrective actions can be taken to establish the OPERABLE status of the CONCRETE CASK within a reasonable period of time.

REFERENCES

1. FSAR Chapter 4 and Chapter 11, Section 11.1.2 and Section 11.2.13.

C 3.2 NAC-UMS® SYSTEM Radiation Protection

C 3.2.1 CANISTER Surface Contamination

BASES

BACKGROUND	A TRANSFER CASK containing an empty CANISTER is immersed in the spent fuel pool in order to load the spent fuel assemblies. The external surfaces of the CANISTER are maintained clean by the application of clean water to the annulus of the TRANSFER CASK. However, there is potential for the surface of the CANISTER to become contaminated with the radioactive material in the spent fuel pool water. Contamination exceeding LCO limits is removed prior to moving the CONCRETE CASK containing the CANISTER to the ISFSI in order to minimize the radioactive contamination to personnel or the environment. This allows the ISFSI to be entered without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.
APPLICABLE SAFETY ANALYSIS	The radiation protection measures implemented at the ISFSI are based on the assumption that the exterior surfaces of the CANISTER are not significantly contaminated. Failure to decontaminate the surfaces of the CANISTER to below the LCO limits could lead to higher-than-projected occupational dose and potential site contamination.
LCO	Removable surface contamination on the exterior surfaces of the CANISTER is limited to 10,000 dpm/100 cm^2 from beta and gamma sources and 100 dpm/100 cm^2 from alpha sources. Only loose contamination is controlled, as fixed contamination will not result from the CANISTER loading process. Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels that could cause significant personnel skin dose.

(continued)

LCO (continued)

LCO 3.2.1 requires removable contamination to be within the specified limits for the exterior surfaces of the CANISTER. Compliance with this LCO may be verified by direct and/or indirect methods. The location and number of CANISTER and TRANSFER CASK surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. The objective is to determine a removable contamination value representative of the entire CANISTER surface area, while implementing sound ALARA practices.

Swipes and measurements of removable surface contamination levels on the interior surfaces of the TRANSFER CASK may be performed to verify the CANISTER LCO limits following transfer of the CANISTER to the CONCRETE CASK. These measurements will provide indirect indications regarding the removable contamination on the exterior surfaces of the CANISTER.

APPLICABILITY

Verification that the exterior surface contamination of the CANISTER is less than the LCO limits is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS and STORAGE OPERATIONS. Measurement of the CANISTER surface contamination is unnecessary during UNLOADING OPERATIONS, as surface contamination would have been measured prior to moving the subject CANISTER to the ISFSI.

(continued)

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER LOADING OPERATION. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the removable surface contamination of the CANISTER that has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the CANISTER and bring the removable surface contamination to within limits. The Completion Time of prior TRANSPORT OPERATIONS is appropriate, given that the time needed to complete the decontamination is indeterminate and surface contamination does not affect the safe storage of the spent fuel assemblies.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This SR verifies (either directly or indirectly) that the removable surface contamination on the exterior surfaces of the CANISTER is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification prior to initiating TRANSPORT OPERATIONS in order to confirm that the CANISTER can be moved to the ISFSI without spreading loose contamination.

REFERENCES

1. FSAR Section 8.1.
2. NRC IE Circular 81-07.

C 3.2 NAC-UMS® SYSTEM Radiation Protection

C 3.2.2 CONCRETE CASK Average Surface Dose Rates

BASES

BACKGROUND	The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10 CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions in accordance with 10 CFR 72.
APPLICABLE SAFETY ANALYSIS	The CONCRETE CASK average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.
LCO	The limits on CONCRETE CASK average surface dose rates are based on the Safety Analysis Report shielding analysis of the NAC-UMS® SYSTEM (Ref. 2). The limits are selected to minimize radiation exposure to the public and to maintain occupational dose ALARA to personnel working in the vicinity of the NAC-UMS® SYSTEM. The LCO specifies sufficient locations for taking dose rate measurements to ensure the dose rates measured are indicative of the effectiveness of the shielding materials.
APPLICABILITY	The CONCRETE CASK average surface dose rates apply during STORAGE OPERATIONS. These limits ensure that the CONCRETE CASK average surface dose rates during STORAGE OPERATIONS are bounded by the shielding safety analyses. Radiation doses during STORAGE OPERATIONS are monitored by the NAC-UMS® SYSTEM user in accordance with the plant-specific radiation protection program as required by 10 CFR 72.212(b)(6) and 10 CFR 20 (Reference 1).
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each loaded CONCRETE CASK. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CONCRETE CASK not meeting the LCO. Subsequent NAC-UMS®

(continued)

ACTIONS (continued) SYSTEMs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the CONCRETE CASK average surface dose rates are not within limits, it could be an indication that a fuel assembly that did not meet the Approved Contents Limits in Section B2.0 of Appendix B was inadvertently loaded into the CANISTER. Administrative verification of the CANISTER fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a misloaded fuel assembly is the cause of the out-of-limit condition. The Completion time is based on the time required to perform such a verification.

A.2

If the CONCRETE CASK average surface dose rates are not within limits and it is determined that the CONCRETE CASK was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the CONCRETE CASK would result in the ISFSI offsite or occupational calculated doses exceeding regulatory limits in 10 CFR Part 72 or 10 CFR Part 20, respectively. If it is determined that the measured average surface dose rates do not result in the regulatory limits being exceeded, STORAGE OPERATIONS may continue.

B.1

If it is verified that the fuel was misloaded, or that the ISFSI offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the CONCRETE CASK average surface dose rates above the LCO limit, the fuel assemblies must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable, based on the time required to transport the CONCRETE CASK, transfer the CANISTER to the TRANSFER CASK, remove the structural lid and vent and drain port cover welds, perform fuel cooldown operations, cut the shield lid weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

This SR ensures that the CONCRETE CASK average surface dose rates are within the LCO limits after transfer of the CANISTER into the CONCRETE CASK and prior to the beginning of STORAGE OPERATIONS. This Frequency is acceptable as corrective actions can be taken before off-site dose limits are compromised. The surface dose rates are measured approximately at the locations indicated on Figure A3-1 of Appendix A of the CoC Number 1015 Technical Specifications, following standard industry practices for determining average surface dose rates for large containers.

REFERENCES

1. 10 CFR Parts 20 and 72.
2. FSAR Sections 5.1 and 8.2.

C 3.3 NAC-UMS® SYSTEM Criticality Control

C 3.3.1 Dissolved Boron Concentration

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into a PWR spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents Limits shown in Table B2-2. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.
APPLICABLE SAFETY ANALYSIS	During loading into, or unloading from, the CANISTER, criticality control of certain PWR fuel requires that the water in the CANISTER contains dissolved boron in a concentration of 1,000 parts per million, or greater. As shown in Table B2-2, spent fuel with the enrichments shown in the "without (w/o) boron" column may be loaded with no assured level of boron in the water in the CANISTER. However, spent fuel with the enrichments shown in the "with boron" column must be loaded or unloaded from the CANISTER when the water in the CANISTER has a boron concentration of 1,000 parts per million or greater. Since boron concentration varies with water temperature, water temperature must be considered in measuring the boron concentration.

(continued)

LCO	The criticality analysis shows that PWR fuel with certain combinations of initial enrichment and fuel content requires credit for the presence of at least 1,000 parts per million of boron in solution in the water in the CANISTER (see Section B3.2.1 for the requirements for assuring soluble boron concentration during loading or unloading). This water must be used to flood the canister cavity during underwater PWR fuel loading or unloading. The boron in the pool water ensures sufficient thermal neutron absorption to preserve criticality control during fuel loading in the basket. Consequently, if boron credit is required for the fuel being loaded or unloaded, the canister must be flooded with water that contains boron in the proper concentration in accordance with the requirements of LCO 3.3.1. Concentration of boron must also be measured and maintained in accordance with LCO 3.3.1. The dissolved boron concentration requirement, and measurement requirement, applies to both the spent fuel pool water and to water in the CANISTER, when pool water is used to fill the CANISTER.
APPLICABILITY	Control of Boron concentration is required during LOADING or UNLOADING OPERATIONS when the CANISTER holds at least one spent fuel assembly that requires dissolved boron for criticality control as described in Table B2-2. This LCO does not apply to spent fuel having an enrichment within the limits specified in the table in the "without (w/o) boron" column.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the required dissolved Boron concentration of the water in the CANISTER is not met, immediate actions must be taken to restore the required dissolved boron concentration. No actions, including continued loading, may be taken that increases system reactivity. <u>AND</u>

(continued)

A.2

The required concentration of dissolved Boron must be restored.

AND

A.3

If the required boron concentration in the water in the CANISTER cannot be established within 24 hours, remove all fuel assemblies that exceed the enrichment limits of Table B2-2 for fuel assemblies taking no boron credit from the CANISTER to bring the system to a safe configuration. The 24 hour period provides adequate time to restore the required boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1

The assurance of an adequate concentration of dissolved boron in the water in the CANISTER must be established once within 4 hours of beginning any LOADING or UNLOADING OPERATION, using two independent measurements of determining boron concentration. During LOADING or UNLOADING OPERATIONS, verification of continued adequate dissolved boron concentration must be performed every 48 hours after the beginning of operations. The 48-hour boron concentration verification is not required when no water is being introduced into the CANISTER cavity. In this situation, no potential exists for the boron in the CANISTER to be diluted, so verification of the boron concentration is not necessary.

REFERENCES

Section B3.2.1 and Table B2-2.

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13.0 QUALITY ASSURANCE

13.1 Introduction

The NAC International (NAC) Quality Assurance (QA) Program is designed and administered to meet all Quality Assurance criteria of 10 CFR 72, Subpart G [1], 10 CFR 50, Appendix B [2], 10 CFR 71, Subpart H [3], and NQA-1 (Basic and Supplemental Requirements) [4]. The program is defined in a QA Program description document that has been reviewed and approved by the Nuclear Regulatory Commission (Approval No. 0018).

The NAC Quality Assurance Program is described in a Quality Assurance Manual. This Quality Assurance Manual, as approved by the company's President and Chief Executive Officer, contains policy as to how NAC intends to comply with the applicable regulatory QA criteria. Detailed implementing quality procedures are used to provide the procedural direction to comply with the policy of the QA Manual.

Employing a graded methodology, as described in USNRC Regulatory Guide 7.10 [5], NAC applies quality controls to items and activities consistent with their safety significance. Table 13.1-1 identifies the NAC Quality Assurance Manual sections, which address the applicable quality criteria.

A synopsis of the NAC Quality Assurance Program is presented in Section 13.2.

Table 13.1-1 Correlation of Regulatory Quality Assurance Criteria to NAC Quality Assurance Program

Regulatory Quality Assurance Criteria*		Corresponding NAC QA Manual Section Number
I.	Organization	1
II.	Quality Assurance Program	2
III.	Design Control	3
IV.	Procurement Document Control	4
V.	Procedures, Instructions, and Drawings	5
VI.	Document Control	6
VII.	Control of Purchased Items and Services	7
VIII.	Identification and Control of Material, Parts and Components	8
IX.	Control of Special Processes	9
X.	Inspection	10
XI.	Test Control	11
XII.	Control of Measuring and Test Equipment	12
XIII.	Handling, Storage and Shipping	13
XIV.	Inspection, Test and Operating Status	14
XV.	Control of Nonconforming Items	15
XVI.	Corrective Action	16
XVII.	Records	17
XVIII.	Audits	18

*The criteria are obtained from 10 CFR 50 Appendix B; 10 CFR 71 Subpart H; and 10 CFR 72 Subpart G.

13.2 NAC Quality Assurance Program Synopsis

Eighteen applicable Quality Assurance criteria are identified in 10 CFR 72, Subpart G; 10 CFR 50, Appendix B; 10 CFR 71, Subpart H; and ASME NQA-1 (Basic and Supplemental Requirements). NAC compliance with each of these criteria is addressed below.

13.2.1 Organization

The President and Chief Executive Officer of NAC has the ultimate authority and responsibility over all organizations and their functions within the corporation. However, the President delegates and empowers qualified personnel with the authority and responsibility over selected key areas, as identified in the NAC Organization Chart, Figure 13.2-1.

The Vice President, Quality, is responsible for definition, development, implementation and administration of the NAC Quality Assurance Program. The Quality Assurance organization is independent from other organizations within NAC and has complete authority to assure adequate and effective program execution, including problem identification, satisfactory corrective action implementation and the authority to stop work, if necessary. The Vice President, Quality, reports directly to the President and Chief Executive Officer of NAC. The Vice President, Quality, has sufficient expertise in the field of quality to direct the quality function and will be capable of qualifying as a lead auditor.

Strategic Business Unit (SBU) Vice Presidents direct operations, utilizing project teams as appropriate for a particular work scope. SBU Vice Presidents are responsible to the President and Chief Executive Officer for the proper implementation of the NAC Quality Assurance Program.

13.2.2 Quality Assurance Program

NAC has established a Quality Assurance Program that meets the requirements of 10 CFR 72, Subpart G, 10 CFR 50 Appendix B, 10 CFR 71, Subpart H, and NQA-1. Employing a grading methodology consistent with U.S. NRC Regulatory Guide 7.10, the Quality Assurance Program provides control over activities affecting quality from the design to fabrication, operation, and maintenance of nuclear products and services for nuclear applications. The Quality Assurance Program is documented in the Quality Assurance Manual and implemented via Quality Procedures. These documents are approved by the Vice President, Quality, and the President and

Chief Executive Officer, as well as the Vice President from each SBU performing activities within the scope of the NAC Quality Assurance Manual.

Personnel assigned responsibilities by the Quality Assurance Program may delegate performance of activities associated with that responsibility to other personnel in their group when those individuals are qualified to perform those activities by virtue of their education, experience and training. Such delegations need not be in writing. The person assigned responsibility by the Quality Assurance Program retains full accountability for the activities.

13.2.3 Design Control

The established Quality Procedures covering design control assure that the design activity is planned, controlled, verified and documented so that applicable regulatory and design basis requirements are correctly translated into specifications, drawings, and procedures with appropriate acceptance criteria for inspection and test delineated.

When computer software is utilized to perform engineering calculations, verifications of the computational accuracy are performed, and error tracking of the software is controlled in accordance with approved Quality Procedures.

Design interface control is established and adequate to assure that the review, approval, release, distribution and revision of design documents involving interfaces are performed by appropriately trained, cognizant design personnel using approved procedures.

Design verification is performed by individuals other than those who performed the original design. These verifications may include design reviews, alternate calculations or qualification tests. Selection of the design verification method is based on regulatory, contractual or design complexity requirements. When qualification testing is selected, the “worst case” scenario will be utilized. The verification may be performed by the originator’s supervisor, provided the supervisor did not specify a singular design approach, rule out certain design considerations, or establish the design inputs used in the design, unless the supervisor is the only individual in the organization competent to perform the verification. When verification is provided by the supervisor, the need shall be so documented in advance and evaluated after performance by internal audit.

Design changes are controlled and require the same review and approvals as the original design.

13.2.4 Procurement Document Control

Procurement documents and their authorized changes are generated, reviewed and approved in accordance with the Quality Procedures. These procedures assure that all purchased material, components, equipment and services adhere to design specification, regulatory and contractual requirements including Quality Assurance Program and documentation requirements.

NAC Quality Assurance personnel review and approve all purchase orders invoking compliance with the Quality Assurance Program for inclusion of quality related requirements in the procurement documents.

13.2.5 Procedures, Instructions, and Drawings

All activities affecting quality are delineated in the Quality Procedures, Specifications, Inspection/Verification Plans or on appropriate drawings. These documents are developed via approved Quality Procedures and include appropriate quantitative and qualitative acceptance criteria. These documents are reviewed and approved by Quality Assurance personnel prior to use.

13.2.6 Document Control

All documents affecting quality, including revisions thereto, are reviewed and approved by authorized personnel, and are issued and controlled in accordance with Quality Procedures by those persons or groups assigned responsibility for the document to be controlled. Transmittal forms, with provisions for receipt acknowledgment, are utilized and controlled document distribution logs are maintained.

All required support documentation for prescribed activities is available at the work location prior to initiation of the work effort.

13.2.7 Control of Purchased Items and Services

Items and services affecting quality are procured from qualified and approved suppliers. These suppliers have been evaluated and selected in accordance with the Quality Procedures based upon their capability to comply with applicable regulatory and contractual requirements.

Objective evidence attesting to the quality of items and services furnished by NAC suppliers is provided with the delivered item or service, and is based on contract requirements and item or service complexity. This vendor documentation requirement is delineated in the procurement documents.

Source inspection, receipt inspection, vendor audits and vendor surveillance are performed as required to assure product quality, documentation integrity, and supplier compliance to the procurement, regulatory and contractual requirements.

13.2.8 Identification and Control of Material, Parts, and Components

Identification is maintained either on the item or in quality records traceable to the item throughout fabrication and construction to prevent the use of incorrect or defective items.

Identification, in accordance with drawings and inspection plans, is verified by Quality Assurance personnel prior to releasing the item for further processing or delivery.

13.2.9 Control of Special Processes

Special processes, such as welding, heat treating and nondestructive testing, are performed in accordance with applicable codes, standards, specifications and contract requirements by qualified personnel. NAC and NAC suppliers' special process procedures and personnel certifications are reviewed and approved by NAC Quality Assurance prior to their use.

13.2.10 Inspection

NAC has an established and documented inspection program that identifies activities affecting quality and verifies their conformance with documented instructions, plans, procedures and drawings.

Inspections are performed by individuals other than those who performed the activity being inspected. Inspection personnel report directly to the Vice President, Quality.

Process monitoring may also be used in conjunction with identified inspections, if beneficial to achieve required quality.

Mandatory inspection hold points are used to assure verification of critical characteristics. Such hold points are delineated in appropriate process control documents.

13.2.11 Test Control

NAC testing requirements are developed and applied in order to demonstrate satisfactory performance of the tested items to design/contract requirements.

The NAC test program is established to assure that preoperational or operational tests are performed in accordance with written test procedures. Test procedures developed in accordance with approved Quality Procedures identify test prerequisites, test equipment and instrumentation and suitable environmental test conditions. Test procedures are reviewed and approved by NAC Quality Assurance personnel.

Test results are documented, evaluated and accepted by qualified personnel as required by the Quality Assurance inspection instructions prepared for the test, as approved by cognizant quality personnel.

13.2.12 Control of Measuring and Testing Equipment

Control of measuring and testing equipment/instrumentation is established to assure that devices used in activities affecting quality are calibrated and properly adjusted at specified time intervals to maintain their accuracy.

Calibrated equipment is identified and traceable to calibration records, which are maintained. Calibration accuracy is traceable to national standards when such standards exist. The basis of calibration shall always be documented.

Whenever measuring and testing equipment is found to be out of calibration, an evaluation shall be made and documented of the validity of inspection or test results performed and of the acceptability of items inspected or tested since the previous calibration.

13.2.13 Handling, Storage and Shipping

Requirements for handling, storage and shipping are documented in specifications and applicable procedures or instructions. These requirements are designed to prevent damage or deterioration to items and materials.

Information pertaining to shelf life, environment, packaging, temperature, cleaning and preservation are also delineated as required.

Quality Assurance Surveillance/Inspection personnel are responsible for verifying that approved handling, storage, and shipping requirements are met.

13.2.14 Inspection, Test and Operating Status

Procedures are established to indicate the means of identifying inspection and test status on the item and/or on records traceable to the item. These procedures assure identification of items that have satisfactorily passed required inspections and/or tests, to preclude inadvertent bypassing of inspection/test.

Inspection, test, and operating status indicators may only be applied or modified by Quality Assurance personnel or with formal Quality Assurance concurrence.

13.2.15 Control of Nonconforming Items

NAC has established and implemented procedures that assure appropriate identification, segregation, documentation, notification and disposition of items that do not conform to specified requirements. These measures prevent inadvertent usage of the item and assure appropriate authorization or approval of the item's disposition.

All nonconformances are reviewed and accepted, rejected, repaired or reworked in accordance with documented approved procedures. If necessary, a Review Board is convened, consisting of engineering, licensing, quality, operations and testing personnel to provide disposition of nonconforming conditions.

NAC procurement documents provide for control, review and approval of nonconformances noted on NAC items, including associated dispositions.

13.2.16 Corrective Action

Conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material/ equipment, and nonconformances are promptly identified, documented and corrected.

Significant conditions adverse to quality will have their cause determined and sufficient corrective action taken to preclude recurrence. These conditions are documented and reported to the Vice President, Quality, who assures awareness by the President and Chief Executive Officer.

13.2.17 Records

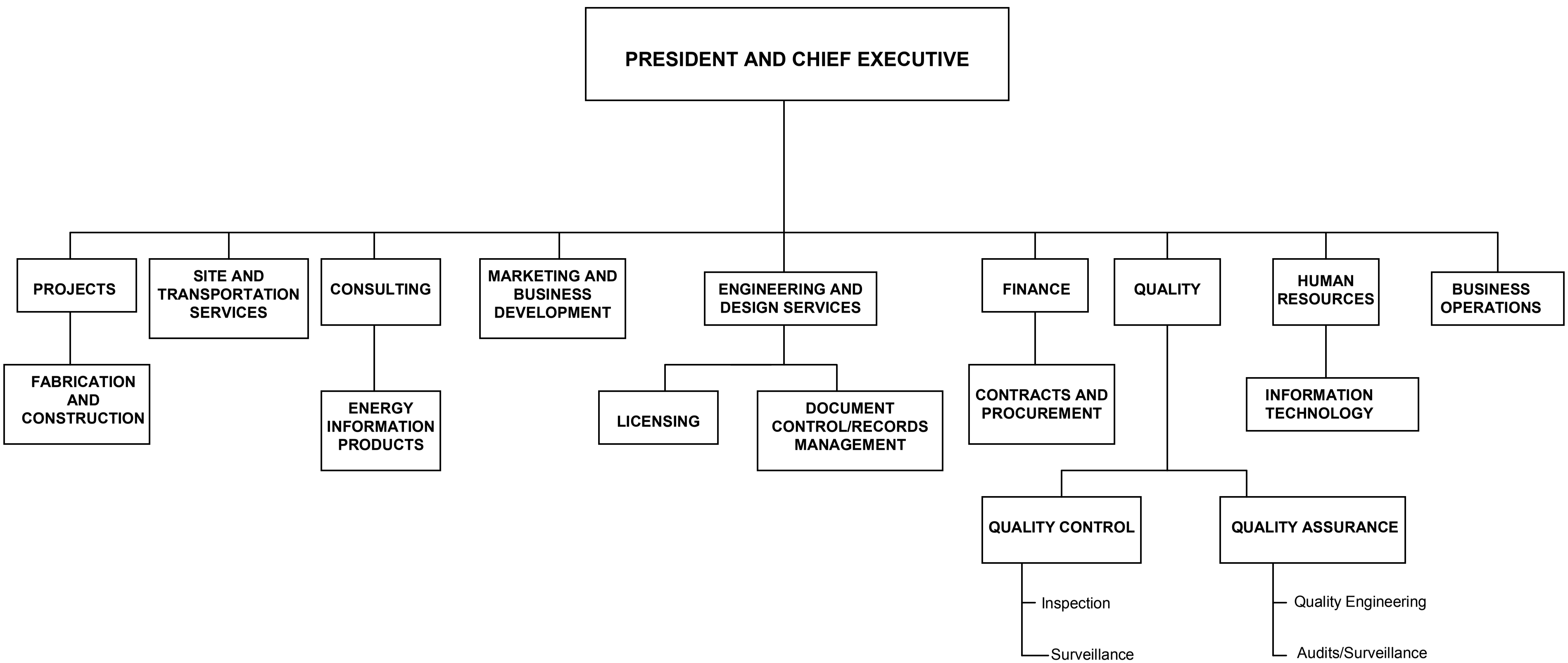
NAC maintains a records system in accordance with approved procedures to assure that documented objective evidence pertaining to quality related activities is identifiable, retrievable and retained to meet regulatory and contract requirements, including retention duration, location and responsibility.

Quality records include, but are not limited to, inspection and test reports, audit reports, quality personnel qualifications, design documents, purchase orders, supplier evaluations, fabrication documents, nonconformance reports, drawings, specifications, etc. Quality Assurance maintains a complete list of records and provides for record storage and disposition to meet regulatory and contractual requirements.

13.2.18 Audits

Approved Quality Procedures provide for a comprehensive system of planned and periodic audits performed by qualified personnel, independent of activities being audited. These audits are performed in accordance with written procedures and are intended to verify program adequacy and its effective implementation and compliance, both internally and at approved-supplier locations. Internal audits are conducted annually, and approved suppliers are audited on a triennial basis, as a minimum.

Figure 13.2-1 NAC Organization Chart



13.3 References

1. U.S. Code of Federal Regulations, "Quality Assurance Requirements," Part 72, Title 10, Subpart G.

2. U.S. Code of Federal Regulations, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Part 50, Title 10, Appendix B.

3. U.S. Code of Federal Regulations, "Quality Assurance," Part 71, Title 10, Subpart H.

4. ASME NQA-1-1994, Part 1, Basic and Supplemental Requirements (as referenced by the ASME Code, including latest accepted addenda), Quality Assurance Program Requirements for Nuclear Facility Applications.

5. U.S. Nuclear Regulatory Commission, "Establishing Quality Assurance Program for Packaging Used in the Transport of Radioactive Material," Regulatory Guide 7.10, Revision 1, June 1986.

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