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7.0 CONFINEMENT

The Universal Storage System Transportable Storage Canister provides confinement for its radioactive contents in long-term storage. The confinement boundary is closed by welding, creating a solid barrier to the release of contents in all of the design basis normal, off-normal and accident conditions. The welds are visually inspected and nondestrucively examined to verify integrity. The containment boundary is leak tight as defined by ANSI N 14.5 [1].

The sealed canister contains an inert gas (helium). The confinement boundary retains the helium and also prevents the entry of outside air into the canister in long term storage. The exclusion of air precludes degradation of the fuel rod cladding, over time, due to cladding oxidation failures.

The Universal Storage System canister confinement system meets the requirements of 10 CFR 72.24 for protection of the public from release of radioactive material [2]. It also meets the requirements of 10 CFR 72.122 for protection of the spent fuel contents in long-term storage such that future handling of the contents would not pose an operational safety concern.

7.1 Confinement Boundary

The transportable storage canister provides confinement of the PWR or BWR contents in long-term storage. The welded canister forms the confinement vessel.

The primary confinement boundary of the canister consists of the canister shell, bottom plate, shield lid, the two port covers, and the welds that join these components. A secondary confinement boundary consists of the canister shell, the structural lid, and the welds that join the structural lid and canister shell. The confinement boundaries are shown in Figures 7.1-1 and 7.1-2. There are no bolted closures or mechanical seals in the primary or secondary confinement boundary. The confinement boundary welds are described in Table 7.1-1.

7.1.1 Confinement Vessel

The canister consists of three principal components: the canister shell, the shield lid, and the structural lid. The canister shell is a right circular cylinder constructed of 0.625-inch thick rolled Type 304L stainless steel plate. The edges of the rolled plate are joined using full penetration welds. It is closed at the bottom end by a 1.75-inch thick circular plate joined to the shell by a

full penetration weld. The inside and outside diameters of the canister are 65.81 inches and 67.06 inches, respectively. The canister has a length that is variable, depending on the class of fuel stored (See Figure 7.1-1).

The canister is fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, except for the field installed structural lid and shield lid closure welds [3]. These welds are not full penetration welds, but are inspected either ultrasonically or by using a progressive liquid penetrant examination.

After loading, the canister is closed at the top by a shield lid and a structural lid. The shield lid is a 7-inch-thick Type 304 stainless steel plate. It is joined to the canister shell using a field installed bevel weld. The shield lid contains the drain and vent penetrations and provides gamma radiation shielding for the operators during the welding, draining, drying and inerting operations. After the shield lid is welded in place, the canister is pressure tested and leak tested to ensure leak tightness. Following draining, drying and inerting operations, the vent and drain penetrations are closed with Type 304 stainless steel port covers that are welded in place with bevel welds. The operating procedures, describing the handling steps to close the canister, are presented in Section 8.1.1. The pressure and leak test procedures are described in Section 9.1.

A secondary, or redundant, confinement boundary is formed at the top of the canister by the structural lid, which is placed over the shield lid. The structural lid is a 3-inch thick Type 304L stainless steel plate. The structural lid provides the attachment points for lifting the loaded canister. The structural lid is welded to the shell using a field installed bevel weld.

The weld specifications and the weld examination and acceptance criteria for the shield lid and stuctural lid welds are presented in Sections 7.1.3.2 and 7.1.3.3, respectively.

The confinement boundaries are shown in Figures 7.1-1 and 7.1-2. As illustrated in Figure 7.1-2, the secondary confinement boundary includes the structural lid, the upper 3.2 inches of the canister shell and the joining weld. This boundary provides additional assurance of the leak tightness of the canister during its service life.

7.1.1.1 Design Documents, Codes and Standards

The canister is constructed in accordance with the license drawings presented in Section 1.6. The principal Codes and Standards that apply to the canister design, fabrication and assembly are described in Sections 7.1.1 and 7.1.3, and are shown on the licensing drawings.

7.1.1.2 Technical Requirements for the Canister

The canister confines up to 24 PWR, or 56 BWR, fuel assemblies. Over its 50-year design life, the canister precludes the release of radioactive contents and the entry of air that could potentially damage the cladding of the stored spent fuel. The design of the canister to the requirements of the ASME Code Section III, Subsection NB ensures that the canister maintains confinement in all of the evaluated normal, off-normal, and accident conditions.

The canister has no exposed penetrations, no mechanical closures, and does not employ seals to maintain confinement. There is no requirement for continuous monitoring of the welded closures. The design of the canister allows the recovery of stored spent fuel should it become necessary.

The minimum helium purity level of 99.9% specified in Section 8.1.1 of the Operating Procedures maintains the quantity of oxidizing contaminants to less than one mole per canister for all loading conditions. Based on the calculations presented in Section 4.4.5, the free gas volume of the empty canister is less than 300 moles. Conservatively assuming that all of the impurities in 99.9% pure helium are oxidents, a maximum of 0.3 moles of oxidants could exist in the largest NAC-UMS canister during storage. By limiting the amount of oxidants to less than one mole, the recommended limits for preventing cladding degradation found in the Pacific Northwest Laboratory, "Evaluation of Cover Gas Impuritities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365 [7] are satisfied.

The design criteria that apply to the canister, as an element of the NAC-UMS dry storage system, are presented in Table 1.2-1. The design basis parameters of the PWR and BWR spent fuel contents are presented in Section 1.3.

7.1.1.3 Release Rate

The primary confinement boundary is formed by joining the canister confinement boundary stainless steel components by welding. The canister shell longitudinal and girth welds are visually inspected, ultrasonically examined and pressure tested as described in Section 7.1.3.3 to confirm integrity. The shield lid welds are liquid penetrant examined following the root and the final weld passes. The shield lid to canister shell weld is pressure tested as described in Section 7.1.3.3. The structural lid to canister shell multi-pass weld is either: 1) progressively liquid penetrant examined; or 2) ultrasonically examined in conjunction with a liquid penetrant examination of the final weld surface.

To demonstrate leak tightness of the shield lid to canister shell weld, the leaktight criteria of 1×10^{-7} ref cm^3/sec, or 2×10^{-7} cm^3/sec (helium) at standard conditions, as defined in Section 2.1 of ANSI N14.5-1997, is applied. "Standard" conditions are defined as the leak rate at 298 K (25°C) with a one atmosphere pressure differential in the test condition. Since helium at approximately 25°C (77°F) is injected into the canister, at the point of the procedure (Section 8.1.1) that the leak test is performed, the actual temperature of the helium is always equal to, or higher than, 25°C due to the decay heat of the contents. This results in a pressure within the canister that is higher than the 0 psig (helium) that is initially established. To ensure that the leak test is conservatively performed, the ANSI N14.5 defined leak rate of 2×10^{-7} cm^3/sec is used. The higher temperature and higher pressure differential that actually exist in the canister, are conservatively ignored. The sensitivity of the leak test is 1×10^{-7} cm^3/sec (helium). Using this criterion, there is no maximum allowable leak rate specified for the canister, and calculation of the radionuclide inventory is not required. The leak test is described in Section 7.1.3.3 and in Section 8.1.1.

These steps provide reasonable assurance that the confinement boundary is leak tight and does not provide a path for the release of any of the content particulates, fission gases, volatiles, corrosion products or fill gases.

7.1.2 Confinement Penetrations

Two penetrations (with quick disconnect fittings) are provided in the canister shield lid for operator use. One penetration is used for draining residual water from the canister. It connects to a drain tube that extends to the bottom of the canister. The other penetration extends only to the underside of the shield lid. It is used to introduce air, or inert gas, into the top of the canister.

Once draining is completed, either penetration may be used for vacuum drying and backfilling with helium. After backfilling, both penetrations are closed with port covers that are welded to the shield lid. When the port covers are in place, the penetrations are not accessible. These port covers are enclosed and covered by the structural lid, which is also welded in place to form the secondary confinement boundary. The structural lid and the remainder of the canister have no penetrations.

7.1.3 Seals and Welds

This section describes the process used to properly assemble the confinement vessel (canister). Weld processes and inspection and acceptance criteria are described in Sections 7.1.3.2 and 7.1.3.3.

No elastomer or metallic seals are used in the confinement boundary of the canister.

7.1.3.1 Fabrication

All cutting, machining, welding, and forming are performed in accordance with Section III, Article NB-4000 of the ASME Code, unless otherwise specified in the approved fabrication drawings and specifications. License drawings are provided in Section 1.6. Code alternatives are listed in Table 4-1 of Chapter 12.

7.1.3.2 Welding Specifications

The canister body is assembled using longitudinal and circumferential welds in the shell and a circumferential weld at the bottom plate/ shell juncture.

Weld procedures and qualifications are in accordance with ASME Code Section IX. The welds joining the canister shell are radiographed in accordance with ASME Code Section V, Article 2. The weld joining the bottom plate to the canister shell is ultrasonically examined in accordance with ASME Code Section V, Article 5 [5]. The acceptance criteria for these welds is as specified in ASME Code Section III, NB-5320 (radiographic) and NB-5330 (ultrasonic). The finished surfaces of these welds are liquid penetrant examined in accordance with ASME Code, Section III, NB-5350.

After loading, the canister is closed by the shield lid and the structural lid using field installed groove welds.

After the shield lid is welded in place, the canister is pneumatically (air over water) pressure tested. Following draining, drying and inerting operations, the vent and drain ports are closed with port covers that are welded in place. The root and final surfaces of the shield lid to port cover welds are liquid penetrant examined in accordance with ASME Code Section V, Article 6. Acceptance is in accordance with ASME Code Section III, NB-5350. The shield lid to canister shell weld is liquid penetrant examined at the root and final surfaces in accordance with ASME Code Section V, Article 6, with acceptance in accordance with ASME Code Section III, NB-5350, and is pressure and leak tested to ensure leaktightness. The operating procedures, describing the handling steps to seal the canister are presented in Section 8.1.1. The pressure and leak test procedures are described in Sections 8.1.1 and 9.1.3.

A redundant confinement boundary is provided at the top of the canister by the structural lid, which is placed over the shield lid. The structural lid is welded to the canister shell using a field-installed groove weld. The structural lid to canister shell weld is either: 1) ultrasonically examined (UT) in accordance with ASME Code Section V, Article 5, with the final weld surface liquid penetrant (PT) examined in accordance with ASME Code Section V, Article 6; or, 2) progressive liquid penetrant examined in accordance with ASME Code Section V, Article 6. Acceptance criteria are specified in ASME Code Section III, NB-5330 (UT) and NB-5350 (PT).

All welding procedures are written and qualified in accordance with Section IX of the ASME Code. Each welder and welding operator must be qualified in accordance with Section IX of the ASME Code.

7.1.3.3 Testing, Inspection, and Examination

The following tests are performed to ensure satisfactory performance of the confinement vessel:

1. All components are visually examined for conformance with the fabrication drawings.

2. All welds that are directly visible are visually examined in accordance with the requirements of ASME Code Section V, Article 9.

3. The acceptance standards for visual examination of canister welded joints are as specified in ASME Code, Section III, Paragraphs NB-4424 and NB-4427. Unacceptable weld defects are repaired in accordance with ASME Code Section III, Subarticle NB-4450 and visually re-examined.

4. Canister welds designated to be examined by radiographic examination are examined in accordance with the requirements of Section V, Article 2 of the ASME Code. The minimum acceptance standards for radiographic examination are as specified in ASME Code Section III, NB-5320. Welds designated for ultrasonic examinations are examined in accordance with the requirements of Section V, Article 5 of the ASME Code. The minimum acceptance standards for ultrasonic examination are as specified in ASME Code Section III, NB-5330. Unacceptable defects in the welds are repaired in accordance with ASME Code Section III, NB-4450 and re-examined.

5. A written report of each weld examined is prepared. At a minimum, the written report will include: identification of part, material, name and level of examiner, NDE procedure used and the findings or dispositions, if any.

6. All personnel performing nondestructive examinations are qualified in accordance with American Society of Nondestructive Testing Recommended Practice No. SNT-TC-1A [6].

7. Field installed welds that are not ultrasonically inspected are root and final surface or progressive (i.e., at weld thickness intervals not exceeding 0.375 inch) liquid penetrant examined to ensure detection of critical weld flaws. As a minimum, liquid penetrant examination is applied to the root pass and final pass of the weld.

8. The results of the liquid penetrant examination, including all relevant indications, are recorded by video, photographic or other means to provide a retrievable record of weld integrity.

9. Individuals qualified for NDT Level I, NDT Level II, or NDT level III may perform nondestructive testing. Only Level II or Level III personnel may interpret the results of an examination or make a determination of the acceptability of examined parts.

10. The vendor completely assembles the canister prior to shipping. The purpose of assembling the canister is to ensure that all items specified have been supplied and to test the fit of the shield lid assembly including the shield lid, drain tube and the structural lid.

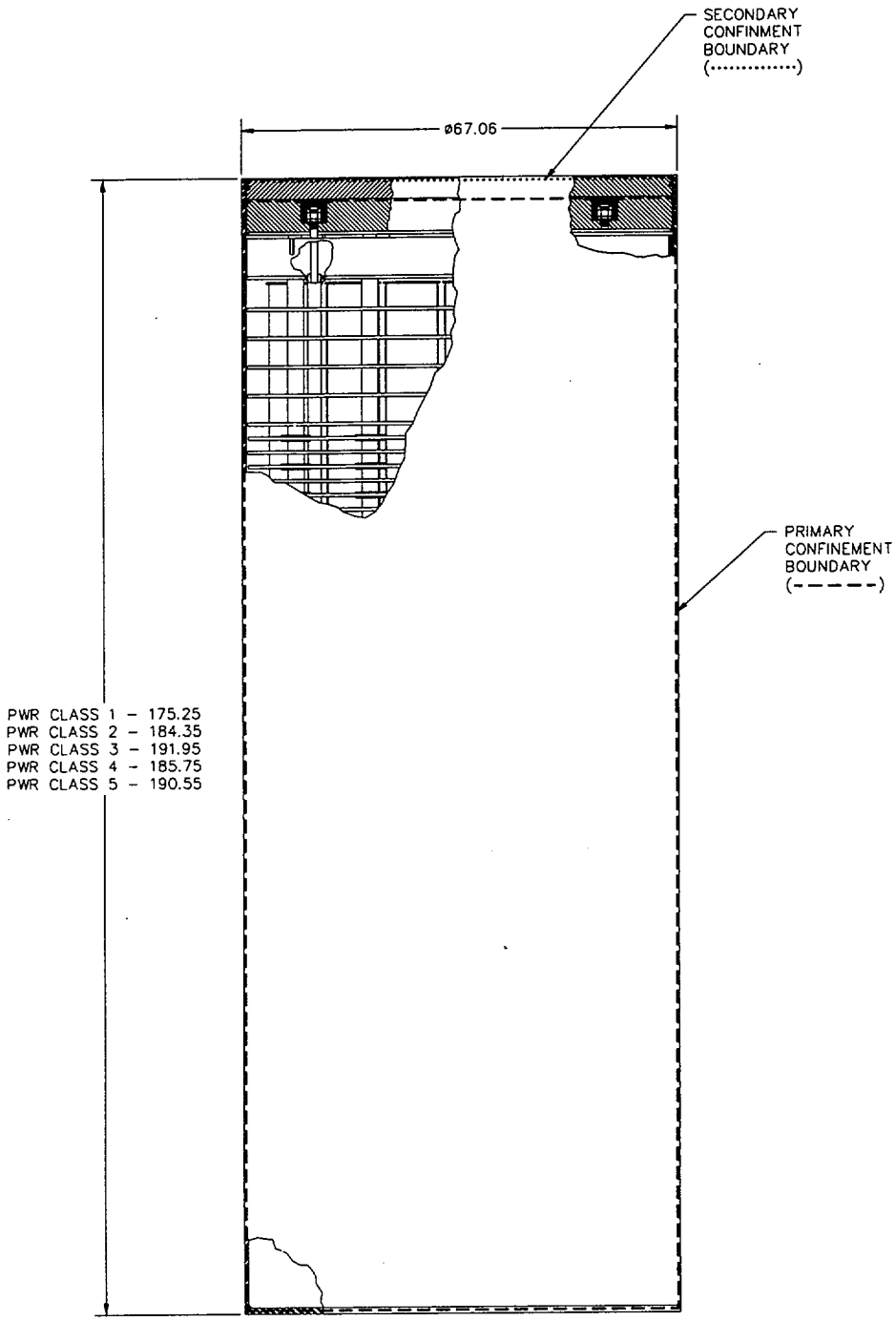
11. A pressure test to 35 psia is conducted after welding of the shield lid following loading of the fuel assemblies. The pressure test is performed in accordance with ASME Code, NB-6321.

12. A helium leak test is used to verify that the shield lid welds are leak tight. The canister is pressurized with helium to 0 psig when the canister is closed. A leak test fixture is used to create a volume above the shield lid, which is evacuated. This volume is then tested, using a mass spectrometer type helium leak detector, to verify that the shield lid welds meet the leak tight criteria to a leak test sensitivity of 1×10^{-7} cm^3/sec (helium). The leak test conforms to the evacuated envelope method of ANSI N14.5. As noted in the procedure presented in Section 8.1.1, a "sniffer detector" test method may be used as an optional informational leak test prior to the installation of the vent and drain port covers. This leak test is intended to ensure that there are no leaks in the shield lid welds at a leak rate of 1×10^{-5} cm^3/sec (helium) based on the detector leak rate sensitivity of 5×10^{-6} cm^3/sec (helium).

7.1.4 Closure

The primary closure of the transportable storage canister consists of the welded shield lid and the two welded port covers. There are no bolted closures or mechanical seals in the primary closure. A secondary closure is provided at the top end of the canister by the structural lid. The structural lid, when welded to the canister shell, fully encloses the shield lid and the port covers.

Figure 7.1-1 Transportable Storage Canister Primary and Secondary Confinement Boundaries



* Dimensions are in inches.

Figure 7.1-2 Confinement Boundary Detail at Shield Lid Penetration

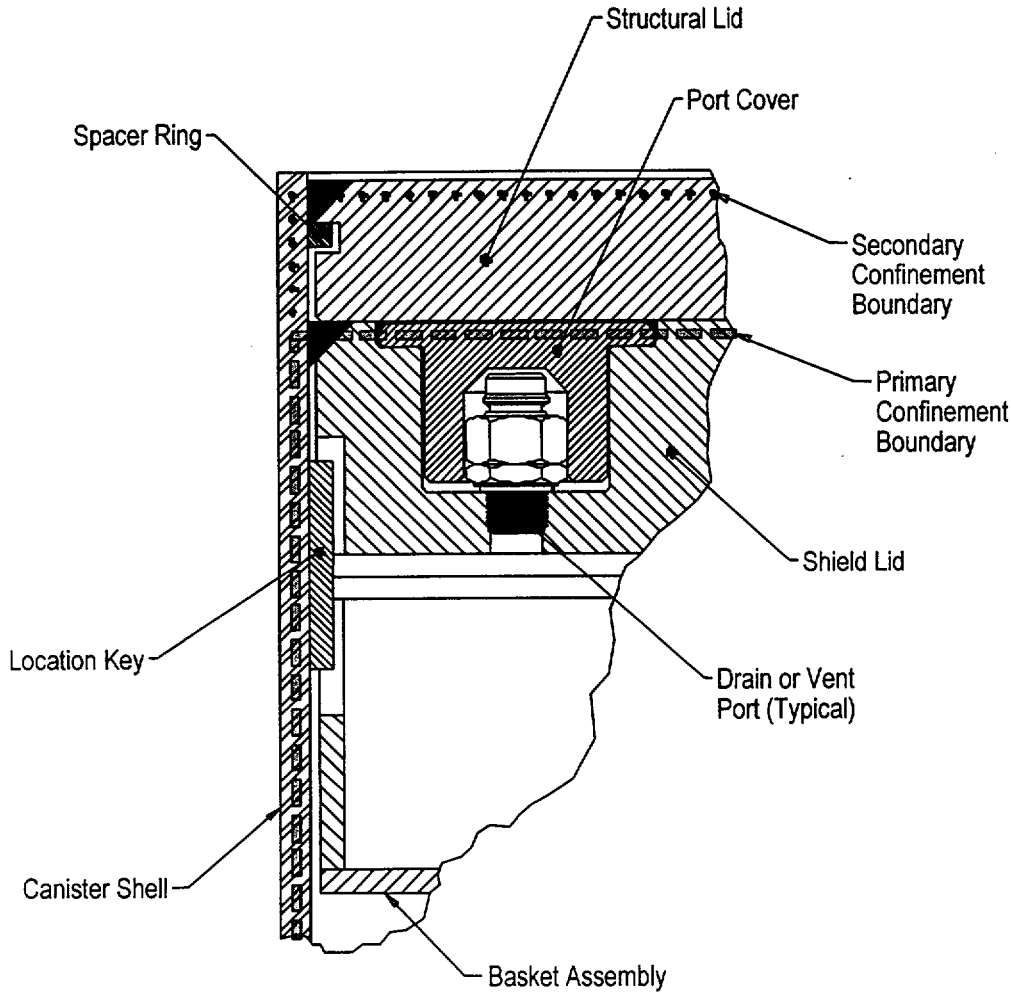


Table 7.1-1 Canister Confinement Boundary Welds

Confinement Boundary Welds		
Weld Location	**Weld Type**	**ASME Code Category (Section III, Subsection NB)**
Shell longitudinal	Full penetration groove (shop weld)	A
Shell circumferential (if used)	Full penetration groove (shop weld)	B
Bottom plate to shell	Full penetration groove (shop weld)	C
Shield lid to shell	Bevel (field weld)	C
Structural lid to shell	Bevel (field weld)	C
Vent and drain port covers to shield lid	Bevel (field weld)	C

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7.2 Requirements for Normal Conditions of Storage

The canister is transferred to a vertical concrete cask using a transfer cask. During this transfer, the canister is subject to handling loads. The evaluation of the canister for normal handling loads is provided in Section 3.4.4. The principal design criteria for the Universal Storage System are provided in Table 2-1.

Once the canister is placed inside of the vertical concrete storage cask, it is effectively protected from direct loading due to natural phenomena, such as wind, snow and ice loading. The principal direct loading for normal operating conditions arises from increased internal pressure caused by decay heat, solar insolation, and ambient temperature. The effect of the normal operating internal pressure is evaluated in Section 3.4.4.

7.2.1 Release of Radioactive Material

The structural analysis of the canister for normal conditions of storage presented in Section 3.4.4 shows that the canister is not breached in any of the normal operating events. Consequently, there is no release of radioactive material during normal conditions of storage.

7.2.2 Pressurization of the Confinement Vessel

The canister is vacuum dried and backfilled with helium at one atmosphere absolute prior to installing and welding the penetration port covers. In normal service, the internal pressure increases due to an increase in temperature of the helium and due to the postulated failure of fuel rod cladding of 3% of the fuel rods, which releases 30% of the available fission gases in those rods.

The canister, shield lid, fittings, and the canister basket are fabricated from materials that do not react with ordinary or borated spent fuel pool water to generate gases. The aluminum heat transfer disks are protected by an oxide film that forms shortly after fabrication. This oxide layer effectively precludes further oxidation of the aluminum components or other reaction with water in the canister at temperatures less than 200°F, which is higher than the typical spent fuel pool water temperature. The neutron absorber criticality control poison plates in the fuel baskets are

enclosed by a welded stainless steel cover. No steels requiring protective coatings or paints are used in the PWR configuration canister, shield lid, fittings, or basket, or in the BWR configuration canister, shield lid, or fittings. Carbon steel support disks are used in the BWR configuration basket. These disks are completely coated to protect the disks in immersion in the spent fuel pool, as defined on Drawing 790-573. The consequence of the use of a coating in BWR spent fuel pools is evaluated in Sections 3.4.1.2.3 and 3.4.1.2.4. That evaluation shows that no adverse interactions result from the use of the coating. The coating does not contain Zinc, and no gases are formed as a result of the exposure of this coating to the neutrally buffered water used in BWR spent fuel pools.

Since the canister is vacuum dried and backfilled with helium prior to sealing, no significant moisture or gases, such as air, remain in the canister. Consequently, there is no potential that radiolytic decomposition could cause an increase in canister internal pressure or result in a build up of explosive gases in the canister.

The calculation of the canister pressure increase based on these conditions is less than the pressure evaluated in Section 3.4.4 for the maximum normal operating pressure. As shown in Section 3.4.4, there are no adverse consequences due to the internal pressure resulting from normal storage conditions.

Since the containment boundary is closed by welding and contains no seals or O-rings, and since the boundary is not ruptured or otherwise compromised in normal handling events, no leakage of contents occurs in normal conditions.

7.3 Confinement Requirements for Hypothetical Accident Conditions

The evaluation of the canister for off-normal and accident condition loading is provided in Sections 11.1 and 11.2, respectively.

Once the canister is placed inside the vertical concrete cask, it is effectively protected from direct loading due to natural phenomena, such as seismic events, flooding and tornado (wind driven) missiles. Accident conditions assume the cladding failure of all the fuel rods stored in the canister. Consequently, there is an increase in canister internal pressure due to the release of a fraction of the fission product and charge gases. The accident conditions internal pressure for the PWR and BWR configurations is calculated in Section 11.2.1.

For evaluation purposes, a class of events identified as off-normal is also considered in Section 11.1. The off-normal class of events is not considered here, since off-normal conditions are bounded by the hypothetical accident conditions.

The structural analysis of the canister for off-normal and accident conditions of storage, presented in Chapter 11, show that the canister is not breached in any of the evaluated events. Consequently, based on a leaktight configuration, there is no release of radioactive material during off-normal or accident conditions of storage.

The resulting site boundary dose due to a hypothetical accident is, therefore, less than the 5 rem whole body or organ (including skin) dose at 100 meter minimum boundary required by 10 CFR 72.106 (b) for accident exposures.

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7.4 Confinement Evaluation for Site Specific Spent Fuel

This section presents the confinement evaluation for fuel assembly types or configurations, which are unique to specific reactor sites. Site specific spent fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel rod or assemblies that are classified as damaged.

The design of the Transportable Storage Canister incorporates a leak tight configuration as described in Section 7.1 and as defined by ANSI N 14.5. Consequently, site specific fuel configurations need be evaluated only if the configuration results in a modification of the confinement boundary of the canister that is intended for use or when the configuration could result in a higher internal pressure or temperature than is used in the design basis analysis.

7.4.1 Confinement Evaluation for Maine Yankee Site Specific Spent Fuel

Maine Yankee site specific spent fuel is to be stored in either the Class 1 or Class 2 Transportable Storage Canister, depending on the overall length of the fuel assembly, including inserted non fuel-bearing components. These canisters are closed by welding and are inspected and tested to confirm the leak tight condition.

Site specific fuel includes fuel having variable enrichment radial zoning patterns and annular axial fuel blankets, removed fuel rods or empty rod positions, fuel rods placed in guide tubes, consolidated fuel, damaged fuel, and high burnup fuel (fuel with a burnup between 45,000 MWD/MTU and 50,000 MWD/MTU). These configurations are not included in the standard fuel analysis, but are present in the site fuel inventory that must be stored. As discussed in Section 4.5.1, the site specific fuel configurations do not result in a canister pressure or temperature that exceeds the canister design basis. Since the canisters are leak tight, there is no release from a canister containing Maine Yankee high burnup fuel rods site specific spent fuel.

Intact site specific fuel is loaded directly into the fuel tubes in the PWR basket. Damaged fuel is inserted into a fuel can, shown in Drawings 412-501 and 412-502, which precludes the release of gross particulate material from the fuel can. The fuel can is sized to allow its insertion into a fuel position in the PWR basket.

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7.5 References

1. ANSI N14.5-1997, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," American National Standards Institute, 1997.

2. Title 10 of the Code of Federal Regulations, Part 72 (10 CFR 72), "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation," April 1996 Edition.

3. ASME Boiler and Pressure Vessel Code, Section III, Division I, "Rules for Construction of Nuclear Power Plant Components," 1995 Edition with 1997 Addenda.

4. ASME Boiler and Pressure Vessel Code, Section IX, "Qualification Standard for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operators," 1995 Edition with 1997 Addenda.

5. ASME Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination," 1995 Edition with 1997 Addenda.

6. SNT-TC-1A, Recommended Standard Practice, " Personnel Qualification and Certification in Nondestructive Testing," American Society for Nondestructive Testing, August 1984.

7. PNL-6365, "Evaluation of Cover Gas Impuritities and Their Effects on the Dry Storage of LWR Spent Fuel," Pacific Northwest Laboratory, Richland, Washington, November, 1987.

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8.0 OPERATING PROCEDURES

This chapter provides general guidance for operating the Universal Storage System. Three operating conditions are addressed. The first is loading the transportable storage canister, installing it in the vertical concrete cask, and transferring it to the storage (Independent Spent Fuel Storage Installation (ISFSI)) pad. The second is the removal of the loaded canister from the concrete storage cask. The third is opening the canister to remove spent fuel in the unlikely event that this should be necessary.

The operating procedure for transferring a loaded canister from a storage cask to the Universal Transport Cask, is described in Section 7.2.2 of the UMS® Universal Transport Cask Safety Analysis Report. [1]

Users shall develop site-specific written and approved procedures that incorporate the requirements presented here, consistent with the Technical Specifications presented in Section 3.0 of Chapter 12 and the Approved Contents and Design Features presented Sections 2.0 and 4.0 of Chapter 12, respectively. Site-specific procedures shall also incorporate site-specific Technical Specifications, surveillance requirements, administrative controls, and other limits appropriate to the use of the NAC UMS® Storage System at the ISFSI. The procedures shall incorporate spent fuel assembly selection and verification requirements to ensure that the spent fuel assemblies loaded into the Transportable Storage Canister are as authorized by the Certificate of Compliance.

Operation of the Universal Storage System requires the use of ancillary equipment items. The ancillary equipment supplied with the system is shown in Table 8.1.1-1. The system does not rely on the use of bolted closures, but bolts are used to secure retaining rings and lids. The hoist rings used for lifting the shield lid and canister have threaded fittings. Table 8.1.1-2 provides the torque values for installed bolts and hoist rings. Supplemental shielding may be employed to reduce radiation exposure for certain of the tasks specified by these procedures. Use of supplemental shielding is at the discretion of the User.

The design of the Universal Storage System is such that the potential for spread of contamination during handling and future transport of the canister is minimized. The transportable storage canister is loaded in the spent fuel pool but is protected from gross contact with pool water by a jacket of clean or filtered pool water while it is in the transfer cask. The top of the canister is

closed by the structural lid, which is not contaminated when it is installed. Consequently, the canister external surface is expected to be essentially free of contamination. There are no radioactive effluents from the canister or the concrete cask in routine operations or in the design basis accident events.

When used in accordance with these procedures, the user dose is As Low As Reasonably Achievable (ALARA).

The Administrative Programs and Controls for the storage system are described in Section 8.4. The Administrative Programs and Controls are intended to assist the User in complying with the training and dry run requirements of 10 CFR 72, present the applicable operational and fuel loading controls and limits, and address the system operational features and requirements.

8.1 Procedures For Loading the Universal Storage System

The Universal Storage System consists of three principal components: the transportable storage canister (canister), the transfer cask, and the vertical concrete cask. The transfer cask is used to hold the canister during loading and while the canister is being closed and sealed. The transfer cask is also used to transfer the canister to the concrete cask and to load the canister into the transport cask. The principal handling operations involve closing and sealing the canister by welding, and placing the loaded canister in the vertical concrete cask. The vent and drain port locations are shown in Figure 8.1.1-1.

The transfer cask is provided in two configurations: A standard transfer cask, weighing approximately 121,500 lbs, and a 100-ton transfer cask weighing approximately 100,000 lbs. The 100-ton transfer cask is designed to accommodate sites having a 100-ton cask handling crane weight limit. It 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 without fuel, and when it holds a canister that is loaded and closed with its structural lid. Canister handling, fuel loading and canister closing are operationally identical for either transfer cask configuration, except that the standard transfer cask can accommodate an extension fixture to allow the use of the next longer length canister.

This procedure assumes that the canister with an empty basket is installed in the transfer cask, that the transfer cask is positioned in the decontamination area or other suitable work station, and that the vertical concrete cask is positioned in the plant cask receiving area or other suitable staging area. The transfer cask extension must be installed on the standard transfer cask if its use is required. To facilitate movement of the transfer cask to the concrete cask, the staging area should be within the operational "footprint" of the cask handling crane. The concrete cask may be positioned on a heavy-haul transporter, or on the floor of the work area.

The User must ensure that the fuel assemblies selected for loading conform to the Approved Contents provisions of Section 2.0 of Chapter 12 and to the Certificate of Compliance. Fuel assembly loading may also be administratively controlled to ensure that fuel assemblies with specific characteristics are preferentially loaded in specified positions in the canister. Preferential loading requirements are described in Sections 2.1.2 and 2.1.3 of Chapter 2.

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8.1.1 Loading and Closing the Transportable Storage Canister

1. Visually inspect the basket fuel tubes to ensure that they are unobstructed and free of debris. Ensure that the welding zones on the canister, shield, and structural lids, and the port covers are prepared for welding. Ensure transfer cask door lock bolts are installed and secure.
2. Fill the canister with clean or filtered pool water until the water is about 4 inches from the top of the canister.

   Note: Do not fill the canister completely in order to avoid spilling water during the transfer to the spent fuel pool.

   Note: If fuel loading requires boron credit, the minimum boron concentration of the water in the canister must be at least 1,000 ppm (boron) in accordance with Section 8.4.4.1.
3. Attach clean or filtered pool water lines to the transfer cask.
4. If it is not already attached, attach the transfer cask lifting yoke to the cask handling crane, and engage the transfer cask lifting trunnions.

   Note: The minimum temperature of the transfer cask (i.e., surrounding air temperature) must be verified to be higher than 0°F prior to lifting.
5. Raise the transfer cask and move it over the pool, following the prescribed travel path.
6. Lower the transfer cask to the pool surface and turn on the clean or filtered pool water line to fill the canister and the annulus between the transfer cask and canister.
7. Lower the transfer cask as the annulus fills with clean or filtered pool water until the trunnions are at the surface, and hold that position until the clean or filtered pool water overflows through the upper fill lines of the transfer cask. Then lower the transfer cask to the bottom of the pool cask loading area.

   Note: If an intermediate shelf is used to avoid wetting the cask handling crane hook, follow the plant procedure for use of the crane lift extension piece.
8. Disengage the transfer cask lifting yoke to provide clear access to the canister.
9. Load the previously designated fuel assemblies into the canister.

   Note: Contents must be in accordance with the Approved Contents provisions of Section 2.0 of Chapter 12 and in Chapter 2, Sections 2.1.1 for PWR fuel and 2.1.2 for BWR fuel.

   Note: Contents may be administratively controlled to ensure that fuel assemblies with certain characteristics are preferentially loaded in specified positions in the basket.
10. Attach a three-legged sling to the shield lid using the swivel hoist rings. Torque hoist rings in accordance with Table 8.1-2. Attach the suction pump fitting to the vent port.

    Caution: Verify that the hoist rings are fully seated against the shield lid.

    Note: Ensure that the shield lid key slot aligns with the key welded to the canister shell.

11. Using the cask handling crane, or auxiliary hook, lower the shield lid until it rests in the top of the canister.
12. Raise the transfer cask until its top just clears the pool surface. Hold at that position, and using a suction pump, drain the pool water from above the shield lid. After the water is removed, continue to raise the cask. Note the time that the transfer cask is removed from the pool. Operations through Step 27 must be completed in accordance with the time limits presented in Section 8.4.2.1.

    Note: Alternately, the temperature of the water in the canister may be used to establish the time for completion through Step 27. Those operations must be completed within 2 hours of the time that the canister water temperature is 200°F. For this alternative, the water temperature must be determined every 2 hours beginning 17 hours after the time the transfer cask is removed from the pool.
13. As the cask is raised, spray the transfer cask outer surface with clean or filtered pool water to wash off any gross contamination.
14. When the transfer cask is clear of the pool surface, but still over the pool, turn off the clean or filtered pool water flow to the annulus, remove hoses and allow the annulus water to drain to the pool. Move the transfer cask to the decontamination area or other suitable work station.

    Note: Access to the top of the transfer cask is required. A suitable work platform may need to be erected.

    Caution: If the 100-ton transfer cask is used, the neutron shield tank must be filled with water to provide neutron shielding for the remaining operations.
15. Verify that the shield lid is level and centered.
16. Attach the suction pump to the suction pump fitting on the vent port. Operate the suction pump to remove free water from the shield lid surface. Disconnect the suction pump and suction pump fitting. Remove any free standing water from the shield lid surface and from the vent and drain ports.
17. Decontaminate the top of the transfer cask and shield lid as required to allow welding and inspection activities.

    Note: Supplemental shielding may be used for activities around the shield lid.
18. Insert the drain tube assembly through the drain port of the shield lid into the basket drain tube sleeve. Torque the drain tube assembly to 125 ± 5 ft-lbs. Install a mating quick-disconnect fitting in the vent line to open the vent.
19. Connect the suction pump to the drain port. Verify that the vent port is open. Remove approximately 50 gallons of water from the canister. Disconnect and remove the pump.

    Caution: Radiation level may increase as water is removed from the canister.

20. Install the automatic welding equipment.
21. Attach the hydrogen gas detector to the vent port. Verify that the concentration of any detectable hydrogen gas is below 2.4%.

    Note: If the concentration exceeds 2.4%, operate the vacuum system to remove gases from the underside of the shield lid and re-verify the hydrogen gas concentration. Disconnect and remove vacuum system.
22. Operate the welding equipment to complete the root weld joining the shield lid to the canister shell following approved procedures to minimize canister shell and weld stress.
23. Examine the root weld using liquid penetrant and record the results.
24. Complete welding of the shield lid to the canister shell. Remove the weld equipment and the hydrogen gas detector.
25. Liquid penetrant examine the final weld surface and record the results.
26. If a pressure test of the canister is not required, proceed to Step 27. If a pressure test is required, perform the following steps:
    - 26a. Attach a regulated air supply line to the vent port. Install a valved fitting on the drain port and ensure the valve is closed.
    - 26b. Pressurize the canister to 35 psia (approximately 20.5 psig) and hold the pressure. There must be no loss of pressure for 10 minutes.
    - 26c. Release the pressure. Visually examine the shield lid to canister shell weld for indications of defects.
27. Drain the canister using either Steps 27a through 27e, or Steps 27c through 27e.
    - 27a. Attach the suction pump to the drain line. Ensure that the vent line is open. Using the pump, remove the remaining free water from the canister cavity.
    - 27b. Remove the suction pump from the drain line and close the drain line.
    - 27c. Using the vent port, pressurize the canister with Nitrogen to 30 (+3, -0) psig.
    - 27d. Open the drain line to blow any remaining free water from the canister.
    - 27e. When free water is no longer present at the drain line, stop the flow of Nitrogen, vent the remaining pressure and remove the Nitrogen supply line.

    Note the time that the last free water is removed from the canister cavity.

    Caution: Radiation levels at the top and sides of the transfer cask may rise as water is removed.

    Note: The time duration from completion of draining through the completion of helium backfill (Step 34) shall be monitored in accordance with LCO 3.1.1 and Section 8.4.2.1.
28. Attach the vacuum equipment to the vent and drain ports. Dry any free standing water in the vent and drain port recesses.

29. Operate the vacuum equipment until a vacuum of 10 mm of mercury exists in the canister in accordance with the requirements of Section 8.4.2.2.
30. Verify that no water remains in the canister by holding the vacuum for 10 minutes. If water is present in the cavity, the pressure will rise as the water vaporizes. Continue the vacuum/hold cycle until the vacuum condition of Step 29 is met.
31. Operate the vacuum system until a vacuum of 3 mm of mercury exists in the canister in accordance with Section 8.4.2.2.
32. Backfill the canister cavity with helium having a minimum purity of 99.9% to a pressure of one atmosphere (0 psig).

    Note: As an option, an informational helium leak test may be conducted at this point of the procedure using the following steps (the record leak test is performed at Step 49):

    32a. Backfill the canister cavity with helium having a minimum purity of 99.9% to a pressure of 15 psig.

    32b. Using a helium leak detector ("sniffer" detector) with a test sensitivity of 5×10^{-6} cm^3/sec (helium), survey the weld joining the shield lid and canister shell.

    32c. At the completion of the survey, vent the canister helium pressure to one atmosphere (0 psig).
33. Restart the vacuum equipment and operate until a vacuum of 3 mm of mercury exists in the canister.
34. Backfill the canister with helium having a minimum purity of 99.9% to a pressure of one atmosphere (0 psig) in accordance with the requirements of Section 8.4.2.3.
35. Disconnect the vacuum and helium supply lines from the vent and drain ports. Dry any residual water that may be present in the vent and drain port cavities.
36. Install the vent and drain port covers.
37. Complete the root pass weld of the drain port cover to the shield lid.
38. Prepare the weld and perform a liquid penetrant examination of the root pass. Record the results.
39. Complete welding of the drain port cover to the shield lid.
40. Prepare the weld and perform a liquid penetrant examination of the drain port cover weld final pass. Record the results.
41. Complete the root pass weld of the vent port cover to the shield lid.
42. Prepare the weld and perform a liquid penetrant examination of the root pass. Record the results.
43. Complete welding of the vent port cover to the shield lid.
44. Prepare the weld and perform a liquid penetrant examination of the weld final surface. Record the results.

45. Remove any supplemental shielding used during shield lid closure activities.
46. Install the helium leak test fixture.
47. Attach the vacuum line and leak detector to the leak test fixture fitting.
48. Operate the vacuum system to establish a vacuum in the leak test fixture.
49. Operate the helium leak detector for 15 minutes to verify that there is no indication of a helium leak exceeding 2×10^{-7} cm^3/second in accordance with the requirements of Section 8.4.2.3.
50. Release the vacuum and disconnect the vacuum and leak detector line from the fixture.
51. Remove the leak test fixture.
52. Attach a three-legged sling to the structural lid using the swivel hoist rings.

    Caution: Ensure that the hoist rings are fully seated against the structural lid. Torque the hoist rings in accordance with Table 8.1.1-2. Verify that the spacer ring is in place on the structural lid.

    Note: Verify that the structural lid is stamped or otherwise marked to provide traceability of the canister contents.
53. Using the cask handling crane or the auxiliary hook, install the structural lid in the top of the canister. Verify that the structural lid does not protrude above the canister shell. If so, remove the lid and inspect the surface of the shield lid for the cause of the interference. Verify that the gap in the spacer ring is not aligned with the shield lid alignment key. Remove the hoist rings.
54. Install the automatic welding equipment on the structural lid.
55. Operate the welding equipment to complete the root weld joining the structural lid to the canister shell, following approved procedures to minimize canister shell and weld stress.
56. Prepare the weld and perform a liquid penetrant examination of the weld root pass. Record the results.
57. Continue with the welding procedure, examining the weld at 3/8-inch intervals using liquid penetrant. Record the results of each intermediate examination.

    Note: If ultrasonic testing of the weld is used, testing is performed after the weld is completed.
58. Remove the weld equipment.
59. Perform a smear survey of the accessible area at the top of the canister to ensure that the surface contamination is less than the limits established for the site. Smear survey results shall meet the requirements of Section 8.4.1.2.
60. Install the transfer cask retaining ring. Torque bolts to 155 ± 10 ft-lbs. (Table 8.1.1-2).
61. Decontaminate the external surface of the transfer cask to the limits established for the site.

Figure 8.1.1-1 Vent and Drain Port Locations

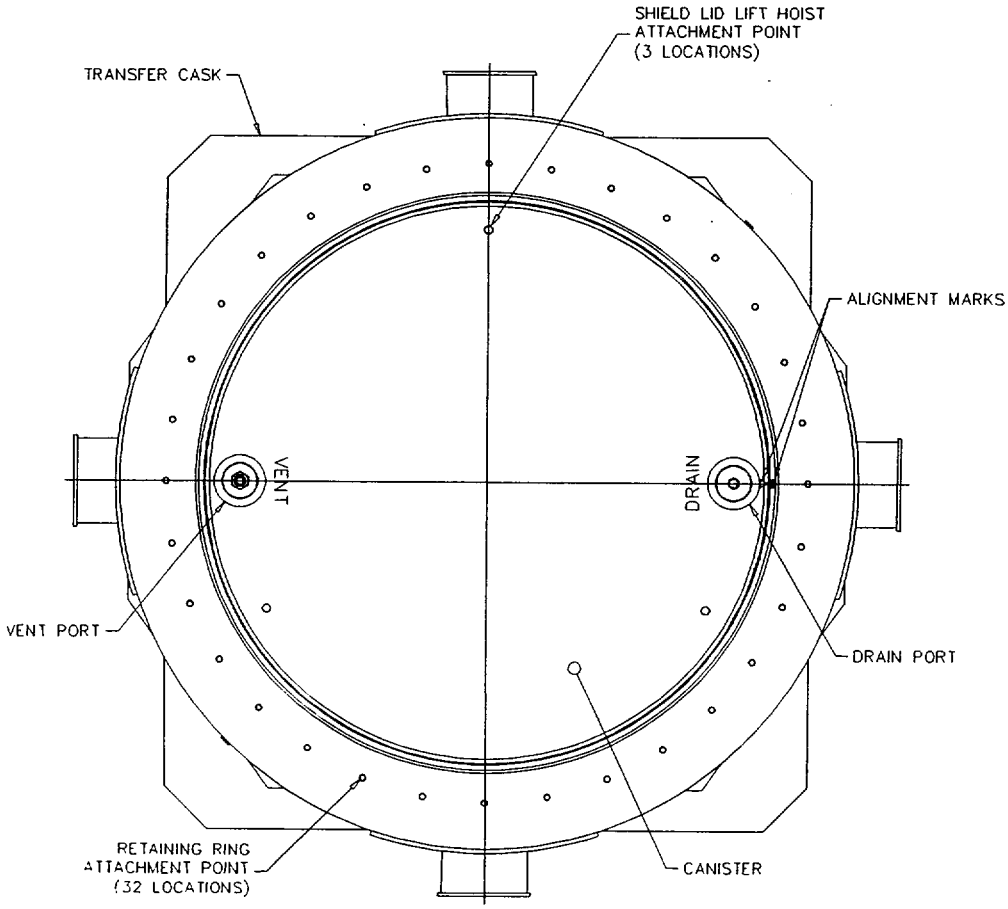


Table 8.1.1-1 List of Ancillary Equipment

Item	Description
Transfer Cask Lifting Yoke	Required for lifting and moving the transfer cask.
Heavy-Haul Transporter (Optional)	Heavy-haul (double drop frame) trailer required for moving the loaded and empty vertical concrete cask to and from the ISFSI pad.
Mobile Lifting Frame (Optional)	A self-propelled or towed A-frame lifting device for the concrete cask. Mobile Lifting Frame is used to lift the cask and move it using two lifting lugs in the top of the concrete cask.
Horizontal Transfer Cradle	A trailer-mounted cradle assembly for the horizontal transfer of the 100-ton transfer cask.
Helium Supply System	Supplies helium to the canister for helium backfill and purging operations.
Vacuum Drying System	Used for evacuating the canister. Used to remove residual water, air and initial helium backfill.
Automated Welding System	Used for welding the shield lid and structural lid to the canister shell.
Self-Priming Pump	Used to remove water from the canister.
Shield Lid Sling	A three-legged sling used for lifting the shield lid. It is also used to lift the concrete cask shield plug and lid.
Canister Sling	A set of 2 three-legged slings used for lifting the structural lid by itself, or for lifting the canister when the structural lid is welded to it. The slings are configured to provide for simultaneously loading during the canister lift.
Transfer Adapter	Used to align the transfer cask to the vertical concrete cask or the Universal Transport Cask. Provides the platform for the operation of the transfer cask shield doors.
Transfer Cask Extension	A carbon steel ring used to extend the height of the transfer cask when using the next larger size canister.
Hydraulic Unit	Operates the shield doors of the transfer cask.
Lift Pump Unit	Jacking system for raising and lowering the concrete cask.
Air Pad Rig Set	Air cushion system used for moving the concrete cask.
Supplemental Shielding Fixture	An optional carbon steel fixture inserted in the Vertical Concrete Cask air inlets to reduce radiation dose rate at the inlets.

Table 8.1.1-2 Torque Values

Fastener	Torque Value (ft-lbs)	Torque Pattern
Transfer Adapter Bolts	40 ± 5	None
Transfer Cask Retaining Ring	155 ±10	None
Transfer Cask Extension	155 ±10	None
Vertical Concrete Cask Lid	40 ± 5	None
Lifting Hoist Rings		
Shield Lid	230 (+30, -0)	None
Structural Lid	800 (+80, -0)	None
Concrete Cask Lid	60 (+10, -0)	None
Shield Plug	230 (+30, -0)	None
Canister Lid Plug Bolts	Hand Tight	None
Shield Lid Plug Bolts	Hand Tight	None
Transfer Cask Door Lock Bolts	Hand Tight	None
Canister Drain Tube	125 ± 5	None

8.1.2 Loading the Vertical Concrete Cask

This section of the loading procedure assumes that the vertical concrete cask is located on the bed of a heavy-haul transporter, or on the floor of the work area, under a crane suitable for lifting the loaded transfer cask. The vertical concrete cask shield plug and lid are not in place, and the bottom pedestal plate cover is installed.

1. Using a suitable crane, place the transfer adapter on the top of the concrete cask.
2. Using the transfer adapter bolt hole pattern, align the adapter to the concrete cask. Bolt the adapter to the cask using four (4) socket head cap screws.
3. Verify that the shield door connectors on the adapter plate are in the fully extended position.
4. If not already done, attach the transfer cask lifting yoke to the cask handling crane. Verify that the transfer cask retaining ring is installed.
5. Install six (6) swivel hoist rings in the structural lid of the canister and torque to the value specified in Table 8.1.1-2. Attach two (2) three-legged slings to the hoist rings.
   Caution: Ensure that the hoist rings are fully seated against the structural lid.
6. Stack the slings on the top of the canister so they are available for use in lowering the canister into the storage cask.
7. Engage the transfer cask trunnions with the transfer cask lifting yoke. Ensure that all lines are disconnected from the transfer cask.
   Note: The minimum temperature of the transfer cask (i.e., temperature of the surrounding air) must be verified to be higher than 0° F prior to lifting.
   Note: Verify that the transfer cask extension is installed if required.
8. Raise the transfer cask and move it over the concrete cask. Lower the transfer cask, ensuring that the transfer cask shield door rails and connector tees align with the adapter plate rails and door connectors. Prior to final set down, remove transfer cask shield door lock bolts.
9. Ensure that the shield door connector tees are engaged with the adapter plate door connectors.
10. Disengage the transfer cask yoke from the transfer cask and from the cask handling crane hook.
11. Return the cask handling crane hook to the top of the transfer cask and engage the two (2) three-legged canister slings attached to the canister to the crane hook.
    Caution: The top attachment point of the two three-legged slings must be at least 75 inches or equivalent above the top of the canister, arranged so the slings load evenly.

12. Lift the canister slightly (about ½ inch) to take the canister weight off of the transfer cask shield doors.

    Note: A load cell may be used to determine when the canister is supported by the crane.

    Caution: Avoid raising the canister to the point that the canister top engages the transfer cask retaining ring, as this could result in lifting the transfer cask.
13. Using the hydraulic system, open the shield doors to access the concrete cask cavity.
14. Lower the canister into the concrete cask, using a slow crane speed as the canister nears the pedestal at the base of the concrete cask.
15. When the canister is properly seated, disconnect the slings from the canister at the crane hook, and close the transfer cask shield doors.
16. Retrieve the transfer cask lifting yoke and attach the yoke to the transfer cask.
17. Lift the transfer cask off of the vertical concrete cask and return it to the decontamination area or designated work station.
18. Using the auxiliary crane, remove the adapter plate from the top of the concrete cask.
19. Remove the swivel hoist rings from the structural lid and replace them with threaded plugs.
20. Install three swivel hoist rings in the shield plug and torque in accordance with Table 8.1.1-2.
21. Using the auxiliary crane, retrieve the shield plug and install the shield plug in the top of the concrete cask. Remove swivel hoist rings and insert threaded plugs.
22. Install seal tape around the diameter of the lid bolting pattern on the concrete cask flange.
23. Using the auxiliary crane, retrieve the concrete cask lid and install the lid in the top of the concrete cask. Secure the lid using six stainless steel bolts. Torque bolts in accordance with Table 8.1.1-2.
24. Ensure that there is no foreign material left at the top of the concrete cask. Install the tamper-indicating seal.
25. If used, install a supplemental shielding fixture in each of the four air inlets.

    Note: The supplemental shielding fixtures may also be shop installed.

8.1.3 Transport and Placement of the Vertical Concrete Cask

This procedure assumes that the loaded vertical concrete cask is positioned on a heavy-haul transporter and is to be positioned on the ISFSI pad using the air pad set. Alternately, the concrete cask may be lifted and moved using a mobile lifting frame. The mobile lifting frame lifts the cask using the two sets of lifting lugs at the top of the concrete cask. The lifting frame may be self-propelled or towed, and does not use the air pad set.

Transport and placement of the vertical concrete cask must be performed in accordance with the requirements of the administrative controls and program plan for ISFSI operations presented in Section 8.4.3. In accordance with Section 8.4.3, the vertical concrete cask lift height limit is 24 inches when the cask is moved using the air pad set or the mobile lifting frame. Because of lift fixture configuration, the maximum lift height of the concrete cask using the jacking arrangement is approximately 6 inches.

The concrete cask surface dose rates must be verified in accordance with the requirements of Section 8.4.3.4. These measurements may be made prior to movement of the cask, at a location along the transport path, or at the ISFSI. An optional supplemental shielding fixture, shown in Drawing 790-613, may be installed in the concrete cask air inlets to reduce the radiation dose rate at the inlets.

1. Using a suitable towing vehicle, tow the heavy-haul transporter to the dry storage pad (ISFSI). Verify that the bed of the transporter is approximately at the same height as the pad surface. Install four (4) hydraulic jacks at the four (4) designated jacking points at the air inlets in the bottom of the vertical concrete cask.
2. Raise the concrete cask approximately 3-inches.
   Caution: Do not exceed a maximum lift height of 24 inches.
3. Move the air-bearing rig set under the cask, if not already in place.
   Note: A hydraulic skid may also be used to move the concrete cask. The height the concrete cask is raised depends upon the height of the skid or air pad set used, but may not exceed 24 inches.
4. Inflate the air-bearing rig set. Remove the four (4) hydraulic jacks.
5. Using a suitable towing vehicle, move the concrete cask from the bed of the transporter to the designated location on the storage pad.
   Note: Spacing between concrete casks must not be less than 15 feet (center-to-center).
6. Turn off the air-bearing rig set, allowing it to deflate.

7. Reinstall the four (4) hydraulic jacks and raise the concrete cask approximately 3 inches.
   Caution: Do not exceed a maximum lift height of 24 inches.
8. Remove the air-bearing rig set pads. Ensure that the surface of the dry storage pad under the concrete cask is free of foreign objects.
9. Lower the concrete cask to the surface and remove the four (4) hydraulic jacks.
10. Install the screens in the inlets.
11. Install/connect temperature monitoring equipment and verify operation in accordance with LCO 3.1.2 and Section 8.4.3.5.
12. Scribe/stamp concrete cask name plate to indicate loading information.

8.2 Removal of the Loaded Transportable Storage Canister from the Vertical Concrete Cask

Removal of the loaded canister from the vertical concrete cask is expected to occur at the time of shipment of the canistered fuel off site. Alternately, removal could be required in the unlikely event of an accident condition that rendered the concrete cask or canister unsuitable for continued long-term storage or for transport. This procedure assumes that the concrete cask is being returned to the reactor cask receiving area. However, the cask may be moved to another facility or area using the same operations. It identifies the general steps to return the loaded canister to the transfer cask and return the transfer cask to the decontamination station, or other designated work. area or facility. Since these steps are the reverse of those undertaken to place the canister in the concrete cask, as described in Section 8.1.2, they are only summarized here.

The concrete cask may be moved using the air pad set or a mobile lifting frame. This procedure assumes the use of the air pad set. If a lifting frame is used, the concrete cask is lifted using four lifting lugs in the top of the cask, and the air pad set and heavy haul transporter are not required. The mobile lifting frame may be self-powered or towed.

At the option of the user, the canister may be removed from the concrete cask and transferred to another concrete cask or to the Universal Transport Cask at the ISFSI site. This transfer is done using the transfer cask, which provides shielding for the canister contents during the transfer.

1. Remove the screens and instrumentation.
2. Using the hydraulic jacking system and the air pad set, move the concrete cask from the ISFSI pad to the heavy-haul transporter. The bed of the transporter must be approximately level with the surface of the pad and sheet metal plates are placed across the gap between the pad and the transporter bed.
   Caution: Do not exceed a maximum lift height of 24 inches when raising the concrete cask.
3. Tow the transporter to the cask receiving area or other designated work area or facility.
4. Remove the concrete cask shield plug and lid. Install the hoist rings in the canister structural lid and torque to the value specified in Table 8.1.1-2. Verify that the hoist rings are fully seated against the structural lid and attach the lift slings. Install the transfer adapter on the top of the concrete cask.

5. Retrieve the transfer cask with the retaining ring installed, and position it on the transfer adapter. Attach the shield door hydraulic cylinders.

   Note: The surrounding air temperature for cask unloading operations shall be ≥ 0°F.
6. Open the shield doors. Attach the canister lift slings to the cask handling crane hook.

   Caution: The attachment point of the two three-legged slings must be at least 75 inches above the top of the canister.
7. Raise the canister into the transfer cask.

   Caution: Avoid raising the canister to the point that the canister top engages the transfer cask retaining ring, as this could result in lifting the transfer cask.
8. Close the shield doors. Lower the canister to rest on the shield doors. Disconnect the canister slings from the crane hook. Install and secure door lock bolts.
9. Retrieve the transfer cask lifting yoke. Engage the transfer cask trunnions and move the transfer cask to the decontamination area or designated work station.

After the transfer cask containing the canister is in the decontamination area or other suitable work station, additional operations may be performed on the canister. It may be opened, transferred to another storage cask, or placed in the Universal Transport Cask.

8.3 Unloading the Transportable Storage Canister

This section describes the basic operations required to open the sealed canister if circumstances arise that dictate the opening of a previously loaded canister and the removal of the stored spent fuel. It is assumed that the canister is positioned in the transfer cask and that the transfer cask is in the decontamination station or other suitable work station in the facility. The principal mechanical operations are the cutting of the closure welds, filling the canister with water, cooling the fuel contents, and removing the spent fuel. Supplemental shielding is used as required. The amount of time the loaded canister may remain in the transfer cask is not limited in accordance with Section 8.4.2.1. The canister cooling water temperature, flow rate and pressure must be limited in accordance with this procedure.

1. Remove the transfer cask retaining ring.
2. Survey the top of the canister to establish the radiation level and contamination level at the structural lid.
3. Set up the weld cutting equipment to cut the structural lid weld (Abrasive grinding, hydrolaser, or similar cutting equipment).
4. Enclose the top of the transfer cask in a radioactive material retention tent, as required.
   Caution: Monitor for any out-gassing. Wear respiratory protection as required.
5. Operate the cutting equipment to cut the structural lid weld.
6. After proper monitoring, remove the retention tent. Remove the cutting equipment and attach a three-legged sling to the structural lid.
7. Using the auxiliary crane, lift the structural lid from the canister and out of the transfer cask.
8. Survey the top of the shield lid to determine radiation and contamination levels. Use supplemental shielding as necessary. Decontaminate the top of the shield lid, if necessary.
9. Reinstall the retention tent. Using an abrasive grinder or hydrolaser, and wearing suitable respiratory protection, cut the welds joining the vent and drain port covers to the shield lid.
   Caution: The canister could be pressurized.
10. Remove the port covers. Monitor for any out-gassing and survey the radiation level at the quick-disconnect fittings. Attach a manually valved line with a vacuum bottle to the vent port quick-disconnect. Open the valve to the vacuum bottle to obtain a gas sample from the vent line. Analyze the gas sample to determine the make up of the canister atmosphere. The presence of fission gases indicates failed fuel and the possible need to handle ruptured fuel.

11. Attach a nitrogen gas line to the drain port quick-disconnect and a discharge line from the vent port quick-disconnect to an off-gas handling system in accordance with the schematic shown in Figure 8.3-1. Set up the vent line with appropriate instruments so that the pressure in the discharge line and the temperature of the discharge gas are indicated. Continuously monitor the radiation level of the discharge line.

    Caution: The discharge gas temperature could initially be above 400°F. The discharge line and fittings may be very hot.

    Note: Any significant radiation level in the discharge gas indicates the presence of fission gas products. The temperature of the gas indicates the thermal conditions in the canister.

12. Start the flow of nitrogen through the line until there is no evidence of fission gas activity in the discharge line. Continue to monitor the gas discharge temperature. When there is no additional evidence of fission gas, stop the nitrogen flow and disconnect the drain and vent port line connections. The nitrogen gas flush must be maintained for at least 10 minutes.

    Note: See Figure 8.3-1 for Canister Reflood Piping and Control Schematic.

13. Perform canister refill and fuel cooldown operations. Attach a source of clean or filtered pool water with a minimum temperature of 70°F and a maximum supply pressure of 25 (+10, -0) psig to the drain port quick-disconnect. Attach a steam rated discharge line to the vent port quick-disconnect and route it to a fuel pool cooler or an in-pool steam condensing unit. Slowly start the flow of clean or filtered pool water to establish a flow rate at 5 (+3, -0) gpm. Monitor the discharge line pressure gage during canister flooding. Stop filling the canister if the canister vent line pressure exceeds 50 psig. Re-establish water flow when the canister pressure is below 30 psig. The discharge line will initially discharge hot gas, but after the canister fills, it will discharge hot water.

    Caution: Relatively cool water may flash to steam as it encounters hot surfaces within the canister.

    Caution: If there are grossly failed or ruptured fuel rods within the canister, very high levels of radiation could rapidly appear at the discharge line. The radiation level of the discharge gas or water should be continuously monitored.

    Caution: Reflooding requires the use of borated water (water with not less than 1,000 parts per million of soluble boron) if borated water was required for the initial fuel loading.

14. Monitor water flow through the canister until the water discharge temperature is below 200°F. Stop the flow of water and remove the connection to the drain line.

    Note: Monitor canister water temperature and reinitiate cooldown operations if temperature exceeds 200°F.

15. Connect a suction pump to the drain port and remove approximately 50 gallons of water. Disconnect and remove the pump.

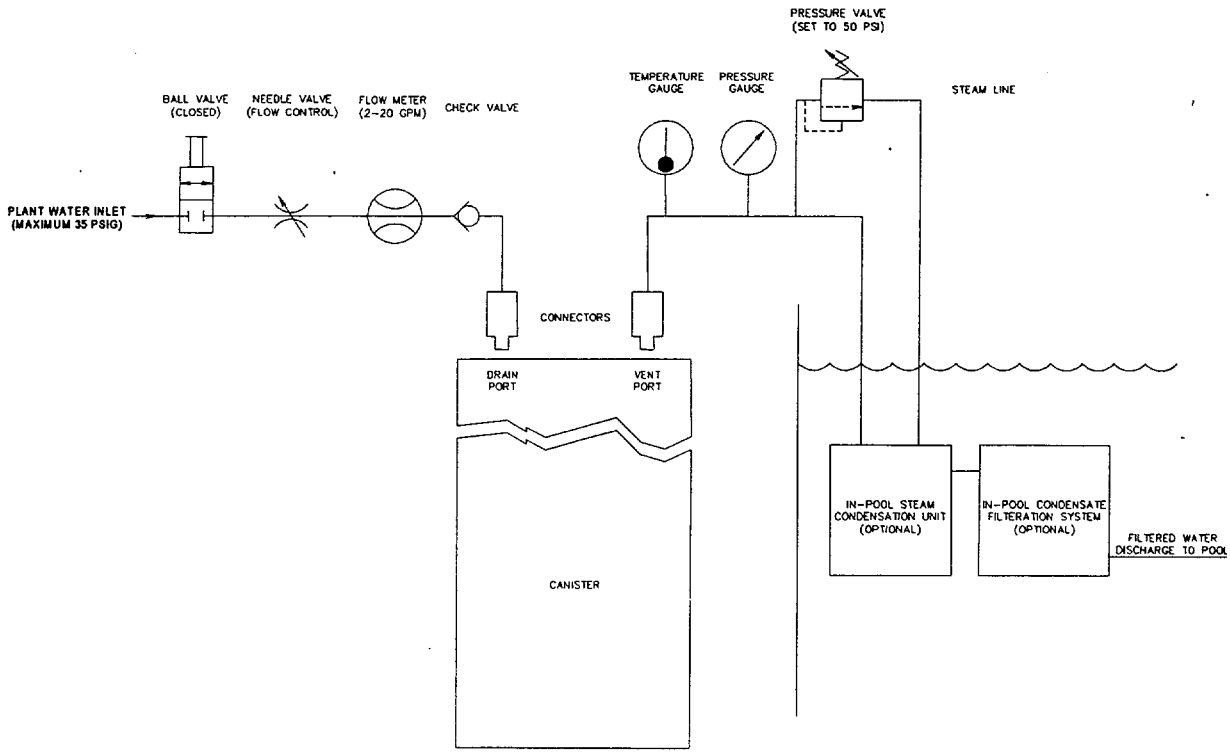
    Note: Air pressure may be used to force water out of the canister by connecting the air line to the vent port and a drain line to the drain port. Air pressure must be regulated to not exceed 50 psig.
16. Set up the weld cutting equipment to cut the shield lid weld (Abrasive grinding, hydrolaser, or similar cutting equipment.). Route the vent line to avoid interference with the weld cutting operation.
17. Tent the top of the transfer cask and wear respiratory protection equipment as required. Attach a hydrogen gas detector to the vent port. Verify that the concentration of hydrogen gas is less than 2.4%.
18. Operate the cutting equipment to cut the shield lid weld.

    Note: Stop the cutting operation if the hydrogen gas detector indicates a concentration of hydrogen gas above 2.4%. Clear the gas before proceeding with the cutting operation.
19. Remove the cutting equipment. Remove supplemental shielding if used. Install the shield lid lifting hoist rings, verifying that the hoist rings are fully seated against the shield lid, and attach a three-legged sling. Attach a tag line to the sling master link to aid in attaching the sling to the auxiliary crane hook (at Step 24).
20. Attach the clean or filtered pool water line to the transfer cask.
21. Retrieve the transfer cask lifting yoke and engage the transfer cask lifting trunnions.
22. Move the transfer cask over the pool and lower the bottom of the transfer cask to the surface. Start the flow of clean or filtered pool water to the transfer cask annulus. Continue to lower the transfer cask, as the annulus fills with water, until the top of the transfer cask is about 4 inches above the pool surface. Hold this position until clean or filtered pool water fills to the top of the transfer cask.
23. Lower the transfer cask to the bottom of the cask loading area and remove the lifting yoke.
24. Attach the shield lid lifting sling to the crane hook.

    Caution: The drain line tube is suspended from the under side of the shield lid. The lid should be raised as straight as possible until the drain tube clears the canister basket. The under side of the shield lid could be highly contaminated.
25. Slowly lift the shield lid. Move the shield lid to one side after it is raised clear of the transfer cask.
26. Visually inspect the fuel for damage.

At this point, the spent fuel could be transferred from the canister to the fuel racks. If the fuel is damaged, special rigging could be required to remove the fuel. In addition, the bottom of the canister could be highly contaminated. Care must be exercised in the handling of the transfer cask when it is removed from the pool. Highly radioactive particles could rest on flat surfaces of the transfer cask resulting in high dose rates.

Figure 8.3-1 Canister Reflood Piping and Controls Schematic



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8.4 Administrative Programs and Controls

This section provides the detailed descriptions of the Administrative Programs described in Section 5.0 of the Technical Specifications presented in Chapter 12.

8.4.1 Radioactive Effluent Control Program

The Radioactive Effluent Control Program implements the requirements of 10 CFR 72.44(d) by helium leak testing the transportable storage canister to ensure that it is leak tight in accordance with ANSI N14.5-1997 and by ensuring that the surface contamination of the transportable storage canister is within the limits evaluated for the site boundary in Section 11.1.5.

The Universal Storage System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required.

This program includes an environmental monitoring program. Each general license user may incorporate storage system operations into their environmental monitoring program for 10 CFR Part 50 operations.

If not provided in accordance with 10 CFR 50 operations, an annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

8.4.1.1 Canister Helium Leak Rate Limit

The canister shield lid weld and port cover welds are helium leak tested in accordance with ANSI N14.5-1997 to demonstrate that the closures are leak tight as defined by the Standard. The leak tight condition provides assurance that there are no radioactive effluents from the contents of the transportable storage canister.

The test condition requires that there be no indication of a helium leak at a test sensitivity of 1×10^{-7} cm^3/sec (helium) through the canister shield lid to canister shell confinement weld, and through the port cover to shield lid welds, to demonstrate a helium leak rate equal to or less than 2×10^{-7} cm^3/sec (helium). Any indication of a leak is unacceptable and repair is required, as no deviation from the leak tight condition is permitted. The test condition, and test method, is described in Sections 8.1.1 and 9.1.3.

As described in Section 8.4.2.1, the amount of time that the canister is in the transfer cask is not limited when the canister is filled with helium. The canister is filled with helium during, and following, the leak test activity. Consequently, adequate time is available to determine the reason for any failure of the helium leak test and for corrective action, if necessary.

8.4.1.2 Canister Surface Contamination Limit

The transportable storage canister surface contamination is controlled during wet loading using design features, procedures and operational controls. The adequacy of the contamination control system and process is demonstrated by measurement of surface contamination prior to installation of the loaded canister in the concrete storage cask.

The surface contamination limits for the accessible canister surface and the accessible inside surface of the transfer cask is 10,000 dpm/100 cm^2 from beta/gamma sources and 100 dpm/100 cm^2 from alpha sources. Meeting the specified limits ensures that the canister surface contamination is very conservatively within the limits evaluated for the site boundary in Section 11.1.5. The evaluation in that section is based upon residual contamination of approximately 1.57×10^5 dpm/100 cm^2 β-γ and 5.24×10^2 dpm/100 cm^2 α activity, on the surface of the design basis canister. The assumptions of this analysis include the whole surface contamination of the loaded canister(s) at the limit allowed and the simultaneous release of all the assumed contamination in a single event. These assumed conditions are highly unlikely, but without these assumed conditions, there is no measurable effect.

Surface contamination levels that exceed these limits must be further decontaminated before the loaded canister may be installed in the concrete storage cask, or the consequences of the higher contamination level must be evaluated using the methodology of Section 11.1.5. This methodology provides for the calculation of residual contamination limits based on the plume dispersion calculations used in Regulatory Guides 1.109 and 1.145. Maximum contamination is limited based on the off-site radiological consequences.

8.4.2 Storage System Operations Program

A training program for the storage system shall be developed under the general licensee's systematic approach to training (SAT). Training modules shall include comprehensive instructions for the operation and maintenance of the storage system and the independent spent fuel storage installation (ISFSI).

A dry run training exercise on preparations, loading, closure, handling, unloading, and transfer of the storage system shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the canister. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to:

a. Moving the concrete cask into its designated loading area
b. Moving the transfer cask with the empty canister into the spent fuel pool
c. Loading one or more dummy fuel assemblies into the canister, including independent verification
d. Selection and verification of fuel assemblies requiring preferential loading
e. Controlling boron concentration in pool water during loading
f. Installing the shield lid
g. Removal of the transfer cask from the spent fuel pool
h. Closing and sealing of the canister to demonstrate pressure testing, vacuum drying, helium backfilling, welding, weld inspection and documentation, and leak testing
i. Supplemental cooling of the loaded canister
j. Transfer cask movement through the designated load path
k. Transfer cask installation on the concrete cask
l. Transfer of the canister to the concrete cask
m. Concrete cask shield plug and lid installation
n. Transport of the concrete cask to the ISFSI
o. Canister removal from the concrete cask
p. Canister unloading, including reflooding and weld removal or cutting

Appropriate mock-up fixtures may be used to demonstrate and/or to qualify procedures, processes or personnel in welding, weld inspection, vacuum drying, helium backfilling, leak testing and weld removal or cutting.

8.4.2.1 Time in the Transfer Cask

This section supports item (h) of the Storage System Operations Program (Section 8.4.2). It specifies the amount of time that a canister loaded with spent fuel may be in the transfer cask in three conditions—with water in the canister, in a vacuum condition and with helium in the canister. These time limits allow the thermal condition of the canister to be controlled such that

the canister components and fuel cladding are protected from unacceptably high temperature or, in the case of water in the canister, that general boiling of water in the canister is precluded.

Canister in the Transfer Cask with Water in the Canister

This section is applicable to Step 12 of Section 8.1.1. The thermal analysis shows that the decay heat of the spent fuel in the canister causes the temperature of the water in the canister to increase once the canister is lifted out of the pool. This control is applied to preclude the occurrence of general boiling of the water in the canister during the canister closing process. It is desirable to avoid general boiling of the water to preclude the occurrence of free moisture in the shield lid weld area, to limit the pressure that could exist in the canister and to avoid the venting of steam at the vent port.

The control of canister water temperature is achieved by monitoring of the time duration of the water condition or by the periodic measuring of the water temperature. The thermal analysis shows that the calculated time to boil increases as the decay heat of the contents decreases, since the heat-up rate of the water is lower with lower decay heat. Section 4.4.3 presents the analytical model and results of that model for the ranges of decay heat load evaluated. The heat load ranges, and time limits associated with those heat loads, are:

Total Heat Load (L) (kW)	PWR Time Limit (Hours)	BWR Time Limit (Hours)
$20.0 < L \leq 23.0$	17	17
$17.6 < L \leq 20.0$	18	18
$14.0 < L \leq 17.6$	20	19
$11.0 < L \leq 14.0$	22	20
$8.0 < L \leq 11.0$	24	23
$L \leq 8.0$	26	31

The indicated time durations, based on heat load, must be applied unless a new time is calculated using the methodology and analytical model presented in Section 4.4.3. However, as specified in the analysis, the temperature of the canister water may periodically be measured to determine when the water is approaching the temperature at which it could boil.

As described in Step 12 of Section 8.1.1, if the time limits shown in the previous table are not used, the water temperature must be monitored beginning 17 hours from the completion of Step 12 and each 2 hours thereafter until the activity is completed. Once the water temperature

reaches 200°F, actions to control the canister water temperature, which may include water recirculation, must be implemented within 2 hours.

Canister in a Vacuum Condition in the Transfer Cask

This section is applicable to LCO 3.1.1 and to Steps 27-32 of Section 8.1.1. The thermal analysis presented in Section 4.4.3 shows that the fuel cladding and canister component temperatures increase during the vacuum drying process. Since the heat-up rate is slower for lower total heat loads, the time required to reach component limits is longer than for the design basis heat load. The spent fuel and the canister component short-term temperature limits will not be exceeded, provided that the time in a vacuum condition is limited based on the canister heat load. The heat load ranges, and time limits associated with those heat loads, are:

Total Heat Load (L) (kW)	PWR Time Limit (Hours)	BWR Time Limit (Hours)
$20.0 < L \leq 23.0$	32	32
$17.6 < L \leq 20.0$	40	38
$14.0 < L \leq 17.6$	48	49
$11.0 < L \leq 14.0$	73	72
$L \leq 11.0$	Not Limited	Not Limited

The indicated time durations, based on heat load, must be applied unless a new time is calculated using the methodology and analytical model presented in Section 4.4.3.

If these time limits are not met, then supplemental cooling of the canister in the transfer cask must be initiated in accordance with LCO 3.1.1. Supplemental cooling may be achieved by returning the loaded canister to the spent fuel pool (in-pool cooling) or initiating forced air flow in the transfer cask annulus (forced air cooling).

Canister Backfilled with Helium in the Transfer Cask

This section is applicable to Step 34 of Section 8.1.1. The evaluation of the loaded canister containing PWR or BWR fuel at the design basis heat load of 23 kW shows that the time the loaded canister, backfilled with helium, may be in the transfer cask, prior to placement in the concrete cask, is not limited. Backfilling the canister with helium ensures that the temperatures of critical components, including the basket support disks, heat transfer disks and fuel cladding, are maintained below short-term limits.

Section 4.4.3 presents the analytical model and results of that model for the decay heat loads evaluated. The design basis heat load limit and time limit associated with those heat loads for PWR and BWR fuel are:

Fuel Type	Total Heat Load (L) (kW)	Canister Condition	Time Limit (Hours)
PWR	$L \leq 23.0$	Helium	Not Limited
BWR	$L \leq 23.0$	Helium	Not Limited

Analyses reported in the Safety Analysis Report conclude that spent fuel cladding and canister material short-term temperature limits will not be exceeded provided that the loaded canister is backfilled with helium when in the transfer cask.

8.4.2.2 Canister Vacuum Drying Pressure

This section supports item (h) of the Storage System Operations Program (Section 8.4.2) and specifies the canister vacuum drying pressure. This section is applicable to Steps 29 and 30 of Section 8.1.1. Heat-up of the canister and contents will occur during vacuum drying, but is controlled by LCO 3.1.1. Dryness of the canister (e.g., no free water) is verified by holding a vacuum of 10 mm Hg or less for a period of not less than 10 minutes with no pressure rise during the 10 minute period after the pressure stabilizes. Operation of the valves to isolate the system can cause fluctuation in the measured pressure. The time period should start after the pressure stabilizes below 10 mm Hg. The vapor pressure of water at 70°F is approximately 30 mm Hg. Selecting a pressure that is 1/3 of the vapor pressure at 70°F ensures that all of the free water in the canister is removed without excursion to a low vacuum condition that could lead to icing. The actual temperature in the loaded canister is expected to be above 70°F, which would result in a higher vapor pressure. Consequently, the vacuum pressure of 10 mm Hg is conservatively selected. Holding the vacuum pressure for 10 minutes demonstrates that there is no free water in the canister since the presence of any free water results in a pressure of at least 30 mm Hg within the canister. A vacuum pressure of 10 mm Hg is above the pressure that could lead to the formation of ice in residual free water.

8.4.2.3 Helium Backfill Pressure and Helium Leak Rate

This section supports item (h) of the Storage System Operations Program (Section 8.4.2) and specifies the canister helium backfill pressure and helium leak rate.

The removal of oxidizing gases that could lead to fuel cladding deterioration is assured by deep vacuum excursion to 3 mm Hg (minimum). After this vacuum condition is achieved the canister is backfilled with helium to one atmosphere. After the backfill, the canister is subjected again to a 3 mm Hg vacuum condition. The canister is then backfilled to one atmosphere (0 [+1, -0] psig) with helium and sealed. The removal of oxidizing gases and the establishment of an inert atmosphere in the canister is controlled by the vacuum excursion to 3 mm Hg in two cycles. These vacuum cycles ensure that the canister contents are dry and that the atmosphere in the canister is essentially free (< one mole) of any gases that could attack the fuel cladding, as recommended by PNL-6365. This control is applicable to Steps 31, 32, 33 and 34 of Section 8.1.1.

The leak tight condition of the canister is verified by performance of a leak test of the installed shield lid and port covers using a test fixture. The leak tight condition ensures that there is no release of contents of the canister in the evaluated normal, off-normal and accident storage conditions. This control is applicable to Step 49 of Section 8.1.1. A leak rate greater than 2 x 10^{-7} cm^3/sec is not permitted, since the analysis for off-site release is based on a leak tight condition. The leak test method must be in accordance with ANSI N14.5-1997 and must be performed by personnel qualified and trained in the test method.

8.4.2.4 Supplemental Cooling in the Transfer Cask

This section supports Item (i) of the Storage System Operations Program (Section 8.4.2). It specifies the amount of time that a canister holding spent fuel may be in the transfer cask following 24 hours of in-pool or forced air supplemental cooling. These time limits allow the thermal condition of the canister to be controlled such that the canister components and fuel cladding are protected from unacceptably high temperature. Supplemental cooling lowers component and fuel cladding temperature so that interrupted operations may be completed.

In-Pool Supplemental Cooling

This section is applicable to LCO 3.1.1. The thermal analysis presented in Section 4.4.3 shows that the fuel cladding and canister component temperatures can be reduced from the values experienced during operations leading to closure of the canister by backfilling the canister with helium and immersing the transfer cask in the spent fuel pool. By reducing the canister component and fuel cladding temperatures, additional time is made available to complete closure operations when the transfer cask is removed from the pool. The amount of additional time

available is dependent on the pool water temperature, on the decay heat load and on the amount of time the in-pool cooling is used. The analysis assumes a 24-hour in-pool cooling period and a pool water temperature of 100°F. Based on the analysis and the heat load ranges, the additional time available as a result of in-pool cooling is:

Total Heat Load (L) (kW)	PWR Time Limit (Hours)	BWR Time Limit (Hours)
$20.0 < L \leq 23.0$	20	20
$17.6 < L \leq 20.0$	27	25
$14.0 < L \leq 17.6$	34	35
$11.0 < L \leq 14.0$	49	56

In-pool cooling is not required for total heat loads less than 11 kW.

The indicated time durations, based on heat load, must be applied unless a new time is calculated using the methodology and analytical model presented in Section 4.4.3.

Forced Air Supplemental Cooling

This section is applicable to LCO 3.1.1. The thermal analysis presented in Section 4.4.3 shows that the fuel cladding and canister component temperatures can be reduced from the values experienced during operations leading to closure of the canister by backfilling the canister with helium and initiating forced air cooling of the canister in the transfer cask. By reducing the canister component and fuel cladding temperature, additional time is made available to complete closure operations when the forced air cooling conditions are met. The amount of additional time available is dependent on the air temperature, air flow rate, on the decay heat load and on the amount of time the forced air cooling is used. The analysis assumes a 24-hour forced air cooling period, an air temperature of 76°F and a flow rate of 375 cubic feet per minute. Based on the analysis and the heat load ranges, the additional time available as a result of forced air cooling is:

Total Heat Load (L) (kW)	PWR Time Limit (Hours)	BWR Time Limit (Hours)
$20.0 < L \leq 23.0$	9	10
$17.6 < L \leq 20.0$	16	15
$14.0 < L \leq 17.6$	23	24
$11.0 < L \leq 14.0$	37	45

Forced air cooling is not required for total heat loads less than 11 kW.

The indicated time durations, based on heat load, must be applied unless a new time is calculated using the methodology and analytical model presented in Section 4.4.3.

8.4.3 ISFSI Operations Program

This section provides the Administrative Program and Controls for implementation of the Safety Analysis Report requirements for ISFSI operations. The ISFSI Operations Program comprises the following operations or activities:

a. ISFSI Pad Construction Parameters
b. Transport to the ISFSI and Associated Lift Height Limits
c. Placement of the Concrete Cask on the ISFSI Pad
d. Concrete Cask Average Surface Dose Rates
e. Air Temperature Monitoring
f. Surveillance After an Off-Normal, Accident or Natural Phenomena Event

8.4.3.1 ISFSI Pad Construction Parameters

This section supports item (a) of the ISFSI Operations Program (Section 8.4.3).

This program provides a means for evaluating ISFSI pad construction parameters to ensure that in the postulated cask tip-over event, the resulting stresses do not exceed those calculated in Section 11.2.12. As described in Section 11.2.12, the postulated tip-over is a hypothetical event. There are no design basis normal, off-normal or accident events, which result in concrete cask tip-over.

The parameters of the ISFSI pad and foundation considered in the analysis are:

Concrete thickness	36 inches maximum
Pad subsoil thickness	10 feet minimum
Specified concrete compressive strength	$\leq$ 5,000 psi at 28 days
Concrete dry density (ρ)	$125 \leq \rho \leq 160$ lbs/ft^3
Soil in place density (ρ)	$100 \leq \rho \leq 160$ lbs/ft^3
Soil modulus of elasticity (PWR)	$\leq$ 60,000 psi
Soil modulus of elasticity (BWR)	$\leq$ 30,000 psi

These parameters must be applied unless a separate analysis is performed, using the methodology of Section 11.2.12, to show that different parameters limit the g-loading of the canister to 60g as assumed in the analysis presented in Section 11.2.12, or to a lessor acceptable value.

8.4.3.2 Transport to the ISFSI and Associated Lift Height Limits

This section supports item (b) of the ISFSI Operations Program (Section 8.4.3).

This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded transfer cask or concrete cask using devices, which are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable when the transfer cask or concrete cask is in the fuel building or is being handled by a device providing support from underneath (i.e., on a rail car, heavy-haul trailer, air pads, etc.). It also provides the means for evaluating concrete cask spacing on the ISFSI pad and concrete cask average surface dose rates after the concrete cask is loaded.

Pursuant to 10 CFR 72.212, this program shall evaluate the site specific transport route conditions.

a. The lift height above a transport surface shall not exceed the limits shown in the following table. The program shall ensure that the transport route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than those prescribed for the reference pad surface.

b. For site specific transport conditions, which are not bounded by the surface characteristics, the program may evaluate the site specific conditions to ensure that the impact loading due to design basis drop events does not exceed 60g. This alternative analysis shall be commensurate with the drop analyses described in Section 11.2.4, using the same methodology for the analysis. The program shall ensure that these alternative analyses are documented and controlled.

c. The transfer cask and concrete cask may be lifted to those heights necessary to perform cask handling operations, including canister transfer, provided the lifts are made with structures and components designed in accordance with the criteria specified in Section 3.4.3.

Item	Orientation	Lifting Height Limit
Transfer Cask	Horizontal	None Established
Transfer Cask	Vertical	None Established
Concrete Cask	Horizontal	Not Permitted
Concrete Cask	Vertical	< 24 inches

8.4.3.3 Placement of the Concrete Cask on the ISFSI Pad

This section supports item (c) of the ISFSI Operations Program (Section 8.4.3).

This program provides a means for evaluating the location of loaded concrete casks on the ISFSI pad to ensure adequate separation of the casks during storage operations. The analysis presented in Section 4.4.1 considers a center-to-center spacing of the concrete casks of not less than 15 feet. This provides a nominal physical separation between concrete casks of 3.7 feet. This spacing is considered in the view factor determination of radiation heat transfer between the concrete casks and must be used unless a different spacing is evaluated using the methodology of Section 4.4.1.1. Spacing greater than 15 feet center-to-center need not be analyzed since this provided greater radiation heat transfer between the concrete casks, improving thermal performance.

In the unlikely event that a spacing ≥ 15 feet is not, or cannot be, achieved in loaded concrete cask placement, a thermal analysis conforming to that provided in Section 4.4.1.1 may be used to evaluate the consequences of the reduced spacing.

8.4.3.4 Concrete Cask Average Surface Dose Rates

This section supports item (d) of the ISFSI Operations Program (Section 8.4.3).

The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public. Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10 CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions in accordance with 10 CFR 72.

The concrete cask average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.

The limits on the concrete cask average surface dose rates are based on the analysis presented in Section 5.4. A sufficient number of locations for taking dose rate measurements are specified as shown in Figure 8.4-1 to ensure the dose rates measured are indicative of the effectiveness of the shielding materials. Dose rates are measured using standard industry practice.

The average surface dose rates are measured on the surface of the concrete cask or at the projected surface at the inlets and outlets. The limits assumed in the design basis are:

a. 50 mrem/hour (neutron + gamma) on the side surface
b. 50 mrem/hour (neutron + gamma) on the top surface
c. 100 mrem/hour (neutron + gamma) at air inlets and outlets

These limits ensure that the concrete cask average surface dose rates during storage operations are bounded by the shielding safety analyses.

If the average surface dose rates are not within these limits, the fuel loading history for the affected canister shall be reviewed to verify correct fuel assembly selection and loading. In the unlikely event that a misloading of the canister is identified in this review, an evaluation must be performed to demonstrate satisfactory performance of the fuel, canister and basket components. This analysis must use the methodology applied in the safety analysis report, primarily considering the thermal analysis of Sections 4.4 or 4.5 as appropriate.

Based on the measured average dose rates, an analysis is performed to determine if the resulting ISFSI off-site or occupational calculated doses exceed regulatory limits in 10 CFR Part 72 or 10 CFR Part 20, respectively. The analysis must use the methodology presented in Section 5.4 and Section 10.3 in determining dose rate consequence. If it is determined that the measured average surface dose rates do not result in the regulatory limits being exceeded, storage operations may continue.

If it is verified that fuel and/or storage system performance are not satisfactory, or that the off-site radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 cannot be met, the fuel assemblies must be placed in a safe condition in the spent fuel pool.

8.4.3.5 Air Temperature Monitoring

This section supports item (e) of the ISFSI Operations Program (Section 8.4.3).

The air temperature monitoring system reports the ISFSI ambient temperature and the outlet temperatures for each of the four outlets on each of the concrete casks in the ISFSI. The ambient temperature is used as the concrete cask inlet air temperature for purposes of analysis. The air temperature monitoring system provides objective evidence of the satisfactory operation of the passive heat removal system of each concrete cask. Satisfactory operation of the heat removal system is required by LCO 3.1.2.

Analyses are performed to evaluate several effects of air temperature and heat removal system operation. The analyses consider the effects of high (106°F) and low temperature (-40°F) (Section 11.1.1), severe ambient temperature (133°F) (Section 11.2.7), blockage of one-half of the air inlet vents (Section 11.1.2), blockage of all of the inlet and outlet vents (Section 11.2.13) and failure of the air temperature monitoring system (Section 11.1.4). Since objective evidence of satisfactory operation may be obtained by physical inspection, the air temperature monitoring system is not classified as important to safety (Section 11.1.4).

As specified in LCO 3.1.2, the maximum difference between the average concrete cask air outlet temperature and ISFSI ambient temperature is monitored to demonstrate satisfactory operation of the concrete cask heat removal system. The temperature difference considered in the heat transfer analysis for the concrete cask air flow is ≤ 102°F (for PWR fuel) and ≤ 92°F (for BWR fuel) based on the maximum fuel decay heat load. So long as this temperature difference is not exceeded, the satisfactory operation of the concrete cask heat removal system is demonstrated.

Blocked air inlets or outlets will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the canister. Based on the analyses, provided the air temperature rise is less than the limits, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long-term fuel cladding integrity.

Since the heat removal system is passive, the air passageway is constructed of carbon steel and stainless steel components, and blockage of the inlets or outlets by external material is unlikely, the allowable temperature difference is unlikely to be achieved. However, should the allowable temperature difference show an unexpected increase or approach the allowable difference, an inspection is required to clear the blockage to restore the system to an operable condition.

The allowable temperature difference values may be revised, provided that the analysis methodology of Section 4.4.1.1 is used to develop the revised temperature differential values.

This methodology ensures the fuel cladding, canister, basket and concrete cask components temperature limits are not exceeded. If a revised temperature differential is not calculated, then the temperature difference of ≤ 102°F (for PWR fuel) and ≤ 92°F (for BWR fuel) provided in the FSAR must be used.

8.4.3.6 Surveillance After an Off-Normal, Accident or Natural Phenomena Event

This section supports item (f) of the ISFSI Operations Program (Section 8.4.3).

A Response Surveillance is required following off-normal, accident or natural phenomena events. The Universal Storage System in use at an ISFSI shall be inspected within 4 hours after the occurrence of an off-normal, accident or natural phenomena event in the area of the ISFSI. This inspection must specifically verify that all the concrete cask inlets and outlets are not blocked or obstructed. At least one-half of the inlets and outlets on each concrete cask must be cleared of blockage or debris within 24 hours to restore air circulation.

The concrete cask and canister shall be inspected if they experience a drop or a tip-over. As described in Section 11.2.12, there are no design basis accident events that result in concrete cask tip-over.

8.4.4 Fuel Selection and Verification Program

A verification program is required to establish that fuel selected for loading in a designated canister meets the constraints and limitations for fuel to be loaded as described in the Certificate of Compliance. The constraints and limitations include, but are not limited to, the fuel descriptions presented in Tables 2.1.1-1 and 2.1.2-1 for PWR and BWR fuels, respectively, and to the loading tables that are applicable to the enrichment, burnup and cool time constraints. The verification programs shall consider the fuel condition and configuration, loading placement within the basket, inserted or replacement components, and configurations of fuel that are explicitly defined.

Based on the fuel parameters considered, the verification program shall establish a loading plan for each canister.

8.4.4.1 Control of Boron Concentration in Pool Water During Loading

This section supports item (e) of the Storage System Operations Program (Section 8.4.2).

The criticality analysis shows that PWR fuel with certain combinations of maximum initial enrichment, fuel content and burnup require credit for the presence of at least 1,000 parts per million of boron in solution in the fuel pool water. This water must be used to flood the canister cavity during underwater PWR fuel loading. The boron in the pool water ensures sufficient thermal neutron absorption so as to preserve criticality control during fuel loading in the basket. Consequently, if boron credit is required for the fuel being loaded, the canister must be flooded with water that contains boron in the proper concentration. Measurement of pool water boron concentration must be done prior to the submergence of the canister in the pool. It is also required any time that the concentration of boron might be diluted by the influx of unborated water (such as washing of the yoke if it is removed from the pool).

8.4.4.2 Fuel Selection Based on Cool Time

A verification program is required to establish that fuel to be loaded in a designated canister is selected based on the uniform loading and preferential loading configurations that are described in Section 2.1. As shown in Tables 2.1.1-2 and 2.1.2-2 for PWR and BWR fuel, respectively, spent fuel with higher burnup must generally be loaded after longer cool times. In addition, fuel with specific combinations of enrichment, burnup and cool time, may be arranged in a given canister so that fuel with dissimilar cool times can be loaded together within the authorized total decay heat load.

The verification program shall require that cool time constraints be considered in the canister fuel loading plan.

8.4.4.3 Verification of Oxide Layer Thickness on High Burnup Fuel

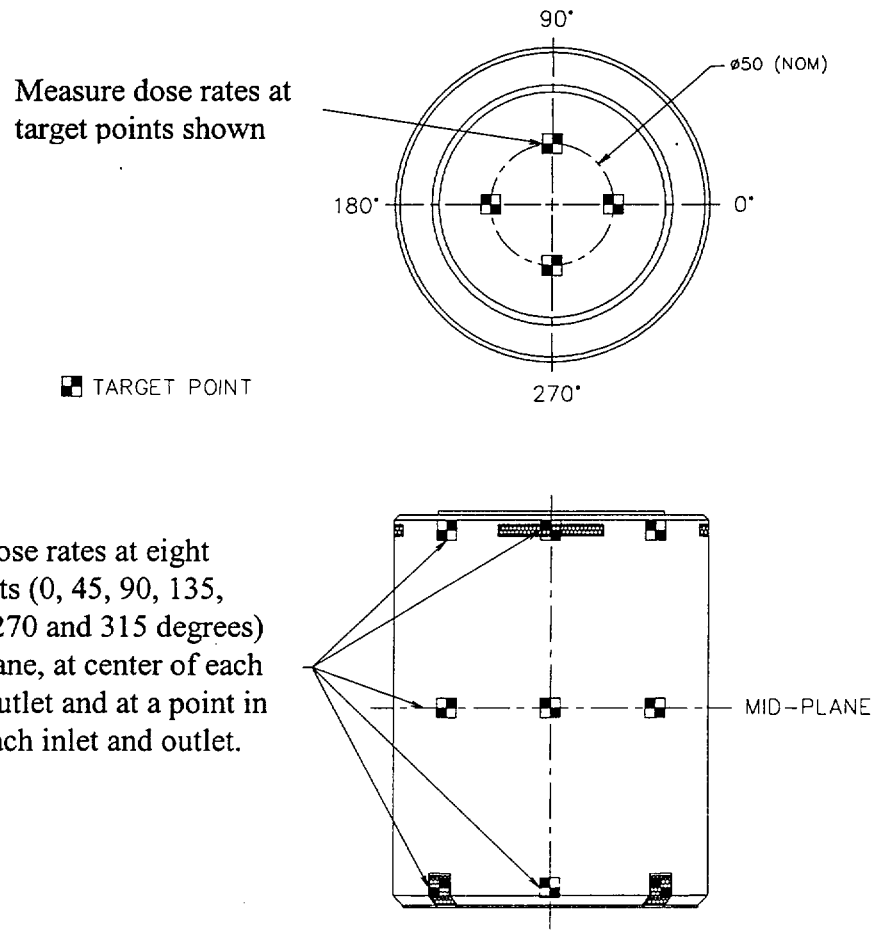
A verification program is required to determine the oxide layer thickness on high burnup fuel by measurement or by statistical analysis. A fuel assembly having a burnup greater than 45,000 MWD/MTU is classified as high burnup. The verification program shall be capable of classifying high burnup fuel as intact fuel or damaged fuel based on the following criteria:

1. A high burnup fuel assembly may be stored as intact fuel provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and that no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, and that the fuel assembly is otherwise intact fuel.

2. A high burnup fuel assembly not meeting the cladding oxide thickness criteria for intact fuel or that has an oxide layer that is detached or spalled from the cladding is stored as damaged fuel.

A fuel assembly with a burnup greater than 45,000 MWD/MTU must be preferentially loaded in periphery positions of the basket.

Figure 8.4-1 Concrete Cask Surface Dose Rate Measurement



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8.5 References

1. "Safety Analysis Report for the UMS® Universal Transport Cask," Docket Number 71-9270, NAC International, April 1997.

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9.0 ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

This chapter specifies the acceptance criteria and the maintenance program for the Universal Storage System primary components - the Vertical Concrete Cask and Transportable Storage Canister. The system components, such as the concrete cask liner, base and air outlets, and the canister shell with the bottom plate, the shield and structural lids, and the basket that holds the spent fuel, are shop fabricated. The concrete cask consists of reinforced concrete placed around the steel liner and base that are integral to its performance. The liner forms the central cavity of the vertical concrete cask, which is mounted on the base. The liner/base interface forms air inlet passageways to the central cavity. The inlets allow cool ambient air to be drawn in and passed by the canister that contains the fuel. Air outlets at the top of the concrete cask allow the air heated by the canister wall and concrete cask liner to be discharged. The base of the concrete cask acts as a pedestal to support the canister during storage.

The concrete reinforcing steel (rebar) is bent in the shop and delivered to the concrete cask construction site. Concrete cask construction begins with the erection of the cask liner onto the steel base. Reinforcing steel is placed around the liner, followed by a temporary outer form which encircles the cask liner and reinforcing steel. The temporary form creates an annulus region between the liner and the form into which the concrete is placed.

As described in Section 8.1.3, the vertical concrete cask may be lifted by: (1) hydraulic jacks and moved by using air pads underneath the base; or (2) lifting lugs and moved by a mobile lifting frame.

9.1 Acceptance Criteria

The acceptance criteria specified below ensure that the concrete cask, including the liner, base, and canister are fabricated, assembled, inspected and tested in accordance with the requirements of this SAR and the license drawings presented in Section 1.6.

9.1.1 Visual and Nondestructive Examination Inspection

The acceptance test program establishes a set of visual inspections, nondestructive examinations and test requirements and corresponding criteria to determine the adequacy of the fabricated components and sub-components. Similar acceptance requirements and criteria are established for the on-site concrete cask construction. Once in service, cask performance monitoring is used

to assure that the cask is operating within the expected temperature range. Satisfactory results for these inspections, examinations and tests demonstrate that the components comply with the requirements of this Safety Analysis Report and the license drawings.

A fit-up test of the canister shell and sub-components is performed during the canister acceptance inspection. The fit-up test demonstrates that the canister, basket, shield lid and structural lid can be properly assembled during canister closure operations, and that the fuel assemblies can be installed in the fuel tubes.

A visual inspection is performed on all materials used for concrete cask, canister and basket fabrication. The visual inspection applies to finished surfaces of the components. All welds (shop and field installed) are visually inspected for defects prior to the nondestructive examinations that may also be specified. The welding of the canister is performed in accordance with ASME Code, Section III, Subsection NB-4000 [1], except as described by this Safety Analysis Report. (See Section 7.1.)

The visual inspections of the canister welds are performed in accordance with the ASME Code, Section V, Article 9 [2]. Acceptance criteria for the visual examinations of the canister welds are in accordance with ASME Code, Section VIII, Division 1, UW-35 and UW-36 [3]. Unacceptable welds in the canister are repaired as required by ASME Code, Section III, Subsection NB-4450 and reexamined in accordance with the original acceptance criteria.

Welding of the vertical concrete cask's steel components, including field installed welds, is performed in accordance with ANSI/AWS D1.1-96 [4], or ASME Code Section VIII, Division 1, Part UW, and inspected in accordance with ANSI/AWS D1.1, Section 8.15.1, or ASME Code Section VIII, Division 1, UW-35 and UW-36. Weld procedures and welder qualifications shall be in accordance with ANSI/AWS D1.1, Section 5 or ASME Code, Section IX [5].

Welding of the basket assemblies for spent fuel is performed in accordance with ASME Code, Section III, Subsection NG-4000 [6]. Visual examination of the welds is performed per the requirements of ASME Code, Section V, Article 9. Acceptance criteria for the visual examination of the basket assembly welds are those of ASME Code, Section III, Paragraphs NB-4424 and NB-4427. Any required weld repairs are performed in accordance with ASME Code, Section III, Subsection NG-4450 and reexamined in accordance with the original acceptance criteria.

All visual inspections are performed by qualified personnel according to written and approved procedures.

9.1.1.1 Nondestructive Weld Examination

The acceptance test program establishes a set of visual inspections, nondestructive examinations and test requirements for the fabrication and assembly of the storage cask, canister and transfer cask. Satisfactory results for these inspections, examinations and tests demonstrate that the components comply with the requirements of the SAR and the license drawings.

A fit-up test of the canister and its components is performed during the acceptance inspection. The fit-up test demonstrates that the canister, basket, shield lid and structural lid can be properly assembled during fuel loading and canister closure operations.

A visual inspection is performed on all materials and welds used for storage cask, canister, basket and transfer cask fabrication. The visual inspection applies to finished surfaces of the components. All welds (shop and field installed) are visually inspected for defects prior to the nondestructive examinations that are specified.

The fabrication of the canister is performed in accordance with ASME Code, Section III, Article NB-4000, except as described in Section 7.1.3 and Table 12B3-1. The visual examinations of the canister welds are performed in accordance with the ASME Code Section V, Article 9 [2]. Acceptance criteria for the visual examinations of the canister welds are in accordance with ASME Code Section III, NB-4424 and NB-4427. Required weld repairs on the canister are performed in accordance with ASME Code Section III, NB-4450, and are reexamined in accordance with the original acceptance criteria.

Fabrication of the storage cask's steel components, including field installed welds, is performed in accordance with either: 1) ANSI/AWS D1.1-96 [4] with visual examination in accordance with ANSI/AWS D1.1, Section 8.15.1; or 2) ASME Code Section VIII with visual examination in accordance with ASME Code Section V, Article 9.

Fabrication of the basket assembly for spent fuel is performed in accordance with ASME Code Section III, NG-4000 [6]. Visual examination of the welds is performed per the requirements of ASME Code Section V, Article 9. Acceptance criteria for the visual examination of the basket assembly welds is that of ASME Code Section III, Subsection NG-5360. Any

required weld repairs are performed in accordance with ASME Code Section III, NG-4450 and the repaired weld is reexamined in accordance with the original acceptance criteria.

Qualified personnel perform all visual inspections according to written and approved procedures. The results of all visual weld inspections are recorded.

9.1.1.2 Fabrication Inspections

Materials used in the fabrication of the vertical concrete cask and transportable storage canister are procured with material certifications and supporting documentation as necessary to assure compliance with procurement specifications. All materials are receipt inspected for appropriate acceptance requirements, and for traceability to required material certification, appropriate for the safety classification of the components.

The canister is fabricated to the requirements of ASME Code, Section III, Subsection NB. Specific alternatives to the ASME Code are described in Table 4-1 of Chapter 12. The basket structure is fabricated to ASME Code, Section III, Subsection NG. Shop fabricated components of the concrete cask are fabricated in accordance with ANSI/AWS D1.1-96, or ASME Code, Section VIII, Part UW.

A complete dimensional inspection of critical components and a components fit-up test is performed on the canister to ensure proper assembly in the field. Dimensions shall conform to the engineering drawings.

On completion of fabrication, the canister, basket and other shop fabricated components are inspected for cleanliness. All components must be free of any foreign material, oil, grease and solvents. All surfaces of carbon steel components assembled for the concrete cask that are not in direct contact with the concrete, are coated with a corrosion-resistant paint.

9.1.1.3 Construction Inspections

Concrete mixing slump, air entrainment, strength and density are field verified using either the American Concrete Institute (ACI) or the American Society for Testing and Materials (ASTM) standard testing methods and acceptance criteria, as appropriate, to ensure adequacy. Reinforcing steel is installed per specification requirements based on ACI-318 [7].

9.1.2 Structural and Pressure Test

The transportable storage canister is pressure tested at the time of use. After loading of the canister basket with spent fuel, the shield lid is welded in place after approximately 50 gallons of water are removed from the canister. Removal of the water ensures that the water level in the canister is below the bottom of the shield lid during welding of the shield lid to the canister shell. Prior to removing the remaining spent fuel pool water from the canister, the canister is pressure tested at 35 psia. This pressure is held for 10 minutes. Any loss of pressure during the test period is unacceptable. The leak must be located and repaired. The pressure test procedure is described in Section 8.1.1.

9.1.2.1 Transfer Casks

The transfer cask lifting trunnions and the bottom shield doors and rails shall be tested in accordance with the requirements of ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4,500 kg) or More for Nuclear Materials" [8].

The standard transfer cask lifting trunnion load test shall consist of applying a vertical load of 660,000 pounds, which is greater than 300 percent of the maximum service load (214,300 pounds) for the transfer cask and loaded canister with the shield lid and full of water. The 100-ton transfer cask lifting trunnion load test shall consist of applying a vertical load of 580,000 pounds, which is greater than 300 percent of the maximum service load (191,900 pounds) for the loaded canister with the shield lid and full of water. The bottom shield door and rail load test shall consist of applying a vertical load of 266,000 pounds, which is over 300 percent of the maximum service load (88,000 pounds) for the loaded canister with the shield lid and full of water. These maximum service loads are based on the Class 5 BWR configuration, which is the heaviest configuration and, thus, bounds all of the other configurations.

The load tests shall be held for a minimum of 10 minutes and shall be performed in accordance with approved, written procedures.

Following completion of the lifting trunnion load tests, all trunnion welds and all load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking. Magnetic particle examinations shall be performed in accordance with ASME Code Section V, Articles 1 and 7, with acceptance in accordance with ASME Code Section III, NF-5340. Similarly,

following completion of the bottom shield door and rail load tests, all door rail welds and all load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking.

Any evidence of permanent deformation, cracking or galling of the load bearing surfaces or unacceptable liquid penetrant examination results, shall be cause for evaluation, rejection, or rework of the affected component. Liquid penetrant examinations of all load bearing welds shall be performed in accordance with ASME Code Section V, Article 6, with acceptance in accordance with ASME Code Section III, NF-5350.

9.1.2.2. Concrete Cask

The concrete cask, at the option of the user/licensee, may be provided with lifting lugs to allow for the vertical handling and movement of the concrete cask. The lifting lugs are provided as two sets of two lugs each, through which a lifting pin is inserted and connected to a specially designed mobile lifting frame. The concrete cask lifting lug system and mobile lifting frame and pins are designed, analyzed, and load tested in accordance with ANSI N14.6. The concrete cask lifting lug load test shall consist of applying a vertical load of at least 520,000 pounds, which is greater than 150 percent of the maximum concrete cask weight of 313,900 pounds plus a 10 percent dynamic load factor.

The test load shall be applied for a minimum of 10 minutes in accordance with approved, written procedures. Following completion of the load test, all load bearing surfaces of the lifting lugs shall be visually inspected for permanent deformation, galling, or cracking. Liquid penetrant examinations of load bearing surfaces shall be performed in accordance with ASME Code, Section V, Article 6, with acceptance criteria in accordance with ASME Code, Section III, Subsection NF, NF-5350.

Any evidence of permanent deformation, cracking, or galling, or unacceptable liquid penetrant examination results for the load bearing surfaces of the lifting anchors shall be cause for evaluation, rejection, or rework and retesting.

9.1.2.3 Transportable Storage Canister

The transportable storage canister shell may be hydrostatically or pneumatically pressure tested during fabrication in accordance with Section NB-6200 or NB-6300 of the ASME Code, respectively. Hydrostatic testing will be performed in accordance with NB-6221 using 1.25 times the design pressure of 15 psig. The test pressure shall be held a minimum of 10 minutes in

accordance with NB-6223. Examination after the pressure test shall be in accordance with NB-6224. Alternately, a pneumatic pressure test may be performed in accordance with NB-6321 using 1.2 times the design pressure of 15 psig. The test pressure shall be held a minimum of 10 minutes in accordance with NB-6323. Examination after the pressure test shall be in accordance with NB-6224.

The canister shell shall consist of the completed Shell Weldment as shown on Drawing 790-582.

If the pressure test is not performed during fabrication, a pressure test must be performed upon closure of the canister with the shield lid as described in Section 8.1.1 of the operating procedures.

9.1.3 Leak Tests

The canister is leak tested at the time of use. After the pressure test described in Section 9.1.2, the canister is drained of residual water, vacuum dried and backfilled with helium. The canister is pressurized with helium to 0 psig. The shield lid to canister shell weld and the weld joining the port covers to the shield lid, are helium leak tested using a leak test fixture installed above the shield lid. The leaktight criteria of 2.0×10^{-7} cm^3/sec (helium) of ANSI N14.5[1] is applied. The leak test is performed at a sensitivity of 1.0×10^{-7} cm^3/sec (helium). Any indication of a leak of 2.0×10^{-7} cm^3/sec (helium), or greater, is unacceptable and repair is required as appropriate.

9.1.4 Component Tests

The components of the Universal Storage System do not require any special tests in addition to the material receipt, dimensional, and form and fit tests described in this chapter.

9.1.4.1 Valves, Rupture Disks and Fluid Transport Devices

The transportable storage canister and the vertical concrete cask do not contain rupture disks or fluid transport devices. There are no valves that are part of the confinement boundary for transport or storage. Quick-disconnect valves are installed in the vent and drain ports of the shield lid. These valves are convenience items for the operator, as they provide a means of quickly connecting ancillary drain and vent lines to the canister. During storage and transport, these fittings are not accessible, as they are covered by port covers that are welded in place when

the canister is closed. As presented for storage and transport, the canister has no accessible valves or fittings.

9.1.4.2 Gaskets

The transportable storage canister and the vertical concrete cask have no mechanical seals or gaskets that form an integral part of the system, and there are no mechanical seals or gaskets in the confinement boundary.

9.1.5 Shielding Tests

Based on the conservative design of the Universal Storage System for shielding criteria and the detailed construction requirements, no shielding tests of the vertical concrete cask are required.

9.1.6 Neutron Absorber Tests

A neutron absorbing material is used for criticality control in the fuel tubes of the PWR, BWR and oversize BWR fuel tubes. The placement and dimensions of the neutron absorber are as shown on the License Drawings for these components. The neutron absorbing material is an aluminum matrix material formed from aluminum and boron carbide, available from a number of qualified vendors. The mixing of the aluminum and boron carbide powder forming the neutron absorber material is controlled to assure the specified ^{10}B areal density, where the areal density is also specified on the component License Drawings. The constituents of the neutron absorber material shall be verified by chemical testing and spectroscopy and by physical property measurement to ensure the quality of the finished plate or sheet. The results of all neutron absorber material tests and inspections, including the results of wet chemistry coupon testing, are documented and become part of the quality records documentation package for the fuel tube and basket assembly.

Aluminum/boron carbide neutron absorbing material is available under trade names such as BORAL® and METAMIC®.

BORAL is manufactured by AAR Advanced Structures (AAR) of Livonia, Michigan, under a Quality Assurance/Quality Control program in conformance with the requirements of 10 CFR 50, Appendix B. AAR uses a computer-aided manufacturing process that consists of several steps. The initial step is the mixing of the aluminum and boron carbide powders that form the core of

the finished material. The amount of each powder is a function of the desired ^{10}B areal density. The methods used to control the weight and blend of the powders are patented and proprietary processes of AAR.

METAMIC is similarly manufactured by California Consolidated Technology, Inc. (CCT). CCT uses patented and proprietary processes to control the weight and blend of the powders used to meet the ^{10}B content specification and also uses a computer-aided manufacturing process to form the neutron absorber plates.

After manufacturing, test samples from each batch of neutron absorber sheets shall be tested using wet chemistry techniques to verify the presence, proper distribution, and minimum weight percent of ^{10}B. The tests shall be performed in accordance with approved written procedures.

9.1.6.1 Neutron Absorber Material Sampling Plan

The neutron absorber sampling plan is selected to demonstrate a 95/95 statistical confidence level in the neutron absorber sheet material compliance with the specification. In addition to the specified sampling plan, each sheet of material is visually and dimensionally inspected using at least 6 measurements on each sheet. No rejected neutron absorber sheet is used. The sampling plan is supported by written and approved procedures.

The sampling plan requires that a coupon sample be taken from each of the first 100 sheets of absorber material. Thereafter, coupon samples are taken from 20 randomly selected sheets from each set of 100 sheets. This 1 in 5 sampling plan continues until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder, aluminum powder, or aluminum extrusion) or a process change. The sheet samples are indelibly marked and recorded for identification. This identification is used to document neutron absorber test results, which become part of the quality record documentation package.

9.1.6.2 Neutron Absorber Wet Chemistry Testing

Wet chemistry testing of the test coupons obtained from the sampling plan is used to verify the ^{10}B content of the neutron absorber material. Wet chemistry testing is applied because it is considered to be the most accurate and practical direct measurement method for determining ^{10}B, boron and B_4C content of metal materials and is considered by the Electric Power Research Institute (EPRI) to be the method of choice for this determination.

An approved facility with chemical analysis capability, which could include the neutron absorber vendor's facility, shall be selected to perform the wet chemistry tests. Personnel performing the testing shall be trained and qualified in the process and in the test procedure.

Wet chemistry testing is performed by dissolving the aluminum in the matrix, including the powder and cladding, in a strong acid, leaving the B_4C material. A comparison of the amount of B_4C material remaining to the amount required to meet the ^{10}B content specification is made using a mass-balance calculation based on sample size.

A statistical conclusion about the neutron absorber sheet from which the sample was taken and that batch of neutron absorber sheets may then be drawn based on the test results and the controlled manufacturing processes.

The adequacy of the wet chemistry method is based on its use to qualify the standards employed in neutron blackness testing. The neutron absorption performance of a test material is validated based on its performance compared to a standard. The material properties of the standard are demonstrated by wet chemistry testing. Consequently, the specified test regimen provides adequate assurance that the neutron absorber sheet thus qualified is acceptable.

9.1.6.3 Acceptance Criteria

The wet chemistry test results shall be considered acceptable if the ^{10}B areal density is determined to be equal to, or greater than, that specified on the fuel tube License Drawings. Failure of any coupon wet chemistry test shall result in 100% sampling, as described in the sampling plan, until compliance with the acceptance criteria is demonstrated.

9.1.7 Thermal Tests

No thermal acceptance testing of the Universal Storage System is required during construction. Thermal performance of the system is confirmed in accordance with the procedure specified in Section 9.2.3. In addition, temperature measurements are taken at the air outlets of the concrete cask(s) placed in service, in accordance with Section 8.4.3.5, as verification of the thermal performance of the storage system.

9.1.8 Cask Identification

A stamped, stainless steel nameplate is permanently attached on the outer surface of the concrete cask as shown on Drawing No. 790-562.

The nameplate is installed at approximately eye level and includes the following information:

Vertical Concrete Cask	
Owner:	(Utility Name)
Designer:	NAC International Inc.
Fabricator:	(Vendor Name)
Date of Manufacture:	(mm/dd/yy)
Model Number:	(UMS-XXX)
Cask No.:	(XXX)
Date of Loading:	(mm/dd/yy)
Empty Weight:	(Pounds [kilograms])

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9.2 Maintenance Program

The Universal Storage System is a passive system. No active components or systems are incorporated in the design. Consequently, only a minimal amount of maintenance is required over its lifetime, as described in Section 9.2.3.

9.2.1 Subsystems Maintenance

The Universal Storage System has no active or passive subsystems that require scheduled maintenance. As described in Section 9.2.3, the air inlets and outlets are subject to daily inspection to identify blockage of the vents, which could result in a reduction in cooling air flow.

9.2.2 Valves, Rupture Discs, and Gaskets on the Containment Vessel

The Universal Storage System has no valves, gaskets, rupture discs, seals, or accessible penetrations. Consequently, there is no maintenance associated with these types of features.

9.2.3 Continuing Maintenance Requirements

Recommended maintenance for the vertical concrete cask in normal conditions is specified below. The surveillance requirements for verifying continued satisfactory performance of the heat removal system are described in Section 8.4.3.5. It is not necessary to inspect the canister during the storage period as long as the thermal performance is normal, based on daily temperature verification.

1. Daily surveillance of the vertical concrete casks:

    - Visual inspection of air inlet and outlets for detection of blockage.
    - Verify that the inlet and outlet screens are in place, and are whole and secure.
    - Record the ambient temperature and air outlet temperature for each vertical concrete cask upon placement in service. Thereafter, the temperatures shall be recorded on a daily basis to verify the continuing thermal performance of the system.
    - Visual inspection of the ISFSI site for security and safeguards.

2. Annual inspection of the vertical concrete cask exterior:

- Visual inspection of surface for chipping, spalling or other surface defects. Any defects larger than one inch in diameter (or width) and deeper than one inch shall be regrouted, according to the grout manufacturer's recommendations.

- Reapplication of corrosion-inhibiting (external) coatings on accessible surfaces, including concrete cask lifting lugs, if present.

9.2.4 Transfer Cask Maintenance

The transfer cask trunnions and shield door assemblies shall be visually inspected for gross damage and proper function prior to each use. Annually, the lifting trunnions, shield doors and shield door rails shall be either dye penetrant or magnetic particle examined, using the examination method appropriate to the material. The examination method shall be in accordance with Section V of the ASME Code. The acceptance criteria shall be in accordance with Section III, Subsection NF, Article NF-5350 or NF-5340 as appropriate to the examination method, as required by ANSI N14.6.

The annual examination may be omitted in periods of nonuse of the transfer cask, provided that the transfer cask examination is performed prior to the next use of the transfer cask.

Annually, the coating applied to the carbon steel surfaces of the transfer cask shall be inspected, and any chips, cracks or other defects in the coating shall be repaired.

9.3 References

1. ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NB, "Class 1 Components," 1995 Edition with 1997 Addenda.

2. ASME Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination," 1995 Edition with 1997 Addenda.

3. ASME Boiler and Pressure Vessel Code, Section VIII, Subsection B, Part UW, "Requirements for Pressure Vessels Fabricated by Welding," 1995 Edition with 1997 Addenda.

4. American Welding Society, Inc., "Structural Welding Code - Steel," AWS D1.1, 1996.

5. ASME Boiler and Pressure Vessel Code, Section IX, "Welding and Brazing Qualifications," 1995 Edition with 1997 Addenda.

6. ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NG, "Core Support Structures," 1995 Edition with 1997 Addenda.

7. American Concrete Institute, "Building Code Requirements for Structural Concrete," ACI-318-95, October 1995.

8. American National Standards Institute, "Radioactive Materials - Special Lifting Devices for Shipping Containers Weighting 10,000 Pounds (4,500 kg) or More," ANSI N14.6-1993, 1993.

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10.0 RADIATION PROTECTION

10.1 Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)

The Universal Storage System provides radiation protection for all areas and systems that may expose personnel to radiation or radioactive materials. The components of the PWR and BWR configurations of the system that require operation, maintenance and inspection are designed, fabricated, located, and shielded so as to minimize radiation exposure to personnel.

10.1.1 Policy Considerations

It is the policy of NAC International (NAC) to ensure that the Universal Storage System is designed so that operation, inspection, repair and maintenance can be carried out while maintaining occupational exposure as low as is reasonably achievable (ALARA).

10.1.2 Design Considerations

The design of the Universal Storage System complies with the requirement of 10 CFR 72.3 [1] concerning ALARA and meets the requirements of 10 CFR 72.126(a) and 10 CFR 20.1101 [2] with regard to maintaining occupational radiation exposures ALARA. Specific design features that demonstrate the ALARA philosophy are:

- Material selection and surface preparation that facilitate decontamination.
- A basket configuration that allows spent fuel canister loading using accepted standard practice and current experience.
- Positive clean water flow in the transfer cask/canister annulus to minimize the potential for contamination of the canister surface during in-pool loading.
- Passive confinement, thermal, criticality and shielding systems that require no maintenance.
- Thick steel and concrete walls to reduce the side surface dose rate of the concrete cask to less than 50 mrem/hr (average).

- Nonplanar cooling air pathways to minimize radiation streaming at the inlets and outlets of the vertical concrete cask.

- Use of remote, automated outlet air temperature measurement to reduce surveillance time.

10.1.3 Operational Considerations

The ALARA philosophy is incorporated into the procedural steps necessary to operate the Universal Storage System in accordance with its design. The following features or actions, which comprise a baseline radiological controls approach, are incorporated in the design or procedures to minimize occupational radiation exposure:

- Use of automatic equipment for welding the shield lid and structural lid to the canister shell.

- Use of automatic equipment for weld inspections.

- Decontamination of the exterior surface of the transfer cask, welding of the shield lid, and pressure testing of the canister while the canister remains filled with water.

- Use of quick-disconnect fittings at penetrations to facilitate required service connections.

- Use of remote handling equipment, where practical, to reduce radiation exposure.

- Use of prefabricated, shaped temporary shielding, if necessary, during automated welding equipment set up and removal, during manual welding, during weld inspection of the shield lid, and during all other canister closing and sealing operations conducted at the shield lid.

The operational procedures at a particular facility are determined by the user's operational conditions and facilities.

10.2 Radiation Protection Design Features

The radiation shielding design description is provided in Sections 5.3.1 and 5.3.2. The design criteria radiation exposure rates are summarized in Table 2-1. The principal radiation protection design features are the shielding necessary to meet the design objectives, the placement of penetrations near the edge of the canister shield lid to reduce operator exposure and handling time, and the use of shaped supplemental shielding for work on and around the shield lid, as necessary. This supplemental shielding reduces operator dose rates during the welding, inspection, draining, drying and backfilling operations that seal the canister. An optional supplemental shielding fixture, shown in Drawing 790-613, may be installed in the air inlets to reduce the radiation dose rate at the base of the vertical concrete cask. Similarly, an alternate baffle design (license drawing 790-614) may be employed. The additional mass in this baffle will also reduce dose rates at the base of the concrete cask. The change in the stand thickness associated with the alternate baffle design produced no significant increase in the cask surface dose rates.

Radiation exposure rates at various work locations are determined for the principal Universal Storage System operational steps using a combination of the SAS4 [3], MCBEND [6] and SKYSHINE III [4] computer codes. The use of SAS4 and MCBEND is described in Section 5.1.2. The SKYSHINE III code is discussed in Section 10.4. The calculated dose rates decrease with time.

10.2.1 Design Basis for Normal Storage Conditions

The radiation protection design basis for the Universal Storage System vertical concrete cask is derived from 10 CFR 72 and the applicable ALARA guidelines. The design basis surface dose rates, and the calculated surface and 1 meter dose rates are:

Vertical Concrete Cask	Design Basis Surface Dose Rate (mrem/hr)	Surface Dose Rate (mrem/hr)		1 Meter Maximum Dose Rate (mrem/hr)	
		PWR	BWR	PWR	BWR
Side wall	50.0 (avg.)	37.3	22.7	25.3	15.4
Air inlet	100.0	6.8	8.5	<5.0	5.0
Air outlet	100.0	65.6	50.6	12.5	7.5
Top lid	50.0 (avg.)	26.1	19.7	13.3	8.5

The calculated dose rates at these, and at other dose points, are reported in Sections 5.1.3 and 5.4.1.3. The dose rates presented are for the design basis 40,000 MWD/MTU, 5-year cooled fuel. These dose rates bound those of the higher burnup, but longer cooled, fuel described in Sections 2.1 and 2.5.

Activities associated with closing the canister, including welding of the shield and structural lids, draining, drying, backfilling and testing, may employ temporary shielding to minimize personnel dose in the performance of those tasks.

10.2.2 Design Basis for Accident Conditions

Damage to the vertical concrete cask after a design basis accident does not result in a radiation exposure at the controlled area boundary in excess of 5 rem to the whole body or any organ. The high energy missile impact is estimated to reduce the concrete shielding thickness, locally at the point of impact, by approximately 6 inches. Localized cask surface dose rates for the removal of 6 inches of concrete are estimated to be less than 250 mrem/hr for the PWR and BWR configurations.

A hypothetical accident event, tip-over of the vertical concrete, is considered in Section 11.2.12. There is no design basis event that would result in the tip-over of the vertical concrete cask.

10.3 Estimated On-Site Collective Dose Assessment

Occupational radiation exposures (person-mrem) resulting from the use of the Universal Storage System are calculated using the estimated exposure rates presented in Sections 5.1.3, 5.4.1.3, 5.4.2.3 and 10.2.1. Exposure is evaluated by identifying the tasks and estimating the exposure duration and number of personnel performing those tasks based on industry experience. The tasks identified are based on the design basis operating procedures, as presented in Chapter 8.0.

Dose rates for the standard transfer cask and the concrete storage cask are calculated using the shielding analysis design basis fuel assemblies. The shielding design basis PWR assembly is the Westinghouse 17x17 standard fuel assembly, with an initial enrichment of 3.7 wt % ^{235}U. The design basis BWR assembly is the GE 9x9, with 79 fuel rods and an initial enrichment of 3.25 wt % ^{235}U. Both design basis fuel assemblies have an assumed burnup of 40,000 MWD/MTU, and a cool time of 5 years. The selection of these assemblies for the shielding design basis is described in Section 5.1. Exposures bound those of the higher burnup, but longer cooled fuel. The principal parameters of these assemblies are presented in Table 2.1-1.

As described in Section 5.1, there are no single PWR or BWR design basis assemblies for the 100-ton transfer cask. Instead, the seven bounding fuel assembly types listed in Section 5.5, are employed to calculate the 100-ton cask bounding dose rates using the minimum allowable cool time tables.

10.3.1 Estimated Collective Dose for Loading a Single Universal Storage System

This section estimates the collective dose due to the loading, sealing, transfer and placement on the independent spent fuel storage installation (ISFSI) pad of the Universal Storage System. The analysis assumes that the exposure incurred by the operators is independent of background radiation, as background radiation varies from site to site. The number of persons allocated to task completion is a typical number required for the task. Working area exposure rates are assigned based on the orientation of the worker with respect to the source and take into account the use of temporary shielding.

Table 10.3-1 summarizes the estimated total exposure by task, attributable to the loading, transfer, sealing and placement of a design basis Universal Storage System based on the use of the standard transfer cask. Table 10.3-2 summarizes total exposure based on the use of the 100-ton transfer cask.

Exposures associated with shield lid operations are based on the presence of a temporary 5-inch thick steel shield.

This estimated dose is considered to be conservative as it assumes the loading of a cask with design basis fuel (or the limiting fuel from the 100-ton transfer cask analysis) and does not account for efficiencies in the loading process that occur with experience.

10.3.2 Estimated Annual Dose Due to Routine Operations

Once in place, the ISFSI requires limited ongoing inspection and surveillance throughout its service life. The annual dose evaluation considers the combination of inspection and surveillance requirements described in the following bulleted list, and those tasks that are anticipated to be representative of an operational facility. Other than an inspection of the vertical concrete cask surface, no annual maintenance of the storage system is required. Collective dose due to design basis off-normal conditions and accident events, such as clearing the blockage of air vents, is accounted for in Chapter 11.0, and is not included in this evaluation.

Routine operations are expected to include:

- Daily electronic measurement of air outlet temperatures. The outlet temperature monitoring station is located away from the cask array. Remote temperature measurement is not assumed to contribute to operator dose.

- A daily security inspection of the fence and equipment surrounding the storage area. The security inspection is assumed to make no significant additional contribution to operator dose.

- Grounds maintenance performed every other week by 1 maintenance technician. Grounds maintenance is assumed to require 0.5 hour.

- Quarterly radiological surveillance. The surveillance consists of a radiological survey comprised of a surface radiation measurement on each cask, the determination and/or verification of general area exposure rates and radiological postings. This surveillance is assumed to require 1 hour and 1 person.

- Annual inspection of the general condition of the casks. This inspection is estimated to require 15 minutes per cask and require 2 technicians.

Calculation of the dose due to annual operation and surveillance requirements is estimated based on a single cask containing design basis fuel and on an ISFSI array of 20 casks that are assumed to be loaded at the rate of 2 casks per year over a ten-year period. Consequently, the casks in the array are assumed to have the cool times as shown in Table 10.3-3. To account for the reduction in source term with cool time, weighting factors are applied to the neutron and gamma radiation spectra as shown in Table 10.3-4.

The annual operation and surveillance requirements result in an estimated annual collective exposure of 26.4 person-mrem for a single PWR cask containing design basis fuel and 17.0 person-mrem for a single design basis BWR cask. The annual operation and surveillance requirements for the assumed single cask and total estimated dose is shown in Table 10.3-5 for the single PWR cask and in Table 10.3-7 for the BWR cask. The annual operation and surveillance requirements for the assumed 20-cask ISFSI are shown in Tables 10.3-6, and 10.3-8 for PWR and BWR configurations, respectively. These tables show an estimated annual collective exposure of 371 person-mrem for the PWR cask configuration and 240 person-mrem for the BWR cask configuration for operation and maintenance of a 20-cask array.

Figure 10.3-1 Typical ISFSI 20 Cask Array Layout

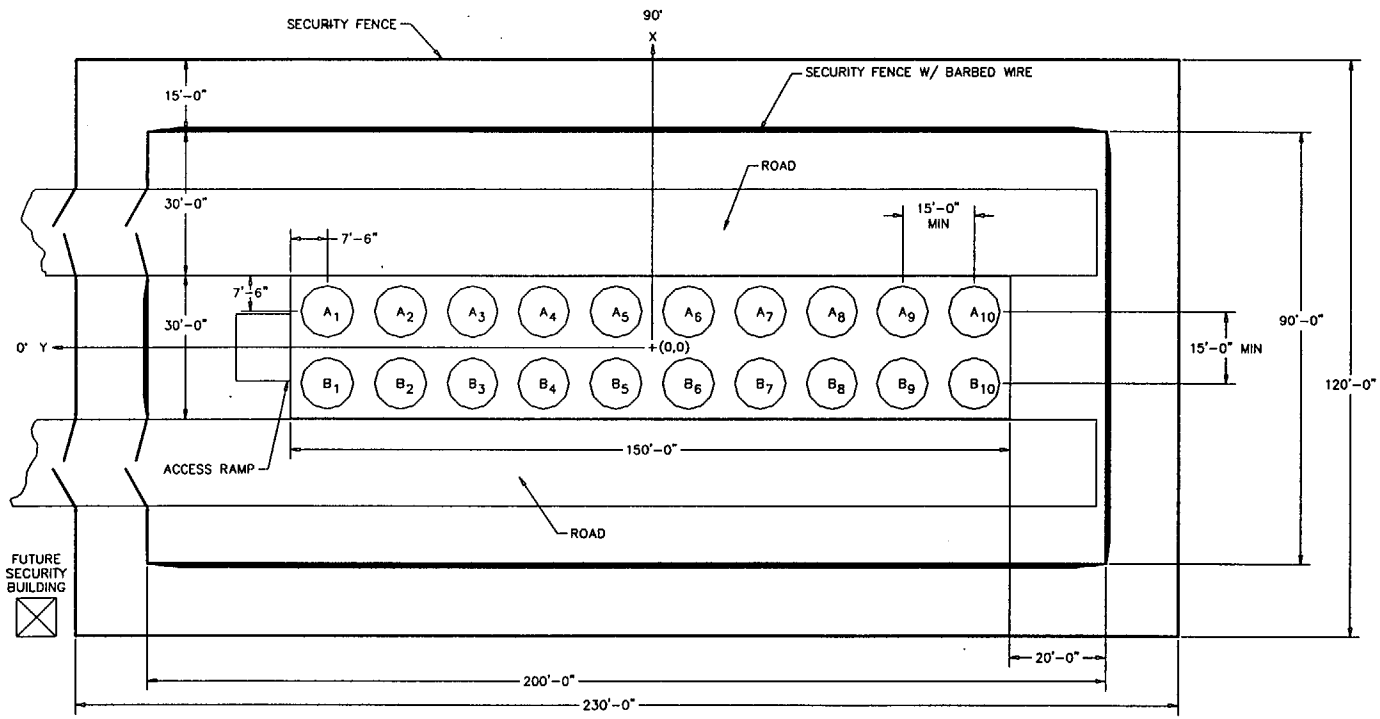


Table 10.3-1 Estimated Person-Mrem Exposure for Operations Using the Standard Transfer Cask

Design Basis Fuel Assemblies Loading and Handling Activity	Estimated Number of Personnel[6]	Exposure Duration (hr)	Average Dose Rate (mrem/hr)		Exposure (person-mrem)	
			PWR	BWR	PWR	BWR
Load Canister[1]	2	9.9/21.9	2.1	2.0	42	88
Move to Decon Area/Prep for Weld	2	0.6	29.1	19.4	33	22
Setup Shield Lid Weld[3]	2	0.5	39.6	25.7	37	24
Welding Operation (Automated)	1	0.3	BDR[2]	BDR[2]	0	0
Weld Inspections[3,4]	1	7.5	10.4	6.6	78	50
Drain/ Vacuum Dry/Backfill and Leak Test[3,5]	2	0.4	30.0	20.4	25	17
Weld and Inspect Port Covers[3,4]	2	2.2	35.1	22.8	151	98
Setup Structural Lid Weld[3]	2	0.3	25.3	15.8	16	10
Welding Operation (Automated)	1	0.3	BDR[2]	BDR[2]	0	0
Weld Inspections[3,4]	1	7.7	6.8	4.0	52	31
Transfer to Vertical Concrete Cask	4	2.8	22.0	13.4	249	152
Position on ISFSI Pad	2	0.8	16.3	11.3	26	18
Total					709	510

1. Assumes 22.5 minutes for the loading of each PWR or BWR fuel assembly with additional time for installation of drain tube and shield lid prior to move to decontamination area.
2. Background Dose Rate (BDR). No exposure is estimated due to the canister contents.
3. Dose rates associated with the presence of a temporary shield on top of the shield lid.
4. Includes root, progressive, and final weld surface inspections.
5. Includes fixturing, connection and monitoring time. Operators not present during routine draining and drying process.
6. Number of personnel shown is a representative number. Personnel vary for the different operation stages, with total exposure divided over a larger number of personnel than the number shown.

Table 10.3-2 Estimated Person-Mrem Exposure for Operations Using the 100-Ton Transfer Cask

Design Basis/100-ton Fuel Assemblies Loading and Handling Activity	Estimated Number of Personnel[6]	Exposure Duration (hr)	Average Dose Rate (mrem/hr)		Exposure (person-mrem)	
			PWR	BWR	PWR	BWR
Load Canister[1]	2	9.9/21.9	3.1	2.4	61	107
Move to Decon Area/Prep for Weld	2	0.6	29.1	19.4	33	22
Setup Shield Lid Weld[3]	2	0.5	39.6	25.7	37	24
Welding Operation (Automated)	1	0.3	BDR[2]	BDR[2]	0	0
Weld Inspections[3,4]	1	7.5	10.4	6.6	78	50
Drain/ Vacuum Dry/Backfill and Leak Test[3,5]	2	0.4	30.0	20.4	25	17
Weld and Inspect Port Covers[3,4]	2	2.2	35.1	22.8	151	98
Setup Structural Lid Weld[3]	2	0.3	25.3	15.8	16	10
Welding Operation (Automated)	1	0.3	BDR[2]	BDR[2]	0	0
Weld Inspections[3,4]	1	7.7	6.8	4.0	52	31
Transfer to Vertical Concrete Cask	4	2.8	37.9	28.6	429	324
Position on ISFSI Pad	2	0.8	16.3	11.3	26	18
Total					908	701

1. Assumes 22.5 minutes for the loading of each PWR or BWR fuel assembly with additional time for installation of drain tube and shield lid prior to move to decontamination area.
2. Background Dose Rate (BDR). No exposure is estimated due to the canister contents.
3. Dose rates associated with the presence of a temporary shield on top of the shield lid.
4. Includes root, progressive, and final weld surface inspections.
5. Includes fixturing, connection and monitoring time. Operators not present during routine draining and drying process.
6. Number of personnel shown is a representative number. Personnel vary for the different operation stages, with total exposure divided over a larger number of personnel than the number shown.

Table 10.3-3 Assumed Contents Cooling Time of the Vertical Concrete Casks Depicted in the Typical ISFSI Array

Cask Number	Cooling Time (yr)		Cask Number	Cooling Time (yr)	
	PWR	BWR		PWR	BWR
A-1	14	14	B-1	14	14
A-2	13	13	B-2	13	13
A-3	12	12	B-3	12	12
A-4	11	11	B-4	11	11
A-5	10	10	B-5	10	10
A-6	9	9	B-6	9	9
A-7	8	8	B-7	8	8
A-8	7	7	B-8	7	7
A-9	6	6	B-9	6	6
A-10	5	5	B-10	5	5

Table 10.3-4 Vertical Concrete Cask Radiation Spectra Weighting Factors

Cask Numbers	Axial Neutron Weighting Factor		Axial Gamma Weighting Factor		Radial Neutron Weighting Factor		Radial Gamma Weighting Factor	
	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
A-1, B-1	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
A-2, B-2	0.96	0.96	0.83	0.84	0.96	0.96	0.83	0.83
A-3, B-3	0.93	0.93	0.72	0.74	0.93	0.93	0.72	0.74
A-4, B-4	0.89	0.89	0.65	0.67	0.89	0.89	0.65	0.67
A-5, B-5	0.86	0.86	0.59	0.62	0.86	0.86	0.59	0.62
A-6, B-6	0.83	0.83	0.55	0.58	0.83	0.83	0.55	0.58
A-7, B-7	0.80	0.80	0.52	0.55	0.80	0.80	0.52	0.55
A-8, B-8	0.77	0.77	0.50	0.52	0.77	0.77	0.50	0.52
A-9, B-9	0.74	0.74	0.47	0.50	0.74	0.74	0.48	0.50
A-10, B-10	0.72	0.72	0.45	0.48	0.72	0.72	0.46	0.48

Table 10.3-5 Estimate of Annual Exposure for the Operation and Surveillance of a Single PWR Cask

Activity	Dose Rate Distance (meters)	Frequency (days)	Time (min)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (Pers-mrem)
Radiological Surveillance	4	4	15	7.40	1	7.4
Annual Inspection						
Operations	1	1	15	25.30	1	6.3
Radiological Support	1	1	3	25.30	1	1.3
Grounds Maintenance	10	26	15	1.76	1	11.4
Total Person-mrem						26.4

Table 10.3-6 Estimate of Annual Exposure for the Operation and Surveillance of a 20-Cask Array of PWR Casks

Activity	Dose Rate Distance (meters)	Frequency (days)	Time (min)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (Pers-mrem)
Radiological Surveillance	4	4	60	5.96	1	23.8
Annual Inspection						
Operations	1	1	15(1)	47.91	1	239.6
Radiological Support	1	1	3(1)	47.91	1	47.9
Grounds Maintenance	10	26	60	2.31	1	60.1
Total Person-mrem for the 20-Cask Array						371.4
Average Total Person-mrem for a Single Cask in the Array						18.6

Table 10.3-7 Estimate of Annual Exposure for the Operation and Surveillance of a Single BWR Cask

Activity	Dose Rate Distance (meters)	Frequency (days)	Time (min)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (mrem)
Radiological Surveillance	4	4	15	4.9	1	4.9
Annual Inspection						
Operations	1	1	15	15.2	1	3.8
Radiological Support	1	1	3	15.2	1	0.8
Grounds Maintenance	10	26	15	1.16	1	7.5
Total Person - mrem						17.0

Table 10.3-8 Estimate of Annual Exposure for the Operation and Surveillance of a 20-Cask Array of BWR Casks

Activity	Dose Rate Distance (meters)	Frequency (days)	Time (min)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (mrem)
Visual Inspection	4	365	1(1)	4.2	1	509.8
Radiological Surveillance	4	4	60	4.2	1	16.8
Annual Inspection						
Operations	1	1	15(1)	29.9	1	149.5
Radiological Support	1	1	3(1)	29.9	1	29.9
Grounds Maintenance	10	26	60	1.7	1	43.2
Total Person - mrem for the 20-Cask Array						240.4
Average Total Person - mrem for a Single Cask in the Array						12.0

(1) Time listed is per cask; it is multiplied by 20 for the cask array.

10.4 Exposure to the Public

The NAC Version 5.0.1 of the SKYSHINE-III code is used to evaluate the placement of the controlled area boundary for a single storage cask containing design basis fuel and for a 20-cask array. For the 20-cask array, the storage casks are assumed to be loaded with design basis fuel at the rate of two casks per year. SKYSHINE III calculates dose rates for user defined detector locations for up to 100 point sources.

Version 5.0.1 of SKYSHINE-III explicitly calculates cask self-shielding based on the storage cask geometry and arrangement of the cask array. A ray tracing technique is utilized. Given the source position on the cask surface and the direction cosines for the source emission, geometric tests are made to see if any adjacent casks are in the path of the emission. If so, the emission history does not contribute to the air scatter dose. Also, given the source position on the cask surface and the direction cosines for the source to detector location, geometric tests are made to see if any adjacent casks are in the source path. If so, the emission position does not contribute to the uncollided dose at the detector location.

The code is benchmarked by modeling a set of Kansas State University ^{60}Co skyshine experiments and by modeling two Kansas State University neutron computational benchmarks. The code compares well with these benchmarks for both neutron and gamma doses versus distance.

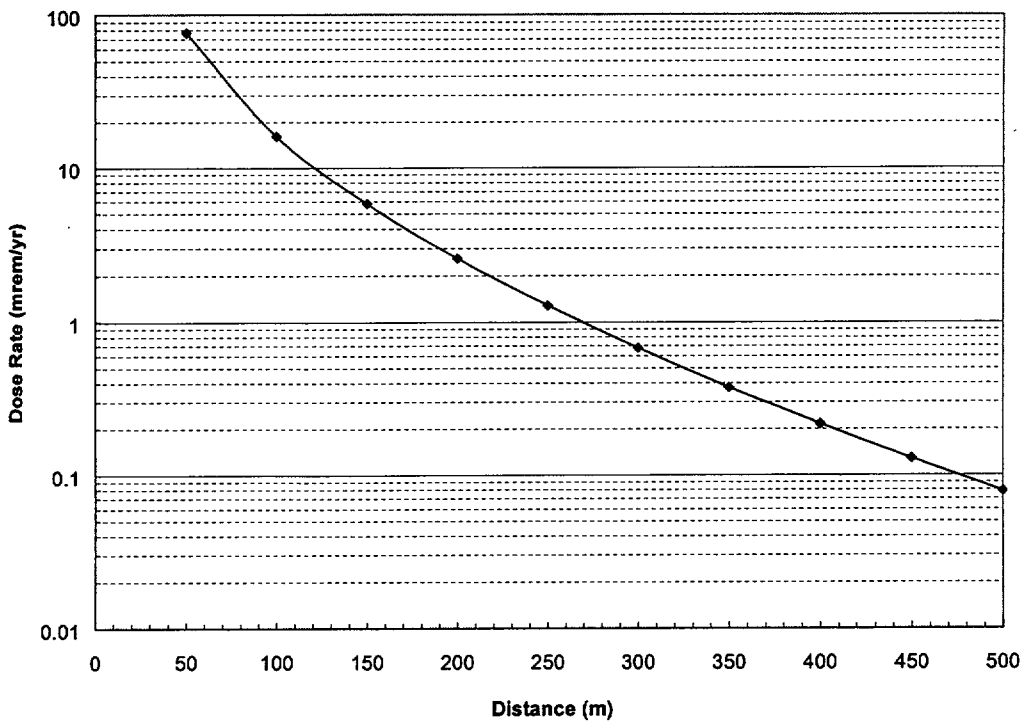
The storage cask array is explicitly modeled in the code, with the source term from each cask represented as top and side surface sources. Surface source emission fluxes are provided from 1-D SAS1 shielding evaluations. The top and side source energy distributions for both neutron and gamma radiation are taken from the design basis cask shielding evaluation. As stated in Section 10.3, the array cask source strengths are multiplied by weighting factors to correct for the differences in cooling times resulting from the assumption of a loading rate of 2 casks per year. The SKYSHINE cask surface fluxes (sources) are adjusted to reflect the higher cask surface fluxes calculated by the SAS4 3-D shielding evaluation. Surface gamma-ray fluxes are also adjusted for dose peaks associate with fuel assembly end-fitting hardware and radiation streaming through the cask vents and canister-to-cask annulus. The 2x10 ISFSI storage cask array layout is presented in Figure 10.3-1. For this analysis the cask-to-cask pitch is conservatively taken at 16 feet, as opposed to the minimum 15 feet, to minimize cask-to-cask shadowing. These results are conservative for the minimum 15-foot cask center-to-center-spacing specified in Section 8.1.3.

Exposures are determined at distances ranging from 50 to 500 meters surrounding a single PWR and BWR storage cask containing design basis fuel. The results are presented graphically in Figures 10.4-1 and 10.4-2, for the PWR or BWR single cask, respectively. The storage casks in the 2 x 10 array are assumed to be loaded at the rate of 2 per year with design basis PWR and BWR spent fuel, with credit taken for the cool time that occurs during the 10-year period that the ISFSI array is completed. For both the single cask and 2 x 10 array calculations, the controlled area boundary is based on the 25 mrem/year limit. Occupancy at the controlled area boundary is assumed at 2,080 hours per year. While higher occupancy may be required at certain sites, the increased exposure time will likely be offset by increased cool time or decreased burnup.

Table 10.4-1 presents a summary of the dose rates versus distance for a single PWR and BWR storage cask containing design basis fuel. Linear interpolation of these results shows that minimum distances from a single cask to the site boundary of 93 meters and 84 meters for the design basis PWR and BWR fuels, respectively, are required for compliance with the requirements of 10 CFR 72.104(a), i.e., a dose rate of 25 mrem/year. Table 10.4-2 results show that a minimum site boundary of ≈195 meters is required for a 2x10 PWR cask array to meet the 10 CFR 72.104(a) 25 mrem/year requirement. The 2x10 BWR cask array requires a minimum site boundary of ≈186 meters to meet 10 CFR 72.104(a).

The distances used in Tables 10.4-1 and 10.4-2 are measured from the center of the 2x10 cask array along a line perpendicular to the center of the 10-cask face of the array.

Figure 10.4-1 SKYSHINE Exposures from a Single Cask Containing Design Basis PWR Fuel

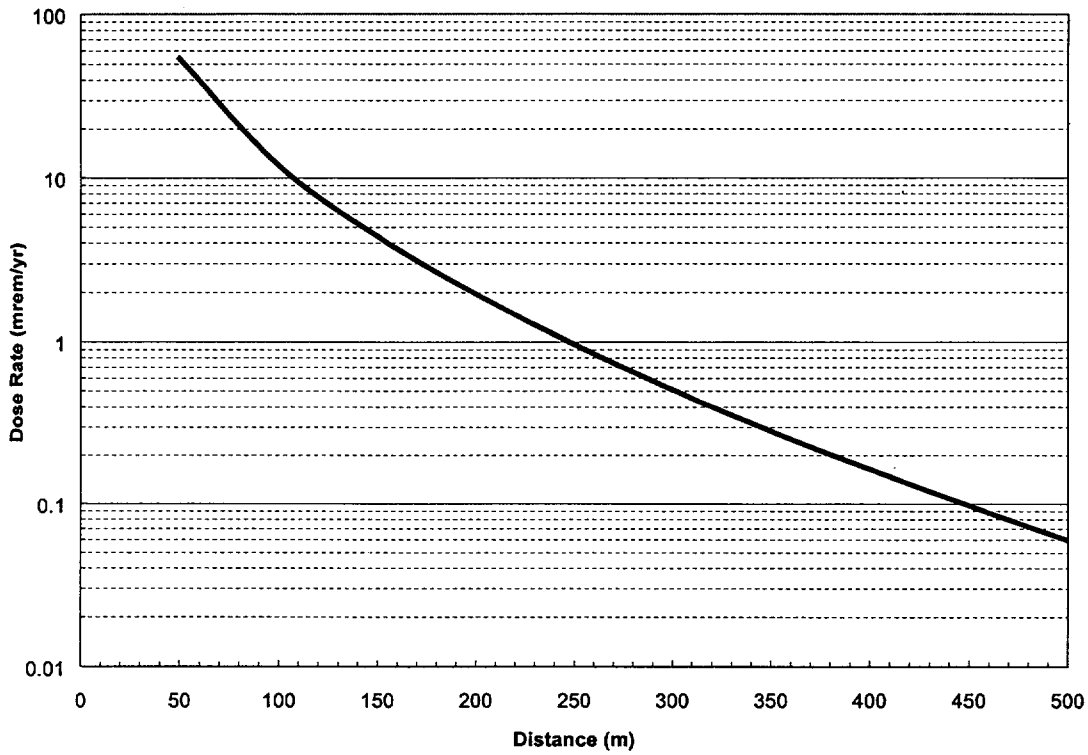


Distance from Center of Cask(m)	Dose Rate (mrem/year)			
	Gamma Dose	Neutron Dose	N-Gamma Dose	Total Dose
50	7.28E+01	3.85E+00	7.93E-04	77
100	1.47E+01	1.34E+00	8.07E-04	16
150	5.25E+00	5.56E-01	8.14E-04	5.8
200	2.32E+00	2.54E-01	7.86E-04	2.6
250	1.15E+00	1.24E-01	7.26E-04	1.3
300	6.12E-01	6.29E-02	6.43E-04	0.68
350	3.40E-01	3.34E-02	5.50E-04	0.37
400	1.97E-01	1.83E-02	4.58E-04	0.22
450	1.18E-01	1.03E-02	3.71E-04	0.13
500	7.19E-02	5.97E-03	2.95E-04	0.08

General Notes:

1. Based on a 2,080-hour exposure.
2. Axial gamma and radial neutron doses are negligible.

Figure 10.4-2 SKYSHINE Exposures from a Single Cask Containing Design Basis BWR Fuel



Distance from Center of Cask(m)	Dose Rate (mrem/year)			
	Gamma Dose	Neutron Dose	N-Gamma Dose	Total Dose
50	4.81E+01	5.80E+00	1.47E-03	54
100	9.86E+00	2.02E+00	1.27E-03	12
150	3.53E+00	8.40E-01	1.25E-03	4.4
200	1.57E+00	3.84E-01	1.20E-03	2.0
250	7.78E-01	1.86E-01	1.10E-03	0.97
300	4.15E-01	9.49E-02	9.78E-04	0.51
350	2.33E-01	5.03E-02	8.37E-04	0.28
400	1.35E-01	2.76E-02	6.96E-04	0.16
450	8.12E-02	1.56E-02	5.64E-04	0.10
500	5.00E-02	9.00E-03	4.48E-04	0.06

General Notes:

1. Based on a 2,080-hour exposure.
2. Axial gamma and radial neutron doses are negligible.

Table 10.4-1 Dose Versus Distance for a Single Cask Containing Design Basis PWR or BWR Fuel

Distance from Center of Cask (m)	PWR Cask Total Dose Rate (mrem/y)[1]	BWR Cask Total Dose Rate (mrem/y)[1]
50	77	54
100	16	12
150	5.8	4.4
200	2.6	2.0
250	1.3	0.97
300	0.68	0.51
350	0.37	0.28
400	0.22	0.16
450	0.13	0.10
500	0.08	0.06

1. 2080-hour exposure.

Table 10.4-2 Annual Exposures from a 2 x 10 Cask Array Containing Design Basis PWR or BWR Fuel

Distance from Center of Array (m)	PWR Cask Total Dose Rate (mrem/y)[1]	BWR Cask Total Dose Rate (mrem/y)[1]
50	600	466
100	135	111
150	49	41
200	22	19
250	11	9.2
300	5.8	4.9
350	3.2	2.7
400	1.9	1.5
450	1.1	0.90
500	0.67	0.55

1. 2080-hour exposure.

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10.5 Radiation Protection Evaluation for Site Specific Spent Fuel

This section presents the radiation protection evaluation of fuel assemblies or configurations, which are unique to specific reactor sites. These site specific configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel that is classified as damaged.

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

10.5.1 Radiation Protection Evaluation for Maine Yankee Site Specific Spent Fuel

The shielding evaluation of Maine Yankee site specific fuel characteristics is presented in Section 5.6.1.1. In the shielding evaluation, the specific fuel assembly and non-fuel hardware sources are shown to be bounded by the design basis fuel assembly characteristics. To ensure that the Maine Yankee contents are bounded by the design basis fuel, specific evaluations are performed and minimum cooling time and loading restrictions are established.

Because the dose rates from the Maine Yankee contents are bounded by the design basis fuel, the radiological evaluations performed for the design basis fuel in Sections 10.3 and 10.4 are also bounding. Therefore, detailed radiological evaluations for the Maine Yankee site specific fuel configurations are not required and the evaluated on-site and off-site doses presented in Sections 10.3 and 10.4 can be used in site planning considerations.

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10.6 References

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11.0 ACCIDENT ANALYSES

The analyses of the off-normal and accident design events, including those identified by ANSI/ANS 57.9-1992 [1], are presented in this chapter. Section 11.1 describes the off-normal events that could occur during the use of the Universal Storage System, possibly as often as once per calendar year. Section 11.2 addresses very low probability events that might occur once during the lifetime of the ISFSI or hypothetical events that are postulated because their consequences may result in the maximum potential impact on the surrounding environment.

The Universal Storage System includes Transportable Storage Canisters and Vertical Concrete Casks of five different lengths to accommodate three classes of PWR fuel or two classes of BWR fuel. In the analyses of this chapter, the bounding concrete cask parameters (such as weight and center of gravity) are conservatively used, as appropriate, to determine the cask's capability to withstand the effects of the analyzed events.

The load conditions imposed on the canisters and the baskets by the design basis normal, off-normal, and accident conditions of storage are less rigorous than those imposed by the transport conditions–including the 30-foot drop impacts and the fire accident (10 CFR 71) [2]. Consequently, the evaluation of the canisters and the baskets for transport conditions bounds those for storage conditions evaluated in this chapter. A complete evaluation of the normal and accident transport condition loading on the PWR and BWR canisters and the baskets is presented in the Safety Analysis Report for the Universal Transport Cask. [3]

This chapter demonstrates that the Universal Storage System satisfies the requirements of 10 CFR 72.24 and 10 CFR 72.122 [4] for off-normal and accident conditions. These analyses are based on conservative assumptions to ensure that the consequences of off-normal conditions and accident events are bounded by the reported results. If required for a site specific application, a more detailed evaluation could be used to extend the limits defined by the events evaluated in this chapter.

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11.1 Off-Normal Events

This section evaluates postulated events that might occur once during any calendar year of operations. The actual occurrence of any of these events is, therefore, infrequent.

11.1.1 Severe Ambient Temperature Conditions (106°F and -40°F)

This section evaluates the Universal Storage System for the steady state effects of severe ambient temperature conditions (106°F and -40°F).

11.1.1.1 Cause of Severe Ambient Temperature Event

Large geographical areas of the United States are subjected to sustained summer temperatures in the 90°F to 100°F range and winter temperatures that are significantly below zero. To bound the expected steady state temperatures of the canister and storage cask during these severe ambient conditions, analyses are performed to calculate the steady state storage cask, canister, and fuel cladding temperatures for a 106°F ambient temperature and solar loads (see Table 4.1-1). Similarly, winter weather analyses are performed for a -40°F ambient temperature with no solar load. Neither ambient temperature condition is expected to last more than several days.

11.1.1.2 Detection of Severe Ambient Temperature Event

Detection of off-normal ambient temperatures would occur during the daily measurement of ambient temperature and storage cask outlet air temperature.

11.1.1.3 Analysis of Severe Ambient Temperature Event

Off-normal temperature conditions are evaluated by using the thermal models described in Section 4.4.1. The design basis heat load of 23.0 kW is used in the evaluation of PWR and BWR fuels. The concrete temperatures are determined using the two-dimensional axisymmetric air flow and concrete cask models (Section 4.4.1.1) and the canister, basket and fuel cladding temperatures are determined using the three-dimensional canister models (Section 4.4.1.2). A steady state condition is considered in all analyses. The temperature profiles for the concrete cask and for the air flow associated with a 106°F ambient condition are shown in Figure 11.1.1-1 and Figure 11.1.1-2, respectively. Temperature profiles for the -40°F ambient temperature condition for the PWR fuel

are shown in Figure 11.1.1-3 and Figure 11.1.1-4. Temperature profiles for the BWR cask are similar.

The principal component temperatures for each of the ambient temperature conditions discussed above are summarized in the following table along with the allowable temperatures. As the table shows, the component temperatures are within the allowable values for the off-normal ambient conditions.

Component	106°F Ambient Max Temp. (°F)		-40°F Ambient Max Temp. (°F)		Allowable Temp. (°F)	
	PWR	BWR	PWR	BWR	PWR	BWR
Fuel Cladding	670	661	557	533	1058	1058
Support Disks	624	637	499	501	800	700
Heat Transfer Disks	622	636	498	500	750	750
Canister Shell	401	422	254	275	800	800
Concrete	228	231	17	20	350	350

The thermal stress evaluations for the concrete cask for these off-normal conditions are bounded by those for the accident condition of "Maximum Anticipated Heat Load (133°F ambient temperature)" as presented in Section 11.2.7. Thermal stress analyses for the canister and basket components are performed using the ANSYS finite element models as described in Section 3.4.4. Evaluations of the thermal stresses combined with the stresses due to other off-normal loads (e.g., canister internal pressure and handling) are shown in Section 11.1.3.

There are no adverse consequences for these off-normal conditions. The maximum component temperatures are within the allowable temperature values.

11.1.1.4 Corrective Actions

No corrective actions are required for this off-normal condition.

11.1.1.5 Radiological Impact

There is no radiological impact due to this off-normal event.

Figure 11.1.1-1 Concrete Temperature (°F) for Off-Normal Storage Condition 106°F Ambient Temperature (PWR Fuel)

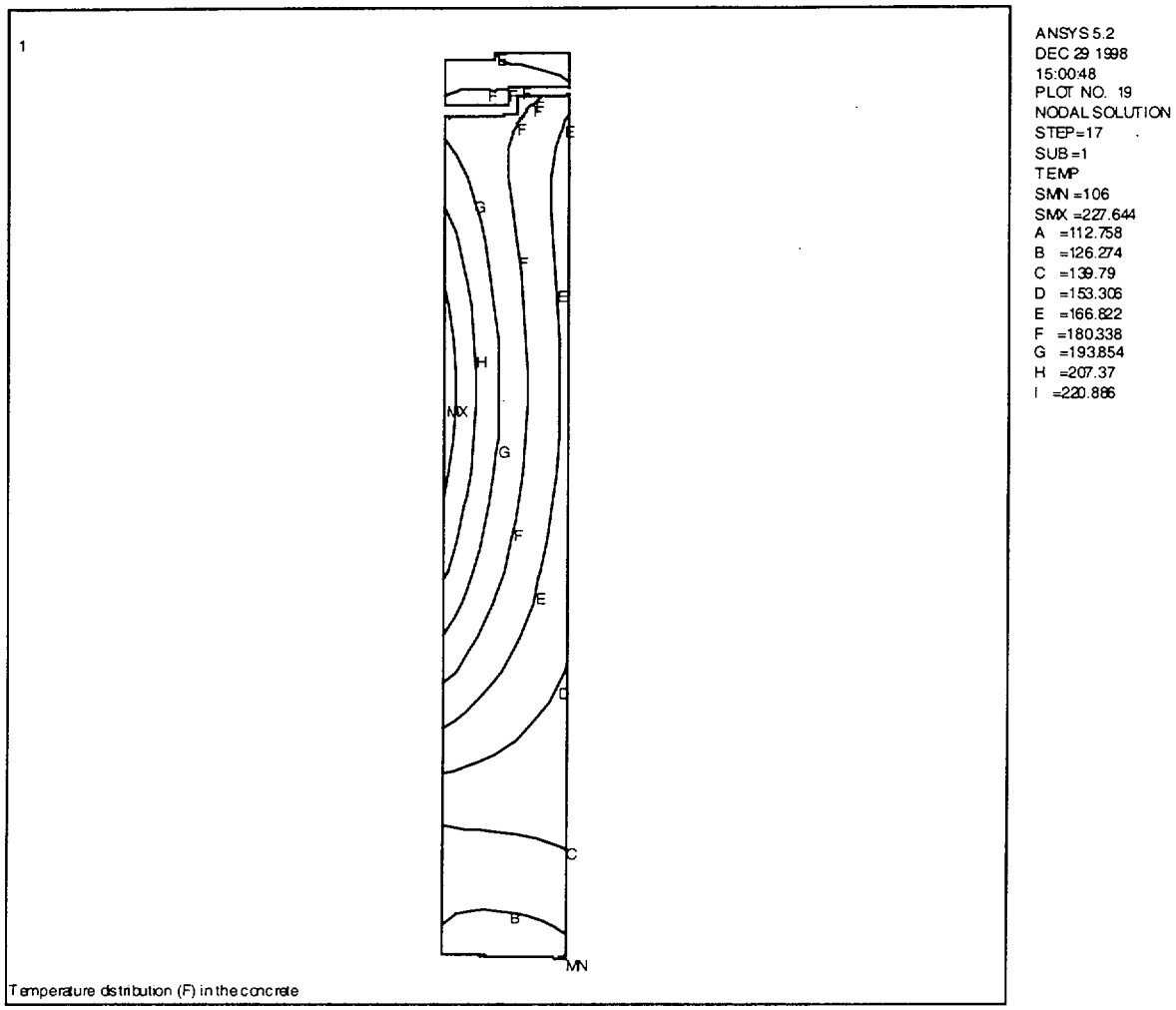


Figure 11.1.1-2 Vertical Concrete Cask Air Temperature (°F) Profile for Off-Normal Storage Condition 106°F Ambient Temperature (PWR Fuel)

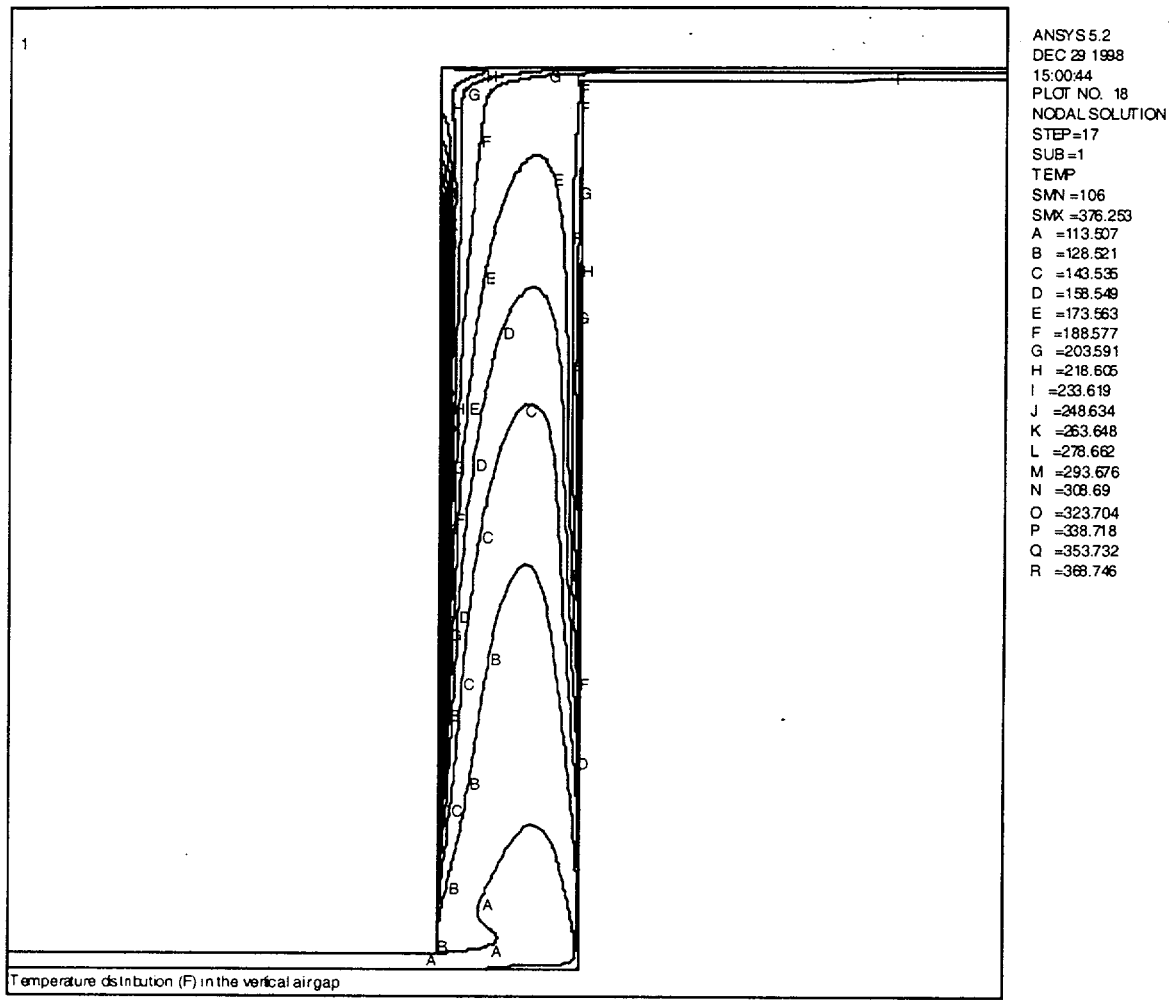


Figure 11.1.1-3 Concrete Temperature (°F) for Off-Normal Storage Condition -40°F Ambient Temperature (PWR Fuel)

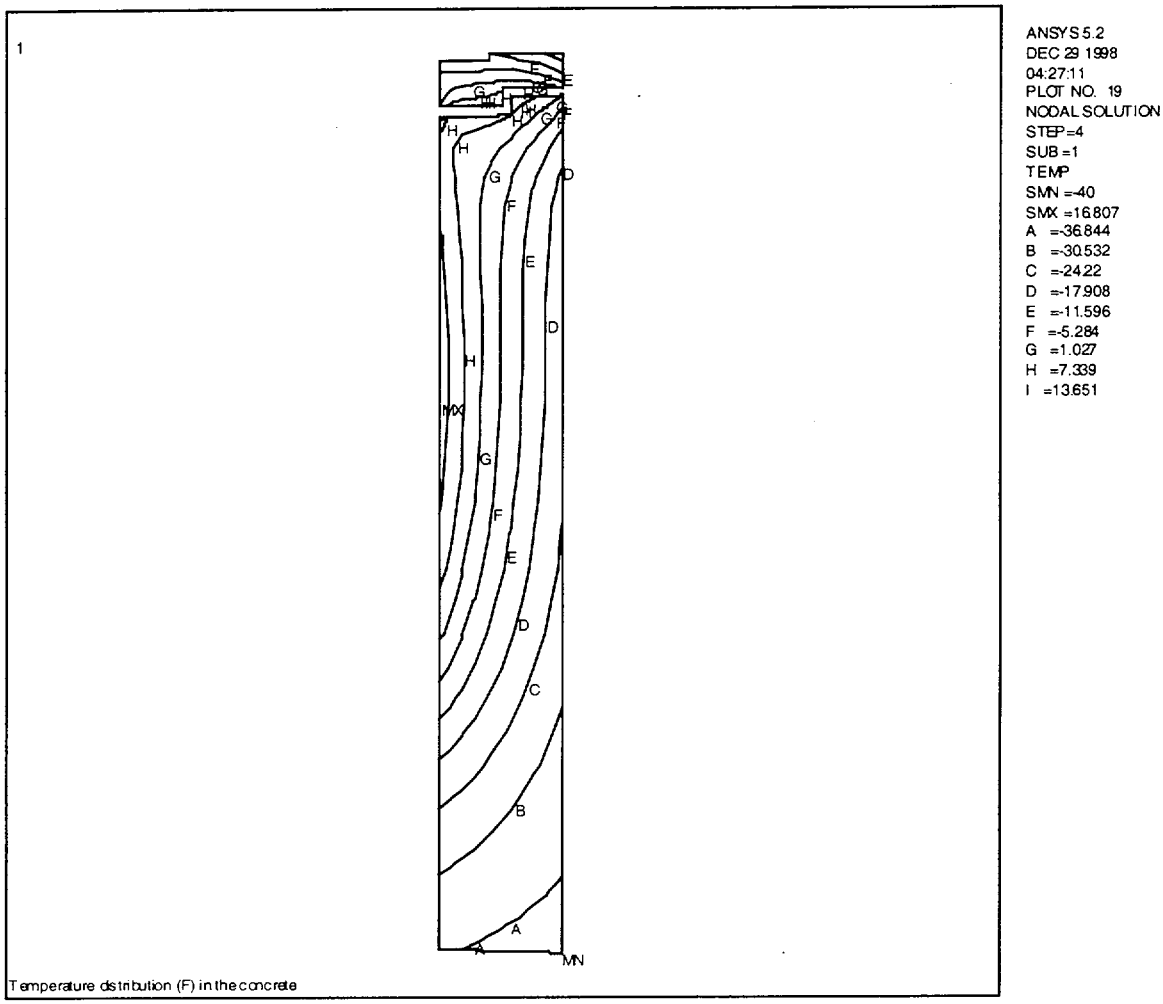
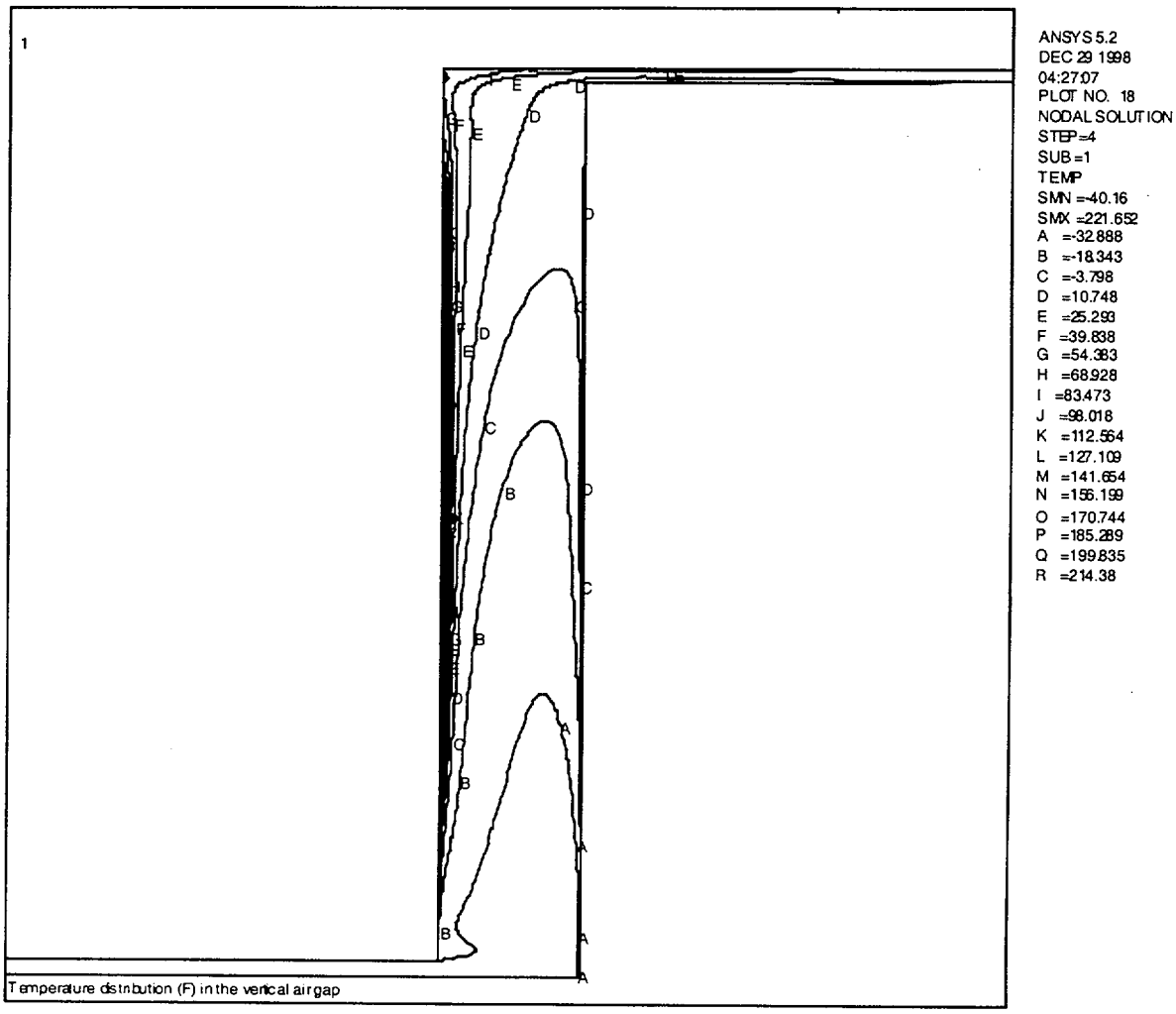


Figure 11.1.1-4 Vertical Concrete Cask Air Temperature (°F) Profile for Off-Normal Storage Condition -40°F Ambient Temperature (PWR Fuel)



11.1.2 Blockage of Half of the Air Inlets

This section evaluates the Universal Storage System for the steady state effects of a blockage of one-half of the air inlets at the normal ambient temperature (76°F).

11.1.2.1 Cause of the Blockage Event

Although unlikely, blockage of half of the air inlets may occur due to blowing debris, snow, intrusion of a burrowing animal, etc. The screens over the inlets are expected to minimize any blockage of the inlet channels.

11.1.2.2 Detection of the Blockage Event

This event would be detected visually by the persons inspecting the air inlets and gathering outlet air temperature data on a daily basis. It could also be detected by security forces, or other operations personnel, engaged in other routine activities such as fence inspection, or grounds maintenance.

11.1.2.3 Analysis of the Blockage Event

Using the same methods and the same thermal models described in Section 11.1.1 for the off-normal conditions of severe ambient temperatures, thermal evaluations are performed for the concrete cask and the canister and its contents for this off-normal condition. The boundary condition of the two-dimensional axisymmetric air flow and concrete cask model is modified to allow only half of the air flow into the air inlet to simulate the half inlets blocked condition. The calculated maximum component temperatures due to this off-normal condition are compared to the allowable component temperatures. Table 11.1.2-1 summarizes the component temperatures for off-normal conditions. As the table demonstrates, the calculated temperatures are shown to be below the component allowable temperatures.

The thermal stress evaluations for the concrete cask for this off-normal condition are bounded by those for the accident condition of "Maximum Anticipated Heat Load (133°F ambient temperature)" as presented in Section 11.2.7. Thermal stress analyses for the canister and basket components are performed using the ANSYS finite element models described in Section 3.4.4. Evaluations of the thermal stresses combined with stresses due to other off-normal loads (e.g., canister internal pressure and handling) are shown in Section 11.1.3.

11.1.2.4 Corrective Actions

The debris blocking the affected air inlets must be manually removed. The nature of the debris may indicate that other actions are required to prevent recurrence of the blockage.

11.1.2.5 Radiological Impact

There are no significant radiological consequences for this event. Personnel will be subject to an estimated maximum contact dose rate of 66 mrem/hr when clearing the PWR cask inlets. If it is assumed that a worker kneeling with his hands on the inlets would require 15 minutes to clear the inlets, the estimated maximum extremity dose is 17 mrem. For clearing the BWR cask inlets, the maximum contact dose rate and the maximum extremity dose are estimated to be 51 mrem/hr and 13 mrem, respectively. The whole body dose in both PWR and BWR cases will be significantly less.

Table 11.1.2-1 Component Temperatures (°F) for Half of Inlets Blocked Off-Normal Event

	Half of Inlets Blocked Max Temperature (°F)		**Allowable Temperature (°F)**	
Component	**PWR**	**BWR**	**PWR**	**BWR**
Fuel Cladding	647	637	1058	1058
Support Disks	599	611	800	700
Heat Transfer Disks	597	610	750	750
Canister Shell	371	391	800	800
Concrete	191	195	350	350

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11.1.3 Off-Normal Canister Handling Load

This section evaluates the consequence of loads on the Transportable Storage Canister during the installation of the canister in the Vertical Concrete Cask, or removal of the canister from the concrete cask or from the transfer cask.

11.1.3.1 Cause of Off-Normal Canister Handling Load Event

Unintended loads could be applied to the canister due to misalignment or faulty crane operation, or inattention of the operators.

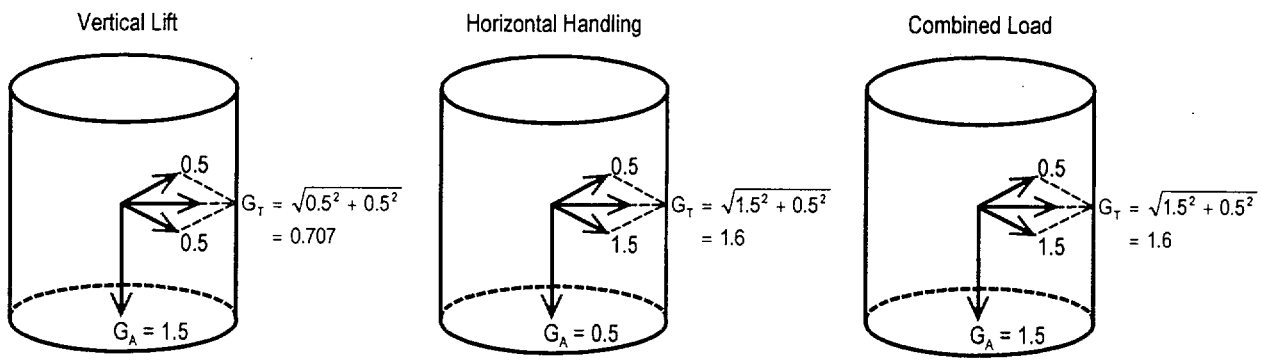
11.1.3.2 Detection of Off-Normal Canister Handling Load Event

The event can be detected visually during the handling of the canister, or banging or scraping noise associated with the canister movement. The event is expected to be obvious to the operators at the time of occurrence.

11.1.3.3 Analysis of Off-Normal Canister Handling Load Event

The canister structural analysis, including lifting loads, is performed by using an ANSYS finite element model as shown in Figure 11.1.3.1-1. The model is based on the canister model presented in Section 3.4.4.1 with the elements fuel basket (support disks and top and bottom weldment disks) added. The disks are modeled with SHELL63 elements. These elements are included to transfer loads from the basket to the canister shell for loads in the canister transverse direction. The interface between the disks and the canister shell is simulated by CONTAC52 elements. For the transverse loads, uniform pressure loads representing the weight (including appropriate g-loading) of the fuel assemblies, fuel tubes, heat transfer disks, tie-rods, spacers, washers, and nuts are applied to the slots of the support/weldment disks. For loads in the canister axial direction, interaction between the fuel basket and the canister is modeled by applying a uniform pressure representing the weight of the fuel assemblies and basket (including appropriate g-loading) to the canister bottom plate. The model is used to evaluate the canisters for both PWR and BWR fuel types by modeling the shortest canister (Class 1 PWR) with the heaviest fuel/fuel basket weight (Class 5 BWR).

The normal handling loads are defined as 1g of dead weight plus a 10% dynamic load factor (a total of 1.1g in the axial direction). The off-normal handling loads are defined as 0.5g in all directions plus 1g of dead weight. Note the canister may be handled in the vertical or horizontal positions. To bound horizontal and vertical handling cases, combined accelerations are applied to the ANSYS model as shown in the following figures.



The boundary conditions (restraints) for the canister model are the same as those described in Section 3.4.4.1.4 for the normal handling condition.

The resulting maximum canister stresses for off-normal handling loads are summarized in Tables 11.1.3-1 and 11.1.3-2 for primary membrane and primary membrane plus bending stresses, respectively.

The resulting maximum canister stresses for combined off-normal handling, maximum off-normal internal pressure (15 psig), and thermal stress loads are summarized in Tables 11.1.3-3, 11.1.3-4, and 11.1.3-5 for primary membrane, primary membrane plus bending, and primary plus secondary stresses, respectively.

The sectional stresses shown in Tables 11.1.3-1 through 11.1.3-5 at 16 axial locations are obtained for each angular division of the model (a total of 19 angular locations for each axial location). The locations of the stress sections are shown in Figure 3.4.4.1-4.

To determine the structural adequacy of the PWR and BWR fuel basket support disks and weldments for off-normal conditions, a structural analysis is performed by using ANSYS to evaluate off-normal handling loads. To simulate off-normal loading conditions, an inertial load of 1.5g is applied to the support disk and the weldments in the axial (canister axial) direction and 0.5g in two orthogonal disk in plane directions (0.7071g resultant), for the governing case (canister handled in the vertical orientation.

Stresses in the support disks and weldments are calculated by applying the off-normal loads to the ANSYS models described in Sections 3.4.4.1.8 and 3.4.4.1.9. An additional in-plane displacement constraint is applied to each model at one node (conservative) at the periphery of the disk or the weldment plate to simulate the side restraint of the canister shell for the lateral load (0.7071g). To evaluate the most critical regions of the support disks, a series of cross sections is considered. The locations of these sections on the PWR and BWR support disks are shown in Figures 3.4.4.1-7, 3.4.4.1-8, and Figures 3.4.4.1-13 through 3.4.4.1-16 (Note: stress allowables for support disks are taken at 800°F). The stress evaluation for the support disk and weldment is performed according to ASME Code, Section III, Subsection NG. For off-normal conditions, Level C allowable stresses are used: the allowable stress is 1.2 S_m or S_y, 1.8 S_m or 1.5S_y, and 3.0 S_m for the P_m, P_m+P_b, and P_m+P_b+Q stress categories, respectively. The stress evaluation results are presented in Tables 11.1.3-6 through 11.1.3-8 for the PWR support disks and in Tables 11.1.3-9 through 11.1.3-11 for the BWR support disks. The tables list the 40 sections with the highest P_m, P_m+P_b, and P_m+P_b+Q stress intensities. All of the support disk sections have large margins of safety. The stress results for the PWR and BWR weldments are shown in Table 11.1.3-12.

The canisters and fuel baskets maintain positive margins of safety for the off-normal handling condition. There is no deterioration of canister or fuel basket performance. The Universal Storage System is in compliance with all applicable regulatory criteria.

11.1.3.4 Corrective Actions

Operations should be halted until the cause of the misalignment, interference or faulty operation is identified and corrected. Since the radiation level of the canister sides and bottom is high, extreme caution should be exercised if inspection of these surfaces is required.

11.1.3.5 Radiological Impact

There are no radiological consequences associated with this off-normal event.

Figure 11.1.3.1-1 Canister and Basket Finite Element Model

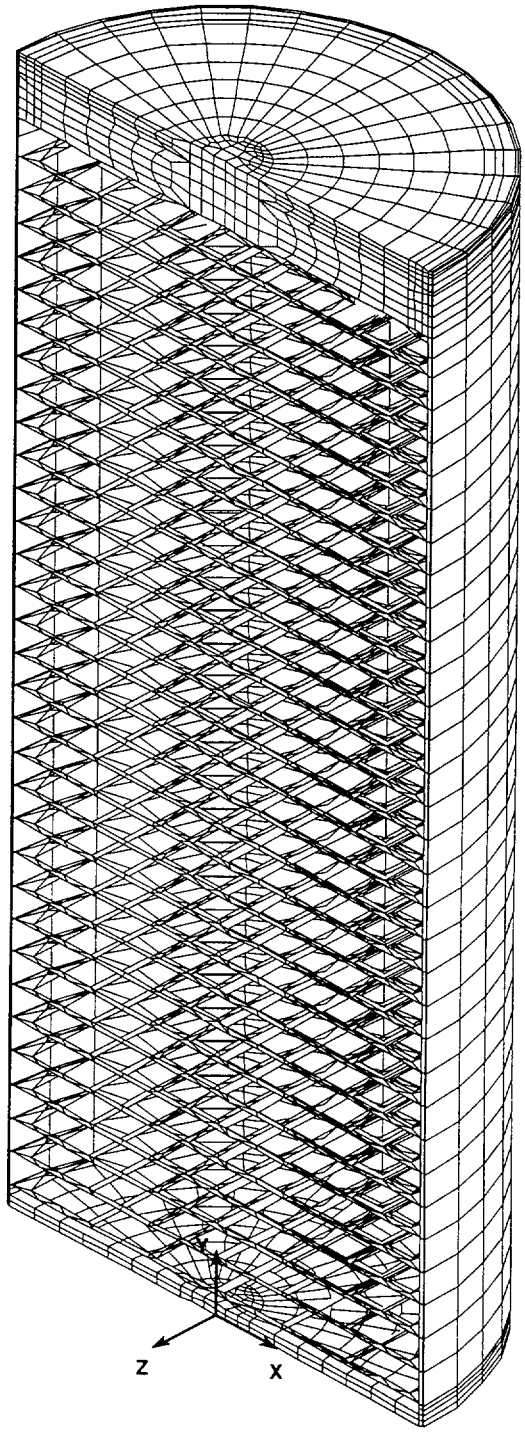


Table 11.1.3-1 Canister Off-Normal Handling (No Internal Pressure) Primary Membrane (P_m) Stresses (ksi)

Section No.(1)	Angle(1) (degrees)	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	0	-1.47	2.90	0.41	0.30	-0.06	-0.20	4.43
2	0	2.40	-1.16	-2.40	-0.09	-0.17	-0.41	4.89
3	0	-0.14	0.82	0.00	-0.01	-0.01	0.02	0.97
4	0	-0.20	0.76	0.00	0.00	-0.01	0.01	0.96
5	0	-0.22	0.78	0.00	0.00	0.00	0.01	1.00
6	0	-0.25	0.83	-0.01	0.00	0.00	0.00	1.08
7	0	-0.27	0.99	-0.05	0.00	0.01	-0.01	1.26
8	0	-0.03	1.94	-0.09	-0.02	0.20	0.03	2.07
9	0	0.39	3.46	0.76	0.20	0.38	0.15	3.19
10	0	-0.32	4.34	0.62	-0.02	0.53	0.17	4.76
11	0	0.33	3.60	1.50	-1.32	0.55	-0.10	4.32
12	120	0.55	3.15	0.08	-0.17	-0.16	-0.35	3.29
13	0	-4.88	0.02	0.69	-2.06	0.14	-0.09	6.53
14	0	0.31	-0.02	0.42	-0.04	-0.17	0.00	0.56
15	170	-0.05	0.00	-0.04	0.00	0.00	0.00	0.04
16	0	-0.07	0.00	0.04	0.00	0.01	0.00	0.11

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

Table 11.1.3-2 Canister Off-Normal Handling (No Internal Pressure) Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No.(1)	Angle(1) (degrees)	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity
1	0	-8.34	-2.16	-0.38	0.06	-0.14	-0.25	7.98
2	0	0.78	-12.39	-6.16	-0.82	-0.41	-0.67	13.36
3	0	-0.12	0.85	0.15	-0.02	0.00	0.03	0.98
4	0	-0.21	0.77	0.05	0.00	-0.01	0.01	0.98
5	0	-0.24	0.82	0.15	0.00	-0.01	0.02	1.06
6	0	-0.27	0.90	0.24	0.00	-0.01	0.03	1.17
7	0	-0.30	1.07	0.27	0.01	0.02	0.02	1.37
8	0	0.01	1.97	-0.15	-0.02	0.18	0.02	2.15
9	0	0.54	5.13	1.12	-0.03	0.46	0.16	4.69
10	0	-0.58	4.18	0.70	0.23	0.37	0.23	4.86
11	0	-0.85	2.51	1.29	-2.13	0.39	-0.17	5.51
12	120	0.68	4.30	-0.06	-0.22	-0.21	-0.56	4.70
13	0	-9.91	-1.78	-0.12	-1.63	-0.04	0.03	10.11
14	180	8.86	0.24	8.88	-0.04	-0.17	0.01	8.65
15	0	-0.25	-0.01	-0.23	0.00	0.00	0.00	0.25
16	0	-1.10	-0.03	-0.97	0.00	0.01	0.01	1.07

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

Table 11.1.3-3 Canister Off-Normal Handling plus Normal/Off-Normal Internal Pressure (15 psig) Primary Membrane (P_m) Stresses (ksi)

Section No.(1)	Angle(1) (degrees)	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity	Stress Allowable(2)	Margin of Safety
1	0	-1.45	4.36	0.94	0.11	-0.04	-0.17	5.83	21.04	2.61
2	0	3.38	-2.09	-3.64	-0.30	-0.16	-0.57	7.14	21.03	1.94
3	0	-0.14	1.28	0.79	0.02	-0.02	0.08	1.43	19.61	12.75
4	0	-0.20	1.18	0.79	0.00	-0.01	0.08	1.38	18.40	12.29
5	0	-0.23	1.18	0.78	0.00	-0.01	0.08	1.42	17.41	11.29
6	0	-0.25	1.24	0.77	0.00	0.00	0.07	1.50	18.26	11.21
7	0	-0.29	1.39	0.73	0.00	0.01	0.06	1.68	19.38	10.54
8	0	-0.05	2.41	0.31	-0.02	0.24	0.08	2.51	20.60	7.22
9	0	0.09	3.69	0.89	0.24	0.38	0.13	3.70	20.94	4.66
10	0	-0.56	4.45	0.75	0.01	0.52	0.17	5.11	20.95	3.10
11	0	-0.24	3.29	1.33	-1.18	0.53	-0.08	4.36	21.06	3.83
12	0	-0.36	2.90	0.62	0.24	0.30	0.17	3.35	20.94	5.24
13	0	-4.45	0.09	0.89	-1.96	0.14	-0.05	6.19	21.07	2.40
14	0	0.55	-0.03	0.67	-0.07	-0.27	0.00	0.90	20.04	21.38
15	0	-0.07	-0.01	-0.06	0.00	0.00	0.00	0.06	20.96	348.83
16	0	-0.04	0.00	0.06	0.00	0.01	0.00	0.11	20.96	195.09

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

(2) ASME Service Level C is used for material allowable stress.

Table 11.1.3-4 Canister Off-Normal Handling plus Normal/Off-Normal Internal Pressure (15 psig) Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No.(1)	Angle(1) (degrees)	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity	Stress Allowable(2)	Margin of Safety
1	0	6.65	11.53	1.34	0.53	0.04	-0.20	10.26	31.23	2.04
2	0	1.23	-19.14	-9.27	-1.30	-0.39	-0.96	20.64	31.21	0.51
3	0	-0.13	1.36	1.03	0.02	0.00	0.10	1.49	27.78	17.63
4	0	-0.21	1.18	0.81	0.00	-0.02	0.08	1.40	25.48	17.20
5	0	-0.25	1.23	0.95	0.00	-0.01	0.09	1.48	24.10	15.25
6	0	-0.28	1.32	1.07	0.00	-0.01	0.10	1.61	25.29	14.75
7	0	-0.32	1.50	1.14	0.01	0.01	0.09	1.82	27.24	13.93
8	0	-0.07	2.48	0.45	-0.02	0.29	0.09	2.61	30.15	10.57
9	0	0.03	5.40	1.22	0.14	0.48	0.14	5.44	30.97	4.69
10	0	-0.24	4.83	0.77	-0.15	0.69	0.13	5.21	31.02	4.95
11	0	-1.09	2.73	1.36	-2.08	0.40	-0.14	5.73	31.28	4.46
12	0	-0.95	3.75	0.64	0.16	0.34	0.22	4.77	30.99	5.49
13	0	-9.02	-1.74	0.16	-1.55	-0.04	0.09	9.51	31.29	2.29
14	130	13.93	0.38	13.96	-0.07	-0.27	0.01	13.60	28.81	1.12
15	0	-0.20	-0.01	-0.22	0.00	0.00	0.00	0.20	31.04	152.95
16	0	0.96	0.03	1.07	0.00	0.01	-0.01	1.05	31.04	28.67

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

(2) ASME Service Level C is used for material allowable stress.

Table 11.1.3-5 Canister Off-Normal Handling plus Normal/Off-Normal Internal Pressure (15 psig) Primary plus Secondary (P + Q) Stresses (ksi)

Section No.(1)	Angle(1) (degrees)	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity	Stress Allowable(2)	Margin of Safety
1	60	3.46	14.31	4.54	-0.08	-0.26	-0.65	11.17	50.10	3.49
2	50	-4.61	-23.19	-2.69	-1.05	1.10	-5.11	24.92	50.10	1.01
3	0	-0.47	1.15	3.82	0.02	-0.43	0.32	4.41	49.02	10.12
4	0	-1.15	0.70	8.50	-0.04	-0.55	0.78	9.81	46.00	3.69
5	0	-2.65	-9.75	-17.93	0.03	0.04	-3.38	16.71	43.52	1.60
6	0	-1.40	1.34	9.43	0.04	0.55	0.83	10.98	45.66	3.16
7	0	-0.97	2.75	5.42	0.04	0.48	0.38	6.52	48.45	6.43
8	0	0.41	6.76	1.16	0.18	1.13	0.34	6.72	50.10	6.45
9	0	1.97	8.26	2.92	1.51	0.46	-0.33	7.14	50.10	6.02
10	0	-8.12	3.90	-1.97	-0.90	0.00	0.26	12.16	50.10	3.12
11	0	2.08	-11.54	-2.14	0.57	-0.29	-0.28	13.70	50.10	2.66
12	0	-8.12	3.90	-1.97	-0.90	0.00	0.26	12.16	50.10	3.12
13	0	2.44	8.20	4.06	-1.97	0.26	0.25	7.03	50.10	6.13
14	0	-15.14	-0.25	-14.60	-0.09	0.02	-0.27	15.00	50.10	2.34
15	180	-8.63	-6.89	-7.75	0.01	-0.50	-0.19	2.02	50.10	23.80
16	50	0.22	-0.57	0.26	0.02	0.06	-0.03	0.85	50.10	57.91

(1) See Figure 3.4.4.1-4 for definition of locations and angles of stress sections.

(2) ASME Service Level C is used for material allowable stress.

Table 11.1.3-6 P_m Stresses for PWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
120	0.8	-0.8	0.1	1.6	77.7	47.6
114	-0.5	1.0	-0.1	1.5	77.7	50.8
21	-0.3	-1.1	0.1	1.1	77.7	69.6
37	-1.1	-0.3	0.1	1.1	77.7	69.6
23	0.0	1.0	0.2	1.1	77.7	69.6
35	1.0	0.0	0.2	1.1	77.7	69.6
111	-0.3	0.5	0.2	0.9	77.7	85.3
112	0.5	-0.3	0.2	0.9	77.7	85.3
98	-0.5	-0.8	-0.2	0.9	77.7	85.3
40	0.1	-0.7	0.1	0.9	77.7	85.3
28	-0.8	0.1	0.1	0.9	77.7	85.3
51	0.8	0.1	0.1	0.8	77.7	96.1
7	0.1	0.8	0.1	0.8	77.7	96.1
110	-0.8	0.0	0.1	0.8	77.7	96.1
72	-0.8	-0.7	0.0	0.8	77.7	96.1
26	-0.8	-0.4	0.1	0.8	77.7	96.1
119	0.0	-0.8	0.1	0.8	77.7	96.1
42	-0.4	-0.8	0.1	0.8	77.7	96.1
95	0.0	-0.8	0.1	0.8	77.7	96.1
64	-0.8	0.0	0.1	0.8	77.7	96.1
49	-0.7	0.0	0.1	0.8	77.7	96.1
9	0.0	-0.7	0.1	0.8	77.7	96.1
94	-0.8	0.0	0.1	0.8	77.7	96.1
71	0.0	-0.7	0.1	0.8	77.7	96.1
46	-0.7	-0.2	0.1	0.7	77.7	110.0
123	-0.3	0.4	-0.1	0.7	77.7	110.0
124	0.4	-0.3	-0.1	0.7	77.7	110.0
96	-0.4	0.1	0.2	0.7	77.7	110.0
63	0.1	-0.4	0.2	0.7	77.7	110.0
92	0.2	-0.4	-0.2	0.7	77.7	110.0
91	-0.4	0.2	-0.2	0.7	77.7	110.0
99	-0.5	0.1	0.0	0.7	77.7	110.0
74	0.1	-0.5	0.0	0.7	77.7	110.0
104	-0.6	0.0	-0.2	0.6	77.7	128.5
106	0.1	-0.5	-0.1	0.6	77.7	128.5
117	-0.4	0.2	0.0	0.6	77.7	128.5
113	0.2	-0.3	0.0	0.6	77.7	128.5
67	-0.5	0.1	-0.1	0.6	77.7	128.5
88	0.5	0.2	-0.2	0.6	77.7	128.5
39	0.0	-0.5	0.1	0.6	77.7	128.5

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.
2. Stress allowables are taken at 800°F.

Table 11.1.3-7 $P_m + P_b$ Stresses for PWR Support Disk Off -Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
37	-2.5	-5.1	0.6	5.3	63.2	10.9
21	-5.1	-2.5	0.6	5.3	63.2	10.9
120	-0.4	-5.1	0.4	5.1	63.2	11.4
23	4.5	2.5	0.6	4.6	63.2	12.7
35	2.4	4.5	0.6	4.6	63.2	12.7
4	3.0	4.3	0.4	4.5	63.2	13.0
1	4.3	3.0	0.4	4.4	63.2	13.4
112	-1.1	-4.7	0.0	4.7	63.2	12.4
111	-4.7	-1.1	0.0	4.7	63.2	12.4
51	2.0	4.3	0.5	4.4	63.2	13.4
7	4.3	2.0	0.5	4.4	63.2	13.4
9	-3.9	-1.9	0.5	4.0	63.2	14.8
49	-1.9	-3.9	0.5	4.0	63.2	14.8
66	4.1	1.0	0.4	4.1	63.2	14.4
3	-3.7	-2.8	0.5	3.9	63.2	15.2
2	-2.8	-3.6	0.5	3.8	63.2	15.6
20	-2.9	-3.7	0.4	3.9	63.2	15.2
34	-3.7	-2.9	0.4	3.9	63.2	15.2
42	-0.9	-4.0	0.2	4.0	63.2	14.8
26	-4.0	-0.9	0.2	4.0	63.2	14.8
96	0.9	3.9	0.0	3.9	63.2	15.2
63	3.9	0.9	0.0	3.9	63.2	15.2
28	-3.6	-0.4	0.1	3.6	63.2	16.6
40	-0.4	-3.6	0.1	3.6	63.2	16.6
95	-3.3	-2.1	0.5	3.5	63.2	17.1
64	-2.1	-3.3	0.5	3.4	63.2	17.6
48	3.1	2.4	0.3	3.2	63.2	18.8
6	2.4	3.1	0.3	3.2	63.2	18.8
14	3.1	0.7	0.2	3.1	63.2	19.4
54	0.7	3.1	0.2	3.1	63.2	19.4
56	0.4	3.1	0.0	3.1	63.2	19.4
12	3.1	0.4	0.0	3.1	63.2	19.4
79	2.9	1.6	0.3	3.0	63.2	20.1
80	1.6	2.9	0.3	3.0	63.2	20.1
122	-2.8	-0.4	0.4	2.9	63.2	20.8
115	-0.4	-2.8	0.4	2.9	63.2	20.8
72	-1.5	-2.6	0.3	2.7	63.2	22.4
82	-2.4	-0.4	0.3	2.4	63.2	25.3
123	-1.9	0.2	-0.6	2.3	63.2	26.5
124	0.2	-1.9	-0.6	2.3	63.2	26.5

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.
2. Stress allowables are taken at 800°F.

Table 11.1.3-8 $P_m + P_b + Q$ Stresses for PWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
44	-9.2	-31.2	6.5	33.0	105.3	2.19
58	-9.0	-29.6	6.2	31.3	105.3	2.36
21	-25.3	-9.2	2.9	25.8	105.3	3.08
37	-9.1	-25.3	2.8	25.8	105.3	3.08
49	-8.5	-23.9	2.7	24.3	105.3	3.33
9	-23.8	-8.6	2.7	24.3	105.3	3.33
112	-8.8	-24.2	2.4	24.5	105.3	3.30
111	-24.1	-8.7	2.4	24.4	105.3	3.32
107	22.9	2.0	-4.2	23.7	105.3	3.44
123	21.9	2.6	5.8	23.5	105.3	3.48
124	2.5	21.9	5.7	23.4	105.3	3.50
76	1.9	22.7	-4.1	23.4	105.3	3.50
75	22.2	1.8	-4.1	22.9	105.3	3.60
80	-8.2	-22.1	2.3	22.5	105.3	3.68
79	-22.0	-8.1	2.3	22.4	105.3	3.70
92	2.1	21.3	5.4	22.7	105.3	3.64
91	21.2	2.3	5.6	22.7	105.3	3.64
108	1.6	21.9	-4.0	22.7	105.3	3.64
32	20.7	-0.4	-1.2	21.2	105.3	3.97
31	20.3	-0.5	1.6	21.1	105.3	3.99
45	-0.5	20.0	-1.5	20.7	105.3	4.09
17	19.9	-0.3	-1.2	20.4	105.3	4.16
18	19.5	-0.5	1.5	20.2	105.3	4.21
60	-0.4	19.2	-1.4	19.9	105.3	4.29
46	-2.3	17.2	0.3	19.5	105.3	4.40
20	-13.7	-13.8	4.9	18.6	105.3	4.66
34	-13.7	-13.7	4.9	18.5	105.3	4.69
59	-2.2	16.6	0.3	18.8	105.3	4.60
6	-13.0	-12.8	4.6	17.5	105.3	5.02
48	-12.7	-13.0	4.6	17.4	105.3	5.05
30	-11.4	-13.9	4.8	17.6	105.3	4.98
7	-16.2	-4.8	-1.9	16.5	105.3	5.38
120	-4.7	-17.0	1.4	17.2	105.3	5.12
42	-6.2	-16.7	1.5	16.9	105.3	5.23
95	-16.1	-7.2	-2.4	16.8	105.3	5.27
51	-4.7	-16.1	-1.9	16.4	105.3	5.42
26	-16.5	-6.1	1.4	16.7	105.3	5.31
64	-7.2	-16.0	-2.4	16.6	105.3	5.34
16	-10.8	-13.5	4.5	16.9	105.3	5.23
23	-16.0	-4.4	-1.8	16.3	105.3	5.46

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.
2. Stress allowables are taken at 800°F.

Table 11.1.3-9 P_m Stresses for BWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
265	-0.9	0.9	0.1	1.9	58.3	29.7
10	0.7	-0.4	-0.7	1.8	58.3	31.4
277	0.9	-0.9	0.1	1.8	58.3	31.4
262	-0.8	0.7	0.1	1.5	58.3	37.9
259	-0.7	0.6	0.1	1.4	58.3	40.6
77	0.6	-0.8	0.0	1.3	58.3	43.8
194	-0.6	0.6	0.1	1.2	58.3	47.6
197	-0.5	0.5	0.1	1.1	58.3	52.0
263	-0.9	-0.9	0.1	1.0	58.3	57.3
12	-0.4	0.0	-0.4	1.0	58.3	57.3
229	-0.8	0.2	0.1	1.0	58.3	57.3
264	-0.9	0.0	0.1	1.0	58.3	57.3
276	0.5	-0.4	0.1	0.9	58.3	63.8
76	0.6	-0.3	0.1	0.9	58.3	63.8
16	-0.3	0.4	-0.3	0.9	58.3	63.8
260	-0.8	-0.8	0.1	0.9	58.3	63.8
286	0.4	-0.5	0.1	0.9	58.3	63.8
85	-0.9	-0.8	0.0	0.9	58.3	63.8
269	-0.8	-0.9	0.0	0.9	58.3	63.8
273	0.0	-0.9	0.0	0.9	58.3	63.8
211	-0.6	0.3	0.1	0.9	58.3	63.8
261	-0.8	0.0	0.1	0.9	58.3	63.8
193	-0.7	-0.8	0.1	0.8	58.3	71.9
289	-0.8	-0.5	0.1	0.8	58.3	71.9
88	0.6	-0.2	0.1	0.8	58.3	71.9
103	-0.8	-0.1	0.1	0.8	58.3	71.9
9	0.0	-0.1	-0.4	0.8	58.3	71.9
14	-0.3	0.0	-0.3	0.8	58.3	71.9
81	0.0	-0.8	0.0	0.8	58.3	71.9
258	-0.7	0.0	0.1	0.8	58.3	71.9
268	-0.7	-0.4	0.1	0.7	58.3	82.3
97	0.6	-0.1	0.1	0.7	58.3	82.3
11	0.0	-0.1	-0.4	0.7	58.3	82.3
294	-0.7	-0.1	0.2	0.7	58.3	82.3
196	-0.6	-0.7	0.1	0.7	58.3	82.3
166	0.7	0.1	0.1	0.7	58.3	82.3
280	-0.7	-0.5	0.1	0.7	58.3	82.3
84	-0.7	-0.3	0.1	0.7	58.3	82.3
246	-0.1	-0.7	0.1	0.7	58.3	82.3
199	-0.5	-0.7	0.1	0.7	58.3	82.3

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.
2. Stress allowables are taken at 800°F.

Table 11.1.3-10 $P_m + P_b$ Stresses for BWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
265	-4.6	0.8	-0.2	5.3	48.6	8.2
295	-1.6	-5.0	0.5	5.1	48.6	8.5
294	-2.2	-4.9	0.5	5.0	48.6	8.7
254	-4.8	-2.2	0.5	4.9	48.6	8.9
257	-4.5	-1.6	0.6	4.6	48.6	9.6
293	-1.9	-4.4	0.4	4.5	48.6	9.8
289	-2.3	-4.3	0.6	4.5	48.6	9.8
243	-4.3	-1.5	0.2	4.3	48.6	10.3
24	-4.3	-1.4	0.1	4.3	48.6	10.3
263	-4.0	-2.4	0.7	4.3	48.6	10.3
275	1.7	4.3	0.3	4.3	48.6	10.3
252	4.2	1.7	0.3	4.3	48.6	10.3
246	-4.1	-1.7	0.5	4.2	48.6	10.6
274	1.7	4.1	0.3	4.2	48.6	10.6
10	-0.3	-2.2	-1.9	4.2	48.6	10.6
267	-1.6	-4.1	0.2	4.2	48.6	10.6
241	4.1	1.5	0.2	4.1	48.6	10.9
288	1.8	4.1	0.4	4.1	48.6	10.9
227	0.9	4.1	0.2	4.1	48.6	10.9
75	-1.7	-4.1	0.3	4.1	48.6	10.9
22	-4.1	-1.7	0.3	4.1	48.6	10.9
208	-1.6	-4.0	0.3	4.1	48.6	10.9
32	4.0	1.6	0.3	4.0	48.6	11.2
51	4.0	1.0	0.1	4.0	48.6	11.2
237	4.0	1.8	0.3	4.0	48.6	11.2
83	-1.6	-4.0	0.3	4.0	48.6	11.2
19	4.0	1.6	0.3	4.0	48.6	11.2
62	3.9	1.4	0.4	4.0	48.6	11.2
228	0.8	3.9	0.3	4.0	48.6	11.2
21	3.9	1.7	0.3	4.0	48.6	11.2
240	3.9	1.8	0.3	4.0	48.6	11.2
74	1.6	3.9	0.3	3.9	48.6	11.5
174	3.9	1.7	0.3	3.9	48.6	11.5
238	3.9	1.4	0.2	3.9	48.6	11.5
209	-1.4	-3.9	0.3	3.9	48.6	11.5
18	3.9	1.6	0.3	3.9	48.6	11.5
266	1.7	3.9	0.3	3.9	48.6	11.5
184	-3.8	-1.6	0.3	3.9	48.6	11.5
137	1.7	3.8	0.3	3.9	48.6	11.5
49	-3.8	-1.5	0.2	3.9	48.6	11.5

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.
2. Stress allowables are taken at 800°F.

Table 11.1.3-11 $P_m + P_b + Q$ Stresses for BWR Support Disk Off-Normal Conditions (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress[2]	Margin of Safety
295	-2.0	-20.5	1.3	20.6	81.0	2.93
268	-9.2	-18.9	2.2	19.4	81.0	3.18
289	-6.6	-18.8	1.6	19.0	81.0	3.26
16	16.0	5.1	5.4	18.3	81.0	3.43
139	-8.7	-17.8	2.1	18.2	81.0	3.45
30	-9.1	-17.2	2.7	18.0	81.0	3.50
14	15.7	4.6	5.2	17.8	81.0	3.55
265	-17.5	-6.3	1.6	17.7	81.0	3.58
276	-6.3	-17.5	1.3	17.7	81.0	3.58
166	-0.3	-17.4	0.9	17.5	81.0	3.63
43	-9.3	-16.5	2.7	17.4	81.0	3.66
266	-9.7	-16.4	2.2	17.0	81.0	3.76
137	-9.6	-16.2	2.1	16.8	81.0	3.82
24	-15.6	-10.2	2.9	16.8	81.0	3.82
18	-16.0	-8.6	2.6	16.8	81.0	3.82
15	13.6	4.8	-6.2	16.8	81.0	3.82
160	-5.5	-16.4	1.4	16.6	81.0	3.88
31	-15.8	-8.6	2.6	16.6	81.0	3.88
21	-16.0	-7.8	2.4	16.6	81.0	3.88
269	-7.8	-15.9	1.9	16.3	81.0	3.97
263	-16.1	-6.6	1.5	16.3	81.0	3.97
147	-6.1	-16.1	1.3	16.3	81.0	3.97
34	-15.6	-7.5	2.4	16.3	81.0	3.97
2	-1.8	14.2	-1.0	16.1	81.0	4.03
1	-1.8	14.2	-1.0	16.1	81.0	4.03
274	-7.8	-15.7	1.9	16.1	81.0	4.03
246	-15.9	-5.2	1.6	16.1	81.0	4.03
13	13.0	4.4	-6.0	16.1	81.0	4.03
37	-14.5	-9.6	2.7	15.7	81.0	4.16
238	-15.3	-8.4	1.8	15.7	81.0	4.16
241	-15.5	-6.8	1.4	15.7	81.0	4.16
145	-7.7	-15.2	1.8	15.6	81.0	4.19
243	-15.4	-6.8	1.3	15.6	81.0	4.19
4	-1.8	13.6	-0.9	15.5	81.0	4.23
3	-1.8	13.6	-0.9	15.5	81.0	4.23
111	-15.0	-8.2	1.8	15.4	81.0	4.26
267	-9.2	-14.8	1.9	15.3	81.0	4.29
277	-3.8	-14.8	1.4	15.0	81.0	4.40
140	-7.4	-14.4	1.7	14.8	81.0	4.47
27	-13.9	-8.4	2.5	14.8	81.0	4.47

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16.
2. Stress allowables are taken at 800°F.

Table 11.1.3-12 Summary of Maximum Stresses for PWR and BWR Fuel Basket Weldments – Off-Normal Condition (ksi)

Component	Stress Category	Maximum Stress Intensity[1]	Node Temperature (°F)	Allowable Stress[2,3]	Margin of Safety
PWR Top Weldment	$P_m + P_b$	0.7	297	20.7	+Large
	$P_m + P_b + Q$	52.1	292	56.1	+0.08
PWR Bottom Weldment	$P_m + P_b$	0.8	179	22.5	+Large
	$P_m + P_b + Q$	20.9	175	60.0	+1.87
BWR Top Weldment	$P_m + P_b$	1.2	226	19.4	+Large
	$P_m + P_b + Q$	14.6	383	52.5	+2.60
BWR Bottom Weldment	$P_m + P_b$	1.5	265	22.5	+Large
	$P_m + P_b + Q$	36.6	203	60.0	+0.64

1. Nodal stresses are from the finite element analysis.
2. Conservatively, stress allowables are taken at 400°F for the PWR top weldment, 300°F for the PWR bottom weldment, 500°F for the BWR top weldment, and 300°F for the BWR bottom weldment.
3. P_m stress allowables are conservatively used for the P_m+P_b evaluation.

11.1.4 Failure of Instrumentation

The Universal Storage System uses an electronic temperature sensing system to read and record the outlet air temperature at each of the four air outlets on each Vertical Concrete Cask. The temperatures are read and recorded daily.

11.1.4.1 Cause of Instrumentation Failure Event

Failure of the temperature measuring instrumentation could occur as a result of component failure, or as a result of another accident condition that interrupted power or damaged the sensing or reader terminals.

11.1.4.2 Detection of Instrumentation Failure Event

The failure is identified by the lack of a reading at the temperature reader terminal. The failure could also be identified by disparities between outlet temperatures in a cask or between similar casks.

11.1.4.3 Analysis of Instrumentation Failure Event

Since the temperature of each outlet of each concrete cask is recorded daily, the maximum time period during which the instrumentation failure may go undetected is 24 hours. Therefore, the maximum time period, during which an increase in the outlet air temperatures may go undetected, is 24 hours. The principal condition that could cause an increase in temperature is the blockage of the cooling air inlets or outlets. Section 11.2.13 shows that even if all of the inlets and outlets of a single cask are blocked immediately after a temperature measurement, it would take longer than 24 hours before any component approaches its allowable temperature limit. Therefore, the opportunity exists to identify and correct a defect prior to reaching the temperature limits. During the period of loss of instrumentation, no significant change in canister temperature will occur under normal conditions.

The purpose of the daily temperature monitoring is to ensure that the passive cooling system is continuing to operate normally. Instrument failure would be of no consequence, if the affected storage cask continued to operate in normal storage conditions.

Because the canister and the concrete cask are a large heat sink, and because there are few conditions that could result in a cooling air temperature increase, the temporary loss of remote sensing and monitoring of the outlet air temperature is not a major concern. No applicable regulatory criteria are violated by the failure of the temperature instrumentation system.

11.1.4.4 Corrective Actions

This event requires that the temperature reporting equipment be either replaced or repaired and calibrated. Prior to repair or replacement, the temperature shall be recorded manually.

11.1.4.5 Radiological Impact

There are no radiological consequences for this event.

11.1.5 Small Release of Radioactive Particulate From the Canister Exterior

The procedures for loading the canister provide for steps to minimize exterior surface contact with contaminated spent fuel pool water, and the exterior surface of the canister is surveyed by smear at the top end to verify canister surface conditions. Design features are also employed to ensure that the canister surface is generally free of surface contamination prior to its installation in the concrete cask. The surface of the canister is free of traps that could hold contamination. The presence of contamination on the external surface of the canister is unlikely, and, therefore, no particulate release from the canister exterior surface is expected to occur in normal use.

11.1.5.1 Cause of Radioactive Particulate Release Event

In spite of precautions taken to preclude contamination of the external surface of the canister, it is possible that a portion of the canister surface may become slightly contaminated during fuel loading by the spent fuel pool water and that the contamination may go undetected. Surface contamination could become airborne and be released as a result of the air flow over the canister surface.

11.1.5.2 Detection of Radioactive Particulate Release Event

The release of small amounts of radioactive particles over time is difficult to detect. Any release is likely to be too low to be detected by any of the normally employed long-term radiation dose monitoring methods (such as TLDs). It is possible that a suspected release could be verified by a smear survey of the air outlets.

11.1.5.3 Analysis of Radioactive Particulate Release Event

A calculation is made to determine the level of surface contamination that if released would result in a dose of one tenth of one (0.1) mrem at a minimum distance of 100 meters from a design basis storage cask. ISFSI-specific allowable dose rates and surface contamination limits will be calculated on a site specific basis to conform to 10 CFR 72. The method for determining the residual contamination limit is based on the plume dispersion calculations presented in U.S. NRC Regulatory Guides 1.109 [9] and 1.145 [13] and is highly conservative. The calculation shows that a residual contamination of approximately 1.57×10^5 dpm/100 cm^2 β-γ and 5.24×10^2 dpm/100 cm^2 α activity, on the surface of the design basis canister, is required to yield a dose of one tenth of one (0.1) mrem at the minimum distance of 100 meters. The canister surface area is inversely

proportional to the allowable surface contamination. The design basis cask is, therefore, the Class 3 PWR cask, which has the largest canister surface area at 3.06×10^5 cm^2.

The above analysis demonstrates that the off-site radiological consequences from the release of canister surface contamination is negligible, and all applicable regulatory criteria can be met for an ISFSI array.

11.1.5.4 Corrective Actions

No corrective action is required since the radiological consequence is negligible.

11.1.5.5 Radiological Impact

As shown above, the potential off-site radiological impact due to the release of canister surface contamination is negligible.

11.1.6 Off-Normal Events Evaluation for Site Specific Spent Fuel

This section presents the off-normal events evaluation of spent fuel assemblies or configurations, which are unique to specific reactor sites. These site specific fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blankets and variable enrichment assemblies, fuel with burnup that exceeds the design basis, and fuel that is classified as damaged.

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

11.1.6.1 Off-Normal Events Evaluation for Maine Yankee Site Specific Spent Fuel

Maine Yankee site specific fuels are described in Section 1.3.2.1. A thermal evaluation has been performed for Maine Yankee site specific fuels that exceed the design basis burnup as shown in Section 4.5.1.2. As shown in that section, loading of fuel with a burnup between 45,000 and 50,000 MWD/MTU is subject to preferential loading in designated basket positions in the Transportable Storage Canister.

With preferential loading, the design basis total heat load of the canister is not changed. Consequently, the thermal performance for the Maine Yankee site specific fuels is bounded by the design basis PWR fuels. Therefore, no further evaluation is required for the off-normal thermal events (severe ambient temperature conditions and blockage of half of the air inlets) as shown in Sections 11.1.1 and 11.1.2. In Section 3.6.1.1, the total weight of the canister contents for Maine Yankee site specific fuels is shown to be bounded by the PWR design basis fuels. Therefore, the evaluation for the off-normal canister handling load in Section 11.1.3 bounds the canister configuration loaded with Maine Yankee fuels.

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