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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 1.8.

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 NG-4424 and NG-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 B3-1 of Appendix B. 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 exceptions to the ASME Code are described in Table B3-1 of Appendix B. 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 70 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 a minimum 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.

If the canister is to be ASME Code N-stamped, the canister shall be hydrostatically tested in accordance with the requirements of ASME Code Subsection NB-6220 and Code Case N-595-2 following fabrication of the canister, insertion of the basket and welding of the lid support ring, and prior to fuel loading. The post-loading pressure test shall also be performed.

9.1.2.1 Transfer Casks

The transfer cask is provided in the Standard or Advanced configuration. The Standard transfer cask is restricted to handling the Standard weight canister. The Advanced transfer cask incorporates a reinforced trunnion design that allows it to handle either the standard weight, or a heavier weight, canister.

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

Standard Transfer Cask

The Standard transfer cask lifting trunnion load test shall consist of applying a vertical load of 630,000 pounds, which is greater than 300% of the maximum service load for the transfer cask and loaded canister with the shield lid and full of water (208,400 lbs). The bottom shield door and rail load test shall consist of applying a vertical load of 265,200 pounds, which is over 300% of the maximum service load (88,400 lbs). These maximum service loads are selected based on the heaviest configuration and, thus, bound all of the other configurations.

Advanced Transfer Cask

The Advanced transfer cask lifting trunnion load test shall consist of applying a vertical load of 690,000 pounds, which is greater than 300% of the maximum service load (225,000 pounds) for the transfer cask and 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 300,000 pounds, which is over 300% of the maximum service load (98,000 lbs). These maximum service loads are based on the heaviest configuration and, thus, bound all 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. Liquid penetrant examination (the magnetic particle method may be used on ferrous material) shall be performed on accessible trunnion and shield door rail load-bearing welds in accordance with ASME Code Section V, Articles 1, 6 and/or 7, with acceptance in accordance with ASME Code Section III, NF-5340 or NF-5350, as applicable. 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 or magnetic particle examinations of all load bearing welds shall be performed in accordance with ASME Code Section V, Articles 1, 6 and/or 7, with acceptance in accordance with ASME Code Section III, NF-5350 or NF-5340, as applicable.

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, which is greater than 150 percent of the maximum concrete cask weight plus a 10 percent dynamic load factor, where the concrete cask weight is determined, based on the class, from Table 3.2-1 or 3.2-2.

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 or magnetic particle examinations of load bearing surfaces shall be performed in accordance with ASME Code, Section V, Articles 1, 6 and/or 7, with acceptance criteria in accordance with ASME Code, Section III, Subsection NF, NF-5350 or NF-5340, as applicable.

Any evidence of permanent deformation, cracking, or galling, or unacceptable liquid penetrant or magnetic particle 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[9] 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) 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 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 required ^{10}B areal density, as specified on the component License Drawings. The constituents of the neutron absorber material shall be verified by chemical testing and/or 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 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 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 in 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 Appendix A of the Amendment 3 Technical Specifications, as verification of the thermal performance of the storage system.

9.1.8 Cask Identification

A 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

Model Number:	(UMS-XXX)
Cask No.:	(XXX)
Empty Weight:	(Pounds [kilograms])

Note: Additional information may be added to the nameplate at the user's/NAC's discretion.

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9.2 Maintenance Program

This section presents the maintenance requirements for the UMS® Universal Storage System and for the transfer cask.

9.2.1 UMS® Storage System Maintenance

The UMS® 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.

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

The routine surveillance requirements are described in Technical Specification LCO 3.1.6 in Appendix A. It is not necessary to inspect the concrete cask or canister during the storage period as long as the thermal performance is normal, based on daily temperature verification.

The ambient air temperature and air outlet temperature of each Vertical Concrete Cask must be recorded upon placement in service. Thereafter, the temperatures shall be recorded on a daily basis to verify the continuing thermal performance of the system.

In the event of a decline in thermal performance, the heat removal system must be restored to acceptable operation. The user should perform a visual inspection of air inlets and outlets for evidence of blockage and verify that the inlet and outlet screens are whole, secure and in place.

The user must also visually inspect the Vertical Concrete Cask within 4 hours of any off-normal, accident or natural phenomena event, such as an earthquake.

An annual inspection of the Vertical Concrete Cask exterior is required, to include:

- Visual inspection of concrete surfaces 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 corroded surfaces, including concrete cask lifting lugs, if present.

9.2.2 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 visually inspected for permanent deformation and cracking. Liquid penetrant examination (the magnetic particle method may be used on ferrous material) shall be performed on all accessible lifting trunnion and shield door rail load-bearing weld surfaces. 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.

9.2.3 Required Surveillance of First Storage System Placed in Service

For the first Universal Storage System placed in service with a heat load equal to or greater than 10 kW, the canister is loaded with spent fuel assemblies and the decay heat load calculated for that canister. The canister is then loaded into the vertical concrete cask, and the cask's thermal performance is evaluated by measuring the ambient and air outlet temperatures for normal air flow. The purpose of the surveillance is to measure the heat removal performance of the Universal Storage System and to establish baseline data. In accordance with 10 CFR 72.4, a letter report summarizing the results of the surveillance and evaluation will be submitted to the NRC within 30 days of placing the loaded cask on the ISFSI pad. The report will include a comparison of the calculated temperatures of the NAC-UMS® system heat load to the measured temperatures. A report is not required to be submitted for the NAC-UMS® systems that are subsequently loaded, provided that the performance of the first system placed in service with a heat load ≥ 10 kW, is demonstrated by the comparison of the calculated and measured temperatures.

9.3 References

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

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

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

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

9. American National Standards Institute, "Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

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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 Section 5.3.1. 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.

Radiation exposure rates at various work locations are determined for the principal Universal Storage System operational steps using a combination of the SAS4 [3] and SKYSHINE III [4] computer codes. The use of SAS4 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-foot dose rates are:

Vertical Concrete Cask	Design Basis Surface Dose Rate (mrem/hr)	Surface Dose Rate (mrem/hr)		1-Foot Maximum Dose Rate (mrem/hr)	
		PWR	BWR	PWR	BWR
Side wall	50.0 (avg.)	37.3	22.7	42.3	24.5
Air inlet(1)	100.0(2)	136	129	47.8	44.9
Air outlet	100.0(2)	63	55	15.7	12.8
Top lid	50.0 (avg.)	26.1	19.7	22.6	15.7

(1) Air inlet dose rates are based on the use of the air inlet shields. Design basis source terms require the use of the inlet shields to remain below the technical specification limits outlined in Appendix A.

(2) An air inlet and outlet average dose rate of 100 mrem/hr.

The calculated dose rates at these, and at other dose points, are reported in Sections 5.1.3 and 5.4.3. 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 Section 2.1.

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, 5.4.3 and 10.2.1. Exposure is evaluated by identifying the tasks and estimating the 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.

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 17×17 Standard fuel assembly, with an initial enrichment of 3.7 wt % ^{235}U. The design basis BWR assembly is the GE 9×9, 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. The principal parameters of these assemblies are presented in Table 2.1.1-1.

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. As documented in Section 5.1, exposures from the advanced transfer cask are not going to differ substantially from exposures documented for the standard 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, 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 evaluations presented in Tables 10.3-4 through 10.3-7 estimate the exposure due to a combination of inspection and surveillance activities and other tasks that are anticipated to be representative of an operational facility. The visual inspection exposure, based on a daily inspection of the storage cask or storage cask array, is provided for information only since a daily inspection is not required as long as the temperature monitoring system is operational. 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 10.3-2. 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 10.3-3.

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 are shown in Table 10.3-4 for the single PWR cask and in Table 10.3-6 for the BWR cask. The annual operation and surveillance requirements for the assumed 20-cask ISFSI are shown in Tables 10.3-5 and 10.3-7 for PWR and BWR configurations, respectively. These tables show an estimated annual collective exposure of 377.6 person-mrem for the PWR cask configuration and 239.4 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

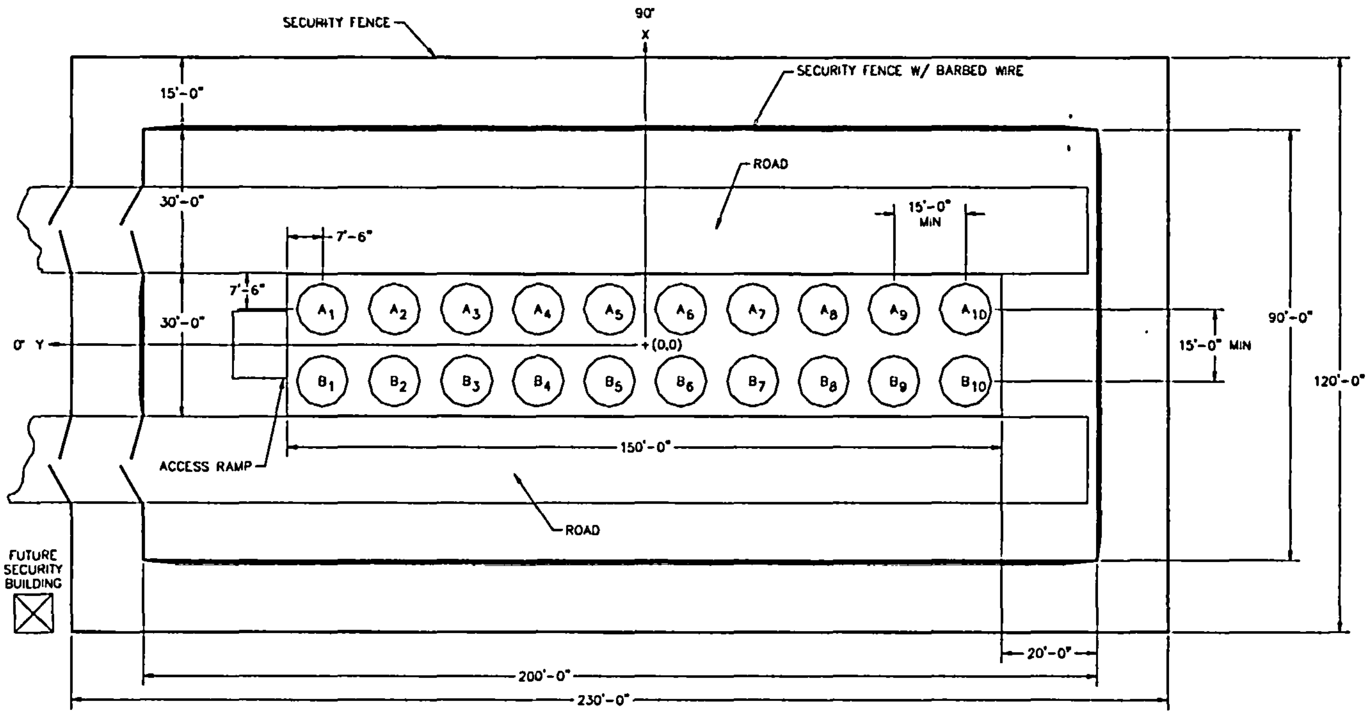

Table 10.3-1 Estimated Exposure for Operations Using the Standard Transfer Cask

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 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 10.3-3 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 10.3-4 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 10.3-5 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	2.55	1	66.3
Total Person-mrem for the 20-Cask Array						377.6
Total Person-mrem for a Single Cask in the Array						18.6

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

Table 10.3-6 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 10.3-7 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)
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						239.4
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.

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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.

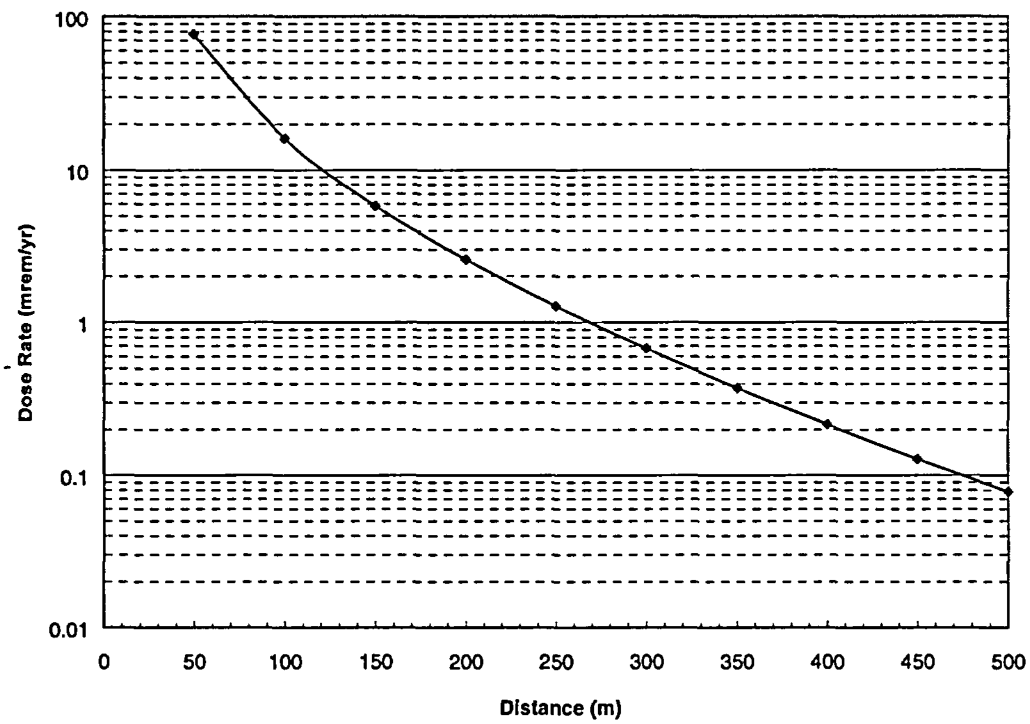
The storage cask array is explicitly modeled in the code, with the source term from each cask represented as top and side surface sources. Surface source emission fluxes are provided from one-dimensional SAS1 shielding evaluations. The top and side source energy distributions for both neutron and gamma 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 three-dimensional shielding evaluation. Surface gamma-ray fluxes are also adjusted for dose peaks associated with fuel assembly end-fitting hardware and radiation streaming through the cask vents and canister-to-cask annulus. Air inlet and outlet dose rates have been recalculated in Section 5.4 based on the use of the MCBEND Monte Carlo code. The MCBEND generated air inlet dose rate results are significantly higher than those obtained from the SAS4 evaluation. Since the air inlets represent less than 0.6% of the total radial surface of the cask, and considering that the 100 mrem/hr air inlet and outlet dose rate limit is retained in the technical specification, an increase in the calculated air inlet dose rate (surface flux) will not significantly impact SKYSHINE results based on the SAS4 evaluation. The 2×10 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 6.3.2.

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 2×10 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 2×10 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 2×10 PWR cask array to meet the 10 CFR 72.104(a) 25 mrem/year requirement. The 2×10 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 2×10 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

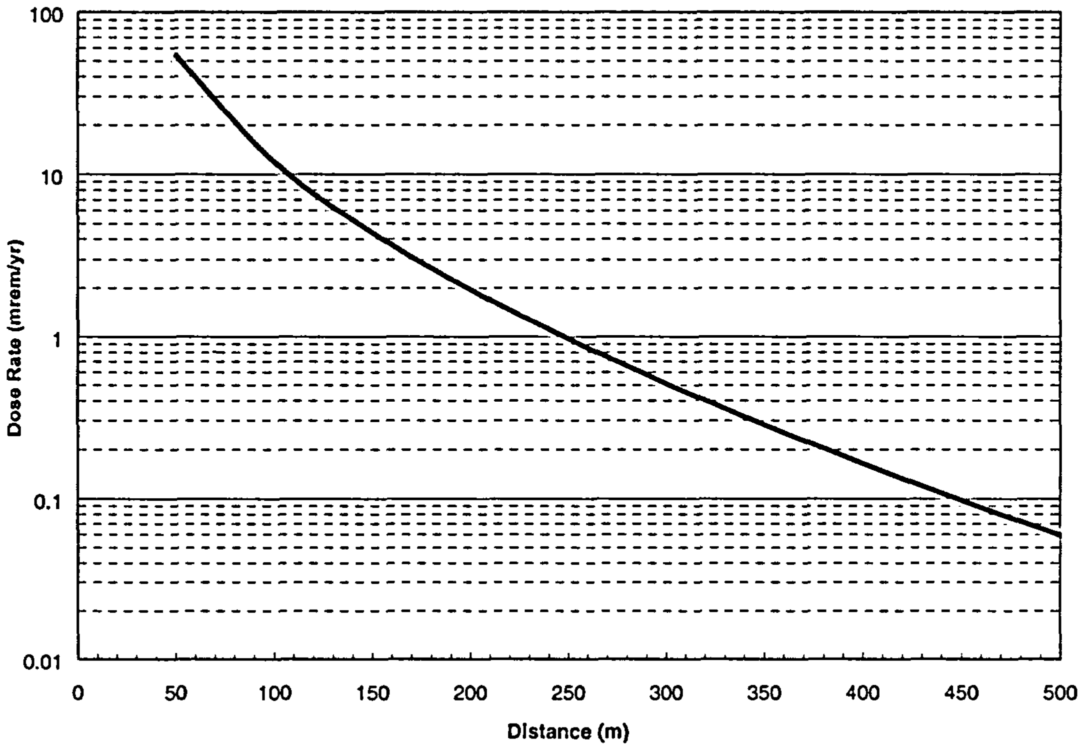


Distance from Center of Cask(m)	Dose Rate (mrem/year)			
	Gamma Dose	Neutron Dose	N-Gamma Dose	Total Dose
50	7.28E+01	3.85E+00	7.93E-04	77
100	1.47E+01	1.34E+00	8.07E-04	16
150	5.25E+00	5.56E-01	8.14E-04	5.8
200	2.32E+00	2.54E-01	7.86E-04	2.6
250	1.15E+00	1.24E-01	7.26E-04	1.3
300	6.12E-01	6.29E-02	6.43E-04	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 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. 2,080-hour exposure.

Table 10.4-2 Annual Exposures from a 2×10 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. 2,080-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

1. 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.

2. Title 10 of the Code of Federal Regulations, Part 20 (10 CFR 20), "Standards for Protection Against Radiation," April 1996.

3. ORNL/NUREG/CSD-2/V1/R5, Volume 1, Section S4, "SAS4: A Monte Carlo Cask Shielding Analysis Module Using an Automated Biasing Procedure," Tang, J. S., September 1995.

4. SKYSHINE III, "Calculation of the Effects of Structure Design on Neutron, Primary Gamma-Ray and Secondary Gamma-Ray Dose Rates in Air," RISC Code Package CCC-289, NAC International, Version 4.0.1, February 1997.

5. ORNL/NUREG/CSD-2/V3/R5, Volume 1, Section S1, "SAS1: A One-Dimensional Shielding Analysis Module," Knight, J.R. et al., September 1995.

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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	672	667	561	540	1058	1058
Support Disks	628	640	505	505	800	700
Heat Transfer Disks	626	638	502	504	750	750
Canister Shell	381	405	226	252	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)

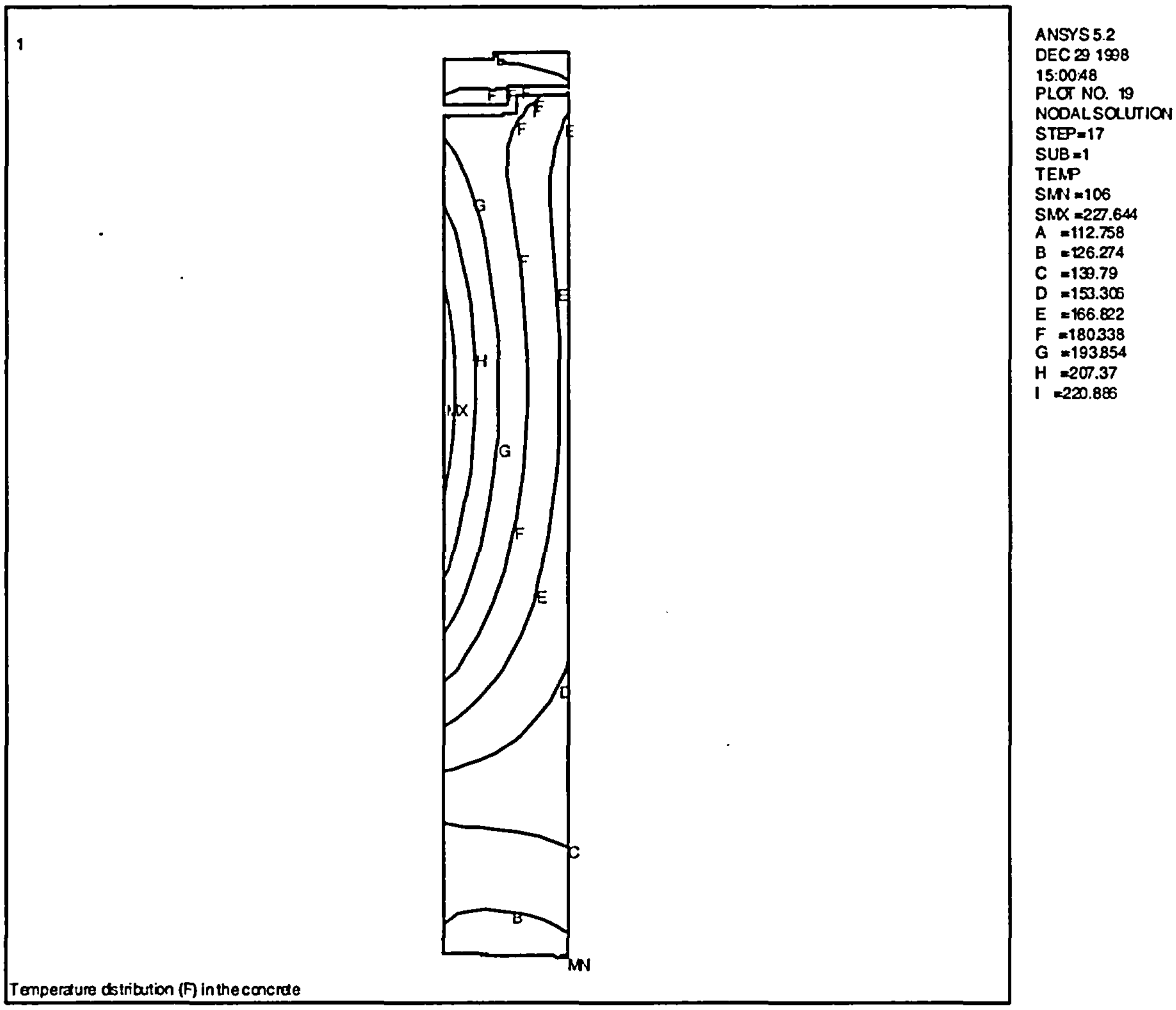


Figure 11.1.1-2 Vertical Concrete Cask Air Temperature (°F) Profile for Off-Normal Storage Condition 106°F Ambient Temperature (PWR Fuel)

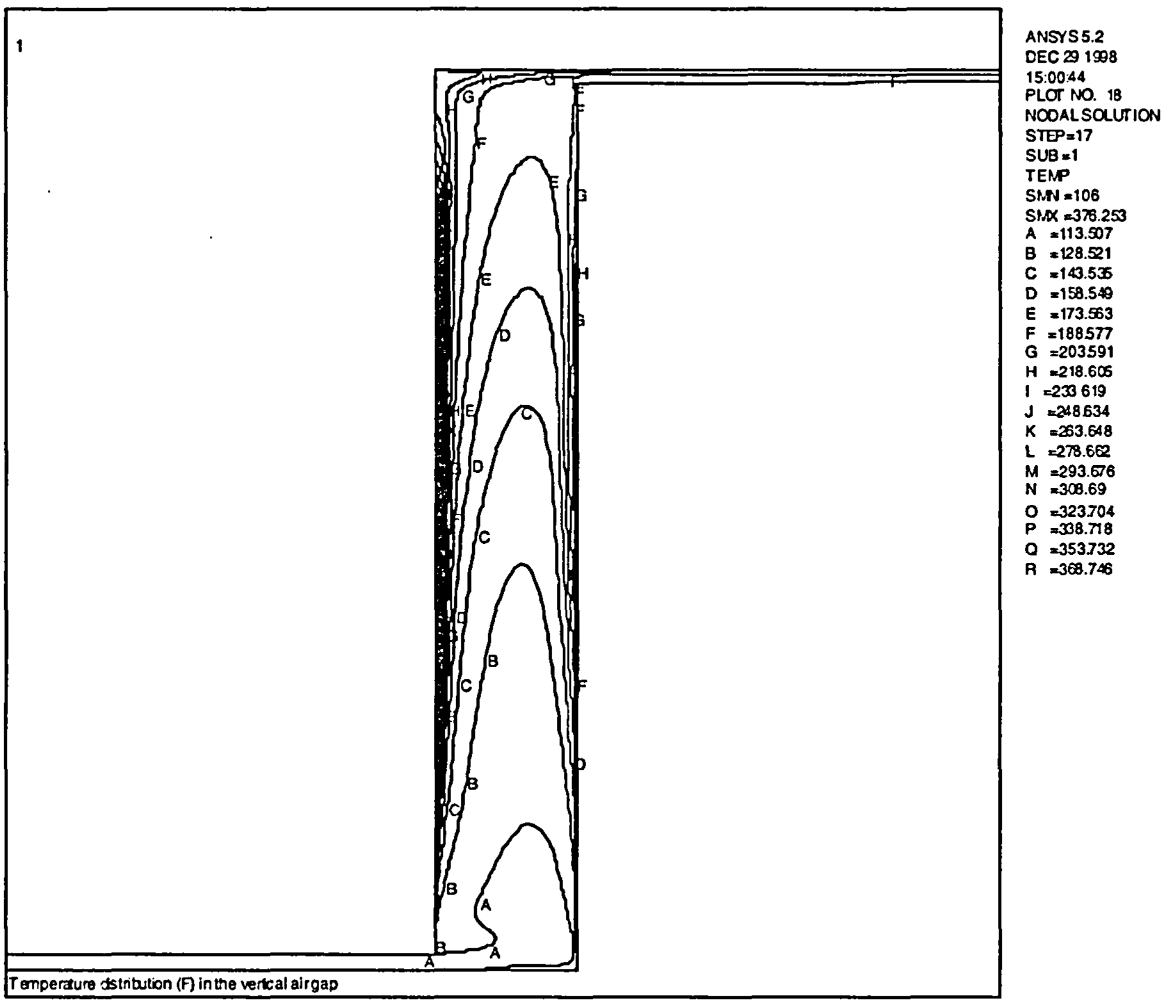


Figure 11.1.1-3 Concrete Temperature (°F) for Off-Normal Storage Condition -40°F Ambient Temperature (PWR Fuel)

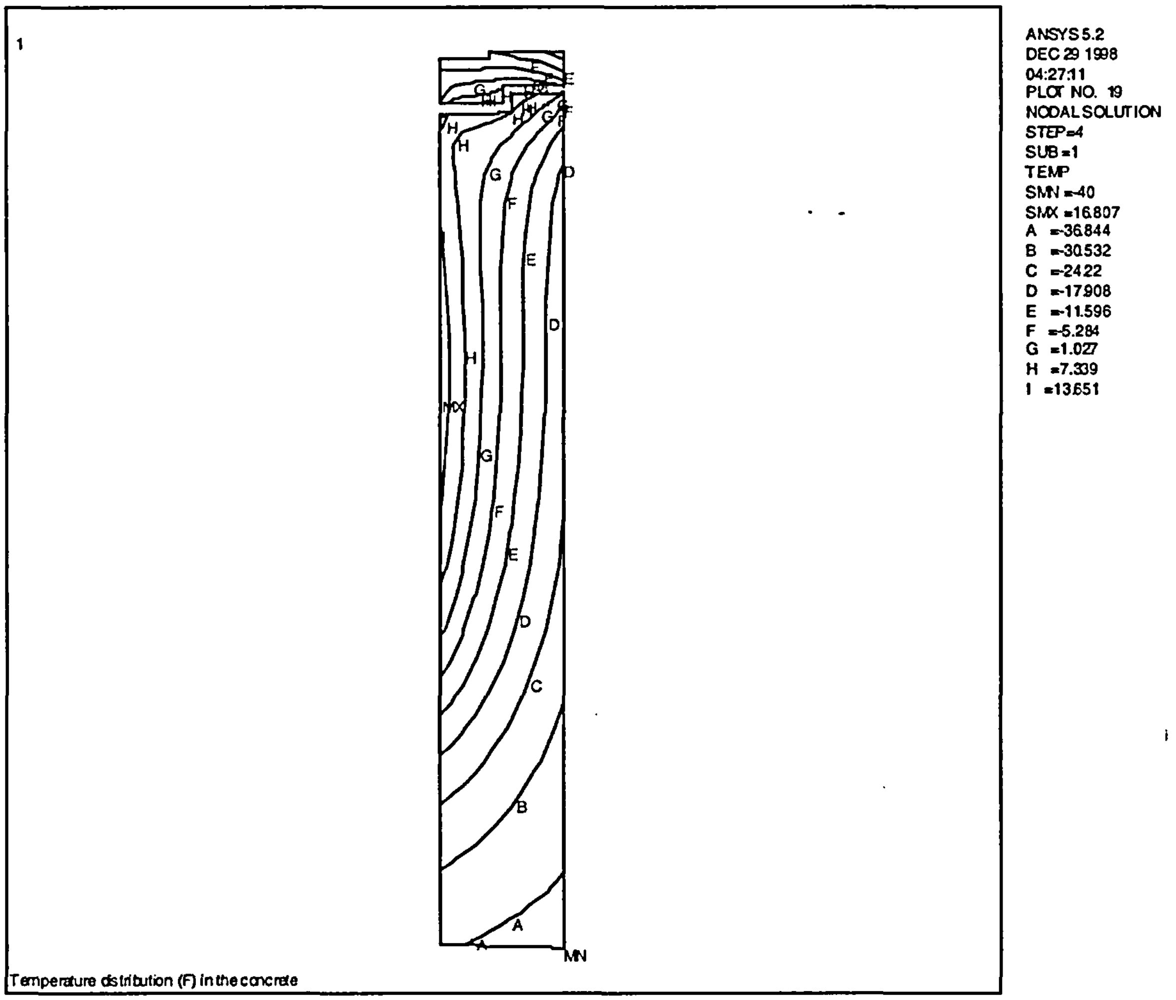
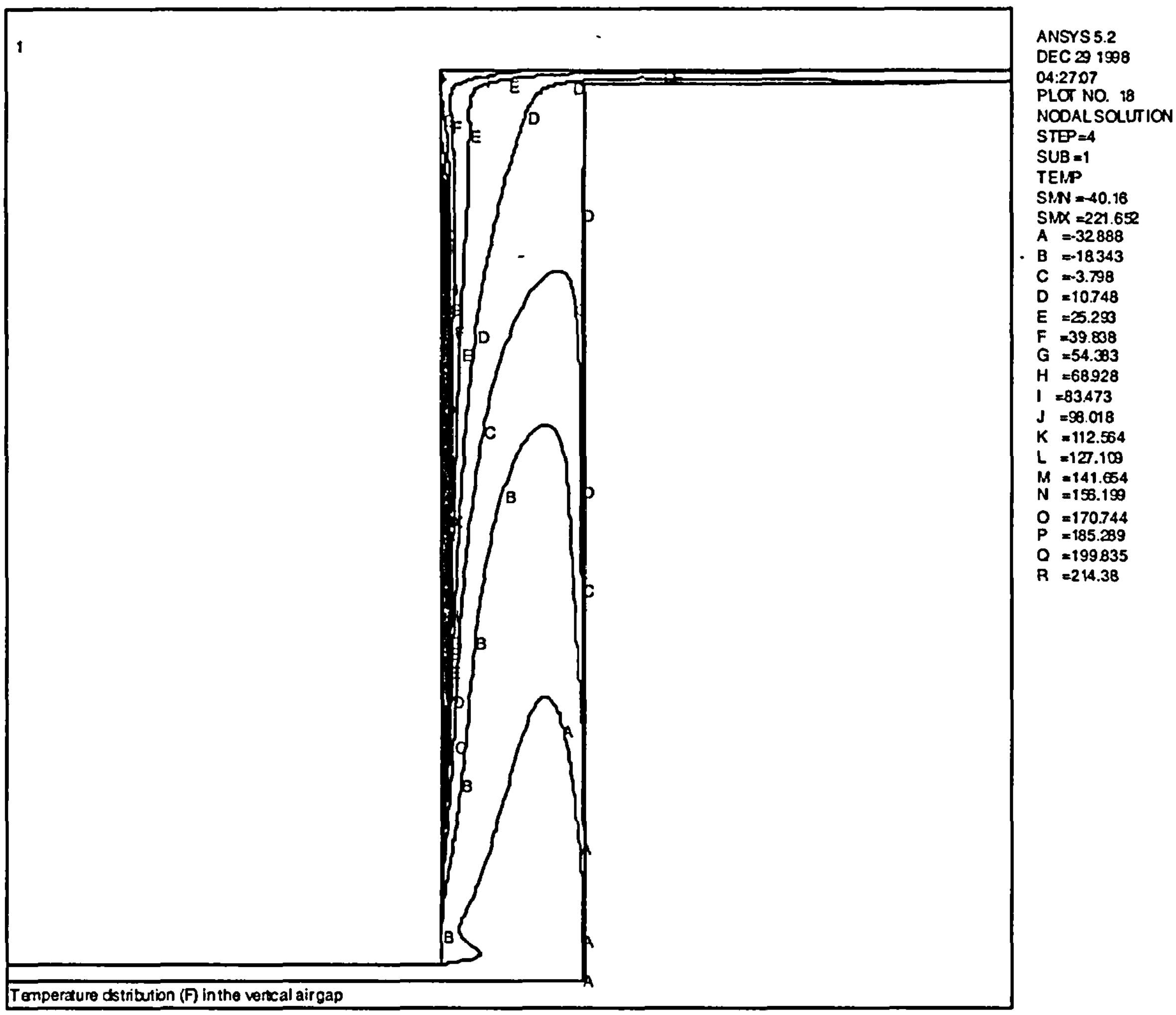


Figure 11.1.1-4 Vertical Concrete Cask Air Temperature (°F) Profile for Off-Normal Storage Condition -40°F Ambient Temperature (PWR Fuel)



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 persons observing an increase in the concrete cask outlet temperature, which would result from the reduced air flow caused by the blockage. 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	649	642	1058	1058
Support Disks	603	614	800	700
Heat Transfer Disks	600	612	750	750
Canister Shell	350	373	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. The canister may be handled vertically in the Standard or Advanced transfer casks. The Standard and Advanced transfer casks are similar, except that the Advanced transfer cask incorporates a reinforcing gusset at the lifting trunnions allowing an increased canister weight.

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.

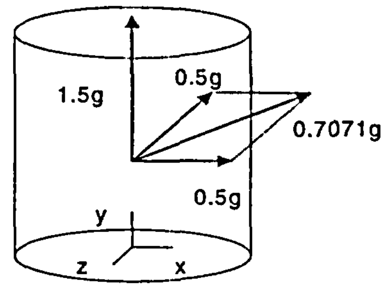
11.1.3.2 Detection of Off-Normal Canister Handling Load Event

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 off-normal handling analysis is performed 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 for the 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. Interaction between the fuel basket and the canister during vertical load conditions 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 with minimum lid-to-shell weld sizes (Class 1 PWR) with the heaviest fuel/fuel basket weight (Class 5 BWR). The material stress allowables used in the analysis consider the higher component temperatures that occur during transfer operations.

The off-normal canister handling loads are defined as 0.5g applied in all directions (i.e., in the global x, y, and z directions) in addition to a 1g lifting load applied in the finite element model. The resulting off-normal handling accelerations are 0.7071g in the lateral direction and 1.5g (0.5g + 1g) in the vertical direction.



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. In addition, for the lateral loading, the canister is assumed to be handled inside the vertical concrete cask. The interface between the canister shell and the concrete cask inner surface is represented using CONTAC52 elements.

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 (0.707g 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

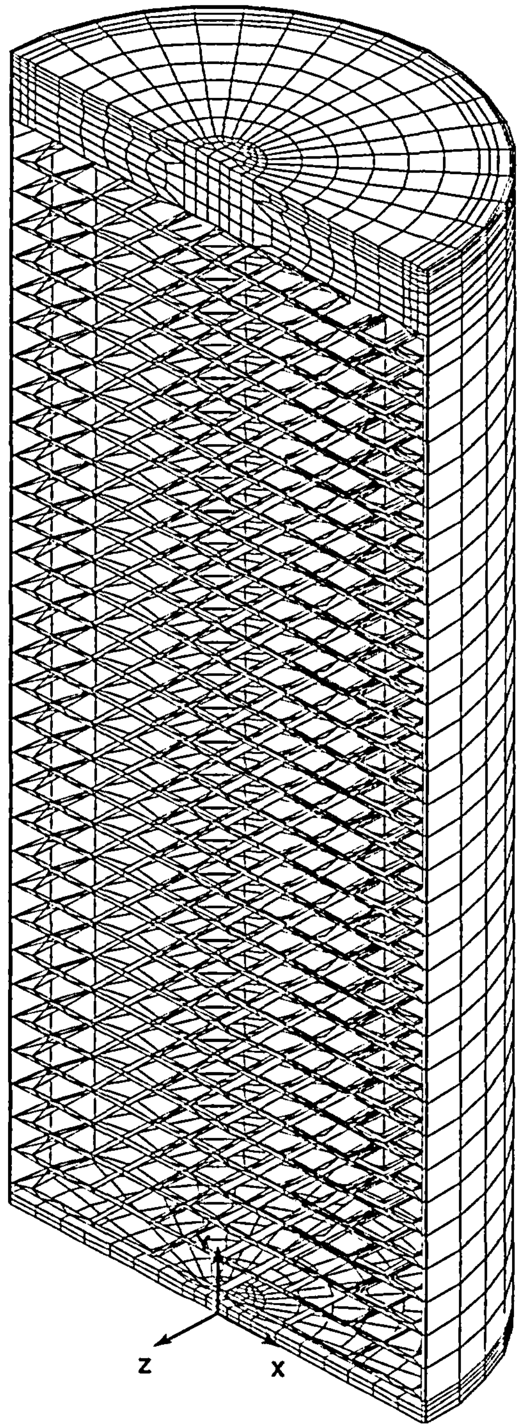


Table 11.1.3-1 Canister Off-Normal Handling (No Internal Pressure) Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	Angle (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0	-0.65	0.66	2.73	0.07	0.02	-0.03	3.39
2	0	2.02	-2.42	-1.40	0.36	0.07	-0.23	4.52
3	0	-0.32	-3.62	1.16	0.28	0.07	0.89	5.23
4	0	-0.04	0.00	0.83	0.00	0.01	-0.02	0.87
5	0	-0.09	0.00	0.76	0.00	0.00	0.00	0.86
6	0	-0.12	-0.01	0.79	0.00	0.00	0.00	0.91
7	0	-0.14	-0.04	0.93	0.01	-0.01	0.00	1.07
8	0	0.05	0.01	1.81	-0.03	-0.16	-0.02	1.85
9	0	0.05	0.55	2.77	-0.04	-0.29	0.10	2.77
10	0	-0.33	0.53	3.51	-0.12	-0.40	0.11	3.91
11	0	-0.62	1.28	2.39	-0.06	-0.31	-0.71	3.41
12	0	-0.14	0.76	3.53	-0.15	-0.21	0.30	3.75
13	0	-2.09	1.36	-0.52	-0.13	-0.05	-1.61	4.46
14	0	0.35	0.40	-0.01	0.00	0.19	-0.03	0.56
15	180	-0.04	-0.04	0.00	0.00	0.00	0.00	0.04
16	0	-0.02	0.03	0.00	0.00	-0.01	0.00	0.05

(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 (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0	3.64	0.54	7.08	0.13	-0.03	0.26	6.57
2	0	0.77	-5.92	-12.15	0.61	0.18	-0.84	13.09
3	0	-1.34	0.67	17.12	-0.15	-0.15	1.08	18.60
4	0	-0.04	-0.24	0.76	0.02	0.03	-0.02	1.01
5	0	-0.09	0.03	0.77	-0.01	0.00	0.00	0.86
6	0	-0.13	0.07	0.81	-0.01	0.00	0.00	0.94
7	0	-0.16	0.13	0.97	0.00	-0.01	0.00	1.13
8	0	0.06	0.14	1.96	-0.04	-0.13	-0.02	1.93
9	0	-0.15	0.50	3.08	0.00	-0.40	-0.06	3.29
10	0	-0.54	1.03	5.09	-0.21	-0.25	0.35	5.71
11	0	-1.12	1.25	1.58	-0.05	-0.28	-1.69	4.38
12	0	-0.58	0.92	4.68	-0.21	-0.24	0.34	5.35
13	0	-4.53	1.12	-1.97	-0.29	0.11	-1.38	6.29
14	180	8.93	8.96	0.25	0.00	0.17	-0.04	8.72
15	0	-0.25	-0.24	-0.01	0.00	0.00	0.00	0.24
16	0	1.02	1.08	0.03	0.01	-0.01	0.00	1.05

(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 (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0	-0.63	1.20	4.20	0.04	0.01	-0.21	4.85	21.04	3.3
2	0	3.00	-3.67	-2.33	0.53	0.06	-0.44	6.79	21.03	2.1
3	0	-0.50	-5.51	1.61	0.44	0.12	1.32	7.80	20.99	1.7
4	0	-0.02	0.78	1.28	-0.06	0.02	-0.04	1.31	19.39	13.8
5	0	-0.09	0.78	1.18	-0.07	0.00	0.00	1.28	17.93	13.1
6	0	-0.12	0.77	1.20	-0.07	0.00	0.00	1.33	17.77	12.4
7	0	-0.16	0.74	1.33	-0.06	-0.01	0.00	1.49	19.12	11.8
8	0	0.01	0.47	2.24	-0.06	-0.18	-0.01	2.26	20.51	8.1
9	0	0.04	0.81	3.18	-0.08	-0.30	0.12	3.19	20.94	5.6
10	0	-0.43	0.74	3.78	-0.14	-0.41	0.04	4.27	20.95	3.9
11	0	-0.49	1.40	2.33	-0.08	-0.30	-0.71	3.23	21.06	5.5
12	0	-0.22	0.79	3.17	-0.16	-0.21	0.20	3.46	20.94	5.1
13	0	-1.83	1.53	-0.35	-0.17	-0.04	-1.56	4.36	21.07	3.8
14	0	0.59	0.65	-0.02	0.00	0.30	-0.05	0.90	20.08	21.4
15	180	-0.06	-0.06	-0.01	0.00	0.00	0.00	0.06	20.96	373.8
16	0	0.01	0.05	0.00	0.00	-0.01	0.00	0.06	21.08	367.5

(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 (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0	4.89	0.68	10.67	0.18	-0.05	0.25	10.01	31.23	2.1
2	0	1.23	-9.06	-18.95	0.91	0.16	-1.32	20.43	31.21	0.53
3	0	-2.06	1.32	26.71	-0.24	-0.11	1.61	28.97	31.11	0.1
4	0	-0.02	1.10	1.36	-0.09	0.00	-0.04	1.39	27.25	18.7
5	0	-0.09	0.82	1.19	-0.07	0.00	0.00	1.28	24.83	18.4
6	0	-0.14	0.89	1.23	-0.08	0.01	0.00	1.38	24.62	16.9
7	0	-0.18	0.99	1.40	-0.08	-0.01	0.00	1.58	26.62	15.8
8	0	0.01	0.47	2.32	-0.06	-0.15	-0.01	2.33	29.94	11.9
9	0	-0.11	0.94	4.09	-0.06	-0.40	0.03	4.25	30.97	6.3
10	0	-0.63	1.00	4.54	-0.21	-0.26	0.22	5.23	31.01	4.9
11	0	-0.93	1.50	2.00	-0.08	-0.29	-1.72	4.56	31.28	5.9
12	0	-0.69	0.89	4.14	-0.21	-0.25	0.20	4.89	30.98	5.3
13	0	-4.11	1.30	-1.86	-0.34	0.12	-1.28	6.04	31.29	4.2
14	170	14.01	14.04	0.40	-0.01	0.27	-0.07	13.66	28.91	1.1
15	0	-0.20	-0.22	-0.02	0.00	0.00	0.00	0.20	31.03	150.7
16	0	1.04	1.11	0.03	0.01	-0.01	0.00	1.08	31.33	28.0

(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 (degrees)[1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	60	4.78	3.21	14.22	0.17	0.20	0.19	11.03	50.10	3.5
2	0	1.64	-8.45	-22.75	0.88	-0.11	-1.36	24.62	50.10	1.03
3	50	-2.19	3.26	30.88	-0.05	0.48	1.50	33.21	50.10	0.5
4	0	-0.07	2.44	1.37	-0.18	0.56	0.00	2.78	48.46	16.5
5	0	-1.39	9.10	0.08	-0.90	0.79	-0.08	10.71	44.83	3.2
6	0	-1.60	9.78	0.13	-0.98	-0.87	0.10	11.62	44.44	2.8
7	0	-0.26	2.93	2.15	-0.20	-0.64	0.03	3.58	47.79	12.4
8	0	0.21	1.55	4.40	-0.11	-0.13	0.03	4.21	50.10	10.9
9	0	1.13	2.00	6.96	0.00	-0.12	1.36	6.44	50.10	6.8
10	0	-7.08	-1.89	2.43	-0.33	-0.11	-0.94	9.71	50.10	4.2
11	140	2.31	-2.03	-10.03	0.10	-0.09	0.99	12.50	50.10	3.01
12	0	-7.08	-1.89	2.43	-0.33	-0.11	-0.94	9.71	50.10	4.2
13	30	-5.47	-0.78	1.84	-0.39	0.07	0.65	7.46	50.10	5.7
14	180	-15.40	-15.03	-0.23	0.26	0.00	-0.11	15.31	50.10	2.27
15	180	-8.41	-7.57	-6.63	0.20	0.49	0.01	2.05	50.10	23.5
16	180	0.33	0.22	-0.56	0.03	-0.06	0.01	0.90	50.10	54.8

(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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11.2 Accidents and Natural Phenomena

This section presents the results of analyses of the design basis and hypothetical accident conditions evaluated for the Universal Storage System. In addition to design basis accidents, this section addresses very low probability events, including natural phenomena, that might occur over the lifetime of the ISFSI, or hypothetical events that are postulated to occur because their consequences may result in the maximum potential impact on the immediate 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 accident analyses of this section, the bounding 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 accidents.

The results of analyses show that no credible potential accident exists that will result in a dose of ≥ 5 rem beyond the postulated controlled area. The Universal Storage System is demonstrated to have a substantial design margin of safety and to provide protection to the public and to occupational personnel during storage of spent nuclear fuel.

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11.2.1 Accident Pressurization

Accident pressurization is a hypothetical event that assumes the failure of all of the fuel rods contained within the Transportable Storage Canister (canister). No storage conditions are expected to lead to the rupture of all of the fuel rods.

Results of analysis of this event demonstrate that the canister is not significantly affected by the increase in internal pressure that results from the hypothetical rupture of all PWR or BWR fuel rods contained within the canister. Positive margins of safety exist throughout the canister.

11.2.1.1 Cause of Pressurization

The hypothetical failure of all of the fuel rods in a canister would release the fission and fill gases to the interior of the canister, resulting in the pressurization of the canister.

11.2.1.2 Detection of Accident Pressurization

The rupture of fuel rods within the canister is unlikely to be detected by any measurements or inspections that could be undertaken from the exterior of the canister or the concrete cask.

11.2.1.3 Analysis of Accident Pressurization

Analysis of this accident involves evaluation of the maximum canister internal pressure and the canister stress due to the maximum internal pressure. These evaluations are provided below.

Maximum Canister Accident Condition Internal Pressure

The analysis requires the calculation of the free volume of the canister, calculation of the releasable quantity of fill and fission gas in the fuel assemblies, BPRA gases, and the subsequent calculation of the pressure in the canister if these gases are added to the backfill helium pressure (initially at 1 atm) already present in the canister (Section 4.4.5). Canister pressures are determined for two accident scenarios, 100 percent fuel failure and a maximum temperature accident. The maximum temperature accident includes the fire accident and full vent blockage. While no design basis event results in a 100 percent fuel failure condition, the pressures from this condition are presented to form a complete licensing basis. The method employed in either of the accident analyses is identical to that employed in the normal condition evaluation of Section 4.4.5.

For the maximum temperature accident condition, the gas quantities are combined with the accident average gas temperatures of 505°F (PWR) and 465°F (BWR) to calculate conservative system pressures. Maximum pressures under the fire accident conditions are 6.14 psig (PWR) and 5.11 psig (BWR).

Canister pressures under the 100 percent fuel failure assumption are 59.2 psig (PWR) and 37.4 psig (BWR). Assemblies producing the maximum pressures are identical to those in the normal condition evaluation, i.e., B&W 17×17 Mark C in UMS® canister Class 2 for PWR assemblies and GE 7×7 (49 fuel rod) assembly in canister class 5 for BWR assemblies. Similar pressures result from the Westinghouse 17×17 standard fuel assembly in UMS® canister Class 1 and the GE 9×9 (79 fuel rod) assembly in canister Class 5.

Maximum Canister Stress Due to Internal Pressure

The stresses that result in the canister due to the internal pressure are evaluated using the ANSYS finite element model that envelops both PWR and BWR configurations as described in Section 3.4.4. The pressure used for the model is 65 psig, which bounds the results of 59.2 and 37.4 psig for the PWR and BWR configurations, respectively.

The resulting maximum canister stresses for accident pressure loads are summarized in Tables 11.2.1-1 and 11.2.1-2 for primary membrane and primary membrane plus bending stresses, respectively.

The resulting maximum canister stresses and margins of safety for combined normal handling (Tables 3.4.4.1-4 and 3.4.4.1-5) and maximum accident internal pressure (65 psig) are summarized in Tables 11.2.1-3 and 11.2.1-4 for primary membrane and primary membrane plus bending stresses, respectively.

The sectional stresses shown in Tables 11.2.1-1 through 11.2.1-4 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.

All margins of safety are positive. Consequently, there is no adverse consequence to the canister as a result of the combined normal handling and maximum accident internal pressure (65 psig).

11.2.1.4 Corrective Actions

No recovery or corrective actions are required for this hypothetical accident.

11.2.1.5 Radiological Impact

There are no dose consequences due to this accident.

Table 11.2.1-1 Canister Accident Internal Pressure (65 psig) Only Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	0.44	2.49	6.33	-0.17	-0.08	-0.91	6.19
2	4.24	-5.27	-4.12	0.71	-0.09	-0.90	9.71
3	-0.77	-8.07	1.82	0.68	0.16	1.82	10.91
4	-0.01	3.43	1.69	-0.30	0.00	0.00	3.49
5	-0.01	3.40	1.70	-0.30	0.00	0.00	3.45
6	0.00	3.40	1.70	-0.30	0.00	0.00	3.45
7	-0.01	3.40	1.70	-0.30	0.00	0.00	3.46
8	-0.01	2.28	1.69	-0.20	0.00	-0.04	2.33
9	0.16	0.90	1.25	-0.07	0.02	0.15	1.14
10	-0.55	0.60	0.84	-0.09	0.00	-0.15	1.42
11	0.71	0.41	-0.11	0.00	0.00	0.08	0.83
12	-0.29	0.17	-0.83	0.00	0.00	-0.27	1.11
13	-0.15	0.45	0.77	-0.06	0.03	0.06	0.94
14	1.05	1.05	-0.06	0.00	0.45	-0.07	1.44
15	-0.12	-0.12	-0.04	0.00	-0.01	0.00	0.08
16	0.09	0.09	0.00	0.00	-0.01	0.00	0.10

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

Table 11.2.1-2 Canister Accident Internal Pressure (65 psig) Only Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity
1	4.85	0.53	15.30	0.22	-0.09	-0.06	14.79
2	2.05	-13.36	-29.48	1.27	-0.15	-2.06	31.90
3	-3.08	2.85	41.20	-0.38	0.17	2.29	44.54
4	-0.02	3.45	1.64	-0.30	0.00	0.00	3.52
5	-0.02	3.44	1.70	-0.30	0.00	0.00	3.51
6	-0.02	3.44	1.70	-0.30	0.00	0.00	3.51
7	-0.02	3.44	1.70	-0.30	0.00	0.00	3.51
8	-0.03	2.31	1.89	-0.20	0.00	-0.04	2.37
9	0.18	1.32	2.67	-0.10	0.03	0.37	2.61
10	-0.41	1.34	3.21	-0.14	0.00	0.11	3.64
11	0.57	-0.13	-1.80	0.00	0.00	0.16	2.39
12	-0.78	-0.17	-1.52	0.00	0.00	-0.46	1.57
13	-1.11	0.07	0.32	-0.09	0.04	0.12	1.46
14	21.95	21.97	0.56	0.01	0.41	-0.09	21.43
15	-1.46	-1.46	-0.08	0.00	-0.01	0.00	1.38
16	0.75	0.75	0.02	0.00	-0.01	0.00	0.73

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

Table 11.2.1-3 Canister Normal Handling plus Accident Internal Pressure (65 psig) Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.55	3.19	8.13	-0.22	-0.10	-1.17	7.95	40.08	4.0
2	5.41	-6.96	-5.28	0.92	-0.11	-1.17	12.63	40.08	2.2
3	-0.97	-10.70	2.35	0.90	0.20	2.30	14.34	40.08	1.8
4	-0.01	3.44	2.20	0.30	0.00	0.00	3.50	38.77	10.1
5	-0.01	3.40	2.18	0.30	0.00	0.00	3.46	35.86	9.4
6	-0.01	3.40	2.13	0.30	-0.01	0.00	3.46	35.55	9.3
7	-0.01	3.40	2.04	0.30	-0.01	0.00	3.46	38.23	10.0
8	0.01	2.24	2.79	-0.20	-0.07	-0.04	2.80	40.08	13.3
9	0.18	1.27	2.68	-0.10	-0.13	0.19	2.54	40.08	14.8
10	-0.78	0.94	2.52	-0.16	-0.22	-0.08	3.36	40.08	10.9
11	0.13	1.12	0.79	-0.09	-0.11	-0.44	1.26	40.08	30.8
12	-0.32	0.19	-1.12	-0.10	-0.23	-0.42	1.57	40.08	24.5
13	0.12	1.40	0.43	-0.22	0.00	-0.47	1.68	40.08	22.9
14	1.35	1.35	-0.03	0.00	0.60	-0.09	1.84	40.08	20.8
15	-0.13	-0.13	-0.06	0.00	-0.01	0.00	0.07	40.08	547.4
16	0.10	0.11	-0.02	0.00	-0.01	0.00	0.13	40.08	299.0

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

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

Table 11.2.1-4 Canister Normal Handling plus Accident Internal Pressure (65 psig) Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	6.17	0.48	19.67	0.29	-0.11	-0.08	19.22	60.12	2.1
2	2.62	-17.34	-37.86	1.65	-0.19	-2.67	40.97	60.12	0.5
3	-3.93	3.38	53.13	-0.46	0.22	2.91	57.38	60.12	0.1
4	-0.03	3.52	2.14	0.31	0.00	-0.01	3.60	58.16	15.2
5	-0.03	3.57	2.23	0.32	-0.01	0.00	3.65	53.79	13.8
6	-0.03	3.62	2.20	0.32	-0.01	0.00	3.70	53.32	13.4
7	-0.02	3.60	2.11	0.31	0.00	0.00	3.67	57.35	14.6
8	-0.01	2.39	3.04	-0.22	-0.08	-0.04	3.07	60.12	18.6
9	0.12	1.68	4.32	-0.10	-0.18	0.37	4.28	60.12	13.0
10	-0.56	1.48	4.32	-0.16	-0.29	0.12	4.93	60.12	11.2
11	0.01	1.63	2.55	-0.13	-0.19	-0.91	3.16	60.12	18.0
12	-0.61	-0.09	-1.90	-0.11	-0.27	-0.61	2.13	60.12	27.2
13	0.09	1.30	1.27	-0.09	-0.12	-0.85	2.10	60.12	27.7
14	28.61	28.64	0.79	0.01	0.55	-0.12	27.88	60.12	1.2
15	-1.53	-1.53	-0.09	0.00	-0.01	0.00	1.44	60.12	40.7
16	0.75	0.74	0.00	0.00	-0.01	0.00	0.75	60.12	79.2

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

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

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11.2.2 Failure of All Fuel Rods With a Ground Level Breach of the Canister

Since no mechanistic failure of the canister occurs and since the canister is leaktight, this potential accident condition is not evaluated.

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11.2.3 Fresh Fuel Loading in the Canister

This section evaluates the effects of an inadvertent loading of up to 24 fresh, unburned PWR fuel assemblies or up to 56 fresh, unburned BWR fuel assemblies in a canister. There are no adverse effects on the canister due to this event since the criticality control features of the Universal Storage System ensure that the k_{eff} of the fuel is less than 0.95 for all loading conditions of fresh fuel.

11.2.3.1 Cause of Fresh Fuel Loading

The cause of this event is operator and/or procedural error. In-plant operational procedures and engineering and quality control programs are expected to preclude occurrence of this event. Nonetheless, it is evaluated here to demonstrate the adequacy of the canister design for accommodating fresh fuel without a resulting criticality event.

11.2.3.2 Detection of Fresh Fuel Loading

This accident is expected to be identified immediately by observation of the condition of the fuel installed in the canister or by a review of the fuel handling records.

11.2.3.3 Analysis of Fresh Fuel Loading

The criticality analysis presented in Chapter 6 assumes the loading of up to 24 design basis PWR or up to 56 design basis BWR fuel assemblies having no burnup. The maximum k_{eff} for the accident conditions remains below the upper safety limit.

The criticality control features of the Transportable Storage Canister and the basket ensure that the k_{eff} of the fuel is less than 0.95 for all loading conditions of fresh fuel. Therefore, there is no adverse impact on the Universal Storage System due to this event.

11.2.3.4 Corrective Actions

This event requires that the canister be unloaded when the incorrect fuel loading is identified. The cause for the error should be identified and procedural actions implemented to preclude recurrence.

11.2.3.5 Radiological Impact

There are no dose implications due to this event.

11.2.4 24-Inch Drop of Vertical Concrete Cask

This analysis evaluates a loaded Vertical Concrete Cask for a 24-inch drop onto a concrete storage pad. The cask containing the Transportable Storage Canister loaded with Class 5 BWR fuel is identified as the heaviest cask, and is conservatively used in the analysis as the bounding case. The results of the evaluation show that neither the concrete cask nor the Transportable Storage Canister experience significant adverse effects due to the 24-inch drop accident.

11.2.4.1 Cause of 24-Inch Cask Drop

The Vertical Concrete Cask may be lifted and moved using either an air pad system, which lifts the concrete cask from the bottom, or a mobile lifting frame, which lifts the concrete casks using lifting lugs in the top of the cask.

Using the air pad system, the concrete cask, containing a loaded canister, must be raised approximately 4 inches to enable installation of the inflatable air pads beneath it. The air pads use pressurized air to allow the cask to be moved across the surfaces of the transporter and the ISFSI pad to the designated position. The cask is raised using hydraulic jacks installed at jack-points in the cask's air inlets. The failure of one or more of the jacks or of the air pad system could result in a drop of the cask.

The concrete cask may be lifted and moved by a mobile lifting frame, which may be self-propelled or towed. The lifting frame uses hydraulic power to raise the cask approximately 24 inches using a lifting attachment that connects to the four cask lifting lugs. The failure of one or more of the lifting lugs, or the failure of the hydraulic pistons, could result in a drop of the cask.

11.2.4.2 Detection of 24-Inch Cask Drop

This event will be detected by the operators as it occurs.

11.2.4.3 Analysis of 24-Inch Cask Drop

A bottom end impact is assumed to occur normal to the concrete cask bottom surface, transmitting the maximum load to the concrete cask and the canister. The energy absorption is computed as the product of the compressive force acting on the concrete cask and its displacement. Conservatively assuming that the storage surface impacted is an infinitely rigid surface, the concrete cask body will crush until the impact energy is absorbed.

A compressive strength of 4,000 psi is used for the cask concrete. The evaluation conservatively ignores any energy absorption by the internal friction of the aggregate as crushing occurs.

The canister rests upon a base weldment designed to allow cooling of the canister. Following the initial impact, the inlet system will partially collapse, providing an energy absorption mechanism that somewhat reduces the deceleration force on the canister.

Evaluation of the Concrete Cask

In the 24-inch bottom drop of the concrete cask, the cylindrical portion of the concrete is in contact with the steel bottom plate that is a part of the base weldment. The plate is assumed to be part of an infinitely rigid storage pad. No credit is taken for the crush properties of the storage pad or the underlying soil layer. Therefore, energy absorbed by the crushing of the cylindrical concrete region of the concrete cask equals the product of the compressive strength of the concrete, the crush depth of the concrete, and the projected area of the concrete cylinder. Crushing of the concrete continues until the energy absorbed equals the potential energy of the cask at the initial drop height. The canister is not rigidly attached to the concrete cask, so it is not considered to contribute to the concrete crushing. The energy balance equation is:

$$w(h+\delta) = P_o A \delta,$$

where:

h = 24 in., the drop height,

δ = the crush depth of the concrete cask,

P_o = 4000 psi, the compressive strength of the concrete,

A = $\pi(R_1^2 - R_2^2)$ = 7,904 in^2, the projected area of the concrete shield wall,

w = 190,000 lbs (bounding concrete plus rebar)

It is assumed that the maximum force that can be exerted on the concrete cask is the compressive strength of the concrete multiplied by the area of the concrete being crushed. The concrete cask's steel shell will not experience any significant damage during a 24-inch drop. Therefore, its functionality will not be impaired due to the drop.

The crush distance computed from the energy balance equation is:

$$\delta = \frac{hw}{P_o A - w} = \frac{(24)(190{,}000)}{(4000)(7{,}904) - (190{,}000)} = 0.145 \text{ inch}$$

where, w = 190,000 lbs (the highest bounding weight is used to obtain the maximum deformation)

The resultant inlet deformation is 0.145 inch.

Evaluation of the Canister for a 24-inch Bottom End Drop

Upon a bottom end impact of the concrete cask, the canister produces a force on the base weldment located near the bottom of the cask (see Figure 11.2.4-1). The ring above the air inlets is expected to yield. To determine the resulting acceleration of the canister and deformation of the pedestal, a LS-DYNA analysis is used.

A half-symmetry model of the base weldment is built using the ANSYS preprocessor (see Figure 11.2.4-2). The model is constructed of 8-node brick and 4-node shell elements. Symmetry conditions are applied along the plane of symmetry (X-Z plane). Lumped mass elements located in the canister bottom plate represent the loaded canister. The impact plane is represented as a rigid plane, which is considered conservative, since the energy absorption due to the impact plane is neglected (infinitely rigid). To determine the maximum acceleration and deformations, impact analyses are solved using LS-DYNA program.

The weldment ring, weldment plate, and the inner cone (see Figure 11.2.4-1) materials are modeled using LS-DYNA's piece-wise linear plasticity model. This material model accepts stress–strain curves for different strain rates. These stress-strain curves were obtained from the Atlas of Stress-Strain Curves [44] and are shown in Figure 11.2.4-3. To ensure that maximum deformations and accelerations are determined, two analyses are performed. One analysis, which uses the static stress-strain curve, envelopes the maximum deformation of the pedestal. The second analysis employs the multiple stress-strain curves to account for different strain rates.

The maximum accelerations of the canister during the 24-inch bottom end impact are 45.0g and 44.5g for the variable strain rate material model and the static stress-strain curve, respectively. The resulting acceleration time histories of the bottom canister plate, which correspond to a filter frequency of 200 Hz, are shown in Figure 11.2.4-4 for the analysis using the static stress-strain curve and Figure 11.2.4-5 for the analysis corresponding to the series of stress-strain curves at different strain rates. These time histories indicate that the maximum accelerations do not occur at the beginning where the strain rate is maximum, but rather, at a time where the strain rate has a marginal effect on the accelerations. Therefore, the use of the multiple strain rate material model is considered to bound the accelerations imposed on the canister, since it considers the effect of strain rate on the stress-strain curves.

The filter frequency used in the LS-DYNA evaluation is determined by performing two modal analyses of a quarter symmetry model of the base weldment. Symmetry boundary conditions are applied on the planes of symmetry of the model for both analyses. The second analysis considers a boundary condition that is the center node of the base weldment bottom plate, restrained in the vertical direction. These analyses result in a modal frequency of 173 Hz and 188 Hz, respectively. Therefore, a filter frequency of 200 Hz is selected.

Results of the LS-DYNA analysis show that the maximum deformation of the base weldment is about 1 inch. This deformation is small when compared to the 12-inch height of the air inlet. Therefore, a 24-inch drop of the concrete cask does not result in a blockage of the air inlets.

The dynamic response of the canister and basket on impact is amplified by the most flexible components of the system. In the case of the canister and basket, the basket support disk bounds this response. To account for the transient response of the support disk, a dynamic load factor (DLF) for the support disk is computed for the inertia loading developed during the deceleration of the canister bottom plate. The DLF is determined using quarter symmetry models of the PWR and BWR disks as shown in Figures 11.2.4-6 and 11.2.4-7, respectively. These models are generated using ANSYS, Revision 5.5.

To support the disks in the models, restraints are applied at the basket tie-rod locations. For each tie-rod location, a single node is restrained in the vertical direction allowing the support disks to vibrate freely when the accelerations are applied at the tie rod locations. A transient analysis using ANSYS, Revision 5.5 is performed which uses the acceleration time histories computed from the LS-DYNA analyses. The time history corresponding to the stress–strain curves at different strain rates is used. This case is considered bounding since the maximum acceleration occurs when the rate dependent stress-strain curves are used.

The DLF is determined to be the maximum deflection of the disk (which occurs at the center of the disk) divided by the static displacement (The static analysis used the maximum acceleration determined from the LS-DYNA analysis). The DLF for the PWR and the BWR are determined to be 1.01 and 1.29, respectively.

Therefore, multiplying the calculated accelerations by the DLF's results in effective accelerations of 45.5g and 57.4g for the PWR and BWR canisters, respectively. These values are enveloped by the 60g acceleration employed in the stress evaluation of the end impact of the canister and support disks. These accelerations are considered to be bounding since they incorporate the effect of the strain rate on the plastic behavior of the pedestal and ignore any energy absorption by the impact plane.

Canister Stress Evaluation

The Transportable Storage Canister stress evaluation for the concrete cask 24-inch bottom end drop accident is performed using a load of 60g. This evaluation bounds the 57.4g load that is calculated for the 24-inch bottom end drop event determined above. This canister evaluation is performed using the ANSYS finite element program. The canister finite element model is shown in Figure 11.2.4-8. The construction and details of the finite element model are described in Section 3.4.4.1.1. Stress evaluations are performed with and without an internal pressure of 15 psig.

The principal components of the canister are the canister shell, including the bottom plate, the fuel basket, the shield lid, and the structural lid. The geometry and materials of construction of the canister, baskets, and lids are described in Section 1.2. The structural design criteria for the canister are contained in the ASME Code, Section III, Subsection NB. This analysis shows that the structural components of the canister (shell, bottom plate, and structural lid) satisfy the allowable stress intensity limits.

The results of the bounding canister analysis for the 60g bottom end impact loading are presented in Tables 11.2.4-1 through 11.2.4-4. These results are for the load case that includes a canister internal pressure of 15 psig, since that case results in the minimum margin of safety.

The minimum margin of safety at each section of the canister is presented by denoting the circumferential angle at which the minimum margin of safety occurs. A cross-section of the canister showing the section locations is presented in Figure 11.2.4-9. Stresses are evaluated at 9° increments around the circumference of the canister for each of the locations shown. The minimum margin of safety is denoted by an angular location at each section.

For the canister to structural lid weld (Section 13, Figure 11.2.4-9), base metal properties are used to define the allowable stress limits since the tensile properties of the weld filler metal are greater than those of the base metal. The allowable stress at Section 13 is multiplied by a stress reduction factor of 0.8 in accordance with NRC Interim Staff Guidance (ISG) No. 15.

The allowable stresses presented in Tables 11.2.4-1 through 11.2.4-4, and in Tables 11.2.4-6 and 11.2.4-7, are for Type 304L stainless steel. Because the shield lid is constructed of Type 304 stainless steel, which possesses higher allowable stresses, a conservative evaluation results. The allowable stresses are evaluated at 380°F. A review of the thermal analyses shows that the maximum temperature of the canister is 351°F (Table 4.1-4) for PWR fuel and 376°F (Table 4.1-5) for BWR fuel, which occurs in the center portion of the canister wall (Sections 5 and 6).

Canister Buckling Evaluation

Code Case N-284-1 of the ASME Boiler and Pressure Vessel Code is used to analyze the canister for the 60g bottom end impact. The evaluation requirements of Regulatory Guide 7.6, Paragraph C.5, are shown to be satisfied by the results of the buckling interaction equation calculations.

The internal stress field that controls the buckling of a cylindrical shell consists of the longitudinal (axial) membrane, circumferential (hoop) membrane, and in-plane shear stresses. These stresses may exist singly or in combination, depending on the applied loading. The buckling evaluation is performed without the internal 15 psig pressure, since this results in the minimum margin of safety.

The primary membrane stress results for the 60g bottom impact with no internal pressure are presented in Table 11.2.4-4.

The stress results from the ANSYS analyses are screened for the maximum values of the longitudinal compression, circumferential compression, and in-plane shear stresses for the 60g bottom end impact. For each loading case, the largest of each of the three stress components, regardless of location within the canister shell are combined.

The maximum stress components used in the evaluation and the resulting buckling interaction equation ratios are provided in Table 11.2.4-5. The results show that all interaction equation ratios are less than 1.0. Therefore, the buckling criteria of Code Case N-284-1 are satisfied, demonstrating that buckling of the canister does not occur.

Basket Stress Evaluation

Stresses in the support disks and weldments are calculated by applying the accident loads to the ANSYS models described in Sections 3.4.4.1.8 and 3.4.4.1.9. An inertial load of 60g is conservatively applied to the support disks and weldments in the axial (out of plane) direction. To evaluate the most critical regions of the support disks, a series of cross sections are 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. The stress evaluations for the support disk and weldments are performed according to ASME Code, Section III, Subsection NG. For accident conditions, Level D allowable stresses are used: the allowable stress is $0.7S_u$ and S_u for P_m and P_m+P_b stress categories, respectively. The stress evaluation results are presented in Tables 11.2.4-6 and 11.2.4-7 for the PWR and BWR support disks, respectively. The tables list the 40 highest P_m+P_b stress intensities. The minimum margins of safety are +1.90 and +0.60 for PWR and BWR disks, respectively. The stress results for the PWR and BWR weldments are shown in Table 11.2.4-3. The minimum margin of safety is +1.31 and +0.26 for the PWR and BWR weldments, respectively. Note that the P_m stresses for the disks and weldments are essentially zero, since there are no loads in the plane of the support disk or weldment for a bottom end impact.

Fuel Basket Tie Rod Evaluation

The tie rods serve basket assembly purposes and are not part of the load path for the conditions evaluated. The tie rods are loaded during basket assembly by a 50 ± 10 ft-lbs torque applied to the tie rod end nut. The tensile pre-load on the tie rod, P_B, is [41]:

$$T = P_B\,(0.159\,L + 1.156\,\mu\,d)$$

where:

$T = 60$ ft-lb
$L = 1/8$
$\mu = 0.15$
$d = 1.625$ in.

Solving for P_B:

$$P_B = 2{,}387 \text{ lbs. per rod}$$

The maximum tensile stress in the tie rod occurs while the basket is being lifted for installation in the canister. The BWR basket configuration is limiting because it has six tie rods, compared to eight tie rods in the PWR basket, and weighs more than the PWR basket. The load on each BWR basket tie rod is:

$$P = 2{,}387 + \frac{1.1 \times 17{,}551}{6} = 5{,}605 \text{ lbs. use } 6{,}000 \text{ lbs.}$$

where the weight of the BWR basket is 17,551 pounds.

The maximum tensile stress, S, at room temperature (70°F) is:

$$S = \frac{6{,}000}{\pi \times 0.25 \times 1.625^2} = 2{,}893 \text{ psi}$$

Therefore, the margin of safety is:

$$MS = \frac{20{,}000}{2{,}893} - 1 = +\text{Large}$$

This result bounds that for the PWR basket configuration. The tie rod is not loaded in drop events; therefore, no additional analysis of the tie rod is required.

PWR and BWR Tie Rod Spacer Analysis

The PWR and BWR basket support disks and heat transfer disks are connected by tie rods (8 for PWR and 6 for BWR) and located by spacers to maintain the disk spacing. The PWR and BWR spacers are constructed from ASME SA479 Type 304 stainless steel or ASME SA312 Type 304 stainless steel. The difference in using the two materials is the cross-sectional area of the spacers.

The geometry of the spacers is:

For SA479 stainless steel:

Spacer:	Outside Diameter	= 3.00 in.
	Inside Diameter	= 1.75 in.
Split Spacer:	Outside Diameter	= 2.50 in. (Machined down section)
	Inside Diameter	= 1.75 in.
	Outside Diameter	= 3.00 in.

For the full spacer, the cross-section area is 4.66 inches^2, and for the split spacer, the cross-section area is 2.5 inches^2.

For SA312 stainless steel:

Spacer:	Outside Diameter	= 2.875 in.
	Inside Diameter	= 1.771 in.
Split Spacer:	Outside Diameter	= 2.50 in. (Machined down section)
	Inside Diameter	= 1.771 in.
	Outside Diameter	= 2.875 in.

For the full spacer, the cross-section area is 4.03 inches^2, and for the split spacer, the cross-section area is 2.45 inches^2.

During a 24-inch drop, the weight of the support disks, top weldment, heat transfer disks, spacers, and end nuts are supported by the spacers on the tie rods. A conservative deceleration of 60g is applied to the spacers. The bounding spacer load occurs at the bottom weldment of the BWR basket. The bounding split-spacer load occurs at the 10th support disk (from bottom of the basket) of the BWR basket.

The applied load on the BWR bottom spacer is 126,000 lbs.

$$P = 60(P_S) + P_T = 125{,}147 \text{ lbs. use } 126{,}000 \text{ lbs.}$$

where:

P_T = 2387 lbs	torque pre-load
P_s = 2046 lbs	load on the spacer due to basket structure above the spacer location

$$P_s = \frac{17{,}551 - 623 - 4651}{6} = 2{,}046 \text{ lbs}$$

where:

17,551 lb.	BWR basket weight
623 lb.	BWR bottom weldment weight
4,651 lb.	BWR fuel tube weight

The applied load on the BWR split spacer is 102,000 lbs.

$$P = 60(P_S) + P_T = 101{,}747 \text{ lbs. use } 102{,}000 \text{ lbs.}$$

where:

P_T = 2387 lbs torque pre-load
P_s = 1656 lbs load on the spacer due to basket structure above the spacer location

$$P_s = \frac{17{,}551 - 623 - 4{,}651 - 10\times 204 - 60\times 5}{6} = 1{,}656 \text{ lbs}$$

17,551	lbs	BWR basket weight
623	lbs	BWR bottom weldment weight
4,651	lbs	BWR fuel tube weight
204	lbs	BWR support disk weight (Qty = 10)
5	lbs	BWR full spacer weight (Qty = 60)

The margins of safety for the spacers are:

	Applied Load (lbs)	Cross-sectional area (in^2)	Stress (psi)	Temperature (°F)	Allowable Stress (psi)	Margin of Safety
Spacer						
SA479	126,000	4.66	27,039	250	47,950	0.77
SA312	126,000	4.03	31,266	250	47,950	0.53
Split Spacer						
SA479	102,000	2.50	40,800	350	45,640	0.12
SA312	102,000	2.45	41,633	350	45,640	0.10

The temperatures used bound the analysis locations for all storage conditions. The actual temperatures at these locations for storage for the BWR spacer at the bottom weldment are 118°F (minimum bottom weldment temperature), and 329°F (minimum temperature of 10^{th} support disk) for the split spacer. The 10^{th} support disk is counted from bottom weldment.

Fuel Tube Analysis

During the postulated 24-inch end drop of the concrete cask, fuel assemblies are supported by the canister bottom plate. The fuel assembly weight is not carried by the fuel tubes in the end drop. Therefore, evaluation of the fuel tube is performed considering the weight of the fuel tube, the canister deceleration and the minimum fuel tube cross-section. The minimum cross-section is located at the contact point of the fuel tube with the basket bottom weldment. The PWR fuel tube analysis is bounding because its weight (153 pounds/tube) is approximately twice that of the BWR fuel tube (83 pounds/tube). The minimum cross-section area of the PWR fuel tube is:

$$A = \text{(thickness)(mean perimeter)}$$

$$A = (0.048 \text{ in.})(8.80 \text{ in.} + 0.048 \text{ in.})(4) = 1.69 \text{ in}^2$$

The maximum compressive and bearing stress in the fuel tube is:

$$S_b = \frac{(60g)(153\text{ lbs})}{1.69 \text{ in}^2} = 5{,}432 \text{ psi}$$

The Type 304 stainless steel yield strength is 17,300 psi at a conservatively high temperature of 750°F. The margin of safety is:

$$MS = \frac{S_y}{S_b} - 1 = \frac{17{,}300 \text{ psi}}{5{,}432 \text{ psi}} - 1 = +2.18 \text{ at } 750°F$$

Summary of Results

Evaluation of the UMS cask and canister during a 24-inch drop accident shows that the resulting maximum acceleration of the canister is 57.4g. The acceleration determined for the canister during the 24-inch drop is less than its design allowable g-load and, therefore, is considered bounded. This accident condition does not lead to a reduction in the cask's shielding effectiveness. The base weldment, which includes the air inlets, is crushed approximately 1-inch as the result of the 24-inch drop. The effect of the reduction of the inlet area by the drop is to reduce cooling airflow. This condition is bounded by the consequences of the loss of one-half of the air inlets evaluated in Section 11.1.2.

11.2.4.4 Corrective Actions

Although the concrete cask remains functional following this event and no immediate recovery actions are required, the canister should be moved to a new concrete cask as soon as one is available. The damaged cask should be inspected for stability, and repaired as required prior to continued use.

11.2.4.5 Radiological Impact

There are no radiological consequences for this accident.

Figure 11.2.4-1 Concrete Cask Base Weldment

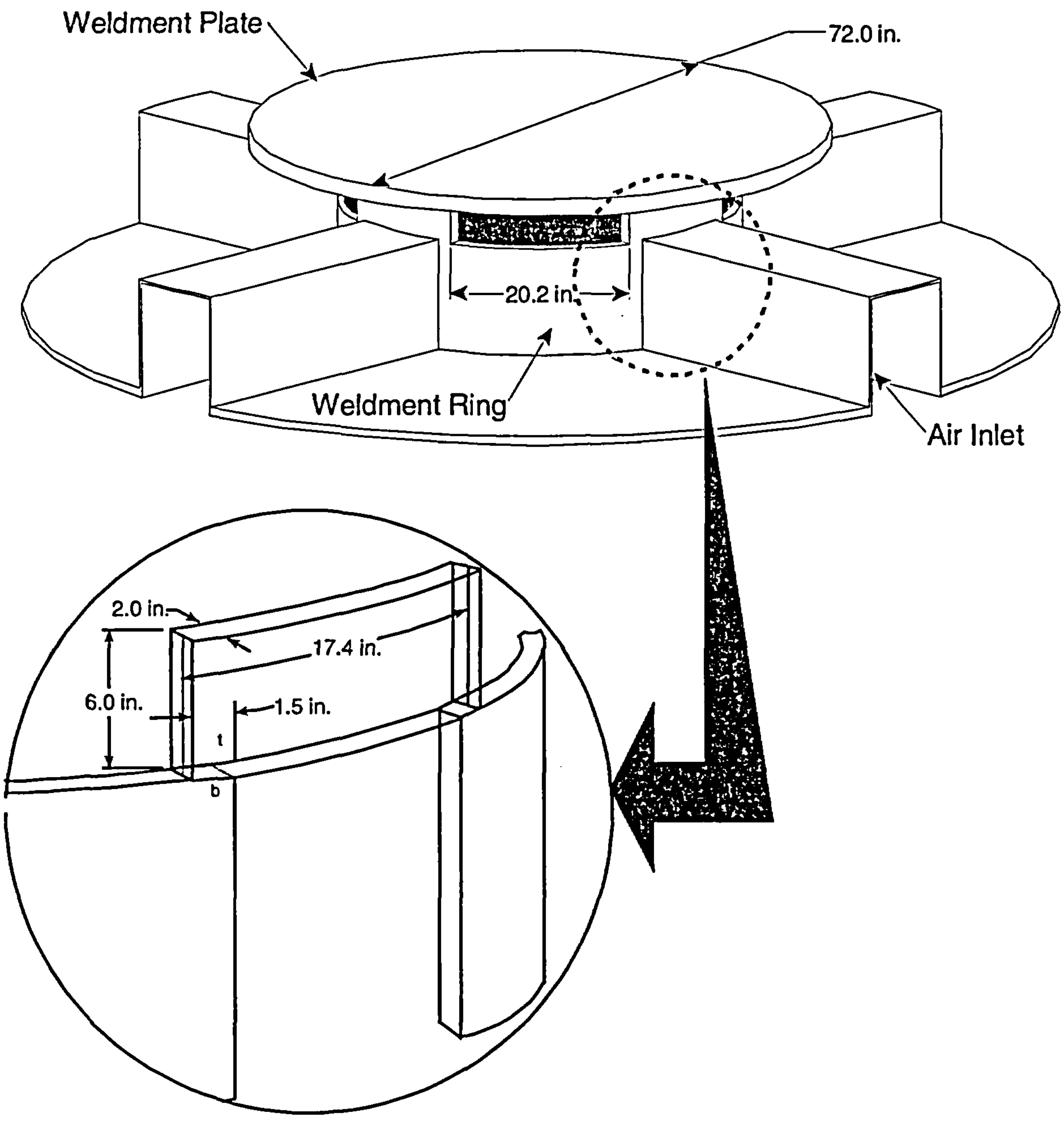


Figure 11.2.4-2 Concrete Cask Base Weldment Finite Element Model

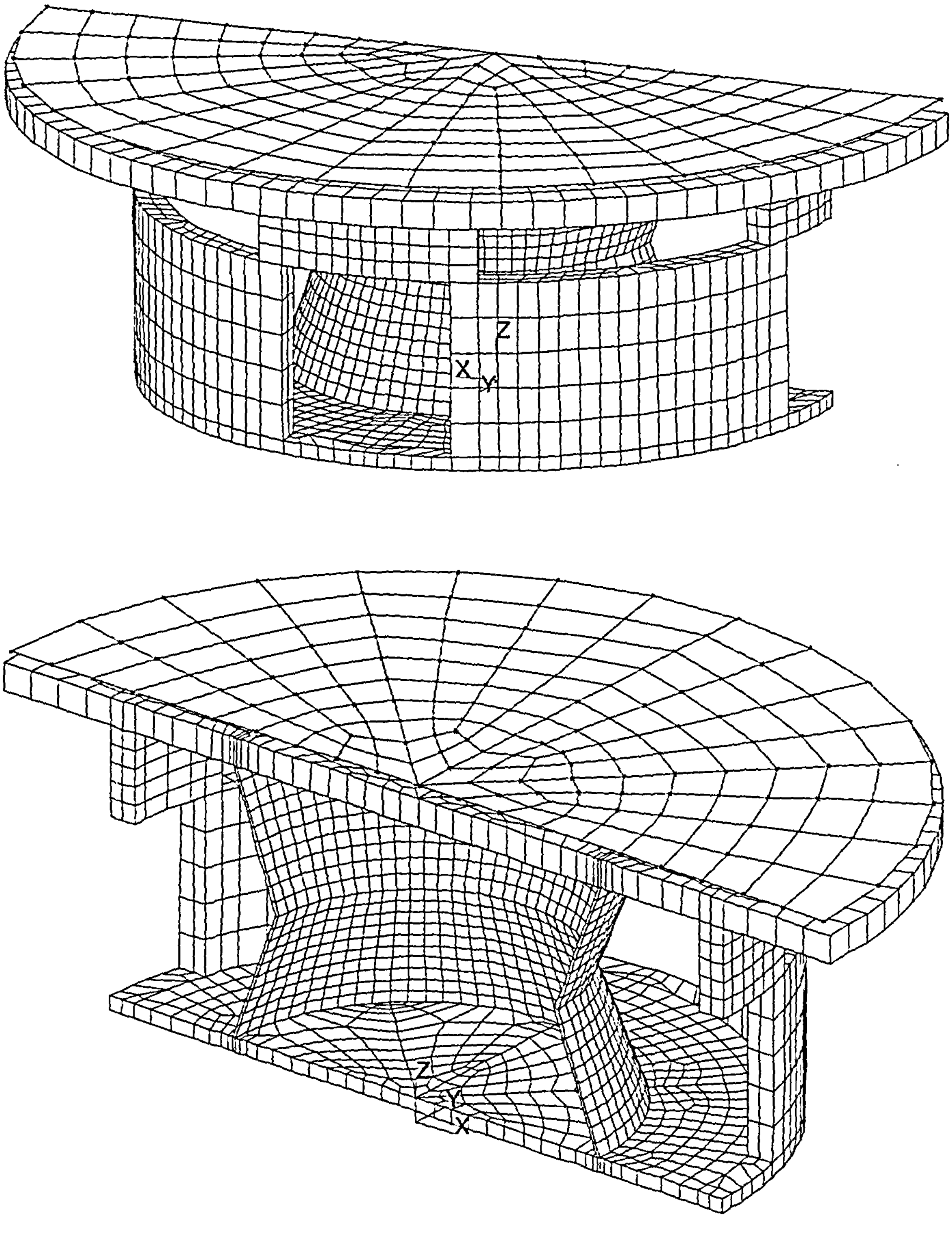


Figure 11.2.4-3 Strain Rate Dependent Stress-Strain Curves for Concrete Cask Base Weldment Structural Steel

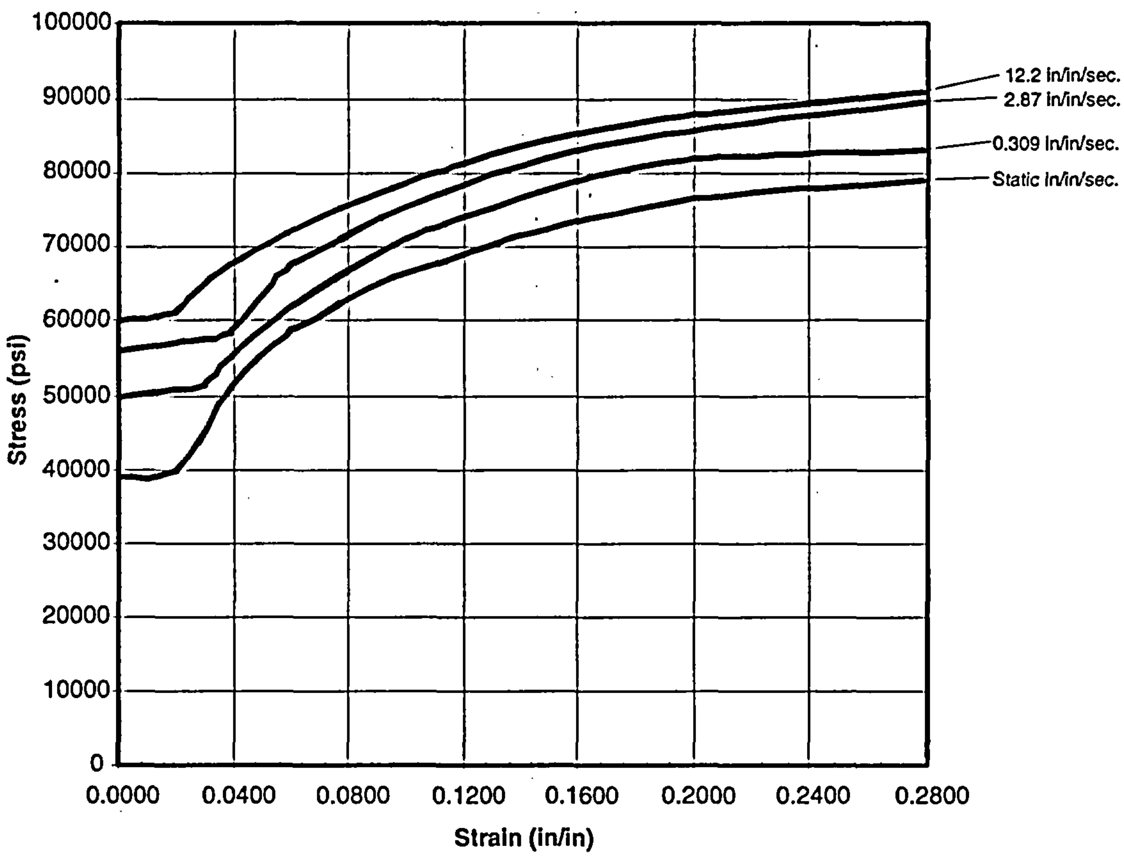


Figure 11.2.4-4 Acceleration Time-History of the Canister Bottom During the Concrete Cask 24-Inch Drop Accident With Static Strain Properties

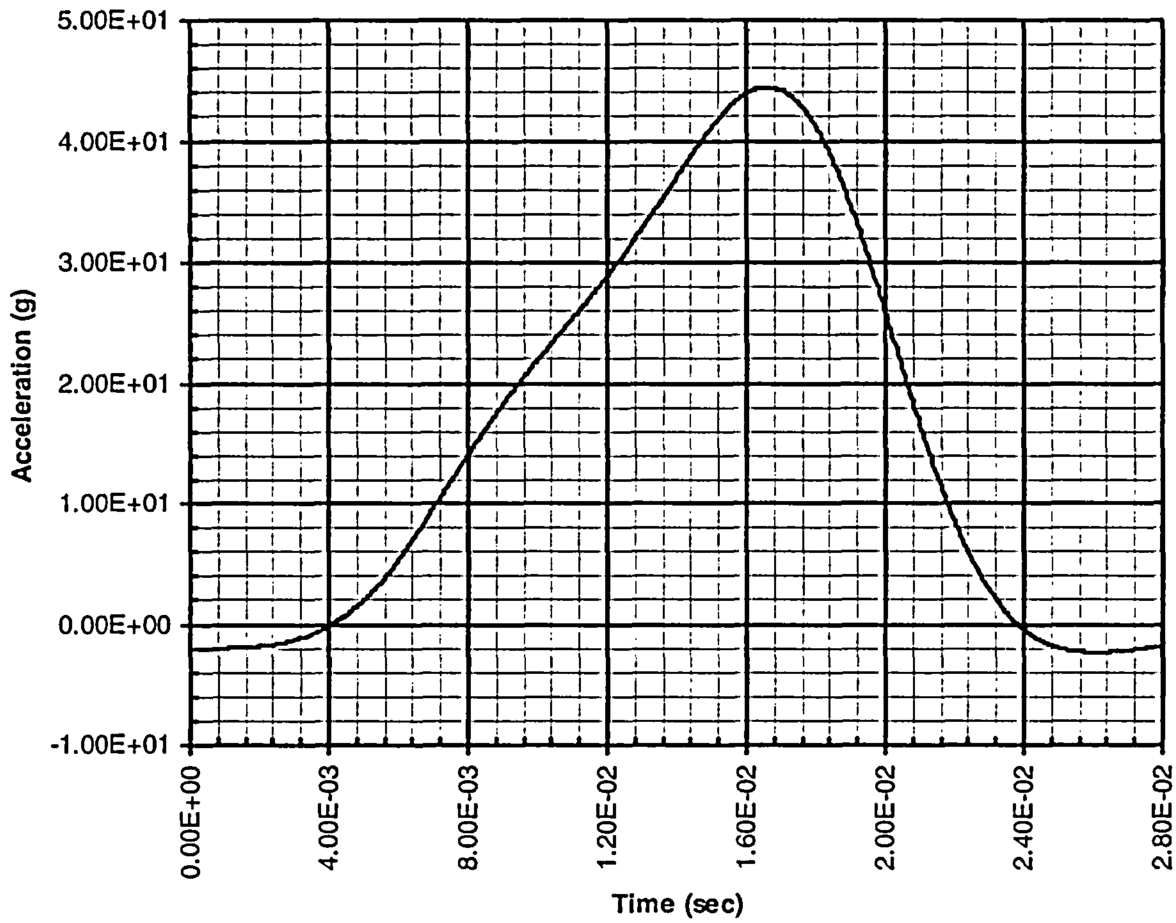


Figure 11.2.4-5 Acceleration Time-History of the Canister Bottom During the Concrete Cask 24-Inch Drop Accident With Strain Rate Dependent Properties

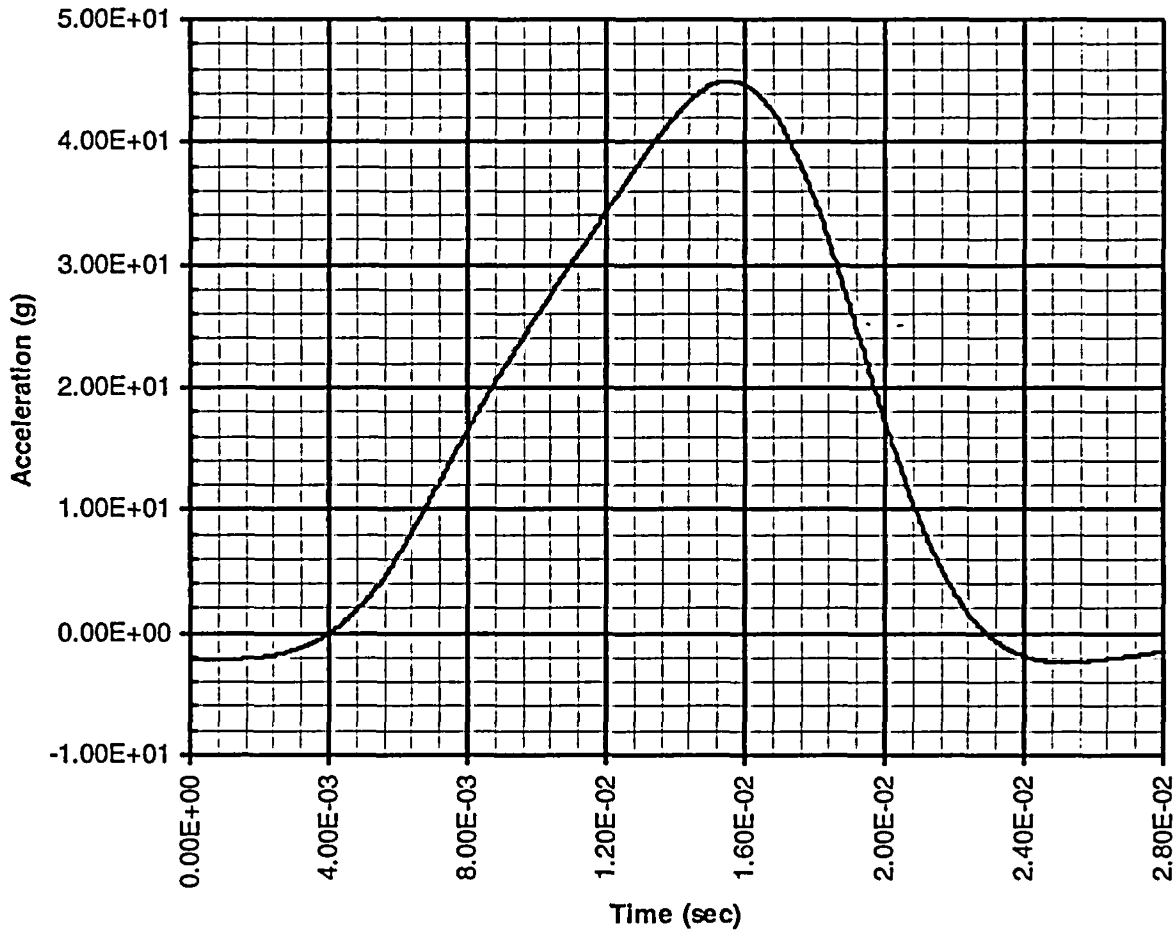


Figure 11.2.4-6 Quarter Model of the PWR Basket Support Disk

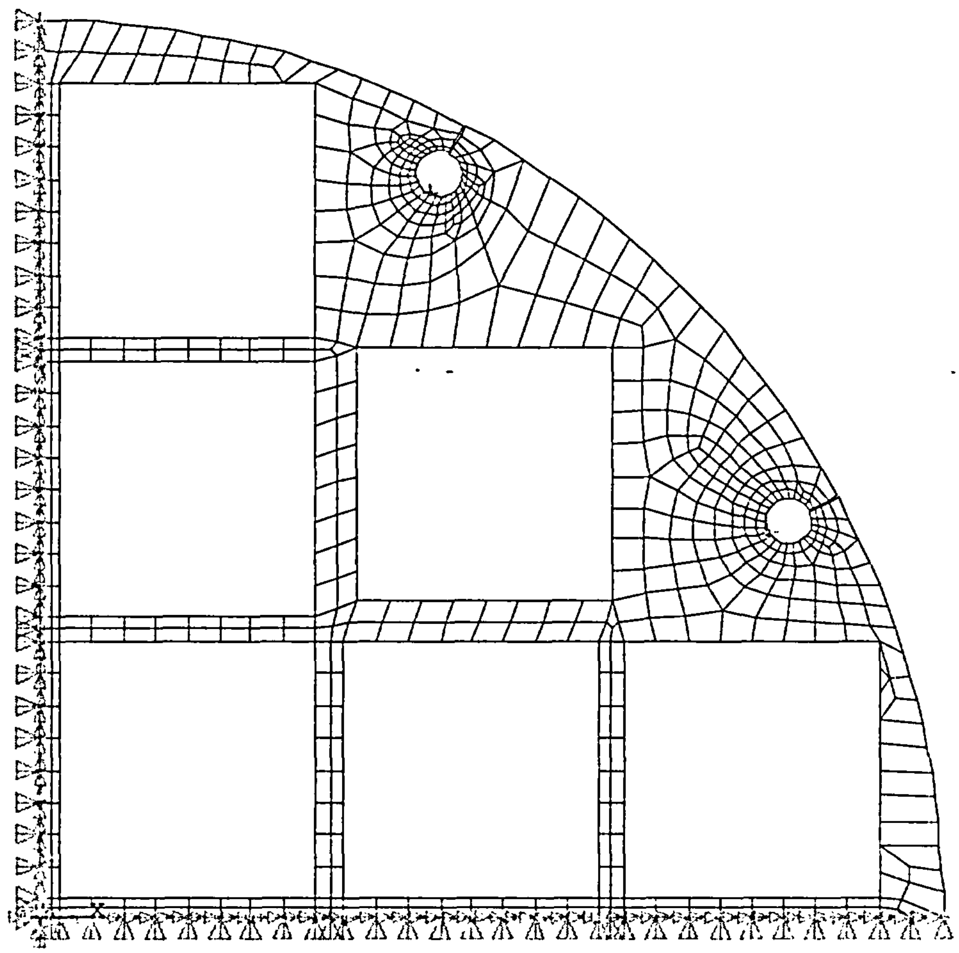


Figure 11.2.4-7 Quarter Model of the BWR Basket Support Disk

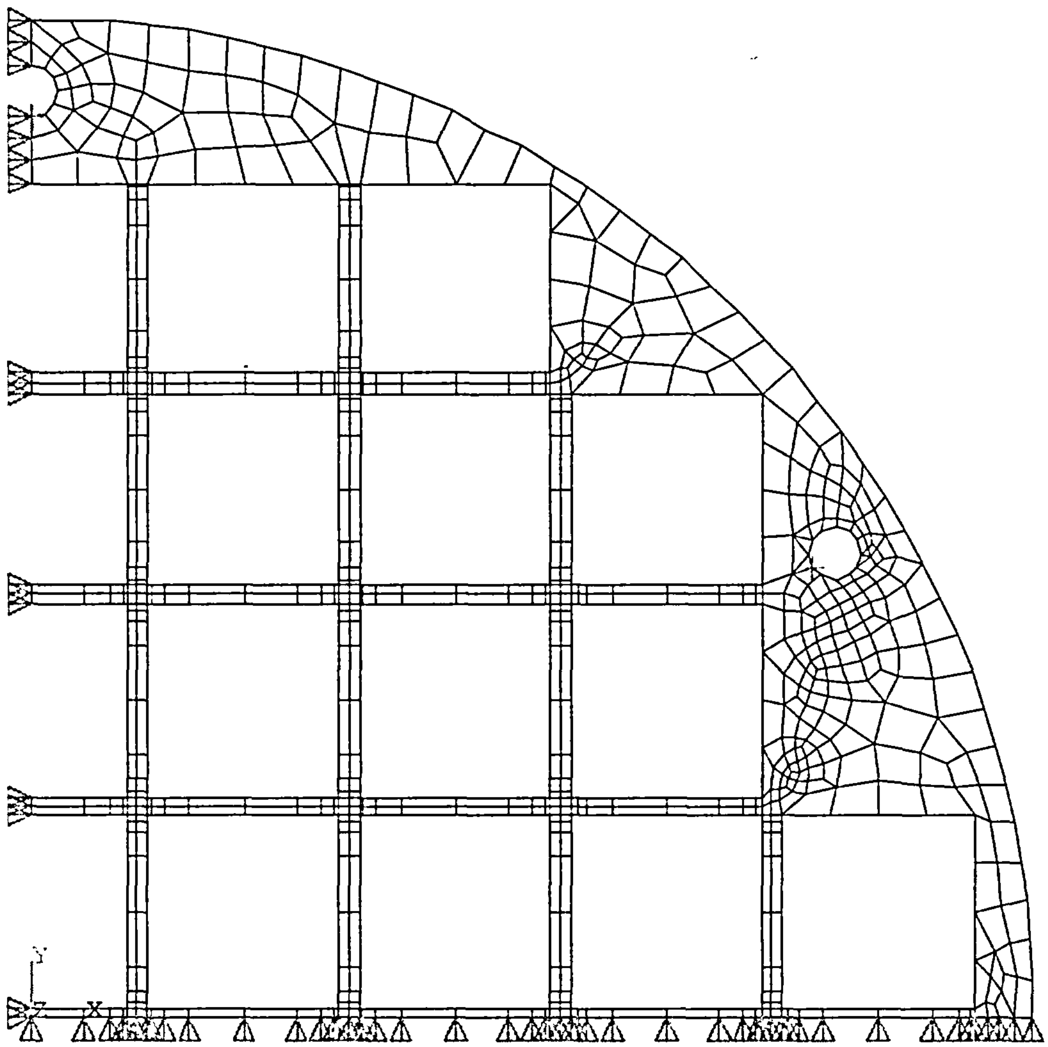


Figure 11.2.4-8 Canister Finite Element Model for 60g Bottom End Impact

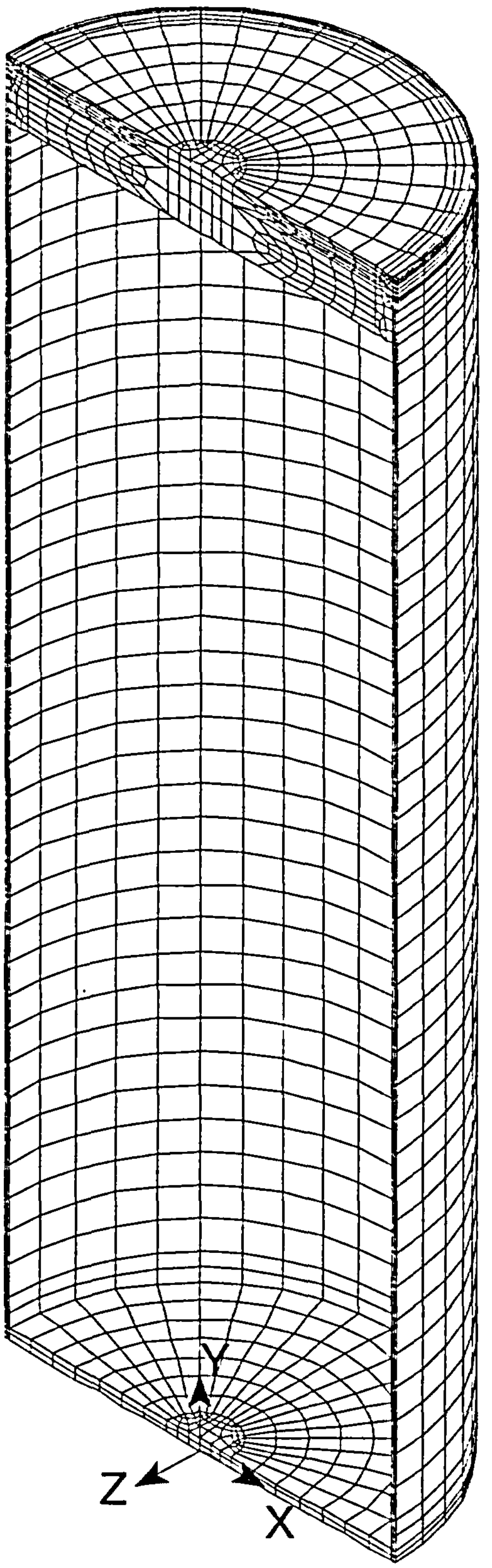
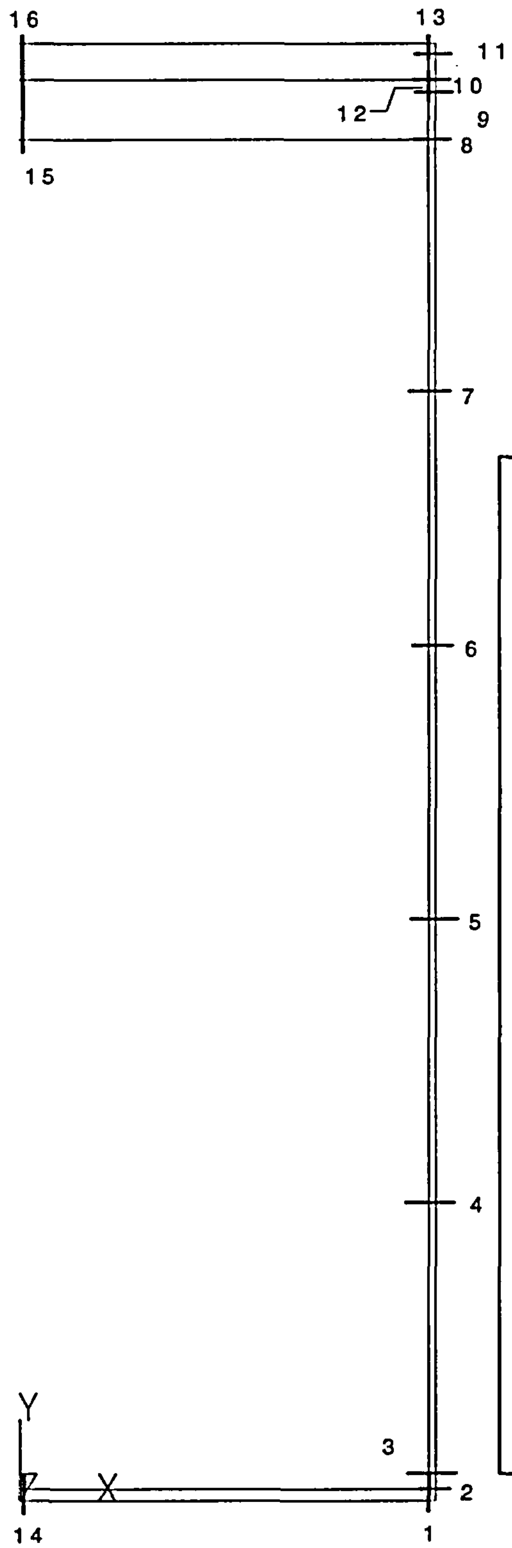
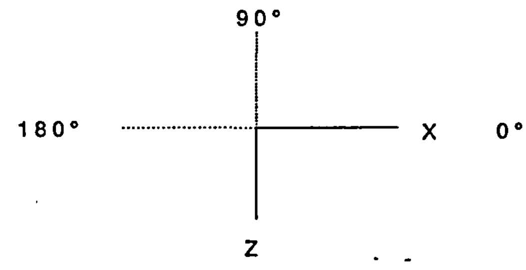


Figure 11.2.4-9 Identification of the Canister Sections for the Evaluation of Canister Stresses due to a 60g Bottom End Impact





Top View of Axis

PWR Section Coordinates at Z = 0

Axial Section	Node 1 x	Node 1 y	Node 2 x	Node 2 y
1	32.905	0.0	32.905	1.75
2	32.905	1.75	33.53	1.75
3	32.905	2.25	33.53	2.25
4	32.905	34.15	33.53	34.15
5	32.905	66.55	33.53	66.55
6	32.905	98.95	33.53	98.95
7	32.905	131.35	33.53	131.35
8	32.905	163.75	33.53	163.75
9	32.905	170.25	33.53	170.25
10	32.905	170.75	33.53	170.75
11	32.905	172.87	33.53	172.87
12	32.905	170.25	32.905	170.75
13	32.905	172.87	32.905	173.75
14	0.1	0.0	0.1	1.75
15	0.1	163.75	0.1	170.73
16	0.1	170.75	0.1	173.75

BWR Section Coordinates at Z = 0

Axial Section	Node 1 x	Node 1 y	Node 2 x	Node 2 y
1	32.905	0.0	32.905	1.75
2	32.905	1.75	33.53	1.75
3	32.905	2.25	33.53	2.25
4	32.905	37.79	33.53	37.79
5	32.905	73.83	33.53	73.83
6	32.905	109.87	33.53	109.87
7	32.905	145.91	33.53	145.91
8	32.905	181.95	33.53	181.95
9	32.905	188.45	33.53	188.45
10	32.905	188.95	33.53	188.95
11	32.905	191.07	33.53	191.07
12	32.905	188.45	32.905	188.95
13	32.905	191.07	32.905	191.95
14	0.1	0.0	0.1	1.75
15	0.1	181.95	0.1	188.73
16	0.1	188.97	0.1	191.95

Table 11.2.4-1 Canister P_m Stresses During a 60g Bottom Impact (15 psig Internal Pressure)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.0	-0.5	-2.9	0.0	-0.1	-0.3	3.0	40.1	12.4
2	0.8	-1.1	-6.2	0.2	0.0	-0.3	7.0	40.1	4.7
3	-0.2	-1.4	-7.2	0.1	0.0	0.2	7.0	40.1	4.7
4	0.0	0.8	-6.6	-0.1	0.0	0.0	7.4	38.8	4.2
5	0.0	0.8	-6.1	-0.1	0.0	0.0	6.9	35.9	4.2
6	0.0	0.8	-5.5	-0.1	0.0	0.0	6.3	35.6	4.7
7	0.0	0.8	-4.9	-0.1	0.0	0.0	5.7	38.2	5.7
8	0.1	0.8	-3.9	-0.1	0.0	0.1	4.7	40.1	7.5
9	-0.7	-2.0	-2.0	0.0	0.0	-0.5	1.6	40.1	24.9
10	1.5	-1.2	-1.3	0.2	0.0	0.2	2.8	40.1	13.2
11	-1.7	-0.9	0.5	0.0	0.0	-0.3	2.2	40.1	17.1
12	0.7	-0.6	1.6	0.1	-0.1	0.4	2.3	40.1	16.4
13	0.5	-1.0	-1.9	0.1	-0.1	-0.3	2.4	32.1[3]	12.4
14	0.1	0.1	-1.0	0.0	0.0	0.0	1.2	40.1	34.0
15	0.3	0.3	0.0	0.0	0.0	0.0	0.3	40.1	134.3
16	-0.2	-0.2	0.0	0.0	0.0	0.0	0.2	40.1	215.9

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level D is used for material allowable stresses.
3. Allowable stress includes a stress reduction factor for the weld: 0.8 × allowable stress.

Table 11.2.4-2 Canister $P_m + P_b$ Stresses During a 60g Bottom Impact (15 psig Internal Pressure)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.7	-0.3	-3.1	0.1	-0.1	-0.4	3.9	60.1	14.3
2	0.4	-2.0	-8.9	0.2	0.0	-0.2	9.4	60.1	5.4
3	-0.1	-1.7	-8.2	0.1	0.0	0.2	8.2	60.1	6.4
4	0.0	0.8	-6.6	-0.1	0.0	0.0	7.4	58.2	6.8
5	0.0	0.8	-6.1	-0.1	0.0	0.0	6.9	53.8	6.9
6	0.0	0.8	-5.5	-0.1	0.0	0.0	6.3	53.3	7.5
7	0.0	0.8	-4.9	-0.1	0.0	0.0	5.7	57.4	9.0
8	0.2	0.6	-4.9	0.0	0.0	0.2	5.5	60.1	9.9
9	-0.5	-2.8	-4.8	0.0	0.0	-0.8	4.6	60.1	12.2
10	0.8	-2.6	-5.6	0.0	0.0	-0.4	6.4	60.1	8.3
11	-1.3	0.4	4.5	-0.1	0.0	-0.4	5.8	60.1	9.3
12	2.5	0.3	2.8	0.2	0.0	0.9	3.2	60.1	17.6
13	2.9	-0.1	-0.8	0.2	-0.1	-0.3	3.8	60.1[3]	11.7
14	0.1	0.1	-1.0	0.0	0.0	0.0	1.2	60.1	51.5
15	3.6	3.6	0.0	0.0	0.0	0.0	3.6	60.1	15.8
16	-1.8	-1.8	-0.1	0.0	0.0	0.0	1.8	60.1	32.8

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level D is used for material allowable stresses.
3. Allowable stress includes a stress reduction factor for the weld: 0.8 × allowable stress.

Table 11.2.4-3 Summary of Maximum Stresses for PWR and BWR Basket Weldments During a 60g Bottom Impact

Case	Stress Category	Maximum Stress Intensity[1] (ksi)	Allowable Stress[2] (ksi)	Margin of Safety
PWR Top Weldment	$P_m + P_b$	27.5	63.5	1.31
PWR Bottom Weldment	$P_m + P_b$	12.0	68.5	+Large
BWR Top Weldment	$P_m + P_b$	34.1	64.0	0.88
BWR Bottom Weldment	$P_m + P_b$	51.9	65.2	0.26

1. Nodal stresses from the finite element analysis results are used.
2. Allowable stresses are conservatively determined at the maximum temperatures of the weldments.

Table 11.2.4-4 Canister P_m Stresses During a 60g Bottom Impact (No Internal Pressure)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	0.0	-0.7	-2.5	0.1	-0.1	-0.4	2.6	40.1	14.6
2	0.8	-1.4	-6.2	0.2	0.0	-0.4	7.1	40.1	4.7
3	-0.2	-1.8	-7.6	0.2	0.0	0.1	7.4	40.1	4.4
4	0.0	0.0	-7.0	0.0	0.0	0.0	7.0	38.8	4.5
5	0.0	0.0	-6.5	0.0	0.0	0.0	6.5	35.9	4.6
6	0.0	0.0	-5.9	0.0	0.0	0.0	5.9	35.6	5.0
7	0.0	0.0	-5.3	0.0	0.0	0.0	5.3	38.2	6.2
8	0.1	0.4	-4.2	0.0	0.0	0.1	4.6	40.1	7.8
9	-0.8	-2.2	-2.1	0.0	0.0	-0.5	1.7	40.1	23.3
10	1.7	-1.3	-1.4	0.2	0.0	0.2	3.1	40.1	12.1
11	-1.8	-0.9	0.5	0.0	0.0	-0.3	2.4	40.1	15.5
12	0.8	-0.6	1.7	0.1	-0.1	0.4	2.5	40.1	15.2
13	0.5	-1.1	-2.0	0.2	-0.1	-0.3	2.6	32.1[3]	11.3
14	0.1	0.1	0.0	0.0	0.0	0.0	0.1	40.1	351.2
15	0.3	0.3	0.0	0.0	0.0	0.0	0.3	40.1	126.8
16	-0.2	-0.2	0.0	0.0	0.0	0.0	0.2	40.1	197.0

1. See Figure 3.4.4.1-4 for definition of locations of stress sections.
2. ASME Code Service Level D is used for material allowable stresses.
3. Allowable stress includes a stress reduction factor for the weld: 0.8 × allowable stress.

Table 11.2.4-5 Canister Buckling Evaluation Results for 60g Bottom End Impact

	Canister Shell[2]
Longitudinal (Axial) Stress[1] S_Z (psi)	9,000
Circumferential (Hoop) Stress[1] S_Y (psi)	3,000
In-Plane Shear Stress S_{YZ} (psi)	500
Elastic Buckling Interaction Equations	
Q1	0.326
Q2	0.193
Q3	0.437
Q4	0.326
Plastic Buckling Interaction Equations	
Q5	0.193
Q6	0.437
Q7	0.193
Q8	0.437

1. Bounding compressive stresses.
2. Component stresses include thermal stresses.

Table 11.2.4-6 $P_m + P_b$ Stresses for PWR Support Disk - 60g Concrete Cask Bottom End Impact (ksi)

Section[1]	S_x	S_y	S_{xy}	Stress Intensity	Allowable Stress	Margin of Safety
66	37.2	18.9	15.6	46.2	135.0	1.9
72	18.1	37.2	15.3	45.7	135.0	2.0
120	17.7	37.3	-15.0	45.5	135.0	2.0
82	36.9	17.9	-15.0	45.1	135.0	2.0
12	-24.1	8.5	2.4	32.9	133.5	3.1
28	-24.1	8.5	2.4	32.9	133.5	3.1
26	-24.0	8.5	-2.3	32.8	133.5	3.1
54	8.5	-24.0	-2.3	32.8	133.5	3.1
14	-23.9	8.5	-2.3	32.8	133.5	3.1
42	8.4	-24.0	-2.3	32.7	133.5	3.1
56	8.5	-23.9	2.3	32.7	133.5	3.1
40	8.4	-24.0	2.3	32.7	133.5	3.1
90	24.5	4.1	-10.4	29.1	135.0	3.6
67	3.3	23.6	10.5	29.1	135.0	3.6
99	3.3	23.5	10.5	29.0	135.0	3.7
106	24.1	3.9	10.4	29.0	135.0	3.7
122	24.4	3.9	-10.3	29.0	135.0	3.7
74	24.1	3.9	10.4	29.0	135.0	3.7
83	3.6	23.7	-10.2	28.6	135.0	3.7
115	3.3	23.6	-10.1	28.6	135.0	3.7
88	12.4	9.5	-14.1	28.4	135.0	3.8
114	9.7	11.9	-14.1	28.4	135.0	3.8
104	11.5	10.4	13.5	27.1	135.0	4.0
98	11.7	11.0	13.1	26.2	135.0	4.2
4	-11.1	-19.7	-7.6	24.1	125.8	4.2
2	-11.1	-19.7	-7.7	24.1	125.8	4.2
3	-19.6	-11.0	-7.6	24.1	125.8	4.2
1	-19.6	-11.0	-7.6	24.0	125.8	4.2
35	-5.3	-22.4	-4.2	23.3	129.9	4.6
37	-5.4	-22.3	4.2	23.3	129.9	4.6
7	-22.3	-5.3	-4.2	23.3	129.9	4.6
51	-5.3	-22.3	-4.1	23.3	129.9	4.6
49	-5.3	-22.3	4.2	23.3	129.9	4.6
23	-22.3	-5.3	-4.2	23.3	129.9	4.6
21	-22.3	-5.3	4.2	23.2	129.9	4.6
9	-22.3	-5.3	4.1	23.2	129.9	4.6
11	-12.3	9.4	-4.3	23.4	133.5	4.7
25	-12.3	9.4	-4.2	23.3	133.5	4.7
53	9.4	-12.3	4.3	23.3	133.5	4.7
39	9.3	-12.3	4.3	23.2	133.5	4.8

1. Section locations are shown in Figures 3.4.4.1-7 and 3.4.4.1-8.

Table 11.2.4-7 $P_m + P_b$ Stresses for BWR Support Disk - 60g Concrete Cask Bottom End Impact (ksi)

Section[1]	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
129	53.2	18.4	10.7	56.2	90.0	0.60
54	52.1	11.4	10.9	54.8	90.0	0.64
171	9.1	52.8	7.7	54.1	90.0	0.66
300	9.1	52.8	7.6	54.1	90.0	0.66
65	50.3	16.0	-10.3	53.2	90.0	0.69
192	49.9	16.8	-10.9	53.1	90.0	0.69
257	45.6	23.2	-14.7	52.9	90.0	0.70
234	11.5	51.7	-6.6	52.8	90.0	0.71
108	9.9	51.6	-6.3	52.6	90.0	0.71
119	50.1	10.2	-9.9	52.5	90.0	0.72
246	49.4	9.1	-9.9	51.7	90.0	0.74
182	49.2	9.5	9.7	51.4	90.0	0.75
103	13.6	16.2	11.6	26.6	90.0	2.39
229	13.6	16.1	11.6	26.5	90.0	2.39
109	-5.3	20.1	2.5	25.9	90.0	2.47
77	10.6	-14.1	3.9	25.9	90.0	2.48
203	10.5	-14.1	3.9	25.7	90.0	2.50
140	10.5	-14.1	-3.8	25.7	90.0	2.50
295	13.4	15.1	-11.4	25.7	90.0	2.50
269	10.5	-14.1	-3.8	25.7	90.0	2.50
166	13.4	15.1	-11.4	25.7	90.0	2.51
301	-4.1	21.1	-2.1	25.6	90.0	2.51
172	-4.3	20.9	-2.2	25.6	90.0	2.52
134	1.7	11.8	-11.6	25.4	90.0	2.55
263	1.6	11.7	-11.6	25.3	90.0	2.55
197	1.6	11.8	11.6	25.3	90.0	2.55
71	1.7	11.8	11.6	25.3	90.0	2.55
235	-3.3	21.5	2.1	25.1	90.0	2.58
27	15.4	-8.9	-2.8	24.9	90.0	2.61
165	-12.3	-4.6	-11.8	24.9	90.0	2.61
228	-12.3	-4.5	11.8	24.9	90.0	2.62
294	-12.3	-4.6	-11.8	24.9	90.0	2.62
40	15.3	-8.9	2.9	24.8	90.0	2.62
102	-12.3	-4.5	11.8	24.8	90.0	2.62
73	4.2	14.1	11.3	24.6	90.0	2.65
199	4.1	14.2	11.2	24.6	90.0	2.66
124	-20.4	-6.4	-8.5	24.5	90.0	2.67
252	-20.4	-6.4	-8.5	24.4	90.0	2.68
60	-20.4	-6.5	8.6	24.4	90.0	2.69
187	-20.4	-6.4	8.5	24.4	90.0	2.69

1. Section locations are shown in Figures 3.4.4.1-13 through 3.4.4.1-16

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11.2.5 Explosion

The analysis of a design basis flood presented in Section 11.2.9 shows that the flood exerts a pressure of 22 psig on the canister, and that the Universal Storage System experiences no adverse effects due to this pressure. The pressure of 22 psig is considered to bound any pressure due to an explosion occurring in the vicinity of the ISFSI.

11.2.5.1 Cause of Explosion

An explosion affecting the Universal Storage System may be caused by industrial accidents or the presence of explosive substances in the vicinity of the ISFSI. However, no flammable or explosive substances are stored or used at the storage facility. In addition, site administrative controls exclude explosive substances in the vicinity of the ISFSI. Therefore, an explosion affecting the site is extremely unlikely. This accident is evaluated in order to provide a bounding pressure that could be used in the event that the potential of an explosion must be considered at a given site.

11.2.5.2 Analysis of Explosion

Pressure due to an explosion event is bounded by the pressure effects of a flood having a depth of 50 feet. The Transportable Storage Canister shell is evaluated in Section 11.2.9 for the effects of the flood having a depth of 50 feet, and the results are summarized in Tables 11.2.9-1 and 11.2.9-2.

There is no adverse consequence to the canister as a result of the 22 psig pressure exerted by a design basis flood. This pressure conservatively bounds an explosion event.

11.2.5.3 Corrective Actions

In the unlikely event of a nearby explosion, inspection of the concrete casks is required to ensure that the air inlets and outlets are free of debris, and to ensure that the monitoring system and screens are intact. No further recovery or corrective actions are required for this accident.

11.2.5.4 Radiological Impact

There are no radiological consequences for this accident.

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11.2.6 Fire Accident

This section evaluates the effects of a bounding condition hypothetical fire accident, although a fire accident is a very unlikely occurrence in the lifetime of the Universal Storage System. The evaluation demonstrates that for the hypothetical thermal accident (fire) condition the cask meets its storage performance requirements.

11.2.6.1 Cause of Fire

A fire may be caused by flammable material or by a transport vehicle. While it is possible that a transport vehicle could cause a fire while transferring a loaded storage cask at the ISFSI, this fire will be confined to the vehicle and will be rapidly extinguished by the persons performing the transfer operations or by the site fire crew. The maximum permissible quantity of fuel in the combined fuel tanks of the transport vehicle and prime mover is the only means by which fuel (maximum 50 gallons) would be next to a cask, and potentially at, or above, the elevation of the surface on which the cask is supported.

The fuel carried by other on-site vehicles or by other equipment used for ISFSI operations and maintenance, such as air compressors or electrical generators, is considered not to be within the proximity of a loaded cask on the ISFSI pad. Site-specific analysis of fire hazards will evaluate the specific equipment used at the ISFSI and determine any additional controls required.

11.2.6.2 Detection of Fire

A fire in the vicinity of the Universal Storage System will be detected by observation of the fire or smoke.

11.2.6.3 Analysis of Fire

The vertical concrete cask with its internal contents, initially at the steady state normal storage condition, is subject to a hypothetical fire accident. The fire is due to the ignition of a flammable fluid, and operationally, the volume of flammable fluid that is permitted to be on the ISFSI pad (at, or above, the elevation of the surface on which a cask is supported and within approximately two feet of an individual cask) is limited to 50 gallons. The lowest burning rate (change of depth per unit time of flammable fluid for a pool of fluid) reported in the 18th Edition of the Fire Protection Handbook [37] is 5 inches/hour for kerosene. The flammable liquid is assumed to cover a 15-foot square area, corresponding to the center to center distance of the concrete casks less the footprint of the concrete cask, which is a 128-inch diameter circle. The depth (D) of the 50 gallons of flammable liquid is calculated as:

$$D = \frac{50\,(\text{gallons}) \times 231\,(\text{in}^3/(\text{gallon})}{15 \times 15 \times 144\,(\text{in}^2) - 3.14 \times 128^2/4\,(\text{in}^2)}$$

D = 0.6 inches

With a burning rate of 5 inches/hour, the fire would continue for 7.2 minutes. The fire accident evaluation in this section conservatively considers an 8-minute fire. The temperature of the fire is taken to be 1475°F, which is specified for the fire accident condition in 10 CFR 71.73c(3).

The fire condition is an accident condition and is initiated with the concrete cask in a normal operating steady state condition. To determine the maximum temperatures of the concrete cask components, the two-dimensional axisymmetric finite element model for the BWR configuration described in Section 4.4.1.1 is used to perform a transient analysis. However, the effective properties for the canister content for specific heat, density and thermal conductivity for the PWR are used, to conservatively maximize the thermal diffusivity, which results in higher temperatures for the canister contents during the fire accident condition.

The initial condition of the fire accident transient analysis is based on the steady state analysis results for the normal condition of storage, which corresponds to an ambient temperature of 76°F in conjunction with solar insolation (as specified in Section 4.4.1.1). The fire condition is implemented by constraining the nodes at the inlet to be 1475°F for 8 minutes (see Figure 11.2.6-1). One of the nodes at the edge of the inlet is attached to an element in the concrete region. This temperature boundary condition is applied as a stepped boundary condition. During the 8-minute fire, solar insolation is also applied to the outer surface of the concrete cask. At the end of the 8 minutes, the temperature of the nodes at the inlet is reset to the ambient temperature of 76°F. The cool down phase is continued for an additional 10.7 hours to observe the maximum canister shell temperature and the average temperature of the canister contents.

The maximum temperatures of the fuel cladding and basket are obtained by adding the maximum temperature change due to the fire transient to the maximum component temperature for the normal operational condition. The maximum component temperatures are presented in Table 11.2.6-1, which shows that the component temperatures are below the allowable temperatures. The limited duration of the fire and the large thermal capacitance of the concrete cask restricted the temperatures above 244°F to a region less than 3 inches above the top surface of the air inlets. The maximum bulk concrete temperature is 138°F during and after the fire accident. This corresponds to an increase of less than 3°F compared to the bulk concrete temperature for normal condition of storage. These results confirm that the operation of the concrete cask is not adversely affected during and after the fire accident condition.

11.2.6.4 Corrective Actions

Immediately upon detection of the fire, appropriate actions should be taken by site personnel to extinguish the fire. The concrete cask should then be inspected for general deterioration of the concrete, loss of shielding (spalling of concrete), exposed reinforcing bar, and surface discoloration that could affect heat rejection. This inspection will be the basis for the determination of any repair activities necessary to return the concrete cask to its design basis configuration.

11.2.6.5 Radiological Impact

There are no significant radiological consequences for this accident. There may be local spalling of concrete during the fire event, which could lead to some minor reduction in shielding effectiveness. The principal effect would be local increases in radiation dose rate on the cask surface.

Figure 11.2.6-1 Temperature Boundary Condition Applied to the Nodes of the Inlet for the Fire Accident Condition

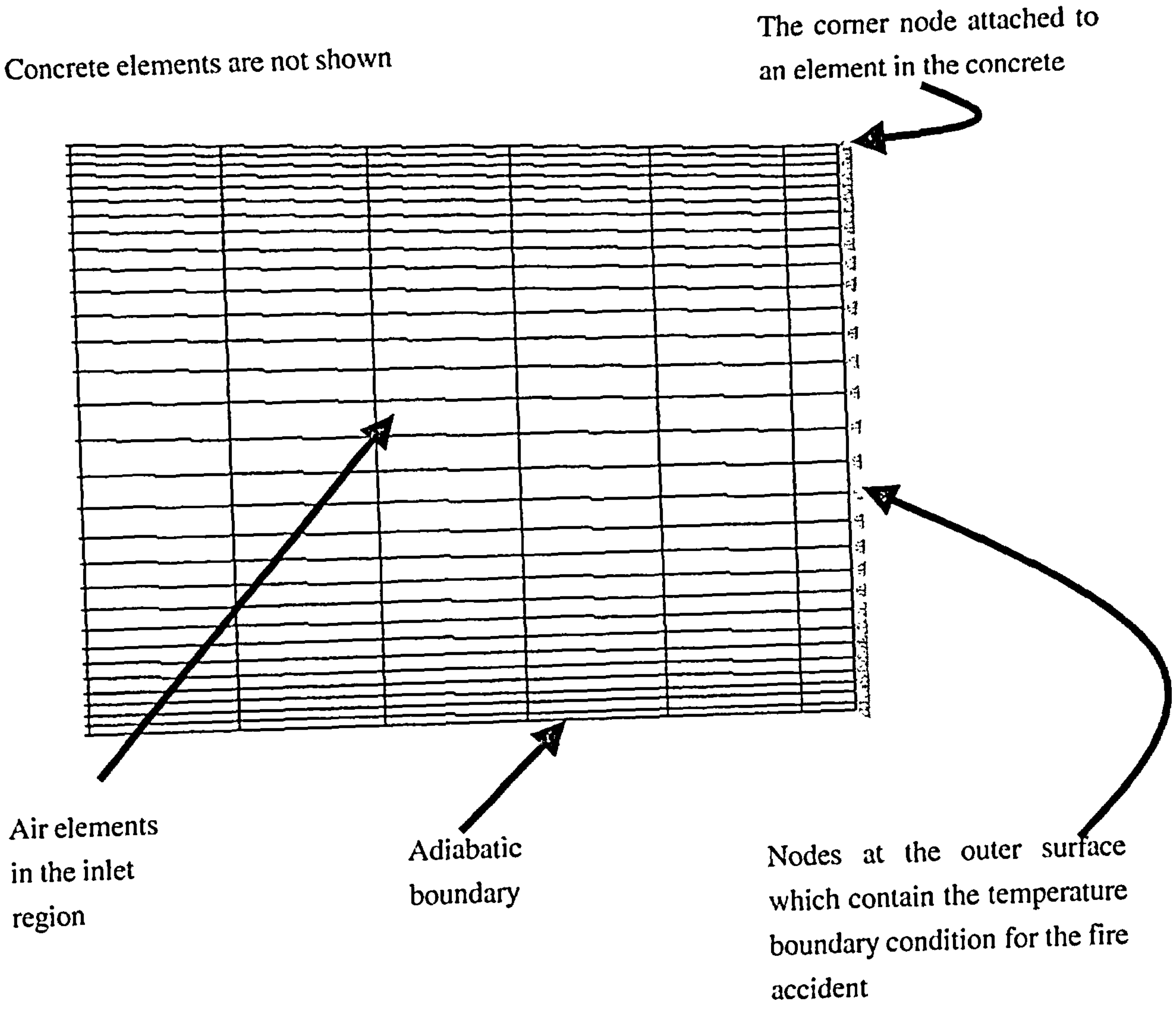

Table 11.2.6-1 Maximum Component Temperatures (°F) During and After the Fire Accident

Component	PWR Maximum temperature (°F)	PWR Allowable temperature (°F)	BWR Maximum temperature (°F)	BWR Allowable temperature (°F)
Fuel clad	688	1058	682	1058
Support disk	641	800	654	700
Heat transfer disk	639	750	652	750
Canister shell	391	800	416	800
Concrete*	244	350	244	350

* Temperatures of 244°F and greater are within 3 inches of the inlet, which does not affect the operation of the concrete cask.

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11.2.7 Maximum Anticipated Heat Load (133°F Ambient Temperature)

This section evaluates the Universal Storage System response to storage operation at an ambient temperature of 133°F. The condition is analyzed in accordance with the requirements of ANSI/ANS 57.9 to evaluate a credible worst-case thermal loading. A steady state condition is considered in the thermal evaluation of the system for this accident condition.

11.2.7.1 Cause of Maximum Anticipated Heat Load

This condition results from a weather event that causes the concrete cask to be subject to a 133°F ambient temperature with full insolation.

11.2.7.2 Detection of Maximum Anticipated Heat Load

Detection of the high ambient temperature condition will be by the daily measurement of ambient temperature and concrete cask outlet air temperature.

11.2.7.3 Analysis of Maximum Anticipated Heat Load

Using the same methods and thermal models described in Section 11.1.1 for the off-normal conditions of severe ambient temperatures (106°F and -40°F), thermal evaluations are performed for the concrete cask and the canister with its contents for this accident condition. The principal PWR and BWR cask component temperatures for this ambient condition are:

Component	133°F Ambient Max Temp. (°F)		Allowable Max Temp. (°F)	
	PWR	BWR	PWR	BWR
Fuel Cladding	693	690	1058	1058
Support Disks	650	664	800	700
Heat Transfer Disks	648	662	750	750
Canister Shell	408	432	800	800
Concrete	262	266	350	350

This evaluation shows that the component temperatures are within the allowable temperatures for the extreme ambient temperature conditions.

Thermal stress evaluations for the concrete cask are performed using the method and model presented in Section 3.4.4. The concrete temperature results obtained from the thermal analysis for this accident condition are applied to the structural model for stress calculation. The maximum stress, 7,869 psi in the reinforcing steel, occurs in the circumferential direction. The margin of safety is 54,000 psi/7,869 psi -1 = +5.9. The maximum compressive stress, 808 psi, in the concrete occurs in the vertical direction. The maximum circumferential compressive stress in the concrete is 116 psi. The margin of safety is [0.7(4,000 psi)/808 psi] -1 = +2.5. These stresses are used in the loading combination for the concrete cask shown in Section 3.4.4.2.

11.2.7.4 Corrective Actions

The high ambient temperature condition is a natural phenomenon, and no recovery or corrective actions are required.

11.2.7.5 Radiological Impact

There are no dose implications due to this event.

11.2.8 Earthquake Event

This section provides an evaluation of the response of the vertical concrete cask to an earthquake imparting a horizontal acceleration of 0.26g and 0.30g at the top surface of the concrete pad. This evaluation shows that the loaded or empty vertical concrete cask does not tip over or slide in the earthquake event. The vertical acceleration is defined as 2/3 of the horizontal acceleration in accordance with ASCE 4-86 [36].

11.2.8.1 Cause of the Earthquake Event

Earthquakes are natural phenomena to which the storage system might be subjected at any U.S. site. Earthquakes are detected by the ground motion and by seismic instrumentation on and off site.

11.2.8.2 Earthquake Event Analysis

In the event of earthquake, there exists a base shear force or overturning force due to the horizontal acceleration ground motion and a restoring force due to the vertical acceleration ground motion. This ground motion tends to rotate the concrete cask about the bottom corner at the point of rotation (at the chamfer). The horizontal moment arm extends from the center of gravity (C.G.) toward the outer radius of the concrete cask. The vertical moment arm reaches from the C.G. to the bottom of the cask. When the overturning moment is greater than or equal to the restoring moment, the cask will tip over. To maximize this overturning moment, the dimensions for the Class 3 PWR configuration, which has the highest C.G., are used in this evaluation. Based on the requirements presented in NUREG-0800 [22], the static analysis method is considered applicable if the natural frequency of the structure is greater than 33 cycles per second (Hz).

The combined effect of shear and flexure is computed as:

$$\frac{1}{f^2} = \frac{1}{f_f^{\,2}} + \frac{1}{f_s^{\,2}} = \frac{1}{348.6} + \frac{1}{150.7} \quad [19]$$

or

$$f = 105.2\ \text{Hz} > 33\ \text{Hz}$$

where:

f_f = frequency for the first free-free mode based on flexure deformation only (Hz),
f_s = frequency for the first free-free mode based on shear deformation only (Hz).

The frequency f_f is computed as:

$$F_f = \frac{\lambda^2}{2\pi L^2}\sqrt{\frac{EI}{M}} = \frac{4.730^2}{2\pi(226)^2}\sqrt{\frac{(3.38\times10^6)\times(1.4832\times10^7)}{2.005}} \quad [19]$$

$$f_f = 348.6 \text{ Hz}$$

where:

$\lambda = 4.730$,

$L = 226$ in, length of concrete cask,

$E = 3.38 \times 10^6$ psi, modulus of elasticity for concrete at 200°F,

$$I = \text{moment of inertia} = \frac{\pi\left(D_o^4 - D_i^4\right)}{64} = \frac{\pi\left[(136 \text{ in})^4 - (79.5 \text{ in})^4\right]}{64} = 1.4832\times10^7 \text{ in}^4,$$

$$\rho = \frac{140}{1728\times386.4} = 2.096\times10^{-4} \text{ lbm/in}^3, \text{ mass density,}$$

$$M = \pi(68^2 - 39.75^2)\times(2.096\times10^{-4}) = 2.005 \text{ lbm/in}$$

The frequency accounting for the shear deformation is:

$$f_s = \frac{\lambda_s}{2\pi L}\sqrt{\frac{KG}{\mu}} = \frac{3.141593}{2(3.141595)(226)}\sqrt{\left(\frac{(0.6947)(1.40\times10^6)}{2.096\times10^{-4}}\right)} \quad [19]$$

$$f_s = \quad 150.7 \text{ Hz}$$

where:

$\lambda_s = \pi$,

$L = 226$ in, length of concrete cask,

$$K = \frac{6(1+\nu)(1+m^2)^2}{(7+6\nu)(1+m^2)+(20+12\nu)m^2}, \text{ shear coefficient,}$$

$= 0.6947,$

$$\mu = \frac{140}{1728 \times 386.4} = 2.096 \times 10^{-4} \text{ lbm/in}^3\text{, mass density of the material,}$$

$$G = \frac{0.5E}{(1+\nu)} = \frac{0.5(3.38 \times 10^6)}{(1+0.2)} = 1.408 \times 10^6 \text{ psi, modulus of rigidity,}$$

and,

$m = R_i/R_o = 39.75/68 = 0.5846,$

$\nu = 0.2$, Poisson's ratio for concrete.

Since the fundamental mode frequency is greater than 33 Hz, static analysis is appropriate.

11.2.8.2.1 Tip-Over Evaluation of the Vertical Concrete Cask

To maintain the concrete cask in equilibrium, the restoring moment, M_R must be greater than, or equal to, the overturning moment, M_o (i.e. $M_R \geq M_o$). Based on this premise, the following derivation shows that 0.26g acceleration of the design basis earthquake at the surface of the concrete pad is well below the acceleration required to tip-over the cask.

The combination of horizontal and vertical acceleration components is based on the 100-40-40 approach of ASCE 4-86 [36], which considers that when the maximum response from one component occurs, the response from the other two components is 40% of the maximum. According to ASCE 4-86, the vertical component of acceleration shall be obtained by scaling the corresponding ordinates of the horizontal components by two-thirds. However, the vertical component of acceleration is conservatively considered to be the same as the horizontal component of acceleration in the evaluation in this section.

Let:

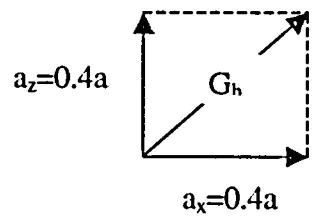
$a_x = a_z = a$ = horizontal acceleration components
$a_y = a$ = vertical acceleration component
G_h = Vector sum of two horizontal acceleration components
G_v = Vertical acceleration component

There are two cases that have to be analyzed:

Case 1) The vertical acceleration, a_y, is at its peak: ($a_y = a$, $a_x = .4a$, $a_z = .4a$)

$$G_h = \sqrt{a_x^2 + a_z^2}$$

$$G_h = \sqrt{(0.4\times a)^2 + (0.4\times a)^2} = 0.566\times a$$

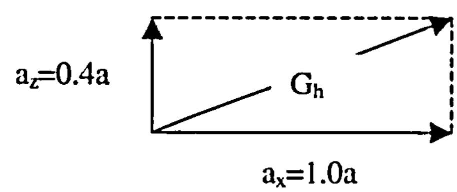


$$G_v = 1.0\times a_y = 1.0\times a$$

Case 2) One horizonal acceleration, a_x, is at its peak: (a_y=.4a, $a_x = a$, $a_z = .4a$)

$$G_h = \sqrt{a_x^2 + a_z^2}$$

$$G_h = \sqrt{(1.0\times a)^2 + (0.4\times a)^2} = 1.077\times a$$



$$G_v = 0.4\times a_y = 0.4\times a$$

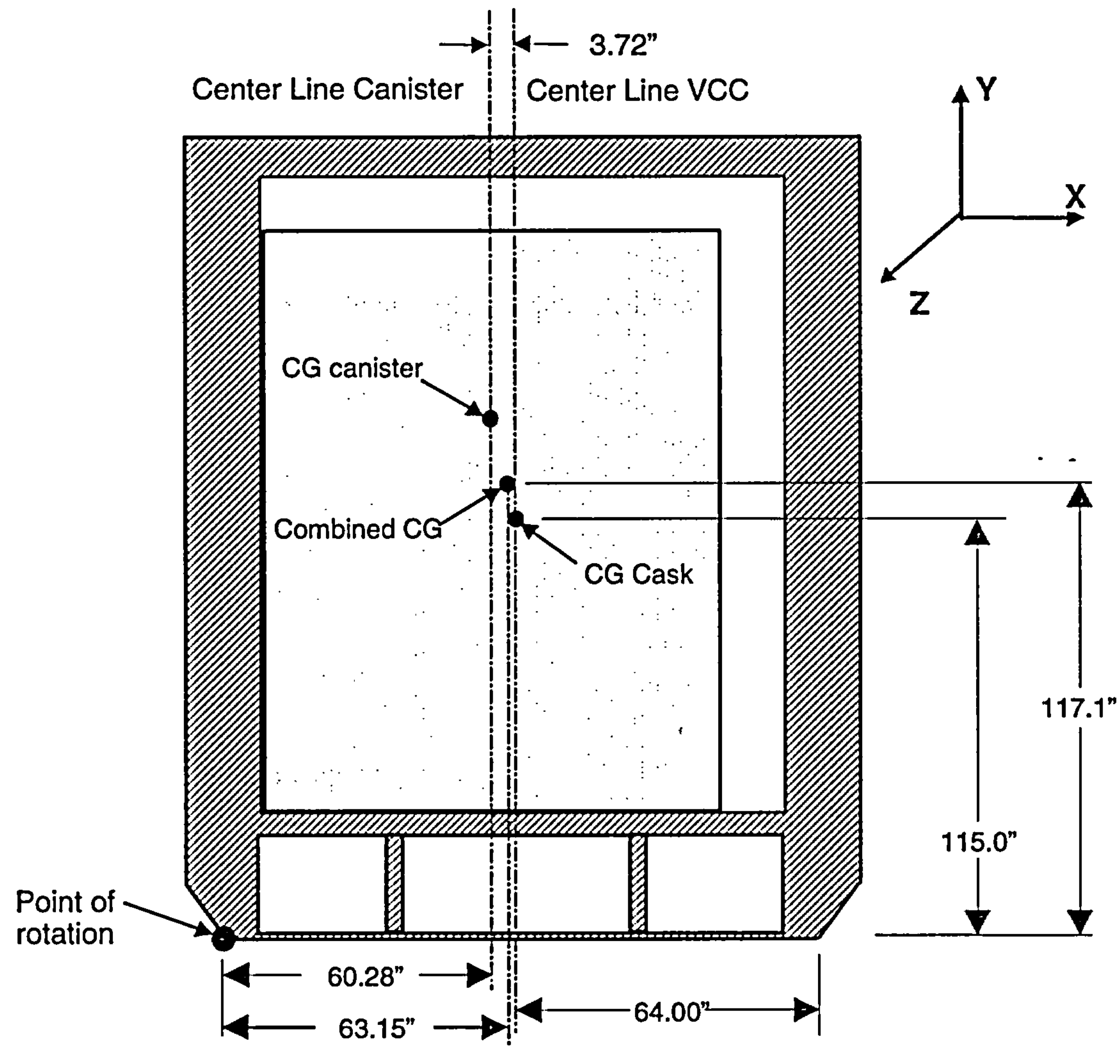
In order for the cask to resist overturning, the restoring moment, M_R, about the point of rotation, must be greater than the overturning moment, M_o, that:

$$M_R \geq M_o \text{, or}$$

$$F_r \times b \geq F_o \times d \Rightarrow (W\times 1 - W\times G_v)\times b \geq (W\times G_h)\times d$$

where:

- d = vertical distance measured from the base of the VCC to the center of gravity
- b = horizontal distance measured from the point of rotation to the C.G.
- W = the weight of the VCC
- F_o = overturning force
- F_r = restoring force



substituting for G_h and G_v gives:

Case 1

$$(1-a)\frac{b}{d} \geq 0.566 \times a$$

$$a \leq \frac{b/d}{0.566 + 1.0\left(b/d\right)}$$

Case 2

$$(1-0.4a)\frac{b}{d} \geq 1.077a$$

$$a \leq \frac{b/d}{1.077 + 0.4\left(b/d\right)}$$

Because the canister is not attached to the concrete cask, the combined center of gravity for the concrete cask, with the canister in its maximum off-center position, must be calculated. The point of rotation is established at the outside lower edge of the concrete cask.

The inside diameter of the concrete cask is 74.5 inches and the outside diameter of the canister is 67.06 inches; therefore, the maximum eccentricity between the two is:

$$e = \frac{74.50\,\text{in} - 67.06\,\text{in}}{2} = 3.72\ \text{in.}$$

The horizontal displacement, x, of the combined C.G. due to eccentric placement of the canister is:

$$x = \frac{70{,}701(3.72)}{310{,}345} = 0.85\ \text{in.}$$

Therefore,

$$b = 64 - 0.85 = 63.15\ \text{in.}$$

$$d = 117.1\ \text{in.}$$

$$1)\ a \le \frac{63.15/117.1}{0.566 + 1.0 \times 63.15/117.1}$$

$$a \le 0.49g$$

$$2)\ a \le \frac{63.15/117.1}{1.077 + 0.4 \times 63.15/117.1}$$

$$a \le 0.42g$$

Therefore, the minimum ground acceleration that may cause a tip-over of a loaded concrete cask is 0.42g. Since the 0.26g design basis earthquake ground acceleration for the UMS® system is less than 0.42g, the storage cask will not tip over.

The factor of safety is 0.42 / 0.26 = 1.61, which is greater than the required factor of safety of 1.1 in accordance with ANSI/ANS-57.9.

Since an empty vertical concrete cask has a lower C.G. as compared to a loaded concrete cask, the tip-over evaluation for the empty concrete cask is bounded by that for the loaded concrete cask.

11.2.8.2.2 Sliding Evaluation of the Vertical Concrete Cask

To keep the cask from sliding on the concrete pad, the force holding the cask (F_s) has to be greater than or equal to the force trying to move the cask.

Based on the equation for static friction:

$$F_s = \mu N \geq G_h W$$
$$\mu\left(1 - G_v\right) W \geq G_h W$$

where:

μ = coefficient of friction
N = the normal force
W = the weight of the concrete cask
G_v = vertical acceleration component
G_h = resultant of horizontal acceleration component

Substituting G_h and G_v for the two cases:

Case 1	Case 2
$\mu\ (1 - 1.0a) \geq 0.556a$	$\mu\ (1 - 0.4a) \geq 1.077a$

For the coefficient of friction of 0.35 [21] between the steel bottom plate of the concrete cask and the concrete surface of the storage pad:

Case 1: $0.35 \times (1\text{-}a) \geq 0.566a$
$a \leq 0.38g$
Case 2: $0.35 \times (1\text{-}0.4a) \geq 1.077a$
$a \leq 0.29g$

For a design acceleration of 0.26g, the minimum factor of safety (FS) for acceleration is:

$$FS = \frac{0.29g}{0.26g} = 1.12$$

For a coefficient of friction of 0.4 between the steel bottom plate of the concrete cask and the concrete surface of the storage pad:

Case 1: $0.4 \times (1\text{-}a) \geq 0.566a$
$a \leq 0.41$
Case 2: $0.4 \times (1\text{-}0.4a) \geq 1.077a$
$a \leq 0.32$

For a design acceleration of 0.29g, the minimum factor of safety (FS) for acceleration is:

$$FS = \frac{0.32g}{0.29g} = 1.10$$

The analysis shows that the minimum safety factor against cask sliding for the design earthquake accelerations is 1.1 and meets the requirements of ANSI/ANS-57.9.

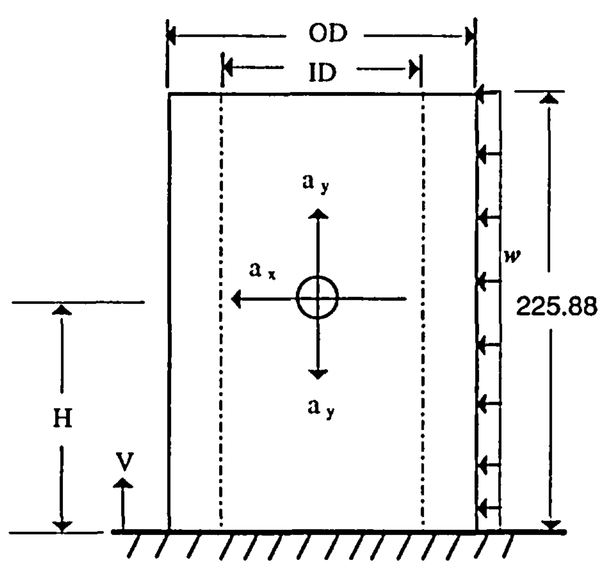
11.2.8.2.3 Stress Generated in the Vertical Concrete Cask During an Earthquake Event

To demonstrate the ability of the concrete cask to withstand earthquake loading conditions, the fully loaded cask is conservatively evaluated for seismic loads of 0.5g in the horizontal direction and 0.5g in the vertical direction. These accelerations reflect a more rigorous seismic loading, and, therefore, bound the design basis earthquake event. No credit is taken for the steel inner liner of the concrete cask. The maximum compressive stress at the outer and inner surfaces of the concrete shell are conservatively calculated by assuming the vertical concrete cask to be a cantilever beam with its bottom end fixed. The maximum compressive stresses are:

$$\sigma_{v\,outer} = (M / S_{outer}) + ((1+a_y)(W_{vcc}) / A) = -84 - 51 = -135 \text{ psi},$$
$$\sigma_{v\,inner} = (M / S_{inner}) + ((1+a_y)(W_{vcc}) / A) = -49 - 51 = -100 \text{ psi},$$

where:

$a =$ 0.50 g, horizontal direction,
$a_y =$ 0.50 g, vertical direction,
H = 117.1 in., fully loaded C.G.,
W_{vcc}= 325,000 lbf, bounding cask weight
OD = 136 in., concrete exterior diameter,
ID = 79.50 in., concrete interior diameter,
$A = \pi (OD^2 - ID^{2)}/4 = 9{,}562.8$ in.^{2},
$I = \pi (OD^4 - ID^{4)}/64 = 14.83 \times 10^6$ in.^{4},
$S_{outer} = 2I/OD = 218{,}088.2$ in.^{3},
$S_{inner} = 2I/ID = 373{,}035.0$ in.^{3},
$w = a_x W_{vcc}/225.88 \approx 720$ lbf / in.
$M = w(225.88)^2/2 = 1.84 \times 10^7$ in.-lbf,
the maximum bending moment at the support.



The calculated compressive stresses are used in the load combinations for the vertical concrete cask as shown in Table 3.4.4.2-1.

11.2.8.3 Corrective Actions

Inspection of the vertical concrete casks is required following an earthquake event. The positions of the concrete casks should be verified to ensure they maintain the 15-foot center-to-center spacing established in Section 8.1.3. The temperature monitoring system should be checked for operation.

11.2.8.4 Radiological Impact

There are no radiological consequences for this accident.

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11.2.9 Flood

This evaluation considers design basis flood conditions of a 50-foot depth of water having a velocity of 15 feet per second. This flood depth would fully submerge the Universal Storage System. Analysis demonstrates that the Vertical Concrete Cask does not slide or overturn during the design-basis flood. The hydrostatic pressure exerted by the 50-foot depth of water does not produce significant stress in the canister. The Universal Storage System is therefore not adversely impacted by the design basis flood.

Small floods may lead to a blockage of concrete cask air inlets. Full blockage of air inlets is evaluated in Section 11.2.13.

11.2.9.1 Cause of Flood

The probability of a flood event at a given ISFSI site is unlikely because geographical features, and environmental factors specific to that site are considered in the site approval and acceptance process. Some possible sources of a flood are: (1) overflow from a river or stream due to unusually heavy rain, snow-melt runoff, a dam or major water supply line break caused by a seismic event (earthquake); (2) high tides produced by a hurricane; and (3) a tsunami (tidal wave) caused by an underwater earthquake or volcanic eruption.

11.2.9.2 Analysis of Flood

The concrete cask is considered to be resting on a flat level concrete pad when subjected to a flood velocity pressure distributed uniformly over the projected area of the concrete cask. Because of the concrete cask geometry and rigidity, it is analyzed as a rigid body. Assuming full submersion of the concrete cask and steady-state flow conditions, the drag force, F_D, is calculated using classical fluid mechanics for turbulent flow conditions. A safety factor of 1.1 for stability against overturning and sliding is applied to ensure that the analyses bound design basis conditions. The coefficient of friction between carbon steel and concrete used in this analysis is 0.35 [23].

Analysis shows that the concrete cask configured for storing the Class 3 PWR spent fuel, because of its center of gravity, weight, and geometry has the least resistance of the five configurations to flood velocity pressure. Conservatively, the analysis is performed for a canister containing no fuel. The Class 3 PWR cask configuration analysis is as follows.

The buoyancy force, F_b, is calculated from the weight of water (62.4 lbs/ft^3) displaced by the fully submerged concrete cask. The displacement volume (vol) of the concrete cask containing the canister is 1,721 ft^3. The displacement volume is the volume occupied by the cask and the transport canister less the free space in the central annular cavity of the concrete cask.

$$F_b = \text{Vol} \times 62.4 \text{ lbs/ft}^3$$
$$= 107{,}383 \text{ lbs.}$$

Assuming the steady-state flow conditions for a rigid cylinder, the total drag force of the water on the concrete cask is given by the formula:

$$F_{D15} = (C_D)(\rho)(V^2)\left(\frac{A}{2}\right) \qquad [24]$$
$$= 32{,}831 \text{ lbs.}$$

where:

C_D = Drag coefficient, which is dependent upon the Reynolds Number (Re). For flow velocities greater than 6 ft/sec, the value of C_D approaches 0.7 [24].

ρ = mass density of water = 1.94 slugs/ft^3

D = Concrete cask outside diameter (136.0 in. / 12 = 11.33 ft)

V = velocity of water flow (15 ft/sec)

A = projected area of the cask normal to water flow (diameter 11.34 ft × overall height 18.95 ft = 214.9 ft^2)

The drag force required to overturn the concrete cask is determined by summing the moments of the drag force and the submerged weight (weight of the cask less the buoyant force) about a point on the bottom edge of the cask. This method assumes a pinned connection, i.e., the cask will rotate about the point on the edge rather than slide. When these moments are in equilibrium, the cask is at the point of overturning.

$$F_D \times \left(\frac{h}{2}\right) = \left(W_{cask} - F_b\right) \times r$$

$F_D = 100{,}314$ lbs

where:

h	=	concrete cask overall height (227.38 in.)
W_{CASK}	=	concrete cask weight = 275,000 lbs
		(Loaded concrete cask – fuel = 310,345 lbs – 35,520 lbs)
F_b	=	buoyant force = 107,383 lbs
r	=	concrete cask radius (5.67 ft)

Solving the drag force equation for the velocity, V, that is required to overturn the concrete cask:

$$V = \sqrt{\frac{2F_D}{C_D \rho A}}$$

= 25.0 ft/sec. (including safety factor of 1.1)

To prevent sliding, the minimum coefficient of friction (with a safety factor of 1.1) between the carbon steel bottom plate of the concrete cask and the concrete surface upon which it rests is,

$$\mu_{min} = \frac{(1.1)F_{D15}}{F_y} = \frac{(1.1)32{,}831 \text{ lb}}{(275{,}000 - 107{,}557)\text{lb}} = 0.22$$

where:

F_y = the submerged weight of the concrete cask.

The analysis shows that the minimum coefficient of friction, μ, required to prevent sliding of the concrete cask is 0.22. For a drag force of 57,160 pounds, the coefficient of friction to prevent sliding is 0.31. The coefficient of friction between the steel bottom plate of the concrete cask and the concrete surface of the storage pad (0.35) is greater than the minimum coefficient of friction required to prevent sliding of the concrete cask. Therefore, the concrete cask does not slide under design-basis flood conditions.

The water velocity required to overturn the concrete cask is greater than the design-basis velocity of 15 ft/sec. Therefore, the concrete cask is not overturned under design basis flood conditions.

The flood depth of 50 feet exerts a hydrostatic pressure on the canister and the concrete cask. The water exerts a pressure of 22 psi (50 × 62.4/144) on the canister, which results in stresses in the canister shell. Canister internal pressure is conservatively taken as 0 psi. The canister structural analysis for the increased external pressure due to flood conditions is performed using an ANSYS finite element model as described in Section 3.4.4.1.

The resulting maximum canister stresses for flood loads are summarized in Tables 11.2.9-1 and 11.2.9-2 for primary membrane and primary membrane plus bending stresses, respectively.

The sectional stresses shown in Tables 11.2.9-1 and 11.2.9-2 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. Consequently, there is no adverse consequence to the canister as a result of the hydrostatic pressure due to the flood condition.

The concrete cask is a thick monolithic structure and is not affected by the hydrostatic pressure due to design basis flood. Nonetheless, the stresses in the concrete due to the drag force (F_D) are conservatively calculated as shown below. The concrete cask is considered to be fixed at its base.

$F_D = 32{,}831$ lbs

$D = 136.0$ in. (concrete exterior diameter)

$ID = 79.5$ in. (concrete interior diameter)

$h = 214.68$ in. (cask overall height)

$A = \pi (D^2 - ID^2) / 4 = 9{,}563$ in.^{2}

(Cross-sectional area)

$I = \pi (D^4 - ID^4) / 64 = 14.83 \times 10^6$ in.^{4}

(Moment of Inertia)

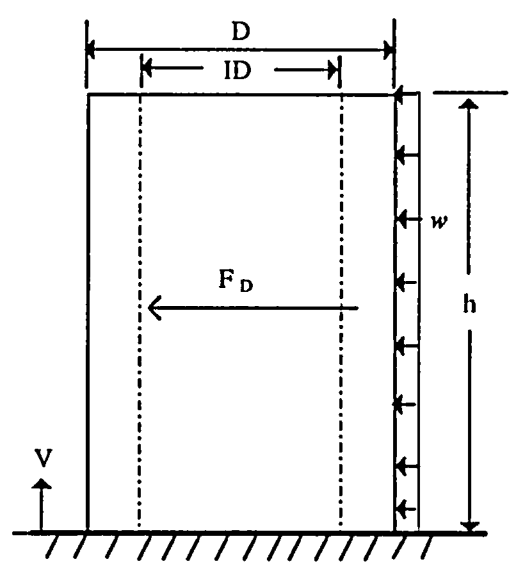
$S = 2I/D = 218{,}088$ in.^{3}

(Section Modulus for outer surface)

$w = F_D/h = 155.0$ lbf / in.

$M = w(h)^2 / 2 = 3.44 \times 10^6$ in.-lbs

(Bending Moment at the base)



Maximum stresses at the base surface:

$\sigma_v = M / S_{outer} \approx 20$ psi (tension or compression)

The compressive stresses are included in load combination No. 7 in Table 3.4.4.2–1. As shown in Table 3.4.4.2–1, the maximum combined stresses for the load combination due to dead, live, thermal and flood loading, are less than the allowable stress.

11.2.9.3 Corrective Actions

Inspection of the concrete casks is required following a flood. While the cask does not tip over or slide, a potential exists for collection of debris or accumulation of silt at the base of the cask, which could clog or obstruct the air inlets. Operation of the temperature monitoring system should be verified, as flood conditions may have impaired its operation.

11.2.9.4 Radiological Impact

There are no dose consequences associated with the design basis flood event.

Table 11.2.9-1 Canister Increased External Pressure (22 psi) with No Internal Pressure (0 psi) Primary Membrane (P_m) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	-0.17	-0.86	-2.17	0.06	0.03	0.31	2.10	40.08	18.1
2	-1.46	1.76	1.37	-0.24	0.03	0.30	3.29	40.08	11.2
3	0.24	2.71	-0.64	-0.23	-0.05	-0.61	3.69	40.08	9.9
4	-0.02	-1.18	-0.60	0.10	0.00	0.00	1.18	38.77	31.8
5	-0.02	-1.17	-0.60	0.10	0.00	0.00	1.17	35.86	29.7
6	-0.02	-1.17	-0.60	0.10	0.00	0.00	1.17	35.55	29.4
7	-0.02	-1.17	-0.60	0.10	0.00	0.00	1.17	38.23	31.7
8	-0.01	-1.13	-0.54	0.10	0.00	0.00	1.13	40.08	34.3
9	-0.28	-0.34	-0.16	0.02	-0.01	-0.12	0.27	40.08	145.6
10	0.32	-0.13	-0.08	0.03	-0.01	-0.07	0.46	40.08	85.5
11	-0.27	-0.13	0.09	-0.01	-0.01	-0.06	0.37	40.08	106.1
12	0.07	-0.23	-0.17	0.03	-0.01	0.02	0.32	40.08	125.6
13	0.06	-0.16	-0.30	0.02	-0.01	-0.06	0.38	40.08	103.4
14	-0.38	-0.38	-0.01	0.00	-0.16	0.02	0.49	40.08	81.5
15	0.02	0.02	-0.01	0.00	0.00	0.00	0.03	40.08	1235.3
16	-0.03	-0.03	-0.02	0.00	0.00	0.00	0.02	40.08	2524.5

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

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

Table 11.2.9-2 Canister Increased External Pressure (22 psi) with No Internal Pressure (0 psi) Primary Membrane plus Bending ($P_m + P_b$) Stresses (ksi)

Section No. [1]	S_X	S_Y	S_Z	S_{XY}	S_{YZ}	S_{XZ}	Stress Intensity	Stress Allowable[2]	Margin of Safety
1	-1.67	-0.20	-5.20	-0.07	0.03	0.02	5.01	60.12	11.0
2	-0.72	4.50	9.96	-0.43	0.05	0.70	10.80	60.12	4.6
3	1.02	-0.99	-13.97	0.13	-0.06	-0.78	15.08	60.12	3.0
4	-0.01	-1.19	-0.58	0.10	0.00	0.00	1.19	58.16	47.8
5	-0.01	-1.18	-0.60	0.10	0.00	0.00	1.19	53.79	44.3
6	-0.01	-1.19	-0.60	0.10	0.00	0.00	1.19	53.32	43.9
7	-0.01	-1.19	-0.60	0.10	0.00	0.00	1.19	57.35	47.3
8	-0.03	-1.16	-0.69	0.10	0.00	0.00	1.15	60.12	51.2
9	-0.19	-0.21	0.16	0.01	-0.01	-0.18	0.50	60.12	119.7
10	0.48	-0.05	0.01	0.04	-0.01	0.06	0.55	60.12	108.1
11	-0.19	0.07	0.69	-0.02	-0.01	-0.07	0.90	60.12	65.8
12	0.54	-0.02	0.07	0.04	-0.01	0.11	0.59	60.12	100.7
13	0.44	-0.01	-0.16	0.04	-0.02	-0.06	0.62	60.12	96.5
14	-7.47	-7.48	-0.23	0.00	-0.14	0.03	7.26	60.12	7.3
15	0.52	0.52	0.01	0.00	0.00	0.00	0.51	60.12	116.4
16	-0.28	-0.28	-0.03	0.00	0.00	0.00	0.25	60.12	240.5

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

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

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11.2.10 Lightning Strike

This section evaluates the impact of a lightning strike on the Vertical Concrete Cask. The evaluation shows that the cask does not experience adverse effects due to a lightning strike.

11.2.10.1 Cause of Lightning Strike

A lightning strike is a random weather-related event. Because the Vertical Concrete Cask is located on an unsheltered pad, the cask may be subject to a lightning strike. The probability of a lightning strike is primarily dependent on the geographical location of the ISFSI site, as some geographical regions experience a higher frequency of storms containing lightning than others.

11.2.10.2 Detection of Lightning Strike

A lightning strike on a concrete cask may be visually detected at the time of the strike, or by visible surface discoloration at the point of entry or exit of the current flow. Most reactor sites in locations experiencing a frequency of lightning bearing storms have lightning detection systems as an aid to ensuring stability of site electric power.

11.2.10.3 Analysis of the Lightning Strike Event

The analysis of the lightning strike event assumes that the lightning strikes the upper-most metal surface and proceeds through the concrete cask liner to the ground. Therefore, the current path is from the lightning strike point on the outer radius of the top flange of the storage cask, down through the carbon steel inner shell and the bottom plate to the ground. The electrical current flow path results in current-induced Joulean heating along that path.

The integrated maximum current for a lightning strike is a peak current of 250 kiloamps over a period of 260 microseconds, and a continuing current of up to 2 kiloamps for 2 seconds in the case of severe lightning discharges [25].

From Joule's Law, the amount of thermal energy developed by the combined currents is given by the following expression [26]:

$$Q = 0.0009478R\left[I_1^2(dt_1) + I_2^2(dt_2)\right]$$
$$= (22.98 \times 10^3)\ R\ \text{Btu} \qquad \text{[Equation 11.2.10.1]}$$

where:

Q = thermal energy (BTU)
I_1 = peak current (amps)
I_2 = continuing current (amps)
dt_1 = duration of peak current (seconds)
dt_2 = duration of continuing current (seconds)
R = resistance (ohms)

The maximum lightning discharge is assumed to attach to the smallest current-carrying component, that is, the top flange connected to the cask lid.

The propagation of the lightning through the carbon steel cask liner, which is both permeable and conductive, is considered to be a transient. For static conditions, the current is distributed throughout the shell. In a transient condition the current will be near the surface of the conductor. Similar to a concentrated surface heat flux incident upon a small surface area, a concentrated current in a confined area of the steel shell will result in higher temperatures than if the current were spread over the entire area, which leads to a conservative result. This conservative assumption is used by constraining the current flow area to a 90 degree sector of the circular cross section of the steel liner as opposed to the entire cross section. The depth of the current penetration (δ in meters) is estimated [27] as:

$$\delta = \frac{1}{\sqrt{\pi \mu f \sigma}}$$

where:

μ = permeability of the conductor = $100\mu_0$ ($\mu_0 = 4\pi \times 10^{-7}$ Henries/m)
σ = electrical conductivity (seimens/meter) = $1/\rho$
= 1/resistivity = $1/9.78\times10^{-8}$ (ohm-m)
f = frequency of the field (Hz)

The pulse is represented conservatively as a half sine form, so that the equivalent $f = 1/2\tau$, where τ is the referenced pulse duration. Two skin depths, corresponding to different pulse duration, are computed. The larger effective frequency will result in a smaller effective area to conduct the current. The effective resistance is computed as:

$$R = \frac{\rho\, l}{a}$$

where:

R = resistance (ohms)
ρ = resistivity = 9.78×10^{-8} (ohm-m)
l = length of conductor path
a = area of conductor (m^2)

Using the current level of the pulse and the duration in conjunction with the carbon steel liner, the resulting energy into the shell is computed using Equation 11.2.10.1.

This thermal energy dissipation is conservatively assumed to occur in the localized volume of the carbon steel involved in the current flow path through the flange to the inner liner. Assuming no heat loss or thermal diffusion beyond the current flow boundary, the maximum temperature increase in the flange due to this thermal energy dissipation is calculated [28] as:

$$\Delta T = \frac{Q}{mc}$$

where:

ΔT = temperature change (°F)
Q = thermal energy (BTU)
C = 0.113 Btu/lbs °F
m = mass (lbm)

The ΔT_1 for the peak current (250KA, 260 μsec) is found to be 4.7°F.

The ΔT_2 for the continuous current (2 kA, 2 sec) is found to be negligible (0.0006°F).

The ΔT_1 corresponds to the increase in the maximum temperature of the steel within the current path. For the concrete to experience an increase in temperature, the heat must disperse from the steel surface throughout the steel. Using the total thickness of the steel, over the 90-degree section, the increase in temperature would be proportional to the volume of steel in this sector resulting in a temperature rise of less than 1°F.

Therefore the increase in concrete temperature attributed to Joulean heating is not significant.

11.2.10.4 Corrective Actions

The casks should be visually inspected for any damage following the lightning event and actions taken as appropriate.

11.2.10.5 Radiological Impact

There are no dose implications due to the lightning event.

11.2.11 Tornado and Tornado Driven Missiles

This section evaluates the strength and stability of the Vertical Concrete Cask for a maximum tornado wind loading and for the impacts of tornado generated missiles. The design basis tornado characteristics are selected in accordance with Regulatory Guide 1.76 [29].

The evaluation demonstrates that the concrete cask remains stable in tornado wind loading in conjunction with impact from a high energy tornado missile. The performance of the cask is not significantly affected by the tornado event.

11.2.11.1 Cause of Tornado and Tornado Driven Missiles

A tornado is a random weather event. Probability of its occurrence is dependent upon the time of the year and geographical areas. Wind loading and tornado driven missiles have the potential for causing damage from pressure differential loading and from impact loading.

11.2.11.2 Detection of Tornado and Tornado Driven Missiles

A tornado event is expected to be visually observed. Advance warning of a tornado and of tornado sightings may be received from the National Weather Service, local radio and television stations, local law enforcement personnel, and site personnel.

11.2.11.3 Analysis of Tornado and Tornado Driven Missiles

Classical techniques are used to evaluate the loading conditions. Cask stability analysis for the maximum tornado wind loading is based on NUREG-0800 [30], Section 3.3.1, "Wind Loadings," and Section 3.3.2, "Tornado Loadings." Loads due to tornado-generated missiles are based on NUREG-0800, Section 3.5.1.4, "Missiles Generated by Natural Phenomena."

The concrete cask stability in a maximum tornado wind is evaluated based on the design wind pressure calculated in accordance with ANSI/ASCE 7-93 [31] and using classical free body stability analysis methods.

Local damage to the concrete shell is assessed using a formula developed for the National Defense Research Committee (NDRC) [32]. This formula is selected as the basis for predicting depth of missile penetration and minimum concrete thickness requirements to prevent scabbing of the concrete. Penetration depths calculated using this formula have been shown to provide reasonable correlation with test results (EPRI Report NP-440) [33].

The local shear strength of the concrete shell is evaluated on the basis of ACI 349-85 [34], Section 11.11.2.1, discounting the reinforcing and the steel internal shell. The concrete shell shear capacity is also evaluated for missile loading using ACI 349-85, Section 11.7.

The cask configuration used in this analysis combines the height of the tallest (Class 3 PWR) cask with the weight and center of gravity of the lightest (Class 1 PWR) cask. This configuration bounds all other configurations for cask stability. The cask properties considered in this evaluation are:

H = Cask Height = 225.88 in (Class 3 PWR)
D_o = Cask Outside Diameter = 136.0 in
D_i = Inside Diameter of concrete shell = 79.5 in
W_{VCC} = Weight of the cask with canister, basket and full fuel load = 285,000 lbs
(285,000 lbs is conservatively used [slightly lighter than the Class 1 PWR cask weight])
A_c = Cross section area of concrete shell = 9,563 in^2
I_c = Moment of inertia of concrete shell = 14.83×10^6 in^4
f_c' = Compressive strength of concrete shell = 4,000 psi

Tornado Wind Loading (Concrete Cask)

The tornado wind velocity is transformed into an effective pressure applied to the cask using procedures delineated in ANSI/ASCE 7-93 Building Code Requirements for Minimum Design Loads in Buildings and Other Structures. The maximum pressure, q, is determined from the maximum tornado wind velocity as follows:

$$q = (0.00256)\ V^2 \text{ psf}$$

where:

V = Maximum tornado wind speed = 360 mph

The velocity pressure exposure coefficient for local terrain effects K, Importance Factor I, and the Gust Factor G, may be taken as unity (1) for evaluating the effects of tornado wind velocity pressure. Then:

$$q = (0.00256)(360)^2 = 331.8 \text{ psf}$$

Considering that the cask is small with respect to the tornado radius, the velocity pressure is assumed uniform over the projected area of the cask. Because the cask is vented, the tornado-induced pressure drop is equalized from inside to outside and has no effect on the cask structure.

The total wind loading on the projected area of the cask, F_w is then computed as:

$$F_w = q \times G \times C_f \times A_p$$
$$= 36{,}100 \text{ lbs}$$

where:

q = Effective velocity pressure (psf) = 331.8 psf.

C_f = Force Coefficient = 0.51 (ASCE 7-93, Table 12 with $D\,q^{1/2} = 206.4$ for a moderately smooth surface, h/D = 18.8 ft /11.3 ft = 1.7)

A_f = Projected area of cask = (225.88 in × 136.0 in)/144 = 213.3 ft^2

G = Constant = 1.0

The wind overturning moment, M_w, is computed as:

$$M_w = F_w \times H/2 = 36{,}100 \text{ lbs} \times 225.88 \text{ in}/12 \times 1/2 \cong 340{,}000 \text{ ft-lbs}$$

where H is the cask height.

The stability moment, M_s, of the cask (with the canister, basket and no fuel load) about an edge of the base, is:

$$M_s = W_{cask} \times D_o/2 = 1.52\times10^6 \text{ ft-lbs}$$

where:

D_o = Cask base plate diameter = 128.0 in

W_{cask} = Weight of the cask with canister
$\cong$ 285,000 lbs

ASCE 7-93 requires that the overturning moment due to wind load shall not exceed two-thirds of the dead load stabilizing moment unless the structure is anchored. Therefore, the margin of safety, MS, against overturning is:

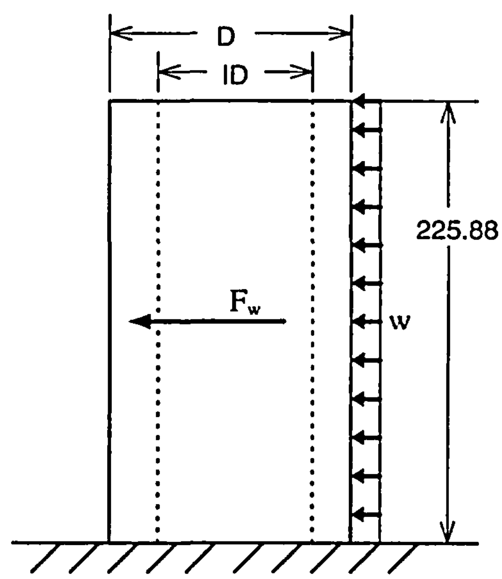
$$MS = \frac{M_S}{M_W} - 1 = \frac{(0.67)1.52\times10^6}{3.40\times10^5} - 1 = +2.00$$

A coefficient of friction of 0.13 (36,100/285,000) between the cask base and the concrete pad on which it rests will inhibit sliding.

Against a coefficient of friction of steel on concrete of approximately 0.35 [23], the margin of safety, MS, against sliding is:

$$MS = \frac{0.35}{0.13} - 1 = +1.69$$

The stresses in the concrete due to the tornado wind load are conservatively calculated below. The concrete cask is considered to be fixed at its base.



F_W = 36,100 lbs

D = 136.0 in. (concrete outside diameter)

ID = 79.5 in. (concrete inside diameter)

H = 225.8 in. /12 = 18.82 ft

A = $\pi (D^2 - ID^{2)} / 4 = 9{,}563 \text{ in}^2$

I = $\pi (D^4 - ID^{4)} / 64 = 14.83\times10^6 \text{ in}^4$

(Moment of Inertia)

M = $\dfrac{F_w \times H}{2} \cong 340{,}000$ lbs-ft

Maximum stresses:

$$\sigma = \frac{Mc}{I} = 18.7 \text{ psi} \qquad \text{(tension or compression)}$$

where:

c = D/2 = 68.0 in.

The compressive stresses are included in the load combination No. 3 in Table 3.4.4.2-1, since they are governing stresses for the load combination. As shown in Tables 3.4.4.2-1 and 3.4.4.2-2, the maximum combined stresses for the load combination of dead, live, thermal and tornado wind are less than the allowable stress.

Tornado Missile Loading (Concrete Cask)

The Vertical Concrete Cask is designed to withstand the effects of impacts associated with postulated tornado generated missiles identified in NUREG-0800, Section 3.5.1.4.III.4, Spectrum I missiles. These missiles consist of: 1) a massive high kinetic energy missile (4,000 lbs automobile, with a frontal area of 20 square feet that deforms on impact); 2) a 280 lbs, 8-inch-diameter armor piercing artillery shell; and 3) a small 1-inch diameter solid steel sphere. All of these missiles are assumed to impact in a manner that produces the maximum damage at a velocity of 126 mph (35% of the maximum tornado wind speed of 360 mph). The cask is evaluated for impact effects associated with each of the above missiles.

The principal dimensions and moment arms used in this evaluation are shown in Figure 11.2.11-1.

The concrete cask has no openings except for the four outlets at the top and four inlets at the bottom. The upper openings are configured such that a 1-inch diameter solid steel missile cannot directly enter the concrete cask interior. Additionally, the canister is protected by the canister structural and shield lids. The canister is protected from small missiles entering the lower inlets by a steel pedestal (bottom plate). Therefore, a detailed analysis of the impact of a 1-inch diameter steel missile is not required.

Concrete Shell Local Damage Prediction (Penetration Missile)

Local damage to the cask body is assessed by using the National Defense Research Committee (NDRC) formula [32]. This formula is selected as the basis for predicting depth of penetration and minimum concrete thickness requirements to prevent scabbing. Penetration depths calculated by using this formula have been shown to provide reasonable correlation with test results [33].

Concrete shell penetration depths are calculated as follows:

$x/2d \leq 2.0$

where:

d = Missile diameter = 8 in

x = Missile penetration depth = $[4KNWd^{-0.8}(V/1000)^{1.8}]^{0.5}$

where:

K= Coefficient depending on concrete strength

= $180/(f_c')^{1/2} = 180/(4000)^{1/2} = 2.846$

N= 1.14 Shape factor for sharp nosed missiles

W= Missile weight = 280 lbs

V= Missile velocity = 126 mph = 185 ft/sec

$x = [(4)(2.846)(1.14)(280)(8^{-0.8})(185/1000)^{1.8}]^{0.5}$

= 5.75 inches

$x/2d = 5.75/(2)(8) = 0.359 < 2.0$

The minimum concrete shell thickness required to prevent scabbing is three times the predicted penetration depth of 5.75 inches based on the NDRC formula, or 17.25 inches. The concrete cask wall thickness includes 28.25 inches of concrete, which is more than the thickness required to prevent damage due to the penetration missile. This analysis conservatively neglects the 2.5-inch steel shell at the inside face of the concrete shell.

Closure Plate Local Damage Prediction (Penetration Missile)

The concrete cask is closed with a 1.5-inch thick steel plate bolted in place. The following missile penetration analysis shows that the 1.5-inch steel closure plate is adequate to withstand the impact of the 280-lbs armor piercing missile, impacting at 126 mph.

The perforation thickness of the closure steel plate is calculated by the Ballistic Research Laboratories Formula with K = 1, formula number 2-7, in Section 2.2 of Topical Report BC-TOP-9A, Revision 2 [35].

$$T = [0.5 m_m V^2]^{2/3}/672d = 0.523 \text{ inch}$$

where:

T = Perforation thickness

m_m = Missile mass = W/g = 280 lbs/32.174 ft/sec^2 = 8.70 slugs

g = Acceleration of gravity = 32.174 ft/sec^2

BC-TOP-9A recommends that the plate thickness be 25% greater than the calculated perforation thickness, T, to prevent perforation. Therefore, the recommended plate thickness is:

$$T \quad = \quad 1.25 \times 0.523 \text{ in.} = 0.654 \text{ in.}$$

The closure plate is 1.5 inches thick; therefore the plate is adequate to withstand the local impingement damage due to the specified armor piercing missile.

Overall Damage Prediction for a Tornado Missile Impact (High Energy Missile)

The concrete cask is a free-standing structure. Therefore, the principal consideration in overall damage response is the potential of upsetting or overturning the cask as a result of the impact of a high energy missile. Based on the following analysis, it is concluded that the cask can sustain an impact from the defined massive high kinetic energy missile and does not overturn.

From the principle of conservation of momentum, the impulse of the force from the missile impact on the cask must equal the change in angular momentum of the cask. Also, the impulse force due to the impact of the missile must equal the change in linear momentum of the missile. These relationships may be expressed as follows:

Change in momentum of the missile, during the deformation phase

$$\int_{t_1}^{t_2} (F)(dt) = m_m(v_2 - v_1)$$

where:

F	=	Impact impulse force on missile
m_m	=	Mass of missile = 4000 lbs/g = 124 slugs/12 = 10.4 (lbs sec^2 /in)
t_1	=	Time at missile impact
t_2	=	Time at conclusion of deformation phase
v_1	=	Velocity of missile at impact = 126 mph = 185 ft/sec
v_2	=	Velocity of missile at time t_2

The change in angular momentum of the cask, about the bottom outside edge/rim, opposite the side of impact is:

$$\int_{t_1}^{t_2} M_c(dt) = \int_{t_1}^{t_2} (H)(F)(dt) = I_m(\omega_1 - \omega_2)$$

Substituting,

$$\int (F)(dt) = m_m(v_2 - v_1) = \frac{I_m(\omega_1 - \omega_2)}{H}$$

where:

M_c	=	Moment of the impact force on the cask
I_m	=	Concrete cask mass moment of inertia, about point of rotation on the bottom rim
ω_1	=	Angular velocity at time t_1
ω_2	=	Angular velocity at time t_2
m_c	=	Mass of concrete cask = $W_c/g = 285{,}000/32.174$
	=	8858.1 slugs/12 = 738.2 lbs sec^2/in)
I_{mx}	=	Mass moment of inertia, VCC cask about x axis through its center of gravity
	≅	$1/12(m_c)(3r^2 + H^2)$ (Conservatively assuming a solid cylinder.)
	≅	$(1/12)(738.2)\,[(3)(68.0)^2 + (225.88)^2] = 3.99\times10^6$ lbs-sec^2-in
I_m	=	$I_{mx} + (m_c)(d_{CG})^2 = 3.99\times10^6 + (738.2)(126.23)^2 = 15.75\times10^6$ lbs-sec^2-in.
d_{CG}	=	The distance between the cask CG and a rotation point on base rim = 126.23 in.

(See Figure 11.2.11-1.)

Based on conservation of momentum, the impulse of the impact force on the missile is equated to the impulse of the force on the cask.

$$m_m(v_2 - v_1) = I_m(\omega_1 - \omega_2)/H$$

at time t_1, v_1 = 185 ft/sec and ω_1 = 0 rad/sec

at time t_2, v_2 = 0 ft/sec

During the restitution phase, the final velocity of the missile depends upon the coefficient of restitution of the missile, the geometry of the missile and target, the angle of incidence, and on the amount of energy dissipated in deforming the missile and target. On the basis of tests conducted by EPRI, the final velocity of the missile, v_f, following the impact is assumed to be zero. Assuming conservatively that all of the missile energy is transferred to the cask, and equating the impulse of the impact force on the missile to the impulse of the force on the cask,

$$(10.4)(v_2 - 185 \text{ ft/sec} \times 12 \text{ in/ ft}) = 15.75\times10^6 \text{ lbs-sec}^2\text{-in } (0 - \omega_2)/225.88$$

$$\omega_2 = 0.331 \text{ rad/sec (when } v_2 = 0)$$

Back solving for v_2

$$v_2 = 261.6 \times \omega_2 = (261.6)(0.331) = 86.6 \text{ in/sec}$$

where the distance from the point of missile impact to the point of cask rotation is $\sqrt{132.0^2 + 225.88^2} = 261.6$ in. (See Figure 11.2.11-1). The line of missile impact is conservatively assumed normal to this line.

Equating the impulse of the force on the missile during restitution to the impulse of the force on the cask yields:

$$-[m_m(v_f - v_2)] = I_m (\omega_f - \omega_2)/H$$

$$-[10.4(0 - 86.6)] = 15.75\times10^6 \text{ lbs-sec}^2\text{-in } (\omega_f - 0.331)/225.88$$

$$\omega_f = 0.344 \text{ rad/sec}$$

where:

v_f = 0
v_2 = 86.6 in/sec
ω_2 = 0.331 rad/sec

Thus, the final energy of the cask following the impact, E_k, is:

$$E_k = (I_m)(\omega_f)^2/(2) = (15.75\times10^6)(0.344)^2/(2) = 9.32\times10^5 \text{ in-lb}_f$$

The change in potential energy, E_p, of the cask due to rotating it until its center of gravity is above the point of rotation (the condition where the cask will begin to tip-over and the height of the center of gravity has increased by the distance, h_{PE}, see Figure 11.2.11-1) is:

$$E_p = (W_{cask})(h_{PE})$$
$$E_p = 285{,}000 \text{ lbs} \times 17.43 \text{ in}$$
$$E_p = 4.97\times10^6 \text{ in-lb}_f$$

The massive high kinetic energy tornado generated missile imparts less kinetic energy than the change in potential energy of the cask at the tip-over point. Therefore, cask overturning from missile impact is not postulated to occur. The margin of safety, MS, against overturning is:

$$MS = \frac{0.67\times4.97\times10^6}{9.32\times10^5} - 1 = +2.57$$

Combined Tornado Wind and Missile Loading (High Energy Missile)

The cask rotation due to the heavy missile impact is calculated as (See Figure 11.2.11-1 for dimensions):

$$h_{KE} = E_k / W_c = 9.32\times10^5 \text{ in-lb}_f / 285{,}000 \text{ lbs} = 3.27 \text{ in}$$

Then

$$\cos\beta = (h_{CG} + h_{KE}) / d_{CG}$$
$$\cos\beta = (108.8 + 3.27) / 126.23 = 0.8878$$
$$\beta = 27.4 \text{ deg}$$
$$\cos\alpha = 108.8 / 126.23 = 0.8619$$
$$\alpha = 30.5 \text{ deg}$$
$$e = d_{CG} \sin\beta$$
$$e = 126.23 \sin 27.4 = 58.1 \text{ in}$$

Therefore, cask rotation after impact = $\alpha - \beta = 30.5 - 27.4 = 3.1$ deg

The available gravity restoration moment after missile impact:

$= (W_c)(e)$
$= 285{,}000 \text{ lbs} \times 58.1 \text{ in}/12$
$= 1.38\times10^6$ ft-lbs >> Tornado Wind Moment $= 3.40\times10^5$ ft-lbs

Therefore, the combined effects of tornado wind loading and the high energy missile impact loading will not overturn the cask. Considering that the overturning moment should not exceed two-thirds of the restoring stability moment, the margin of safety, MS, is:

$$MS = \frac{0.67\left(1.38\times10^6\right)}{3.40\times10^5} - 1 = +1.72$$

Local Shear Strength Capacity of Concrete Shell (High Energy Missile)

This section evaluates the shear strength of the concrete at the top edge of the concrete shell due to a high energy missile impact based on ACI 349-85, Chapter 11, Section 11.11.2.1, on concrete punching shear strength.

The force developed by the massive high kinetic energy missile having a frontal area of 20 square feet, is evaluated using the methodology presented in Topical Report, BC-TOP-9A.

$F = 0.625(v)(W_M)$
$F = 0.625(185 \text{ ft/sec})(4{,}000 \text{ lbs}) = 462.5 \text{ kips}$
$F_u = LF \times F = 1.1 \times 462.5 = 508.8 \text{ kips}$

Based on a rectangular missile contact area, having proportions of 2 (horizontal) to 1 (vertical) and the top of the area flush with the top of the concrete cask, the required missile contact area based on the concrete punching shear strength (neglecting reinforcing) is calculated as follows.

$V_c = (2+4/\beta_c)\,(f_c')^{1/2} b_o\, d$, where $\beta_c = 2/1 = 2$

$V_c = 4\,(f_c')^{1/2} b_o\, d$

$d = 28.25 \text{ in} - 3.25 \text{ in} = 25 \text{ in}$

$(f_c')^{1/2} = 63.24$ psi, where $f_c' = 4{,}000$ psi

b_o = perimeter of punching shear area at d/2 from missile contact area

$b_o = (2b + 25) + 2(b + 12.5) = 4b + 50$

$V_u = \Phi(V_c + V_s)$, where $V_s = 0$, assuming no steel shear

$V_u = \Phi V_c = \Phi\, 4\,(f_c')^{1/2} b_o\, d = (0.85)(4)(63.24)(4b + 50)(25) = 21{,}501\, b + 268{,}770.$

Setting, V_u equal to F_u and solving for b

$508.8\times10^3 = 21{,}501\, b + 268{,}770$

$b = 11.12$ inches (say 1.0 ft)

The implied missile impact area required $= 2b \times b = 2 \times 1 \times 1 = 2.0$ sq ft < 20.0 sq ft

Thus, the concrete shell alone, based on the concrete conical punching strength and discounting the steel reinforcement and shell, has sufficient capacity to react to the high energy missile impact force.

The effects of tornado winds and missiles are considered both separately and combined in accordance with NUREG-800, Section 3.3.2 II.3.d. For the case of tornado wind plus missile loading, the stability of the cask is assessed and found to be acceptable. Equating the kinetic energy of the cask following missile impact to the potential energy yields a maximum postulated rotation of the cask, as a result of the impact, of 3.0 degrees. Applying the total tornado wind load to the cask in this configuration results in an available restoring moment considerably greater than the tornado wind overturning moment. Therefore, overturning of the cask under the combined effects of tornado winds, plus tornado-generated missiles, does not occur.

Tornado Effects on the Canister

The postulated tornado wind loading and missile impacts are not capable of overturning the cask, or penetrating the boundary established by the concrete cask. Consequently, there is no effect on the canister. Stresses resulting from the tornado-induced decreased external pressure are bounded by the stresses due to the accident internal pressure discussed in Section 11.2.1.

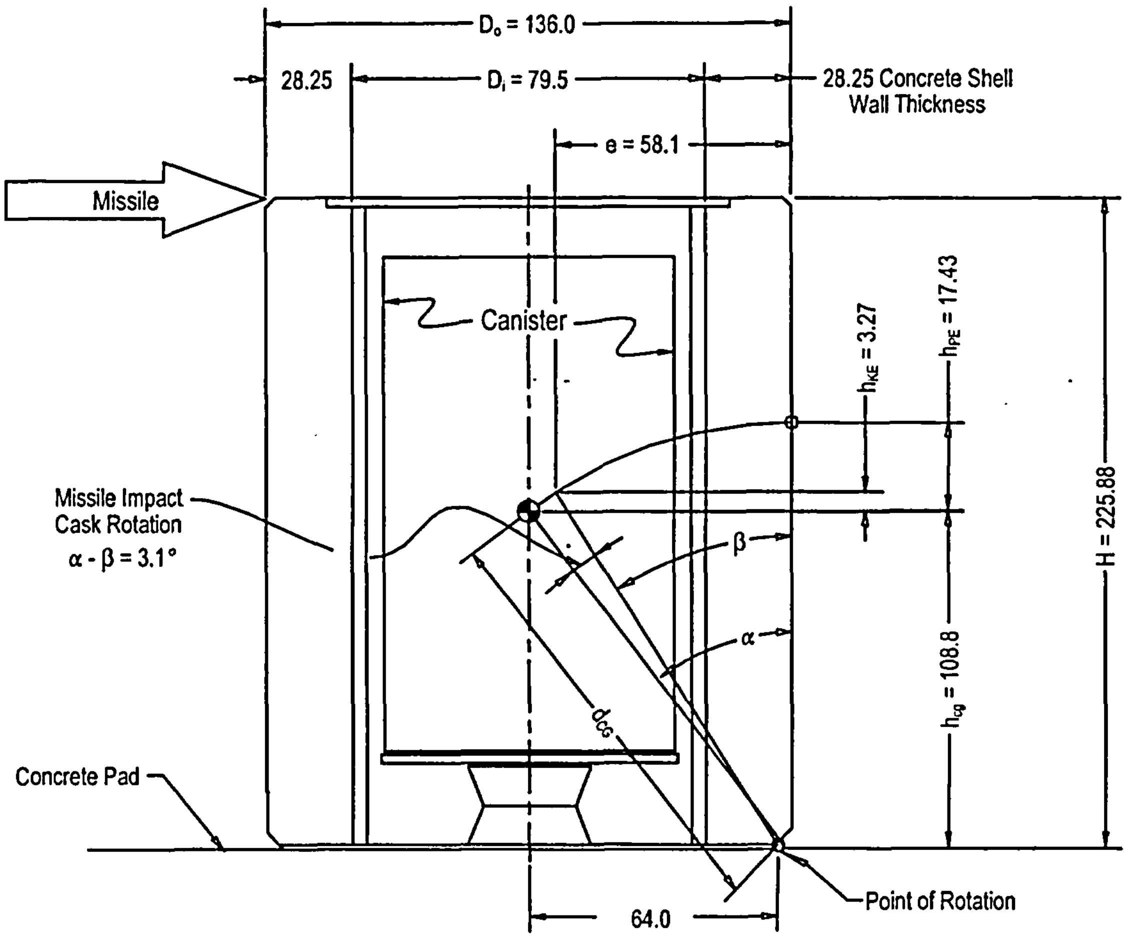
11.2.11.4 Corrective Actions

A tornado is not expected to result in the need to take any corrective action other than an inspection of the ISFSI. This inspection would be directed at ensuring that inlets and outlets had not become blocked by wind-blown debris and at checking for obvious (concrete) surface damage.

11.2.11.5 Radiological Impact

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 penetrating 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.

Figure 11.2.11-1 Principal Dimensions and Moment Arms Used in Tornado Evaluation



11.2.12 Tip-Over of Vertical Concrete Cask

Tip-over of the Vertical Concrete Cask (cask) is a non-mechanistic, hypothetical accident condition that presents a bounding case for evaluation. There are no design basis accidents that result in the tip-over of the cask.

Functionally, the cask does not suffer significant adverse consequences due to this event. The concrete cask, canister, and basket maintain design basis shielding, geometry control of contents, and contents confinement performance requirements.

Results of the evaluation show that supplemental shielding will be necessary, following the tip-over and until the cask can be righted, because the bottom ends of the concrete cask and the canister have significantly less shielding than the sides and tops of these components.

11.2.12.1 Cause of Cask Tip-Over

A tip-over of the cask is possible in an earthquake that significantly exceeds the design basis described in Section 11.2.8. No other events related to design bases are expected to result in a tip-over of the cask.

11.2.12.2 Detection of Cask Tip-Over

The tipped-over configuration of the concrete cask will be obvious during site inspection following the initiating event.

11.2.12.3 Analysis of Cask Tip-Over

For a tip-over event to occur, the center of gravity of the concrete cask and loaded canister must be displaced beyond its outer radius, i.e., the point of rotation. When the center of gravity passes beyond the point of rotation, the potential energy of the cask and canister is converted to kinetic energy as the cask and canister rotate toward a horizontal orientation on the ISFSI pad. The subsequent motion of the cask is governed by the structural characteristics of the cask, the ISFSI pad and the underlying soil.

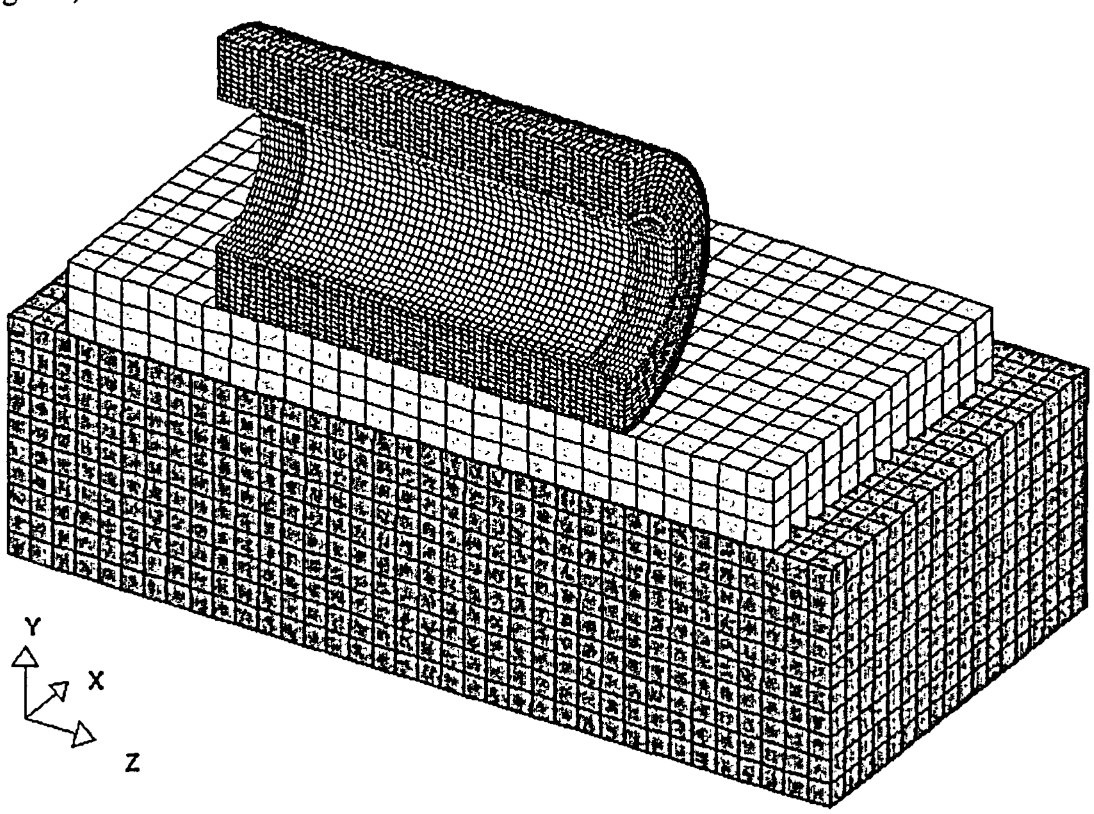
The objective of the evaluation of the response of the concrete cask in the tip-over event is to determine the maximum acceleration to be used in the structural evaluation of the loaded canister and basket (Section 11.2.12.4). The methodology to determine the concrete cask response follows the methodology contained in NUREG/CR-6608, "Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet Onto Concrete Pads" [38]. The LS-DYNA program is used in the evaluation. The validation of the analysis methodology is shown in Section 11.2.12.3.3.

The parameters of the ISFSI pad and foundation are:

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

11.2.12.3.1 Analysis of Cask Tip-Over for PWR Configurations

The finite element model includes a half section of the concrete cask, the concrete ISFSI pad and soil subgrade, as shown:



The concrete pad in the model corresponds to a pad 30-feet by 30-feet square and 3-feet thick, supporting one concrete cask in the center of the pad. The soil under the concrete pad is considered to be 35 feet by 35 feet square and 10 feet thick. Only one-half of the concrete cask, pad and soil configuration is modeled due to symmetry.

The concrete is represented as a homogeneous isotropic material. The concrete cask (outer shell) and the pad are modeled as material Type Number 16 in LS-DYNA. The values for concrete pad and soil properties provided below are typical values for the input to the LS-DYNA model. The material properties used in the model for the concrete ISFSI pad are:

Compressive Strength (f'_c)	$= 5{,}000$ psi	
Density (ρ_c)	$= 125$ pcf	
Poisson's Ratio (ν_c)	$= 0.22$	(NUREG/CR-6608 [38])
Modulus of Elasticity (E_c)	$= 33\,\rho_c^{1.5}\sqrt{f'_c} = 3.261\text{E}6$ psi	(ACI 318-95)
Bulk Modulus (K_c)	$= \dfrac{E_c}{3(1-2\nu_c)} = 1.941\text{E}6$ psi	(Blevins [19])

The material properties used in the model for the soil below the ISFSI pad are:

Density	$= 160$ pcf	
Poisson's Ratio (ν_s)	$= 0.45$	(NUREG/CR-6608)
Modulus of Elasticity	$= 60{,}000$ psi	

The concrete cask steel liner has the properties:

Density	$= 0.284$ lbs/in^3
Poisson's ratio	$= 0.31$
Modulus of elasticity	$= 2.9\text{E}7$ psi

To account for the weight of the shield plug, the loaded canister, and the concrete cask pedestal, effective densities are used for the elements in the first row of the steel liner in the model adjacent to the impact plane of symmetry. These densities represent the regions (6° in the circumferential direction) of the steel liner subjected to the weight of the shield plug, the loaded canister and the pedestal, during the side impact (tip-over) condition. The contact angle (6°) is determined based on the canister/basket analysis for the tip-over condition (Section 11.2.12.4).

Boundary Conditions and Initial Conditions

A friction coefficient of 0.25 is used at the interface between the steel liner and the concrete shell, between the concrete cask and the pad, and between the pad and the soil. For all the embedded faces (three side surfaces and the bottom surface) of the soil in the model, the displacements in the direction normal to the surface are restrained. The symmetry boundary conditions are applied for all nodes at the plane of symmetry.

The initial condition corresponds to the concrete cask in a horizontal position with an initial vertical velocity into the concrete pad. The pad and soil are initially at rest.

The distribution of initial velocity of the concrete cask is simulated by applying an angular velocity (ω) to the entire cask. The point of rotation is taken to be the lower edge of the base of the concrete cask. The angular velocity value is computed by considering energy conservation at the cask "center of gravity over corner" tip condition versus the side impact condition.

From energy conservation:

$$mgh = \frac{I\omega^2}{2}$$

where:

mg = conservative, bounding weight of the loaded concrete cask
= 307,000 lbs (PWR Class 1*)
= 319,000 lbs (PWR Class 2*)
= 324,000 lbs (PWR Class 3*)

h = height change of the concrete cask center of gravity (L_{CG}) = $\sqrt{R^2 + \left(\frac{L_{CG}}{2}\right)^2} - R$

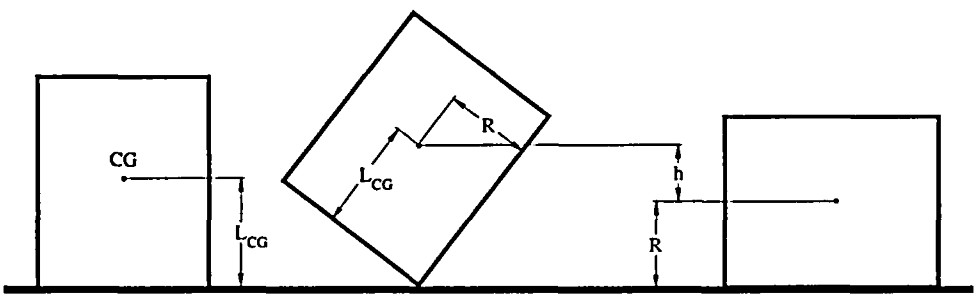
= 60.47 inches (PWR Class 1)
= 63.88 inches (PWR Class 2)
= 67.33 inches (PWR Class 3)

where:

L_{CG} = location of the center of gravity above the pad for the concrete cask
= 109.0 inches (PWR Class 1)
= 113.0 inches (PWR Class 2)
= 117.0 inches (PWR Class 3)

R = radius of the concrete cask = 68 inches

I = total mass moment of inertia of the concrete cask about the point of rotation
= 16,338,092 lbs-sec^2-inch (PWR Class 1)
= 18,091,985 lbs-sec^2-inch (PWR Class 2)
= 19,470,873 lbs-sec^2-inch (PWR Class 3)



The mass moment of inertia for the concrete shell and the steel liner is calculated using the formula for a hollow right circular cylinder (Blevins).

$$I = \frac{m}{12}(3R_1^2 + 3R_2^2 + 4L^2) + md^2$$

where:

m = mass (lbs-sec^2/in)
R_1 and R_2 = the outer and inner radius of the cylinder (inch)
L = height of the cylinder (inch)
d = distance between the center of gravity and the point of rotation (inch)

For the mass of the shield plug, loaded canister and the pedestal, the formula for the moment of inertia for a solid cylinder is used:

$$I = \frac{m}{12}(3R^2 + 4L^2) + md^2$$

where:

m = mass of the cylinder ($lbs\text{-}sec^2/in$)
R = radius of the cylinder (inch)
L = height of the cylinder (inch)
d = distance between the two pivot axes (inch)

The angular velocity is given by $\omega = \sqrt{\frac{2mgh}{I}}$

$= 1.51$ radians/sec (PWR Class 1)
$= 1.50$ radians/sec (PWR Class 2)
$= 1.50$ radians/sec (PWR Class 3)

Filter Frequency

The accelerations are evaluated at the inner surface of the cask liner, which physically corresponds to the interface of the liner and the loaded canister nearest the plane of impact. Following the methodology contained in NUREG/CR-6608, the Butterworth filter is applied to the nodal accelerations. The filter frequency is based on the fundamental mode of the cask.

The fundamental natural frequency of a beam in transverse vibration due to flexure only is given by Blevins as:

$$f = \frac{\lambda^2}{2\pi}\sqrt{\frac{EI}{\rho A L^4}}$$

where:

$\lambda = 3.92660231$ for a pin-free beam

The frequencies of the concrete (f_c) and the steel liner (f_s) are computed as:

Area of concrete cask = $\pi\ \{(68)^2 - (39.75)^2\} = 9562.8\ in^2$

Moment of inertia of concrete cask = $\frac{\pi}{4}\{(68)^4 - (39.75)^4\} = 14{,}832{,}070\ in^4$

$$f_c = 823{,}568 \frac{\lambda^2}{L^2}$$
= 290 Hz (PWR Class 1)
= 267 Hz (PWR Class 2)
= 249 Hz (PWR Class 3)

Area of steel liner = $\pi\ \{(39.75)^2 - (37.25)^2\} = 604.8\ \text{in}^2$

Moment of inertia of steel liner = $\frac{\pi}{4}\{(39.75)^4 - (37.25)^4\} = 448{,}673\ \text{in}^4$

$$f_s = 861{,}707 \frac{\lambda^2}{L^2}$$
= 304 Hz (PWR Class 1)
= 279 Hz (PWR Class 2)
= 260 Hz (PWR Class 3)

Since the concrete cask is short compared to its diameter, the contribution of the flexibility due to shear is also incorporated. This is accomplished by using Dunkerley's formula (Blevins). The system frequency is:

$$\frac{1}{f^2} = \frac{1}{f_c^2} + \frac{1}{f_s^2}$$

Thus, the system frequencies are 210 Hz (PWR Class 1), 193 Hz (PWR Class 2), and 180 Hz (PWR Class 3). Cut-off frequencies of 210 Hz (PWR Class 1), 195 Hz (PWR Class 2), and 180 Hz (PWR Class 3) are applied to filter the analysis results and measure the peak accelerations.

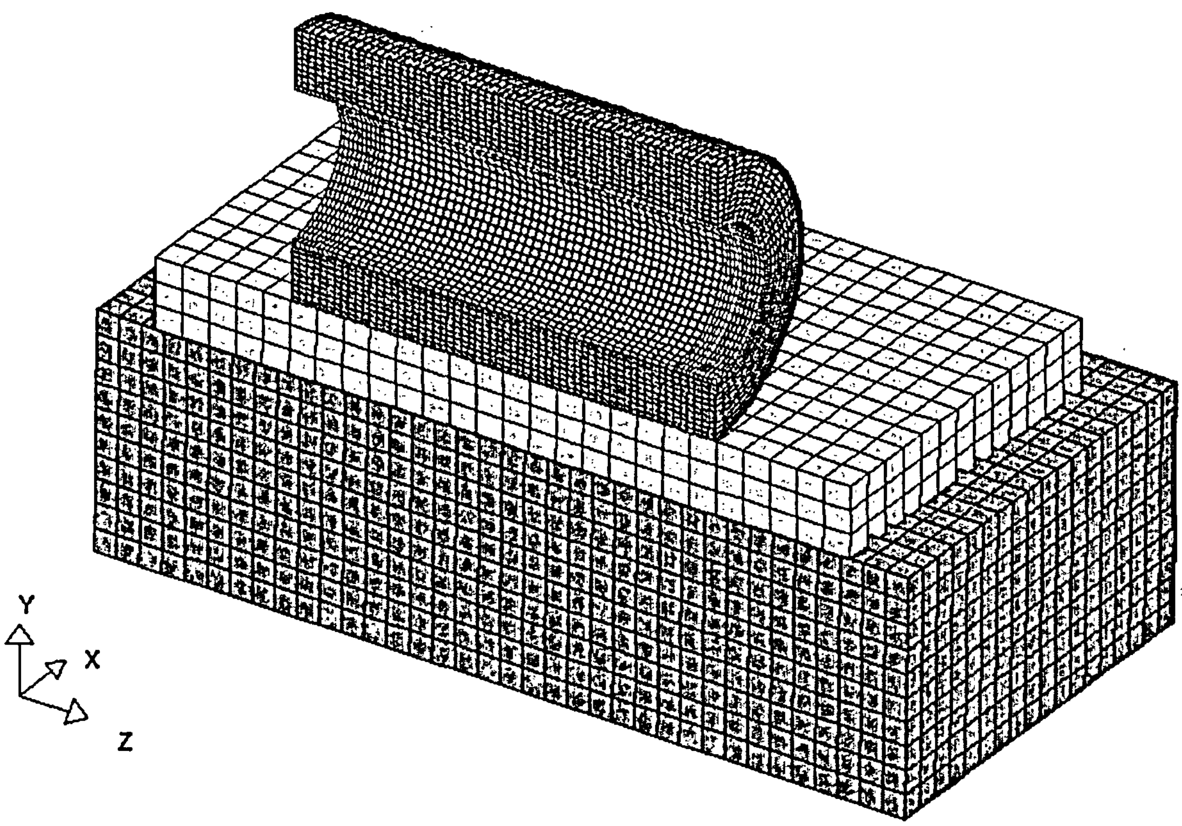
Results of the Transient Analysis

The maximum accelerations at key locations of the concrete cask liner that are required in the evaluation of the loaded canister/basket model (Section 11.2.12.4) are:

Location on Component	Position Measured from the Bottom of the Concrete Cask (inches) PWR Class 1	PWR Class 2	PWR Class 3	Acceleration (g) PWR Class 1	PWR Class 2	PWR Class 3
Top support disk	176.7	185.2	196.3	30.0	31.3	33.4
Top of the canister structural lid	197.9	207.0	214.6	32.8	34.2	35.7

11.2.12.3.2 Analysis of Cask Tip-Over for BWR Configurations

The BWR finite element model is similar to that for the PWR configuration. The concrete pad in this model corresponds to a pad 30-feet by 30-feet and 3-feet thick, supporting one concrete cask in the center of the pad. The soil under the concrete pad is considered to be 35-feet by 35-feet in area and 10-feet thick.



The material properties used in this model for the soil below the ISFSI pad are the same as those for the PWR model, except the modulus of elasticity of the soil is 30,000 psi.

Initial Conditions

The initial velocity for the BWRs was calculated in the same fashion as for the PWRs, but using the following data:

mg = total weight of the loaded concrete cask
= 322,000 lbs (BWR Class 4)
= 328,000 lbs (BWR Class 5)

h = height change of the concrete cask center of gravity (L_{CG}) = $\sqrt{R^2 + L_{CG}^2} - R$
= 64.74 inches (BWR Class 4)
= 66.46 inches (BWR Class 5)

where:

L_{CG} = location of the center of gravity above the pad for the concrete cask
= 114.0 inches (BWR Class 4)
= 116.0 inches (BWR Class 5)

I = total mass moment of inertia of the concrete cask about the point of rotation
= 18,437,994 lbs-sec^2-inch (BWR Class 4)
= 19,422,461 lbs-sec^2-inch (BWR Class 5)

The angular velocity is given by $\omega = \sqrt{\frac{2mgh}{I}}$
= 1.50 radians/sec (BWR Class 4)
= 1.50 radians/sec (BWR Class 5)

Conservatively, an angular velocity of 1.51 rad/sec is applied to the entire cask of each Class.

Filter Frequency

The filter frequency for the BWRs was calculated in the same fashion as for the PWRs but using the following data:

$f_c = 823{,}568 \frac{\lambda^2}{L^2}$
= 263 Hz (BWR Class 4)
= 252 Hz (BWR Class 5)

$$f_s = 861{,}707\,\frac{\lambda^2}{L^2}$$
$$= 275 \text{ Hz (BWR Class 4)}$$
$$= 264 \text{ Hz (BWR Class 5)}$$

Thus, the system frequencies are 190 Hz (BWR Class 4), and 182 Hz (BWR Class 5). Cut-off frequencies of 190 Hz (BWR Class 4) and 185 Hz (BWR Class 5) are conservatively applied to filter the analysis results and measure the peak accelerations.

Results of the Transient Analysis

The maximum accelerations at key locations of the concrete cask liner that are required in the evaluation of the loaded canister/basket model (Section 11.2.12.4) are:

Location on Component	Position Measured from the bottom of the Concrete Cask (inches)		Acceleration (g)	
	BWR-4	BWR-5	BWR-4	BWR-5
Top support disk	178.7	182.9	24.2	24.2
Top of the canister structural lid	208.4	213.2	27.9	28.0

11.2.12.3.3 Validation of the Analysis Methodology

Tip-over tests of a steel billet onto a concrete pad were conducted and reported in NUREG/CR-6608. The purpose of the tests was to provide data, against which, analysis methodology could be validated. Using the geometry described in the benchmark along with the modeling methodology, these analyses were re-performed using the LS-DYNA program.

Using the filter frequency reported in the NUREG/CR-6608 benchmark, the following results are obtained:

Nodes / Gauge Location	Maximum Experiment (g)	NAC Analysis (g)
16115 / A1	237.5	237.1
17265 / A5	231.5	229.4

11.2.12.4 Analysis of Canister and Basket for Cask Tip-Over Event

Structural evaluations are performed for the transportable storage canister and fuel basket support disks for tip-over accident conditions for both PWR and BWR fuel configurations. ANSYS finite element models are used to evaluate this side impact loading condition.

Comparison of maximum stress results to the allowable stress intensities shows that the canister and support disks are structurally adequate for the concrete cask tip-over condition and satisfies the stress criteria in accordance with the ASME Code, Section III, Division I, Subsection NB and NG, respectively.

The structural response of the PWR and BWR canisters and fuel baskets to the tip-over condition is evaluated using ANSYS three-dimensional finite element models consisting of the top portion of the canister, the top five fuel basket support disks, and the fuel basket top weldment disk. The PWR with Fuel Class 1 configuration is used to evaluate the PWR canister and fuel basket, and the BWR with Fuel Class 4 configuration is used to evaluate the BWR canister and fuel basket. These two representative configurations are chosen because they bound the maximum load-per-support disk for the respective fuel configurations. For each fuel configuration analyzed, the structural analyses are performed for various fuel basket drop orientations in order to ensure that the maximum primary membrane (P_m) and primary membrane plus primary bending ($P_m + P_b$) stresses are evaluated. For the PWR fuel configuration, fuel basket drop orientations of 0°, 18.22°, 26.28°, and 45° are evaluated (see Figure 11.2.12.4.1-1). For the BWR fuel configuration, fuel basket drop orientations of 0°, 31.82°, 49.46°, 77.92°, and 90° are evaluated (see Figure 11.2.12.4.2-1).

11.2.12.4.1 Analysis of Canister and Basket for PWR Configurations

Four three-dimensional models of the PWR canister and fuel basket are evaluated for side loading conditions that conservatively simulate a tip-over event while inside the concrete cask. In each model, a different fuel basket drop orientation is used. Three-dimensional half-symmetry models are used for the basket orientation of 0° and 45°, since half-symmetry is applicable based on the support disk geometry and the drop orientation. Three-dimensional full-models are used for the basket drop orientations of 18.22° and 26.28°. Representative figures for the models are presented in this section (three-dimensional full-model with a basket orientation of 18.22°).

Model Description

The finite element model used to evaluate the PWR canister and fuel basket for the tip-over event is presented in Figure 11.2.12.4.1-2 through Figure 11.2.12.4.1-5. The figures presented are for the PWR canister and fuel basket model with a fuel basket drop orientation of 18.22° and are representative of the models for all drop orientations analyzed. Only half of the canister is shown in the figures to present the view of the fuel basket.

The canister shell, shield lid, and structural lids are constructed of SOLID45 elements, which have three degrees-of-freedom (UX, UY, and UZ) per node (see Figure 11.2.12.4.1-3). The interaction of the shield lid and structural lid with the canister shell (below the lid welds) is modeled using CONTAC52 elements with a gap size based on nominal dimensions. The interaction of the bottom edge of the shield lid with the support ring is modeled using COMBIN40 gap elements with a gap size of 1×10^{-8} inch. The interaction of the shield and structural lids is modeled using COMBIN40 gap elements with a conservative gap size of 0.08 inch, based on the flatness tolerance of the two lids. The interaction of the canister shell with the inner surface of the concrete cask is modeled using CONTAC52 elements with an initial gap size equal to the difference in the nominal radial dimensions of the outer surface of the canister and the inner surface of the concrete cask. A gap stiffness of 1×10^{6} lbs/inch is assigned to all CONTAC52 and COMBIN40 elements.

The top five fuel basket support disks and top weldment disk are modeled using SHELL63 elements, which have six degrees-of-freedom per node (UX, UY, UZ, ROTX, ROTY, and ROTZ). For the top (first) and fifth support disk, a refined mesh density is used (see Figure 11.2.12.4.1-4). The remaining support disks and the top weldment disk incorporate a course mesh density to account for the load applied to the canister shell. For the fine-meshed support disks, the tie-rod holes are modeled. CONTAC52 elements are included in the slits at the tie-rod holes. The interaction between the fuel basket support disks and top weldment disk and the canister shell is modeled using CONTAC52 elements with an initial gap size based on the nominal radial difference between the disks and canister shell. A gap stiffness of 1×10^{6} lbs/inch is assigned to all CONTAC52 elements.

The lower boundary of the canister shell (near the 5th support disk) is restrained in the axial (Y) direction. For the half-symmetry models (0° and 45° basket drop orientations), symmetry boundary conditions are applied at the plane of symmetry of the model. Since gap elements are used to represent the contact between the canister shell and the inner surface of the concrete cask, the nodes corresponding to the concrete cask are fixed in all degrees of freedom (UX, UY and UZ). In addition, the axial (UY) and in-plane rotational degrees of freedom (ROTX and ROTZ) of the basket nodes are fixed since there is no out-of-plane loading for the support disk for a side impact condition.

Loading of the model includes an internal pressure of 15 psig (design pressure for normal condition of storage) applied to the inner surfaces of the canister, pressure loads applied to the support disk slots, and the inertial loads. The pressure load applied to the support disk slots represents the weight of the fuel assemblies, fuel tubes, and aluminum heat transfer disks multiplied by the appropriate acceleration (see Figure 11.2.12.4.1-5). For the inertial loads, a maximum acceleration of 40g is conservatively applied to the entire model in the X-direction (see Figure 11.2.12.4.1-2) to simulate the side impact during the cask tip-over event.

As shown in Section 11.2.12.3.1, the maximum acceleration of the concrete cask steel liner at the locations of the top support disk and the top of the canister structural lid during the tip-over event is determined to be 33.4g and 35.7g, respectively. To determine the effect of the rapid application of the inertia loading for the support disk, a dynamic load factor (DLF) is computed using the mode shapes of a loaded support disk. The mode shapes corresponding to the in-plane motions of the disk are extracted using ANSYS. However, only the dominant modes with respect to modal mass participation factors are used in computing the DLF. The dominant resonance frequencies and corresponding modal mass participation factors from the finite element modal analyses of the PWR support disk are:

Frequency (Hz)	% Modal Mass Participation Factor
109.7	85.8
370.1	2.7
371.1	7.2

The mode shapes for these frequencies are shown in Figures 11.2.12.4.1-8 through 11.2.12.4.1-10. The displacement depicted in these figures is highly exaggerated by the ANSYS program in order to illustrate the modal shape. The stresses associated with the actual displacement are shown in Tables 11.2.12.4.1-4 through 11.2.12.4.1-8.

Using the acceleration time history of the concrete cask steel liner at the top support disk location developed from Section 11.2.12.3.1, the DLF is computed to be 1.18. Applying the DLF to the 33.4g results in a peak acceleration of 39.4g for the top support disk. The DLFs for the canister lids are considered to be unity since the lids have significant in-plane stiffness and are considered to be rigid (the structural lid is 3 inches thick and shield lid is 7 inches thick). Therefore, applying 40g to the entire canister/basket model is conservative.

A uniform temperature of 75°F is applied to the model to determine material properties during solution. During post processing for the support disk, temperature distribution with a maximum temperature of 700°F (at the center) and a minimum temperature of 400°F (at the outer edge) are conservatively used to determine the allowable stresses. A constant temperature of 500°F is used for the canister to determine the allowable stresses. These temperatures are the bounding temperatures for the normal, off-normal and accident conditions of storage.

Analysis Results for the Canister

The sectional stresses at 13 axial locations of the canister are obtained for each angular division of the model (a total of 80 angular locations for the full-models and 41 angular locations for the half-symmetry models). The locations for the stress sections are shown in Figure 11.2.12.4.1-6.

The stress evaluation for the canister is performed in accordance with the ASME Code, Section III, Subsection NB, by comparing the linearized sectional stresses against the allowable stresses. Allowable stresses are conservatively taken at a temperature of 500°F, except that 300°F and 250°F are used for the shield lid weld (Section 10) and the structural lid weld (Section 11). The calculated maximum temperatures for the shield lid and structural lid are 212°F and 204°F, respectively (Table 4.4.3-1). The allowable stresses for accident conditions are taken from Subsection NB as shown below. S_m and S_u are 14.8 ksi and 57.8 ksi, respectively, for Type 304L stainless steel (canister shell and structural lid). S_m and S_u are 17.5 ksi and 63.5 ksi, respectively, for Type 304 stainless steel (shield lid).

Stress Category	Accident (Level D) Allowable Stress
P_m	Lesser of 0.7 S_u or 2.4 S_m
P_m+P_b	Lesser of 1.0 S_u or 3.6 S_m

The primary membrane and primary membrane plus bending stresses for the PWR configuration for a 45° basket drop orientation are summarized in Table 11.2.12.4.1-1 and Table 11.2.12.4.1-2, respectively. The stress results for the canister are similar for all four basket drop orientation evaluations. The 45° basket orientation results are presented because this drop orientation results in the minimum margins of safety in the canister.

During the tip-over accident, the canister shell at the structural and shield lids is subjected to the inertial loads of the lids, which results in highly localized bearing stresses (Sections 7 through 9 at angular locations of approximately ± 4.5 degrees from the impact location). This stress is predominant because the weights of the structural and shield lids are transferred to the canister shell near these section locations. According to ASME Code Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions. Therefore, the stresses are not presented for the lid-bearing regions of the canister shell (Sections 7 through 9) in Tables 11.2.12.4.1-1 and 11.2.12.4.1-2. The stresses at the structural lid/canister shell weld region (Section 11) are determined by averaging the stresses over the impact region where the weld is in compression in the radial direction ($\sigma_x \leq 0.0$ psi). In accordance with ISG-15, Revision 0 [60], a 0.8 weld reduction factor is applied to the allowable stresses for the structural lid / canister shell weld. Use of the 0.8 factor is valid because the ultimate tensile strength of the weld material exceeds the base metal strength.

The stress evaluation results for the tip-over accident condition show that the minimum margin of safety in the canister for the PWR configuration is +0.29 for P_m stresses (Section 11). For P_m+P_b stresses, the margin of safety at is +0.64 (Section 11).

Analysis Results for the Support Disks

To evaluate the most critical regions of the support disk, a series of cross sections are considered. To aid in the identification of these sections, Figure 11.2.12.4.1-7 shows the locations on a support disk for the full-models. Table 11.2.12.4.1-3 lists the cross sections versus Point 1 and Point 2, which spans the cross section of the ligament in the plane of the support disk. Note that a local coordinate system (x and y parallel to the support disk ligaments) is used for the stress evaluation.

The stress evaluation for the support disk is performed according to ASME Code, Section III, Subsection NG. According to this subsection, linearized sectional stresses are to be compared against the allowable stresses. The allowable stresses for tip-over accident conditions are taken from Subsection NG as shown below, at the temperature of the Section. The temperature distribution of the disk is determined by a thermal conduction solution for a single disk with the maximum temperature of 700°F specified at the center and the minimum temperature of 400°F specified at the outer edge as boundary conditions.

Stress Category	Accident (Level D) Allowable Stresses
P_m	Lesser of 0.7 S_u or 2.4 S_m
P_m+P_b	Lesser of 1.0 S_u or 3.6 S_m

The shield lid and structural lid provide additional stiffness to the upper portion of the canister shell, which limits the shell and support disk deformations. Therefore, the maximum $P_m + P_b$ stress, and the minimum margin of safety, occur in the 5^{th} support disk (from the top of the basket), where the stiffness effect of the shield and structural lids is not present.

The stress evaluation results for the 5^{th} support disk for the tip-over condition are summarized in Table 11.2.12.4.1-4 for the four basket drop orientations evaluated. As shown in Table 11.2.12.4.1-4, the 26.28° drop orientation case generates the minimum margin of safety in the support disk; therefore, the P_m and $P_m + P_b$ stress intensities for the 26.28° basket drop orientation case are presented in Tables 11.2.12.4.1-6 and 11.2.12.4.1-7, respectively. These tables list stress results with the 30 lowest margins of safety for the 5^{th} support disk. The highest P_m stress occurs at Section 18, with a margin of safety of +0.97 (See Table 11.2.12.4.1-6 for stresses and Figure 11.2.12.4.1-7 for section locations). The highest P_m+P_b stress occurs at Section 61, with a margin of safety of +0.05 (see Table 11.2.12.4.1-7 for stresses and Figure 11.2.12.4.1-7 for section locations).

Support Disk Buckling Evaluation

For the tip-over accident, the support disks experience in-plane loads. The in-plane loads apply compressive forces and in-plane bending moments on the support disk. Buckling of the support disk is evaluated in accordance with the methods and acceptance criteria of NUREG/CR-6322 [39]. Because the ASME Code identifies 17-4PH disk material as ferritic steel, the formulas for non-austenitic steel are used.

The buckling evaluation of the support disk ligaments is based on the Interaction Equations 31 and 32 in NUREG/CR-6322. These two equations adopt the "Limit Analysis Design" approach. Other equations applicable to the calculations are noted as they are applied. The maximum forces and moments for the tip-over accident are based on the finite element analysis stress results.

Symbols and Units

- P = applied axial compressive load, kip
- M = applied bending moment, kip-inch
- P_a = allowable axial compressive load, kip
- P_{cr} = critical axial compression load, kip
- P_e = Euler buckling loads, kip

P_y = average yield load, equal to profile area times specified minimum yield stress, kips (for normal operating condition)

C_c = column slenderness ratio separating elastic and inelastic buckling

C_m = coefficient applied to bending term in interaction equation

M_m = critical moment that can be resisted by a plastically designed member in the absence of axial load, kip-in.

M_p = plastic moment, kip-in.

F_a = axial compressive stress permitted in the absence of bending moment, ksi

F_e = Euler stress for a prismatic member divided by factor of safety, ksi

k = ratio of effective column length to actual unsupported length

l = unsupported length of member, in.

r = radius of gyration, in.

S_y = yield stress, ksi

A = cross sectional area of member, in^2

Z_x = plastic section modulus, in^3

λ = allowable reduction factor, dimensionless

From NUREG/CR-6322, the following equations are used to evaluate the support disk:

$$\frac{P}{P_{cr}} + \frac{C_m M}{M_m\left[1 - \frac{P}{P_e}\right]} \leq 1.0 \qquad \text{(Equation 31)}$$

$$\frac{P}{P_y} + \frac{M}{1.18 M_p} \leq 1.0 \qquad \text{(Equation 32)}$$

where:

$$P_{cr} = 1.7 \times A \times F_a$$

$$F_a = \frac{P_a}{A} \quad \text{for } P_a = P_y\left[\frac{1 - \frac{\lambda^2}{4}}{1.11 + 0.5\lambda + 0.17\lambda^2 - 0.28\lambda^3}\right]$$

and $\lambda = \frac{1}{\pi}\left(\frac{kl}{r}\right)\sqrt{\frac{S_y}{E}}$ (accident conditions)

$$Pe = 1.92 \times A \times Fe$$

$$F_e = \frac{\pi^2 \cdot E}{1.3\left(\frac{k \cdot l}{r}\right)^2} \quad \text{(Level D–Accident)}$$

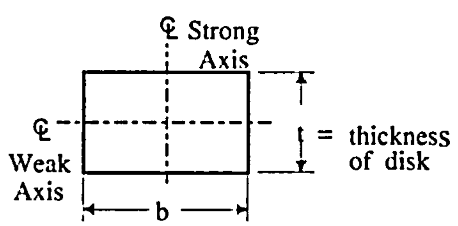
$$P_y = S_y \times A$$

$C_m =$ 0.85 for members with joint translation (sideways)

$$M_p = S_y \times Z_x$$

$$M_m = M_p \cdot \left(1.07 - \frac{\left(\frac{l}{r}\right) \cdot \sqrt{S_y}}{3160}\right) \le M_p$$

Buckling evaluation is performed in all sections in the disk ligaments defined in Figure 11.2.12.4.1-7. Using the cross-sectional stresses calculated at each section located in the ligament for each loading condition, the maximum corresponding compressive force (P) and bending moment (M) are determined as:



$$P = \sigma_m A$$

$$M = \sigma_b S$$

where, σ_m is the membrane stress, σ_b is the bending stress, A is the area (b × t), and S is the section modulus ($tb^2/6$). Note that the strong axis bending is considered in the buckling evaluation since the disk is only subjected to in-plane load during the tip-over event.

To determine the margin of safety:

$$P_1 = P/P_{cr} \qquad M_1 = \frac{C_m M}{(1 - P/P_e)M_m} \qquad (P_1 + M_1 \leq 1)$$

and

$$P_2 = P/P_y \qquad M_2 = \frac{M}{1.18 M_p} \qquad (P_1 + M_1 \leq 1)$$

The margins of safety are:

$$MS1 = \frac{1}{P_1 + M_1} - 1$$

and

$$MS2 = \frac{1}{P_2 + M_2} - 1$$

The support disk buckling evaluation results for the 5th support disk (the 5th support disk experiences the highest stresses) for the tip-over impact condition are summarized in Table 11.2.12.4.1-5 for the four basket drop orientations evaluated. As shown in Table 11.2.12.4.1-5, the 26.28° case generates the minimum margin of safety for buckling; therefore, the results of the buckling analysis for the 26.28° basket drop orientation case are presented in Table 11.2.12.4.1-8. This table presents the 30 minimum margins of safety for this drop orientation. As the tables demonstrate, the support disks meet the requirements of NUREG/CR-6322.

Fuel Tube Analysis

The fuel tube provides structural support and a mounting location for neutron absorber plates. The fuel tube does not provide structural support for the fuel assembly. To ensure that the fuel tube remains functional during a tip-over accident, a structural evaluation of the tube is performed for a side impact assuming a deceleration of 60g. This g-load bounds the maximum g-load (40g) calculated to occur for the PWR basket in a vertical concrete cask tipover event.

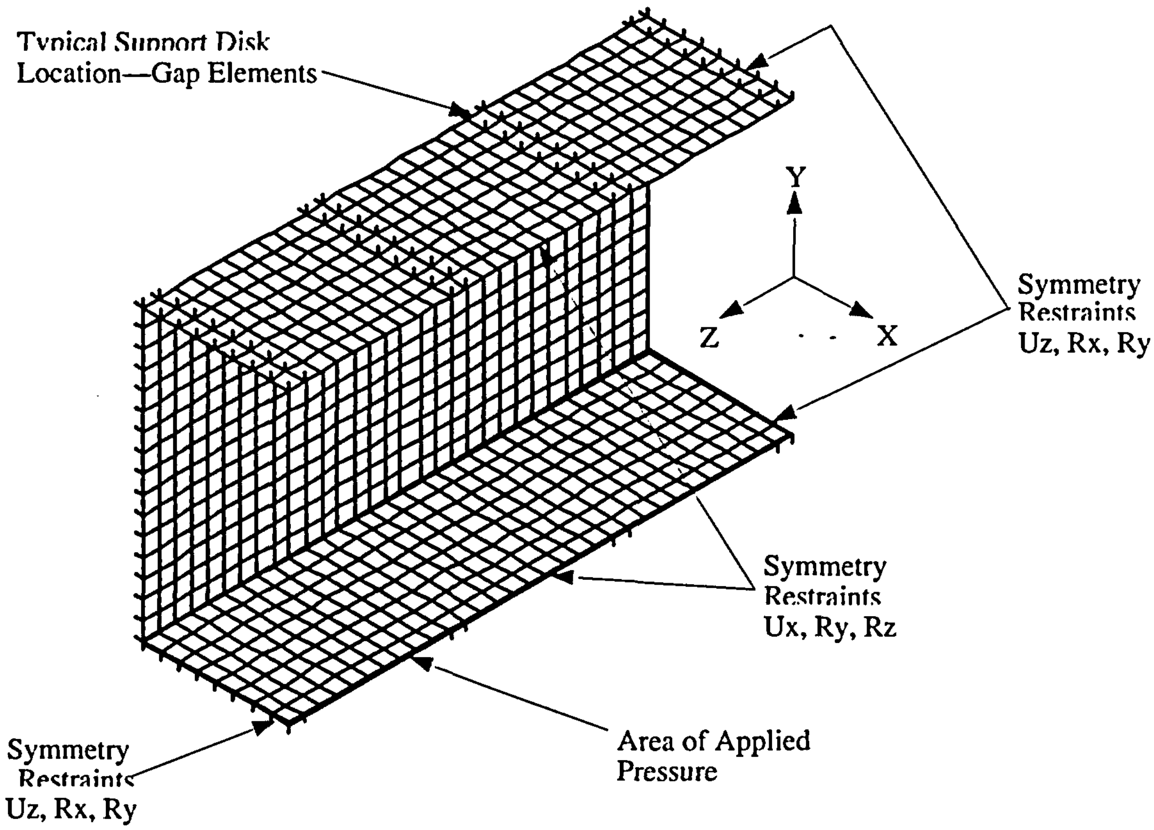
In the tip-over event, the stainless steel support disks in the fuel basket support the fuel tube. The fuel basket support disks, which support the full length of the fuel tube, are spaced 4.42-inches apart (which is less than one half of the fuel tube width of 8.8 inch). Considering the fuel tube subjected to a maximum PWR fuel assembly weight of 1,602 pounds with a 60g load factor and the 30 support locations provided by the basket support disks, the fuel tube shear stress is calculated as:

Shear load = (60g)(1,602)/30 = 3,204 lbs
Area = (0.048)(8.8)(2) = 0.845 in^2
Shear Stress = 3,204/0.845 = 3,792 psi

The yield strength of the tube material, Type 304 stainless steel, is 17,300 psi at 750°F. Conservatively, using the allowable shear stress as one-half the yield strength of the tube material (8,650 psi) results in a large positive margin of safety. Conservative evaluation of the tube loading resulting from its own mass during a side-impact shows that the tube structure maintains position and function.

The load transfer of the weight of the fuel assembly to the fuel basket support disk in the side impact is through direct bearing and compression of the distributed load of the fuel assembly through the fuel tube to the support disk web. Two load conditions are considered in the fuel tube evaluation. The first considers the fuel assembly load as a distributed pressure on the inside surface of the fuel tube. The second postulates that the fuel assembly grid is located at the center of the span between the support disks and produces a localized distributed load over the effective area of the grid.

Two different ANSYS finite element models of the tube are developed for these two load conditions since the fuel tube structural performance for either load is nonlinear. As shown below, the first model represents a fuel tube section with a length of three spans, i.e., the model is supported at four locations by support disks. The model conservatively considers the fuel tube wall thickness of 0.048 inch as the only material subjected to a distributed pressure load representative of the fuel assembly deceleration of 60g. Fuel assembly stiffness is not considered in the development of the imposed pressure load on the fuel tube.



The tube is modeled with the ANSYS plastic, quadrilateral shell element (SHELL43). The support disks are represented by gap elements (CONTAC52). The outer nodes of the gap elements are fully restrained in all three translational directions. Edge restraints were applied to the model to represent symmetry boundary conditions. The effective load on the fuel tube due to the 60g deceleration of the fuel assembly is applied as a pressure to the inside area of the fuel tube.

The finite element analysis results show that the maximum stress in the tube is 23.8 ksi, which is local to the sections of the tube resting on the support disks. At 750°F the ultimate strength for Type 304 stainless steel is 63.1 ksi. The margin of safety is

$$MS = \frac{63.1}{23.8} - 1 = +1.65$$

The analysis shows that the maximum total strain is 0.026 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F [42], the resulting margin of safety is:

$$MS = \frac{0.40/2}{0.026} - 1 = +\text{large}$$

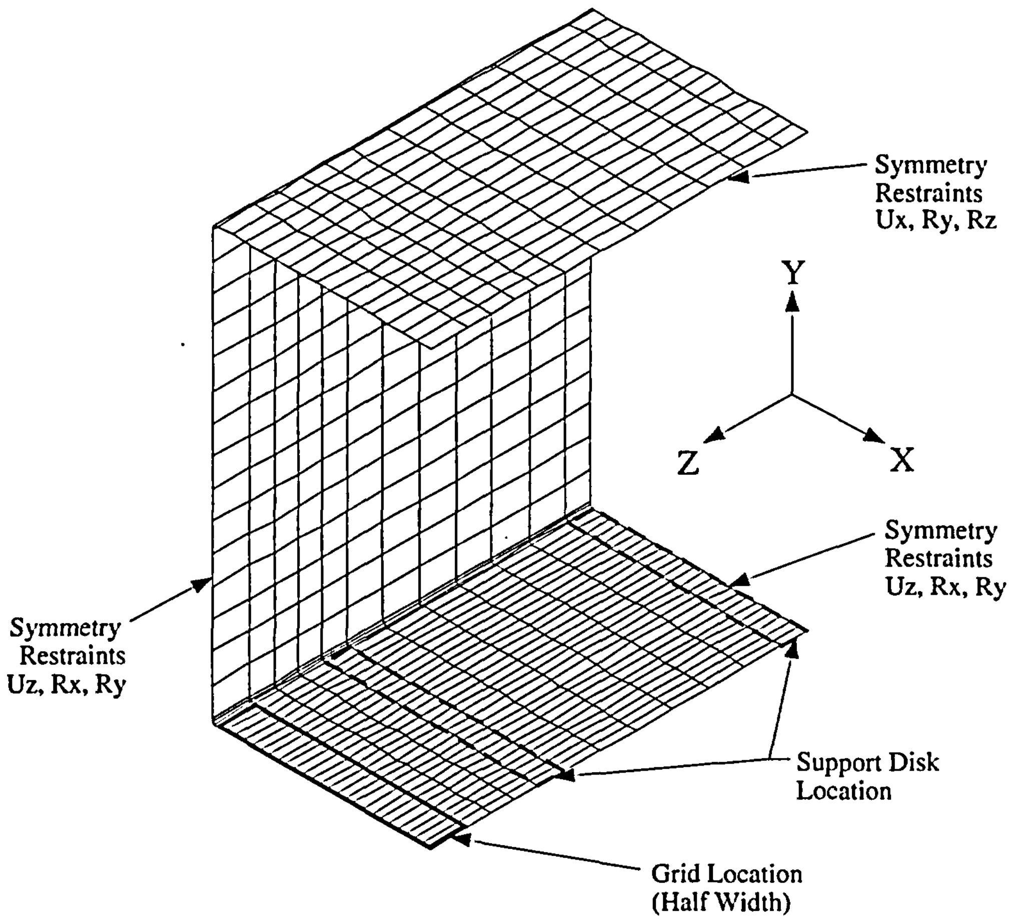
Similarly, the margin of safety for elastic-plastic stress becomes

$$MS = \frac{63.1 - 17.3}{23.8 - 17.3} - 1 = 6.05$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

The second finite element model is used to evaluate the load condition with the fuel assembly grid located at the center of the span between two support disks. The fuel tube is subjected to a localized distributed load over the effective area of the grid. As shown below, the model is a quarter-symmetry periodic section of the fuel tube. As in the finite element model used for the distributed pressure case, this model conservatively considers a fuel tube wall thickness of 0.048 inch. The neutron absorber plate (0.075 inch) and stainless steel cover plate (0.018 inch) are conservatively not included in the model. The tube wall is modeled with ANSYS SHELL43 elements. The support disks are modeled with CONTAC52 elements.

Based on the Lawrence Livermore evaluation of the fuel rods for a side impact (UCID-21246), the fuel rods and fuel assemblies maintain their structural integrity during the side impact resulting from a cask tip-over accident and the displacement of the fuel tube is limited. The maximum displacement of the fuel tube section between the support disks will not exceed the "thickness" of the grid spacer, which is the distance between the outer surface of the grid and the outer surface of the fuel rod array. When the displacement of the fuel tube reaches the "thickness" of the grid spacer, the fuel rods will be in contact with the inner surface of the fuel tube and the weight of the fuel rods will be transferred through the tube wall to the support disks. Therefore, a bounding load condition for this model is simulated by applying a constant displacement of 0.08 inch in the negative Y direction to the nodes corresponding to the grid location in the model. Note that 0.08 inch displacement bounds all PWR fuel assemblies. It is assumed that the fuel assembly grid spacer is rigid and therefore a constant displacement is conservatively applied.



The finite element analysis results show that the maximum stress in the tube is 38.4 ksi, which is local to the corner of the tube at the grid spacer location of the model close to the side wall of the tube. At 750°F the ultimate strength for Type 304 stainless steel is 63.1 ksi. The margin of safety is

$$MS = \frac{63.1}{38.4} - 1 = +0.64$$

The analysis shows that the maximum total strain is 0.11 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F [42], the resulting margin of safety is:

$$MS = \frac{0.40/2}{0.11} - 1 = 0.82$$

Similarly, the margin of safety for elastic-plastic stress becomes

$$MS = \frac{63.1 - 17.3}{38.4 - 17.3} - 1 = 1.17$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

Both the maximum total strain and the elastic-plastic stress analyses indicate that the tube position within the support basket is maintained.

Fuel Tube Yielding

Using the displacement of the fuel rod, a check of the fuel tube is performed to verify that the fuel tube remains elastic during a side-drop. The fuel rod displacement loading is a more realistic loading condition because the load is transmitted from the fuel rods to the fuel tube. The analysis is conservative as it assumes the cumulative displacement of 17 fuel rods (stacked on top of each other) in a 17×17 PWR fuel assembly.

The displacement of a single fuel rod assumed as a four-span continuous beam is calculated as:

$$\Delta_{max} = 0.0065 \frac{wL^4}{EI} = 2.2014 \times 10^{-5} \text{ in}$$

where:

w = mass/length = $\rho_{zirc} A_{zirc} + \rho_{UO2} A_{UO2}$ = 0.0404 lb/in × 17 rods = 0.6868 lb/in

Rod OD = 0.379 in

Rod ID = 0.379-2 × 0.024 = 0.331 in

Rod Density (Zirc-4) = ρ_{zirc} = 0.237 lb/in^3

Rod Area = $A_{zirc} = \frac{\pi}{4}(0.379^2 - 0.331^2) = 0.0268 \text{ in}^2$

UO_2 Density = ρ_{UO2} = 0.396 lb/in^3

$$UO_2 \text{ Area} = A_{UO2} = \frac{\pi}{4} x 0.331^2 = 0.086 \text{ in}^2$$

L = Distance between support disks = 4.42 in

$E_{zirc} = 10.75 \times 10^6$ psi

$$I_{zirc} = \frac{\pi}{64}(0.379^4 - 0.331^4) = 4.236 \text{ x } 10^{-4} \text{ in}^4 \text{ x } 17 \text{ rods} = 0.0072 \text{ in}^4$$

Using the E_{zirc} and I_{zirc} as conservative assumptions, the maximum displacement is estimated as 2.2014×10^{-5} in. For 60g acceleration, this displacement becomes 1.321×10^{-3} inch.

Applying the displacement midway between support disks, the maximum stress intensity is 12,062 psi. The yield stress for the fuel tube (Type 304 stainless steel) is 17,300 psi at 750°F degrees; therefore, during a 60g side-drop, the fuel tube remains elastic.

Assurance that the neutron absorber remains attached to the fuel tube is evaluated by considering that loads produced by the neutron absorber plate and stainless steel attachment plate, assuming a 60g load, are carried by the attachment plate weld. Total load and resultant stress on the weld are calculated as:

$F_{b/ss} = (g)(\rho)(t)(w)(l)$ Load exerted by neutron absorber/stainless steel attachment plate

where:

- g = acceleration (g)
- ρ = density of material (lb/in^3) (The density of aluminum (0.098 lb/in^3) is conservatively used for the neutron absorber.
- t = thickness of material (in.)
- w = width of material (in.)
- l = length of material section (in.)

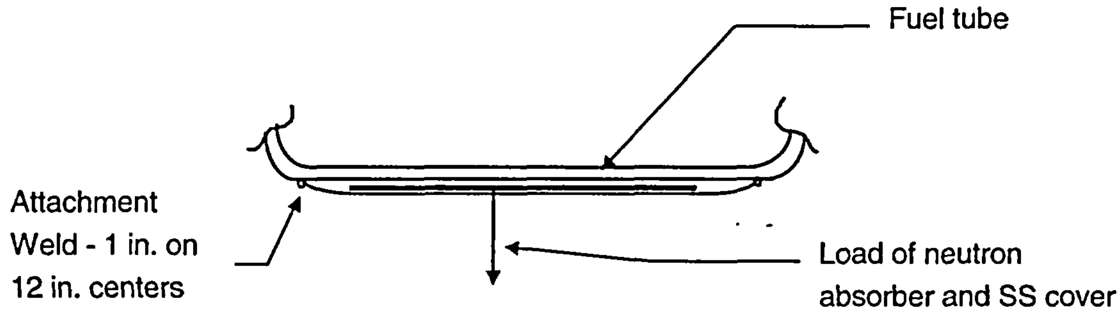
The forces on the weld due to a 12-inch section of neutron absorber (F_b) and a 12-inch section of stainless steel plate (F_{ss}) are:

$$F_b = (60g)(0.098 \text{ lb/in}^3)(0.075 \text{ in.})(8.2 \text{ in.})(12 \text{ in.}) = 43.4 \text{ lbs}$$

$$F_{ss} = (60g)(0.291 \text{ lb/in}^3)(0.018 \text{ in.})(8.7 \text{ in.})(12 \text{ in.}) = 32.8 \text{ lbs}$$

The total load (F_t) on a 1-inch attachment weld for a 12-inch section is:

$$F_t = 43.4 \text{ lbs} + 32.8 \text{ lbs} = 76.2 \text{ lbs}$$



The resulting weld stress is: $\sigma = P/A = (76.2 \text{ lb}/2) / (1 \text{ in.}) (0.018 \text{ in.}) = 2,117$ psi

Since the weld material is Type 304 stainless steel, the margin of safety (at 750°F) is:

$$MS = \frac{17,300}{2,117} - 1 = +7.2$$

Therefore, the neutron absorber remains enclosed on each outer surface of the fuel tube wall.

Figure 11.2.12.4.1-1 Basket Drop Orientations Analyzed for Tip-Over Conditions - PWR

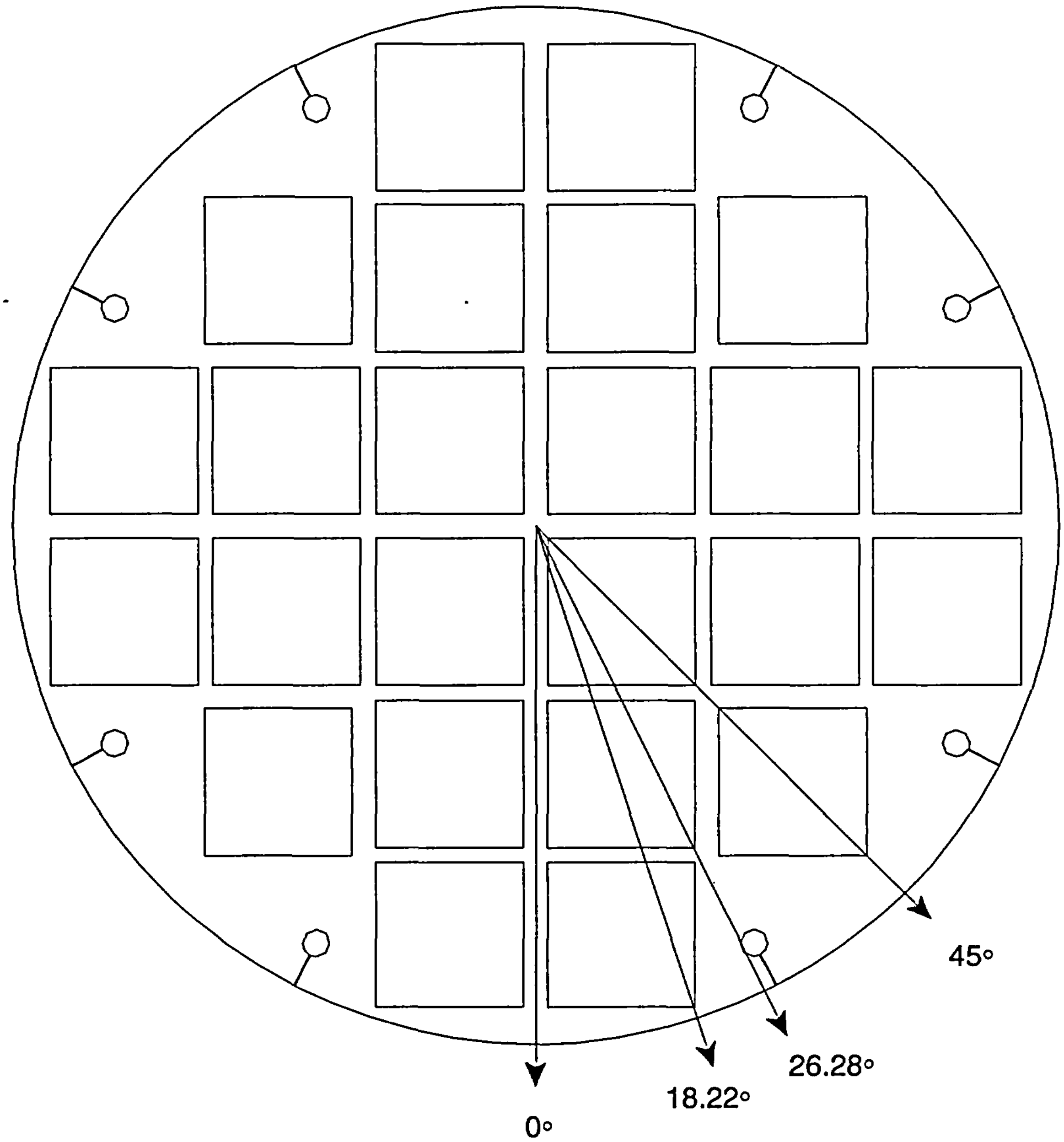
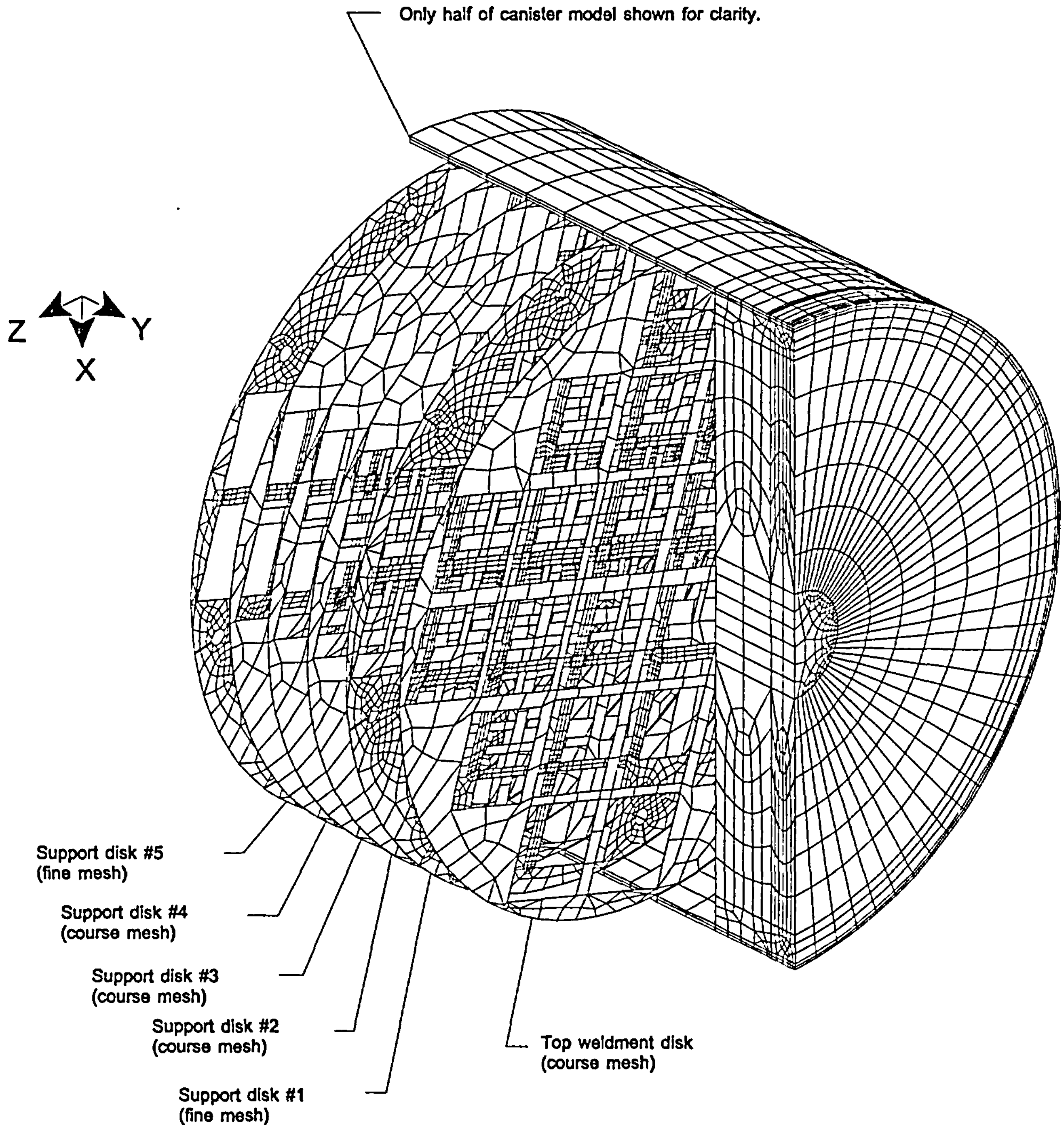
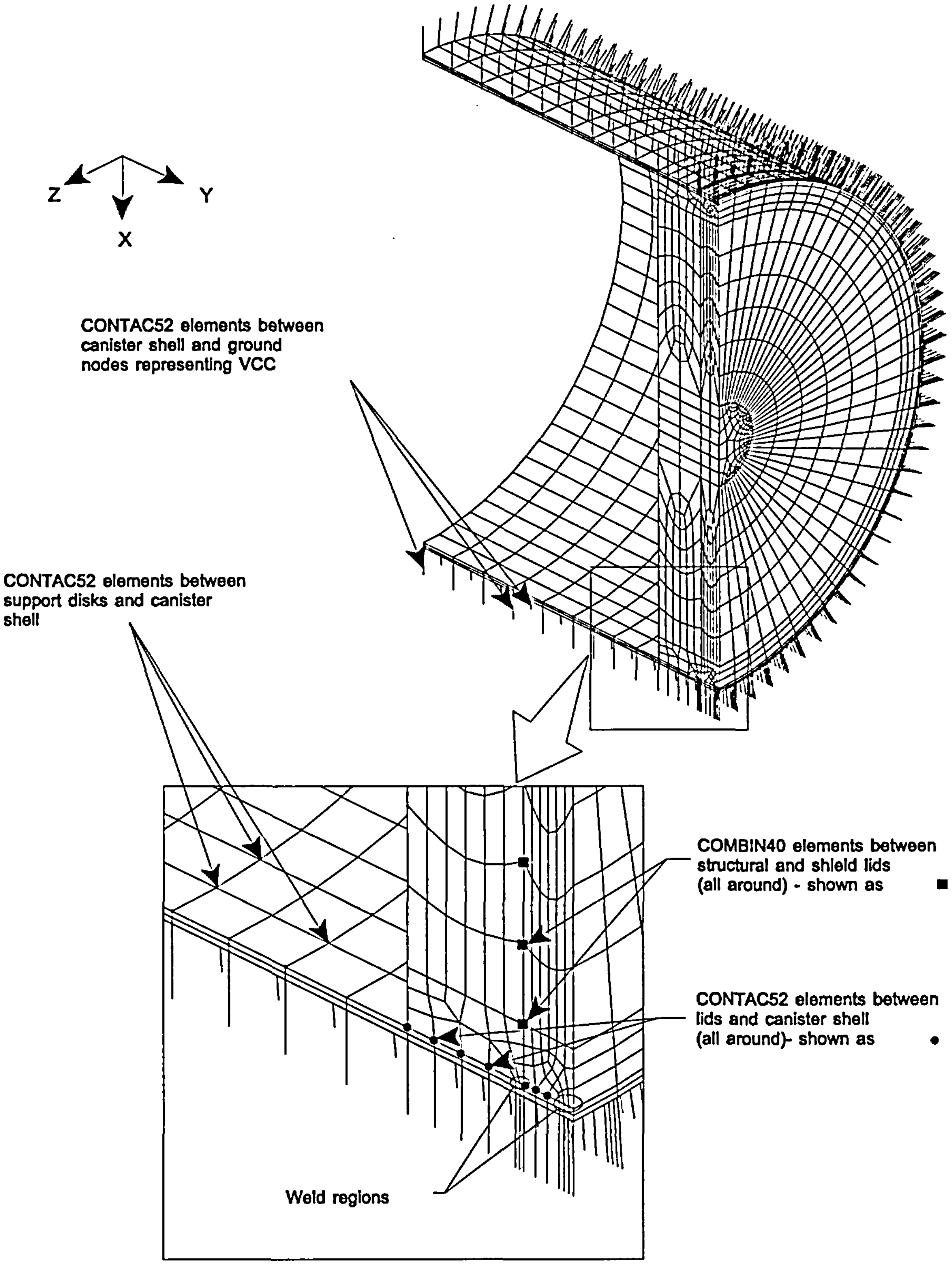


Figure 11.2.12.4.1-2 Fuel Basket/Canister Finite Element Model - PWR



18.22° Basket Drop Orientation

Figure 11.2.12.4.1-3 Fuel Basket/Canister Finite Element Model - Canister



Only Half of Canister Shown for Clarity

Figure 11.2.12.4.1-4 Fuel Basket/Canister Finite Element Model - Support Disk - PWR

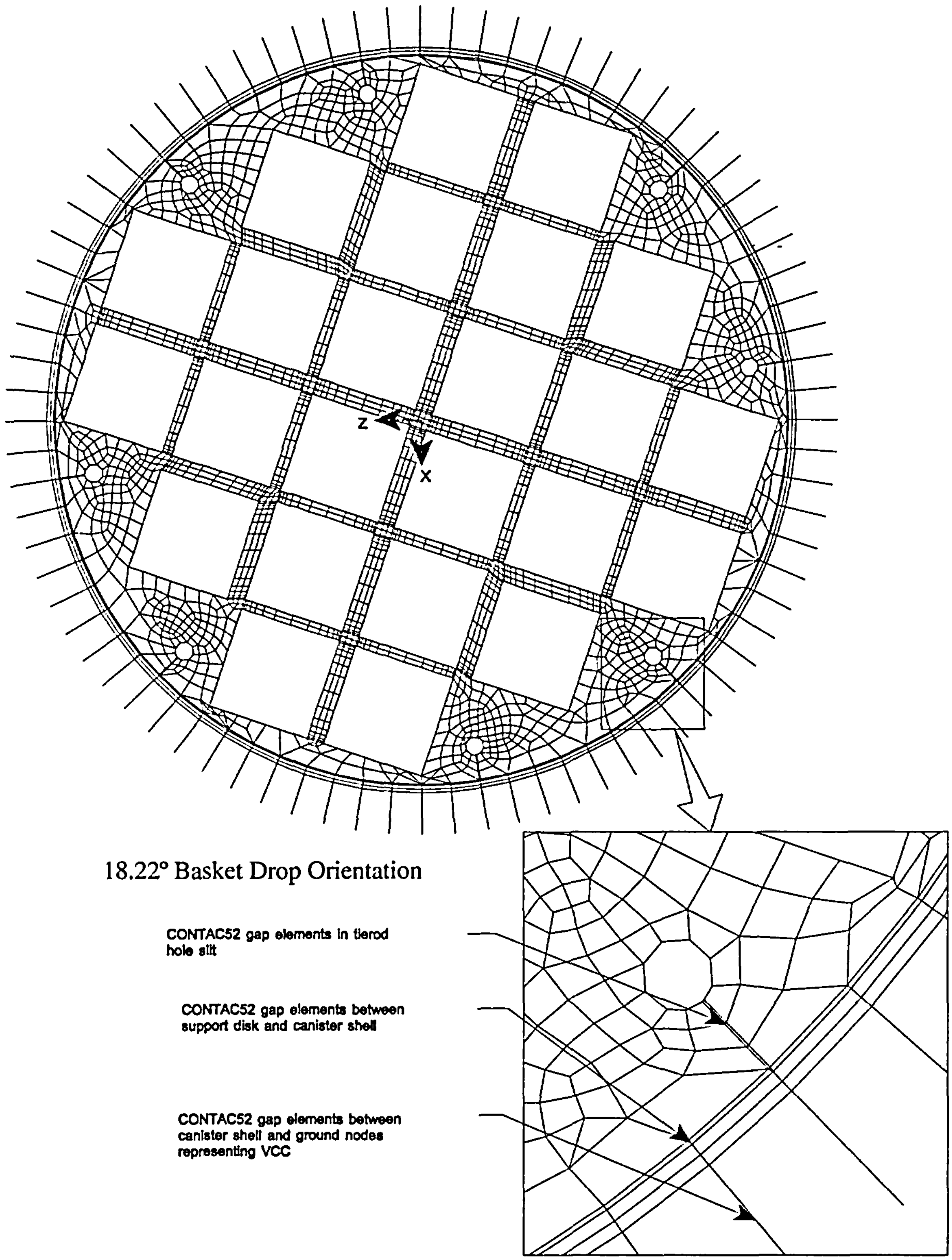
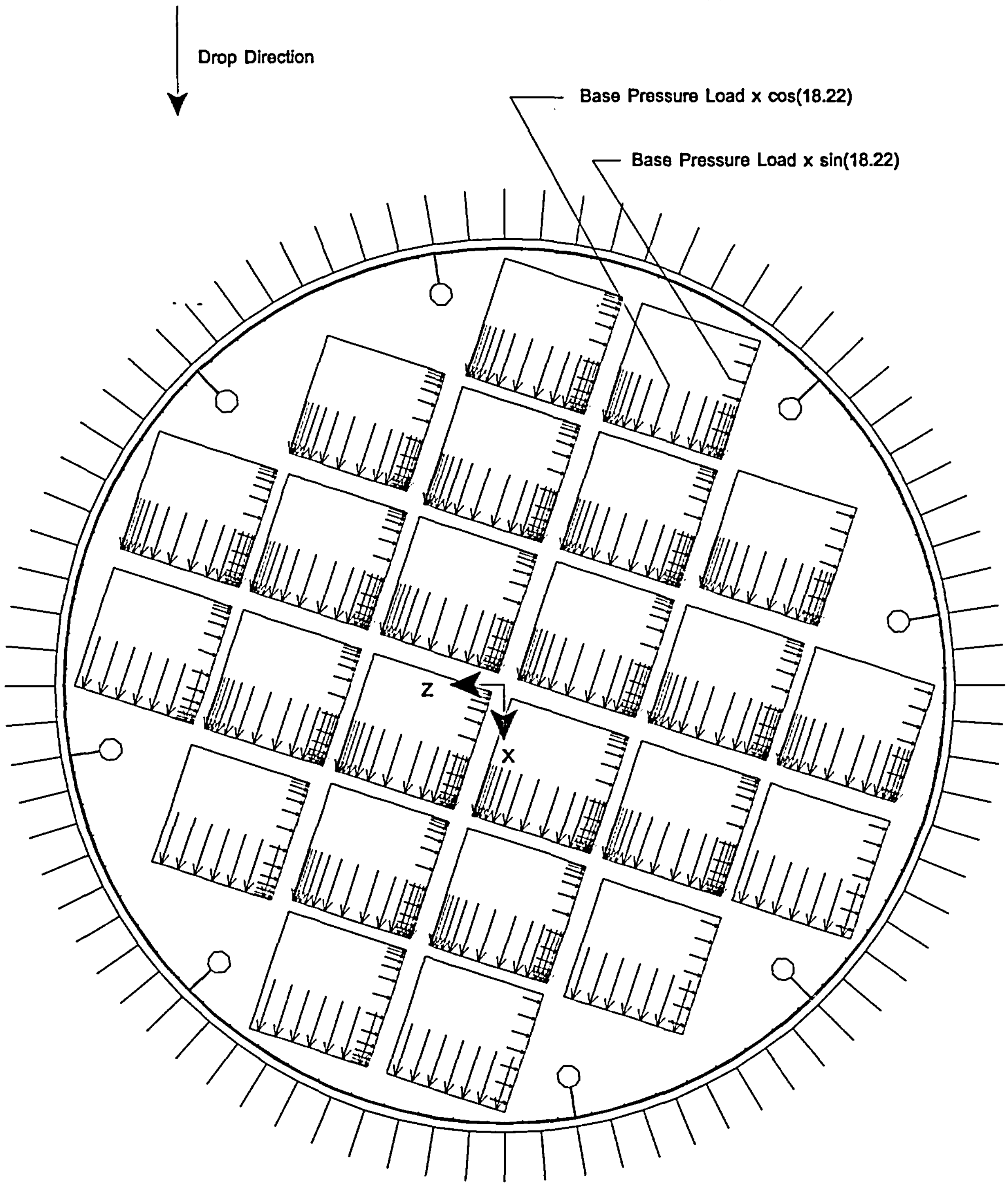
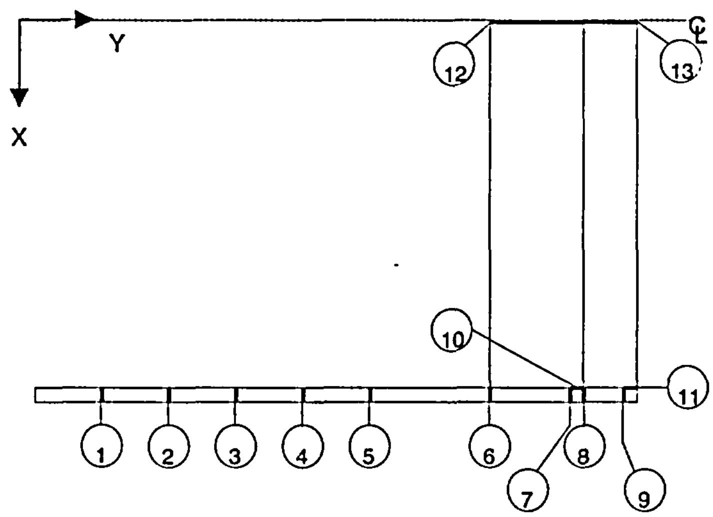


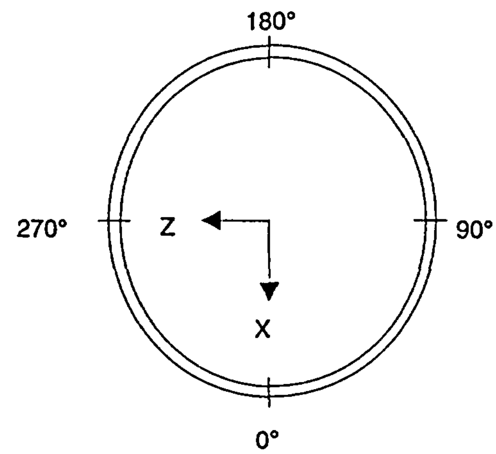
Figure 11.2.12.4.1-5 Fuel Basket/Canister Finite Element Model - Support Disk Loading - PWR



18.22° Basket Drop Orientation
Note: Finite Element Mesh Not Shown

Figure 11.2.12.4.1-6 Canister Section Stress Locations





Top View of Axis

PWR 1
Section Coordinates at Z = 0 and X > 0

Location	Point 1 X	Point 1 Y	Point 2 X	Point 2 Y
1	32.905	131.42	33.53	131.42
2	32.905	136.34	33.53	136.34
3	32.905	141.26	33.53	141.26
4	32.905	146.18	33.53	146.18
5	32.905	151.10	33.53	151.10
6	32.905	165.25	33.53	165.25
7	32.905	171.75	33.53	171.75
8	32.905	172.25	33.53	172.25
9	32.905	174.37	33.53	174.37
10	32.905	171.75	32.905	172.25
11	32.905	174.37	32.905	175.25
12	0.1	165.25	0.1	172.23
13	0.1	172.27	0.1	175.25

BWR 4
Section Coordinates at Z = 0 and X > 0

Location	Point 1 X	Point 1 Y	Point 2 X	Point 2 Y
1	32.905	144.32	33.53	144.32
2	32.905	148.15	33.53	148.15
3	32.905	151.98	33.53	151.98
4	32.905	155.81	33.53	155.81
5	32.905	159.64	33.53	159.64
6	32.905	175.25	33.53	175.25
7	32.905	182.25	33.53	182.25
8	32.905	182.75	33.53	182.75
9	32.905	184.87	33.53	184.87
10	32.905	182.25	32.905	182.75
11	32.905	184.87	32.905	185.75
12	0.1	175.75	0.1	182.73
13	0.1	182.77	0.1	185.75

General Notes:

1) Impact from the tipover condition is at 0° (in the circumferential direction).

2) For the full 360° models, there are 80 sections at each location for a total of 1040 sections. For the half 180° models, there are 41 sections at each location for a total of 533 sections.

3) Location 10 is through the length of the shield lid weld. Locations 8 and 7 are through the canister shell at top and bottom of the shield lid weld, respectively.

4) Location 13 is through the length of the structural lid weld. Location 9 is through the canister shell at the bottom of the structural lid weld.

Figure 11.2.12.4.1-7 Support Disk Section Stress Locations - PWR – Full Model

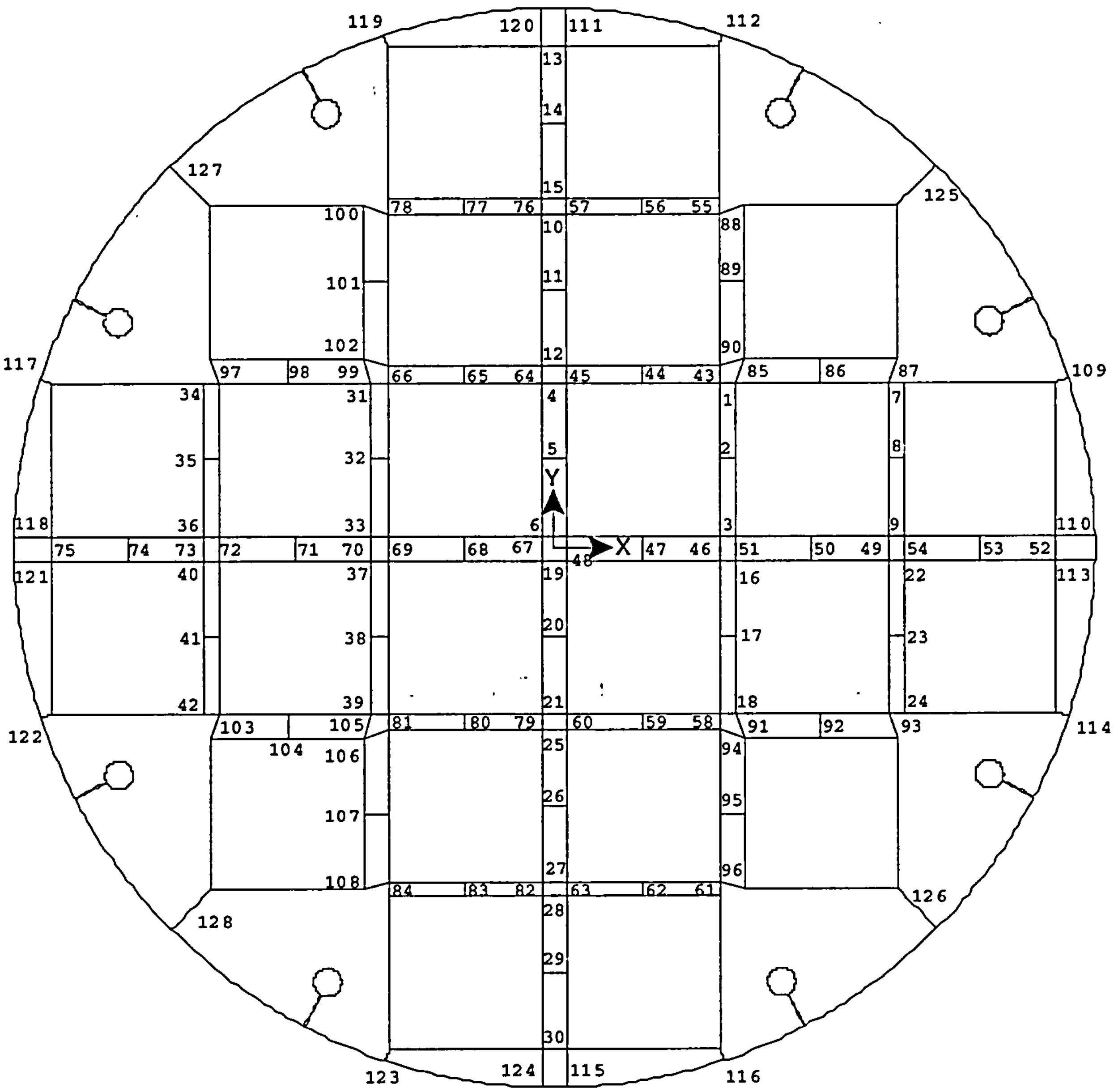
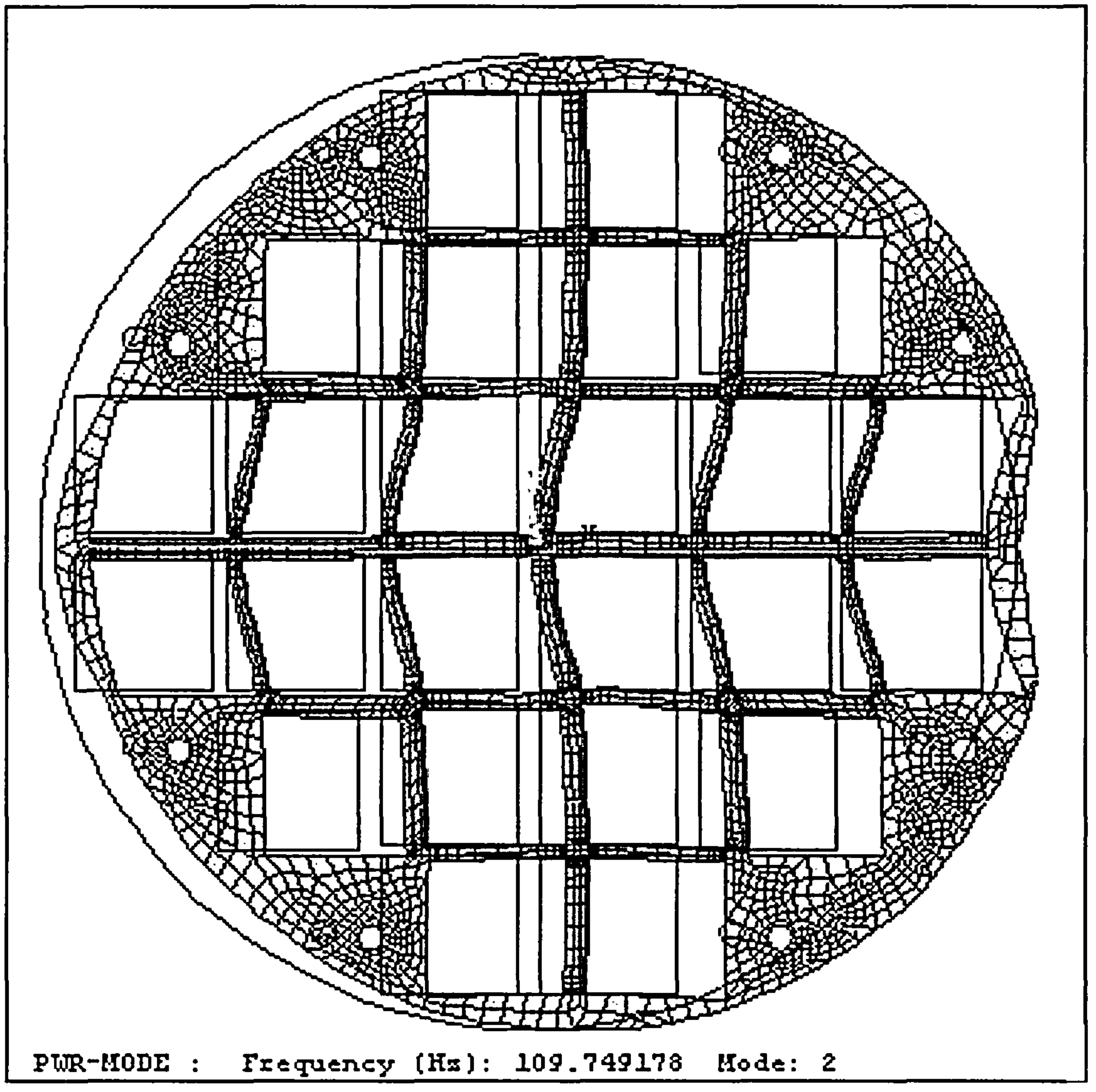
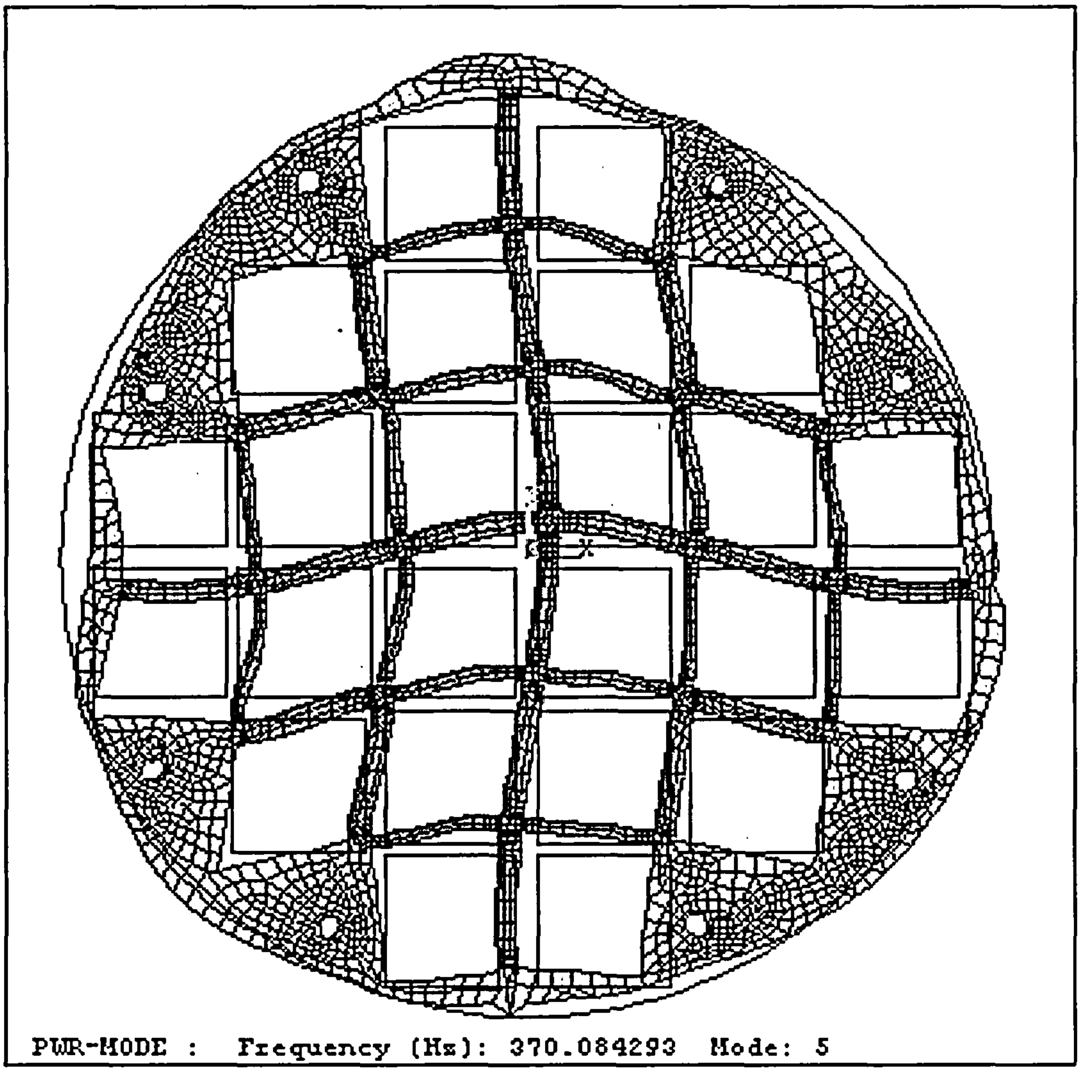


Figure 11.2.12.4.1-8 PWR - 109.7 Hz Mode Shape



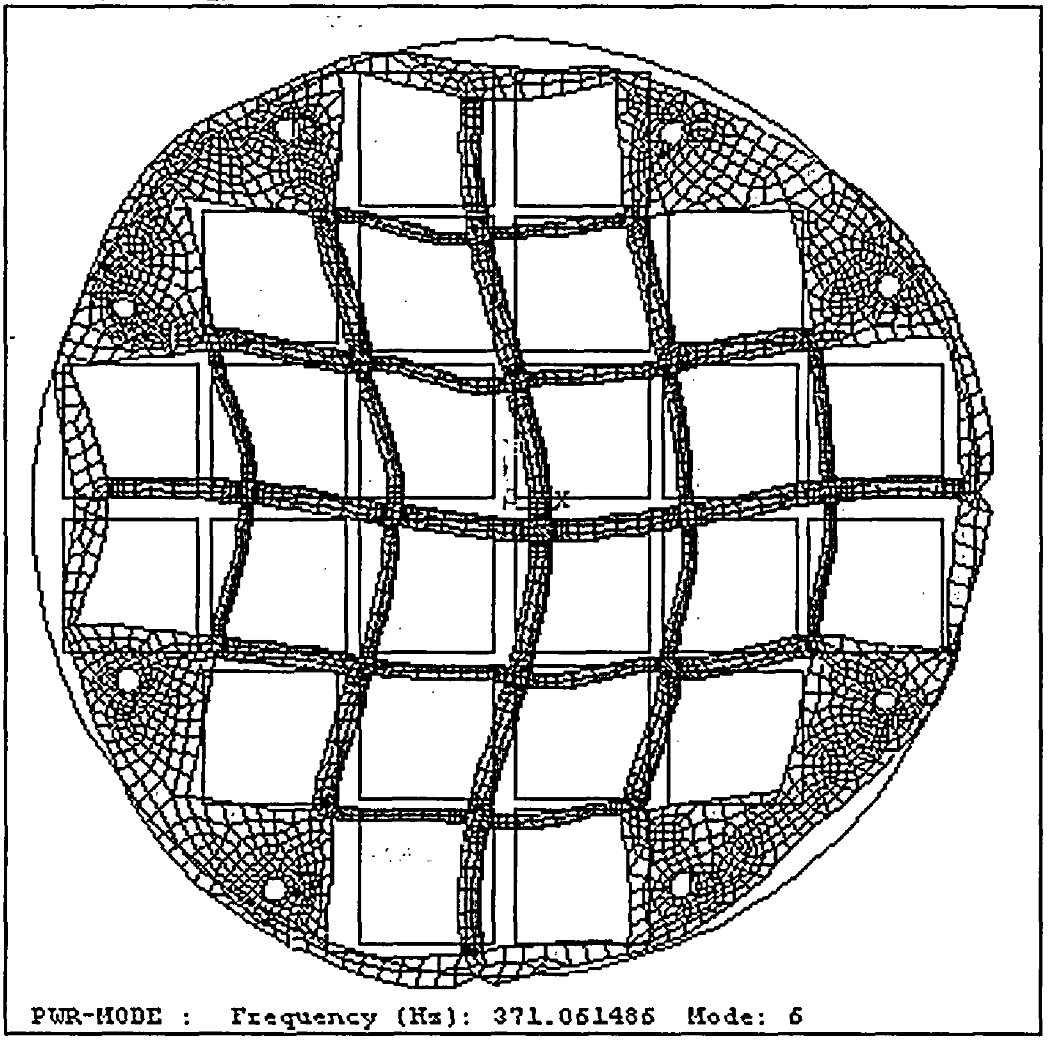
Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Figure 11.2.12.4.1-9 PWR – 370.1 Hz Mode Shape



Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Figure 11.2.12.4.1-10 PWR – 371.1 Hz Mode Shape



Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Table 11.2.12.4.1-1 Canister Primary Membrane (P_m) Stresses for Tip-Over Conditions – PWR - 45° Basket Drop Orientation (ksi)

Section[(1)] Location	Section Angle (deg)	S_x	S_y	S_z	S_{xy}	S_{yz}	S_{xz}	Stress Intensity	Allowable Stress	Margin of Safety
1	0	-1.5	6.4	1.4	-0.1	0	-0.2	7.98	35.52	3.45
2	0	-1.7	9.2	1.5	0.1	0	0.3	10.88	35.52	2.26
3	49.5	-0.2	9.3	6.3	-0.1	1.1	0	9.81	35.52	2.62
4	63	-0.3	8.9	5	0	3.4	0.4	11.22	35.52	2.17
5	90	0.1	2.8	-1	-0.3	6	0.1	12.6	35.52	1.82
6	85.5	0	0.3	0.1	-0.1	7.8	0	15.62	35.52	1.27
7[(2)]	9	1.0	0.6	7.0	2.7	-5.1	0.7	13.61	35.52	1.61
8[(2)]	9	6.8	0	6.9	0.6	-3.2	-1.0	10.09	35.52	2.52
9[(2)]	9	5.8	-3.4	1.0	2.4	-3.8	0	12.50	35.52	1.84
10[(4)]	0–9	-29.7	-15.7	-20.6	6.7	-0.8	-2.0	19.87	40.08[(3)]	1.02
11[(4)]	0–8.4	-30.0	-15.3	-8.8	7.1	-1.8	2.0	24.80	32.06[(5)]	0.29
12	0	-0.7	0.2	0	0	0	-0.1	0.93	35.52	37.05
13	0	-1.5	0.5	0	0	0	-0.1	1.98	35.52	16.92

Stresses are presented in the cylindrical coordinate system, x = radial, y = circumferential and z = axial directions.

1. Section locations are shown in Figure 11.2.12.4.1-6.
2. Stresses are not presented for the sections with localized bearing stress. In accordance with ASME Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions.
3. Allowable stress at 300°F.
4. Stresses are determined by averaging the stresses over the impact region.
5. A stress reduction factor of 0.8 is applied to the allowable stress at 250°F.

Table 11.2.12.4.1-2 Canister Primary Membrane + Primary Bending ($P_m + P_b$) Stresses for Tip-Over Conditions – PWR - 45° Basket Drop Orientation (ksi)

Section[1] Location	Section Angle (deg)	S_x	S_y	S_z	S_{xy}	S_{yz}	S_{xz}	Stress Intensity	Allowable Stress	Margin of Safety
1	0	-2.1	19.3	4.4	-0.6	-0.1	-0.1	21.38	53.28	1.49
2	0	-1.9	22.3	3	-0.3	0.1	0.2	24.26	53.28	1.2
3	0	-2.6	22.3	6.2	0.2	0	-0.1	24.92	53.28	1.14
4	0	-1.8	21	3.9	-0.8	-0.1	-0.3	22.88	53.28	1.33
5	72	-0.7	20.5	12.4	0.1	3.8	-0.9	22.8	53.28	1.34
6	0	0.6	-29.8	-7.6	2.3	-1.1	-0.9	30.93	53.28	0.72
7[2]	9	0.6	9.3	23.7	0.2	-4.0	1.6	24.32	53.28	1.19
8[2]	9	6.7	9.0	23.6	-0.8	-5.3	-3.7	21.08	53.28	1.53
9[2]	9	8.0	-5.9	4.8	4.4	-4.5	-0.3	18.42	53.28	1.89
10[4]	0-8.8	-42.5	-19.4	-24.1	7.1	0.4	-3.6	27.78	60.12[3]	1.16
11[4]	0-8.4	-26.6	-12.0	-1.2	8.0	-0.8	2.0	29.25	48.09[5]	0.64
12	0	-0.9	0	0	0	0	-0.1	0.95	53.28	54.84
13	0	-2.3	-0.7	0	0	0	-0.1	2.33	53.28	21.84

Stresses are presented in the cylindrical coordinate system, x = radial, y = circumferential and z = axial directions.

1. Section locations are shown in Figure 11.2.12.4.1-6.
2. Stresses are not presented for the sections with localized bearing stress. In accordance with ASME Code Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions.
3. Allowable stress at 300°F.
4. Stresses are determined by averaging the stresses over the impact region.
5. A stress reduction factor of 0.8 is applied to the allowable stress at 250°F.

Table 11.2.12.4.1-3 Support Disk Section Location for Stress Evaluation - PWR - Full Model

Sec. No.	Point 1		Point 2		Sec. No.	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
1	10.02	10.02	11.02	10.02	45	0.75	10.02	0.75	11.02
2	10.02	5.39	11.02	5.39	46	10.02	0.75	10.02	-0.75
3	10.02	0.75	11.02	0.75	47	5.39	0.75	5.39	-0.75
4	0.75	10.02	-0.75	10.02	48	0.75	0.75	0.75	-0.75
5	0.75	5.39	-0.75	5.39	49	20.29	0.75	20.29	-0.75
6	0.75	0.75	-0.75	0.75	50	15.66	0.75	15.66	-0.75
7	20.29	10.02	21.17	10.02	51	11.02	0.75	11.02	-0.75
8	20.29	5.39	21.17	5.39	52	30.44	0.75	30.44	-0.75
9	20.29	0.75	21.17	0.75	53	25.81	0.75	25.81	-0.75
10	0.75	20.29	-0.75	20.29	54	21.17	0.75	21.17	-0.75
11	0.75	15.66	-0.75	15.66	55	10.02	20.29	10.02	21.17
12	0.75	11.02	-0.75	11.02	56	5.39	20.29	5.39	21.17
13	0.75	30.44	-0.75	30.44	57	0.75	20.29	0.75	21.17
14	0.75	25.81	-0.75	25.81	58	10.02	-10.02	10.02	-11.02
15	0.75	21.17	-0.75	21.17	59	5.39	-10.02	5.39	-11.02
16	10.02	-0.75	11.02	-0.75	60	0.75	-10.02	0.75	-11.02
17	10.02	-5.39	11.02	-5.39	61	10.02	-20.29	10.02	-21.17
18	10.02	-10.02	11.02	-10.02	62	5.39	-20.29	5.39	-21.17
19	0.75	-0.75	-0.75	-0.75	63	0.75	-20.29	0.75	-21.17
20	0.75	-5.39	-0.75	-5.39	64	-0.75	10.02	-0.75	11.02
21	0.75	-10.02	-0.75	-10.02	65	-5.39	10.02	-5.39	11.02
22	20.29	-0.75	21.17	-0.75	66	-10.02	10.02	-10.02	11.02
23	20.29	-5.39	21.17	-5.39	67	-0.75	0.75	-0.75	-0.75
24	20.29	-10.02	21.17	-10.02	68	-5.39	0.75	-5.39	-0.75
25	0.75	-11.02	-0.75	-11.02	69	-10.02	0.75	-10.02	-0.75
26	0.75	-15.66	-0.75	-15.66	70	-11.02	0.75	-11.02	-0.75
27	0.75	-20.29	-0.75	-20.29	71	-15.66	0.75	-15.66	-0.75
28	0.75	-21.17	-0.75	-21.17	72	-20.29	0.75	-20.29	-0.75
29	0.75	-25.81	-0.75	-25.81	73	-21.17	0.75	-21.17	-0.75
30	0.75	-30.44	-0.75	-30.44	74	-25.81	0.75	-25.81	-0.75
31	-10.02	10.02	-11.02	10.02	75	-30.44	0.75	-30.44	-0.75
32	-10.02	5.39	-11.02	5.39	76	-0.75	20.29	-0.75	21.17
33	-10.02	0.75	-11.02	0.75	77	-5.39	20.29	-5.39	21.17
34	-20.29	10.02	-21.17	10.02	78	-10.02	20.29	-10.02	21.17
35	-20.29	5.39	-21.17	5.39	79	-0.75	-10.02	-0.75	-11.02
36	-20.29	0.75	-21.17	0.75	80	-5.39	-10.02	-5.39	-11.02
37	-10.02	-0.75	-11.02	-0.75	81	-10.02	-10.02	-10.02	-11.02
38	-10.02	-5.39	-11.02	-5.39	82	-0.75	-20.29	-0.75	-21.17
39	-10.02	-10.02	-11.02	-10.02	83	-5.39	-20.29	-5.39	-21.17
40	-20.29	-0.75	-21.17	-0.75	84	-10.02	-20.29	-10.02	-21.17
41	-20.29	-5.39	-21.17	-5.39	85	11.02	10.02	11.52	11.52
42	-20.29	-10.02	-21.17	-10.02	86	16.16	11.52	16.16	10.02
43	10.02	10.02	10.02	11.02	87	20.29	10.02	20.79	11.52
44	5.39	10.02	5.39	11.02	88	10.02	20.29	11.52	20.79

Note: See Figure 11.2.12.4.1-7 for section location.

Table 11.2.12.4.1-4 Summary of Maximum Stresses for PWR Support Disk for Tip-Over Condition

Drop Orientation	P_m			$P_m + P_b$		
	Stress Intensity (ksi)	Allowable Stress (ksi)	Margin of Safety	Stress Intensity (ksi)	Allowable Stress (ksi)	Margin of Safety
0°	58.2	90.8	+0.56	81.9	129.8	+0.58
18.22°	47.5	90.4	+0.91	111.6	130.8	+0.17
26.28°	46.0	90.4	+0.97	124.6	130.8	+0.05
45°	34.4	91.5	+1.66	101.4	129.1	+0.27

Note: See Figure 11.2.12.4.1-1 for Drop Orientation.

Table 11.2.12.4.1-5 Summary of Buckling Evaluation of PWR Support Disk for Tip-Over Condition

Drop Orientation	MS1	MS2
0°	+0.98	+0.96
18.22°	+0.31	+0.36
26.28°	+0.10	+0.15
45°	+0.31	+0.34

Note: See Figure 11.2.12.4.1-1 for Drop Orientation.

Table 11.2.12.4.1-6 Support Disk Primary Membrane (P_m) Stresses for Tip-Over Condition - PWR Disk No. 5 - 26.28° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
18	19.5	-26.1	3.1	46.0	90.4	0.97
3	27.1	-14.8	2.7	42.2	89.3	1.12
16	-38.3	-25.9	1	38.4	89.3	1.32
1	-33.5	-14.7	0.5	33.5	90.4	1.70
94	-28.3	-21.4	2.9	29.4	90.5	2.08
17	-0.1	-26	2	26.2	89.8	2.42
96	6.1	-16.4	-3.1	23.3	91.5	2.92
95	-0.1	-22.4	1.7	22.6	91.1	3.04
88	-18.4	-7	-7	21.7	91.5	3.21
84	-17.1	-20.7	-0.8	20.9	91.5	3.38
61	-17.8	-9.7	5.1	20.3	91.5	3.51
90	15	-5	0.6	20.1	90.5	3.51
60	-11.3	-18.4	1.1	18.6	89.3	3.80
30	-18	-10.1	3	19.0	91.9	3.83
82	-17.2	-7	4.1	18.7	90.8	3.87
62	-17.8	-0.2	2.6	18.4	91.2	3.97
58	-11.4	-13.8	5.4	18.2	90.4	3.97
91	-8.2	-17.5	-1.4	17.7	90.5	4.11
63	-17.8	-12.3	0.2	17.8	90.8	4.11
83	-17.2	-0.2	1.7	17.3	91.2	4.26
7	-16.5	-12.6	-0.8	16.7	91.5	4.49
24	-1.2	-15.8	2	16.1	91.5	4.69
28	-15.4	-10	1.6	15.8	90.9	4.74
23	-0.1	-15.8	0.8	15.8	91.2	4.78
22	-9.1	-15.7	-0.5	15.7	90.8	4.78
51	-3.6	-15.1	-2	15.4	89.4	4.79
37	11.1	-4.3	0.6	15.4	89.3	4.80
79	-6	6.5	4.5	15.4	89.3	4.82
2	-0.1	-14.7	1.6	15.0	89.8	5.00
85	-4.6	-11.2	-6.4	15.1	90.5	5.00

Note: See Figure 11.2.12.4.1-2 for disk location and Figure 11.2.12.4.1-7 for section locations.

Table 11.2.12.4.1-7 Support Disk Primary Membrane + Primary Bending ($P_m + P_b$) Stresses for Tip-Over Condition - PWR Disk No. 5 - 26.28° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
61	-123.4	-34.3	10.4	124.6	130.8	0.05
58	-115.3	-47.4	9.6	116.6	129.1	0.11
43	-95.4	-34.6	6.8	96.1	129.1	0.34
82	-92.1	-27.8	7.2	92.9	129.8	0.40
79	-86.9	-19.9	2.3	87.0	127.6	0.47
16	-54.3	-76.8	15.6	84.8	127.6	0.50
60	-82.9	-41	7.8	84.3	127.6	0.51
18	-4.1	-84.9	-2.5	85.0	129.1	0.52
46	-79.1	-52.5	10.4	82.7	127.6	0.54
55	-84.2	-31.4	5	84.7	130.8	0.54
3	9.1	-71.1	-5.7	81.0	127.6	0.57
64	-79.8	-32.4	7.2	80.9	127.6	0.58
30	-40.2	-74.7	11.7	78.3	131.3	0.68
63	-75.2	-27.9	4.9	75.7	129.8	0.71
76	72.6	21.9	5.2	73.1	129.8	0.77
48	-66.5	-43.2	3.9	67.1	125.7	0.87
19	-39.5	-66.4	2.9	66.7	125.7	0.88
6	-43.6	-63.2	5.2	64.5	125.7	0.95
94	-59.5	-44.7	11.1	65.5	129.3	0.97
21	-48.3	-59.4	5.2	61.5	127.6	1.08
45	-61.2	-14.4	-0.6	61.2	127.6	1.09
67	-56.6	-43.3	5.4	58.6	125.7	1.15
1	-49.4	-43.6	13.2	60.0	129.1	1.15
51	26.3	-30.4	4.7	57.5	127.7	1.22
33	-29.3	-54.9	7.1	56.7	127.6	1.25
39	-29.2	-52.9	6.2	54.5	129.1	1.37
24	-8.5	-52.1	4.1	52.5	130.8	1.49
81	-49.2	-30.8	5.5	50.7	129.1	1.55
4	-43.3	-43.7	5.8	49.3	127.6	1.59
28	-46.3	-28.1	9.2	50.1	129.9	1.59

Note: See Figure 11.2.12.4.1-2 for disk location and Figure 11.2.12.4.1-7 for section locations.

Table 11.2.12.4.1-8 Summary of Support Disk Buckling Evaluation for Tip-Over Condition - PWR Disk No. 5 - 26.28° Drop Orientation

Section Number	P (kip)	Pcr (kip)	Py (kip)	M (in-kip)	Mp (in-kip)	Mm (in-kip)	MS1	MS2
61	7.80	44.18	38.91	6.74	8.51	8.18	0.10	0.15
58	5.69	51.79	43.78	8.66	10.94	10.67	0.23	0.25
82	7.52	43.76	38.54	4.78	8.43	8.10	0.44	0.48
18	13.04	51.79	43.78	4.90	10.94	10.67	0.51	0.48
43	1.95	51.79	43.78	7.62	10.94	10.67	0.54	0.58
16	12.97	50.82	42.93	4.24	10.73	10.47	0.62	0.57
79	3.00	50.82	42.93	6.74	10.73	10.47	0.63	0.66
60	5.66	50.82	42.93	5.96	10.73	10.47	0.65	0.66
63	7.78	43.76	38.54	3.66	8.43	8.10	0.73	0.75
55	0.92	44.18	38.91	5.24	8.51	8.18	0.76	0.83
64	2.18	50.82	42.93	6.29	10.73	10.47	0.79	0.83
3	7.40	50.82	42.93	4.69	10.73	10.47	0.86	0.84
46	1.85	83.64	64.39	14.37	24.15	24.15	0.89	0.88
30	7.60	87.05	67.05	12.10	25.14	25.14	1.00	0.92
19	3.78	81.50	62.70	11.51	23.51	23.51	1.15	1.10
48	1.80	81.50	62.70	12.01	23.51	23.51	1.19	1.17
6	2.46	81.50	62.70	11.23	23.51	23.51	1.29	1.25
45	1.91	50.82	42.93	4.78	10.73	10.47	1.34	1.37
21	3.89	83.64	64.39	10.16	24.15	24.15	1.47	1.40
24	6.92	44.18	38.91	2.31	8.51	8.18	1.46	1.45
67	1.00	81.50	62.70	10.37	23.51	23.51	1.58	1.57
33	1.95	50.82	42.93	4.25	10.73	10.47	1.59	1.63
84	7.49	44.18	38.91	1.82	8.51	8.18	1.73	1.67
39	2.19	51.79	43.78	4.04	10.94	10.67	1.72	1.75
17	13.00	51.32	43.37	0.79	10.84	10.58	2.13	1.77
1	7.33	51.79	43.78	2.41	10.94	10.67	1.95	1.82
81	2.97	51.79	43.78	3.61	10.94	10.67	1.88	1.88
37	2.13	50.82	42.93	3.24	10.73	10.47	2.26	2.27
4	2.35	83.64	64.39	7.60	24.15	24.15	2.37	2.30
66	2.15	51.79	43.78	3.25	10.94	10.67	2.31	2.33

Note: See Figure 11.2.12.4.1-2 for disk location and Figure 11.2.12.4.1-7 for section locations.

11.2.12.4.2 Analysis of Canister and Basket for BWR Configurations

Five three-dimensional models of the BWR canister and fuel basket are evaluated for the cask tip-over event. Each model corresponds to a different fuel basket drop orientation. For the BWR fuel configuration, fuel basket drop orientations of 0°, 31.82°, 49.46°, 77.92°, and 90° are evaluated, as shown in Figure 11.2.12.4.2-1. Three-dimensional half-symmetry models are used for the basket drop orientations of 0° and 90°. Three-dimensional full-models are used for the basket orientations of 31.82°, 49.46° and 77.92°.

Model Description

The models used for the evaluation of the canister and basket for BWR configuration are similar to those used for the PWR (Section 11.2.12.4.1). The three-dimensional model used for the basket drop orientation of 31.82° is presented in Figure 11.2.12.4.2-2 and Figure 11.2.12.4.2-3.

The same modeling and analysis techniques described for the PWR model (see Section 11.2.12.4.1) are used for the BWR models. Loading of the BWR models includes an internal pressure of 15 psig (design pressure for normal condition of storage) applied to the inner surfaces of the canister, pressure loads applied to the support disk slots and the inertial loads. The pressure load applied to the support disk slots represents the combined weight of the BWR fuel assemblies, fuel tubes and aluminum heat transfer disks multiplied by 30g. Note that the BWR fuel assembly weight is 702 pounds.

For the inertial loads, a maximum acceleration of 30g is conservatively applied to the entire model. As shown in Section 11.2.12.3.2, the maximum acceleration of the concrete cask steel liner at the locations of the top support disk and the top of the canister structural lid during the tip-over event is determined to be 24.2g and 28.0g, respectively. Using the same method described in Section 11.2.12.4.1 for the PWR models, the DLF for the acceleration at the top support disk is computed to be 1.09. Applying the DLF to the 24.2g results in a peak acceleration of 26.4g for the top support disk.

The dominant resonance frequencies and corresponding modal mass participation factors from the finite element modal analyses of the BWR support disk are:

Frequency (Hz)	% Modal Mass Participation Factor
79.3	38.4
80.2	54.9
210.9	3.4

The mode shapes for these frequencies are shown in Figures 11.2.12.4.2-5 through 11.2.12.4.2-7. The displacement depicted in these figures is highly exaggerated by the ANSYS program in order to illustrate the modal shape. The stresses associated with the actual displacement are shown in Tables 11.2.12.4.2-4 through 11.2.12.4.2-8.

The DLFs for the canister lids are considered to be unity since the lids have significant in-plane stiffness and are considered to be rigid. Therefore, applying 30g to the entire canister/basket model is conservative.

A uniform temperature of 75°F is applied to the model to determine material properties during solution. During post processing for the support disk, temperature distribution with a maximum temperature of 700°F (at the center) and a minimum temperature of 400°F (at the outer edge) are conservatively used to determine the allowable stresses. A constant temperature of 500° is used for the canister to determine the allowable stresses. These temperatures are the bounding temperatures for the normal, off-normal and accident conditions of storage.

Analysis Results for Canister

The sectional stresses at 13 axial locations of the canister are obtained for each angular division of the model (a total of 80 angular locations for the full-models and a total of 41 angular locations for the half-symmetry models). The locations for the stress sections are shown in Figure 11.2.12.4.1-6.

The same stress allowables used in the evaluation of the PWR canister (see Section 11.2.12.4.1) are used in evaluating the BWR canister.

The primary membrane and primary membrane plus bending stresses for the BWR configuration for a 49.46° basket drop orientation are summarized in Table 11.2.12.4.2-1 and Table 11.2.12.4.2-2, respectively. The stress results of the canister are similar for all five models. Only the 49.46° basket drop orientation results are presented for the canister because this drop orientation generates the minimum margin of safety in the canister. The stress evaluation results for tip-over accident conditions show that the minimum margin of safety in the canister for BWR configurations is +0.35 for P_m (Section 10) and +0.46 for P_m+P_b (Section 10).

Analysis Results for Support Disks

To evaluate the most critical regions of the support disk, a series of cross sections are considered. To aid in the identification of these sections, Figure 11.2.12.4.2-4 shows the locations on a support disk for the full-models. Table 11.2.12.4.2-3 lists the cross-sections with their end point locations (Point 1 and Point 2), which spans the cross section of the ligament in the plane of the support disk. Note that a local coordinate system (x and y parallel to the support disk ligaments) is used for the stress evaluation.

The stress evaluation for the support disk is performed according to ASME Code, Section III, Subsection NG. The allowable stresses for each section are determined based on the temperature of the support disk at the section location. The temperature distribution of the disk is determined by a thermal conduction solution for a single disk with a temperature of 700°F specified at the center of the disk and a temperature of 400°F specified at the outer edge of the disk as boundary conditions. These temperatures are bounding temperatures for the normal, off-normal and accident conditions of storage.

The highest stress occurs at the 5th support disk. The stress evaluation results for the 5th support disk are summarized in Table 11.2.12.4.2-4 for the five basket drop orientations evaluated. As shown in Table 11.2.12.4.2-4, the 77.92° drop orientation case generates the minimum margin of safety in the support disk; therefore, the P_m and $P_m + P_b$ stress intensities for the 77.92° basket drop orientation case are presented in Table 11.2.12.4.2-6 and Table 11.2.12.4.2-7, respectively. These tables list the stresses with the 30 lowest margins of safety for the 5th support disk. The highest P_m stress occurs at Section 202, with a margin of safety of +0.33 (See Table 11.2.12.4.2-6 for stresses and Figure 11.2.12.4.2-4 for section locations). The highest P_m+P_b stress occurs at Section 169, with a margin of safety of +0.04 (see Table 11.2.12.4.2-7 for stresses and Figure 11.2.12.4.2-4 for section locations).

Support Disk Buckling Evaluation

The support disk buckling evaluation for the BWR support disks is performed using the same method as that presented for the PWR support disks (see Section 11.2.12.4.1). The support disk buckling evaluation results for the 5th support disk (the 5th support disk experiences the highest stresses) for the tip-over impact condition are summarized in Table 11.2.12.4.2-5 for the five basket drop orientations evaluated. As shown in Table 11.2.12.4.2-5, the 77.92° drop orientation case generates the minimum margin of safety for buckling; therefore, the results of the buckling analysis for the 77.92° basket drop orientation case are presented in Table 11.2.12.4.2-8. This table presents the results for 30 minimum margins of safety for this drop orientation. As the tables demonstrate, the support disks meet the requirements of NUREG/CR-6322.

Fuel Tube Analysis

The fuel tube provides structural support and a mounting location for neutron absorber plates. The fuel tube does not provide structural support for the fuel assembly. To ensure that the fuel tube remains functional during a tip-over accident, a structural evaluation of the tube is performed for a side impact assuming a deceleration of 60g. This g-load bounds the maximum g-load (30g) calculated to occur for the BWR basket in a vertical concrete cask tipover event.

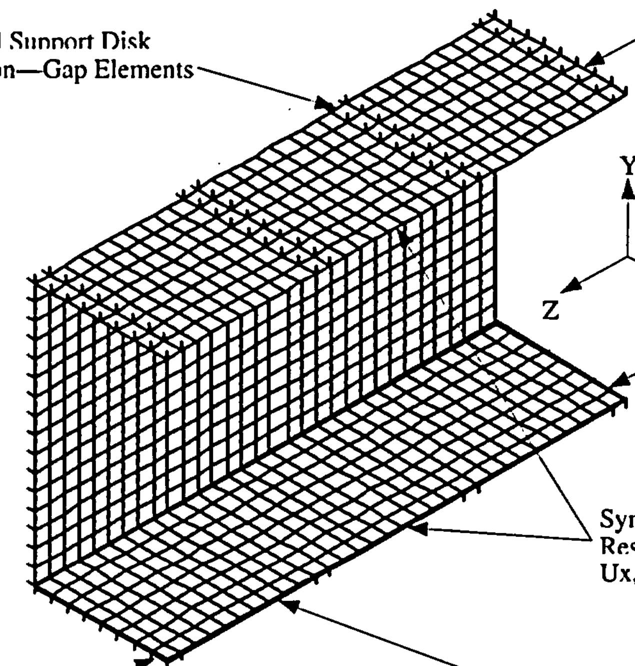
In the tipover event, the stainless steel support disks in the fuel basket support the fuel tube. The fuel basket support disks, which support the full length of the fuel tube, are spaced 3.205-inches apart (which is slightly more than one half of the fuel tube width of 5.9 inch). Considering the fuel tube subjected to a maximum BWR fuel assembly weight of 702 pounds with a 60g load factor and the 40 support locations provided by the basket support disks, the fuel tube shear stress is calculated as:

Shear load = (60g)(702)/40 = 1,053 lbs
Area = (0.048)(5.9)(2) = 0.566 in^2
Shear Stress = 1,053/0.566 = 1,860 psi

The yield strength of the tube material, Type 304 stainless steel, is 17,300 psi at 750°F. Conservatively using the allowable shear stress as one- half the yield strength of the tube material (8,650 psi) results in a large positive margin of safety. Conservative evaluation of the tube loading resulting from its own mass during a side impact shows that the tube structure maintains position and function.

The load transfer of the fuel assembly to the weight of the fuel basket support disk in the side impact is through direct bearing and compression of the distributed load of the fuel assembly through the fuel tube to the support disk web. Two load conditions are considered in the fuel tube evaluation. The first considers the fuel assembly load as a distributed pressure on the inside surface of the fuel tube. The second postulates that the fuel assembly grid is located at the center of the span between the support disks and produces a localized distributed load over the effective area of the grid.

Two different ANSYS finite element models of the tube are developed for these two load conditions since the fuel assembly structural performance for either load is nonlinear. As shown below, the first model represents a fuel tube section with a length of three spans, i.e., the model is



supported at four locations by support disks. The model conservatively considers the fuel tube wall thickness of 0.048 inch as the only material subjected to a distributed pressure load representative of the fuel assembly deceleration of 60g. Fuel assembly stiffness is not considered in the development of the imposed pressure load on the fuel tube.

The fuel tube is modeled with the ANSYS plastic, quadrilateral shell element (SHELL43). The support disks are represented as rigid gap elements (CONTAC52). The outer nodes of the gap elements are fully restrained in all three translational directions. Edge restraints were applied to the model to represent symmetry boundary conditions. The effective load on the fuel tube due to the 60g deceleration of the assembly is applied as a pressure to the inside area of the fuel tube.

The finite element analysis results show that the maximum stress in the tube is 19.5 ksi, which is local to the sections of the tube resting on the support disks. At 750°F the ultimate strength for Type 304 stainless steel is 63.1 ksi. The margin of safety is:

$$MS = \frac{63.1}{19.5} - 1 = +2.24$$

The analysis shows that the maximum total strain is 0.0078 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F [42], the resulting margin of safety is:

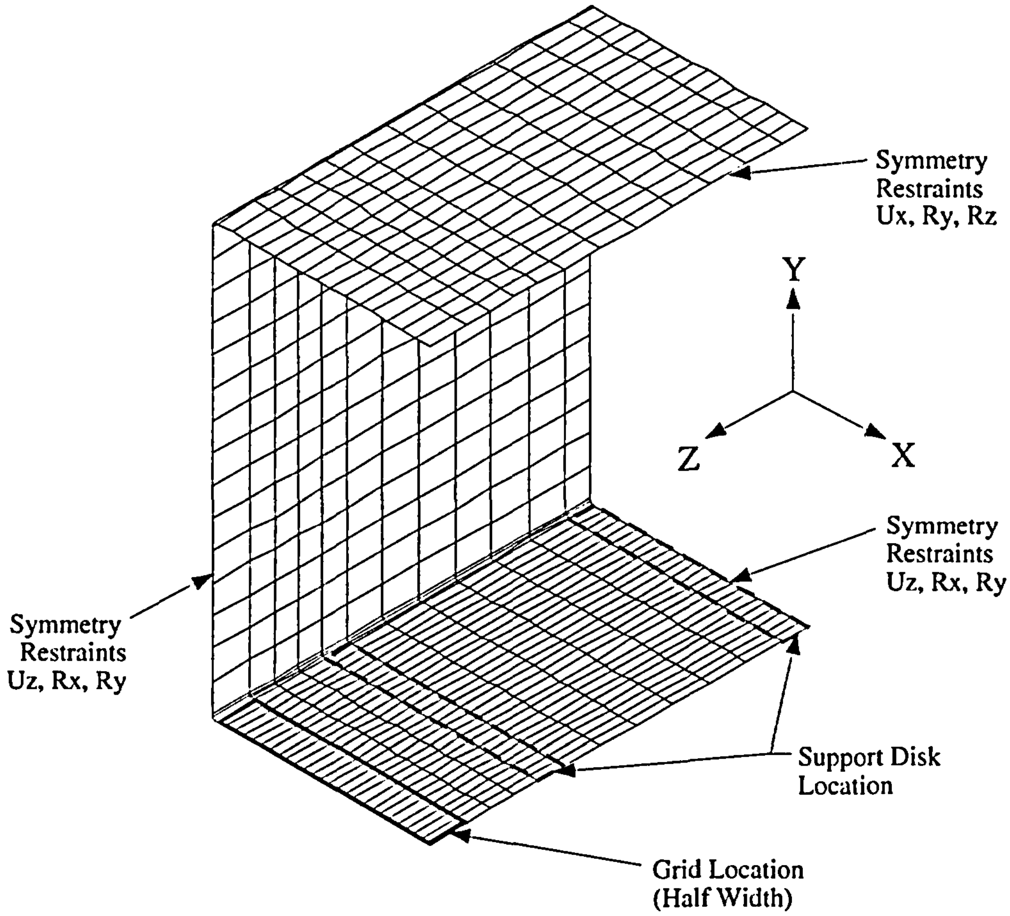
$$MS = \frac{0.40/2}{0.0078} - 1 = +\text{Large}$$

Similarly, the margin of safety for elastic-plastic stress becomes

$$MS = \frac{63.1 - 17.3}{19.5 - 17.3} - 1 = +\text{Large}$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

The second finite element model is used to evaluate the load condition with the fuel assembly grid located at the center of the span between two support disks. The fuel tube is subjected to a localized distributed load over the effective area of the grid. As shown below, the model is a quarter-symmetry periodic section of the fuel tube. As in the finite element model used for the distributed pressure case, this model conservatively considers a fuel tube wall thickness of 0.048 inch. The neutron absorber plate (0.135 inch) and stainless steel cover plate (0.018 inch) are conservatively not included in the model. The tube wall is modeled with ANSYS SHELL43 elements. The support disks are modeled with CONTAC52 elements. A uniform pressure corresponding to the fuel assembly weight with the 60g load is applied to the elements at the grid location of the model. The displacement in the Y-direction for the nodes at the grid location of the model are coupled to represent the structural rigidity of the spacer grid.



The finite element analysis results show that the maximum stress in the tube is 40.8 ksi. At 750°F, the ultimate strength for Type 304 stainless steel is 63.1 ksi. The margin of safety is

$$MS = \frac{63.1}{40.8} - 1 = +0.54$$

The analysis shows that the maximum total strain is 0.10 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F [42], the resulting margin of safety is:

$$MS = \frac{0.40/2}{0.127} - 1 = +0.57$$

Similarly, the margin of safety for elastic-plastic stress becomes

$$MS = \frac{63.1 - 17.3}{40.8 - 17.3} - 1 = +0.94$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

Fuel Tube Yielding

Using the displacement of the fuel rod, a check of the fuel tube is performed to verify that the fuel tube remains elastic during a side-drop scenario. The fuel rod displacement loading is a more realistic loading condition because the load is transmitted from the fuel rods to the fuel tube. The analysis is conservative as it assumes the cumulative displacement of 9 fuel rods (stacked on top of each other) in a 9×9 PWR fuel assembly.

The displacement of a single fuel rod assumed as a four-span continuous beam is calculated as

$$\Delta_{max} = 0.0065\frac{wL^4}{EI} = 4.415 \times 10^{-6} \text{ in}$$

where:

w = mass/length = $\rho_{zirc}A_{zirc} + \rho_{UO2}A_{UO2}$ = 0.05 lb/in × 9 rods = 0.4498 lb/in

Rod OD = 0.424 in

Rod ID = 0.424-2 × 0.03 = 0.364 in

Rod Density (Zirc-4) = ρ_{zirc} = 0.237 lb/in^3

Rod Area = $A_{zirc} = \frac{\pi}{4}(0.424^2 - 0.364^2) = 0.0371 \text{ in}^2$

UO_2 Density = ρ_{UO_2} = 0.396 lb/in^3

UO_2 Area = $A_{UO_2} = \frac{\pi}{4}x0.364^2 = 0.104 \text{ in}^2$

L = Distance between support disks = 3.205 in

E_{zirc} = 10.75 × 10^6 psi

$I_{zirc} = \frac{\pi}{64}(0.424^4 - 0.364^4) = 7.247 \times 10^{-4} \text{ in}^4 \times 9 \text{ rods} = 0.0065 \text{ in}^4$

Using the E_{zirc} and I_{zirc} as conservative assumptions, the maximum displacement is estimated as 4.415×10^{-6} in. For 60g acceleration, this displacement becomes 0.0003 inch.

Applying the displacement midway between support disks, the maximum stress intensity is 5,812 psi. The yield stress for the fuel tube (Type 304 stainless steel) is 17,300 psi at 750°F degrees; therefore, during a 60g side-drop, the fuel tube remains elastic.

Both the maximum total strain and the elastic-plastic stress analyses indicate that the tube position within the support basket is maintained.

Assurance that the neutron absorber remains attached to the fuel tube is evaluated by considering that loads produced by the neutron absorber plate and stainless steel attachment plate, assuming a 60g load, are carried by the attachment plate weld. Total load and resultant stress on the weld are calculated as:

$F_{b/ss} = (g)(\rho)(t)(w)(l)$ Load exerted by neutron absorber/stainless steel attachment plate

where:

- g = acceleration (g)
- ρ = density of material (lb/in^3) (The density of aluminum (0.098 lb/in^3) is conservatively used for the neutron absorber.
- t = thickness of material (in.)
- w = width of material (in.)
- l = length of material section (in.)

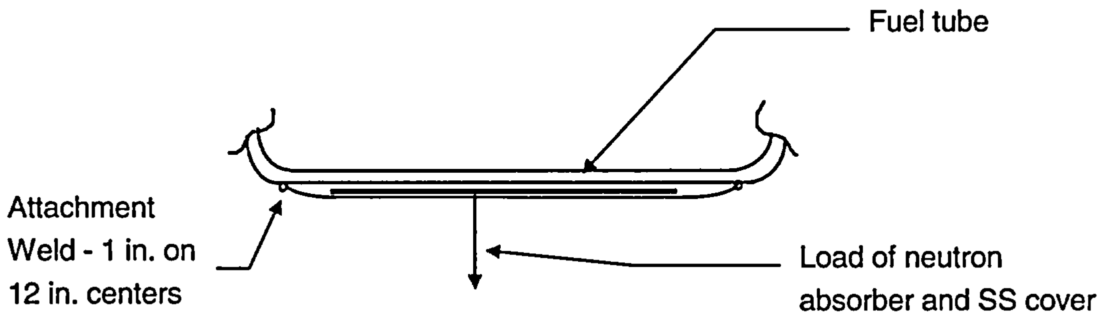
The forces on the weld due to a 12-inch section of neutron absorber (F_b) and a 12-inch section of stainless steel plate (F_{ss}) are:

$$F_b = (60g)(0.098\ \text{lb/in}^3)(0.135\ \text{in})(5.45\ \text{in})(12\ \text{in}) = 51.9\ \text{lbs}$$

$$F_{ss} = (60g)(0.291\ \text{lb/in}^3)(0.018\ \text{in})(5.79\ \text{in})(12\ \text{in}) = 21.8\ \text{lbs}$$

The total load (F_t) on a 1-inch attachment for a 12-inch section is:

$$F_t = 57.9\ \text{lbs} + 21.8\ \text{lbs} = 73.7\ \text{lbs}$$



The resulting weld stress is: $\sigma = P/A = (73.7 \text{ lbs}/2) / (1 \text{ in}) (0.018 \text{ in}) = 2{,}074$ psi

Since the weld material is Type 304 stainless steel, the margin of safety (at 750°F) is:

$$MS = \frac{17{,}300}{2{,}047} - 1 = +7.5$$

Therefore, the neutron absorber remains enclosed on each outer surface of the fuel tube wall.

Figure 11.2.12.4.2-1 Fuel Basket Drop Orientations Analyzed for Tip-Over Condition - BWR

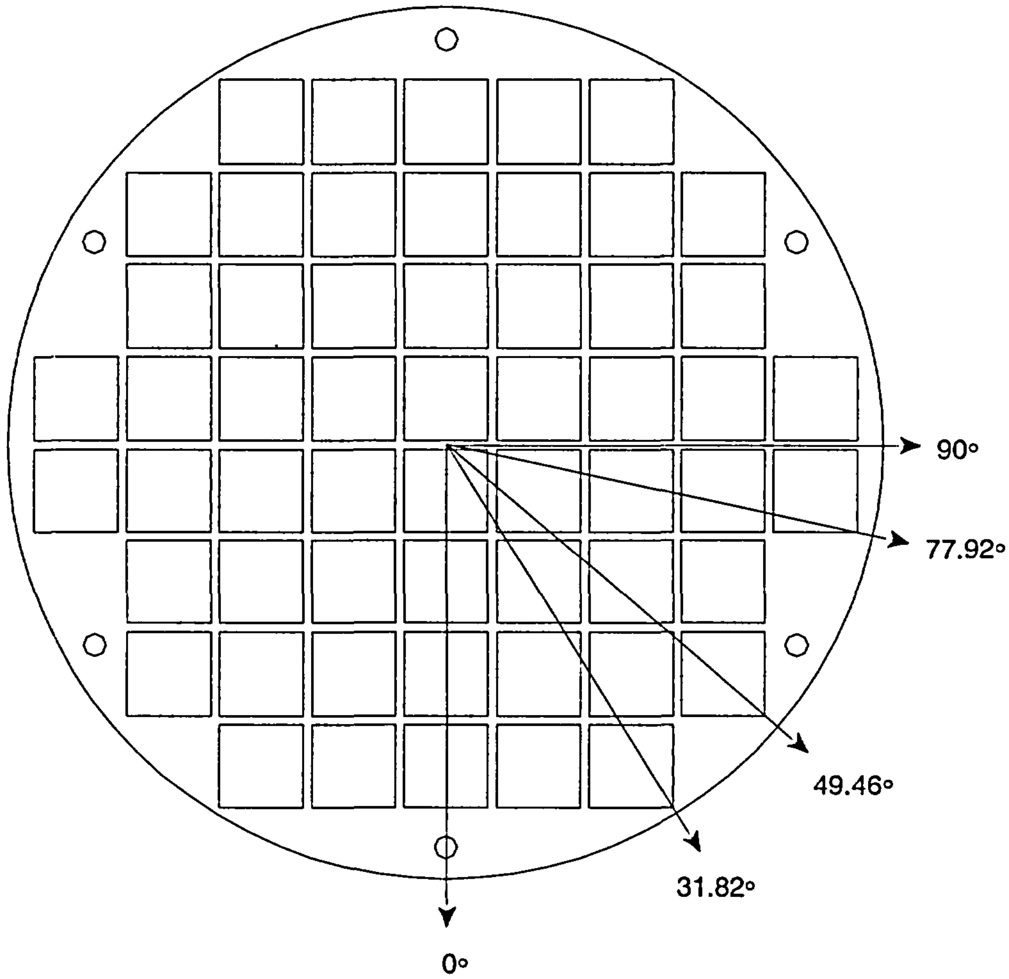
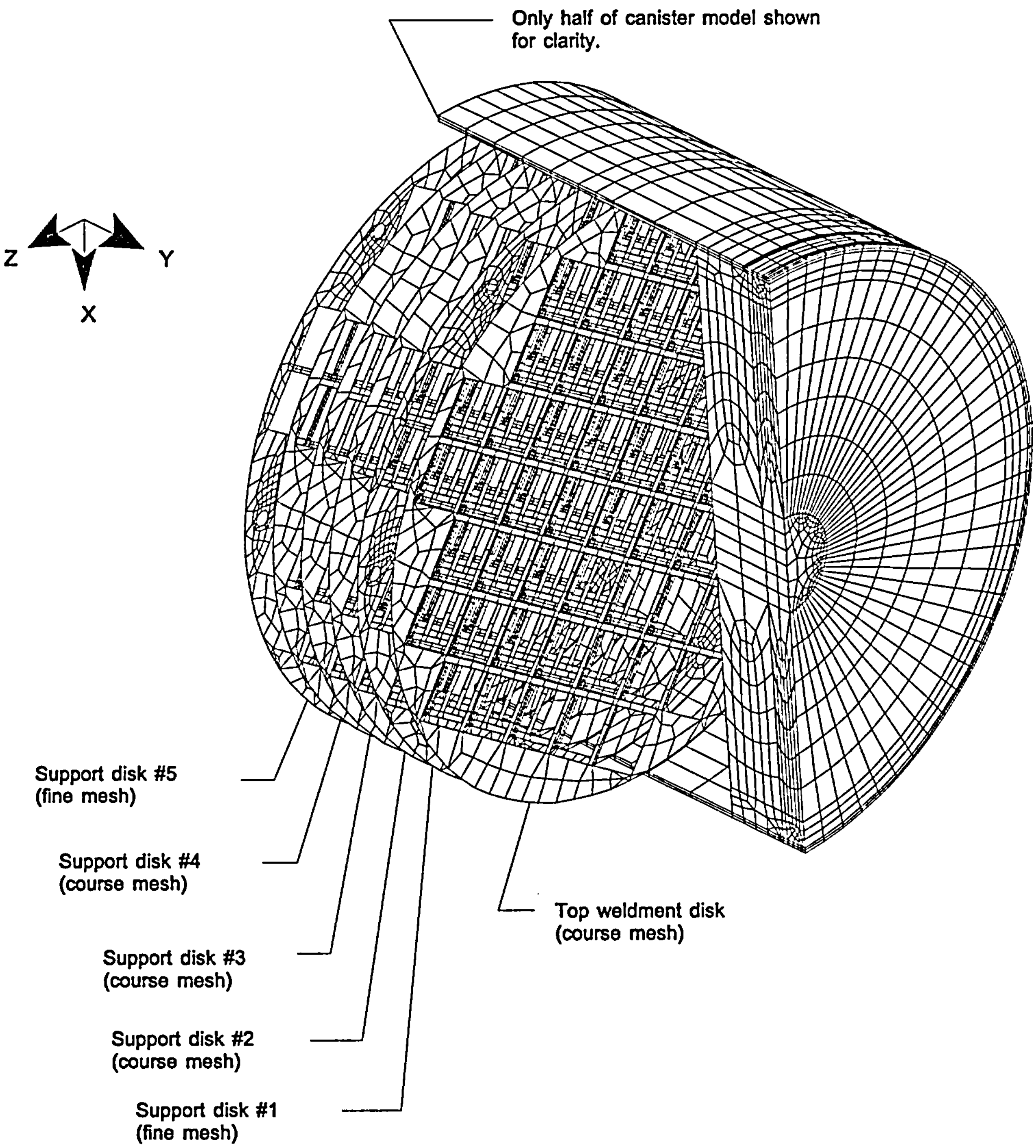


Figure 11.2.12.4.2-2 Fuel Basket/Canister Finite Element Model - BWR



31.82° Basket Drop Orientation

Figure 11.2.12.4.2-3 Fuel Basket/Canister Finite Element Model - Support Disk - BWR

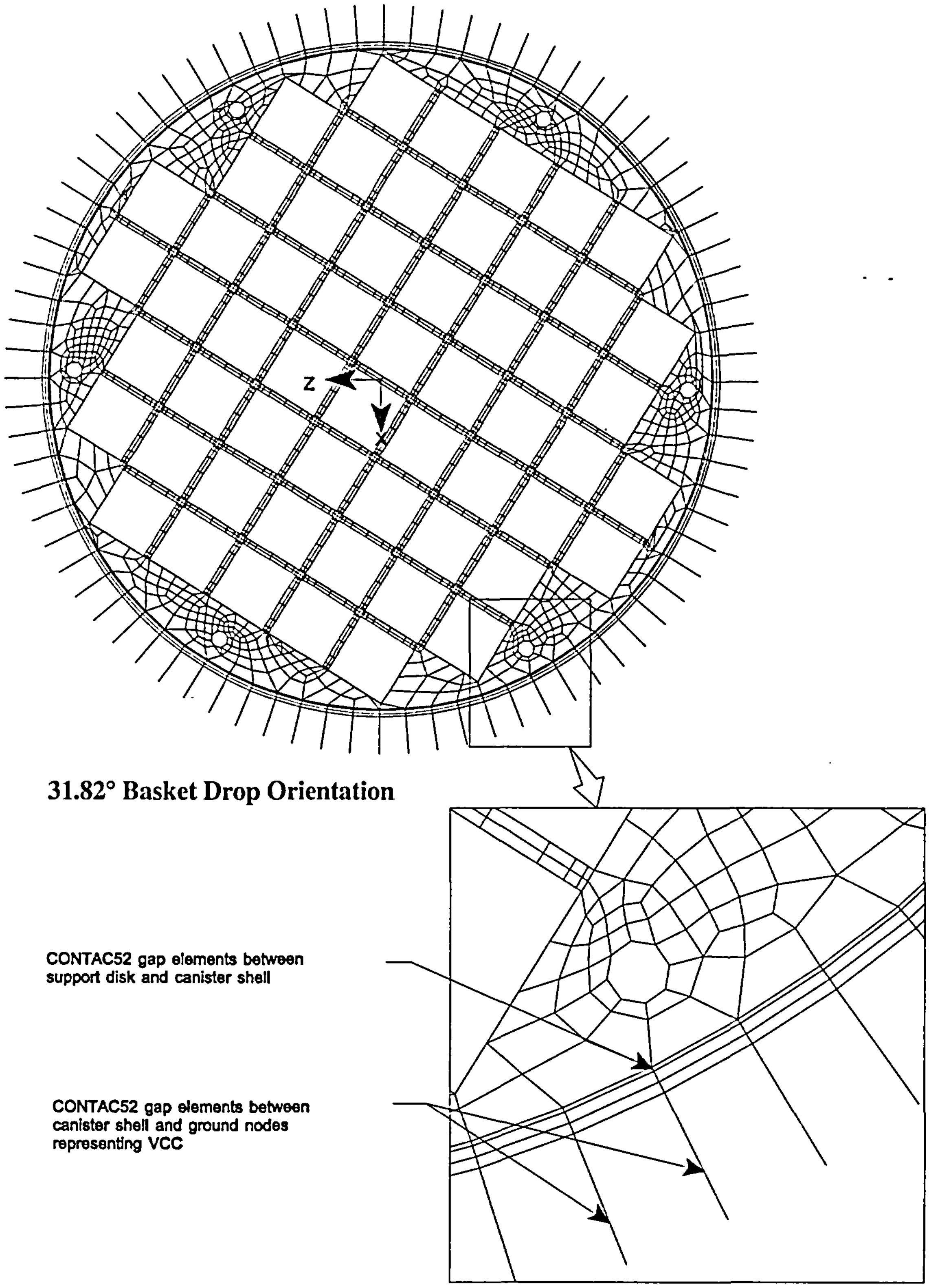


Figure 11.2.12.4.2-4 Support Disk Section Stress Locations - BWR - Full Model

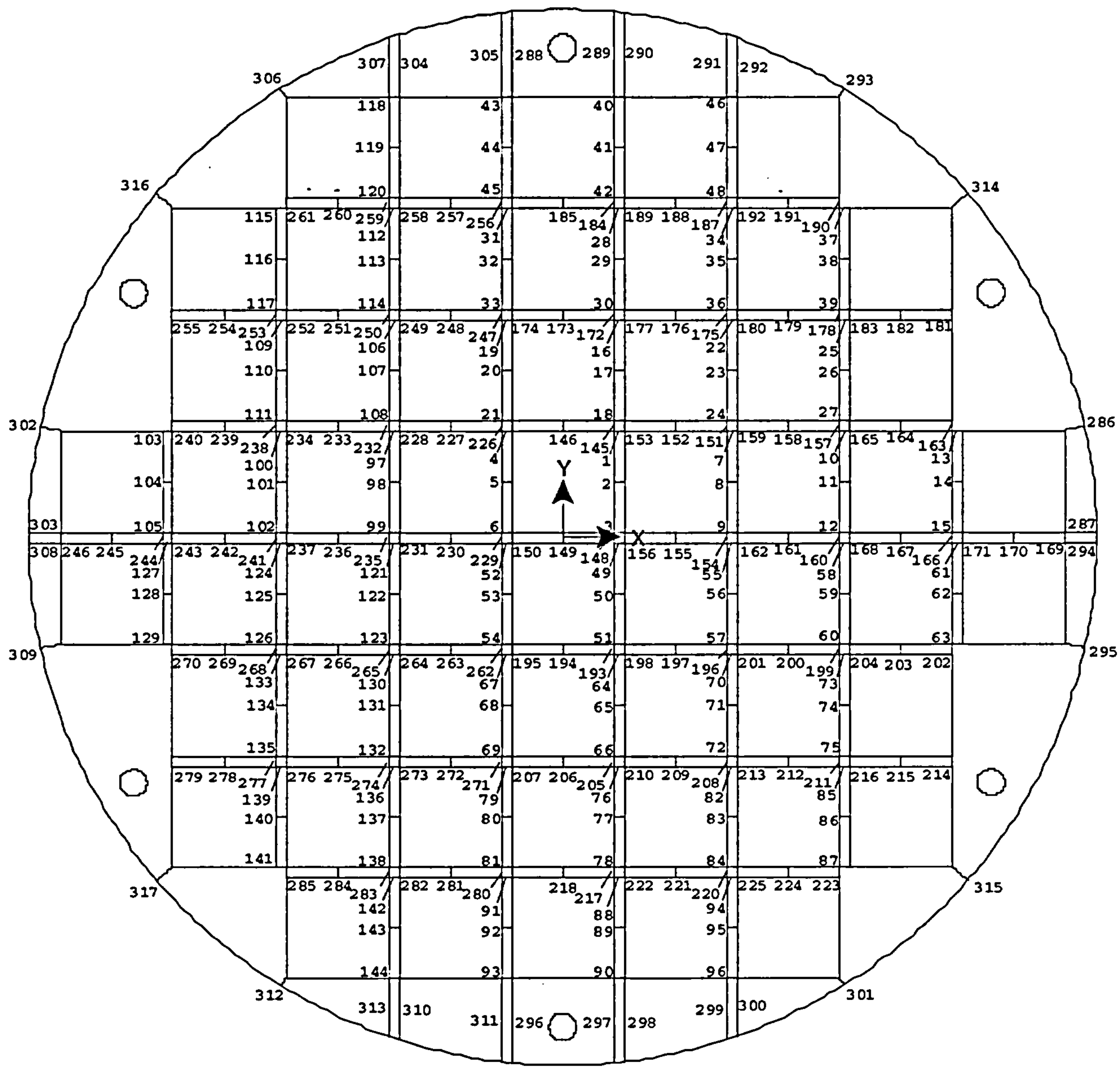
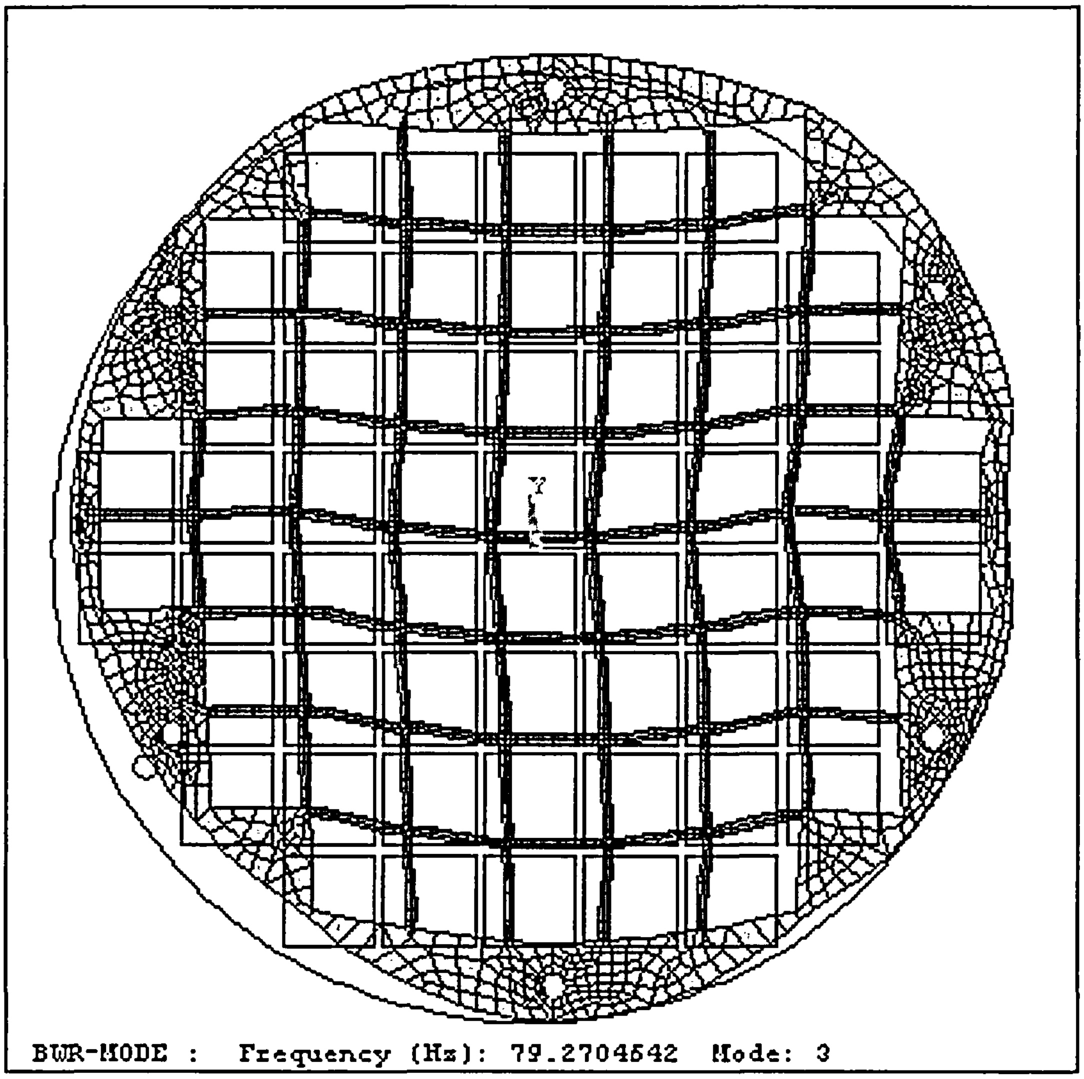
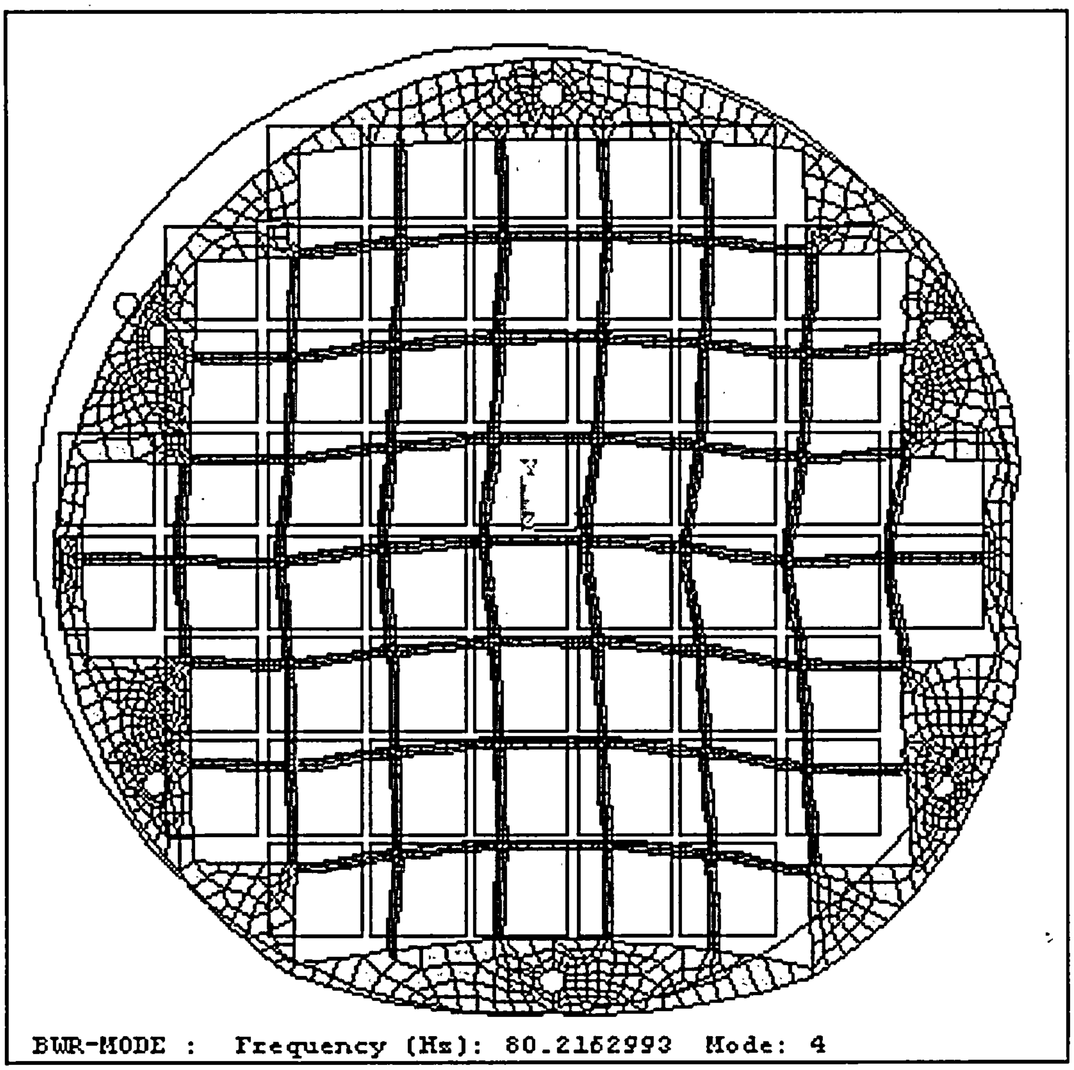


Figure 11.2.12.4.2-5 BWR – 79.3 Hz Mode Shape



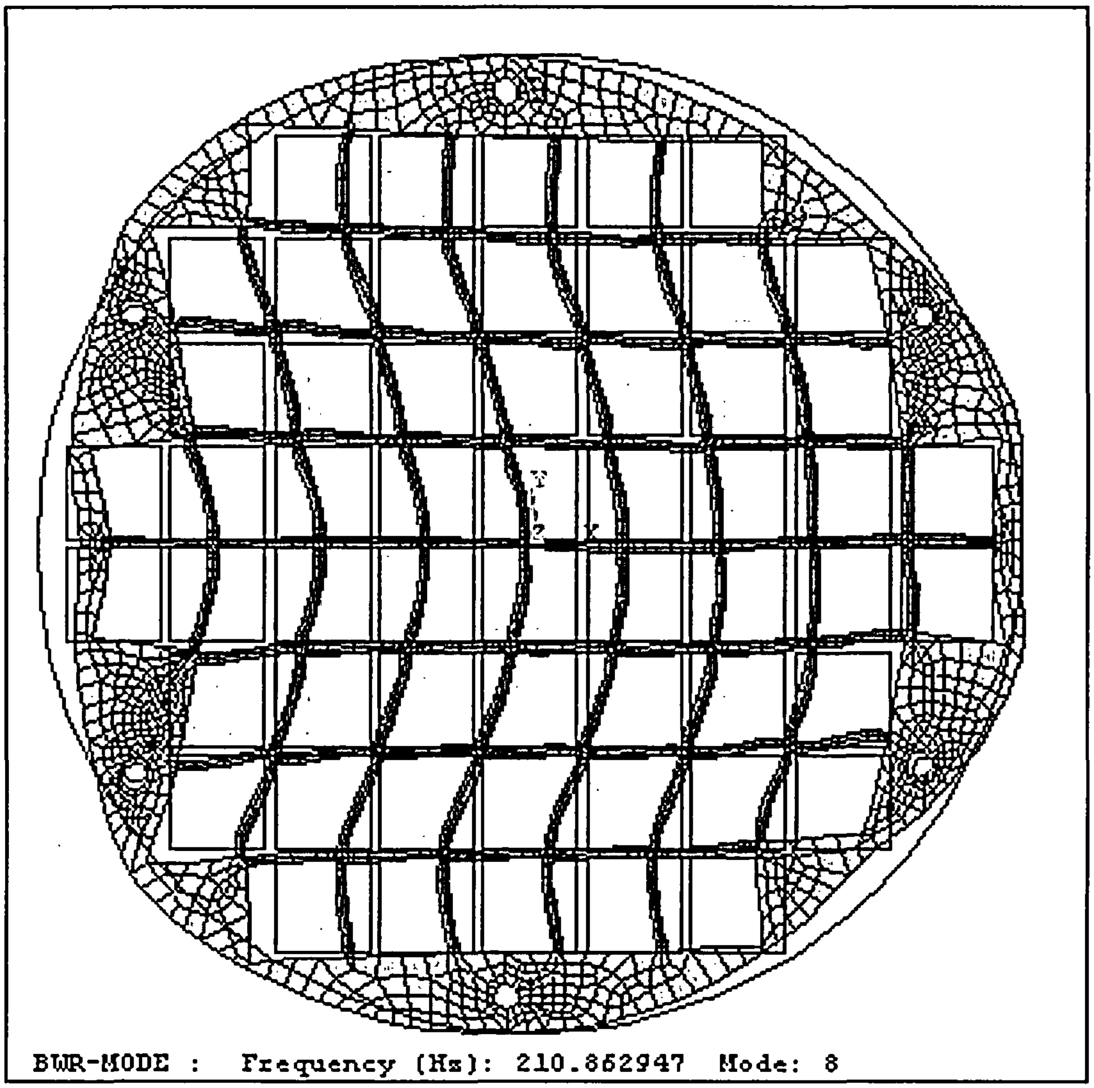
Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Figure 11.2.12.4.2-6 BWR – 80.2 Hz Mode Shape



Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Figure 11.2.12.4.2-7 BWR – 210.9 Hz Mode Shape



Note: Displacements are greatly exaggerated by the ANSYS program to illustrate the mode shapes.

Table 11.2.12.4.2-1 Canister Primary Membrane (P_m) Stresses for Tip-Over Conditions - BWR - 49.46° Basket Drop Orientation (ksi)

Section Location(1)	Section Angle (deg)	Sx	Sy	Sz	Sxy	Syz	Sxz	Stress Intensity	Allowable Stress	Margin of Safety
1	0	-1.2	6.2	1.4	-0.1	-0.1	0.0	7.46	35.52	3.76
2	0	-1.6	8.2	1.4	0.0	-0.2	0.1	9.77	35.52	2.63
3	0	-1.5	7.9	1.4	0.0	-0.2	-0.1	9.41	35.52	2.78
4	90	-0.1	3.0	-2.1	-0.2	3.7	0.1	8.92	35.52	2.98
5	85.5	0.0	2.8	-1.0	-0.2	4.8	-0.1	10.29	35.52	2.45
6	76.5	0.0	0.3	-0.4	0.0	6.0	0.0	12.09	35.52	1.94
7(2)	9.0	0.6	0.3	4.8	1.6	-3.8	-0.2	9.60	35.52	2.70
8(2)	351.0	4.5	0.1	5.2	-0.1	2.3	-0.6	7.06	35.52	4.03
9(2)	351.0	4.5	-1.0	1.5	-1.6	2.8	-0.2	8.17	35.52	3.35
10	0	-38.6	-16.2	-30.4	0.5	0.0	-10.7	29.74	40.08(3)	0.35
11(4)	351.9 – 8.2	-22.1	-9.9	-6.7	-0.1	0.0	1.1	15.51	32.06(4)	1.07
12	0	-0.6	0.2	0.0	0.0	0.0	-0.3	0.92	35.52	37.66
13	0	-1.0	0.3	0.0	0.0	0.0	-0.4	1.46	35.52	23.31

Stresses are presented in the cylindrical coordinate system, x = radial, y = circumferential and z = axial directions.

1. Section locations are shown in Figure 11.2.12.4.1-6.
2. Stresses are not presented for the sections with localized bearing stress. In accordance with ASME Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions.
3. Allowable stress at 300°F.
4. Stresses are determined by averaging the stresses over the impact region. A stress reduction factor of 0.8 is applied to the allowable stress at 250°F.

Table 11.2.12.4.2-2 Canister Primary Membrane + Primary Bending ($P_m + P_b$) Stresses for Tip-Over Conditions - BWR - 49.46° Basket Drop Orientation (ksi)

Section Location(1)	Section Angle (deg)	Sx	Sy	Sz	Sxy	Syz	Sxz	Stress Intensity	Allowable Stress	Margin of Safety
1	0.0	-1.6	18.5	4.6	-0.2	-0.4	0.1	20.13	53.28	1.65
2	0.0	-1.8	20.2	2.7	0.0	-0.4	0.1	22.01	53.28	1.42
3	0.0	-2.3	20.6	4.8	-0.1	-0.3	-0.1	22.92	53.28	1.32
4	0.0	-1.8	20.2	3.9	-0.2	-0.4	-0.1	22.00	53.28	1.42
5	0.0	-2.2	19.7	6.4	-0.1	-0.6	0.1	21.94	53.28	1.43
6	0.0	0.0	-21.0	-3.8	0.0	-0.7	-0.7	21.21	53.28	1.51
7(2)	351.0	0.1	6.4	17.2	0.2	2.3	0.2	17.50	53.28	2.04
8(2)	351.0	3.3	5.2	13.5	0.7	3.6	-2.1	13.02	53.28	3.09
9(2)	351.0	5.9	-3.0	3.6	-3.0	3.2	-0.6	12.44	53.28	3.28
10	0.0	-42.9	-15.8	-27.8	0.4	0.3	-19.1	41.17	60.12(3)	0.46
11(4)	351.9 – 8.1	-18.8	-7.2	-1.7	-0.1	0.0	2.6	17.86	48.09(4)	1.69
12	0.0	-0.9	0.1	-0.1	0.0	0.0	-0.5	1.37	53.28	37.81
13	0.0	-1.1	0.4	0.0	0.0	0.0	-0.1	1.56	53.28	33.07

Stresses are presented in the cylindrical coordinate system, x = radial, y = circumferential and z = axial directions.

1. Section locations are shown in Figure 11.2.12.4.1-6.
2. Stresses are not presented for the sections with localized bearing stress. In accordance with ASME Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions.
3. Allowable stress at 300°F.
4. Stresses are determined by averaging the stresses over the impact region. A stress reduction factor of 0.8 is applied to the allowable stress at 250°F.

Table 11.2.12.4.2-3 Support Disk Section Locations for Stress Evaluation - BWR - Full Model

Section[1]	Point 1		Point 2		Section[1]	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
1	3.14	6.6	3.79	6.6	44	-3.14	24.25	-3.79	24.25
2	3.14	3.46	3.79	3.46	45	-3.14	21.11	-3.79	21.11
3	3.14	0.33	3.79	0.33	46	10.07	27.39	10.72	27.39
4	-3.14	6.6	-3.79	6.6	47	10.07	24.25	10.72	24.25
5	-3.14	3.46	-3.79	3.46	48	10.07	21.11	10.72	21.11
6	-3.14	0.33	-3.79	0.33	49	3.14	-0.33	3.79	-0.33
7	10.07	6.6	10.72	6.6	50	3.14	-3.46	3.79	-3.46
8	10.07	3.46	10.72	3.46	51	3.14	-6.6	3.79	-6.6
9	10.07	0.33	10.72	0.33	52	-3.14	-0.33	-3.79	-0.33
10	17	6.6	17.65	6.6	53	-3.14	-3.46	-3.79	-3.46
11	17	3.46	17.65	3.46	54	-3.14	-6.6	-3.79	-6.6
12	17	0.33	17.65	0.33	55	10.07	-0.33	10.72	-0.33
13	23.92	6.6	24.57	6.6	56	10.07	-3.46	10.72	-3.46
14	23.92	3.46	24.57	3.46	57	10.07	-6.6	10.72	-6.6
15	23.92	0.33	24.57	0.33	58	17	-0.33	17.65	-0.33
16	3.14	13.53	3.79	13.53	59	17	-3.46	17.65	-3.46
17	3.14	10.39	3.79	10.39	60	17	-6.6	17.65	-6.6
18	3.14	7.25	3.79	7.25	61	23.92	-0.33	24.57	-0.33
19	-3.14	13.53	-3.79	13.53	62	23.92	-3.46	24.57	-3.46
20	-3.14	10.39	-3.79	10.39	63	23.92	-6.6	24.57	-6.6
21	-3.14	7.25	-3.79	7.25	64	3.14	-7.25	3.79	-7.25
22	10.07	13.53	10.72	13.53	65	3.14	-10.39	3.79	-10.39
23	10.07	10.39	10.72	10.39	66	3.14	-13.53	3.79	-13.53
24	10.07	7.25	10.72	7.25	67	-3.14	-7.25	-3.79	-7.25
25	17	13.53	17.65	13.53	68	-3.14	-10.39	-3.79	-10.39
26	17	10.39	17.65	10.39	69	-3.14	-13.53	-3.79	-13.53
27	17	7.25	17.65	7.25	70	10.07	-7.25	10.72	-7.25
28	3.14	20.46	3.79	20.46	71	10.07	-10.39	10.72	-10.39
29	3.14	17.32	3.79	17.32	72	10.07	-13.53	10.72	-13.53
30	3.14	14.18	3.79	14.18	73	17	-7.25	17.65	-7.25
31	-3.14	20.46	-3.79	20.46	74	17	-10.39	17.65	-10.39
32	-3.14	17.32	-3.79	17.32	75	17	-13.53	17.65	-13.53
33	-3.14	14.18	-3.79	14.18	76	3.14	-14.18	3.79	-14.18
34	10.07	20.46	10.72	20.46	77	3.14	-17.32	3.79	-17.32
35	10.07	17.32	10.72	17.32	78	3.14	-20.46	3.79	-20.46
36	10.07	14.18	10.72	14.18	79	-3.14	-14.18	-3.79	-14.18
37	17	20.46	17.65	20.46	80	-3.14	-17.32	-3.79	-17.32
38	17	17.32	17.65	17.32	81	-3.14	-20.46	-3.79	-20.46
39	17	14.18	17.65	14.18	82	10.07	-14.18	10.72	-14.18
40	3.14	27.39	3.79	27.39	83	10.07	-17.32	10.72	-17.32
41	3.14	24.25	3.79	24.25	84	10.07	-20.46	10.72	-20.46
42	3.14	21.11	3.79	21.11	85	17	-14.18	17.65	-14.18
43	-3.14	27.39	-3.79	27.39	86	17	-17.32	17.65	-17.32

1. See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-3 Support Disk Section Locations for Stress Evaluation - BWR - Full Model (Continued)

Section[1]	Point 1		Point 2		Section[1]	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
87	17	-20.46	17.65	-20.46	130	-10.07	-7.25	-10.72	-7.25
88	3.14	-21.11	3.79	-21.11	131	-10.07	-10.39	-10.72	-10.39
89	3.14	-24.25	3.79	-24.25	132	-10.07	-13.53	-10.72	-13.53
90	3.14	-27.39	3.79	-27.39	133	-17	-7.25	-17.65	-7.25
91	-3.14	-21.11	-3.79	-21.11	134	-17	-10.39	-17.65	-10.39
92	-3.14	-24.25	-3.79	-24.25	135	-17	-13.53	-17.65	-13.53
93	-3.14	-27.39	-3.79	-27.39	136	-10.07	-14.18	-10.72	-14.18
94	10.07	-21.11	10.72	-21.11	137	-10.07	-17.32	-10.72	-17.32
95	10.07	-24.25	10.72	-24.25	138	-10.07	-20.46	-10.72	-20.46
96	10.07	-27.39	10.72	-27.39	139	-17	-14.18	-17.65	-14.18
97	-10.07	6.6	-10.72	6.6	140	-17	-17.32	-17.65	-17.32
98	-10.07	3.46	-10.72	3.46	141	-17	-20.46	-17.65	-20.46
99	-10.07	0.33	-10.72	0.33	142	-10.07	-21.11	-10.72	-21.11
100	-17	6.6	-17.65	6.6	143	-10.07	-24.25	-10.72	-24.25
101	-17	3.46	-17.65	3.46	144	-10.07	-27.39	-10.72	-27.39
102	-17	0.33	-17.65	0.33	145	3.14	6.6	3.14	7.25
103	-23.92	6.6	-24.57	6.6	146	0	6.6	0	7.25
104	-23.92	3.46	-24.57	3.46	147	-3.14	6.6	-3.14	7.25
105	-23.92	0.33	-24.57	0.33	148	3.14	0.33	3.14	-0.33
106	-10.07	13.53	-10.72	13.53	149	0	0.33	0	-0.33
107	-10.07	10.39	-10.72	10.39	150	-3.14	0.33	-3.14	-0.33
108	-10.07	7.25	-10.72	7.25	151	10.07	6.6	10.07	7.25
109	-17	13.53	-17.65	13.53	152	6.93	6.6	6.93	7.25
110	-17	10.39	-17.65	10.39	153	3.79	6.6	3.79	7.25
111	-17	7.25	-17.65	7.25	154	10.07	0.33	10.07	-0.33
112	-10.07	20.46	-10.72	20.46	155	6.93	0.33	6.93	-0.33
113	-10.07	17.32	-10.72	17.32	156	3.79	0.33	3.79	-0.33
114	-10.07	14.18	-10.72	14.18	157	17	6.6	17	7.25
115	-17	20.46	-17.65	20.46	158	13.86	6.6	13.86	7.25
116	-17	17.32	-17.65	17.32	159	10.72	6.6	10.72	7.25
117	-17	14.18	-17.65	14.18	160	17	0.33	17	-0.33
118	-10.07	27.39	-10.72	27.39	161	13.86	0.33	13.86	-0.33
119	-10.07	24.25	-10.72	24.25	162	10.72	0.33	10.72	-0.33
120	-10.07	21.11	-10.72	21.11	163	23.92	6.6	23.92	7.25
121	-10.07	-0.33	-10.72	-0.33	164	20.78	6.6	20.78	7.25
122	-10.07	-3.46	-10.72	-3.46	165	17.65	6.6	17.65	7.25
123	-10.07	-6.6	-10.72	-6.6	166	23.92	0.33	23.92	-0.33
124	-17	-0.33	-17.65	-0.33	167	20.78	0.33	20.78	-0.33
125	-17	-3.46	-17.65	-3.46	168	17.65	0.33	17.65	-0.33
126	-17	-6.6	-17.65	-6.6	169	30.85	0.33	30.85	-0.33
127	-23.92	-0.33	-24.57	-0.33	170	27.71	0.33	27.71	-0.33
128	-23.92	-3.46	-24.57	-3.46	171	24.57	0.33	24.57	-0.33
129	-23.92	-6.6	-24.57	-6.6	172	3.14	13.53	3.14	14.18

1. See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-3 Support Disk Section Locations for Stress Evaluation - BWR - Full Model (Continued)

Section[1]	Point 1		Point 2		Section[1]	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
173	0	13.53	0	14.18	216	17.65	-13.53	17.65	-14.18
174	-3.14	13.53	-3.14	14.18	217	3.14	-20.46	3.14	-21.11
175	10.07	13.53	10.07	14.18	218	0	-20.46	0	-21.11
176	6.93	13.53	6.93	14.18	219	-3.14	-20.46	-3.14	-21.11
177	3.79	13.53	3.79	14.18	220	10.07	-20.46	10.07	-21.11
178	17	13.53	17	14.18	221	6.93	-20.46	6.93	-21.11
179	13.86	13.53	13.86	14.18	222	3.79	-20.46	3.79	-21.11
180	10.72	13.53	10.72	14.18	223	17	-20.46	17	-21.11
181	23.92	13.53	23.92	14.18	224	13.86	-20.46	13.86	-21.11
182	20.78	13.53	20.78	14.18	225	10.72	-20.46	10.72	-21.11
183	17.65	13.53	17.65	14.18	226	-3.79	6.6	-3.79	7.25
184	3.14	20.46	3.14	21.11	227	-6.93	6.6	-6.93	7.25
185	0	20.46	0	21.11	228	-10.07	6.6	-10.07	7.25
186	-3.14	20.46	-3.14	21.11	229	-3.79	0.33	-3.79	-0.33
187	10.07	20.46	10.07	21.11	230	-6.93	0.33	-6.93	-0.33
188	6.93	20.46	6.93	21.11	231	-10.07	0.33	-10.07	-0.33
189	3.79	20.46	3.79	21.11	232	-10.72	6.6	-10.72	7.25
190	17	20.46	17	21.11	233	-13.86	6.6	-13.86	7.25
191	13.86	20.46	13.86	21.11	234	-17	6.6	-17	7.25
192	10.72	20.46	10.72	21.11	235	-10.72	0.33	-10.72	-0.33
193	3.14	-6.6	3.14	-7.25	236	-13.86	0.33	-13.86	-0.33
194	0	-6.6	0	-7.25	237	-17	0.33	-17	-0.33
195	-3.14	-6.6	-3.14	-7.25	238	-17.65	6.6	-17.65	7.25
196	10.07	-6.6	10.07	-7.25	239	-20.78	6.6	-20.78	7.25
197	6.93	-6.6	6.93	-7.25	240	-23.92	6.6	-23.92	7.25
198	3.79	-6.6	3.79	-7.25	241	-17.65	0.33	-17.65	-0.33
199	17	-6.6	17	-7.25	242	-20.78	0.33	-20.78	-0.33
200	13.86	-6.6	13.86	-7.25	243	-23.92	0.33	-23.92	-0.33
201	10.72	-6.6	10.72	-7.25	244	-24.57	0.33	-24.57	-0.33
202	23.92	-6.6	23.92	-7.25	245	-27.71	0.33	-27.71	-0.33
203	20.78	-6.6	20.78	-7.25	246	-30.85	0.33	-30.85	-0.33
204	17.65	-6.6	17.65	-7.25	247	-3.79	13.53	-3.79	14.18
205	3.14	-13.53	3.14	-14.18	248	-6.93	13.53	-6.93	14.18
206	0	-13.53	0	-14.18	249	-10.07	13.53	-10.07	14.18
207	-3.14	-13.53	-3.14	-14.18	250	-10.72	13.53	-10.72	14.18
208	10.07	-13.53	10.07	-14.18	251	-13.86	13.53	-13.86	14.18
209	6.93	-13.53	6.93	-14.18	252	-17	13.53	-17	14.18
210	3.79	-13.53	3.79	-14.18	253	-17.65	13.53	-17.65	14.18
211	17	-13.53	17	-14.18	254	-20.78	13.53	-20.78	14.18
212	13.86	-13.53	13.86	-14.18	255	-23.92	13.53	-23.92	14.18
213	10.72	-13.53	10.72	-14.18	256	-3.79	20.46	-3.79	21.11
214	23.92	-13.53	23.92	-14.18	257	-6.93	20.46	-6.93	21.11
215	20.78	-13.53	20.78	-14.18	258	-10.07	20.46	-10.07	21.11

1. See Figure 11.2.12.4.2-4 for section locations.

Table 11.12.12.4.2-3 Support Disk Section Locations for Stress Evaluation - BWR - Full Model (Continued)

Section[1]	Point 1		Point 2		Section[1]	Point 1		Point 2	
	X	Y	X	Y		X	Y	X	Y
259	-10.72	20.46	-10.72	21.11	289	3.14	27.39	3.14	32.63
260	-13.86	20.46	-13.86	21.11	290	3.79	27.39	3.79	32.56
261	-17	20.46	-17	21.11	291	10.07	27.39	10.07	31.2
262	-3.79	-6.6	-3.79	-7.25	292	10.72	27.39	10.72	30.98
263	-6.93	-6.6	-6.93	-7.25	293	17	27.39	17.29	27.86
264	-10.07	-6.6	-10.07	-7.25	294	30.85	-0.33	32.78	-0.33
265	-10.72	-6.6	-10.72	-7.25	295	30.85	-6.6	32.06	-6.86
266	-13.86	-6.6	-13.86	-7.25	296	-3.14	-27.39	-3.14	-32.63
267	-17	-6.6	-17	-7.25	297	3.14	-27.39	3.14	-32.63
268	-17.65	-6.6	-17.65	-7.25	298	3.79	-27.39	3.79	-32.56
269	-20.78	-6.6	-20.78	-7.25	299	10.07	-27.39	10.07	-31.2
270	-23.92	-6.6	-23.92	-7.25	300	10.72	-27.39	10.72	-30.98
271	-3.79	-13.53	-3.79	-14.18	301	17	-27.39	17.29	-27.86
272	-6.93	-13.53	-6.93	-14.18	302	-30.85	6.6	-32.06	6.86
273	-10.07	-13.53	-10.07	-14.18	303	-30.85	0.33	-32.78	0.33
274	-10.72	-13.53	-10.72	-14.18	304	-10.07	27.39	-10.07	31.2
275	-13.86	-13.53	-13.86	-14.18	305	-3.79	27.39	-3.79	32.56
276	-17	-13.53	-17	-14.18	306	-17	27.39	-17.29	27.86
277	-17.65	-13.53	-17.65	-14.18	307	-10.72	27.39	-10.72	30.98
278	-20.78	-13.53	-20.78	-14.18	308	-30.85	-0.33	-32.78	-0.33
279	-23.92	-13.53	-23.92	-14.18	309	-30.85	-6.6	-32.06	-6.86
280	-3.79	-20.46	-3.79	-21.11	310	-10.07	-27.39	-10.07	-31.2
281	-6.93	-20.46	-6.93	-21.11	311	-3.79	-27.39	-3.79	-32.56
282	-10.07	-20.46	-10.07	-21.11	312	-17	-27.39	-17.29	-27.86
283	-10.72	-20.46	-10.72	-21.11	313	-10.72	-27.39	-10.72	-30.98
284	-13.86	-20.46	-13.86	-21.11	314	23.92	20.46	24.92	21.31
285	-17	-20.46	-17	-21.11	315	23.92	-20.46	24.92	-21.31
286	30.85	6.6	32.06	6.86	316	-23.92	20.46	-24.92	21.31
287	30.85	0.33	32.78	0.33	317	-23.92	-20.46	-24.92	-21.31
288	-3.14	27.39	-3.14	32.63					

1. See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-4 Summary of Maximum Stresses for BWR Support Disk for Tip-Over Condition

Drop Orientation	P_m			$P_m + P_b$		
	Stress Intensity (ksi)	Allowable Stress (ksi)	Margin of Safety	Stress Intensity (ksi)	Allowable Stress (ksi)	Margin of Safety
0°	35.1	63.0	+0.80	46.1	90.0	+0.95
31.82°	25.8	63.0	+1.44	65.7	90.0	+0.37
49.46°	23.7	63.0	+1.65	55.5	90.0	+0.62
77.92°	47.5	63.0	+0.33	86.6	90.0	+0.04
90°	58.4	63.0	+0.08	69.6	90.0	+0.29

Note: See Figure 11.2.12.4.2-1 for Drop Orientation.

Table 11.2.12.4.2-5 Summary of Buckling Evaluation of BWR Support Disk for Tip-Over Condition

Drop orientation	MS1	MS2
0°	1.17	1.03
31.82°	0.56	0.53
49.46°	0.86	0.81
77.92°	0.18	0.16
90°	0.38	0.58

Table 11.2.12.4.2-6 Support Disk Primary Membrane (P_m) Stresses for Tip-Over Condition – BWR Disk No. 5 - 77.92° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
202	-24.9	22.5	1	47.5	63.0	0.33
199	-21.8	14.8	1.3	36.6	63.0	0.72
196	-18.8	12.5	1.3	31.4	63.0	1.01
193	-16	11.2	1.3	27.2	62.8	1.30
63	-18.3	8.5	2.4	27.2	63.0	1.32
203	-24.9	-0.1	0.8	24.9	63.0	1.53
204	-24.8	-16.1	0.7	24.9	63.0	1.53
262	-13.2	10.3	1.3	23.7	62.8	1.65
201	-21.7	-16	1	21.9	63.0	1.88
200	-21.7	0	1.1	21.8	63.0	1.89
73	-18.6	2.1	-0.6	20.8	63.0	2.03
265	-10.6	9.8	1.2	20.6	63.0	2.06
166	-12.3	7.9	1.6	20.4	63.0	2.09
169	-13.9	-19.2	2.3	20.0	63.0	2.15
198	-18.7	-15.1	1	19.0	62.8	2.31
197	-18.8	0	1.1	18.9	63.0	2.34
295	-6	-15.6	-6.3	18.7	63.0	2.37
15	-9.1	8.2	2.5	18.0	63.0	2.50
268	-8.1	9.7	0.9	17.8	63.0	2.53
195	-15.9	-14.2	1	16.3	62.8	2.85
194	-15.9	0	1.1	16.1	62.8	2.91
211	-12.2	3.6	0.6	15.8	63.0	2.98
60	-12.3	2.7	2.5	15.8	63.0	2.99
61	-6.8	8.5	1	15.5	63.0	3.06
160	-10.7	4.2	1.9	15.4	63.0	3.10
171	-13.8	0.8	2	15.2	63.0	3.15
70	-14.6	0.2	-0.3	14.9	63.0	3.24
170	-13.9	0	2.1	14.5	63.0	3.34
264	-13.2	-13.2	1	14.1	63.0	3.46
13	-5.7	8.2	1	14.1	63.0	3.48

See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-7 Support Disk Primary Membrane + Primary Bending (P_m+P_b) Stresses for Tip-Over Condition - BWR Disk No. 5 - 77.92° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
169	-85.6	-34.9	7.1	86.6	90.0	0.04
202	-50.9	15.4	-2.3	66.5	90.0	0.35
63	1.2	63.9	-1.5	63.9	90.0	0.41
160	-61.6	-14.9	1.5	61.7	90.0	0.46
171	-60	-17.6	3	60.2	90.0	0.49
60	3.8	59.5	0.4	59.5	90.0	0.51
57	4.8	59.1	0.1	59.1	90.0	0.52
15	10.2	58.9	1.1	59.0	90.0	0.53
51	-28.2	-57	4.7	57.7	89.5	0.55
154	-57.6	-16.5	1.6	57.7	89.8	0.56
199	-54.3	3	-1.4	57.3	90.0	0.57
162	-56.8	-22.8	3.4	57.1	89.9	0.57
54	-26	-55.3	4.3	55.9	89.5	0.60
156	-54.4	-22.8	3.3	54.8	87.8	0.60
148	-54.3	-16.2	1.5	54.4	87.6	0.61
9	14.6	54.1	1.5	54.1	89.8	0.66
166	-54.1	-9.7	0.5	54.1	90.0	0.66
3	-25.2	-52.1	3.5	52.6	87.6	0.67
13	3.7	53.7	1.1	53.7	90.0	0.68
12	15.2	53.5	2.1	53.6	90.0	0.68
123	-23.9	-52.9	3.9	53.4	90.0	0.69
150	-51.3	-22.4	3.2	51.7	87.6	0.69
6	-23.6	-51.1	3.3	51.5	87.6	0.70
229	-51.1	-15.6	1.3	51.2	87.8	0.71
201	-50.2	-27.9	6.7	52.0	90.0	0.73
196	-51.2	-0.2	-1	51.3	90.0	0.76
168	-50.4	-19.2	2.9	50.7	90.0	0.78
198	-48.4	-27.4	6.3	50.1	89.5	0.79
99	-22.1	-49.4	3.1	49.7	89.8	0.81
231	-48.5	-21.6	3	48.8	89.8	0.84

See Figure 11.2.12.4.2-4 for section locations.

Table 11.2.12.4.2-8 Summary of Support Disk Buckling Evaluation for Tip-Over Condition - BWR Disk No. 5 - 77.92° Drop Orientation

Section Number	P (kip)	Pcr (kip)	Py (kip)	M (in-kip)	Mp (in-kip)	Mm (in-kip)	MS1	MS2
169	5.65	31.59	25.67	3.15	4.17	4.11	0.18	0.16
199	8.84	31.4	25.52	1.43	4.15	4.09	0.69	0.57
171	5.62	31.52	25.62	2.03	4.16	4.1	0.64	0.58
160	4.34	31.35	25.48	2.24	4.14	4.08	0.63	0.59
202	10.12	31.55	25.64	1.14	4.17	4.11	0.76	0.59
201	8.82	31.23	25.38	1.25	4.12	4.07	0.80	0.65
196	7.63	31.22	25.37	1.43	4.12	4.07	0.81	0.68
162	4.32	31.1	25.28	2.03	4.11	4.05	0.74	0.70
154	3.7	31.07	25.26	2.14	4.1	4.05	0.74	0.70
204	10.09	31.41	25.53	0.88	4.15	4.09	0.95	0.74
198	7.61	30.97	25.18	1.31	4.09	4.04	0.89	0.75
156	3.67	30.35	24.73	2	4.02	3.97	0.80	0.75
166	4.98	31.51	25.61	1.84	4.16	4.1	0.82	0.76
148	3.05	30.27	24.67	2.06	4.01	3.96	0.82	0.79
193	6.48	30.96	25.18	1.41	4.09	4.04	0.94	0.82
168	4.96	31.36	25.49	1.68	4.14	4.08	0.94	0.86
150	3.02	30.27	24.67	1.93	4.01	3.96	0.92	0.88
51	0.11	30.96	25.18	2.5	4.09	4.04	0.89	0.92
195	6.46	30.96	25.18	1.3	4.09	4.04	1.04	0.90
229	2.39	30.35	24.73	1.99	4.02	3.97	0.96	0.94
54	0.26	30.96	25.18	2.4	4.09	4.04	0.94	0.97
262	5.37	30.97	25.18	1.39	4.09	4.04	1.11	0.99
123	0.25	31.22	25.37	2.3	4.12	4.07	1.04	1.07
6	0.14	30.27	24.67	2.24	4.01	3.96	1.06	1.09
231	2.36	31.07	25.26	1.88	4.1	4.05	1.11	1.08
264	5.35	31.22	25.37	1.29	4.12	4.07	1.23	1.10
99	0.15	31.07	25.26	2.16	4.1	4.05	1.18	1.22
235	1.73	31.1	25.28	1.87	4.11	4.05	1.21	1.20
265	4.31	31.23	25.38	1.32	4.12	4.07	1.38	1.27
237	1.7	31.35	25.48	1.82	4.14	4.08	1.29	1.28

See Figure 11.2.12.4.2-4 for section locations.

11.2.12.5 Corrective Actions

The most important recovery action required following a concrete cask tip-over is the uprighting of the cask to minimize the dose rate from the exposed bottom end. The uprighting operation will require a heavy lift capability and rigging expertise. The concrete cask must be returned to the vertical position by rotation around a convenient bottom edge, and by using a method and rigging that controls the rotation to the vertical position.

Surface and top and bottom edges of the concrete cask are expected to exhibit cracking and possibly loss of concrete down to the layer of reinforcing bar. If only minor damage occurs, the concrete may be repairable by using grout. Otherwise, it may be necessary to remove the canister for installation in a new concrete cask. If the canister remains in the cask, it should be returned to its centered storage position within the cask.

The storage pad must be repaired to preclude the intrusion of water that could cause further deterioration of the pad in freeze-thaw cycles.

11.2.12.6 Radiological Impact

There is an adverse radiological consequence in the hypothetical tip-over event since the bottom end of the concrete cask and the canister have significantly less shielding than the sides and tops of these same components. The dose rate at 1 meter is calculated, using a 1–D analysis, to be approximately 34 rem/hour, and the dose at 4 meters is estimated to be approximately 4 rem/hour. Consequently, following a tip-over event, supplemental shielding should be used until the concrete cask can be uprighted. Stringent access controls must be applied to ensure that personnel do not enter the area of radiation shine from the exposed bottom of the tipped-over concrete cask.

Damage to the edges or surface of the concrete cask may occur following a tip-over, which could result in marginally higher dose rates at the bottom edge or at surface cracks in the concrete. This increased dose rate is not expected to be significant, and would be dependent on the specific damage incurred.

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11.2.13 Full Blockage of Vertical Concrete Cask Air Inlets and Outlets

This section evaluates the Vertical Concrete Cask for the steady state effects of full blockage of the air inlets and outlets at the normal ambient temperature (76°F). It estimates the duration of the event that results in the fuel cladding, the fuel basket and the concrete reaching their design basis limiting temperatures (See Table 4.1-3 for the allowable temperatures for short term conditions).

The evaluation demonstrates that there are no adverse consequences due to this accident, provided that debris is cleared within 24 hours.

11.2.13.1 Cause of Full Blockage

The likely cause of complete cask air inlet and outlet blockage is the covering of the cask with earth in a catastrophic event that is significantly greater than the design basis earthquake or a land slide. This event is a bounding condition accident and is not credible.

11.2.13.2 Detection of Full Blockage

Blockage of the cask air inlets and outlets will be visually detected during the general site inspection following an earthquake, land slide, or other events with a potential for such blockage.

11.2.13.3 Analysis of Full Blockage

The accident temperature conditions are evaluated using the thermal models described in Section 4.4.1. The analysis assumes initial normal storage conditions, with the sudden loss of convective cooling of the canister. Heat is then rejected from the canister to the Vertical Concrete Cask liner by radiation and conduction. The loss of convective cooling results in the fairly rapid and sustained heat-up of the canister and the concrete cask. To account for the loss of convective cooling in the ANSYS air flow model (Section 4.4.1.1), the elements in the model are replaced with thermal conduction elements. This model is used to evaluate the thermal transient resulting from the postulated boundary conditions. The analysis indicates that the maximum basket temperature (support disk and heat transfer disk) remain less than the allowable temperature for 24 hours after the initiation of the event. The maximum fuel cladding temperature and the maximum concrete bulk temperature remain less than the allowable temperatures for about 6 days (150 hours) after the initiation of the event. The heat up of the fuel cladding, canister shell and concrete (bulk temperature) are shown in Figures 11.2.13-1 and 11.2.13-2, for the PWR and BWR configurations, respectively.

11.2.13.4 Corrective Actions

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

11.2.13.5 Radiological Impact

There are no significant radiological consequences for this event, as the Vertical Concrete Cask retains its shielding performance. Dose is incurred as a consequence of uncovering the concrete cask and vent system. Since the dose rates at the air inlets and outlets are higher than the nominal rate (35 mrem/hr) at the cask wall, personnel will be subject to an estimated maximum dose rate of 100 mrem/hr when clearing the inlets and outlets. If it is assumed that a worker kneeling with his hands on the inlets or outlets requires 15 minutes to clear each inlet or outlet, the estimated extremity dose is 200 mrem for the 8 openings. The whole body dose will be slightly less. In addition, some dose is incurred clearing debris away from the cask body. This dose is estimated at 50 mrem, assuming 2 hours is spent near the cask exterior surface.

Figure 11.2.13-1 PWR Configuration Temperature History—All Vents Blocked

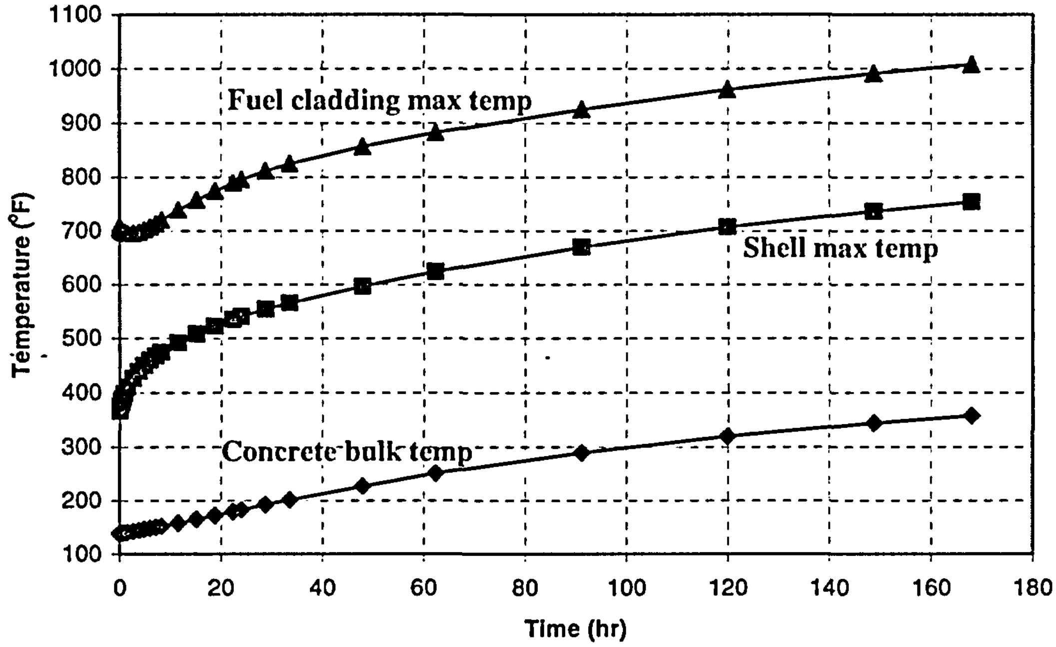
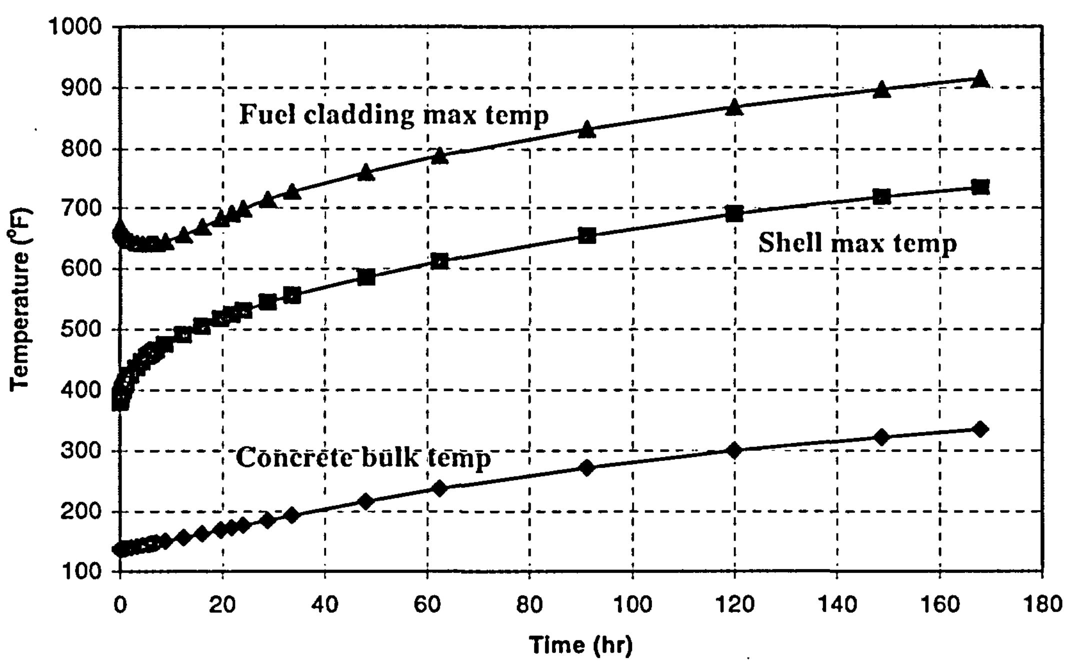


Figure 11.2.13-2 BWR Configuration Temperature History—All Vents Blocked



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11.2.14 Canister Closure Weld Evaluation

The closure weld for the canister is a groove weld with a thickness of 0.75 inches. The evaluation of this weld, in accordance with NRC guidance, is to incorporate a 0.8 stress reduction factor. Applying a factor of 0.8 to the weld stress allowable incorporates the stress reduction factor.

The stresses for the canister are evaluated using sectional stresses as permitted by Subsection NB of the ASME Code. Canister stresses resulting from the concrete cask tip-over accident (Section 11.2.12.4) are used for evaluation. The location of the section for the canister weld evaluation is shown in Figure 11.2.12.4.1-6 and corresponds to Section 11. The governing P_m and $P_m + P_b$ stress intensities for Section 11 and the associated allowables are listed in Tables 11.2.12.4.1-1 and Table 11.2.12.4.1-2, respectively. The factored allowables, incorporating a 0.8 stress reduction factor, and the resulting controlling Margins of Safety are:

Stress Category	Analysis Stress (ksi)	0.8 × Allowable Stress (ksi)	Margin of Safety
P_m	24.80	32.06	0.29
$P_m + P_b$	29.25	48.09	0.64

This confirms that the canister closure weld is acceptable for accident conditions.

Critical Flaw Size for the Canister Closure Weld

The closure weld for the canister is comprised of multiple weld beads using a compatible weld material for Type 304L stainless steel. An allowable (critical) flaw evaluation has been performed to determine the critical flaw size in the weld region. The result of the flaw evaluation is used to define the minimum flaw size, which must be identifiable in the nondestructive examination of the weld. Due to the inherent toughness associated with Type 304L stainless steel, a limit load analysis is used in conjunction with a J-integral/tearing modulus approach. The safety margins used in this evaluation correspond to the stress limits contained in Section XI of the ASME Code.

One of the stress components used in the evaluation for the critical flaw size is the radial stress component in the weld region of the structural lid. For an accident (Level D) event, in accordance with ASME Code Section XI, a safety factor of $\sqrt{2}$ is required. For the purpose of identifying the stress for the flaw evaluation, the weld region corresponds to Section 11 in Figure 11.2.12.4.1-6 is considered.

The maximum tensile radial stress at Section 11 is 6.9 ksi, based on the analysis results of the tip-over accident (Section 11.2.12.4). To perform the flaw evaluation, a 10 ksi stress is conservatively used, resulting in a significantly larger safety factor than the required safety factor of $\sqrt{2}$. Using 10 ksi as the basis for the evaluation, the minimum detectable flaw size is 0.44 inch for a flaw that extends 360 degrees around the circumference of the canister. Stress components for the circumferential and axial directions are also reported in the concrete cask tip-over analysis, which would be associated with flaws oriented in the radial or horizontal directions respectively. The maximum stress for these components is 4.0 ksi, which is also enveloped by the value of 10 ksi used in the critical flaw evaluation for stresses in the radial direction. The 360-degree flaw employed for the circumferential direction is considered to be bounding with respect to any partial flaw in the weld, which could occur in the radial and horizontal directions. Therefore, using a minimum detectable flaw size of 0.375 inch is acceptable, since it is less than the 0.44-inch critical flaw size.

11.2.15 Accident and Natural Phenomena Events Evaluation for Site Specific Spent Fuel

This section presents the accident and natural phenomena 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. Damaged fuel includes fuel rods with cladding that exhibits defects greater than pinhole leaks or hairline cracks.

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.

11.2.15.1 Accident and Natural Phenomena 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, and certain high burnup fuel may require loading in the Maine Yankee Fuel Can. The fuel can is provided in two configurations that differ only in the square cross-section of the can body. In both configurations, the walls of the body of the fuel can are 0.048-inch thick Type 304 stainless steel (18 gauge), have a length of 162.8 inches and have a bottom plate that is 0.63 inch thick. One configuration has a minimum square internal width of 8.52 inches; the second has a minimum square internal width of 8.32 inches.

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 thermal accident events, as presented in Sections 11.2.6, 11.2.7, and 11.2.13.

As shown in Section 3.6.1.1, the total weight of the contents of the Transportable Storage Canister for Maine Yankee fuels is bounded by the total weight for the PWR design basis fuels. However, some design parameters for the Maine Yankee site ISFSI pad are different from those for the design basis ISFSI pad. Therefore, the hypothetical accident (non-mechanistic) tip-over event is evaluated to ensure that the maximum tip-over g-load remains below the bounding g-load (40g) used in the evaluation of the PWR canister and basket in Section 11.2.12.4. The evaluation of the UMS® Vertical Concrete Cask tip-over event on the Maine Yankee site ISFSI pad is presented in Section 11.2.15.1.1. The methodology used is similar to that used in Section 11.2.12.3.1.

Although the total weight, and the maximum g-load, for the Maine Yankee fuel is bounded by the PWR design basis fuels, the maximum weight of the consolidated fuel lattices (2,100 lbs) is larger than that of a single PWR Class 1 design basis fuel assembly (1,567 lbs). This additional weight need only be considered in the support disk evaluation for a side impact condition, similar to the analysis presented in Section 11.2.12.4.1. A parametric study is presented in Section 11.2.15.1.2 to demonstrate that the maximum stress in the support disk due to the consolidated fuel lattice remains bounded by the maximum stress for the support disk for the PWR design basis fuels for a side impact condition.

Section 11.2.15.1.3 provides the structural evaluation for the Maine Yankee fuel can for the 24-inch drop (Section 11.2.4) and the tip-over (Section 11.2.12) accident events.

A Maine Yankee site earthquake evaluation is presented in Section 11.2.15.1.4 to demonstrate the stability of the Vertical Concrete Cask on the Maine Yankee site ISFSI pad.

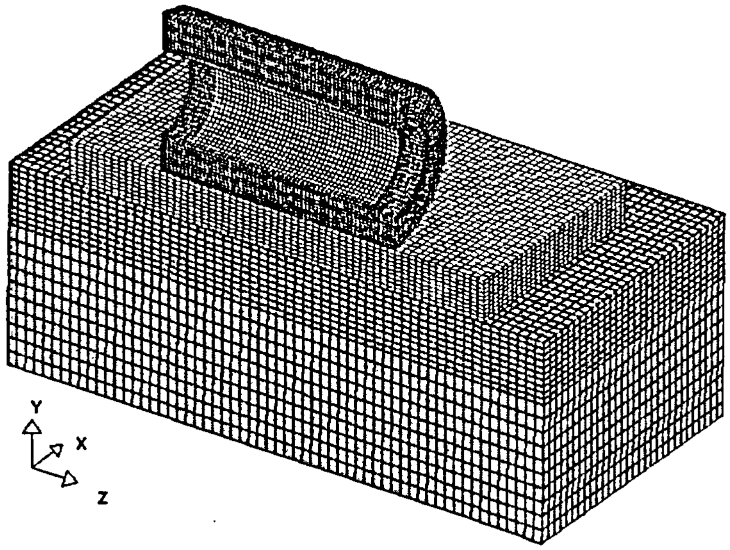
11.2.15.1.1 Maine Yankee Vertical Concrete Cask Tip-Over Analysis

This section evaluates the maximum acceleration of the Transportable Storage Canister and basket during the Vertical Concrete Cask tip-over event on the Maine Yankee site ISFSI pad. This evaluation applies the methodology of Section 11.2.12 for the design basis cask tip-over evaluation.

A finite element model is generated using the LS-DYNA program to determine the acceleration of the vertical concrete cask during the tip-over event.

The concrete pad in the model corresponds to a pad 31-feet by 31-feet square and 3-feet thick, supporting one concrete cask in the center of the pad. The soil under the concrete pad is considered to be 40-feet by 40-feet square and made up of two layers: a 4.5-foot thick upper layer and a 10-foot thick lower layer. Only one-half of the concrete cask, pad and soil configuration is modeled due to symmetry. Both the Class 1 and Class 2 UMS® configurations are evaluated.

The model includes a half section of the concrete cask, the concrete ISFSI pad and soil subgrade, as shown:



Concrete Pad Properties

Vertical concrete cask tip-over analyses are performed for ISFSI pad concrete compressive strengths of 3,000 and 4,000 psi. The Poisson's Ratio (ν_c) is 0.22. The concrete dry density is considered to be between 135 pcf and 145 pcf. To account for the weight of reinforcing bar in the pad, three values of Density (ρ) are used in the model:

ρ (lbs/ft^3)	E_c (psi)	K_c (psi)
140	2.994×10^6	1.782×10^6
145	3.156×10^6	1.879×10^6
152	3.387×10^6	2.016×10^6

The corresponding values of Modulus of Elasticity (E_c) and Bulk Modulus (K_c) are also provided, where:

Modulus of Elasticity (E_c) $= 33\rho_c^{1.5}\sqrt{f_c'}$ (ACI 318-95)

Bulk Modulus (K_c) $= \dfrac{E_c}{3(1-2\nu_c)}$ (Blevins [19])

Soil Properties

The soil properties used in the model are based on three soil sets. The vertical concrete cask tip-over analyses are performed for three different combinations of soil densities: (1) 4.5-foot thick upper layer density of 135 pcf (Modulus of Elasticity, E = 162,070 psi), with a 10-foot thick lower layer density of 127 pcf (E = 31,900 psi); (2) 4.5-foot thick upper layer density of 130 pcf, with a 10-foot thick lower layer density of 127 pcf; and (3) 15-foot depth with density of 145 pcf (E ≤ 60,000 psi). The Poisson's Ratio (ν_s) of the soil is 0.45.

Summary of Design Basis ISFSI Pad Parameters

The ISFSI pads and foundation shall include the following characteristics as applicable to the end drop and tip-over analyses:

Concrete thickness	36 inches maximum
Pad subsoil thickness	15 feet minimum
Specified concrete compressive strength	≤ 4,000 psi at 28 days
Soil in place density (ρ)	$\rho \leq 145$ lbs/ft^3 (upper layer)
Concrete dry density (ρ)	$135 \leq \rho \leq 145$ lbs/ft^3
Soil Modulus of Elasticity	≤ 60,000 psi

The concrete pad maximum thickness excludes the ISFSI pad footer. The compressive strength of the concrete is determined in accordance with Section 5.6 of ACI-318 with concrete acceptance in accordance with the same section. Steel reinforcement is used in the pad and footer. The soil modulus of elasticity is determined according to the test method described in ASTM D4719.

Vertical Concrete Cask Properties

The material properties used in the model for the Vertical Concrete Cask are the same as the properties used in the PWR models in Section 11.2.12.3. The tip-over impact is simulated by applying an initial angular velocity of 1.485 rad/sec (PWR Class 1) and 1.483 rad/sec (PWR Class 2), respectively, to the entire cask. The angular velocity values are determined by the method used in Section 11.2.12 based on the weight of the loaded concrete cask with Maine Yankee fuel (285,513 pounds and 297,509 pounds for PWR Class 1 and PWR Class 2, respectively).

A cut-off frequency of 210 Hz (PWR Class 1) and 190 Hz (PWR Class 2) is applied to filter the analysis results from the LS-DYNA models and determine the peak accelerations. The resulting calculated accelerations on the canister at the location of the top support disk and of the top of the structural lid are tabulated for all of the analysis cases that were run. The maximum accelerations at the two key locations on the canister for the PWR Class 1 and Class 2 configurations are:

Component Location	Position Measured from the Bottom of the Concrete Cask (inches)		Acceleration (g)	
	Class 1	Class 2	Class 1	Class 2
Top Support Disk	176.7	185.2	32.3	34.2
Top of the Canister Structural Lid	197.9	207.0	35.3	37.6

The impact accelerations for the vertical concrete cask tip-over on the Maine Yankee ISFSI pad site are observed to be slightly higher than those reported in Section 11.2.12.3.1 for the design-basis ISFSI pad. Therefore, peak accelerations are calculated for the top support disk and are evaluated with respect to the analysis presented in Section 11.2.12.4.1.

To determine the effect of the rapid application of the inertia loading for the support disk, a dynamic load factor (DLF) is computed using the method presented in Section 11.2.12.4. The DLF is computed to be 1.07 and 1.02 for PWR Class 1 and Class 2, respectively. Applying the DLFs to the 32.3g and 35.4g results in peak accelerations of 34.6g and 36.1g for the top support disk PWR Class 1 and Class 2, respectively. The DLFs for the canister lids are considered to be unity since the lids have significant in-plane stiffness and are considered to be rigid. Additional sensitivity evaluations considering varying values of the ISFSI concrete pad density have been performed. The results of those evaluations demonstrate that the maximum acceleration for the canister and basket are below 40g. Therefore, the maximum acceleration for the canister and basket for the cask tipover accident on the Maine Yankee site ISFSI pad is bounded by the 40g used in Section 11.2.12.4.1 (analysis of canister and basket for PWR configurations for tip-over event).

11.2.15.1.2 Parametric Study of Support Disk Evaluation for Maine Yankee Consolidated Fuel

A parametric study is performed to show that the PWR basket loaded with a Maine Yankee consolidated fuel lattice is bounded by the PWR basket design basis loading for a side impact condition. Only one consolidated fuel lattice, in a Maine Yankee Fuel Can, will be loaded in any single Transportable Storage Canister. However, Maine Yankee Fuel Cans holding other intact or damaged fuel can be loaded in the other three corner positions of the basket. (Maine Yankee Fuel

Cans may be loaded only in the four corner positions of the basket. See Figure 11.2.15.1.2-2 for corner positions. Therefore, the bounding case for Maine Yankee is the basket configuration with twenty (20) Maine Yankee fuel assemblies, three (3) fuel cans containing spent fuel, and one (1) fuel can containing consolidated fuel.

A two-dimensional ANSYS model is employed for the parametric study as shown in Figure 11.2.15.1.2-1. The load from a PWR fuel assembly is modeled as a pressure load at the inner surface of each support disk slot opening. The design basis fuel pressure loading (1g) is 12.26 psi. Based on the same design parameters (slot size = 9.272 in., disk thickness = 0.5 inch, and the number of disks = 30), the pressure load corresponding to a Maine Yankee standard CE 14×14 fuel assembly is 10.3 psi. The pressure load is 11.3 psi for a Maine Yankee fuel can holding an intact or damaged fuel assembly. For a Maine Yankee fuel can holding consolidated fuel the pressure load is 17.0 psi.

This study considers a 60g side impact condition for four different basket orientations: 0°, 18.22°, 26.28° and 45°, as shown in Figure 11.2.15.1.2-2. The 60g bounds the g-load for the PWR support disks (40g) due to the Vertical Concrete Cask tip-over accident as shown in Section 11.2.12.

A total of five cases are considered in the study. Inertial loads are applied to the support disk in all cases. The base case considers that all 24 fuel positions hold design basis PWR fuel assemblies. The other four cases (Cases 1 through 4) represent four possible load combinations for the placement of four Maine Yankee fuel cans in the corner positions, one of which holds consolidated fuel. The remaining twenty basket positions hold Maine Yankee standard 14×14 fuel assemblies. The basket loading positions are shown in Figure 11.2.15.1.2-2. The load combinations evaluated in the four Maine Yankee fuel can loading cases are:

Case	Basket Position 1	Basket Position 2	Basket Position 3	Basket Position 4
1	Consolidated	Damaged	Damaged	Damaged
2	Damaged	Consolidated	Damaged	Damaged
3	Damaged	Damaged	Damaged	Consolidated
4	Damaged	Damaged	Consolidated	Damaged

Table 11.2.15.1.2-1 provides a parametric comparison between the Base Case and the four cases evaluated, based on the maximum sectional stress in the support disk. As shown in the table, the maximum stress in the PWR basket support disk loaded with 20 standard fuel assemblies and four Maine Yankee fuel cans, including one holding consolidated fuel, is bounded by that for the support disk loaded with the design basis PWR fuel.

Additionally, a three-dimensional analysis was performed for Case 4 with a 26.28° drop orientation using the three-dimensional canister/basket model presented in Section 11.2.12.4.1. Results of the analysis for the top support disk, where maximum stress occurs, are presented in Tables 11.2.15.1.2-2 and 11.2.15.1.2-3. The minimum margin of safety is +1.12 and +0.11 for P_m stresses and $P_m + P_b$ stresses, respectively. The minimum margin of safety for the corresponding analysis for the design basis PWR configuration is +0.97 and +0.05 for P_m and $P_m + P_b$ stresses, respectively (see Table 11.2.12.4.1-4). Therefore, it is further demonstrated that the maximum stress in the PWR support disk loaded with Maine Yankee fuel with consolidated fuel is bounded by the stress for the PWR support disk loaded with the design basis PWR fuel.

Since no credit is taken for the structural integrity of the consolidated fuel or damaged fuel inside the fuel can, it is assumed that 100% of the fuel rods fail during an accident. For a Maine Yankee standard 14×14 fuel assembly, the volume of 176 fuel rods (100%) and 5 guide tubes will fill up the lower 103.6 inches (about at the elevation of the 21st support disk) assuming a 50% volume compaction factor. For the consolidated fuel, the volume of 283 rods (100%) and 4 connector rods will fill up the lower 109.6 inches (about at the elevation of the 22nd support disk) assuming a 75% compaction factor. The compaction factor of 75% for the consolidated fuel considers that the number of rods in the consolidated fuel is approximately 1.5 times of the number of rods in the standard Maine Yankee fuel and these rods are initially more closely spaced.

During a tip-over accident of the vertical concrete cask, the maximum total load on the support disk (top/30th disk) for the design basis PWR basket is 54.6 kips (12.26 psi × 9.272-inch × 0.5-inch × 24 × 40g), considering the design deceleration of 40g (Section 11.2.12.4). With the assumption of 100% rod failure for the damaged fuel and consolidated fuel in the Maine Yankee fuel can, the 21st disk is subjected to the maximum total load (including weight from 20 standard fuel assemblies, 3 damaged fuel assemblies and the consolidated fuel). The pressure load (1g) on the support disk corner slot corresponding to 100% failed damaged fuel is 15.3 psi (load distributed to 21 support disks) and the pressure load corresponding to the 100% failed consolidated fuel is 22.6 psi (load distributed on 22 support disks). In the tip-over accident, the g-load at the 21st disk is 30g, based on the design deceleration of 40g at the top (30th) disk. The total load (W_{21}) on the 21st support disk is:

$$W_{21} = (10.3\times20+15.3\times3+22.6\times1)\times 9.272 \times 0.5 \times 30 = 38{,}200 \text{ pounds} = 38.2 \text{ kips}$$

The support disk load is only 70% (38.2/54.6 = 0.7) of the maximum total load on the support disk due to the design basis PWR fuel load. Consequently, the maximum stress in the support disk, assuming 100% rod failure of the damaged and consolidated fuel in Maine Yankee fuel cans, is bounded by the maximum stress in the support disk calculated for the design basis fuel.

Figure 11.2.15.1.2-1 Two-Dimensional Support Disk Model

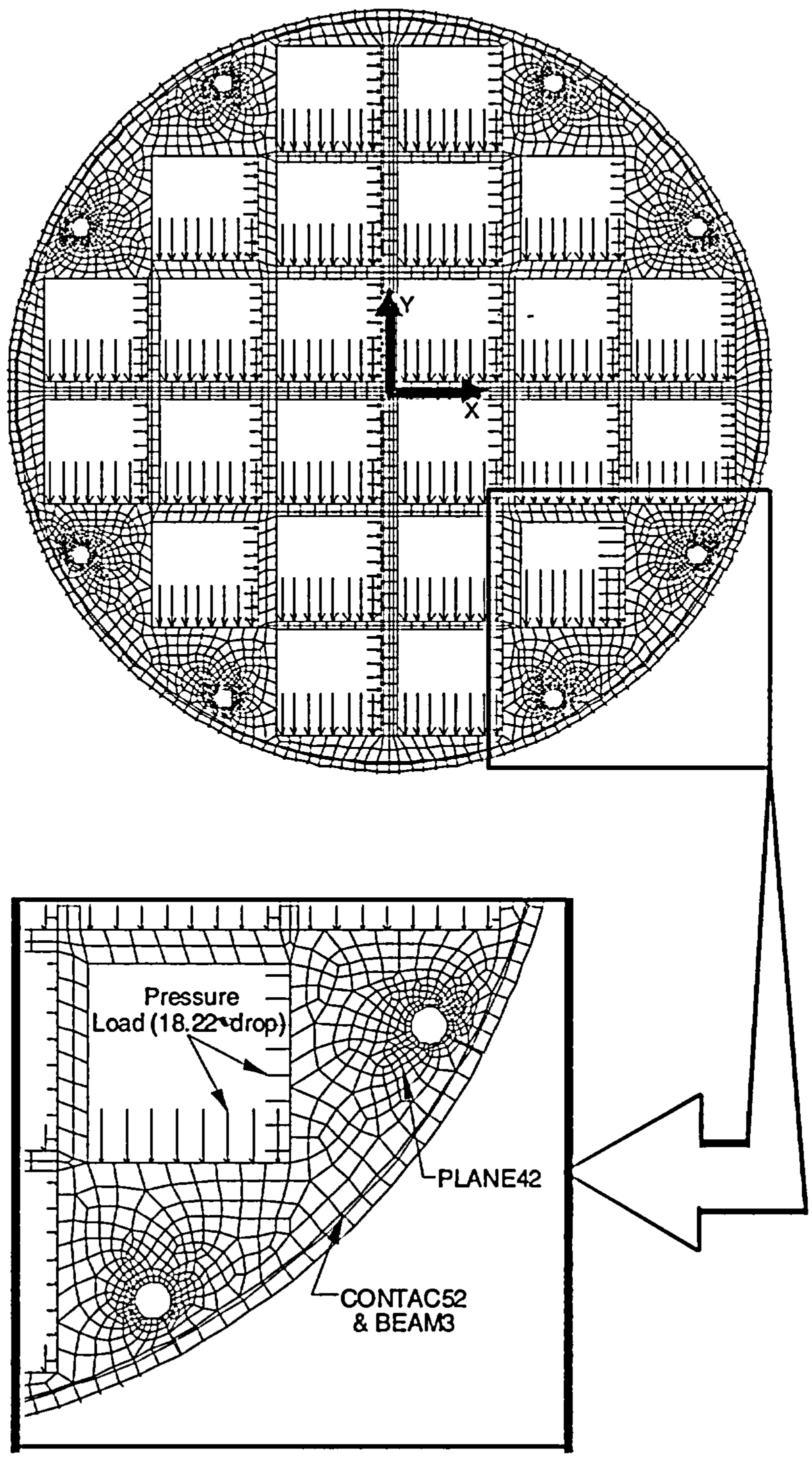


Figure 11.2.15.1.2-2 PWR Basket Impact Orientations and Case Study Loading Positions for Maine Yankee Consolidated Fuel

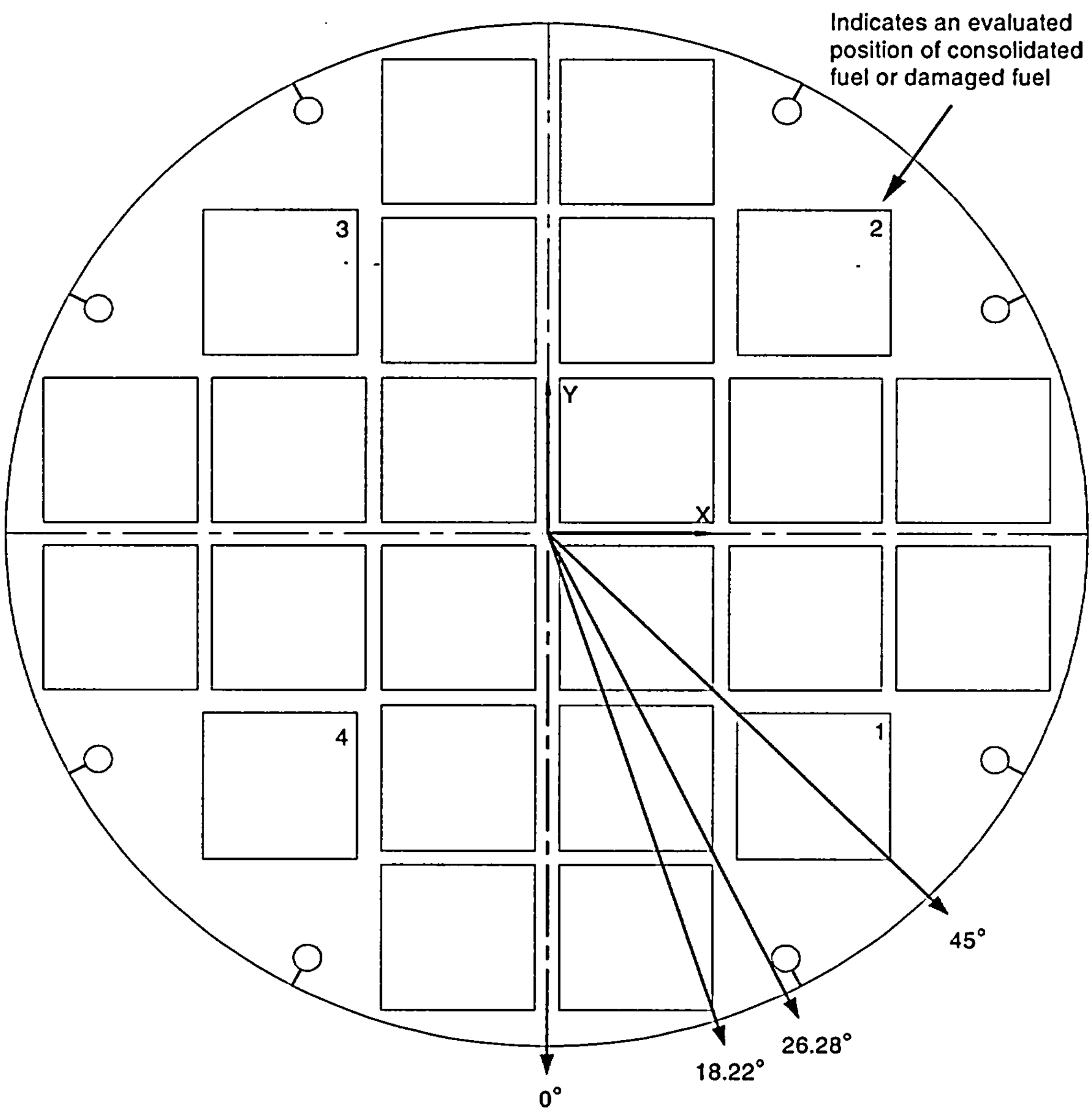


Table 11.2.15.1.2-1 Normalized Stress Ratios – PWR Basket Support Disk Maximum Stresses

Orientation[1]	Membrane Stress Ratio[2]				Membrane + Bending Stress Ratio[2]			
	0°	18.22°	26.28°	45°	0°	18.22°	26.28°	45°
Base Case	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
Case 1	0.91	0.94	0.94	0.94	0.96	0.94	0.94	0.94
Case 2	0.91	0.94	0.94	0.95	0.95	0.95	0.95	0.95
Case 3	0.91	0.95	0.95	0.95	0.96	0.95	0.95	0.95
Case 4	0.91	0.95	0.95	0.96	0.96	0.98	0.98	0.97

1. Orientations correspond to those shown in Figure 11.2.15.1.2-2.
2. Stress ratios are based on the maximum sectional stresses of the support disk.

Table 11.2.15.1.2-2 Support Disk Primary Membrane (P_m) Stresses for Case 4, 26.28° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
18	19.3	-22.9	2.8	42.6	90.4	1.12
3	27.1	-12.2	2.4	39.6	89.3	1.26
16	37.1	-22.8	1	37.2	89.3	1.4
1	32.3	-12.1	0.6	32.3	90.4	1.8
94	26.8	-19	2.7	27.6	90.5	2.28
17	-0.1	-22.8	1.9	23.1	89.8	2.9
88	18.3	-5.6	-7.3	21.6	91.5	3.23
96	6.7	-13.8	-3.2	21.4	91.5	3.27
95	-0.1	-19.9	1.5	20	91.1	3.55
90	15.3	-3.5	0.8	18.9	90.5	3.8
84	15.6	-18.5	-0.4	18.6	91.5	3.93
61	15.7	-10.5	4.7	18.5	91.5	3.96
60	10.2	-17.5	1.3	17.7	89.3	4.03
82	15.7	-7.8	3.8	17.2	90.8	4.27
37	11.9	-4.3	0.6	16.3	89.3	4.49
58	10.3	-12.1	5	16.3	90.4	4.54
62	15.7	-0.2	2.6	16.3	91.2	4.59
83	15.7	-0.2	1.7	15.8	91.2	4.75
91	-7.4	-15.4	-1.5	15.7	90.5	4.78
63	15.6	-9.9	0.5	15.7	90.8	4.8
30	14.1	-9.3	3.1	15.6	91.9	4.89
33	14.6	-4.7	2.3	15.1	89.3	4.93
108	13.5	-5.6	-3.9	15.1	91.5	5.07
24	-2	-14.3	1.7	14.5	91.5	5.31
79	-5.3	6.3	4.1	14.2	89.3	5.31
23	-0.1	-14.2	0.7	14.2	91.2	5.41
22	-7.3	-14.1	-0.4	14.2	90.8	5.42
28	13.2	-9.1	1.8	13.9	90.9	5.56
7	13.6	-11.9	-0.7	13.8	91.5	5.62
46	-2.4	-10.8	5.1	13.2	89.3	5.74

Note: See Figure 11.2.12.4.1-7 for Section locations.

Table 11.2.15.1.2-3 Support Disk Primary Membrane + Primary Bending ($P_m + P_b$) Stresses for Case 4, 26.28° Drop Orientation (ksi)

Section Number	Sx	Sy	Sxy	Stress Intensity	Allowable Stress	Margin of Safety
61	-116.4	-39.3	10.1	117.7	130.8	0.11
58	-109.5	-43.9	8.7	110.6	129.1	0.17
43	-92.6	-32.4	6.2	93.2	129.1	0.39
82	-87.8	-27.9	7	88.6	129.8	0.46
60	-81.6	-39.9	7.7	83	127.6	0.54
79	-82	-18.9	2	82	127.6	0.56
55	-83.5	-29.3	4.6	83.9	130.8	0.56
16	-52.5	-71.9	15	80.1	127.6	0.59
46	-77.1	-49.3	9.5	80	127.6	0.59
64	-76.2	-31.8	7	77.2	127.6	0.65
30	-34.4	-75.2	13.1	79.1	131.3	0.66
18	-2.8	-77.6	-2.9	77.8	129.1	0.66
3	10.1	-65.4	-6	76.5	127.6	0.67
63	-75.4	-26	4.3	75.8	129.8	0.71
76	69	21	4.7	69.5	129.8	0.87
48	-66	-42.7	4	66.7	125.7	0.89
19	-38.2	-65.3	2.6	65.5	125.7	0.92
6	-43.2	-62	5.4	63.4	125.7	0.98
45	-63.2	-15.3	-0.2	63.2	127.6	1.02
94	-56.3	-40.8	10.4	61.5	129.3	1.1
21	-47.1	-57.5	5.3	59.7	127.6	1.14
67	-54.5	-42.3	5.3	56.5	125.7	1.22
1	-47.7	-40.7	12.7	57.3	129.1	1.25
33	-29.7	-52.9	7.4	55	127.6	1.32
51	26.7	-27.3	3.9	54.5	127.7	1.34
39	-29	-49.8	6.3	51.6	129.1	1.5
81	-49.9	-29.5	5.3	51.2	129.1	1.52
84	-48	-26.1	6.2	49.7	130.8	1.63
4	-41.7	-43.6	5.3	48	127.6	1.66
28	-44.6	-29.6	8.3	48.2	129.9	1.69

Note: See Figure 11.2.12.4.1-7 for Section locations.

11.2.15.1.3 Structural Evaluation for the Maine Yankee Fuel Can

Twenty-Four Inch Drop of the Vertical Concrete Cask

The 24-inch drop of the Vertical Concrete Cask onto an unyielding surface (Section 11.2.4) results in accelerations that are bounded by the 60g acceleration used in this structural evaluation for the Maine Yankee Fuel Can. The compressive load (P) on the tube is the combined weight of the lid, side plates and tube body. The Maine Yankee Fuel Can having the smallest internal cross-section (8.32 inches) is used in this analysis. This bounds the condition for the larger fuel can.

The compressive load (P) is:

$$P = (17.89 + 6.57 + 78.77) \times 60 = 6{,}193.8 \text{ lbs, use } 8{,}500 \text{ lbs.}$$

The compressive stress (S_c) in the tube body is:

$$S_c = \frac{P}{A} = \frac{8{,}500}{1.674} = 5{,}078 \text{ psi}$$

The margin of safety (MS) is determined based on the accident condition allowable primary membrane stress (0.7 S_u) at a bounding temperature of 600°F for Type 304 stainless steel:

$$MS = \frac{0.7S_u}{S_c} - 1 = \frac{0.7(63{,}300)}{5{,}078} - 1 = +7.7$$

The potential buckling of the tube is evaluated, using the Euler formula, to determine the critical buckling load (P_{cr}):

$$P_{cr} = \frac{\pi^2 EI}{L_e^2} = \frac{\pi^2 (25.2\times10^6)(19.55)}{2(157.8)} = 48{,}817 \text{ lbs}$$

where:

$E = 25.2\times10^6$ psi

$$I = \frac{8.62^4 - 8.32^4}{12} = 19.55 \text{ in.}^4$$

L_e = 2L (worst case condition)

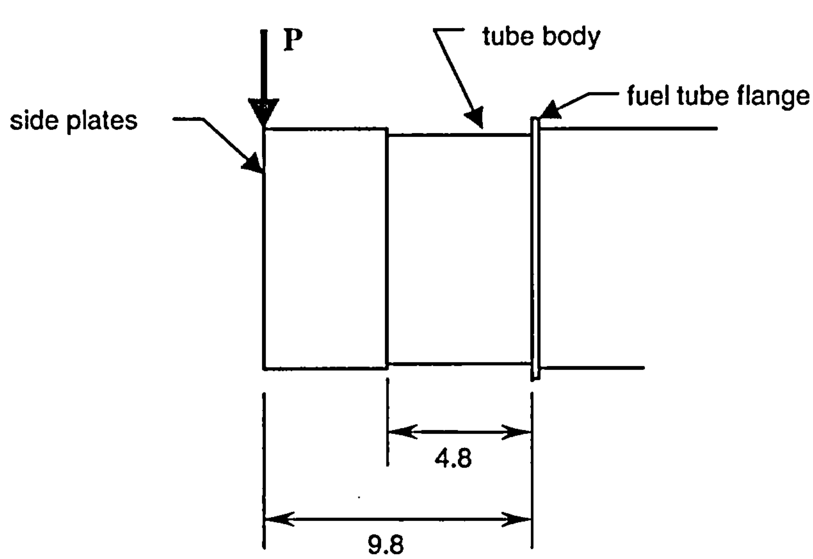
L = tube body length (157.8 in.)

Because the maximum compressive load (8,500 lbs under the accident condition) is much less than the critical buckling load (16.5×10^6 psi) the tube has adequate resistance to buckling.

Tip–Over of the Vertical Concrete Cask

The majority of the fuel can tube body is contained within the fuel tube in the basket assembly. Because both the tube body of the fuel can and the fuel tube have square cross sections, they are effectively in full contact (for 153.0 in. longitudinally) during a side impact and no significant bending stress is introduced into the tube body. The last 4.8 inches of the tube body and the 5.0 inches length of the side plates are unsupported past the fuel tube flange in the side impact orientation.

The tube body is evaluated as a cantilevered beam with the combined weight (P) of the overhanging tube body and side plates and conservatively, concentrated at the top end of the side plates multiplied by a deceleration factor of 60g. Note that the maximum g-load for the PWR basket is 40g for the tip-over accident (Section 11.2.12).



The maximum bending moment (M) is:

$$M = Pg \times L = 35(60)(9.8) = 20{,}581 \text{ lbs·in.}$$

where:

P = 35 lbs (weight of the overhung tube and side plates)

g = 60 (conservative g-load that bounds the tip over condition)

L = 9.8 in. (the total overhung length of the tube body and side plates)

The maximum bending stress, f_b, is:

$$f_b = \frac{Mc}{I} = \frac{20{,}581(4.21)}{19.55} = 4{,}432 \text{ psi}$$

where:

c = half of the outer dimension of the tube

I = the moment of inertia

The shear stress (τ) is:

$$\tau = \frac{Pg}{A} = \frac{35(60)}{1.674} = 1{,}254 \text{ psi}$$

where:

A = the cross-sectional area of the tube = 1.674 in^2

The principal stresses are calculated to be 4,762 psi and -330 psi, and the corresponding stress intensity is determined to be 5,092 psi.

The margin of safety (MS) is calculated based on the allowable primary membrane plus bending stress (1.0 S_u) at a bounding temperature of 600°F for Type 304 stainless steel:

$$MS = \frac{1.0S_u}{\sigma_{max}} - 1 = \frac{63,300 \text{ psi}}{5,092 \text{ psi}} - 1 = +11.4$$

As discussed in Section 11.2.15.1.2, the Maine Yankee fuel can may hold a 100% failed damaged fuel lattice or consolidated fuel lattice. An evaluation is performed to demonstrate that the fuel can maintains its integrity during a tip-over accident for this condition. The fuel can is evaluated using the methodology presented in Section 11.2.12.4.1 for the PWR Fuel Tube Analysis for a 60-g side impact condition. This g-load bounds the maximum g-load (40g) for the PWR basket in the concrete cask tip-over event. Similar to the finite element model used for the PWR fuel tube analysis for the uniform pressure case (see Section 11.2.12.4.1), an ANSYS finite element model is generated to represent a section of the damaged fuel can with a length of three spans, i.e., the model is supported at four locations by the support disks. The fuel tube, the neutron absorber plate, and its stainless steel cover plate are conservatively ignored in the model. A bounding uniform pressure is applied to the lower inside surface of the fuel can wall. The pressure is determined based on the weight of the 100% failed consolidated fuel (2,100 lbs × 60g) occupying a length of 109.6 inches (see Section 11.2.15.1.2) as shown below. The inside dimension of the larger fuel can (8.52-inches) is conservatively used in the analysis, as it bounds the bending stress condition of the fuel can with the smaller cross-section.

$$P = \frac{2,100}{109.6(8.52)} \times 60 = 135 \text{ psi}$$

The finite element analysis results show that the maximum stress in the fuel can is 25.4 ksi, which is local to the sections of the tube resting on the support disks. At 750°F the ultimate strength for Type 304 stainless steel is 63.1 ksi. The Margin of Safety is:

$$MS = \frac{63.1}{25.4} - 1 = +1.48$$

The analysis shows that the maximum total strain is 0.05 inch/inch. Defining the acceptable elastic-plastic response of the stainless steel as one half of the material failure strain of 0.40 in./in. at 750°F, the resulting Margin of Safety is:

$$MS = \frac{0.40/2}{0.05} - 1 = +3.0$$

Similarly, the Margin of Safety for elastic-plastic stress is:

$$MS = \frac{63.1 - 17.3}{25.4 - 17.3} - 1 = +4.65$$

where the yield strength of Type 304 stainless steel is 17.3 ksi at 750°F.

Therefore the Maine Yankee fuel can maintains its integrity for the accident conditions.

11.2.15.1.4 Maine Yankee Site Specific Earthquake Evaluation of the Vertical Concrete Cask

This section provides an evaluation of the response of the vertical concrete cask to an earthquake imparting a horizontal acceleration of 0.38g at the top surface of the concrete pad. The evaluation shows that the loaded or empty vertical concrete cask does not tip over or slide in the earthquake event. The methodology used in this evaluation is identical to that presented in Section 11.2.8.

Tip-Over Evaluation of the Vertical Concrete Cask

To maintain the concrete cask in equilibrium, the restoring moment, M_R must be greater than, or equal to, the overturning moment, M_o (i.e. $M_R \geq M_o$). Based on this premise, the following derivation shows that a 0.38g acceleration of the design basis earthquake at the surface of the concrete pad is well below the acceleration required to tip-over the cask.

The combination of horizontal and vertical acceleration components is based on the 100-40-40 approach of ASCE 4-86 [36], which considers that when the maximum response from one component occurs, the response from the other two components are 40% of the maximum. The vertical component of acceleration is obtained by scaling the corresponding ordinates of the horizontal components by two-thirds.

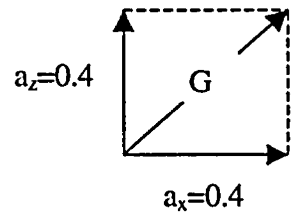
Using this method, two cases are evaluated where:

$a_x = a_z = a$ = horizontal acceleration components
$a_y = (2/3)\, a$ = vertical acceleration component
G_h = Vector sum of two horizontal acceleration components
G_v = Vertical acceleration component

In the first case, the horizontal acceleration is at its maximum. In the second, one horizontal acceleration is at its maximum.

Case 1) The vertical acceleration, a_y, is at its peak: ($a_y = 2/3a$, $a_x = 0.4a$, $a_z = 0.4a$)

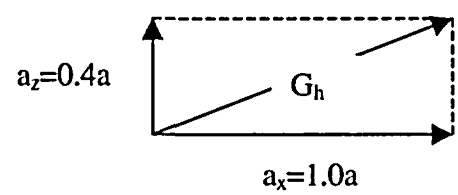
$$G_h = \sqrt{a_x^2 + a_z^2}$$
$$G_h = \sqrt{(0.4\times a)^2 + (0.4\times a)^2} = 0.566\times a$$



$$G_v = 1.0\times a_y = 1.0\times\left(a\times\frac{2}{3}\right) = 0.667\times a$$

Case 2) One horizonal acceleration, a_x, is at its peak: ($a_y = 0.4 \times 2/3a$, $a_x = a$, $a_z = 0.4a$)

$$G_h = \sqrt{a_x^2 + a_z^2}$$
$$G_h = \sqrt{(1.0\times a)^2 + (0.4\times a)^2} = 1.077\times a$$



$$G_v = 0.4\times a_y = 0.4\times\left(a\times\frac{2}{3}\right) = 0.267\times a$$

In order for the cask to resist overturning, the restoring moment, M_R, about the point of rotation, must be greater than the overturning moment, M_o, that:

$$M_R \geq M_o \text{, or}$$
$$F_r \times b \geq F_o \times d \Rightarrow (W\times 1 - W\times G_v)\times b \geq (W\times G_h)\times d$$

where:

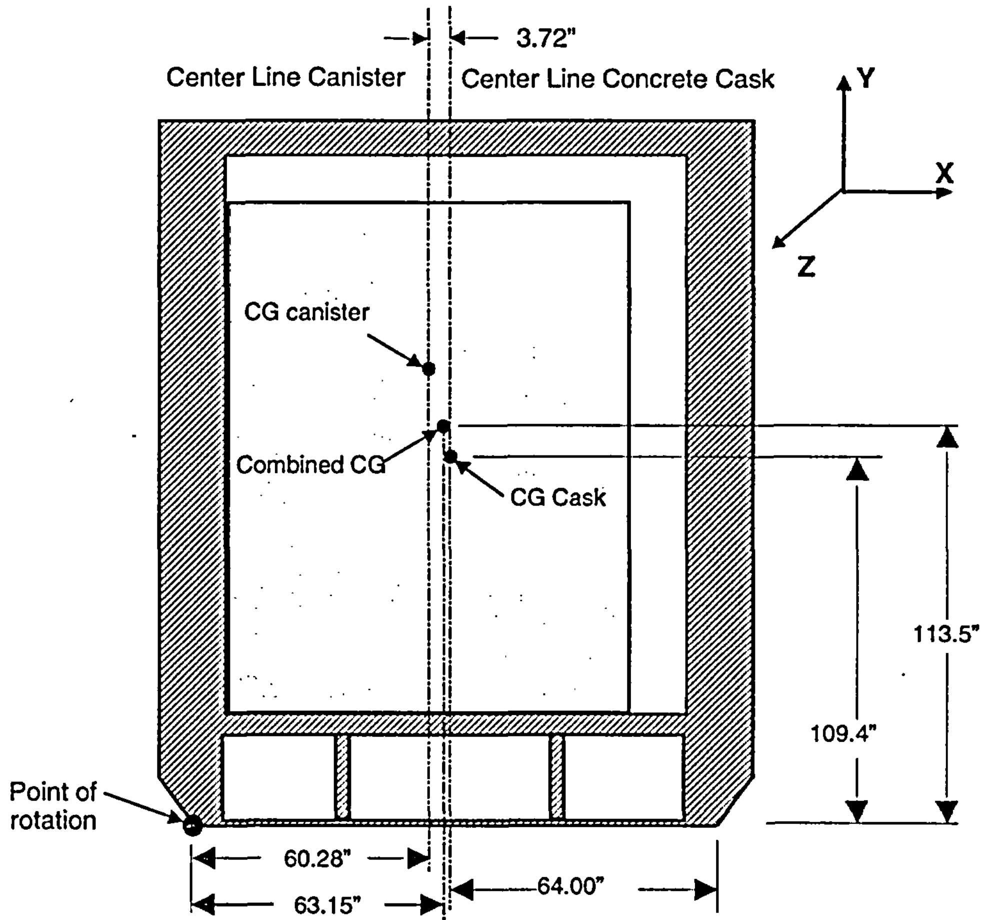
d = vertical distance measured from the base of the Vertical Concrete Cask to the center of gravity

b = horizontal distance measured from the point of rotation to the C.G.

W = the weight of the Vertical Concrete Cask

F_o = overturning force

F_r = restoring force



Substituting for G_h and G_v gives:

Case 1

$$(1 - 0.667a)\ \frac{b}{d} \geq 0.566 \times a$$

$$a \leq \frac{b/d}{0.566 + 0.667\,(b/d)}$$

Case 2

$$(1 - 0.267a)\frac{b}{d} \geq 1.077a$$

$$a \leq \frac{b/d}{1.077 + 0.267(b/d)}$$

Because the canister is not attached to the concrete cask, the combined center of gravity for the concrete cask, with the canister in its maximum off-center position, must be calculated. The point of rotation is established at the outside lower edge of the concrete cask.

The inside diameter of the concrete cask is 74.5 inches and the outside diameter of the canister is 67.06 inches; therefore, the maximum eccentricity between the two is:

$$e = \frac{74.50\,\text{in} - 67.06\,\text{in}}{2} = 3.72\ \text{in.}$$

The horizontal displacement, x, of the combined C.G. due to eccentric placement of the canister is

$$x = \frac{70{,}783(3.72)}{308{,}432} = 0.85\ \text{in}$$

Therefore,

$$b = 64 - 0.85 = 63.15\ \text{in.}$$

and

$$d = 113.5\ \text{in.}$$

The C.G. of the loaded Maine Yankee Vertical Concrete Cask is conservatively assumed to be 113.5 inches, which bounds all of the Maine Yankee UMS® Storage System configurations.

1) $$a \le \frac{63.15/113.5}{0.566 + 0.667 \times \left(63.15/113.5\right)}$$

$a \le 0.59g$

2) $$a \le \frac{63.15/113.5}{1.077 + 0.267 \times \left(63.15/113.5\right)}$$

$a \le 0.45g$

Therefore, the minimum ground acceleration that may cause a tip-over of a loaded concrete cask is 0.45g. Since the 0.38g design basis earthquake ground acceleration for the UMS® System at the Maine Yankee site is less than 0.45g, the storage cask will not tip-over.

The factor of safety is 0.45 / 0.38 = 1.18, which is greater than the required factor of safety of 1.1 in accordance with ANSI/ANS-57.9.

Since an empty vertical concrete cask has a lower C.G. as compared to a loaded concrete cask, the tip-over evaluation for the empty concrete cask is bounded by that for the loaded concrete cask.

Sliding Evaluation of the Vertical Concrete Cask

To keep the cask from sliding on the concrete pad, the force holding the cask (F_s) has to be greater than or equal to the force trying to move the cask.

Based on the equation for static friction:

$$F_s = \mu N \geq G_h W$$
$$\mu(1 - G_v)W \geq G_h W$$

Where:

μ = coefficient of friction
N = the normal force
W = the weight of the concrete cask
G_v = vertical acceleration component
G_h = resultant of horizontal acceleration component

Substituting G_h and G_v for the two cases:

Case 1) $\mu(1 - 0.667a) \geq 0.566\,a$

$$\mu \geq \frac{0.566a}{1 - 0.667a}$$

Case 2) $\mu(1 - 0.267a) \geq 1.077\,a$

$$\mu \geq \frac{1.077a}{1 - 0.267a}$$

For a = 0.38g

Case 1) $\mu \geq 0.29$

Case 2) $\mu \geq 0.45$

The analysis shows that the minimum coefficient of friction, μ, required to prevent sliding of the concrete cask is 0.45. The coefficient of friction between the steel bottom plate of the concrete cask and the concrete surface (broom finish) of the storage pad, 0.50, is greater than the coefficient of friction required to prevent sliding of the concrete cask [45,46]. Therefore, the concrete cask will not slide under design-basis earthquake conditions. The factor of safety is 0.50 / 0.45 =1.11 which is greater than the required factor of safety of 1.1 in accordance with ANSI/ANS-57.9 [1].

11.2.15.1.5 Buckling Evaluation for Maine Yankee High Burnup Fuel Rods

This section presents the buckling evaluation for Maine Yankee high burnup fuel (burnup between 45,000 and 50,000 MWD/MTU) having cladding oxide layers that are 80 and 120 microns thick. A similar evaluation is presented in Section 11.2.15.1.6 for Maine Yankee high burnup fuel with an oxide layer thickness of 80 microns that is also mechanically damaged. These analyses show that the high burnup fuel and the damaged high burnup fuel do not buckle in the design basis accident events. An end drop orientation is considered with an acceleration of 60 g, which subjects the fuel rod to axial loading. A reduced clad thickness is assumed, due to the cladding oxide layer.

In the end drop orientation, the fuel rods are laterally restrained by the grids and may come into contact with the fuel assembly base. The only vertical constraint for the fuel rod is the base of the assembly. The weight of the fuel pellets is included in this evaluation, as the pellets are considered to be vertically supported by the cladding. A two-dimensional model comprised of ANSYS BEAM3 elements, shown in Figure 11.2.15.1.5-1, is used for the evaluation. This evaluation is considered to be the bounding condition (as opposed to an evaluation, which considers the cladding only).

80 Micron Oxide Layer Thickness Evaluation

During the end drop, the fuel rod impacts the fuel assembly base. The fuel rod itself will respond as an elastic bar under a sudden compression load at its bottom end. The duration of this impact is bounded by the first extentional mode shape of the fuel rod. Contribution of higher frequency extentional modes of the rod would tend to shorten the duration of impact of the fuel rod with the fuel assembly base. The fuel rod, upon initiation of impact, corresponds to an undeformed state. In the process of the impact, the compression of the fuel rod will increase to a maximum and then return to a near uncompressed state, at which point the time of impact has been completed. This actually represents half of a cycle of the lowest frequency mode shape of the fuel rod. The shape of the time dependence of the deformation is sinusoidal. The single extentional mode shape can also be considered to be a single degree of freedom with a corresponding mass and stiffness. In viewing such an event as a spring mass system, the time variation of the deformation during the impact is expected to be sinusoidal.

The buckling mode for the fuel rod is governed by the boundary conditions. For this configuration, the grids provide a lateral support, but no vertical support. The only vertical restraint is considered to be at the point of contact of the fuel rod and the base of the assembly. The weight of the fuel rod pellets and cladding is assumed to be uniformly distributed along the length of the fuel rod. In the end drop, this results in the maximum compressive load occurring at the base of the fuel rod. The first buckling mode shape corresponding to these conditions is computed as shown in Figure 11.2.15.1.5-2.

Typically eigenvalue buckling is applied for static environments. For dynamic loading, it is assumed that the duration of the loading is sufficiently long to allow the system to experience the complete load, even as the deformation associated with the buckling is commenced. For dynamic loading, the lateral motion, which would correspond to the buckled shape, will correspond to the lowest mode shape. This lowest frequency mode shape is shown in Figure 11.2.15.1.5-2 and corresponds to a frequency of 25.9 Hz. The similarity of the two shapes shown in Figure 11.2.15.1.5-2 is expected, since both have the same displacement boundary conditions, the same stiffness matrix, and the same governing finite element equations, i.e.,

$$[K]\{\phi_i\} = \lambda_i \, [A]\{\phi_i\}$$

where:

$[K]$ = structure stiffness matrix
$\{\phi_i\}$ = eigenvector
λ_i = eigenvalue
$[A]$ = mass matrix for the mode shape calculation or stress stiffening matrix for the buckling evaluation

Based on the time duration of the impact and the inherent inability of the fuel rod to rapidly displace in the lateral direction, the effect of the actual lateral motion of buckling can be computed with a dynamic load factor (DLF) [47]. The expression for the DLF for a half-sine loading for a single degree of freedom is given by

$$\text{DLF} = \frac{2\beta \cos(\pi/2\beta)}{1 - \beta^2}$$

where:

β = ratio of the first extentional mode frequency to the first lateral mode frequency

These values, computed in this section, are $\beta = 8.32$ and DLF = 0.244.

This DLF is applied to the end drop acceleration of 60g, which is the bounding load to potentially result in the buckling of the fuel rod. The product of 60g × DLF (= 14.6g) is well below the vertical acceleration corresponding to the first buckling mode shape, 37.9g as computed in this section. This indicates that the time duration of the impact of the fuel onto the fuel assembly base is of sufficiently short nature that buckling of the fuel rod cannot occur.

An effective cross-sectional property is used in the model to consider the properties of the fuel pellet and the fuel cladding. The modulus of elasticity (EX) for the fuel pellet has a nominal value of 26.0×10^6 psi [48]. To be conservative, only 50 percent of this value is used in the evaluation. The EX for the fuel pellet was, therefore, taken to be 13.0×10^6 psi. The value of EX (10.47×10^6 psi) was used for the irradiated Zircaloy cladding (ISG-12). Reference information shows that there is no additional reduction of the ductility of the cladding due to extended burnup into the 45,000 – 50,000 MWD/MTU range [49].

The bounding dimensions and physical data (minimum clad thickness, maximum rod length and minimum number of support grids) for the Maine Yankee fuel rod used in the model are:

Outer diameter of cladding (inches)	0.434
Cladding thickness (inches)	0.023
Cladding density (lb/in^3)	0.237
Fuel pellet density (lb/in^3)	0.396

The cladding is reduced from its nominal value of 0.026 inches by the assumed 80 micron oxidation layer (0.003 inches) to 0.023 inches. Similarly, the fuel rod outer diameter is reduced from the nominal value of 0.44 inches to 0.434 inches.

The elevation of the grids, measured from the bottom of the fuel assembly are: 2.3, 33.0, 51.85, 70.7, 89.6, 108.4, 127.3 and 144.9 (inches).

The effective cross-sectional properties (EI_{eff}) for the beam are computed by adding the value of EI for the cladding and the pellet, where:

E = modulus of elasticity (lb/in^2)
I = cross-sectional moment of inertia (in^4)

The lowest frequency for the extentional mode shape was computed to be 219.0 Hz. The first mode shape corresponds to a frequency of 25.9 Hz. Using the expression for the DLF previously discussed, the DLF is computed to be 0.240 ($\beta = 8.44$).

120 Micron Oxide Layer Thickness Evaluation

The buckling calculation used the same model employed for the mode shape calculation. The load that would potentially buckle the fuel rod in the end drop is due to the deceleration of the rod. This loading was implemented by applying a 1g acceleration in the direction that would result in compressive loading of the fuel rod. The acceleration required to buckle the fuel rod is computed to be 37.3g.

Using the same fuel rod model, the acceleration required to buckle the fuel rods is found to be 37.3g, which is much higher than the calculated effective g-load (14.3g) due to the 60g end drop. Therefore, the fuel rods with a 120 micron cladding oxide layer do not buckle in the 60g end drop event.

Figure 11.2.15.1.5-1 Two-Dimensional Beam Finite Element Model for Maine Yankee Fuel Rod

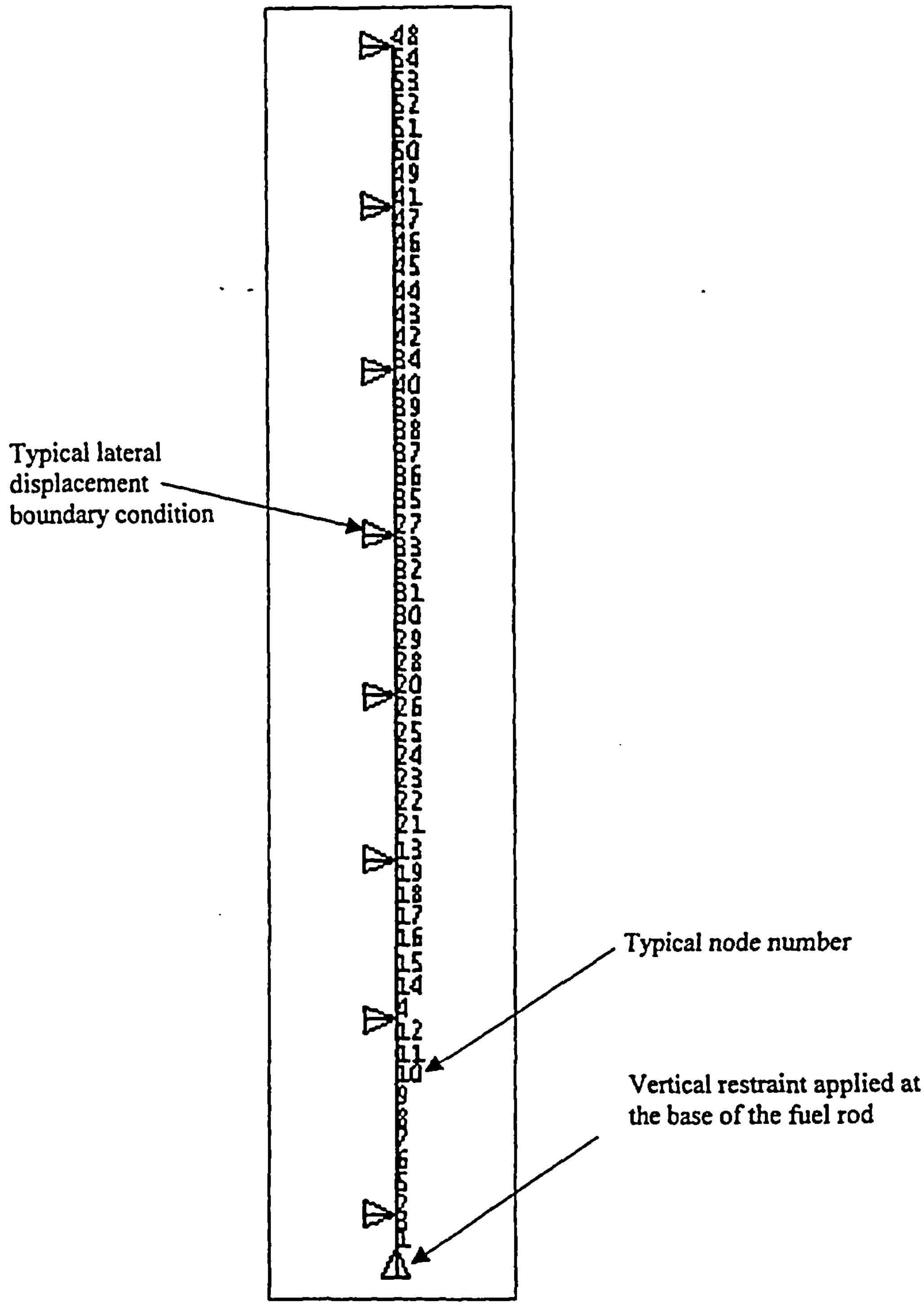
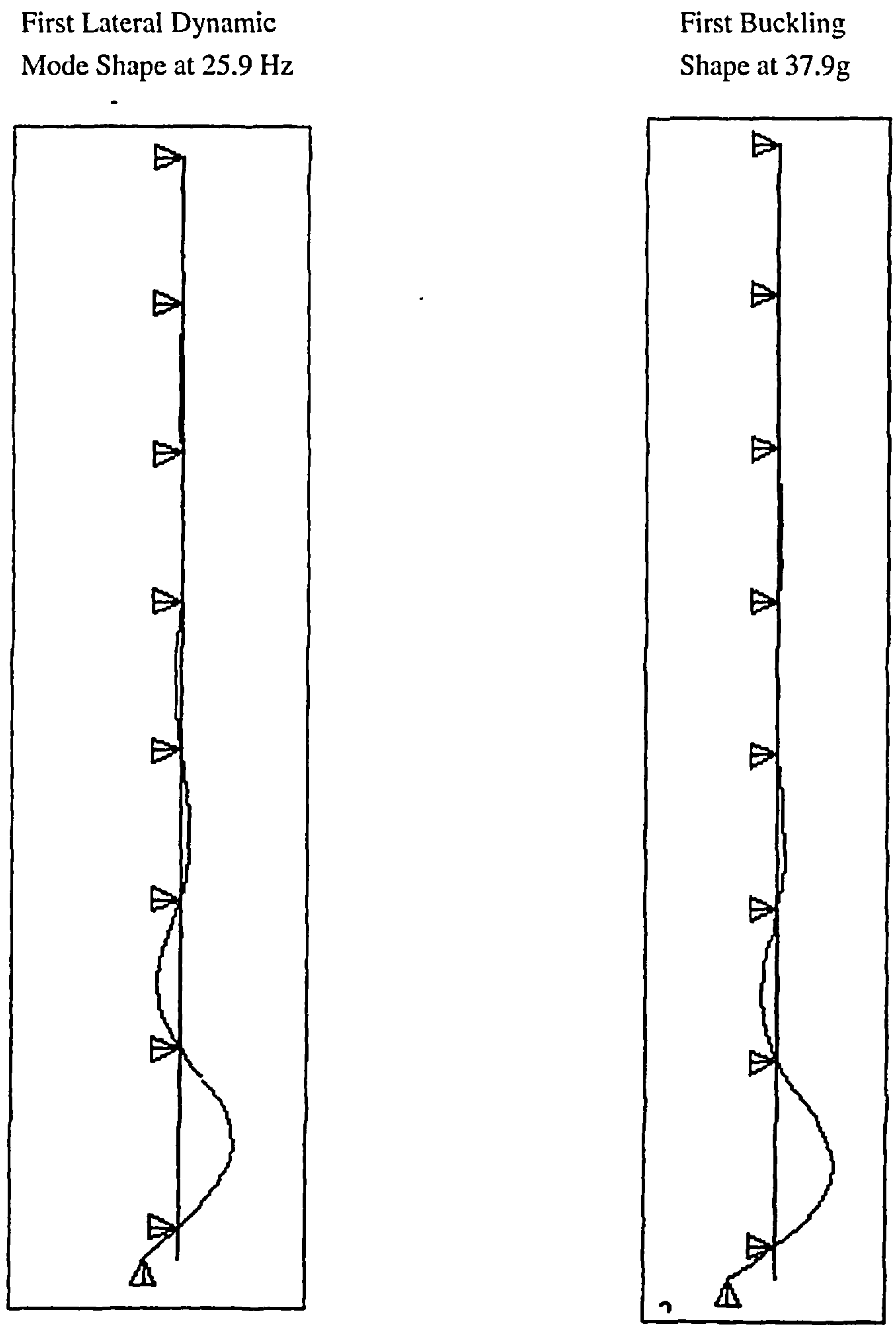


Figure 11.2.15.1.5-2 Mode Shape and First Buckling Shape for the Maine Yankee Fuel Rod



11.2.15.1.6 Buckling Evaluation for High Burnup Fuel with Mechanical Damage

This section presents the buckling evaluation for high burnup fuel having an 80 micron cladding oxide layer thickness and with mechanical damage consisting of one or more missing support grids up to an unsupported fuel rod length of 60 inches.

End Drop Evaluation

The buckling load is maximized at the bottom of the fuel assembly. The bounding evaluation is the removal of the grid strap that maximizes the spacing at the lowest vertical elevation. The elevations of the grids in the model, measured from the bottom of the fuel assembly are: 2.3, 51.85, 70.7, 89.6, 108.4, 127.3 and 144.9 inches (Figure 11.2.15.1.6-1). The grid at the 33.0-inch elevation is removed, resulting in a grid spacing of approximately 50.0 inches. The grid located at 51.85 inches is conservatively assumed to be located at 62.3 inches, resulting in an unsupported rod length of 60.0 inches.

The case of the missing grid is evaluated using the methodology presented in Section 11.2.15.1.5 for the fuel assembly with the grids being present. The dimensions and physical data for the Maine Yankee fuel rod used in the model are:

Outer diameter of cladding (inches)	0.434
Cladding thickness (inches)	0.023
Cladding density (lb/in^3)	0.237
Fuel pellet density (lb/in^3)	0.396
Fuel pellet Modulus of Elasticity (psi)	13.0×10^6
Zircaloy cladding Modulus of Elasticity (psi)	10.47×10^6

The cladding is reduced from its nominal value of 0.026 inches by the assumed 80 micron oxidation layer thickness (0.003 inches) to 0.023 inches. Similarly, the fuel rod outer diameter is reduced from the nominal value of 0.44 inches to 0.434 inches. The fuel pellet modulus of elasticity is conservatively reduced 50%. The modulus of elasticity of the Zircaloy cladding is taken from ISG-12 [50].

With the grid missing, the frequency of the fundamental lateral mode shape is 7.8 Hz. The natural frequency of the fundamental extensional mode was determined to be 218.9 Hz. The DLF is computed to be 0.072, resulting in an effective acceleration of $0.072 \times 60 = 4.3$ g. Using the same method to compute the acceleration at which buckling occurs, the lowest buckling acceleration is 14.4 g, which is significantly greater than 4.3 g. Therefore, the fuel rod does not buckle during an end drop. Figures 11.2.15.1.6-1 and 11.2.15.1.6-2 show the finite element model and buckling results and mode shape.

Side Drop Evaluation

The Maine Yankee fuel rod is evaluated for a 60 g side drop with a missing support grid in the fuel assembly. Using the same assumptions as for the end drop evaluation, the span between support grids is assumed to be 60.0 inches.

For this analysis, the dimensions and physical data used are:

Fuel rod OD	0.434 in. (80 micron oxidation layer)
Clad ID	0.388 in.
E_{clad}	10.47E6 psi
E_{fuel}	13.0E6 psi
Clad density	0.237 lb/in^3
Fuel density	0.396 lb/in^3
A_{clad}	0.030 in^2 (cross-sectional area)
A_{fuel}	0.118 in^2 (cross-sectional area)

The mass of the fuel rod per unit length is:

$$m = \frac{0.396(0.122) + 0.237(0.030)}{386.4} = 0.000143 \text{ lb-s}^2/\text{in}^2$$

For the fuel rod, the product of the Modulus of Elasticity (E) and Moment of Inertia (I), is:

$$EI_{clad} = 10.47E6 \frac{\pi(0.217^4 - 0.194^4)}{4} = 6{,}586 \text{ lb-in}^2$$

$$EI_{fuel} = 13.0E6 \frac{\pi(0.194^4)}{4} = 14{,}462 \text{ lb-in}^2$$

$$EI = 6{,}586 + 14{,}462 = 21{,}048 \text{ lb-in}^2$$

During a side drop, the maximum deflection of a fuel rod is based on the fuel rod spacing of the fuel assembly. The pitch (center-to-center spacing) of fuel rods is 0.58 inches [51]. The maximum pitch is across the diagonal of the fuel assembly. The maximum pitch is:

$$dp = \frac{0.58}{\sin 45} = 0.82 \text{ in.}$$

The maximum deflection of a fuel rod is at the top of the fuel assembly and the minimum deflection is at the bottom of the fuel assembly.

Assuming a 17 × 17 array (which envelops the Maine Yankee 14 × 14 array), the maximum fuel rod deflection is:

$$(17\text{-}1) \times (0.82\text{-}0.43) = 6.18 \text{ in.}$$

The deflection of a simply supported beam with a distributed load is given by the equation:

$$\Delta = \frac{5\omega l^4}{384EI} = \frac{5(g\omega)l^4}{384\,(EI_{total})} \qquad [52]$$

$$g = \frac{384\Delta(EI_{total})}{5\omega l^4}$$

The cladding bending stress is given by the equation:

$$S = \frac{Mc}{I} = \frac{\left(\left(g\omega l^2\right)/8\right)c}{I_{clad}}\left(\frac{EI_{clad}}{EI_{total}}\right)$$

Inserting the equation for 'g':

$$S = \frac{384\Delta cE_{clad}}{40 \times L^2}$$

where:

$c = 0.217$ inch distance from center of fuel rod to extreme outer fiber

$L = 60$ inches (the unsupported fuel rod length)

$\Delta = 6.18$ inches (the maximum deflection)

The bending stress in the fuel rod is:

$$S = \frac{384 \times 6.18 \times 0.217 \times 10.47E6}{40(60)^2} = 37.4\ \text{ksi}$$

The maximum hoop stress due to the fuel rod internal pressure is determined to be 19.1 ksi (131.4 MPa per Tables 4.4.7-3 and 4.5.1.2-1). Therefore, the maximum axial stress is 9.6 ksi (one half of the hoop stress [53]).

The bearing stress between two fuel rods under a 60 g load is:

$$S_{brg} = 0.591\sqrt{\frac{\omega E}{K_D}} = 0.591\sqrt{\frac{(0.000143 \times 386.4) \times 60 \times 10.47E6}{0.22}} = 7.4\ \text{ksi} \quad [53]$$

where:

$$K_D = \frac{D_1 D_2}{D_1 + D_2} = \frac{0.434 \times 0.434}{0.434 + 0.434} = 0.22$$

The total stress is:

$$S = 37.4 + 9.6 + 7.4 = 54.4\ \text{ksi}$$

The ultimate strength allowable for irradiated Zircaloy–4 is 83.4 ksi (Figure 3-2 [54]). Therefore, the margin of safety for ultimate strength is:

$$MS = \frac{83.4}{54.4} - 1 = 0.53$$

The yield strength allowable for irradiated Zircaloy–4 is 78.3 ksi (Figure 3-2 [54]). Therefore, the margin of safety for yield strength is:

$$MS = \frac{78.3}{54.4} - 1 = 0.44$$

The maximum bearing stress occurs between the bottom fuel rod and the fuel tube. The bearing stress is:

$$S_{brg} = 0.591\sqrt{\frac{17\times 0.000143\times 386.4\times 60\times 10.47E6}{0.44}} = 21.6\,\text{ksi}$$

The bending stress is negligible because the maximum deflection is equal to the spacing of the fuel rods established by the grid. Therefore, the top fuel rod is bounding.

Consequently, the fuel rods are demonstrated to be structurally adequate for the 60g side drop loading condition.

Figure 11.2.15.1.6-1 Two-Dimensional Beam Finite Element Model for a Fuel Rod with a Missing Grid

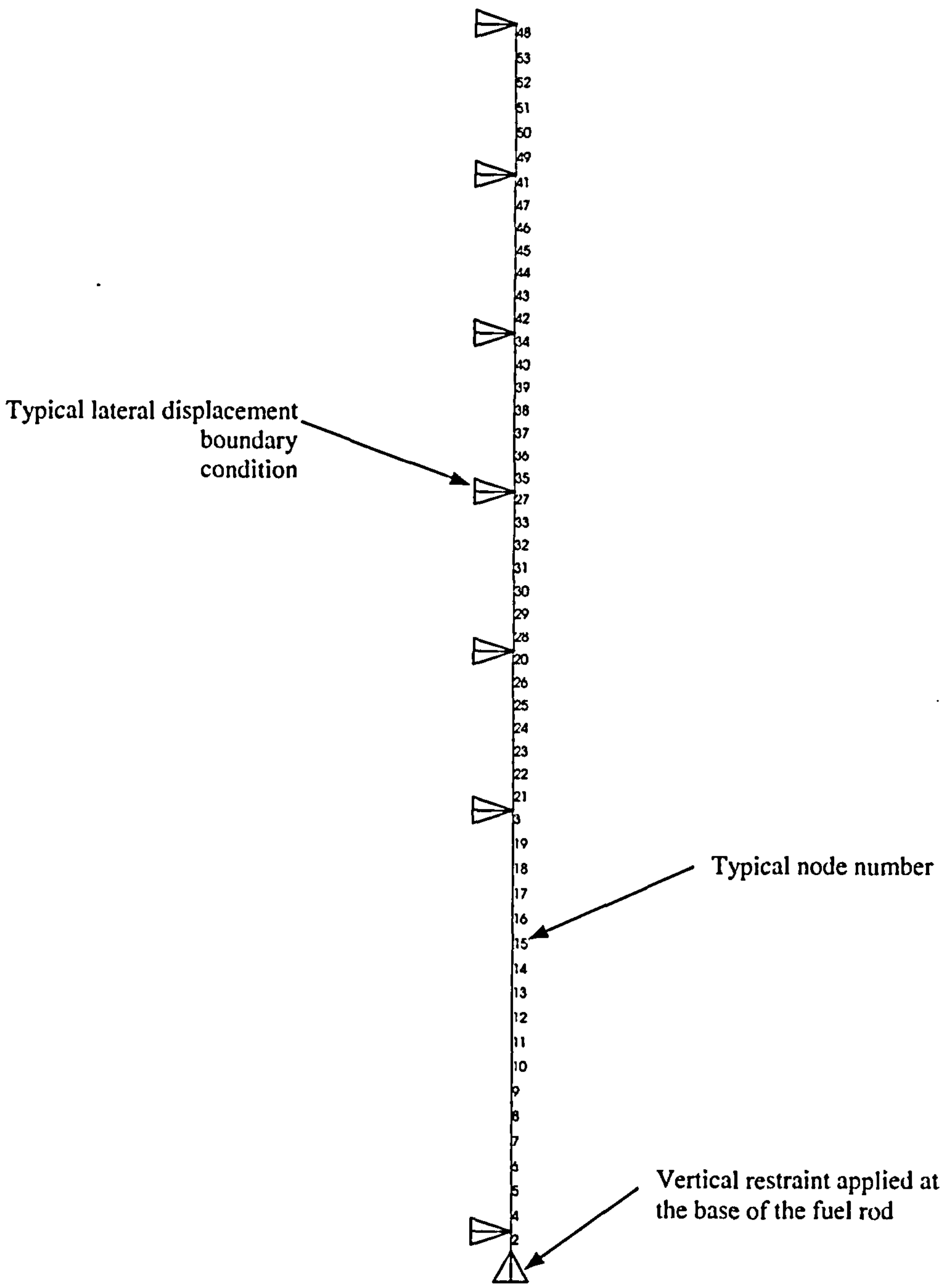


Figure 11.2.15.1.6-2 Modal Shape and First Buckling Mode Shape for a Fuel Rod with a Missing Grid

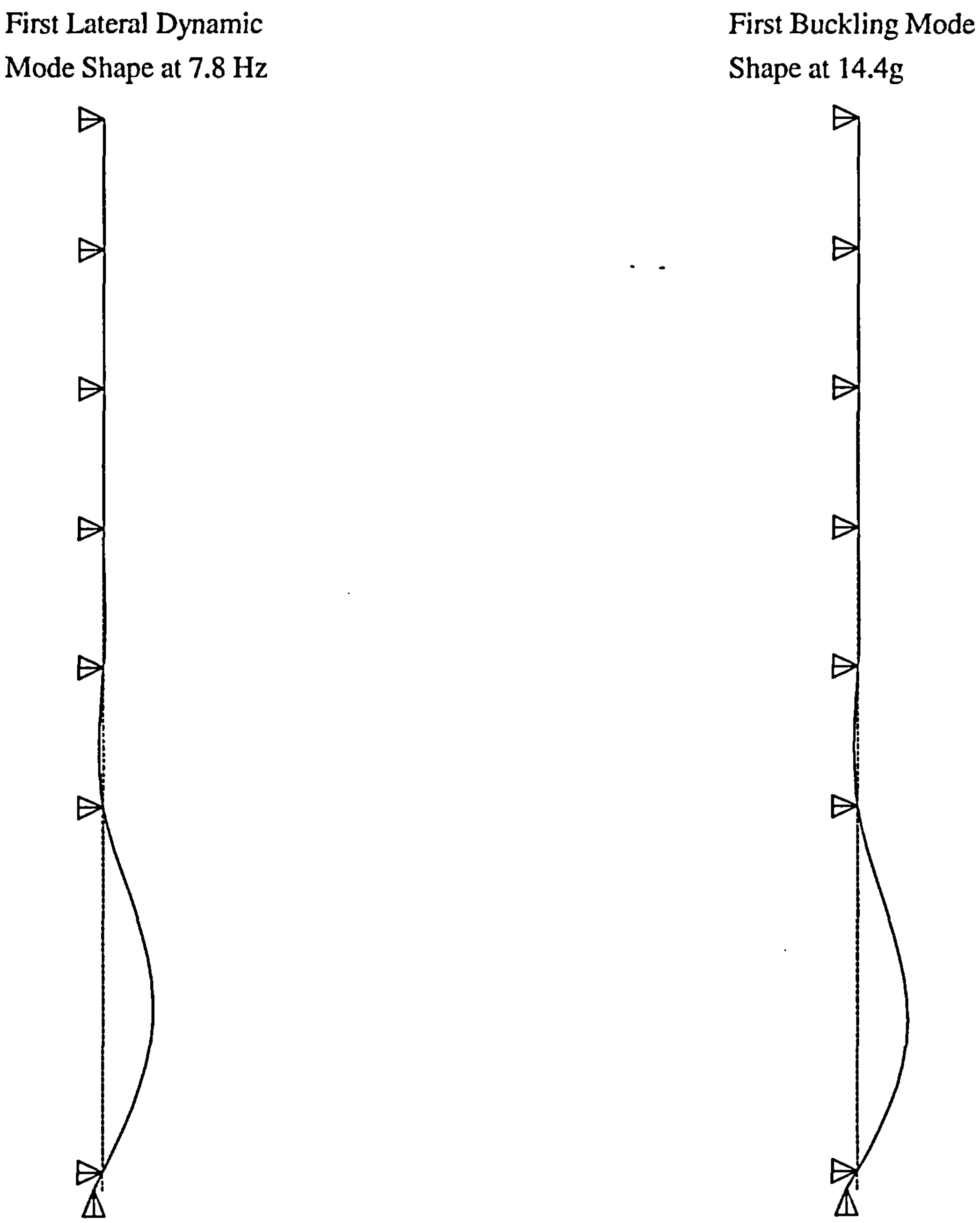


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Table of Contents

List of Tables

12.0 OPERATING CONTROLS AND LIMITS

This chapter identifies operating controls and limits, technical parameters and surveillance requirements imposed to ensure the safe operation of the NAC-UMS® System.

Controls used by NAC International (NAC) as part of the NAC-UMS® design and fabrication are provided in the NAC Quality Assurance Manual and Quality Procedure. The NAC Quality Assurance Program is discussed in Chapter 13.0. If procurement and fabrication of the NAC-UMS® System is performed by others, a Quality Assurance Program prepared in accordance with 10 CFR 72 Subpart G shall be implemented. Site specific controls for the organization, administrative system, procedures, record keeping, review, audit and reporting necessary to ensure that the NAC-UMS® storage system installation is operated in a safe manner, are the responsibility of the user of the system.

12.1 Administrative and Operating Controls and Limits for the NAC-UMS® System

The NAC-UMS® Storage System operating controls and limits are summarized in Table 12-1. Appendix A of the Amendment 3 Technical Specifications provides the proposed Limiting Conditions for Operations (LCO). The Approved Contents and Design Features for the NAC-UMS® System are presented in Appendix B of the Amendment 3 Technical Specifications. The Bases for the specified controls and limits are presented in Appendix 12C.

Section 3.0 of Appendix B presents Design Features that are important to the safe operation of the NAC-UMS® System, but that are not included as Technical Specifications. These include items which are singular events, those that cannot be readily determined or re-verified at the time of use of the system, or that are easily implemented, verified and corrected, if necessary, at the time the action is undertaken.

12.2 Administrative and Operating Controls and Limits for SITE-SPECIFIC FUEL

This section describes the administrative and operating controls and limits placed on the loading of fuel assemblies that are unique to specific reactor sites. SITE-SPECIFIC FUEL configurations result from conditions that occurred during reactor operations, participation in research and development programs, testing programs intended to improve reactor operations, from the placement of control components or other items within the fuel assembly and from the disposition of damaged fuel assemblies or fuel rods.

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. Separate evaluation may establish different limits, which are maintained by administrative controls for preferential loading. The preferential loading controls take advantage of design features of the UMS® Storage System to allow the loading of fuel configurations that may have higher burnup or additional hardware material that is not specifically considered in the design basis fuel evaluation.

Unless specifically excepted, SITE SPECIFIC FUEL must meet all of the conditions specified for the design basis fuel presented in Table 12-1.

12.2.1 Operating Controls and Limits for Maine Yankee SITE-SPECIFIC FUEL

The fuel design used at Maine Yankee is the Combustion Engineering (CE) 14 × 14 fuel assembly. The CE 14 × 14 fuel assembly is one of those included in the design basis evaluation of the UMS® Storage System as shown in Table B2-2 of Appendix B of Certificate of Compliance No. 72-1015. The estimated Maine Yankee SITE-SPECIFIC FUEL inventory is shown in Table B2-6. Except as noted in this section, the spent fuel in this inventory meets the Fuel Assembly Limits provided in Table B2-1.

As shown in Table B2-6, certain of the Maine Yankee fuel has characteristics, such as fuel assembly lattice configurations, different from STANDARD FUEL, from PWR INTACT FUEL ASSEMBLIES - including CONSOLIDATED FUEL, DAMAGED FUEL and fuel with higher burnup or enrichment, that differs from the characteristics of the fuel considered in the design basis. As shown in Table B2-6, certain fuel configurations must be preferentially loaded in corner or peripheral fuel tube positions in the fuel basket based on the shielding, criticality or thermal evaluation of the fuel configuration.

The corner positions are used for the loading of fuel configurations with missing fuel rods, and for DAMAGED FUEL and CONSOLIDATED FUEL in the MAINE YANKEE FUEL CAN. Specification for placement in the corner fuel tube positions results primarily from shielding or criticality evaluations of the designated fuel configurations.

Spent fuel having a burnup from 45,000 to 50,000 MWD/MTU is assigned to peripheral locations, and may require loading in a Maine Yankee fuel can. The interior locations must be loaded with fuel that has lower burnup and/or longer cool times in order to maintain the design basis heat load and component temperature limits for the basket and canister.

The Fuel Assembly Limits for the Maine Yankee SITE SPECIFIC FUEL are shown in Table B2-7 of Appendix B of the Amendment 3 Technical Specifications. Part A of the table lists the STANDARD, INTACT FUEL ASSEMBLY and SITE SPECIFIC FUEL that does not require preferential loading.

Part B of the table lists the SITE SPECIFIC FUEL configurations that require preferential loading due to the criticality, shielding or thermal evaluation. The loading pattern for Maine Yankee SITE SPECIFIC FUEL that must be preferentially loaded is presented in Section B 2.1.2. The preferential loading controls take advantage of design features of the UMS® Storage System to allow the loading of fuel configurations that may have higher burnup or additional hardware or fuel source material that is not specifically considered in the design basis fuel evaluation.

Fuel assemblies with a Control Element Assembly (CEA) or a CEA plug inserted are loaded in a Class 2 canister and basket due to the increased length of the assembly with either of these components installed. However, these assemblies are not restricted as to loading position within the basket.

The Transportable Storage Canister loading procedures for Maine Yankee SITE SPECIFIC FUEL are administratively controlled in accordance with the requirements of Section B 2.1.2 for the loading of: (1) a fuel configuration with removed fuel or poison rods, (2) a MAINE YANKEE FUEL CAN, or (3) fuel with burnup between 45,000 MWD/MTU and 50,000 MWD/MTU.

Table 12-1 NAC-UMS® System Controls and Limits

Control or Limit	Applicable Technical Specification	Condition or Item Controlled
1. Fuel Characteristics	Table B2-1	Type and Condition
	Table B2-2	Class, Dimensions and Weight for PWR
	Table B2-3	Class, Dimensions and Weight for BWR
	Table B2-4	Minimum Cooling Time for PWR Fuel
	Table B2-5	Minimum Cooling Time for BWR Fuel
	Table B2-7	Maine Yankee Site Specific Fuel Limits
	Table B2-8	Minimum Cooling Time for Maine Yankee Fuel – No CEA
	Table B2-9	Minimum Cooling Time for Maine Yankee Fuel – With CEA
2. Canister	LCO 3.1.4	Time in Transfer Cask (fuel loading)
Fuel Loading	Table B2-1	Weight and Number of Assemblies
	Table B2-7	Maine Yankee Site Specific Fuel Limits
	Table B2-4	Minimum Cooling Time for PWR Fuel
	Table B2-5	Minimum Cooling Time for BWR Fuel
Drying	LCO 3.1.2	Vacuum Drying Pressure
Backfilling	LCO 3.1.3	Helium Backfill Pressure
Sealing	LCO 3.1.5	Helium Leak Rate
Vacuum	LCO 3.1.1	Time in Vacuum Drying
External Surface	LCO 3.2.1	Level of Contamination
Unloading	Note 1	Fuel Cooldown Requirement
3. Concrete Cask	LCO 3.2.2	Surface Dose Rates
	Note 1	Cask Spacing
	Note 2	Cask Handling Height
4. Surveillance	LCO 3.1.6	Heat Removal System
5. Transfer Cask	B 3.4(8)	Minimum Temperature
6. ISFSI Concrete Pad	B3.4.1(6) B3.4.2(7)	Seismic Event Performance

1. Procedure and/or limits are presented in the Operating Procedures of Chapter 8.
2. Lifting height and handling restrictions are provided in Section A5.6 of Appendix A.

APPENDIX 12A

TECHNICAL SPECIFICATIONS FOR THE NAC-UMS® SYSTEM

The Technical Specifications for the NAC-UMS® storage system, including the Limiting Conditions for Operation (LCOs), Surveillance Requirements (SRs) and the Administrative Controls and Programs, are incorporated in Appendix A of Certificate of Compliance No. 1015.

APPENDIX 12B

APPROVED CONTENTS AND DESIGN FEATURES FOR THE NAC-UMS® SYSTEM

The NAC-UMS® storage system Approved Contents and Design Features are incorporated in Appendix B of Certificate of Compliance No. 1015.

APPENDIX 12C

TECHNICAL SPECIFICATION BASES
FOR THE NAC-UMS® SYSTEM

Appendix 12C
Table of Contents

C 1.0 Introduction

This Appendix presents the design or operational condition, or regulatory requirement, which establishes the bases for the Technical Specifications provided in Appendix A of Certificate of Compliance No. 1015.

The section and paragraph numbering used in this Appendix is consistent to the numbering used in Appendix A, Technical Specifications for the NAC-UMS® SYSTEM, and Appendix B, Approved Contents and Design Features for the NAC-UMS® System, of Certificate of Compliance No. 1015.

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C 2.0 APPROVED CONTENTS

C 2.1 Fuel to be Stored in the NAC-UMS® SYSTEM

BASES

BACKGROUND	The NAC-UMS® SYSTEM design requires specifications for the spent fuel to be stored, such as the type of spent fuel, minimum and maximum allowable enrichment prior to irradiation, maximum burnup, minimum acceptable post-irradiation cooling time prior to storage, maximum decay heat, and condition of the spent fuel (i.e., INTACT FUEL). Other important limitations are the dimensions and weight of the fuel assemblies. The approved contents, which can be loaded into the NAC-UMS® SYSTEM are specified in Section B2.0 of Appendix B. Specific limitations for the NAC-UMS® SYSTEM are specified in Table B2-1 of Appendix B. These limitations support the assumptions and inputs used in the thermal, structural, shielding, and criticality evaluations performed for the NAC-UMS® SYSTEM.
APPLICABLE SAFETY ANALYSES	To ensure that the shield lid is not placed on a CANISTER containing an unauthorized fuel assembly, facility procedures require verification of the loaded fuel assemblies to ensure that the correct fuel assemblies have been loaded in the canister.
APPROVED CONTENTS	C 2.1.1 Approved Contents Section B2.0 refers to Table B2-1 in Appendix B for the specific fuel assembly characteristics for the PWR or BWR fuel assemblies authorized for loading into the NAC-UMS® SYSTEM. These fuel assembly characteristics include parameters such as cladding material, minimum and maximum enrichment, decay heat generation, post-irradiation cooling time, burnup, and fuel assembly length, width, and weight. Tables B2-2 through B2-5 are referenced from Table B2-1 and provide additional specific fuel characteristic limits for the fuel assemblies based on the fuel assembly class type, enrichment, burnup and cooling time.

(continued)

APPROVED CONTENTS (continued)	The fuel assembly characteristic limits of Tables B2-1 through B2-5 must be met to ensure that the thermal, structural, shielding, and criticality analyses supporting the NAC-UMS® SYSTEM Safety Analysis Report are bounding. <u>C 2.1.2</u> Approved Contents Section B2.0 in Appendix B requires preferential loading of Maine Yankee SITE SPECIFIC FUEL assemblies with significantly different post-irradiation cooling times. This preferential loading is required to prevent a cooler assembly from heating up due to being surrounded by hotter fuel assemblies. For the purposes of complying with this Approved Contents limit, only fuel assemblies with post-irradiation cooling times differing by one year or greater need to be loaded preferentially. This is based on the fact that the heat-up phenomenon can only occur with significant differences in decay heat generation characteristics between adjacent fuel assemblies having different post-irradiation cooling times.
APPROVED CONTENT LIMITS AND VIOLATIONS	<u>C 2.2.1</u> If any Approved Contents limit of B2.1.1 or B2.1.2 in Appendix B is violated, the limitations on fuel assemblies to be loaded are not met. Action must be taken to place the affected fuel assembly(s) in a safe condition. This safe condition may be established by returning the affected fuel assembly(s) to the spent fuel pool. However, it is acceptable for the affected fuel assemblies to temporarily remain in the NAC-UMS® SYSTEM, in a wet or dry condition, if that is determined to be a safe condition. <u>C 2.2.2 and C 2.2.3</u> NRC notification of the Approved Contents limit violation is required within 24 hours. A written report on the violation must be submitted to the NRC within 30 days. This notification and written report are independent of any reports and notification that may be required by 10 CFR 72.216.
REFERENCES	1. FSAR, Sections 2.1, 4.4; Chapters 5 and 6.

C 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the NAC-UMS® SYSTEM is in the specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within the specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and, b. Completion of the Required Actions is not required when an LCO is meet within the specified Completion Time, unless otherwise specified. There are two basic Required Action types. The first Required Action type specifies a time limit, the Completion Time to restore a system or component or to restore variables to within specified limits, in which the LCO must be met. Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second Required Action type specifies the remedial measures that permit continued activities that are not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

(continued)

LCO 3.0.2 (continued)	Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications. The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillance, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.
LCO 3.0.3	This specification is not applicable to the NAC-UMS® SYSTEM because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.
LCO 3.0.4	LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the facility in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist: a. NAC-UMS® SYSTEM conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in NAC-UMS® SYSTEM activities being required to exit the Applicability desired to be entered to comply with the Required Actions. Compliance with Required Actions that permit continued operation for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the NAC-UMS® SYSTEM. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

(continued)

LCO 3.0.4 (continued)

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of the NAC-UMS® SYSTEM.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with the ACTIONS. The sole purpose of the Specification is to provide an exception to LCO 3.0.2 (e.g. to not comply with the applicable Required Action[s]) to allow the performance of testing to demonstrate:

a. The equipment being returned to service meets the LCO; or

b. Other equipment meets the applicable LCOs.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

C 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

Surveillance Requirements (SRs)	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillance is performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to meet Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or,

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the NAC-UMS® SYSTEM is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including those invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance

(continued)

SR 3.0.1 (continued)	testing may not be possible in the current specified conditions in the Applicability, due to the necessary NAC-UMS® SYSTEM parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a specified condition where other necessary post-maintenance tests can be completed.
SR 3.0.2	SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities). The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable." As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.

(continued)

SR 3.0.2 (continued)	The provisions of SR 3.0.2 are not intended to be used repeatedly, merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes: consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency, based not on time intervals, but upon specified NAC-UMS® SYSTEM conditions, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility, which is not intended to be used as an operational convenience to extend Surveillance intervals.

(continued)

SR 3.0.3 (continued)	If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance. Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.
SR 3.0.4	SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability. This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe operation of NAC-UMS® SYSTEM activities. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability. However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO.

(continued)

SR 3.0.4 (continued)

When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s), since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in a SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not in this situation, LCO 3.0.4 will govern any restrictions that may be (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of the NAC-UMS® SYSTEM.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances, when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO, prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering LCO Applicability, would have its Frequency specified such that is not "due" until the specific conditions needed are met.

Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or to be performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.1 CANISTER Maximum Time in Vacuum Drying

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.

Limiting the elapsed time from the end of CANISTER draining operations through dryness verification testing and subsequent backfilling of the CANISTER with helium ensures that the short-term temperature limits established in the Safety Analyses Report for the spent fuel cladding and CANISTER materials are not exceeded and that the test duration of 30 days (720 hours) considered in PNL-4835 for Zircaloy clad fuel for storage in air is not exceeded.

APPLICABLE SAFETY ANALYSIS

Limiting the total time for loaded CANISTER vacuum drying operations ensures that the short-term temperature limits for the fuel cladding and CANISTER materials are not exceeded. If vacuum drying operations are not completed in the required time period, the CANISTER is backfilled with helium, the TRANSFER CASK and loaded CANISTER are placed in the spent fuel pool and the TRANSFER CASK and loaded CANISTER are kept in the pool for a minimum of 24 hours.

(continued)

APPLICABLE SAFETY ANALYSIS (continued)	Analyses reported in the Safety Analysis Report conclude that spent fuel cladding and CANISTER material short-term temperature limits will not be exceeded for total elapsed time in the vacuum drying operation and in the TRANSFER CASK with the CANISTER filled with helium. Since the rate of heat up is slower for lower total heat loads, the time required to reach component limits is longer than for the design basis heat load. Consequently, longer time limits are specified for heat loads below the design basis for the PWR and BWR fuel configurations as shown in LCO 3.1.1. As shown in the LCO, for total heat loads not specified, the time limit for the next higher specified heat load is conservatively applied. Analysis also shows that the fuel cladding and CANISTER component temperatures are well below the allowable temperatures for the time durations specified from the end of in-pool cooling, or end of forced air cooling, of the CANISTER through the completion of the vacuum drying and for the time specified in LCO 3.1.4 for the CANISTER in the TRANSFER CASK when backfilled with helium.
LCO	Limiting the length of time for vacuum drying operations for the CANISTER ensures that the spent fuel cladding and CANISTER material temperatures remain below the short-term temperature limits for the NAC-UMS® SYSTEM.
APPLICABILITY	The elapsed time restrictions for vacuum drying operations on a loaded CANISTER apply during LOADING OPERATIONS from the completion point of CANISTER draining operations through the completion point of the CANISTER dryness verification testing. The LCO is not applicable to TRANSPORT OPERATIONS or STORAGE OPERATIONS.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each NAC-UMS® SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each NAC-UMS® SYSTEM not meeting the LCO. Subsequent NAC-UMS® SYSTEMS that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

(continued)

ACTIONS (continued)

A.1

If the LCO time limit is exceeded, the CANISTER will be backfilled with helium to a pressure of 0 psig (+1,-0).

AND

A.2.1.1

The TRANSFER CASK and loaded CANISTER shall be placed in the spent fuel pool with the water level above the top of the CANISTER, a maximum water temperature of 100°F and with the annulus fill system operating for in-pool cooling operations.

AND

A.2.1.2

The TRANSFER CASK and loaded CANISTER shall be maintained in the spent fuel pool with the water level above the top of the CANISTER, and a maximum water temperature of 100°F for a minimum of 24 hours prior to the restart of LOADING OPERATIONS.

OR

A.2.2.1

A cooling air flow of 375 CFM at a maximum temperature of 76°F shall be initiated. The airflow will be routed to the annulus fill/drain lines of the TRANSFER CASK and will flow through the annulus and cool the CANISTER.

AND

A.2.2.2

The cooling air flow shall be maintained for a minimum of 24 hours prior to restart of LOADING OPERATIONS.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The elapsed time shall be monitored from completion of CANISTER draining through completion of the CANISTER vacuum dryness verification testing. Monitoring the elapsed time ensures that helium backfill and in-pool cooling operations can be initiated in a timely manner during LOADING OPERATIONS to prevent fuel cladding and CANISTER materials from exceeding short-term temperature limits.

SR 3.1.1.2

The elapsed time shall be monitored from the end of in-pool cooling through completion of the CANISTER vacuum dryness verification testing. Monitoring the elapsed time ensures that helium backfill and in-pool cooling operations can be initiated in a timely manner during LOADING OPERATIONS to prevents fuel cladding and CANISTER materials from exceeding short-term temperature limits.

REFERENCES

1. FSAR Sections 4.4 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.2 CANISTER Vacuum Drying Pressure

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents Limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad. CANISTER cavity vacuum drying is utilized to remove residual moisture from the CANISTER cavity after the water is drained from the CANISTER. Any water not drained from the CANISTER cavity evaporates due to the vacuum. This is aided by the temperature increase, due to the heat generation of the fuel.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of design basis spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on storage in an inert atmosphere. This is accomplished by removing water from the CANISTER and backfilling the cavity with helium. The thermal analysis assumes that the CANISTER cavity is dried and filled with helium.

(continued)

APPLICABLE SAFETY ANALYSIS (continued)	The heat-up of the CANISTER and contents will occur during CANISTER vacuum drying, but is controlled by LCO 3.1.1.
LCO	A vacuum pressure of 3 mm of mercury, as specified in this LCO, indicates that liquid water has evaporated and been removed from the CANISTER cavity. Removing water from the CANISTER cavity helps to ensure the long-term maintenance of fuel cladding integrity.
APPLICABILITY	Cavity vacuum drying is performed during LOADING OPERATIONS before the TRANSFER CASK holding the CANISTER is moved to transfer the CANISTER into the CONCRETE CASK. Therefore, the vacuum requirements do not apply after the CANISTER is backfilled with helium and leak tested prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the CANISTER cavity vacuum drying pressure limit cannot be met, actions must be taken to meet the LCO. Failure to successfully complete cavity vacuum drying could have many causes, such as failure of the vacuum drying system, inadequate draining, ice clogging of the drain lines, or leaking CANISTER welds. The Completion Time is sufficient to determine and correct most failure mechanisms. Excessive heat-up of the CANISTER and contents is precluded by LCO 3.1.1. <u>B.1</u> If the CANISTER fuel cavity cannot be successfully vacuum dried, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met.

(continued)

ACTIONS (continued)

A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1. The Completion Time is reasonable, based on the time required to reflood the CANISTER, perform fuel cooldown operations, cut the shield lid weld, move the TRANSFER CASK into the spent fuel pool, and remove the CANISTER shield lid in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Cavity dryness is demonstrated by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time. A low vacuum pressure is an indication that the cavity is dry. The surveillance must verify that the CANISTER cavity vacuum drying pressure is within the specified limit prior to backfilling the CANISTER with helium.

REFERENCES

1. FSAR Sections 4.4, 7.1 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.3 CANISTER Helium Backfill Pressure

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad. Backfilling of the CANISTER cavity with helium promotes heat transfer from the spent fuel to the CANISTER structure and the inert atmosphere protects the fuel cladding. Providing a helium pressure equal to atmospheric pressure ensures that there will be no in-leakage of air over the life of the CANISTER, which might be harmful to the heat transfer features of the NAC-UMS® SYSTEM and harmful to the fuel.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on the ability of the NAC-UMS® SYSTEM to remove heat from the CANISTER and reject it to the

(continued)

APPLICABLE SAFETY ANALYSIS (continued)	environment. This is accomplished by removing water from the CANISTER cavity and backfilling the cavity with an inert gas. The heat-up of the CANISTER and contents will continue following backfilling with helium, but is controlled by LCO 3.1.4. The thermal analyses of the CANISTER assume that the CANISTER cavity is dried and filled with dry helium.
LCO	Backfilling the CANISTER cavity with helium at a pressure equal to atmospheric pressure ensures that there is no air in-leakage into the CANISTER, which could decrease the heat transfer properties and result in increased cladding temperatures and damage to the fuel cladding over the storage period. The helium backfill pressure of 0 psig specified in this LCO was selected based on a minimum helium purity of 99.9% to ensure that the CANISTER internal pressure and heat transfer from the CANISTER to the environment are maintained consistent with the design and analysis basis of the CANISTER.
APPLICABILITY	Helium backfill is performed during LOADING OPERATIONS, before the TRANSFER CASK and CANISTER are moved to the CONCRETE CASK for transfer of the CANISTER. Therefore, the backfill pressure requirements do not apply after the CANISTER is backfilled with helium and leak tested prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent condition entry and application of associated Required Actions. A.1 If the backfill pressure cannot be established within limits, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which would prevent backfilling of the CANISTER cavity with helium. These actions include identification and repair of helium leak paths or replacement of the helium backfill equipment.

(continued)

ACTIONS (continued)

B.1

If the CANISTER cavity cannot be backfilled with helium to the specified pressure, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 cannot be extended by reperforming A.1. The Completion Time is reasonable based on the time required to re-flood the CANISTER, perform cooldown operations, cut the CANISTER shield lid weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the CANISTER shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert atmosphere and maintenance of adequate heat transfer mechanisms. Filling the CANISTER cavity with helium at a pressure within the range specified in this LCO will ensure that there will be no air in-leakage, which could potentially damage the fuel. This pressure of helium gas is sufficient to maintain fuel cladding temperatures within acceptable levels.

Backfilling of the CANISTER cavity must be performed successfully on each CANISTER before placing it in storage. The surveillance must verify that the CANISTER helium backfill pressure is within the limit specified prior to installation of the structural lid.

REFERENCES

1. FSAR Sections 4.4, 7.1 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.4 CANISTER Maximum Time in the TRANSFER CASK

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.

Backfilling the CANISTER cavity with helium promotes heat transfer from the fuel and the inert atmosphere protects the fuel cladding. The cumulative time a loaded, helium backfilled CANISTER may remain in the TRANSFER CASK is limited to 600 hours. This limit ensures that the test duration of 30 days (720 hours) considered in PNL-4835 for Zircaloy clad fuel for storage in air is not exceeded and ensures that the TRANSFER CASK is used as intended. The time limit is established to preclude long-term storage of a loaded CANISTER in the TRANSFER CASK.

Intermediate time limits are established for CANISTERS with heat loads above 20 kW (PWR) or 17 kW (BWR) if they are not in either forced air cooling or in-pool cooling. These intermediate limits assure that the short-term temperature limits established in the Safety Analysis Report for the spent fuel cladding and CANISTER materials are not exceeded. Placing the CANISTER in either forced air cooling or in-pool cooling for a minimum of 24 hours maintains temperatures within the short-term limits. For heat loads less than or equal to 20kW (PWR) or 17kW (BWR), neither forced air cooling nor in-pool cooling is required.

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APPLICABLE SAFETY ANALYSIS

Analyses reported in the Safety Analysis Report conclude that for heat loads greater than 20 kW (PWR) or greater than 17 kW (BWR), spent fuel cladding and CANISTER material short-term temperature limits will not be exceeded for the total elapsed times specified in LCO 3.1.4. As shown in the LCO, for total heat loads not specified, the time limit for the next higher specified heat load is conservatively applied. The thermal analysis shows that the fuel cladding and CANISTER component temperatures are below their allowable temperatures for the time durations specified, with the CANISTER in the TRANSFER CASK and backfilled with helium, after completion of 24 hours of in pool cooling with the annulus fill system in operation, or forced air cooling. For lower heat loads, the steady state fuel cladding and component temperatures are below the allowable temperatures.

The basis for forced air cooling is an inlet maximum air temperature of 76°F which is the maximum normal ambient air temperature in the thermal analysis. The specified 375 CFM air flow rate exceeds the CONCRETE CASK natural convective cooling flow rate by a minimum of 10 percent. This comparative analysis conservatively excludes the higher flow velocity resulting from the smaller annulus between the TRANSFER CASK and CANISTER, which would result in improved heat transfer from the CANISTER.

From calculated temperatures reported in the Safety Analysis Report, it can be concluded that spent fuel cladding and CANISTER material short-term temperature limits will not be exceeded for a total elapsed time of greater than 20 hours for PWR fuel or 30 hours for BWR fuel for high heat loads, if the loaded CANISTER backfilled with helium is in the TRANSFER CASK. A 2 hour completion time is provided to establish in-pool or forced airflow cooling to ensure cooling of the CANISTER.

For heat loads of 20 kW or less (PWR), or 17 kW or less (BWR), and with the CANISTER backfilled with helium, the analysis shows that the fuel cladding and CANISTER components reach a steady-state temperature below the short-term allowable temperatures. Therefore, the time in the TRANSFER CASK is limited to 600 hours. For heat loads greater than 20 kW (PWR), or greater than 17 kW (BWR), the analysis shows that if forced air cooling at 375 CFM with air at 76°F is used, the temperatures of the fuel cladding and CANISTER components are at or below the values calculated for the CONCRETE CASK normal conditions.

(continued)

APPLICABLE SAFETY ANALYSIS (Continued)	This limit ensures that the test duration of 30 days (720 hours) considered in PNL-4835 for Zircaloy clad fuel for storage in air is not exceeded and ensures that the TRANSFER CASK is used as intended. Since the 600 hours is significantly less than the 720 hours considered in PNL-4835, operation in the TRANSFER CASK to this period is acceptable. Since the cooling provided by the forced air is equivalent to the passive cooling provided by the CONCRETE CASK and TRANSPORT CASK, relocation of a loaded and helium-filled CANISTER to a CONCRETE CASK or TRANSPORT CASK ensures that the fuel cladding and CANISTER component short-term temperature limits are not exceeded.
LCO	For PWR heat loads less than or equal to 20 kW, and BWR heat loads less than or equal to 17 kW, the thermal analysis shows that the presence of helium in the CANISTER is sufficient to maintain the fuel cladding and CANISTER component temperatures below the short-term temperature limits. Therefore, forced air cooling or in-pool cooling is not required for these heat load conditions. For higher heat loads of these fuels, as shown in the LCO, once forced air cooling or in-pool cooling is established, the amount of time the CANISTER resides in the TRANSFER CASK is not limited by the intermediate time limits, since the cooling provided by the forced air or water is equivalent to the passive cooling that is provided by the CONCRETE CASK or TRANSPORT CASK. If forced air flow or in-pool cooling is continuously maintained for a period of 24 hours, or longer, then the temperatures of the spent fuel cladding and CANISTER components are at, or below, the values calculated for the CONCRETE CASK normal conditions. Therefore, forced air cooling or in-pool cooling may be ended, allowing a new entry into Condition A of this LCO. This provides a new period in which continuation of LOADING OPERATIONS, TRANSFER OPERATIONS or UNLOADING OPERATIONS for high heat load PWR and BWR fuel may occur. Similarly, in LOADING OPERATIONS, TRANSFER OPERATIONS or UNLOADING OPERATIONS for heat loads up to the design basis, continuous forced air cooling or in-pool cooling maintains the fuel cladding and CANISTER component temperatures below the short-term temperature limits. Therefore, the CANISTER may remain in the TRANSFER CASK for up to 600 hours, where the time limit is based on the test duration of 30 days (720 hours) considered in PNL-4835 for Zircaloy clad fuel for storage in air rather than on temperature limits.

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APPLICABILITY	For LOADING OPERATIONS, the elapsed time restrictions on the loaded CANISTER apply from the completion point of the CANISTER helium backfilling through completion of the transfer from the TRANSFER CASK to the CONCRETE CASK. For TRANSFER OPERATIONS, the elapsed time restrictions on the loaded CANISTER apply from the completion point of the closing of the TRANSFER CASK shield doors through completion of the unloading of the CANISTER from the TRANSFER CASK. For UNLOADING OPERATIONS, the elapsed time restrictions on the loaded CANISTER apply from the completion point of the closing of the TRANSFER CASK shield doors through initiation of CANISTER cooldown.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each NAC-UMS® SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each NAC-UMS® SYSTEM not meeting the LCO. Subsequent NAC-UMS® SYSTEMS that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. A note has been added to Condition A that reminds users that all time spent in Condition A is included in the 600-hour cumulative limit. If LCO 3.1.4 intermediate time is not exceeded: A.1.1 The TRANSFER CASK containing the loaded CANISTER backfilled with helium will be placed in the spent fuel pool with the water level above the top of the CANISTER, a maximum water temperature of 100°F, and with the annulus fill system operating to allow the cooler water to reduce the TRANSFER CASK, CANISTER, and spent fuel cladding temperatures to below the short-term temperature limits. AND A.1.2 The TRANSFER CASK and loaded CANISTER shall be kept in the spent fuel pool for a minimum of 24 hours prior to restart of LOADING OPERATIONS, TRANSFER OPERATIONS and UNLOADING OPERATIONS.

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ACTIONS (continued)	<u>OR</u> <u>A.2.1</u> A cooling air flow of 375 CFM at a maximum temperature of 76° F shall be initiated. The airflow will be routed to the annulus fill/drain lines in the TRANSFER CASK and will flow through the annulus and cool the CANISTER. <u>AND</u> <u>A.2.2</u> The cooling air flow shall be maintained for a minimum of 24 hours prior to restart of LOADING OPERATIONS, TRANSFER OPERATIONS and UNLOADING OPERATIONS. If the LCO 3.1.4. 600-hour cumulative time limit is exceeded: <u>B.1</u> The CANISTER shall be placed in an operable CONCRETE CASK. <u>OR</u> <u>B.2</u> The CANISTER shall be placed in an operable TRANSPORT CASK. <u>OR</u> <u>B.3</u> The CANISTER shall be unloaded. The 5-day Completion Time for Required Actions B.1, B.2, and B.3 assures that the PNL-4835 30-day test duration used to establish the LCO limit will not be exceeded, taking into account the 600 hours allowed by the LCO.
SURVEILLANCE REQUIREMENTS	<u>SR 3.1.4.1</u> The elapsed time from entry into the LCO conditions of Applicability until placement of the CANISTER in an operable CONCRETE CASK or TRANSPORT CASK, or until CANISTER cooldown is initiated for UNLOADING OPERATIONS shall be monitored. This SR ensures that the fuel cladding and CANISTER component temperature limits are not exceeded.
REFERENCES	1. FSAR Sections 4.4, 8.1 and 8.2.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.5 CANISTER Helium Leak Rate

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.

Backfilling the CANISTER cavity with helium promotes heat transfer from the fuel to the CANISTER shell. The inert atmosphere protects the fuel cladding. Prior to transferring the CANISTER to the CONCRETE CASK, the CANISTER helium leak rate is verified to meet leaktight requirements to ensure that the fuel and helium backfill gas is confined.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on maintaining an inert atmosphere, and maintaining the cladding temperatures below established long-term limits. This is accomplished by removing water from the CANISTER and backfilling the cavity with helium. The heat-up of the CANISTER and contents will continue following backfilling the cavity and leak testing the shield lid-to-shell weld, but is controlled by LCO 3.1.4.

(continued)

LCO

Verifying that the CANISTER cavity helium leak rate is below the leaktight limit specified in this LCO ensures that the CANISTER shield lid is sealed. Verifying that the helium leak rate is below leaktight levels will also ensure that the assumptions in the accident analyses and radiological evaluations are maintained.

APPLICABILITY

The leaktight helium leak rate verification is performed during LOADING OPERATIONS before the TRANSFER CASK and integral CANISTER are moved for transfer operations to the CONCRETE CASK. TRANSPORT OPERATIONS would not commence if the CANISTER helium leak rate was not below the test sensitivity. Therefore, CANISTER leak rate testing is not required during TRANSPORT OPERATIONS or STORAGE OPERATIONS.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the helium leak rate limit is not met, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which could cause a helium leak rate in excess of the limit. Actions to correct a failure to meet the helium leak rate limit would include, in ascending order of performance, 1) verification of helium leak test system performance; 2) inspection of weld surfaces to locate helium leakage paths using a helium sniffer probe; and 3) weld repairs, as required, to eliminate the helium leakage. Following corrective actions, the helium leak rate verification shall be reperformed.

(continued)

ACTIONS (continued)

B.1

If the CANISTER leak rate cannot be brought within the limit, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated as necessary prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1. The Completion Time is reasonable based on the time required to re-flood the CANISTER, perform fuel cooldown operations, cut the CANISTER shield lid weld, move the TRANSFER CASK into the spent fuel pool, remove the CANISTER shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

The primary design consideration of the CANISTER is that it is leaktight to ensure that off-site dose limits are not exceeded and to ensure that the helium remains in the CANISTER during long-term storage. Long-term integrity of the stored fuel is dependent on storage in a dry, inert environment.

Verifying that the helium leak rate meets leaktight requirements must be performed successfully on each CANISTER prior to TRANSPORT OPERATIONS. The Surveillance Frequency allows sufficient time to backfill the CANISTER cavity with helium and performs the leak test, while minimizing the time the fuel is in the CANISTER and loaded in the TRANSFER CASK.

REFERENCES

1. FSAR Sections 7.1 and 8.1.

C 3.1 NAC-UMS® SYSTEM Integrity

C 3.1.6 CONCRETE CASK Heat Removal System

BASES

BACKGROUND	The CONCRETE CASK Heat Removal System is a passive, air-cooled convective heat transfer system, which ensures that heat from the CANISTER is transferred to the environment by the upward flow of air through the CONCRETE CASK. Relatively cool air is drawn into the annulus between the CONCRETE CASK and the CANISTER through the four air inlets at the bottom of the CONCRETE CASK. The CANISTER transfers its heat from the CANISTER surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air flows back into the environment through the four air outlets at the top of the CONCRETE CASK.
APPLICABLE SAFETY ANALYSIS	The thermal analyses of the CONCRETE CASK take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the CONCRETE CASK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and CANISTER component temperatures do not exceed applicable limits. Under normal storage conditions, the four air inlets and four air outlets are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs. Analyses have been performed for the complete obstruction of all of the air inlets and outlets. The complete blockage of all air inlets and outlets stops air cooling of the CANISTER. The CANISTER will continue to radiate heat to the relatively cooler inner shell of the CONCRETE CASK. With the loss of air cooling, the CANISTER component temperatures will increase toward their respective short-term temperature limits. The limiting component is the CANISTER basket support and heat transfer disks, which, by analysis, approach their temperature limits in 24 hours, if no action is taken to restore air flow to the heat removal system.
LCO	The CONCRETE CASK Heat Removal System must be verified to be OPERABLE to preserve the assumptions of the thermal analyses.

(continued)

LCO (continued)	Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environment at a sufficient rate to maintain fuel cladding and CANISTER component temperatures within design limits.
APPLICABILITY	The LCO is applicable during STORAGE OPERATIONS. Once a CONCRETE CASK containing a CANISTER loaded with spent fuel has been placed in storage, the heat removal system must be OPERABLE to ensure adequate heat transfer of the decay heat away from the fuel assemblies.
ACTIONS	A note has been added to ACTIONS which states that, for this LCO, separate Condition entry is allowed for each CONCRETE CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each CONCRETE CASK not meeting the LCO. Subsequent CONCRETE CASKs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the heat removal system has been determined to be inoperable, it must be restored to OPERABLE status within 8 hours. Eight hours is reasonable based on the accident analysis which shows that the limiting CONCRETE CASK component temperatures will not reach their temperature limits for 24 hours after a complete blockage of all inlet air ducts. <u>B.1</u> SR 3.1.6.1 is performed to document the continuing status of the operability of the CONCRETE CASK Heat Removal System. <u>AND</u> <u>B.2</u> Efforts must continue to restore the heat removal system to OPERABLE status by removing the air flow obstruction(s).

(continued)

ACTIONS (continued)	B.2 (continued) This Required Action must be completed in 12 hours. The Completion Time reflects a conservative total time period without any cooling of 24 hours. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short-term temperature limit for more than 24 hours. It is also unlikely that an unforeseen event could cause complete blockage of all four air inlets and outlets immediately after the last successful Surveillance.
SURVEILLANCE REQUIREMENTS	SR 3.1.6.1 The long-term integrity of the stored fuel is dependent on the ability of the CONCRETE CASK to reject heat from the CANISTER to the environment. The temperature rise between ambient and the CONCRETE CASK air outlets shall be monitored to verify operability of the heat removal system. 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 stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long-term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility. The Frequency of 24 hours is reasonable based on the time necessary for CONCRETE CASK components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of the blockage of the air inlets and outlets.
REFERENCES	1. FSAR Chapter 4 and Chapter 11, Section 11.2.13.

C 3.2 NAC-UMS® SYSTEM Radiation Protection

C 3.2.1 CANISTER Surface Contamination

BASES

BACKGROUND	A TRANSFER CASK containing an empty CANISTER is immersed in the spent fuel pool in order to load the spent fuel assemblies. The external surfaces of the CANISTER are maintained clean by the application of clean water to the annulus of the TRANSFER CASK. However, there is potential for the surface of the CANISTER to become contaminated with the radioactive material in the spent fuel pool water. This contamination is removed prior to moving the CONCRETE CASK containing the CANISTER to the ISFSI in order to minimize the radioactive contamination to personnel or the environment. This allows the ISFSI to be entered without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.
APPLICABLE SAFETY ANALYSIS	The radiation protection measures implemented at the ISFSI are based on the assumption that the exterior surfaces of the CANISTER are not contaminated. Failure to decontaminate the surfaces of the CANISTER could lead to higher-than-projected occupational dose and potential site contamination.
LCO	Removable surface contamination on the exterior surfaces of the CANISTER is limited to 10,000 dpm/100 cm^2 from beta and gamma sources and 100 dpm/100 cm^2 from alpha sources. Only loose contamination is controlled, as fixed contamination will not result from the CANISTER loading process. Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels, which would cause significant personnel skin dose.

(continued)

LCO (continued)

LCO 3.2.1 requires removable contamination to be within the specified limits for the exterior surfaces of the CANISTER. Compliance with this LCO may be verified by direct or indirect methods. The location and number of CANISTER and TRANSFER CASK surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. Accessible portions of the CANISTER are the upper portion of the CANISTER external shell wall accessible after draining of the TRANSFER CASK annulus and the top surface of the structural lid. The user shall determine a reasonable number and location of swipes for the accessible portion of the CANISTER. The objective is to determine a removable contamination value representative of the entire upper circumference of the CANISTER and the structural lid, while implementing sound ALARA practices.

Verification swipes and measurements of removable surface contamination levels on the accessible interior surfaces of the TRANSFER CASK may be performed following transfer of the CANISTER to the CONCRETE CASK. These measurements will provide indirect evidence that the exterior surfaces of the CANISTER do not have removable contamination levels exceeding the limit.

APPLICABILITY

Verification that the exterior surface contamination of the CANISTER is less than the LCO limits is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS and STORAGE OPERATIONS. Measurement of the CANISTER and TRANSFER CASK surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject CANISTER to the ISFSI.

(continued)

ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER LOADING OPERATION. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the removable surface contamination of the CANISTER that has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the CANISTER and bring the removable surface contamination to within limits. The Completion Time of prior TRANSPORT OPERATIONS is appropriate, given that the time needed to complete the decontamination is indeterminate and surface contamination does not affect the safe storage of the spent fuel assemblies.
SURVEILLANCE REQUIREMENTS	SR 3.2.1.1 This SR verifies (either directly or indirectly) that the removable surface contamination on the exterior surfaces of the CANISTER is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification prior to initiating TRANSPORT OPERATIONS in order to confirm that the CANISTER can be moved to the ISFSI without spreading loose contamination.
REFERENCES	1. FSAR Section 8.1. 2. NRC IE Circular 81-07.

C 3.2 NAC-UMS® SYSTEM Radiation Protection

C 3.2.2 CONCRETE CASK Average Surface Dose Rates

BASES

BACKGROUND	The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). 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.
APPLICABLE SAFETY ANALYSIS	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.
LCO	The limits on CONCRETE CASK average surface dose rates are based on the Safety Analysis Report shielding analysis of the NAC-UMS® SYSTEM (Ref. 2). The limits are selected to minimize radiation exposure to the public and to maintain occupational dose ALARA to personnel working in the vicinity of the NAC-UMS® SYSTEM. The LCO specifies sufficient locations for taking dose rate measurements to ensure the dose rates measured are indicative of the effectiveness of the shielding materials.
APPLICABILTY	The CONCRETE CASK average surface dose rates apply during STORAGE OPERATIONS. These limits ensure that the CONCRETE CASK average surface dose rates during STORAGE OPERATIONS are bounded by the shielding safety analyses. Radiation doses during STORAGE OPERATIONS are monitored by the NAC-UMS® SYSTEM user in accordance with the plant-specific radiation protection program as required by 10 CFR 72.212(b)(6) and 10 CFR 20 (Reference 1).
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each loaded CONCRETE CASK. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CONCRETE CASK not meeting the LCO. Subsequent NAC-UMS®

(continued)

ACTIONS (continued)

SYSTEMs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the CONCRETE CASK average surface dose rates are not within limits, it could be an indication that a fuel assembly that did not meet the Approved Contents Limits in Section B2.0 of Appendix B was inadvertently loaded into the CANISTER. Administrative verification of the CANISTER fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a misloaded fuel assembly is the cause of the out-of-limit condition. The Completion time is based on the time required to perform such a verification.

A.2

If the CONCRETE CASK average surface dose rates are not within limits and it is determined that the CONCRETE CASK was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the CONCRETE CASK would result in the ISFSI offsite or occupational calculated doses exceeding regulatory limits in 10 CFR Part 72 or 10 CFR Part 20, respectively. If it is determined that the measured average surface dose rates do not result in the regulatory limits being exceeded, STORAGE OPERATIONS may continue.

B.1

If it is verified that the fuel was misloaded, or that the ISFSI offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the CONCRETE CASK average surface dose rates above the LCO limit, the fuel assemblies must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable, based on the time required to transport the CONCRETE CASK, transfer the CANISTER to the TRANSFER CASK, remove the structural lid and vent and drain port cover welds, perform fuel cooldown operations, cut the shield lid weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

(continued)

SURVEILLANCE REQUIREMENTS	SR 3.2.2.1 This SR ensures that the CONCRETE CASK average surface dose rates are within the LCO limits after transfer of the CANISTER into the CONCRETE CASK and prior to the beginning of STORAGE OPERATIONS. This Frequency is acceptable as corrective actions can be taken before off-site dose limits are compromised. The surface dose rates are measured approximately at the locations indicated on Figure A3-1 of Appendix A of the Amendment 3 Technical Specifications, following standard industry practices for determining average surface dose rates for large containers.
REFERENCES	1. 10 CFR Parts 20 and 72. 2. FSAR Sections 5.1 and 8.2.

C 3.3 NAC-UMS® SYSTEM Criticality Control

C 3.3.1 Dissolved Boron Concentration

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into a PWR spent fuel pool and loaded with fuel assemblies meeting the requirements of the Approved Contents Limits shown in Table B2-2. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid weld is examined, pressure tested, and leak tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is then backfilled with helium. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, the CONCRETE CASK is then moved to the ISFSI. Average CONCRETE CASK dose rates are measured at the ISFSI pad.
APPLICABLE SAFETY ANALYSIS	During loading into, or unloading from, the CANISTER, criticality control of certain PWR fuel requires that the water in the CANISTER contains dissolved boron in a concentration of 1,000 parts per million, or greater. As shown in Table B2-2, spent fuel with the enrichments shown in the "without (w/o) boron" column may be loaded with no assured level of boron in the water in the CANISTER. However, spent fuel with the enrichments shown in the "with boron" column must be loaded or unloaded from the CANISTER when the water in the CANISTER has a boron concentration of 1,000 parts per million or greater. Since boron concentration varies with water temperature, water temperature must be considered in measuring the boron concentration.

(continued)

LCO	The criticality analysis shows that PWR fuel with certain combinations of initial enrichment and fuel content requires credit for the presence of at least 1,000 parts per million of boron in solution in the water in the CANISTER (see Section B3.2.1 for the requirements for assuring soluble boron concentration during loading or unloading). This water must be used to flood the canister cavity during underwater PWR fuel loading or unloading. The boron in the pool water ensures sufficient thermal neutron absorption to preserve criticality control during fuel loading in the basket. Consequently, if boron credit is required for the fuel being loaded or unloaded, the canister must be flooded with water that contains boron in the proper concentration in accordance with the requirements of LCO 3.3.1. Concentration of boron must also be measured and maintained in accordance with LCO 3.3.1. The dissolved boron concentration requirement, and measurement requirement, applies to both the spent fuel pool water and to water in the CANISTER, when pool water is used to fill the CANISTER.
APPLICABILITY	Control of Boron concentration is required during LOADING or UNLOADING OPERATIONS when the CANISTER holds at least one spent fuel assembly that requires dissolved boron for criticality control as described in Table B2-2. This LCO does not apply to spent fuel having an enrichment within the limits specified in the table in the "without (w/o) boron" column.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the required dissolved Boron concentration of the water in the CANISTER is not met, immediate actions must be taken to restore the required dissolved boron concentration. No actions, including continued loading, may be taken that increases system reactivity. <u>AND</u>

(continued)

A.2

The required concentration of dissolved Boron must be immediately restored.

AND

A.3

If the required boron concentration in the water in the CANISTER cannot be established, all fuel assemblies must be removed from the CANISTER within 24 hours to bring the system to a safe configuration. The 24 hour period provides adequate time to restore the required boron concentration.

SURVEILLANCE REQUIREMENTS	SR 3.3.1.1 The assurance of an adequate concentration of dissolved boron in the water in the CANISTER must be established within 4 hours of beginning any LOADING or UNLOADING OPERATION, using two independent measurements of determining boron concentration. During LOADING or UNLOADING OPERATIONS, verification of continued adequate dissolved boron concentration must be performed every 48 hours after the beginning of operations. The 48-hour boron concentration verification is not required when no water is being introduced into the CANISTER cavity. In this situation, no potential exists for the boron in the CANISTER to be diluted, so verification of the boron concentration is not necessary.
REFERENCES	Section B3.2.1 and Table B2-2.

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13.0 QUALITY ASSURANCE

13.1 Introduction

The NAC International (NAC) Quality Assurance (QA) Program is designed and administered to meet all Quality Assurance criteria of 10 CFR 72, Subpart G [1], 10 CFR 50, Appendix B [2], 10 CFR 71, Subpart H [3], and NQA-1 (Basic and Supplemental Requirements) [4]. The program is defined in a QA Program description document that has been reviewed and approved by the Nuclear Regulatory Commission (Approval No. 0018).

The NAC Quality Assurance Program is described in a Quality Assurance Manual. This Quality Assurance Manual, as approved by the company's President and Chief Executive Officer, contains policy as to how NAC intends to comply with the applicable regulatory QA criteria. Detailed implementing quality procedures are used to provide the procedural direction to comply with the policy of the QA Manual.

Employing a graded methodology, as described in USNRC Regulatory Guide 7.10 [5], NAC applies quality controls to items and activities consistent with their safety significance. Table 13.1-1 identifies the NAC Quality Assurance Manual sections, which address the applicable quality criteria.

A synopsis of the NAC Quality Assurance Program is presented in Section 13.2.

Table 13.1-1 Correlation of Regulatory Quality Assurance Criteria to NAC Quality Assurance Program

Regulatory Quality Assurance Criteria*		Corresponding NAC QA Manual Section Number
I.	Organization	1
II.	Quality Assurance Program	2
III.	Design Control	3
IV.	Procurement Document Control	4
V.	Procedures, Instructions, and Drawings	5
VI.	Document Control	6
VII.	Control of Purchased Items and Services	7
VIII.	Identification and Control of Material, Parts and Components	8
IX.	Control of Special Processes	9
X.	Inspection	10
XI.	Test Control	11
XII.	Control of Measuring and Test Equipment	12
XIII.	Handling, Storage and Shipping	13
XIV.	Inspection, Test and Operating Status	14
XV.	Control of Nonconforming Items	15
XVI.	Corrective Action	16
XVII.	Records	17
XVIII.	Audits	18

*The criteria are obtained from 10 CFR 50 Appendix B; 10 CFR 71 Subpart H; and 10 CFR 72 Subpart G.

13.2 NAC Quality Assurance Program Synopsis

Eighteen applicable Quality Assurance criteria are identified in 10 CFR 72, Subpart G; 10 CFR 50, Appendix B; 10 CFR 71, Subpart H; and ASME NQA-1 (Basic and Supplemental Requirements). NAC compliance with each of these criteria is addressed below.

13.2.1 Organization

The President and Chief Executive Officer of NAC has the ultimate authority and responsibility over all organizations and their functions within the corporation. However, the President delegates and empowers qualified personnel with the authority and responsibility over selected key areas, as identified in the NAC Organization Chart, Figure 13.2-1.

The Vice President, Quality, is responsible for definition, development, implementation and administration of the NAC Quality Assurance Program. The Quality Assurance organization is independent from other organizations within NAC and has complete authority to assure adequate and effective program execution, including problem identification, satisfactory corrective action implementation and the authority to stop work, if necessary. The Vice President, Quality, reports directly to the President and Chief Executive Officer of NAC. The Vice President, Quality, has sufficient expertise in the field of quality to direct the quality function and will be capable of qualifying as a lead auditor.

Strategic Business Unit (SBU) Vice Presidents direct operations, utilizing project teams as appropriate for a particular work scope. SBU Vice Presidents are responsible to the President and Chief Executive Officer for the proper implementation of the NAC Quality Assurance Program.

13.2.2 Quality Assurance Program

NAC has established a Quality Assurance Program that meets the requirements of 10 CFR 72, Subpart G, 10 CFR 50 Appendix B, 10 CFR 71, Subpart H, and NQA-1. Employing a grading methodology consistent with U.S. NRC Regulatory Guide 7.10, the Quality Assurance Program provides control over activities affecting quality from the design to fabrication, operation, and maintenance of nuclear products and services for nuclear applications. The Quality Assurance Program is documented in the Quality Assurance Manual and implemented via Quality Procedures. These documents are approved by the Vice President, Quality, and the President and

Chief Executive Officer, as well as the Vice President from each SBU performing activities within the scope of the NAC Quality Assurance Manual.

Personnel assigned responsibilities by the Quality Assurance Program may delegate performance of activities associated with that responsibility to other personnel in their group when those individuals are qualified to perform those activities by virtue of their education, experience and training. Such delegations need not be in writing. The person assigned responsibility by the Quality Assurance Program retains full accountability for the activities.

13.2.3 Design Control

The established Quality Procedures covering design control assure that the design activity is planned, controlled, verified and documented so that applicable regulatory and design basis requirements are correctly translated into specifications, drawings, and procedures with appropriate acceptance criteria for inspection and test delineated.

When computer software is utilized to perform engineering calculations, verifications of the computational accuracy are performed, and error tracking of the software is controlled in accordance with approved Quality Procedures.

Design interface control is established and adequate to assure that the review, approval, release, distribution and revision of design documents involving interfaces are performed by appropriately trained, cognizant design personnel using approved procedures.

Design verification is performed by individuals other than those who performed the original design. These verifications may include design reviews, alternate calculations or qualification tests. Selection of the design verification method is based on regulatory, contractual or design complexity requirements. When qualification testing is selected, the "worst case" scenario will be utilized. The verification may be performed by the originator's supervisor, provided the supervisor did not specify a singular design approach, rule out certain design considerations, or establish the design inputs used in the design, unless the supervisor is the only individual in the organization competent to perform the verification. When verification is provided by the supervisor, the need shall be so documented in advance and evaluated after performance by internal audit.
Design changes are controlled and require the same review and approvals as the original design.

13.2.4 Procurement Document Control

Procurement documents and their authorized changes are generated, reviewed and approved in accordance with the Quality Procedures. These procedures assure that all purchased material, components, equipment and services adhere to design specification, regulatory and contractual requirements including Quality Assurance Program and documentation requirements.

NAC Quality Assurance personnel review and approve all purchase orders invoking compliance with the Quality Assurance Program for inclusion of quality related requirements in the procurement documents.

13.2.5 Procedures, Instructions, and Drawings

All activities affecting quality are delineated in the Quality Procedures, Specifications, Inspection/Verification Plans or on appropriate drawings. These documents are developed via approved Quality Procedures and include appropriate quantitative and qualitative acceptance criteria. These documents are reviewed and approved by Quality Assurance personnel prior to use.

13.2.6 Document Control

All documents affecting quality, including revisions thereto, are reviewed and approved by authorized personnel, and are issued and controlled in accordance with Quality Procedures by those persons or groups assigned responsibility for the document to be controlled. Transmittal forms, with provisions for receipt acknowledgment, are utilized and controlled document distribution logs are maintained.

All required support documentation for prescribed activities is available at the work location prior to initiation of the work effort.

13.2.7 Control of Purchased Items and Services

Items and services affecting quality are procured from qualified and approved suppliers. These suppliers have been evaluated and selected in accordance with the Quality Procedures based upon their capability to comply with applicable regulatory and contractual requirements.

Objective evidence attesting to the quality of items and services furnished by NAC suppliers is provided with the delivered item or service, and is based on contract requirements and item or service complexity. This vendor documentation requirement is delineated in the procurement documents.

Source inspection, receipt inspection, vendor audits and vendor surveillance are performed as required to assure product quality, documentation integrity, and supplier compliance to the procurement, regulatory and contractual requirements.

13.2.8 Identification and Control of Material, Parts, and Components

Identification is maintained either on the item or in quality records traceable to the item throughout fabrication and construction to prevent the use of incorrect or defective items.

Identification, in accordance with drawings and inspection plans, is verified by Quality Assurance personnel prior to releasing the item for further processing or delivery.

13.2.9 Control of Special Processes

Special processes, such as welding, heat treating and nondestructive testing, are performed in accordance with applicable codes, standards, specifications and contract requirements by qualified personnel. NAC and NAC suppliers' special process procedures and personnel certifications are reviewed and approved by NAC Quality Assurance prior to their use.

13.2.10 Inspection

NAC has an established and documented inspection program that identifies activities affecting quality and verifies their conformance with documented instructions, plans, procedures and drawings.

Inspections are performed by individuals other than those who performed the activity being inspected. Inspection personnel report directly to the Vice President, Quality.

Process monitoring may also be used in conjunction with identified inspections, if beneficial to achieve required quality.

Mandatory inspection hold points are used to assure verification of critical characteristics. Such hold points are delineated in appropriate process control documents.

13.2.11 Test Control

NAC testing requirements are developed and applied in order to demonstrate satisfactory performance of the tested items to design/contract requirements.

The NAC test program is established to assure that preoperational or operational tests are performed in accordance with written test procedures. Test procedures developed in accordance with approved Quality Procedures identify test prerequisites, test equipment and instrumentation and suitable environmental test conditions. Test procedures are reviewed and approved by NAC Quality Assurance personnel.

Test results are documented, evaluated and accepted by qualified personnel as required by the Quality Assurance inspection instructions prepared for the test, as approved by cognizant quality personnel.

13.2.12 Control of Measuring and Testing Equipment

Control of measuring and testing equipment/instrumentation is established to assure that devices used in activities affecting quality are calibrated and properly adjusted at specified time intervals to maintain their accuracy.

Calibrated equipment is identified and traceable to calibration records, which are maintained. Calibration accuracy is traceable to national standards when such standards exist. The basis of calibration shall always be documented.

Whenever measuring and testing equipment is found to be out of calibration, an evaluation shall be made and documented of the validity of inspection or test results performed and of the acceptability of items inspected or tested since the previous calibration.

13.2.13 Handling, Storage and Shipping

Requirements for handling, storage and shipping are documented in specifications and applicable procedures or instructions. These requirements are designed to prevent damage or deterioration to items and materials.

Information pertaining to shelf life, environment, packaging, temperature, cleaning and preservation are also delineated as required.

Quality Assurance Surveillance/Inspection personnel are responsible for verifying that approved handling, storage, and shipping requirements are met.

13.2.14 Inspection, Test and Operating Status

Procedures are established to indicate the means of identifying inspection and test status on the item and/or on records traceable to the item. These procedures assure identification of items that have satisfactorily passed required inspections and/or tests, to preclude inadvertent bypassing of inspection/test.

Inspection, test, and operating status indicators may only be applied or modified by Quality Assurance personnel or with formal Quality Assurance concurrence.

13.2.15 Control of Nonconforming Items

NAC has established and implemented procedures that assure appropriate identification, segregation, documentation, notification and disposition of items that do not conform to specified requirements. These measures prevent inadvertent usage of the item and assure appropriate authorization or approval of the item's disposition.

All nonconformances are reviewed and accepted, rejected, repaired or reworked in accordance with documented approved procedures. If necessary, a Review Board is convened, consisting of engineering, licensing, quality, operations and testing personnel to provide disposition of nonconforming conditions.

NAC procurement documents provide for control, review and approval of nonconformances noted on NAC items, including associated dispositions.

13.2.16 Corrective Action

Conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material/ equipment, and nonconformances are promptly identified, documented and corrected.

Significant conditions adverse to quality will have their cause determined and sufficient corrective action taken to preclude recurrence. These conditions are documented and reported to the Vice President, Quality, who assures awareness by the President and Chief Executive Officer.

13.2.17 Records

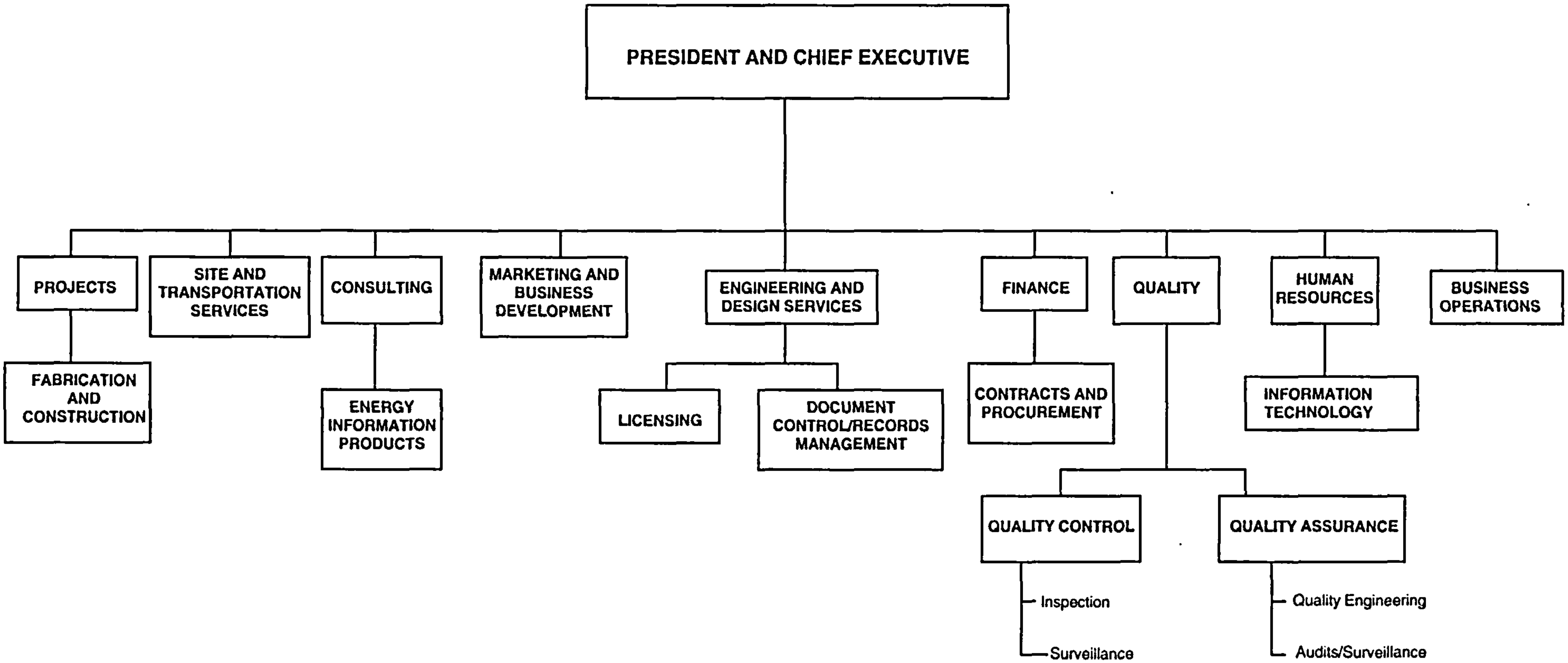
NAC maintains a records system in accordance with approved procedures to assure that documented objective evidence pertaining to quality related activities is identifiable, retrievable and retained to meet regulatory and contract requirements, including retention duration, location and responsibility.

Quality records include, but are not limited to, inspection and test reports, audit reports, quality personnel qualifications, design documents, purchase orders, supplier evaluations, fabrication documents, nonconformance reports, drawings, specifications, etc. Quality Assurance maintains a complete list of records and provides for record storage and disposition to meet regulatory and contractual requirements.

13.2.18 Audits

Approved Quality Procedures provide for a comprehensive system of planned and periodic audits performed by qualified personnel, independent of activities being audited. These audits are performed in accordance with written procedures and are intended to verify program adequacy and its effective implementation and compliance, both internally and at approved-supplier locations. Internal audits are conducted annually, and approved suppliers are audited on a triennial basis, as a minimum.

Figure 13.2-1 NAC Organization Chart



13.3 References

1. U.S. Code of Federal Regulations, "Quality Assurance," Part 71, Title 10, Subpart H.

2. U.S. Code of Federal Regulations, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Part 50, Title 10, Appendix B.

3. U.S. Code of Federal Regulations, "Quality Assurance Requirements," Part 72, Title 10, Subpart G.

4. ASME NQA-1-1994, Part 1, Basic and Supplemental Requirements (as referenced by the ASME Code, including latest accepted addenda), Quality Assurance Program Requirements for Nuclear Facility Applications.

5. U.S. Nuclear Regulatory Commission, "Establishing Quality Assurance Program for Packaging Used in the Transport of Radioactive Material," Regulatory Guide 7.10, Revision 1, June 1986.

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