December 2008

Revision MPC-08A

NAC-MPC

NAC Multi-Purpose Canister

FINAL SAFETY ANALYSIS REPORT

MPC-LACBWR Amendment

Volume 1 of 2

Docket No. 72-1025

NAC INTERNATIONAL

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1.0 GENERAL DESCRIPTION

NAC International (NAC) has designed a Multi-Purpose Canister system (NAC-MPC) for the long-term storage of spent nuclear fuel. The NAC-MPC system consists of a 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 NAC Storage Transport Cask (NAC-STC) and to be compatible with the U.S. Department of Energy MPC Design Procurement Specification 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 storage cask also provides protection during storage for the transportable storage canister under adverse environmental conditions. The NAC-MPC 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 NAC-STC for transport.

This Safety Analysis Report demonstrates the ability of the NAC-MPC system to satisfy the Nuclear Regulatory Commission (NRC) requirements for the storage of spent fuel, as prescribed by 10 CFR 72.

This chapter provides a general description of the major components of the system and a description of the system operation. The terminology used throughout this report is summarized in Table 1-1. Table 1.5-1 provides a compliance matrix to the regulatory requirements and acceptance criteria specified in NUREG-1536. This matrix describes how the NAC-MPC Safety Analysis Report complies with each requirement and criterion listed in NUREG-1536. Table B3-1 of the Certificate of Compliance provides a list of the alternatives from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) that are applicable for the canister.

Appendix A of this chapter contains the general description of the MPC-LACBWR system.

Table 1-1 Terminology

NAC-STC Cask	The licensed spent-fuel transport cask consisting of a spent fuel storable transport cask body with dual closure lids and energy-absorbing impact limiters (Certificate of Compliance No. 71-9235).
Confinement System	The components of the transportable storage canister intended to retain the radioactive material during storage.
Transportable Storage Canister (Canister)	The stainless steel cylindrical shell, bottom end plate, shield lid, and structural lid that holds the spent fuel in the canister basket.
Yankee MPC Contents	Up to 36 intact Yankee Class pressurized-water reactor (PWR) spent fuel assemblies, Reconfigured Fuel Assemblies and Recaged Fuel Assemblies, including up to four fuel assemblies loaded in Damaged Fuel Cans. The maximum total contents weight in the Transportable Storage Canister is 30,600 pounds, not including the weight of the Damaged Fuel Cans, which are defined to be components of the fuel basket.
CY- MPC Contents	Up to 26 Connecticut Yankee PWR spent fuel assemblies (including fuel inserts), CY-MPC Reconfigured Fuel Assemblies and CY-MPC Damaged Fuel Cans containing Connecticut Yankee intact or damaged fuel to a maximum total contents weight of 35,100 pounds.
Yankee Class Spent Fuel	Fuel that includes United Nuclear Type A and Type B, Combustion Engineering Type A and Type B, Exxon-ANF Type A and Type B, and Westinghouse Type A and Type B spent fuel assemblies.
CY-MPC Reconfigured Fuel Assembly	A stainless steel container, having external dimensions that are slightly larger than a standard Connecticut Yankee fuel assembly, that ensures criticality control geometry and which permits gaseous and liquid media to escape while minimizing dispersal of gross particulate. It may contain a maximum of 100 intact fuel rods or damaged fuel rods, or fuel debris from any Connecticut Yankee spent fuel assembly.
Connecticut Yankee Spent Fuel	15 x 15 PWR fuel assemblies manufactured by Westinghouse, Gulf Nuclear, Gulf General Atomic, NUMEC and Babcock and Wilcox.

Table 1-1 Terminology (continued)

Recaged Fuel Assembly	A Yankee Class Combustion Engineering fuel assembly lattice (skeleton) holding United Nuclear fuel rods with no empty fuel rod positions.
Retainer	A stainless steel component used to secure removable fuel rods in a United Nuclear assembly.
Connecticut Yankee Fuel Inserts	Reactor Control Cluster Assemblies, flow mixers or stainless steel rods that may be stored with the Connecticut Yankee spent fuel.
Intact Fuel Assembly	A fuel assembly without known or suspected cladding defects greater than pinhole leaks or a hairline cracks. Connecticut Yankee fuel assemblies with missing fuel rods, or with missing fuel rods replaced with solid filler rods, or with structural damage, are considered INTACT FUEL ASSEMBLIES, provided that they have no DAMAGED FUEL RODS. Yankee Class fuel assemblies with missing fuel rods replaced with Zircaloy or stainless steel rods, or with structural damage, are considered intact fuel assemblies provided that they have no damaged fuel rods.
Intact Fuel Rod	A fuel rod without known or suspected cladding defects greater than a pinhole leak or a hairline crack.
Yankee Damaged Fuel Assembly	A fuel assembly containing up to 20 missing or damaged fuel rods with known or suspected cladding defects greater than a hairline crack or a pinhole leak.
Connecticut Yankee Damaged Fuel Assembly	A fuel assembly with damaged fuel rods, or that cannot be handled by normal means, or both.
Damaged Fuel Rod	A fuel rod with known or suspected cladding defects greater than a hairline crack or a pinhole leak.

Table 1-1 Terminology (continued)

Damaged Fuel Can	A stainless steel container that is similar to an enlarged fuel tube that confines a Yankee Class Intact Fuel Assembly, Damaged Fuel Assembly, Recaged Fuel Assembly or a Reconfigured Fuel Assembly. A damaged fuel can is closed on its bottom end by a stainless steel bottom plate having screened openings and on its top end by a stainless steel lid that also has screened openings. The screened openings allow gaseous and liquid media to escape, but minimizes the dispersal of gross particulate. Use of the Damaged Fuel Can requires that four cans be used in the canister in conjunction with the use of a special shield lid machined to accept the cans.
Fuel Debris	Fuel in the form of particles, loose pellets and fragmented rods or assemblies.
Lattice	A fuel assembly structure that is used to hold up to 204 Intact Fuel Rods or Damaged Fuel Rods from other fuel assemblies. A Lattice is sometimes called a fuel skeleton, cage or structural cage. It is built from the same components as a standard fuel assembly, but some of those components may be modified slightly, such as relaxed grids, to accommodate the distortion that may be present in a Damaged Fuel Rod. The outside dimensions are identical to a standard fuel assembly.
Failed Rod Storage Canister	A handling container for moving up to 60 individual intact or damaged fuel rods in stainless steel tubes into a CY-MPC Damaged Fuel Can. The steel tubes are held in place by regularly spaced plates welded in an open stainless steel frame. The failed rod storage canister, which is closed at the top end by a bolted closure and at the bottom by a welded plate to capture the fuel rods in the tubes, must be loaded in a CY-MPC Damaged Fuel Can.

Table 1-1 Terminology (continued)

Structural Damage	Damage to the fuel assembly that does not prevent handling the fuel assembly by normal means. Structural damage is defined as partially torn, abraded, dented or bent grid straps, end fittings or guide tubes. The damaged grid straps or end fittings must continue to provide support to the fuel rods, as designed, and may not be completely torn or missing. Guide tubes cannot be ruptured and must be continuous between the upper and lower end fittings. Fuel assemblies with structural damage are considered to be intact fuel assemblies provided that they do not have failed or damaged fuel rods.
Canister Basket	The structure placed in the transportable storage canister to support the fuel assemblies (fuel basket).
-Support Disk	A circular stainless steel plate with square holes machined in a symmetrical pattern that provides the primary lateral load-bearing component of the canister basket.
-Heat Transfer Disk	A circular aluminum plate with 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 that may have BORAL neutron poison material on its exterior surfaces.
-Tie Rod	A stainless steel rod used to align the support disks and heat transfer disks in the fuel basket structure.
-Split Spacer	Spacers installed on the tie rod between the support disks to properly position, and provide axial support for, the support disks and the heat transfer disks.
Shield Lid	The primary confinement boundary for the canister. It is located directly above the canister basket and is provided in two configurations. The Damaged Fuel Can configuration may not be used interchangeably with the intact fuel configuration.
-Drain Port	A penetration located in the shield lid to permit draining of the canister cavity.

Table 1-1 Terminology (continued)

-Vent Port	A penetration located in the shield lid to aid in draining and backfilling the canister cavity.
-Port Cover	The stainless steel covers that close the vent and drain ports, which are welded in place following draining, drying, and backfilling operations.
-Quick Disconnect	The quick-disconnect valved nipple used in the vent and drain ports to facilitate operations.
Structural Lid	The secondary confinement boundary for the canister. The structural lid provides the lifting point for the loaded canister.
Vertical Concrete Cask (Concrete Cask) (Storage Cask)	A reinforced concrete cylinder closed at the top end by a shield plug and lid that holds the transportable storage canister during storage. The vertical concrete cask is formed around a steel inner liner and base.
- Shield Plug	A thick carbon steel plug installed in the top end of the storage cask to reduce skyshine radiation. The shield plug also contains a neutron shield material.
- Lid	A thick carbon steel bolted closure for the storage 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 storage 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 inlet air vents, the storage cask jacking points, and the pedestal that supports the canister inside the storage cask.
Transfer Cask	A shielded lifting device for the empty and loaded canister. It is used for the vertical transfer of the canister between work stations and the storage cask or the transport cask. The transfer cask incorporates bottom doors that permit the vertical loading of the storage and transport casks.
- Lifting Trunnions	Carbon steel trunnions used to lift and move the transfer cask.

Table 1-1 Terminology (continued)

Adapter Plate	A carbon steel plate that attaches to the top of the transport or storage cask to facilitate the installation and alignment of the transfer cask. It also provides the operating mechanism for the transfer cask bottom doors.
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 of a material divided by its actual (calculated) stress.
Yankee-MPC Reconfigured Fuel Assembly	A stainless steel canister having the same external dimensions as a standard Yankee Class fuel assembly, that ensures criticality control geometry and which permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. It may contain a maximum of 64 intact fuel rods or damaged fuel rods, or fuel debris from any type of Yankee Class spent fuel assembly.
MPC-LACBWR	MPC-LACBWR is a NAC-MPC system having a fuel basket designed to accommodate Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) spent fuel. The MPC-LACBWR meets the NAC-MPC system requirements.

1.1 Introduction

The NAC-MPC system is a transport compatible dry storage system that uses a vertical concrete storage cask and a stainless steel transportable storage canister (canister) with a welded closure to safely store irradiated nuclear fuel (spent fuel). The canister is stored in the central cavity of the concrete cask and is compatible with the NAC-STC transport cask for future off-site shipment. The concrete storage 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 NAC-MPC is designed and analyzed for a minimum 50-year life.

The principal components of the NAC-MPC system are the canister, the vertical concrete cask and the transfer cask. The loaded canister is moved to and from the concrete cask with 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 NAC-MPC system and shows the transfer cask positioned on the top of the concrete cask.

The fuel is initially loaded into a canister containing a fuel basket. Figure 1.1-2 depicts the canister and the spent fuel basket. The design characteristics of the NAC-MPC system are shown in Table 1.1-1.

The system design and analyses were performed in accordance with Title 10, Code of Federal Regulations, Part 72 (10 CFR 72), ANSI/ANS 57.9-1984 and the applicable sections of the ASME Boiler and Pressure Vessel Code and the American Concrete Institute Code.

The NAC-MPC is provided in three configurations. The first is designed to store up to 36 intact Yankee Class spent fuel and reconfigured fuel assemblies and is referred to as the Yankee-MPC. The second is designed to store up to 26 Connecticut Yankee fuel assemblies, reconfigured fuel assemblies and damaged fuel in CY-MPC damaged fuel cans, and is referred to as the CY-MPC.

The third configuration, referred to as MPC-LACBWR, is designed to store up to 68 Dairyland Power Cooperative (DPC) La Crosse Boiling Water Reactor (LACBWR) spent fuel assemblies with up to 32 damaged fuel cans. The MPC-LACBWR system is described in Appendix 1.A.

Yankee Class fuel includes United Nuclear, Combustion Engineering, Exxon-ANF, and Westinghouse Type A and Type B fuel designs. The Type A and Type B fuel designs are complementary configurations that accommodate the use of a cruciform control blade in reactor operations. The fuel specifications that serve as the design basis for the Yankee-MPC are presented in Sections 1.3.1 and 2.1.1.

Connecticut Yankee spent fuel includes 15 x 15 PWR fuel assemblies having a square cross-section. The fuel specifications that serve as the design basis for the CY-MPC are presented in Sections 1.3.2 and 2.1.2. The Connecticut Yankee fuel consists of fuel assemblies manufactured by Westinghouse, Gulf Nuclear/Gulf General Atomic, NUMEC and by Babcock & Wilcox.

The MPC-LACBWR spent fuel is described in Section 1.A.3 and Table 1.A.3-1.

1.2 The NAC-MPC System

The NAC-MPC system is provided in three configurations, the Yankee-MPC, the CY-MPC, and the MPC-LACBWR, which have similar components and operating features, but different physical dimensions, weights and storage capacities. All configurations provide long-term storage and subsequent transport of the stored spent fuel using the certified NAC-STC. 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 NAC-MPC System Components

The NAC-MPC system consists of three principal components:

- Transportable storage canister (canister),
- Vertical concrete cask, and
- Transfer cask.

Ancillary equipment needed to use the NAC-MPC system is:

- Automated or manual welding equipment;
- An air pallet or hydraulic roller skid (used to move the storage cask on and off the heavy haul transfer trailer and to position the storage cask on the storage pad);
- Suction pump, vacuum drying, helium backfill and leak detection equipment;
- A heavy haul trailer or cask transporter (for storage cask transport to the storage pad);
- 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, air and water), common tools and fittings, and miscellaneous hardware.

The transportable storage canister is designed to be transported in the NAC-STC (Certificate of Compliance No. 71-9235). The transport load conditions produce higher stresses in the canister than would be produced by the storage load conditions alone. Consequently, the canister design is conservative with respect to storage conditions. The evaluation of the canister for transport conditions is found in the NAC-STC Safety Analysis Report, Docket No. 71-9235.

1.2.1.1 Transportable Storage Canister and Baskets

The Transportable Storage Canister (canister) contains a basket that is designed to accommodate either Yankee Class or Connecticut Yankee (CY) spent fuel. The Yankee-MPC basket holds up to 36 intact Yankee Class spent fuel assemblies and reconfigured fuel assemblies (RFAs) up to a total contents weight of 30,600 pounds, including up to four fuel assemblies or RFAs loaded in damaged fuel cans. The CY-MPC basket holds up to 26 spent fuel assemblies and RFAs up to a total contents weight of 35,100 pounds, including up to four fuel assemblies or RFAs loaded in damaged fuel cans.

The canister assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The cylindrical shell, plus the bottom plate and lids, constitutes the confinement boundaries. The fuel basket is based on the directly loaded fuel basket design used in the certified NAC-STC. This basket features the NAC-patented poison tubes and stacked disk design with heat transfer disks. The basket was analyzed using the ANSYS computer code to demonstrate that it can withstand the horizontal drop loads without deforming in a way that damages or constrains a fuel assembly. This tube and disk design has been accepted and approved by the NRC, pursuant to 10 CFR 71 and 10 CFR 72. Table 1.2-1 summarizes the major physical design parameters of the canister configurations.

The fuel basket design is a right-circular cylinder configuration with either 24, 26, or 36 fuel tubes laterally supported by a series of support disks, which are retained by spacers on radially located tie rods. Connecticut Yankee fuel may be stored in either a 24- or 26-assembly basket configuration, while Yankee Class fuel may be stored in the 36-assembly configuration. Eight tie rods are used in the Yankee Class basket design. Six tie rods are used in the CY-MPC basket. The support disks are stainless steel (17-4 PH) with holes for the poison fuel tubes or damaged fuel cans. The basket top and bottom weldments are fabricated from Type 304 stainless steel. The tie rods and spacer sleeves are also fabricated from Type 304 stainless steel. The fuel assemblies are contained in fuel tubes. The CY-MPC fuel tubes are fabricated from Type 304 stainless steel with encased BORAL sheets on all four outside surfaces of the fuel tube. The BORAL provides criticality control in the basket.

1.3 NAC-MPC Storage System Contents

The NAC-MPC is provided in three configurations designated the Yankee-MPC, the CY-MPC and the MPC-LACBWR. The Yankee-MPC is designed to hold up to 36 Yankee Class fuel assemblies as described in Section 1.3.1. The CY-MPC is designed to hold up to 26 Connecticut Yankee fuel assemblies as described in Section 1.3.2.

The MPC-LACBWR is designed to hold up to 68 Dairyland Power Cooperative La Crosse Boiling Water Reactor spent fuel assemblies with up to 32 LACBWR damaged fuel cans. The MPC-LACBWR contents are described in Section 1.A.3 and Table 1.A.3-1.

1.3.1 Yankee-MPC Storage System Contents

The Yankee-MPC Storage System is designed to accommodate intact fuel assemblies and individual intact and damaged fuel rods used during the operation of the Yankee facility.

1.3.1.1 Yankee Class Spent Fuel

The Yankee-MPC is designed to store up to 36 Yankee Class spent fuel assemblies. The Yankee Class fuel consists of fuel assemblies manufactured by Westinghouse, United Nuclear, Exxon, and Combustion Engineering. The assemblies vary in initial enrichment from 3.5 to 4.94 wt % ^{235}U. Each manufacturer's types of assemblies include two configurations identified as Type A and Type B. The arrangement of fuel rods differs in each type to allow the fuel assembly to accept a segment of a control blade used for criticality control. The characteristics of the Yankee Class spent fuels are presented in Table 1.3-1. Unenriched fuel assemblies are not evaluated and are not included as proposed contents.

1.3.1.2 Yankee Class (Yankee-MPC) Reconfigured Fuel Assembly

A Yankee-MPC canister may contain one or more Yankee-MPC Reconfigured Fuel Assemblies designed to confine Yankee Class intact or damaged spent fuel rods and fuel debris. The Yankee-MPC reconfigured fuel assembly is designed to confine its contents during all storage and transport conditions. The assembly can accept up to 64 full length spent fuel rods in an eight by eight array of tubes. A sketch of the assembly is provided in Figure 1.3-1.

The Yankee-MPC reconfigured fuel assembly consists of a shell (square tube with end fittings), a basket assembly and 64 fuel tubes. The external dimensions of the shell are the same as those of a standard Yankee Class spent fuel assembly and all materials are stainless steel. It is designed such that it can be handled in the same manner as a standard Yankee Class spent fuel assembly.

The spent fuel is confined in the fuel tubes. The tubes are supported by a basket assembly within the shell and have end plugs with drilled holes to permit draining drying and inerting with helium. The shell has holes in the top and bottom fittings to permit draining, drying and inerting of the assembly. The total number of full length rods that can be placed in the reconfigured fuel assembly is less than the number that are in the Yankee Class fuel assemblies (maximum of 64 versus 236 rods of the most reactive fuel). Consequently, the effects of a Yankee-MPC reconfigured fuel assembly placed in a canister (e.g., criticality, thermal output, source term) are significantly less than the effects of a design basis Yankee Class spent fuel assembly.

1.3.1.3 Yankee – MPC Damaged Fuel Cans

The NAC-MPC also has a damaged fuel configuration to provide storage for Yankee fuel classified as damaged. This configuration provides four stainless steel damaged fuel cans located in the four corner positions of the basket, as shown in Figure 2.1-1. A damaged fuel can is similar to an enlarged fuel tube, except that the can is closed on the bottom end with a screened plate that is welded into place and closed on the top by a screened lid, which is held in place by the damaged fuel shield lid once the canister is closed.

The square holes in the four corner positions of the basket top and bottom weldments are enlarged to allow the damaged fuel can to be removed or inserted after the basket is assembled. The damaged fuel can is captured between the canister bottom plate and the damaged fuel shield lid to preclude vertical movement and to hold the can lid in place. The shield lid has four machined recesses on the underside to mate with the damaged fuel can lids.

The damaged fuel can may also contain an intact fuel assembly to allow the efficient use of the NAC-MPC system.

1.3.2 CY-MPC Storage System Contents

The CY-MPC Storage System is designed to accommodate intact fuel assemblies, damaged fuel assemblies, individual intact and damaged fuel rods, and non-fuel assembly hardware used during the operation of the Connecticut Yankee facility.The Connecticut Yankee fuel inventory includes a number of fuel assemblies with one or more damaged fuel rods, and individual intact and damaged fuel rods that are not a part of a fuel assembly. Damaged fuel assemblies, lattices and the failed rod storage canister will be placed into a damaged fuel can prior to storage, while individual fuel rods will be placed into a CY-MPC reconfigured fuel assembly for storage in the CY-MPC system. The criteria for storage of fuel assemblies in damaged fuel cans is described in Section 2.1.2.3. The criteria for storage of individual fuel rods in CY-MPC reconfigured fuel assemblies is described in Section 2.1.2.2.

1.3.2.1 Spent Fuel Assemblies

The CY-MPC is designed to store up to 26 Connecticut Yankee spent fuel assemblies. The Connecticut Yankee fuel consists of 15x15 PWR fuel assemblies manufactured by Westinghouse, Gulf Nuclear/Gulf General Atomic, NUMEC and by Babcock & Wilcox. Approximately 10% of the Connecticut Yankee spent fuel inventory is Zircaloy-clad. The remaining assemblies are stainless steel clad. The Zircaloy clad assemblies vary in initial enrichment from 2.95 to 4.61 wt % ^{235}U and have a maximum burnup of 43,000 MWD/MTU. The stainless steel clad assemblies vary in initial enrichment from 3.0 to 4.03 wt % ^{235}U and have a maximum burnup of 38,000 MWD/MTU.

The characteristics of the Connecticut Yankee spent fuels are presented in Tables 1.3-2 and 2.1-3. The Zircaloy clad Westinghouse Vantage 5H fuel assembly enriched to 4.61 wt % ^{235}U is the most reactive fuel and is the design basis fuel for the criticality evaluation. The shielding evaluation uses the Westinghouse stainless steel clad fuel assembly with a minimum enrichment of 3.65 wt % ^{235}U, a fuel mass of 432 kg U and a burnup of 38,000 MWD/MTU as the design basis stainless steel clad fuel. This fuel assembly is also the design basis for the structural and thermal evaluations, using an assembly weight of 1,350 lbs and decay heat up to 840 watts. The shielding design basis Zircaloy clad fuel uses a minimum enrichment of 3.59 wt % ^{235}U, a fuel mass of 395 kg U and a burnup of 43,000 MWD/MTU. These parameters are selected to represent the bounding mix of those that could occur in loading Zircaloy clad fuel but they do not match those of any single Connecticut Yankee fuel assembly.

Unenriched fuel assemblies are not evaluated and are not included as a proposed contents. Zircaloy clad fuel assemblies with enrichments greater than 3.93 wt % ^{235}U may only be placed in the 24-assembly basket configuration, while the remaining fuel may be placed into the 26-assembly baskets. Connecticut Yankee fuel assemblies may have a Flow Mixer/Thimble Plug assembly or a Reactor Control Cluster Assembly component inserted as described in Section 1.3.2.4.

To achieve greater flexibility in loading the Connecticut Yankee fuel assemblies, a set of parameters have been established to restrict loading of certain fuel assemblies into particular locations in the fuel basket, or into a particular basket configuration (24 or 26 assembly capacity) based on enrichment, burnup, cooling time, and cladding type. A description of the preferential fuel loading requirements is presented in Section 2.1.2.1.

Solid stainless steel rods, approximately 21 inches long, may be inserted into Connecticut Yankee intact and damaged fuel assembly Reactor Control Cluster Assemblies (RCCA) guide tubes not containing a RCCA. The stainless steel rods are intended to displace the water from the lower end of the RCCA guide tubes during draining of the canister. The height of the first drainage hole of the RCCA guide tube is over 21 inches from the bottom of the tube. The 20 RCCA guide tubes per assembly could retain significant amounts of water. This water would be required to be removed by vacuum drying. The small diameter of the RCCA guide tubes, the height of water in the tube, and the minimal decay heat in the location of the tubes would make removal of the water by the vacuum drying process extremely difficult. The rods will be installed in the assemblies prior to loading the assemblies into the canister.

1.3.2.2 Connecticut Yankee (CY-MPC) Reconfigured Fuel Assembly

The CY-MPC reconfigured fuel assembly consists of a stainless steel 10 x 10 array of tubes attached to upper and lower end fittings that are similar to those used on standard fuel assemblies. This allows handling using standard fuel assembly handling equipment. The tubes are designed to hold individual fuel rods that have been removed from fuel assemblies. The diameter of the tubes is sized to allow the insertion of individual damaged or bowed fuel rods.

The CY-MPC reconfigured fuel assembly is fabricated from stainless steel and has top and bottom closures that allow the release of gaseous products and liquids but minimizes the dispersal of particulates. The cross-section dimension restricts loading to one of the four corner "oversized fuel" basket positions.

A sketch of the CY-MPC reconfigured fuel assembly is provided in Figure 1.3-2. The major physical design parameters are presented in Table 1.3-3. The design and fabrication specification summary is provided in Table 1.3-6. A discussion of the preferential loading of the CY-MPC reconfigured fuel assembly is presented in Section 2.1.2.2.

Appendix 1.A GENERAL DESCRIPTION – MPC-LACBWR MPC STORAGE SYSTEM FOR DAIRYLAND POWER COOPERATIVE LA CROSSE BOILING WATER REACTOR

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1.A GENERAL DESCRIPTION OF THE MPC-LACBWR STORAGE SYSTEM

NAC International (NAC) is amending the design of the Multi-Purpose Canister system (NAC-MPC) for the long-term storage of Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) spent nuclear fuel. The MPC-LACBWR system consists of a 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 NAC Storage Transport Cask (NAC-STC).

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 storage cask also provides protection during storage for the transportable storage canister under adverse environmental conditions. Employing the guidance in ISG-18, Rev 1, the MPC-LACBWR closure design uses a single lid with a redundant weld closure 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 NAC-STC for transport.

This Safety Analysis Report demonstrates the ability of the NAC-MPC system to satisfy the Nuclear Regulatory Commission (NRC) requirements for the storage of spent fuel, as prescribed by 10 CFR 72.

This Appendix provides a general description of the major components of the MPC-LACBWR system and a description of the system operation. The terminology used throughout this Appendix is summarized in Table 1.A-1. Table 1.A.5-1 provides a compliance matrix to the regulatory requirements and acceptance criteria specified in NUREG-1536. This matrix describes how the MPC-LACBWR Safety Analysis Report complies with each requirement and criterion listed in NUREG-1536. Table B.3-1 of Appendix 12.B of the Technical Specifications provides a list of the alternatives from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) that are applicable for the canister.

Table 1.A-1 Terminology

NAC-STC Cask	The licensed spent-fuel transport cask consisting of a spent fuel storable transport cask body with dual closure lids and energy-absorbing impact limiters (Certificate of Compliance No. 71-9235).
Confinement System	The components of the transportable storage canister intended to retain the radioactive material during storage.
Transportable Storage Canister (Canister)	The stainless steel cylindrical shell, bottom end plate, closure lid, closure ring and redundant port covers that holds the spent fuel in the canister basket.
MPC-LACBWR Contents	Up to 68 Dairyland Power Cooperative La Crosse BWR spent fuel assemblies, including up to 32 Damaged Fuel Cans.
LACBWR Damaged Fuel Assembly	Spent nuclear fuel (SNF) that cannot fulfill its fuel-specific or system-related function. SNF is classified as a LACBWR damaged fuel assembly 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 a solid dummy rod that displaces a volume equal to, or greater than, the original fuel rod. 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). 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) are classified as undamaged. 5. The SNF 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).

LACBWR Undamaged Fuel Assembly	A spent nuclear fuel assembly that can meet all fuel-specific and system-related functions. A LACBWR undamaged fuel assembly is spent nuclear fuel that is not a LACBWR damaged fuel assembly, as defined herein, and does not contain assembly structural defects that adversely affect radiological and/or criticality safety. As such, a LACBWR undamaged fuel assembly may contain breached spent fuel rods (i.e., rods with minor defects up to hairline cracks or pinholes), but cannot contain grossly breached fuel rods.
LACBWR Damaged Fuel Can	A stainless steel container that is similar to an enlarged fuel tube that confines a LACBWR Fuel Assembly, Damaged Fuel Assembly, or debris. A damaged fuel can is closed on its bottom end by a stainless steel bottom plate having screened openings and on its top end by a stainless steel lid that also has screened openings. The screened openings allow gaseous and liquid media to escape, but minimizes the dispersal of gross particulate.
Fuel Debris	Fuel in the form of particles, loose pellets and fragmented rods or assemblies.
Canister Basket	The structure placed in the transportable storage canister to support the fuel assemblies (fuel basket).
-Support Disk	A circular stainless steel plate with square holes machined in a symmetrical pattern that provides the primary lateral load-bearing component of the canister basket.
-Heat Transfer Disk	A circular aluminum plate with 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 that may have BORAL neutron poison material on its exterior surfaces. Certain MPC-LACBWR fuel tubes may have an aluminum sheet in place of a BORAL sheet on one side.
-Tie Rod	A stainless steel rod used to align the support disks and heat transfer disks in the fuel basket structure.
-Split Spacer	Spacers installed on the tie rod between the support disks to properly position, and provide axial support for, the support disks and the heat transfer disks.
Closure Lid	The primary confinement boundary for the canister. It is located directly above the canister basket and provides the lifting point for the loaded canister.

-Drain Port	A penetration located in the closure lid to permit draining of the canister cavity.
-Vent Port	A penetration located in the closure lid to aid in draining and backfilling the canister cavity.
-Port Cover	The stainless steel dual covers that close the vent and drain ports, which are welded in place following draining, drying, and backfilling operations and provide both a confinement boundary and redundant closure .
-Quick Disconnect	The quick-disconnect valved nipple used in the vent and drain ports to facilitate operations.
Closure Ring	The steel bar that is positioned and welded above the closure lid to canister shell weld that provides redundant confinement.
Vertical Concrete Cask (Concrete Cask) (Storage Cask)	A reinforced concrete cylinder closed at the top end by a shield plug and lid that holds the transportable storage canister during storage. The vertical concrete cask is formed around a steel inner liner and base.
- Lid	A thick carbon steel bolted closure with incased concrete for shielding for the storage cask. The lid precludes access to the canister and provides radiation shielding to reduce skyshine.
- Liner	A thick carbon steel shell that forms the annulus of the concrete storage 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 inlet air vents` and the pedestal that supports the canister inside the storage cask.
Transfer Cask	A shielded lifting device for the empty and loaded canister. It is used for the vertical transfer of the canister between work stations and the storage cask or the transport cask. The transfer cask incorporates bottom doors that permit the vertical loading of the storage and transport casks.
- Lifting Trunnions	Carbon steel trunnions used to lift and move the transfer cask.
Adapter Plate	A carbon steel plate that attaches to the top of the transport or storage cask to facilitate the installation and alignment of the transfer cask. It also provides the operating mechanism for the transfer cask bottom doors.
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 of a material divided by its actual (calculated) stress.

MPC-LACBWR	MPC-LACBWR is a NAC-MPC system having a fuel basket designed to accommodate Dairyland Power Coöperative La Crosse BWR (LACBWR) reactor spent fuel. The MPC-LACBWR meets the NAC-MPC system requirements.

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1.A.1 Introduction

The MPC-LACBWR system is a transport compatible dry storage system that uses a vertical concrete storage cask and a stainless steel transportable storage canister (canister) with a welded closure to safely store irradiated nuclear fuel (spent fuel). The canister is stored in the central cavity of the concrete cask and is compatible with the NAC-STC transport cask for future off-site shipment. The concrete storage 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 MPC-LACBWR system is designed and analyzed for a minimum 50-year life.

The principal components of the MPC-LACBWR system are the canister, the vertical concrete cask and the transfer cask. The loaded canister is moved to and from the concrete cask with 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.A.1-1 depicts the major components of the MPC-LACBWR system and shows the transfer cask positioned on the top of the concrete cask.

The fuel is initially loaded into a canister containing a fuel basket. Figure 1.A.1-2 depicts the canister and the spent fuel basket. The design characteristics of the MPC-LACBWR system are shown in Table 1.A.1-1.

The system design and analyses were performed in accordance with Title 10, Code of Federal Regulations, Part 72 (10 CFR 72), ANSI/ANS 57.9-1992 and the applicable sections of the ASME Boiler and Pressure Vessel Code, 1995 Edition with 1995 Addenda, and the American Concrete Institute Code, edition as referenced in this application.

The MPC-LACBWR is designed to store up to 68 LACBWR spent fuel assemblies including up to 32 damaged fuel cans.

Figure 1.A.1-1 Major Components of the MPC-LACBWR System

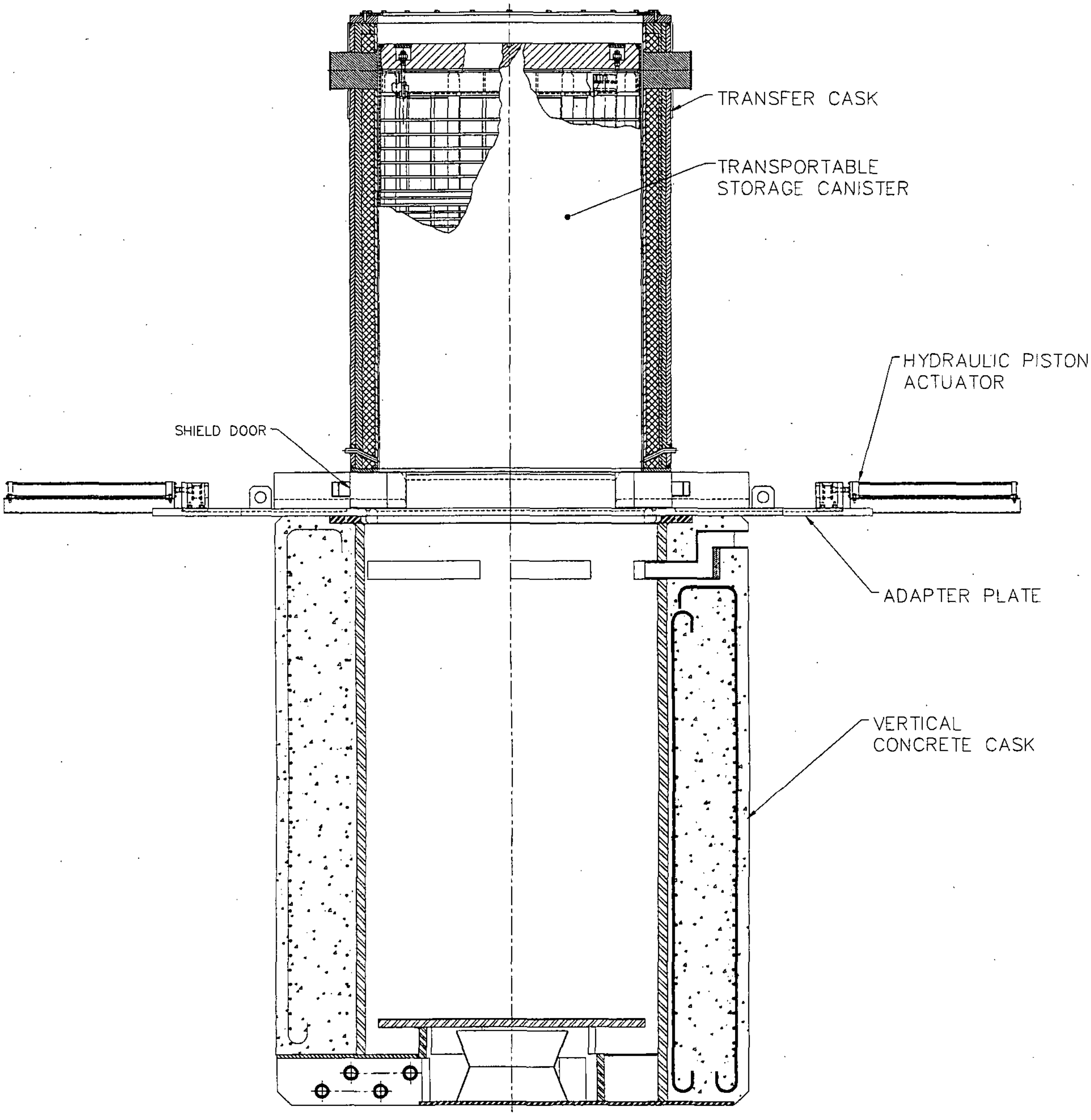

Figure 1.A.1-2 MPC-LACBWR Transportable Storage Canister Showing the Spent Fuel Basket

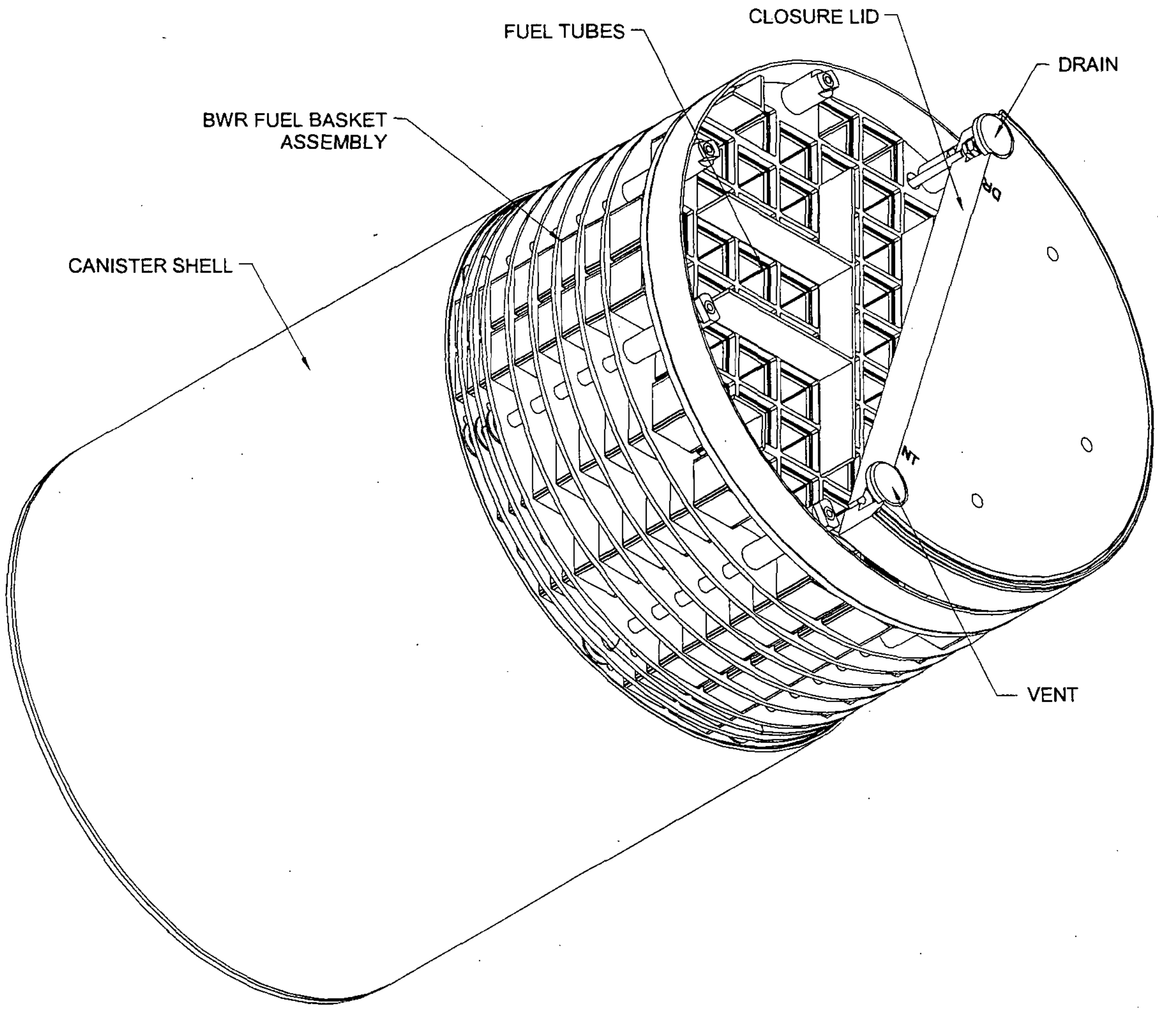

Table 1.A.1-1 Design Characteristics of the MPC-LACBWR System

Design Characteristic	Dimension[1]	Material
MPC-LACBWR Canister		
- Shell	1/2 thick Plate	Type 304L Stainless Steel
- Bottom	1.25 thick Plate	Type 304/304L Stainless Steel
- Closure Lid	7.0 thick Plate	Type 304/304L Stainless Steel
MPC-LACBWR Fuel Basket		
- End Weldments	1.0 × 69.3 dia.	Type 304 Stainless Steel
- Support Disks	1.25 × 69.4 dia 0.625 × 69.4 dia 0.75 × 69.4 dia.	Type 17-4 PH Stainless Steel
- Heat Transfer Disks	0.5 × 69.13 dia.	Type 6061-T651 Aluminum Alloy
-Fuel Tube		
Standard	5.85 × 5.85 × 0.048	Type 304 Stainless Steel
Enlarged	6.10 × 6.10 × 0.048	Type 304 Stainless Steel
- Spacers	3.2 diameter	Type 304 Stainless Steel
- Tie Rods (8)	1-5/8 diameter	Type 304 Stainless Steel

1. Dimensions in inches unless otherwise noted.

Table 1.A.1-1 Design Characteristics of the MPC-LACBWR System (continued)

Design Characteristic	Dimension[1]	Material
MPC-LACBWR Transfer Cask		
- Outer Shell	1.25 x 86.5 dia.	ASTM A588 Low Alloy Steel
- Inner Shell	0.75 x 73.0 dia.	ASTM A588 Low Alloy Steel
- Retaining Ring	0.75 x 80.8 dia.	ASTM A588 Low Alloy Steel
- Trunnions	10.0 dia.	ASTM A350 LF2 Carbon 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	9.5 thick	ASTM A350 LF2 Carbon Steel
- Door Rails	9.88 x 6.5	ASTM A350 LF2 Carbon Steel
- Gamma Shield	3.5 thick	ASTM B29, Chem. Copper Grade Lead
- Neutron Shield	2.0 thick	NS-4-FR, Solid Synthetic Polymer
Transfer Adapter		
- Base Plate	2.0 thick Plate	ASTM A36 Carbon Steel
- Locating Ring	2.0 wide x 78.25 dia.	ASTM A36 Carbon Steel

1. Dimensions in inches unless otherwise noted.

Table 1.A.1-1 Design Characteristics of the MPC-LACBWR System (continued)

Design Characteristic	Dimension[1]	Material
MPC-LACBWR Vertical Concrete Cask		
Weldment Structure		
- Shell	2.5 thick x 84.0 dia.	ASTM A36 Carbon Steel
- Top Flange	2.0 thick x 97.9 dia.	ASTM A36 Carbon Steel
- Base Plate	2.0 thick x 72.0 dia.	ASTM A36 Carbon Steel
Concrete Cask		
- Concrete Shell	22.0 thick x 128 dia.	Type II Portland Cement
- Lid Shield	8.38 x 78.5 dia.	ASTM A36, Carbon Steel and Type II Portland Cement
- Lid Plate	1.5 thick x 91.5 dia.	ASTM A36, Carbon Steel
- Rebar	Various	ASTM A615, GR 60, Carbon Steel

1. Dimensions in inches unless otherwise noted.

1.A.2 The MPC-LACBWR Storage System

The MPC-LACBWR system is similar to the Yankee-MPC and the CY-MPC system components and operating features with specific enhancement to improve ALARA operations and storage capacities. The MPC-LACBWR system provides long-term storage and subsequent transport of the stored spent fuel using the certified NAC-STC. 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.A.2.1 MPC-LACBWR System Components

The MPC-LACBWR system consists of three principal components:

- Transportable storage canister (canister),
- Vertical concrete cask, and
- Transfer cask.

Ancillary equipment needed to use the MPC-LACBWR system is:

- Automated or manual welding equipment;
- An air pallet or hydraulic roller skid (used to move the storage cask on and off the heavy haul transfer trailer and to position the storage cask on the storage pad);
- Suction pump, vacuum drying, helium backfill and leak detection equipment;
- A heavy haul trailer or cask transporter (for storage cask transport to the storage pad);
- 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 closure lid.

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

The transportable storage canister is designed to be transported in the NAC-STC (Certificate of Compliance No. 71-9235). The transport load conditions produce higher stresses in the canister than would be produced by the storage load conditions alone. Consequently, the canister design is conservative with respect to storage conditions.

1.A.2.1.1 Transportable Storage Canister and Baskets

The Transportable Storage Canister (canister) contains a basket that is designed to accommodate up to 68 LACBWR spent fuel assemblies, including up to 32 damaged fuel cans.

The canister assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a closure lid, closure ring and two redundant sets of penetration port covers. The cylindrical shell, plus the bottom plate, closure lid and inner port covers constitute the confinement boundary. The fuel basket design and configuration is similar to and based on the directly loaded fuel basket design used in the certified NAC-STC and the certified NAC-MPC and NAC-UMS canister based spent fuel storage and transport systems. This basket features the NAC-patented poison tubes and stacked disk design with heat transfer disks. The basket was analyzed using the ANSYS computer code to demonstrate that it can withstand the horizontal drop loads without deforming in a way that damages or constrains a fuel assembly. This tube and disk design has been accepted and approved by the NRC, pursuant to 10 CFR 71 and 10 CFR 72. Table 1.A.2-1 summarizes the major physical design parameters of the canister configurations.

The fuel basket design is a right-circular cylinder configuration with 68 fuel tubes laterally supported by a series of support disks, which are retained by spacers on radially located tie rods. Damaged fuel cans may be placed in 32 peripheral oversized fuel tubes. Eight tie rods are used in the MPC-LACBWR basket design. The support disks are stainless steel (17-4 PH) with standard and oversized holes for the poison fuel tubes and damaged fuel cans. The first top and bottom end support disks are thicker than the intermediate support disks to accommodate postulated rubblized fuel in the 32 damaged fuel cans. The basket top and bottom weldments are fabricated from Type 304 stainless steel. The tie rods and spacer sleeves are also fabricated from Type 304 stainless steel. The fuel assemblies are contained in fuel tubes. The MPC-LACBWR fuel tubes are fabricated from Type 304 stainless steel with stainless steel clad covered BORAL sheets on defined outside surfaces of the fuel tube. The BORAL provides criticality control in the basket.

The MPC-LACBWR fuel tubes are fabricated from 18-gauge Type 304 stainless steel sheet. The standard fuel tube has a square interior cross-section of 5.75 inches and supports a clad covered BORAL sheet on defined outside surfaces of the fuel tube. The enlarged fuel tube has a square interior cross-section of 6.0 inches, and supports a clad covered BORAL sheet on three or four sides. Enlarged fuel tubes with BORAL sheets on three sides have an aluminum sheet on the fourth side in order to provide a symmetric interface between the fuel tube and the top basket support disk. These larger cross-section fuel tubes can accommodate damaged fuel cans and fuel assemblies that exhibit slight physical effects (e.g., twist, bow) that could preclude loading in the smaller cross-section standard fuel tubes. The enlarged fuel tubes are located in the 32 periphery fuel cell positions of the basket as shown in Figure 2.A.1-1. When installed, the standard and enlarged fuel tubes are captured between the top and bottom weldments of the fuel basket.

The damaged fuel can is similar to a fuel tube without exterior BORAL sheets on the sides and is closed on its bottom end by a stainless steel bottom plate having screened openings. After loading, the can is closed on its top end by a stainless steel lid that also has screened openings. The top plate and can body incorporate lifting fixtures that allow movement of the loaded can, and installation and removal of the can lid. The damaged fuel can extends through the bottom and top weldments of the basket, and is captured between the closure lid and the canister bottom plate. The damaged fuel can lid is held in place by the closure lid. The screened openings in the damaged fuel can lid and bottom plate allow the filling, draining and vacuum drying of the damaged fuel can, but preclude the release of gross particulate matter to the canister interior.

The heat transfer disks are aluminum plates with holes for the standard and enlarged fuel tubes. The heat transfer disks are spaced midway between the support disks and are the primary path for conducting the heat from the fuel assemblies to the canister wall. Holes in the heat transfer disks for the tubes, damaged fuel cans, and tie rods are sized to accommodate thermal expansion occurring after the fuel is placed into the basket.

The fuel basket tube-and-disk design provides the structural integrity to maintain the spent fuel in a subcritical configuration during normal operations and the hypothetical accident events, even if optimum moderator condition and fresh fuel are assumed. With the most reactive fuel, the fuel basket maintains $k_{eff} \leq 0.95$. Subcriticality is assured assuming fresh fuel loading and no soluble boron in the spent fuel pool water during fuel loading operations.

The transportable storage canister assembly is designed to facilitate filling with water and subsequent draining and drying. Each fuel tube is supported by the basket bottom weldment, ensuring free flow of water between the inner tube regions and the bottom of the canister. The top lid and bottom plate of the damaged fuel can incorporate screened openings to allow water to fill and drain during loading and canister closure operations. In addition, the bottom weldment is positioned by supports above the bottom of the canister to facilitate water flow to the drain line.

The canister is fabricated from 1/2-inch-thick dual certified Type 304/304L stainless steel rolled plate, joined at its edges by a full penetration weld, which is radiographed. The bottom closure is a 1.25-inch-thick Type 304/304L stainless steel plate joined to the canister shell by a full penetration weld, which is ultrasonically examined. The design of the closure lid and closure ring with dual redundant port covers provides a redundant confinement boundary at the top of the canister. The closure lid weld to the canister shell is inspected using liquid penetrant examination on the root, intermediate, and final passes.

The LACBWR closure lid design includes a 4-inch-thick, 38.3-inch-square aluminum spacer plate attached to the underside of the lid to limit axial movement of the fuel assemblies placed in the 36 basket locations that do not contain damaged fuel cans. Axial movement of the damaged fuel cans is limited by the position of the closure lid bottom surface.

The vent and drain ports through the closure lid allow the inner cavity to be drained, evacuated, and backfilled with helium to provide an inert atmosphere for long-term dry storage. The drain port is equipped with a quick disconnect fitting and a drain tube that extends nearly to the bottom of the canister. The vent port extends to the underside of the closure lid and is equipped with a quick disconnect fitting used for vacuum drying and helium backfilling. After draining, drying, backfilling, and testing operations are complete, port covers are installed and welded to the closure lid to seal the penetration. Leak testing is performed on both inner port cover welds followed by installation of a second redundant port cover for each port.

A summary of the canister fabrication specifications is presented in Table 1.A.2-2.

1.A.2.1.2 Vertical Concrete Cask

The vertical concrete cask (storage cask) is the storage overpack for the transportable storage canister. It provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage. Table 1.A.2-3 lists the major physical design parameters of the storage cask for the MPC-LACBWR configuration. The storage cask is a reinforced concrete (Type II Portland cement) structure with a structural steel inner liner. The concrete wall and steel liner provide neutron and gamma radiation shielding. 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 storage cask incorporates reinforced chamfered corners at the edges to facilitate construction. The MPC-LACBWR base weldment base plate is covered with a ¼-inch-thick stainless steel plate to prevent contact between the stainless steel canister and the carbon steel pedestal. The storage cask is shown in Figure 1.A.2-1.

The storage cask has 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 from the concrete cask cavity to minimize radiation streaming. The decay heat is transferred from the fuel assembly to the fuel tube or damaged fuel can and fuel tube 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 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 storage cask is closed by a lid with integral radiation shield. The radiation shield is approximately 8-inch thick concrete encased in a carbon steel shell extending into the cask cavity from the bottom surface of the 1.5-inch-thick carbon steel lid.

Fabrication of the storage cask involves no unique or unusual forming, concrete placement, or reinforcement requirements. The concrete portion of the storage cask is constructed by placing concrete between a reusable, exterior form and the inner metal liner. Reinforcing bars are placed near the inner and outer concrete surfaces to provide structural integrity. The inner liner and base of the storage cask are shop fabricated. Radiation shielding is installed in the air inlets to reduce dose rates local to the air inlets at the base of the cask.

The principal fabrication specifications for the storage cask are shown in Table 1.A.2-4.

1.A.2.1.3 Transfer Cask

The MPC-LACBWR transfer cask is the same transfer cask that was designed, licensed and used for loading operations for the Yankee-MPC system at Yankee Rowe. New shield doors, with the ALARA enhancement of door stops, are fabricated for the MPC-LACBWR transfer cask. The transfer cask drawing is provided in the MPC FSAR, Section 1.7.1, Yankee-MPC License Drawing No. 455-860.

The transfer cask, with its lifting yoke, is primarily a lifting device used to move the canister assembly. It provides biological shielding when it contains a loaded canister. The transfer cask is used for the vertical transfer of the canister between work stations and the storage cask, or transport cask. The general arrangement of the transfer cask and canister is shown in Figure 1.A.2-4, and the arrangement of the transfer cask and concrete cask is shown in Figure 1.A.2.-5. The configuration of the transfer cask, canister and concrete cask during loading of the concrete cask is shown in Figure 1.A.2-6.

Table 1.A.2-5 shows the principal design parameters of the transfer cask used for the Yankee-MPC and MPC-LACBWR systems.

The transfer cask is a multiwall (steel/lead/NS-4-FR neutron shield/steel) design, which limits the average contact radiation dose rate to less than 100 mrem/hr. The transfer cask design incorporates a top retaining ring, which is bolted in place preventing a loaded canister from being inadvertently removed 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 stops, 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 casks. The transfer cask with adaptor plate is shown in Figure 1.A.2-2.

To qualify the transfer cask as a heavy lifting device, it is designed, fabricated, and proof load tested to the requirements of NUREG-0612 and ANSI N14.6. Maintenance is to be performed in accordance with site specific procedures that meet the requirements of NUREG-0612.

To minimize potential contamination of the canister and the transfer cask during loading operations in the spent fuel pool, clean water is circulated in the gap between the transfer cask interior surface and the canister exterior surface using fill and drain lines in the wall of the transfer cask. The clean water flow precludes the intrusion of pool water when the canister is submerged. Clean water is processed or filtered pool water, or any water external to the spent fuel pool that is compatible.

Exposed surfaces of the transfer cask, other than the load-bearing surfaces of the trunnions and the bottom door rails, are coated with Keeler & Long E-Series epoxy enamel or Carboline 890 to protect the carbon steel and to provide a smooth surface to facilitate decontamination.

1.A.2.1.4 Ancillary Equipment

This section presents a brief description of the principal ancillary equipment needed to operate the MPC-LACBWR system in accordance with its design.

1.A.2.1.4.1 Adapter Plate

The adapter plate is a carbon steel table that mates the transfer cask to either the vertical concrete (storage) cask or the NAC-STC transport cask. 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 adapter plate to guide and support the bottom shield doors of the transfer cask when they are in the open position. The adapter plate also supports the hydraulic system and the actuators that open and close the transfer cask bottom doors. The adapter plate drawing is provided in the MPC FSAR, Section 1.7.1, Yankee-MPC License Drawing No. 455-859.

1.A.2.1.4.2 Air Pad Rig Set

The air pad rig set (air pad set) is a commercially available device, sometimes referred to as an air pallet or air bearing. When inflated, the air pad set lifts the concrete cask by using high volume air. 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 filler 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, a suitable towing vehicle, such as a commercial tug or forklift can move the storage cask.

1.A.2.1.4.3 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.A.2.1.4.4 Draining and Drying System

The draining and drying system consists of a suction pump, vacuum pump and helium backfill system. The suction pump and/or helium backfill system 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.A.2.1.4.5 Helium Leak Test Equipment

A helium leak detector and leak test fixture are required to verify the integrity of the welds between the inner port covers and closure lid. The helium leak detector is the mass spectrometer type.

1.A.2.1.4.6 Heavy-Haul Trailer

The heavy haul trailer is used to move the vertical concrete storage cask. A special trailer has been designed for transport of the empty or loaded storage cask. However, any commercial double-drop-frame trailer having a deck height approximately matching that of the storage pad could be used.

1.A.2.1.4.7 Lifting Jacks

Hydraulic jacks are installed in the air inlets to lift the storage 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 four air pads under the concrete cask.

1.A.2.1.4.8 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 B30.9 and NUREG-0612. The slings include a concrete cask lid sling, canister closure lid sling, loaded canister transfer sling (also used to handle the closure lid), and canister retaining ring sling. The appropriate rings or eye bolts are provided to accommodate each sling and component.

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 single-failure-proof by design. The transfer cask lifting yoke is initially load tested to 300 percent of the design load.

1.A.2.1.4.9 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 screen blockage to ensure the cask heat removal system remains operable.

1.A.2.1.5 Transport Cask

The transportable storage canister is designed to be transported in the NAC-STC. The canister is positioned in the NAC-STC cavity with two axial spacers. The spacers are required because the transport cask cavity length is 165 inches, while the length of the MPC-LACBWR canister is 116.3 inches.

The NAC-STC is licensed by the NRC pursuant to 10 CFR 71 (Certificate of Compliance No. 71-9235) for shipment of the MPC canister. The NAC-STC is designed for free interchange/rail shipment and transport by heavy-haul truck or barge. The rail transport configuration is shown in Figure 1.A.2-3.

1.A.2.2 Operational Features

This section outlines the principal handling activities of the MPC-LACBWR storage system. The system provides passive long-term storage of spent fuel in an inert environment.

The principal activities associated with the use of the system are closing the canister and loading the canister in the storage 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 storage cask. The canister assembly consists of four principal components: the canister shell (side wall and bottom), closure lid, closure ring and redundant vent and drain port covers. A drain tube extends from the closure lid drain port to the bottom of the canister. The location of the drain and vent ports is shown in MPC FSAR Figure 8.1-1.

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 inner vent and drain port covers, the closure lid, the canister shell, and the joining welds form the primary confinement boundary. A secondary or redundant welded boundary is formed by the closure ring welds to the canister shell and closure lid and the second redundant port cover welds to the closure lid. This boundary is shown in Figure 7.A.1-1.

The closure 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 may be filled with optional 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 use of the MPC-LACBWR system are presented in Appendix A to Chapter 8. The following list presents a brief description of the principal activities. This list assumes that the empty canister is installed in the transfer cask for spent fuel pool loading.

- Lift the transfer cask over the pool and start the flow of water to the transfer cask annulus and canister. After the annulus and canister are filled, lower the cask to the bottom of the pool.
- Load the selected spent fuel assemblies into the canister and set the closure 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.

- Weld the closure lid to the canister shell. Inspect the weld. Pressure test the weld. Weld the closure ring to the canister shell and closure lid and inspect welds. Drain the pool water from the canister while backfilling the cavity with helium. Attach the vacuum system to the drain line, and operate the system to achieve a vacuum.
- Hold the vacuum and backfill with helium to 1 atmosphere. Restart the vacuum system and remove the helium. After achieving vacuum, backfill the canister with helium.
- Weld the inner port covers to the closure lid and helium leak check the welds. Install the redundant vent and drain port covers and weld them to the closure lid.
- Install the hoist rings, and attach the canister lifting sling. Install the adapter plate on the storage cask.
- Lift the transfer cask to the top of the storage cask and set it on the adapter plate, ensuring that the bottom door hydraulic actuators are engaged.
- Attach the canister lifting slings to the crane hook and lift the canister.
- Open the bottom doors of the transfer cask.
- Lower the canister into the storage cask. Detach the canister slings from the hook.
- Remove the transfer cask and adapter plate. Remove the canister lifting slings.
- Install the lid on the concrete cask.
- Move the loaded storage cask to the storage pad.
- Using the air pad rig set and a towing vehicle, move the storage cask to its designated location on the storage pad.
- During storage operations, the operability of the concrete cask is verified 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 are required.

The ancillary equipment needed to operate the MPC-LACBWR system has been described in Section 1.A.2.1.4. Other items required are miscellaneous hardware, connection hose and fittings, and hand tools typically found at a reactor site.

Figure 1.A.2-1 MPC-LACBWR Vertical Concrete Storage Cask

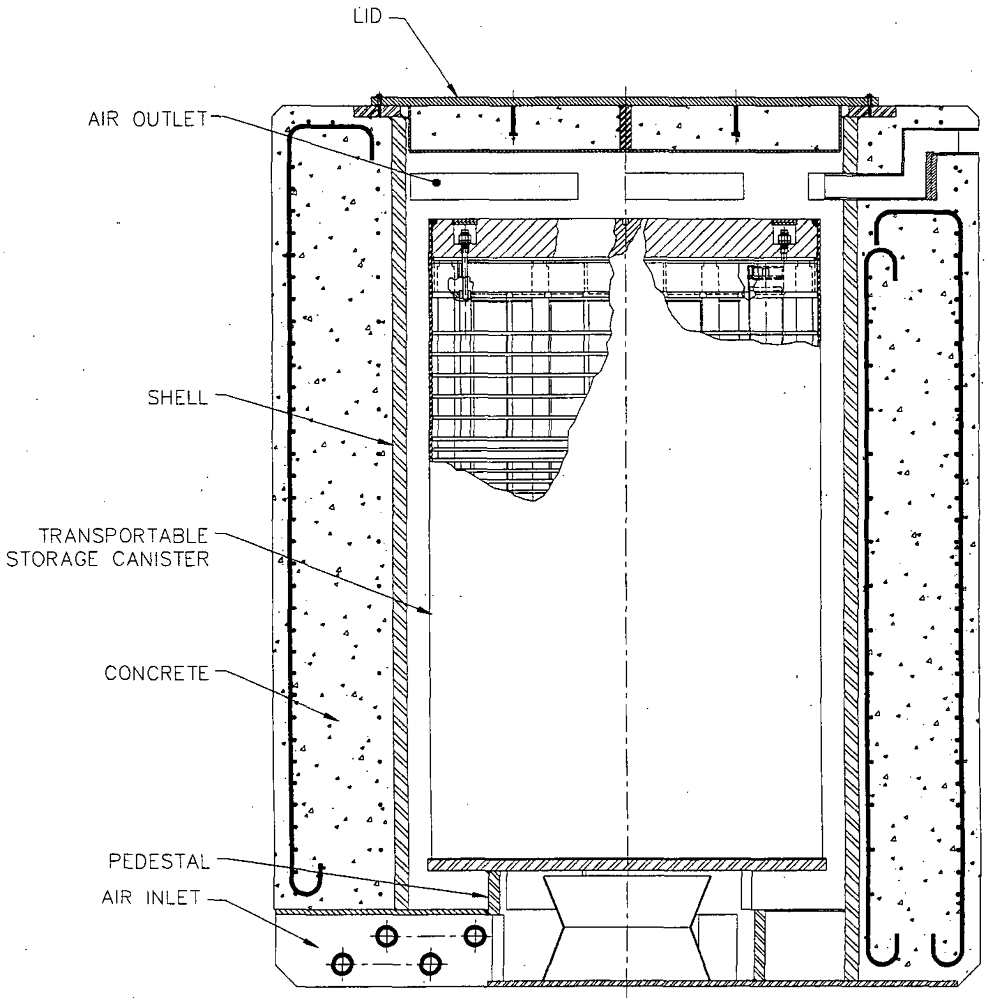

Figure 1.A.2-2 MPC-LACBWR Transfer Cask with Adapter Plate

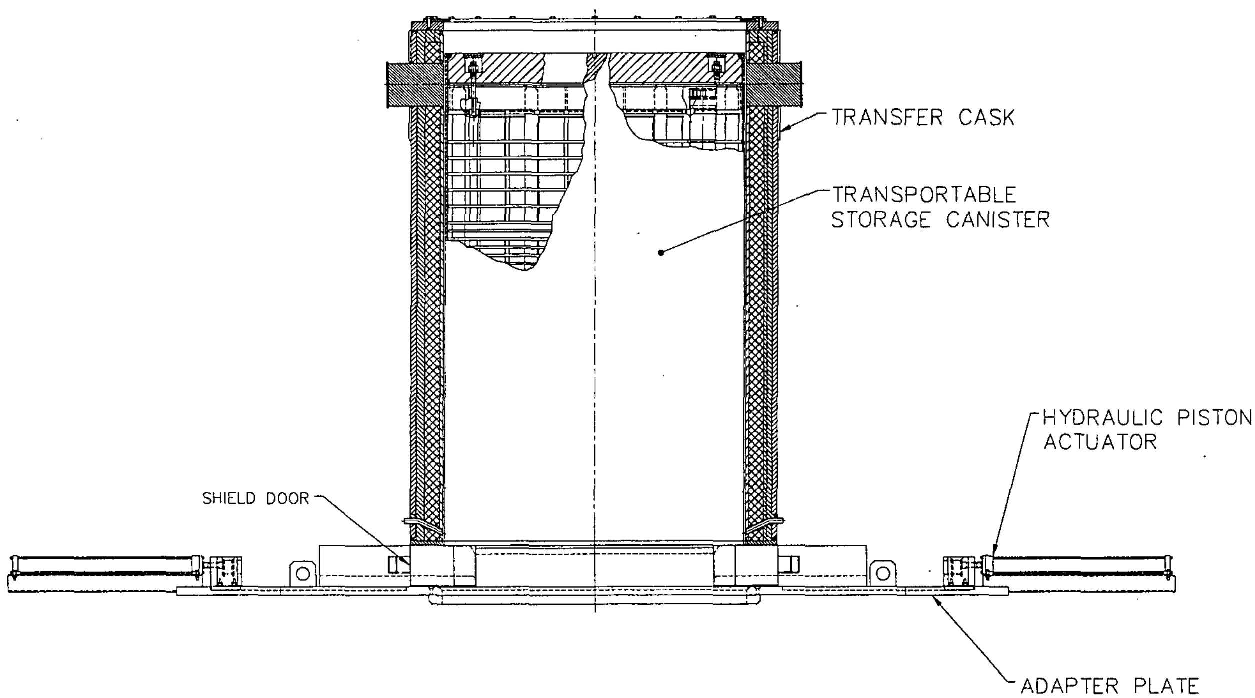

Figure 1.A.2-3 NAC-STC Transport Configuration

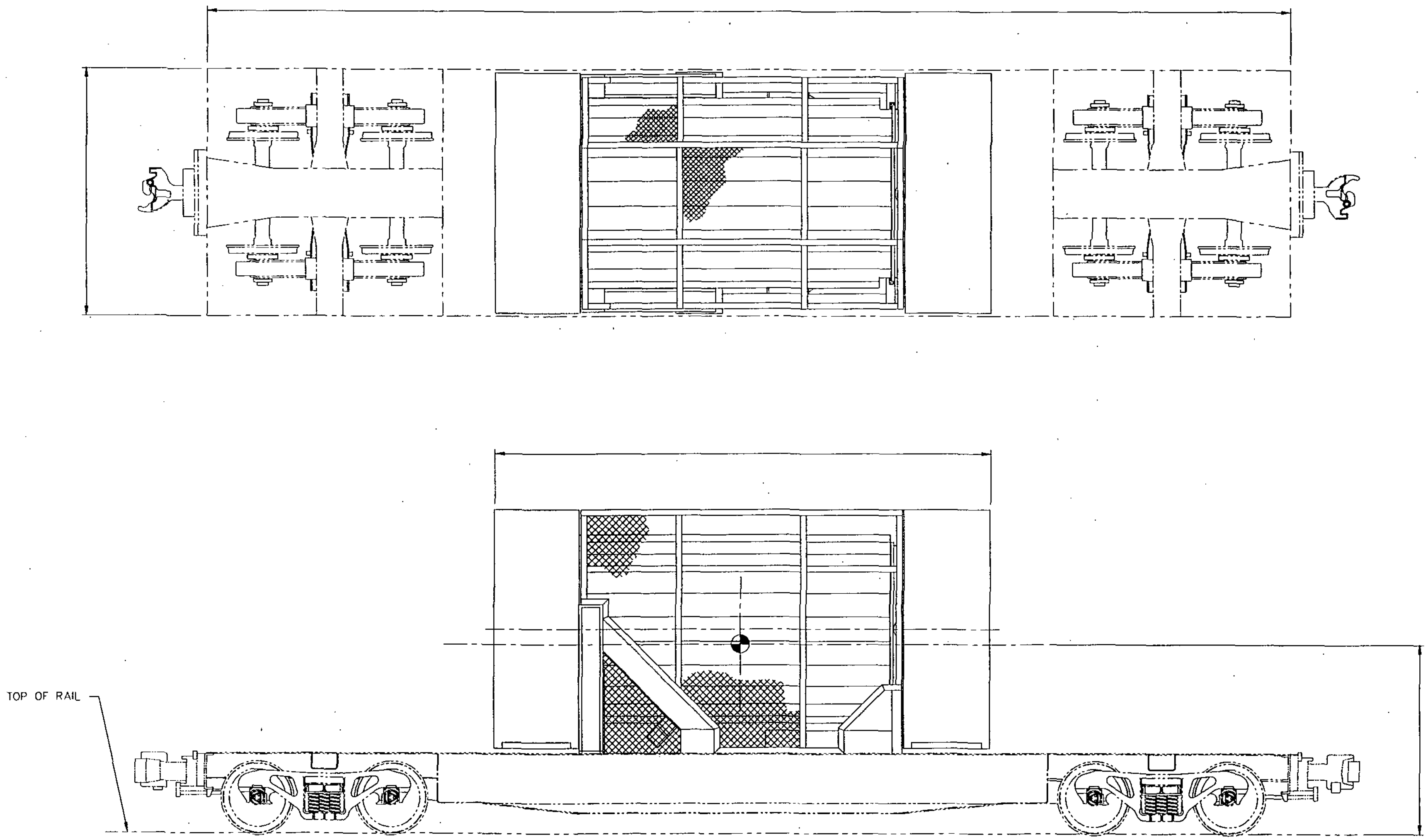

Figure 1.A.2-4 MPC-LACBWR Transfer Cask and Canister Arrangement (for illustration purposes only)

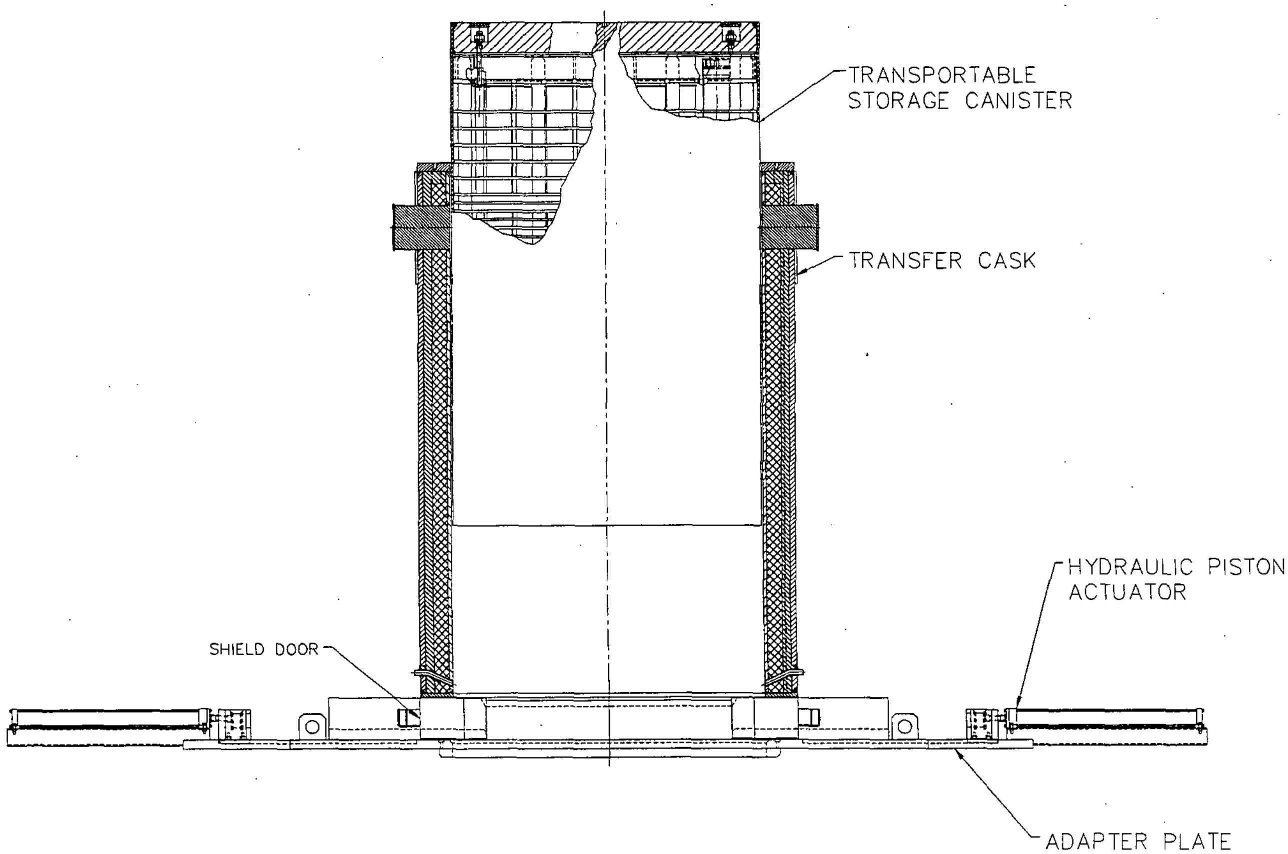

Figure 1.A.2-5 Vertical Concrete Cask and Transfer Cask Arrangement

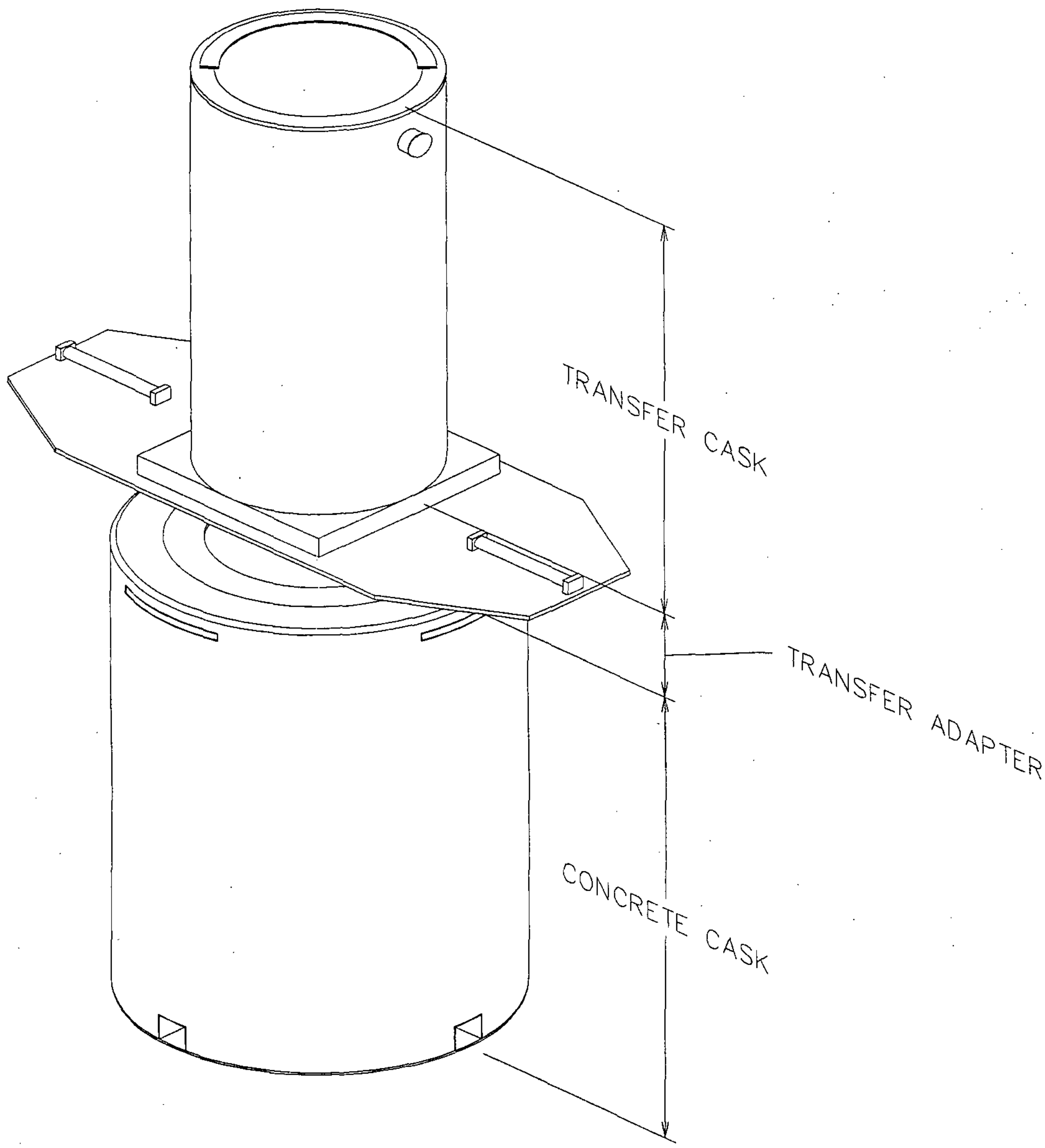

Figure 1.A.2-6 Major Component Configuration for Loading the MPC-LACBWR Vertical Concrete Cask

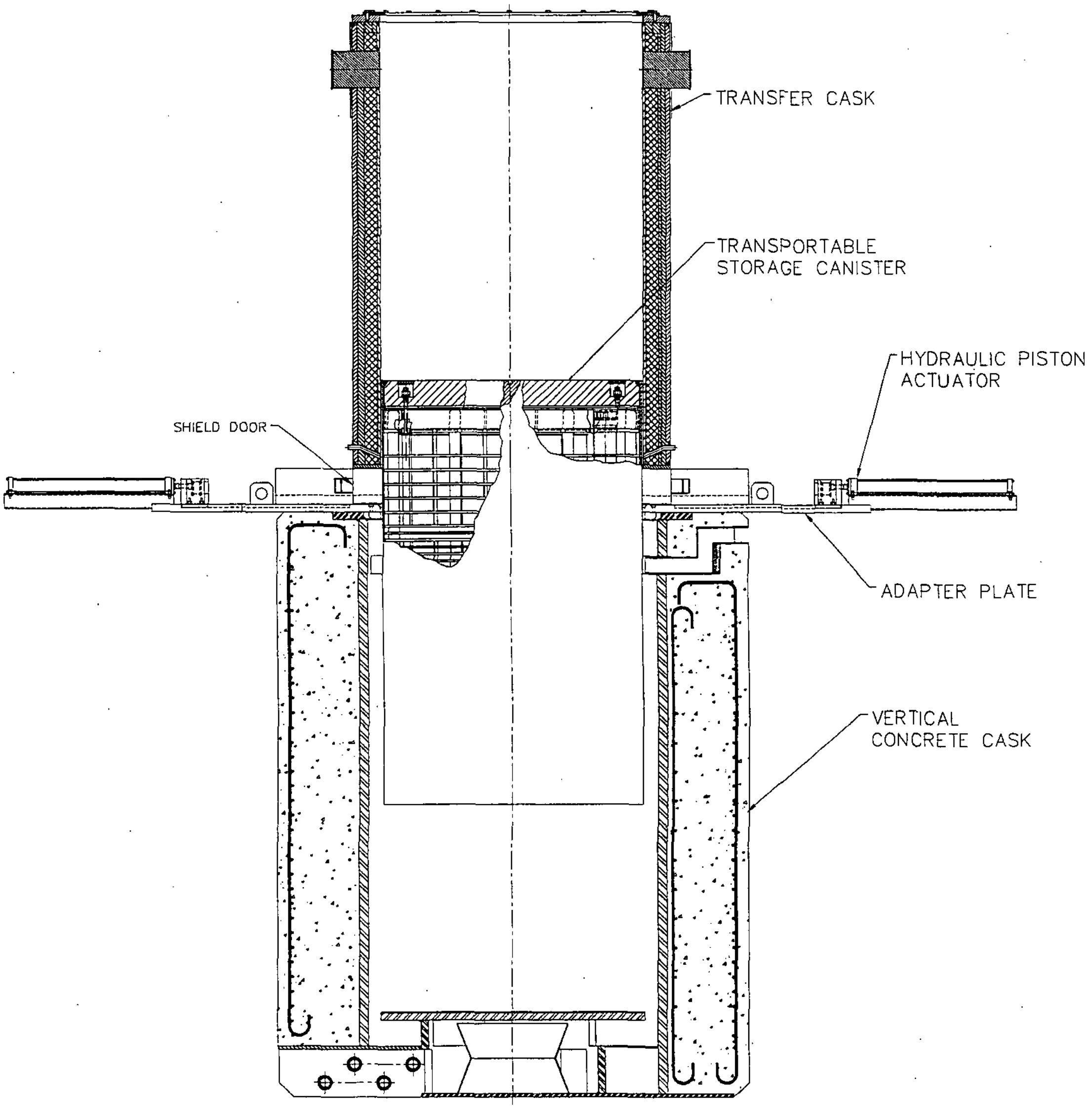

Table 1.A.2-1 Major Physical Design Parameters for the MPC-LACBWR Transportable Storage Canister

Parameters	MPC-LACBWR
Outside Diameter	70.64 in.
Length	116.3 in.
Capacity	68 LACBWR spent fuel assemblies
Weight	54,650 lbs. loaded, with lid, no water
Maximum heat load	4.5 kW (fuel)
Maximum Cladding Temperature Stainless Steel Normal Conditions Off-normal and Accident	 430°C[1] 430°C
Internal Atmosphere	Helium

1. See Section 2.A.1 and Table 2.A-1.

Table 1.A.2-2 MPC-LACBWR Transportable Storage Canister Fabrication Specification Summary

Materials

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

Welding

- All welds shall be in accordance with the referenced drawings.
- All filler metals shall be appropriate ASME material.
- All welders and welding operators shall be qualified in accordance with ASME Code Section IX.
- All welding procedures shall be written and qualified in accordance with ASME Code Section IX.
- All welds specified to be visually examined shall be examined as specified in ASME Code Section V, Article 9 with acceptance per ASME Code Section III, NB-4424, NB-4426 and NB-4427.
- All welds specified to be liquid penetrant examined shall be examined in accordance with the requirements of ASME Code Section V, Article 6, with acceptance in accordance with ASME Code Section III, NB-5350.
- All personnel performing examinations shall be qualified in accordance with the NAC International Quality Assurance program and SNT-TC-1A, as appropriate.
- 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 in accordance with ASME Code Section V, Article 5, with acceptance in accordance with ASME Code Section III, NB-5330.
- Canister weldment shall be helium leakage tested using the evacuated envelope method as described in the ASME Code, Section V, Article 10 and ANSI 14.5.

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, Section 2.
- All fabrication tolerances shall meet the requirements of the referenced drawings after fabrication.

Packaging

- Packaging and shipping shall be in accordance with ANSI N45.2.2, Level D.

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 NAC International prior to initiation of work.
- Hold points are established by NAC and contractually imposed on the fabricator to assure the completed hardware complies with the licensed configuration. Hold points may include verification of the basket assembly diameter and length, insertion of a "dummy" fuel assembly into each fuel tube, and insertion of the basket into the canister shell.

A Certificate of Conformance (or Compliance) shall be issued by the fabricator stating that the canister meets the specifications and drawings.

Table 1.A.2-3 Major Physical Design Parameters for the MPC-LACBWR Vertical Concrete Cask

Vertical Concrete Cask Parameters	**MPC-LACBWR**
Height (including lif)	162 in.
Outside diameter	128 in.
Shielding (side wall) Concrete thickness Steel thickness	 22 in. 2.50 in.
Radiation dose rate (average): Side surface Top surface Air inlet/outlet vents	 ≤ 20 mrem/hr ≤ 25 mrem/hr ≤ 100 mrem/hr
Weight	141,200 lbs. (nominal)
Material of construction Concrete Reinforcing steel Steel liner	 Type II Portland Cement A615 Grade 60 A36 Carbon Steel
Service life	50 years
Maximum concrete temperatures for normal operation	150°F bulk 200°F local

Table 1.A.2-4 MPC-LACBWR Concrete Cask Fabrication Specification Summary

Materials

- Concrete mix shall be in accordance with the requirements of ACI 318 and ASTM C94.
- Type II Portland Cement, ASTM C150.
- Fine aggregate ASTM C33 and C637.
- Coarse aggregate ASTM C33 and C637.
- Admixtures
  - Water Reducing ASTM C494.
  - Pozzolanic Admixture ASTM C618.
- Compressive Strength 4000 psi at 28 days.
- Specified Air Entrainment in accordance with 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 8.15.

Construction

- Specimens shall be obtained or prepared for each batch or truck load of concrete per ASTM C172 and ASTM C192.
- Test specimens shall be tested in accordance with ASTM C39.
- Formwork shall be in accordance with ACI 318.
- All sidewall formwork and shoring shall remain in accordance with the requirements of ACI 318.
- All bottom formwork and shoring shall remain in place for 14 days.
- 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 prior to initiation of the work.

Table 1.A.2-5 Major Physical Design Parameters for the MPC-LACBWR Transfer Cask

Transfer Cask Parameters	MPC-LACBWR
Inside Diameter	71.5 in
Outside Diameter	86.5 in
Height	133.38 in
Empty Weight (nominal)	80,800 lbs
Side Wall Dose Rate (average)	$\leq$ 100 mrem/hr

1.A.3 MPC-LACBWR Storage System Contents

The MPC-LACBWR storage system is designed to hold up to 68 spent fuel assemblies with 32 oversized peripheral cell locations to accommodate damaged fuel cans.

1.A.3.1 MPC-LACBWR Spent Fuel

The MPC-LACBWR is designed to store up to 68 LACBWR spent fuel assemblies. The LACBWR fuel consists of Allis Chalmers fuel assemblies with enrichments of 3.64 wt % and 3.94 wt % and Exxon fuel assemblies with planar average enrichment of 3.71 wt %. The characteristics of the MPC-LACBWR spent fuels are presented in Table 1.A.3-1. Unenriched fuel assemblies are not evaluated and are not included as proposed contents.

1.A.3.2 MPC-LACBWR Damaged Fuel Can

The damaged fuel can is designed to hold a complete fuel assembly or debris. LACBWR damaged fuel includes fuel assemblies that cannot be handled with normal fuel handling equipment and may be placed in an ancillary fuel handling sleeve with drainable bottom. The damaged fuel can has a square cross-section that is slightly larger than a standard MPC-LACBWR fuel assembly. Consequently, loading of the damaged fuel can into the MPC-LACBWR canister basket is restricted to one of the 32 peripheral oversized cell basket positions. The damaged fuel can is fabricated from stainless steel and has top and bottom closures that allow the release of gaseous products and liquids but minimizes the dispersal of particulates.

A sketch of the MPC-LACBWR damaged fuel can is provided in Figure 1.A.3-1. The major physical design parameters are presented in Table 1.A.3-2. The design and fabrication specification summary is provided in Table 1.A.3-3.

Figure 1.A.3-1 MPC-LACBWR Damaged Fuel Can

Figure Withheld Under 10 CFR 2.390

Table 1.A.3-1 MPC-LACBWR Design Basis Fuel Characteristics

Parameter	Allis Chalmers	Exxon
Number of Assemblies per Canister [1]	32	68
Assembly Weight, lbs.	400	400
Assembly Length, in.	103	103
Fuel Rod Cladding	Stainless Steel	Stainless Steel
Maximum Uranium, kgU [2]	121.4	111.9
Maximum Initial ^{235}U, wt %	3.94/3.64	3.71 [3]
Maximum Burnup, MWd/MTU	22,000	21,000
Maximum Assembly Decay Heat, W	63	62
Minimum Cool Time, yr	28	23

1. Maximum 68 assemblies per canister. Allis Chalmers fuel is restricted to DFCs. Therefore, Allis Chalmers fuel is limited to 32 assemblies per canister.
2. DFCs have been evaluated for 5% additional fuel rod mass.
3. Represents planar average enrichment.

Table 1.A.3-2 Major Physical Design Parameters of the MPC-LACBWR Damaged Fuel Can

Parameter	**Value**
Overall Length (in.)	106.75
Inside Cross Section (in.)	5.75
Outside Cross Section (in.) (1)	5.85
Can Wall Thickness	18 gauge (0.05 in)
Internal Cavity Length (in.)	103.75
Empty Weight (nominal) (lbs.)	55

1. Outside cross section of the Damaged Fuel Can upper structure is 6.05 x 6.05 in. at top (4.5 in.) for lid engagement and fuel can lifting. This upper structure is located above the top weldment plate of the fuel basket assembly.

Table 1.A.3-3 MPC-LACBWR Damaged Fuel Can Design and Fabrication Specification Summary

Design

- The MPC-LACBWR damaged fuel can shall be designed in accordance with ASME Code, Section III, Subsection NG, 1995 Edition with 1995 Addenda, with the alternatives specified in Table B.3-1 of Appendix 12.B of the Technical Specifications for fuel basket structures.
- The damaged fuel can will have screened vents in the lid and base plate. Stainless steel fine mesh screens shall cover all openings to minimize the dispersal of particulates from damaged fuel assemblies to the canister cavity.
- The damaged 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 Type 304 stainless steel.

Welding

- All welds shall be in accordance with the referenced drawings.
- All welds that are specified to be visually examined shall be examined as specified in 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 first unit of the MPC-LACBWR damaged fuel can shall be load tested at 150% of design load and visually inspected at the completion of fabrication.

Quality Assurance

- The MPC-LACBWR damaged 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 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.

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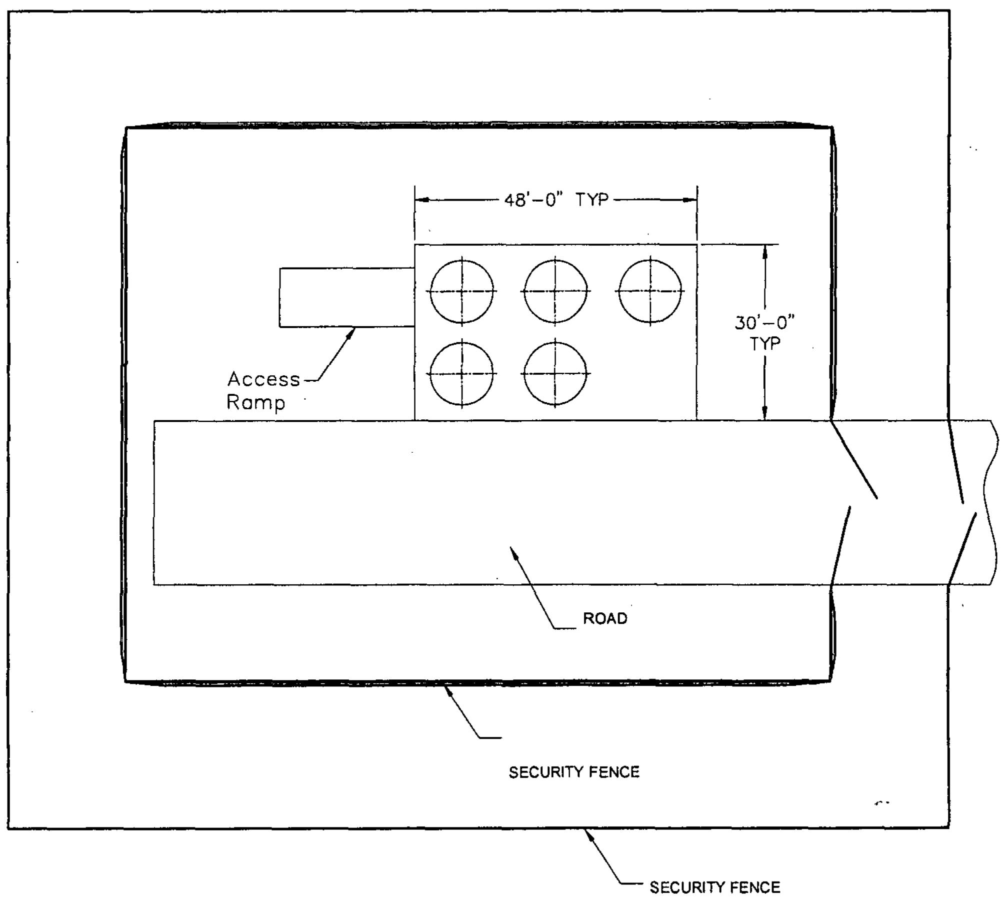
1.A.4 MPC-LACBWR Storage Cask Arrays

The MPC-LACBWR is designed for long-term storage at an ISFSI. At the ISFSI site, the loaded concrete storage casks are placed in the vertical position on a concrete pad in a linear array. The array size for the LACBWR ISFSI is limited to 5 MPC-LACBWR casks to accommodate all the spent fuel in the LACBWR spent fuel pool. The reinforced concrete foundation of the ISFSI pad is capable of sustaining the transient loads from the air pad and the general loads of the stored casks. Figure 1.A.4-1 shows cask spacing and representative site dimensions for a 5-cask array. This configuration is used in the controlled site boundary dose calculations presented in Section 10.A.4.

Figure 1.A.4-1 Conceptual MPC-LACBWR ISFSI Storage Pad Layout



1.A.5 MPC-LACBWR Storage System Compliance with NUREG-1536

The design of the MPC-LACBWR Storage System meets the regulatory requirements and acceptance criteria specified in NUREG-1536 as shown in Table 1.A.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 NAC-MPC Storage System Safety Analysis Report where each of the requirements or criteria are addressed.

Table 1.A.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, non-proprietary 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.A.2.
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.A.7. Safety classifications are provided in Table 2.A.3-1.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 1 – General Description			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
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., intact assembly or consolidated fuel rods), weight and nature of non-spent 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 1.A.3.

Table 1.A.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.A.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.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 1 – General Description			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
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 SAR and Certificate of Compliance are discussed in Section 1.A.2.1.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 2 – Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
1. Structures, Systems, and Components 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 for 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 safety classification of system components are described in Table 2.A.3-1. The design bases and criteria for the system are specified in Table 2.A-1. Detailed design criteria are presented in Section 2.A.2.

Table 1.A.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., intact 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 and 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., intact 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.A.1. Specific physical parameters of the fuel are listed in Table 2.A.1-1.

Table 1.A.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 **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.A.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 are specified in Section 3.A.1.2.

Table 1.A.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 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 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)] 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)] 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)] Storage systems must be designed to allow ready retrieval of spent fuel waste for further processing or disposal.	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.A.2. Combined loadings are addressed specifically in Section 2.2.5 and in Table 2.2-1. The design-basis earthquake is specified in Section 2.A.2.1 in accordance with 10 CFR 72.102 criteria. Because the system maintains adequate margins of safety during normal (Section 3.A.4.4), off-normal (Section 11.A.1) and accident condition (Section 11.A.2) events, confinement of the radioactive material is assured. Because 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 Appendix 6.A. The maximum allowable cladding temperatures are specified in Tables 2.A-1 and 4.A.3-3. The temperature results for the fuel cladding listed in Table 4.A.3-3 show that the allowable cladding temperatures are not exceeded. Therefore, the fuel cladding is protected against degradation during storage. As described in Section 1.A.2, the system is designed to be readily transported as necessary for further processing or disposal.

Table 1.A.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 DCSS must be designed to provide adequate heat removal capacity without active cooling systems. [10 CFR 72.236(f)]	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 passive design and reliability of the MPC-LACBWR system heat removal capability is demonstrated by analyses presented in Appendix 4.A. Routine surveillance of the operability of the concrete cask heat removal system is described in Section 2.A.3.3. As shown in Table 4.A.3-3 the storage system provides adequate heat removal through the passive cooling design features described in Section 1.A.2.1.2.

Table 1.A.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)] 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)] 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)] The DCSS must be designed to provide redundant sealing of confinement systems. [10 CFR 72.236(e)] 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)] 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)]	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.A.3.2.1, while the radiation shielding design features are described in Section 2.A.3.5. Section 10.A.4 presents the necessary minimum site boundary distances from an array of loaded storage systems to meet the controlled area dose limits. As stated in Section 10.A.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 redundant sealing features of the confinement system are presented in Section 2.A.3.2.1. As described in Section 2.A.3.1, the system is passive and can operate through all postulated normal, off-normal, and accident events while maintaining safe storage conditions for the fuel.

Table 1.A.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.A.3.4.

Table 1.A.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 facilities. [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 Appendix 8.A. The system is compatible with wet and dry spent fuel loading and unloading 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.A.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.A.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.A.3.5.3. Fuel assembly specifications are provided in Appendix 12.B, Section B2.0 of the Technical Specifications to ensure that doses from effluents and direct radiation are maintained ALARA.

Table 1.A.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 of the system are listed in Appendix 9.A, including the associated commitment, industry standard or NRC regulation.

Table 1.A.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(l)] 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.A.4.

Table 1.A.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.A.2. Description of the structural design is provided in Section 3.A.1.1.
	10 CFR 72.24(c)(4) Contents of Application: Applicable Codes and Standards	The applicable codes and standards are specified in Sections 3.A.1.1 and 3.A.1.2.
	10 CFR 72.122(a) Overall Requirements: Quality Standards	The quality standards of the system are provided in Table 2.A.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 Section 3.A.4.4. Off-normal and accident loads are evaluated in Sections 11.A.1 and 11.A.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.A.2.

Table 1.A.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 Section 3.A.4.4. Off-normal and accident condition margins of safety are presented in Sections 11.A.1 and 11.A.2, respectively. Adequate safety margins are maintained for all events, ensuring the mitigation of accident consequences, and 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.A.3.4 and 6.A.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.A.4.
	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.A.4, 10.A.3 and 10.A.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.A.2 and 7.A.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.A.4.4, 11.A.1 and 11.A.2. Operating procedures for removing spent fuel from the system are presented in Sections 8.A.2 and 8.A.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.A.2.1.1, the system is designed for a seismic event that is greater than regulatory requirements.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 3 – Structural Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
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.A.1 and Table 2.A-1 specify a 50-year design life for the system.
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).	A general description of the Vertical Concrete Cask (VCC) is provided in Section 1.A.2.1.2.
	10 CFR 72.24(c) Contents of Application: Design Criteria, Design Bases, Component Descriptions, Codes and Standards	The design criteria for the VCC is presented in Table 2.A-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.A.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.A.1.2 and Section 3.1.2, the canister and basket structure are designed in accordance with the ASME Code, Section III, Division I, 1995 Edition, 1995 Addenda. The canister is designed using Subsection NB from the code, while the basket structure is designed using Subsection NG criteria. A list of alternatives from the ASME Code is provided in Appendix 12.B, Table B.3-1 of the Technical Specifications.

Table 1.A.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 NAC-MPC 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.A.1.2 and Section 3.1.2, the VCC 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 NAC-MPC system has no concrete structures other than that addressed in #2 above.

Table 1.A.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 NAC-MPC system are evaluated in accordance with NUREG-0612 and ANSI N14.6, as specified in Section 3.A.1.2 and Section 3.1.2.

Table 1.A.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.A.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.A.1 and Table 2.A-1 specify a 50-year design life for the system. Table 4.A.3-3 demonstrates that the concrete 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.	Table 4.A.3-3 demonstrates 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.A.3. Table 4.A.3-3 demonstrates 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. The temperatures of the system are maintained within allowable limits, and do not preclude retrieval of spent fuel from the system. As specified in the Technical Specifications, Section A3.1.6, the air temperatures of the outlet vents and ISFSI ambient are measured to verify operation of the heat removal system of the concrete casks or the air inlet and outlet screens are visually inspected to ensure that they are unobstructed. Section 1.A.1 and Table 2.A-1specify a 50-year design life for the system. Table 4.A.3-3 demonstrates that the concrete temperatures are maintained within their allowable limits. The operating procedures for the system are presented in Appendix 8.A. The system is compatible with wet or dry spent fuel loading and unloading facilities.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 4 – Thermal Evaluation		
Area	**Acceptance Criteria**	***Description of Compliance***
1. Long-term Cladding Temperatures	Fuel cladding (Zircaloy) temperature at the beginning of dry cask storage should generally be below the anticipated damage-threshold temperatures for normal conditions and a minimum of 20 years of cask storage (Refs. 13 and 14). Ref 13: UCID-21181, "Spent Fuel Cladding Integrity During Dry Storage" Ref 14: PNL-6189, "Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircaloy Clad Fuel Rods in Inert Gas"	As shown in Table 4.A.3-3 the fuel cladding temperatures are maintained below 806°F for stainless steel-clad MPC-LACBWR fuel. This temperature is within the recommended temperature limits for stainless steel-clad fuel (EPRI TR-106440) for long-term conditions.
2. Short-Term Cladding Temperatures	Fuel cladding temperature should generally be maintained below 430°C (806°F) for short-term accident conditions, short-term off-normal conditions, and fuel transfer operations (e.g., vacuum drying of the cask or dry transfer). (PNL-4835)	As shown in Table 4.A.3-3, the fuel cladding temperature for stainless steel are maintained below 806°F for MPC-LACBWR short-term off-normal or accident condition events.
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 normal condition pressure calculation is presented in Section 4.A.3.5. The accident condition pressure calculation is presented in Section 11A.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.	Table 4.A.3-3 demonstrates 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	For each fuel type proposed for storage, the DCSS should ensure a very low probability (e.g., 0.5 percent per fuel rod) of cladding breach during long-term storage.	As concluded in EPRI TR-106449 (stainless steel), the probability of cladding breech is very low when the cladding temperature is maintained below allowable limits.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 4 – Thermal Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
6. Long-Term Cladding Damage	Fuel cladding damage resulting from creep cavitation should be limited to 15 percent of the original cladding cross-sectional area during dry storage. (UCID-21181)	The maximum fuel cladding temperatures are determined in accordance with EPRI TR-106440 (Stainless Steel).
Passive Cooling	The cask system should be passively cooled. [10 CFR 72.236(f)]	As stated in Sections 1.A.2 and 4.A.3, 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 VCC for normal conditions are provided in Section 3.A.4.4. The canister is evaluated for off-normal thermal stresses in Section 11.A.1.4.2, and the VCC is analyzed for accident thermal stresses in Section 11.A.2.10.

Table 1.A.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.A.2. A detailed description of the shielding features of the system are provided in Section 5.A.1.

Table 1.A.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.A.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.A.2.2. Specific details of the dose rates due to the tip-over accident are presented in Section 11.A.2.12.3.
	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.A.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.A.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.A.1 and 11.A.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.A.2.5. The radiological consequences of an explosion are provided in Section 11.A.2.3.
	10 CFR 72.128(a)(2) Criteria for Spent Fuel Storage and Handling: Radiation Protection	The dose rate results demonstrating the radiation protection of the system are presented in Section 5.A.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.A.4. The accident condition doses are discussed in Section 10.A.2.2.

Table 1.A.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.	Section 10.A.4 presents the controlled area boundary dose rate evaluation for a representative array configuration.
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.A.4 presents the controlled area boundary dose rate evaluation for a representative array configuration.
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.A.1. System design and operations are based on ALARA principles as discussed in Section 10.A.1.
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.A.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 doses for typical loading operations are provided in Section 10.A.3. In practice, occupational doses would be controlled on a site-specific basis by the operator of the ISFSI.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 6 – Criticality Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
Criticality Control	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.A.2, with a detailed description of the criticality safety features of the system provided in Appendix 6.A.
	10 CFR 72.24(d) Contents of Application: Margins of Safety / Mitigation of Accident Consequences	Section 6.A.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.A.3.4 and 6.A.1.
	10 CFR 72.236(c) Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Subcritical Configuration	Section 6.A.4 presents the results of the criticality evaluation of the storage cask for the most reactive credible conditions.
	10 CFR 72.236(g) Specific Requirements for Spent Fuel Storage Cask Approval: Minimum 20-year Lifetime	Section 1.A.1 and Table 2.A-1 specify a 50-year design life for the system.

Table 1.A.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 Sections 6.A.1 the maximum allowable multiplication factor for the system is less than 0.95, including adjustment for all biases and uncertainties.
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.A.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. Therefore, criticality cannot occur unless two separate events, such as (1) misloading a higher than design-basis enrichment, unirradiated fuel assembly and (2) water intrusion, occur.
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.A.1, the criticality safety of the design is based on geometry and fixed neutron poisons. The continued efficacy of the neutron poison material required by 10 CFR 72.124(b) is assured by the vacuum drying and atmosphere inerting that occurs in the canister sealing process. These steps remove free water and gases that could potentially degrade the aluminum and ensure the continued performance of the neutron poison material in storage. Further, the aluminum that covers the B_4C material experiences only very limited reaction with water and air environments (See Section 3.4.1.2.3). Demonstration of performance prior to use is provided for in Section 9.A.1.6.
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.A.3.2 provides a list of 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 Boral plates is used. Also, no integral fuel burnable neutron absorbers, nor fission product neutron poisons, are considered in the analysis.

Table 1.A.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.A.2, with a detailed description of the confinement features of the system provided in Section 7.A.1.1.
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.A.2.1, 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.A.1.4, 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 I component designed and fabricated in accordance with the ASME Code, Section III, Subsection NB. It is closed by a fully welded, redundant closure system. Therefore, in accordance with 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(i)]	As monitoring is not required, no instrumentation and controls are needed.
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.A.2.1 and 7.A.3, the leaktight integrity of the confinement boundary is maintained during all postulated normal and accident condition events. Therefore, there is no release of radionuclides to the environment.

Table 1.A.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.A.4.4, and for off-normal, and accident conditions in Sections 11.A.1 and 11.A.2, respectively.
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.A.4.

Table 1.A.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.A.1.4, the canister is sealed after loading by means of a redundant lid 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.A.1.1. ASME Code Section III, Subsection NB is utilized for the design 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.A.2.1 and 7.A.3, leakage from the confinement system under normal, off-normal, and accident conditions is not credible because the canister is demonstrated to be leaktight.

Table 1.A.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.A.1.1. Therefore, no monitoring system is required. Routine surveillance is required as specified in Section A3.1.6 of the Technical Specifications.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 7 – Confinement Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
5. Nonreactive 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 nonreactive environment within the confinement cask typically include draining, vacuum drying and backfilling with a nonreactive 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 Section 7.A.1.1, the confinement system is vacuum dried and backfilled with inert helium gas during loading operations.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 8 – Operating Procedures		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Health and Safety	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 Appendix 8.A Notes and Cautions are listed among the steps provided to emphasize steps important to maintaining health and safety.
2. ALARA	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)]	Appendix 8.A 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. Section A3.2.2 of the Technical Specifications specifies average external dose rates to maintain reasonable dose level within a cask array for routine surveillance and inspection activities.
3. Control of Radioactive Effluents	The applicant must describe all equipment and processes used to maintain control of radioactive effluents. [10 CFR 72.24(l)(2)]	As described in Sections 7.A.1.1, 7.A.2.1, and 7.A.3, 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.
4. Written Procedures	The general licensee shall conduct activities related to storage of spent fuel in accordance with written procedures. [10 CFR 72.212(b)(9)] 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 Appendix 8.A. These procedures are intended to provide general operational guidance for use of the system. These procedures would be used by an ISFSI operator to develop detailed, site specific procedures for use of the system.
5. Wet or Dry Loading and Unloading Facilities	The cask must be compatible with wet or dry spent fuel loading and unloading facilities. [10 CFR 72.236(h)]	The operating procedures are provided in Appendix 8.A. The system is compatible with wet or dry spent fuel loading and unloading facilities.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 8 – Operating Procedures		
Area	**Regulatory Requirement**	**Description of Compliance**
6. Decontamination Features	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 2A.3.5.3. As described in Section 8.A.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.
7. Ready Retrieval of Spent Fuel	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 Sections 8.A.2 specifies the steps necessary for retrieval of the spent fuel from the system for further processing or disposal.
8. Radioactive Waste Generation	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.
9. Inspection, Maintenance, and Testing	The design of structures, systems, and components (SSCs) that are important to safety must permit inspection, maintenance, and testing. [10 CFR 72.122(f)]	Sections 9.A.2 and 9.A.3 specify the inspection and maintenance activities required for the system.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Scope of Application	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 Appendix 8.A cover all planned operations of the system, including loading of spent fuel, placement of the system at the site, and unloading of the system.
2. Process Control and Hazard Mitigation	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 Appendix 8.A include Notes and Cautions to indicate steps important in maintaining safety.
3. Operating Controls and Limits	Operating procedure descriptions should ensure conformance with the applicable operating controls and limits described in the technical specifications provided in SAR Section 12.	The operating controls and limits specified in the Technical Specifications are referred to in the appropriate steps of the procedures in Appendix 8.A.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
4. Operational Planning	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.A, 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.A.1.1 and 8.A.3 include steps to preclude releases of radioactive material during loading and unloading operations. As stated in Sections 7.A.2.1 and 7.A.3, release of radioactive material from the system under normal, off-normal and accident conditions is not a credible event.
	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.A.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.A.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 Appendix 8.A and the previous cask loading and unloading experience of NAC support the calculation of occupational dose rates presented in Section 10.A.3.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
5. 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	Sections 9.A.2 and 9.A.3 specify the inspection and maintenance activities required for the system during storage. The limits established in Appendix 12.A, Section A3.0, of the Technical Specifications 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.
	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.).	These activities include contingency actions that may be required as a result of the inspection.
6. 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 12.A, Sections A3.1.1, A3.1.2 and A3.1.3, of the Technical Specifications, the canister is vacuum dried to eliminate water and backfilled with inert helium gas during fuel loading operations to protect the fuel cladding against oxidation.

Table 1.A.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)]	Sections 9.A.1 and 9.A.2 present the acceptance testing and criteria for the system.
	b. The cask design must permit maintenance as required. [10 CFR 72.236(g)]	Section 9.A.3 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.A.1, 9.A.2 and 9.A.3 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.A.1, 9.A.2 and 9.A.3 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.A.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.A.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.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Testing and Maintenance	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)] 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)]	Records of maintenance activities would be maintained by the ISFSI user, and thus are not applicable. As specified in Section 9.A.2.9, each system is to be marked with the model number, unique cask number, empty weight, and additional information
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.A.1 and 3.A.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.A.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.A.2.9, each system is to be marked with the model number, unique cask number, empty weight, and additional information.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

Chapter 9 – Acceptance Tests and Maintenance Program		
Area	**Acceptance Criteria**	**Description of Compliance**
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 N14.5-1987)	As specified in Sections 3.A.1.2 and 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 12.B, Table B.3-1 of the Technical Specifications. The confinement system is leak tested in accordance with ANSI N14.5 following closure lid welding as specified in Appendix 12.A, Section A3.1.5, of the Technical Specifications.
Confinement Internals (e.g., basket)	ASME B&PV Code, Section III, Subsection NG	As specified in Section 3.A.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 Sections 3.A.1.2 and 3.1.2, the Vertical 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.A.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.A.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	Occupational doses for typical loading operations are provided in Section 10.A.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.A.4. Section 10.A.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, including skin.

Table 1.A.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.A.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.A.2.
	10 CFR 72.126(a) Criteria for Radiological Protection: Exposure Control	Operational methods utilized to provide radiation protection are discussed in Section 10.A.1.3.

Table 1.A.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.A.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	Occupational doses for typical loading operations are provided in Section 10.A.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 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. 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.	The controlled area boundary dose calculations and minimum controlled area boundary distances under normal conditions are presented in Section 10.A.4. 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. Section 10.A.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, including skin.

Table 1.A.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.A.1. The operating procedures provided in Appendix 8.A are developed to keep occupational doses ALARA.

Table 1.A.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.A.1 and 11.A.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.A.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.A.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, including 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.A.4 presents the results of the criticality evaluation of the storage cask for the most reactive credible 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.A.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.A.1 and 11.A.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.A.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 and is leak tested to leaktight criteria as specified in Appendix 12.A of the Technical Specifications, LCO 3.1.5. 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.A.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.A.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.A.2.2 indicates that the controlled area boundary dose as a result of an accident will not exceed 5 rem to any organ, including 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 reactive credible conditions, including the consequences of the off-normal and accident condition events evaluated in Sections 11.A.1 and 11.A.2, respectively. As stated in Section 6.A.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. Therefore, criticality cannot occur unless two separate events, such as (1) misloading a higher than design-basis enrichment, unirradiated fuel assembly and (2) water intrusion, occur.

Table 1.A.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.A.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.A.1 and 11.A.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 utilizes instrumentation and control systems for routine inspection and surveillance to verify operation of the system, but does not utilize instrumentation and control systems that must remain operational under accident conditions.

Table 1.A.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 12.B, Section B2.0.
b. limiting conditions	Limiting conditions for operation are specified in Appendix 12.A, Section A3.0.
c. surveillance requirements	Surveillance requirements are specified in Appendix 12.A, Section A3.0.
d. design features	Design features are specified in Appendix 12.B, Section B3.0.
e. administrative controls	Administrative controls are specified in Appendix 12.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 12.A are intended to ensure that the system is maintained within its design basis through all normal, off-normal, and accident conditions.

Table 1.A.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 h. condition of the spent fuel (i.e., intact assembly or consolidated fuel rods) i. inerting atmosphere requirements	Specifications for the spent fuel contents are provided in Appendix 12.B, Tables B.2-1 through B.2-4 of the Technical Specifications. As specified in Appendix 12.A, Section A3.1.3, of the Technical Specifications, 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 Appendix 2.A.
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.A.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 12.A, Section A3.2.2 of the Technical Specifications. These limits are established to ensure that, for the minimum controlled area boundary distance presented in Section 10.A.4, the controlled area boundary annual dose will be maintained within allowable limits.

Table 1.A.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)] • Provide adequate heat removal capacity without active cooling systems. [10 CFR 72.236(f)] • Safely store the spent fuel for a minimum of 20 years and permit maintenance as required. [10 CFR 72.236(g)] • Facilitate decontamination to the extent practicable. [10 CFR 72.236(i)]	 The redundant sealing features of the confinement system are presented in Section 2.A.3.2.1 and Appendix 7.A. As shown in Table 4.A.3-3, the system provides adequate heat removal through the passive cooling design features described in Section 4.3. Section 1.A.1 and Tables 2.A-1 and 2.A-2 specify a 50-year design life for the system. Routine maintenance is permitted as specified by Section 9.2. Decommissioning of the system is discussed in Section 2.A.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 Appendix 8.A. The system is compatible with wet or dry spent fuel loading and unloading facilities.
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.A.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.A.4.4, and for off-normal and accident conditions in Sections 11.A.1 and 11.A.2, respectively. Because the canister maintains adequate positive margins of safety, the system will reasonably maintain confinement under all credible conditions.

Table 1.A.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 Quality Assurance Program is approved by the NRC under 10 CFR 71, Subpart H.

Table 1.A.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.A.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.A.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.A.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.A.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.A.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 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.

Table 1.A.5-1 NUREG-1536 Compliance Matrix (continued)

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.A.4.
2.	Classification and disposal of wastes, as contained in 10 CFR 61.55.	Not applicable.

1.A.6 Agents and Contractors

The prime contractor for the MPC-LACBWR design is NAC International (NAC). All design and specification activities are performed by NAC. Fabrication of the steel components will be by qualified vendors. A qualified concrete contractor will perform fabrication of the vertical concrete storage cask. All fabrication activities will be performed in accordance with quality assurance programs meeting the requirements of 10 CFR 71 and 10 CFR 72.

NAC was founded as a private corporation in 1968, with the primary focus of tracking, inspection, handling, storage, 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 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 25 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)**
8 NAC-LWT	7 UMS®/MPC transfer casks
16 TRUPACT-II	2 NAC-I28 S/T metal casks
6 RH-TRU 72B	1 NAC-I26 S/T metal cask
2 NAC-STC	> 210 UMS®/MPC TSCs
	> 212 UMS®/MPC concrete casks

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1.A.7 MPC-LACBWR License Drawings

This section presents the License Drawings for the MPC-LACBWR System.

Drawing Number	Title	Revision No.	No. of Sheets
630045-861	Weldment, Structure, Vertical Concrete Cask (VCC), MPC-LACBWR	1	3
630045-862	Loaded Vertical Concrete Cask (VCC), MPC-LACBWR	0	1
630045-863	Lid Assembly, Vertical Concrete Cask (VCC), MPC-LACBWR	0	1
630045-864	Name Plate, Vertical Concrete Cask (VCC) MPC-LACBWR	0	1
630045-866	Reinforcing Bar and Concrete Placement, Vertical Concrete Cask (VCC), MPC-LACBWR	1	5
630045-870	Shell Weldment Canister (TSC), MPC-LACBWR	0	1
630045-871	Details TSC, MPC-LACBWR	0	4
630045-872	Assembly, Transportable Storage Canister (TSC), MPC-LACBWR	0	2
630045-873	Assembly, Drain Tube, TSC, MPC-LACBWR	0	1
630045-877	Bottom Weldment, Fuel Basket, MPC-LACBWR	0	1
630045-878	Top Weldment, Fuel Basket, MPC-LACBWR	0	1
630045-881	Fuel Tube Assembly, MPC-LACBWR	0	2
630045-893	Support Disk, Fuel Basket, MPC-LACBWR	0	1
630045-894	Heat Transfer Disk, Fuel Basket, MPC-LACBWR	0	1
630045-895	Fuel Basket Assembly, 68 Element BWR, MPC-LACBWR	0	3
630045-901	Assembly, Damaged Fuel Can (DFC), MPC-LACBWR	0	1
630045-902	Details, Damaged Fuel Can (DFC), MPC-LACBWR	0	2

Note: The Transfer Adapter and Transfer Cask License Drawings No. 455-859 and 455-860, respectively, are included in the MPC FSAR, Section 1.7.1.

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Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

WELDMENT, STRUCTURE,
VERTICAL CONCRETE CASK (VCC),
MPC–LACBWR

PROJECT 630045		DRAWING 861	REV 1
SCALE N.T.S.	EST.WT. AS NOTED	SH 1 OF 3	7:46AM 1-14-2009

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

WELDMENT, STRUCTURE, VERTICAL CONCRETE CASK (VCC), MPC-LACBWR

PROJECT 630045		DRAWING 861	REV 1
SCALE N.T.S.	WEIGHT AS NOTED	SH 2 OF 3	7:47AM 1-14-2009

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

WELDMENT, STRUCTURE, VERTICAL CONCRETE CASK (VCC), MPC-LACBWR

PROJECT 630045		DRAWING 861	REV 1
SCALE N.T.S.	WEIGHT AS NOTED	SH 3 OF 3	7:49AM 1-14-2009

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LOADED VERTICAL
CONCRETE CASK (VCC)
MPC-LACBWR

PROJECT 630045		DRAWING 862	REV 0
SCALE N.T.S.	EST.WT. N/A	SH 1 OF 1	7:29AM 12-29-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LID ASSEMBLY,
VERTICAL CONCRETE CASK (VCC),
MPC–LACBWR

PROJECT 630045		DRAWING 863	REV 0
SCALE N.T.S.	EST.WT. AS NOTED	SH 1 OF 1	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

NAMEPLATE
VERTICAL CONCRETE CASK (VCC)
MPC-LACBWR

PROJECT 630045		DRAWING 864	REV 0
SCALE N.T.S.	EST.WT. N/A	SH 1 OF 1	1:22PM 12-29-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

REINFORCING BAR AND
CONCRETE PLACEMENT,
VERTICAL CONCRETE CASK (VCC)
MPC–LACBWR

PROJECT 630045		DRAWING 866	REV 1
SCALE N.T.S.	EST.WT. N/A	SH 1 OF 5	3:02PM 1-14-2009

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC), MPC–LACBWR

PROJECT 630045		DRAWING 866	REV 1
SCALE N.T.S.	WEIGHT N/A	SH 2 OF 5	2:59PM 1–14–2009

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) MPC–LACBWR

PROJECT 630045		DRAWING 866	REV 1
SCALE N.T.S.	WEIGHT N/A	SH 3 OF 5	2:58PM 1-14-2009

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

REINFORCING BAR AND
CONCRETE PLACEMENT,
VERTICAL CONCRETE CASK (VCC)
MPC-LACBWR

PROJECT 630045		DRAWING 866		REV 1
SCALE N.T.S.	WEIGHT N/A	SH 4 OF 5		2:57PM 1-14-200

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

REINFORCING BAR AND CONCRETE PLACEMENT, VERTICAL CONCRETE CASK (VCC) MPC-LACBWR

PROJECT 630045		DRAWING 866	REV 1
SCALE N.T.S.	WEIGHT N/A	SH 5 OF 5	3:04PM 1-14-2009

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

SHELL WELDMENT,
CANISTER (TSC),
MPC-LACBWR

PROJECT 630045		DRAWING 870			REV 0
SCALE N.T.S.	EST.WT. AS NOTED	SH 1	OF 1		2:30PM 12-29-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

DETAILS TSC,
MPC-LACBWR

PROJECT 630045		DRAWING 871	REV 0
SCALE NTS	EST.WT. AS NOTED	SH 1 OF 4	8:23AM 12-17-2005

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

DETAILS TSC,
MPC-LACBWR

PROJECT 630045		DRAWING 871	REV 0
SCALE N.T.S.	WEIGHT N/A	SH 2 OF 4	8:26AM 12-17-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

DETAILS TSC,
MPC–LACBWR

PROJECT 630045		DRAWING 871		REV 0
SCALE N.T.S.	WEIGHT NOTED	SH 3 OF 4		8:26AM 12-17-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

DETAILS TSC,
MPC-LACBWR

PROJECT 630045		DRAWING 871	REV 0
SCALE N.T.S.	WEIGHT NOTED	SH 4 OF 4	3:50PM 12-29-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY, TRANSPORTABLE STORAGE CANISTER (TSC), MPC–LACBWR

PROJECT 630045		DRAWING 872	REV 0
SCALE NTS	EST.WT. AS NOTED	SH 1 OF 2	2:37PM 12-29-2006

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL				
ASSEMBLY, TRANSPORTABLE STORAGE CANISTER (TSC), MPC-LACBWR				
PROJECT 630045		DRAWING 872		REV 0
SCALE N.T.S.	WEIGHT AS NOTED	SH 2 OF 2		2:33PM 12-29-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
DRAIN TUBE TSC,
MPC–LACBWR

PROJECT 630045		DRAWING 873		REV 0
SCALE	EST.WT. 12.5#	SH 1 OF 1	6:28AM 12-17-2008	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BOTTOM WELDMENT,
FUEL BASKET,
MPC–LACBWR

PROJECT 630045		DRAWING 877	REV 0
SCALE N.T.S.	EST.WT. 485#	SH 1 OF 1	10:30AM 12-29-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

TOP WELDMENT,
FUEL BASKET,
MPC-LACBWR

PROJECT 630045		DRAWING 878	REV 0
SCALE NTS	EST.WT. 600#	SH 1 OF 1	2:37PM 12-29-2006

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL TUBE ASSEMBLY,
MPC–LACBWR

PROJECT 630045		DRAWING 881	REV 0
SCALE N.T.S.	EST.WT. AS NOTED	SH 1 OF 2	10:03AM 12–30–2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL TUBE ASSEMBLY,
MPC–LACBWR

PROJECT 630045		DRAWING 881		REV 0
SCALE N.T.S.	WEIGHT AS NOTED	SH 2	OF 2	10:04AM 12-30-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

SUPPORT DISK,
FUEL BASKET,
MPC–LACBWR

PROJECT 630045		DRAWING 893	REV 0
SCALE N.T.S.	EST.WT. 204#	SH 1 OF 1	9:53AM 12-30-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

HEAT TRANSFER DISK,
FUEL BASKET,
MPC-LACBWR

PROJECT 630045		DRAWING 894	REV 0
SCALE N.T.S.	EST.WT. 68#	SH 1 OF 1	9:55AM 12-30-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL BASKET ASSEMBLY,
68 ELEMENT BWR
MPC–LACBWR

PROJECT 630045		DRAWING 895		REV 0
SCALE N.T.S.	EST.WT. 14,600#	SH 1 OF 3		10:07AM 12-19-2005

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL BASKET ASSEMBLY,
68 ELEMENT BWR
MPC-LACBWR

PROJECT 630045		DRAWING 895	REV 0
SCALE N.T.S.	WEIGHT AS NOTED	SH 2 OF 3	11:56AM 12-19-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL BASKET ASSEMBLY,
68 ELEMENT BWR
MPC-LACBWR

PROJECT 630045		DRAWING 895	REV 0
SCALE N.T.S.	WEIGHT AS NOTED	SH 3 OF 3	10:07AM 12-19-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
DAMAGED FUEL CAN (DFC),
MPC–LACBWR

PROJECT 630045		DRAWING 901	REV 0
SCALE N.T.S.	EST. WT. 55#	SH 1 OF 1	8:33AM 12–17–2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

DETAILS,
DAMAGED FUEL CAN (DFC),
MPC-LACBWR

PROJECT 630045		DRAWING 902	REV 0
SCALE N.T.S.	EST.WT. 45#	SH 1 OF 2	2:58PM 12-29-2008

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

DETAILS,
DAMAGED FUEL CAN (DFC),
MPC–LACBWR

PROJECT 630045		DRAWING 902	REV 0
SCALE N.T.S.	WEIGHT AS NOTED	SH 2 OF 2	3:06PM 12-29-2008

Table of Contents

Table of Contents (Continued)

2.0 PRINCIPAL DESIGN CRITERIA

The NAC-MPC is a canister-based dry storage cask system that is designed to be transported in the NAC-STC licensed transport cask.

This chapter presents the design basis, including the principal design criteria, limiting load conditions, and operational parameters of the NAC-MPC dry storage system. The NAC-MPC is provided in three configurations. The Yankee-MPC for Yankee Class spent fuel, the CY-MPC for Connecticut Yankee spent fuel, and MPC-LACBWR for Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) spent fuel. The principal design criteria for the Yankee-MPC system are described in Table 2-1. The CY-MPC system criteria are presented in Table 2-2. The principal design criteria for MPC-LACBWR system are described in Table 2.A-1 in Appendix 2.A.

The design criteria for the spent fuel to be stored in the Yankee-MPC and CY-MPC configurations are described in Section 2.1. Except as noted, the design criteria presented in Section 2.2, the Safety Protection Systems described in Section 2.3, and the Decommissioning Considerations discussed in Section 2.4, apply to both configurations.

The design criteria for the spent fuel to be stored in the MPC-LACBWR configuration are described in Section 2.A.1.

Table 2-1 Summary of the Yankee-MPC Design Criteria

Yankee-MPC Design Criteria	
Design Life	50 years
Design Code - Confinement	ASME Code, Section III, Subsection NB for confinement boundary
Design Code - Nonconfinement	
Basket	ASME Code, Section III, Subsection NG and NUREG/CR-6322
Vertical Concrete Cask	ACI-349, ACI-318, ANSI/ANS 57.9
Transfer Cask	ANSI N14.6 and NUREG-0612
Design Weight:	
Canister Assembly with fuel	54,730 lbs.
Transfer Cask	80,743 lbs.
Vertical Concrete Cask	151,364 lbs.
Thermal:	
Maximum Temperature, Zircaloy Cladding	340°C for 10-yr. Cooled 380°C for 5-yr. Cooled 430°C Off-Normal/Accident/Transfer
Maximum Temperature, Stainless Steel Cladding	340°C for 10-yr. Cooled 430°C Off-Normal/Accident/Transfer
Ambient Temperature Range	-40° to 125°F
Average Annual Ambient Temperature	75°F
Concrete Temperature:	
Normal Conditions	≤ 150°F; ≤ 200°F local
Off-Normal/Accident Conditions	≤ 350°F local/ surface
Canister Cavity Atmosphere	Helium

2.1 Spent Fuel To Be Stored

The NAC-MPC is provided in three configurations. The Yankee Class MPC (Yankee-MPC) is designed to store up to 36 Yankee Class spent fuel assemblies. The Connecticut Yankee MPC (CY-MPC) is designed to store up to 26 Connecticut Yankee spent fuel assemblies, but is provided with either a 26-assembly or a 24-assembly basket. The Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) is designed to store up to 68 LACBWR spent fuel assemblies including up to 32 LACBWR damaged fuel cans. The spent fuel to be stored in MPC-LACBWR is described in Section 2.A.1. The spent fuel assemblies stored in all configurations are delineated by various factors including manufacturer, type, enrichment, burnup, cool time, and cladding material.

The Yankee Class fuel consists of two types, designated A and B. The Type A assembly incorporates a protruding corner of fuel rods while the Type B assembly omits one corner of the fuel rods. These fuel types, as well as minor differences among manufacturers, are illustrated in Figures 6.2.1-1 through 6.2.1-3. During reactor operations, the symmetric stacking of the alternating assemblies permitted the insertion of cruciform control blades between the assemblies. Table 2.1-1 lists the nominal design parameters and the maximum and minimum enrichments of each fuel design type. These parameters exclude the loading of unenriched fuel assemblies in the transportable storage canister. Not listed in the table are the various inert rod configurations employed in the CE and Exxon fuel types. Yankee Class fuel is described in Section 2.1.1.

There are three configurations of the basket. The first configuration includes 36 fuel tubes with BORAL on all four sides (standard fuel tubes). The second configuration includes 32 fuel tubes with BORAL on all four sides and four enlarged fuel tubes located in the four corner positions without BORAL as shown in Figure 2.1-1. The third configuration includes 32 fuel tubes with BORAL on all four sides and four damaged fuel cans without BORAL located in the four corner positions. The enlarged fuel tubes have a larger cross-section than the standard fuel tubes to accommodate the loading of fuel assemblies that exhibit slight physical deformations (e.g., bow, twist) that could preclude loading in the smaller cross-section standard fuel tubes. However, fuel assemblies that can be loaded in standard fuel tubes may also be loaded in the enlarged fuel tubes. In the damaged fuel can configuration, the basket top and bottom weldment openings at the four corner locations are enlarged, allowing the loaded or empty damaged fuel can to be removed from the basket if necessary. The damaged fuel can(s) is captured between the canister bottom plate and the shield lid configured for the damaged fuel cans. The standard and enlarged fuel tubes are captured between the basket top and bottom weldments and cannot be removed.

The Connecticut Yankee fuel is a 15 × 15 square array PWR assembly. The majority of the Connecticut Yankee fuel is stainless steel clad. About 15% of the fuel to be stored is Zircaloy clad. The 15 × 15 array incorporates 20 guide tubes for the insertion of control components. Table 2.1-3 lists the nominal design parameters of each fuel design type. The Connecticut Yankee fuel is described in Section 2.1.2.

The stored fuel assemblies must be structurally intact, or separately packaged, so that special handling of an assembly is not required. Intact fuel assemblies may not have cladding defects greater than pin holes or hairline cracks. Unenriched fuel assemblies may not be stored in the NAC-MPC system.

The short-term and long-term temperature limits for stainless steel-clad fuel are derived based on the limits presented in EPRI report TR-106440, "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage," April 1996. In this report, the potential failure modes in both wet and dry storage environments were assessed to develop the bounding conditions for the prevention of any potential cladding degradation phenomena and cladding failure modes of stainless steel clad fuel. The potential cladding degradation mechanisms evaluated include: general corrosion, stress corrosion cracking, localized corrosion, mechanical failures and chemical/metallurgical-based failure mechanisms. The EPRI report is based on several types of stainless steel cladding, including Type 304, 348, 348H and modified 348H, which includes Yankee Class and Connecticut Yankee fuel.

The long-term temperature limit is conservatively established as 340°C for the Yankee-MPC System for both stainless steel and Zircaloy clad fuel. This value is significantly lower than the 430°C temperature limit defined by the EPRI report for stainless steel cladding. The short-term temperature limit for stainless steel clad fuel is 430°C for the Yankee-MPC System. This short-term limit is also conservatively used for Zircaloy clad fuel. The CY-MPC cladding temperature limits are shown in Table 2-2 and described in Section 4.5.7.

2.1.1 Yankee Class Fuel - Bounding Fuel Evaluation

The criticality evaluations show that the United Nuclear Type A 16 × 16 fuel assembly is the most reactive fuel type, even though the stainless steel clad Westinghouse fuel has a higher enrichment (4.94 wt % ^{235}U). The criticality evaluation considers a complete assembly fuel rod matrix. Consequently, either solid rods fabricated from Zircaloy or stainless steel, or hollow Zircaloy rods with either Zircaloy or stainless steel slugs, must replace any fuel rod that is removed from an intact fuel assembly.

The shielding evaluations show that the Combustion Engineering Type A has the largest dose rates. The United Nuclear assemblies are evaluated for a source term based on a nominal initial enrichment of 4.0 wt %, a maximum burnup of 32,000 MWd/MTU, and a minimum cool time of 13 years after reactor discharge. Exxon fuel with Zircaloy or stainless steel hardware is evaluated at 36,000 MWd/MTU and 10 or 16 years' cool time, respectively, with a nominal initial enrichment of 3.5 wt % ^{235}U. Westinghouse fuel is evaluated at 32,000 MWd/MTU and a 24-year cool time with a nominal initial enrichment of 4.94 wt % ^{235}U. Combustion Engineering fuel is evaluated for a source term based on an initial nominal enrichment of 3.7 wt % ^{235}U, a maximum burnup of 36,000 MWd/MTU, and a minimum cool time of 8.1 years after reactor discharge. For Combustion Engineering assemblies at a maximum burnup of 32,000 MWD/MTU and nominal initial enrichment of 3.5 wt % ^{235}U, a minimum cool time of 8 years is required.

The NAC-MPC maximum decay heat load is 12.5 kilowatts. This results in a maximum decay heat load for the design basis fuel assemblies of 0.347 kilowatt per assembly, based on 36 fuel assemblies per canister.

2.1.1.1 Yankee-MPC Reconfigured Fuel Assembly

One or more transportable storage canisters may hold Yankee-MPC Reconfigured Fuel Assemblies containing up to 64 intact or damaged whole spent fuel rods, rod segments or fuel debris. The fuel rods, rod segments or debris are held in individual tubes in an 8 x 8 array. Damaged fuel rods, rod segments or fuel debris must be loaded in the Reconfigured Fuel Assembly. The array of tubes is positioned in a stainless steel container having the same external dimensions as a standard fuel assembly. It has a top end fitting that has the same configuration as a standard fuel assembly. The container is closed on the top and bottom ends by perforated plates, which act as a barrier to the release of gross particles to the canister, but allow the draining and drying of the container. The tubes are stainless steel and are closed on each end by a plug. Each plug has a small hole drilled through it. The perforated plate screens the drilled hole. The hole allows the draining and drying of the individual tubes during routine closing of the canister. The perforated plate precludes the release of gross particles to the canister. The effects of the reconfigured fuel assembly are evaluated in the appropriate sections. The structural, thermal, shielding, confinement, and criticality effects of the reconfigured fuel assembly are bounded by those of an intact fuel assembly.

The physical parameters of the Yankee-MPC Reconfigured Fuel Assembly are provided in Table 2.1-2.

2.1.1.2 Yankee Stainless Steel-Clad Fuel

The short-term and long-term temperature limits for stainless steel-clad fuel are derived based on the limits presented in EPRI report TR-106440, "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage," April 1996. In this report, the potential failure modes in both wet and dry storage environments were assessed to develop the bounding conditions for the prevention of any potential cladding degradation phenomena and cladding failure modes of stainless steel-clad fuel. The potential cladding degradation mechanisms evaluated include: general corrosion, stress corrosion cracking, localized corrosion, mechanical failures and chemical/metallurgical-based failure mechanisms. The EPRI report is based on several types of stainless steel cladding, including 304, 348, 348H and modified 348H, which includes Yankee Class fuel.

The long-term temperature limit is conservatively established as 340°C for the NAC-MPC System for both stainless steel and Zircaloy-clad fuel. This value is significantly lower than the 430°C temperature limit defined by the EPRI report for stainless steel cladding. The short-term temperature limit for stainless steel-clad fuel is 430°C for the NAC-MPC System. This short-term limit is also conservatively used for Zircaloy-clad fuel.

2.1.2 Connecticut Yankee Fuel – Bounding Fuel Evaluation

The criticality evaluations show that the Westinghouse Vantage 5H 15 x 15 fuel assembly with Zircaloy cladding with an enrichment of 4.61 wt % ^{235}U is the most reactive fuel type. This fuel must be loaded in the 24-assembly basket configuration as described in Section 2.1.2.1. The remainder of the Connecticut Yankee fuel inventory may be loaded into the 26-assembly basket configuration. The most reactive fuel type for the 26-assembly basket configuration is the Babcock and Wilcox Zircaloy-clad fuel enriched to 3.93 wt % ^{235}U. The criticality evaluation for Connecticut Yankee fuel considers the effect of missing fuel rods in the fuel assembly array. However, as shown in Table B2-3 of the Certificate of Compliance, fuel assemblies having missing fuel rods not replaced by filler rods, must be preferentially loaded in corner positions in the basket.

The shielding evaluations show that the stainless steel clad fuel assemblies with burnups of up to 38,000 MWD/MTU and Zircaloy-clad fuel assemblies with burnups of up to 43,000 MWD/MTU, with minimum cooling times of 5 years may be accommodated in the CY-MPC. The shielding evaluation considers the minimum initial enrichments for the fuel assemblies in the design basis analyses. For stainless steel clad fuel assemblies, the analysis considers the gamma dose rate contributions from the cobalt impurities present in the cladding.

The CY-MPC maximum decay heat load is 17.5 kilowatts. This results in a maximum decay heat load for the design basis fuel assemblies of 674 watts per assembly, assuming a uniform heat load in all assemblies and 26 fuel assemblies in each canister. Section 2.1.2.1 describes the preferential loading of fuel assemblies with variable decay heats, with a maximum individual fuel assembly decay heat of up to 840 watts.

The inventory of Connecticut Yankee fuel includes damaged fuel to be stored in CY-MPC damaged fuel cans as described in Section 2.1.2.3, and individual fuel rods to be packaged into CY-MPC reconfigured fuel assemblies as described in Section 2.1.2.2. The storage of non-fuel hardware (flow mixers and control clusters) with the fuel assemblies is described in Section 2.1.2.4.

2.1.2.1 Connecticut Yankee Preferential Fuel Loading

The CY-MPC is designed to accommodate the entire spent fuel inventory currently in storage in the Connecticut Yankee spent fuel pool. To provide the greatest flexibility in loading this spent fuel inventory, the CY-MPC is designed to accommodate certain fuel assemblies using preferential loading schemes. The first such scheme is a 24-assembly basket design (rather than the standard 26-assembly configuration) designed to accommodate the higher enrichment Westinghouse Vantage 5H (Zircaloy-clad) fuel assemblies utilized in the last cycle at Connecticut Yankee. The second scheme is intended to permit flexibility in loading fuel assemblies with decay heats higher than the average 674 watt per assembly value for the design basis 17.5 kilowatt heat load. The two preferential loading schemes are further described in Sections 2.1.2.1.1 and 2.1.2.1.2.

2.1.2.1.1 CY-MPC 24-Assembly Basket Configuration

To accommodate the limited quantity of Westinghouse Vantage 5H fuel assemblies at Connecticut Yankee, Zircaloy-clad fuel assemblies with enrichments greater than 3.93 wt % ^{235}U

must be preferentially loaded in the 24-assembly basket configuration, which is equivalent to the 26-assembly configuration with two fuel positions blocked, as shown in Figures 2.1-2 and 2.1-3.

It is important to note that, of the entire 1019 Connecticut Yankee spent fuel assembly inventory, fewer than 55 assemblies require loading in the 24-assembly configuration. These Westinghouse Vantage 5H fuel assemblies were utilized only during the last cycle of operation of the reactor, and are uniquely identifiable by Batch identifier number. Consequently, only three CY-MPC storage systems will utilize the 24-assembly basket. Administrative controls shall ensure that the entire inventory of Vantage 5H fuel assemblies are loaded into the three canisters. Additionally, controls shall be utilized to ensure that Vantage 5H fuel assemblies are not inadvertently loaded into systems containing 26-assembly baskets.

2.1.2.1.2 CY-MPC Preferential Decay Heat Loading

Preferential fuel loading is also used to control the decay heat load within a given canister. This loading allows the storage of spent fuel assemblies that have decay heats larger than the uniform design value of 674 watts. A number of spent fuel assemblies are projected to have decay heat values in excess of 674 watts at the proposed loading date of August 2002. To accommodate fuel assemblies with higher decay heats, a variable decay heat loading pattern is used, as shown in Figure 2.1-4. Using this loading pattern, fuel assemblies with decay heats of up to 840 watts can be accommodated, enveloping the decay heat of the design basis Connecticut Yankee fuel assemblies with maximum burnup and a minimum cooling time of 6 years. As shown in Figure 2.1-4, preferential loading to accommodate this higher fuel assembly heat load is restricted to the 26-assembly basket configuration. Preferential fuel loading is not required for thermal performance if each fuel assembly in the basket is less than 674 watts. The 24-assembly configuration will be loaded with fuel assemblies having decay heats below 674 watts, as the Westinghouse Vantage 5H fuel assemblies to be loaded into the 24-assembly configuration were only exposed to one cycle of operation. Other fuel assemblies to be loaded into the 24-assembly baskets will be cooled sufficiently to ensure that the decay heat is below 674 watts.

The preferential decay heat loading pattern utilizes two distinct decay heat limits of 840 watts and 600 watts. Analyses performed on the design basis Connecticut Yankee spent fuel assemblies demonstrate that for the maximum design basis burnup, fuel assemblies with 6 years cooling (≤ 840 watts) may only be loaded into fuel positions 7, 8, 12, 13, 14, 15, 19 and 20. This position restriction is conservative, as it assumes that each fuel assembly has been burned to the design basis, maximum burnup. Fuel assemblies with lower burnups will require less cooling time prior to loading. Fuel assemblies ≤ 600 watts may be loaded into any fuel position. Depending on fuel type and maximum burnup, fuel assemblies require between 5 and 12 years of cool time to meet the ≤ 600 watt load restriction.

Absent any other preferential loading requirement, fuel assemblies with the shorter cooling time (thus, a higher fuel clad temperature allowable) and higher radiation source strength are placed toward the center of the basket in order to minimize external dose rates. With this loading arrangement, the temperature of the fuel assemblies in the peripheral positions are lower than those in the center, based on the physical geometry of the system and the propagation of heat generated within the system.

This preferential decay heat loading pattern is expected to be used only on 8 of the approximately 40 NAC-MPC systems to be utilized at Connecticut Yankee. To minimize the number of systems requiring preferential decay heat loading, fuel from the last 3 core discharges (6, 7, and 9 years cooled as of August 2002) are expected to be loaded entirely into 8 canisters. Of these assemblies, approximately 111 are Zircaloy-clad, excluding the Vantage 5H fuel assemblies that will be loaded separately. Other than the 8 Zircaloy-clad fuel assemblies discharged prior to 1990, only about 100 Zircaloy-clad fuel assemblies will require preferential loading, and will have a cooling time between 6 and 7 years at the projected start of storage in August 2002. Approximately 52 Zircaloy-clad assemblies from the final core discharge were operational for only two cycles and have maximum burnups significantly less than 35,000 MWd/MTU. As such, these assemblies will have decay heats below 840 watts, and may be loaded into the 840-watt positions.

2.1.2.2 CY-MPC Reconfigured Fuel Assembly

The CY-MPC Reconfigured Fuel Assembly is considered to be preferentially loaded because it can only be installed in the fuel positions that are oversized. These are positions 1, 4, 23 and 26 of the 26-assembly basket and positions 1, 4, 21 and 24 of the 24-assembly basket, shown in Figures 2.1-2 and 2.1-3, respectively.

Individual intact or damaged fuel rods are installed in tubes in a CY-MPC Reconfigured Fuel Assembly to maintain configuration control of the rods during handling and storage. The assembly consists of a 10 × 10 array of stainless steel tubes supported on either end by stainless steel end fittings that allow it to be handled in the same way as a spent fuel assembly. The end fittings are screened to allow water to drain from the tubes holding the fuel rods, but to preclude the release of gross particulate material to the interior of the canister.

The CY-MPC Reconfigured Fuel Assembly is shown in Drawings 414-903 and 414-904, and the major physical parameters are provided in Table 2.1-4. The array of stainless steel tubes is held in place by top and bottom housings, which are attached using full-length corner angle members. The bottom housing forms a plenum region below the ends of the tubes, using a retaining plate, backing screen, and filter screen, to trap loose material that passes through the 0.2 inch holes at the end of the tube in the bottom housing. The top housing, together with a guide plate, form the top end of the assembly, that is closed by a lid weldment. The lid weldment includes a rod retaining plate with four screened holes for draining. The lid is attached to the top housing using five hex head bolts.

The CY-MPC Reconfigured Fuel Assembly accommodates the following contents:

- Intact Fuel Rods – A Connecticut Yankee spent fuel rod without known or suspected cladding defects greater than pinhole leaks or hairline cracks.
- Damaged Fuel Rods – A Connecticut Yankee spent fuel rod with known or suspected cladding defects greater than a hairline crack or a pinhole leak.
- Fuel Debris or Fuel Pellets – Any Connecticut Yankee fuel, which is outside of its original fuel rod. The Reconfigured Fuel Assembly tube ensures that the pellets or debris maintain regular array geometry.

Individual damaged fuel rods, fuel debris or fuel pellets must be loaded in the Reconfigured Fuel Assembly.

To account for the possible presence of fuel debris or pellets, the criticality evaluation of the reconfigured fuel assembly conservatively assumes that each tube of the assembly is filled with fuel enriched to 4.61 wt % ^{235}U. Consequently, fuel rod parameters of individual fuel assembly types need not be specified.

The CY-MPC Reconfigured Fuel Assembly design and fabrication specification summary is provided in Table 2.1-4. The structural evaluation is provided in Sections 3.4.4.5 and 11.4.2.

2.1.2.3 CY-MPC Damaged Fuel Can

The CY-MPC Damaged Can is considered to be preferentially loaded because it can be only installed in the fuel positions that are oversized. These are positions 1, 4, 23, and 26 of the 26-assembly basket and positions 1, 4, 21 and 24 of the 24-assembly basket.

Single fuel assemblies with damage to structural components that prevents handling in the normal manner, fuel assemblies that have one or more damaged fuel rods, lattices holding intact or damaged fuel rods, and the failed rod storage canister must be loaded in the Damaged Fuel Can. However, an intact, undamaged fuel assembly could also be loaded into the Damaged Fuel Can. The CY-MPC Damaged Fuel Can is shown in Drawings 414-901 and 414-902. As shown in the drawings, the can is 141.5 inches in length, and has an internal square dimension of 8.7 inches.

The can is closed on the bottom end by a 0.5-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 particulate material from the fuel assembly. A lid having an overall depth dimension of 2.3 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 assembly is inserted in the can and the can lid is installed. Lifting lugs in the can shell allow the loaded can to be lifted and installed in the canister basket. Alternately, the Damaged Fuel Can may be inserted in a basket corner position before the designated fuel assembly is inserted in the Damaged Fuel Can.

The CY-MPC Damaged Fuel Can design and fabrication specification summary is provided in Table 1.3-6. The major physical design parameters are provided in Table 2.1-5. The structural evaluation is provided in Sections 3.4.4.6 and 11.4.3.

2.1.2.4 Connecticut Yankee Fuel with Inserted Hardware

Connecticut Yankee fuel assemblies may have a Reactor Control Cluster Assembly (control cluster), a Flow Mixer/Thimble Plug Assembly (flow mixer) or stainless steel rods inserted in the fuel assembly. These components add weight, and the control clusters and flow mixer add gamma radiation source term to the standard fuel assembly.

The control cluster consists of 20 control rods mounted on a Type 304 stainless steel spider assembly and weighs about 140 pounds. In some fuel designs, these components are known as Control Element Assemblies (CEAs). The control rods are inserted in the fuel assembly guide tubes when the cluster is inserted in the fuel assembly. When fully inserted, the cluster 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 control clusters installed do not require preferential loading. Some fuel assemblies may have flow mixers inserted in the top of the fuel assembly for water flow control in the reactor. The mixers are an array of thimble plugs attached to a top spider assembly and weigh about 15 pounds.

Fuel assemblies with control clusters or flow mixers installed may not be loaded in a Damaged Fuel Can.

The number and positioning of control clusters and flow mixers loaded in the canister are administratively controlled to minimize external dose rates, and to maintain overall payload weights below the canister weight limit of 35,100 pounds. Flow mixers are limited to the central 8 positions in the 26-assembly basket configuration (6 positions in the 24-assembly basket configuration).

Solid stainless steel rods, approximately 21 inches long, may be inserted into Connecticut Yankee intact and damaged fuel assembly Reactor Control Cluster Assemblies (RCCA) guide tubes not containing a RCCA. The weight of the contents, including the installed stainless steel rods, will not exceed the canister weight limit of 35,100 pounds.

2.1.3 Damaged Fuel

A transportable storage canister configured for damaged fuel holds four damaged fuel cans located in the corner positions of the basket as shown in Figure 2.1-1. A damaged fuel can may contain either an intact or a damaged spent fuel assembly of the types described in Table 2.1-1, but may not contain individual fuel rods not in an assembly array. Fuel assemblies classified as damaged may have up to 20 fuel rods missing or with defects greater than pinhole leaks or hairline cracks. A canister configured for damaged fuel has a basket design that allows the damaged fuel can to be placed in the basket and a canister shield lid design that is machined on its underside to mate with the lid of the damaged fuel can. The shield lid and basket designed for the damaged fuel can cannot be used interchangeably with other canister configurations. The damaged fuel can is constructed of stainless steel and has a welded closure on its bottom end and a removable lid on the top end. The damaged fuel can lid and bottom closure are screened to allow the filling, draining and vacuum drying of the can, and to preclude the release of gross particulate to the canister during operations or storage events. The damaged fuel can has the same cross-section dimensions as the enlarged fuel tube and does not have attached neutron absorber plates on its exterior. The corner fuel positions of the basket top and bottom weldments are enlarged so that the damaged fuel can may be removed from the basket if necessary. The can is captured between the canister shield lid and bottom plate to limit axial movement. The structural, thermal, shielding, confinement, and criticality effects of the damaged fuel cans are separately evaluated in the appropriate sections.

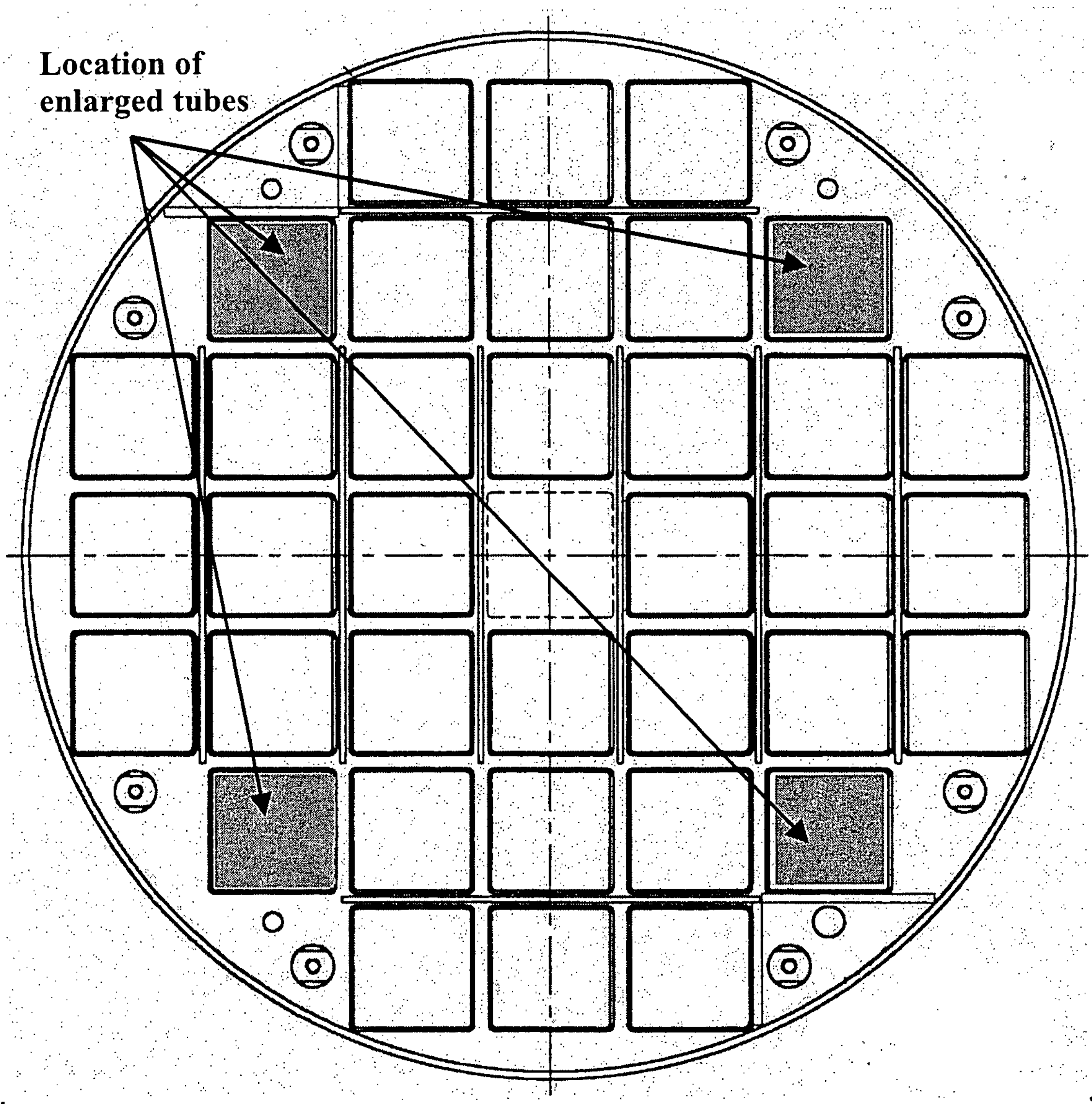
2.1.4 Recaged Fuel Assemblies

Certain United Nuclear fuel assemblies may be destructively disassembled for the purpose of inspection and testing of individual fuel rods. The United Nuclear fuel assembly lattice from which the fuel rods are removed will not likely be suitable for reuse. Consequently, the United Nuclear fuel rods may be installed in a Combustion Engineering fuel assembly lattice and placed in pool storage. The Combustion Engineering fuel assembly lattice will provide the same grid support structure as did the original United Nuclear fuel assembly lattice, but it will not have the shroud fixture used to preclude water impingement on the fuel rods. No empty fuel rod positions are permitted in the recaged Combustion Engineering fuel assembly lattice. The maximum heat load of the recaged fuel assemblies is bounded by the design basis heat load of 0.347 kW/assembly.

2.1.5 Fuel Assemblies with Removable Fuel Rods

Twelve fuel assemblies supplied by United Nuclear each have 12 fuel rods that are removable for inspection and test purposes. These fuel rods are not secured on either the top or bottom end, but are held in the grid structure of the fuel assembly lattice. The bottom end fitting prevents removal of the 12 fuel rods at the bottom end of the fuel assembly, but cutouts in the top end fitting allow the fuel rod to be grappled and removed. During loading of these United Nuclear fuel assemblies, unirradiated stainless steel retainers are used to block the cutouts in the top end fitting and prevent vertical movement of the 12 removable fuel rods in the fuel assembly grid.

Figure 2.1-1 Yankee-MPC Enlarged Fuel Tube Locations



APPENDIX 2.A PRINCIPAL DESIGN CRITERIA – MPC-LACBWR MPC STORAGE SYSTEM FOR DAIRYLAND POWER COOPERATIVE LA CROSSE BOILING WATER REACTOR

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2.A MPC-LACBWR PRINCIPAL DESIGN CRITERIA

The MPC-LACBWR storage system is one of three configurations of the canister-based dry storage cask system designed to be transported in the NAC-STC licensed transport cask.

This Appendix presents the design basis, including the principal design criteria, limiting load conditions, and operational parameters of the MPC-LACBWR dry storage system. The principal design criteria for the MPC-LACBWR system are described in Table 2.A-1.

The design criteria for the spent fuel to be stored in the MPC-LACBWR configuration are described in Section 2.A.1. Except as noted in this Appendix, the design criteria presented in MPC FSAR Section 2.2, the Safety Protection Systems described in Section 2.3, and the Decommissioning Considerations discussed in Section 2.4, apply to the MPC-LACBWR configuration.

Table 2.A-1 Summary of the MPC-LACBWR Design Criteria

MPC-LACBWR Design Criteria	
Design Life	50 years
Design Code[1] - Confinement	ASME Code, Section III, Subsection NB for confinement boundary
Design Code[1] - Nonconfinement	
Basket	ASME Code, Section III, Subsection NG and NUREG/CR-6322
Vertical Concrete Cask	ACI-349, ACI-318, ANSI/ANS 57.9 (1992)
Transfer Cask	ANSI N14.6 (1993) and NUREG-0612 (1980)
Design Weight:	
Canister Assembly (loaded, dry, with lid)	54,650 lbs.
Transfer Cask (empty)	80,740 lbs.
Vertical Concrete Cask (empty with lid)	141,200 lbs.
Thermal:	
Maximum Temperature,	430°C Normal, Off-Normal/Accident
Stainless Steel Cladding	(EPRI TR-106440)
Ambient Temperature Range	-40° to 125°F
Average Annual Ambient Temperature	75°F
Concrete Temperature:	
Normal Conditions	≤ 150°F; ≤ 200°F local
Off-Normal/Accident Conditions	≤ 350°F local/ surface
Canister Cavity Atmosphere	Helium

1. ASME and ACI Code editions are as specified in Section B3.3 of Appendix 12.B.

Table 2.A-1 Summary of the MPC-LACBWR Design Criteria (continued)

MPC-LACBWR Design Criteria (Continued)	
RADIATION PROTECTION/SHIELDING	
Concrete Cask Side Wall Contact Dose Rate	≤ 20 mrem/hr
Concrete Cask Top Lid Contact Dose Rate	≤ 25 mrem/hr
Concrete Cask Air Inlet/Outlet	≤ 100 mrem/hr
Owner Controlled Area Boundary	
Normal/Off-Normal	
Annual Whole Body Dose	25 mrem/yr
Accident Whole Body Dose	5 rem
MPC-LACBWR SPENT FUEL SPECIFICATIONS	
Spent Fuel	
Fuel Configuration/Vendor	Allis Chalmers 10 × 10, 3.64 (Type 1)/3.94 (Type 2) wt % ^{235}U (maximum enrichment) Exxon 10 × 10, 3.71 wt % ^{235}U (maximum planar average enrichment)
Fuel Cladding	Stainless Steel
Spent Fuel Capacity – Fuel Assemblies (may include up to 32 undamaged or damaged fuel assemblies, or fuel debris, in damaged fuel cans)	68
Spent Fuel Assembly Burnup (max)	22,000 MWd/MTU Allis Chalmers 21,000 MWd/MTU Exxon
Decay Heat per Fuel Assembly/DFC	63 W

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2.A.1 Spent Fuel To Be Stored in the MPC-LACBWR Storage System

The MPC-LACBWR storage system is designed to store up to 68 Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) fuel assemblies. The spent fuel assemblies stored in MPC-LACBWR are delineated by various factors including manufacturer, type, enrichment, burnup, cool time, and cladding material.

The LACBWR fuel consists of two types, Allis Chalmers and Exxon fuel. LACBWR fuel assemblies are comprised of 10x10 array of rods, with Allis Chalmers fuel containing 100 fuel rods and Exxon fuel containing 96 fuel rods and four inert rods. All fuel assemblies are steel clad. Table 2.A.1-1 lists the nominal design parameters of each fuel type.

The stored fuel assemblies must be undamaged or must be placed inside damaged fuel cans (DFC). Undamaged fuel assemblies may not have cladding defects greater than pin holes or hairline cracks. Unenriched fuel assemblies may not be stored in the MPC-LACBWR system.

The short-term and long-term temperature limits for stainless steel-clad fuel are derived based on the limits presented in EPRI Report TR-106440, "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage," April 1996. In this report, the potential failure modes in both wet and dry storage environments were assessed to develop the bounding conditions for the prevention of any potential cladding degradation phenomena and cladding failure modes of stainless steel clad fuel. The potential cladding degradation mechanisms evaluated include: general corrosion, stress corrosion cracking, localized corrosion, mechanical failures and chemical/metallurgical-based failure mechanisms. The EPRI report is based on several types of stainless steel cladding, including Type 304, 348, 348H and modified 348H, which includes LACBWR fuel. The temperature limit for stainless steel clad fuel is 430°C for the MPC-LACBWR system.

2.A.1.1 LACBWR Fuel - Bounding Fuel Evaluation

The criticality evaluations show that the Allis Chalmers fuel, with its maximum fuel mass and enrichment is the most reactive fuel type. All Allis Chalmers fuel is required to be placed inside a DFC and is therefore restricted to the 32 outer fuel tubes. Only Exxon assemblies are permitted in the interior 36 fuel tubes, but Exxon fuel may be loaded into the 32 peripheral (outer) tubes and DFCs as well. The criticality evaluation considers a complete assembly fuel rod matrix for undamaged fuel assemblies. Consequently, either solid rods fabricated from Zircaloy or stainless steel, or hollow Zircaloy rods with either Zircaloy or stainless steel slugs, must replace any fuel rod that is removed from an undamaged fuel assembly. This requirement does not apply to damaged fuel placed in DFCs.

The shielding evaluations demonstrate maximum dose rates for a combination load of Exxon and Allis Chalmers fuel assemblies. Allis Chalmers fuel assemblies are located in the basket peripheral (DFC) locations. Exxon fuel assemblies are primarily loaded in the center 32 fuel tubes. A limited number (5) of Exxon assemblies located in canister outer DFC locations are documented in the shielding valuation to have no effect on system dose rates. Allis Chalmers and Exxon fuel assemblies are evaluated for a minimum assembly average enrichment of 3.6 wt % ^{235}U to produce maximum source term. Allis Chalmers assemblies are evaluated at a maximum burnup of 22,000 MWd/MTU and 28 years minimum cool time, while Exxon assemblies are evaluated at 21,000 MWd/MTU and 23 years minimum cool time.

The NAC-MPC maximum decay heat load evaluated is 4.5 kilowatts. Based on source term evaluations the maximum heat load produced by the assemblies is 63W for a total Canister heat load of 4.3 kW for MPC-LACBWR.

2.A.1.2 MPC-LACBWR Preferential Fuel Loading

The MPC-LACBWR is designed to accommodate all fuel types representing the entire spent fuel inventory currently in storage in the LACBWR spent fuel pool while meeting all storage and transport requirements. Due to the large number of DFCs and the increase in reactivity associated with DFC preferential flooding, a preferential loading pattern for the two Allis Chalmers fuel types (3.64 wt % and 3.94 wt % ^{235}U maximum enrichments) is required. The pattern allowed is depicted in Figure 2.A.1-1.

2.A.1.3 MPC-LACBWR Damaged Fuel Can

The LACBWR Damaged Fuel Can (DFC) is required to be preferentially loaded because it can only be installed in the fuel positions that are oversized. Available DFC positions are identified as "B" and "C" locations in Figure 2.A.1-1.

Single fuel assemblies with damage to structural components that prevents handling the fuel assembly in the normal manner, fuel assemblies that have one or more damaged fuel rods, and fuel debris (including optional fuel debris handling canister) must be loaded in DFCs. However, an undamaged fuel assembly may also be loaded into a DFC.

The DFC is closed on the bottom end by a 0.5-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 DFC. A screen covers the holes to preclude the release of gross particulate material from the fuel assembly. The DFC lid is not secured to the DFC shell, but is held in place when the closure lid is installed in the canister. The DFC lid also has four drilled and screened holes. The damaged fuel assembly is inserted in the DFC. Lifting lugs in the DFC shell allow the loaded DFC to be lifted and placed in the canister basket. Alternately, the DFC may be inserted in a basket corner position before the designated fuel assembly is inserted into the DFC. The DFC lid is installed after fuel loading is completed.

The MPC-LACBWR DFC design and fabrication specification summary is provided in Table 1.A.3-3. The major physical design parameters are provided in Table 1.A.3-2. The structural evaluation is provided in Sections 3.A.4.4.2.4, 11.A.2.11.2.2 and 11.A.2.12.2.2.

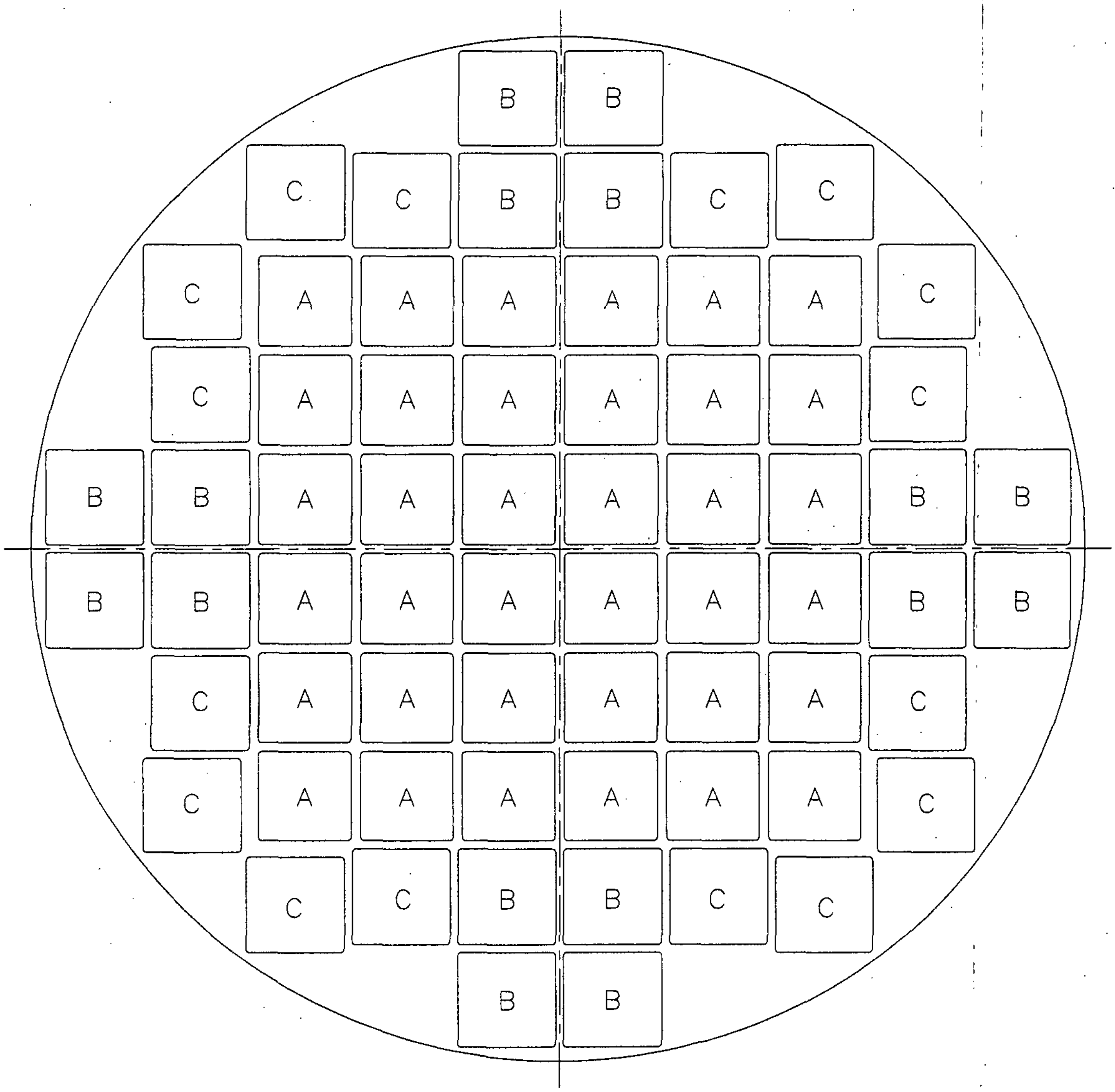
2.A.1.4 LACBWR Nonfuel Hardware

No nonfuel hardware is included in the MPC-LACBWR system design. BWR channels are not permitted for storage.

2.A.1.5 LACBWR Damaged Fuel

A transportable storage canister configured for damaged fuel holds up to 32 damaged fuel cans (DFCs) located in the "B" and "C" positions of the basket shown in Figure 2.A.1-1. A DFC may contain either an undamged or a damaged LACBWR spent fuel assembly of the types described in Table 2.A.1-1, and may also contain fuel debris. To analytically bound loose fuel material potentially trapped in a damaged fuel assembly, but not originating from the damaged fuel assembly, evaluations increased the spent fuel rod material quantity inside the DFC by 5% over that of a LACBWR undamaged assembly. The structural, thermal, shielding, confinement, and criticality effects of the damaged fuel cans are separately evaluated in the appropriate chapters.

Figure 2.A.1-1 Loading Diagram for MPC-LACBWR Fuel Basket



Slot A: Undamaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.

Slot B: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.
Damaged Allis Chalmers fuel maximum enrichment 3.64 wt % ^{235}U.

Slot C: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.
Damaged Allis Chalmers fuel maximum enrichment 3.94 wt % ^{235}U.

Table 2.A.1-1 LACBWR Fuel Parameters

	Allis Chalmers	**Exxon**
ASSEMBLY CONFIGURATION		
Assembly Length (in)	102.5	102.5
Assembly Width (in)	5.6	5.6
Assembly Array	10x10	10x10
Assembly Weight (lb)	400	400
Enrichment-wt. % ^{235}U		
Maximum	3.64/3.94[1]	3.71[2]
Minimum	3.6	3.6
Initial Fuel Weight (KgUO_2/Assembly)	138	127
Initial Heavy Metal (KgU/Assembly)	122	112
Max. Burnup (MWd/MTU)	22,000	21,000
Min. Cool Time (yr)	28	23
Max. Decay Heat (W)	63	62
FUEL ROD CONFIGURATION		
Fuel Rod Pitch (cm)	0.565	0.557
Active Fuel Length (in)	83	83
Rod OD (in)	0.396	0.394
Clad Thickness (cm)	0.02	0.022
Pellet OD (in)	0.35	0.343
Rods per Assembly	100	96
Fuel Material	UO_2	UO_2
Clad Material	Stainless Steel	Stainless Steel
INERT RODS		
Displacement Rod Material	N/A	SS Clad / Zirc Slug
Displacement Rod Diameter (cm)	N/A	0.394
Number Per Assembly	N/A	4

1. Enrichments are for Type 1 and Type 2 Allis Chalmers fuel respectively.
2. Represents maximum planar average enrichment.

Note: Values represent unirradiated fuel conditions.

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2.A.2 Design Criteria for Environmental Conditions and Natural Phenomena for the MPC-LACBWR Dry Storage System

The design criteria defined in this section identify the site environmental conditions and natural phenomena to which the storage system could reasonably be exposed during the period of storage. Analyses to demonstrate that the MPC-LACBWR design meets these design criteria are presented in the appropriate chapters of this report.

The design criteria that are defined in Section 2.2 of the NAC-MPC FSAR apply to the MPC-LACBWR system in their entirety for: Tornado and Wind Loading; Applicable Design Parameters; Determination of Forces on Structures and Tornado Missiles; Water Level (Flood) Design, including Flood Elevations; Phenomena Considered in Design Load Calculations; Flood Force Application and Flood Protection; Snow and Ice Loading Combined Criteria and Environmental Temperatures, except that the off-normal-severe heat environmental temperature is 105°F for MPC-LACBWR.

2.A.2.1 MPC-LACBWR Seismic Design

The MPC-LACBWR may be exposed to a seismic event (earthquake) during storage on an unsheltered concrete pad at an ISFSI site. The seismic response spectra experienced by the cask will depend 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 MPC-LACBWR would be a possible tip-over; however, tip-over does not occur in the evaluated design basis earthquake. Seismic response of the MPC-LACBWR is presented in Section 11.A.2.2.

2.A.2.1.1 Input Criteria

The magnitude of the maximum seismic accelerations to which the MPC-LACBWR may be subjected are site specific. 10 CFR 72.102 defines a 0.10 g horizontal ground motion design earthquake as the minimum allowable seismic design criteria, and 0.25 g is suggested for sites east of the Rocky Mountain front. The MPC-LACBWR is designed to 0.45 g horizontal and 0.3 g vertical seismic acceleration.

2.A.2.1.2 Seismic - System Analyses

Using quasi-static analysis methods, the seismic ground acceleration that would be required to cause the MPC-LACBWR to tip over is calculated in Section 11.A.2.2 to establish the factor of safety against tip-over. Both horizontal and vertical acceleration components are considered in the analyses. These components are calculated and combined according to Section 3.7.1 of NUREG-0800. Evaluation of the consequences of a tip-over event is provided in Section 11.A.2.12.

2.A.3 Safety Protection Systems

The MPC-LACBWR 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 of the MPC FSAR, fire and explosion events are effectively precluded by site administrative controls that prevent the introduction of flammable and explosive materials into areas where an explosion or fire could damage installed NAC-MPC systems. Quantities of transient combustibles are controlled to ensure that the design bases are not violated. The use of passive systems provides protection from mechanical or equipment failure.

2.A.3.1 General

The MPC-LACBWR is designed for safe, long-term storage of spent nuclear fuel. The MPC-LACBWR will survive 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 have been incorporated in the MPC-LACBWR system to assure safe long-term fuel storage are:

1. Continued confinement in postulated accidents.
2. Thick concrete and steel biological shield.
3. Passive systems that ensure reliability.
4. Inert atmosphere to provide corrosion protection for stored fuel cladding.

Each MPC-LACBWR system storage component is classified with respect to its function and corresponding effect on public safety. In accordance with Regulatory Guide 7.10, each system component is assigned a safety classification into Category A, B or C, as shown in Table 2.A.3-1. The safety classification is based on review of each component's function and the assessment of the consequences of component failure following the guidelines of NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

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.

As discussed in the following sections, the MPC-LACBWR design incorporates features addressing the above design considerations to assure safe operation during fuel loading, handling, and storage.

2.A.3.2 Protection by Multiple Confinement Barriers and Systems

2.A.3.2.1 Confinement Barriers and Systems

The radioactivity that the MPC-LACBWR must confine originates from the LACBWR spent fuel assemblies to be stored and residual contamination that may remain inside the canister as a result of contact with the water in the fuel pool where the canister loading is conducted.

The MPC-LACBWR is designed to confine the radioactive fuel. The canister is closed by welding. The closure lid weld is pressure tested. The closure lid weld is liquid penetrant examined following the root, intermediate, and final weld passes. A closure ring provides redundant closure to the closure lid. The closure lid inner port covers and outer port covers, which provide a redundant closure for the confinement boundary, are sealed by welding and are liquid penetrant examined on the root and/or final surface. The inner port cover is leak tested to 1.0×10^{-7} cm^3/s (air). The canister shell is leak tested at fabrication to 1.0×10^{-7} cm^3/s (air). The longitudinal and girth welds, and bottom welds of the canister shell are full penetration welds. The longitudinal and girth welds are radiographically inspected during fabrication. The bottom weld is ultrasonically inspected 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 a transportation cask without precluding the subsequent removal of the fuel (i.e., the fuel tubes do not deform such that they bind the fuel).

Personnel radiation exposure during handling and closure of the canister is minimized by the following steps:

1. Placing the closure 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 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.

2.A.3.3 Protection by Equipment and Instrumentation Selection

The MPC-LACBWR 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 in the following sections.

2.A.3.3.1 Equipment

The only important-to-safety equipment employed in the use and operation of the MPC-LACBWR is the lifting yoke used to lift the transfer cask. The transfer cask lifting yoke is designed to meet the requirements of ANSI N14.6 and NUREG-0612. It is single failure-proof by design. The lifting yoke is proof load tested to 300 percent of design load when fabricated. The lifting yoke is inspected for visible defects prior to each use and is inspected annually.

Additional handling equipment (such as trailers, skids, air pads, portable cranes, or cask transporters) are not important to safety as the MPC-LACBWR system is designed to withstand the failure of any of these components.

2.A.3.3.2 Instrumentation

No instrumentation is required for the safe storage operations of the MPC-LACBWR. 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 verification 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.A.3.4 Nuclear Criticality Safety

The primary nuclear criticality safety design criterion of the MPC-LACBWR is to provide features that ensure that the cask remains subcritical under normal, off-normal, and accident conditions. Neutron absorber sheets (BORAL) are employed in the basket design to capture thermalized neutrons, and preclude uncontrolled fission events. BORAL sheets are attached to the side of fuel tubes to have each fuel assembly separated from the adjacent assembly by at least one neutron absorber sheet. Fuel tubes containing damaged fuel cans (DFCs) have additional absorber sheets attached to provide flux traps, two absorber sheets, between assemblies. The BORAL sheets are mechanically supported by the fuel tube structure to ensure that the absorber sheets remain in place during the design basis normal, off-normal, and accident events.

The efficiency of the BORAL sheets in preserving nuclear criticality safety for the MPC-LACBWR is demonstrated by the Criticality Evaluation presented in Appendix A of Chapter 6.

2.A.3.4.1 Error Contingency Criterion

The design of the canister and fuel basket for the MPC-LACBWR is such that, under all conditions, the highest neutron multiplication factor (k_{eff}) will be less than 0.95. The criticality evaluation for the design basis fuel is presented in Section 6.A.4. Assumptions made in the analyses used to demonstrate conformance to this criterion include:

1. Most reactive fuel assembly type with maximum ^{235}U loading;
2. 75 percent of the nominal ^{10}B loading in the BORAL;
3. Infinite storage cask array of casks;
4. No structural material present in the assembly;
5. No credit taken for boron in the cask cavity or surrounding loading or storage area (BWR facilities typically do not contain borated pools); and

6. No credit taken for fuel burnup or for the buildup of fission product neutron poisons.

These assumptions demonstrate adequate controls to assure subcriticality in the use of the MPC-LACBWR system.

2.A.3.5 Radiological Protection

The MPC-LACBWR system, in keeping with the As Low As Reasonably Achievable (ALARA) philosophy, is designed to minimize, to the extent practicable, operator radiological exposure.

2.A.3.5.1 Access Control

Access to the LACBWR ISFSI site is controlled by a peripheral fence to meet the requirements of 10 CFR 72 and 10 CFR 20. Access to the storage area, and its designation as to the level of radiation protection required, is established by site procedure. The storage area will be surrounded by a fence, having lockable truck and personnel access gates. The fence will have intrusion-detection features as determined by the appropriate site procedure.

2.A.3.5.2 Shielding

10 CFR 72.104 and 72.106 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 that predict the normal and accident MPC-LACBWR doses are included in Appendices 5.A and 11.A. As shown in those appendices, the MPC-LACBWR meets these limits. The design basis average contact dose rate limits are:

Location	MPC-LACBWR (mrem/hr)
Storage Cask Top	25
Storage Cask Sides	20
Storage Cask Air Inlets and Outlets	100
Transfer Cask Side Wall	100
Top of Canister Structure	600

2.A.3.5.3 Ventilation Off-Gas

The MPC-LACBWR is passively cooled by radiant and natural convection heat transfer at the outer surface of the canister and natural convective heat transfer in the canister-concrete cask annulus. The bottom of the cask is conservatively assumed to be an adiabatic surface. The design criterion for the air-flow in the annulus is that the pressure difference, due to the buoyancy effect created by the heating of the air, is equal to the flow pressure drop. The details of the passive ventilation system design are provided in Section 4.0 of the MPC FSAR. Note that no convection credit is taken for the MPC-LACBWR system.

There are no radioactive releases during normal operations. Also, there are no credible accidents that cause significant releases of radioactivity from the MPC-LACBWR and, hence, there are no off-gas system requirements for the MPC-LACBWR 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 will be used.

The surface of the canister is exposed to cooling air when the canister is placed in the storage cask. If the surface is contaminated, the possibility exists that contamination could be carried aloft by the cooling air stream. To ensure that the canister surface is free of contamination, pool water is prevented from contacting 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 preclude 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 upper end. The upper end of the canister may be contaminated. The evaluated upper limit on surface contamination is presented in Section 11.A.1.5.2. The upper limit specified in LCO 3.2.1 is one-half of the value used in the evaluation in Section 11.A.1.5.2. If this limit is exceeded, then 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.

2.A.3.5.4 Radiological Alarm Systems

There are no radiological alarms required on the MPC-LACBWR. Justification for this is provided in analysis in Chapters 5.0 (Shielding Evaluation), 10.0 (Radiation 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 at the ISFSI. The TLDs are read periodically to provide a record of boundary dose.

2.A.3.6 Fire and Explosion Protection

Fire and explosion protection of the MPC-LACBWR is primarily provided by administrative controls applied at the site, which preclude the introduction of any explosive and any excessive flammable materials into the ISFSI area.

2.A.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 will be put in place to ensure that the presence of combustibles is minimized. A hypothetical fire event is evaluated as an accident condition in Section 11.2.5 of the MPC-FSAR.

2.A.3.6.2 Explosion Protection

The cask and associated systems are analyzed to ensure their proper function under an overpressure condition. As described in Section 11.2.3 of the MPC-FSAR, 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 storage cask.

For the same reasons as 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.

Table 2.A.3-1 Safety Classification of MPC-LACBWR Components

Component Description	Reference Drawings	Safety Function	Safety Classification
TSC Assembly	630045-872	Structural and Confinement	A
Shell and Base Plate	630045-870		
Closure Lid	630045-871		
Port Covers	630045-871		
Fuel Basket Assembly	630045-895	Criticality, Structural and Thermal	A
Basket Weldments	630045-877 and 630045-878		
Fuel Tube Assemblies	630045-881		
Neutron Absorbers	630045-881		
Damaged Fuel Can	630045-901 and 630045-902		
Support Disk	630045-893		
Heat Transfer Disk	630045-894		
Transfer Cask Assembly	455-860	Structural, Shielding and Operations	B
Trunnions			
Inner and Outer Shells			
Shield Doors and Rails			
Lead Gamma Shield			
Neutron Shield			
Adapter Plate Assembly	455-859	Operations and Shielding	NQ
Base Plate			
Door Rails			
Hydraulic Operating System			
Side Shields			
Concrete Cask Assembly	630045-862	Structural, Shielding, Operations and Thermal	B
Structural Weldments and Base Plate	630045-861		
Lid Assembly	630045-863		
Reinforcing Bars	630045-866		
Concrete			

2.A.4 Decommissioning Considerations

The principal elements of the MPC-LACBWR storage system are the vertical concrete cask (storage cask) and the transportable storage canister (canister).

The storage cask provides biological shielding and physical protection for the contents of the canister during long-term storage. The storage 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 storage cask, the top surface that supports the transfer cask during loading and unloading operations, and the pedestal of the storage cask that supports the canister. All of these surfaces are carbon steel, and it is anticipated that these surfaces could be decontaminated as necessary for decommissioning. A ¼-inch stainless steel plate is placed on the carbon steel pedestal of the MPC-LACBWR storage cask to separate it from the stainless steel canister bottom. Contamination of these surfaces is expected to be minimal, since the canister is isolated from spent fuel pool water during loading in the pool and the transfer cask is decontaminated prior to transfer of the canister to the storage cask.

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.

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 storage cask is expected to occur. The activity concentrations from activation of storage cask and canister components are listed in Table 2.A.4-1 for the MPC-LACBWR configuration. This table includes 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 by electron capture and is not of radiological concern.

Decommissioning of the storage cask would involve the removal of the canister and the subsequent disassembly of the storage cask. It is expected that the concrete would 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 transportable storage canister is designed and fabricated to be suitable for use as a waste package for permanent disposal in a deep Mined Geological Disposal System, in that it meets the requirements of the DOE MPC Design Procurement Specification. The canister is fabricated from materials having high long-term corrosion resistance, and the canister contains no paints or coatings that could adversely affect the permanent disposal of the canister. Consequently, decommissioning of the canister would occur only if the fuel contained in the canister had to be removed. Decommissioning would require that the closure welds at the canister closure lid and port covers be cut, so that the spent fuel could be removed. Removal of the contents of the canister would 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.

The Dairyland Power Cooperative La Crosse BWR ISFSI storage pad, fence and supporting utility fixtures are not expected to require decontamination as a result of use of the MPC-LACBWR 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.A.4-1 Activation of the MPC-LACBWR Concrete Cask and Canister

	Activity Concentration[1] (Ci/m^3)			
	Vertical Concrete Cask		**Transportable Storage Canister**	
Isotope	Cask Body[2]	Lid	Shell	Lid
^{40}K[3]	2.7E-07	3.8E-05	--	--
^{60}Co	5.1E-06	--	2.1E-04	4.1E-05
^{55}Fe	1.3E-04	1.3E-07	6.5E-05	1.3E-05
^{58}Co	--	--	2.7E-05	--
^{142}Ce	--	--	1.2E-05	1.2E-05
^{63}Ni	2.2E-07	--	1.1E-05	2.1E-06
^{87}Rb	--	--	1.9E-06	1.9E-06
^{54}Mn	3.6E-07	--	1.1E-05	5.4E-07
Total	1.4E-04	3.8E-05	3.4E-04	7.1E-05

Notes:

1. Seven days after removal of spent fuel.
2. Includes liner, rebar, concrete, and weldments.
3. K-40 is a naturally occurring radionuclide. It is present in the concrete and rebar due to the inclusion of potassium as a trace constituent. It is not present as a result of activation.

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3.0 STRUCTURAL EVALUATION

This section describes the design and analyses of the principal structural components of the NAC-MPC system under normal operating conditions. It demonstrates that the NAC-MPC system meets the requirements to assure confinement of contents, criticality control, radiological shielding, and contents retrievability as required by 10 CFR 72 for the design basis operating conditions. Off-normal and accident conditions are evaluated in Chapter 11.

3.1 Structural Design

The NAC-MPC is provided in three configurations. The first is designed to store up to 36 intact Yankee Class spent fuel and Yankee Class reconfigured fuel assemblies and is referred to as the Yankee-MPC. The second is designed to store up to 26 Connecticut Yankee fuel assemblies, CY-MPC reconfigured fuel assemblies and damaged fuel cans, and is referred to as the Connecticut Yankee-MPC (CY-MPC). The third is the La Crosse BWR MPC, referred to as the MPC-LACBWR, designed to store up to 68 Dairyland Power Cooperative La Crosse Boiling Water Reactor spent fuel assemblies, including up to 32 LACBWR damaged fuel cans. The three configurations have similar structural design and design criteria. They differ primarily in capacity and in certain principal dimensions and weight due to the physical parameters of the contents. The description and evaluation of the Yankee-MPC and CY-MPC systems are described in the appropriate sections of this chapter and the description and evaluation of the MPC-LACBWR system are provided in the Appendix A of this Chapter (3.A).

3.1.1 Discussion

The NAC-MPC system consists of three major components: 1) the vertical concrete cask (storage cask); 2) the transportable storage canister (canister); and 3) the transfer cask. These components are shown in Figure 3.1-1. The principal structural member of the vertical concrete cask is the reinforced concrete shell. The principal structural members of the canister are the shell, structural lid, bottom plate, the welds joining these components, and the basket assembly. The primary structural components of the transfer cask are its trunnions, inner and outer steel walls and the bottom doors and their support rails. All of the components are shown on the license drawings provided in Section 1.7.

The NAC-MPC components evaluated in this chapter are:

- Canister lifting devices
- Canister shell, bottom, and structural lid
- Canister shield lid support ring
- Damaged fuel can
- Basket assembly
- Transfer cask trunnions, shells, retaining ring, bottom doors, and support rails
- Vertical concrete cask body
- Concrete cask steel components (reinforcement, liner, lid, bottom plate, bottom, etc.)

All other NAC-MPC system components shown on the drawings presented in Section 1.7 are either nonstructural or not classified as important to safety. They are appropriately included as loads in the evaluation of the components listed above.

The structural evaluations demonstrate that all of the NAC-MPC components meet their structural design criteria and are capable of safely storing the design basis spent fuel.

3.1.1.1 Description of the Yankee-MPC

The concrete cask is a reinforced concrete cylinder with an outside diameter of 128 inches and an overall height of 160 inches. A 3.5-inch thick cylindrical carbon steel liner having an inside diameter of 79 inches forms the internal cavity of the concrete cask. The liner is a stay-in-place form. Its thickness is primarily determined by shielding requirements, but is related to the need to establish a practical limit to the diameter of the concrete shell. The concrete is Type II Portland Cement, having a nominal density of 140 lbs/ft^3, and a nominal compressive strength of 4000 psi. The inner and outer reinforcing bar assemblies are formed by vertical hook bars and horizontal hoop bars. The air flow path is formed by channels at the bottom and top of the concrete cask that provide the entrance and exit for cooling air. The air inlets admit the air to the storage cask interior cavity (i.e., the annular gap between the canister outer surface and the concrete cask liner interior surface). The air outlets allow the heated air to exhaust to the environment. A 5-inch thick carbon steel shield plug, that encloses a 1-inch thick layer of NS-4-FR neutron shield material, is installed in the concrete cask cavity above the canister. The plug is supported by a support ring welded to the liner. A 1.5-inch 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. The lid is bolted in place.

The canister consists of a cylindrical shell assembly closed at its top end by an inner shield lid and an outer structural lid. The canister contains a basket assembly that holds the spent fuel. The canister shell is 122.5 inches long and is fabricated from Type 304L stainless steel plate. The canister shield lid is 5-inch thick Type 304 stainless steel, and the structural lid is 3.0-inch thick Type 304L stainless steel. Both lids are welded to the canister shell to close the canister.

The shield lid is supported from below, prior to welding, by a support ring. The structural lid is supported, prior to welding, by the shield lid. The bottom of the canister is a 1-inch thick, Type 304L stainless steel plate that is welded to the canister shell.

The basket assembly is designed to hold up to 36 Yankee Class fuel assemblies. It incorporates 22 Type 17-4 PH stainless steel support disks and 14 Type 6061- T651 aluminum alloy heat transfer disks. The remaining components of the basket assembly are Type 304 stainless steel. These disks, together with the top and bottom weldments, are positioned by tie rods (with spacers and washers) that extend the length of the basket and clamp the components together. The support disks provide heat removal and support the fuel tubes that pass through the disks. The heat transfer disks provide the heat removal capability, but are not considered to be structural components.

There are three basket configurations that incorporate two fuel tube configurations and a damaged fuel can configuration. The three basket configurations accommodate 36 standard fuel tubes, 32 standard fuel tubes and four enlarged fuel tubes at the four basket corner positions or 32 standard fuel tubes and four damaged fuel cans at the four basket corner positions. The three basket configurations are not interchangeable. The standard fuel tube has a square interior cross-section of 7.8 inches and has BORAL sheets attached on all four outside surfaces of the fuel tube. No structural credit is taken for the BORAL sheet. The enlarged fuel tube has a square interior cross-section of 8.0 inches, but does not have exterior BORAL sheets on the sides. These larger cross-section fuel tubes can accommodate fuel assemblies that exhibit slight physical deformations (e.g., twist, bow) that could preclude loading in the smaller cross-section standard fuel tubes. The enlarged fuel tubes are restricted to the four corner positions of the basket as shown in Figure 2.1-1. When installed, the standard and enlarged fuel tubes are captured between the top and bottom weldments of the fuel basket. To permit full access to the enlarged fuel tubes, the corner positions of the top and bottom weldments used in this basket configuration are also enlarged. However, the enlarged fuel tubes remain captured between the basket top and bottom weldments.

The damaged fuel can is similar to the enlarged fuel tube in that it does not have exterior BORAL sheets on the sides and is restricted to the four corner positions of the basket. The damaged fuel can is closed on its bottom end by a stainless steel bottom plate having screened openings. After loading, the can is closed on its top end by a stainless steel lid that also has screened openings. The top plate and can body incorporate lifting fixtures that allow movement of the loaded can, if necessary, and installation and removal of the can lid. The damaged fuel can extends through the bottom and top weldments of the basket, and is captured between the damaged fuel shield and canister bottom plate. The damaged fuel can lid is held in place by the damaged fuel shield lid, which is machined on the underside in four places to mate with the damaged fuel can lid. The screened openings allow the filling, draining, and vacuum drying of the damaged fuel can, but preclude the release of gross particulate material to the canister interior. The damaged fuel can may also hold an intact fuel assembly.

To permit removal, if necessary, of the damaged fuel can, the top and bottom weldment openings in the four corner positions of the damaged fuel basket configuration are sized to allow the can to be inserted or removed with the basket assembled. Consequently, the damaged fuel can is not captured between the weldments.

Since the standard fuel tube with BORAL sheets attached, the enlarged fuel tube and the damaged fuel can have the same external dimensions, the support disks and heat transfer disks used in the three basket configurations are identical.

A transportable storage canister containing spent fuel may also contain one or more Yankee Class Reconfigured Fuel Assemblies. The Reconfigured Fuel Assembly is designed to contain Yankee Class spent fuel rods, or portions thereof, which are classified as failed, and to maintain the geometric positions of the rods. The assembly has a capacity of 64 full length spent fuel rods in an eight by eight array of tubes. As shown in Figure 1.3-1, the reconfigured fuel assembly consists of a shell (square tube with end fittings), a basket assembly, and 64 fuel tubes. All of the materials are stainless steel.

The Yankee Class Reconfigured Fuel Assembly is designed to contain failed fuel rods, in fuel tubes, during all storage and transport conditions. The Reconfigured Fuel Assembly is designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article NG-3000 and NUREG/CR-6322, "Buckling Analysis of Spent Fuel Baskets," and using the additional guidance contained in ASME Code Section III, Article NF-3000 and in ASME Code Section III,

Appendix F. The structural evaluation of the Reconfigured Fuel Assembly is presented in Section 11.4.1.

The external dimensions of the Yankee Class Reconfigured Fuel Assembly are the same as those of other Yankee Class fuel assemblies. The weight of a loaded reconfigured fuel assembly (approximately 550 pounds) is less than the weight of other Yankee Class fuel assemblies (approximately 850 pounds). The maximum temperature of the Reconfigured Fuel Assembly components is determined by the thermal analyses presented in Section 4.4.

The Yankee Class Reconfigured Fuel Assembly has been evaluated and is capable of withstanding, within code allowable limits (Service Level A/B), a postulated end impact resulting in a deceleration of 20g. It is also, when located in a fuel slot in the transportable storage container, capable of withstanding, within code allowable limits (Service Level A/B), a postulated side impact resulting in a deceleration of 20g. This analysis bounds the design conditions of the Reconfigured Fuel Assembly for normal conditions of storage.

The Reconfigured Fuel Assembly has also been evaluated for accident conditions and is capable of withstanding, within code allowable limits (Service Level D), a postulated end impact resulting in a deceleration of 57 g. It is also, when located in a fuel slot in the transportable storage container, capable of withstanding, within code allowable limits (Service Level D), a postulated side impact resulting in a deceleration of 55 g. This analysis bounds the design conditions of the Reconfigured Fuel Assembly for accident conditions of storage.

Therefore, the structural evaluations of the Yankee-MPC system containing other Yankee Class fuel assemblies (Chapters 3.0 and 11.0) bound those of the Yankee-MPC system containing one or more Yankee Class Reconfigured Fuel Assemblies.

3.1.1.2 Description of the CY-MPC

The CY-MPC system is similar in design and configuration to the Yankee-MPC system. The principal differences are due to the characteristics and parameters of the Connecticut Yankee and the Yankee Class fuels. Since the two systems are nearly identical, only the differences between the systems are presented in this section.

The overall length of the CY-MPC concrete cask is 190.6 inches, compared to a length of 160 inches for the Yankee-MPC. Otherwise, the dimensions and features of the CY-MPC concrete cask are the same as those for the Yankee-MPC. The shield plug of the CY-MPC concrete cask consists of 3.75 inches of ASTM A36 carbon steel and 2 inches of NS-4-FR or NS-3 neutron shielding material. The neutron shielding material is enclosed by 3/8-inch of ASTM A36 carbon steel, resulting in an overall shield plug thickness of 6.125 inches. The concrete cask lid is 1.5 inches of ASTM A36 carbon steel.

The CY-MPC canister shell is 151.75 inches long, compared to a length of 122.5 inches for the Yankee-MPC, and is also fabricated from Type 304L stainless steel plate. The bottom of the CY-MPC canister is a 1.75-inch thick, Type 304L stainless steel plate that is welded to the canister shell. The minimum structural lid weld and shield lid weld are 0.75 inch and 0.375 inch, respectively.

The CY-MPC basket is provided in two designs. The first is a 24-assembly basket intended to hold Zircaloy clad fuel with enrichment greater than 3.93 wt % ^{235}U. The second is a 26 assembly basket designed for Connecticut Yankee Zircaloy clad fuel with enrichments of 3.93 wt % ^{235}U, or less, and stainless steel clad fuel with enrichments of 4.03 wt % ^{235}U, or less. The two basket configurations are identical in principal dimensions. The loaded weight of the 24-assembly basket is less than that of the 26-assembly basket as it holds less fuel.

Both CY-MPC basket designs incorporate 28 Type 17-4 PH stainless steel support disks and 27 Type 6061-T651 aluminum alloy heat transfer disks. The remaining components of the basket assembly are Type 304 stainless steel. The fuel tubes have an inside square dimension of 8.72 inches and a composite wall thickness of 0.14 inches. All walls of each fuel tube contain a sheet of BORAL neutron poison material. No structural credit is taken for the BORAL sheet.

CY-MPC Reconfigured Fuel Assemblies

The CY-MPC canister containing spent fuel may also contain one or more CY-MPC Reconfigured Fuel Assemblies. The reconfigured fuel assembly is designed to contain Connecticut Yankee spent fuel rods, or portions thereof, which are classified as failed, and to maintain the geometric positions of the rods. The assembly has a capacity of 100 full length spent fuel rods in a ten by ten array of tubes. As shown in Figure 1.3-2, the CY-MPC reconfigured fuel assembly consists of a shell (square tube with end fittings), a basket assembly, and 100 fuel tubes. All of the materials are stainless steel.

The Reconfigured Fuel Assembly is designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Article NG-3000 and NUREG/CR-6322, "Buckling Analysis of Spent

Fuel Baskets," and using the additional guidance contained in ASME Code Section III, Article NF-3000 and in ASME Code Section III, Appendix F. The structural evaluation of the CY-MPC Reconfigured Fuel Assembly is presented in Section 11.4.2.

The CY-MPC reconfigured fuel assembly is 8.9 inches square, which is slightly larger than a standard fuel assembly and, thus, requires that it be loaded in one of the oversized loading positions in the basket. These are positions 1, 4, 23 and 26 in the 26-assembly basket (Figure 2.1-2) or positions 1, 4, 21 and 24 in the 24-assembly basket (Figure 2.1-3). The weight of a loaded reconfigured fuel assembly (approximately 1,150 pounds) is less than the weight of other Connecticut Yankee fuel assemblies. The maximum temperature of the Reconfigured Fuel Assembly components is determined by the thermal analyses presented in Section 4.5.3.

The CY-MPC reconfigured fuel assembly has been evaluated for a normal condition of storage end impact of 20g and a side impact resulting in a deceleration of 20g. These analyses, presented in Section 3.4.4.5, bound the design conditions of the reconfigured fuel assembly for normal conditions of storage.

The CY-MPC reconfigured fuel assembly has also been evaluated for accident conditions, i.e., an end impact of 60g and a side impact of 60g. The analysis, presented in Section 11.4.2, bounds the design conditions of the CY-MPC reconfigured fuel assembly for accident conditions of storage.

Therefore, the structural evaluations of the NAC-MPC system containing other Connecticut Yankee fuel assemblies (Chapters 3.0 and 11.0) bound those of the NAC-MPC system containing one or more CY-MPC reconfigured fuel assemblies.

CY-MPC Damaged Fuel Cans

The CY-MPC damaged fuel can is installed in the basket fuel positions that are oversized. These are positions 1, 4, 23, and 26 of the 26-assembly basket and positions 1, 4, 21 and 24 of the 24-assembly basket. It is designed to hold a single fuel assembly that may be classified as intact or as damaged. The CY-MPC damaged fuel can is shown in Figure 1.3-3 and in Drawings 414-901 and 414-902. As shown in the drawings, the can is 141.5 inches in length, and has an internal square dimension of 8.7 inches. The shell of the damaged fuel can, which has the shape of a square tube, is 18 gauge stainless steel.

The can is closed on the bottom end by a 0.5-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 particulate material from the fuel assembly. A lid having an overall depth dimension of 2.3 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 assembly is inserted in the can and the can lid is installed. Lifting lugs in the can shell allow the loaded can to be lifted and installed in the canister basket. Alternately, the fuel can may be inserted in a basket corner position before the damaged fuel assembly is inserted in the fuel can. The loaded weight of the damaged fuel can is 1,500 pounds.

The CY-MPC damaged fuel can design and fabrication specification summary is provided in Table 1.3-6. As shown in that table, the damaged fuel can and the reconfigured fuel assembly are designed to the same criteria. The major physical design parameters are provided in Table 2.1-5. The structural evaluation of the damaged fuel can is provided in Sections 3.4.4.6 and 11.4.3.

3.1.2 Design Criteria

The NAC-MPC structural design criteria are specified in Section 2.2. The load combinations of normal, off-normal, and accident loadings have been evaluated in accordance with ANSI 57.9 and ACI 349 for the concrete cask, and in accordance with the ASME Code, Section III, Division I, Subsection NB for Class 1 components for the canister, as listed in Tables 2.2-1 and 2.2-2, respectively. The basket is evaluated in accordance with ASME Code, Section III, Subsection NG, and NUREG-6322. The transfer cask and the lifting yoke are lifting devices that are designed to NUREG-0612 and ANSI N14.6.

3.4 General Standards for Casks

The NAC-MPC is provided in three configurations. All three configurations, the Yankee-MPC, the CY-MPC and the MPC-LACBWR, are constructed from the same materials, are used in the same environments and are exposed to the same operating conditions. The evaluations provided in Sections 3.4.1 and 3.4.2 apply to the Yankee-MPC and the CY-MPC configurations. Separate evaluations of each configuration are provided in Sections 3.4.3 and 3.4.4, since the configurations have different weight and length dimensions, and different design basis pressures due to the physical parameters of the contents. The MPC-LACBWR is evaluated in Appendix 3.A.

3.4.1 Chemical and Galvanic Reactions

The materials used in the fabrication and operation of the NAC-MPC system have been 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—have been considered for the environments that may be encountered under normal, off-normal, or accident conditions. Based on the evaluation, there is one potential reaction that could adversely affect the overall integrity of the storage cask, the fuel basket, the transportable storage canister, or the structural integrity and retrievability of the fuel from the canister. That potential reaction, between aluminum and spent fuel pool water, which may produce hydrogen is mitigated by the specific canister loading procedures presented in Section 8.1.1.

3.4.1.1 Component Operating Environment

Most of the component materials of the NAC-MPC are exposed to two typical operating environments: 1) an open canister containing 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 the canister in environs that include air, rain water/snow/ice, and marine (salty) water/air. The spent fuel assemblies consist of Zircaloy or stainless steel clad fuel and other fuel assembly components of stainless steel.

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.

One of the operating environments to which the canister internal component materials are exposed does not provide the conditions necessary for a reaction (corrosion); i.e., both moisture and oxygen must be present for corrosion to occur. This long-term environment is the sealed canister, backfilled with helium. The free volume in the canister is drained, vacuum dried and backfilled with helium, effectively precluding corrosion. Galvanic corrosion (i.e., between dissimilar metals that are in contact) could occur, but only if there is water present at the point of contact and the metals are in electrical contact with each other (i.e., mechanically held together). NAC's operating procedures provide two helium backfill cycles in series separated by a vacuum-drying cycle for the canister during the preparation of the canister for storage. Therefore, the canister cavity is effectively dry and galvanic corrosion is precluded.

3.4.1.2 Component Material Categories

The component materials evaluated are categorized based on similarity of physical and chemical properties and/or on similarity of component functions. The categories of materials that are considered are stainless/nickel alloy steels, nonferrous metals, and criticality control materials. These categories are evaluated based on the environment to which they 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, except aluminum, during any phase of normal, off-normal, or accident condition loading, unloading, handling, or storage operations. Therefore, only the potential aluminum reaction with spent fuel pool water is evaluated.

3.4.1.2.1 Stainless Steels

No reaction of the canister component stainless steel is expected in any environment, except for the marine environment, where chloride-containing salt spray might 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 NAC-MPC has smooth external surfaces to minimize the collection of such materials as salts.

There is a significant electrochemical potential difference between austenitic (300 series) stainless steel and aluminum. If aluminum is in electrical contact with the austenitic stainless steel, the aluminum could be expected to exhibit corrosion driven by electrochemical EMF when immersed in water. Pressurized water reactor (PWR) pool water does provide a conductive potential. The only aluminum components that will be in contact with stainless steel and exposed to the pool water are the heat transfer disks in the fuel basket. Since the fuel basket is not welded or bolted to the canisters, poor, if any, electrical contact with the stainless steels is

Appendix 3.A STRUCTURAL EVALUATION – MPC-LACBWR MPC STORAGE SYSTEM FOR DAIRYLAND POWER COOPERATIVE LA CROSSE BOILING WATER REACTOR

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3.A STRUCTURAL EVALUATION OF THE MPC-LACBWR STORAGE SYSTEM

This section describes the design of the principal structural components of the MPC-LACBWR storage system and provides the structural evaluation of the MPC-LACBWR storage system for normal operating conditions. The structural evaluation of the MPC-LACBWR storage system for off-normal and accident conditions is provided in Appendix 11.A. The structural evaluation demonstrates that the MPC-LACBWR storage system will provide confinement of the radioactive material, maintain criticality control, provide biological shielding, and assure retreivability of the spent nuclear fuel (SNF) contents, in accordance with the requirements of 10 CFR 72.

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3.A.1 Structural Design of the MPC-LACBWR Storage System

The MPC-LACBWR storage system, which is designed to store up to 68 La Crosse Boiling Water Reactor (LACBWR) spent fuel assemblies, is similar to the Yankee-MPC storage system in most respects. The structural design of the MPC-LACBWR storage system is discussed in Section 3.A.1.1. The structural design criteria for the MPC-LACBWR storage system are discussed in Section 3.A.1.2.

3.A.1.1 Discussion

The principal components of the MPC-LACBWR storage system are the concrete cask, transportable storage canister, and the transfer cask. The structural design features of these components are discussed in this section.

Vertical Concrete Cask

The MPC-LACBWR concrete cask is almost identical to the Yankee-MPC concrete cask described in Section 3.1.1.1. It is constructed from the same materials and has the same canister pedestal design, inlet and outlet ventilation duct geometry, lid attachment details, inside and outside diameters, overall height and concrete reinforcment as the Yankee-MPC concrete cask. The primary differences between the structural design of the MPC-LACBWR concrete cask and Yankee-MPC concrete cask are the thicknesses of the steel and concrete shells that form the cylindrical wall of the concrete cask and the design and support conditions of the shielding at the top end of the concrete cask.

The MPC-LACBWR concrete cask has the same overall wall thickness as the Yankee-MPC concrete cask, but a thinner steel liner (2.5-inch vs. 3.5-inch) and a thicker reinforced concrete shell (22-inch vs. 21-inch). As a result, the MPC-LACBWR concrete cask weighs slightly less than the Yankee-MPC concrete cask. The thickness of the concrete cask steel liner is primarily driven by shielding requirements. These differences between the steel liner and concrete shell thicknesses are addressed in the MPC-LACBWR concrete cask structural evaluation.

The lid of the MPC-LACBWR concrete cask differs from that of the Yankee-MPC concrete cask in that shielding material is attached to the under-side of the MPC-LACBWR concrete cask cover plate by carbon steel plates and stud anchors, whereas the Yankee-MPC concrete cask shield plug is a separate assembly that is supported by a steel ring attached to the inside of the concrete cask steel liner. Furthermore, the shielding material used for the MPC-LACBWR concrete cask lid is different than that of the Yankee-MPC concrete cask shield plug. The MPC-LACBWR concrete cask lid includes steel-encased concrete shielding, whereas the Yankee-MPC concrete cask shield plug assembly is fabricated from carbon steel and NS-4-FR. The MPC-LACBWR concrete cask lid concrete shielding material is thicker than the MPC-LACBWR concrete cask shield plug (8.38 inches vs. 5.13 inches), but weighs less due to the materials of construction. Because the MPC-LACBWR concrete cask shield plug is integral with the concrete cask cover plate, cask loading operations are simplified.

Transportable Storage Canister

The MPC-LACBWR canister design is similar to the Yankee-MPC canister design. It consists of a cylindrical canister shell assembly and internal basket assembly. The canister shell forms the pressure-retaining boundary that is relied upon to confine the radioactive contents of the canister. The canister basket assembly provides lateral support of the spent fuel assemblies and maintains their geometric spacing for criticality control.

The MPC-LACBWR canister shell assembly has a 70.64-inch outside diameter and a 116.3-inch overall length. The MPC-LACBWR canister shell assembly is formed by a ½-inch-thick cylindrical shell, a 1.25-inch-thick bottom plate, and a 7-inch-thick closure lid; all of which are Type 304/304L austenitic stainless steel. The bottom plate is attached to the cylindrical shell by a complete joint penetration weld. The closure lid is attached to the cylindrical shell by a ½-inch-thick partial penetration groove weld. The top closure weld is backed by a secondary closure weld that is formed by connecting a ¾-inch-thick ring between the closure lid and cylindrical shell with ¼-inch-thick partial penetration groove welds. The six threaded holes that are located on the top surfaces of the MPC-LACBWR closure lid near its perimeter are used to lift the lid and canister assembly. A 4-inch-thick by 38.3-inch-square spacer, which is fabricated from aluminium, is bolted to the underside of the closure lid. The spacer minimizes fuel assembly free movement by eliminating the excess gap above the spent fuel assemblies stored in the central 6 × 6 array of the basket assembly. The spacer does not have a structural function for storage conditions; its purpose is to eliminate secondary fuel impact effects under the transportation hypothetical accident free drop conditions that are postulated to result from excessive axial gaps above the fuel.

The MPC-LACBWR basket assembly is a tube-and-disk design that is similar to the Yankee-MPC basket assembly. The MPC-LACBWR basket assembly is designed to accommodate up to 68 LACBWR fuel assemblies. Up to 36 fuel assemblies are stored directly in the central 6 × 6 array of the basket and up to 32 fuel assemblies may be placed inside damaged fuel cans and stored in the perimeter basket locations (i.e., cells located outside the central 6 × 6 array.)

The main structure of the MPC-LACBWR basket assembly is formed by 26 support disks and a top and bottom weldment that are positioned and supported axially by eight tie rods. The basket assembly also includes 14 nonstructural aluminum heat transfer disks that are supported and positioned between the support disks located in the middle of the basket. Fuel tube assemblies are positioned inside each opening of the basket assembly and captured axially by the top and bottom weldments. The fuel tube assemblies are slightly smaller versions of the Yankee-MPC fuel tube design described in MPC FSAR Section 3.1.1.1. The cross-section and overall length of the MPC-LACBWR fuel tubes are adjusted to accommodate MPC-LACBWR fuel assemblies. The MPC-LACBWR fuel tube assemblies are provided in two sizes; a standard size for undamaged fuel and an enlarged size for fuel stored in damaged fuel cans.

All 26 support disks are circular plates having a 69.4-inch outside diameter and 68 square cutouts. The 36 square cutouts that form the center 6×6 array are sized to accommodate standard sized fuel tubes and the remaining 32 square cutouts are sized to accommodate enlarged fuel tubes. With the exception of the support disks located at the top and bottom ends of the basket, all support disks are fabricated from 5/8-inch-thick plate. The top and bottom support disks are fabricated from 1-¼-inch and ¾-inch-thick plate, respectively. All support disks are fabricated from Type 17-4 PH stainless steel. The top and bottom weldments and the tie rods are fabricated from Type 304 stainless steel.

The MPC-LACBWR damaged fuel can design is a slightly smaller version of the Yankee-MPC damaged fuel can design described in MPC FSAR Section 3.1.1.1. The square cavity and overall length of the MPC-LACBWR damaged fuel can are adjusted to accommodate LACBWR fuel assemblies.

Transfer Cask

The same transfer cask is used for the MPC-LACBWR that was used for the Yankee-MPC storage system at the Yankee Rowe site. The structural evaluation of the Yankee-MPC transfer cask lifting devices (i.e., lifting trunnions, retaining ring and bolts, and cask rails) is presented in MPC FSAR Section 3.4.3.3. The structural analysis of the Yankee-MPC transfer cask is performed using transfer cask and canister weights that envelope those of the MPC-LACBWR transfer cask and canister. Therefore, the results of the Yankee-MPC transfer cask structural analysis are bounding.

3.A.1.2 Design Criteria

The structural design criteria for the MPC-LACBWR storage system are the same as those used for design and license of the Yankee-MPC system discussed in MPC FSAR Section 3.1.2.

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3.A.2 Weights and Centers of Gravity

The weights and centers of gravity for the MPC-LACBWR system components and assemblies are summarized in Table 3.A.2-1.

Table 3.A.2-1 MPC-LACBWR System Weights and Centers of Gravity

Item Description	**Calculated Weight (pounds)**	**Center of Gravity[1] (inches)**
Concrete Cask Lid	6,970	160.7
Canister Lid	7,700	112.8[2]
Transfer Adapter Plate	12,660	--
Canister (with basket, without lid)	18,080	--
Canister (loaded with contents, with water and lid)	64,800	60.6[2]
Canister (loaded with contents, with lid, no water)	54,650	61.7[2]
Concrete Cask (empty, without lid)	134,230	78.2
Concrete Cask and Canister (loaded with contents, with lid)	195,850	83.0
Transfer Cask (empty)	80,740	57.0[3]
Transfer Cask and Canister (with basket, without lid)	98,820	61.1[3]
Transfer Cask and Canister (loaded with contents, with water and lid)	145,550	58.6[3]
Transfer Cask and Canister (loaded with contents, dry with lid)	135,390	58.9[3]
Water in Canister	10,160	--
Fuel (68 at 400 lbs. per fuel assembly)	27,200	53.0[2]
Damaged Fuel Cans (DPC) (32 at 52 lbs. each)	1,670	64.0[2]

1. All centers of gravity are measured from the bottom of the Concrete Cask base plate, unless otherwise noted.
2. Measured from the bottom of the Canister bottom plate.
3. Measured from the bottom of the Transfer Cask.

3.A.3 Mechanical Properties of Materials

The material properties used in the strucutral evaluation of the MPC-LACBWR storage system are presented in MPC FSAR Section 3.3.

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3.A.4 General Standards for Casks

3.A.4.1 Chemical and Galvanic Reactions

The MPC-LACBWR storage system materials of construction are the same materials used in the Yankee-MPC storage system. The potential chemical and galvanic reactions between these materials are evaluated in MPC FSAR Section 3.4.1 for all phases of storage operations.

3.A.4.2 Positive Closure

Positive closure of the MPC-LACBWR storage system is provided by the same means as the Yankee-MPC storage system, as discussed in MPC FSAR Section 3.4.2.

3.A.4.3 Lifting Devices

The MPC-LACBWR storage system components are lifted using the same methods desribed in MPC FSAR Section 3.4.3. This section presents the structural evaluation of the MPC-LACBWR storage system lifting devices.

3.A.4.3.1 MPC-LACBWR Storage Cask Bottom Lift

The MPC-LACBWR concrete cask is designed to be lifted vertically from its bottom end using four hydraulic jacks (one jack in each air inlet opening) to allow insertion of an air pad system under the bottom end of the concrete cask. The structural evaluation of the concrete cask for this bottom lift condition considers the required hydraulic jack size, the capacity of the Nelson studs that secure the concrete cask bottom weldment to the concrete shell, and the stresses in the concrete cask pedestal that supports the canister. The evaluation of the MPC-LACBWR concrete cask for the bottom lift condition is based on comparison to the Yankee-MPC concrete cask design.

The minimum hydraulic jack piston diameter required to lift the loaded Yankee-MPC concrete cask is shown to be 3.73 inches in MPC FSAR Section 3.4.3.1.1. The required jack piston diameter for the MPC-LACBWR is smaller since the weight of the loaded MPC-LACBWR concrete cask is lower. Therefore, the hydraulic jack piston size used to lift the Yankee-MPC concrete cask is adequate for lifting the MPC-LACBWR concrete cask.

The structural analysis of the Nelson studs that secure the concrete cask concrete shell to the steel base plate is presented in MPC FSAR Section 3.4.3.1.1 for the bottom lift of the Yankee-MPC concrete cask. The Nelson studs are designed to withstand a total pull-out force that is based on the combined weight of the loaded canister, concrete cask pedestal, and air inlet vent liners. Since the MPC-LACBWR canister weighs less than the Yankee-MPC canister, and designs of the concrete cask bottom end steel weldments are structurally the same, including the material, size, number, and spacing of the Nelson studs, the margin of safety against failure of the Nelson studs is higher for the MPC-LACBWR concrete cask.

The stresses in the Yankee-MPC concrete cask pedestal for the bottom lift condition are shown to satisfy the applicable allowable stress design criteria in MPC FSAR Section 3.4.3.1.1. Since the MPC-LACBWR canister weighs less than the Yankee-MPC canister, and designs of the concrete cask bottom pedestals are structurally the same, the maximum stresses in the MPC-LACBWR concrete cask pedestal due to the bottom lift are bounded by those calculated for the Yankee-MPC concrete cask pedestal.

Lastly, the air pad system used to move the Yankee-MPC concrete cask is adequate for use with the MPC-LACBWR concrete cask since it has the same bottom end details as the Yankee-MPC concrete cask and a lower loaded weight.

3.A.4.3.2 MPC-LACBWR Canister Lift

The MPC-LACBWR canister lifting devices are identical to the Yankee-MPC canister lifting devices. The MPC-LACBWR canister is lifted using redundant three-legged lifting slings attached to hoist rings that are screwed into the six 1-½-inch-diameter threaded holes located near the perimeter of the lid. For this redundant lift configuration, the canister lifting devices are designed to provide factors of safety of 3 against yield and 5 against ultimate in accordance with ANSI N14.6. The structural evaluation of the MPC-LACBWR canister lifting devices, which considers the load capacity of the hoist rings that attach to the lid and the stresses in the canister lid and closure welds, demonstrates that it satsifies the structural stress design criteria for the contolling lift configuration.

The lifting slings and hoist rings used to lift the MPC-LACBWR canister are the same as those used to the lift the Yankee-MPC canister. Each hoist ring is required to have a working load limit (i.e., ultimate load is 5 times the working load limit) of 24,000 pounds or higher and a minimum thread engagement length of 2.23 inches. The required working load limit of each lifting sling is a function of its angular rotation from vertical (θ), which is calcuated based on its length as follows:

$$T = \frac{F}{\sin(\theta)}$$

where:

F = 20,057 lb., vertical load of the sling (54,700 lb. × 1.1 / 3)

θ = $\tan^{-1}(H/R)$

H = Height from canister top end to master link

R = 30.25 in., lifting attachment bolt circle radius

For a sling configuration in which the sling master link is located 5-feet above the top end of the canister (i.e., H = 60 inches), each sling must have a minimum working load limit of 22,462 pounds (< 24,000 pounds).

The stresses in the MPC-LACBWR lid and closure welds due to the combined effects of dead weight, normal handing (canister vertical lift), and internal pressure loads are calculated using finite element analysis, as discussed in Section 3.A.4.4.1.5. The analysis conservatively assumes that the canister lift load is supported by only one of the two 3-legged slings used to lift the canister. Therefore, due to the redundancy provided by two 3-legged lift slings, the canister is designed to provide factors of safety of 3 against yield and 5 against ultimate for this condition.

The analysis results show that the maximum nodal stress intensity in the MPC-LACBWR canister closure weld is 3.69 ksi. The yield strength and ultimate tensile strength of the MPC-LACBWR canister Type 304 stainless steel material at the bounding design temperature of 300°F are 22.5 ksi and 61.5 ksi, respectively. Therefore, the MPC-LACBWR canister provides factors of safety of 6.1 (>3) against yield and 16.7 (>5) against ultimate for the vertical lift condition.

3.A.4.3.3 MPC-LACBWR Transfer Cask Lift

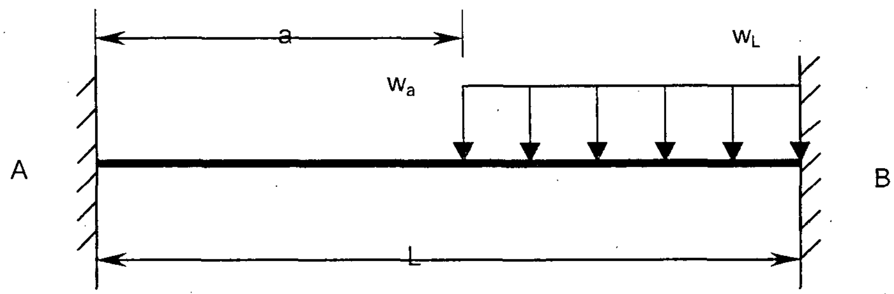
The same transfer cask design and hardware are used for the MPC-LACBWR and Yankee-MPC storage systems. The structural analysis of the Yankee-MPC transfer cask lifting devices (i.e., lifting trunnions, retaining ring and bolts, and cask rails), which is conservatively based on bounding empty and loaded transfer cask weights of 81 kips and 150 kips, respectively, is presented in MPC FSAR Section 3.4.3.3. As shown in Table 3.A.2-1, these weights bound the corresponding MPC-LACBWR transfer cask weights. Therefore, the stresses in the MPC-LACBWR transfer cask are bounded by those calculated for the Yankee-MPC transfer cask in MPC FSAR Section 3.4.3.3.

3.A.4.3.4 MPC-LACBWR Damaged Fuel Can Lift

The MPC-LACBWR damaged fuel can (DFC), containing a fuel assembly, is designed to be lifted vertically using a lifting tool with 1-inch wide lugs that engage with the four lifting slots in the top end of the DFC side plates. The DFC lifting slot dimensions are the same as those of the MPC-CY DFC design shown in MPC FSAR Section 3.4.3.7. The DFC is designed with a minimum factor of safety of 3 against yield for normal lifting loads. The stresses in the DFC side plates, the welds that joins the DFC tube body to the side plates and bottom plate, and the lifting tool lugs that engage with the DFC are evaluated using classical methods as discussed below.

Damaged Fuel Can Side Plates:

The stress in the DFC side plate above the lifting slot is determined by analyzing the section above the slot as a 0.15-inch wide × 1.88-inch long × 1.25-inch deep beam that is fixed at both ends. The lifting tool lug is 1.0-inch wide and engages the last one inch of the slot. The load supported by the lifting tool is equal to the combined weight of the DFC and fuel assembly, or 462.2 pounds, increased by 10% to account for handling loads. The following figure represents the configuration evaluated:



where:

a = 0.88 in.

L = 1.88 in.

w_a = w_L = ((462.2 lbs x 1.1*g*)/4)/1.0 in. = 127.1 lbs/in.

Reactions and moments at the fixed ends of the beam are calculated per Roark, Table 3, Case 2d. The reaction at the left end of the beam (R_A) is:

$$R_A = \frac{w_a}{2L^3}(L-a)^3(L+a) = 26.4 \text{ lbs.}$$

The moment at the left end of the beam (M_A) is:

$$M_A = \frac{-w_a}{12L^2}(L-a)^3(L+3a) = -13.5\,\text{lbs}\cdot\text{in.}$$

The reaction at the right end of the beam (R_B) is:

$$R_B = w_a\,(L-a) - R_A = 100.7\text{ lbs}$$

The moment at the right end of the beam (M_B) is:

$$M_B = R_A L + M_A - \frac{w_a}{2}(L-a)^2 = -27.4\text{ lbs}\cdot\text{in.}$$

The maximum bending stress (σ_b) in the side plate is:

$$\sigma_b = \frac{Mc}{I} = 562\,\text{psi}$$

where:

$M = M_B = 27.4$ lb-in.

$c = 0.5$ in.

$I = (0.15)(1.25^3)/12 = 0.0244\text{ in}^3$

The maximum shear stress (τ) occurs at the right end of the slot:

$$\tau = \frac{R_B}{A} = 536\,\text{psi}$$

where:

$A = (0.15)(1.25) = 0.188\text{ in}^2$

The Von Mises stress (σ_{max}) is:

$$\sigma_{max} = \sqrt{\sigma_b^2 + 3\tau^2} = 1085\text{ psi}$$

The yield strength (S_y) for Type 304 stainless steel is 22,500 psi at 300°F. Therefore, the factor of safety provided against yield is:

$$FS = \frac{S_y}{\sigma_{max}} = 20.7 > 3$$

Therefore, the DFC side plates satisfy the allowable stress design criteria for the vertical lift condition.

Damaged Fuel Can Weld Evaluation

The DFC tube body is joined to the DFC side plates and bottom plate by full penetration welds that receive surface visual examination. In accordance with NG-3352-1, a weld quality factor (n) of 0.5 is used for these welds. The average tensile stress in these welds due to the lift load is:

$$\sigma_w = \frac{1.1(P)}{A} = 438 \text{ psi}$$

where:

P = 462 lb., combined weight of the tube body, bottom weldment and fuel assembly

A = 1.16 in^2, cross-sectional area of DFC tube body

The factor of safety (FS) against yield, which includes the weld quality factor, is:

$$FS = \frac{n \cdot S_y}{\sigma_w} = 21.2 > 3$$

Therefore, the DFC tube body welds satisfy the allowable stress design criteria for the vertical lift condition.

Damaged Fuel Can Lifting Tool

The lifting tool uses four 1.0-inch wide × 0.25-inch thick lugs that engage the DFC lifting slots. The lugs are analyzed in shear loading with four lugs carrying the design-lifting load.

The shear stress (τ) in the lug is:

$$\tau = \frac{1.1P}{A} = 508 \text{ psi}$$

where:

$A = 0.25 \text{ in}^2$.

$P = 462/4 = 116$ lbs.

The factor of safety (FS) against shear stress allowable (i.e., $0.6S_y$), based on a yield strength of 22.5 ksi for mild stainless steel (e.g., Type 304) at 300°F, is:

$$FS = \frac{0.6Sy}{\tau} = 26.6 > 3$$

Therefore, the design condition that lifting stresses have a load factor of 3 on the basis of yield strength is met.

3.A.4.4 MPC-LACBWR Components under Normal Operating Loads

The structural evaluation MPC-LACBWR storage system for normal conditions of storage is discussed in the following sections. The results of the MPC-LACBWR storage system structural evaluation demonstrate that the canister and concrete cask satisfy the applicable structural design criteria for all normal conditions of storage.

3.A.4.4.1 MPC-LACBWR Canister Analysis

3.A.4.4.1.1 Canister Thermal Stress Analysis

This section discusses the structural analysis of the MPC-LACBWR canister for the normal thermal condition. Stresses in the MPC-LACBWR canister due to thermal loading are determined using the ANSYS finite element analysis program. The finite element model of the MPC-LACBWR canister is constructed from solid elements, which represent the canister shell, bottom plate, closure lid and closure ring. A full 360° finite element model of the canister shell is used for the analysis. Figure 3.A.4.4.1-1 shows a half-symmetry view of the MPC-LACBWR canister finite element model, with a detail of the canister top closure region.

The ANSYS thermal stress analysis was performed with canister temperatures that enveloped the canister temperature gradients for normal storage (105°F and -40°F ambient temperatures) and transfer conditions. Prior to performing the thermal stress analysis, the steady-state temperature distribution was determined using temperature information from the storage and transfer thermal analyses (MPC FSAR Chapter 4). This was 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 were applied at six key locations (i.e., top-center of the structural lid, top-outer diameter of the lid, bottom-center of the lid, bottom-center of the bottom plate, bottom-outer diameter of the bottom plate, and mid-elevation of the canister shell). The temperatures of the key locations used in the analysis are:

Top center of the lid	=	290°F
Top outer diameter of the lid	=	270°F
Bottom center of the lid	=	310°F
Bottom center of the bottom plate	=	250°F
Bottom outer diameter of the bottom plate	=	170°F
Mid-elevation of the canister shell	=	500°F

The temperatures for all nodes in the canister model were obtained by the solution of the steady state thermal conduction problem. These temperatures were selected to envelope the temperature differences experienced by the canister for storage and transfer conditions as calculated in the thermal analysis presented in MPC FSAR Chapter 4. Additionally, canister temperatures used for determining allowable stress values were selected to envelope the maximum temperatures experienced by the canister during storage and transfer conditions. Specifically, allowable stresses were selected at temperatures of 400°F and 350° for the bottom plate center and outer edge, respectively; 400°F and 300°F for canister shell and the lid respectively.

The maximum stresses in the MPC-LACBWR canister resulting from the bounding applied temperature gradient, which are classified as secondary stresses in accordance with ASME Code, are summarized in Table 3.A.4.4.1-1. The canister thermal load is evaluated in combination with dead weight, internal pressure, and normal handling loads, as described in Section 3.A.4.4.1.5.

3.A.4.4.1.2 Canister Dead Weight Load Analysis

The MPC-LACBWR canister is evaluated for dead weight load using the finite element model described in Section 3.A.4.4.1.1. Under storage dead load, the canister rests on the concrete cask pedestal base plate. The canister support condition is simulated in the model using gap elements on the outside surface of the canister bottom plate.

The dead weight of the canister contents (i.e., fuel and basket assembly) are modeled as a uniform pressure load of 10.8 psi acting on the inside surface of the canister bottom plate. The canister contents dead weight pressure is based on a bounding canister contents weight of 41.3 kips. An acceleration load of 1g is applied to the model in the axial direction (Y) to simulate the dead load of the canister. Dead weight is evaluated in combination with normal thermal, internal pressure, and normal handling loads, as described in Section 3.A.4.4.1.5.

The MPC-LACBWR canister includes a support ring that is designed to support the weight of the lid prior to making the closure weld. The MPC-LACBWR and Yankee-MPC canister lid support rings, their attachment welds, and their structural interfaces with the lids are the same. Furthermore, the weight of the MPC-LACBWR canister lid (7,700 lb.) is less than the combined weight of the Yankee-MPC canister shield lid and structural lid (8,639 lb.), which was used in the evaluation of the support ring for Yankee-MPC canister. Therefore, the dead weight stresses in the MPC-LACBWR canister lid support ring and its attachment welds are bounded by those calculated for the Yankee-MPC canister support ring in MPC FSAR Section 3.4.4.1.2.

3.A.4.4.1.3 Canister Internal Pressure Analysis

The MPC-LACBWR canister is evaluated for a design pressure load of 12 psig using the finite element model described in Section 3.A.4.4.1.1. This internal pressure load bounds the calculated internal pressure for normal and off-normal conditions of storage. The design internal pressure of 12 psig is applied to the inside surfaces of the canister shell, bottom plate and the lid.

The maximum membrane and membrane plus bending stresses in the MPC-LACBWR canister resulting from the 12 psig internal pressure load are summarized in Tables 3.A.4.4.1-2 and 3.A.4.4.1-3, respectively. The canister internal pressure load is evaluated in combination with normal thermal, dead weight, and normal handling loads, as described in Section 3.A.4.4.1.5.

3.A.4.4.1.4 Canister Handling Analysis

The MPC-LACBWR canister is evaluated for handling loads using the finite element model described in Section 3.A.4.4.1.1. Normal handling of the canister was simulated by restraining the model at three lift points and applying a 1.1g acceleration load to the model in the axial direction, which includes a 10% dynamic load factor. Although the canister is designed to be lifted using a 6-point lifting configuration, the analysis is conservatively performed based on a 3-point lifting configuration. Thus, the finite element model is restrained in the axial direction at the three nodes (120° apart) located on the lid outside surface at the locations of the attachment holes for the lifting devices.

The maximum membrane and membrane plus bending stresses in the MPC-LACBWR canister resulting from the dead weight plus normal handling load are summarized in Tables 3.A.4.4.1-4 and 3.A.4.4.1-5, respectively. The canister normal handling load is evaluated in combination with normal thermal, dead weight, and internal loads, as described in Section 3.A.4.4.1.5.

3.A.4.4.1.5 Canister Load Combination

The MPC-LACBWR canister is evaluated for the combined effects of thermal, dead weight, internal pressure, and handling loads for normal conditions of storage using the finite element model described in Section 3.A.4.4.1.1. The controlling load combination for normal conditions includes vertical handling, in which the canister is suspended by the lid, as discussed in Section 3.A.4.4.1.4. The thermal, internal pressure, and dead weight/handling loads discussed in Sections 3.A.4.4.1.1, 3.A.4.4.1.3, and 3.A.4.4.1.4, respectively, are simultaneously applied to the model.

The resulting maximum stresses in the canister for combined loads are summarized in Tables 3.A.4.4.1-6, 3.A.4.4.1-7, and 3.A.4.4.1-8 for primary membrane, primary membrane plus primary bending, and primary membrane plus primary bending plus secondary stresses, respectively. The results show that the MPC-LACBWR canister maintains positive margins of safety for the controlling normal load combination.

As noted in Table 3.A.4.4.1-7, the bending stress at the bottom plate-to-shell junction is classified as secondary stress in accordance with Table NB-3217-1 if the edge moment is not required to maintain the bending stress at the center of the bottom plate to within acceptable limits. To demonstrate this, the bending stress at the center of the bottom plate is determined using hand calculations (Roark), treating it as a simply supported circular plate subjected to a uniform pressure load. For a combined pressure load of 23.88 psi from dead weight plus normal handling (11.88 psi) and design internal pressure (12 psig), the membrane plus bending stress at the center of the bottom plate is calculated to be 23.87 ksi. The allowable primary membrane plus bending stress intensity, based on Type 304 stainless steel properties at a bounding design temperature of 400°F, is 27.9 ksi. Therefore, the design margin for primary membrane plus bending at the center of the bottom plate is 0.17 (= 27.9/23.87 – 1).

3.A.4.4.1.6 Canister Fatigue Evaluation

This section discusses the fatigue evaluation of the MPC-LACBWR canister for the effects of thermal and mechanical cyclic loading conditions of storage. The fatigue evaluation of the canister and basket are based on the criteria presented in ASME Code, Section III, Subparagraphs NB-3222.4 and NG-3222.4, respectively.

In accordance with Subparagraph NB-3222.4(d), the MPC-LACBWR canister shell is concluded to satisfy the limits on peak stress intensity, as governed by fatigue, because the following six conditions are met:

Condition 1 – Atmospheric to Service Pressure Cycle:

The specified number of times that the pressure will be cycled from atmospheric pressure to service pressure and back to atmospheric pressure during normal service must not exceed the number of allowable cycles on the applicable ASME code fatigue curve corresponding to an S_a value of $3S_m$. The MPC-LACBWR canister pressure is cycled between atmospheric and service pressure only twice during its 50-year life (i.e., once when the canister is sealed and once when it is opened). Therefore, the atmospheric to service pressure cycle loading of the canister and basket does not cause fatigue concerns, and this condition is satisfied.

Condition 2 – Normal Service Pressure Fluctuation:

The specified full range of pressure fluctuations during normal service (based upon 10^6 service cycles) must not exceed:

$$P_f = \frac{1}{3} \times P_d \times \left(\frac{S_a}{S_m}\right) = \frac{12.0 \times 28.2}{3 \times 19.4} = 5.81 \text{ psig}$$

where:

S_a = 28.2 ksi, Value from the design fatigue curve for 10^6 service cycles.

S_m = 19.4 ksi, Allowable stress intensity for Type 304 stainless steel at 350°F.

P_d = 12.0 psig, Design pressure.

The maximum pressure differential for the canister occurs between off-normal and storage conditions. For normal, off-normal, and transfer conditions the maximum pressure differential is:

ΔP = 12.0 - 8.7 = 3.3 psig < 5.81 psig.

Therefore, this criterion is satisfied.

Condition 3 – Temperature Difference: Startup and Shutdown:

This condition is not applicable. It is only required for power plant startup and shutdown processes.

Condition 4 – Temperature Difference: Normal and Off-Normal Service:

The ASME Code specifies that temperature excursions are not significant if the temperature difference between two adjacent points does not change by more than the quantity:

$$\Delta T = \frac{S_a}{2E\alpha} = 58°F$$

where:

S_a = 28,200 psi, Value obtained from the design fatigue curve for 10^6 service cycles

E = 26.8×10^6 psi, Modulus of elasticity at 350°F

α = 9.1×10^{-6} in./in./°F, Coefficient of linear thermal expansion at 350°F

For temperature differences on surfaces of revolution in the meridional (axial) direction, adjacent points are defined as points that are less than the distance $2\sqrt{Rt}$ apart, where R is the radius measured normal to the surface from the axis of rotation to the mid-wall and t is the thickness of the part at the point under consideration. For temperature differences on surfaces of revolution in the circumferential direction and on flat parts, such as flanges and flat heads, adjacent points are defined as any two points on the same surface.

The greatest cyclic temperature difference will occur between the off-normal, severe hot (105°F ambient temperature) and the off-normal, severe cold (-40°F ambient temperature) conditions as evaluated in the thermal evaluation. Accident temperature conditions are not applicable.

At the hot condition, the canister bottom plate temperature varies from 347°F at its center to 288°F at its outer radius, a ΔT of 59°F. At the cold condition, the canister bottom plate temperature varies from 272°F at its center to 213°F at its outer radius, a ΔT of 59°F. Therefore, in cycling from 105°F ambient to -40°F ambient conditions, the ΔT between adjacent points changes by less than 1°F. Since this is less than the 58°F, it is not considered to be a significant excursion. Heat transfer is uniform around the circumference; therefore, no cyclic ΔT exists between adjacent points on a circumference of the canister shell.

At the hot condition the canister shell temperature varies linearly from 365°F at its center to 250°F at its top, a ΔT of 115°F. At the cold condition, the canister shell temperature varies from 280°F at its center to 130°F at its top, a ΔT of 150°F. The distance between adjacent points is:

$$d_p = 2\sqrt{Rt} = 8.375 \text{ in.}$$

where:

R = 35.07 in, Mean radius of the canister shell.

t = 0.50 in., Thickness of the canister shell.

For the severe hot condition (105°F ambient temperature), the ΔT between the center of the canister and the end of the canister is 115°F. Thus, the ΔT of adjacent points is 17.0°F [i.e., 115°F/58 in.)(8.375 in.)].

For the severe cold condition (-40°F ambient temperature), the ΔT between the center of the canister and the end of the canister is 150°F. Thus, the ΔT of adjacent points is 21.7°F (150°F/58 in.)(8.375 in.).

Therefore, in cycling from severe hot to severe cold conditions, the ΔT between adjacent points changes by 4.7°F (i.e., 21.7°F - 17°F). Since this is less than the 58°F, it is not considered to be a significant excursion. Therefore, this condition is satisfied for the canister.

In storage, the basket is isolated from the influence of environmental temperature excursions by the canister. Any temperature differences within the basket structure are bounded by the evaluation of the temperature differences evaluated for the canister.

Condition 5 – Temperature Difference Between Dissimilar Materials:

The canister is constructed of stainless steels that have the same Young's modulus and coefficient of thermal expansion. The basket is constructed of several materials. However, all materials except the support disks are free to expand, thus relieving any thermal stress concentration.

Condition 6 – Mechanical Loads:

Mechanical loads are not applied to the storage cask and canister during storage conditions. Therefore, no further evaluation is required.

3.A.4.4.1.7 Canister Pressure Test

The MPC-LACBWR canister is hydrostatically pressure tested following closure lid welding in accordance with ASME Code, Section III, NB-6000. The design pressure is 12 psig for the MPC-LACBWR canister. In accordance with NB-6221, the test pressure applied is 25% higher than the design pressure, or 15.0 psig. The MPC-LACBWR canister stresses resulting from the pressure test are evaluated in accordance with the requirements of NB-3226. The pressure test is conducted only once, and therefore, need not be considered in the fatigue analysis per NB-3226(e).

The maximum primary membrane (P_m) and primary membrane plus bending (P_m+P_b) stress intensities in the MPC-LACBWR canister due to the 15 psig test internal pressure are determined by multiplying the maximum stress intensities calculated for normal design internal pressure in Section 3.A.4.4.1.3 by the ratio of the pressure loads (i.e., 15/12). Therefore, the maximum calculated P_m and P_m+P_b stress intensities in the MPC-LACBWR canister due to the 15 psig test internal pressure are 5.0 ksi (4.0 ksi × 15/12) and 20.7 ksi (16.57 ksi × 15/12), respectively.

NB-3226 requires that the primary membrane stress intensity (P_m) not exceed 90% of the material yield strength (S_y) at the test temperature, or 19.4 ksi (i.e., 0.9×21.6 ksi) based on Type 304 stainless steel at a bounding temperature of 350°F. Since the calculated P_m stress is significantly less than the allowable stress, the allowable stress criteria is satisfied.

NB-3226 also requires that the primary membrane plus bending stress intensity (P_m+P_b) not exceed 135% of the material yield strength (S_y) at the test temperature when P_m is lower than 0.67S_y. Therefore, P_m+P_b is limited to 29.2 ksi (i.e., 1.35×21.6 ksi) based on Type 304 stainless steel at a bounding temperature of 350°F. Therefore, the allowable the allowable stress criteria for P_m+P_b is satisfied.

Figure 3.A.4.4.1-1 MPC-LACBWR Canister Shell Assembly Finite Element Model

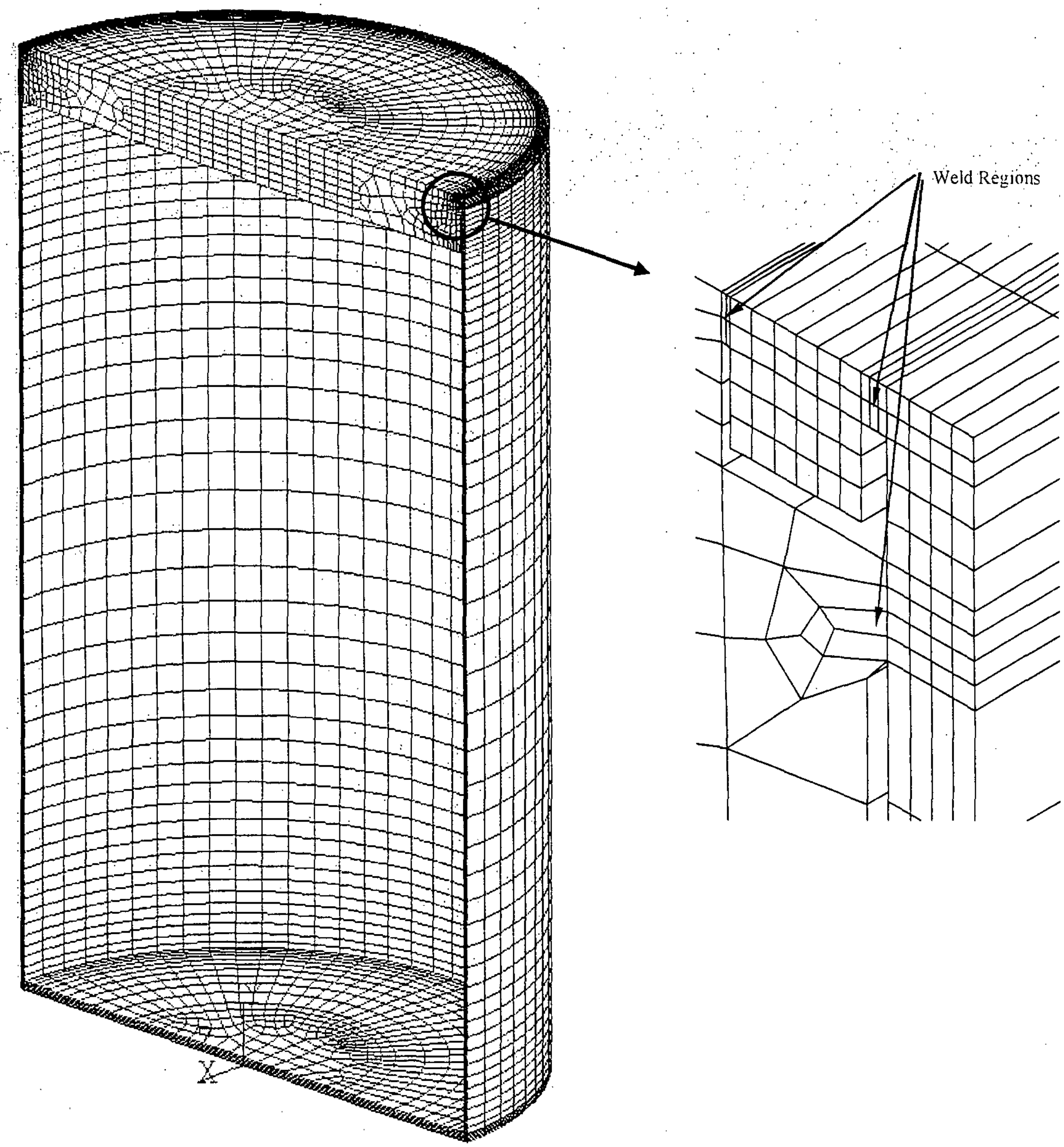
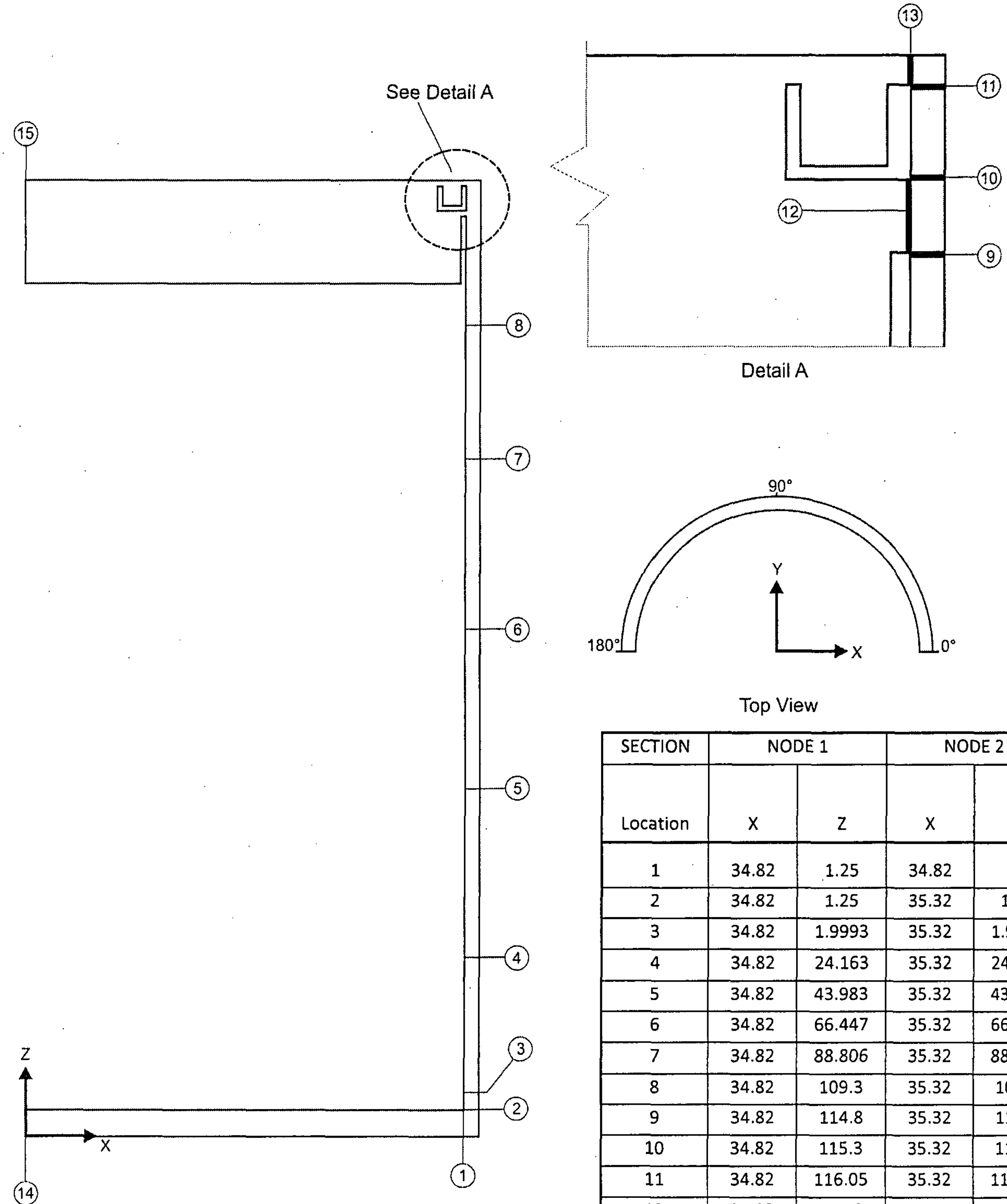


Figure 3.A.4.4.1-2 MPC-LACBWR Canister Finite Element Model Stress Section Locations



SECTION	NODE 1		NODE 2	
Location	X	Z	X	Z
1	34.82	1.25	34.82	0
2	34.82	1.25	35.32	1.25
3	34.82	1.9993	35.32	1.9993
4	34.82	24.163	35.32	24.163
5	34.82	43.983	35.32	43.983
6	34.82	66.447	35.32	66.447
7	34.82	88.806	35.32	88.806
8	34.82	109.3	35.32	109.3
9	34.82	114.8	35.32	114.8
10	34.82	115.3	35.32	115.3
11	34.82	116.05	35.32	116.05
12	34.82	114.8	34.82	115.3
13	34.82	116.05	34.82	116.3
14	0	1.25	0	0
15	0	116.3	0	109.3

Table 3.A.4.4.1-1 Summary of MPC-LACBWR Secondary Stresses, Thermal Load

Section No.[1]	Component Stresses (ksi)						Stress Intensity (ksi)
	SX	SY	SZ	SXY	SYZ	SXZ	
1	-0.1	1.6	0.4	0.0	0.0	-0.1	1.70
2	0.3	1.3	-0.3	0.0	0.0	-0.1	1.59
3	0.0	0.6	0.0	0.0	0.0	0.0	0.70
4	0.0	0.0	0.0	0.0	0.0	0.0	0.00
5	0.0	0.3	0.0	0.0	0.0	0.0	0.33
6	0.0	-0.1	0.0	0.0	0.0	0.0	0.12
7	0.0	0.0	0.0	0.0	0.0	0.0	0.00
8	0.0	-0.5	0.0	0.0	0.0	0.0	0.50
9	0.6	0.1	-0.3	0.0	0.0	0.2	1.03
10	-0.2	0.3	-0.1	0.0	0.0	0.2	0.66
11	-0.2	0.9	0.0	0.0	0.0	0.1	1.22
12	0.0	0.6	1.2	0.0	0.0	0.1	1.14
13	-0.3	0.8	-0.1	0.0	0.0	0.0	1.08
14	-7.6	-7.7	-0.8	0.0	0.0	0.0	6.89
15	-1.4	-1.5	-0.4	0.0	0.0	0.0	1.02

1. Canister section locations are shown in Figure 3.A.4.4.1-2.

Table 3.A.4.4.1-2 Summary of MPC-LACBWR Membrane (P_m) Stresses, Internal Pressure Load

Section No.[1]	Component Stresses (ksi)						Stress Intensity (ksi)
	SX	SY	SZ	SXY	SYZ	SXZ	
1	-0.2	0.9	2.6	0.0	0.0	-0.4	2.91
2	1.9	-1.6	-1.2	0.0	0.0	-0.6	3.55
3	-0.3	-3.3	0.4	0.0	0.0	0.6	4.00
4	0.0	0.8	0.4	0.0	0.0	0.0	0.83
5	0.0	0.8	0.4	0.0	0.0	0.0	0.84
6	0.0	0.8	0.4	0.0	0.0	0.0	0.84
7	0.0	0.8	0.4	0.0	0.0	0.0	0.84
8	0.0	0.7	0.4	0.0	0.0	0.0	0.72
9	0.1	0.2	0.2	0.0	0.0	0.0	0.12
10	-0.1	0.1	0.0	0.0	0.0	0.0	0.20
11	0.1	0.2	0.0	0.0	0.0	0.0	0.17
12	0.0	0.1	0.1	0.0	0.0	-0.2	0.32
13	0.0	0.2	0.1	0.0	0.0	0.0	0.17
14	0.2	0.2	-0.1	0.0	0.0	0.0	0.31
15	0.0	0.0	0.0	0.0	0.0	0.0	0.00

1. Canister section locations are shown in Figure 3.A.4.4.1-2.

Table 3.A.4.4.1-3 Summary of MPC-LACBWR Membrane + Bending (P_m+P_b) Stresses, Internal Pressure Load

Section No.[1]	Component Stresses (ksi)						Stress Intensity (ksi)
	SX	SY	SZ	SXY	SYZ	SXZ	
1	2.0	0.9	6.7	0.0	0.0	0.2	5.88
2	0.6	-5.9	-14.1	0.0	0.0	-1.3	15.01
3	-1.0	1.2	15.5	0.0	0.0	0.8	16.57
4	0.0	0.8	0.4	0.0	0.0	0.0	0.84
5	0.0	0.8	0.4	0.0	0.0	0.0	0.85
6	0.0	0.8	0.4	0.0	0.0	0.0	0.85
7	0.0	0.8	0.4	0.0	0.0	0.0	0.85
8	0.0	0.8	0.6	0.0	0.0	0.0	0.78
9	0.1	0.4	0.8	0.0	0.0	-0.1	0.77
10	-0.2	0.0	-0.1	0.0	0.0	-0.1	0.25
11	0.2	0.2	0.1	0.0	0.0	0.0	0.18
12	0.1	0.2	0.3	0.0	0.0	-0.3	0.54
13	-0.2	0.1	0.0	0.0	0.0	0.0	0.32
14	7.4	7.5	-0.1	0.0	0.0	0.0	7.67
15	0.4	0.4	0.0	0.0	0.0	0.0	0.38

1. Canister section locations are shown in Figure 3.A.4.4.1-2.

Table 3.A.4.4.1-4 Summary of MPC-LACBWR Membrane (P_m) Stresses, Dead Load + Normal Handling

Section No.[1]	Component Stresses (ksi)						Stress Intensity (ksi)
	SX	SY	SZ	SXY	SYZ	SXZ	
1	-0.2	0.9	2.7	0.0	0.0	-0.4	2.98
2	1.9	-1.8	-1.2	0.0	0.0	-0.6	3.76
3	-0.3	-3.6	0.5	0.0	0.0	0.6	4.37
4	0.0	0.0	0.5	0.0	0.0	0.0	0.50
5	0.0	0.0	0.5	0.0	0.0	0.0	0.54
6	0.0	0.0	0.6	0.0	0.0	0.0	0.62
7	0.0	0.0	0.7	0.0	0.0	0.0	0.72
8	0.0	-0.1	0.9	0.0	0.0	0.0	0.96
9	-0.2	0.3	0.7	0.0	0.0	-0.1	0.91
10	-0.1	0.3	0.2	0.0	0.0	-0.2	0.52
11	0.3	0.6	0.0	0.0	0.0	-0.1	0.56
12	0.0	0.1	-0.5	0.0	0.0	-0.5	1.03
13	0.2	0.6	0.2	0.0	0.0	0.0	0.43
14	0.2	0.2	0.0	0.0	0.0	0.0	0.25
15	0.0	0.0	0.0	0.0	0.0	0.0	0.00

1. Canister section locations are shown in Figure 3.A.4.4.1-2.

Table 3.A.4.4.1-5 Summary of MPC-LACBWR Membrane + Bending (P_m+P_b) Stresses, Dead Load + Normal Handling

Section No.[1]	Component Stresses (ksi)						Stress Intensity (ksi)
	SX	SY	SZ	SXY	SYZ	SXZ	
1	2.0	0.7	6.9	0.0	0.0	0.2	6.17
2	0.7	-6.2	-14.5	0.0	0.0	-1.3	15.34
3	-1.0	1.0	16.0	0.0	0.0	0.8	17.08
4	0.0	-0.1	0.5	0.0	0.0	0.0	0.54
5	0.0	-0.1	0.5	0.0	0.0	0.0	0.60
6	0.0	-0.1	0.6	0.0	0.0	0.0	0.69
7	0.0	-0.1	0.7	0.0	0.0	0.0	0.79
8	0.0	0.0	1.0	0.0	0.0	0.0	1.04
9	-0.5	0.2	0.3	0.0	0.0	-0.4	1.19
10	-0.1	0.7	1.2	0.0	0.0	0.0	1.28
11	0.8	0.7	0.0	0.0	0.0	-0.1	0.81
12	-0.3	0.0	-0.3	0.0	0.0	-0.6	1.25
13	-0.7	0.3	0.0	0.0	0.0	0.1	0.97
14	8.0	8.2	0.0	0.0	0.0	0.0	8.22
15	-0.1	-0.1	0.0	0.0	0.0	0.0	0.09

1. Canister section locations are shown in Figure 3.A.4.4.1-2.

Table 3.A.4.4.1-6 Summary of MPC-LACBWR Primary Membrane (P_m) Stresses, Dead Load + Normal Handling + Internal Pressure

Section No.[1]	Component Stresses (ksi)						Stress Intensity (ksi)	Allowable Stress Intensity (ksi)	Margin of Safety
	SX	SY	SZ	SXY	SYZ	SXZ			
1	-0.4	1.8	5.3	0.0	0.0	-0.9	5.92	19.27	2.26
2	3.8	-3.3	-2.3	0.0	0.0	-1.1	7.35	19.27	1.62
3	-0.7	-6.9	0.9	0.0	0.0	1.1	8.42	19.26	1.29
4	0.0	0.8	0.9	0.0	0.0	0.0	0.91	18.97	19.90
5	0.0	0.8	1.0	0.0	0.0	0.0	0.96	18.71	18.44
6	0.0	0.8	1.0	0.0	0.0	0.0	1.03	18.89	17.27
7	0.0	0.8	1.1	0.0	0.0	0.0	1.14	19.38	16.05
8	0.0	0.7	1.3	0.0	0.0	0.0	1.33	19.84	13.90
9	-0.1	0.5	0.9	0.0	0.0	-0.1	1.03	19.96	18.40
10	-0.2	0.4	0.2	0.0	0.0	-0.2	0.71	19.97	27.26
11	0.4	0.7	0.0	0.0	0.0	-0.1	0.73	19.98	26.47
12	0.0	0.2	-0.4	0.0	0.0	-0.6	1.29	15.97[2]	11.39
13	0.2	0.8	0.3	0.0	0.0	0.0	0.60	19.98	32.44
14	0.5	0.5	0.0	0.0	0.0	0.0	0.52	18.60	35.05
15	0.0	0.0	0.0	0.0	0.0	0.0	0.00	20.00	>100

1. Canister section locations are shown in Figure 3.A.4.4.1-2.
2. The closure weld allowable stress intensity includes a weld strength reduction factor of 0.8.

Table 3.A.4.4.1-7 Summary of MPC-LACBWR Primary Membrane + Bending (P_m+P_b) Stresses, Dead Load + Normal Handling + Internal Pressure

Section No.[1]	Component Stresses (ksi)						Stress Intensity (ksi)	Allowable Stress Intensity (ksi)	Margin of Safety
	SX	SY	SZ	SXY	SYZ	SXZ			
1	4.1	1.6	13.7	0.0	0.0	0.4	12.12	28.91	1.39
2	3.8	-3.3	-2.3	0.0	0.0	-1.1	7.35[2]	28.91	2.93
3	-0.7	-6.9	0.9	0.0	0.0	1.1	8.42[2]	28.89	2.43
4	0.0	0.8	0.9	0.0	0.0	0.0	0.92	28.46	29.84
5	0.0	0.9	1.0	0.0	0.0	0.0	0.99	28.07	27.34
6	0.0	0.9	1.1	0.0	0.0	0.0	1.08	28.34	25.34
7	0.0	0.9	1.2	0.0	0.0	0.0	1.19	29.07	23.43
8	0.0	0.7	1.5	0.0	0.0	0.0	1.48	29.75	19.17
9	-0.4	0.5	1.2	0.0	0.0	-0.6	1.92	29.94	14.58
10	-0.1	0.8	1.4	0.0	0.0	0.0	1.43	29.95	19.96
11	1.0	0.9	0.1	0.0	0.0	-0.2	0.96	29.96	30.19
12	-0.2	0.2	0.0	0.0	0.0	-0.9	1.77	23.95[3]	12.54
13	-0.9	0.4	0.0	0.0	0.0	0.1	1.30	29.96	22.14
14	15.8	16.1	-0.1	0.0	0.0	0.0	16.14	27.90	0.73
15	0.5	0.5	0.0	0.0	0.0	0.0	0.47	30.00	63.39

1. Canister section locations are shown in Figure 3.A.4.4.1-2.
2. The bending stress at a gross structural discontinuity (flat head at junction to shell) is classified as a secondary stress. Reference 1995 ASME Boiler & Pressure Vessel Code, Division 1– Subsection NB, Table NB-3217-1.
3. The closure weld allowable stress intensity includes a weld strength reduction factor of 0.8.

Table 3.A.4.4.1-8 Summary of MPC-LACBWR Primary Membrane + Bending + Secondary (P_m+P_b+Q) Stresses, Dead Load + Normal Handling + Internal Pressure + Normal Thermal

Section No.[1]	Component Stresses (ksi)						Stress Intensity (ksi)	Allowable Stress Intensity (ksi)	Margin of Safety
	SX	SY	SZ	SXY	SYZ	SXZ			
1	4.4	3.5	14.8	0.0	0.0	0.3	11.36	57.82	4.09
2	1.4	-11.8	-31.4	0.0	0.0	-2.9	33.30	57.82	0.74
3	-2.2	3.7	34.8	0.0	0.0	1.6	37.09	57.79	0.56
4	0.0	0.8	0.9	0.0	0.0	0.0	0.94	56.91	59.49
5	-0.2	1.2	0.9	0.0	0.0	0.0	1.39	56.14	39.36
6	0.0	0.7	1.2	0.0	0.0	0.0	1.21	56.68	45.81
7	0.0	1.0	1.2	0.0	0.0	0.0	1.19	58.14	47.94
8	0.0	0.3	1.5	0.0	0.0	0.0	1.54	59.51	37.59
9	0.2	-1.0	-3.8	0.0	0.0	0.4	4.07	59.87	13.73
10	-0.7	0.5	0.9	0.0	0.0	0.1	1.65	59.90	35.23
11	-0.1	1.6	-0.2	0.0	0.0	0.1	1.85	59.93	31.32
12	0.8	1.2	1.9	0.0	0.0	-0.9	2.09	47.91[2]	21.94
13	-0.8	1.3	0.0	0.0	0.0	0.1	2.14	59.93	27.06
14	-18.4	-18.9	0.6	0.0	0.0	0.0	19.45	55.80	1.87
15	-3.3	-3.3	-1.1	0.0	0.0	0.0	2.21	60.00	26.14

1. Canister section locations are shown in Figure 3.A.4.4.1-2.
2. The closure weld allowable stress intensity includes a weld strength reduction factor of 0.8.

3.A.4.4.2 MPC-LACBWR Fuel Basket Analysis

3.A.4.4.2.1 Fuel Basket Support Disk Evaluation

Under normal conditions, the MPC-LACBWR canister is stored in a vertical orientation. In this orientation each basket support disk is supported axially by the eight basket tie-rods and must support its own weight. Since all of the support disks experience the same loading and boundary conditions during storage and handling, a bounding evaluation is performed for a single 5/8-inch thick support disk. The stresses in the support disks located at the top and bottom ends of the basket are bounded by the stresses calculated for the 5/8-inch thick interior support disk because they are thicker and experience much lower temperature gradients than the support disks located at the center of the basket.

The stresses in the MPC-LACBWR fuel basket support disks due to thermal, dead weight, and handling loads for normal conditions of storage are calculated using the ANSYS finite element model shown in Figure 3.A.4.4.2-1. The model is constructed from ANSYS SHELL63 elastic shell elements with a thickness of 5/8 inch. The support disk model nodes located at the centerline of each tie-rod are restrained in the vertical direction (i.e., UY=0).

Dead weight and normal handling loads are accounted for by applying a 1.1*g* vertical acceleration to the model. Thermal stresses in the support disk are conservatively calculated using a temperature distribution that envelopes those experienced by the support disk during storage and transfer conditions. The bounding thermal gradient ranges from 500°F at the disk center to 150°F at the disk outer diameter. Since general thermal stress is classified as secondary by the ASME Code, thermal loads are only combined with dead weight and handling loads for the evaluation of primary plus secondary stresses.

Linearized stresses in the MPC-LACBWR canister support disk are evaluated at its critical sections; those which are located at the mid-length and ends of the disk ligaments and between the corners of the exterior slots and the support disk outer edge. The maximum stresses in the MPC-LACBWR canister support disk due to the combined effects of dead weight, normal handling, and thermal load are presented in Table 3.A.4.4.2-1. The results show that the support disk maintains large positive margins of safety for normal storage conditions.

3.A.4.4.2.2 Fuel Basket Weldment Evaluation

This section discusses the stress analysis of the MPC-LACBWR canister top and bottom weldments for normal thermal, dead weight, and normal handling loads. Under normal conditions, the top weldment only supports its own weight, whereas the bottom weldment supports its own weight and the weight of the 68 fuel tubes. The fuel assemblies and damaged fuel cans, which extend through the disk slots in the bottom weldment disk, are supported axially by the canister bottom end plate. The top and bottom weldments are both supported axially by the eight tie-rods. In addition, the bottom weldment is supported by the six ligament support plates.

The stresses in the MPC-LACBWR canister basket top and bottom weldments due to normal storage and handling conditions are determined using the ANSYS finite element analysis program. The finite element models used for the stress analyses of the top and bottom end weldments are shown in Figures 3.A.4.4.2-2 and 3.A.4.4.2-3, respectively. Both models are constructed entirely from ANSYS SHELL63 elastic shell elements with thicknesses input as real constants. The top weldment model includes the 1-inch thick disk plate, 3/8-inch thick outer ring, 3/8-inch thick ligament support plates, and 3/8-inch thick stiffener angles. The top weldment model nodes located at the centerline of each tie-rod are restrained in the vertical direction (i.e., UY=0.) The bottom weldment model includes the 1-inch thick disk plate and eight ½-inch thick ligament support plates. The bottom weldment model nodes located at the centerline of each tie-rod are restrained in the vertical direction (i.e., UY=0). In addition, the bottom weldment model is supported axially at the end nodes of each ligament support plate.

Dead weight and normal handling loads are accounted for by applying a 1.1*g* vertical acceleration to the models. In addition, a vertical line load is applied to the perimeter of each opening in the bottom weldment to account for the 1.1*g* dead weight plus handing load from the fuel tubes. The line loads applied to the 36 opening on the interior of the bottom weldment disk (i.e., the innermost 6×6 array) are based on a bounding standard fuel tube weight of 44 pounds. The line loads on the remaining 32 exterior openings are conservatively based on a bounding DFC fuel tube weight of 63 pounds. Thermal stresses in the top and bottom weldments are conservatively calculated using the same bounding temperature distribution used for the analysis of the support disks, as discussed in Section 3.A.4.4.2.1. Since general thermal stress is classified as secondary by the ASME Code, thermal loads are only combined with dead weight and handling loads for the evaluation of primary plus secondary stresses.

Linearized stresses in the MPC-LACBWR canister top and bottom weldments are evaluated at its critical sections; those which are located at the mid-length and ends of the disk ligaments and between the corners of the exterior slots and the support disk outer edge. The results of the MPC-LACBWR canister top and bottom weldment structural analyses for dead weight, normal handling, and thermal load are presented in Table 3.A.4.4.2-1. The results show that the top and bottom weldments both maintain large positive margins of safety for normal storage conditions.

3.A.4.4.2.3 Fuel Tube Evaluation

During storage, the fuel tubes in the MPC-LACBWR canister basket support their own weight. The fuel assemblies are supported axially by the canister bottom plate and do not load the fuel tubes. The structural evaluation of the MPC-LACBWR fuel tubes for normal conditions considers the stresses due to dead weight and normal handling loads. Thermal stress in the fuel tubes is considered to be negligible since the tubes are free to expand within the basket in both the axial and radial directions. The combined load due to dead weight and normal handling is a 1.1*g* vertical acceleration.

The maximum compressive stresses at the bottom end of the standard fuel tubes and fuel tubes for Damaged Fuel Can (DFC) due to a 1.1*g* vertical load are calculated by dividing the total weight of the fuel tube (including the neutron absorbers and cladding) by the tube cross-section area. The maximum compressive stress in the standard fuel tube is calculated to be 45 psi, based on a weight of 46 pounds and a 1.114 in^2 cross-section area. Similarly, the maximum compressive stress in the DFC fuel tube is calculated to be 59 psi, based on a weight of 62 pounds and a 1.162 in^2cross-section area. The maximum stresses in the fuel tubes are much lower than the yield strength of Type 304 stainless steel material (e.g., S_y = 17,300 psi at 750°F) from which they are made. Therefore, it is concluded that the fuel tubes will remain in position and maintain the position of the neutron absorbers under normal storage conditions.

3.A.4.4.2.4 Damage Fuel Can Evaluation

The MPC-LACBWR damaged fuel can (DFC) design is similar to the fuel tube design, but has a screened bottom plate and a removable lid assembly. Under normal storage and transfer conditions, the weight of the fuel assembly inside the DFC loads the DFC bottom plate, which is uniformly supported by the canister bottom plate. Thus, the only stresses in the DFC during normal storage conditions are those in the DFC tube and the lid assembly. This section discusses the stress analysis of the DFC for normal conditions of storage. The structural evaluation of the DFC lifting devices is presented in Section 3.A.4.3.4.

The stresses in the DFC are conservatively calculated for a 20*g* vertical acceleration load, which bounds the 1.1*g* vertical acceleration due to combined dead weight and normal handling loads. Thermal stress in the DFC is considered to be negligible since the DFC is free to expand within the basket and canister in both the axial and radial directions.

The compressive load (P) on the DFC tube is the combined weight of the lid, side plates, and tube body (49 pounds) multiplied by the 20g acceleration load, or 980 pounds. A bounding compressive load of 1,000 pounds is conservatively assumed. The resulting compressive stress at the bottom end of the DFC tube is 862 psi, based on a 1.16 in^2 tube cross-section area. This stress is classified as primary membrane. The allowable primary membrane stress intensity is limited to S_m for normal conditions, which is 16,700 psi for the DFC tube Type 304 stainless steel material at an upper bound temperature of 600°F. Therefore, the margin of safety for primary membrane stress intensity in the DFC tube for normal storage conditions is large (+18.4).

The maximum bending stress for the DFC lid is conservatively calculated assuming that it behaves as a 5.6-inch long by 1-inch wide by 0.31-inch thick simply supported beam under a uniform line load. The DFC lid line load (w) due to the 20*g* vertical acceleration, conservatively calculated based on a bounding DFC lid weight of 11 lbs, is 7 lb/inch (= 11 x $20g/5.6^2$). The corresponding bending stress in the DFC lid is:

$$f_b = \frac{6(27.5)}{1.0(0.31^2)} = 1{,}717 \text{ psi}$$

The allowable primary membrane plus bending stress intensity is limited to $1.5S_m$ for normal conditions, or 25,050 psi for the DFC tube Type 304 stainless steel material at an upper bound temperature of 600°F. Therefore, the margin of safety for primary membrane stress intensity in the DFC tube for normal storage conditions is large (+13.6).

Figure 3.A.4.4.2-1 MPC-LACBWR Support Disk Finite Element Model

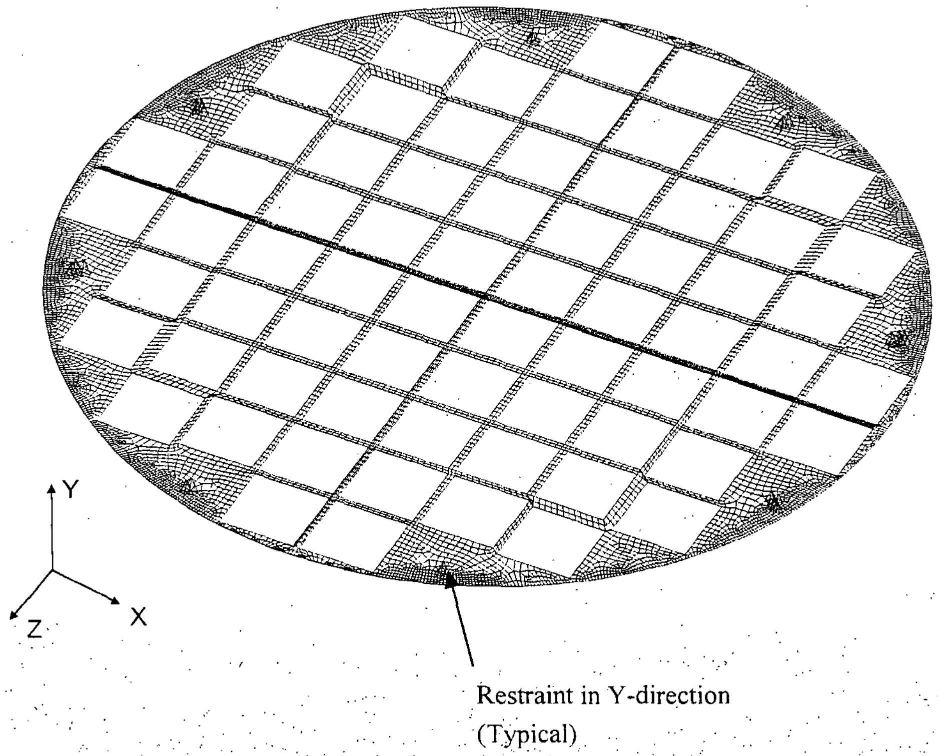


Figure 3.A.4.4.2-2 MPC-LACBWR Top Weldment Finite Element Model

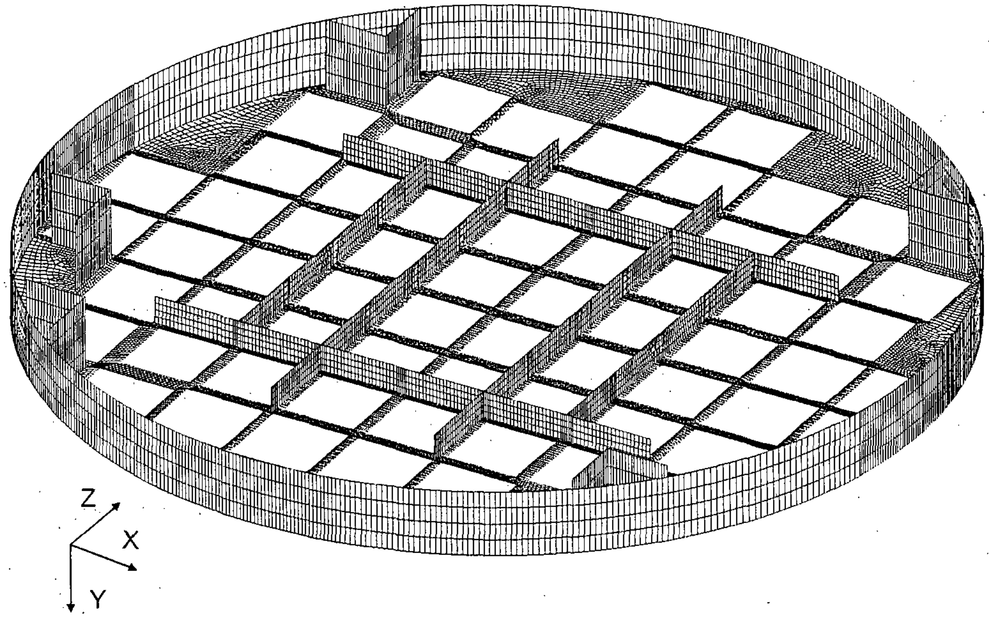


Figure 3.A.4.4.2-3 MPC-LACBWR Bottom Weldment Finite Element Model

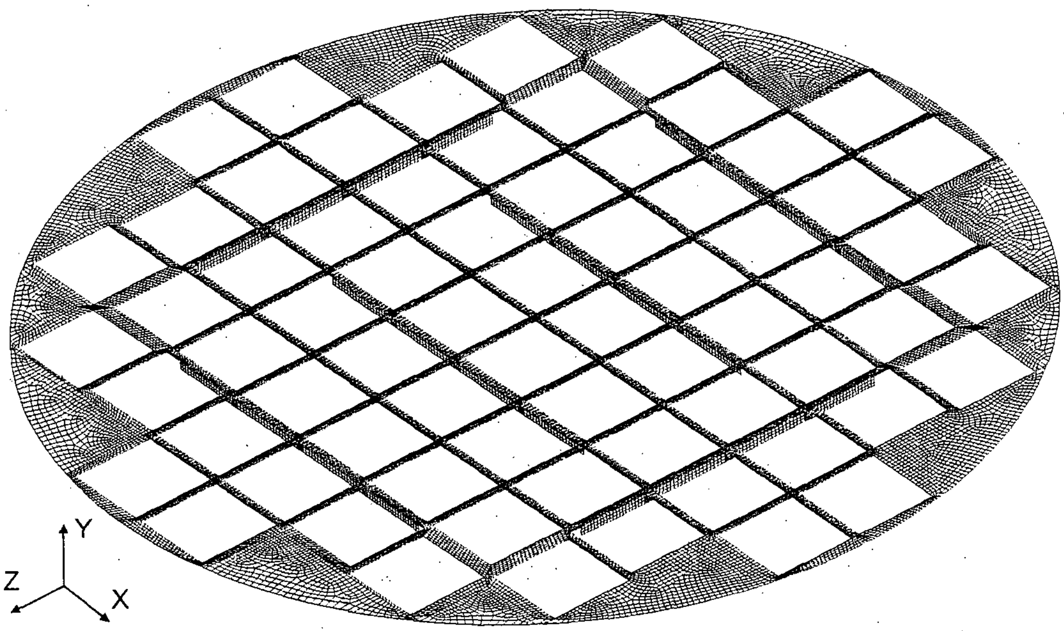


Table 3.A.4.4.2-1 Summary of Maximum Stresses for the MPC-LACBWR Fuel Basket Support Disks and Weldments, Normal Conditions

Canister Basket Component	Load Combination	Stress Intensity		Allowable Stress		Margin of Safety
		Stress Type	Maximum Value (ksi)	Criteria	Allowable Value (ksi)	
Top Weldment	Dead Load	P_m	0.45	S_m	17.5	Large
		P_m+P_b	0.68	$1.5S_m$	26.25	Large
	Dead Load + Thermal	P_m+P_b+Q	31.93	$3.0S_m$	52.5	0.64
	Dead Load + Handling	P_m	0.49	S_m	17.5	Large
		P_m+P_b	0.75	$1.5S_m$	26.25	Large
	Dead Load + Handling + Thermal	P_m+P_b+Q	31.92	$3.0S_m$	52.5	0.64
Bottom Weldment	Dead Load	P_m	1.41	S_m	17.5	Large
		P_m+P_b	1.97	$1.5S_m$	26.25	Large
	Dead Load + Thermal	P_m+P_b+Q	27.45	$3.0S_m$	52.5	0.91
	Dead Load + Handling	P_m	1.41	S_m	17.5	Large
		P_m+P_b	1.98	$1.5S_m$	26.25	Large
	Dead Load + Handling + Thermal	P_m+P_b+Q	27.45	$3.0S_m$	52.5	0.91
Support Disk	Dead Load	P_m	0.5	S_m	42.9	Large
		P_m+P_b	0.75	$1.5S_m$	64.35	Large
	Dead Load + Thermal	P_m+P_b+Q	23.17	$3.0S_m$	128.6	4.55
	Dead Load + Handling	P_m	0.55	S_m	42.9	Large
		P_m+P_b	0.82	$1.5S_m$	64.35	Large
	Dead Load + Handling + Thermal	P_m+P_b+Q	23.16	$3.0S_m$	128.6	4.55

3.A.4.4.3 MPC-LACBWR Vertical Concrete Cask Evaluation

The structural evaluation of the MPC-LACBWR concrete cask for normal conditions of storage is discussed in this section. As discussed in the following sections, the stresses in the concrete cask concrete shell due to dead load, live load, and normal thermal load are shown to be bounded by those calculated for the Yankee-MPC concrete cask. The bounding concrete shell stresses due to normal loads are combined with the MPC-LACBWR concrete cask shell stresses calculated for off-normal and accident conditions in Section 11.A. The maximum stresses for each load condition are factored and combined in accordance with the load combinations defined in MPC FSAR Table 2.2-1, as summarized in Table 3.A.4.4.3-1. The results of the load combination evaluation show that the maximum combined stresses in the MPC-LACBWR concrete cask concrete shell result from load combination 3, and are identical to those from the Yankee-MPC concrete cask concrete shell evaluation. As shown in MPC FSAR Table 3.4.4.2-2, the maximum compressive stress in the concrete and maximum tensile loads in the steel reinforcing bars for this controlling load combination satisfy the applicable requirements of ACI 349-85.

3.A.4.4.3.1 MPC-LACBWR Concrete Cask Dead Load

Under dead weight loading, the concrete cask concrete shell is conservatively assumed to support the entire weight of the concrete cask assembly, including the weight of the inner steel liner shell and the steel pedestal upon which the concrete shell and canister rest. No structural credit is taken for the concrete cask steel liner in the dead weight stress analysis. Since the MPC-LACBWR concrete cask weighs less than the Yankee-MPC concrete cask, and the MPC-LACBWR concrete cask concrete shell has a greater cross-section area than the Yankee-MPC concrete cask, the dead weight stress in the MPC-LACBWR concrete cask concrete shell is bounded by the maximum dead weight stress of -21.4 psi (i.e., compressive) calculated for the Yankee-MPC concrete cask concrete shell in MPC FSAR Section 3.4.4.2.1. The bounding compressive stress in the Yankee-MPC concrete cask concrete shell is conservatively used for the evaluation of the MPC-LACBWR concrete cask concrete shell.

3.A.4.4.3.2 MPC-LACBWR Concrete Cask Live Load

The MPC-LACBWR concrete cask is evaluated for two different design basis live load conditions: (1) snow live load, and (2) transfer cask live load (i.e., weight of the fully loaded transfer cask resting on the top end of the concrete cask.) Live loads are assumed to be resisted only by the concrete cask concrete shell, conservatively neglecting support provided by the concrete cask steel liner shell. The compressive stress in the concrete shell due to live load is calculated by dividing the highest live load force by the concrete shell cross section area. The results of the live load evaluation show that the stresses in the MPC-LACBWR concrete cask concrete shell are not significant.

Since the surface area on the top end of the MPC-LACBWR concrete cask is the same as that of the Yankee-MPC concrete cask, both designs are subjected to the same design basis snow live load, which is shown to be 9 kips in MPC FSAR Section 3.4.4.2.2. Since the snow live load is much lower than the transfer cask live load, the stresses in the concrete cask concrete shell due to snow live load are bounded by those due to the transfer cask live load.

The compressive stress in the MPC-LACBWR concrete cask concrete shell due to transfer cask live load is bounded by the maximum compressive stress of -19.2 psi calculated for the Yankee-MPC concrete cask concrete shell in MPC FSAR Section 3.4.4.2.2. The weight of the loaded MPC-LACBWR transfer cask is slightly lower than the weight of the loaded Yankee-MPC transfer cask. Furthermore, the cross section area of the MPC-LACBWR concrete cask concrete shell is larger than that of the Yankee-MPC concrete cask concrete shell. However, the bounding compressive stress in the Yankee-MPC concrete cask concrete shell is conservatively used for the evaluation of the MPC-LACBWR concrete cask concrete shell.

3.A.4.4.3.3 MPC-LACBWR Concrete Cask Thermal Load

Under normal thermal conditions, the decay heat load from the canister inside the concrete cask is transferred to the environment primarily by passive convective air flow through the concrete cask ventilation ducts. However, some portion of the decay heat is transferred to the environment by conduction through the concrete cask. The thermal resistance of the concrete cask steel and concrete shells produces axial and radial temperature gradients that generate stress within the concrete cask concrete shell. Given that the MPC-LACBWR concrete cask and Yankee-MPC concrete cask designs are very similar; their heat transfer characteristics are also very similar. Therefore, since the design basis canister heat load for the MPC-LACBWR canister is much lower than that of the Yankee-MPC canister (i.e., 4.2 kW vs. 12.5 kW), the temperature gradients and thermal stresses in the MPC-LACBWR concrete cask concrete shell will be much lower than those calculated for the Yankee-MPC concrete cask concrete shell. However, the bounding thermal stresses calculated for the Yankee-MPC concrete cask concrete shell are conservatively used for evaluation of the MPC-LACBWR concrete cask concrete shell. The bounding compressive stresses in the Yankee-MPC concrete cask concrete and tensile loads in the Yankee-MPC concrete cask steel reinforcing bars are calculated in MPC FSAR Section 3.4.4.2.3.

Table 3.A.4.4.3-1 Stress Summary for MPC-LACBWR Concrete Cask Concrete Shell Load Combinations

Load Comb.[1]	**Stress Direction**	**Stress**[2] **(psi)**							**Total Stress**
		Dead	**Live**	**Wind**[3]	**Thermal**[4]	**Earth-quake**[5]	**Tornado**[6]	**Flood**[7]	
Outside Diameter of Concrete Shell									
1	Vertical	-30.0	-32.6	---	---	---	---	---	-62.6
2	Vertical	-22.5	-24.5	---	---	---	---	---	-47.0
3	Vertical	-22.5	-24.5	-14.7	---	---	---	---	-61.7
4	Vertical	-21.4	-19.2	---	---	---	---	---	-40.6
5	Vertical	-21.4	-19.2	---	---	-65.4	---	---	-106.0
7	Vertical	-21.4	-19.2	---	---	---	---	-10.6	-51.2
8	Vertical	-21.4	-19.2	---	---	---	-11.5	---	-52.1
Inside Diameter of Concrete Shell									
1	Vertical	-30.0	-32.6	---	---	---	---	---	-62.6
	Circumferential	---	---	---	---	---	---	---	0.0
2	Vertical	-22.5	-24.5	---	-669.9	---	---	---	-716.9
	Circumferential	---	---	---	-226.6	---	---	---	-226.6
3	Vertical	-22.5	-24.5	-9.9	-669.9	---	---	---	-726.8
	Circumferential	---	---	---	-226.6	---	---	---	-226.6
4	Vertical	-21.4	-19.2	---	-660.1	---	---	---	-700.7
	Circumferential	---	---	---	-127.5	---	---	---	-127.5
5	Vertical	-21.4	-19.2	---	-525.4	-44.9	---	---	-610.9
	Circumferential	---	---	---	-177.7	---	---	---	-177.7
7	Vertical	-21.4	-19.2	---	-525.4	---	---	-7.1	-573.1
	Circumferential	---	---	---	-177.7	---	---	---	-177.7
8	Vertical	-21.4	-19.2	---	-525.4	---	-7.7	---	-573.7
	Circumferential	---	---	---	-177.7	---	---	---	-177.7

Notes:

1 Load combinations are defined in MPC FSAR Table 2.2-1. See Sections 11.A.2.11 and 11.A.2.12 for the evaluation of drop/impact conditions included in load combination 6.
2 Positive stress values indicate tensile stress and negative values indicate compressive stress.
3 Stress results from Section 11.A.2.13 (Tornado) are conservatively used with a load factor of 1.275.
4 Tensile loads developed in the outer region of the concrete cask concrete shell are resisted by the steel reinforcing bars, and therefore, are not shown in this table. Stresses in the concrete cask concrete shell due to accident thermal loading are from Section 11.A.2.10.
5 Earthquake stresses are from Section 11.A.2.2.
6 Tornado stresses are from Section 11.A.2.13.
7 Flood stresses are from Section 11.A.2.6.

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3.A.5 Fuel Rods

The MPC-LACBWR storage system is designed to maintain the maximum temperatures of the fuel cladding below temperature limits established for both long-term and short-term conditions to prevent fuel clad failure. The maximum temperature of the stainless steel LACBWR fuel cladding is limited to 806°F for both short-term and long-term conditions. The thermal evaluation of the MPC-LACBWR storage system demonstrates that the maximum temperature of the LACBWR fuel cladding remains below the applicable temperature temperatures limits for all normal and accident conditions. Therefore, the LACBWR fuel rod cladding will not fail due to long-term or short-term exposure to high temperatures. Furthermore, the structural evaluation of the LACBWR fuel rods demonstrates that they will not fail due to the concrete cask drop or tipover events, as discussed in Section 11.A.4.

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3.A.6 Canister Closure Weld Evaluation

This section discusses the results of the MPC-LACBWR canister closure weld stress analysis for normal conditions of storage. In addition, this section discusses the closure weld critical flaw size evaluation. The corresponding evaluation for accident conditions is presented in Section 11.A.5. This evaluation confirms that the canister closure weld is acceptable for normal conditions of storage.

Closure Weld Stress Evaluation

The MPC-LACBWR canister closure weld is a ½-inch thick partial penetration groove weld that is designed using a weld stress reduction factor of 0.8 per ISG-4. This stress reduction factor is incorporated by applying a factor of 0.8 to the allowable stresses for the closure weld. The results of the MPC-LACBWR canister stress analyses for normal conditions demonstrate that the maximum stresses in the closure weld satisfy the applicable allowable stress design criteria, which includes the 0.8 stress reduction factor.

The maximum stresses in the MPC-LACBWR canister closure weld are evaluated for the controlling normal load combination of thermal, internal pressure, dead weight, and handling loads, as discussed in Section 3.A.4.4.1.5. The results of the normal load combination stress analyses show that the maximum primary membrane (P_m), primary membrane plus bending (P_m+P_b), and primary plus secondary (P_m+P_b +Q) stress intensities in the closure weld (i.e., section 12) are 1.29 ksi, 1.77 ksi, and 2.09 ksi, respectively. The corresponding normal condition allowable stress intensities for the closure weld, which are based on Type 304 stainless steel material properties at a bounding temperature of 300°F and include the 0.8 stress reduction factor, are 15.97 ksi, 22.0 ksi, and 39.9 ksi, respectively. Therefore, the results of the evaluation show that the margins of safety in the MPC-LACBWR canister closure weld for normal storage conditions are large.

Closure Weld Critical Flaw Size Evaluation

The MPC-LACBWR canister closure weld is comprised of multiple weld beads using a compatible weld material for Type 304 stainless steel base metal. The ½-inch thick closure weld is PT-examined on the root, mid-plane, and final passes. An allowable (critical) flaw evaluation has been performed to determine the critical flaw size in the weld region. The results of the flaw evaluation are used to define the minimum flaw size that must be identifiable in the nondestructive examination (NDE) of the weld. Due to the inherent toughness associated with Type 304 stainless steel, a limit load analysis is conservatively used in conjunction with a J-integral/tearing modulus approach. The maximum radial tensile stress in the closure weld (i.e., SX at Section 12 in Figure 3.A.4.4.1-2) is used for the critical flaw size evaluation. The safety margins used in this evaluation correspond to the stress limits contained in Section XI of the ASME Code. In accordance with ASME Code Section XI, the required safety factor for normal condition is 3.

The maximum calculated radial P_m+P_b tensile stress for normal conditions is 0.76 ksi. To perform the flaw evaluation, a bounding 4.16 ksi tensile stress is conservatively used, resulting in a larger safety factor than the required safety factor of 3. Using 4.16 ksi as the basis for the evaluation, the critical flaw size is 0.34 inch for a flaw that extends 360° around the circumference of the canister. The 360° 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 0.30 inch is acceptable, since it is less than the conservatively determined 0.34-inch critical flaw size.

3.A.7 References

The references for the MPC-LACBWR storage system structural evaluation are the same as those listed in MPC FSAR Section 3.7.

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3.A.8 Coating Specifications

The coatings specifications for the MPC-LACBWR concrete cask and transfer cask are the same as those of the Yankee-MPC concrete cask and transfer cask. The technical data sheets for these coatings are provided in MPC FSAR Section 3.8.

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4.0 THERMAL EVALUATION

The NAC-MPC is provided in three configurations. The first is designed to safely store up to 36 intact Yankee Class spent fuel and reconfigured fuel assemblies and is referred to as the Yankee-MPC. The second is the Connecticut-Yankee MPC, referred to as the CY-MPC, is designed to store up to 26 Connecticut Yankee fuel assemblies, CY-MPC reconfigured fuel assemblies and CY-MPC damaged fuel cans. The third is the La Crosse BWR MPC, referred to as the MPC-LACBWR, designed to store up to 68 La Crosse fuel assemblies, including MPC-LACBWR damaged fuel cans.

The Yankee-MPC system is designed to store Yankee class spent fuel with a maximum heat load of 12.5 kW and reconfigured fuel assemblies with a maximum heat load of 0.102 kW per assembly. The temperatures produced by the design basis fuel bound the temperature effects due to the reconfigured fuel assemblies.

The CY-MPC system is designed to store Connecticut Yankee spent fuel with a maximum total heat load of 17.5 kW, or an average heat load of 0.674 kW per assembly. The maximum heat load of a CY-MPC damaged fuel can, as well as CY-MPC reconfigured fuel assembly, is 0.674 kW.

The MPC-LACBWR system is designed to store Dairyland Power Cooperative La Crosse BWR spent fuel with a maximum total heat load of 4.5 kW, or an average heat load of 66.2 W per assembly for all locations with or without damaged fuel can confinement.

The thermal evaluation of the Yankee-MPC configuration for normal conditions of storage is presented in Section 4.4. The thermal evaluation for normal conditions of the CY-MPC configuration is presented in Section 4.5. The thermal evaluation for the MPC-LACBWR configuration for normal conditions is presented in Section 4.A.3 of Appendix 4.A.

4.1 Discussion

The significant thermal design feature of the NAC-MPC system is the passive convective air flow up along the side of the canister. Cool (ambient) air enters at the bottom of the vertical concrete cask (storage cask) through four inlets. Heated air exits through the four outlets at the top of the storage 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 (storage cask), in conjunction with radiation from the canister surface, maintains the fuel cladding temperature and all of the storage cask component temperatures below their design limits.

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 insolance, and the condition of the air inlets and outlets, as shown in Table 4.1-1. The design conditions for transfer for the Yankee-MPC and CY-MPC configurations are presented in Table 4.1-2.

This evaluation applies different component temperature limits and different material stress limits for long-term (steady-state) conditions and for short-term (transient) conditions. Normal storage is considered to be a steady-state condition. Off-normal and accident events, as well as the vacuum condition that temporarily occurs during the preparation of the canister while it is in the transfer cask and the time the canister is in the transfer cask while filled with helium, are evaluated as transient conditions. The maximum allowable material temperatures for long-term and for transient conditions are provided in Table 4.1-3.

Table 4.1-4 summarizes the results of the thermal evaluation of the Yankee-MPC system and presents the maximum temperatures for principal components. The thermal results and principal component temperatures for the CY-MPC system are shown in Table 4.1-5. Thermal results for the MPC-LACBWR system are presented in Appendix 4.A.

As shown in these tables, the calculated temperatures are well below the allowable component temperatures for normal (long-term) storage conditions and for short-term events.

APPENDIX 4.A THERMAL EVALUATION – MPC-LACBWR MPC STORAGE SYSTEM FOR DAIRYLAND POWER COOPERATIVE LA CROSSE BOILING WATER REACTOR

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4.A THERMAL EVALUATION

4.A.1 Thermal Evaluation of the MPC-LACBWR System

The MPC Storage System for Dairyland Power Cooperative La Crosse Boiling Water Reactor, referred to as the MPC-LACBWR, is designed to safely store up to 68 Allis Chalmers (AC) and Exxon Nuclear (EN) undamaged and damaged spent fuel assemblies. Among the 68 loaded fuel assemblies, 32 of them can be loaded as damaged fuel assemblies in damaged fuel cans (DFCs).

The MPC-LACBWR system is designed to store the spent fuel with a maximum canister heat load of 4.5 kW, and each single fuel assembly with a maximum heat load of 66.2 W.

The thermal evaluation of the MPC-LACBWR for normal conditions of storage is presented in Section 4.A.3.

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4.A.2 Specification of Components

There are three major components that must be maintained within their safe operating temperature ranges: the lead gamma shield and the NS-4-FR solid neutron shield in the transfer cask, and the aluminum heat transfer disk in the canister basket.

The safe operating ranges for the lead gamma shield, solid neutron shield and aluminum heat transfer disk are:

Component	Safe Operating Range
Lead gamma shield	-40°F to +600°F
NS-4-FR solid neutron shield	-40°F to +300°F
Aluminum heat transfer disk	-40°F to +650°F (long-term); +700°F (short-term)

The safe operating range of the lead gamma shield is based on preventing the lead from reaching its melting point of 620°F (Baumeister).

The maximum operating temperature limit of the NS-4-FR solid neutron shield material is specified by the product supplier to be 300°F to ensure sufficient neutron shielding capability.

The safe operating range of the aluminum heat transfer disk is based on the integrity of the aluminum being maintained. The aluminum heat transfer disk is not a structural component to transfer load within the basket. Based on the MIL-HDBK-5F [A1], aluminum at 700°F retains component performance. The maximum long-term and short-term operating temperatures for the aluminum heat transfer disk are 650°F and 700°F, respectively.

The maximum operating temperatures for other materials are shown in Table 4.A.3-3.

As shown in Table 4.A.3-3 for the MPC-LACBWR system, the maximum temperatures for these materials in the normal (long-term) conditions of storage are well below the allowable maximum temperatures. Maximum temperatures for MPC-LACBWR off-normal and accident conditions are addressed in Appendix 11.A.

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4.A.3 Thermal Evaluation of the MPC-LACBWR for Normal Conditions

The significant thermal design feature of the MPC-LACBWR system is the passive convective air flow up along the side of the canister. Cool (ambient) air enters at the bottom of the vertical concrete cask (storage cask) through four inlets. Heated air exits through the four outlets at the top of the storage cask. Radiant heat transfer also occurs from the canister shell to the concrete cask liner. In the thermal analyses performed in this appendix, the air in the inlets, annulus region and the outlets is conservatively not modeled. Consequently, the convection and conduction heat transfer due to the air flow and air are conservatively neglected. Radiation alone, from the canister to the concrete cask, maintains the fuel cladding temperature and all of the storage cask component temperatures below their design limits.

The design basis decay heat for the MPC-LACBWR is 4.5 kW with an axial distribution of the decay heat as shown in Figure 4.A.3-1. A uniform decay heat distribution is considered in which the maximum decay heat for each fuel assembly is 66.2 W.

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 insolance, and the condition of the air inlets and outlets, as shown in Table 4.A.3-1. The design conditions for transfer condition are presented in Table 4.A.3-2. Since no air or air flow is modeled between the concrete cask and the loaded canister, the analysis result for the normal storage condition with an ambient temperature of 75°F bounds the analysis results for the off-normal, half-inlet-blocked condition and the accident conditions of all vents block and the cask burial.

Normal storage is considered to be a steady-state condition. Off-normal and accident events, the vacuum condition that temporarily occurs during the preparation of the canister while it is in the transfer cask, and the time the canister is in the transfer cask while filled with helium are also evaluated as steady-state conditions due to the low heat load. The maximum allowable material temperatures for long-term and for transfer conditions are provided in Table 4.A.3-3.

Table 4.A.3-3 summarizes the results of the thermal evaluation of the MPC-LACBWR system and presents the maximum temperatures for principal components. As shown in the table, the calculated temperatures are well below the allowable component temperatures for normal storage and transfer conditions.

Table 4.A.3-1 Summary of Thermal Design Conditions for Storage

DESIGN CONDITION	ENVIRONMENTAL TEMPERATURE (°F)	SOLAR INSOLANCE [(1)]	STATUS OF INLETS AND OUTLETS
Normal	75	Yes	All open
Off-Normal [(2)]	75	Yes	Two inlets blocked
Off-Normal - Severe Heat	105	Yes	All open
Off-Normal - Severe Cold	-40	No	All open
Accident - Extreme Heat	125	Yes	All open
Accident [(2)]	75	Yes	All blocked
Accident [(2)] - Cask Burial Under Debris [(3)]	75	Yes	All blocked

(1) Solar Insolance 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.

(2) This condition is bounded by the normal condition since no air flow is modeled in the analysis for the normal condition.

(3) In the burial under debris condition, the inlets/outlets are blocked and, in addition, the solar insolance is included in the analysis. These are highly conservative assumptions.

Table 4.A.3-2 Summary of Thermal Design Conditions for Transfer Condition

CONDITION(1)	**DURATION (Hours)**
Design Basis Heat Load (kW)	4.5
Water Filled	600(2)
Vacuum Drying	600(2)
Canister filled with Helium	600(2)

(1) The canister is inside the transfer cask, with an ambient temperature of 75°F.

(2) Duration is unlimited. The 600 hours (25 days) is based on the 30-day time test for abnormal regimes as described in PNL-4835.

Table 4.A.3-3 Summary of Thermal Evaluation for the MPC-LACBWR System

Design Conditions	Material Temperature (°F)									
	Concrete(1)	Fuel Clad	6061- T651 Al Alloy(2)	NS-4-FR(3)	Lead(3)	SA693 630 SS(2)	SA240 SA182 304 SS(4)	SA240 SA182 304L SS(4)	A36 Steel(1)	A588/A350 LF2 Steel(3)
Allowable		SS								
Long-Term	150 (Bulk) 200 (Local)	806	650	300	600	650	800	800	700	700
Short-Term	350	806	700	300	600	800	800	800	700	700
Long-Term Conditions										
Normal (75°F Ambient)	133 (Bulk) 165 (Local)	443	436	N/A	N/A	437	404	349	165	N/A
Short-Term Conditions										
Off-Normal -Half Inlets Blocked (75°F Ambient)	168	443	436	N/A	N/A	437	404	349	168	N/A
Off-Normal -Severe Heat (105°F Ambient)	196	459	452	N/A	N/A	454	420	365	196	N/A
Off-Normal -Severe Cold (-40°F Ambient)	5	377	368	N/A	N/A	370	336	280	5	N/A
Accident -Extreme Heat (125°F Ambient)	228	470	463	N/A	N/A	465	431	377	228	N/A
Transfer -Vacuum Drying	N/A	329	321	140	141	323	288	236	N/A	156
Transfer -Backfilled with Helium	N/A	329	321	140	141	323	288	236	N/A	156

(1) Concrete cask components: Concrete cask and steel liner (ASTM A36). The bounding concrete temperatures for MPC-LACBWR (MPC FSAR Table 4.1-4) with the heat load of 12.5 kW are used.
(2) Fuel basket components: Heat transfer disks (6061- T651) and support disks (ASME SA 693, Type 630 stainless steel).
(3) Transfer cask components: Shells (ASTM A588); bottom doors (A350LF2), neutron shield (NS-4-FR); and, gamma shield (lead)/concrete cask lid shield.
(4) Canister components: Closure Lid (ASME SA 240/182, Type 304/304L stainless steel), shell (ASME SA 240, Type 304/304L stainless steel), bottom plate (ASME SA 240, Type 304L stainless steel), and damaged fuel can (ASME SA 240, Type 304 stainless steel).

4.A.3.1 MPC-LACBWR Thermal Models

As follows, the thermal evaluation of the MPC-LACBWR system utilizes seven finite element models, generated by the ANSYS program, for normal conditions of storage.

1. Three-Dimensional Concrete Cask and Canister Model
2. Three-Dimensional Transfer Cask and Canister Model
3. Two-Dimensional Fuel Model
4. Four Two-Dimensional Fuel Tube Models

The three-dimensional concrete cask and canister model includes: the concrete storage cask, fuel assemblies, fuel tubes, stainless steel support disks, aluminum heat transfer disks, canister shell, lids and bottom plate. Based on symmetry, only one-fourth of the concrete cask and the loaded canister are modeled, and the plane of symmetry is considered to be adiabatic. The air in the inlets annulus between the canister and the concrete cask and the outlets is conservatively not modeled. A conservative distribution of the Boral plates is modeled with a total number of 24 Boral plates.

In the three-dimensional models, fuel regions and the fuel tubes are modeled using effective conductivities, which are determined by the two-dimensional fuel model and the two-dimensional fuel tube models.

The three-dimensional transfer cask and canister model comprise the transfer cask and the loaded canister. The canister and its internals are modeled the same as the three-dimensional concrete cask and the canister model. The model is used to perform steady-state analyses for the transfer condition when the canister is in the loading, vacuum drying, and helium backfill processes. Since the steady state temperatures calculated for water, vacuum and helium conditions are less than the limiting component allowable temperatures, no transient analyses are needed for the transfer conditions.

The effective conductivity of the fuel is determined using the two-dimensional fuel model, which is a detailed two-dimensional thermal model of the fuel assembly. The model includes the fuel pellets, the cladding, the helium gas between the fuel pellets and the fuel rod cladding, and the media occupying the space between the fuel rods. For normal operational conditions, the media between the fuel rods is considered to be helium. For the canister loading and transfer operations, the media is also considered to be water or vacuum. Based on Reference [A2], helium properties are used for the vacuum media since the pressure of the LACBWR during vacuum drying is in the low pressure range where helium gas thermal conductivity is not reduced as a variable of pressure.

The two-dimensional fuel tube models are used to determine the effective conductivities of the tube walls (with or without BORAL plate) and fuel tube walls (normal fuel tube or fuel tubes in the basket slots containing DFC). There are four models for the fuel tubes. The media is in the gaps between the layers comprising the fuel tube and BORAL.

These models are described in Sections 4.A.3.1.1 through 4.A.3.1.4.

4.A.3.1.1 Three-Dimensional MPC-LACBWR Concrete Cask and Canister Model

The concrete cask design of the MPC-LACBWR and the Yankee-MPC are similar. The storage cask has four air inlets at the bottom and four air outlets at the top. Since the air in the annulus region between the canister and the concrete cask is not modeled for the LACBWR heat transfer analysis, the conduction and convection heat transfer by air are conservatively neglected. The model includes the concrete cask and loaded canister from the bottom of the canister bottom plate as shown in Figure 4.A.3-2. The concrete cask and canister model include: the concrete cask, concrete liner, fuel assemblies, fuel tubes, support disks, heat transfer disks, canister shell, lids, bottom plate and helium.

ANSYS SOLID70 three-dimensional thermal elements and MATRIX50 thermal radiation super element are used to construct the three-dimensional concrete cask and loaded canister finite element model as shown in Figures 4.A.3-2 through 4.A.3-4.

In this model, radiation heat transfer across the following regions is considered:

- Thermal radiation across the vertical air gap surfaces (between the canister shell outer surface and the liner inner surface)
- Thermal radiation across the region below the active fuel (axial direction)
- Thermal radiation across the region above the active fuel (axial direction)
- Thermal radiation between the heat transfer disk and the support disk in cask axial direction, or between two support disks when the heat transfer disk is not present in the cask axial direction
- Thermal radiation across the gaps between the disks and the inner surface of the canister shell

In the model, the fuel assemblies are considered to be centered in the fuel tubes. Likewise, the fuel tubes and damaged fuel can are centered in the slots of the basket disks and the basket is centered in the canister. These assumptions are conservative since any contact between components will provide a more efficient path to reject heat.

The gaps included in the canister part of the concrete cask and the canister model are conservatively computed based on nominal dimensions. Gas inside the canister is modeled as helium.

All material properties used in the model, except the following thermal conductivities for concrete and the effective properties discussed in the following paragraphs, are shown in MPC FSAR Tables 4.2-1 through 4.2-11.

Thermal Conductivities of Concrete

	Value at Temperature (°F)		
Property (units)	100	200	300
Conductivity (Btu/hr-in-°F) [a]	0.091	0.089	0.086

[a] Handbook of Concrete Engineering [A3], Figure 6-31, Curve 1.

Temperature dependent thermal conductivity is defined in greater detail for the thermal analysis performed for MPC-LACBWR to permit an improved representation of the heat transfer path through the concrete structure with the consideration of an all-vents-blocked heat transfer model.

The fuel regions are modeled as homogenous regions with effective conductivities, determined by the two-dimensional fuel model as described in Section 4.A.3.1.3.

Four configurations (composite walls) are considered for fuel tubes. The fuel tube is modeled as a homogeneous region in the three-dimensional loaded concrete cask model and the loaded transfer cask model with effective orthotropic thermal properties. These effective thermal properties account for the thermal conductivities of the various components that make up the fuel tube (including the media in the gaps), as well as thermal radiation between components separated by gaps. Each fuel tube region in the model represents a thickness from the inside surface of the fuel tube to the edge of the support/heat transfer disk slot for normal fuel tubes, or a thickness from the inside surface of the DFC to the edge of the support/heat transfer disk slot for slots with DFCs.

The fuel tube effective properties are established using the two-dimensional fuel tube models described in Section 4.A.3.1.4.

MPC-LACBWR Loads and Boundary Conditions

In the normal storage condition, the concrete cask has the following 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 Figure 4.A.3-1, which was used to distribute the decay heat over the active fuel length of 83 inches.

2. Solar insolance to the outer surfaces of the storage cask

   The solar insolance to the storage cask outer surface is considered in the model. The same incident solar energy is applied as described in MPC FSAR Section 4.4.1.1.

3. Natural convection heat transfer at the outer surfaces of the storage cask

   Natural convection heat transfer at the outer surfaces of the storage cask is evaluated using the following heat transfer correlation.

$$Hc = 0.00132 \Delta T^{1/3} \quad \text{Btu/hr-in}^2\text{-°F}$$

4. Radiation heat transfer at the storage cask outer surfaces

   The radiation heat transfer between the outer surfaces and ambient is evaluated in the model by calculating an equivalent radiation heat transfer coefficient using the same methodology as shown in MPC FSAR Section 4.4.1.1.2.

   At the storage cask side surface, an emissivity for the concrete surface of $\varepsilon_1 = 0.9$ is used and a calculated view factor (F_{12}) = 0.197 (MPC FSAR Section 4.4.1.1.2) is applied. This view factor is conservative for the MPC-LACBWR analysis since there are only five casks in the LACBWR ISFSI cask array. At the top, an emissivity of $\varepsilon_1 = 0.8$ for a carbon steel surface is used, and a view factor (F_{12}) = 1 is applied.

5. Other thermal boundary conditions

Fixed-temperature boundary conditions are applied to the model bottom of the concrete cask to simulate the ambient temperature for the normal, off-normal and accident storage conditions. Ambient temperatures are applied for each case. The bottom surface of the canister bottom plate is conservatively assumed to be adiabatic.

Figure 4.A.3-1 Axial Power Distribution for MPC-LACBWR Fuel

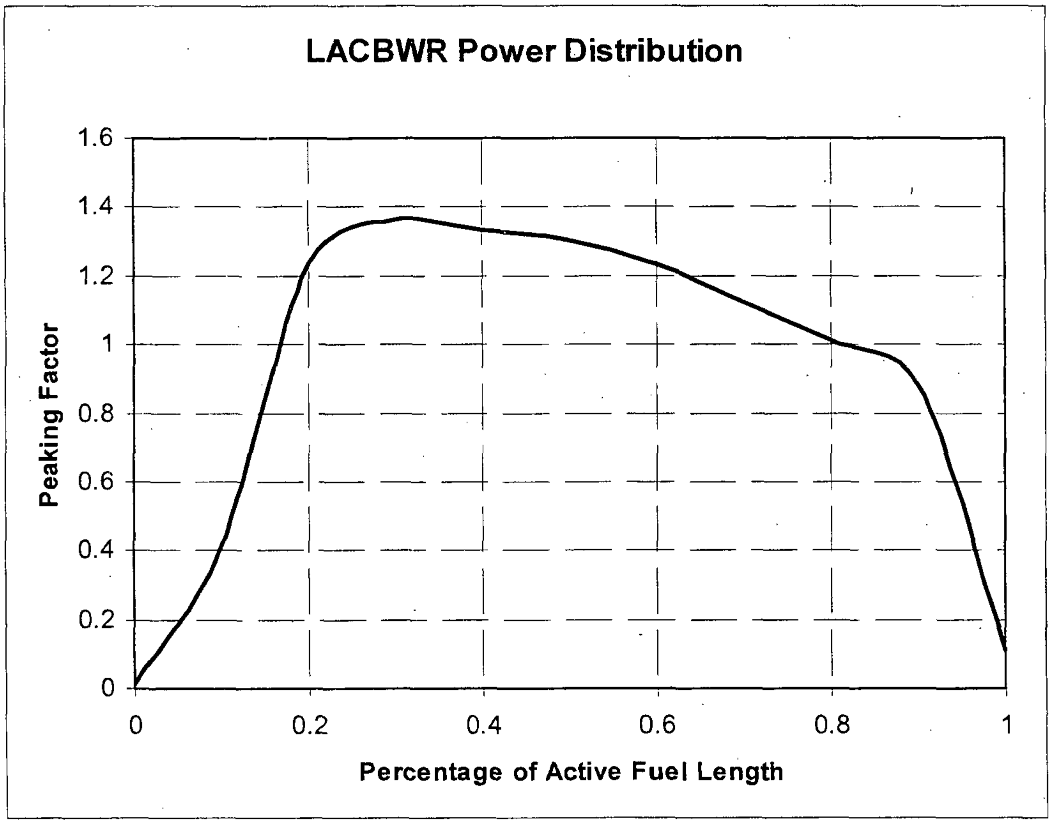


Figure 4.A.3-2 MPC-LACBWR Three-Dimensional Concrete Cask and Canister Model

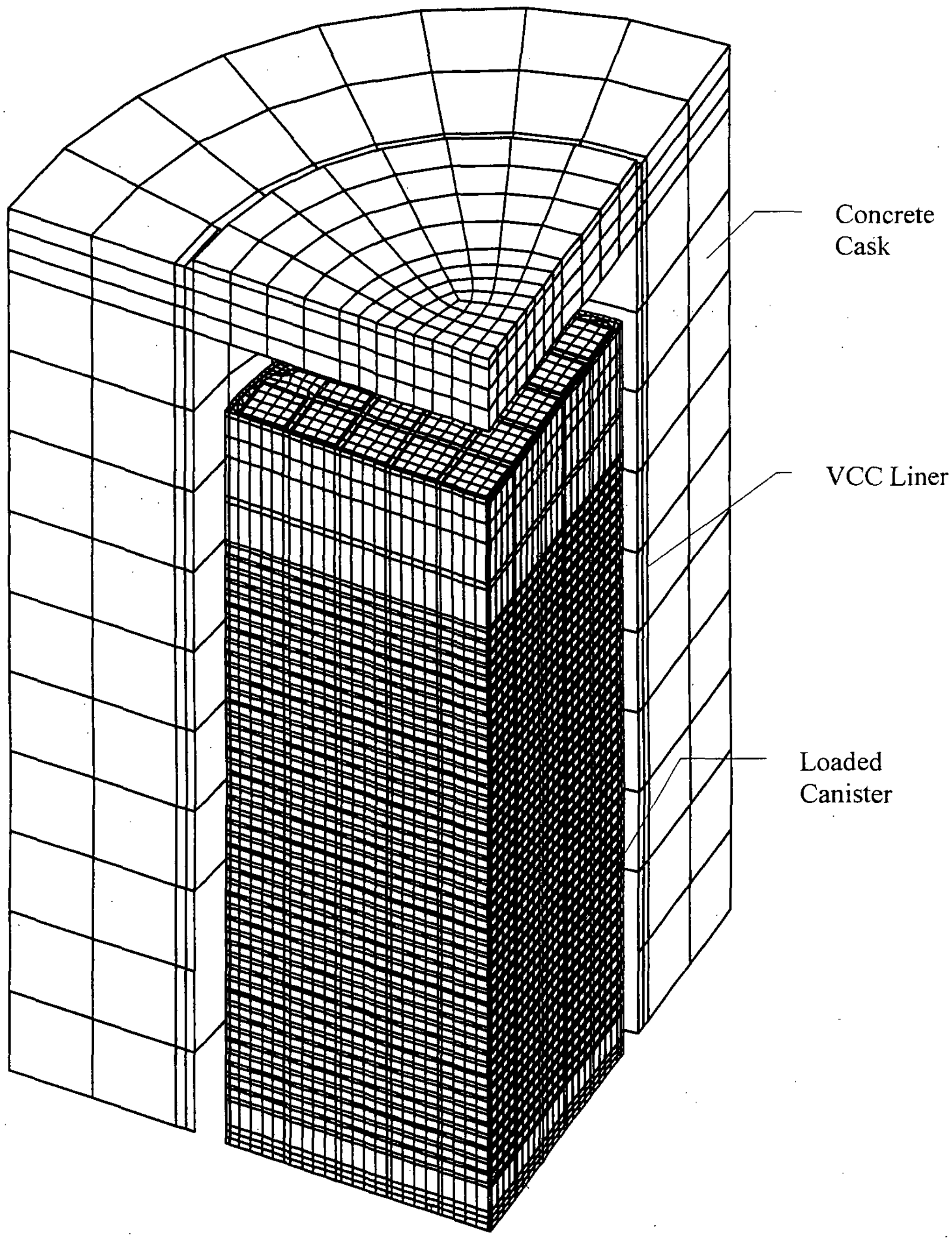


Figure 4.A.3-3 MPC-LACBWR Three-Dimensional Concrete Cask and Canister Model (Canister Portion)

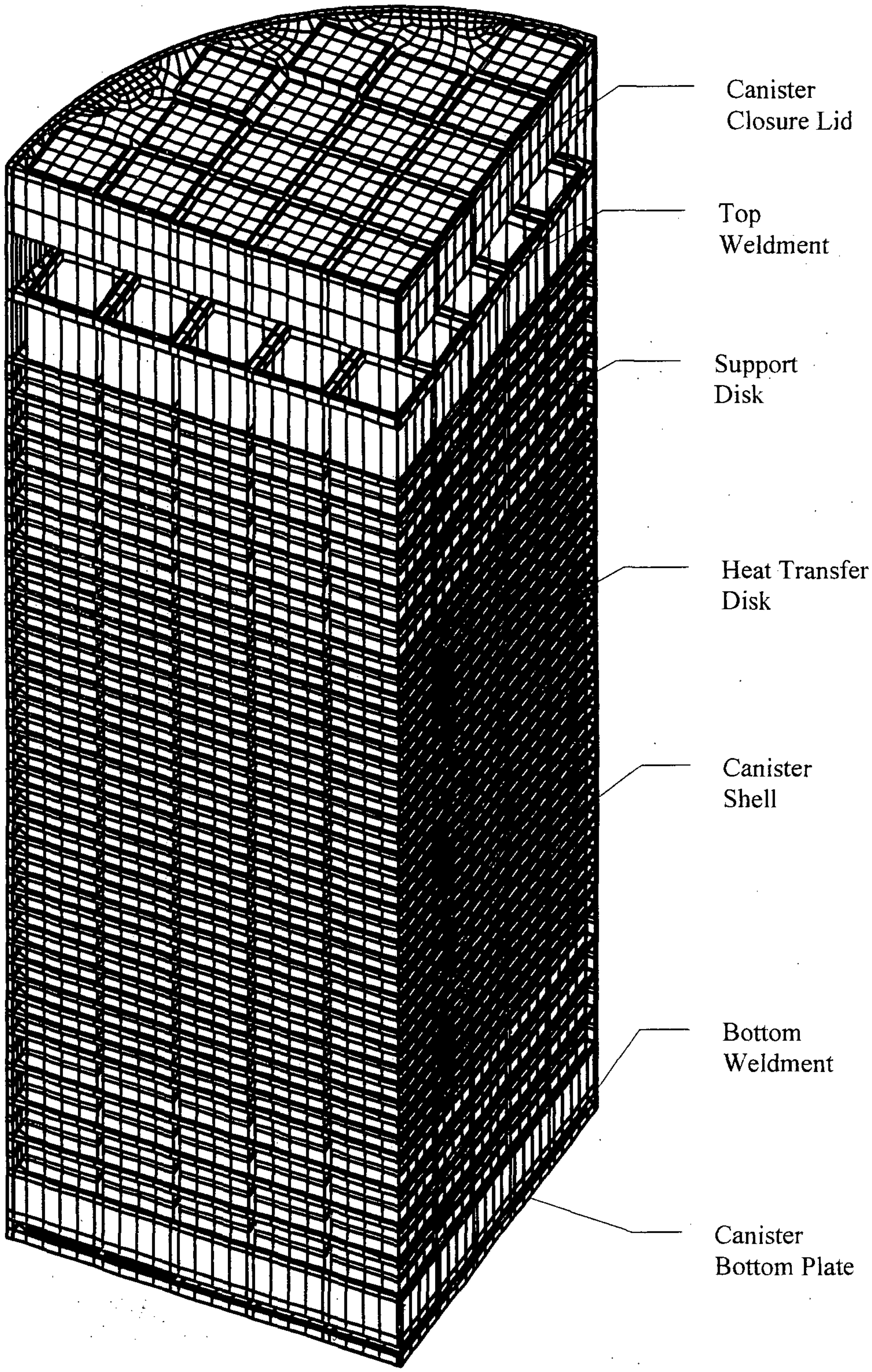
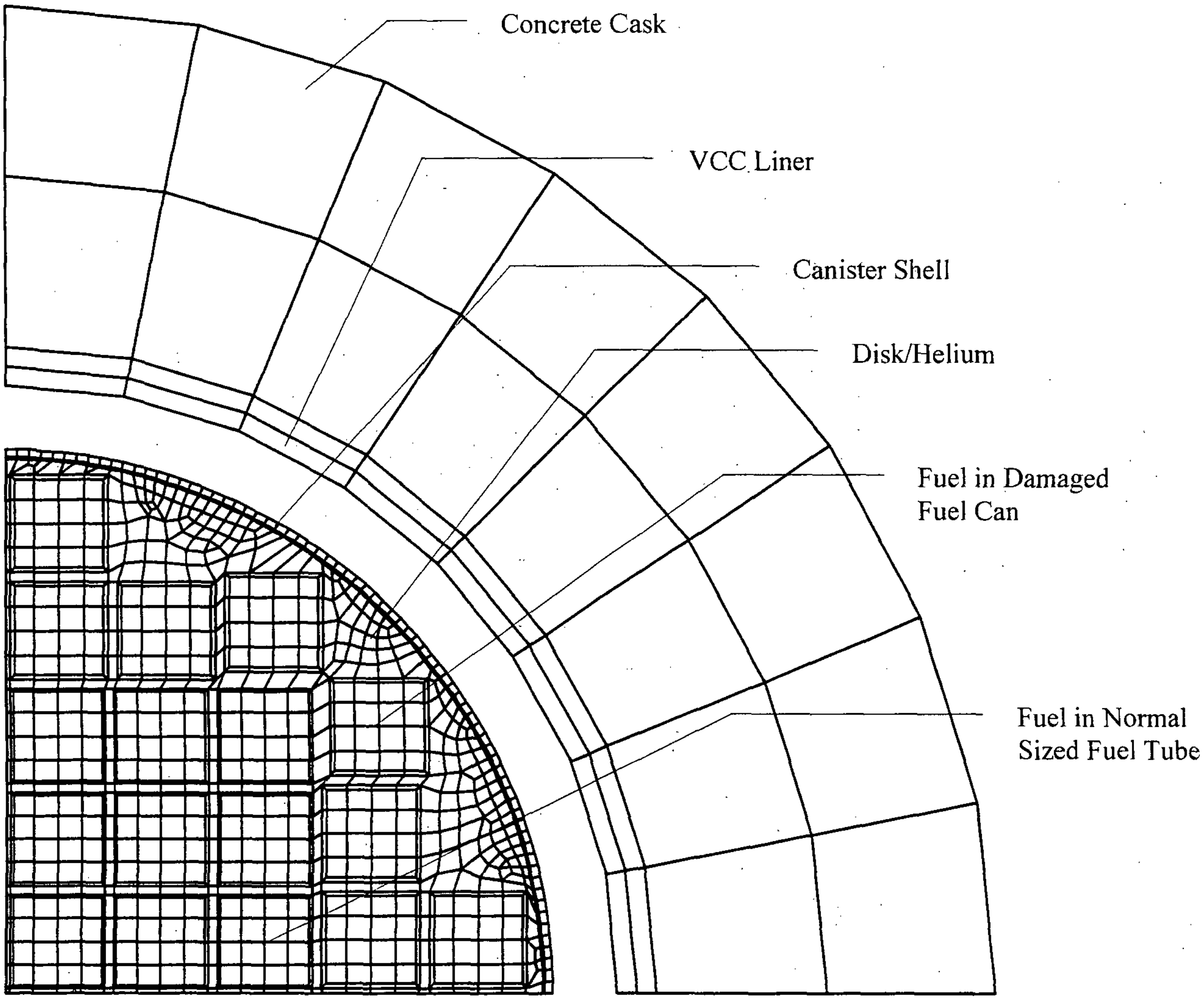


Figure 4.A.3-4 MPC-LACBWR Three-Dimensional Canister Model – Cross-Section



4.A.3.1.2 Three-Dimensional MPC-LACBWR Transfer Cask and Canister Model

The three-dimensional MPC-LACBWR transfer cask and canister model is shown in Figures 4.A.3-5 and 4.A.3-6. ANSYS SOLID70 three-dimensional conduction elements and MATRIX50 radiation super elements are used to construct the model. The model includes the fuel assemblies, the fuel basket (fuel tubes, support disks, heat transfer disks, and top and bottom weldment plates), the canister (shell, bottom plate, and closure lid), the air gap between the canister and the transfer cask, and the transfer cask (top and bottom plates, inner and outer shells, neutron and gamma shields, and doors). The loaded canister in this model is modeled identically to the loaded canister model in the concrete cask and the canister model described in Section 4.A.3.1.1. Based on symmetry, only one-fourth of the transfer cask and canister are modeled, and the plane of symmetry is considered to be adiabatic.

A 0.43-inch air radial gap, based on the nominal dimensions of the canister shell and the transfer cask inner shell, is included. For normal transfer operations, when either vacuum or helium exists in the canister, only conduction and radiation are considered across the gaps inside the canister and convection is conservatively neglected. Conduction helium properties are utilized during the low-pressure drying process [A1]. 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 content.

The heat load of 4.5kW is conservatively used in the analyses of the transfer conditions. For the heat load analyzed, a volumetric heat generation (Btu/hr-in^3) is applied to each fuel assembly based on a active fuel length of 83 inches along with an axial power distribution as shown in Figure 4.A.3-1.

Natural convection and radiation heat transfer modes are considered at the surfaces of the transfer cask and on the top of the canister lid. The same methodology as described in Section 4.A.3.1.1 for concrete cask surfaces is used. The same convection film correlation ($Hc = 0.00132 \Delta T^{1/3}$ Btu/hr-in^2-°F) is used at the transfer cask outer surfaces and the canister lid. Solar insolance is applied at the surfaces of the transfer cask and on the top of the canister lid for helium condition, while solar insolance is not considered for the water condition since the draining process occurs inside a building. The same incident solar energy is applied as described in MPC FSAR Section 4.4.1.1. An ambient temperature of 75°F is assumed. The ambient temperature is applied to the bottom surface of the transfer cask doors as a fixed temperature boundary condition.

Since the steady-state temperatures calculated for water, vacuum and helium conditions are less than the limiting component allowable temperatures, transient analyses are not needed for the transfer conditions.

Figure 4.A.3-5 Three-Dimensional MPC-LACBWR Transfer Cask and Canister Model

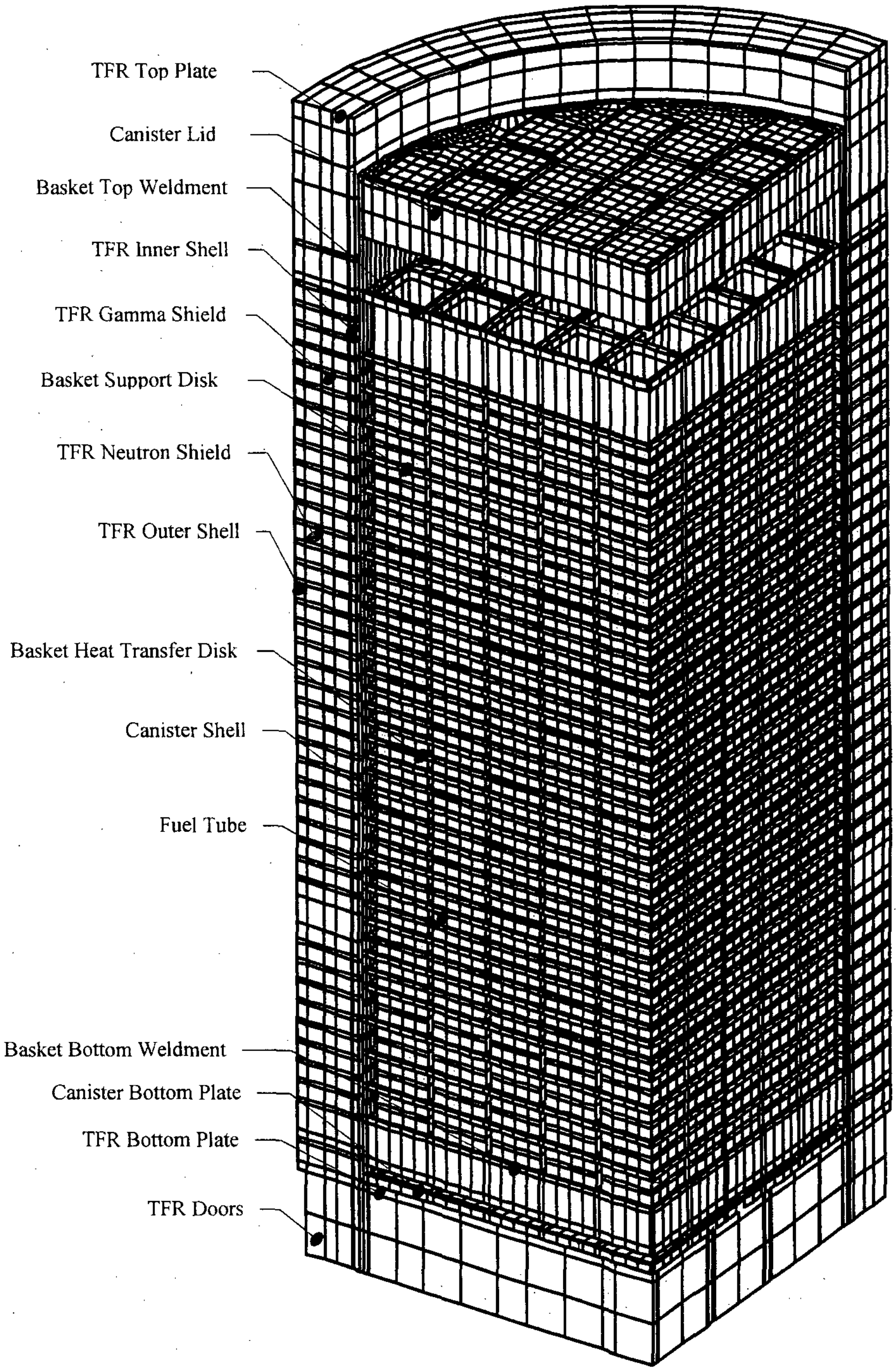
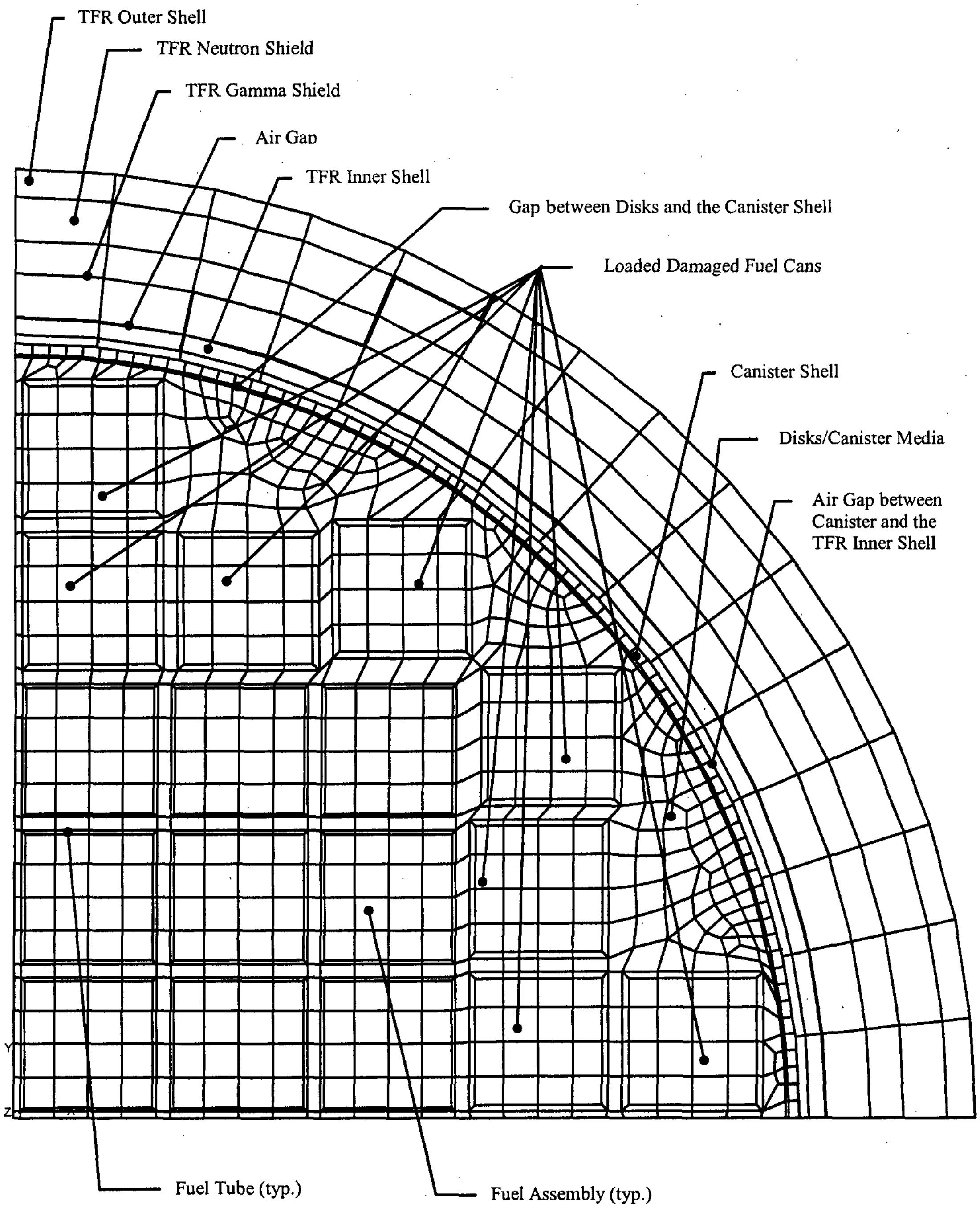


Figure 4.A.3-6 Three-Dimensional MPC-LACBWR Transfer Cask and Canister Model – Cross-Section



4.A.3.1.3 Two-Dimensional MPC-LACBWR Fuel Model

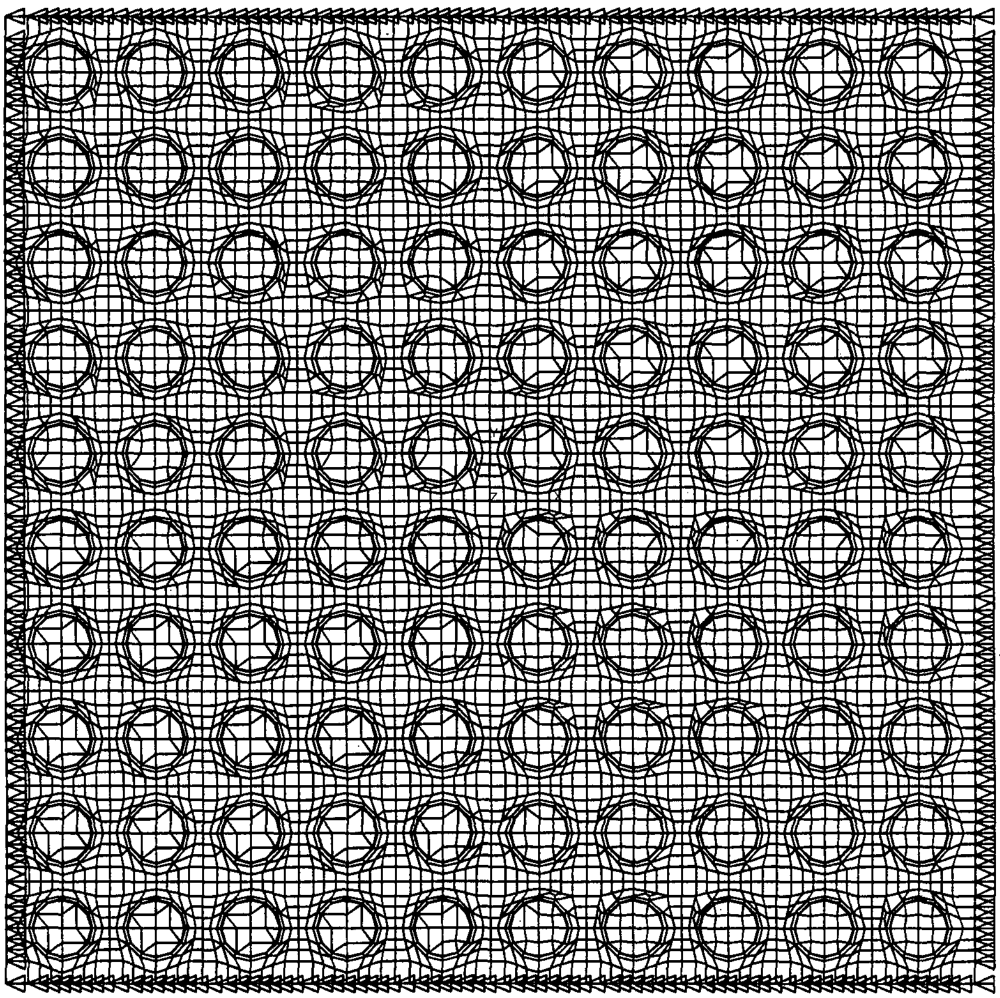
The effective thermal property of the BWR fuel is determined using a two-dimensional finite element model of the 10×10 fuel assembly. The finite element model includes: the fuel pellets, cladding, the gas between fuel pellets and the clad, and the media surrounding the fuel rod. The edge of the fuel model extends to inner surface of the fuel tube for the normal size slots or the inner surface of the damaged fuel can for the oversize slots. The inner dimension for both normal size fuel tubes and the damaged fuel can is 5.75 inches. The model is shown in Figure 4.A.3-7. The fuel cladding material for the fuel is stainless steel.

Modes of heat transfer modeled include conduction and radiation between individual fuel rods when helium is inside the canister for the steady-state condition, and convection is conservatively neglected. Water circulation is considered for the water condition. ANSYS PLANE55 conduction elements and MATRIX50 radiation elements are used in the model, which represents a 10×10 fuel assembly. Each fuel rod consists of the pellet, stainless steel cladding, and a gap between the pellet and cladding. The gas in the gap between the pellet and cladding is considered to be helium. Three types of media between the fuel rods are considered: helium for long-term storage and backfill, water for the drain condition, and vacuum for the vacuum/drying condition. Based on Reference [A1], helium properties are used for the vacuum media since the pressure of the LACBWR during vacuum drying is in the low pressure range where helium gas thermal conductivity is not reduced as a variable of pressure. For the helium and vacuum media, radiation elements are defined between fuel rods and from the fuel rods to the boundary of the model (inside surface of the fuel tube or inner surface of the damaged fuel can). Radiation effects at the gap between the pellets and the cladding are conservatively ignored. Effective emissivities are determined using the formula shown in MPC FSAR Section 4.4.1.2, and the emissivities are specified in MPC FSAR Section 4.2. For the water condition, the radiation matrix elements are effectively removed from the model.

Two kinds of media in the fuel assembly models are considered to accommodate the media of water and helium.

The effective properties for the fuel are determined using the method described in MPC FSAR Section 4.4.1.3 for the Yankee Class fuel model.

Figure 4.A.3-7 Two-Dimensional MPC-LACBWR 10×10 Fuel Model



4.A.3.1.4 Two-Dimensional MPC-LACBWR Fuel Tube Model

The two-dimensional fuel tube model represents one side of the four-sided fuel tubes to determine the effective thermal property of the fuel tube. The calculated effective thermal property is used in the three-dimensional concrete cask and canister model, and the three-dimensional loaded transfer cask model.

There are four fuel tube side configurations represented in the heat transfer thermal model as shown in Figures 4.A.3-8 through 4.A.3-11 and discussed in the following paragraphs. The standard fuel tube in the normal basket slots has a surface with and without BORAL plate. The fuel tube in the slots containing a DFC similarly has a side of the fuel tube with and without BORAL plate. The conductive media between material components is modeled as helium or water.

The model of the standard fuel tube with BORAL plate is shown in Figure 4.A.3-8. This model has six layers, which includes the fuel tube, the BORAL plate (including the core matrix sandwiched by aluminum claddings), media gaps on both sides of the BORAL plate, the stainless cladding, and the media gap between the stainless steel cladding and the support disk or heat transfer disk.

The model of the standard fuel tube without BORAL plate is shown Figure 4.A.3-9. This model has two layers, which includes the fuel tube and the media gap between the fuel tube and the support disk or heat transfer disk.

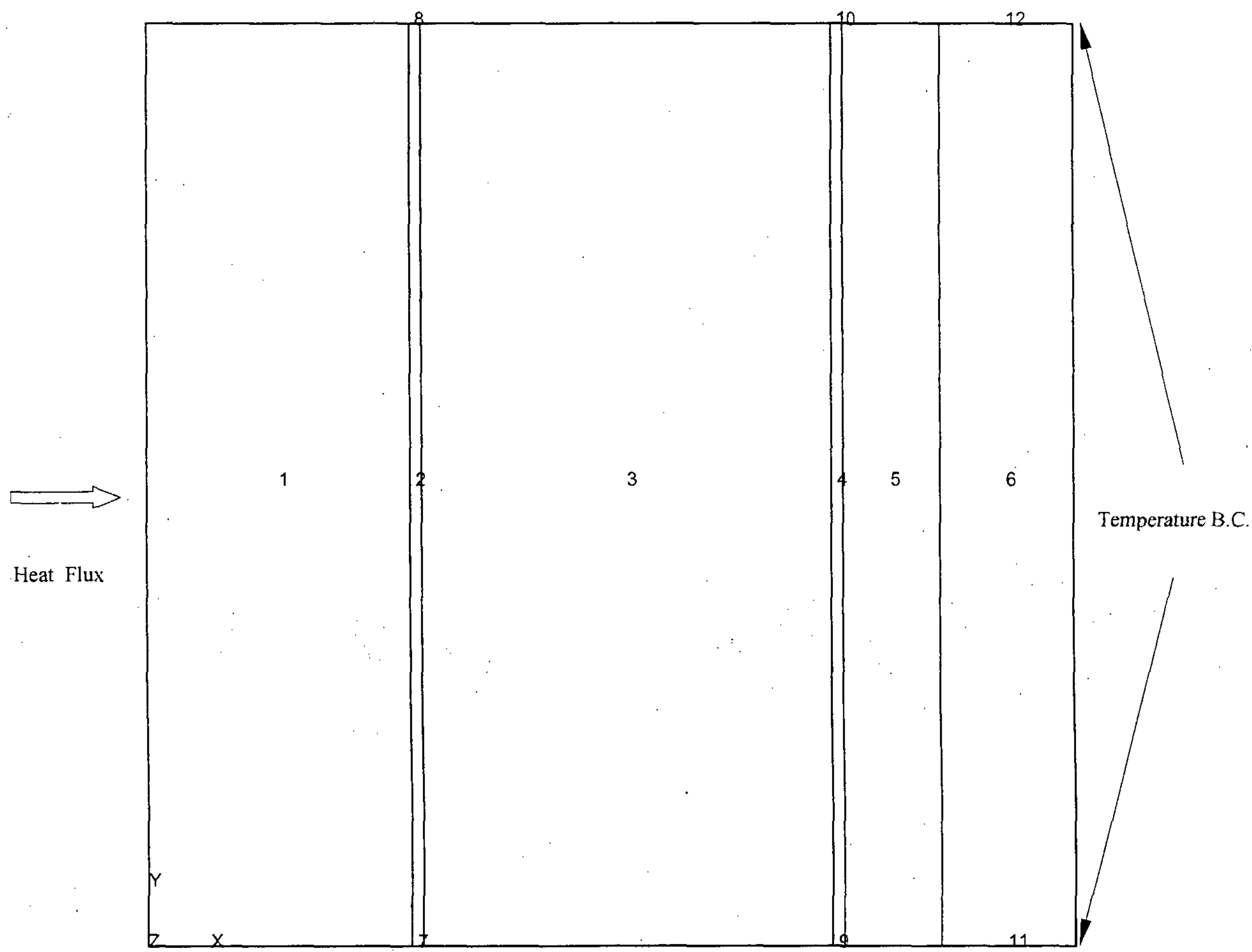
As shown in Figure 4.A.3-10, the model of fuel tube in the slots containing a DFC with BORAL plate has eight layers. This model includes the DFC shell, a gap between the DFC shell and the fuel tube, the fuel tube, the BORAL plate (including the core matrix sandwiched by aluminum claddings), media gaps on both sides of the BORAL plate, the stainless cladding, and the media gap between the stainless steel cladding and the support disk or heat transfer disk.

The model of the fuel tube in the slots containing a DFC without BORAL plate is shown in Figure 4.A.3-11. This model has four layers, which include the DFC shell, a gap between the DFC shell and the fuel tube, the fuel tube, and the media gap between the fuel tube and the support disk or heat transfer disk.

Modes of heat transfer modeled include conduction and radiation. The radiation is considered only for the helium condition. Convection is conservatively neglected for both helium and water conditions. ANSYS PLANE55 conduction elements and LINK31 radiation elements are used to construct the model. The model consists of layers of conduction elements and radiation elements that are defined at the helium gaps (two for each gap). No radiation links are used in the model representing the water condition. The thickness of the model is the distance measured from the inside face of the fuel tube, or the inner surface of the DFC, to the inner face of the slot in the support disk or heat transfer disk (assuming the fuel tube is centered in the hole in the disk). The height of the model is defined as being equal to the width of the model.

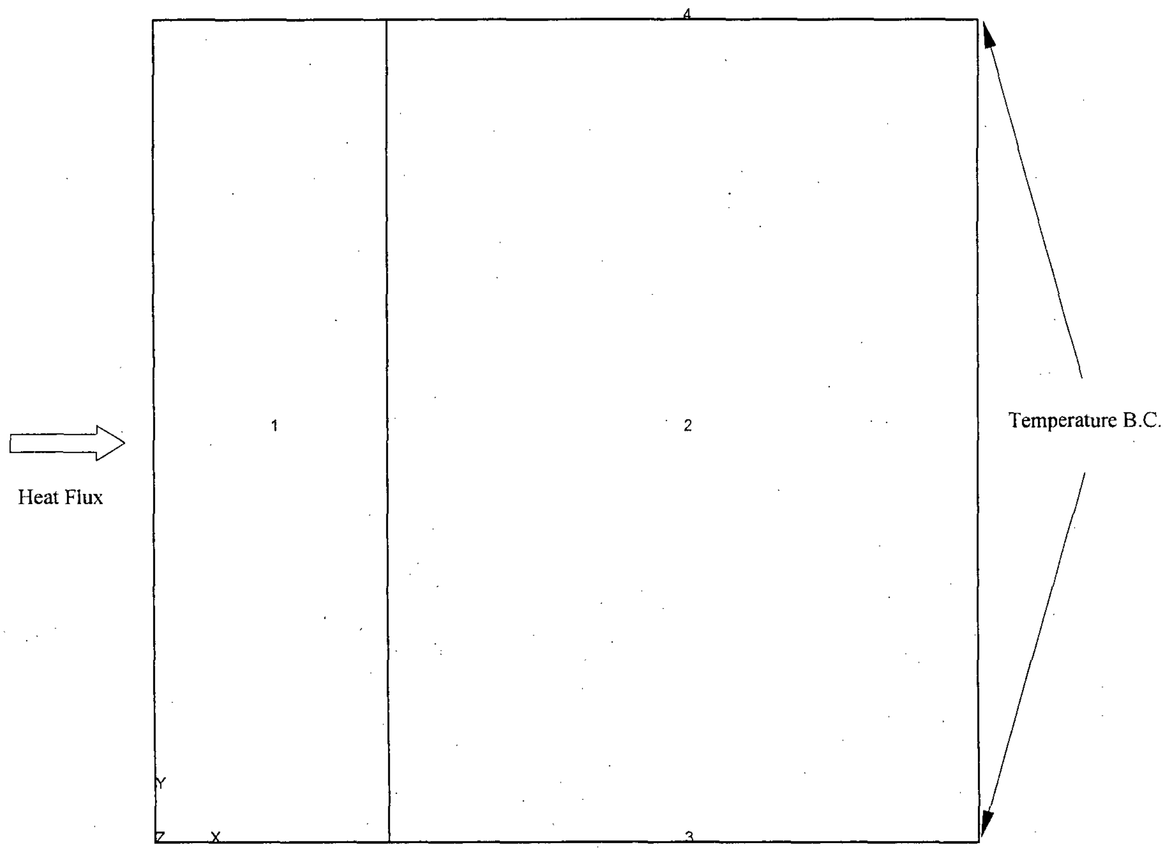
Heat flux is applied at the left side of the model, and the temperature at the right boundary of the model is constrained. The heat flux is determined based on 4.5 kW with a peaking factor of 1.36. The maximum temperature of the model (at the left boundary) and the temperature difference (ΔT) across the model are calculated by ANSYS. The effective conductivities for the fuel tube are determined using the same methodology described in MPC FSAR Section 4.4.1.4. The effective densities and effective specific heats are computed as described in MPC FSAR Section 4.5.1.5.

Figure 4.A.3-8 Two-Dimensional MPC-LACBWR Fuel Tube Model (Standard Fuel Tube with BORAL Plate)



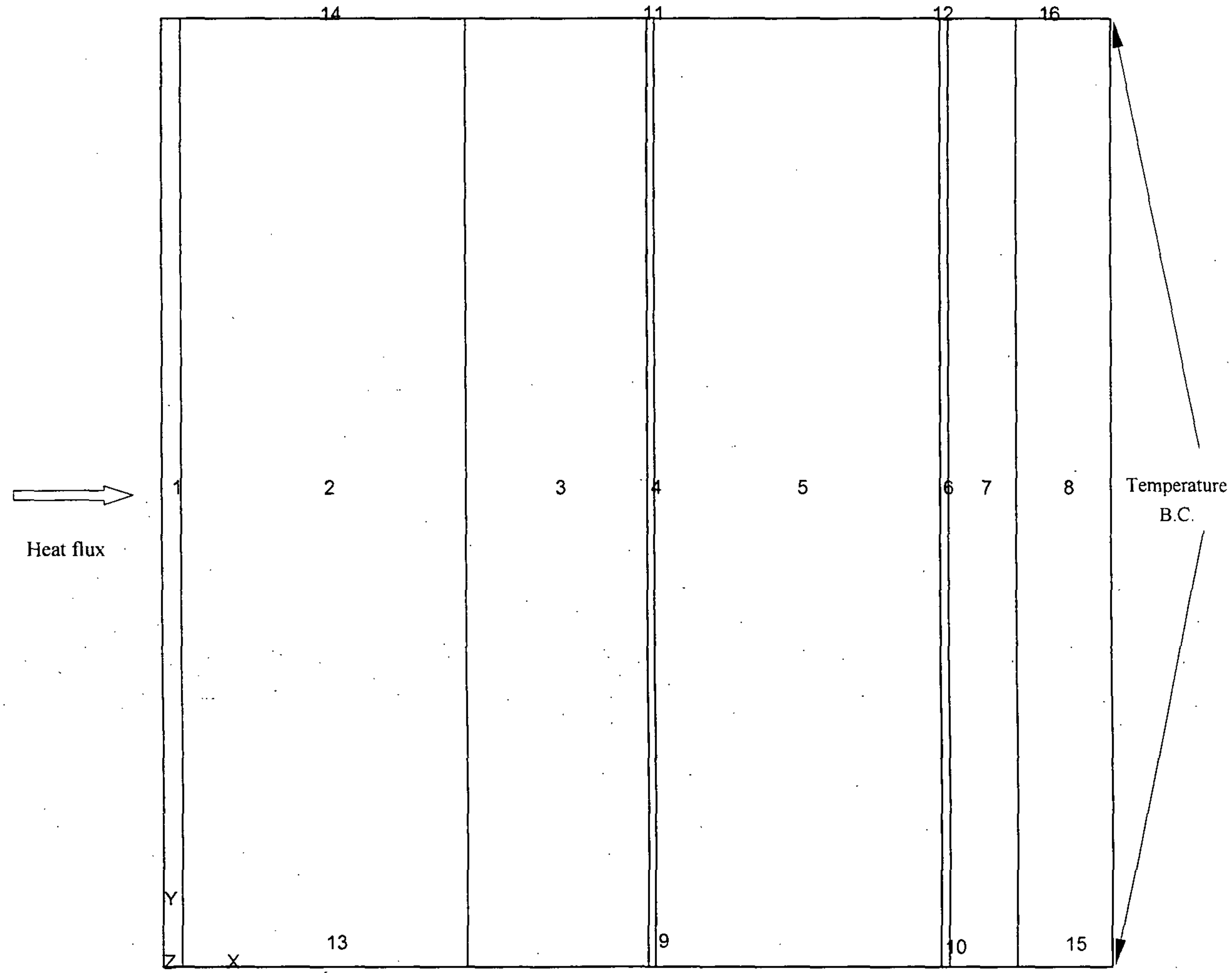
Element Number	Material
1	Stainless steel tubing
2, 4, 6	Gap – helium or water
3	BORAL plate
5	Stainless steel cladding
7 - 120	Radiation links for helium media

Figure 4.A.3-9 Two-Dimensional MPC-LACBWR Fuel Tube Model (Standard Fuel Tube without BORAL Plate)



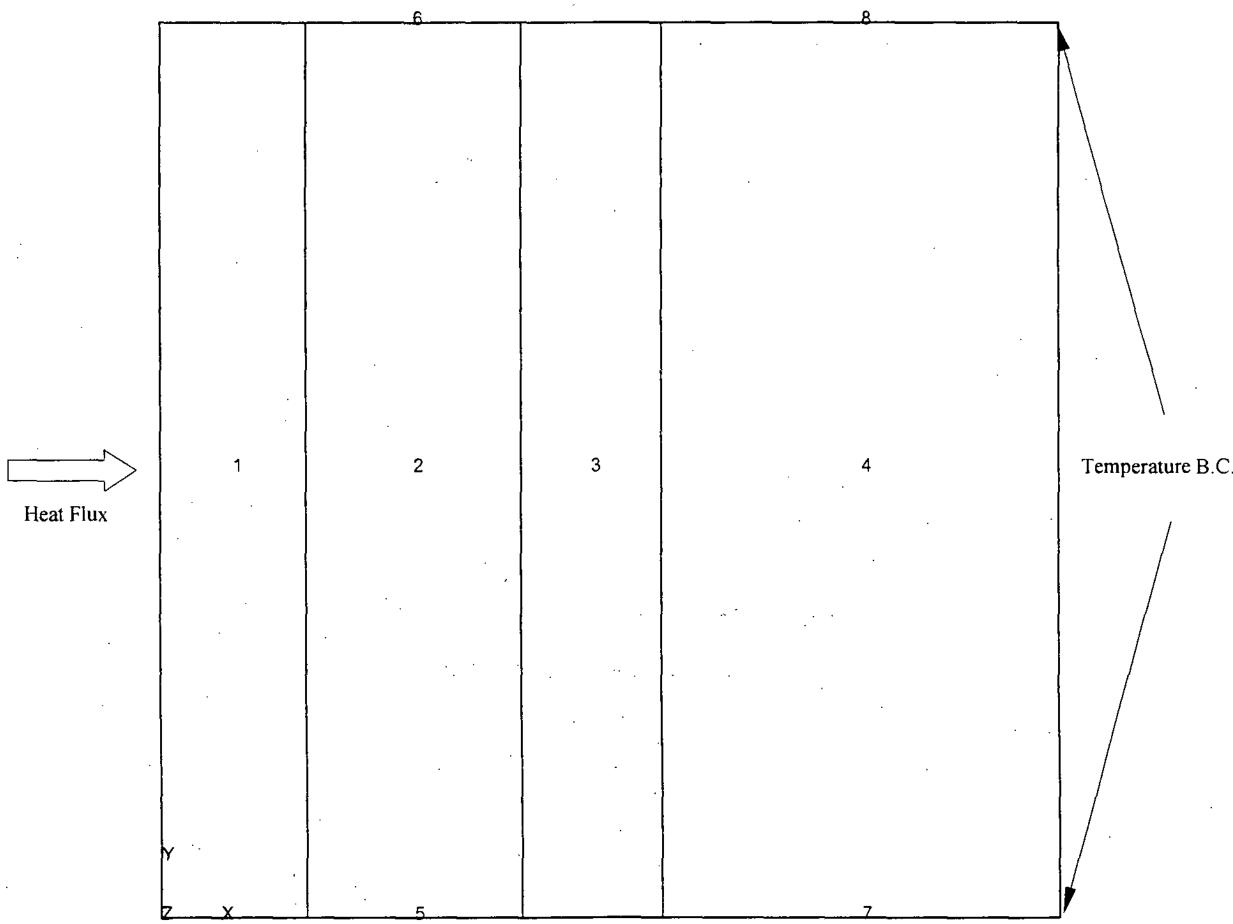
Element Number	Material
1	Stainless steel tubing
2	Gap – helium or water
3, 4	Radiation links for helium media

Figure 4.A.3-10 Two-Dimensional MPC-LACBWR Fuel Tube Model (Fuel Tube in the Slots Containing DFC with BORAL)



Element Number	Material
1	Stainless steel damaged fuel can
2, 4, 6, 8	Gap – helium or water
3	Stainless steel tubing
5	BORAL plate
7	Stainless steel cladding
9 - 16	Radiation links for helium media

Figure 4.A.3-11 Two-Dimensional MPC-LACBWR Fuel Tube Model
(Fuel Tube in the Slots Containing DFC without BORAL)



Element Number	Material
1	Stainless steel damaged fuel can
2, 4	Gap – helium or water
3	Stainless steel tubing
5 - 8	Radiation links for helium media

4.A.3.2 MPC-LACBWR Test Model

The MPC-LACBWR system is conservatively designed by analysis so that testing is not required.

4.A.3.3 Maximum Temperatures for MPC-LACBWR Normal Conditions

The routine operations of loading, closing and transferring the canister to the concrete cask result in different temperature conditions, depending on the configuration of the canister. In the transfer conditions, the maximum temperature of the fuel is maintained below the maximum allowable short-term temperature limit (Table 4.A.3-3). In the normal storage condition, the maximum temperature of the fuel is maintained below the maximum allowable long-term temperature limit (Table 4.A.3-3). Temperature results for the off-normal and accident conditions are presented in Appendix A of Chapter 11 of the MPC FSAR.

4.A.3.3.1 MPC-LACBWR Maximum Temperatures in Long-Term Storage

The design heat load for the MPC-LACBWR fuel is 4.5 kW, with a uniformly distributed fuel load of 66.2 watts per assembly, and is used as the heat source for all thermal analyses for the MPC-LACBWR system. The maximum component temperatures for the normal conditions of storage are shown in Table 4.A.3-4. The maximum and bulk temperatures calculated from the analysis of the concrete cask of the NAC-MPC storage system with a 12.5 kW heat load are conservatively used for defining maximum concrete temperatures for the MPC-LACBWR system with a 4.5 kW heat load. Similarly, using the heat transfer boundary conditions where heat transfer in the annulus region between the canister shell and concrete cask liner is limited to radiation permits system operability to be bounded by air temperature differences calculated between the inlet and outlet vents for the same 12.5 kW system configuration.

4.A.3.3.2 MPC-LACBWR Maximum Temperatures in Transfer Operations

Thermal analyses are performed to establish the allowable time limits for the vacuum and helium conditions in the canister while in the transfer cask loading operation configuration. Since the steady state temperatures calculated are less than the limiting component allowable temperatures, maximum component temperature is not a system control limit for transfer operation. An imposed limit of 600 hours (25 days) is defined based on the 30-day time test for abnormal regimes as described in PNL-4835. The maximum component temperatures for the helium transfer conditions are shown in Table 4.A.3-5. The maximum component temperatures for the water drain condition are shown in Table 4.A.3-6.

Table 4.A.3-4 MPC-LACBWR Maximum Component Temperatures for the Normal Conditions of Storage

Component	Maximum Temperature (°F)	Allowable Temperature (°F)(1)
Fuel Cladding	443	806
Aluminum Disk	436	700
Support Disk	437	650
Canister	349	800
Concrete Liner (steel)	165(2)	700
Concrete	165 (local)(2) 133 (bulk)(2), (3)	200 (local) 150 (bulk)

(1) The allowable temperatures are defined and referenced in MPC FSAR Table 4.1-3.

(2) Concrete temperatures are from MPC FSAR Table 4.1-4, which is for a heat load of 12.5 kW.

(3) The average temperature of the concrete region is used as the bulk concrete temperature.

Table 4.A.3-5 MPC-LACBWR Maximum Component Temperatures for the Transfer Condition – Helium and Vacuum in Canister

Component	Maximum Temperature (°F)	Allowable Temperature (°F)(1)
Fuel Cladding	459	806
Gamma Shield (Lead)	338	600
Neutron Shield	347	300
Heat Transfer Disk (Aluminum)	452	700
Support Disk	454	800
Canister	384	800
Transfer Cask Inner Shell	345	700

(1) The allowable temperatures are defined and referenced in MPC FSAR Table 4.1-3.

Table 4.A.3-6 MPC-LACBWR Maximum Component Temperatures for the Transfer Condition – Water in Canister

Component	Maximum Temperature (°F)	Allowable Temperature (°F)(1)
Fuel Cladding	139	806
Gamma Shield (Lead)	95	600
Neutron Shield	95	300
Heat Transfer Disk (Aluminum)	138	700
Support Disk	138	800
Canister	120	800
Transfer Cask Inner Shell	98	700

(1) The allowable temperatures are defined and referenced in MPC FSAR Table 4.1-3.

4.A.3.4 MPC-LACBWR Minimum Temperatures

MPC FSAR Section 11.1.4 provides the temperature distribution for the off-normal severe cold environmental conditions of -40°F. At this extreme condition, the components are above their minimum material limits.

4.A.3.5 Maximum MPC-LACBWR Internal Pressure for Normal Conditions

The MPC-LACBWR canister is backfilled with helium to atmospheric pressure (0.0 psig) and closed by welding. Normal condition pressure comprises the pressure due to the heating of the backfilled helium plus the pressure due to the postulated failure of 1 percent of the stored fuel rods with the subsequent release of 30 percent of the fission gas and all of the rod charge gas to the canister cavity, at temperature, from those failed rods. All of the gases, except the fission gases are helium. The total pressure for each volume is found by calculating the molar quantity of each gas and summing those directly. The calculated average normal condition temperature of the canister cavity gas is 359°F based on the thermal analysis results using the three-dimensional canister model described in MPC FSAR Section 4.4.1.2. The canister pressure is calculated using the Ideal Gas Law and applying a conservative average temperature of 370°F. The gas constant, R, is 0.0821 (atm x liters)/(Mole °K). The design basis fuel assembly for the internal pressure calculation is the Allis Chalmers (AC) assembly. This assembly has the highest burnup (22,000 MWd/MTU) and the highest fuel mass, and therefore generates the maximum fission gas quantities.

The number of moles of the canister and rod backfill gases is calculated using the Ideal Gas Law, PV = NRT. Backfill gas for the canister is assumed to be initially at 1 atmosphere absolute. The LACBWR fuel rod backfill pressure is also 1 atmosphere. The quantity of fission gas is derived from the SAS2H generated isotopics of the AC fuel assembly. For normal operating conditions, 1 percent of the fuel rods are assumed to fail, releasing 30 percent of their total fission gas and all of the backfill helium.

Rod Backfill Gas

The fuel rod plenum volume is:

$$V_1 = \pi r^2 L - \frac{M_{Spring}}{\rho}$$

where

M_{Spring} = the mass of the springs
ρ = the density of stainless steel.

The pellet clad gap volume is:

$$V_2 = \pi L\left(r^2_{Clad\ ID} - r^2_{Pellet\ OD}\right)$$

The total fuel rod backfill volume is:

$$V_{Rod\ Back\text{-}Fill} = V_1 + V_2$$

For the loaded canister, the total backfill gas volume is:

$$V = \text{Total Back-Fill}$$
$$= V_{Rod\ Back\text{-}Fill}\ \text{inches}^3 \times 100\frac{\text{Rods}}{\text{Assembly}} \times 68\frac{\text{Assemblies}}{\text{Canister}} \times \left(2.54\frac{\text{cm}}{\text{inch}}\right)^3 \times \frac{0.001\,\ell}{\text{cm}^3}$$

From the rod backfill volume and pressure, the quantity of rod backfill gas is calculated using the ideal gas law and a temperature of 68°F.

$$N = \frac{Pv}{RT}$$

$$N = 2.46\frac{\text{moles}}{\text{Canister}}$$

Conservatively rounded to:

$$N = 3\frac{\text{moles}}{\text{Canister}}$$

Fuel Fission Gas

The number of moles of fission gas per assembly is:

Isotope	Atomic Weight [grams/mole]	Mass [grams]	Number of Moles
H	3	0.0006	0.0002
HE	4	0.1360	0.0340
KR	82	0.0344	0.0004
KR	83	3.9400	0.0475
KR	84	9.5700	0.1139
KR	85	0.2190	0.0026
KR	86	16.3000	0.1895
I	127	3.0300	0.0239
I	129	13.9000	0.1078
XE	128	0.1090	0.0009
XE	129	0.0004	0.0000
XE	130	0.3210	0.0025
XE	131	38.7000	0.2954
XE	132	80.8000	0.6121
XE	134	122.0000	0.9104
XE	136	163.0000	1.1985
Total:	--	--	3.5396

For a full cask of 68 assemblies, the number of moles of fission gas per cask is 68 times that of the single assembly.

$$N = 68\frac{\text{Assemblies}}{\text{Cask}} \times 3.5396\frac{\text{Moles}}{\text{Assembly}} = 241\frac{\text{Moles of Fission Gas}}{\text{Cask}}$$

Canister (TSC) Backfill Gas

Canister backfill quantity is based on the canister free volume and backfill temperature. Backfill temperature applied is 68°F, which is conservative for a canister containing a heat-generating spent fuel payload.

$$V^{TSC}_{\text{Free Gas Volume}} = V_{\text{Canister}} - \left(V^{TSC}_{\text{Basket}} + V_{\text{Fuel}}\right)$$

$$V_{Canister} = \pi \frac{d^2}{4} \left(L_{Canister} - L_{TSC\ Bottom\ Plate}\right)$$

$$V_{Canister} = 411{,}160 \text{ inches}^3$$

Conservatively rounded to:

$$V_{Canister} = 410{,}000 \text{ inches}^3$$

$$V_{Basket}^{TSC} = 61{,}286.87 \text{ inches}^3$$

Conservatively rounded to:

$$V_{Basket}^{TSC} = 65{,}000.00 \text{ inches}^3$$

$$V_{Free\ Gas\ Volume}^{TSC} = 410{,}000 - \left(65{,}000 \text{ inches}^3 + 78{,}250 \text{ inches}^3\right)$$

$$V_{Free\ Gas\ Volume}^{TSC} = 266{,}750 \frac{\text{inches}^3}{\text{Canister}}$$

$$V_{Free\ Gas\ Volume}^{TSC} = 266{,}750 \frac{\text{inches}^3}{\text{Canister}} \times \frac{1\,\ell}{61.02 \text{ inches}^3} = 4{,}371 \frac{\ell}{\text{Canister}}$$

$$N = \frac{1 \text{ atm} \times 4{,}371 \dfrac{\ell}{\text{Cask}}}{0.0821 \dfrac{\text{atm}\ \ell}{\text{Mole K}} \times 293 \text{ K}} = 181.62 \frac{\text{Moles of TSC Backfill Gas}}{\text{Canister}}$$

Canister Pressure

The maximum normal operating pressure (MNOP) in the canister is calculated using the ideal gas law where:

$$N = N_{TSC\ Back\text{-}Fill} + 0.01\left(N_{Rod\ Back\text{-}Fill}\right) + 0.3(0.01)\left(N_{Fission\ Gas}\right)$$

$$N = 182.3 \frac{\text{Moles}}{\text{Canister}}$$

Therefore, the maximum normal operating condition canister internal pressure is:

$$P = \frac{\left(182.3\ \frac{\text{Moles}}{\text{Canister}}\right)\times\left(0.0821\,\frac{\text{atm}\ \ell}{\text{mole K}}\right)\times 465\ \text{K}}{\left(4{,}371\,\frac{\ell}{\text{Canister}}\right)} = 1.59\ \text{atm} \approx 23.4\ \text{psia} \approx 8.7\ \text{psig}$$

4.A.3.6 Maximum MPC-LACBWR Thermal Stresses for Normal Conditions

The canister and concrete storage cask thermal stresses are evaluated in MPC FSAR Section 3.4.4.3 and Section 3.4.4.4.

4.A.3.7 Evaluation of MPC-LACBWR Performance for Normal Conditions

As shown in the preceding sections, the MPC-LACBWR system operates within the thermal design limits. Therefore, no degradation due to temperature effects on material or components is expected over the lifetime of the cask.

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4.A.4 References

[A1] Military Handbook, MIL-HDBK-5F, November 1990.

[A2] "The Properties of Gases and Liquids," Bruce E. Poling, et al, 5th Edition, McGraw Hill, 2001

[A3] "Handbook of Concrete Engineering," M. Fintel, Van Nosttrand, Reinhold Co., New York, Second Edition, 1985.

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