PRELIMINARY SAFETY EVALUATION REPORT

NAC INTERNATIONAL, INC.

MPC-LACBWR STORAGE SYSTEM

CERTIFICATE OF COMPLIANCE NO. 1025

AMENDMENT NO. 6

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INTRODUCTION

By application dated January 16, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No.: ML090270151), as supplemented February 11, 2009 (ML090490720), April 1, 2009 (ML090960575), April 30, 2009 (ML091250187), September 22, 2009 (ML092680315), and January 8, 2010 (ML100120178), NAC International, Inc. (NAC) requested approval of an amendment, under the provisions of 10 CFR Part 72, Subpart K, to the NAC-MPC Certificate of Compliance (CoC) No. 1025 for the Multi-Purpose Canister (MPC) Storage System. NAC, via this amendment to CoC No. 1025, requested approval to store, in the NAC-MPC system, spent fuel assemblies from the decommissioned Dairyland Power Cooperative (DPC) La Crosse Boiling Water Reactor (LACBWR) nuclear power plant. The storage system for DPC is designated MPC-LACBWR.

The changes proposed for Amendment No. 6, constitute the third configuration of the NAC-MPC storage system include:

a. Incorporation into the Transportable Storage Canister (TSC) design a single closure lid with a welded closure ring for redundant closure (design features from the MAGNASTOR system);
b. Modification of the TSC and basket design to accommodate up to 68 LACBWR spent fuel assemblies (36 undamaged Exxon fuel assemblies) and up to 32 damaged fuel cans (in a preferential loading pattern) that may contain undamaged Exxon fuel assemblies, damaged Exxon and Allis Chalmers fuel assemblies and/or fuel debris;
c. Minor design modifications to the Vertical Concrete Cask (VCC) incorporating design features from the MAGNASTOR system that improve operability of the system while adhering to ALARA principles;
d. Request for the addition of zirconium alloy shroud compaction debris to be stored with undamaged and damaged fuel assemblies;
e. To change concrete pad compressive strength from 4,000 psi to 6,000 psi;
f. Proposed justification for the 6-foot soil depth as being conservative; and
g. Other changes to incorporate minor editorial corrections.

The Limiting Condition for Operation (LCO) and associated Surveillance Requirement (SR) Applicability of the Technical Specifications affected include the Canister, Concrete Cask, and Fuel Cooldown Requirements.

a. CANISTER
   LCO 3.1.1 – Maximum Time in Vacuum Drying
   LCO 3.1.2 – Vacuum Drying Pressure
   LCO 3.1.3 – Helium Backfill Pressure
   LCO 3.1.5 – Helium Leak Rate
   LCO 3.2.1 – Surface Contamination

b. CONCRETE CASK
   LCO 3.1.6 – Heat Removal System
   LCO 3.2.2 – Average Surface Dose Rates

c. LCO 3.1.7 – Fuel Cooldown Requirements

Definitions, Section A 1.1, contain modifications and newly defined terms. Two sections of the Administrative Controls and Program section of the TS are also affected: A 5.2 Preoperational Testing and Training Exercises and A 5.4 Radioactive Effluent Control Program (editorial). Practically every section of the Approved Contents and Design Features of the NAC-MPC System – Amendment 6 (Appendix B) has been modified and lists the specifications associated with dry storage of LACBWR spent fuel.

This Safety Evaluation Report (SER) documents the review and evaluation of Revisions MPC-08A, 09A, 09B, 09C and 10A to the Final Safety Analysis Report (FSAR) for the NAC-MPC System, MPC-LACBWR Amendment. The FSAR was submitted by NAC following the format of Regulatory Guide 3.61. This SER primarily uses the section-level format of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, with some differences implemented for clarity and consistency.

The staff's assessment is based on whether the applicant meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection.

While components of the MPC-LACBWR system are designed to be used in conjunction with the NAC Storage Transport Cask (NAC-STC) for a dual purpose function, the use or certification of the NAC-STC under 10 CFR Part 71 for the off-site transport of MPC-LACBWR authorized spent fuel contents is not a subject of this SER. Certification for transportation of the MPC-LACBWR authorized spent fuel contents occurs upon the completion of a separate staff review to amend the NAC-STC 10 CFR Part 71 CoC for transportation.

LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BWR	Boiling Water Reactor
CoC	Certificate of Compliance
DCSS	Dry Case Storage System
FSAR	Final Safety Analysis Report
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance
k_{eff}	k-effective
kW	Kilowatt
MPC	Multi-Purpose Canister
NAC	NAC International, Inc.
NRC	Nuclear Regulatory Commission
PT	Liquid Penetrant Examination
QA	Quality Assurance
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SSCs	Structures, Systems and Components
STC	Storage Transport Cask
TC	Transfer Cask
TS	Technical Specifications
TSC	Transportable Storage Canister
VCC	Vertical Concrete Cask

1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the NAC International Inc. (NAC) Multi-Purpose Canister La Crosse Boiling Water Reactor (MPC-LACBWR) storage system is to ensure that NAC has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.1 MPC-LACBWR System Description and Operational Features

NAC's amended design of the Multi-Purpose Canister (MPC) system (NAC-MPC) for the long-term storage of Dairyland Power Cooperative (DPC) La Crosse Boiling Water Reactor (LACBWR) irradiated nuclear fuel (spent fuel) is designated the MPC-LACBWR system.

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. The canister is stored in the central cavity of the concrete cask and is compatible with the NAC-STC transport cask (Certificate of Compliance No. 71-9235) for future offsite 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 consist of a transportable storage canister (TSC), vertical concrete cask (VCC), and a 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 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 1-2). The re-designed MPC-LACBWR fuel basket can store up to sixty-eight (68) LACBWR spent fuel assemblies: thirty-six (36) undamaged Exxon fuel assemblies and up to thirty-two (32) damaged fuel cans (DFC). The DFCs are to be placed in peripheral oversized fuel tubes. The DFC contents may contain undamaged Exxon fuel assemblies, damaged Exxon and Allis Chalmers fuel assemblies and/or fuel debris; and zirconium alloy shroud compaction debris stored with undamaged and damaged fuel assemblies. The MPC-LACBWR storage system is designed to store DPC La Crosse BWR spent fuel with a maximum total heat load of 4.5 kW, or an average heat load of 0.0662 kW per assembly for all locations with or without damaged fuel can confinement.

1.1.1 MPC-LACBWR System Components

The MPC-LACBWR system consists of three principal components: 1) transportable storage canister (TSC); 2) vertical concrete cask (VCC); and 3) transfer cask.

1.1.1.1 Transportable Storage Canister (TSC) and Baskets

The TSC (canister) contains a basket that is designed to accommodate up to 68 LACBWR spent fuel assemblies, including up to 32 damaged fuel cans and is designed to be transported in the NAC-STC.

The TSC has an outside diameter of about 71 inches and is about 116 inches long. The weight of the loaded TSC is slightly less than 55, 000 lbs. As shown in Figure 1-2, the TSC contains a

basket that accommodates up to 68 LACBWR spent fuel assemblies. 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, constitutes 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. 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. 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 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 MPC-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.

1.1.1.2 Vertical Concrete Cask (VCC)

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. The VCC is approximately 160 inches in height, has an outside diameter of about 128 inches, and weighs about 141,000 lbs. The VCC side walls consists of about 22 inches of reinforced concrete (Type II Portland cement), with a 2.5 inch carbon steel liner. It provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the TSC during long-term storage. The VCC has an annular air passage to allow the natural circulation of air around the TSC to remove the decay heat from the spent fuel. The air inlets and outlets are steel-lined penetrations that take non-planar 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. The storage cask is shown in Figure 1-3.

1.1.1.3 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 has an outside diameter of about 87 inches, wall thickness of 15 inches, height of approximately 134 inches, and empty weighs about 81,000 lbs.

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 1-4 and the arrangement of the transfer cask and concrete cask is shown in Figure 1-5.

The transfer cask is a multi-wall (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 cask.

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.

1.1.1.4 Ancillary Equipment

Section 1.A.2.1.4 of the SAR presents a brief description of the principal ancillary equipment needed to operate the MPC-LACBWR system in accordance with its design:

- Adapter Plate – mates the transfer cask to either the VCC or the NAC-STC transport cask.
- Air Pad Rig Set – allows movement of the VCC on the storage pad.
- Automatic Welding System – minimizes radiation exposure during TSC closure welding.
- Draining and Drying System – used to remove moisture and establish a TSC vacuum.
- Helium Leak Test Equipment – mass spectrometer to verify the integrity of the shield lid weld.
- Heavy-Haul Trailer – used to move the VCC.
- Lifting Jacks – used to lift the VCC to insert/remove the Air Pad Rig Set.
- Rigging and Slings – provided for major components such as shield and structural lids and the transfer cask.

- Temperature Instrumentation – located at VCC outlets and inlet for local and/or remote temperature indications.

1.1.1.5 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 about 116 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 of the SAR.

1.1.2 Operational Features

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 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 of the SAR.

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 of the SAR.

1.2 MPC-LACBWR Storage System Contents

The MPC-LACBWR storage system is designed to hold up to sixty-eight (68) spent fuel assemblies with thirty-two (32) oversized peripheral cell locations to accommodate damaged fuel cans. Of the 68 spent fuel assemblies, thirty-six (36) contain undamaged Exxon fuel assemblies and up to thirty-two (32) damaged fuel cans. The damaged fuel cans are placed in peripheral oversized fuel tubes. The contents may contain undamaged Exxon fuel assemblies, damaged Exxon and Allis Chalmers fuel assemblies and/or fuel debris; and zirconium alloy shroud compaction debris stored with undamaged and damaged fuel assemblies.

1.2.1 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 as follows.

Design Basis Fuel Characteristics Parameter	**Allis Chalmers**	**Exxon**
Number of Assemblies per Canister[1]	32	68
Maximum 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.

Unenriched fuel assemblies are not evaluated and are not included as proposed contents.

1.2.2 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 minimize the dispersal of particulates. A sketch of the MPC-LACBWR damaged fuel can is provided in Figure 1-6. The MPC-LACBWR damaged fuel can has an overall length of about 107 inches, outside cross-section of 5.85 inches with an 18-gauge wall thickness (about 0.05 inch), the internal cavity length is about 104 inches and empty, the nominal weight is about 55 lbs. The design and fabrication specification summary is provided in Table 1.A.3-3 of the SAR.

1.3 MPC-LACBWR Storage Cask Arrays

Section 1.A.4 of the SAR describes and depicts a typical storage pad layout for an Independent Spent Fuel Installation (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 1-7 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 of the SAR.

1.4 Qualifications of the Applicant

NAC is the prime contractor for the MPC-LACBWR design, and all design and specification activities are performed by NAC. Fabrication of 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 (QA) programs meeting the requirements of 10 CFR Parts 71 and 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-128 S/T metal casks
6 RH-TRU 72B	1 NAC-126 S/T metal cask
2 NAC-STC	> 210 UMS®/MPC TSCs
	> 212 UMS®/MPC concrete casks

1.5 MPC-LACBWR License Drawings

The drawings associated with the MPC-LACBWR structures, systems, and components (SSCs) important to safety are provided in Section 1.A.7 of the SAR. Sufficiently detailed drawings regarding dimensions, materials, and specifications were provided by the applicant and allow a thorough evaluation of the entire system. Specific SSCs are evaluated in Sections 3 through 14 of this SER.

1.6 Quality Assurance

The QA program is evaluated in Section 13 of this SER.

1.7 Security Assessment

NRC staff evaluated the Model No. MPC-LACBWR storage cask and documented the security assessment separately, as it contains sensitive information that cannot be made publicly available. The security assessment should be reviewed prior to approval of any amendment to this application.

1.8 Evaluation Findings

F1.1 A general description and discussion of the dry cask storage system (DCSS) is presented in Section 1 of the NAC-MPC FSAR, MPC-LACBWR Amendment with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

F1.2 Drawings for structures, systems, and components (SSCs) important to safety are presented in Section 1.A.7 of the NAC-MPC FSAR, MPC-LACBWR Amendment. A listing of those drawings that were relied upon as a basis for approval appears in Section 12 of the Safety Evaluation Report (SER).

F1.3 Specifications for the spent fuel to be stored in the DCSS are provided in SAR Section 2.A.1. Additional details concerning these specifications are presented in Chapter 2 of both the SAR and SER.

F1.4 The technical qualifications of the applicant to engage in the proposed activities are identified in Section 1.A.6 of the SAR.

F1.5 The quality assurance program, and implementing procedures, is described in Section 13 of the SAR.

F1.6 The MPC-LACBWR is not being certified under 10 CFR Part 71 for use in transportation with this application.

The staff finds that the information presented in Chapter 1, “General Information” of the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, Regulatory Guide 3.61, and accepted practices.

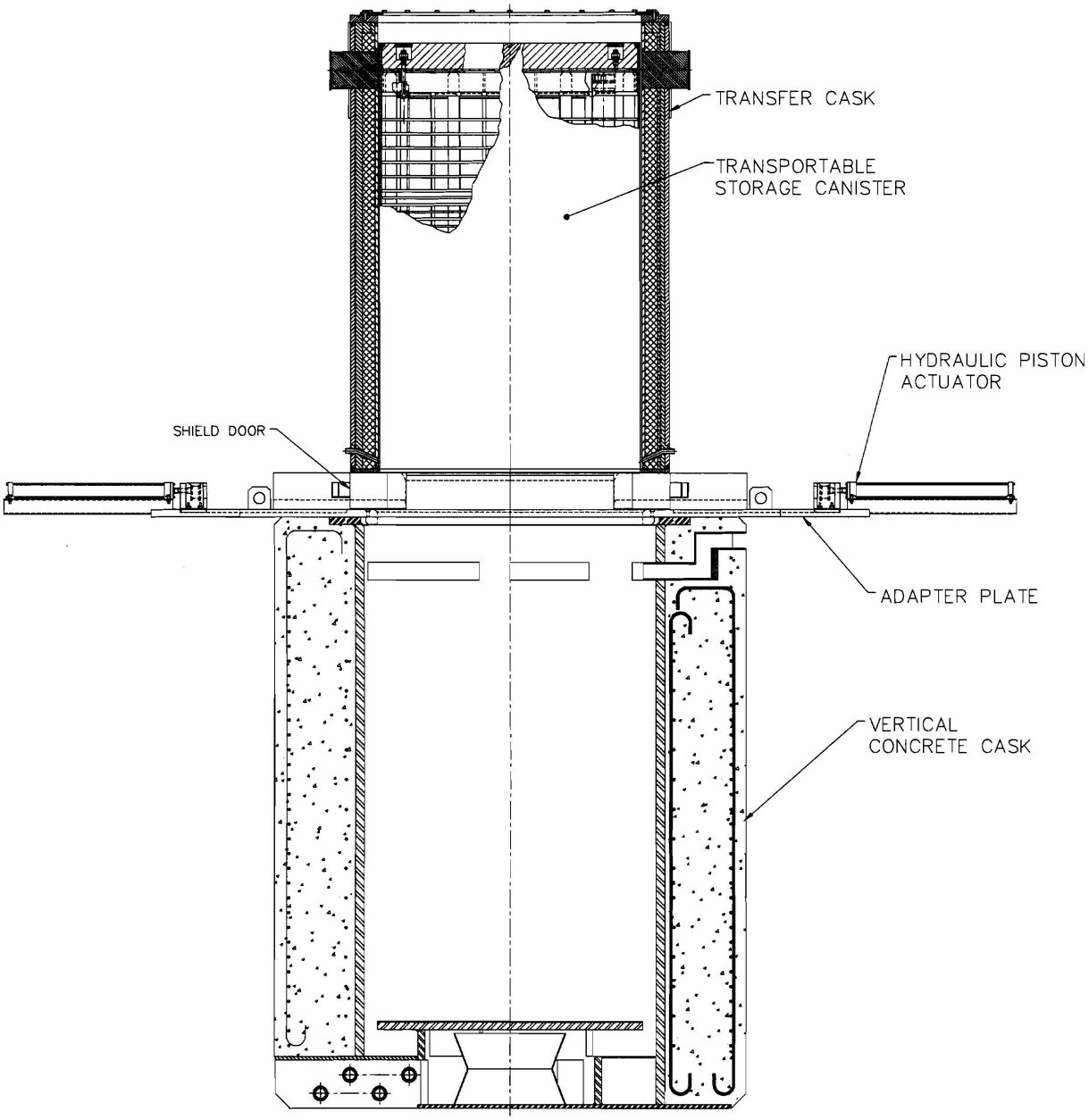

Figure 1-1 Major Components of the MPC-LACBWR System (Figure 1.A.1-1)

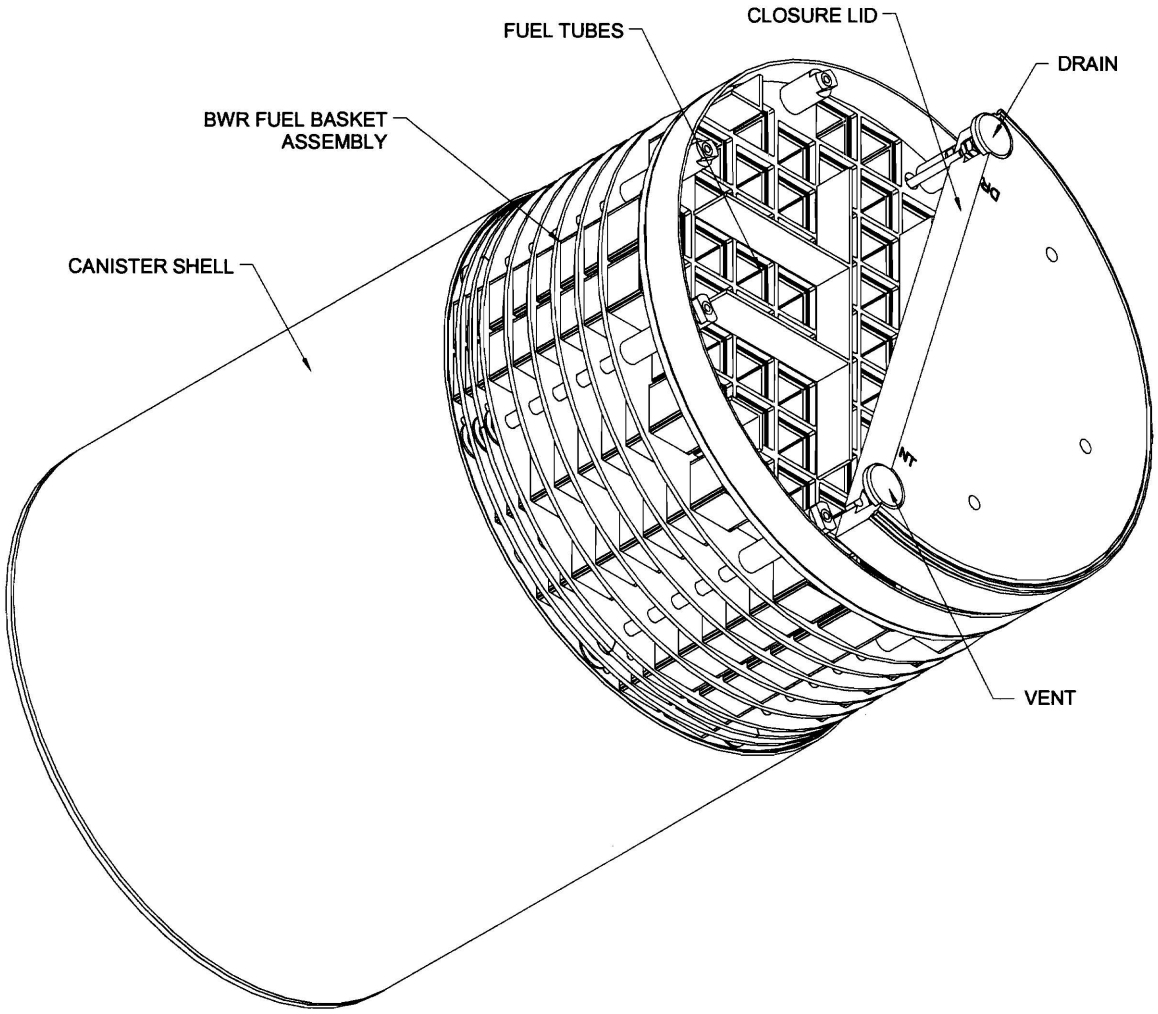

Figure 1-2 MPC-LACBWR Transportable Storage Canister Showing the Spent Fuel Basket (Figure 1.A.1-2)

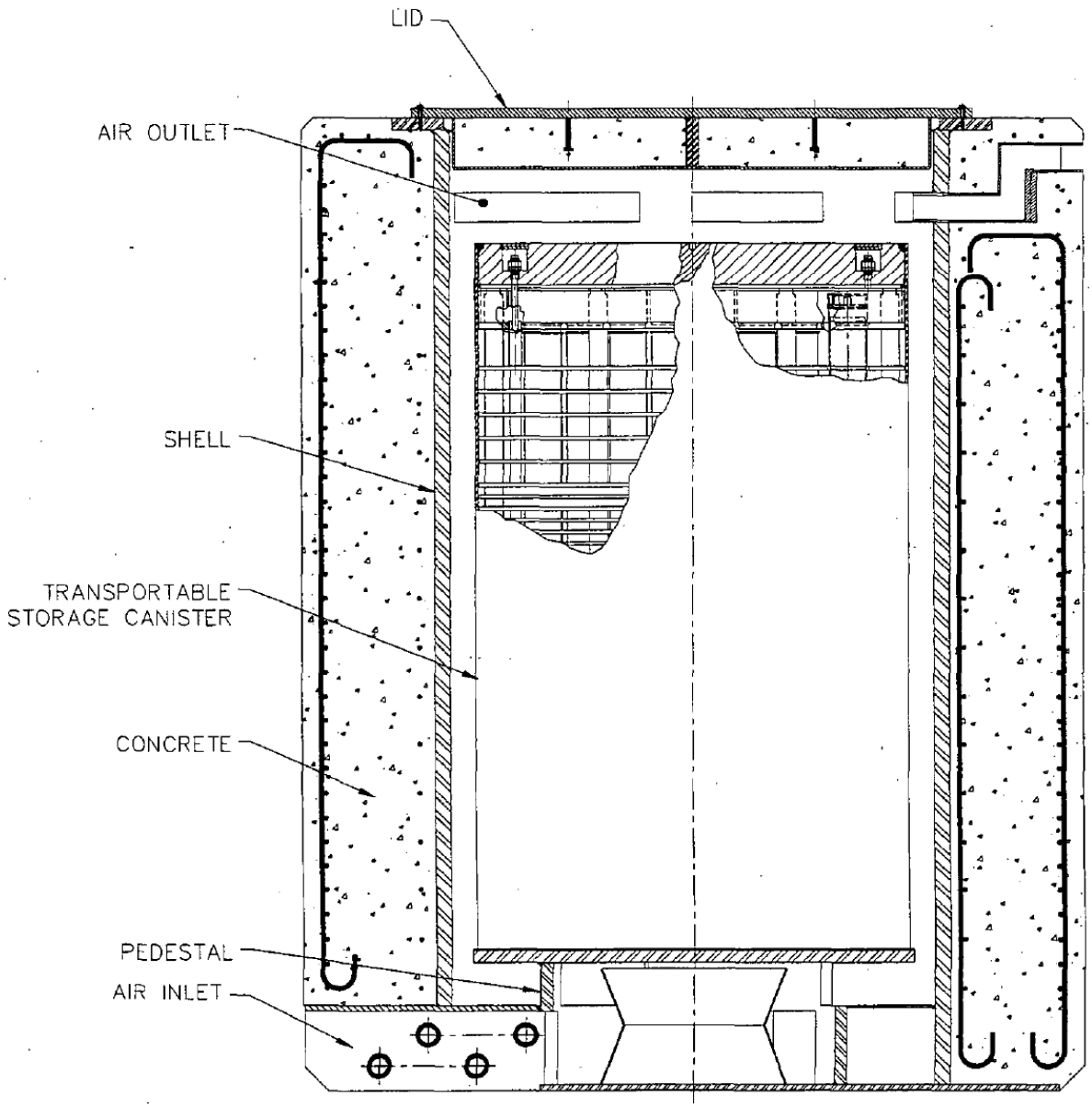

Figure 1-3 MPC-LACBWR Vertical Concrete Storage Cask (Figure 1.A.2-1)

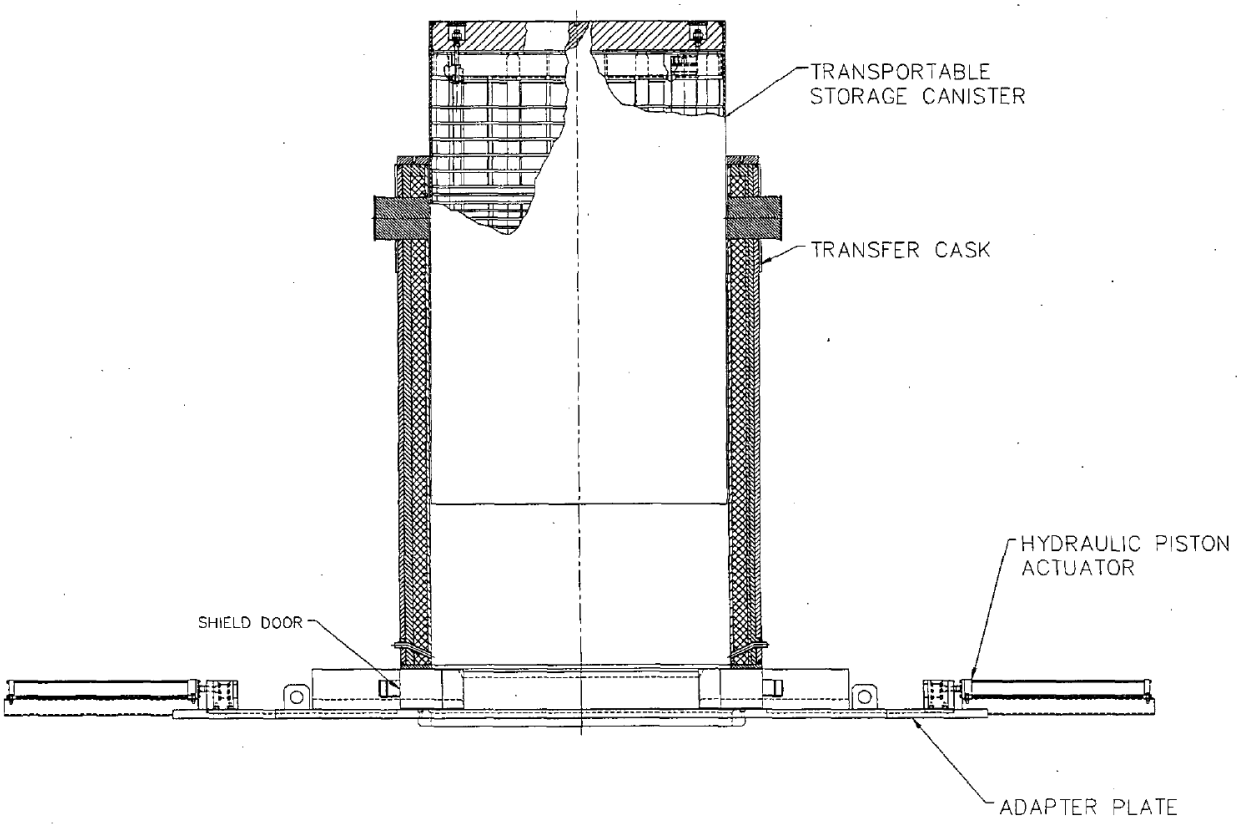

Figure 1-4 MPC-LACBWR Transfer Cask and Canister Arrangement (Figure 1.A.2-4)

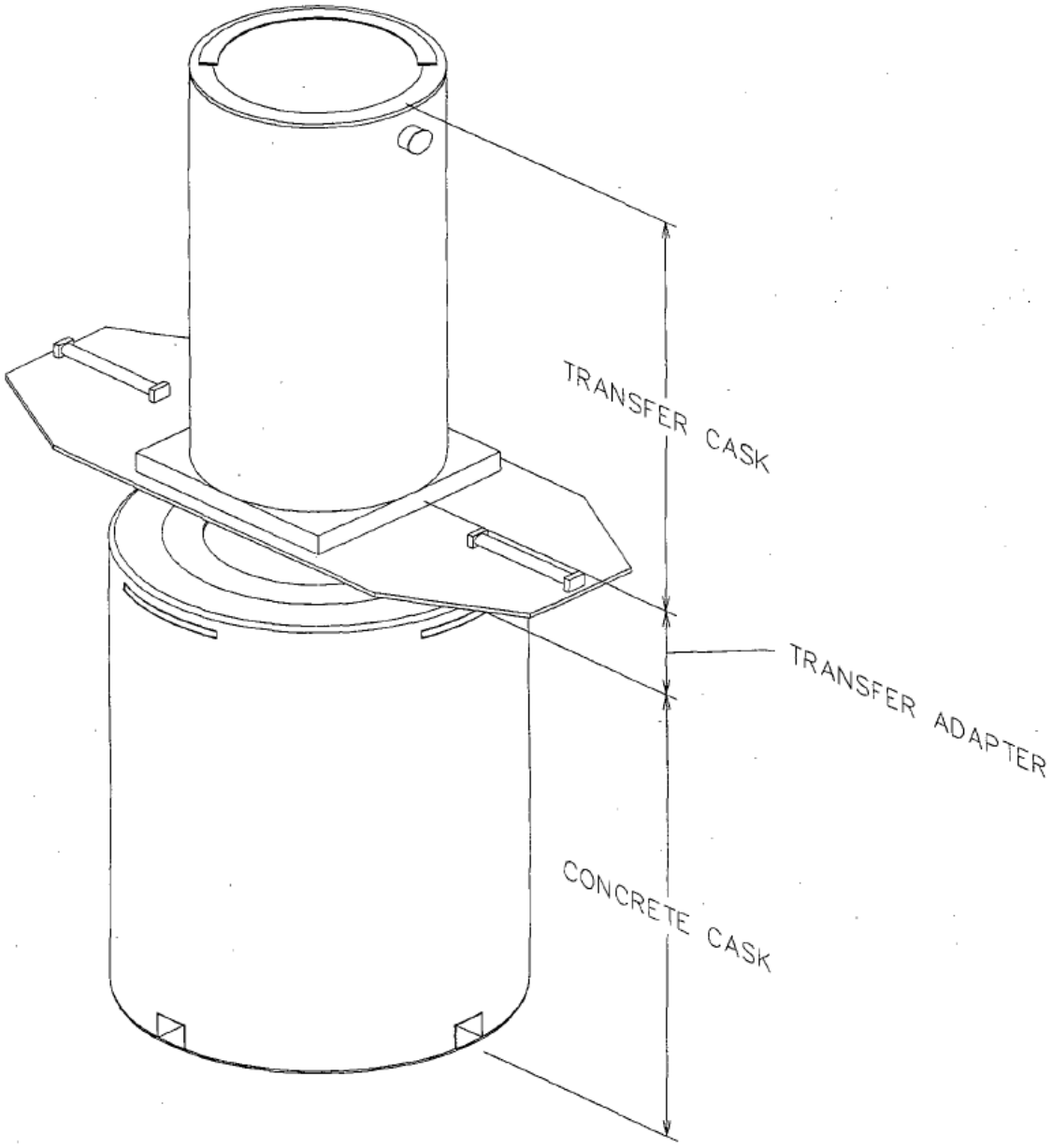

Figure 1-5 MPC-LACBWR Vertical Concrete Cask (VCC) and Transfer Cask (TC) Arrangement

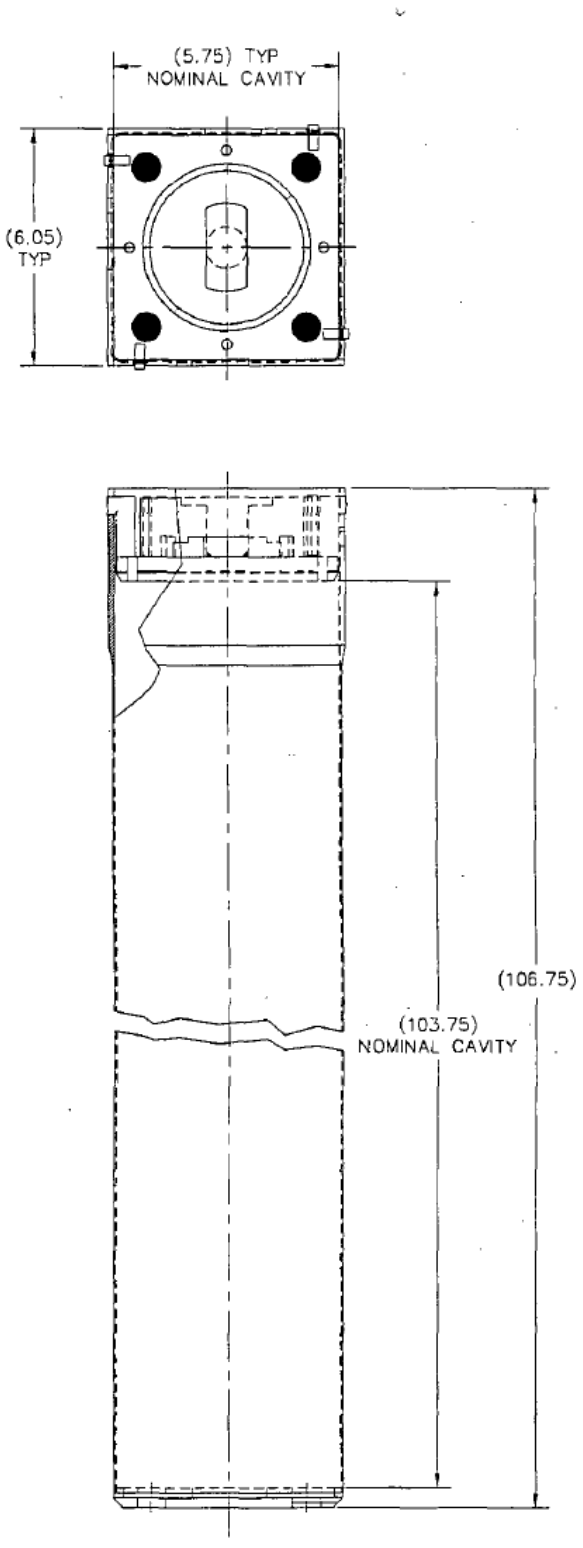

Figure 1-6 MPC-LACBWR Damaged Fuel Can

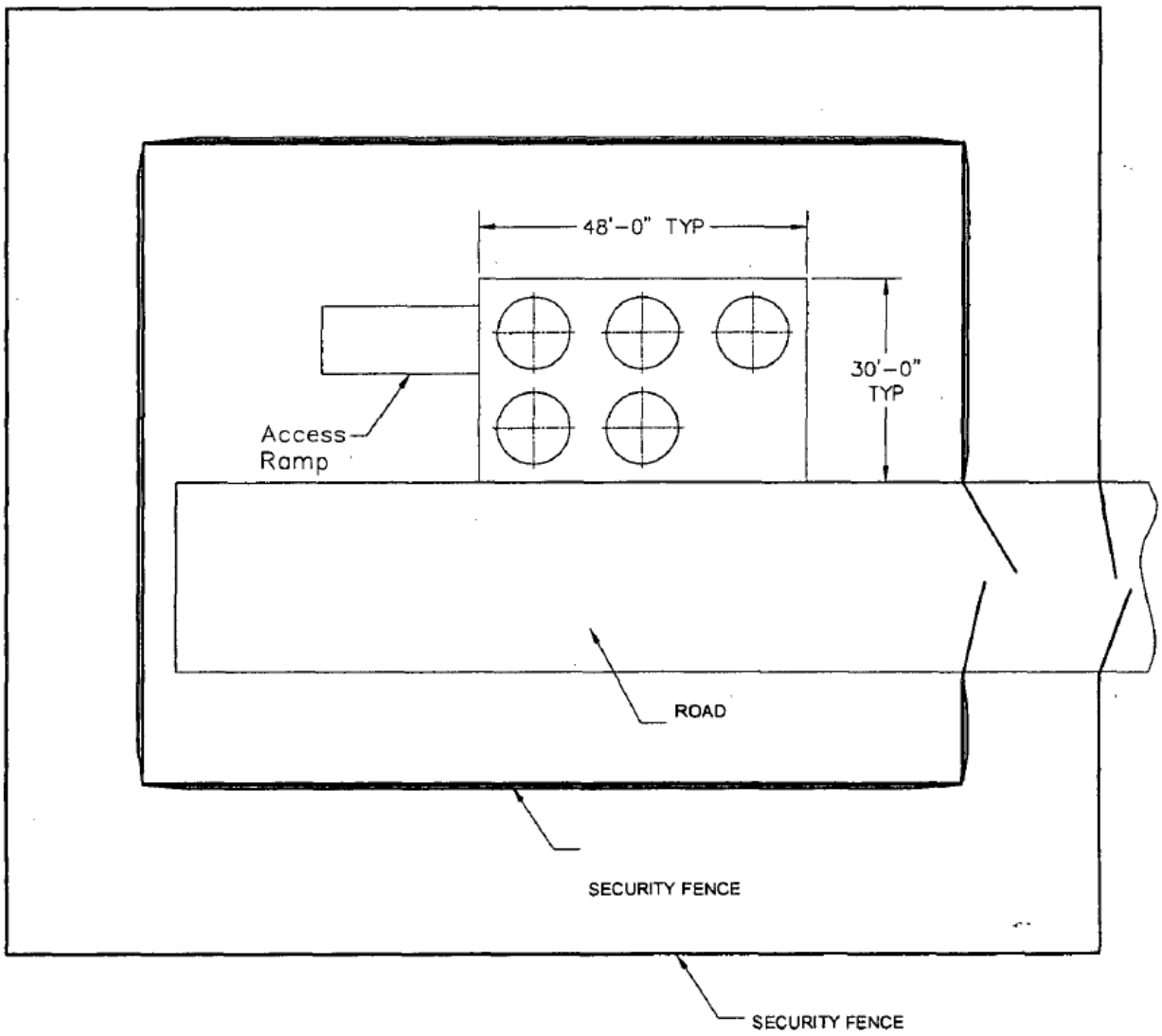

Figure 1-7 Conceptual ISFSI storage pad layout

2.0 PRINCIPAL DESIGN CRITERIA

The objective of evaluating the principal design criteria related to structures, systems, and components (SSC) important to safety is to ensure that they comply with the relevant general criteria established in 10 CFR Part 72.

2.1 Structures, Systems and Components Important to Safety

NAC presented a summary of the principal MPC-LACBWR storage system design criteria in Table 2.A-1 of the SAR. Each MPC-LACBWR storage system component is assigned, in Table 2.A.3-1 of the SAR, a safety classification based on the components function and an assessment of the consequences of component failure. The component safety classifications are based on the guidance of NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety.”

2.2 Design Bases for Structures, Systems and Components Important to Safety

NAC’s design bases summary for the MPC-LACBWR system identified the range of spent fuel configurations and characteristics, the enveloping conditions of use, and the bounding site characteristics.

2.2.1 Spent Fuel Specifications

The MPC-LACBWR is designed to store up to 68 Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) fuel assemblies. Table 2.A.1-1 of the SAR and the TS provide detailed fuel assembly characteristics for the authorized contents. These characteristics include: manufacturer, assembly array, physical assembly dimensions, maximum and minimum enrichments, maximum burnup, minimum cool time, maximum decay heat, total weights per assembly, and initial uranium weight per assembly. Detailed parameters regarding the configuration of individual fuel rods, fill gas pressures, and fuel assembly hardware are also provided.

The MPC-LACBWR is also designed to store up to 32 damaged fuel cans (DFC). The contents may be either an undamaged or a damaged LACBWR spent fuel assembly and/or fuel debris. The LACBWR 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 of the SAR. The MPC-LACBWR DFC design and fabrication specification summary is provided in Table 1.A.3-3 of the SAR. The major physical design parameters are provided in Table 1.A.3-2 of the SAR.

2.2.2 LACBWR Nonfuel Hardware

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

2.2.3 LACBWR Damaged Fuel

A transportable storage canister configured for damaged fuel holds up to 32 damaged fuel cans located in the "B" and "C" positions of the basket shown in Figure 2.A.1-1 of the SAR. A DFC may contain either an undamaged or a damaged LACBWR spent fuel assembly of the types described in Table 2.A. 1-1 of the SAR, 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.

2.2.4 Design Criteria for Environmental Conditions and Natural Phenomena for the MPC-LACBWR Dry Storage System

Section 2.A.2 of the SAR identifies the site environmental conditions and natural phenomena for which the storage system is analyzed during the period of storage. The SAR presents analyses that demonstrate that the MPC-LACBWR system meets the design criteria in subsequent SAR sections, and which is further evaluated in Sections 3 through 14 of this SER.

2.3 Safety Protection Systems

A summary of the Safety Protection Systems is provided in Section 2.A.3 of the SAR. These systems are further evaluated in Sections 3 through 14 of this SER.

2.3.1 General

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 single-failure proof and designed to meet the requirements of American National Standards Institute (ANSI) N14.6 and NUREG-0612. The lifting yoke is proof load tested to 300% of design load upon fabrication and visually inspected prior to each use and inspected annually.

2.3.2 Structural

The structural design criteria presented in SAR Section 2.2 of the NAC-MPC FSAR applies in its entirety to the MPC-LACBWR. The seismic event (earthquake) during storage on an unsheltered concrete pad at an ISFSI site depends on geographic location and distance from the epicenter of the earthquake. 10 CFR 72.102 defines a 0.10g horizontal ground motion design earthquake as the minimum allowable seismic design criteria, and 0.25g is suggested for sites east of the Rocky Mountain front. MPC-LACBWR is designed to 0.45g horizontal and 0.3g vertical seismic acceleration. Confinement of the stored radioactive material is assured by adequate margins of safety during normal, off-normal, and accident conditions and is evaluated in Sections 2.A.3.2, 11.A.1, and 11.A.2 of the SAR, respectively. Similarly, adequate structural margins of safety ensure adequate criticality control as evaluated in Section 6 of the SAR. The thermal analysis (Section 4 of the SAR) determined that cladding temperatures will not exceed the specified maximum allowable cladding temperatures, thereby, protecting the fuel cladding against degradation during storage.

2.3.3 Thermal

The passive heat removal capabilities of the MPC-LACBWR are described in Section 4 of the SAR. Operating limits and verification are established in the TS to ensure continued safe operation. A remote temperature monitoring system may be used to measure the outlet air temperature of the system during 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.

2.3.4 Shielding/Confinement/Radiation Protection

The MPC-LACBWR confinement system is closed by welding. The TSC closure lid weld is liquid penetrant examined. The closure lid inner port covers and outer port covers are sealed by welding and are liquid penetrant tested on the root and/or final surface. The inner port cover is leak tested to 1.0×10^{-7} cubic centimeters per second (air). The canister shell is leak tested at

fabrication to 1.0 x 10^{-7} cubic centimeters per second (air). The longitudinal and girth welds, and bottom welds of the TSC shell are full penetration welds. The longitudinal and girth welds are radiographically inspected during fabrication. Bottom welds are ultrasonically inspected during fabrication. None of the evaluated normal, off-normal, or accident conditions result in a breach of the TSC. The TSC is designed to withstand hypothetical 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 TSC is minimized by both design features and operational procedures. The closure lid is placed on the TSC while the transfer cask and canister are under water in the spent fuel pool. The exterior of the transfer cask is decontaminated prior to draining the TSC to preserve the shielding benefit of the water. Temporary shielding is used. The retaining ring on the transfer cask prevents inadvertent lifting of the TSC beyond the transfer casks shielding.

Access to the LACBWR ISFSI site is controlled by a peripheral fence to meet the requirements of 10 CFR Part 72 and 10 CFR Part 20. The shielding associated with the system design is in accordance with the requirements of 10 CFR 72.104 and 72.106 for normal and accident conditions, respectively.

2.3.5 Criticality

Neutron absorber sheets (BORAL) are employed in the basket design. BORAL sheets are attached to the side of fuel tubes to have each fuel assembly separated from the adjacent assembly. Fuel tubes containing damaged fuel cans have additional absorber sheets attached to provide flux traps. 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.

2.3.6 Operating Procedures

The operating procedures descriptions are discussed in Section 8 of the SAR and include procedures for wet and dry loading and unloading operations. Radiation protection design features, including features to facilitate decontamination, are incorporated in both the physical design and the operating procedures.

2.3.7 Acceptance Tests and Maintenance

The acceptance tests and maintenance of the MPC-LACBWR system are described in Section 9 of the SAR, including the commitments, industry standard, and regulatory requirements used to establish the acceptance, maintenance, and periodic surveillance tests.

2.3.8 Decommissioning Considerations

Decommissioning of the MPC-LACBWR system is described in Section 2.A.4 of the NAC-MPC FSAR but is not evaluated in this SER.

2.4 Evaluation Findings

The staff finds that the principal design criteria for the MPC-LACBWR are acceptable with regard to meeting the regulatory requirements of 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. A more detailed evaluation of design criteria and an assessment of compliance with those criteria as presented in Sections 3 through 14 of the SER.

3.0 STRUCTURAL

3.1 Structural Design of the MPC-LACBWR Storage System

This section describes the design of the principal structural components of the Multi- Purpose Canister La Crosse Boiling Water Reactor (MPC-LACBWR) storage system and provides the structural evaluation for environmental conditions and natural phenomena. The applicant uses as a reference, two previously approved configurations (Yankee-MPC and CY-MPC) to bound and support, as appropriate, the evaluation and design of the MPC-LACBWR. Primary comparisons and bounding evaluations are performed utilizing the Yankee MPC. The basket assembly for this configuration is designed to store up to 68 Dairyland Power Cooperative LACBWR spent fuel assemblies, including up to 32 LACBWR damage fuel cans.

Vertical Concrete Cask (VCC)
The VCC for the MPC-LACBWR is nearly identical to the Yankee-MPC in overall dimensions, materials of construction, and functional design. Exceptions are that the MPC-LACBWR has a thinner steel liner, 2.5 inches instead of 3.5 inches and a thicker reinforced concrete shell, 22 inches instead of 21 inches. Section 3.A.1.1 of the NAC-MPC FSAR notes that the MPC-LACBWR concrete cask weighs slightly less that the Yankee-MPC concrete cask.

Transportable Storage Canister (TSC)
The TSC is similar in materials and construction to the Yankee-MPC canister design. The primary components are a welded right circular cylinder, which provides the containment boundary, and a basket assembly, composed of fuel compartment tubes supported by 26 support disks and 8 tie rods. The support disks have variably sized square cutouts to accommodate up to 68 fuel assemblies or a combination of fuel assemblies and damaged fuel cans. The MPC-LACBWR damaged fuel can design is a slightly smaller version of the Yankee-MPC and the basket cells are sized to accommodate this variation.

Transfer Cask (TC)
The same transfer cask that was used for the Yankee-MPC storage system is used for the MPC-LACBWR. Yankee-MPC structural evaluations are bounding for the TC, therefore no additional evaluation is required.

3.2 Weights and Centers of Gravity

Table 3.A.2-1 summarizes the weights and centers of gravity for the MPC-LACBWR system components and subassemblies.

3.3 Mechanical Properties of Materials

See Materials Section of this SER.

3.4 General Standards for All Casks

3.4.1 Chemical and Galvanic Reactions

The materials of construction for the MPC-LACBWR are the same as for the previously approved Yankee-MPC; therefore no further evaluation is required.

3.4.2 Positive Closure

Positive closure for the MPC-LACBWR is provided by the same means as for the previously approved Yankee-MPC; therefore no further evaluation is required.

3.4.3 Lifting Devices

The MPC-LACBWR storage system components are lifted using the same methods described in NAC-MPC FSAR Section 3.4.3.

MPC-LACBWR Storage Cask Bottom Lift - The applicant demonstrates the following for the MPC-LACBWR concrete storage cask for the bottom lift: (1) the hydraulic jack piston size used to lift the Yankee-MPC concrete cask is adequate for lifting the MPC-LACBWR concrete cask, (2) the margin of safety against failure of the Nelson studs used to secure the cask bottom closure plate is higher for the MPC-LACBWR concrete cask, (3) 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, and (4) the air pad system used to move the Yankee-MPC concrete cask is adequate for use with the MPC-LACBWR concrete cask. These conclusions are based on the as loaded design weight for the Yankee MPC which exceeds that of the MPC-LACBWR which is at a minimum, geometrically identical, and in some cases more structurally robust.

MPC-LACBWR Canister Lift – The dimensions of the lifting apparatus are the same as for the previously approved Yankee-MPC Canister design. The applicant demonstrated that the canister lifting components were designed to a minimum factor of safety of 3 against yielding and a minimum factor of safety of 5 against ultimate tensile strength. This evaluation was conservatively performed by assuming that only one of two lifting slings was engaged and that with this arrangement, the design basis allowable stresses were satisfactory.

MPC-LACBWR Transfer Cask Lift - The applicant demonstrated that the stresses in the MPC-LACBWR Transfer Cask are bounded by those calculated for the Yankee-MPC. The basis for this conclusion is that the empty and loaded Yankee-MPC Transfer Cask weights exceed those for the MPC-LACBWR.

MPC-LACBWR Damaged Fuel Can Lift - The dimensions of the lifting apparatus are the same as for the previously approved CY-MPC DFC design, however, the applicant through classical calculation methods demonstrated that all DFC lifting components are designed to a minimum factor of safety of 3 against yielding. Staff evaluated the calculations and found them to be satisfactory.

3.4.4 MPC-LACBWR Components under Normal Operating Loads

3.4.4.1 MPC-LACBWR Canister Analysis

Thermal Stress – Thermal stresses for use in Normal Conditions which enveloped ambient conditions were obtained from the thermal analysis. Key temperatures from those analyses were applied as nodal loads in an ANSYS Finite Element (FE) model and a steady state conduction solution was obtained. The stresses due to thermal effects were classified as secondary stresses and were applied accordingly to applicable load combinations.

Dead Weight Load – The dead weight load stresses for the canister are derived by applying a pressure based on a maximum bounding payload weight in addition to applying a 1g acceleration in the axial direction to the FE model to simulation the dead load of the canister body. Element stress intensities were obtained and applied accordingly to applicable load combinations.

Internal Pressure – The canister is evaluated with a design internal pressure intended to bound both normal and off normal conditions. This pressure was applied to the interior of the FE model and the element stress intensity results were applied accordingly to applicable load combinations.

Handling – See section 3.4.3 of this SER.

Load Combinations – The MPC-LACBWR canister is evaluated against allowable stresses for the effects of thermal, dead weight, internal pressure and handling loads. The stress intensity FE results from the individual loads are summarized in Tables 3.A.4.4.1-1 through 3.A.4.4.1-5 and the load combination stress intensities and governing Margins of Safety are presented in Tables 3.A.4.4.1-6 through 3.A.4.4.1-8. The results indicate that in no case does the Margin of Safety yield negative values and this demonstrates the ability of the canister to maintain structural integrity under Normal Conditions.

Fatigue – The canister was evaluated for fatigue effects due to thermal and mechanical cycling in accordance with the ASME Code, Section III, Subsections NB and NG. As specified by Subsection NB 3222.4, if the following conditions are evaluated and the criteria established by them met, then it can be concluded that the peak stress intensity limits due to fatigue are satisfied.

Condition 1: Atmospheric to Service Pressure
Condition 2: Normal Service Pressure Fluctuation
Condition 3: Temperature Difference: Startup and Shutdown
Condition 4: Temperature Difference: Normal and Off-Normal
Condition 5: Temperature Difference between Dissimilar Materials
Condition 6: Mechanical Loads

The applicant demonstrated by calculation that the six conditions could be satisfactorily met, therefore peak stress intensities due to fatigue are below allowable limits.

Canister Pressure Test – The canister was evaluated for an internal pressure, which was 125% of design internal pressure in accordance with the ASME Code, Section III, NB-6000. The applicant demonstrated positive Margins of Safety for the primary membrane stress intensity as well as the primary membrane plus primary bending stress intensities; therefore the allowable stress criteria are satisfied.

3.4.4.2 MPC-LACBWR Fuel Basket Analysis

Fuel Basket Support Disks – A component FE model of a single fuel basket support disk was evaluated for combined loading due to dead weight, handling loads and bounding thermal stresses. The resultant stress intensities and associated Margins of Safety demonstrate that support disk function is not compromised during Normal Conditions.

Fuel Basket Weldments – Two component FE models of the top and bottom weldments were evaluated for combined loading due to dead weight, handling loads and bounding thermal stresses. The resultant stress intensities and associated Margins of Safety demonstrate that the structural integrity of the weldments is not compromised during Normal Conditions.

Fuel Tubes – A classical hand calculation evaluating handling and dead weight demonstrates that the structural integrity of the fuel tube under Normal conditions is not compromised.

Damaged Fuel Can (DFC) - A classical hand calculation was presented to evaluate the Margin of Safety of the Damaged Fuel Can and lid. The applicant assumed a conservative compressive load of 1000 pounds (> 20g), which resulted in Margins of Safety of greater than 10 for both the DFC tube and lid.

3.4.4.3 MPC-LACBWR Vertical Concrete Cask Evaluation

The applicant thoroughly demonstrated that this evaluation was bounded by the results of Yankee-MPC.

3.5 Fuel Rods

Fuel rod cladding under normal conditions is only affected by temperature for both long and short-term timeframes. The thermal evaluation demonstrated that peak cladding temperatures remain below acceptance criteria limits, therefore the structural integrity of the cladding is not compromised under normal conditions.

3.6 Canister Closure Weld Evaluation

The applicant used classical hand calculations to demonstrate that Margins of Safety were adequate for the controlling load combination of thermal, internal pressure, dead weight, and handling loads. The applicant compared stress intensities derived from the structural analysis with allowable stresses including maximum primary membrane, primary membrane plus bending, and primary bending plus secondary stresses and included a 0.8 reduction factor for those allowable stresses.

3.7 Evaluation Findings

Based on the applicant's structural evaluation for this amendment and the review of the statements and methodologies employed in the application, the staff finds that the package has been adequately described and that the proposed additions and changes will not affect the ability of the package to meet the regulatory requirements of 10 CFR Part 72.

F3.1 The SAR adequately describes all SSCs that are important to safety, providing drawings and text in sufficient detail to allow evaluation of their structural effectiveness.

F3.2 The applicant has met the requirements of 10 CFR Part 72.236(b). The SSCs important to safety are designed to accommodate the combined loads of normal or off-normal operating conditions and accidents or natural phenomena events with an adequate margin of safety. Stresses at various locations of the cask for various design loads are determined by analysis. Total stresses for the combined loads of normal, off-normal, accident, and natural phenomena events are acceptable and are found to be within limits of applicable codes, standards, and specifications.

F3.3 The applicant has met the requirements of 10 CFR Part 72.236(c), for maintaining subcritical conditions. The structural design and fabrication of the DSS includes structural margins of safety for those SSCs important to nuclear criticality safety. The applicant has demonstrated adequate structural safety for the handling, packaging, transfer, and storage under normal, off-normal, and accident conditions.

F3.4 The applicant has met the requirements of 10 CFR 72.236(l), "Specific Requirements for Spent Fuel Storage Cask Approval." The design analysis and submitted bases for evaluation acceptably demonstrate that the cask and other systems important to safety

will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.5 The applicant has met the requirements of 10 CFR 72.236 with regard to inclusion of the following provisions in the structural design:

- Design, Fabrication, Erection, and Testing to Acceptable Quality Standards.
- Adequate Structural Protection Against Environmental Conditions and Natural Phenomena, Fires, and Explosions.
- Appropriate Inspection, Maintenance, and Testing.
- Adequate Accessibility in Emergencies.
- A Confinement Barrier that Acceptably Protects the Cladding during Storage.

4.0 THERMAL

The thermal review verifies that the cask and fuel material temperatures of the MPC-LACBWR system will remain within the allowable values or criteria for normal, off-normal, and accident conditions. The objective of the thermal evaluation includes confirmation that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation which could lead to gross rupture. This portion of the review also confirms that the thermal design of the cask has been evaluated using acceptable analytical and/or testing methods.

4.1 Thermal Load Specification

The design basis heat load for the MPC-LACBWR spent fuel is 4.5 kW for the entire 68 fuel assembly canister with a maximum per assembly heat load limit of 0.0662 kW. This heat load is very low compared to the two previously licensed NAC-MPC configurations (i.e., Yankee-MPC with 36 assemblies and the Connecticut Yankee (CY) -MPC with 26 assemblies). The previously licensed canister heat loads for the Yankee-MPC and the CY-MPC, were 12.5 kW and 17.5 kW, respectively. The previously licensed maximum heat load per assembly for the Yankee-MPC and the CY-MPC, were 0.347 kW and 0.674 kW, respectively, with a reconfigured fuel assembly heat load limit of 0.102 kW/assembly. Since the previously licensed MPC configurations for the Yankee and CY class fuels have significantly higher thermal loadings than the MPC-LACBWR, and the configurations are similar, it is reasonable to conclude that the thermal loading of the MPC-LACBWR is also acceptable, as well as the component temperatures.

4.2 Model Configuration

In the specific thermal calculations performed for the MPC-LACBWR, the applicant conservatively ignored convection and conduction heat transfer from the canister and relied solely on radiation heat transfer from the canister to the concrete cask to maintain component temperatures below their limits. In the model for the MPC-LACBWR the applicant relied upon an axial power distribution shown in Figure 4.A.3-1 of the SAR, which was used to distribute the heat load over the active fuel length of 83 inches. However, the applicant initially failed to consider the possibility of the potential compaction of the fuel in the damaged fuel cans and was requested to evaluate the thermal effect of all 32 damaged fuel cans containing fuel that had been compacted to 50% of its original length. Compaction of the damaged fuel needs to be evaluated because damaged fuel cannot be quantitatively evaluated for its structural integrity due to the uncertainty of the condition of the cladding. After evaluating the effect of compaction of the fuel and conservatively applying it to the axial mid-plane of the basket, the applicant identified that it raised the normal condition cladding temperature only 6°F to 449°F which remains well below the previously approved fuel cladding allowable temperature of 806°F.

4.3 Pressure Analysis

The staff reviewed the internal pressure calculation for normal conditions and found it to be in compliance with the guidance of standard review plan, NUREG-1536. The maximum normal operating condition internal canister pressure was determined to be 8.7 psig based on the ideal gas law with a gas temperature of 370°F, 30% of fission gas escaping from 1% of failed rods, and 182.3 moles of gas into a volume of 4371 liters. However, when verifying the maximum canister pressure under accident conditions (Section 11.A.2.1.1 of the SAR) the staff determined that the pressure should be 30 psig rather than the 20 psig used in Section 11.A.2.1.2 of the SAR “MPC-LACBWR Canister Maximum Stress Due to Internal Pressure”. The increase is due to 100% of the spent fuel tubes failing with 30% of the fission gas escaping

into the canister and the accident temperature increasing to 680°F. In response to the staff's request for additional information, the applicant reviewed the accident pressure calculation and corrected the accident temperature from 680°F to 400°F, as well as, increased the fission gas released from 30% to 70%. This resulted in changing the maximum pressure for the canister from 20 psig to 32.1 psig and the structural margin of safety was reduced from 0.47 to 0.05.

4.4 Evaluation Findings

The staff finds that the thermal design of the MPC-LACBWR is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the MPC-LACBWR will allow safe storage of spent fuel for a licensed (certified) life of 20 years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

F4.1 Structures, systems, and components (SSCs) important to safety are described in sufficient detail in Sections 4.4 and 4.5 of the SAR to enable an evaluation of their thermal effectiveness. Cask SSCs important to safety remain within their operating temperature ranges.

F4.2 The MPC-LACBWR is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.

F4.3 The spent fuel cladding is protected against degradation leading to gross ruptures by maintaining the cladding temperature below its normal (400 °C) and accident (570 °C) temperature limits in a helium environment. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

5.0 SHIELDING

The staff's evaluation of the shielding aspects of the NAC-MPC system with the addition of the LACBWR spent fuel assemblies is discussed in this section of the safety evaluation report.

5.1 Shielding Design Description

5.1.1 Design Criteria

The staff reviewed the principal design criteria for the MPC-LACBWR system located in Table 2.A-1 and in Section 2.A.3.5.2 in Appendix 2-A of the SAR. The applicant established maximum dose rates for the contact dose rate of cask side wall and top lid, the transfer cask side wall, the top of the canister structure, as well as the cask air inlet/outlet and the whole body dose of the owner controlled area.

The staff verified that applicant states that the system was designed with the As Low As Reasonably Achievable (ALARA) philosophy to minimize operator radiological exposure. This is stated in Section 2.A.3.5 of the SAR.

5.1.2 Design Features

The staff reviewed the general description of the cask in Chapter 1 of the SAR. The staff determined that all figures, drawings, tables describing the shielding features are sufficiently detailed to support an in-depth evaluation.

5.2 Radiation Source Definition

The staff examined the description of the design basis fuel in Chapter 2 of the SAR to determine that the applicant used a bounding source term. In addition to previously approved contents, the applicant is adding Allis Chalmers and Exxon fuel to the NAC-MPC. The fuel assemblies are 10x10 with steel clad. The minimum enrichment, maximum burnup and cooling time are as follows:

Fuel Type	Minimum Enrichment [wt % U-235]	Maximum Burnup [MWD/MTU]	Minimum Cooling Time [yr]
Allis Chalmers	3.6	22,000	28
Exxon Nuclear Co.	3.6	21,000	23

The applicant models the central 36 fuel assemblies as Exxon Nuclear Company assemblies, and the outer 32 fuel assemblies as Allis Chalmers fuel.

The applicant performed an analysis where there is a single Exxon Nuclear Company assembly placed in the peripheral locations (B and C locations in Figure B2-3 of the Technical Specifications) and found there is a slight increase (about 1%) in the dose at the radial location of the assembly. This value is within the uncertainty of the calculation. Therefore, the staff finds that having up to 4 damaged or undamaged Exxon Nuclear Company assemblies in the peripheral locations, as allowed by the TS, would give a minimal increase in gamma dose rates and is acceptable.

Damaged fuel canisters are allowed in the 32 peripheral locations. The applicant considered two damaged fuel configurations. In the first configuration, the applicant assumes that the damaged fuel collects over the active fuel length of the fuel assembly. They increase the mass

of UO_2 to fill the fuel rod interstitial volume with UO_2. The source term is also increased with this increase in mass. Using this model, the applicant calculates lower dose rates due to the increase in self shielding from the 32 peripheral damaged fuel assemblies.

In the second configuration, the applicant assumes the damaged fuel migrated from the active fuel and settles into the lower end fitting of the assembly and fills all of the void space with UO_2. The applicant states that the storage cask inlet and transfer cask bottom surface dose rates increase due to the addition of the damaged fuel. The staff finds that the applicant has appropriately considered the damaged fuel configurations.

The staff examined the MPC-LACBWR loading pattern shown in Figure B2-3 in the Technical Specifications and finds that the loading pattern chosen by the applicant for the shielding analysis is acceptable.

5.2.1 Computer Codes for Radiation Source Definition

The applicant determined the source term for the fuel and fuel assembly hardware using SAS2H as part of the SCALE 4.3 code package. The staff finds that the SCALE 4.3 code package is acceptable for use in this application.

The applicant is using a 27-group library composed primarily of ENDF/B-IV cross sections with ENDF/B-V data for a large number of fission product isotopes. The staff finds that this cross section set is appropriate.

5.2.2 Gamma Source

The staff verified that the applicant specified the gamma source term as a function of energy for both the fuel and the hardware. These values are listed for a single assembly in Table 5.A.2-4 and Table 5.A.2-5 for the fuel and hardware, respectively.

The staff verified that the applicant appropriately considered the Co-60 contained in the fuel assembly hardware. The applicant states that they used a 2 g/kg Co-60 impurity within the stainless steel and that this is the maximum impurity allowed per manufacturer specifications. The staff finds this value to be reasonable and acceptable since it is within manufacturer specifications as stated by the applicant.

The staff reviewed the energy group spectra of the gamma source to determine if it is appropriate. The applicant used SAS2H to determine a source term and a grouped energy spectra for the source term. This was then used in the MCNP code for the shielding calculation which uses continuous energy cross sections. Although the applicant is mixing a grouped energy source with a continuous energy shielding calculation, the staff accepts that using continuous energy cross sections is always more accurate and finds this combination acceptable.

5.2.3 Neutron Source

The staff verified that the applicant specified the neutron source as a function of energy. This is listed in Table 5.A.2-3 for a single assembly. The applicant used the MCNP code for the shielding calculation and the sub-critical multiplication was not accounted for within the code. The applicant used a scaled factor based on the system multiplication factor (as discussed in the Criticality evaluation, Chapter 6 of the SAR) to account for this effect. This method is discussed in Section 5.A.2.2 of the SAR and is based on a multiplication factor of 0.4 for a dry cask and 0.95 for a flooded cask. This results in the neutron source being scaled up by 1.67 for

a dry cask and 20 for a flooded cask. This approach has been used for the NAC-MPC for previously approved contents and for other approved applications (NAC MAGNASTOR Dry Cask Storage System, Docket No. 72-1031). The staff finds this acceptable for the MPC-LACBWR application.

5.3 Shielding Model Specification

The applicant modeled the transfer and the concrete casks using the MCNP code. The applicant evaluated the shielding of the cask under normal and accident conditions.

5.3.1 Configuration of the Shielding and Source

The staff examined the sketches of the shielding model as represented by Figures 5.A.3-1 through 5.A.3-5. The staff verified that the dimensions were consistent with the cask drawings presented in Section 1.A.7.

The applicant modeled the Allis Chalmers fuel in the 32 peripheral basket locations and the Exxon Nuclear Company fuel in the 36 interior basket locations. The fuel and hardware are homogenized and placed into regions defined by the fuel assembly width and height which is subdivided axially into source regions for the active fuel, upper plenum, upper and lower end fittings.

Homogenization of the fuel assemblies may not be appropriate when the cask is flooded and there are significant amounts of moderating materials present. Homogenization may cause the neutron dose to be under-represented. This is true for low enriched fuel, which means there is a high amount of U-238. U-238 has a strong resonance absorption peak just above thermal energies. In a heterogeneous system the neutrons can fully moderate within the water between the fuel rods before reaching the U-238 and avoid the resonance absorption. Homogenization of the fuel with the moderator allows for more neutrons to be absorbed by the U-238 resonance and can under-represent the neutron dose.

The applicant performed an analysis to justify the use of homogenized fuel for certain shielding calculations. The applicant performed a calculation of the flooded cask configuration to include the detailed (heterogeneous) basket and fuel. For this calculation the applicant used the sub-critical multiplication within the MCNP code rather than the scaled factor (as discussed in Section 5.2.3 of this SER). The results of the applicant's calculations show that the heterogeneous fuel rod array in conjunction with the MCNP calculated sub-critical multiplication produces a lower dose rate than the homogeneous fuel rod array and the scaled factor to account for sub-critical multiplication. This demonstrates that the applicant's method is conservative and that the assumption of homogeneous fuel assemblies in conjunction with the scaled factor for sub-critical neutron multiplication is acceptable for use in this application.

The applicant created an axial profile based on measured burnup profile data. The staff viewed the axial burnup profiles as well as the axial gamma and neutron source profiles. The staff finds that they are acceptable and would provide representative results for the dose rates where there is axial peaking.

The staff verified that the applicant adequately described the locations of the dose calculations. The applicant used detector surfaces at 1 foot, 1 meter, 2 meters and 4 meters from the cask. In addition the applicant modeled streaming paths for the inlet and outlet air vents and performed separate calculations to evaluate the dose rates at these locations.

5.3.2 Material Properties

The Transfer Cask used with the MPC-LACBWR has a multi-wall radial shield comprised of carbon steel, lead, solid borated polymer (NS-4-FR), and carbon steel. There is additional radial stainless steel shielding provided by the canister shell. The bottom of the shield is low alloy steel, and the top shield consists of stainless steel. The storage cask radial shield design has carbon steel inner liner surrounded by concrete. The top shield consists of stainless steel and carbon steel. The bottom of the cask sits on a concrete pad and therefore has a relatively thin bottom that consists of stainless steel and carbon steel.

The staff verified that the SAR provides information on all of these material compositions and densities used in the calculational model and that they are reasonable. The applicant also provided information and material densities for the fuel and non-fuel hardware components. For the neutron absorber (BORAL), the staff additionally verified that Chapter 9 of the SAR provided the source of the data and also has a testing program and acceptance criteria for validating boron content.

For the neutron shield (NS-4-FR) the staff did not find that there is appropriate information in Chapter 9 of the SAR regarding their testing program and acceptance criteria for validating material uniformity and boron content. However, the transfer cask design is unchanged from that of the previously approved contents and the LACBWR fuel has a lower neutron source. In response to staff questions, the applicant indicated that they will be using the same transfer cask as used in previous loading campaigns where no dose rate anomalies had been detected due to non-uniform materials. Therefore the staff finds that even though the applicant does not provide adequate information regarding testing for the material uniformity and boron content of the neutron shield, the previous approval is acceptable for the LACBWR fuel.

The staff does not find that there are any temperature sensitive materials important to the shielding of the MPC-LACBWR that will exceed their design limitations during normal or accident conditions.

5.3.3 Accident Conditions

The accident conditions analyzed by the applicant involve a projectile impact and a loss of 6 inches of concrete. The applicant determined that the dose rate increases.

The applicant did perform an analysis of the tip-over event in Section 11 of the SAR and determined that the dose rates determined for previously approved payloads bound that of the MPC-LACBWR. However, high dose rates are expected at the cask bottom and that following this event supplemental shielding would be needed until the cask can be returned to an upright position. The applicant determined that there are no design basis accidents that result in a tip-over of the MPC-LACBWR storage cask.

5.4 Shielding Analysis

5.4.1 Computer Programs

For the shielding analysis the applicant uses the MCNP5 Release 1.30 code with default neutron and photon cross sections that are from various releases of the ENDF/B-V and ENDF/B-VI libraries and the MCNPLIB04 photoatomic data set. MCNP is a widely used and recognized code for shielding analyses. The staff has previously accepted the use of MCNP for similar shielding evaluations.

5.4.2 Flux-to-Dose-Rate Conversion

The applicant states that they use the ANSI/ANS 6.1.1-1977 [1] flux-to-dose rate conversion factors in all of the cask shielding evaluations. The staff finds this acceptable.

5.4.3 Dose Rates

The applicant calculated the following dose rates for the MPC-LACBWR assuming undamaged fuel:

Storage Cask

Location	Source	Surface Dose [mrem/hr]	1 meter [mrem/hr]
Top Axial	Neutron	0.5	0.1
	Gamma	18.2	5.5
	Total	18.7	5.6
Side	Neutron	0.2	0.1
(Normal)	Gamma	28.7	11.4
	Total	28.9	11.5
Side	Neutron	0.7	0.2
(Accident)	Gamma	277.1	104.5
	Total	277.8	104.7

Storage Cask – Inlet and Outlet Maximum Dose Rates

Source	Inlet Average [mrem/hr]	Outlet Average [mrem/hr]
Fuel Neutron	1.3	0.1
Fuel Gamma	6.8	0.1
Fuel Hardware	3.2	0.1
Lower End Fitting	27.0	--
Upper Plenum	--	1.2
Upper End Fitting	--	0.5
Total	38.3	2.0

Transfer Cask – Wet Conditions, Port Covers Off

Location	Source	Surface Dose [mrem/hr]	1 meter [mrem/hr]
Top Axial	Neutron	0.2	0.1
	Gamma	471.0	205.7
	Total	471.2	205.8
Side	Neutron	22.0	4.7
(Normal)	Gamma	46.2	16.7
	Total	68.2	21.4
Side	Neutron	0.1	0.1
(Accident)	Gamma	23.7	10.0
	Total	23.8	10.1

Transfer Cask – Dry Conditions, Port Covers On

Location	Source	Surface Dose [mrem/hr]	1 meter [mrem/hr]
Top Axial	Neutron	7.9	2.4
	Gamma	590.8	251.7
	Total	598.7	254.1
Side	Neutron	19.6	4.4
(Normal)	Gamma	82.6	31.1
	Total	102.2	35.5
Side	Neutron	17.6	4.9
(Accident)	Gamma	36.6	15.4
	Total	54.2	20.3

For damaged fuel that collects at the bottom of the fuel canister and into the lower end fitting of the fuel assembly, the applicant calculates higher Inlet dose rates and higher dose rates at the bottom of the transfer cask. The results of the damaged fuel calculations are as follows:

Inlet dose Rates – Damaged Fuel

Source	Inlet Average [mrem/hr]
Undamaged Fuel	38.3
Damaged Neutron	2.3
Damaged Gamma	34.4
Total	75.0

Transfer Cask Bottom – Damaged Fuel, Dry Conditions, Port covers On

Source	Surface dose [mrem/hr]
Undamaged Fuel	45.1
Damaged Neutron	28.5
Damaged Gamma	2.6
Total	76.2

The staff reviewed the technical specifications and LCO 3.2.2 requires that the MPC-LACBWR adhere to specific dose rates at the side, top and air inlet and outlet (average) for the concrete cask. These dose rates are as follows:

Technical Specification Dose Rates

Location	TS Dose Rate, [mrem/hr] (neutron + gamma)
Side (on the concrete surfaces)	20
Top	25
Average of the measurements at the air inlet and outlet	100

Based on a comparison to similar cask systems, the staff finds that the calculated dose rates are reasonably low and acceptable. The staff finds that the dose rate limits in the technical specifications are also low and reasonable. The staff has discontinued its practice of allowing averaged dose rate limits within the TS because the averaging process may negate any loading errors in which the TS measurements are supposed to detect. However the staff is making an exception for this license amendment. For this particular cask system, there are other allowed fuel types that have averaged dose rate limits in the technical specifications and since the LACBWR fuel is not a modern fuel type, it has a lower source term due to its relatively low

burnup and longer cooling time. The staff finds that that the TS limits would provide an additional layer of assurance should there be any errors in the calculated dose rates.

5.4.4 Confirmatory Calculations

The staff performed confirmatory calculations to verify the applicant's source term. The results of the staff's calculations give the staff reasonable assurance that the source term was calculated conservatively. The staff focused on the source term because the cask and shielding had previously been reviewed and approved by the NRC staff using a higher activity source.

The transfer cask for the previously analyzed Yankee-MPC system and the proposed La Crosse BWR system (MPC-LACBWR) are nearly identical. The canister shell for the MPC-LACBWR is 0.125 inch thinner than the Yankee-MPC. While the radial carbon steel liner for the MPC-LACBWR is one inch thinner, the concrete over pack is one inch thicker. Staff initially performed calculations using MicroShield 5.05 to provide a rough estimate of the effect, given a fixed source term, the change in shield material dimensions would have on surface dose rate.

Staff conducted confirmatory source-term analyses using SAS2H with ENDF-V cross-section libraries in the SCALE 5.1 package. Inlet and outlet moderator density were analyzed to capture the bounding flux spectra expected within the assembly during operation. Power, cycle time and down time were calculated as presented by the applicant. ORIGEN-S was used to determine the combined source term at the minimum cooling time for each LACBWR assembly type. Staff also re-ran the bounding Combustion Engineering 16x16 Type A Connecticut Yankee assembly source-term calculation for comparison.

It was not possible to conduct a direct confirmation. The applicant uses a version of SAS2H with a cross-section library unavailable to the staff. The changes required to the input and to conform to the newer versions would significantly change the problem being investigated. The staff's own conclusions were compared separately, and an identical comparison was made using the provided output files. The trend among each group of analyses with a single variable change is the same for both the Allis Chalmers and Exxon fuel assemblies.

Comparing the staff's source term calculation for the CE Type A fuel with the staff's analysis of the LACBWR fuel indicate that the applicant's analysis has resulted in a reasonable estimate of cask dose rates.

5.5 Evaluation Findings

Based on the above statements, the staff finds that:

F5.1 Sections 1 and 5 of the SAR describe shielding structures, systems, and components (SSCs) important to safety in sufficient detail to allow evaluation of their effectiveness. The following drawings in Section 2 of the SAR define SSCs for shielding: 630045-861, 630045-862, 630045-863, 630045-870, 630045-871, 630045-872, and 630045-878. The following drawing No. 455-859 and 455-860 show the Transfer Adapter and Transfer Cask License Drawings are included in Section 1.7.1 of the NAC-MPC FSAR.

F5.2 Sections 5 and 10 of the SAR demonstrate the radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104 and 10 CFR 72.106.

F5.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The MPC-LACBWR shielding features are designed to assist in meeting these requirements.

The staff finds that the design of the shielding system of the MPC-LACBWR is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system design provides reasonable assurance that the MPC-LACBWR will allow safe storage of spent fuel in accordance with 10 CFR 72.236(d). This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

6.0 CRITICALITY

The staff's evaluation of the criticality of the NAC-MPC system with the addition of the LACBWR spent fuel assemblies is discussed in this section of the safety evaluation report. In reviewing this application, the staff followed the guidance in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."

6.1 Criticality Design Criteria and Features

The staff viewed information in Section 1, 2 and 6 of the SAR and verified that the information is consistent as well as all descriptions, drawings, figures and tables are sufficiently detailed to support an in-depth staff evaluation.

The staff reviewed the principle design criteria for criticality as presented in Section 2.A.3.4 of the SAR. The applicant uses neutron absorber sheets (BORAL) to ensure that the cask remains subcritical under normal, off-normal, and accident conditions.

The spent fuel pool does not contain borated water, therefore the applicant does not use borated water as a means of criticality control during loading and unloading operations of the cask.

6.2 Fuel Specification

The applicant proposes to add two new fuel types as part of this amendment request. These are Allis Chalmers 10x10 fuel with a maximum enrichment of either 3.64% or 3.94%, and Exxon Nuclear Company 10x10 fuel with a maximum enrichment of 3.71%. Both fuel types are stainless steel clad. Table 2.A.1-1 of the SAR lists the nominal design parameters for each fuel type.

The applicant evaluated each of the new fuel types individually and determined that the 3.94% enriched Allis Chalmers fuel has the highest reactivity. In addition the applicant evaluated the effect of assuming a planar average enrichment of the Exxon type fuel rather than the varied radial enrichments. The applicant determined that using the planar averaged enrichment for the Exxon type fuel is statistically equivalent to having discrete radial enrichments.

For the maximum reactivity calculation results shown in the table on Page 6.A.1-2, the applicant states that Exxon fuel is located in the 36 interior locations and damaged Allis Chalmers fuel is located in the exterior 32 locations. Based on the results of the reactivity calculations for the individual fuel types and the allowed loading pattern in Figure B2-3 in the Technical Specifications, the staff finds that the analyzed loading pattern is acceptable and that it bounds all other possible loading patterns.

The applicant assumes 96% theoretical density. The staff finds that this is conservative and acceptable because it bounds LACBWR fuel assembly material mass.

The applicant does not take credit for burn-up. All assemblies are assumed to be fresh fuel. The staff finds this conservative and acceptable.

The staff viewed the information in Section 1 and 6 of the SAR and verified that the description of the fuel used in the criticality analysis bounds that of the allowable fuel contents.

6.2.1 Non-fuel Hardware

There is no non-fuel hardware permitted for storage in the MPC-LACBWR system. However, fuel assembly channels were removed from the spent fuel assemblies prior to dry fuel storage. These are made of zirconium alloy and were compacted in the spent fuel pool. Compaction debris is present within the fuel assemblies. The staff finds that the presence of the zirconium alloy debris would not negatively impact the criticality analysis and finds its presence within the assemblies acceptable.

6.2.2 Fuel Condition

The fuel contains a significant amount of Allis Chalmers fuel classified as damaged due to concerns on clad stability. The MPC-LACBWR system allows up to 32 damaged fuel canisters (DFCs) positioned in the periphery locations. The applicant evaluated cask loadings considering the damaged fuel assemblies. The applicant evaluated configurations where (1) fuel rods were removed from the assembly, (2) the rods have no cladding in conjunction with having missing rods and varied pitch, and (3) a homogenous fuel water mixture to simulate fuel that has lost its geometry. The applicant also determined the optimum moderator density. The applicant found that maximum system reactivity was achieved when the DFCs contain fuel that has all cladding and any other assembly hardware removed, and the pellets are floating in a fully flooded DFC with the maximum allowed square pitch. In response to staff questions, the applicant provided more information on analyses performed for the damaged fuel. The staff finds that the damaged fuel has been conservatively analyzed and is acceptable.

6.3 Model Specification

The applicant states that they performed sensitivity studies evaluating the effects of fabrication tolerances. The applicant states that the combined tolerance model significantly increases system reactivity and produces the maximum reactivity configuration. The applicant discusses the maximum reactivity configuration in Section 6.A.4.3.1. The staff finds that this has been adequately addressed by the applicant and that their maximum reactivity model is acceptable.

6.3.1 Configuration

The staff confirmed that the model used in the criticality evaluation is adequately described for normal conditions and finds that the model for normal conditions bounds that for off-normal and accident conditions.

The staff viewed the sketches of the model used for the criticality calculations. The staff verified that the dimensions and materials of the model are consistent with the engineering drawings.

The applicant states that they performed various moderator density studies. They show the results of varying the moderator density and concluded that full density moderator is the most reactive for the undamaged fuel configurations. For the damaged fuel configuration the applicant varied moderator density within the transportable storage canister (TSC) and determined that the most reactive condition occurs when the TSC is dry and the DFC's are flooded at maximum water density. The applicant also performed partial drain down studies to show that when steel from the lid is acting as a reflector rather than the water above the top of the fuel, the steel reflector does not produce a significant change in reactivity.

The staff verified that the applicant performed the criticality evaluations using a heterogeneous model of the fuel. The applicant did perform studies where the fuel was homogenized, however this was only a sensitivity study for determining maximum reactivity of the damaged fuel

assemblies. The staff finds that the heterogeneous treatment of the fuel is appropriate and acceptable.

6.3.2 Material Properties

The staff verified that compositions and densities are provided for all materials used in the computational model. This information is provided in Tables 6.A.3-8 and 6.A.3-9. The staff viewed this information and finds that the material compositions and densities are reasonable.

The staff viewed the information in Chapter 9 which shows that that the applicant has tests to confirm the areal density of the neutron absorber panels (BORAL). Table 6.A.3-2 as well as Drawing 630045-881 state that the areal density of Boron-10 will be confirmed as 0.02 g/cm^2. The staff finds this value conservative and acceptable.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant performs the criticality evaluations using the MCNP5 three-dimensional Monte Carlo code and continuous energy cross sections. The MCNP5 code is widely used in these types of applications and the staff finds it is appropriate for this application.

Review of the representative MCNP input file from the SAR shows that the applicant is using cross section data from various revisions of the ENDF/B-VI library with the exception of tin, where ENDF data is not available and so ENDL data is used. The applicant states that these are the same cross sections used to perform the validation of the code. The staff finds the cross sections used are appropriate for use with the MPC-LACBWR application.

6.4.2 Multiplication Factor

The applicant calculated the k_{eff} for the transfer and storage casks for both normal and accident conditions. The values of the maximum k_{eff} for the MPC-LACBWR systems are calculated by the applicant. The results are as follows:

Cask Body	$k_{eff} + 2\sigma$
Transfer	0.93014
Storage	0.34222

The applicant performed various sensitivity and parametric analyses to determine the conditions that give the maximum reactivity. The staff finds that the results provided above represent a reasonably bounding configuration.

The staff reviewed some of the results in the representative output files submitted by the applicant and determined that they used an appropriate number of neutron histories to achieve good statistics for the value of k_{eff}.

The applicant's k_{eff} values are below the applicant's upper subcriticality limit (USL) of 0.9372 and therefore the staff finds them acceptable.

6.4.3 Benchmark Comparisons

The applicant performed benchmark comparisons and determined a USL based on the guidance published in NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor

Fuel in Transportation and Storage Packages." The staff finds the use of this guidance acceptable.

In response to staff questions, the applicant stated that they performed benchmarks with the same computer code and cross section set that were used in the licensing calculations.

The staff verified that the following important design parameters for the MPC-LACBWR system were within the benchmark experiments cited by the applicant.

- Enrichment
- Type of fissile material
- Fuel rod pitch and diameter
- B-10 plate loading
- EALF
- H/U-235 ratio

The fuel pellet outer diameter range of the benchmarks is larger than that of the Exxon fuel. In addition the applicant does not have any benchmark comparisons for stainless steel clad fuel. The staff does not find these differences significant and that any potential negative effects would be compensated for by conservative assumptions within their analysis (i.e. fresh fuel assumption, etc.).

The applicant calculated a USL of 0.9376 using the USLSTATS code. This includes the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any k_{eff} less than the USL is less than 0.95. The staff finds this acceptable.

6.5 Independent Staff Calculations

The staff performed independent calculations to verify the k_{eff} of the MPC-LACBWR. The staff constructed its model using design information found in the SAR. The staff used the KENO6 code with the 238-group cross section library derived from ENDF-VI data.

The staff used the fuel loading pattern as described in Figure 6.A.1-1. In Slot A the staff assumed undamaged Exxon Fuel assemblies. In Slot B the staff assumed damaged Allis Chalmers Fuel assemblies with an enrichment of 3.64%. In Slot C, the staff assumed Damaged Allis Chalmers fuel assemblies with a maximum enrichment of 3.94%.

The damaged fuel assemblies were modeled as a bare array (no cladding material) of UO_2 rods. Based on the sensitivity studies from the applicant, the staff assumed the following for the damaged fuel assemblies:

- Heterogeneous array of rods (page 6.A.4-7 of the SAR states that the heterogenous array gives a higher reactivity than a homogeneous mixture)
- Pitch of 0.6 inches (Table 6.A.4-9 of the SAR shows reactivity is increased with increased pitch)
- All rods present within the damaged fuel rod assemblies (Table 6.A.4-9 of the SAR shows that reactivity increased with one missing rod. The increase was not substantial so for simplicity, the staff assumed all rods were present)

The staff's model was based on the transfer cask since this cask would see flooding and provides a more reactive condition. The positioning of the assemblies of the staff's model was

based on Figure 6.A.3-2. The staff assumes void within the empty spaces within the undamaged (Exxon) fuel assemblies, and basket, including the gap between the fuel rod and the cladding. The empty spaces within the damaged (Allis Chalmers) fuel assembly locations are assumed to be flooded with full density water. The staff performed an additional calculation where all empty spaces were fully flooded with full density water and found that the partial flooding condition was more reactive.

The staff used several simplifying assumptions similar to that of the applicant. The staff assumed no structural materials within the basket besides the fuel tubes. The staff assumed that all assembly hardware could be modeled as a top and bottom cap of equivalent stainless steel. The staff assumed no aluminum or additional absorber sheets associated with the enlarged and damaged fuel cans.

The k_{eff} from the staff's calculation generally agrees with that of the applicant's. The staff finds that this helps to demonstrate that the features important to criticality are sufficiently described and that the applicant has addressed the most reactive conditions and that the reported k_{eff} is conservative and that the applicant has appropriately modeled the cask geometry and materials.

6.6 Evaluation Findings

Based on the above statements, the staff has the following evaluation findings with respect to the criticality analysis:

F6.1 Structures, systems, and components important to criticality safety are described in sufficient detail in Chapters 1, 2 and 6 of the SAR to enable an evaluation of their effectiveness.

F6.2 The cask and its spent fuel transfer systems are designed to be subcritical under all credible conditions.

F6.3 The criticality design is based on favorable geometry, and fixed neutron poisons. An appraisal of the fixed neutron poisons has shown that they will remain effective for the term requested in the CoC application and there is no credible way for the fixed neutron poisons to significantly degrade during the requested term in the CoC application; therefore, there is no need to provide a positive means to verify their continued efficacy as required by 10 CFR 72.124(b).

F6.4 The analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the storage of spent fuel for the term requested in the CoC application.

The staff finds that the criticality design features for the MPC-LACBWR are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the MPC-LACBWR will allow safe storage of spent fuel. These findings are reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7.0 CONFINEMENT

The objective of the confinement review is to ensures that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

7.1 Confinement Design Characteristics - Design Features

The confinement boundary for the LACBWR storage system consists of a welded stainless steel transportable storage canister (TSC). The TSC is designed to preclude release of radioactive material for all design basis conditions, including preventing failure from maximum internal pressure.

The TSC is a welded ductile stainless steel canister, composed of a ½-inch thick cylindrical shell, a 1 ¼-inch thick bottom plate, and a 7-inch thick closure lid. The closure lid meets the redundant sealing requirements of 10 CFR 72.236(e) by having a welded closure ring (¾-inch thick) behind the lid-to-shell weld, and the vent and drain port covers have dual welded plates (each ½-inch thick).

All welds except for the closure lid assembly are full penetration and volumetrically examined. The closure lid-to-shell weld is a partial penetration weld and is liquid penetrant examined on its root, mid-plane, and final surfaces. The closure ring is attached to the lid and shell via two partial penetration welds that have their final surface liquid penetrant examined. The port cover plates are beveled seal welds and are liquid penetrant examined on their final surface.

Other testing is performed on the confinement boundary to ensure its integrity. During manufacture, the canister assembly (i.e. shell and bottom plate) is leak tested to the ANSI 14.5-1997 leak tight criterion of 10^{-7} ref cm^3/sec. Subsequent to making the closure lid-to-shell weld, but prior to installing the closure ring, a pressure test is performed to a minimum pressure of 15 psig, in accordance with Subsection NB of the ASME Code. The Code acceptance criterion for the field pressure test is “no visible leakage” and that leakage examination is made after the 10 minute pressure test duration and at a pressure equal to the design pressure or 3/4ths of the test pressure. Potential leakage is readily visible for the closure weld, but is not for the shell welds since the TSC is located within the transfer cask when the pressure test is performed. However, the staff accepts the previously performed shop helium leakage test of the canister assembly with a sensitivity of $1x10^{-7}$ ref cm^3/s (as specified in Sections 9.A.2.4 and 2.A.3.2.1 of the SAR) as reasonable assurance that the shell is free from flaws that would cause any leakage. The applicant changed their pressure test acceptance criteria from no loss in pressure to be in agreement with the standard ASME Code practice as described above.

In accordance with ISG-18, no leak test is to be performed on the closure lid to shell weld because it is a multiple pass weld of ductile stainless steel material made in accordance with the guidance in ISG-15 and it is not pressurized at the time of welding. However, this exception from leak testing does not apply to the welds for the port cover plates that could potentially be under pressure. Therefore, the inner port cover plates are leak tested to the leak tight criteria.

7.2 Evaluation Findings

The staff finds that the design of the confinement system of the MPC-LACBWR is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the MPC-

LACBWR will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant's analysis and the staff's confirmatory analysis, and accepted engineering practices.

F7.1 Section 7.1 of the SAR describe(s) confinement structures, systems, and components (SSCs) important to safety in sufficient detail in to permit evaluation of their effectiveness.

F7.2 The design of the MPC-LACBWR adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the safety evaluation report (SER) discusses the relevant temperature considerations.

F7.3 The design of the MPC-LACBWR provides redundant sealing of the confinement system closure joints by having a welded closure ring (¾-inch thick) behind the lid-to-shell weld, and the vent and drain port covers have dual welded plates (each ½-inch thick).

F7.4 The cask confinement system has been helium leak tested (except as permitted by ISG-18 for the closure weld) to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

8.0 OPERATING PROCEDURES

The applicant revised this section to incorporate changes related to the following activities for the LACBWR fuel and to incorporate lessons-learned through interactions with potential users:

- Loading the MPC-LACBWR storage system;
  - Loading and closing the TSC;
  - Transferring the TSC to the concrete cask;
  - Transporting and placing the load concrete cask;
- Removal of the loaded MPC-LACBWR TSC from the VCC; and
- Wet unloading the MPC-LACBWR TSC.

The staff concludes that the revisions to the operating procedures as presented in Chapter 8 of the SAR are acceptable, and continue to meet the requirements of 10 CFR Part 72.

8.1 Evaluation Findings

The staff finds that the generic procedures and guidance for the operation of the MPC-LACBWR are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

F8.1 The MPC-LACBWR is compatible with wet loading and unloading. General procedure descriptions for these operations are summarized in Sections 8.A.1 and 8.A.3 of the applicant's safety analysis report (SAR). Detailed procedures will need to be developed and evaluated on a site specific basis.

F8.2 The smooth surface of the cask is designed to facilitate decontamination. Only routine decontamination will be necessary after the cask is removed from the spent fuel pool.

F8.3 No significant radioactive waste is generated during operations associated with the independent spent fuel storage installation (ISFSI). Contaminated water from the spent fuel pool will be governed by the 10 CFR Part 50 license conditions.

F8.4 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading will be governed by the 10 CFR Part 50 license conditions.

F8.5 The general operating procedures described in the SAR are adequate to protect health and minimize danger to life and property. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.6 Section 10 of the safety evaluation report (SER) assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The applicant revised this section to incorporate changes related to the following acceptance test activities for the LACBWR fuel and to incorporate lessons-learned through interactions with potential users:

- Visual and Nondestructive Examination Inspections
  - Nondestructive Weld Examination
  - Fabrication Inspections
- Structural and Pressure Tests
- Leak Tests
- Component Tests
  - Valves, Rupture Disks and Fluid Transport Devices
  - Gaskets
- Shielding Tests
- Neutron-Absorber Tests
  - Neutron Absorber Material Sampling Plan
  - Wet Chemistry Test Performance
  - Neutron Absorption Test Performance
  - Acceptance Criteria
- Thermal Tests
- Cask Identification

The applicant also revised this section to incorporate changes related to the maintenance program activities.

The staff concludes that the revisions to the acceptance tests and maintenance program as presented in Chapter 9 of the SAR are acceptable, and continue to meet the requirements of 10 CFR Part 72.

9.1 Evaluation Findings

The staff finds that the acceptance tests and maintenance program for the MPC-LACBWR are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the cask will allow safe storage of spent fuel throughout its licensed or certified term. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

F9.1 Sections 9.A.1 and 9.A.2 of the SAR describe(s) the applicant's proposed program for preoperational testing and initial operations of the MPC-LACBWR. Section 9.A.3 discusses the proposed maintenance program.

F9.2 Structures, systems, and components (SSCs) important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Section 2.A.3 of the SAR identifies the safety importance of SSCs, and Section 9.A.1 present(s) the applicable standards for their design, fabrication, and testing.

F9.3 The applicant/licensee will examine and/or test the MPC-LACBWR to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 7.A.1 of the SAR describe(s) this inspection and testing.

F9.4 The applicant/licensee will mark the cask with a data plate indicating its model number, unique identification number, and empty weight. Drawing 864 in SAR Section 1.A.7 illustrates and/or describes this data plate.

10.0 RADIATION PROTECTION

Using the guidance in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," the staff evaluated the radiation protection features of the NAC-MPC system with the addition of the LACBWR spent fuel assemblies. The radiation protection aspects are discussed in this section of the safety evaluation report.

10.1 Radiation Protection Design Features for the Transfer and Storage Cask

The staff verified that applicant states that the system was designed with the As Low As Reasonably Achievable (ALARA) philosophy to minimize operator radiological exposure. This is stated in Section 2.A.3.5 of the SAR.

The MPC-LACBWR cask is leak tight per ANSI 14.5, so the applicant did not evaluate any radiological exposure due to leakage of radioactive effluents. Therefore the calculated exposure is only due to direct radiation exposure. The staff finds this acceptable.

The applicant has evaluated distances at which the limits of 10 CFR 72.104(a) and 10 CFR 72.106(b) will be met due to direct radiation exposure.

10.2 Occupational Exposures

The staff viewed the Operating Procedures (Chapter 8) in the SAR as well as the direct radiation dose calculations in Chapter 5 to determine that this data was used to estimate doses received by operational personnel during cask loading and transfer to the ISFSI. The staff verified that the applicant also estimated the doses for periodic maintenance as well as surveillance activities.

The activities for which the applicant estimated the dose to the workers include the fuel assembly loading and removal of the transfer cask from the pool, the closing, welding and draining of the cask, the drying of the cavity, and placing the canister inside the storage cask and moving the cask to the ISFSI. The applicant also evaluated the dose to the workers for surveying and decontaminating the cask.

The applicant based the various exposure times on previous experience. The staff reviewed these times and finds them reasonable. Based on the dose evaluations in Chapter 5 (Shielding), and the operating procedures in Chapter 8 of the SAR, the staff finds that the personnel locations relative to the casks are reasonable.

10.3 Exposures at or Beyond the Controlled Area Boundary

The staff reviewed the SAR to ensure that the applicant demonstrated that the shielding and confinement features of the cask are sufficient to meet the requirements for real individuals in 10 CFR 72.104, and for DBA conditions in 10 CFR 72.106.

The applicant used their own Version 6.0.1 of the SKYSHINE-III code (NAC-CASC) to evaluate the placement of the controlled area boundary for the cask array. This code is benchmarked by modeling a set of Kansas State University Co-60 skyshine experiments. The code has been used in previously approved applications for the NAC-MPC.

A general licensee who intends to use the MPC-LACBWR must perform a site-specific evaluation, as required by 10 CFR 72.212(b) demonstrating compliance with 10 CFR 72.104(a). The limit of 25 mrem/year cited in 10 CFR 72.104(a) shall include all site sources. The actual

doses to individuals beyond the controlled area boundary depend on site specific conditions such as cask-array configuration, topography, demographics, and use of engineered features.

10.3.1 Normal Conditions

The calculated exposures are presented in the form of a chart calculated as dose versus distance from a single cask and a cask array (Figures 10.A.4-2 and 10.A.4-3). The applicant evaluated the minimum distance that is required to meet the dose rates in 10 CFR 72.104.

The results show that the annual dose limit of 25 mrem would be met for a single cask at approximately 125 meters from the cask side and for an array at approximately 200 meters from the cask array center. The applicant does not specify supplemental shielding at the ISFSI.

The staff verified that the applicant provided a dose rate versus distance curve for a theoretical cask array. The storage array consists of 5 casks. The applicant states that this is the maximum number of casks that will be stored.

10.3.2 Accident Conditions and Natural Phenomenon Events

The staff reviewed the accident events in Chapter 11. The applicant evaluated an event which accounts for a small release of radioactive particulate from the canister exterior (Section 11.A.1.5 of the SAR). The event was analyzed assuming that 100% of the surface area of the canister was contaminated at a level twice that of the allowance of Technical Specification (TS) 3.2.1. The analysis shows that the dose at 100 m is less than 1 mrem due to contamination. The staff finds that the analysis is conservative and that the limits in the TS are reasonable.

The applicant did perform an analysis of the tip-over event in Chapter 11 of the SAR and determined that the dose rates determined for previously approved payloads bound that of the MPC-LACBWR. However, high dose rates are expected at the cask bottom and that following this event supplemental shielding would be needed until the cask can be returned to an upright position. The applicant determined that there are no design basis accidents that result in a tip-over of the MPC-LACBWR storage cask.

The applicant estimated the dose incurred as a result of the event that all air inlets and outlets are blocked. The applicant assumes the cask is covered with earth from a catastrophic event such as a landslide and that dose is incurred by uncovering the cask. The personnel and the times estimated to perform the task appear reasonable. The staff finds that the dose estimated by the applicant is reasonable.

The applicant also estimated the dose incurred that would result from repairing the damage caused by a projectile. The staff finds the estimated dose is reasonable for this event.

The results of the structural analyses show that the transportable storage canister (TSC) is not breached in any of the evaluated events and that there is no credible leakage and therefore no release of radioactive material during off-normal or accident events of storage.

10.4 ALARA

10.4.1 Design Considerations

The staff reviewed the drawings to ensure that the features for which credit is taken have been clearly identified. This includes the shielding features for both the transfer and storage casks.

The application takes no credit for, and includes no commitments to implement additional features to show compliance with regulatory requirements or ALARA goals.

10.4.2 Procedures and Engineering Controls

The staff determined that the descriptions of the proposed cask loading and un-loading operations adequately demonstrate that ALARA principles have been incorporated into operational procedures and engineering controls.

10.5 Evaluation Findings

Based on the statements in this safety evaluation, the staff has made the following findings about the MPC-LACBWR system:

F10.1 The MPC-LACBWR system provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 72.106.

F10.2 The design and operating procedures of the MPC-LACBWR system provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR 20 and for meeting the objective of maintaining exposures ALARA.

The staff finds that the design of the radiation protection system of the MPC-LACBWR is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the MPC-LACBWR will allow safe storage of SNF. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted health physics practices.

11.0 ACCIDENT ANALYSIS

11.1 Review Objective

In this portion of the MPC-LACBWR dry cask storage system (DCSS) review, the NRC evaluated NAC's identification and analysis of hazards, as well as the summary analysis of system responses to both off-normal and accident or design-basis events. This review ensures that the applicant has conducted thorough accident analyses, as reflected by the following factors:

1. identified all credible accidents
2. provided complete information in the safety analysis report (SAR)
3. analyzed the safety performance of the cask system in each review area
4. fulfilled all applicable regulatory requirements

11.2 Off-Normal Events for the MPC-LACBWR Storage System

11.2.1 Blockage of Half of the Air Inlets

See Thermal Evaluation of this SER.

11.2.2 Canister Off-Normal Handling Load

The applicant evaluated the off normal handling load using the same methodology employed for Normal conditions. The only exception was that additional acceleration loads were applied to simulate canister misalignment movement, faulty equipment operation causing movement, or operator error causing movement. These loads were evaluated in conjunction with dead weight and internal pressure. The applicant demonstrated that positive Margins of Safety were achieved for the canister and internal basket structure.

11.2.3 Failure of Instrumentation

See Thermal Evaluation of this SER.

11.2.4 Severe Environmental Conditions (100°F and -40°F)

Not Applicable to Structural Performance per ASME Code. See Thermal Evaluation of this SER.

11.3 Accident Conditions for the MPC-LACBWR Storage System

11.3.1 Accident Pressurization

The applicant calculated the maximum pressure due to gas release inside the sealed canister. This calculated maximum pressure was the applied in conjunction with dead weight and handling loads in the previously used FE model of the canister. The applicant demonstrated that accident pressurization would result in a positive Margin of Safety.

11.3.2 Earthquake Event

The evaluation for an earthquake event is performed using the same moment-equilibrium methods used for the previously approved Yankee-MPC. The design basis load is defined as a horizontal acceleration load of 0.45g at the top surface of the ISFSI pad and 0.3g for the vertical component consistent with ASCE 4-86 [3]. The applicant demonstrates that the minimum horizontal acceleration load that is required to overturn the loaded 195,850 lbs concrete cask is 0.56g, which is higher than the 0.45g, meaning that the concrete cask will not overturn due to

the design basis earthquake. The maximum horizontal sliding is determined using Appendix VII NUREG/CR-6865 [7] due to the horizontal acceleration load. The applicant demonstrates that the minimum factor of safety for sliding leading to cask impact is 5.91. This factor of safety is larger than the minimum requirement of 1.1 per ANSI/ANS-57.9 [2]. Maximum stresses due to an earthquake event are combined with stresses due to other design basis load with appropriate load combinations and the resulting maximum stresses are below applicable allowable stresses.

11.3.3 Explosion

Explosive overpressure is bounded by the design basis flood event pressure of 22 psig.

11.3.4 Failure of All Fuel Rods with Subsequent Ground Level Breach of the Canister

The canister is shown to be robust enough such that a mechanistic breach of the confinement boundary is not credible; therefore, no further evaluation is required.

11.3.5 Fire

See Thermal Evaluation of this SER.

11.3.6 Flood

The applicant evaluated the storage system for a design flood with a 50-ft water depth, a steady-sate flow velocity of 15ft/sec and a design basis hydrostatic pressure of 22 psig. The drag force of the concrete cask is the same horizontal drag force calculated for the Yankee-MPC storage system (21.72 kips per MPC FSAR Section 11.2.6.2). The minimum force required to cause the cask to overturn is 98.7 kips, therefore the drag force is not large enough to cause the cask to overturn. The minimum required coefficient of static friction to prevent sliding is 0.18 which is less than the lower bound coefficient of static friction for steel-on-concrete (0.35); the design basis flood will not cause the MPC-LACBWR concrete cask to slide. The flood hydrostatic pressure load (22 psig) applied to the exterior surfaces of the canister model result in a maximum primary membrane stress of 7.28 ksi and a primary membrane plus bending stress of 30.14 ksi, which are below the corresponding allowable stress intensities. The analysis demonstrates the concrete cask will not overturn or slide and the structural integrity of the MPC-LACBWR will not be compromised under the design basis flood conditions.

11.3.7 Fresh Fuel Loading in Canister

See Criticality Evaluation of this SER.

11.3.8 Full Blockage of Air Inlets and Outlets

See Thermal Evaluation of this SER.

11.3.9 Lightning

See Thermal Evaluation of this SER.

11.3.10 Maximum Anticipated Heat Load

The applicant did not evaluate the canister for the maximum anticipated heat load because of its classification as a secondary stress. The concrete cask shell was evaluated and the applicant determined that the stresses in the concrete, due to the similarities to the Yankee-MPC and lower design basis heat load of the MPC-LACBWR, are bounded by the previously approved evaluation for Yankee-MPC.

11.3.11 Storage Cask 6 Inch Drop

The applicant used a previously approved energy balance approach to determine the bounding g-loads on the concrete cask and canister due to a 6-inch drop. The average decelerations were then used to evaluate stress intensities in the canister as well as internal canister components, including basket support disks, basket weldments, basket tie rods, basket fuel tubes, and damaged fuel cans. In all cases, the applicant demonstrated positive Margins of Safety.

11.3.12 Tip-Over of the Concrete Cask

The applicant utilized the Yankee-MPC concrete cask tip-over finite element model, with modifications to the density, the elastic modulus of the soil and compressive strength of the the concrete pad, 120 pcf, 10,000 psi and 6000 psi respectively, to evaluate the MPC-LACBWR concrete cask for a tip-over event (SAR Section 11.A.2.12).

SAR Tables 11.A.2.12-1 and 11.A.2.12-2 illustrate the stress intensities in the MPC-LACBWR canister at 15 axial locations due to cask tip over load. Maximum primary membrane and primary membrane- plus-bending stress components for the 5 drop orientations are then compared with allowable stresses. The stress criteria are in accordance with the ASME Code Appendix F, except for consideration of a stress reduction factor of 0.8 per NRC Interim Staff Guidance-4 Rev.1 [5] for the structural lid weld. The lowest stress margin of safety for the primary membrane stress is 0.40 and for the primary membrane-plus-bending stress is 0.63.

The SAR shows that the peak acceleration at the top support disk and at the canister top end is 21.9g and 24.6g, respectively. For the canister tip over stress analysis a dynamic load factor of 1.02 was calculated with a static equivalent load of 25g. The analysis was performed for five basket impact orientations (0°, 11.2°, 15.2°, 37°, and 45°) and the highest stresses were found to occur in the top support disks.

SAR Table 11.A.2.12-4 list the results for the margin of safety against buckling for the top support disks in each tip over impact orientation evaluated. The applicant indicates that the lower margin of safety against buckling is 0.16 and occurs at section 111 of support disk 25 for the 15.2° impact orientation. The support disk evaluation follows the buckling acceptance criteria of NUREG/CR-6322 [6].

11.3.13 Tornado and Tornado Driven Missiles

The stresses used to evaluate the MPC-LACBWR are bounding peak stresses developed for the Yankee-MPC evaluation. When combined with peak stresses from other loadings the conservative maximum stresses are demonstrated to be below the allowable material stresses. Damage due to tornado missiles is expected to be equal to or less severe than the Yankee-MPC because of similarities in geometry and mechanical configuration.

11.4 Evaluation of Site Specific Fuel Components

End Drop Evaluation

The applicant used bounding stress results obtained from 6-inch concrete cask drop evaluation to determine the stress intensity in the fuel cladding. The evaluation was performed with an explicit finite element analysis to evaluate dynamic effects. The model geometry consisted of a half-symmetry section of a fuel assembly, which included ten idealized fuel rods, the fuel tube walls, and idealized grid spacers. Fuel rod bow was also incorporated which lowers the critical buckling load and increases the displacements such that stress intensities are more realistic

than geometrically idealized fuel rods. The applicant reported that the maximum stress intensity produced a significant positive Margin of Safety against yielding of the cladding.

11.5 Tip-Over Evaluation

The tip-over evaluation was performed in a similar manner to the end drop evaluation. The impact is idealized as a side-drop subject to a boundary condition of a constant deceleration of 21g at the support together with a uniform initial velocity of 243.6 inch/sec, which is associated with the tangential component of the angular velocity of the VCC rotating about its base. The applicant reported that the maximum stress intensity produced a positive Margin of Safety against yielding of the cladding.

11.6 Evaluation Findings

The staff finds that the accident design criteria for the MPC-LACBWR cask design are in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the cask adequately demonstrates that it will provide for safe storage of spent fuel during credible accident situations. This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices.

F11.1 Structures, systems, and components of the MPC-LACBWR cask design are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that do occur.

F11.2 The spacing of casks, discussed in Section 1.3 of the safety evaluation report (SER) and included as an operating limit in Section 12 of the SAR will ensure accessibility of the equipment and services required for emergency response.

F11.3 The applicant has evaluated the MPC-LACBWR cask design to demonstrate that it will reasonably maintain confinement of radioactive material under credible accident conditions.

F11.4 An accident or natural phenomena event will not preclude the ready retrieval of spent fuel for further processing or disposal.

F11.5 The spent fuel will be maintained in a subcritical condition under accident conditions.

F11.6 Neither off-normal nor accident conditions will result in a dose, to an individual outside the controlled area, that exceeds the limits of 10 CFR 72.104(a) or 72.106(b), respectively.

F11.7 No instruments or control systems are required to remain operational under accident conditions.

12.0 CONDITIONS FOR CASK USE – OPERATING CONTROLS AND LIMITS OR TECHNICAL SPECIFICATIONS

The purpose of the review of the conditions for cask use is to determine whether the applicant has fully evaluated the TS and to ensure that the SER incorporates any additional operating controls and limits that the staff deems necessary.

12.1 Condition of Use

The conditions for use of the MPC-LACBWR system are fully defined in the CoC and the TS which are appended to it. The Certificate of Compliance Conditions has been revised to reflect the use of the MPC system to include storage of undamaged Exxon fuel assemblies; damaged fuel cans containing undamaged Exxon fuel assemblies, damaged Exxon and Allis Chalmers fuel assemblies and/or fuel debris; and zirconium alloy shroud compaction debris stored with undamaged and damaged fuel assemblies. The aforementioned content is from the decommissioned Dairyland Power Cooperative La Crosse Boiling Water Reactor.

12.2 Technical Specifications

The applicant revised Chapter 12 to include all MPC-LACBWR specific technical specifications, approved contents and design features, and technical specification bases. Section A1.1, Definitions, of Appendix A contains newly defined terms and editorial modifications. Changes to the limiting condition for operation (LCO) and surveillance requirements (SR) applicability in Appendix A of the technical specifications is indexed in the table below. Affected sections include NAC-MPC System Integrity and Radiation Protection. Preoperational Testing and Training Exercises, Section A 5.2 (e, k) of the Administrative Control and Programs was also modified in this application. Table 12-1 lists the TS for the MPC-LACBWR system changed with the original application and each supplement. Practically every section of the Approved Contents and Design Features for the NAC-MPC System – Amendment 6 (Appendix B) has been modified and lists the specifications associated with dry storage of LACBWR spent fuel. The staff has appended these TS to the CoC for the MPC-LACBWR.

Table 12-2 lists the license drawings for the MPC-LACBWR system that were relied upon as a basis for approval.

Table 12-1 MPC-LACBWR SYSTEM TECHNICAL SPECIFICATIONS – Changes

List of Changes, MPC-LACBWR Amendment, Revision 08A
Appendix A – Technical Specifications for the NAC-MPC System

- A 1.1 Definitions – added LACBWR fuel related definitions
- A 3.1.1 CANISTER Maximum Time in Vacuum Drying, LCO 3.1.1 – added Item “1.c” for MPC-LACBWR, no time limit, and clarified required action and surveillance requirements;
- A 3.1.2 CANISTER Vacuum Drying Pressure, LCO 3.1.2 – clarified vacuum drying pressure and note;
- A 3.1.3 CANISTER Helium Backfill Pressure, LCO 3.1.3 – clarified helium backfill pressure;
- A 3.1.5 CANISTER Helium Leak Rate, LCO 3.1.5 – clarified applicability;
- A 3.1.6 CONCRRETE CASK Heat Removal System, LCO 3.1.6, SR 3.1.6.1 – clarified surveillance requirement

- A 3.1.7 Fuel Cooldown Requirements, LCO 3.1.7 – clarified completion time;
- A 3.2.1 CANISTER Surface Contamination, LCO 3.2.1 – clarified contamination surfaces and surveillance requirements;
- A 3.2.2 CONCRETE CASK Average Surface Dose Rates, LCO 3.3.2 – added Item "C" average surface dose rates for MPC-LACBWR and clarified surveillance requirements, revised Figure A3-1;
- A 5.0, A 5.2 Preoperational Testing and Training Exercises, Items "e" and "k" – clarified lid options; and
- A 5.4 Radioactive Effluent Control Program – clarified implementation of 10 CFR 72.44(d)

Appendix B – Approved Contents and Design Features for the NAC-MPC System

- B 2.1 Fuel Specifications and Loading Conditions – added MPC-LACBWR as the third configuration of the NAC-MPC System; B 2.1.3 – added new section describing fuel to be stored in the MPC-LACBWR System, added Table B2-7 and Table B2-8 describing the LACBWR fuel assembly limits and characteristics, respectively; Added Figure B2-3 depicting the MPC-LACBWR loading pattern;
- B 2.2 Violations, B 2.2.3 updated reporting requirement;
- B 3.2 Design Features Significant to Safety, B 3.2.1 – added criticality control requirements for MPC-LACBWR; Added Figure B3-1 depicting the MPC-LACBWR minimum web thickness; revised Table B3-1 to update ASME code alternatives; and
- B 3.4 Site Specific Parameters and Analyses – updated as applicable to MPC-LACBWR.

List of Changes, MPC-LACBWR Amendment, Revision 09A
Proposed Technical Specifications – Appendix B

- Page B2-18, Table B2-7 – added new Item F to address zirconium alloy compaction debris

List of Changes, MPC-LACBWR Amendment, Revision 09B
Proposed Technical Specifications – Appendix B

- Page B3-9, Section B 3.4, Site Specific Parameters and Analyses (continued), Item 6, last column of table l – changed specified concrete compressive strength from "≤ 4,000 psi at 28 days" to "≤ 6,000 psi at 28 days"

List of Changes, MPC-LACBWR Amendment, Revision 09C

- Page B3-8, Section B 3.4, Item 3 – a) is revised by adding "For Yankee-MPC and CY-MPC,"; b) is new to address MPC-LACBWR; former b) is now c)

List of Changes, MPC-LACBWR Amendment, Revision 10A
Appendix 12.B

- Page B2-21, Figure B2-3, MPC-LACBWR Loading Pattern – notes below the figure were revised to clarify the loading pattern for LACBWR undamaged fuel assemblies and LACBWR damaged fuel cans

Table 12-2 MPC-LACBWR License Drawings

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

12.3 Evaluation Findings

The staff finds that the conditions for use of the MPC-LACBWR identify necessary technical specifications to satisfy 10 CFR Part 72, and that the applicable acceptance criteria have been satisfied. The proposed technical specifications provide reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

F10.1 Table 12.1 of the safety evaluation report (SER) lists the technical specifications for the MPC-LACBWR. These technical specifications are further discussed in Section 12 of the safety analysis report (SAR).

13.0 QUALITY ASSURANCE

The purpose of this review and evaluation is to determine whether NAC has a Quality Assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G.

13.1 Areas Reviewed

QA Organization
QA Program
Design Control
Procurement Document Control
Instructions, Procedures, and Drawings
Document Control
Control of Purchased Material, Equipment, and Services
Identification and Control of Materials, Parts, and Components
Control of Special Processes
Licensee and Certificate Holder Inspection
Test Control
Control of Measuring and Test Equipment
Handling, Storage, and Shipping Control
Inspection, Test, and Operating Status
Nonconforming Materials, Parts, or Components
Corrective Action
Quality Assurance Records
Audits

NUREG-1536 provides the criteria for evaluating the above 18 areas. As indicated in Section 13.1 of the SAR, the NRC has issued a QA program approval (Approval No. 0018) for activities conducted under Subpart H of 10 CFR Part 71. Based on the review of the QA program described in the SAR and previous NRC determinations regarding NAC's 10 CFR Part 72 QA program, the staff has determined that it meets the requirements of Subpart G of 10 CFR Part 72. The NRC has also performed periodic inspections, the most recent in January 2009, of NAC's implementation of the 10 CFR Part 71 and 72 QA programs at their headquarters facility and at cask fabrication facilities domestically and in Japan. While these inspections assessed that NAC's QA program is being acceptably implemented, continued proper implementation of the QA program will continue to be assessed during future NRC inspections.

13.2 Evaluation Findings

F13.1 The QA program describes the requirements, procedures, and controls that, when properly implemented, comply with the requirements of 10 CFR Part 72, Subpart G and 10 CFR Part 21, "Reporting of Defects and Noncompliance."

F13.2 The structure of the organization and assignment of responsibility for each activity ensure that designated parties will perform the work to achieve and maintain specified quality requirements.

F13.3 Conformance to established requirements will be verified by qualified personnel and groups not directly responsible for the activity being performed. These personnel and groups report through a management hierarchy which grants the necessary authority

and organizational freedom and provides sufficient independence from economic and scheduling influences.

F13.4 The QA program is well documented and provides adequate control over activities affecting quality, as well as SSCs important to safety, consistent with their relative importance to safety (graded approach).

F13.5 NAC's QA program complies with the applicable NRC regulations and can be implemented for the design, fabrication, testing, modification, and use of the NAC MPC system.

F13.6 This SAR can be referenced without further QA review in a license application to receive and store spent fuel under 10 CFR Part 72, provided the applicant applies its NRC-approved QA program meeting the requirements of 10 CFR Part 50, Appendix B, to the design, construction, and use of SSCs that are important to safety for a spent fuel storage installation.

14.0 MATERIALS

The NAC-MPC system is currently licensed to store fuel from the Connecticut Yankee and Yankee class Nuclear Plants. The current amendment requests that both damaged and undamaged fuel from the La Crosse boiling water reactor (BWR) plant be added as approved content. This content will be stored using a new basket design and a new double closure seal weld on the transportable storage canister (TSC). In addition, there were other non-materials minor changes to the vertical concrete cask (VCC).

This materials review was limited to parts of the system affected by these changes. The VCC and transfer cask are constructed the same as the VCC used with the Connecticut Yankee fuel and were not reviewed further. The gamma shielding that consists of the canister steel and American Society for Testing and Materials (ASTM) B29 chemical copper grade lead is part of the canister and not reviewed further since the age of the fuel precludes the lead reaching melting temperatures. There is no reflood analysis because the fuel is too old and cold for reflood to be of concern. No source terms are given or expected because there is a welded container that is leak tight so there is no credible leak path.

In a supplemental request dated April 1, 2009, the applicant requested that "Addition of zirconium alloy shroud compaction debris to be stored with undamaged and damaged fuel assemblies." The request further said "small quantities of zirconium alloy shroud compaction debris are present in the La Crosse boiling water reactor (LACBWR) fuel pool and are expected to be present within the assembly boundary." This request has no bearing on the materials behavior of the contents during storage. The fuel will be under a static load during normal dry storage, so any metallic debris within the fuel assembly boundary is not expected to have any affect.

14.1 CONTENTS

Undamaged Exxon Nuclear Company (ENC) fuel assemblies, damaged ENC fuel assemblies, damaged Allis Chalmers (AC) fuel assemblies and/or AC fuel debris are being added as approved contents. Both fuel types consist of UO_2 pellets in a 348H stainless steel cladding. They are enriched to a maximum of 3.94% and have a maximum burnup of 22 GWd/MTU. The assembly length of.262 m (103 inches) is ~ 2/3 as long as a modern BWR fuel assembly. Fuel parameters (SAR Table 5.A.2.1, Table 6A.1-1, Table 6.A.2-2, 6.A.-1, 6.A.4-16) were spot checked and are within specifications. SAR Section 11.A.4 gives the modulus of elasticity of the 348H cladding as 242 MPa (35.3 ksi) which is in the expected range. It also gives a pellet density of 10.97 g/cm^3 (0.396 lbs/in^3). This actually is the theoretical density of the UO_2. A 95% theoretical density pellet would have a density of 10.42 g/cm^3 (0.372 lbs/in^3). This error has an insignificant effect on the structural or containment analysis. A yield strength of 254 MPa (37 Ksi) is used for the cladding at 353°C (667°F). This is significantly below the yield of the irradiated cladding thus analysis of the cladding based on this yield strength will be conservative. SAR Section 11.A.2.1.1 uses a 70% fission gas release (FGR) to bound the FGR expected from the La Crosse fuel to estimate the internal TSC pressure. In the original post-irradiation evaluation of the La Crosse BWR fuel, measurements of less than 1% FGR were made on two rods [8]. The staff has verified that the FGR for the La Crosse fuel is bounded by 70%.

The LACBWR fuel has no fuel channels or burnable absorbers. Up to 68 LACBWR assemblies can be stored but no more than 32 assemblies can be in damaged fuel cans. The latter restriction is due to the oversized nature of the basket tubes that can hold damaged fuel cans. SAR Section 11.A.4 indicates that a maximum assembly bow of 6.25mm (0.25 inch) is expected

for the ENC assemblies. The difference between the inside dimension of a standard fuel tube (or damaged fuel can) and the fuel assembly dimension is 3.454 mm (0.136 inch). Therefore the maximum bow can be 454 mm (0.136 inch) so the staff finds the use of a larger bow is conservative.

The staff found the definitions of a LACBWR damaged fuel assembly, LACBWR undamaged fuel assembly, and fuel debris, including zirconium alloy shroud compaction debris, that is present within the assembly boundary, to be adequate. They have been added to the Technical Specifications of the CoC.

The AC fuel has cooled for 28 years and the Exxon fuel for 23 years. The SAR (Section 11.A.2.10.2 has an allowable fuel cladding temperature of 430°C (805°F), obtained from EPRI Report TR-106440 [4]. No judgment is made on whether this limit is correct, but it is not being questioned at this time since the expected cladding temperature of 243°C (469°F) is so much lower than the 400°C (752°F) limit for Zircaloy which should be lower than the acceptable temperature for the stainless steel due to the higher yield strength for stainless steel. This review does not endorse or make a judgment about the accuracy of the EPRI temperature limit for stainless steel cladding other than it is higher than Zircaloy as one would expect, and much higher than the expected temperature of the stainless steel cladding.

Water is removed from TSC to allow welding of the upper lid but at no time are the fuel rods exposed to air. A hydrogen detector is used to sample the H_2 gas below the closure lid before welding. Welding is stopped if >2.3% H_2 is detected (SAR 8.A.1.1). After draining, the cask is blown down with pressurized helium. A vacuum is drawn until the TSC attains a pressure (P) < 10 torr for 10 minutes with the pump off (SAR 8.A.1.1). During this time the cladding temperature (<250°C) is maintained <430°C (806°F) (SAR 8.A.1.1). There is no time limit on vacuum drying since the temperature of the fuel is so low (SAR Section C.3.1.1). While this is not the standard drying procedure of P <3 torr for 30 minutes, the staff has approved this method in the past. After vacuum drying, a vacuum of 3 torr is pulled and the canister backfilled with 99.9% pure He fill gas. The staff finds the operating procedures for the drying are satisfactory.

14.2 FUEL BASKET

The basket is constructed of Type 304 stainless steel except for the support disks of Type 17-4 PH stainless steel, and the heat transfer disks of Type 6061-T651 aluminum alloy. The fuel tubes are made from Type 304 stainless steel and are covered with stainless steel encased BORAL sheets on one or more outer tube surfaces. The sides not covered with BORAL sheets have aluminum coverings for symmetry.

The basket is held together with 8 tie rods. The nuts are lubricated with Neolube or other spent fuel pool compatible lubricant prior to welding. All the basket welding procedures and welder qualifications are in accordance with ASME Section IX. The welds are visually examines in accordance with ASME code section V, Article 9, and accepted per ASME Code Section III, Subsection NG (article NG-5360).

The BORAL sheets on one or more sides of the basket tubes act as the neutron absorber. They have a ^{10}B loading of 0.015 g/cm^2 based on a minimum areal density of 0.02 g/cm^2 ^{10}B and a 75% efficiency factor. These plates are covered with stainless steel. An acceptance and sampling plan, acceptable to the staff, are in the SAR (Section 9.1.6 and Section 9.A.2.7) and are incorporated into the CoC by reference. The acceptance plan has been modified from

previous amendments by requiring the chemical analysis tests be benched marked against neutron attenuation measurements.

All the AC fuel and the ENC fuel that is deemed damaged will be placed in a damaged fuel can. Damaged fuel cans are constructed of 304 stainless steel. The ends will be covered with a stainless steel mesh (mesh size = 250 x 250) screen to retain any particulate fuel. A definition of a LACBWR damaged fuel can has been added to the Technical Specifications of the CoC. Load bearing welds will be inspected using dye penetrant (PT) in accordance with ASME Code Section V, Article 6, and accepted according to ASTM Code Section III, NG-5350 (SAR Section 9.A.2.3). These welds will also be visually inspected as specified in ASME code Section V, Article 9, and accepted in accordance with ASME Code Section III, NG-4000 (SAR Table 1.A.3-3).

14.3 GALVANIC and CHEMICAL INTERACTIONS

Under the expected loading and storage environments, the analysis of potential galvanic and chemical interactions in the original FSAR (NAC-MPC FSAR, April 2000) indicates no galvanic or chemical interactions between the fuel contents and any of the materials of construction of the basket or canister are expected. In that case, all four sides of the fuel tubes were covered with stainless steel encased BORAL and no aluminum plates were used. The operating procedures mitigate the effects of the generation of the hydrogen.

14.4 CANISTER TO UPPER LID SEAL WELDS

The MPC-LACBWR canister is constructed of Type 304 or 304L stainless steel.

The closure lid-to-TSC shell weld is analyzed, installed and examined in accordance with ISG-15 “Materials Evaluation”, and ISG-18 “The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage” guidance, (SAR Section 7.A.1.1). The maximum flaw size is consistent with the ISG-15 guidance (CoC Table B3-1). The closure welds will be either groove or bevel welds made according to ASTM Section III, Subsection NB, category C (SAR Table 7.A.1-1). These partial penetration welds will be examined with PT after each of three passes. The inner port cover welds will be helium leak tested, and the final surface welds will be PT examined. The examination of the weld will be performed by qualified personnel per ASTM Code Section V, Article 1 and 6 with acceptance per ASTM Code Section 111, NB-5350 (CoC Table B3-1). The welds will be qualified and tested according to ASME Code Section IX. Examination of field welds of TSC confinement barrier will be performed by personnel qualified in accordance with American Society of Nondestructive Testing practice No. SNT-TC-1A (SAR Section 7.A.1.3).

14.5 DRAWINGS

All drawings relevant to the MPC-LACBWR, (i.e. closure welds, basket, and damaged fuel can) are included in the SAR. The drawings list all materials specifications for construction of the parts, along with weld specifications.

14.6 Evaluation Findings

The following findings refer to the materials review of the systems affected by the requested changes to the TSC, and contents and VCC. Materials not affected by these changes were previously reviewed and deemed acceptable.

F14.1 The SAR adequately describes the materials used for TSC and/or VCC important to safety and the suitability of those materials for their intended functions in sufficient detail

to evaluate their effectiveness.

F14.2 The applicant has met the requirements of 10 CFR 72.140(a). The material properties of TSC and/or VCC important to safety conform to quality standards commensurate with their safety function.

F14.3 The applicant has met the requirements of 10 CFR 72.124. Materials used for criticality control are adequately designed and specified to perform their intended function.

F14.4 The applicant has met the requirements of 10 CFR 72.122(h)(1) and 236(h). The design of the TSC and/or VCC and the selection of materials adequately protect the SNF cladding against degradation that might otherwise lead to damaged fuel.

F14.5 The applicant has met the requirements of 10 CFR 72.236(h) and 236(m). The material properties of TSC and/or VCC important to safety will be maintained during normal, off-normal, and accident conditions of operation so the SNF can be readily retrieved without posing operational safety problems.

F14.6 The applicant has met the requirements of 10 CFR 72.236(g). The material properties of TSC and/or VCC important to safety will be maintained during all conditions of operation so the SNF, including any zirconium alloy shroud compaction debris, that is present within the assembly boundary, can be safely stored for a minimum of 20 years and maintenance can be conducted as required.

F14.7 The applicant has met the requirements of 10 CFR 72.236(h). The MPC-LACBWR TSC employs materials that are compatible with wet SNF loading and unloading operations and facilities. These materials should not degrade over time or react with one another during any conditions of storage.

The staff finds the material properties of the structures, systems, and components of the MPC-LACBWR TSC and/or VCC are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the material properties provides reasonable assurance the MPC-LACBWR TSC and/or VCC will allow safe storage of SNF for a licensed life of 20 years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

CONCLUSION

The staff has reviewed the amendment for the MPC-LACBWR system. Based on the statements and representations contained in the SAR, as amended, and the conditions specified in the Certificate of Compliance, as amended, the staff concludes that the MPC-LACBWR system meets the requirements of 10 CFR Part 72.

Conclusions Regarding Analytical Methods

The staff determined that, unless otherwise noted in this SER, all analytical methods used by the applicant in this amendment application for the design modifications to the NAC-MPC system proposed in the amendment request are acceptable. However, the staff did not review any methodologies used in the original NAC-MPC system application and did not make a determination on the adequacy of the previous methodologies.

Issued with Certificate of Compliance No. 1025, Amendment No. 6,
on xxx xx, 2010.

REFERENCES

[1] ANSI/ANS-6.1.1-1977, “Neutron and Gamma-Ray Flux-to-Dose-Rate Factors,” American Nuclear Society Committee Working Group, 1997.

[2] ANSI/ANS-57.9-1992, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type),” American Nuclear Society Committee Working Group, 1992.

[3] ASCE 4-86, “Seismic Analysis of Safety-Related Nuclear Structures: Working Group on Seismic Analysis of Safety Related Nuclear Structures,” American Society of Civil Engineers Staff, 1987.

[4] EPRI Report TR-106440, “Evaluation of Expected Behavior of LWR Stainless Steel Clad Fuel in Long-term Storage,” April 1996.

[5] NRC Interim Staff Guidance-4 Rev.1, “Cask Closure Weld Inspections.”

[6] NUREG/CR-6322, “Buckling Analysis of Spent Fuel Basket,” May 1995.

[7] NUREG/CR-6865, “Parametric Evaluation of Seismic Behavior of Freestanding Spent Fuel Dry Cask Storage Systems (NUREG/CR-6865, SAND2004-5794P),” February 2005.

[8] Rieger, G.F., “Post-Irradiation Evaluation of Fuel Rods from the La Crosse Boiling Water Reactor,” June 1976, NEDC-12633.