SAFETY EVALUATION REPORT

Docket No. 72-1025
NAC-MPC STORAGE SYSTEM
Certificate of Compliance No. 1025
Amendment No. 2

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SUMMARY

By application dated May 19, 2000, as supplemented September 6, October 2 and 12, 2000; and April 13, September 6, October 5, 10 and 15, and November 21, 2001, NAC International, Inc. (NAC) requested approval of an amendment, under the provisions of 10 CFR Part 72, Subpart K, to Certificate of Compliance No. 1025 for the Multi-Purpose Canister (MPC) Storage System.

The NAC-MPC system (the cask) consists of the following components: (1) transportable storage canister (TSC), which contains the spent fuel; (2) vertical concrete cask (VCC), which contains the TSC during storage; and (3) a transfer cask, which contains the TSC during loading, unloading, and transfer operations. The cask stores up to 36 fuel assemblies from the Yankee Rowe pressurized water reactor (PWR).

For this amendment to the Certificate of Compliance, NAC requested approval to store the spent nuclear fuel from the decommissioned Connecticut Yankee (CY) Haddam Neck power plant in the MPC system. The changes included increasing the length of the TSC, VCC and Transfer Cask to accommodate the longer CY fuel. A new fuel basket was designed for 26 CY fuel assemblies with an alternate 24 fuel-assembly configuration. The Transfer Cask shielding was also increased to accommodate the CY fuel. Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features) of the certificate were revised in their entirety following the standard technical specification format in NUREG-1745, “Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance.” Furthermore, the certificate format was revised to make the conditions more accurate and eliminate duplication.

NRC staff reviewed the application using the guidance in NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems.” Based on the statements and representations in the application, as supplemented, and the conditions discussed in this Safety Evaluation Report (SER), the staff concluded that the NAC-MPC system meets the requirements of 10 CFR Part 72. The changes to the Certificate of Compliance are indicated by change bars in the margins.

1.0 GENERAL DESCRIPTION

1.1 System Description

The NAC-MPC system is intended to be a transport-compatible dry storage system that uses a stainless steel TSC stored within the central cavity of a VCC. The TSC is intended to be compatible with the NAC-STC transport cask to allow future shipment. The VCC provides radiation shielding and contains internal air flow paths that allow decay heat from the TSC spent fuel contents to be removed by natural air circulation around the canister wall.

The principal components of the NAC-MPC system are the TSC, the VCC, and the transfer cask. The transfer cask is used to move the loaded TSC to and from the VCC and provides radiation shielding while the TSC is being closed and sealed. The TSC is placed in the VCC by positioning the transfer cask on top of the VCC and lowering the TSC in.

1.2 Contents

The applicant requested changes to the authorized contents to include spent nuclear fuel from the CY Haddam Neck power plant. The application contained a redesigned fuel basket to store 26 CY fuel assemblies with an alternate 24 fuel-assembly configuration.

2.0 PRINCIPAL DESIGN CRITERIA

The applicant revised this section to include storage of the CY fuel assemblies. The MPC changes to include the CY fuel are technically evaluated in the sections that follow. Based on those reviews, the staff concludes that the revisions to Safety Analysis Report (SAR) Chapter 2 are acceptable, and continue to meet the requirements of 10 CFR Part 72.

3.0 STRUCTURAL

The CY-MPC configuration has components and operating features similar to the approved MPC design. However, the physical dimensions, weights, and storage capacities are different from the approved MPC. In the following evaluation, the staff focuses on the unique design features and corresponding structural performance of the CY-MPC.

3.1 Structural Design

<u>Design Features</u>

SAR Section 1.2 provides a general description of the MPC, which identifies also the design features unique to the amended MPC to accommodate the CY fuel assemblies. Compared to the approved MPC, the amended MPC is configured with (1) an increased length and weight of fuel assemblies, (2) a varied cutout size and pattern of the fuel basket support and heat transfer disks, and (3) an increased overall length for the TSC, VCC, and transfer cask by about 27 to

30 inches. SAR Tables 1.2-1, 1.2-3, and 1.2-5 list the major physical design parameters of the TSC, VCC, and transfer cask of the amended MPC, respectively.

The application also addresses the storage of CY Reconfigured Fuel Assemblies (CY-RFAs) in the MPC. A CY-RFA is designed to hold damaged fuel rods and fuel debris and consists of a 10 x 10 array of stainless steel tubes attached to the upper and lower end fittings that are similar to those used on a standard fuel assembly. The CY-RFA measures 8.9 inches square by 141.5 inches long to allow its placement only in the four basket corner locations with the oversized support and heat transfer disk cutout of 9.12 inches square. Four ½-inch-thick grid-plates provide lateral support for the tubes, and four 2 x 2 x 3/16 angles, one at each corner, are used to provide additional strength to the assembly. SAR Table 1.3-3 lists the major physical design parameters of the CY-RFA.

The 8.9 inches square by 141.5 inches long CY Damaged Fuel Can (CY-DFC), which is to be placed in the basket corner positions, consists of an 18-gauge square stainless steel shell body, a bottom weldment and a top closure assembly. It is designed to hold a CY damaged fuel assembly, Lattice, or Failed Rod Storage Canister by providing a confinement function while allowing release of gaseous products and liquids. SAR Table 1.3-4 lists the major physical design parameters of the CY-DFC.

Design Criteria

The existing design criteria for the approved MPC apply also to the amended MPC. This includes consideration of codes and standards, environmental and natural phenomenon loads, load combinations, and stress allowables.

Weights and Centers of Gravity

SAR Table 3.2.2 lists the calculated weights and centers of gravity for the major components, and the system as a whole, for the amended MPC. This information provides the basis for structural evaluations including lifting devices, impact responses associated with the VCC drop and tipover accidents, and stability of the VCC against sliding and overturning under analyzed accident conditions.

Finite Element Analysis Codes and Models

The structural analysis of the amended MPC continues to use the ANSYS and LS-DYNA codes and corresponding modeling approaches developed for the previously approved MPC. In general, the approved-MPC series of structural analyses were repeated, but with CY-fuel-specific modeling attributes, such as geometry, material properties, and boundary/loading conditions.

3.2 Normal Operating and Design Conditions

3.2.1 Lifting Devices

Transfer Cask Lift

The SAR presents evaluation of the transfer cask shell and trunnions for a total load of 185,000 lbs, which bounds the combined weight of 183,218 lbs of the transfer cask, loaded TSC, water, and shield lid. SAR Tables 3.4.3.6-1 and -2 summarize stress results for the transfer cask outer and inner shells, respectively. Except for localized over-stresses, as permitted by ANSI N14.6, stresses in the shells are shown to meet the stress design factors of 6 and 10 against the respective yield and ultimate strengths. For the trunnions, the SAR calculates a maximum linearized stress of 3,798 psi in bending and 1,472 psi in shear, which corresponds to the stress design factors of 8.4 and 18.4 against the respective material yield and ultimate strengths, and is acceptable.

The SAR uses stress formulas to evaluate other load bearing components, including the shield door and bolted-in-place retaining ring assemblies, to demonstrate that the transfer cask is structurally adequate in meeting the non-redundant, critical-lifting requirements of ANSI N14.6 and NUREG-0612.

TSC Lift

SAR Section 3.4.3.5 addresses the structural performance of the 2-inch-diameter hoist ring, the structural lid, and the weld that joins the structural lid to the shell for lifting a load of 66,000 lbs. The analysis demonstrates that those components meet the redundant load path strength criteria. Other canister load bearing members are capable of supporting three times the weight of the loaded canister without generating a stress in excess of the material yield strength. This is acceptable.

VCC Bottom-Lift

SAR Section 3.4.3.4 provides an evaluation of the structural performance of the VCC components for a design weight of 252,000 lbs, which bounds the combined weight of 251,771 lbs for the VCC and loaded TSC. The evaluation includes concrete bearing stresses at the lifting jack locations, the size and spacing of Nelson stud anchors for the cask base, and stresses in the canister support pedestal. The stress margins are acceptable.

3.2.2 Hot and Cold Temperature Effects

The SAR continues to consider the design basis ambient temperatures of 75°, 100°, -40°, and 125°F for the normal, off-normal severe heat, off-normal severe cold, and accident extreme heat conditions, respectively, for the amended MPC system.

SAR Section 3.4.4.3 presents thermal stress analyses of the MPC components, including the TSC, fuel basket top and bottom weldments, and fuel basket support disk. SAR Section 3.4.4.4.3 contains a thermal stress evaluation for the VCC. The analyses use the same modeling approaches as those for the approved MPC, but with the CY-fuel-specific bounding thermal and pressure loadings calculated in SAR Sections 4.5.3 and 4.5.5, respectively. The stress results, including differential thermal expansion evaluations, are acceptable.

3.2.3 CY-MPC System Components Structural Analysis

SAR Section 3.4.4 presents normal condition analyses of the CY-MPC components, under the individual and combined dead weight, thermal, pressure, and handling loads.

TSC

SAR Table 3.4.4.3-1 summarizes the maximum TSC thermal stresses under normal operating conditions. SAR Tables 3.4.4.3-2 and -3 present the respective TSC primary membrane and primary membrane-plus-bending stress summaries for the dead weight load. SAR Tables 3.4.4.3-4 and -5 present stress results for the TSC subject to an internal pressure of 15 psig, which bounds the maximum calculated internal pressure of 9 psig. SAR Tables 3.4.4.3-6 and -7 list normal handling stresses. SAR Tables 3.4.4.3-8 through -10 summarize the maximum stresses in the TSC for the respective primary membrane, primary membrane-plus-bending, and primary-plus-secondary stresses at key TSC locations with a minimum design margin of 0.23. The structural performance of the TSC is acceptable for normal operating conditions.

Fuel Basket Support Disk

SAR Section 3.4.4.3.8 addresses the fuel basket support disk for the storage and handling conditions. SAR Table 3.4.4.3-11 summarizes the combined load effects with a minimum design margin of 5.68.

Fuel Basket Top and Bottom Weldments

SAR Section 3.4.4.3.9 provides an evaluation of the top and bottom weldments of the fuel basket for storage and handling conditions. For both configurations, the bounding thermal gradient is based on a temperature of 500°F at the center and 200°F at the circumference of the weldments. SAR Table 3.4.4.3-11 summarizes the combined load effects with a minimum design margin of 2.28.

VCC

SAR Section 3.4.4.4 addresses the structural performance of the VCC for normal conditions of storage by considering the dead and live weight loads, and differential thermal expansion loads. SAR Table 3.4.4.4-1 provides a stress summary for the load combinations defined in SAR Table 2.2-2. The concrete stresses related to the dead weight and live loads are negligibly small. The maximum compressive thermal stresses for the off-normal severe heat condition are 657 psi and 115 psi in the cask axial and circumferential directions, respectively. SAR Table 3.4.4.4-2 summarizes the maximum stresses in the VCC concrete and rebar. The minimum margin of safety for concrete compressive stress is 2.78, and stress margin for the rebar is large. The VCC is structurally acceptable for normal operating conditions.

<u>CY-RFA</u>

SAR Section 3.4.4.5 addresses the structural performance of the CY-RFA for a 20-g end impact, which bounds the dead load plus handling load of 1.1 g for normal operation conditions. The SAR considers NUREG/CR-6322 criteria to demonstrate buckling strength of the corner angles and fuel tubes. For the finite element analysis of the grid plate, in addition to an inertia load of 20 g, the thermal solution considers the temperatures of 750°F and 650°F at the grid plate center and perimeter, respectively, for bounding conditions on the thermal gradient. The maximum stress due to the combined inertia and thermal loads is 9.2 ksi, which is less than the at-temperature allowable of 46.8 ksi, and is acceptable.

<u>CY-DFC</u>

SAR Section 3.4.4.6 addresses the structural performance of the CY-DFC for a 20-g end impact, which bounds the dead load plus handling load of 1.1 g for normal operation conditions. The SAR presents analyses to demonstrate large stress margins for the tube body and top lid. The staff concurs with the SAR assessment that the CY-DFC is free to expand and thermal stresses are negligible.

3.3 Materials

3.3.1 General

Changes to the approved MPC system were reviewed to determine whether there were changes to the types of materials used in the amended MPC design. The staff found that no significant changes in materials of construction occurred. Thus, no re-analysis of material suitability was necessary for the amended MPC system.

3.3.2 CY-RFAs and CY-DFCs

The CY-RFAs and CY-DFCs are fabricated from type 304 stainless steel. This is the same material used in the TSC shell. Thus, no new or different materials are introduced into the TSC as a result of the amendment to require staff re-analysis of potential deleterious chemical or galvanic reactions which could occur during loading operations or storage.

3.3.3 Ferritic Steels–Brittle Fracture Control Measures

Certain transfer cask components, including the lifting trunnions, are fabricated from ferritic steels. Since ferritic materials are potentially susceptible to brittle fracture in cold conditions, specific measures for avoiding or controlling this possibility must be incorporated into the design. The applicant used the recommended practices of NUREG/CR-1815, “Recommendations for Protecting Against Failure By Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick.” By choosing the materials design methods of this document, the applicant has taken measures to prevent brittle failure under the coldest design operating temperatures.

The lifting trunnions are made of a ferritic material not listed in NUREG/CR-1815. However, the ASTM SA-350 forgings used for the lifting trunnions are required to pass certain low temperature impact tests in compliance with the design guidance of the NUREG. In addition, Technical Specification B 3.4 prohibits MPC users from moving the transfer cask when the ambient temperature is below 0°F.

3.3.4 Fuel Element Maximum Temperatures

The MPC is designed for both Zircaloy and stainless-steel clad spent fuel. For the stainless-steel clad spent fuel, the licensee has adopted the guidance of EPRI TR-106440, “Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage,” for establishing the short-term and long-term temperature limits for the cladding. In accordance with this guidance, both the long-term and short-term temperature limits for stainless-steel clad fuel are set at 806°F (430°C).

Zircaloy clad fuel long term temperature limits are determined in accordance with the methods of PNL-6189, “Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas.” Short-term temperature limits are established in accordance with PNL-4835, “Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gasses.” Since the maximum allowable cladding temperature is a function of several variables, including burn-up and cool time before loading into a dry storage cask, SAR Tables 4.5.3-4, 4.5.3-5, 4.5.3-6, Figure 4.5.3-4, and others, provide the temperature limits under a variety of design and operating conditions for both damaged and undamaged fuel.

3.3.5 Proposed Alternatives to the ASME Code

A number of alternatives to the ASME Code were proposed in the application. With one exception, these proposed alternatives (listed in SAR Table 12 B 3-1) involve alternatives previously reviewed and found by the staff as acceptable alternatives to the Code requirements.

The exception was a proposed alternative for the use of Code materials procured from non-Code-approved suppliers. Under the ASME Code, Section III, a non-Code supplier may be used as a source of Code-materials provided that the ASME Code-Stamp holder purchasing the material performs the qualification and audit functions normally performed by ASME for Code-approved suppliers. However, if the purchaser is also not an ASME stamp holder, the provisions of certain Code paragraphs (such as WA-3842 or NCA-3842) that allow for substitution of a non-Code approved supplier cannot be invoked without prior approval by NRC.

3.4 Accident Analyses

3.4.1 Off-Normal Conditions and Accident Events

<u>Canister Off-Normal Handling Load</u>

SAR Section 11.1.2.2.2 provides stress analysis of the TSC for the combined effects of off-normal handling loads and the design off-normal internal pressure of 15 psig. The stress results are summarized in SAR Tables 11.1.2-1 and -2 with the minimum safety margins of 2.21

and 0.32, which are acceptable, for the primary membrane and primary membrane-plus-bending stresses, respectively.

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

SAR Section 11.1.4.2.2 references SAR Section 3.4.4.3 in evaluating thermal stress effects for the TSC and its fuel basket. For thermal stress analysis of the VCC, the SAR states that the severe environmental conditions are bounded by the accident extreme heat condition at an ambient temperature of 125°F with the maximum concrete compressive stresses of 583 psi and 106 psi in the cask axial and circumferential directions, respectively. The stress results demonstrate that the MPC continues to be structurally adequate to withstand the severe and extreme temperature conditions.

Accident Pressurization

SAR Section 11.2.1.2.2 assumes a bounding temperature of 550°F to compute the maximum pressure of 49 psig in the TSC. Considering an internal pressure of 55 psig, which bounds the maximum pressure, the SAR calculates the minimum margins of 3.1 and 0.6 for the primary membrane and primary membrane-plus-bending stresses, respectively. This demonstrates that the TSC performance is not significantly affected by the increase in internal pressure that results from the hypothetical rupture of all of the fuel rods in the canister.

Explosion

SAR Section 11.2.3 references the SAR Section 11.2.6 analysis which demonstrates acceptable structural performance of the TSC when subjected to an external static pressure of 22 psig. This demonstrates that the canister will suffer no adverse consequences as a result of an explosion which exerts an equivalent static pressure of less than 22 psig on the canister.

Storage Cask 6-Inch Drop

SAR Section 11.2.11.2.2 follows the same energy balance approach for the approved VCC to evaluate the amended VCC 6-inch drop accident. For the VCC shell, the SAR presents a maximum deceleration of 138 g, which corresponds to a concrete crush depth of 0.044 inch. Considering a flow stress of 47,000 psi in a portion of the pedestal weldment, the SAR reports a maximum end impact deceleration of 18 g for the canister and a crush of about 0.36 inches for the support base air inlet. The effect of the reduction of the air inlet area by the 6-inch drop is bounded by the loss of the one-half of the air inlets, which is evaluated in SAR Section 11.1.1.

SAR Tables 11.2.11-1 and -2 summarize primary membrane and primary membrane-plus-bending stresses, respectively, for the canister subject to an end impact force of 60 g plus an internal pressure of 15 psig. The minimum stress margin is 3.20. Table 11.2.11-3 presents stress results with positive margins for the basket support disk and the top and bottom weldments for an end impact of 60 g. This demonstrates that the TSC and VCC will continue to perform satisfactorily during a 6-inch VCC drop accident.

Cask Tipover

SAR Section 11.2.12.3 follows the same LS-DYNA analysis approach for the approved MPC to compute tipover deceleration g-loads for the amended MPC. The amended MPC cask-pad-soil interaction model consists of a 30 ft x 30 ft x 3 ft concrete pad, a 45 ft x 45 ft x 5 ft soil subgrade, and a circular hollow concrete cylinder with an inner steel liner to simulate the cask system mass and its distribution. Other specific modeling features for the amended MPC include: (1) concrete compressive strength of 4,000 psi for the cask and pad, (2) subgrade soil modulus of elasticity equal to or less than 30,000 psi, (3) pad concrete density between 135 pcf and 150 pcf, (4) soil density between 85 pcf and 130 pcf, and (5) initial cask angular velocity of 1.55 rad/sec.

The SAR reports the side impact decelerations of 28.4 g and 25.3 g for the locations corresponding to the TSC structural lid and the top-most fuel basket support disk, respectively. For the support disk, the applicant continued to use the existing transient response approach for the approved MPC to calculate a dynamic load factor of 1.07, which results in a static equivalent load of 27.1 g (25.3 x 1.07 = 27.1) for the amended MPC.

SAR Section 11.2.12.3.2 provides stress analysis of the TSC and its fuel basket. The modeling approaches for the approved MPC continued to be used except for consideration of the modeling details unique to the amended MPC, such as the TSC internal pressure of 15 psig and a design basis tipover deceleration load of 40 g, which bounds the static equivalent load of 27.1 g. To provide reasonable assurance that the maximum stresses are evaluated, the SAR provides four series of structural analyses defined by the fuel basket drop orientations of 0^o, 38^o, 63^o, and 90^o.

SAR Tables 11.2.12.3.2-1 through -8 present, for 13 axial locations of the TSC body, the primary membrane and primary membrane-plus-bending stresses for the four drop orientations. The stress criteria are in accordance with ASME Code, Section III, Subsection NB, except for consideration of a stress reduction factor of 0.8 (per NRC Interim Staff Guidance - 4, Rev. 1) for the structural lid weld. The analysis demonstrates the minimum stress margins of 0.31 and 0.12 for the respective primary membrane stress, which occurs in the structural lid-to-canister shell weld, and primary membrane-plus-bending stress, which occurs in the TSC shell.

SAR Tables 11.2.12.3.2-10 through -17 list stress results for the fifth support disk, which has maximum stress intensities, from the top of the TSC for all four drop orientations. The stress criteria are in accordance with ASME Code, Section III, Subsection NG. The results demonstrate the minimum margins of 0.92 and 0.13 for the primary membrane and primary membrane-plus-bending stresses, respectively.

The applicant continued to use the NUREG/CR-6322 interaction equations approach to demonstrate acceptable buckling strength for the support disk. The evaluation results are listed in SAR Tables 11.2.12.3.2-18 through -21.

RFA and DFC

SAR Sections 11.4.2 and 11.4.3 address structural performance of the CY-RFA and CY-DFC, respectively, for a 60-g inertia load for both the end and side impacts. The inertia load bounds the 6-inch end-drop deceleration of 18 g and a tipover side impact design basis deceleration of 40 g. The applicant applied beam models in hand calculations for the side impact. The end impact analysis follows the same approaches for the normal operation end impact except for consideration of the accident-level stress allowables. The analysis demonstrates structural adequacy of the CY-RFA and CY-DFC for the 6-inch VCC drop and tipover accidents.

Fuel Tubes

SAR Section 11.4.5 provides a structural performance evaluation of the fuel tubes for the end-drop and side-drop design impact deceleration of 55 g, which bounds the TSC 6-inch end-drop deceleration of 18 g and the tipover side-impact design basis deceleration of 40 g. For the end impact, the SAR reports a bearing stress margin of 3.46. For the side impact, the SAR presents finite element non-linear analyses for two loading cases: (1) pressure loading to simulate a uniform distribution of the fuel assembly weight, and (2) line loading to simulate the load exerted through the grid spacer. The analyses show the maximum total strains in the tube of 0.019 for the pressure loading and 0.047 for the grid loading. These correspond to the strain margins of 9.53 and 3.26, respectively, considering an acceptable strain to be one-half of the material failure strain of 0.4 in/in. This is acceptable.

Fuel Rod Buckling

SAR Section 11.4.4 addresses fuel rod buckling for four fuel assembly types: Westinghouse Zircaloy, Westinghouse SS303, B&W Zircaloy, and B&W SS303. The design basis end-impact pulse has a peak deceleration of 60 g, which bounds the 6-inch TSC end-drop impact of 18 g. Consistent with NRC's Interim Staff Guidance - 12, Revision 1, the SAR includes pellet weight and derated material properties of irradiated fuel for calculating clad section properties. The SAR models the full length of the fuel rod to be laterally constrained at grid spacers for the ANSYS analyses to obtain buckling g-load capabilities. By considering the axial impact load as impulsive and short duration in nature, the applicant calculated dynamic load factors for the four fuel rod types. The results showed the equivalent static g-loads all being less than the corresponding first-mode buckling capabilities of the fuel rods. This demonstrates that the fuel rods will not buckle during the 6-inch VCC drop accident.

3.4.2 Natural Phenomenon Events

The SAR addresses effects of natural phenomenon events by repeating analyses performed on the approved MPC, including consideration of the identical design bases, for the amended MPC configuration.

Flood

The design basis flood depth of 50 ft and water velocity of 15 ft/sec correspond to a design basis hydrostatic pressure of 22 psig and drag force of 25.9 kips on the VCC. The drag force is less than the minimum force of 109.6 kips required to cause the cask to overturn and is not large enough to overcome the friction between the cask and the concrete pad to cause the cask to slide. At an external pressure of 22 psig, SAR Section 11.2.6.2.2 reports the maximum TSC primary membrane stress of 3.91 ksi and primary membrane-plus-bending stress of 15.18 ksi, which are below the stress intensity limits. The analysis demonstrates that the VCC will not overturn or slide and the MPC will not suffer adverse structural consequences under the design basis flood conditions.

Tornado Wind and Tornado Driven Missiles

At a tornado wind velocity of 360 mph, SAR Section 11.2.13.2.2 shows an effective pressure load of 29.2 kips on the VCC. This results in a safety factor of 3.61 per ASCE 7, which requires that the overturning moment due to wind load shall not exceed two-thirds of the dead load stabilizing moment. A friction coefficient of 0.12 between the cask bottom and support pad is required to prevent cask sliding.

For penetration missiles, the applicant calculated a penetration depth for an armor-piercing shell and determined that scabbing will not result in the 21-inch-thick concrete shell. For the same armor-piercing shell impacting the VCC closure plate, the applicant estimated a penetration depth less than the plate thickness.

The SAR addresses the effects of a high-energy deformable-missile impacting the VCC and finds that the cask can sustain the impact and does not overturn. Considering an estimated impact force on the cask top and the punching-shear capacity, the applicant determined that the VCC would have sufficient capacity to withstand the high-energy missile impact.

The above analyses demonstrate that the design basis tornado wind pressure and tornado driven missiles are not capable of overturning the cask or penetrating the VCC. Therefore, the TSC confinement boundary remains intact.

Earthquake

SAR Section 11.2.2.2.2 provides an evaluation of the VCC's stability against sliding and tipover by considering a design earthquake motion, at the top of the storage pad, of 0.25 g for the two horizontal components and 0.167 g for the vertical component. For the fully loaded VCC of 251,771 lbs, the SAR states that a minimum horizontal component of 0.48 g is required to cause the cask to tipover. This corresponds to a margin of 1.92. The SAR also reports that a friction coefficient of 0.29 is required to prevent the VCC from sliding. On the basis of a commonly considered friction coefficient of 0.35 between the steel and concrete surfaces characteristic of the VCC base and the storage pad, the applicant calculated a margin of 1.21 against sliding. These margins are larger than the minimum requirement of 1.1 per ANSI/ANS-57.9, and are acceptable.

Snow and Ice

SAR Section 3.4.4.4 states that the maximum VCC snow load of 9,007 lbs is much smaller than the design basis live-load of 175,000 lbs exerted by the loaded transfer cask during the TSC loading operation. Therefore, the staff agrees that the snow and ice load has little structural effects on the VCC.

3.5 Conclusions

The NRC staff reviewed the SAR evaluation of the structural performance of the amended MPC for compliance with 10 CFR Part 72. The review considered the regulations, appropriate Regulatory Guides, applicable codes and standards, and common engineering practices. The NRC staff concludes that the amended MPC system is structurally adequate to ensure safe storage of spent fuel.

4.0 THERMAL

The thermal review ensures that the cask and fuel material temperatures of the MPC system will remain within the allowable values or criteria for normal, off-normal, and accident conditions. This objective 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 review also confirms that the thermal design of the cask has been evaluated using acceptable analytical and/or testing methods.

Over 80% of the CY spent fuel assemblies to be stored are stainless-steel clad with a cladding temperature limit of 806°F. The remainder of the assemblies are Zircaloy clad and have a lower long-term temperature limit (633°F used in the application for all basket locations except for Loading Positions 13 & 14 of the 26 assembly basket, which use 694°F). The Zircaloy-clad assemblies also have a higher short term temperature limit (i.e., 1058°F) than the stainless-steel assemblies. The applicant conservatively used in its analysis the lower long-term cladding temperature limits of Zircaloy and the short-term limit of 806°F.

The major thermal changes requested include, but are not limited to: (1) instituting a preferential loading scheme to accommodate heat loads per assembly up to 0.840 kW; (2) increasing the heat load rating of the canister from 12.5 kW to 17.5 kW (for preferential loading) or 13 kW (for non-preferential loading); (3) decreasing the minimum cooling time prior to loading from 8 to 6 years; (4) decreasing the number of assemblies to be loaded in one canister from 36 or 34, to 26 or 24 (note that preferential loading can only be accommodated via the 26-assembly TSC); (5) increasing the heat load per assembly from 0.347 kW or 0.247 kW (depending on fuel cladding material), to 0.674 kW (for non-preferentially loaded fuel) or 0.840 kW (for preferential loaded fuel); (6) consideration of the effects of failed fuel relocating within its can; and (7) increasing the times for completing loading operations (i.e., with water in the transfer cask, vacuum drying, and transfer to the storage pad) to permit more time for these operations before entering the technical-specification-required actions. Regarding the latter, the applicant has analyzed the MPC to increase the operational time limits for heat loads of 17.5 kW (for preferential loading only), 13.0 kW (with and without preferential loading), and 9.0 kW (without preferential loading). In some instances, the increased loading operational-time limits resulted in an increase in the component (fuel cladding and heat transfer disk) temperatures, but the component temperatures remain within their allowable temperature limits.

In reviewing this amendment, the staff requested that a sensitivity analysis be performed due to the relative closeness of the calculated temperatures to their limits. In response, the applicant evaluated the combined effects of a reduction in emissivities. This resulted in increasing the gaps within the basket to account for the worst case tolerances, decreasing the natural convection-film coefficient in the VCC model, and increasing the number of modeling elements in the air-gap between the TSC and the VCC inner-liner. The applicant demonstrated that the calculated temperatures may have a variance of approximately 21°F for normal conditions. For transfer operations, the applicant showed that the calculated temperatures may have a variance of 13°F at end of vacuum drying and 36°F at end of the time in transfer cask. These variances are within the calculated margins.

The applicant also demonstrated that the MPC with Zircaloy-clad fuel is evaluated using a cooling rate that is conservative with respect to that assumed in PNL-6189, "Recommended Temperature Limits for Dry Storage of Spent Fuel Light Water Reactor Zircaloy Clad Fuel Rods in Inert Gas." Therefore, the cladding temperature limits established using this guidance are bounding.

Table 4-1
Maximum Temperatures - Vacuum Drying or Transfer Operations

<table>
<tr><th colspan="2" rowspan="2"></th><th colspan="2">MPC Component</th></tr>
<tr><th>Fuel Cladding</th><th>Heat Transfer Disk</th></tr>
<tr><td colspan="2">Allowable (Short Term) Temperature (°F)</td><td>806</td><td>700</td></tr>
<tr><td rowspan="4">Maximum Temperature During Vacuum Drying or Transfer Operations within the time limits established by the technical specifications (refer to SAR Table 4.5.3-7) (°F)</td><td>17.5 kW (preferential load)</td><td>767</td><td>656</td></tr>
<tr><td>13.0 kW (preferential load)</td><td>784</td><td>612</td></tr>
<tr><td>13.0 kW</td><td>767</td><td>593</td></tr>
<tr><td>9.0 kW</td><td>771</td><td>535</td></tr>
<tr><td rowspan="4">Maximum Temperature During Vacuum Drying or Transfer Operations post in-pool or post forced air cooling when technical specification time limits are exceeded (refer to SAR Table 4.5.3-8 or Table 4.5.3-9) (°F)</td><td>17.5 kW (preferential load)</td><td>766</td><td>665</td></tr>
<tr><td>13.0 kW (preferential load)</td><td>766</td><td>612</td></tr>
<tr><td>13.0 kW</td><td>774</td><td>593</td></tr>
<tr><td>9.0 kW</td><td>771</td><td>543</td></tr>
</table>

Table 4-2
Maximum Temperatures - Normal, Off-Normal, and Accident Conditions

MPC Component	Normal Conditions (°F)			Off-Normal & Accident Conditions (°F)			
	Uniform Loading (<0.674 kW/ assembly)	Preferential Loading (17.5 kW/ canister)	Allowable (Long-Term)	Blockage of Half of Air Inlets (Off-Normal)	Extreme Environmental Conditions (Accident)	Fire (Accident)	Allowable (Short-Term)
Fuel Cladding	592	629	633 (694 for 2 center-most locations, Zircaloy-clad fuel must be cooled < 7 years)	632	649	704	806
Aluminum Heat Transfer Disk	515	534	700	538	557	609	700
Stainless Steel Support Disk	517	538	650	541	560	613	800
TSC Shell	312	312	800	315	337	387	800
VCC Liner	171	171	700	NA	NA	NA	NA
VCC Concrete	171 (local) 130 (bulk)	171 (local) 130 (bulk)	200 (local) 150 (bulk)	178	203	141	350

4.1 Conclusions

Staff has determined that the overall risk associated with a misloading event does not warrant special consideration within the thermal analysis arena because there is no adverse safety impact. A misloading, in a worst case scenario, would cause some rods to overheat and possibly result in fuel cladding rupture. However, the fuel would remain in its analyzed configuration since the rupture openings would be small for this ductile cladding and the containment boundary of the TSC would remain intact for pressure increases resulting from a postulated failure of all fuel rods. Since risk is defined as frequency multiplied by the consequence, the overall risk associated with a misloading event would be acceptable because there are no adverse consequences as long as the containment boundary remains intact.

Additionally, the staff reviewed the applicant’s proprietary calculations for the transfer cask’s thermal transient analysis and the steady-state thermal analysis for the TSC and fuel basket under normal, off-normal and accident conditions. The staff found that these calculations adequately supported the conclusions presented in the SAR. The staff performed confirmatory calculations regarding the decay heat associated with the proposed contents and concluded that the design basis heat loads were bounding. Based on the information presented in the SAR, the applicant’s analysis, the staff’s independent calculations and NRC’s position on misloading risk, the staff finds that this amendment presents reasonable assurance that the storage cask’s material temperatures will be maintained below their limits.

5.0 SHIELDING

5.1 Source Term

CY fuel consists of 15 x 15 PWR fuel assemblies manufactured by Westinghouse, Gulf Nuclear/Gulf General Atomic, NUMEC, and Babcock & Wilcox. Approximately 15 percent of the assemblies have Zircaloy-clad fuel rods, the rest have stainless-steel-clad fuel rods. The initial enrichment of the Zircaloy-clad fuel varies from 2.95 to 4.61 wt. percent ^{235}U, and the maximum burnup is 43,000 MWd/MTU. Stainless-steel-clad fuel varies in enrichment from 3.0 to 4.0 wt. percent ^{235}U and has a maximum burnup of 38,000 MWd/MTU.

The amended MPC is designed to hold up to 26 intact spent fuel assemblies, damaged fuel assemblies, individual intact and damaged fuel rods, and non-fuel assembly hardware used during plant operation. Damaged fuel assemblies, lattices, and failed rod storage canisters will be placed in DFCs prior to being placed in the TSC. Individual intact or damaged fuel rods will be installed in tubes in a reconfigured 10 x 10 array, stainless steel fuel assembly as a RFA. The DFCs and RFAs can only be loaded into the oversized fuel positions in the TSC fuel basket.

The amended MPC also has a 24-assembly basket to be used for loading Westinghouse Vantage 5H Zircaloy clad fuel.

The non-fuel hardware which may be inserted into CY fuel assemblies are either reactor control-cluster assemblies or Flow Mixer/Thimble Plug Assemblies. The reactor control-cluster assemblies consist of 20 control rods mounted on a stainless-steel spider-assembly. The 15 x 15 array has 20 guide tubes for inserting the control rods. The flow mixer was inserted into the top of the fuel during reactor operations to control the water flow in the reactor.

5.2 VCC

The VCC is comprised of a 3.5-inch thick carbon steel inner liner surrounded by 21 inches of concrete. The loaded TSC shell provides an additional 0.625 inches of stainless steel. Gamma shielding is provided by both the carbon steel and concrete. Neutron shielding is provided by the concrete. The shielding for the top of the VCC is provided by the 5 inches of stainless steel in the TSC shield-lid, the 3 inches in the structural lid, a 3.75-inch thick carbon-steel shield-plug, a 2.0-inch thick layer of either NS-4-FR or NS-3 shielding material, a 0.375-inch thick carbon-steel cover and a 1.5-inch thick carbon-steel lid. The bottom of the VCC rests on a concrete pad, so the bottom shielding is comprised of a 1.75-inch thick stainless-steel canister-bottom-plate, a 4.0-inch thick carbon-steel weldment-base-plate, and a 1.0-inch thick carbon-steel bottom-plate. The base plate and bottom plate are structural components that position the TSC above the VCC air inlets.

Fuel source-terms were determined using the SAS2H module of the SCALE code. The source term includes neutron and gamma contributions from the fuel, gamma contribution from the non-fuel hardware, and the contribution from neutron-alpha interactions with the oxygen in the fuel. Since the CY fuel inventory consists of both stainless-steel-clad and Zircaloy-clad fuel, source terms were developed for each type of fuel.

The source term for the stainless-steel-clad fuel was based on a maximum burnup of 38,000 MWd/MTU and with a minimum cooling time of 5 years. The Zircaloy-clad fuel's source-term was developed using a maximum burnup of 43,000 MWd/MTU and 5 years cooling time. The fuel assembly hardware gamma source contribution is primarily from the ^{60}Co, which is an activation product of Type 304 stainless steel, with minor contributions from ^{59}Ni and ^{58}Fe. A ^{59}Co impurity level of 0.5 g/kg was used to determine the ^{60}Co level from stainless-steel-clad fuel. A ^{59}Co impurity level of 1.2 g/kg was used to determine the ^{60}Co level from Zircaloy-clad fuel.

Analyses were performed to determine the gamma source-term contribution to the activated non-fuel hardware to be stored in the fuel assemblies. The source term for the non-fuel hardware was developed assuming that each reactor control cluster assembly was positioned in the reactor during operation so as to achieve the maximum possible activation throughout its lifetime and each flow mixer was assumed to have been in place in the top nozzle of a fuel assembly for every operating cycle.

A three-dimensional, Monte Carlo type, shielding code (MCBEND) was used to model the fuel assemblies, basket, and storage cask. Monte Carlo calculations were performed for each source type present in the different source regions. The MCBEND code allows detailed modeling of the fuel assemblies, basket, heat transfer annulus, and storage cask features, including streaming paths from VCC air inlet and outlet vents, and TSC drain ports.

The neutron shield evaluation for the VCC top is based on a 1-inch layer of NS-4-FR or a 1.5-inch layer of NS-3. The dose rate calculations are conservative because there is actually 2-inches of neutron shielding (either NS-4-FR or NS-3) incorporated into the VCC shield lid.

Table 5-1 lists the maximum surface dose rates for the VCC loaded with the design basis stainless-steel-clad fuel. Table 5-2 lists the maximum surface dose rates for the VCC loaded with the design basis Zircaloy-clad fuel.

Table 5-1
Maximum Dose Rates with Design Basis Stainless-Steel-Clad Fuel

	Cask Surface (mrem/hr)		1 Meter from Surface (mrem/hr)	
Source	Side	Top	Side	Top
Neutron	2	2	1	6
Gamma	165	30	81	6
Total	167	32	82	12

Table 5-2
Maximum Dose Rates with Design Basis Zircaloy-Clad Fuel

	Cask Surface (mrem/hr)		1 Meter from Surface (mrem/hr)	
Source	Side	Top	Side	Top
Neutron	2	3	1	8
Gamma	138	33	68	6
Total	140	36	69	14

5.3 Transfer Cask

The CY-MPC transfer cask is a multi-walled radial shield with 0.75 inches of low alloy steel, 4.0 inches of lead, 2.75 inches of NS-4-FR, and 1.25 inches of low-steel alloy. The TSC shell provides an additional 0.625 inches of stainless steel. The gamma shielding is provided by the stainless steel and lead. The NS-4-FR is a solid, borated polymer which provides the neutron shielding. The bottom of the transfer cask is a solid section of 9.5 inches of low-alloy steel. The top shielding is provided by 5 inches of stainless steel in the TSC shield-lid and by 3 inches of stainless steel in the structural lid. A temporary shield of 5 inches of carbon steel is used during welding, draining, drying, and helium backfill operations.

The dose rate evaluation for the transfer cask was analyzed for various lid configurations, with and without covers over the TSC vent and drain ports, and also under either wet or dry conditions. For wet conditions, the TSC is assumed to be filled with water to the appropriate level to allow for welding the shield lid. The maximum dose rates have been identified at the top surface of the transfer cask for both stainless-steel-clad and Zircaloy-clad fuels. The maximum dose rate for the stainless-steel-clad fuel is 3,830 mrem/hr and 4,050 mrem/hr for the Zircaloy-clad fuel.

5.3 Conclusions

The staff performed confirmatory analyses of the design basis gamma and neutron source-terms for the design basis stainless-steel and Zircaloy-clad fuels. Staff used SAS2H and Origen-S of the SCALE-4.4 computer code. The results of the confirmatory analyses correlate with the results obtained by the applicant. Staff analyses resulted in a minimally higher source term, which is a result of using the SCALE 4.4 for the personal computer version and the 44GROUPNDF5 cross-section library. The staff reviewed the fuel parameters listed in the SAR and has reasonable assurance that the design basis gamma and neutron source terms are adequate for the shielding analysis.

Based upon the information provided by the applicant and the confirmatory calculations, staff has reasonable assurance that the dose rates presented in the SAR are representative of dose rates which would occur when the transfer cask and the VCC are filled with design basis stainless-steel or Zircaloy-clad fuel. The staff has reviewed the input to the codes used by the applicant to determine the source terms and dose rates for the VCC and the transfer cask and has found that the code appears to have been used appropriately for this system.

6.0 CRITICALITY

The staff reviewed the SAR criticality analysis to determine whether all credible normal, off-normal and accident conditions had been addressed and whether potential consequences on criticality met the following regulatory requirements: 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c) and 72.236(g). The amendment was also reviewed to determine whether it fulfilled the following acceptance criteria listed in Section 6 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems:"

a. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95% confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

b. At least two unlikely, independent and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed possible.

c. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons) or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.

d. Criticality safety of the cask system should not rely on use of the following credits:
   - Burnup of the fuel;
   - Fuel-related burnable neutron absorbers; and
   - More than 75% for fixed neutron absorbers when subject to standard acceptance tests.

Observations and conclusions from the staff review are summarized below.

6.1 Criticality Design Criteria and Features

Criticality safety of the amended MPC basket depends on the geometry of the fuel basket and the use of fixed Boral panels for absorbing neutrons. The basket features square fuel tubes, each with Boral panels fixed to the four outer-walls. The primary design parameters that ensure subcriticality are the minimum flux-trap width between fuel tubes and the minimum ^{10}B content of 0.02 g/cm^2 in each of the Boral panels. The flux trap widths in the MPC basket vary from 1.00 to 3.50 inches. SAR Section 6.3 provides sketches of the MPC basket.

The most reactive credible configurations of the MPC system occur when the cask is flooded with water. The MPC system does not rely on borated water as a means of criticality control. Therefore, the MPC remains subcritical when flooded with pure water.

The staff verified that the design-basis off-normal and postulated accident events would not adversely affect the design features important to criticality safety. In the flooded normal configurations with waterlogged fuel rods, the maximum system reactivity was determined to be identical to, or bounded by, the credible configurations resulting from an off-normal or accident event. Based on the information provided in the SAR, the staff concludes that the design of the MPC basket within the MPC system meets the double-contingency requirements of 10 CFR 72.124(a).

6.2 Fuel Specification

The MPC system is designed to transfer and store up to 26 CY spent fuel assemblies enriched up to 3.93 wt% ^{235}U or up to 24 CY spent fuel assemblies enriched up to 4.61 wt% ^{235}U. Seven CY fuel assembly vendor categories, which are presented in SAR Table 2.1-3, constitute the allowed fuel contents. In addition, up to four CY RFAs or DFCs are permitted. The MPC system may contain fuel assemblies with either stainless-steel or Zircaloy cladding.

The applicant condensed the seven vendor categories into four basic fuel groupings. The staff reviewed these groupings and agrees that they are acceptable. SAR Table 6.2.2-1 lists the fuel parameters used for each of the fuel groupings. In that table the basic fuel parameters are as follows:

- maximum initial enrichment;
- maximum MTU;
- cladding material (Zircaloy or Stainless Steel);
- number of fueled rods;
- maximum fuel rod pitch;
- maximum pellet diameter;
- maximum active fuel length; and
- minimum rod outer diameter and minimum cladding thickness.

SAR Table 6.2.2-2 provides the reconfigured fuel assembly parameters.

Specifications on the condition of the fuel are also included in the MPC amendment, including the proposed Technical Specifications in Chapter 12. The MPC system is designed to accommodate intact fuel assemblies, DFCs containing damaged fuel or fuel debris, and RFAs as defined in SAR Chapter 12. The four oversized corner locations of the amended MPC basket are designed to store RFAs or DFCs.

The staff reviewed the fuel specifications considered in the criticality analysis and verified that they bound the fuel specifications presented in SAR Chapters 2 and 12. The staff verified that fuel assembly parameters important to criticality safety have been included in the proposed Technical Specifications.

6.3 Model Specification

6.3.1 Configuration

The applicant modeled the amended MPC basket configuration, including the basket tubes, Boral sheets, and water gaps. Sketches of the MPC basket are provided in SAR Section 6.3. The sketches are based on the engineering drawings provided in SAR Section 1.5.

The calculational models conservatively assumed the following:

- fresh fuel isotopics;
- 75% credit for the ^{10}B loading in the Boral panels; and
- flooding of the fuel rod-gap regions with pure water.

The MPC system was modeled using periodic axial and reflective boundary conditions to simulate an infinite array of casks.

For intact fuel assemblies, the applicant modeled the fuel rods and homogenized the end-fitting regions. A number of parametric cases were analyzed to determine the most reactive model for intact spent fuel. The staff reviewed the applicant's methods, calculations and results for determining the most reactive fuel assembly; the worst-case manufacturing tolerances; and the most reactive assembly shift within the basket. The applicant used minimum tolerances in the Boral width and the maximum fuel tube opening to model the system as more reactive than with the use of nominal dimensions. In addition, the assemblies were shifted towards the center, which was found to be more reactive than the case with the assemblies centered in the basket compartments. This case increases the interaction between the assemblies in the basket, which is expected to increase the reactivity.

The applicant also evaluated missing fuel rods from the CY fuel assemblies. This case used the most reactive 24-assembly basket configuration loaded with Zircaloy clad fuel assemblies enriched to 4.61 wt% ^{235}U. The analysis removed up to 24 rods from each assembly, which in effect optimized the H/U ratio and resulted in an increase in system reactivity. The applicant further evaluated this case limiting the assemblies with missing rods to only the four corner locations. The calculation demonstrates that loading assemblies with missing rods in corner locations does not significantly effect the overall reactivity of the MPC system.

To bound the most reactive configuration of a CY-DFC, the applicant modeled the CY-MPC 24-assembly basket loaded with 20 fuel assemblies enriched to 4.61 wt% ^{235}U with four DFCs in each corner location. Each DFC was modeled as a homogeneous mixture of fuel and water and the volume fractions of fuel/water were varied. To bound the most reactive configuration of an RFA, the applicant's model uniformly distributed the maximum allowed mass of the most reactive fuel material within each of the RFA's 100 tubes. The analysis demonstrates that loading the DFCs or RFAs in the corner locations does not significantly impact the overall reactivity of the MPC system.

The analysis considered varying moderating conditions from full density to reduced density water, including uneven drain-down conditions between the DFCs and the basket.

The staff reviewed the applicant's models and agrees that they are consistent with the design descriptions in SAR Chapters 1 and 2, including engineering drawings. The staff also reviewed the applicant's analyses and results. Based on the information presented, the staff agrees with the applicant's conclusions.

The staff performed confirmatory analyses using the information provided in the MPC amendment, including drawing numbers: 414-881, Rev. 2; 414-882, Rev. 2; 414-893, Rev. 1; 414-894, Rev. 0; 414-895, Rev. 2; 414-901, Rev. 0; 414-902, Rev. 1; 414-903, Rev. 1; and 414-904, Rev. 0. The staff's fuel assembly models were based on the fuel parameters given in SAR Table 2.3-1, Table 2.3-2, Table 6.2.2-1, Table 6.2.2-2 and SAR Section 12. The staff results were comparable to those of the applicant.

6.3.2 Material Properties

The compositions and densities for the materials used in the criticality safety analysis computer models are provided in SAR Section 6.3.2 of the MPC amendment. The minimum required areal density of the ^{10}B in the fixed neutron poison plates is 0.02 mg/cm^2 for the boral. The calculations modeled 75% of the ^{10}B. SAR Section 9.1.6.1 of the MPC amendment shows the acceptance tests for the boral.

The compositions and densities for the materials in the computer models were reviewed by the staff and determined to be acceptable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant used the MONK8A and the accompanying JEF 2.2-based point energy neutron library for the criticality analysis and benchmark calculations.

The staff performed confirmatory analysis with the CSAS modules of the SCALE Version 4.4 computer codes and the accompanying 27 group cross-section library for the MPC basket analysis.

The MONK8A and SCALE codes are both acceptable for performing criticality analyses. The staff agrees that the codes and cross-section sets used are appropriate for this particular application and fuel system.

6.4.2 Multiplication Factor

Results of the applicant's criticality analysis show that k_{eff} of the MPC system will remain below 0.95 for all allowed fuel loadings. The staff reviewed the applicant's calculated k_{eff} values and Upper Subcritical Limit (USL) and agrees that these values have been appropriately calculated to include all biases and uncertainties at a 95% confidence level or better.

The staff performed independent calculations using SCALE4.4 to confirm the applicant's analysis. Results of the staff's confirmatory calculation were in close agreement with the applicant's results.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the MPC system will remain subcritical with an adequate safety margin under all credible normal, off-normal, and accident conditions.

6.4.3 Benchmark Comparisons

The applicant performed benchmark comparisons on critical experiments that were selected to bound the variables in the MPC design. The three most important parameters are the ^{10}B loading of the neutron absorbers, the flux-trap size, and the fuel enrichment. Parameters such as reflector material and spacing, fuel pellet diameter and fuel-rod pitch were also considered in selecting the critical experiments. The staff found no significant trends in the bias.

The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data on the same computer hardware used in the criticality calculations. The staff reviewed the benchmark comparisons in the SAR and agrees that the CSAS module of the SCALE computer codes used for the analysis was adequately benchmarked to representative critical experiments.

A USL of 0.9425 was calculated by the applicant. The USL incorporates 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 reviewed the applicant's method for determining the USL and found it to be acceptable and conservative. The staff also verified that only biases that increase k_{eff} were applied.

6.5 Conclusions

Based on the staff's review of the amended MPC system and the staff's own confirmatory analyses, the staff concludes that the amended MPC system meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds the following:

– Structures, systems and components important to criticality safety are described in sufficient detail in Chapters 1, 2, and 6 of the MPC amendment and on the design drawings to enable an evaluation of their effectiveness.

– The MPC system is designed to be subcritical under all normal, off-normal, and accident conditions.

– The criticality design is based on favorable geometry and fixed neutron poisons in the basket. An appraisal of the fixed neutron poisons has shown that they will remain effective for the 20-year storage period.

– The staff concludes that the criticality design features for the MPC system with the CY fuel configurations 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 system will allow safe storage of the CY spent fuel. This finding is reached on the basis of a staff review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7.0 CONFINEMENT

The confinement review 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. The staff reviewed the information provided in the SAR to determine whether the amended MPC system fulfills the following acceptance criteria:

– The SAR must describe the confinement SSCs important to safety in sufficient detail to facilitate evaluation of their effectiveness [10 CFR 72.24(c)(3) and 10 CFR 72.24(l)].

– The design must adequately protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures during storage, or the fuel must be confined through other means such that fuel degradation during storage will not pose operational safety problems with respect to removal of the fuel from storage [10 CFR 72.122(h)(1)].

– The cask design must provide redundant sealing of the confinement boundary [10 CFR 72.236(e)].

– Storage confinement systems must allow continuous monitoring, such that the licensee will be able to determine when to take corrective action to maintain safe storage conditions [10 CFR 72.122(h)(4) and 10 CFR 72.128(a)(1)].

– The design must provide instrumentation and controls to monitor systems that are important to safety over anticipated ranges for normal and off-normal operations. In addition, the applicant must identify those control systems that must remain operational under accident conditions [10 CFR 72.122(i)].

– The applicant must estimate the quantity of radionuclides expected to be released annually to the environment [10 CFR 72.24(l)(1)].

– The applicant must evaluate the cask and its systems important to safety, using appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions [10 CFR 72.24(d)].

- Systems, structures, and components important to safety must be designed to withstand the effects of hypothetical accident conditions and severe natural phenomena without impairing their capability to perform safety functions [10 CFR 72.122(b)].

- During normal operations and anticipated occurrences, the annual dose equivalent to any individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ [10 CFR 72.104(a)].

- From any design basis accident, an individual at or beyond the controlled area boundary may not receive the more limiting of the following conditions: (1) the total effective dose equivalent must not exceed 5 rem, or (2) the sum of the deep-dose equivalent plus the committed dose equivalent to any organ may not exceed 50 rem. Additionally, the shallow dose equivalent to the skin or any extremity shall not exceed 50 rem, and the lens dose equivalent shall not exceed 15 rem [10 CFR 72.106(a)].

7.1 Confinement Design Characteristics

The staff reviewed the applicant's confinement analyses in SAR Chapter 7 and the drawings in SAR Chapter 1. The applicant has clearly identified the confinement boundary. The confinement boundary for the amended MPC system is the same as that for the previously approved MPC system. The amended MPC confinement boundary includes the TSC shell, bottom baseplate, shield lid (including the vent and drain port cover plates), and the associated welds. There are no bolted closures or mechanical seals in the primary confinement boundary. The TSC is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code, Section III, Subsection NB. Exceptions to the ASME Code, with respect to the confinement boundary, are identified in Table 12B3-1 of the SAR. The shield lid (with the vent and drain port cover plates welded to the lid) and the structural lid are independently welded to the upper part of the TSC shell. This design provides redundant sealing of the confinement boundary and satisfies the requirements of 10 CFR 72.236(e). The design, testing, inspection, and examination of the welds forming the confinement boundary are described in detail in Section 7.1.3 of the SAR.

The staff reviewed the cask vacuum drying and backfilling procedures that are used during loading operations. The procedures require that a vacuum pressure of 10 mm Hg be maintained for 10 minutes without the aid of vacuum equipment, followed by an excursion to 3 mm Hg, helium backfill to 0 psig, and another excursion to 3 mm Hg, to ensure that an acceptably low amount of water and potentially oxidizing material remain in the TSC. The combination of the all-welded cask design and the use of these procedures will ensure that both the cladding and the confinement boundary integrity are maintained during normal, off-normal, and hypothetical accident conditions.

The staff also reviewed the applicant's helium leak testing procedures. A leak test of the shield lid is performed to an "as tested leakage rate" of 2×10^{-7} cm^3/sec (helium), which corresponds to a reference air leakage rate less than the leaktight standard of 1×10^{-7} ref·cm^3/sec. The tests will be performed in accordance with ANSI N14.5-1997 using a helium leak detector having a sensitivity of 1×10^{-7} cm^3/sec (helium). During the test, the TSC cavity will be pressurized with helium to one atmosphere through the vent port quick disconnect valve. All fittings and connectors used to attach the helium leak testing equipment will be tested to ensure that (1) there is minimal leakage from these sources, and (2) the shield lid-to-shell weld can be tested to leaktight criteria of ANSI N14.5-1997. Any indication of a leak is unacceptable and repair of the leak will be done in accordance with ASME Code Section III. The helium leak test also confirms that the amount of helium lost from the TSC over the license period, at a leakage rate of $2x10^{-7}$ cm^3/sec (helium), would be less than 1% of the initial amount of helium. Thus, there will be an adequate amount of helium in the TSC to maintain an inert atmosphere and the heat removal capability over the lifetime of the cask.

For normal conditions of storage, the TSC relies on the fuel cladding and the TSC shell cavity as multiple confinement barriers to assure that there is no release of radioactive material to the environment. The TSC is backfilled with an inert gas (helium) to protect against degradation of the cladding. As discussed in Sections 3 and 11 of this SER, there is reasonable assurance that the confinement boundary maintains its structural integrity during normal, off-normal, and hypothetical accident storage conditions. Further, Section 4 of this SER shows that the peak confinement boundary component temperatures and pressures are within the design basis limits for normal conditions of storage. The integrity of the TSC confinement boundary is assured through (1) nondestructive examinations (NDE), including multiple surface and/or volumetric examinations, of the TSC shield lid, structural lid, and vent and drain port cover plate welds; (2) leakage rate testing; and (3) pneumatic testing. The TSC inspection and test acceptance criteria are described in Section 9.1 of the SAR. TSC closure weld examination and acceptance criteria are described in detail in Section 9.1.1.

The staff concludes that the all-welded construction of the TSC with redundant welded shield and structural lids and associated inspection and testing programs ensure that no release of radioactive material will occur under normal, off-normal, and hypothetical accident conditions.

7.2 Confinement Monitoring Capability

For cask systems having canisters with seal weld closures, continuous monitoring of the weld closures is unnecessary because there is no known plausible, long-term degradation mechanism which would cause the seal welds to fail. However, other licensee monitoring programs, including periodic surveillance, inspection, and radiological and environmental surveys, will ensure that the operating controls and limits are met to maintain safe storage conditions.

7.3 Nuclides with Potential for Release

The confinement boundary of the TSC is designed to be leaktight (i.e., maximum allowable leakage rate of $1x10^{-7}$ ref·cm^3/sec) in accordance with ANSI N14.5-1997. In this consensus standard, the definition of leaktight (e.g., “a degree of package containment that, in a practical sense, precludes any significant release of radioactive materials”) precludes the need for the applicant to determine the releaseable radiological source term and the corresponding dose consequence. Therefore, the staff concludes that it was unnecessary for the applicant to specify the source term for the confinement analyses.

7.4 Confinement Analysis

The confinement boundary is completely welded, and the stresses, temperatures, and pressures of the TSC are within the design basis limits under normal, off-normal, and hypothetical accident conditions. The TSC is vacuum-dried and backfilled with helium gas prior to final canister closure, so there is no potential for an increase in the canister pressure or degradation of the cladding due to radiolytic decomposition or other adverse reactions.

The staff concludes that (1) no discernable leakage of radioactive material from the TSC is credible, (2) the dose consequence due to leakage of radioactive material from the all-welded canister is negligible, and (3) the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(a) are met.

7.5 Supplemental Information

Supplemental information, or documentation, in the form of justifications of assumptions and analytical procedures were provided as requested to complete this review.

7.6 Conclusions

The staff concludes that the design of the MPC adequately protects the spent fuel cladding against degradation that might otherwise lead to gross rupture. The design of the MPC provides redundant sealing of the confinement system closure joints using dual welds on the TSC shield and structural lids. The staff also concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. The MPC satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b). The staff finds that the design of the confinement system of the MPC 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 will allow safe storage of spent fuel. This finding is based on the applicant's compliance with the regulations, use of appropriate regulatory guides, applicable codes and standards, the applicant's analysis, the staff's confirmatory analysis, and accepted engineering practices.

8.0 OPERATING PROCEDURES

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

- Loading the MPC system;
- Loading and closing the TSC;
- Loading the VCC;
- Transport and placement of the VCC;
- Removal of the TSC from the VCC; and
- Unloading the 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.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

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

- Visual and nondestructive examination inspections;
- Nondestructive weld examinations;
- Fabrication inspections;
- Structural and pressure tests;
- Neutron absorber tests; and
- 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.

10.0 RADIATION PROTECTION

The staff evaluated the changes made to the MPC design to accommodate CY fuel to ensure regulatory dose requirements are met. The regulatory requirements for providing adequate radiation protection to site licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d).

10.1 Occupational Exposures

Occupational radiation exposures from cask loading, normal operation, and fuel unloading are based upon the calculated dose rates from SAR Chapter 5. Occupational exposures from the MPC loaded with CY fuel will be higher than exposures previously analyzed for the Yankee Class fuel because the source term from the fuel is higher. Due to the higher surface dose rates on the VCC surface, the use of automated equipment will be used to compensate for the higher dose rates. Using automated equipment should reduce exposure time during canister transfer operations and lid-welding operations. Additionally, three-dimensional analyses were used to determine the occupational dose rates, rather than the one-dimensional evaluation performed for the Yankee Class fuel.

Once the MPC is loaded and in place on the ISFSI pad, maintenance and surveillance activities will be minimal. A daily electronic measurement of the ambient air and outlet temperatures for each VCC will be made and the results will be reviewed by an operator located away from the cask storage location. By employing a remote readout of the temperature measurement, the operator is not expected to incur a dose.

The storage array for the CY fuel was evaluated assuming a mixture of 5- and 10-year cooled stainless-steel fuel, with 32 casks containing 10-year cooled fuel and eight casks containing 5-year cooled fuel. Dose rates from the SKYSHINE code were used to determine the annual exposure from routine operations of the 40-cask array. Annual operations and surveillance requirements result in an estimated annual collective exposure of 1.54 person-rem for the ISFSI. Actual occupational exposure to ISFSI workers is required to be measured and documented in accordance with the facility's Radiation Protection Program, and actual occupational exposures must be within the dose-rate limits specified in 10 CFR Part 20.

10.2 Dose to Members of the Public

Based on the containment evaluation of SAR Chapter 7, no effluents are expected from the MPC during normal, off-normal, or accident conditions because of the leaktight configuration of the TSC. Therefore, direct radiation (including skyshine) is the primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions.

The 40 cask array was modeled using the SKYSHINE-III code. Surface-source emission-fluxes were provided from the MCBEND shielding evaluations. Annual exposures, based on a 8,760-hour/year duration, were evaluated. To maintain an annual boundary dose rate in compliance with the regulatory limit of 25 mrem/year or less, the site controlled-area boundary for the MPC with bounding CY fuel loaded would have to be 426 meters from the cask top and bottom sides and at a radius of 455 meters from the sides of the casks.

The MPC user is required to have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and must demonstrate compliance with dose limits to individual members of the public, as required by 10 CFR Part 20, Subpart D, by evaluations and measurements.

Chapter 11 of the SAR contains a description of accident conditions and natural phenomena events which could affect the ISFSI. Damage to the MPC cask from a high-energy tornado-missile impact is estimated to reduce the concrete shielding thickness at the point of impact by 6 inches. This reduction in shielding results in a calculated surface dose of 1000 mrem/hr for the MPC loaded with CY fuel, which is less than the regulatory limit of 5 rem to the whole body or any organ as specified in 10 CFR 72.106.

10.3 Conclusions

The staff reviewed the estimated occupational exposures and found them to be acceptable. The occupational exposure dose estimates provide reasonable assurance that occupational limits in 10 CFR Part 20, Subpart C can be achieved. Actual occupational doses will depend on site-specific parameters, including special measures taken to maintain exposures as low as is reasonably achievable. The licensee is required to have an established radiation protection program, as required in 10 CFR Part 20, Subpart B. In addition, each licensee must demonstrate compliance with all dose limits in 10 CFR Part 20, 10 CFR Part 72, and any site-specific 10 CFR Part 50 license requirements with evaluations prior to loading of the casks.

The staff evaluated the public dose estimates from direct and reflected radiation from normal and off-normal (anticipated occurrences) conditions and found them to be acceptable. The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by the licensee.

The staff evaluated the public dose estimates from direct radiation from accident conditions and natural phenomena events and found them acceptable. The staff has reasonable assurance that the effects of direct radiation from bounding design basis accidents and natural phenomena will be below the regulatory limit of 5 rem specified in 10 CFR 72.106(b).

11.0 ACCIDENT ANALYSES

SAR Chapter 11 provides accident analyses of the amended MPC. To evaluate structural performance, the analyses with applicable design basis loads and structural details for the approved MPC series were repeated for the amended MPC. For off-normal conditions and accident events, the SAR evaluations include off-normal handling load, severe environmental conditions, accident pressurization, explosion, VCC 6-inch drop and tipover. For natural phenomena, the SAR addresses loads associated with design basis floods, tornado wind and tornado driven missiles, earthquake, and snow and ice. As discussed in SER Section 3, these analyses demonstrate that the amended MPC is structurally adequate to satisfy the requirements of 10 CFR Part 72 to provide for safe storage of spent fuel under design basis off-normal, accident, and natural phenomenon events.

12.0 CONDITIONS FOR CASK USE - TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised in their entirety to include the new fuel type specifications and operational limits, and to follow the standard technical specification format in NUREG-1745, “Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance.” The conditions in Table 12-1 were removed from the Technical Specifications and moved to SAR Chapter 8, “Operating Procedures.”

Previous Technical Specification	Description	SAR Chapter 8 Requirement
A4.5.1.1	Spacing of MPC systems shall be 15 feet center-to-center.	8.1.3.10
A4.5.1.2	Helium shall have a minimum purity of 99.9%.	8.1.1.32a 8.1.1.34
A4.5.2	Minimum distance from master link of the canister lifting slings to the top of the canister shall be 67 inches.	8.1.2.10 8.2.5
A5.3	Heat transfer characteristics of the MPC will be recorded by temperature measurements of the first MPC placed in service with a heat load equal to or greater than 7.5 kW.	*

* - SAR Section 8.1.3.13 requires cask users to install and connect temperature monitoring equipment and to verify its operation after placement of the cask on the concrete storage pad. Temperature measurements are required to be taken at the VCC outlet vents for all loaded casks in accordance with Technical Specification A 3.1 to ensure that the temperature difference between the ambient and the outlet vent air does not exceed 110°F.

13.0 CONDITIONS

The Certificate of Compliance Conditions have been renumbered and revised to reflect the standardized certificate format, expand the use of the MPC system to include storage of CY fuel, make the conditions more accurate, and eliminate duplication.

14.0 CONCLUSIONS

14.1 Overall Conclusion

The staff has reviewed the amendment to the SAR for the NAC-MPC 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 NAC-MPC system meets the requirements of 10 CFR Part 72.

14.2 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 are acceptable. However, the staff did not review any methodologies used in the original 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. 2,
on May 30, 2002.