NAC INTERNATIONAL

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July 27, 2001

U.S. Nuclear Regulatory Commission
11555 Rockville Pike
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Attn: Document Control Desk

Subject: Request for an Amendment of the Certificate of Compliance (CoC) for the NAC-UMS® Universal Storage System to Incorporate Enhanced Design Features

Docket No. 72-1015

Reference:
1. Certificate of Compliance No. 1015, Amendment 1, for the NAC International UMS® Universal Storage System, United States Nuclear Regulatory Commission (NRC), February 20, 2001
2. Final Safety Analysis Report (FSAR) for the UMS® Universal Storage System, Amendment 1, NAC International, May 2001

NAC International (NAC) herewith requests that Reference 1 (CoC for the UMS® Universal Storage System) be amended to incorporate enhanced design features, as described herein and documented in the enclosed proposed FSAR. This amendment is being requested to support implementation of the UMS® Universal Storage System at the Maine Yankee Nuclear Plant, the Palo Verde Nuclear Plant, the McGuire Nuclear Plant and other sites where dry storage may be implemented.

This submittal includes ten copies of the request for the amendment and the complete UMS® FSAR Amendment 1, with Revision UMSS-01C changed pages inserted. Complete books are provided to support an efficient review of the submittal, because a large number of changed pages are involved.

The Revision UMSS-01C changed pages, which incorporate the requested amendment, have been prepared in accordance with the following conventions:

- The changed pages for this submittal are designated as Revision UMSS-01C to provide a unique identification of the pages and changes.
- Revision bars are used in the page margin to indicate changes. Revision bars are not used to indicate text flow. All previous revision bars on the Amendment 1 pages have been deleted, so that only the revisions associated with this amendment request are marked.
- All of the pages in the List of Effective Pages are designated Revision UMSS-01C, but no revision bars are used on those pages.
- All of the pages in Chapter 12 are designated Revision UMSS-01C to incorporate the Standardized Technical Specifications format, but no revision bars are used on those pages.

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The significant changes incorporated into the NAC-UMS® SAR in this proposed revision are:

- Revised Technical Specifications to incorporate the Standardized Technical Specifications format of NUREG-1745
- Added optional 100-Ton Transfer Cask (TFR) Design
- Added optional poison material, i.e., METAMIC
- Increased BWR fuel assembly weight (from 696 pounds to 702 pounds)
- Revised fuel assembly dimensions (length and width) for more comprehensive coverage (BWR and PWR)
- Revised thermal analyses and extended operating time limits for vacuum drying and helium backfill (BWR and PWR) and for the closed transportable storage canister in the transfer cask
- Revised allowable temperatures for aluminum basket components to reflect aluminum creep testing (includes revised ANSYS models)
- Included optional inlet supplemental shield and revised concrete cask pedestal baffle
- Determined maximum permissible enrichment for criticality analysis (loading of 5 wt. % enriched ^{235}U with 1000 ppm soluble boron)

This submittal includes revisions to the UMS® Universal Storage Cask Licensing Drawings, many based on fabrication experience at Maine Yankee. The drawing changes do not affect the form, fit or function of the components, and they do not change the component designs, as analyzed in the FSAR. The detailed descriptions of the drawing revisions are provided in Attachment A.

A list of the significant revisions incorporated in this proposed amendment is provided in Attachment B. Also, administrative/editorial changes were made throughout the proposed FSAR, as appropriate.

If you have any comments or questions, please contact me directly at (678) 328-1321.

Sincerely,

T. C. Thompson

Thomas C. Thompson
Director, Licensing
Engineering & Design Services

Enclosures

cc: P. Plante - MY (w/o Enclosure)
T. Williamson –MY (w/o Enclosure)

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ATTACHMENT A

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-501, Revision 3 — Canister/Basket Assembly Table, NAC-UMS®

- Replace the words "OUT OF SPEC" in three places

Drawing 790-559, Revision 4— Assembly, Transfer Adapter, NAC-UMS®

- Sheet 3, Zone B3, Change dimension IS) 2.35 WAS) .35
- Sheet 3, Zone B3, Change dimension IS) 4.75 WAS) 6.75

Drawing 790-560, Revision 9 — Assembly, Transfer Cask (TFR), NAC-UMS®

- Change Delta Note 16 IS) Cut one... WAS) Shear one...
- Change dimension callout Sheet 4, Zone E-7 IS) (1.2) cut plane... WAS) 1.2 shear plane...
- In general Note 2, add the words "load bearing" between accessible and welds
- Re-dimension Item 36, Gamma Shield Brick
- Assy –99, Add qty (4) 5/8" diameter dowel pins to the bill of matl', com'l grade, st.stl., 2" long
- Add Delta Note for installation of the dowel pins as follows: Locate at assembly, with doors centered, fully closed, and with the dowel pin in contact with the Transfer Cask Base/Outer shell. Maintain a .25" minimum outer edge distance. Drill and ream for press-fit 1.0 deep into Item 29.
- Add Delta Note as follows: Grind transition chamfers on the leading and trailing edges of the 2.3" x 2.0" shoulder 1/8" x 30-45 degrees. Add this Delta Note to assy's –93 and –94.
- Add Delta Note as follows: Grind transition chamfers on the leading and trailing edges of the Transfer Cask door rail (Item 16) 1/8" x 30-45 degrees. Add this Delta Note to Item 16 detail.
- Sheet 5 of 5, Zone D7 and D4, change diameter .53" to .61"
- DCR 790-560-7A, Delete Item 1 under Description of Requested Change
- Note 2, change "ACCESABLE" to "ACCESSIBLE"
- Sheet 4 of 5, Zone E-8, Change "9°" to "8.9°"
- Add the following to Delta Note 15: Alternatively information may be steel stamped/engraved onto an 11 gauge stainless steel sheet and seal welded all around to the outer shell
- Item 41, Transfer Cask Extension, change Ø77.4 to Ø77.2
- Sheet 4 of 5, Zone A-8, Item 19, Modify dimension IS) 5.0 WAS) 6.7

ATTACHMENT A

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-561, Revision 7 — Weldment, Structure, Vertical Concrete Cask (VCC), NAC-UMS®

- Revise dimension in Zone E-8, Sheet 2 IS) 135.4 WAS) 136.0
- Revise dimension in Zone D-8, Sheet 2 IS) 29.2 WAS) 29.5
- Revise dimension in Zone E-5, Sheet 3 IS) 43.4 WAS) 43.7
- Revise dimension in Zone E-7, Sheet 2 IS) (43.4 TYP) WAS) (43.7 TYP)
- Add Hex Nut and revise assembly –93 "Outlet Weldment"
- Sheet 1, add threaded holes and detail F-F
- Sheet 2, Zone B8 IS) 4x 1-8 UNC-2B OPTIONAL WAS) 4x 1-8 UNC-2B
- Update BOM to include Hex Nut as Item 31
- Revise two dimensions in Assembly –93 "Outlet Weldment" that were inadvertently changed on DCR No. 790-561-5D: Change "17.86" back to "17.9" and "11.61" back to "11.6"
- Add to BOM Item 33; Qty: 1; Name: Baffle Assembly; Material: Blank; Spec: Blank; Drawing No.: 790-614-99; Description: Blank
- Add to BOM Item 34; Qty: 4; Name: Supplemental Shielding; Material: Blank; Spec: Blank; Drawing No.: 790-613-99; Description: Blank
- Add Detail G-G to Sheet 3
- Add Delta Note 10 to read: "Tack weld 3 places approximately equal spaced." Add Delta Notes to weld callouts on Detail G-G of Sheet 3
- Add BOM Item 32: Qtys: A/R; Name: Coatings; Material: Blank; Spec: COML; Drawing No.: Blank; Description: See Note 3
- Sheet 2, Zones D4 and A4, is "Detail E-E" was "Detail D-D"
- Add Delta Note 11 note to read: "Items 33 and 34 can be used at the customers request. Item 15 shall be replaced with Item 35 when Item 33 is utilized."
- Add to BOM Item 35; Qty: 1; Name: Stand; Material: Carbon Steel; Spec: ASTM A36; Drawing No.: Blank; Description: 1/2 Plate
- Add Delta Note 12 to read: "Shim plates may be utilized to facilitate field welding operations on one or both sides of the Supplemental Shielding Weldment. Locate welds approx. as shown." Add Delta Note to weld callout to Sheet 2 Zone B6

Drawing 790-562, Revision 8 — Reinforcing Bar and Concrete Placement, Vertical Concrete Cask (VCC), NAC-UMS®

- Change dimension Sht1, Zone A-7 IS) ∅136.0 +3/4, -1/2 WAS) ∅136.0
- Add tolerance of +/- 1" to Rebar table shown on Sheet 3
- Change description in the BOM for item 16 IS) 4x4 milling grade wire cloth WAS) 5 mesh .025 welded wire

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-562, Revision 8 — Reinforcing Bar and Concrete Placement, Vertical Concrete Cask (VCC), NAC-UMS® (continued)

- Sheet 3 – Add angle of bends for Items 10 and 11 to read 45 ± 5 degrees
- Sheet 3 – Revise the note on Items 7, 8 and 9 to refer to both Notes 5 and 7
- Sheet 3 – Add a dimension for the length of the top 90 degree turndown hook for Items 2, 4 and 6 to be 4 inches
- Sheet 1 – Add note as follows: "Location of horizontal reinforcement (hoop bars) can be reversed with the vertical reinforcement, both inside and outside reinforcement curtain"
- Sheet 3 – Revise dimension in Zone A-8 IS) ∅85± 1/2 inside face, ∅86-1/2±1/2 outside face WAS) ∅85±1/2
- Sheet 3 – Revise dimension in Zone A-6 IS) 128-1/2±1/2 inside face WAS) ∅1281/2±1/2
- 790-562 Sheet 3, add for returns after radius for Items 2, 4 and 6 to be 4" minimum (NOTE: This item previously addressed as Item 3 on DCR 790-062-9B)
- Sheet 2 of 4, three locations (Zone 3F, 3E/D and 5E), change rebar location tolerance: WAS: ±1/2 IS: ±1
- Sheet 4 of 4, Zone 2C, Replace Items 1, 3 and 5 rebar spacing note with the following; "SPACING BETWEEN REBAR ITEMS 1, 3 AND 5 ARE AS FOLLOWS: ITEMS 1 AND 5 @ 5" ± 1", ITEM 5 AND 5 @ 5" ± 1", ITEM 3 AND 3 @ 8" ± 1" AND BETWEEN ITEM'S 3 AND 1 @ 8" ± 1". ITEM 3 TO BE LOCATED WITH RESPECT TO AXIS AS SHOWN."
- Sheet 4 of 4, Zone 5C, change Items 2, 4 and 6 location tolerance: WAS: ± 1/2 IS) ± 1
- Add to Note 1: REINFORCEMENT LATERAL SPACING TO BE IN ACCORDANCE WITH ACI 117 (± 1")
- Sheet 1 of 4, Change Note 1, 2nd sentence to read: "Reinforcement placement shall be in accordance with ACI 117-90 tolerances
- Sheet 1 of 4, Change Note 4 to read: "A 3 inch concrete cover shall be maintained for reinforcement at the exterior concrete surfaces, 2 inch concrete cover between the cask liner and the reinforcement, and 3/4 inch concrete cover between the other non-exposed surfaces and the reinforcement unless otherwise noted, in accordance with the tolerances as allowed by ACI 117-90."
- Sheet 2 of 4, Section A-A, delete in 6 places "MIN." Add the following to 3/4 MIN TYP, "SEE NOTE 4".
- Sheet 2 of 4, Section B-B, delete 3 places "± 1"
- Sheet 4 of 4, OUTER HOOPS AND VERTICALS AND VENT REINFORCEMENT note, delete "to within ± 1." INNER HOOPS AND VERTICALS note, delete 4 places "± 1".
- Sheet 4, C-2: revise note to read as follows: "Spacing between rebar Items 1, 3 and 5 are as follows: Items 1 and 5 equally spaced between air outlets. Spacing may deviate to clear nelson studs provided two Item 1 and two Item 3 are installed for each quadrant. Item 3 equally spaced between the Item 1 bars such that five bars are installed under each air outlet."

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-562, Revision 8 — Reinforcing Bar and Concrete Placement, Vertical Concrete Cask (VCC), NAC-UMS® (continued)

- Sheet 4, C-4: revise note as follows: "Items 2, 4 and 6 are to be located equally spaced per quadrant (14 bars/quadrant) as shown on drawing, 56 places total."
- Sheet 4, C-2: correct typo to change 3 to 5: Spacing between rebar Items 1, 3 and 5 are as follows: Items 1 and 6 equally spaced between air outlets. Spacing may deviate to clear nelson studs provided two Item 1 and two Item "5" are installed for each quadrant. Item 3 equally spaced between the Item 1 bars such that five bars are installed under each air outlet. Typo was added in DCR No.: 790-562-6B.
- Sheet 3 of 4, Change dimensions for Assembly-94
- Add Delta Note 13 to BOM Item 18
- Add Delta Note 13 to read: Item 18, Flat Washer, may be replaced as needed with a 5/8 DIA, Stainless Steel, bevel washer
- Add new Delta Note 14, Sheet 1: "Item #5 may be installed with the hook at the top of the cask. Alternately, the 90° bend may be substituted with a 180° hook."
- Add to Item 5 detail, Sheet 3, E-5, "Delta Note 14 callout"
- Add note to Item 4 on Sheet 4, E-2, "Delta Note 14 callout"
- Add new note 15, Sheet 1: "Secure the air outlets to prevent both upward displacement during concrete placement and downward displacement due to fabrication prior to concrete placement
- Revise Item 10, Sheet 3 of 4, D-6, dimensions as follows WAS) 42 1/2 ± 2 IS) 45 ± 2, detail as bend-to-bend, delete the 4 1/4 ± 2. Add new out-to-out dimension of 53 1/2 ± 2
- Revise tolerances on Item 11 to read: 13 1/2 ± 2 bend-to-bend instead of 11 ± 2; and 32 1/2 +4 1/2/ -2 (out-to-out) instead of 32 1/2 ± 2.

Drawing 790-564, Revision 5 — Shield Plug, Vertical Concrete Cask (VCC), NAC-UMS®

- Create a second, alternate lid utilizing 1.5 inches of NS-3 material for shielding
- Sheet 1, Zone D-4 IS) 64.00+.50/-.13 WAS) 64.0
- Use same tolerancing on lid created by DCR 790-564-4A

ATTACHMENT A

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-565, Revision 2 — Nameplate, Vertical Concrete Cask, NAC-UMS®

- Change Note 3 to Delta Note 3 and to read as follows, "Sheet metal may vary from 10 gauge to 18 gauge."
- Add Delta Note 3 to Item 1 of BOM
- Change Nameplate Cask No.: to read as follows, "CASK NO.: XX-VCC-YY"
- Revise Delta Note 1 to read as follows: Each nameplate for its respective cask to be uniquely identified, where XX is a unique ID for each Customer site and YY is a unique consecutive number beginning with 01 for each Customer site
- Update title block and rev. block to current NAC standard, including changing TM to ® in title
- Add ∅5/16" holes located in each corner, horizontal distance between holes to be 10.5 and vertical distance between holes to be 4.5

Drawing 790-566, Revision 1 — Assembly, 100-Ton Transfer Cask (TFR)

- Delete Item 45, Name: "Liquid Neutron Shield" on BOM and update drawing.

Drawing 790-573, Revision 7 — Support Disk and Misc. Basket Details, 56 Element BWR, NAC-UMS®

- Modify Delta Note to Item 1 as follows: WAS) -40°F IS) -50°F

Drawing 790-575, Revision 5 — BWR Fuel Tube, NAC-UMS®

- Items 3 and 4 IS) Neutron Absorber WAS) Neutron Poison and update Item/Assembly names and notes accordingly
- Replace material in BOM IS) Boral/Metamic WAS) Boral

Drawing 790-581, Revision 6 — PWR Fuel Tube, NAC-UMS®

- Items 4, 5 and 6 IS) Neutron Absorber WAS) Neutron Poison and update Item/Assembly names and notes accordingly
- Replace material in BOM IS) Boral/Metamic WAS) Boral

Drawing 790-582, Revision 7 — Shell Weldment, Canister, NAC-UMS®

- Item 7, Zone A/B-8, Reduce 1.2 to .8
- Item 7 and Item 6, Zone D-1, Move weld callout from 1.0 side to 2.5 side

ATTACHMENT A

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-584, Revision 12 — Details, Canister, NAC-UMS®

- Add Delta Note 6 to Item 7
- Modify Delta Note 6 to address the inclusion of Item 7 by revising the first sentence of Note 6 to read "May be fabricated using multiple sections."
- Item 4, Zone F4, change depth of threads from 2.50 to 2.25 min.
- Add Assembly –98, "SHIELD LID ASSEMBLY – GTCC", assembly is to be identical to Assembly –99 with exception of 3X 1-8 UNC-2B tapped holes (Note: holes are to be true positioned to align with matching holes in 790-111-96).
- Delta Note 7 IS) Items 1, 4 and 9 WAS) Items 1 and 4
- Sheet 3 of 3, for lid support ring, Item 6, remove the 0.38" bevel and show the lid support ring as a square bar
- Sheet 3 of 3, in Zone B-6, delete Detail F-F
- Sheet 3 of 3, in Zone D-5/6, delete the dashed circle and the words, "See Detail F-F"
- Sheet 3 of 3, in Zone C-7, add Delta Note 8 next to Delta Note 6
- Sheet 1 of 3, add Delta Note 8 to read "Weld preparation shall be determined by the fabricator based upon the weld process used. See Drawing 790-585 and 790-612 for effective throat size of the weld."
- Revise Sheet 1 of 3, Note 2 to read. . . Engrave Delta .5" per side and .03" deep, not to infringe on the weld bevel, and fill with weather resistant black paint
- Sheet 2 of 3, Revise Detail C-C, to reflect changed diameter of weld prep, diameter of backing bar groove, and diameter of material below backing bar groove
- Sheet 2 of 3, Revise Structural Lid, to change diameter from "65.5" to "65.1"
- Sheet 3 of 3, Revise Backing Ring, to change diameter from "64.8" to "64.4"
- Sheet 1 of 3, Add Delta Note 9 to read: "Minimum of 0.125 of material is required to be underneath bolt hole"
- Sheet 2 of 3, Section F-6, Add Delta Note 9 callout at structural lid bolt-hole callout
- Sheet 1, Assy –98 top view, rotate holes in bottom of shield lid to match view G-G
- Sheet 1, View G-G, rotate notch 180° to match Assy –98 top view
- Sheet 2, Detail E-E, Change "30°±5°" to "30°+20°-5°"
- Sheet 2, Detail E-E, Add chamfer callout for outside edge that meets up with the key: Optional" 45°±5° x .13
- Sheet 1, Zone F-6 and F-3, change 1.5 TYP to 1.5+1, -1 TYP
- Revise Sheet 3, Zone D-7, lid support ring diameter from hard dimension to reference dimension, WAS) ∅65.8, IS) (∅65.8) and add Delta Note 10 callout

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-584, Revision 12 — Details, Canister, NAC-UMS® (continued)

- Revise Sheet 1, add Delta Note 10 to read. . . Item 16, lid support ring to be fit-up and welded to shell inside diameter in a manner to assure maintenance to the 1" nominal gap
- Add Delta Note 11 as follows: Tool marks and other marks are acceptable on all unspecified machined surfaces as long as required thickness/diameter of items are met
- Revise BOM, Item 7 Name to read: Spacer Ring
- Sheet 3, Zone C4, Revise name of Item 7 to read Spacer Ring

Drawing 790-585, Revision 9 — Transportable Storage Canister (TSC), NAC-UMS®

- Sheet 1 of 2, Zone F-5, change the 5/16" partial pen with 1/8" fillet weld symbol to a 1/8" effective throat weld all around, except for key slot region, geometry optional
- Change Delta Note 9 to read as follows: "At the option of the user, Stainless Steel (ASTM/ASME A/SA 240, Type 304/304/L) Shims of appropriate thickness may be used in the welding of the shield lid (Item 17) to the shell weldment (Items 1-5)"
- Revise welding symbol, drawing zone F-5, Sheet 1, to delete 1/8" square groove portion of the symbol
- Revise Delta Note 6 to read: At the option of the user, stainless steel shims (ASME SA240/479, Type 304L) of appropriate thickness may be used in the welding of the structural lid (Item 19) to the shell weldment (Item 1 – 5). Also add Delta Note 6 callout at Zone F6, next to structural lid to shell weld callout.
- Revise Item 20 Name to : Spacer Ring

Drawing 790-590, Revision 3 — Loaded Vertical Concrete Cask (VCC), NAC-UMS®

- Add stainless steel tabs to Item 15
- BOM, Item 19, Change Spec to read, "A240/A276/A479." Note: Item 19 was added on DCR No. 790-590-1A.
- Add Delta Note 3 to BOM Item 14 to read, "at constructor's option, an additional washer may be added to facilitate lid to lid bolt fit-up"
- Add the following to the end of Delta Note 1, "Minimum thread length of bolt is 1.75"
- Revise Sheet 1 Item 13, Lid Bolt, description to: See Note 5
- Add Note 5 to read: Item 14 to be 1/2-13 UNC-2A X 3-1/4 LG Hex Hd. with minimum thread length of 1.75 or 1/2-13 UNC-2A X 2 1/2 LG Hex Hd.
- Change Delta Note 1 to read. . . Drill a 1/16 diameter hole thru the middle of the bolt head, from the middle of one flat to the opposite flat for minimum of two bolts per assembly
- Add Item 20, Cover Plate, Assembly 94 with quantity of 1 for Assemblies 95 thru 99. Change quantity of Item 15 to 1 for Assembly 94 only and quantity of Item 19 to 3 for Assembly 94. Add 2nd sheet for Assembly 94 detail.

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-590, Revision 3 — Loaded Vertical Concrete Cask (VCC), NAC-UMS® (continued)

- Update graphics to include new optional Supplemental Shielding (790-613) and Baffle Assembly (790-614)
- Add Delta Note 6 to read: "Shim plates may be utilized to facilitate field welding operations on one or both sides of the Supplemental Shielding Weldment. Locate welds approx. as shown." Add Delta Note to weld on Sheet 1 in Zone D6.
- Add reference dimension "(5.0) TYP" for placement of supplemental shielding in Detail D-D
- Add to BOM Item 21; Qty: Blank; Name: Supplemental Shielding; Material: Blank; Drawing No.: 790-613-99; Description: Blank

Drawing 790-591, Revision 3 — Bottom Weldment, Fuel Basket, 24 Element PWR NAC-UMS®

- Add Delta Note 5 as follows: The 3X ∅ 1.3 holes may be replaced with holes of ∅ 2.0. Also add Delta Note callout in Zone E7, to 3X ∅ 1.3

Drawing 790-592, Revision 6 — Top Weldment, Fuel Basket, 24 Element PWR, NAC-UMS®

- Dwg. Zone E5 dimension IS) .2" TYP WAS) .4" TYP
- Add chamfer to Items 3, 4 and 7, at interfaces with Item 2, with the following call-out in Section A-A: "45°±5° x .3"
- Add Note 5 as follows: Tolerance for fuel tube openings, dimensions 5.39, 15.66, 16.16, and 25.81 is ± .04
- Add Note 6 as follows: Minimum thickness of Item 2 may be reduced to .355 for a length of up to 31 inches measured along the outer circumference

Drawing 790-595, Revision 6 — Fuel Basket Assembly, 24 Element PWR, NAC-UMS®

- Add the second sentence to Note 3: "This dimension applies to the gap between the tallest fuel tube and the top weldment only"
- Add 45° x .3 chamfer to graphics for Items 2, 17 and 18, that was added per DCR No. 790-592-4B
- Modify Delta Note 4 to read: Item 4 length to extend beyond Item 1 surface by .25 + .02, -.25. Add Delta Note 4 in detail C-C, Sheet 2, Zone B5.
- Sheet 1, Zone C6, delete the fillet weld symbol
- Sheet 2, Zone C4, delete TYP near .25 in detail C-C

ATTACHMENT A

NAC-UMS® STORAGE SYSTEM DRAWING CHANGES

Drawing 790-605, Revision 6 — BWR Fuel Tube, Over-Sized Fuel, NAC-UMS®

- Items 3 and 4 IS) Neutron Absorber WAS) Neutron Poison and update Item/Assembly names, and notes accordingly
- Replace material in BOM IS) Boral/Metamic WAS) Boral

Drawing 790-613, Revision 0 — Supplemental Shielding, VCC Inlets, NAC-UMS®

- New drawing

Drawing 790-614, Revision 0 — Baffle Assembly, Vertical Concrete Cask (VCC), NAC-UMS®

- New drawing

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ATTACHMENT B

TABLE OF SIGNIFICANT CHANGES FOR UMS® REVISION UMSS-01C

Section/Page	Change
CHAPTER 1	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout.
Table 1-1/1-4	Changed "50,000 MWD/MTU" to "60,000 MWD/MTU."
Table 1-1/1-6	Changed "BORAL neutron poison" to "neutron absorber." BASIS: This change is intended to provide for the use of METAMIC.
Table 1-1/1-7	Deleted dimensions of neutron shield material in the Shield Plug.
Table 1-1/1-7	Added reference to the 100-ton transfer cask configuration.
Table 1-1/1-7	Added reference to the two lifting trunnions on the 100-ton transfer cask.
Table 1-1/1-7	Added description of the transfer cask cradle.
Table 1-1/1-8	Added NS-3 to the NS-4-FR definition section.
1.1/1.1-1	Changed Table references to Tables 6.2-1 and 6.2-2 for typical fuel allocation by Universal Storage System Class. The fuel parameter tables in Section 2.1 no longer present the System Class for allowed fuel. Additional explanation was required to address System Class.
1.2.1.1/1.2-3	Added text to clarify configuration and inspection of structural lid weld. Changed nomenclature FROM "backing ring" TO "spacer ring."
1.2.1.2.1/1.2-4	Revised text description to delete component dimensions. BASIS: Component dimensions are specified on the License Drawings and in Design Characteristics tables.
1.2.1.2.2/1.2-5	Revised text description to delete component dimensions. BASIS: Component dimensions are specified on the License Drawings and in Design Characteristics tables.
1.2.1.3/1.2-7	Added alternate baffle configuration (Drawing 790-614) description. Added NS-3 shield plug configuration. Added optional supplemental shielding fixture (Drawing 790-613) description.
1.2.1.4/1.2-7, -8, -9	Added 100-ton transfer cask description.
1.2.1.5.8/1.2-11	Added 100-ton transfer cask cradle description. Added transfer cask extension description to apply to the standard transfer cask.
Table 1.2-1/1.2-21	Changed reference FROM "BORAL" TO "neutron absorber."
Table 1.2-1/1.2-22	Added 100-ton transfer cask design characteristics.
Table 1.2-1/1.2-23	Deleted reference to rebar size.
Table 1.2-4/1.2-26	Changed reference FROM "BORAL" TO "neutron absorber."

ATTACHMENT B

TABLE OF SIGNIFICANT CHANGES FOR UMS® REVISION UMSS-01C

Section/Page	Change
CHAPTER 1 (CONTINUED)	
Table 1.2-5/1.2-27	Changed nominal weights of the concrete cask. BASIS: Conforms to weight tables in Chapter 3.
Table 1.2-7/1.2-29	Added physical parameters of 100-ton transfer cask.
1.3.1/1.3-1	Revises the allowable fuel parameter description. Primarily deletes reference to specific fuel vendors. This change results in renumbering of the fuel limits. BASIS: Use of NEI Standard Technical Specification fuel descriptions.
1.3.1/1.3-1	Added requirement for boron credit for some fuel enrichments.
Table 1.5-1/All	References to Chapter 12 have been revised throughout. BASIS: Use of NEI Standard Technical Specification format.
CHAPTER 2	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout.
Table 2-1/2-2	Changed weights in accordance with Chapter 3 weight and cg analysis. Added weight of the 100-ton transfer cask.
2.1/2.1-1	Revised evaluated spent fuel description to conform to presentation required by Standardized Technical Specifications. Deleted reference to design basis fuel assemblies. BASIS: NUREG-1745 ("Standard Format and Content for Technical Specifications for 10 CFR 72 Cask Certificates of Compliance").
2.1.1/2.1.1-1	Revised evaluated PWR fuel description to conform to presentation required by Standardized Technical Specifications. Deleted reference to design basis fuel assemblies. BASIS: NUREG-1745
Table 2.1.1/2.1.1-2	Revised evaluated PWR fuel description to conform to presentation required by Standardized Technical Specifications. Deleted reference to design basis fuel assemblies. BASIS: NUREG-1745
Table 2.1.1-2/2.1.1-3	Revised PWR Loading Table to account for higher evaluated enrichment (5.0) and for higher evaluated burnup (60,000). Added provision for loading using soluble Boron credit.

ATTACHMENT B

TABLE OF SIGNIFICANT CHANGES FOR UMS® REVISION UMSS-01C

Section/Page	Change
CHAPTER 2 (CONTINUED)	
2.1.2/2.1.2-1	Revised evaluated BWR fuel description to conform to presentation required by Standardized Technical Specifications. Deleted reference to design basis fuel assemblies. BASIS: NUREG-1745
Table 2.1.2/2.1.2-2	Revised evaluated BWR fuel description to conform to presentation required by Standardized Technical Specifications. Deleted reference to design basis fuel assemblies. BASIS: NUREG-1745
Table 2.1.2-2/2.1.2-3	Revised BWR Loading Table to account for higher evaluated enrichment (4.9) and for higher evaluated burnup (60,000).
2.1.3/2.1.3-1	Revised MY site specific fuel description to conform to presentation required by Standardized Technical Specifications. BASIS: NUREG-1745
2.1.3.1.3/2.1.3-3	Revised inserted hardware description to clarify use of the Class 2 canister.
Figure 2.1.3.1-1/2.1.3-8	Revised figure to allow references to loading positions by number. No previously approved loading positions for MY fuel are changed.
Table 2.1.3.1-4/2.1.3-12	Added Loading Table for MY fuel with no inserts. This table was previously in Chapter 12. BASIS: NUREG-1745
Table 2.1.3.1-5/2.1.3-13	Added Loading Table for MY fuel with CEA inserts. This table was previously in Chapter 12. BASIS: NUREG-1745
2.3.3.1/2.3-4	Added 100-ton transfer cask description and test requirement.
2.3.4.1/2.3-6	Changed text reference from "BORAL" to "neutron absorber." BASIS: This change is intended to provide for the use of METAMIC.
Table 2.3-1/2.3-12 to 2.3-19	Table is revised throughout to include the description of additional items added to the drawing Bill of Material and to add new drawings. Some component safety classifications have been revised based on evaluation.

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Section/Page	Change
CHAPTER 3	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout.
Various	Weights of components and assembled configurations are revised to reflect the results of updated calculations and to provide rounded weight values. The use of revised weights results in extensive recalculation of stresses and margins of safety throughout this Chapter.
3.1.1/3.1-2	Describes the use of NS-3 as a neutron shield material in the concrete cask shield plug.
3.1.1/3.1-2 (and throughout)	Changes nomenclature of the ring on the structural lid FROM "Backing Ring" TO "Spacer ring." BASIS: Deletes use of an ASME defined term from a component that is not intended to be an ASME Code based item.
3.1.1/3.1-2 (and throughout)	Deletes reference to BORAL as a trade name for a neutron absorber where appropriate. BASIS: Provides for the future use of METAMIC as a neutron absorber material.
3.1.1/3.1-2 (and throughout)	Describes the "100-ton" transfer cask. This configuration of the transfer cask is intended for use where there is a cask handling crane having a weight limit of 100-tons. Establishes nomenclature for the previously approved transfer cask as the "standard" transfer cask. The design of the 100-ton transfer cask provides for horizontal transfer of the loaded and sealed canister and for the unloaded canister.
Tables 3.2-1, 3.2-2, and 3.2-3	Weights presented in the tables are revised throughout to reflect updated component weight calculations and to present rounded values. The rounded values allow simplified comparison for bounding analysis.
Table 3.2-4	Table is added to present the under-the-hook weights for the 100-ton transfer cask.
Table 3.3-1	Adds a column to present material property values at 900°F.
Table 3.3-6	Adds property values of A-706 reinforcing bar used for lifting lug attachment.
Table 3.3-7	Adds a column to present material property values at 800°F.
Table 3.3-11	Adds a column to present material property values at 750°F.

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Section/Page	Change
CHAPTER 3 (CONTINUED)	
Table 3.3-13	Revised to include the material properties of NS-3.
Table 3.3-14	Table added to present property values for SA-516 carbon steel.
3.3.2/3.3-16	Added fracture toughness description for A-706 lifting lug rebar material.
3.4.1.2.4/3.4.1-8	Added description for alternate coating for exposed carbon steel surfaces.
3.4.3/3.4.3-1	This section revised throughout to include discussion of the 100-ton transfer cask.
3.4.3.1/3.4.3-6	Concrete cask lift evaluation is revised based on revised weight calculations for components and assembled components.
3.4.3.3/3.4.3-26 (and following pages)	This section is revised throughout to rename the previously approved transfer cask as the "standard" transfer cask and changes section, table and figure titles, where appropriate, to indicate the section applies to the standard transfer cask. This section is also revised to reflect the recalculation of component and assembled component weights. The revised weights result in the recalculation of stresses and margins of safety throughout the standard transfer cask evaluation.
3.4.3.4/3.4.3-55	This section is added to provide the lifting and horizontal handling evaluation of the 100-ton transfer cask. The function of the 100-ton transfer cask is the same as that of the standard transfer cask except that the 100-ton transfer cask may be handled horizontally, moved on a wheeled cradle. For vertical handling modes, the analysis methodology is the same as that for the standard transfer cask. Horizontal handling is described in Section 3.4.3.4.2.
3.4.4.1.1/3.4.4-3	Temperatures used in the thermal stress analysis are changed based on an improved fuel model to establish conductivities used in the analysis model. This section is revised throughout based on the use of the temperatures calculated by the new model.
3.4.4.1.2/3.4.4-5	This section is revised to include the evaluation of the canister in the horizontal orientation when in the 100-ton transfer cask.
3.4.4.1.7/3.4.4-11	This section is revised to include consideration of the pressure exerted by the weight of the water during the pressure test of the canister. The analysis is also revised to include the use of 1.25 times the normal pressure as bounding condition.
3.4.4.1.8/3.4.4-12	This section is revised to include the evaluation of the fuel basket support disks in the horizontal orientation when in the 100-ton transfer cask. Stress allowables are based on a temperature of 800°F.

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Section/Page	Change
CHAPTER 3 (CONTINUED)	
3.4.4.1.9/3.4.4-15	This section is revised to refer to temperatures established by the new fuel conductivities model.
3.4.4.1.10/3.4.4-18	This section is revised to use increased weights for the PWR and BWR fuel tubes to provide a bounding fuel tube analysis.
3.4.4.1.11/3.4.4-19	The canister closure weld stress intensities are revised based on the horizontal handling condition for the canister.
Tables 3.4.4.1-1 to 3.4.4.1-10 and Tables 3.4.4.1-12 and 3.4.4.1-13	These tables are revised to show new stress intensities based primarily on the bounding condition that occurs in horizontal transfer, but including consideration of the revised temperatures and component weights (PWR case). (Note: Table 3.4.4.1-11 is a listing of the PWR model sections and is not changed. Table 3.4.4.1-14 is the corresponding BWR listing.)
Tables 3.4.4.1-15 and 3.4.4.1-16	These tables are revised to show new stress intensities based primarily on the bounding condition that occurs in horizontal transfer, but including consideration of the revised temperatures and component weights. (BWR case).
Table 3.4.4.1-17/3.4.4-63	This table is revised to provide the stress summary for the PWR and BWR weldment. These revisions are due primarily to recalculation of temperatures.
3.4.4.2.1/3.4.4-65	The concrete cask evaluation is revised to incorporate revisions to calculated weights of components and assembled components. The revised weights represent a bounding condition.
Table 3.4.4.2-1/3.4.4-76	This table is revised to summarize the results of the concrete cask evaluation incorporating revisions to calculated weights of components and assembled components. The revised weights represent a bounding condition.

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Section/Page	Change
CHAPTER 4	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout.
Various	Changed FROM "BORAL" TO "neutron absorber" throughout
Various	Changed FROM "backing ring" TO "spacer ring" throughout
TOC	Updated to reflect any changes
4.1/4.1-2	Revised text description to delete reference to time limits and add reference to Administrative Controls. BASIS: NUREG-1745 ("Standard Format and Content for Technical Specifications for 10 CFR Part 72 Certificates of Compliance"). This NUREG provides for the application of administrative controls for the control of designated temperature limits.
4.1/4.1-3	Added 100-ton transfer cask description.
Table 4.1-2/4.1-5	Revised table to report maximum durations for canister conditions based on revised thermal analysis. The revised thermal analysis considers improved fuel thermal conductivities.
Table 4.1-3/4.1-6	Revised short-term temperature limit for Aluminum 6061-T651 to 750°F.
Table 4.1-4/4.1-7	Revised PWR maximum component temperatures based on model results using improved fuel thermal conductivities.
Table 4.1-5/4.1-8	Revised BWR maximum component temperatures based on model results using improved fuel thermal conductivities.
Table 4.2-1/4.2-2	Added thermal properties for NS-3.
Table 4.2-2/4.2-2	Added properties for 800°F and 900°F.
Tables 4.2-3 and 4.2-5/4.2-3	Added properties for 800°F.
4.4.1/4.4.1-1	
Table 4.2-13/4.2-7	Added thermal properties for METAMIC.
4.3/4.3-1	Added heat transfer disk creep evaluation for higher allowable temperature based on ANSYS models in Figures 4.3-1 and 4.3-2.
Table 4.3-1/4.3-4	Added deflection results.
4.4.1/4.4.1-1	Removed reference to the two-dimensional axisymmetric transfer cask and canister models. (Analysis is performed using the three-dimensional transfer cask and canister model.)

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Section/Page	Change
CHAPTER 4 (CONTINUED)	
4.4.1/4.4.1-2	Revised to clarify use of models.
4.4.1/4.4.1-3	Revised to include reference to METAMIC material.
4.4.1.1/4.4.1-8	Added supplemental shielding fixture discussion. Provides analysis showing that there is no significant temperature effect based on the insertion of the supplemental shielding fixture in the inlets.
4.4.1.2/4.4.1-13	Thermal evaluation using the three-dimensional canister model is revised to include the assumption of a single point of contact between the heat transfer disks and the canister shell and between the support disks and canister shell. A sensitivity analysis is preformed to support the single point assumption.
4.4.1.2/4.4.1-15	Added description of emissivity differences.
Figure 4.4.1.2-2/4.4.1-19	Added location of coupled nodes.
Figure 4.4.1.2-4/4.4.1-21	Added location of coupled nodes.
Tables 4.4.1.2-1 and 4.4.1.2-5/4.4.1-22 and 23	Revised to report fuel thermal conductivities for PWR and BWR fuel, respectively, based on improved model results.
Tables 4.4.1.2-3 and 4.4.1.2-4/4.4.1-24 and 25	Revised to report fuel thermal conductivities for PWR and BWR fuel tubes, respectively, based on improved model results.
4.4.1.3/4.4.1-26	Revised model description throughout to delete reference to the two-dimensional axisymmetric model and refer to the three-dimensional model. This model incorporates the revised fuel and fuel tube thermal conductivities.
4.4.1.3/4.4.1-27	Dispositioned 100-ton transfer cask analysis as bounded by the standard transfer cask.
Figure 4.4.1.3-1/4.4.1-28	Revised figure to show PWR three-dimensional standard transfer cask and canister model.
Figure 4.4.1.3-2/4.4.1-29	Revised figure to show BWR three-dimensional standard transfer cask and canister model.
4.4.1.4/4.4.1-30	Revised analysis model reference.
4.4.1.5/4.4.1-34	Fuel model is revised to more accurately model fuel configuration, improving fuel thermal conductivity results.
4.4.1.5/4.4.1-37	Fuel tube model is revised to more accurately model fuel tube configuration, improving fuel tube thermal conductivity results.
Figure 4.4.1.6-2/4.4.1-41 Figure 4.4.1.6-3/4.4.1-42	Deletes reference to BORAL and shows location of LINK31 elements used in fuel tube model.

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Section/Page	Change
CHAPTER 4 (CONTINUED)	
4.4.3/4.4.3-1	This section is revised throughout to report the results of the thermal analysis based on the improved fuel and fuel tube thermal conductivity results.
4.4.3.1/4.4.3-2	This section is revised to increase the time allowed in the vacuum condition based on reduced heat loads. It incorporates reference to the LCO and administrative controls that are applied to ensure that allowable temperatures for fuel cladding, and other components are not exceeded. It deletes reference to control of the time in helium condition. BASIS: The revised analysis is based on improved fuel thermal conductivities that extend the time the fuel can be in the vacuum condition and establish an unlimited time that the fuel can be in the helium condition. This allows the deletion of LCO 3.1.4, controlling the time the canister can be in the transfer cask.
4.4.3.1/4.4.3-4	This section is revised to provide extended times for the condition in which supplemental cooling (either in-pool or forced air) is applied on failure to meet the time in vacuum condition. The extended times are applied within an administrative control in Section 8.4.2. BASIS: The revised analysis is based on improved fuel thermal conductivities.
Figure 4.4.3-5/4.4.3-10	Revised figure shows extended times for PWR fuel activities based on revised thermal analysis.
Figure 4.4.3-6/4.4.3-11	Revised figure shows extended times for BWR fuel activities based on revised thermal analysis.
Tables 4.4.3-1 through 4.4.3-10/4.4.3-14 through 4.4.3-20	These tables report the revised temperatures for major components based on the revised fuel thermal conductivity models, and report the times associated with activities based on the revised maximum temperatures. They include the temperature conditions for all of the nominal activities and for supplemental cooling.
Section 4.4.5.1/4.4.5-1	The maximum pressure of the PWR canister is revised to include consideration of the effects of burnable poison rod assemblies on canister pressure. The calculation detail is deleted. The revised analysis results in a lower canister pressure. BASIS: Reduced helium temperature based on lower internal temperatures. Lower temperatures result from improved fuel thermal conductivities.

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<table>
<tr><th>Section/Page</th><th>Change</th></tr>
<tr><td colspan="2" align="center">CHAPTER 4 (CONTINUED)</td></tr>
<tr><td>Section 4.4.5.2/4.4.5-1</td><td>The maximum pressure of the BWR canister is revised to include consideration of the effects of burnable poison rod assemblies on canister pressure. The revised analysis results in a lower canister pressure.
BASIS: Reduced helium temperature based on lower internal temperatures. Lower temperatures result from improved fuel thermal conductivities.</td></tr>
<tr><td>4.4.7/4.4.7-1</td><td>Revises upper burnup limit to 60,000 MWD/MTU.</td></tr>
<tr><td>Figures 4.4.7-1 and 4.4.7-2/4.4.7-6</td><td>Figures revised to include high burnup curves.</td></tr>
<tr><td>Table 4.4.7-3/4.4.7-9</td><td>Tables revised to include higher burnup cladding stress limits.</td></tr>
<tr><td>Table 4.4.5-5/4.4.7-10</td><td>Table is revised to include maximum allowable cladding temperatures for PWR and BWR fuel.</td></tr>
<tr><td>Table 4.4.8-5/4.4.7-10</td><td>Table is revised to include maximum allowable decay heat for PWR and BWR fuel based on cool time at higher burnups.</td></tr>
<tr><td colspan="2" align="center">CHAPTER 5</td></tr>
<tr><td colspan="2"><u>Note:</u> Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.</td></tr>
<tr><td>Various</td><td>References to Chapter 12 have been revised throughout</td></tr>
<tr><td>Various</td><td>References to BORAL have been changed to neutron absorber, where appropriate, throughout</td></tr>
<tr><td>5.1/5.1-1</td><td>Inclusion of transfer cask in either a standard or 100-ton configuration</td></tr>
<tr><td>5.1/5.1-2</td><td>Brief description of 100-ton transfer cask</td></tr>
<tr><td>5.1/5.1-2</td><td>Inclusion of either NS-4-FR or NS-3 neutron shielding</td></tr>
<tr><td>5.1/5.1-3</td><td>Clarifies that the shielding evaluation for PWR fuel is based on the standard transfer cask and vertical concrete cask</td></tr>
<tr><td>5.1/5.1.3</td><td>Analysis of 7 PWR and BWR fuel types for shielding evaluation of 100-ton transfer cask</td></tr>
<tr><td>5.1/5.1.4</td><td>Inclusion of two canister and neutron shield conditions for the 100-ton transfer cask: wet canister with dry neutron shield and dry canister with wet neutron shield</td></tr>
<tr><td>5.1.2/5.1-7</td><td>Inclusion of MCBEND Monte Carlo transport code for shielding evaluations of 100-ton transfer cask</td></tr>
</table>

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Section/Page	Change
CHAPTER 5 (CONTINUED)	
5.1.3.3/5.1-10	Inclusion of dose rates for the 100-ton transfer cask for PWR & BWR fuel
Table 5.1-5/5.1-14	Summary of maximum dose rates for 100-ton transfer cask with PWR fuel
Table 5.1-6/5.1-14	Summary of average dose rates for 100-ton transfer cask with PWR fuel
Table 5.1-7/5.1-15	Summary of maximum dose rates for 100-ton transfer cask with BWR fuel
Table 5.1-8/5.1-15	Summary of average dose rates for 100-ton transfer cask with BWR fuel
5.2.2/5.2.2-1	Detailed description of 100-ton transfer cask source descriptions, including gamma source, neutron source, PWR & BWR fuel assemblies
Table 5.2-27/5.2.3-27	100-ton transfer cask three-dimensional PWR fuel assembly descriptions
Table 5.2-28/5.2.3-28	100-ton transfer cask three-dimensional BWR fuel assembly descriptions
Table 5.2-29/5.2.3-29	PWR fuel reactor operating conditions for 100-ton transfer cask analysis
Table 5.2-30/5.2.3-29	BWR fuel reactor operating conditions for 100-ton transfer cask analysis
Table 5.2-31/5.2.3-30	PWR cycle length calculation for 100-ton transfer cask source terms
Table 5.2-32/5.2.3-31	BWR cycle length calculation for 100-ton transfer cask source terms
Table 5.2-33/5.2.3-32	MCBEND standard 28 group neutron boundaries
Table 5.2-34/5.2.3-33	MCBEND standard 22 group gamma boundaries
Table 5.2-35/5.2.3-34	PWR fuel assembly hardware mass and mass scale factors by source region
Table 5.2-36/5.2.3-35	BWR fuel assembly hardware mass and mass scale factors by source region
5.3.2/5.3.2-1	Detailed description of 100-ton transfer cask model specification
Figure 5.3-7/5.3.2-11	MCBEND three-dimensional 100-ton transfer cask model – axial dimensions
Figure 5.3-8/5.3.2-12	MCBEND three-dimensional 100-ton transfer cask model – radial dimensions

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CHAPTER 5 (CONTINUED)	
Figure 5.3-9/5.3.2-13	MCBEND three-dimensional 100-ton transfer cask model – upper trunnion detail
Table 5.3-6/5.3.2-21	MCBEND PWR fuel region homogenization
Table 5.3-7/5.3.2-22	MCBEND BWR fuel region homogenization
Table 5.3-8/5.3.2-23	MCBEND PWR fuel assembly hardware region homogenization
Table 5.3-9/5.3.2-24	MCBEND BWR fuel assembly hardware region homogenization
Table 5.3-10/5.3.2-25	MCBEND homogenized PWR fuel regional densities
Table 5.3-11/5.3.2-25	MCBEND homogenized BWR fuel regional densities
Table 5.3-12/5.3.2-26	MCBEND regional densities for 100-ton transfer cask structural and shield materials
5.4.2/5.4.2-1	Detailed description of 100-ton transfer cask shielding evaluation
Figure 5.4-29/5.4.2-20	100-ton transfer cask radial surface dose rate profile at various radial positions: azimuthal average, wet canister/dry neutron shield, PWR fuel
Figure 5.4-30/5.4.2-20	100-ton transfer cask radial surface dose rate profile by source component: azimuthal average, wet canister/dry neutron shield, PWR fuel
Figure 5.4-31/5.4.2-21	100-ton transfer cask azimuthal surface dose rate profile at fuel midplane elevation: wet canister/dry neutron shield, PWR fuel
Figure 5.4-32/5.4.2-22	100-ton transfer cask radial surface dose rate profile at various radial positions: azimuthal average, dry canister/wet neutron shield, PWR fuel
Figure 5.4-33/5.4.2-22	100-ton transfer cask radial surface dose rate profile by source component: azimuthal average, dry canister/wet neutron shield, PWR fuel
Figure 5.4-34/5.4.2-23	100-ton transfer cask azimuthal surface dose rate profile at fuel midplane elevation: dry canister/wet neutron shield, PWR fuel
Figure 5.4-35/5.4.2-24	100-ton transfer cask radial surface dose rate profile at various radial positions: azimuthal average, wet canister/dry neutron shield, BWR fuel
Figure 5.4-36/5.4.2-24	100-ton transfer cask radial surface dose rate profile by source component: azimuthal average, wet canister/dry neutron shield, BWR fuel
Figure 5.4-37/5.4.2-25	100-ton transfer cask azimuthal surface dose rate profile at fuel midplane elevation: wet canister/dry neutron shield, BWR fuel

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Section/Page	Change
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Figure 5.4-38/5.4.2-26	100-ton transfer cask radial surface dose rate profile at various radial positions: azimuthal average, dry canister/wet neutron shield, BWR fuel
Figure 5.4-39/5.4.2-26	100-ton transfer cask radial surface dose rate profile by source component: azimuthal average, dry canister/wet neutron shield, BWR fuel
Figure 5.4-40/5.4.2-27	100-ton transfer cask azimuthal surface dose rate profile at fuel midplane elevation: dry canister/wet neutron shield, BWR fuel
Table 5.4-3/5.4.2-30	ANSI standard neutron flux-to-dose rate factors in MCBEND group structure
Table 5.4-4/5.4.2-31	ANSI standard gamma flux-to-dose rate factors in MCBEND group structure
Table 5.4-5/5.4.2-32	Summary of maximum 100-ton transfer cask dose rates
Table 5.4-6/5.4.2-33	Summary of maximum 100-ton transfer cask dose rates for PWR fuel
Table 5.4-7/5.4.2-33	Summary of maximum 100-ton transfer cask dose rates for BWR fuel
Table 5.4-8/5.4.2-34	Summary of average 100-ton transfer cask dose rates for PWR fuel
Table 5.4-9/5.4.2-34	Summary of average 100-ton transfer cask dose rates for BWR fuel
5.5/5.5-1	Incorporation of higher burnup fuel—to 60 GWD/MTU
Table 5.5-2/5.5-6	Decay heat limits on a per assembly basis for PWR and BWR fuels
Table 5.5-6/5.5-9	Westinghouse 17x17 minimum cooling time evaluation (higher enriched fuel added)
Table 5.5-7/5.5-10	GE 17x17 minimum cooling time evaluation (higher enriched fuel added)
Table 5.5-8/5.5-11	Loading table for PWR fuel (higher enriched fuel added; higher burnup fuel added)
Table 5.5-9/5.5-13	Loading table for BWR fuel (higher enriched fuel added; higher burnup fuel added)
5.7/5.7-3	Two additional references added (numbers 23 and 24)
CHAPTER 6	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout
Various	References to BORAL have been changed to neutron absorber, where appropriate, throughout

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Section/Page	Change
CHAPTER 6 (CONTINUED)	
Various	Inclusion of the ANSWERS MONK module throughout to perform criticality analysis (in addition to SCALE 4.3 CSAS)
6.1/6.1-1	Definitions given for maximum initial enrichment for PWR and BWR fuel assemblies with and without soluble boron credit
6.1/6.1-2	Brief description of 100-ton transfer cask
6.1/6.1-4	Inclusion of MONK8A and JEF 2.2 neutron cross-section libraries to determine K_{eff}
Table 6.1-1/6.1-6	PWR fuel assembly maximum allowed enrichment – no soluble boron
Table 6.1-2/6.1-7	BWR fuel assembly maximum allowed enrichment
Table 6.2-1/6.2-2	PWR fuel assembly characteristics (Zirc-4 clad)
Table 6.2-2/6.2-3	BWR fuel assembly characteristics (Zirc-4 clad)
6.3.4/6.3-7	Inclusion of METAMIC as alternate neutron absorber material
Figure 6.3-7/6.3-16	100-ton transfer cask geometry
Figure 6.3-10/6.3-18	Transfer cask containing a PWR basket
Figure 6.3-11/6.3-19	Concrete cask containing a BWR basket
6.4.3.2/6.4-14	Inclusion of criticality models for PWR canister configurations in the 100-ton transfer cask
6.4.3.3/6.4-15	Inclusion of criticality models for BWR canister configurations in the 100-ton transfer cask
6.4.5/6.4-19	Detailed discussion of PWR and BWR fuel assembly specific maximum initial enrichment
Table 6.4-25/6.4-39	PWR maximum allowable enrichment – no soluble boron
Table 6.4-26/6.4-39	BWR maximum allowable enrichment – no soluble boron
Table 6.4-27/6.4-40	Most reactive geometry for a borated water PWR canister
Table 6.4-28/6.4-40	Moderator density versus reactivity for the borated water cases
Table 6.4-29/6.4-41	PWR maximum allowable enrichment – soluble boron
6.5.5/6.5-9	MONK validation in accordance with NUREG/CR-6361
Figure 6.5-10/6.5-21	MONK8A – JEF 2.2 Library – k_{eff} versus rod pitch
Figure 6.5-11/6.5-22	MONK8A – JEF 2.2 Library – k_{eff} versus H/U (fissile) atom ratio
Figure 6.5-12/6.5-23	MONK8A – JEF 2.2 Library – k_{eff} versus ^{10}B plate loading
Figure 6.5-13/6.5-24	MONK8A – JEF 2.2 Library – k_{eff} versus mean neutron log(E) causing fission
Figure 6.5-14/6.5-25	MONK8A – JEF 2.2 Library – k_{eff} versus cluster gap thickness

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CHAPTER 6 (CONTINUED)	
Figure 6.5-15/6.5-26	MONK8A – JEF 2.2 Library – k_{eff} versus fuel pellet outside diameter
Figure 6.5-16/6.5-27	MONK8A – JEF 2.2 Library – k_{eff} versus fuel rod outside diameter
Figure 6.5-17/6.5-28	MONK8A – JEF 2.2 Library – k_{eff} versus soluble boron PPM in moderator
Figure 6.5-18/6.5-29	USLSTATS Output – k_{eff} versus gap thickness
Table 6.5-3/6.5-37	SCALE 4.3 range of correlated parameters of most reactive configurations
Table 6.5-4/6.5-38	MONK 8A range of correlated parameters for design basis fuel
Table 6.5-5/6.5-38	MONK8A – correlation coefficient for linear curve-fit of critical benchmarks
Table 6.5-6/6.5-39	MONK8A – JEF 2.2 Library validation statistics
6.7/6.7-2	One additional reference added (number 20)
Figure 6.8-9/6.8-52	MONK8A input for PWR transfer cask with soluble boron
Figure 6.8-10/6.8-58	MONK8A input for BWR transfer cask
CHAPTER 7	
There are no material changes to Chapter 7.	
CHAPTER 8	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout.
8.1/8.1-1	Added description and intended use for 100-ton transfer cask.
8.1.1/8.1.1-1	Added Note for Boron credit for certain fuel loading conditions.
8.1.1 (Step 12) /8.1.1-2	Added reference to an Administrative Control in Section 8.4.2.1.
8.1.1 (Step 26) /8.1.1-3	Added optional pressure test of the canister in the event that the canister is not tested at fabrication in accordance with Section 9.1.2.3.
8.1.1 (Steps 27 to 31) / 8.1.1-3.	Revised references FROM LCOs previously specified in Chapter 12 TO Administrative Controls described in Section 8.4. In Step 27, a reference to LCO 3.1.1 is used to control the time in vacuum drying.
8.1.1 (Steps 34 and 49) / 8.1.1-4	Revised references FROM LCOs previously specified in Chapter 12 TO Administrative Controls described in Section 8.4.
8.1.1 (Step 60) / 8.1.1-5	Revised reference FROM LCO 3.2.1 previously specified in Chapter 12 TO Administrative Controls described in Section 8.4.1.1.

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Section/Page	Change
CHAPTER 8 (CONTINUED)	
Table 8.1.1-1/8.1.1-7 (2 places and in procedures)	Deleted reference to Master Link. BASIS: The weight of the Master Link component was too high to be safely handled by field personnel.
Table 8.1.1-1	Added description of the optional supplemental shielding fixture.
8.1.3/8.1.3-1	Changed the number of concrete cask lifting lugs FROM 4 TO 2.
8.1.3/8.1.3-1	Changed allowable lift height FROM 20 inches TO 24 inches. Changed Administrative Control reference to 8.4.3.
8.1.3/8.1.3-1	Added reference to the optional supplemental shielding fixture.
8.1.3/8.1.3-2	Changed allowable lift height FROM 20 inches TO 24 inches.
8.1.3/8.1.3-2	Changed LCO reference FROM 3.1.6 TO 3.1.2.
8.2/8.2-1	Changed allowable lift height FROM 20 inches TO 24 inches.
8.3/8.3-1	Changed handling time FROM 4 hours TO control by an Administrative Program in Section 8.4.2.1.
8.3 (Step 13) / 8.3-2	Added Note to address the possible use of borated water for boron credit.
8.4/8.4-1	This section is added to Chapter 8. It presents the Administrative Programs and Controls that were previously included in Chapter 12 as Technical Specifications, Limiting Conditions of Operation (LCO) or as Program Controls. BASIS: This new section incorporates certain Limiting Condition of Operations (LCOs), Design Features and Administrative Programs formerly in Chapter 12, Technical Specifications. These requirements were converted to Administrative Controls as described in Section 8.4 in accordance with NUREG-1745.
8.4.1/8.4-1	This section presents the Radioactive Effluent Control Program that was previously provided in Appendix 12A, Section A 5.5. Section 8.4.1 is includes the controls placed on helium leak rate (Old LCO 3.1.5 and on canister surface smearable contamination limits (Old LCO 3.2.1). Administrative controls on these items are included as they are considered to be potential contributors to radioactive effluent. Section 8.4.1.2 increases the allowable surface contamination limits.

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Section/Page	Change
CHAPTER 8 (CONTINUED)	
8.4.2/8.4-2	This section presents the Storage System Operations Program that was previously provided in Appendix 12A, Sections A5.1 and A5.2. Section 8.4.2 includes the controls on time in transfer cask for the vacuum condition (Old LCO 3.1.1), the control on vacuum drying pressure (Old LCO 3.1.2), the control on helium backfill pressure (Old LCO 3.1.3) and the control on allowable helium leak rate (Old LCO 3.1.4). This section also presents the controls on supplemental (in-pool or forced air) cooling of the canister in the unlikely event that time in vacuum condition limits are not met.
8.4.3/8.4-9	This section presents the ISFSI Operations Program that was previously provided in Appendix 12A, Sections A5.2 and the Design Features descriptions in Section B3.0. Section 8.4.3 includes the controls on ISFSI pad construction, lift height limits, concrete cask positioning, concrete cask surface dose rate limits (Old LCO 3.2.2) and the air temperature-monitoring requirement of (Old LCO 3.1.6). The air temperature monitoring control, which demonstrates that the heat removal system is operable is included as (New) LCO 3.1.2. This section presents revised parameters for concrete (ISFSI) pad construction. This section also presents the control requirement for surveillance after an off-normal or accident event, or a natural phenomena event.
8.4.4/8.4-15	This section presents the fuel selection criteria that are considered in the approved contents. It includes the criteria previously provided in Appendix 12B, Section B2.0. These criteria include the requirement to verify fuel based on cool time and, for high burnup fuel, to verify oxide layer thickness. This section points to Sections 2.1.1 and 2.1.2 of Chapter 2, as the fuel descriptions have been moved from Section B2.0 to these sections for PWR and BWR fuel, respectively. It includes a boron concentration measurement requirement for boron credit.

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Section/Page	Change
CHAPTER 9	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout.
9.1.2.1/9.1-5	Added load test requirement for 100-ton transfer cask.
9.1.2.1/9.1-5	Revised weights used in load tests based on weights revised in Chapter 3.
9.1.2.3/9.1-6	Added an optional provision for conducting a pressure test of the transportable storage canister during fabrication. Either a hydrostatic or pneumatic pressure test may be performed in accordance with NB-6200 or NB-6300. (Steps remain in the operating procedure for conducting the pressure test during the canister loading and closing sequence in the event the test is not performed at fabrication.)
9.1.6/9.1-8	Added provision for the use of METAMIC® neutron absorber material.
9.1.6.1/9.1-9	Added description of the neutron absorber material sampling plan. This plan is applied to ensure that the required level of ^{10}B is obtained in fabricated neutron absorber plates.
9.1.6.2/9.1-9	Added description of the wet chemistry testing method for determination of boron content. BASIS: Wet chemistry is an industry accepted method of determining ^{10}B content and is used to qualify thermal neutron absorption (neutron blackness testing) targets.
N/A/9.2-2	Deleted the requirement to verify and report the thermal performance of the first storage unit placed in service. BASIS: The heat removal system is verified to be operational in accordance with LCO 3.1.2. The heat removal is accomplished using a chimney effect, passive system constructed of structural steel components. There are no components of the system that can fail to operate in normal conditions. The adequacy of the chimney effect system is demonstrated by the successful operation of installed system of similar design.
CHAPTER 10	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout.

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Section/Page	Change
CHAPTER 10 (CONTINUED)	
10.1.310.1-2	This section is revised to add the use of automated equipment for weld inspection. This use decreases operational dose. BASIS: Field experience.
10.2/10.2-1	Added descriptions of the alternate baffle design and the supplemental shielding fixture. The supplemental shielding fixture may be field installed. Added reference to the MCBEND code used for shielding evaluations of the 100-ton transfer cask.
10.3/10.3-1	Added description of fuel configuration used to establish the bounding dose rates for the 100-ton transfer cask.
10.3.1/10.3-1	Added reference to summary tables for dose rates associated with the use of the standard and 100-ton transfer casks.
10.3.2/10.3-2	Deleted requirement for daily inspections of the concrete cask inlets and outlets. BASIS: Daily recording of air ambient temperature and the air temperature at all four air outlets on each of the concrete cask provides adequate early indication of loss of cooling capability. Since the system is passive, and has structural integrity, clearing of inlets or outlets is expected to be readily accomplished. Administrative controls establish the maximum allowable temperature difference between the ambient and outlet air temperatures. Revised total dose rates for loading single PWR and BWR systems and 20 cask arrays. The revised total dose rates are significantly less than the previously reported dose rates. BASIS: The revised dose rate analysis considers recent operational experience and consideration of "time in exposure zone" rather than the total time required to perform an activity. This distinction occurs because personnel are not continuously present in the maximum radiation zone during a specific activity or sequence of related activities.
Table 10.3-1/10.3-5	This table is revised to reflect reduced exposure during weld inspection through the use of automated equipment and reduced exposure during weld equipment set through the use of supplemental shielding. The time of exposure is revised to reflect the duration of the exposure rather than the total time taken for an activity. Exposure is based on the standard transfer cask.

Section/Page	Change
CHAPTER 10 (CONTINUED)	
Table 10.3-2/10.3-6	This table is added to present calculated exposure based on the 100-ton transfer cask.
Tables 10.3-5 through 10.3-8	These tables are revised to reflect the reduction in total dose and average total dose based on deleting the daily inspection requirement.
10.4/10.4-2	A description of how distances are measured is added.
Figures 10.4-1 and 10.4-2 and Tables 10.4-1 and 10.4-2	These figures and tables are revised to provided recalculated dose rates based on a corrected grouping of neutron energy spectra.
CHAPTER 11	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
Various	References to Chapter 12 have been revised throughout.
11.1.1/11.1-2	Temperatures revised due to improved fuel modeling to establish conductivities. Increased heat transfer disk allowable temperature.
11.1.2/11.2-3	Temperatures revised due to improved fuel modeling to establish conductivities. Increased heat transfer disk allowable temperature.
11.1.3.3/11.1.3-1	Off-normal handling case is revised to account for horizontal handling of the canister, using the 100-ton transfer cask (revised tabular results).
11.2.1.3/11.2.1-1	Canister pressure analysis is revised to account for the increased pressure from the BPRA gases. Details of the classical analysis are removed. The canister stress evaluation is revised based on minor revisions in the canister model.
11.2.3.3/11.2.3-1	Deletes reference to specific k_{eff} values for normal and accident conditions.
11.2.4.3/11.2.4-7	Revises reported stresses based on slightly increased weights.
11.2.4.3.1/11.2.4-12	This section is added to present the analysis of an alternate baffle assembly design. The alternate baffle assembly deforms under impact loading to reduce the g-loading on the concrete cask and canister. The reduced impact loading allows a higher lift height of the concrete cask, increasing the options for its movement. The added analysis results in text flow.

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CHAPTER 11 (CONTINUED)	
11.2.6.1/11.2.6-1	The fire analysis is revised to consider a fire consisting of 350 gallons of flammable liquid. This assumption allows the use of additional equipment within the ISFSI enclosure during transfer of the loaded concrete cask. There is no change in the assumed design basis fire duration.
11.2.7.3/11.2.7-1	Temperatures revised due to improved fuel modeling. Increased heat transfer disk allowable temperature.
11.2.8.2.1/11.2.8-5	The earthquake analysis is slightly revised to consider slight increases in the center of gravity height of the concrete cask based on slight component weight increases. The allowable earthquake loading is increased. BASIS: The controlling factor is sliding. As coefficient of friction increases the allowable earthquake load increases.
11.2.9.2/11.2.9-2	Revised to reflect slight changes in total weight and volume. Tabular stress results revised to reflect changes. Margins of Safety of 10 or greater as specified as "large."
11.2.11.3/11.2.11-2	Revised to reflect slight change in weight. Weight used is conservatively less.
11.2.12.3/11.2.12-2	Revised to evaluate the use of 28 day concrete compressive strength to 5,000 psi, and dry concrete and soil densities to 160 lbs/ft^3. Required reevaluation of component stresses in tip-over event.

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CHAPTER 12	
Note: Chapter Table of Contents, List of Figures and List of Tables updated to reflect chapter revisions.	
All	Chapter 12 is revised throughout to incorporate NUREG-1745 ("Standard Format and Content for Technical Specifications for 10 CFR 72 Cask Certificates of Compliance"). The principal revisions include the conversion of most LCOs to Administrative or Program Controls, which are added to Section 8.4 of the Operating Procedures in Chapter 8. The format of Chapter 12 is revised to facilitate its attachment to a future Certificate of Compliance. Due to the extent of the revision to Chapter 12, revision bars are not used.
1.1/12-3	Definitions have been rearranged and minor revisions have been made to the definitions for "Operable" (removed temperature limits), for "High Burnup Fuel" (extended burnup to 60,000 MWD/MTU), "Loading Operations" (remove reference to post-storage operations), and "Transport Operations" (included reference to use of lifting lugs).
1.2/12-8	No material changes have been made to this Section.
1.3/12-11	No material changes have been made to this Section.
1.4/12-15	No material changes have been made to this Section.
2.0/12-18	This section is revised throughout. Fuel Specifications and Loading Conditions are moved to Section 2.0 of Chapter 2. PWR and BWR fuel specifications and loading conditions are presented in Sections 2.1.1 and 2.1.2 of Chapter 2, respectively. Site Specific (Maine Yankee) fuel description is provided in Section 2.1.3 of Chapter 2. This revision eliminates Appendix 12B. Section of fuel, including high burnup fuel is supported by the Administrative Control provision of Section 8.4.4.
3.0/12-19	No material changes have been made to this Section.
LCO 3.1.1/12-22	This LCO is revised by moving the time limits to an Administrative Control provision in Section 8.4.2.1.

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CHAPTER 12 (CONTINUED)	
LCO 3.1.2/NA	This LCO is deleted. It is replaced by an Administrative Control provision in Section 8.4.2.2.
LCO 3.1.3/NA	This LCO is deleted. It is replaced by an Administrative Control provision in Section 8.4.2.3.
LCO 3.1.4/NA	This LCO is deleted. It is replaced by an Administrative Control provision in Section 8.4.2.1.
LCO 3.1.5/NA	This LCO is deleted. It is replaced by an Administrative Control provision in Section 8.4.1.1.
LCO 3.1.6/NA	This LCO is renumbered to LCO 3.1.2. It is supported by an Administrative Program provision in Section 8.4.3.5.
LCO 3.1.7/NA	This LCO is deleted. It is replaced by an Administrative Control provision in Section 8.4.2.1.
LCO 3.2.1/NA	This LCO is deleted. It is replaced by an Administrative Control provision in Section 8.4.1.2.
LCO 3.2.2/NA	This LCO is deleted. It is replaced by an Administrative Control provision in Section 8.4.3.4.
4.0/12-26	This section is revised to include the Design Features previously reported in Appendix 12B, Section B3 and the alternatives to the ASME Code employed in the system design. Section 4.3.1 revises the earthquake loading for the UMS system. BASIS: Section 11.2.8 shows that the controlling factor is sliding. As coefficient of friction increases the allowable earthquake load increases.
5.0/12-33	This section is revised throughout, but retains the requirement for dry run training for system operations including loading, closing, sealing, inspection, testing, and movement. The requirement for reporting on the thermal performance of the first system placed in service is deleted. BASIS: LCO 3.1.2, formed, steel structure of air flow path, ease of clearing obstructions, performance of systems already in service.
CHAPTER 13	
Section/Page	**Change**
13.3/13.3-1	Dates removed from CFR references.

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DOCKET NO. 72-1015

UMS®

UNIVERSAL MPC SYSTEM®

SAFETY ANALYSIS REPORT

for the

UMS® Universal Storage System

JUNE 2001 REVISION UMSS-01C

NAC INTERNATIONAL

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1.0 GENERAL DESCRIPTION

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

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

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

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

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

Table 1.5-1 provides a compliance matrix to the regulatory requirements and acceptance criteria specified in NUREG-1536. This matrix describes how the Universal Storage System Safety Analysis Report addresses and demonstrates compliance with each requirement and criterion listed in NUREG-1536. Table 4-1 in Chapter 12 provides a list of approved alternatives to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Table 1-1 Terminology

Universal Storage System	The storage component of the Universal MPC System (UMS®) designed by NAC for the storage and transportation of spent nuclear fuel.
Universal Transport Cask	The packaging consisting of a Universal Transport Cask body with a closure lid and energy-absorbing impact limiters. The Universal Transport Cask is used to transport a Transportable Storage Canister containing spent fuel. The cask body provides the primary containment boundary during transport.
Confinement System	The components of the Transportable Storage Canister intended to retain the radioactive material during storage.
Contents	Twenty-four PWR fuel assemblies, or fifty-six BWR fuel assemblies. The fuel assemblies may be configured as Site Specific Fuel. The fuel assemblies are contained in a Transportable Storage Canister.
Standard Fuel	Irradiated fuel assemblies with a burnup less than, or equal to, 45,000 MWD/MTU and having the same configuration as when originally fabricated consisting generally of the end fittings, fuel rods, guide tubes, and integral hardware. For BWR fuel, the channel is considered to be integral hardware. The design basis fuel characteristics and analysis are based on the standard fuel configuration.
Consolidated Fuel	A nonstandard fuel configuration in which the individual intact fuel rods from one or more fuel assemblies are placed in a single container or a lattice structure that is similar to a fuel assembly.
Intact Fuel (Assembly or Rod) (Undamaged Fuel)	A fuel assembly or fuel rod with no fuel rod cladding defects, or with known or suspected fuel rod cladding defects not greater than pinhole leaks or hairline cracks.

Table 1-1 Terminology (Continued)

Damaged Fuel (Failed Fuel)	A fuel assembly or fuel rod with known or suspected cladding defects greater than pinhole leaks or hairline cracks. Damaged Fuel must be placed in a Maine Yankee Fuel Can.
High Burnup Fuel	A fuel assembly having a burnup between 45,000 and 60,000 MWD/MTU, which must be preferentially loaded in periphery positions of the basket. Intact High Burnup Fuel having a cladding oxide layer thickness of 80 microns or less, as determined by measurement and statistical analysis, may be stored as intact fuel. High Burnup Fuel having a cladding oxide layer thickness greater than 80 microns is stored as damaged fuel.
Site Specific Fuel	Spent fuel configurations that are unique to a site or reactor due to the addition of other components or reconfiguration of the fuel assembly at the site. It includes fuel assemblies, which hold nonfuel-bearing components, such as control components or instrument and plug thimbles, or which are modified as required by expediency in reactor operations, research and development or testing. Modification may consist of individual fuel rod removal, fuel rod replacement of similar or dissimilar material or enrichment, the installation, removal or replacement of burnable poison rods, or containerizing damaged (failed) fuel. Site specific fuel includes irradiated fuel assemblies designed with variable enrichments and/or axial blankets, fuel that is consolidated and fuel that exceeds design basis fuel parameters.

Table 1-1 Terminology (Continued)

Maine Yankee Fuel Can	A specially designed stainless steel screened can sized to hold an intact fuel assembly, consolidated fuel, or damaged fuel. The can screens permit draining and drying, while precluding the release of gross particulates into the canister cavity.
Transportable Storage Canister (Canister)	The stainless steel cylindrical shell, bottom end plate, shield lid, and structural lid that contain the fuel basket structure and the contents.
Shield Lid	A thick stainless steel disk that is located directly above the fuel basket. The shield lid comprises the first part of a double-welded closure system for the Transportable Storage Canister. The shield lid provides a containment/confinement boundary for storage and shielding for the contents.
- Drain Port	A penetration located in the shield lid to permit draining of the canister cavity.
- Vent Port	A penetration located in the shield lid to aid in draining and in vacuum drying and backfilling the canister with helium.
- Port Cover	The stainless steel covers that close the vent and drain ports, and that are welded in place following draining, drying, and backfilling operations.
- Quick Disconnect	The valved nipple used in the vent and drain ports to facilitate operations.
Structural Lid	A thick stainless steel disk that is positioned on top of the shield lid and welded to the canister. The structural lid is the second part of a double-welded closure system for the Transportable Storage Canister. The structural lid provides a confinement boundary for storage, shielding for the contents, and canister lifting/handling capability.

Table 1-1 Terminology (Continued)

Fuel Basket (Basket)	The structure located within the Transportable Storage Canister that provides structural support, criticality control, and primary heat transfer paths for the fuel assemblies.
- Support Disk	The primary lateral load-bearing component of the fuel basket. The PWR support disk is a circular stainless steel plate with 24 square holes machined in a symmetrical pattern. The BWR support disk is a circular carbon steel plate with 56 square holes machined in a symmetrical pattern. Each square hole is a location for a fuel tube.
- Heat Transfer Disk	A circular aluminum plate with 24 (PWR basket) or 56 (BWR basket) square holes machined in a symmetrical pattern. The heat transfer disk enhances heat transfer in the fuel basket.
- Fuel Tube	A stainless steel tube having a square cross-section. One fuel tube is inserted through each square hole in the support disks and heat transfer disks. Fuel assemblies are loaded into the fuel tube. A fuel tube may have neutron absorber material enclosed by a stainless steel sheet on one or more of its external faces depending on fuel type and the position of the fuel tube in the basket.
- Tie Rod	A stainless steel rod used to align, retain, and support the support disks and the heat transfer disks in the fuel basket structure. The tie rods extend from the top weldment to the bottom weldment.
- Spacer	Installed on the tie rod between the support disks (BWR only) or between the support disks and top and bottom weldments (BWR and PWR) to properly position the disks and provide axial support for the support disks.
- Split Spacer	Spacers installed on the tie rod between the support disks and the heat transfer disks to properly position the disks and provide axial support for the support disks and the heat transfer disks.

Table 1-1 Terminology (Continued)

Vertical Concrete Cask (Concrete Cask)	A concrete cylinder that contains the Transportable Storage Canister during storage. The Vertical Concrete Cask is formed around a steel inner liner and base and is closed by a shield plug and lid.
- Shield Plug	A thick carbon steel plug, which also contains a neutron shield material, installed in the top end of the Vertical Concrete Cask to reduce skyshine radiation.
- Lid	A thick carbon steel plate that serves as the bolted closure for the Vertical Concrete Cask. The lid precludes access to the canister and provides additional radiation shielding.
- Liner	A thick carbon steel shell that forms the annulus of the concrete cask. The liner serves as the inner form during concrete pouring and provides radiation shielding of the canister contents.
- Base	A carbon steel weldment that contains the air inlets, the concrete cask jacking points and the pedestal that supports the canister inside of the concrete cask.
Transfer Cask	A shielded lifting device for handling of the Transportable Storage Canister during loading of spent fuel, canister closure operations, and transfer of the canister into or out of the Vertical Concrete Cask during storage, or into or out of the Universal Transport Cask during transportation. The transfer cask incorporates bottom doors that permit the vertical loading of the storage and transport casks. The transfer cask is provided in either the standard or 100-ton configuration.
- Transfer Cask Lifting Trunnions	Low alloy steel trunnions used to lift and move the transfer cask in a vertical orientation. The standard transfer cask has four trunnions. The 100-ton transfer cask has two trunnions.
- Transfer Cask Cradle	A wheeled transfer trailer used to move the 100-ton transfer cask in the horizontal orientation.

Table 1-1 Terminology (Continued)

Transfer Adapter	A carbon steel plate assembly that attaches to the top of the transport or concrete cask to facilitate installation and alignment of the transfer cask. It also provides the operating mechanism for the transfer cask bottom doors.
NS-4-FR **NS-3**	Solid, hydrogenous material with neutron absorption capabilities.
Air Pad Rig Set (Air Pallet)	A device used to lift the Vertical Concrete Cask by using high volume air.
Heavy Haul Trailer	The trailer used to transport the empty or loaded Vertical Concrete Cask.
Margin of Safety	An analytically determined value defined as the "factor of safety" minus 1. Factor of safety is also analytically determined, and is defined as the allowable stress or displacement of a material divided by its actual (calculated) value.

1.1 Introduction

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

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

The Universal Storage System is designed to safely store up to 24 PWR or up to 56 BWR spent fuel assemblies. The fuel specifications and parameters that serve as the design basis are presented in Tables 2.1.1-1 and 2.1.2-1 for PWR and BWR fuel, respectively. The spent fuel considered in the design basis includes fuel assemblies that have different overall lengths. The range of overall lengths of the PWR fuel population is grouped in three classes. To accommodate the three classes, the Universal Storage System principal components, the transportable storage canister, transfer cask and vertical concrete cask, are provided in three different lengths. Similarly, BWR fuel is grouped into two classes, which are also accommodated by two different lengths of the principal components. The class designations of these principal components, and corresponding lengths, are shown on the License Drawings. The allocation of representative fuel by class is shown in Tables 6.2-1 and 6.2-2 for PWR and BWR fuel, respectively.

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

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

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

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

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

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

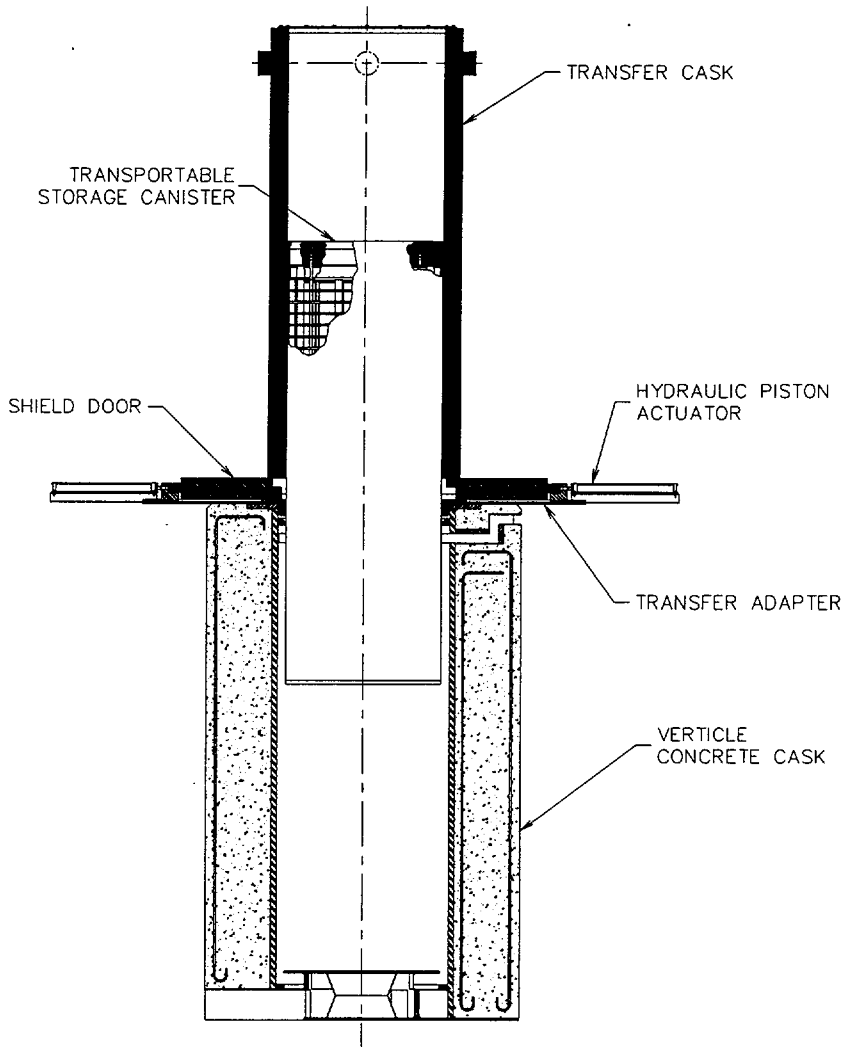


Figure 1.1-2 Transportable Storage Canister Containing PWR Spent Fuel Basket

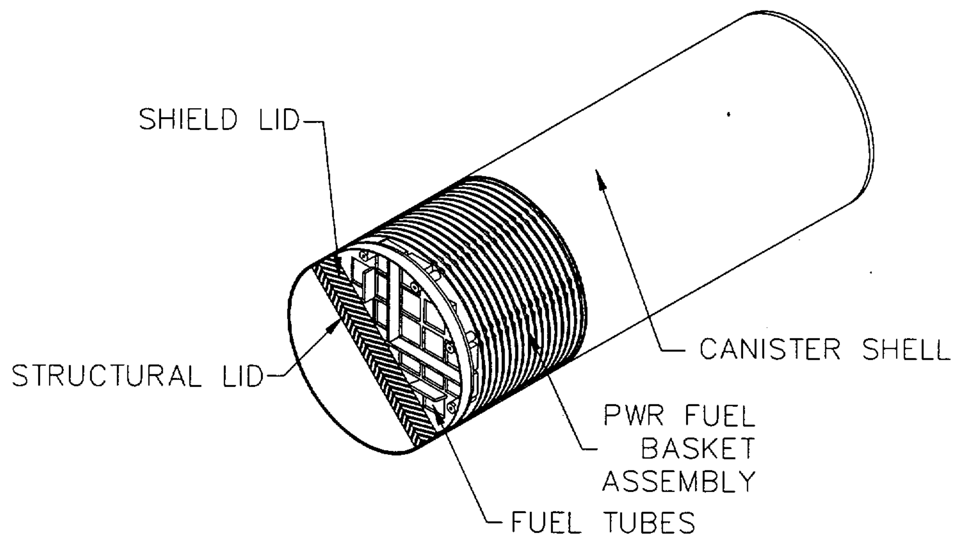
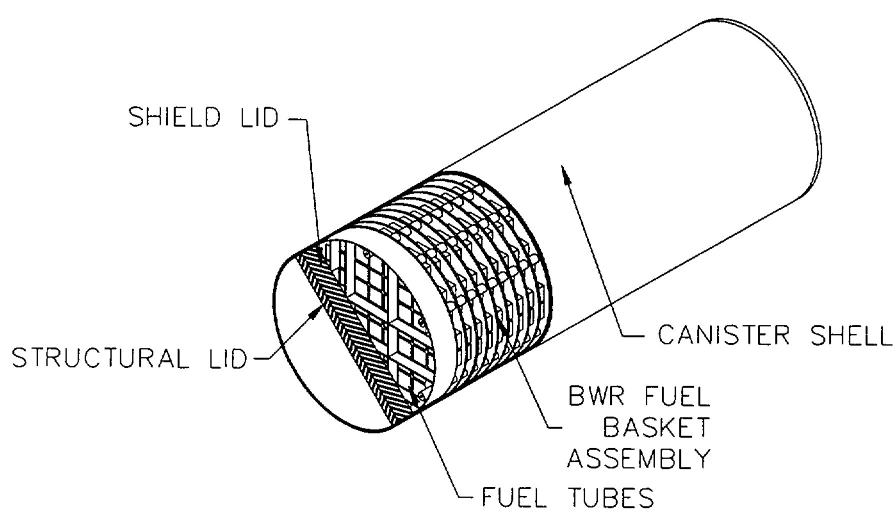


Figure 1.1-3 Transportable Storage Canister Containing BWR Spent Fuel Basket



1.2 General Description of the Universal Storage System

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

1.2.1 Universal Storage System Components

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

The Universal Storage System consists of three principal components:

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

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

Ancillary equipment needed to use the Universal Storage System are:

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

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

1.2.1.1 Transportable Storage Canister

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

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

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

The canister consists of a cylindrical, 5/8 in.-thick Type 304L stainless steel shell with a 1.75 in.-thick Type 304L stainless steel bottom plate and a Type 304 stainless steel shield lid support ring. A basket assembly is placed inside the canister. The shield lid assembly is a 7 in.-thick Type 304 stainless steel disk that is positioned on the shield lid support ring above the basket assembly. The shield lid is welded to the canister after the canister is loaded and removed from the pool. Two penetrations through the shield lid are provided for draining, vacuum drying, and backfilling the canister with helium. The drain penetration has a pipe thread fitting. The drain pipe is threaded into the shield lid after removal of the canister from the spent fuel pool. To facilitate water removal, the pipe extends to within 1/8 in. of the bottom of the canister. The vent penetration in the shield lid is used for water removal and for vacuum drying and backfilling the canister with helium. After the shield lid is welded in place, it is pressure-tested and leak-tested to ensure that the required leak tightness is achieved.

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

The Transportable Storage Canister is designed to the requirements of the ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Division I, Subsection NB [8]. It is fabricated and assembled in accordance with the requirements of Subsection NB to the maximum extent practicable, consistent with the conditions of use. Alternatives to the ASME Code are noted in Table 4-1 in Chapter 12.

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

The design of the transportable storage canister and its fabrication controls allows the canister to be ASME Code stamped in accordance with ASME Code Case N-595 [29].

1.2.1.2 Fuel Baskets

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

1.2.1.2.1 PWR Fuel Basket

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

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

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

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

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

1.2.1.2.2 BWR Fuel Basket

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

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

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

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

1.2.1.3 Vertical Concrete Cask

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

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

The Vertical Concrete Cask forms an annular air passage to allow the natural circulation of air around the canister to remove the decay heat from the spent fuel. The air inlets and outlets are steel-lined penetrations that take nonplanar paths to the concrete cask cavity to minimize radiation streaming. A baffle assembly directs inlet air upward and around the pedestal that supports the canister. The pedestal may incorporate either of two optional baffle configurations as shown on Drawings 790-561 or 790-614. The decay heat is transferred from the fuel assemblies to the tubes in the fuel basket and through the heat transfer disks to the canister wall. Heat flows by radiation and convection from the canister wall to the air circulating through the concrete cask annular air passage and is exhausted through the air outlet vents. This passive cooling system is designed to maintain the peak cladding temperature of the Zircaloy-clad fuel well below acceptable limits during long-term storage. This design also maintains the bulk concrete temperature below 150°F and localized concrete temperatures below 200°F in normal operating conditions.

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

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

1.2.1.4 Transfer Cask

The transfer cask is a heavy lifting device, which is designed, fabricated, and load-tested to meet the requirements of NUREG-0612 [11] and ANSI N14.6 [12]. The transfer cask is provided in either a standard or 100-ton configuration. The 100-ton transfer cask weighs less than the standard transfer cask and is designed to accommodate sites having a 100-ton cask handling crane weight limit. Canister handling, fuel loading and canister closing are operationally identical for either transfer cask configuration.

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

The transfer cask design incorporates a top retaining ring, which is bolted in place to prevent a loaded canister from being inadvertently 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 pins so they cannot inadvertently open. During unloading, the doors are retracted using hydraulic cylinders to allow the canister to be lowered into a concrete cask for storage or into a transport cask. A typical transfer cask is shown in Figure 1.2-2. The principal design parameters of the transfer casks are shown in Table 1.2-7.

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

Standard Transfer Cask

The standard transfer cask is designed for lifting and handling in the vertical orientation only. It has four lifting trunnions, which allows for redundant load path lifting. It incorporates a multiwall (steel/lead/NS-4-FR/steel) design, which limits the contact radiation dose rate to less than 300 mrem/hr. It has a maximum empty weight of approximately 120,000 lbs. The standard transfer cask design is shown in Drawing 790-560.

100-Ton Transfer Cask

The 100-ton transfer cask is also a multiwall design but uses a water jacket to provide neutron radiation shielding. The 100-ton transfer cask has two trunnions for handling in the vertical orientation, but it may also be moved in a horizontal orientation using a wheeled cradle. Horizontal movement can only be used when the transfer cask is empty, when it holds a canister that is empty, except for its basket, and when it holds a canister that is loaded and closed with its structural lid. The 100-ton transfer cask has an empty weight of approximately 100,000 lbs. This weight allows the use of the system in facilities having a handling crane weight limit of 100-tons. The lower weight of the 100-ton transfer cask is partly achieved by reducing the amount of radiation shielding. Consequently, the surface dose rates are higher than those of the standard transfer cask. The 100-ton transfer cask design is shown in Drawing 790-566.

1.2.1.5 Auxiliary Equipment

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

1.2.1.5.1 Transfer Adapter

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

1.2.1.5.2 Air Pad Rig Set

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

1.2.1.5.3 Automatic Welding System

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

1.2.1.5.4 Draining and Drying System

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

1.2.1.5.5 Lifting Jacks

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

1.2.1.5.6 Heavy-Haul Trailer

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

1.2.1.5.7 Helium Leak Test Equipment

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

1.2.1.5.8 Rigging and Slings

Load rated rigging attachments and slings are provided for major components. The rigging attachments are swivel hoist rings that allow attachment of the slings to the hook. All slings are commercially purchased to have adequate safety margin to meet the requirements of ANSI N14.6 and NUREG-0612. The slings include a concrete cask lid sling, concrete cask shield plug sling, canister shield lid sling, loaded canister transfer sling (also used to handle the structural lid), and a canister retaining ring sling. The appropriate rings or eye bolts are provided to accommodate each sling and component.

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

A horizontal handling cradle may be used with the 100-ton transfer cask. The cradle is a wheeled trailer or dolly, which incorporates a rotation fixture for the transfer cask with either an unloaded canister or a loaded canister that is closed with its structural lid.

1.2.1.5.9 Transfer Cask Extension

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

1.2.1.5.10 Temperature Instrumentation

The Vertical Concrete Cask has four air outlets near the top of the cask and four air inlets at the bottom. Each outlet is equipped with a permanent remote temperature detector mounted in the outlet air plenum. The detector is used to measure the outlet air temperature, which can be read at a display device located on the outside surface of the concrete cask or at a remote location.

The detectors are installed on all of the concrete casks at the Independent Spent Fuel Storage Installation (ISFSI) facility.

1.2.1.6 Universal Transport Cask

The Universal Transport Cask is designed to transport the Transportable Storage Canister. The canister, which may contain PWR or BWR spent fuel, is positioned in the Universal Transport Cask cavity by axial spacer(s) at the bottom of the cavity. The spacer(s) are required because the Universal Transport Cask cavity length is 192.5 in. while the lengths of the canisters containing different classes of fuel vary from 175.3 in. to 192.0 inch.

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

1.2.2 Operational Features

In storage, the only active system is for temperature monitoring of the outlet air. This temperature is recorded daily as a check of the thermal performance of the concrete casks. This system does not penetrate the confinement boundary and is not essential to the safe operation of the Universal Storage System.

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

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

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

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

- Lift the transfer cask over the pool and start the flow of clean or filtered pool water to the transfer cask annulus and canister. After the annulus and canister fill, lower the cask to the bottom of the pool.
- Load the selected spent fuel assemblies into the canister and set the shield lid.
- Raise the transfer cask from the pool. Decontaminate the transfer cask exterior as it clears the pool surface. Drain the annulus. Place the transfer cask in the decontamination area.
- Weld the shield lid to the canister shell. Inspect and pressure test the weld. Drain the pool water from the canister. Attach the vacuum system to the drain line, and operate the system to achieve a vacuum.
- Hold the vacuum and backfill with helium to 1 atmosphere. Restart the vacuum system and remove the helium. After achieving vacuum, backfill and pressurize the canister with helium.
- Helium leak check the shield lid welds. Vent the helium pressure to 1 atmosphere (absolute). Install the vent and drain port covers and weld them to the shield lid. Inspect the port cover welds.
- Install the structural lid and weld it to the canister shell. Inspect the structural lid weld. Install the hoist rings, and attach the canister lifting sling. Install the adapter plate on the concrete cask.
- Lift the transfer cask to the top of the concrete cask and set it on the adapter plate. (See Figure 1.2-5). Ensure that the bottom door hydraulic actuators are engaged.
- Attach the canister lifting slings to the crane hook and lift the canister.
- Open the bottom doors of the transfer cask.
- Lower the canister into the concrete cask (See Figure 1.2-6). Remove the canister lifting slings.
- Remove the transfer cask and adapter plate.
- Install the shield plug and lid on the concrete cask.
- Move the loaded concrete cask to the storage pad.

- Using the air pad rig set and a towing vehicle, move the concrete cask to its designated location on the storage pad.

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

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

Figure 1.2-1 Vertical Concrete Cask

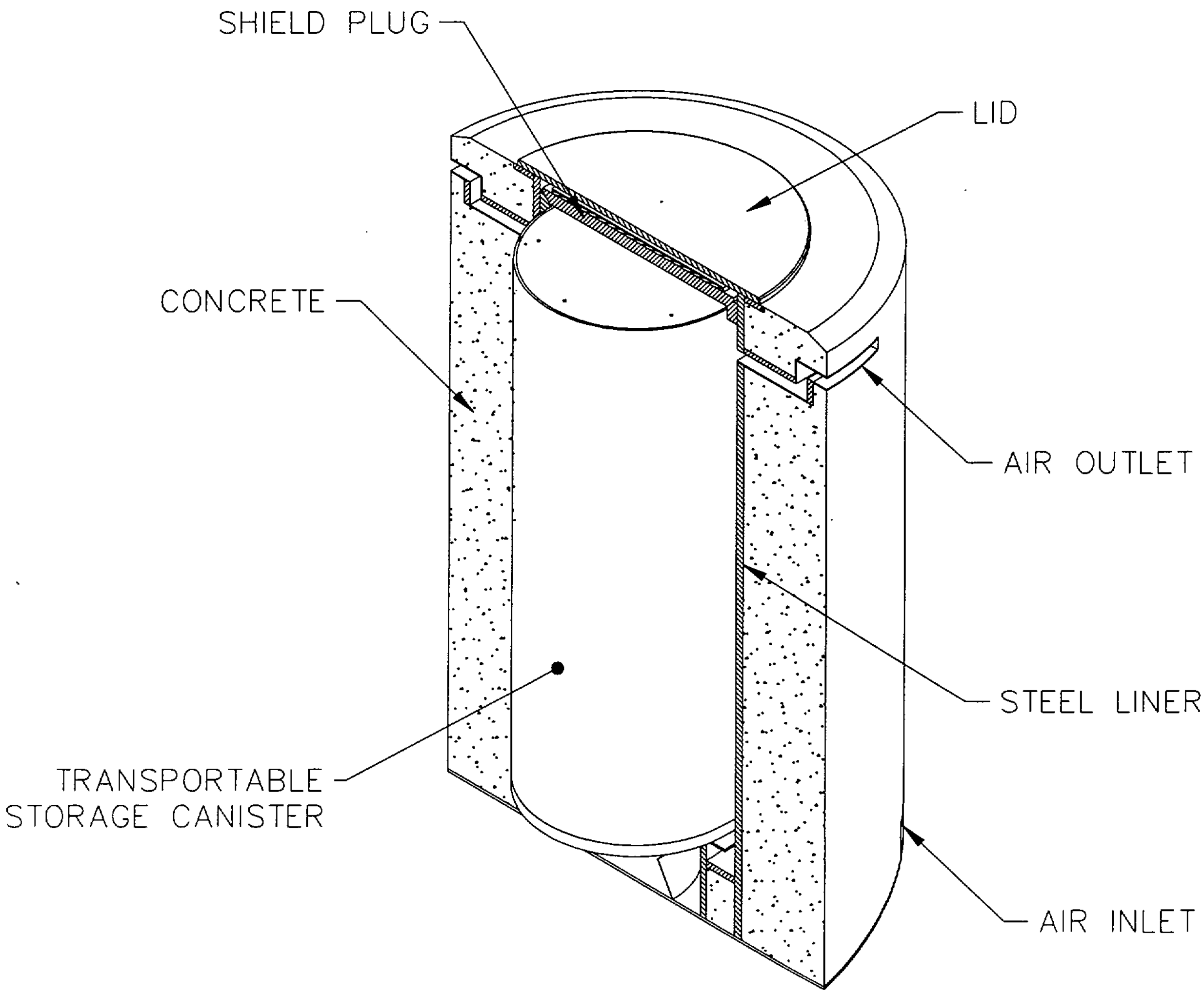
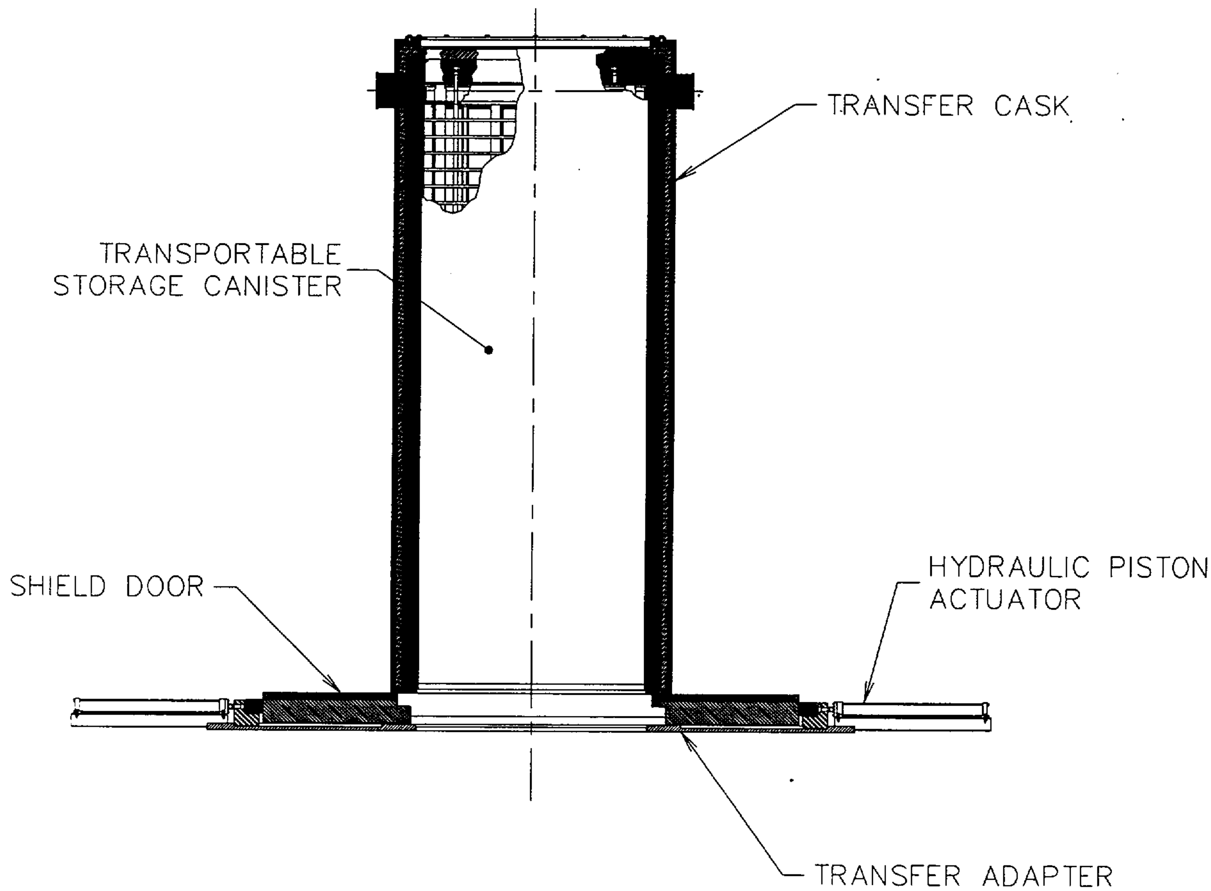


Figure 1.2-2 Transfer Cask



Typical Transfer Cask with Transfer Adapter Plate

Figure 1.2-3 Transport Configuration of the Universal Transport Cask

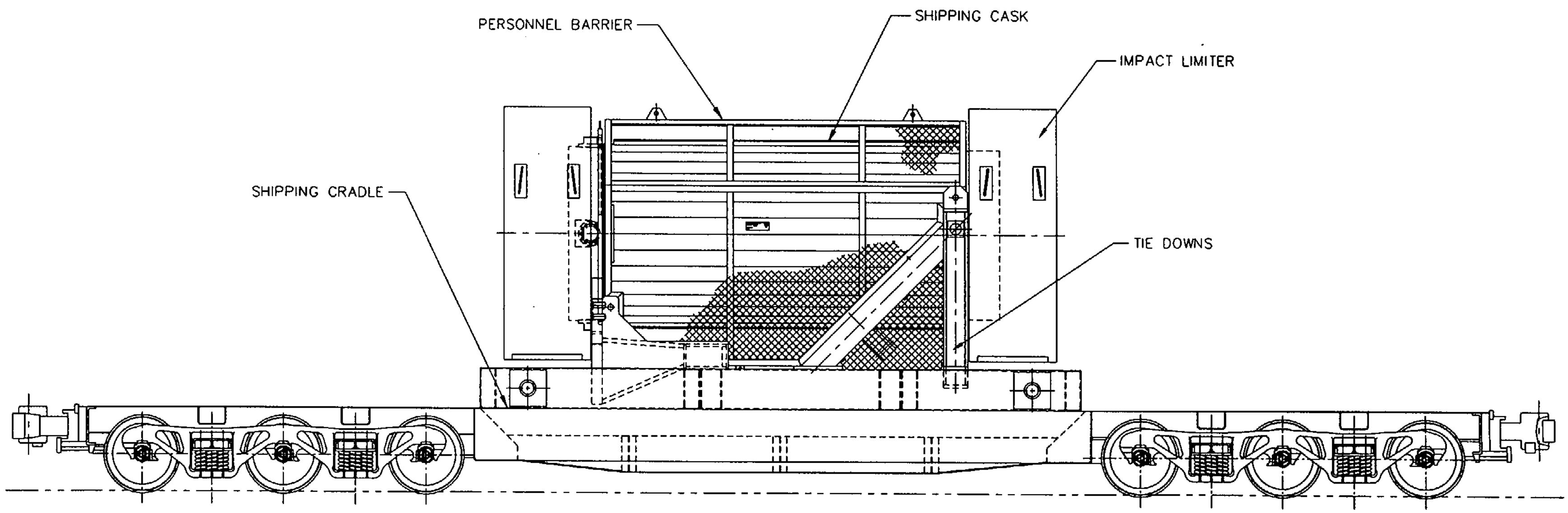
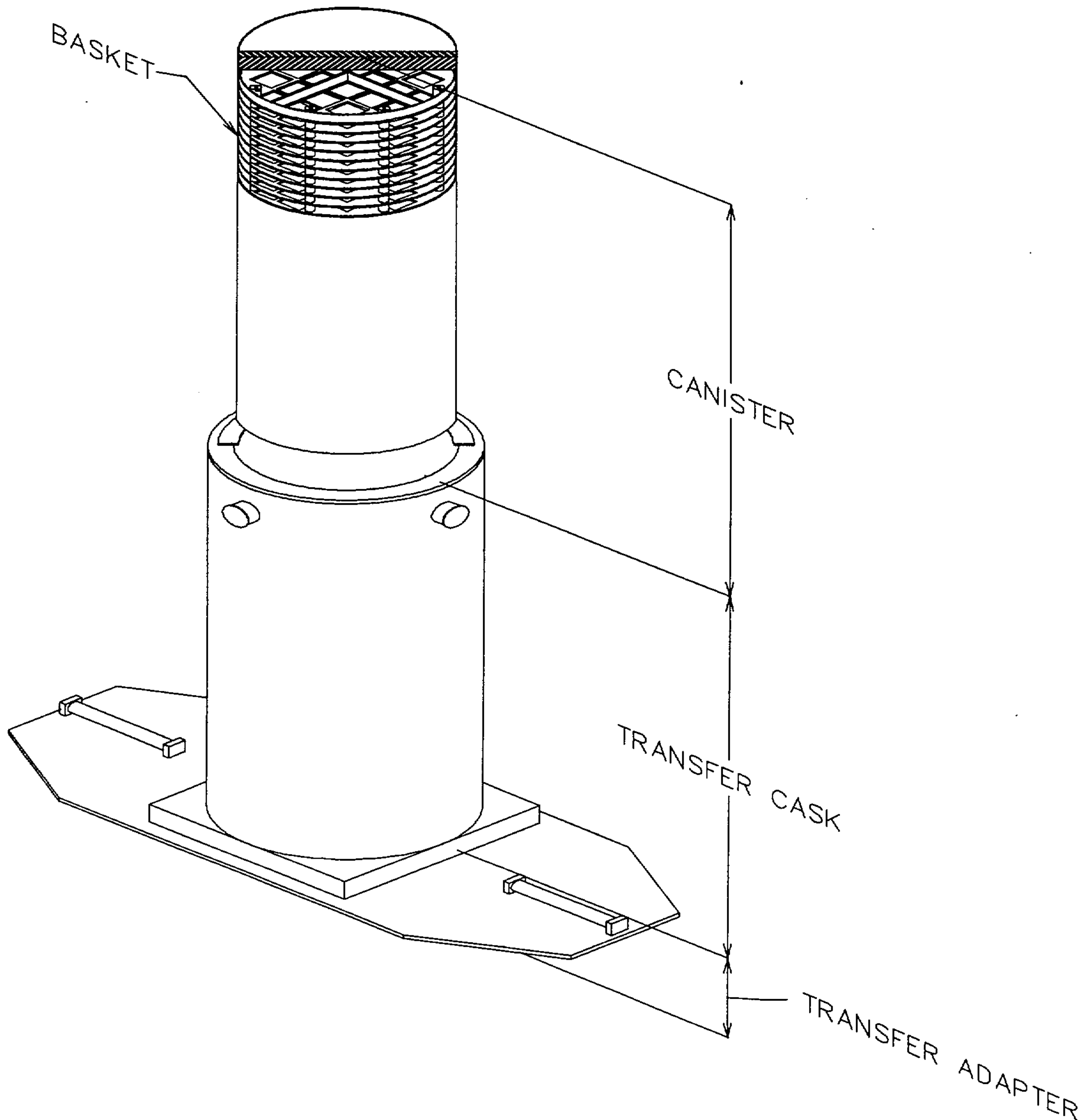
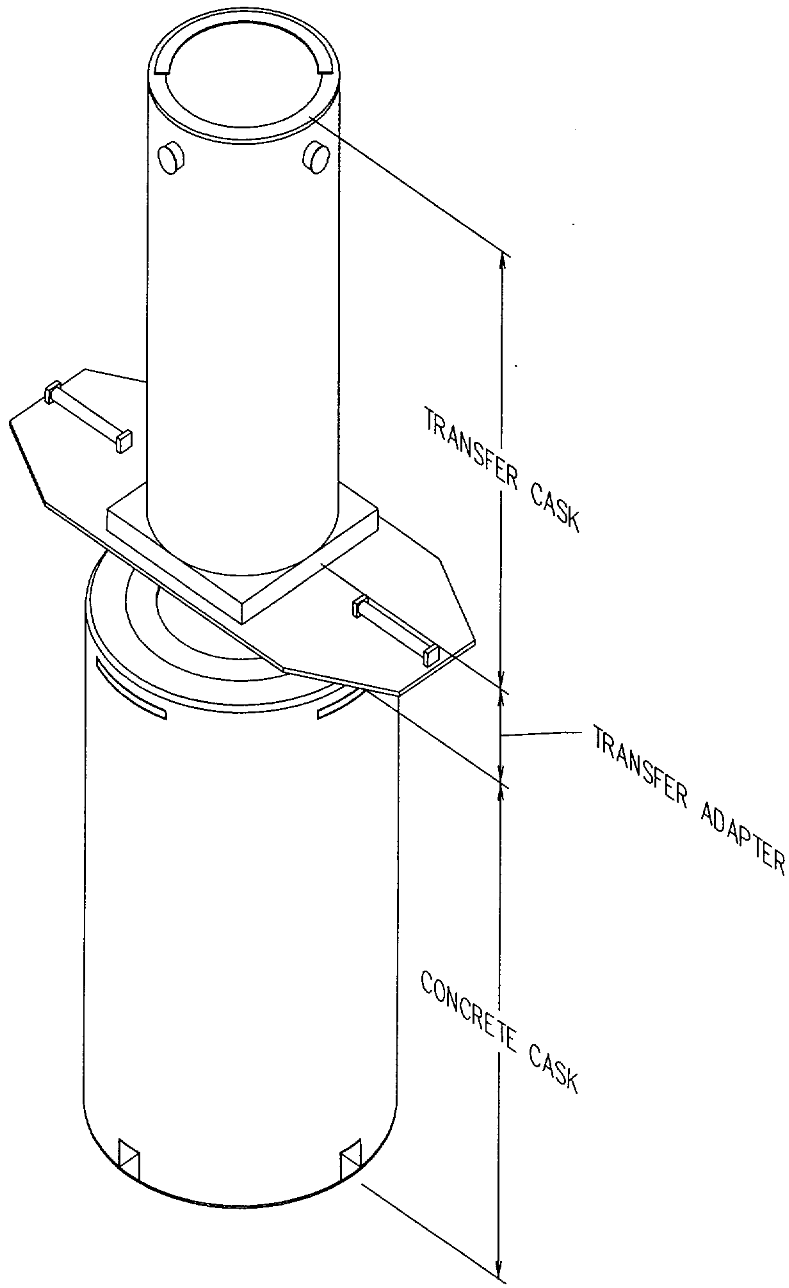


Figure 1.2-4 Transfer Cask and Canister Arrangement



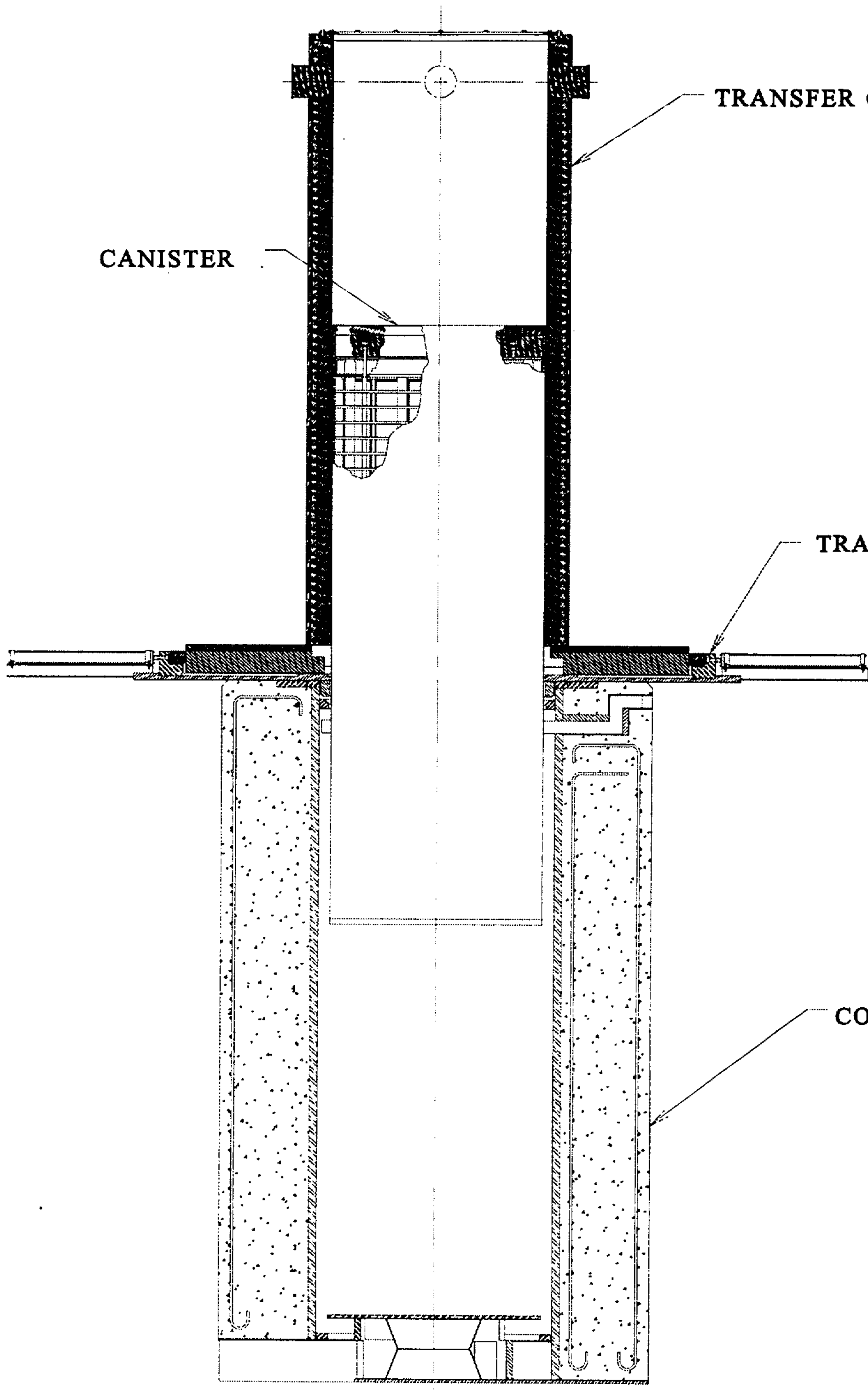
Note: Standard transfer cask shown. The 100-ton transfer cask is similar.

Figure 1.2-5 Vertical Concrete Cask and Transfer Cask Arrangement



Note: Standard transfer cask shown. The 100-ton transfer cask is similar.

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



Note: Standard transfer cask shown. 100-ton transfer cask is similar.

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

Design Characteristic	Value (in.)	Material
Transportable Storage Canister		
Shell thickness	0.625	Type 304L Stainless Steel
Shell bottom thickness	1.75	Type 304L Stainless Steel
Shield lid thickness	7	Type 304 Stainless Steel
Structural lid thickness	3	Type 304L Stainless Steel
Canister Fuel Basket		
Top weldment PWR thickness	1.25	Type 304 Stainless Steel
Bottom weldment PWR thickness	1.0	Type 304 Stainless Steel
Top and bottom weldment BWR thickness	1.0	Type 304 Stainless Steel
Support disks thickness		
- PWR	0.5	Type 17-4 PH Stainless Steel
- BWR	0.625	SA-533, Type B Class 2 Carbon Steel
Heat transfer disk thickness	0.5	Type 6061-T651 Aluminum Alloy
Fuel tube dimensions		
- PWR (inside)	8.8 x 8.8	Type 304 Stainless Steel Enclosing neutron absorber
- BWR Standard (inside)	5.9 x 5.9	Type 304 Stainless Steel Enclosing neutron absorber
- BWR Over-Sized Fuel (inside)	6.05 x 6.05	Type 304 Stainless Steel Enclosing neutron absorber
Spacer(s) diameter	3.0	Type 304 Stainless Steel
Tie rod diameter		
- PWR	1-5/8	Type 304 Stainless Steel
- BWR	1-5/8	Type 304 Stainless Steel

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

Design Characteristic	Value (in.)	Material
Standard Transfer Cask		
Outer Shell	1.25 x 85.3 dia.	ASTM A588 Low Alloy Steel
Inner Shell	0.75 x 67.8 dia.	ASTM A588 Low Alloy Steel
Retaining Ring	0.75 x 77.1 dia.	ASTM A588 Low Alloy Steel
Trunnions	10.0 dia.	A350 LF2 Low Alloy Steel
Bottom Plate	1.0 thick plate	ASTM A588 Low Alloy Steel
Top Plate	2.0 thick plate	ASTM A588 Low Alloy Steel
Shield Doors	9.0 thick	A350 LF2 Low Alloy Steel and NS-4-FR
Door Rails	9.4 x 6.5	A350 LF2 Low Alloy Steel
Gamma Shield	3.75 thick	ASTM B29, Chemical Copper Grade Lead
Neutron Shield	3.0 thick	NS-4-FR, Solid Synthetic Polymer
100-Ton Transfer Cask		
Outer Shell	0.5 x 83.3 dia.	ASTM A240 Type 304
Inner Shell	1.25 x 67.8 dia.	ASTM A588 Low Alloy Steel
Retaining Ring	0.75 x 77.1 dia.	ASTM A588 Low Alloy Steel
Trunnions	8.0 dia.	A350 LF2 Low Alloy Steel
Bottom Plate	1.0 thick plate	ASTM A588 Low Alloy Steel
Top Plate	2.0 thick plate	ASTM A588 Low Alloy Steel
Shield Doors	8.75 thick	A350 LF2 Low Alloy Steel and NS-4-FR
Door Rails	9.4 x 6.5	A350 LF2 Low Alloy Steel
Gamma Shield	2.2 thick	ASTM B29, Chemical Copper Grade Lead
Neutron Shield	2.5 thick	Water
Transfer Adapter		
Base Plate	2.0 thick plate	ASTM A36 Carbon Steel
Locating Ring	2.75 wide x 73.75	ASTM A36 Carbon Steel

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

Design Characteristic	Value (in.)	Material
Vertical Concrete Cask		
Weldment Structure		
Shell	2.5 thick x 79.50 dia	ASTM A36 Carbon Steel
Top Flange	2.0 thick x 101.40 dia.	ASTM A36 Carbon Steel
Support Ring	2.5 thick x 74.50 dia.	ASTM A36 Carbon Steel
Base Plate	2.0 thick x 67.50 dia.	ASTM A36 Carbon Steel
Concrete Cask		
Concrete Shell	28.3 thick x 136 dia.	Type II Portland Cement
Shield Plug (NS-4-FR)	5.13 x 74.0 dia.	ASTM A36 Carbon Steel and NS-4-FR
Shield Plug (NS-3)	5.63 x 74.0 dia.	ASTM A36 Carbon Steel and NS-3
Cask Lid	1.50 thick x 85.6 dia.	ASTM A36 Carbon Steel
Rebar	Various Lengths	ASTM A615, GR60, Carbon Steel

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

Canister Parameter	Value
Canister Shell	
Outside Diameter (in.)	67.1
Thickness (in.)	0.625
Overall Length (in.)	
Class 1 (PWR)	175.1
Class 2 (PWR)	184.2
Class 3 (PWR)	191.8
Class 4 (BWR)	185.6
Class 5 (BWR)	190.4
Capacity (No. of fuel assemblies)	
Classes 1 - 3 (PWR)	24
Classes 4 - 5 (BWR)	56
Maximum Heat Load (kW)	
PWR	23.0
BWR	23.0
Maximum Long-Term Fuel Cladding Temperature - 5-year cooled fuel (°F [°C])	
Classes 1 - 3 (PWR)	716 (380)
Classes 4 - 5 (BWR)	716 (380)
Internal Atmosphere	Helium

Table 1.2-3 Transportable Storage Canister Fabrication Specification Summary

Materials

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

Welding

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

Fabrication

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

Packaging

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

Quality Assurance

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

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

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

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

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

Table 1.2-6 Vertical Concrete Cask Fabrication Specification Summary

Materials

- Concrete mix shall be in accordance with the requirements of ACI 318 and ASTM C94 [19].
- Type II Portland Cement, ASTM C150 [20].
- Fine aggregate ASTM C33 [21] and C637 [22].
- Coarse aggregate ASTM C33.
- Admixtures
  - Water Reducing and Superplasticizing ASTM C494 [23].
  - Pozzolanic Admixture (Loss on Ignition 6% or less) ASTM C618 [24].
- Compressive Strength 4000 psi at 28 days.
- Specified Air Entrainment 3% - 6%.
- All steel components shall be of material as specified in the referenced drawings.

Welding

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

Construction

- Specimens shall be obtained or prepared for each batch or truck load of concrete per ASTM C172 [26] and ASTM C31 [27].
- Test specimens shall be tested in accordance with ASTM C39 [28].
- Formwork shall be in accordance with ACI 318.
- All sidewall formwork and shoring shall remain in place for at least 24 hours.
- Grade, type, and details of all reinforcing steel shall be in accordance with the referenced drawings.
- Embedded items shall conform to ACI 318 and the referenced drawings.
- The placement of concrete shall be in accordance with ACI 318.
- Surface finish shall be in accordance with ACI 318.

Quality Assurance

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

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

Parameter	Transfer Cask Configuration	
	Standard	**100-Ton**
Inside Diameter (in.)	67.8	67.8
Outside Diameter (in.)	85.3	85.3
Cavity Height (nominal) (in.)		
Class 1	177.3	177.3
Class 2	186.4	186.4
Class 3	194.0	194.0
Class 4	187.8	187.8
Class 5	192.6	192.6
Empty Weight (nominal) (lbs)		
Class 1	112,300	91,600
Class 2	117,300	95,500
Class 3	121,500	98,800
Class 4	118,100	96,100
Class 5	120,700	98,200
Side Wall Dose Rate (Average, mrem/hr)	< 300	< 1,300

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1.3 Universal Storage System Contents

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

1.3.1 Design Basis Spent Fuel

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

The Universal Storage System fuel limits are:

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

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

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

4. The maximum UO_2 weight (MTU) shall not exceed 11.53 MTU for PWR and 11.08 MTU for BWR fuel assemblies.

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

6. The maximum initial enrichment of the PWR fuel is based on a pool/canister water boron content of at least 1,000 parts per million for some fuel parameter combinations. The maximum initial enrichment of the BWR fuel is defined as the maximum initial peak planar-

average enrichment. The initial peak planar-average enrichment is the maximum initial peak planar-average enrichment at any height along the axis of the fuel assembly. The initial peak planar-average may be higher than the bundle average enrichment value that appears in fuel design or plant documents. Unenriched fuel assemblies are not evaluated and are not included as a proposed content.

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

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

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

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

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

1.3.2 Site Specific Spent Fuel

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

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

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

1.3.2.1 Maine Yankee Site Specific Spent Fuel

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

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

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

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

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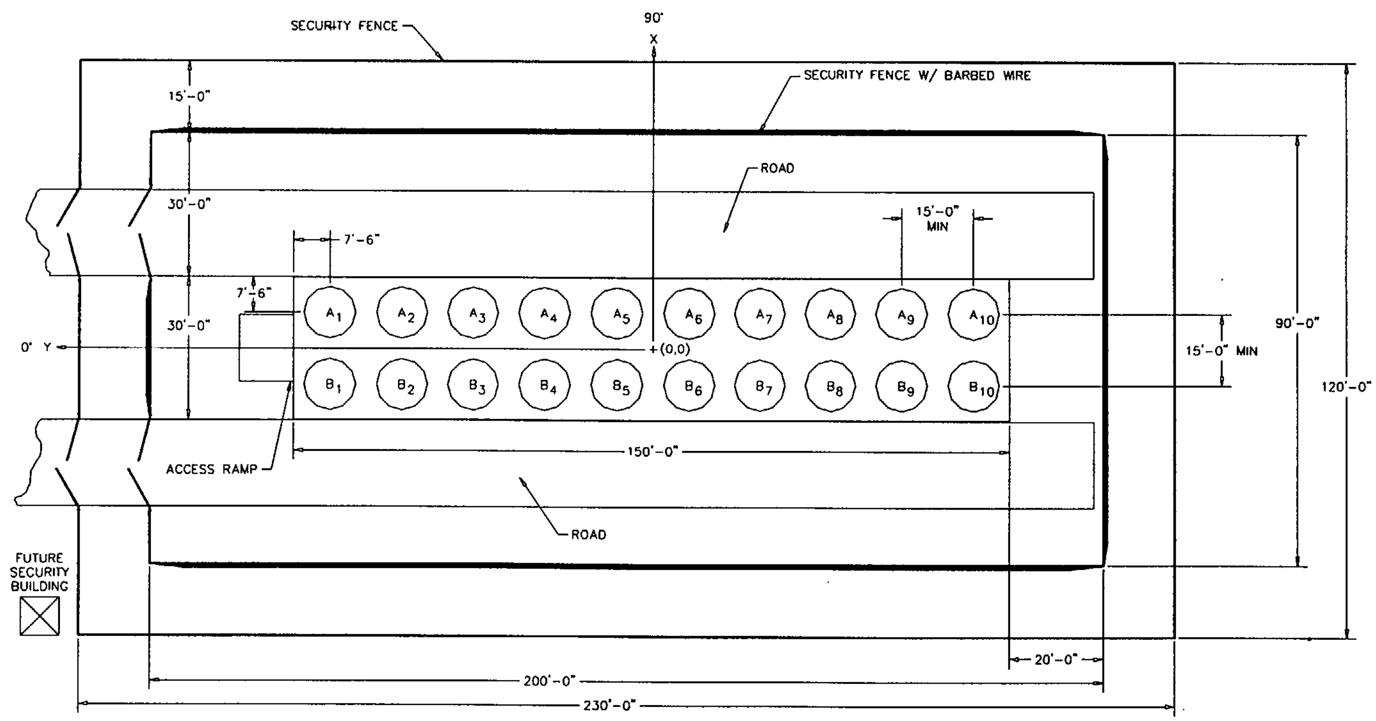
1.4 Generic Vertical Concrete Cask Arrays

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

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

Figure 1.4-1 Typical ISFSI Storage Pad Layout



1.5 UMS® Universal Storage System Compliance With NUREG-1536

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

Table 1.5-1 NUREG-1536 Compliance Matrix

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

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

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

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

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

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

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

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

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

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

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

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

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **c. Thermal**	Each spent fuel storage or handling system must be designed with a heat removal capability having testability and reliability consistent with its importance to safety. [10 CFR 72.128(a)(4)]	The applicant should provide a general discussion of the proposed heat removal mechanisms, including the reliability and verifiability of such mechanisms and any associated limitations. All heat removal mechanisms should be passive and independent of intervening actions under normal and off-normal conditions.	The testability of the heat removal capability of the storage system is described in Section 2.3.3.2. The reliability of the heat removal system is demonstrated in Chapter 4, and operating limits are established in Chapter 12 consistent with the temperature monitoring and routine surveillance described in Section 2.3.3.2 and the Technical Specifications to ensure continued safe operation.
	The DCSS must be designed to provide adequate heat removal capacity without active cooling systems. [10 CFR 72.236(f)]		As shown by the results of the thermal evaluation of the system reported in Tables 4.1-4 and 4.1-5, the storage system provides adequate heat removal through the passive cooling design features described in Section 1.2.1.3.

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

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **d. Shielding/Confinement/ Radiation Protection**	The proposed DCSS design must provide radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 72.106. [10 CFR 72.126(a), 72.128(a)(2), 72.128(a)(3), and 72.236(d)]	The applicant should describe those features of the cask that protect occupational workers and members of the public against direct radiation dosages and releases of radioactive material, and minimize the dose after any off-normal or accident conditions.	The confinement design features are described in Section 2.3.2.1, while the radiation shielding design features are described in Section 2.3.5.
	During normal operations and other anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to (1) planned discharges to the general environment of radioactive materials except radon and its decay products, (2) direct radiation from operations of the ISFSI or monitored retrievable storage (MRS), and (3) any other radiation from uranium fuel cycle operations within the region. [10 CFR 72.24(d), 72.104(a), and 72.236(d)]		Section 10.4 presents the necessary minimum site boundary distances from an array of loaded storage systems to meet the controlled area dose limits.
	Any individual located at or beyond the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design-basis accident. The minimum distance from the spent fuel handling and storage facilities to the nearest boundary of the controlled area shall be 100 meters. [10 CFR 72.24(d), 72.24(m), 72.106(b), and 36(d)]		As stated in Section 10.2.2, there is no postulated accident condition that would result in a release of radioactive materials. Therefore, the accident dose limit is met.
	The DCSS must be designed to provide redundant sealing of confinement systems. [10 CFR 72.236(e)]		The redundant sealing features of the confinement system are presented in Section 2.3.2.1.
	Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. [10 CFR 72.122(h)(4) and 72.128(a)(1)]		As described in Section 2.3.1, the system is fully welded and can operate through all postulated normal, off-normal, and accident events while confining of the stored radioactive material. Therefore, continuous monitoring is not required.
	The DCSS design must include inspections, instrumentation and/or control (I&C) systems to monitor the SSC that are important to safety over anticipated ranges for normal and off-normal operation. In addition, the applicant must identify those control systems that must remain operational under accident conditions. [10 CFR 72.122(i)]		LCO 3.1.2 in Chapter 12 specifies the surveillance requirements for the system under normal conditions and after an accident, respectively. These activities are specified to ensure that the system is operated within its design parameters at all times.

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

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

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

Chapter 2 - Principal Design Criteria			
Area	**Requirement**	**Acceptance Criteria**	**Description of Compliance**
3. Design Criteria for Safety Protection Systems **f. Operating Procedures**	The DCSS must be compatible with wet or dry spent fuel loading and unloading procedures. [10 CFR 72.236(h)] Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)] The DCSS must be designed to minimize the quantity of radioactive waste generated. [10 CFR 72.24(f) and 72.128(a)(5)] The applicant must describe equipment and processes proposed to maintain control of radioactive effluents. [10 CFR 72.24(l)(2)] To the extent practicable, the DCSS must be designed to facilitate decontamination. [10 CFR 72.236(I)] The applicant must establish operational restrictions to meet the limits defined in 10 CFR Part 20 and to ensure that radioactive materials in effluents and direct radiation levels associated with ISFSI operations will remain as low as is reasonably achievable (ALARA). [10 CFR 72.24(e) and 72.104(b)]	The applicant should provide potential licensees with guidance regarding the content of normal, off-normal, and accident response procedures. Cautions regarding both loading, unloading, and other important procedures should be mentioned here. Applicants may choose to provide model procedures to be used as an aid for preparing detailed site-specific procedures.	The operating procedures for the system are presented in Chapter 8, and include procedures for wet loading and unloading operations. Discussion is provided on the development of operating procedures for dry cask handling facilities. The procedures include methods for retrieving the spent fuel after storage for off-site transport or for return to the spent fuel pool. The decommissioning considerations of the system are described in Section 2.4. Operation of the system generates no radioactive waste, other than a limited amount of protective clothing and tools used during loading operations that could be easily disposed or decontaminated. The radiation protection design features of the system are presented in Section 2.3.5. Operating procedures for the system include provisions for controlling potential effluents from the system. The canister is designed to facilitate decontamination, as described in Section 2.3.5.3. Fuel assembly specifications are provided in Section 2.1 to ensure that doses from direct radiation are maintained ALARA. There are no radioactive effluents from the canister or concrete cask in storage operations.

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

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

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

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

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

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

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

<table>
<tr><th colspan="4">Chapter 3 – Structural Evaluation</th></tr>
<tr><th>Area</th><th colspan="2">Regulatory Requirement</th><th>Description of Compliance</th></tr>
<tr><td rowspan="6">2. Radiation, Shielding, Confinement, and Subcriticality</td><td colspan="2">Radiation shielding, confinement, and subcriticality must meet the regulatory requirements defined in 10 CFR 72.24(d); 10 CFR 72.124(a); and 10 CFR 72.236(c), (d), and (l).</td><td rowspan="2">The margins of safety for normal conditions are listed in Sections 3.4.4.1 and 3.4.4.2. Off-normal and accident condition margins of safety are presented in Sections 11.1 and 11.2, respectively. Adequate safety margins are maintained for all events, ensuring the mitigation of accident consequences, and maintaining the shielding, confinement, and criticality analyses presented in the SAR.</td></tr>
<tr><td>10 CFR 72.24(d)</td><td>Contents of Application: Margins of Safety / Mitigation of Accident Consequences</td></tr>
<tr><td>10 CFR 72.124(a)</td><td>Criteria for Nuclear Criticality Safety: Design for Criticality Safety</td><td>The nuclear criticality safety design of the system is discussed in Sections 2.3.4 and 6.1.</td></tr>
<tr><td>10 CFR 72.236(c)</td><td>Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Subcritical Configuration</td><td>Subcriticality of the system is demonstrated in Section 6.4.3.</td></tr>
<tr><td>10 CFR 72.236(d)</td><td>Specific Requirements for Spent Fuel Storage Cask Approval: Radiation Protection</td><td>Radiation protection of the system is demonstrated in Sections 5.4, 10.3 and 10.4.</td></tr>
<tr><td>10 CFR 72.236(l)</td><td>Specific Requirements for Spent Fuel Storage Cask Approval: Maintain Confinement</td><td>Confinement of the spent fuel is discussed in Sections 7.2 and 7.3.</td></tr>
<tr><td>3. Removal of Spent Fuel</td><td colspan="2">As stated in 10 CFR 72.122(f) and (h)(l), the storage system design must allow ready retrieval of spent fuel without posing operational safety problems.</td><td>The system is not adversely affected by normal, off-normal, or accident condition events as demonstrated in Sections 3.4.4.1, 3.4.4.2, 11.1 and 11.2. Operating procedures for removing spent fuel from the system are presented in Sections 8.2 and 8.3.</td></tr>
<tr><td>4. Design Basis Earthquake</td><td colspan="2">As stated in 10 CFR 72.102(f), the design-basis earthquake (DBE) must be equal to or greater than the safe-shutdown earthquake (SSE) of nuclear plant sites previously evaluated under 10 CFR Part 100 or, in the case of sites licensed before the implementation of 10 CFR Part 100, developed under Topic III-2 of the Systematic Evaluation Program (SEP).</td><td>As described in Section 2.2.3.1, the system is designed for a seismic event that meets the regulatory requirements.</td></tr>
<tr><td>5. Minimum Lifetime</td><td colspan="2">As stated in 10 CFR 72.24(c) and 10 CFR 72.236(g), the analysis and evaluation of the structural design and performance must demonstrate that the cask system will allow storage of spent fuel for a minimum of 20 years with an adequate margin of safety.</td><td>Section 1.1 and Table 2-1 specify a 50-year design life for the system.</td></tr>
</table>

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

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

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

Chapter 3 – Structural Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. Confinement Cask **a. Steel Confinement Cask**	The structural design, fabrication, and testing of the confinement system and its redundant sealing system should comply with an acceptable code or standard, such as Section III of the Boiler and Pressure Vessel Code (B&PV) promulgated by the American Society of Mechanical Engineers (ASME). (The NRC has accepted use of either Subsection NB or Subsection NC of this code.) Other design codes or standards may be acceptable depending on their application. i. The NRC staff evaluates the proposed limitations on allowable stresses and strains in the confinement cask, reinforced concrete components, system components important to safety, and other components subject to review, by comparison with those specified in applicable codes and standards. Where certain proposed load combinations will exceed the accepted limits for localized points on the structure, the applicant should provide adequate justification to show that the deviation will not affect the functional integrity of the structure. ii. The NRC has accepted the use of applicable subsections of the ASME B&PV Code, Division 1, for components used within the confinement cask but not integrated with it. This includes the "basket" structure used in casks to restrain and position multiple fuel elements.	As specified in Section 3.1.2, the canister and basket structure are designed in accordance with the ASME Code, Section III, Division I, 1995 Edition. The canister is designed in accordance with Subsection NB of the ASME Code, while the basket structure is designed in accordance with Subsection NG criteria. A list of alternatives from the ASME code is provided in Table 4-1 of Chapter 12.

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

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

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

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

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

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

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

Chapter 4 – Thermal Evaluation		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Minimum Lifetime	10 CFR Part 72 requires an analysis and evaluation of DCSS thermal design and performance to demonstrate that the cask will permit safe storage of the spent fuel for a minimum of 20 years.	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Tables 4.1-4 and 4.1-5 demonstrate that the system's temperatures are maintained within their allowable limits.
2. Spent Fuel Cladding Protection	The spent fuel cladding must be protected against degradation that may lead to gross ruptures.	Tables 4.1-4 and 4.1-5 demonstrate that the fuel cladding temperatures are maintained within allowable limits.
3. Thermal Structures, Systems, and Components	Thermal structures, systems, and components important to safety must be described in sufficient detail to permit evaluation of their effectiveness. Applicable thermal requirements are identified, in part, in 10 CFR 72.24(c)(3), 72.24(d), 72.122(h)(1), 72.122(l), 72.128(a)(4), 72.236(f), 72.236(g), and 72.236(h).	The discussion of the thermal design features of the system is presented in Section 4.1.
	10 CFR 72.24(c)(3) Contents of Application: Descriptions of Components Important to Safety 10 CFR 72.24(d) Contents of Application: Margins of Safety / Mitigation of Accident Consequences	Tables 4.1-4 and 4.1-5 demonstrate that the temperatures of SSCs are maintained within allowable limits for all components of the system, including the fuel cladding. Therefore, the system is not adversely affected by normal, off-normal, or accident condition events.
	10 CFR 72.122(h)(1) Overall Requirements: Confinement Barriers and Systems 10 CFR 72.122(l) Overall Requirements: Retrievability	The temperatures of the system are maintained within allowable limits, and do not preclude retrieval of spent fuel from the system.
	10 CFR 72.128(a)(4) Criteria for Spent Fuel Storage and Handling: Testable Heat Removal Capacity	As specified in Sections 9.1.7 and in LCO 3.1.2, the air temperature at the air outlets is measured to ensure proper operation of the passive heat removal system.
	10 CFR 72.236(f) Specific Requirements for Spent Fuel Storage Cask Approval: Passive Heat Removal 10 CFR 72.236(g) Specific Requirements for Spent Fuel Storage Cask Approval: Minimum 20-year Lifetime	Section 1.1 and Table 2-1 specify a 50-year design life for the system. Tables 4.1-4 and 4.1-5 demonstrate that the system's temperatures are maintained within their allowable limits.
	10 CFR 72.236(h) Specific Requirements for Spent Fuel Storage Cask Approval: Wet/Dry Loading and Unloading Compatibility	The operating procedures for the system, presented in Chapter 8, include procedures for wet and dry loading and unloading operations. A discussion is provided for development of dry loading and unloading procedures for dry cask handling facilities.

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

Chapter 4 – Thermal Evaluation		
Area	**Acceptance Criteria**	**Description of Compliance**
1. **Long-term Cladding Temperatures**	Fuel cladding (zircalloy) temperature at the beginning of dry cask storage should generally be below the anticipated damage-threshold temperatures for normal conditions and a minimum of 20 years of cask storage (Refs. 13 and 14). Ref 13: UCID-21181, "Spent Fuel Cladding Integrity During Dry Storage" Ref 14: PNL-6189, "Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircalloy clad fuel Rods in Inert Gas"	As shown in Tables 4.1-4 and 4.1-5, the fuel cladding temperatures are maintained below allowable temperature limits for zircaloy clad fuel as determined in accordance with PNL-6189.
2. **Short Term Cladding Temperatures**	Fuel cladding temperature should generally be maintained below 570 °C (1058 °F) for short-term accident conditions, short-term off-normal conditions, and fuel transfer operations (e.g., vacuum drying of the cask or dry transfer). (PNL-4835)	As shown in Tables 4.1-4 and 4.1-5, the fuel cladding temperatures are maintained below 570°C (1058°F) for short term off-normal or accident condition events.
3. **Maximum Internal Pressure**	The maximum internal pressure of the cask should remain within its design pressures for normal, off-normal, and accident conditions assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods.	The maximum normal condition pressure calculation is presented in Section 4.4.5. The accident condition pressure calculation is presented in Section 11.2.1. The off-normal condition is bounded by the accident condition, which assumes 100% failure of the cladding.
4. **Maximum Material Temperatures**	Cask and fuel materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions in order to enable components to perform their intended safety functions.	Tables 4.1-4 and 4.1-5 demonstrate that the temperatures are maintained within allowable limits for all components of the system, including the fuel cladding. Therefore, the system is not adversely affected by normal, off-normal, or accident condition events.
5. **Fuel Cladding Protection**	For each fuel type proposed for storage, the DCSS should ensure a very low probability (e.g., 0.5 percent per fuel rod) of cladding breach during long-term storage.	As concluded in PNL-6189 (Zircaloy), the probability of cladding breach is very low when the cladding temperature is maintained below allowable limits.
6. **Long-Term Cladding Damage**	Fuel cladding damage resulting from creep cavitation should be limited to 15 percent of the original cladding cross-sectional area during dry storage. (UCID-21118)	The maximum fuel cladding temperatures are determined in accordance with PNL-6189 (Zircaloy). Calculations were not performed using the methodology of UCID-21181.
7. **Passive Cooling**	The cask system should be passively cooled. [10 CFR 72.236(f)]	As stated in Sections 1.2 and 4.1, the system is passively cooled.
8. **Thermal Operating Limits**	The thermal performance of the cask should be within the allowable design criteria specified in SAR Section 2 (e.g., materials, decay heat specifications) and SAR Section 3 (e.g., thermal stress analysis) for normal, off-normal, and accident conditions.	The thermal stress analyses of the canister and Vertical Concrete Cask for normal conditions are provided in Sections 3.4.4.1.1 and 3.4.4.2.3, respectively. The system is evaluated for off-normal thermal loading in Section 11.1.2, and the system is analyzed for accident thermal loading in Sections 11.2.6, 11.2.7 and 11.2.13.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Chapter 8 – Operating Procedures		
Area	**Regulatory Requirement**	**Description of Compliance**
Health and Safety	1. The applicant must develop operating procedures that adequately protect health and minimize danger to life or property. [10 CFR 72.40(a)(5)]	Operating procedures are provided in Chapter 8. Notes and Cautions are listed among the steps to emphasize steps important to maintaining health and safety.
ALARA	2. The applicant must establish operational restrictions to meet the regulatory requirements of 10 CFR Part 20 and objective limits that are as low as is reasonably achievable (ALARA) for radioactive materials in effluents and direct radiation levels associated with ISFSI operations. [10 CFR 72.104(b) and 10 CFR 72.24(e)]	Section 8.0 specifies that the procedures are developed to maintain occupational dose ALARA. Automated welding systems and temporary shielding are utilized to minimize worker dose during canister loading operations. Section 8.4.3.4 specifies maximum external dose rates to maintain reasonable dose level within a cask array for routine surveillance and inspection activities.
Control of Radioactive Effluents	3. The applicant must describe all equipment and processes used to maintain control of radioactive effluents. [10 CFR 72.24(l)(2)]	As described in Section 8.0, there are no radioactive effluents in routine operations other than pool water and helium gas that are removed from the canister. These effluents are routinely handled in Licensee operations.
Written Procedures	4. The general licensee shall conduct activities related to storage of spent fuel in accordance with written procedures. [10 CFR 72.212(b)(9)] 5. Vendors seeking approval of a cask design shall ensure that written procedures and appropriate tests are established before initial use of the casks. In addition, the vendor must provide a copy of these procedures and tests to each prospective cask user. [10 CFR 72.234(f)]	Written procedures for the system are provided in Chapter 8. These procedures are intended to provide general operational guidance for use of the system. These procedures will be used by an ISFSI operator to develop detailed, site specific procedures for use of the system.
Wet or Dry Loading and Unloading Facilities	6. The cask must be compatible with wet or dry spent fuel loading and unloading facilities. [10 CFR 72.236(h)]	The system design is compatible with both wet or dry loading and unloading facilities.
Decontamination Features	7. To the extent practicable, the design of the cask must facilitate decontamination. [10 CFR 72.236(i)]	The canister is designed to facilitate decontamination as described in Section 2.3.5.3. As described in Section 8.1.1, the annulus between the canister and transfer cask is filled with clean water prior to placement in the fuel pool to minimize the potential for contamination of the surface of the canister.
Ready Retrieval of Spent Fuel	8. The design of storage systems must allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)]	The procedure provided in Section 8.2 and 8.3 specify the steps necessary for retrieval of the spent fuel from the system for further processing or disposal.

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

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

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

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

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

Chapter 8 – Operating Procedures		
Area	**Acceptance Criteria**	**Description of Compliance**
Surveillance, Maintenance, and Contingency Plans	5. Operating procedure descriptions should include provisions for the following activities:	
	a. testing, surveillance, and monitoring of the stored material and casks during storage and loading and unloading operations	The testing and inspection requirements during loading and unloading operations are specified in Section 8.1 and 8.3. Section 9.2 specifies the inspection and maintenance activities required for the system during storage. The limits established in Chapter 12 are provided to ensure that the spent fuel is protected during loading and unloading operations.
	b. maintenance of casks and cask functions during storage c. contingency actions triggered by inspections, checks, observations, instrument readings, and so forth. (Some of these may involve off-normal conditions addressed in SAR Section 11.)	Normal operational maintenance and surveillance activities are specified in Section 9.2. These activities include contingency actions that may be required as a result of the inspection.
Cladding Protection	6. As required by 10 CFR 72.122(h)(1), the operating procedure descriptions should facilitate reducing the amount of water vapor and oxidizing material within the confinement cask to an acceptable level to protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures.	As specified in Section 8.4.2, the canister is vacuum dried to eliminate water, the cavity dryness is verified, and the cavity is then backfilled with inert helium gas during fuel loading operations to protect the fuel cladding against oxidation.

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

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Testing and Maintenance	a. The SAR must describe the applicant's program for preoperational testing and initial operations. [10 CFR 72.24(p)]	Section 9.1 presents the acceptance testing for the system.
	b. The cask design must permit maintenance as required. [10 CFR 72.236(g)]	Section 9.2 presents the maintenance activities for the system.
	c. Structures, systems, and components (SSCs) important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. [10 CFR 72.122(a), 10 CFR 72.122(f), 10 CFR 72.128(a)(1), and 10 CFR 72.24(c)]	The acceptance tests and maintenance activities presented in Sections 9.1 and 9.2 are performed to verify compliance with the design bases and criteria, and that the system continues to perform as designed.
	d. The applicant or licensee must establish a test program to ensure that all required testing is performed to meet applicable requirements and acceptance criteria. In addition, at least 30 days before the receipt of spent fuel, the licensee must submit to the NRC a report concerning the pre-operational test acceptance criteria and test results. [10 CFR 72.162 and 10 CFR 72.82(e)]	The testing and maintenance provided in Sections 9.1 and 9.2 are intended to be used by an ISFSI user in the development of site-specific programs.
	e. The applicant or licensee must evaluate the cask and its systems important to safety, using appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR 72.236(l)]	The acceptance tests presented in Section 9.1 demonstrate that the system will maintain confinement of the spent fuel under normal, off-normal, and accident conditions.
	f. The applicant or licensee must inspect the cask to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce confinement effectiveness. [10 CFR 72.236(j)]	As described in Section 9.1.1, the canister is visually and non-destructively examined prior to use.
	g. The applicant must perform, and make provisions that permit the Commission to perform, tests that the Commission deems necessary or appropriate. [10 CFR 72.232(b)]	As described in LCO 3.1.2, performance of the heat removal system is verified by temperature measurement, however, provisions shall be made, as necessary, to facilitate additional NRC imposed testing as required.
	h. The general licensee must accurately maintain the record provided by the cask supplier showing any maintenance performed on each cask. This record must include evidence that any maintenance and testing have been conducted under an NRC-approved quality assurance (QA) program. [10 CFR 72.212(b)(8)]	Records of maintenance activities would be maintained by the ISFSI user, and thus are not applicable.
	The applicant or licensee must assure that the casks are conspicuously and durably marked with a model number, unique identification number, and the empty weight. [10 CFR 72.236(k)]	As specified in Section 9.1.8, each system is to be marked with the model number, unique cask number, empty weight, and additional information.

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

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
2. Resolution of Issues Concerning Adequacy or Reliability	The SAR must identify all SSCs important to safety for which the applicant cannot demonstrate functional adequacy and reliability through previous acceptable evidence. For this purpose, acceptable evidence may be established in any of the following ways: • prior use for the intended purpose • reference to widely accepted engineering principles • reference to performance data in related applications In addition, the SAR should include a schedule showing how the applicant or licensee will resolve any associated safety questions before the initial receipt of spent fuel. [10 CFR 72.24(i)]	As described in Sections 3.1 and 3.3, the design of the system is based on industry standard codes and standards for materials and margins of safety. The acceptance tests specified in Section 9.1 are performed to demonstrate the adequacy of each fabricated system in accordance with applied Codes and Standards. The system does not rely on any materials or design standards that lack acceptable evidence of functional adequacy.
3. Cask Identification	The applicant or licensee must conspicuously and durably mark the cask with a model number, unique identification number, and empty weight. [10 CFR 72.236(k)]	As specified in Section 9.1.8, each system is to be marked with the model number, unique cask number, empty weight, and additional information.
Confinement System	American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel (B&PV) Code," Section III, Subsection NB or NC "American National Standard for Radioactive Materials— Leakage Tests on Packages for Shipment" (ANSI N14.5-1987)	As specified in Section 3.1.2, the canister is designed in accordance with the ASME Code, Section III, Subsection NB. Alternatives to the Code are provided in Table 4-1 of Chapter 12. The confinement system is leak tested in accordance with ANSI N14.5 following shield lid welding as specified in Sections 8.4.1.1 and 8.4.2.3.
Confinement Internals (e.g., basket)	ASME B&PV Code, Section III, Subsection NG	As specified in Section 3.1.2, the basket structure is designed in accordance with the ASME Code, Section III, Subsection NG.
Metal Cask Overpack	ASME B&PV Code, Section VIII	Not applicable.
Concrete Cask Overpack	American Concrete Institute (ACI) Standards 318 and 349, as appropriate	As stated in Section 3.1.2, the concrete cask is designed in accordance with ACI-349 and ANSI/ANS-57.9.
Other Metal Structures	ASME B&PV Code, Section III, Subsection NF American Institute of Steel Construction (AISC), "Manual of Steel Construction"	Not applicable.

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

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

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

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

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

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

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

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

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

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

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

Chapter 11 – Accident Analysis		
Area	**Regulatory Requirement**	**Description of Compliance**
7. Monitoring Systems	Instrumentation and control systems must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation. Those instruments and control systems that must remain operational under accident conditions must be identified in the Safety Analysis Report. [10 CFR 72.122(i)]	The system utilizes temperature monitoring instrumentation but utilizes routine inspection and surveillance to verify proper thermal operation of the system. The confinement system is fully welded and is leak tested to leaktight criteria as specified in Section 8.4.1.1. No seal monitoring is required.
8. Surveillance	Where instrumentation and control systems are not appropriate, storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. [72.122(h)(4)]	No active, continuous monitoring systems are required. Licensee radiological monitoring programs assure ISFSI operations meet 10 CFR 72.104 and 72.106 requirements.

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

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

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

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

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

Chapter 12 – Operating Controls and Limits	
Regulatory Requirement	**Description of Compliance**
1. General Requirement for Technical Specifications	
The applicant shall propose technical specifications (complete with acceptable bases and adequate justification). These specifications must include the following five areas [10 CFR 72.44(c), 10 CFR 72.24(g), and 10 CFR 72.26]:	
a. functional/operating limits, monitoring instruments, and limiting controls	Functional and operating limits are specified in Chapter 12.
b. limiting conditions	Limiting conditions for operation are also specified in Chapter 12.
c. surveillance requirements	Surveillance requirements are specified in LCO 3.1.2 and in Section 8.4.2.
d. design features	Design features are specified in Section 4.0 of Chapter 12.
e. administrative controls	Administrative controls are specified in Section 5.0 of Chapter 12.
Subpart E, "Siting Evaluation Factors," and Subpart F, "General Design Criteria," to 10 CFR Part 72, provide the bases for the cask system design and, hence, are applicable as bases for appropriate technical specifications.	
2. Specific Requirements for Technical Specifications — Storage Cask Approval	
As a condition of approval, the design, fabrication, testing, and maintenance of a spent fuel DCSS must comply with the requirements of 10 CFR 72.236. [10 CFR 72.234(a)] 10 CFR 72.236 Specific Requirements for Spent Fuel Storage Cask Approval	The operating controls, limits, and surveillance activities specified in Chapter 12 are intended to ensure that the system is maintained within its design basis through all normal, off-normal, and accident conditions.

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

Chapter 12 – Operating Controls and Limits	
Regulatory Requirement	**Description of Compliance**
The applicant must provide specifications for the spent fuel to be stored in the DCSS. At a minimum, these specifications should include, but not be limited to the following details [10 CFR 72.236(a)]: a. type of spent fuel (i.e., BWR, PWR, or both) b. maximum allowable enrichment of the fuel prior to any irradiation c. burn-up (i.e., megawatt-days/MTU) d. minimum acceptable cooling time of the spent fuel prior to storage in the DCSS (minimum 1 year) e. maximum heat that the DCSS system is designed to dissipate f. maximum spent fuel loading limit weights and dimensions h. condition of the spent fuel (i.e., intact assembly or consolidated fuel rods) i. inerting atmosphere requirements	Specifications for the spent fuel contents are provided in Section 2.2 as described in Section 2.0 of Chapter 12. As specified in Section 8.4.2.3, the canister is backfilled with helium gas to maintain an inert atmosphere for the spent fuel.
The applicant must provide design bases and design criteria for structures, systems, and components (SSCs) important to safety. [10 CFR 72.236(b)]	The design bases and criteria for the system are specified in Section 2.2.
The applicant must design and fabricate the DCSS so that the spent fuel will be maintained in a subcritical condition under credible conditions. [10 CFR 72.236(c)]	As shown in Section 6.4, the spent fuel is maintained in a subcritical configuration under all credible configurations.
The applicant must provide radiation shielding and confinement features that are sufficient to meet the requirements in 10 CFR 72.104 and 72.106 regarding radioactive material in effluents, direct radiation, and area control. [10 CFR 72.236(d) and 10 CFR Part 20] 10 CFR 72.104 Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS 10 CFR 72.106 Controlled Area of an ISFSI or MRS	The maximum external dose rates for the system are specified in Section 8.4.3.4. These limits are established to ensure that, for the minimum controlled area boundary distance presented in Section 10.4, the controlled area boundary annual dose will be maintained within allowable limits.

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

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

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

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

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

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

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

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

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

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

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

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

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