72-1007

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June 30, 2005
BFS/NRC 05-014
Docket No. 72-1007
File Nos. WEP-09
WEP01.26

ATTN: Document Control Desk
Director, Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: **Amendment Request for the VSC-24 Ventilated Storage System Certificate of Compliance**

References:
1) Certificate of Compliance for Spent Fuel Storage Casks, Certificate No. 1007, Docket No. 72-1007, Amendment 5, Package Identification No. USA/72-1007.
2) Letter from BNFL Fuel Solutions to USNRC, "Notice of Company Name Change," BFS/NRC 05-009, April 12, 2005.

Dear Sir or Madam:

BNG Fuel Solutions Corporation (BFS) hereby submits an application to amend the VSC-24 Ventilated Storage System Certificate of Compliance (CoC No. 1007) (Reference 1) in accordance with the requirements of 10 CFR 72.244. The license amendment request (LAR 1007-006) requests to eliminate Technical Specification (TS) 1.3.4, which requires daily temperature measurement of the cask. Daily temperature measurement is not required because the daily visual inspection of the cask inlet and outlet vent screens required by TS 1.3.1 provides the capability to determine when corrective action needs to be taken to maintain safe storage conditions in accordance with the requirements of 10 CFR 72.122(h)(4).

In addition, the LAR incorporates editorial changes associated with the company name change (Reference 2) that have been made in accordance with the BFS Quality Assurance Program.

A summary of the changes requested by LAR 1007-006 and the reasons for the changes are provided in the Attachment to this letter.

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Enclosed are two (2) copies of a CD-ROM containing an electronic file of LAR 1007-006. The LAR is being filed electronically in accordance with the requirements of 10 CFR 72.4. Please note that all of the enclosed information is publicly available.

Should you or any member of your staff have any questions, please contact the undersigned at:

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Sincerely,

Steven E. Sisley
Licensing/Regulatory Compliance Manager

Attachment:
Summary of Changes for VSC-24 Ventilated Storage Cask System LAR 1007-006

Enclosure:
CD-ROM, labeled "VSC-24 Ventilated Storage Cask System LAR 1007-006, June 30, 2005 – Publicly Available" (2 copies), containing the following file:

001_LAR1007006_R0.pdf, 7,801 KB, publicly available

cc: Mr. J. Cuadrado, NRC NMSS, w/ attachment and enclosure
Mr. R. J. Lewis, NRC NMSS w/ attachment

Summary of Changes for VSC-24 Ventilated Storage Cask System
LAR 1007-006

Section	Page	Description
1.2.1.1	1-4	Editorial revision for company name change
1.2.1.5	1-7	Editorial revision for company name change
1.3	1-8	Editorial revision for company name change
2.3.3.2	2-9	Revise method of thermal performance verification consistent with proposed TS monitoring requirements
3.4.3.2	3-5	Editorial revision for company name change
7.1.5	7-4	Revise method of thermal performance verification consistent with proposed TS monitoring requirements
10.1.1	10-1	Editorial revision for company name change
12.3	12-2	Editorial revision for company name change
TS TOC	---	Revise to remove deleted sections
TS 1.2.3	TS-21	Revise TS to specify method of thermal performance verification
TS 1.3.4	TS-45	Delete
TS Table 3	TS-46	Delete TS 1.3.4 from summary of surveillance requirements
B.1.3.4	TS-67	Delete
13.0	13-1	Editorial revision for company name change
14.2	14-4	Editorial revision for company name change

Docket 72-1007
LAR 1007-006, Revision 0

FINAL SAFETY ANALYSIS REPORT
FOR THE
VSC-24 VENTILATED STORAGE CASK SYSTEM

Prepared by:

BNG FUEL SOLUTIONS CORPORATION
CAMPBELL, CALIFORNIA

June 2005

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1.0 GENERAL DESCRIPTION

1.1 INTRODUCTION

The Ventilated Storage Cask (VSC-24) System is a second-generation dry storage system using a concrete storage cask and a steel, seal-welded basket to safely store irradiated nuclear fuel. The Multi-assembly Sealed Basket (MSB) is stored in the central cavity of the Ventilated Concrete Cask (VCC). The VCC is ventilated by internal airflow paths, which allow the decay heat to be removed by natural circulation around the metal basket wall. Figure 1.1-1 pictorially shows the major components of the VSC-24 System.

The VSC System can be sized to hold from 4 to 24 Pressurized Water Reactor (PWR) assemblies. For this Final Safety Analysis Report (FSAR), a VSC-24 system that holds 24 PWR assemblies has been chosen for analysis. This system takes full advantage of the economies of scale and is the predominant size preferred by utilities where site-specific conditions do not limit the storage cask size.

The *technical specifications* for the fuel to be stored in the VSC-24 System, including heat generation rate, radiation source strength, and neutron multiplication factor (k_{eff}), may be met by a wide variety of combinations of enrichment, burnup, and cooling time. The exact fuel specifications that serve as the basis for this FSAR are listed in Section 2.1. Generally, the VSC-24 System can accommodate fuel that has received its design basis burnup without further analysis.

The VSC-24 System has a number of unique design features. Among these are the vertical loading of the MSB into the VCC. This is accomplished by employing the MSB Transfer Cask (MTC) (or shielding bell) to move a loaded MSB from the fuel pool to the VCC (located in the fuel building truck bay). This vertical loading greatly simplifies the transfer of fuel to a storage cask (compared to horizontally loaded systems). The potential for spreading of contamination is minimized because the vessel that goes in the pool (the MTC) is kept inside the fuel and auxiliary building at all times. Vertical loading also allows the transfer to be conducted within the controlled environment of the truck bay instead of outside on the storage pad.

Another important feature is the simplified fabrication of the MSB using mostly welded components, minimal machining, a minimum of horizontal surfaces to hold water during drying operation, and high-strength pressure vessel steel.

The VCC has chamfered edges to mitigate potential damage due to a cask drop. These chamfered edges eliminate the sharp corners at the cask top and bottom, where chipping, spalling, and loss of material predominately occur in a drop accident. The chamfered edges are reinforced and greatly increase the impact area, thus spreading the load throughout a much larger section of the cask and reducing (if not preventing) concrete material loss from a drop accident.

The VSC-24 handling system employs a hydraulic roller skid and a heavy haul trailer (or alternatively, an engineered cask transporter, as described in Appendix B) in addition to the MTC. The MTC is used to move a filled MSB from the fuel pool to the concrete cask, and the hydraulic roller skid (or transporter) is used to move the concrete cask on and off the storage pad onto a heavy haul trailer. Moving all 24 assemblies at one time saves many hours of fuel handling and the associated radiation dosage to personnel. This handling system is shown pictorially in Figure 1.1-2. Only the MSB, VCC, and MTC are important to safety. They are designed to withstand any failure of the handling equipment. It should be noted that the handling system shown and discussed here is only one of many possible handling systems that may be selected by a utility. The handling equipment, including the transfer cask lifting yoke, is addressed on a site-specific basis in site safety reviews.

The VSC-24 System has been designed and analyzed for a lifetime of 50 years. However, it is expected that the VSC-24 components will significantly out-perform this conservative analysis. It is expected that life extension will be possible with future inspections.

The MSB is designed by analysis to meet material and stress requirements of the American Society of Mechanical Engineers (ASME) Code, Section III, Division 1. The VCC is designed by analysis to meet the American Concrete Institute Code ACI-349 and the American Nuclear Society ANS-57.9. The MTC is designed as a lifting device to meet NUREG-0612 and ANSI N14.6 and as a shielding bell for radiological safety. Chapter 2 further describes the design criteria for the VSC-24 System. The following subsections provide a general description.

1.2 GENERAL DESCRIPTION OF THE STORAGE CASK

1.2.1 CASK SYSTEM CHARACTERISTICS

The VSC-24 System includes:

1. Ventilated Concrete Cask (VCC)
2. 24-Assembly Multi-Assembly Sealed Basket (MSB)
   (placed inside the central cavity of the VCC)
3. One MSB Transfer Cask (MTC)
   (used to move the loaded MSB from the spent fuel pool to the concrete cask)
4. One Hydraulic Roller Skid*
   (used to move the concrete cask on and off the heavy haul transfer trailer)
5. One Heavy Haul Transfer Trailer*
   (for cask transport from the auxiliary building to the storage location)
6. Vacuum Drying and Helium Back-fill System with a Helium Sniffer for Leak Detection
7. Automatic (or Manual) Welding Equipment

* The handling system discussed here is only one of many possible systems that may be selected by a utility. Such systems are to be addressed by the utility on a site-specific basis.

The following subsections describe the proposed hardware and its operation.

The overall design criteria of the VSC-24 System are shown in Table 1.2-1. The design accounts for both normal and off-normal conditions, including a range of credible and postulated accidents. The system design and analyses are performed in accordance with Title 10, Code of Federal Regulations, Part 72 (10CFR72), ANS-57.9-1984, and the applicable sections of the ASME and ACI codes. Other applicable ANSI, ANS, AWS, and ASTM standards and NRC regulatory guides are also used in the VSC-24 design. These codes, standards, and regulatory guides are discussed in the applicable sections describing the component designs.

The MSB is also designed to be compatible with future shipping casks. Therefore, a hypothetical drop accident is evaluated as the structurally limiting condition for the MSB. The MSB is first designed to accommodate hypothetical drop accident loads. Stresses from other loads (thermal, internal pressure, etc.) are then determined. The loads are combined as required per the ASME Code, Section III, and the material stress allowables are shown to bound all combined loads.

The shielding and thermal load requirements are the structurally determining criteria for the VCC. Once the MSB design has been shown to provide adequate shielding and thermal load capacity, calculations are then performed to show adequate margin of safety for the remaining loading conditions (e.g., seismic, tornado, flood). The applicable load combinations are determined (per ANS-57.9) and compared to the allowable stresses. The concrete cask also meets the load combinations in ACI-349 (these are slightly different from those of ANS-57.9.)

The VSC-24 system employs several features that facilitate decontamination. The metal surfaces of VSC-24 cask system components are coated with industry standard coatings, such as Carbo-Zinc, Dimetcote 6, or the equivalent (as is discussed in more detail for the MSB in Section 1.2.1.1). These coatings prevent corrosion of the metal components, as well as facilitating decontamination. However, significant contamination of the radial surfaces of the MSB exterior and MTC interior is not expected because a continuous flow of demineralized water is maintained in the annulus between the MSB shell and the MTC inner liner during fuel loading operations. Another feature of the VSC-24 system that facilitates decontamination is placement of ceramic tiles on the VCC bottom plate. These ceramic tiles prevent the MSB bottom plate from contacting the VCC bottom plate, thereby preventing galvanic reactions and potential contamination of the VCC bottom plate.

1.2.1.1 Description of the MSB

Design

The 24 PWR element MSB consists of an outer shell assembly, a shielding lid, a structural lid, and the fuel basket assembly. The MSB is shown in Figure 1.1-1. General arrangement drawings of the MSB are provided in Section 1.5. Table 1.2-2 summarizes the main physical design parameters. Each major component of the MSB is described in the paragraphs below.

The MSB shell is fabricated from SA-516 Gr 70 pressure vessel steel. This material shall be low-carbon, low-sulfur, calcium-treated and vacuum-degassed.* The MSB shell is 62.5 inches in diameter with a 1.0-inch wall thickness. The MSB length depends on fuel type, with or without control elements, and varies from 164.2 to 192.25 inches.

The shell and the internals are coated to prevent detrimental effects from the fuel pool water chemistry during the four to eight hours the MSB is in the pool for loading. A coating with a proven history of nuclear applications and compatible with existing site painting requirements and pool chemistry requirements will be used. Everlube 812, Everlube 823, Carbo-Zinc 11, Dimetcote 6, or an equivalent coating is recommended. These are radiation-resistant hard film coatings that can withstand high temperatures. The shell thickness was designed to withstand more than 50 years of corrosion in an uncoated condition in a coastal, marine environment. The exterior is coated, however, so that no corrosion is expected.

The pressure boundary is designed to withstand a hypothetical cask tip-over or drop without loss of containment.

The fuel basket is a welded assembly fabricated from square steel tubes. Structural support in the horizontal direction is provided by the curved horizontal support assemblies located at each end and at the center of the basket assembly. All material is SA-516 Gr 70 or equivalent. The coating is also applied to the interior basket to protect the fuel pool chemistry. The basket is designed by analysis using the ANSYS computer code to demonstrate that it can take the horizontal drop loads without deformation that would cause a fuel assembly to be damaged or constrained.

The basic MSB design is for fuel that has received up to its design basis burnup. Therefore, criticality control is by administrative procedures and by taking credit for the negative reactivity effect due to the steel in the basket assembly. The administrative controls include limitations on the boron concentration in the MSB cavity and a minimum burnup for analyzed fuel types other than CE 16x16 fuel. For CE 16x16 fuel, the administrative controls include a minimum boron concentration relative to initial enrichment, rather than reliance on burnup. The boron concentration will be maintained at the level which would provide subcritical configuration if the MSB were loaded with unburned fuel. The calculations showing the criticality safety of the VSC-24 System are provided in Chapters 6 and 11 of this FSAR.

The shield lid is made of steel plate and RX-277 neutron shielding. To prevent binding during installation, the diameter and the thickness of the shield lid are such that the diagonal from the bottom edge to the top edge through the center is less than the MSB internal diameter. The shield lid sits on the shield lid support ring and is held securely in place between the support ring and the structural lid. Two penetrations for draining, vacuum drying, and backfilling with helium are also located in the shield lid.

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* For material that has already been procured prior to issuance of Amendment 2, the top 4 inches of the MSB shell inside shall be inspected for flaws and defects by acid etching and the MSB shell shall be UT inspected in accordance with ASTM A435. Alternatively, the MSB shell may be UT inspected in accordance with ASME Section III, Division 1, NB-2532.1, 1986 Edition with the 1988 Addenda, or later Editions or Addenda.

The shield lid is placed in the MSB body after the fuel is loaded. A guide tube is screwed into a threaded hole on the backside of the draining penetration. This tube is installed at the pool surface. The tube reaches to within 1/16 inch of the MSB bottom to facilitate removal of the water from the MSB after fuel loading. The structural lid is a 3-inch thick steel disk that has a penetration for access to the fittings. This penetration is sealed via multiple welds once the helium backfill process has been completed. The MSB bottom plate is 0.75 inches thick. The bottom plate is welded to the shell in the fabrication shop. The shield and structural lids are welded to the MSB shell after the fuel is inserted.

Fabrication

The detailed fabrication requirements are described in the general arrangement drawings in Section 1.5. The major points of fabrication are listed in Table 1.2-3. Additional details are also provided in Section 7.1.

1.2.1.2 Description of the VCC

Design

The VCC is shown in Figure 1.1-1. General arrangement drawings of the VCC are included in Section 1.5. Table 1.2-4 lists the major physical design parameters. The VCC provides structural support, shielding, and natural convection cooling for the MSB. The steel and concrete walls of the cask are sufficiently thick to limit side surface radiation dose rates to less than 20 mrem/hr.

The internal cavity of the VCC is formed by a thick steel cylinder. The concrete is Type II Portland Cement, 144 lb/ft^3, 4000 psi concrete. Outer and inner re-bar cages are formed by vertical hook bars and horizontal ring bars. The concrete mix has been specifically selected to assure strength and long life at the elevated temperatures expected during normal operations (100 to 200°F) and the higher short-term temperatures that could potentially occur during off-normal and accident conditions. Specific properties selected are the use of Type II Portland Cement, matching of the aggregate's and its carrier's thermal expansion coefficients, and low water/cement ratio.

The air flow path is formed by the skid channels at the bottom (air entrance), the air inlet ducts, the gap between the MSB exterior and the concrete cask interior, and the air outlet ducts. The air inlet and outlet vents are steel lined penetrations that take non-planar paths to minimize radiation streaming. The cask cover plate is a thick plate which provides additional shielding to reduce the skyshine radiation and provides a cover and seal to protect the MSB from the environment and postulated tornado missiles. The cover is bolted in place and has tamper indications on two of the nuts. The bottom of the VCC has a quarter-inch thick steel plate which covers the entire bottom and prevents any loss of material during a bottom drop accident. Also, as previously pointed out, the VCC has reinforced chamfered corners at the top and bottom to minimize the damage due to a drop.

Construction

The VCC is constructed by pouring concrete between a re-usable form and the inner metal liner. Reinforcing bars are used at the inner and outer concrete surfaces. The air flow embedments are inserted into the inner shell and tied to the outer re-bar frame and outer form. Normal density, Type II Portland cement concrete is used. This construction can be performed at the reactor site or at a contractor's facility. It is recommended that a local constructor be used to build the VCC on a 50 ft x 50 ft "construction section" of the Independent Spent Fuel Storage Installation (ISFSI) storage pad. This construction section will be outside of the security fence. However, the VCC can also be constructed off-site at a local contractor's facility and "heavy hauled" to the plant site.

The VCC construction is detailed in the general arrangement drawings in Section 1.5. Table 1.2-5 provides a summary of the VCC construction requirements.

1.2.1.3 Description of the MTC

Table 1.2-6 gives the important design parameters of the MTC. General arrangement drawings of the MTC are provided in Section 1.5. The MTC consists of a cylinder (the length is site-specific and depends on the MSB length) with a steel-lead-steel-RX-277 neutron shield-steel sandwich wall. The thick-walled cylinder reduces the dose rate to below 250 mrem/hr at one meter from its surface. The top cover of the MTC extends over the MSB to prevent the MSB from being inadvertently lifted out of the top of the MTC while being lowered into the VCC. The MTC has a movable shield door at the lower end to allow lowering of the MSB into the VCC. The doors slide in steel guides along each side of the cask. Four steel pins are used to prevent inadvertent opening of the doors. Hydraulic pistons are used to open the doors for the MSB transfer. The yoke provided with the MTC is used to interface with the existing cask crane.

The MTC is a special lifting device designed and fabricated to the requirements of NUREG-0612 and ANSI N14.6 so that a crane designed to the same requirements can be used during transfer operation without a cask drop being considered (per NUREG-0612). If the cask is used with a crane which does not meet NUREG-0612, site-specific heavy loads requirements will be followed to conform to the existing 10CFR50 license.

During operation, an empty MSB is inserted into the MTC and filled with borated water. Steel shielding segments are placed in the top of the MSB-MTC gap. The gap between the inner MTC surface and the outer MSB surface is filled with clean demineralized water as the cask is being lowered into the pool. A water supply hose is connected to the side of the MTC. This allows clean water to be injected into the MSB-MTC gap during the entire time it is submerged in the pool. This prevents the outside of the MSB from becoming contaminated from contact with the pool water and loosened crud from the fuel assemblies. After loading fuel into the MSB, the MTC containing the loaded MSB with the shield lid is lifted with the shield lid from the pool and placed in the decontamination area. The MSB is then seal welded, dried, backfilled with helium, and structurally welded.

At this point the MTC and its MSB payload are moved from the decontamination area to the top of the VCC. The bottom doors of the MTC are unpinned and the MTC supports are placed into position. Next the hydraulically operated lower shield doors are opened and the MSB is lowered into the VCC.

1.2.1.4 Description of the Roller Skid

The loaded VCC is too large to be lifted by most in-plant cranes. Hence, a roller skid with hydraulic lifting cylinders (or engineered transporter vehicle) is used to move the cask.

The skid is composed of two steel forks which ride on eight 50-ton Hilman rollers. The roller skid also has a detachable, rear stabilizing beam and a permanently mounted front beam. The rollers are capable of moving the loaded VCC over smooth metal or concrete surfaces.

With the hydraulic pistons lowered and the rear stabilizing beam removed, the two forks of the skid can be rolled under the loaded VCC through the two skid channels located at the cask base. Once the skid is under the cask, the rear stabilizing beam is re-bolted to the protruding forks. When the hydraulic pistons are energized, they lift the cask two inches above the floor. In this position, the skid and its cask load can be towed or pushed with a heavy haul truck (or other suitable vehicle). The skid is removed from the cask by lowering the pistons, removing the rear beam, and pulling the skid from underneath the cask. The use of the Hilman rollers in this manner to lift and move heavy loads is a standard practice among fabricators and constructors of large equipment.

If an engineered transporter is used, the vehicle would be similar to those previously developed for metal casks. Trunnions or other lifting devices would also have to be provided on the VCC.

1.2.1.5 Transfer Trailer

A number of different transfer trailers can be used for the on-site movement of the loaded VCC from the fuel building to the ISFSI pad. A three-axle trailer with 24 foam-filled tires that can support the stationary weight of the VCC, MTC, loaded MSB, and auxiliary equipment is recommended. The trailer has stabilizing/load spreading jacks which will spread the floor loadings to below the truck bay limits to prevent vertical trailer movement during loading or unloading of a VCC.

1.2.2 OPERATIONAL FEATURES

Fuel assemblies are stored in the VSC-24 System according to the following sequence of main events:*

1. Position transfer cask containing empty MSB in fuel pool.

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* See Chapter 8 for a more detailed presentation of VSC system operations.

2. Load fuel assemblies in the MSB.
3. Place shielding lid on MSB and use transfer cask to remove MSB from fuel pool.
4. Close MSB. (See discussion in Chapter 8.)
5. Transfer MSB to concrete cask.
6. Close VCC.
7. Move VCC to storage location via truck trailer.
8. Position VCC at storage site via hydraulic roller skid.

Safe storage of the irradiated fuel in the VSC-24 System is provided by the removal of decay heat by convection, radiation, and conduction from the fuel rods to the MSB shell wall and the subsequent natural convection in the MSB-VCC annulus. Radiation exposure to site personnel is limited by the steel and concrete shielding. VSC-24 System operation is totally passive. No active systems are required.

The handling equipment required to implement the VSC-24 System is site-specific. This equipment includes an overhead handling crane at the reactor fuel pool, one transfer cask, one hydraulic roller skid, one yoke, and one transfer trailer. All equipment is designed and tested to applicable government and industrial standards and will be maintained and operated to the owner's specifications.

1.2.3 CASK CONTENTS

The fuel to be stored in the VSC-24 System is described in Section 2.1 of this FSAR. In addition to the fuel, the MSB will contain a helium cover gas.

1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

The owner of the design and Certificate of Compliance for the VSC-24 Ventilated Storage System is BNG Fuel Solutions Corporation (BFS). BFS is a wholly owned subsidiary of BNG America. All design and specification activities including quality assurance services continue to be performed by BFS. Fabrication of the steel components (MSB, MTC) will be by a qualified steel fabrication shop chosen through a competitive bidding process. Fabrication of the VCC will be performed by a local concrete contractor, also chosen by the bid process. BFS will retain full responsibility and control over all design, analysis, and fabrication activities.

Additional agents/contractors that may build and deliver VSC-24 systems and components are Wisconsin Electric Power Company of Milwaukee, Wisconsin. Wisconsin Electric Power Company will only build the VSC-24 systems and components for their own use and will build them under their own 10 CFR 50 Appendix B QA Program. BFS will build casks and components for other users under the BFS Manual of Quality Assurance referenced in Chapter 13 of this FSAR.

1.4 GENERIC CASK ARRAYS

A typical ISFSI storage pad layout is provided in Figure 1.4-1. This layout is for 69 casks. As Figure 1.4-1 shows, the pad has three major sections: the truck/trailer loading area, the cask construction area, and the cask storage area. Casks are placed in the vertical position on the pad in linear arrays as defined by the owner utility. Actual array sizes could range from 20 to more than 200 total casks. Figure 1.4-1 shows typical spacing and overall site dimensions. However, these are heavily dependent on the general site layout, access roads, site boundaries and transfer equipment selection.

The reinforced concrete foundation of the storage pad is capable of handling the transient loads from the roller skid and the general loads of the stored cask. Furthermore, the pad can be poured in phases to specifically meet utility required expansions. However, any number of alternative layouts and expansion plans are available.

Table 1.2-1 - Summary of VSC-24 System Design Criteria (2 pages)

Component: VCC (Major Design Codes: ANS-57.9, ACI 349-80)

Design Load Type	Design Parameters	Applicable Codes
Design Basis Tornado (DBT)	Maximum wind pressures Maximum speeds Maximum differential pressures	NRC Reg. Guide 1.76 ASCE 7-93
DBT Missile Types:	Maximum speed Automobile 8 in dia. shell 1 in solid sphere	NUREG-0800 Section 3.5.1.4
Flood	Maximum water height Maximum velocity	10CFR72.122
Seismic	Design horiz. acceleration: 0.25g Design vert. acceleration: 0.17g	10CFR72, NRC R.G. 1.60 10CFR72, NRC R.G. 1.61
Dead Loads	Dead weight, including MSB weight (concrete density with rebar at 150 lb/ft^3)	ANS-57.9
Design Basis Normal Operating Temperature	Maximum concrete temperature 150°F (bulk), 225°F (locally)	ACI-349 and NRC Guidance
Snow and Ice Loads	Design load of 403 psf Included in live loads	ASCE 7-93 and ANS-57.9

Table 1.2-1 - Summary of VSC-24 System Design Criteria (2 pages)

Component: MSB (Major Design Code: ASME, Section III, NC Class 2 Components)

Design Load Type	Design Parameters	Applicable Codes
Flood	Maximum water height	10CFR72.122
Seismic	Design horiz. Acceleration: 0.25g Design vert. Acceleration: 0.17g	10CFR72, NRC R.G. 1.60
Dead Load	Weight of MSB loaded	ASME Sec. III
Design Basis Internal Press	Design pressure	ASME Sec. III
Normal and Off-Normal Operating Temperature	MSB w/fuel generating 24 kW decay heat. Ambient temperature -40°F to 100°F	ASME Sec. III
Normal Operation Load	±0.5g applied in all directions simultaneously	ASME Sec. III
Operation Handling Accident Load	MSB moving at 2 ft/sec hitting cask Limited to 0.75 ft/sec when loading MSB into storage cask	ASME Sec. III
Accident Drop	Peak vert. deceleration: 60g (120g applied statically) Peak horiz. deceleration: 22g (44g applied statically)	
	Containment Boundary (shell, structural lid)	ASME Sec. III, 10CFR72
	Sleeve Assembly	10CFR72 Deformation limited so that fuel is not touched by sleeves, ASME Section III material properties used.
Accident Pressure	Maximum internal pressurization from rod ruptures	ASME Sec. III
Load Combinations	See Table 2.2-3.	ASME Sec. III

Table 1.2-2 - Major Physical Design Characteristics for the 24 Assembly MSB

Parameter	Value
Outside Diameter	62.5 inches
Length	164.2 to 192.25 inches (depending on fuel length)*
Capacity	24 PWR assemblies
Maximum Heat Load	24 kW
Maximum Fuel Cladding Temperature	**
Material	SA-516 Gr 70 coated with Everlube 812, Everlube 823, Carbo-Zinc 11, Dimetcote, 6 or equivalent.
Internal Atmosphere	Helium

* Dimensions of all MSB configurations are shown in the general arrangement drawings in Section 1.5.

** Maximum cladding temperature is dependent on many parameters. The methodology developed in Reference 1.1 and further described in Reference 4.1 is used for this temperature determination. The range of acceptable temperatures for most PWR fuel is between 712 and 752°F (370 and 400°C).

Table 1.2-3 - MSB Fabrication Summary

EXAMINATION

- Dye penetrant or magnetic particle examination of welds shall be performed in accordance with the requirements of ASME Section V, Article 6, and Section III, NC-5350 (dye penetrant), or Article 7 and NC-5340 (magnetic particle), respectively.
- Welds that are not dye penetrant or magnetic particle examined shall be visually examined in accordance with ASME Section V, Article 9 and Section III, NF-5360, or in accordance with AWS D1.1, "Structural Welding Code – Steel" (1983 to 1995 Editions), Section 8.15.
- Personnel performing examinations shall be qualified in accordance with the quality assurance program and either ASNT Recommended Practice No. SNT-TC-1A (1984, 1988, 1992, or 1996 Edition) or ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel, CP-189 (1991 or 1995 Edition).
- Welds to be radiographed shall be examined in accordance with the requirements of ASME Section V, Article 2, and Section III, NC-5320.

FABRICATION

- Except as provided in Table 1.2-7, the MSB confinement boundary shall be constructed in accordance with ASME Section III, Subsection NC, 1986 Edition, or later Editions and Addenda through the 1998 Edition with the 2000 Addenda. Welding of temporary attachments shall meet NC-4435 of the 1992 Edition with the 1992 Addenda, or later Editions and Addenda through the 1998 Edition with the 2000 Addenda.
- Surfaces shall be cleaned to a surface cleanness classification C or better as defined in ANSI/ASME N45.2.1, Section 2.

PACKAGING AND SHIPPING

Packaging and shipping shall be in accordance with ANSI/ASME N45.2.2.

QUALITY ASSURANCE

The MSB shall be fabricated under a quality assurance program that meets 10 CFR 72, Subpart G.

Table 1.2-4 - Major Physical Design Parameters for the 24 Assembly VCC

Parameter	Value
Height	196.7 to 225.1 inches*
Outside Diameter	132 inches
Capacity	1 MSB containing 24 PWR Fuel Assemblies
Shielding	
Concrete Thickness	29 inches
Steel Thickness	1.75 inches
Surface Average Radiation Dose:	
Side	<100 mrem/hr
Top	<200 mrem/hr
Air Flow at Design Heat Load	1.1 lb_m/sec
Material of Construction	Normal Density, Type II Portland Cement, 4000 psi Reinforced Concrete ASTM A-615 grade 60 Reinforcing Steel and A36 Structural Steel
Service Life	>50 years
Maximum Concrete Temperatures for Normal Operation	150°F Bulk 225°F Local Hot Spots

* Dimensions of all VCC configurations are shown in the general arrangement drawings in Section 1.5.

Table 1.2-5 - VCC Construction Summary

MATERIALS

- Concrete mix shall be in accordance with ACI 318, Chapters 4 and 5.
- Type II Portland Cement, ASTM C150.
- Fine aggregate ASTM C33.
- Coarse aggregate ASTM C33.
- Admixtures
  - Water Reducing ASTM C494
  - Pozzolanic Admixture ASTM C618
- Air Entrainment 3% to 6% by volume.
- Compressive Strength 4000 psi.
- Slump and air content shall be in accordance with ACI 301, Chapter 3.
- Steel reinforcement A615, Grade 60 or equivalent.
- Density 144 lb/ft^3.

WELDING

- Welding shall meet ASME Section III, NF-4000.
- Visual inspection of all welds shall be performed to the requirements of ASME Section V, Article 9, and the acceptance criteria of ASME Section III, NF-5360 or AWS D1.1.

CONSTRUCTION

- The quantities and intervals of compressive test cylinder tests shall be in accordance with ACI 318, Chapter 5, with a minimum of one set of test cylinders per cask.
- Test specimens shall be tested in accordance with ASTM C39.
- Formwork to be in accordance with ACI 318, Chapter 6, and ACI 301.
- Embedded items shall be in accordance with ACI 318, Chapter 6.
- Reinforcement shall be in accordance with ACI 318, Chapter 7, and ACI 301, Chapter 5.
- All sidewall formwork and shoring to remain in place for at least 24 hours.
- The placement of concrete shall be in accordance with ACI 318, Chapter 5, and ACI 301, Chapter 8.
- Surface finish shall be in accordance with ACI 301.
- Curing shall be in accordance with ACI 318, Chapter 5, and ACI 301, Chapter 12.

QUALITY ASSURANCE

- The concrete cask shall be constructed under a quality assurance program that meets 10CRF72, Subpart G.
- Parameters important to safety that are covered by the QA program are density, wall thickness, compressive strength, and reinforcing material strength.

Table 1.2-6 - Major Physical Design Parameters for the 24 Assembly MTC

Parameter	Value
Inside Diameter	63.0 to 63.5 inches*
Outside Diameter	82.0 to 83.5 inches*
Height	164.7 to 192.8 inches*
Weight	110,000 to 119,000 lbs
Side Wall Dose Rate	250 mrem at 1 meter from surface

* Dimensions of all MTC configurations are shown in the general arrangement drawings in Section 1.5.

Table 1.2-7 - Exceptions to ASME Section III Requirements for the MSB (2 pages)

Section	Requirement	Exception
Subsection NCA	Miscellaneous administrative requirements	The MSB is not Code Symbol stamped. Therefore: 1. No Design Specification is prepared. Design conditions are specified by this Safety Analysis Report. A fabrication specification is prepared. 2. No Design Report is prepared. Design calculations are used instead. 3. Manufacturers are not required to have a Certificate of Authorization or an NCA-4000 quality assurance program. A 10CFR72, Subpart G Quality Assurance program is applied. 4. Material Organizations are not required to have an NCA-3800 quality assurance program. A 10CFR72, Subpart G Quality Assurance program is applied. 5. Code Authorized Inspection is not required, and Code Data Reports are not prepared. These activities are accomplished using a 10CFR72, Subpart G Quality Assurance program.
NC-2330, NC-4335	Impact Testing	For MSB pressure-retaining material, Charpy V-Notch impact testing is performed at a temperature no higher than –50°F. Acceptance criteria are 15 ft-lbs average, 10 ft-lbs minimum. The upper shell course, structural lid, and weld filler metal used to join the shell and structural lid shall also exhibit at least 45 ft-lbs when tested at no higher than 0°F. For the HAZ and unaffected base material, lateral expansion or absorbed energy values meet the requirements of NC-4335.2, using these test temperatures and acceptance criteria.
NC-3211.1, NC-3254, NC-4267	Types of attachment welds in NC-3200 vessels	The shield lid support ring is joined to the pressure boundary by a weld that is not continuous on all sides. This attachment does not serve a pressure-retaining function, and, when fuel is loaded, could be subjected to accident loads that occur only once. If this attachment is subjected to such a load, the MSB would be examined prior to continued use. Therefore, cyclic loading, stress ratchet, and fatigue are not credible events. Detailed drop analysis includes weld configuration and load transfer to the pressure-retaining boundary.

Table 1.2-7 - Exceptions to ASME Section III Requirements for the MSB (2 pages)

Section	Requirement	Exception
NC-3252, NC-5253	Category C welded joints for NC-3200 vessels	Subsection NC requires Category C full-penetration corner-welded joints to be examined by the radiographic or ultrasonic method. Because of the difficulty of performing a meaningful radiographic examination, the Category C structural lid closure weld is not volumetrically examined in accordance with NC-5253. Instead, this weld is examined by time-of-flight-diffraction ultrasonic examination method. To ensure a leak-tight boundary, the partial-penetration weld between the shield lid and the shell is examined by the liquid penetrant method (as required by NC-5260), with an additional root-pass PT examination, and is helium leak tested.
NC-3258	Design of head attachments using corner joints	When the head-to-shell weld is a corner joint, NC-3258.3 requires that the through-thickness dimension of the weld exceed the thinner of the head or shell thickness by an amount that varies with the specific joint design. Due to the geometry of the internals and due to lack of access to the inside surface of the structural lid closure weld, the shell-to-bottom-plate weld and the structural-lid-to-shell closure weld do not have the required ¼ inch fillet weld or other weld reinforcement on the ID surface. This is acceptable because there is no significant cyclic loading on these joints.
NC-6000	Hydrostatic pressure test	The vessel shell is not hydrostatically tested. Structural shell welds are volumetrically examined. The structural-lid-to-shell weld is examined by the time-of-flight-diffraction UT and by the liquid penetrant method (root, middle, and final layers). The partial-penetration shield lid weld is helium leak tested at the Code-required pneumatic test pressure. The MSB design pressure is very low; the pressure stresses are less than 10% of the ASME Code allowable stresses.

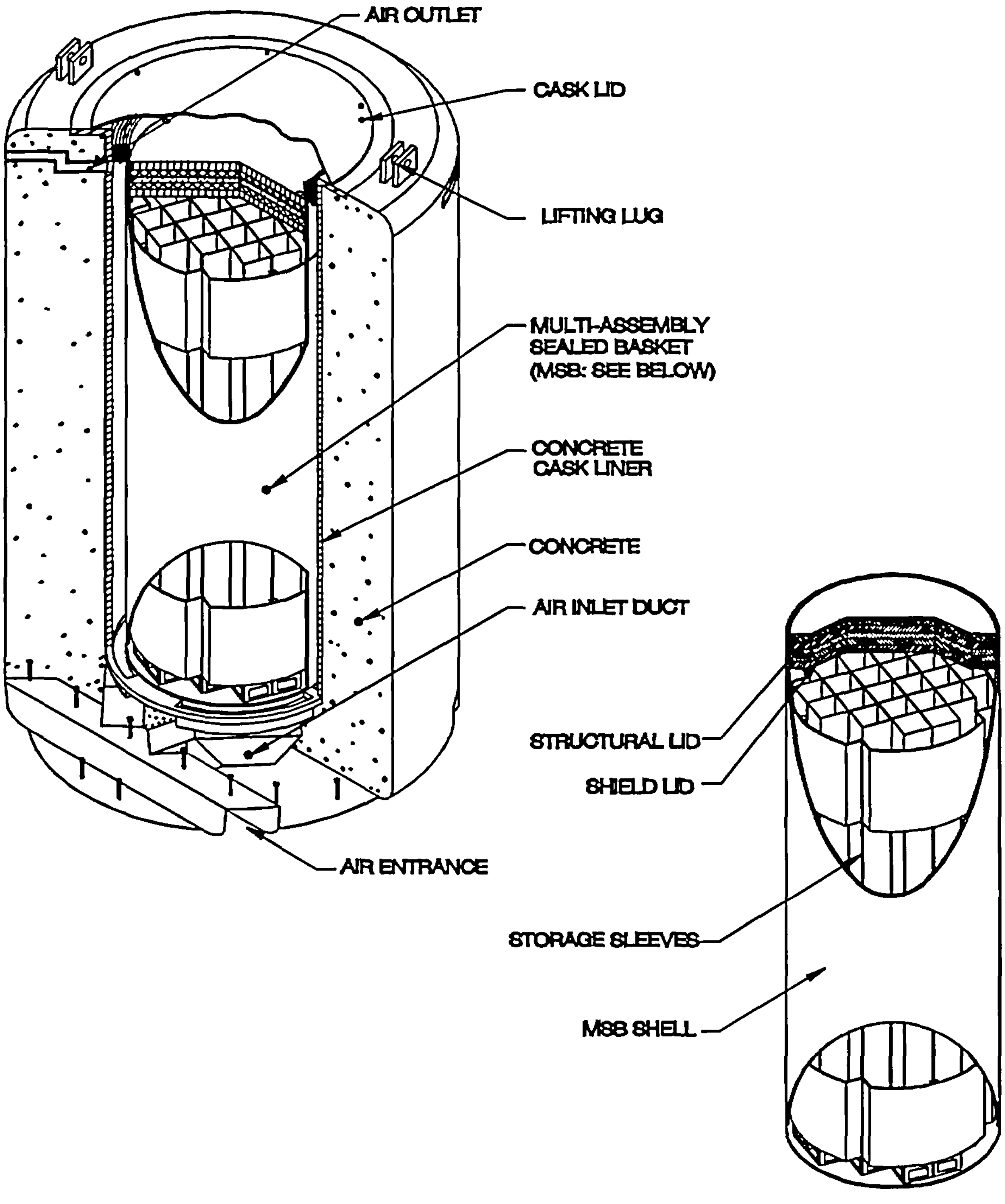


Figure 1.1-1 - VSC-24 System Components

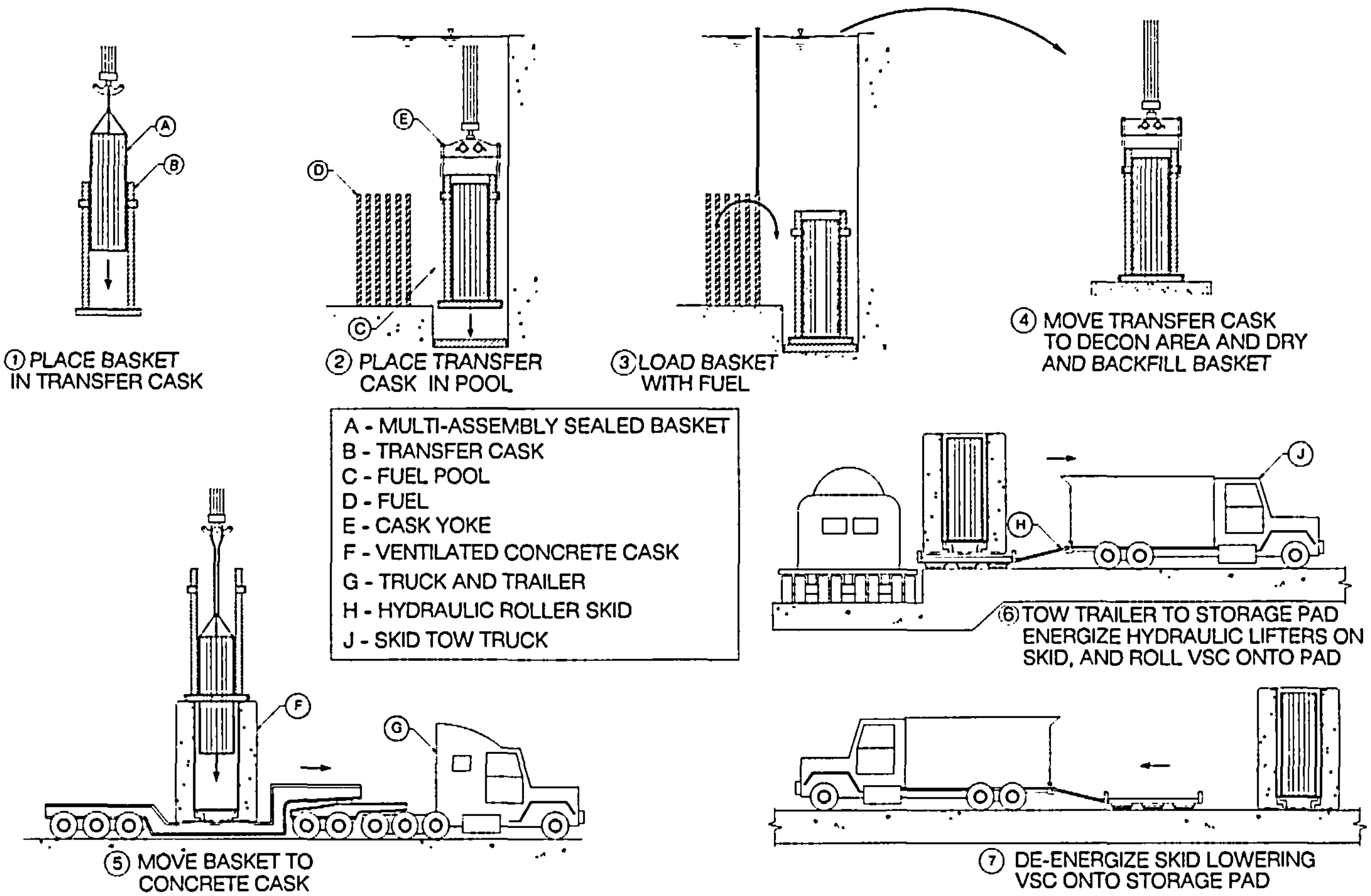


Figure 1.1-2 - VSC-24 Operations Using MTC

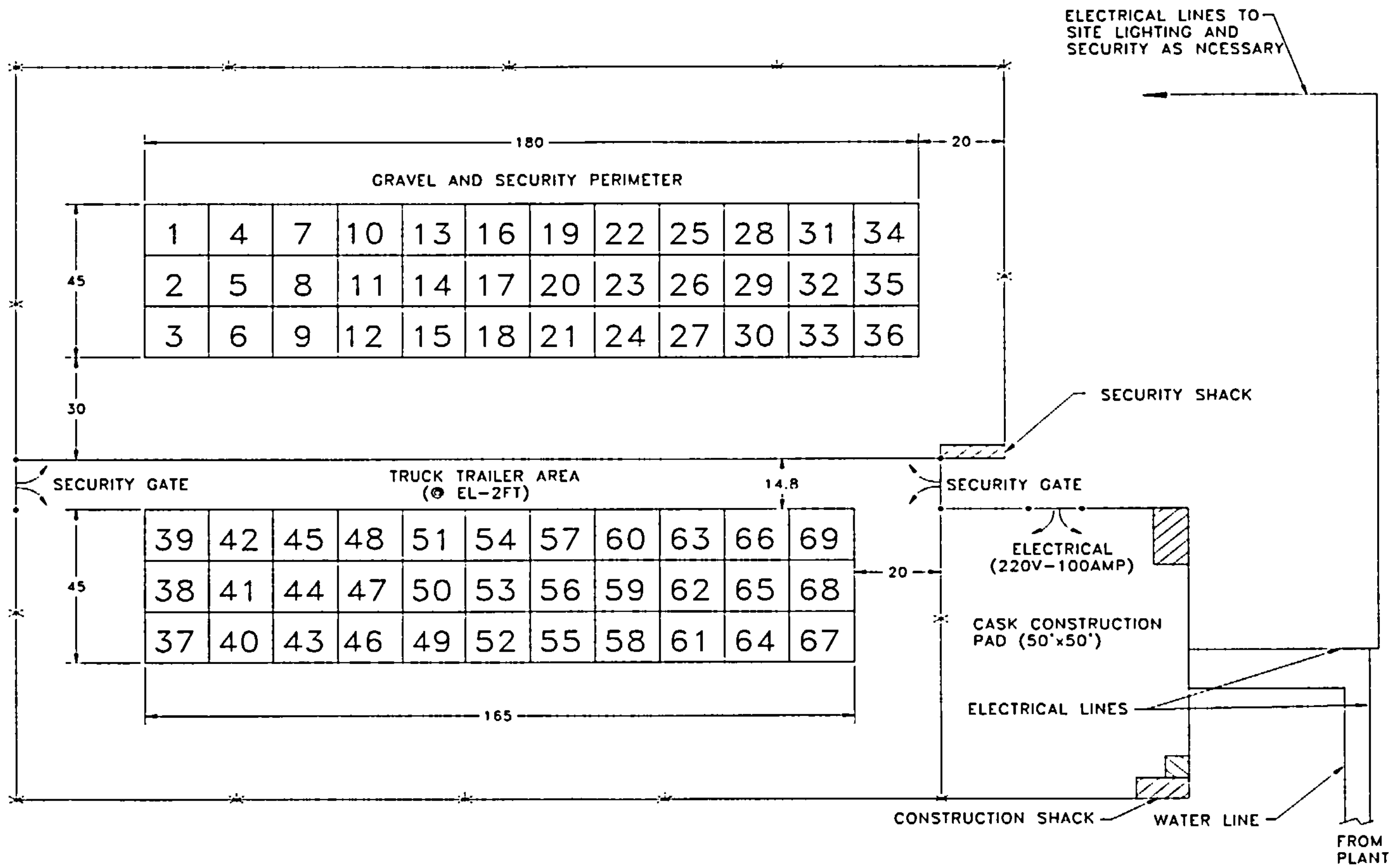


Figure 1.4-1 - Typical ISFSI Pad Layout

1.5 SUPPLEMENTAL DATA

General arrangement drawings of all the equipment items are provided in this section. Operational schematics are provided in Chapter 8.

LIST OF DRAWINGS

Drawing Title	Drawing Number	Sheet	Revision
Ventilated Concrete Cask (VCC) Assembly	VCC-24-001	1/2	6
	VCC-24-001	2/2	4
Cask Liner and Lid Assembly	VCC-24-002	1/3	5
	VCC-24-002	2/3	5
	VCC-24-002	3/3	3
Air Inlet Assembly	VCC-24-003	1/1	2
Air Outlet Assembly	VCC-24-004	1/1	2
VCC Bottom Plate Assembly	VCC-24-005	1/1	2
VCC Reinforcement	VCC-24-006	1/2	5
	VCC-24-006	2/2	3
Miscellaneous Steel Components	VCC-24-008	1/1	3
MSB Assembly	MSB-24-001	1/2	6
	MSB-24-001	2/2	5
MSB Shell and Structural Plates	MSB-24-002	1/2	6
	MSB-24-002	2/2	4
Shield Lid Assembly	MSB-24-003	1/1	6
Storage Sleeve Assembly	MSB-24-004	1/3	6
	MSB-24-004	2/3	1
	MSB-24-004	3/3	4

LIST OF DRAWINGS (continued)

Figure Withheld Under 10 CFR 2.390

PSN

VENTILATED CONCRETE CASK (VCC) ASSEMBLY

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE

FRACTION	DECIMALS	ANGLES
± 1/16	.X ±.1	± 1°

PSN

VENTILATED CONCRETE CASK (VCC) ASSEMBLY

Figure Withheld Under 10 CFR 2.390

PSN

CASK LINER AND LID ASSEMBLY

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE

FRACTION ± 1/16 DECIMALS ANGLES ± 1°

PSN

CASK LINER AND LID ASSEMBLY

Figure Withheld Under 10 CFR 2.390

PSN

Figure Withheld Under 10 CFR 2.390

PSN

IMPORTANT TO SAFETY

AIR INLET ASSEMBLY

Figure Withheld Under 10 CFR 2.390

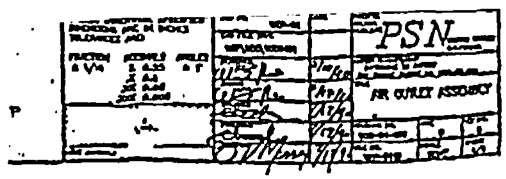


Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE

FRACTION	DECIMALS	ANGLES
± 1/16	.X ± 0.1 .XX ± 0.05 .XXX ± 0.005	± 1°

7/4/03

7/4/03

3/4/03

3/4/03

3/4/03

PSN

VCC BOTTOM PLATE ASSEMBLY

SHEET 1/1

Figure Withheld Under 10 CFR 2.390

PSN

VCC

REINFORCEMENT

Figure Withheld Under 10 CFR 2.390

Figure Withheld Under 10 CFR 2.390

PSN

MISC. STEEL COMPONENTS

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE	[illegible]	PSN
FRACTION ± 1/16 DECIMALS ANGLES ± 1°	3/4/03	[illegible]
[illegible]	3/4/03	NSB ASSEMBLY
[illegible]	3/6/03	[illegible]
[illegible]	3/6/03	[illegible]
[illegible]	3/6/03	1/2

Figure Withheld Under 10 CFR 2.390

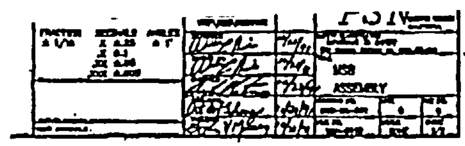


Figure Withheld Under 10 CFR 2.390

PSN

MSR SHELL AND STRUCTURAL PLATES

Figure Withheld Under 10 CFR 2.390

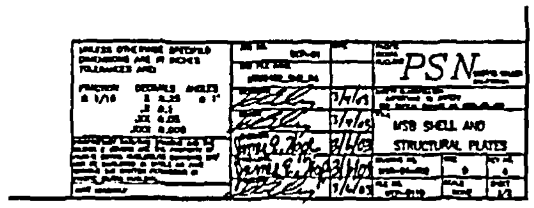


Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
TOLERANCES ARE

FRACTION	DECIMALS	ANGLES
± 1/16	X 0.25	± 1°
	.X 0.1	
	.XX 0.03	
	.XXX 0.005	

3/6/03

3/6/03

3/6/03

3/6/03

3/6/03

PSN

SHIELD LID ASSEMBLY

1/1

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE

FRACTION ± 1/16 DECIMALS .X ± 0.1 .XX ± 0.06 .XXX ± 0.005 ANGLES ± 1°

PSN

STORAGE SLEEVE ASSEMBLY

3/4/03

3/4/03

3/1/03

3/6/03

D

0

1/1

Figure Withheld Under 10 CFR 2.390

PSN

STORAGE SLEEVE ASSEMBLY

D.V. Massey

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE

FRACTION ± 1/16

ANGLES ± 1°

PSN

STORAGE SLEEVE ASSEMBLY

Figure Withheld Under 10 CFR 2.390

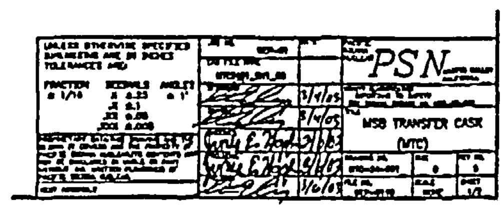


Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE: FRACTION ± 1/16, DECIMALS .X ±0.1, ANGLES ± 1°		7/9/03	PSN
		7/9/03	MSB TRANSFER CASK (MTC)
		7/6/03	
		7/6/03	
		7/6/03	

Figure Withheld Under 10 CFR 2.390

FRACTION ± 1/16	DECIMALS .X ±.1	ANGLES ± 1°	PSN
			CASK WALL ASSEMBLY

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE:			PSN
FRACTION ± 1/16	DECIMALS .X ± 0.25 .XX ± 0.1 .XXX ± 0.03 .XXXX ± 0.005	ANGLES ± 1°	
	1/6/13		CASK WALL ASSEMBLY
	1/6/13		
	1/6/13		

Figure Withheld Under 10 CFR 2.390

PSN

OUTER SHELL

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
TOLERANCES ARE:

FRACTION ± 1/16 DECIMALS ANGLES ± 1°

PSN

OUTER SHELL

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
TOLERANCES ARE

FRACTION	DECIMALS	ANGLES
± 1/16	.X ± 0.25	± 1°
	.XX ± 0.1	
	.XXX ± 0.05	
	.XXXX ± 0.005	

PSN

INNER SHELL

Figure Withheld Under 10 CFR 2.390

PSN

Figure Withheld Under 10 CFR 2.390

FRACTION ± 1/16	DECIMALS .X ± 0.25, .X ± 0.1, .XX ± 0.03, .XXX ± 0.005	ANGLES ± 1°
	1/4/03	PSN
	1/4/03	RAIL ASSEMBLY
	3/6/03	
	3/6/03	
	3/6/03	

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE		PSN
FRACTION ± 1/16	DECIMALS	ANGLES ± 1°
	3/4/03	TRUNNION ASSEMBLY
	3/4/03	
	3/6/03	
	3/6/03	
	3/6/03	

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE

FRACTION ± 1/16 DECIMALS ANGLES ± 1°

PSN

MTC SHIELD DOOR ASSEMBLY

Figure Withheld Under 10 CFR 2.390

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE

FRACTION	DECIMALS	ANGLES
± 1/16	.X ±.25	± 1°
	.X ±.1	
	.XX ±.05	
	.XXX ±.005	

PSN

MTC SHIELD

DOOR ASSEMBLY

Figure Withheld Under 10 CFR 2.390

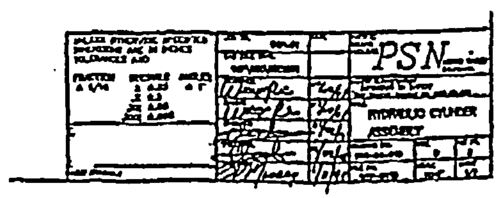
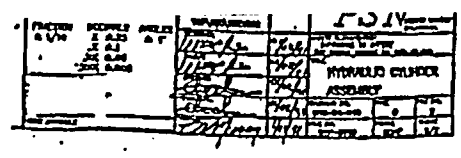


Figure Withheld Under 10 CFR 2.390



2.0 PRINCIPAL DESIGN CRITERIA

The most limiting physical parameters of the different VSC-24 versions are used in the design basis analyses. Tables 2.0-1 through 2.0-3 show the ranges of physical parameters (material, nominal dimensions, and nominal weight) considered in the structural and thermal calculations. Table 2.0-4 shows which combinations of these parameters were determined to be limiting for these analyses.

2.1 IRRADIATED FUEL TO BE STORED

The design payload for the VSC system is PWR fuel with the characteristics listed in Table 2.1-1. The main physical parameters are the fuel assembly length, weight, and envelope (transverse dimension). These parameters define the mechanical and structural design of the VCC and MSB. Each structural analysis is performed using the combination of fuel assembly parameters limiting for that specific analysis. The other important characteristics are the initial enrichment, burnup, and time since discharge from the reactor (subcritical time). The *technical specifications* provided in Section 12.4 of this FSAR provide specific limits on these characteristics, which define the thermal load, radiation source strength, and neutron multiplication factor of the loaded cask.

The only thermal design limit of the fuel to be stored in the VSC is that the maximum heat generation rate per assembly be such that the fuel cladding temperature is below its limit. The limitation in the decay power per assembly specified in the *technical specifications* in Section 12.4 includes the use of the combination of fuel assembly parameters that is limiting for a specific analysis. The thermal analysis (described in Chapter 4) shows that if the cladding limit is not violated, other temperatures (concrete and RX-277) will also be below their limits.

The thermal design limit is that the fuel cladding temperature is below the temperature limit as determined by the methodology developed in References 1.1 and 4.1. This methodology is described in detail in Appendix C. The analyses in Appendix C show that a fuel cladding temperature limit of 712°F is bounding for all types of PWR fuel. The thermal analyses presented in Chapter 4 show that the fuel cladding does not exceed this temperature, provided that the loaded assemblies have a decay power not greater than 1.0 kW, and provided that the uranium loading does not exceed 0.471 MTU per fuel assembly or 3.27 kg/inch of fuel in the assembly. The 3.27 kg/inch limit ensures that assemblies with fuel lengths shorter than 144 inches will not produce heat generation levels exceeding the 6.94 W/inch value used in the analyses presented in Chapter 4.

The principal radiological design criteria for loaded spent fuel is that the gamma and neutron sources are such that the cask exterior dose rates are not greater than the values given in Section 2.3.5.2. Analyses presented in Chapter 5 show that the Section 2.3.5.2 dose rate limits are not exceeded for MSBs containing design basis, 35 GWd/MTU, 5-year-cooled PWR fuel.

Supplementary analyses presented in Section 5.5 determine the minimum cooling time at which both the thermal and radiological criteria are met, as a function of PWR assembly burnup and initial enrichment. The PWR assemblies that meet the requirements defined in Section 5.5 would have a decay power of 1.0 kW or less, and would have gamma and neutron sources that would not exceed any of the cask exterior dose rate limits given in Section 2.3.5.2.

The criticality design criterion requires that k_{eff} remain below 0.95 under normal, off-normal, and accident conditions. The VSC-24 cask system employs soluble boron in the pool water during cask loading as the primary means of criticality control. The minimum boron concentration required to maintain criticality control varies with fuel assembly physical parameters and with assembly initial enrichment. The required boron concentration is presented as a function of initial enrichment for each PWR assembly type in Figures 6.4-1 through 6.4-8. The physical parameters important to criticality that define each PWR assembly type are listed in Tables 6.2-1(a) and 6.2-1(b).

2.2 DESIGN CRITERIA FOR ENVIRONMENTAL CONDITIONS AND NATURAL PHENOMENA

The VSC system is designed to be stored outdoors without additional weather protection. Therefore, the cask is designed to withstand the design basis daily and seasonal temperature fluctuations, and tornado, wind, flood, seismic, snow, and ice loads. Loads from these various phenomena are combined as directed in ANS-57.9, and variation of live loads from 0% to 100% is considered.

2.2.1 ENVIRONMENTAL TEMPERATURES

The normal, long-term design temperature was selected to model the expected average ambient temperature seen by a cask over its lifetime. A temperature of 75°F was selected to bound most annual average temperatures in the United States. The following list shows a representative selection of sites in the hot regions of the country (Reference 2.1):

Location	Annual Average Temperature (°F)
Columbia, SC	64
Mobile, AL	68
Shreveport, LA	66
Miami, FL	75
Laredo, TX	74
Yuma, AZ	74
Needles, CA	73
Las Vegas, NV	66

The 75°F normal temperature was used to evaluate the long-term concrete properties and to serve as the base temperature for thermal cycle evaluations. The evaluation of this environmental condition is discussed along with the thermal analysis models in Chapter 4. The thermal stress evaluation to define the normal operating thermal stress load (T_0) is provided in Chapter 3. Normal temperature fluctuations about this temperature are bounded by the severe ambient temperature cases that were evaluated as off-normal and accident conditions.

Off-normal, severe environmental conditions were defined as -40°F with no solar loads and 100°F with solar loads. An extreme environmental case of 125°F with maximum solar loads for 12 hours (analyzed as 125°F steady state) was also evaluated to show compliance with the maximum heat load accident case required by ANS-57.9. Furthermore, the cases of the complete and half-blockage of the air inlets were considered. Thermal analyses for the above described cases are presented in Chapter 11. The ambient temperature design conditions are further discussed in Chapter 4. The case with a maximum temperature gradient is chosen for the thermal stress analysis so that the obtained thermal stresses are used in the load combinations as either normal operating stress (T_0) or accident thermal stress (T_a).

The calculations using the above-defined ambient conditions are described in Chapters 3 (Structural), 4 (Thermal Hydraulics), and 11 (Accident Analysis).

The cask is also designed for tornado, wind, flood, seismic, and off-normal loads as well as snow and ice conditions as described in the subsections that follow. Appropriate combinations of normal, off-normal, and accident loadings are also defined in the Section 2.2.6, Combined Load Criteria.

2.2.2 TORNADO AND WIND LOADINGS

The VSC cask is designed to withstand loads associated with the most severe meteorological conditions, including extreme wind and tornado, which are postulated to occur at the storage site. Tornado design parameters used to evaluate the suitability of the cask include high winds, wind generated pressure differentials and tornado generated missiles. The design basis tornado maximum wind speed is 360 mph. The design basis tornado missiles are described in Table 2.2-1. These tornado and tornado missile parameters were used to determine the resulting loads on the VSC and assess any damage that could be caused. A full evaluation of the tornado event is presented in Chapter 11. NUREG-0800, Regulatory (Reg.) Guide 1.76, ANS-57.9, ASCE 7-93, and National Defense Research Committee (NDRC) methodologies were used to guide the tornado analyses. Furthermore, all missiles were assumed to impact in a manner that produces the maximum damage to the VSC.

In summary of the analysis given in Chapter 11, the maximum penetration depth for local damage is 5.69 inches (due to an 8-inch diameter missile). Combined effects of the wind loading and the high-energy missile were also evaluated and the cask was shown to be stable (i.e., no overturning) even under these severe loadings.

2.2.3 WATER LEVEL (FLOOD) DESIGN

The VCC and MSB have been evaluated for forces associated with a probable maximum flood (PMF). For the purpose of these analyses, the PMF has been assumed to result in a maximum water level which completely inundates the cask. Resultant loads on the cask consist of buoyancy effects, static pressure loads and velocity pressures. Wind wave effects and the dynamic effects of vortex shedding have been neglected in these assessments. The results of the analyses indicate that the VCC and MSB can safely withstand flood water levels up to 120 feet and water stream velocities up to 17.7 ft/sec. The analyses which support these design bases are provided in Chapter 11. Site-specific safety reviews will need to confirm that flood parameters do not exceed the results shown in Chapter 11 (i.e., 17.7 ft/sec and submerged depth of 120 feet).

2.2.4 SNOW AND ICE LOADINGS

The criteria for determining design snow loads is based on ASCE 7-93, Section 7.0. Including the worst-case ground snow load in Alaska, the live load is 403 psf. Stresses from the snow load are combined with stresses from other loads as described in Chapter 3. The following design and environmental factors are used.

C_e	=	Exposure factor = 1.0
C_t	=	Thermal factor = 1.2
I	=	Importance factor = 1.2

The exposure factor accounts for wind effects. The VSC is assumed to have a site location typical for siting category C, which is defined to be a location in which "snow removal by wind cannot be relied on to reduce roof loads because of terrain, higher structures, or several trees nearby."

The thermal factor accounts for the thermal condition of the structure. The VSC is classified as an unheated structure.

The importance factor accounts for the importance of buildings and structures in relation to public health and safety. The VSC is conservatively classified as Category III.

2.2.5 SEISMIC DESIGN

The VSC is designed to withstand a maximum horizontal ground acceleration of 0.25g in each of two orthogonal horizontal directions and a maximum vertical ground acceleration of 0.17g occurring simultaneously. A horizontal ground acceleration of 0.25g was selected as the basis of the design in accordance with 10 CFR Part 72, 72.102 (a) as being appropriate for the majority of sites east of the Rocky Mountains. Site-specific analyses will be necessary for sites whose design basis earthquake is larger than 0.25g. Analysis of the seismic event is presented in Chapter 11.

2.2.6 COMBINED LOAD CRITERIA

The cask is subjected to normal, off-normal, and accident loads. These loads are defined as follows:

Normal Loads	-	Dead Weight, Pressure, Handling, Thermal, Snow, Winds, etc.
Off-Normal Loads	-	Off-Normal Severe Environmental Conditions, Interference During MSB Lowering From MTC to VCC, Blockage of One-Half of Air Inlets, Off-Normal Handling Load, Off-Normal Pressurization
Accident Loads	-	Full Blockage of Air Inlets, Maximum Heat Load, MSB Drop Accident, Tornado (wind and missiles), Flood, Earthquake, Accident Pressurization

Normal loads due to pressure, temperature, and dead weight act in combination with all other loads, except that snow, ice, and wind loads are negligible and are bounded by other loads. No two accident events are postulated to occur simultaneously. However, loads due to one event, such as tornado wind and tornado missile loads, are assumed to act in direct combination.

2.2.6.1 Load Combinations and Design Strength - Concrete Cask

As shown in Table 2.2.2, the load combinations for the concrete cask are those specified for concrete structures in ANS-57.9. The live loads are considered to vary from 0% to 100% to ensure that the worst-case condition is evaluated. In each case, use of 100% of the live load produces the maximum load condition. The steel components of the concrete cask are stay-in-place forms and radiation shielding. Therefore, they are designed to the ACI 349 requirements for steel forms and reinforcement.

Design Strength Reduction Factors - Concrete

In calculating the design strength of the VCC concrete body, nominal strength values are multiplied by a strength reduction factor Ø per Section 9.3 of ACI 349.

2.2.6.2 Load Combinations and Design Strength - MSB Steel Vessel

The MSB shell and basket internals are designed to the 1986 Edition, or later Editions and Addenda through the 1998 Edition with the 2000 Addenda of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC for Class 2 components, with the exceptions listed in Table 1.2-7. The load combinations for all normal, off-normal and accident conditions and corresponding Service Levels are shown in Table 2.2-3, which defines the MSB design and service loadings. Level A Service Limits are used for normal conditions, and Level B and C Service Limits are used for off-normal conditions. The analysis methods allowed by the ASME Code are used. Stress intensities caused by pressure, temperature, and mechanical loads are combined before comparing to ASME Code allowables.

For accident conditions (Level D Service), the stress limits are specified in the ASME Code, Section III, Appendix F. Stresses caused by normal condition loads are combined with the stresses caused by accident or off-normal loads.

The design strength criteria for the drop evaluation is that any damage to the cask will not prevent the cask from performing the following safety functions: heat dissipation, criticality prevention, confinement, and shielding. Local stresses may exceed allowables. Internal basket deformation is limited such that fuel assemblies shall be removable.

The structural design criteria are summarized in Table 2.2-4.

2.2.6.3 Design Strength - MSB Transfer Cask

The MTC is a special lifting device designed and fabricated to the requirements of ANSI N14.6 and NUREG-0612. The criteria are:

- Maximum stress during the lift (with 10% dynamic load factor) less than or equal to ($S_y/6$ or $S_u/10$) for nonredundant load path or ($S_y/3$ or $S_u/5$) for redundant load path.
- Load bearing members of the MTC shall be subject to drop weight test (ASTM E208) or Charpy impact test (ASTM A370) per ANSI N14.6 paragraph 4.2.6.

The MTC cover plate and bolts are not designed as part of a special lifting device; they are designed to protect against excessive radiation exposure to workers. Therefore, they are designed to meet the AISC Manual of Steel Construction, not NUREG-0612.

2.3 SAFETY PROTECTION SYSTEMS

2.3.1 GENERAL

The VSC is designed for safe, long-term storage of spent nuclear fuel. The VSC will survive all normal, off-normal, and postulated accident conditions without any uncontrolled release of radioactive material or excessive radiation exposure to workers or members of the general public. The major design considerations that have been incorporated in the VSC system to assure safe long-term fuel storage are:

1. Leak-tight/multi-pass seal welds on MSB structural lid, shield lid and bottom-end plate.
2. Thick shielding lid to minimize radiation exposure during MSB closure.
3. Design of MSB body and internals to withstand a hypothetical cask drop accident.
4. Use of sealing shims to minimize contamination of the MSB exterior by fuel pool water.

As discussed in the following sub-sections, the VSC design incorporates features addressing each of the above design considerations to assure safe operation during fuel loading, VSC handling, and storage.

2.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

2.3.2.1 Confinement Barriers and Systems

The radioactivity which the VSC must confine originates from the spent fuel assemblies to be stored and the contaminated water in the fuel pool where the MSB loading is conducted. This radioactivity is confined by the multiple barriers listed in Table 2.3-1.

The VSC is provided with multiple systems to confine the radioactive fuel. First, only non-failed* fuel assemblies are to be stored within the cask, so that the cladding material provides the first level of confinement for the fission products. Second, the MSB is sealed by a weld on the shield lid tested in accordance with the *technical specification* in Section 12.4, and tested to a leak tightness of 10^{-4} standard cubic centimeters (scc) per sec at a pressure differential of 0.5 atmospheres in accordance with the *technical specification* in Section 12.4. Also, the MSB structural closure is accomplished by multi-pass welding that is tested in accordance with the *technical specification* in Section 12.4. The longitudinal and girth welds and bottom welds of the MSB shell are radiographed at the fabrication shop.

Radioactive contamination from the fuel pool water is minimized by restricting its contact with the MSB exterior. Pool water is prevented from contacting the MSB exterior by filling the MSB with demineralized borated water prior to placing the MSB in the fuel pool, filling the transfer cask-MSB annular gap with clean demineralized water as it is being lowered into the fuel pool, placing steel shielding pieces in the transfer cask-MSB annular opening (to prevent entrainment of contaminated fuel pool water in the gap) and injecting clean water into the gap during the entire submerged time.

The MSB is designed to withstand a postulated drop accident without damaging the stored fuel (i.e., the storage sleeves do not deform such that they bind the fuel). A detailed evaluation of the drop accident is included in Chapter 11.

Personnel radiation exposure during handling and closure of the MSB is minimized by the following steps.

1. Placing the shield lid on the MSB while the MTC and MSB remain in the fuel pool.
2. Decontaminating the MTC exterior prior to draining the MSB
3. Draining the MSB while still housed in the MTC.
4. Using portable shielding as necessary and available.

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* Failed fuel is an assembly which is structurally deformed or has damaged cladding or spacers to the extent that special handling may be required

5. Using the shielded MTC that is remotely operated to transfer the MSB to the VSC.
6. Placing a shielding ring over annular gap between the VSC and MSB.
7. Swiping the VSC exterior for contamination prior to leaving the auxiliary building.

2.3.2.2 Ventilation Off-Gas

The VSC is passively cooled by radiant and natural convection heat transfer at the outer surface and natural convective heat transfer in the MSB-VSC annulus. The bottom of the cask is conservatively assumed to be an adiabatic surface. The design criterion for the air-flow in the VSC-MSB annulus is that the pressure difference due to the buoyancy effect due to the heating of the air is equal to the flow pressure drop. The details of the passive ventilation system design are provided in Chapter 4.

There are no radioactive releases during normal operations. Also, there are no credible accidents which cause significant releases of radioactivity from the VSC and, hence, there are no off-gas system requirements for the VSC during normal storage operation. The only time an off-gas system is required is during the MSB drying phase. During this operation the off-gas system at the licensed reactor site will be used.

2.3.3 PROTECTION BY EQUIPMENT AND INSTRUMENTATION SELECTION

2.3.3.1 Equipment

The VSC system may include several pieces of support equipment. However, the only important to safety equipment used is the handling gear required to lift the transfer cask in and out of the pool. This equipment is by nature site-specific and must be addressed in site safety reviews.

Additional handling equipment (such as trailers, skids, portable cranes, or cask transporters) are not important to safety as the VSC system is designed to withstand the failure of any of these components.

2.3.3.2 Instrumentation

The VSC does not require any instrumentation to assure the safe storage of spent fuel.

Continuous monitoring of the MSB closure welds (for radionuclide leakage) is not necessary because there are no long-term degradation mechanisms that could cause the closure welds to fail within the design lifetime of the MSB, and because the possibility (and effects) of corrosion have already been considered in the MSB confinement boundary design. Application of existing plant 10CFR50 environmental monitoring requirements, along with periodic surveillance, inspection, and survey requirements, are sufficient to satisfy the requirements of 10CFR72.122(h)(4).

The thermal performance of the cask system is verified through daily visual inspections of the storage cask inlet and outlet ducts that are performed to detect and prevent any airflow blockages. Such a blockage would lead to a rise in outlet duct air temperature. The daily temperature measurements and visual inspections ensure that the temperatures of all cask system components remain within their specified thermal design limits as analyzed in this SAR.

2.3.4 NUCLEAR CRITICALITY SAFETY

The VSC is designed to maintain nuclear criticality safety (subcriticality) under all applicable regulatory conditions. These conditions include normal handling and storage conditions, off-normal handling and component functioning, and hypothetical accident conditions.

The principal criticality design criterion is that k_{eff} remain below 0.95 during normal operation and during accident conditions where optimum moderation is assumed. These values of k_{eff} also include error contingencies and calculational and modeling biases.

The methods of criticality control in the VSC are the use of soluble boron in the fuel pool water and the neutron absorption properties of the fuel and the steel basket of the MSB. The administrative controls ensure that fuel placed in the VSC and the boron concentration meet the requirements described in Chapter 6.

2.3.5 RADIOLOGICAL PROTECTION

2.3.5.1 Access Control

Access to a VSC installation site is controlled by a peripheral fence so as to meet 10 CFR 72 requirements. The details of this access control and the division of the installation site into radiation protection areas will be on a site-specific basis and described in an applicant's 10 CFR 50 license documents or safety review.

2.3.5.2 Shielding

The VSC is designed to provide an average external side surface dose (gamma and neutron) of less than 100 mrem/hr on the sides and 200 mrem/hr on the top. Dose rates of less than 350 mrem/hr at the inlet duct and 100 mrem/hr at the outlet vent are also provided. The actual VSC shielding calculations show dose rates lower than these limits. The design maximum dose rate at the top of the MSB structural lid is 1000 mrem/hr to allow limited personnel access during MSB closure operations. Chapter 5 presents analyses verifying that the VSC loaded with the design basis payload (35 GWd/MTU, 5-year-cooled, PWR fuel) meets these criteria. Supplementary analyses presented in Section 5.5 give the cooling time required, as a function of assembly burnup and initial enrichment, to meet the above criteria and the assembly decay power limit of 1.0 kW.

Sections 72.104 and 72.106 of 10 CFR 72 set whole body dose limits for an individual located beyond the controlled area at 25 millirems per year during normal operations and 5 rem (5,000 millirems) from any design basis accident. The radiation emanating from the VSC system results in much lower doses. The analyses showing the actual VSC doses are included in Chapters 10 and 11. These analyses conservatively assume the maximum allowable average surface dose rates on the cask side and top, which are 100 and 200 mrem/hr, respectively.

2.3.5.3 Radiological Alarm Systems

There are no radiological alarms required on the VSC. Justification for this is provided in analysis in Chapters 5 (Shielding), 10 (Radiological Protection) and 11 (Accident Analysis).

2.3.6 FIRE AND EXPLOSION PROTECTION

The VSC design is highly resistant to the effects of fire and explosion. The thick concrete walls are not significantly affected by short-term exposure to temperatures in excess of 2000°F and the thermal diffusivity is such that any fire would be required to burn for a long time (days) before much of the wall thickness would be affected. This can be seen from the transient analysis presented in Chapter 11. The resistance of the cask design to explosive loadings can be implied from its performance under other impulsive loadings, such as missile impacts. However, the VSC has not been specifically analyzed for these events, because it is not expected that either fire or explosive events would be considered credible at most storage sites. It is expected that no fire and explosion protection will be required, although this will be determined on a site-specific basis as necessary.

2.3.7 LIGHTNING

The possibility of lightning strike must be addressed, on a site-specific basis, by all VSC-24 cask system licensees, in accordance with the requirements of the Lightning Protection Code and the National Electric Code. These site-specific evaluations must show either that lightning strike on a cask is not credible or that the cask is capable of withstanding a lightning strike without significant damage.

2.4 DECOMMISSIONING CONSIDERATIONS

The first step in decommissioning the VSC is to remove the fuel. This can be done in a number of ways. Three potentials are discussed in Reference 2.2.

Unless the MSB is licensed for transportation at some later date, the MSB will be reopened after the storage period, and the spent fuel will be unloaded and placed in another storage or transportation system. This process will most likely involve "wet" fuel unloading and transfer in some form of spent fuel storage pool and, thus, involves re-flooding the MSB.

Analysis of the re-flooding process is required to demonstrate that fuel cladding integrity will be maintained (i.e., to show that the cladding will not be damaged by "thermal shock" effects.) The analysis must also show that the MSB will not be over-pressurized by the boiling of water that may occur during the re-flood process. In addition to demonstrating cladding integrity and ensuring against MSB over-pressure, such analyses may be used to determine specifications for the MSB re-flood process, such as minimum temperatures and maximum flow rates for the injected water.

The MSB re-flooding analyses described above must be performed, on a site-specific basis, by all VSC-24 licensees. These evaluations must be performed before placing any spent fuel into storage at the site in question.

The VSC system has been designed to minimize contamination of the cask during loading and unloading operations. No contamination is expected on the VCC. A corrosion and radiation resistant coating is applied to the cask interior. Therefore, it is anticipated that, if the interior were to become contaminated, it could be decontaminated and the complete VCC broken up (or left whole) and shipped to a normal landfill.

Activation of the concrete or steel is not a concern because the neutron flux is only on the order of 10^3 n/sec-cm^2. Also, this is a fast neutron flux with energy around 1 Mev. Hence, activation of the VSC components would be on the order of 10^{10} less than the activation around a reactor (where the flux is 10^{13} n/cm^2-sec). Activation is, therefore, insignificant.

The MSB interior is expected to be highly contaminated with fuel crud. At the time of cask decommissioning a determination of the amount of crud or other contamination would be made. If necessary, MSB flushing could be performed to reduce most of the interior contamination. It is not expected that large quantities of crud would remain in the MSB, but should quantities sufficient to create a hazard to site workers be present, the cask interior could be further decontaminated using one of the currently available solvent based systems. After the cask has been made sufficiently safe to handle, it could be cut into large pieces and shipped as Low Specific Activity (LSA) material to a low-level disposal site. Alternatively, the MSB could be shipped (without cutting) to a burial site and buried. It could even be used to hold other waste for burial.

Table 2.0-1 - VCC Design Parameters (2 pages)

Component	Design Input	Value
VCC CASK	Concrete/rebar Density	150 lbf/ft3
	Concrete Density	144 lbf/ft3
	Concrete Strength	4,000 psi
	OD	132 in.
	Length (w/o cover plate or lid)	196.7 - 225.1 in.
	Corner Chamfer	3 in. x 45 deg. bevel
VCC BOTTOM PLATE	Material	A 36
	OD	126 in.
	Thickness	0.25 in.
VCC LID	OD	82 in.
	Thickness	0.75 in.
VCC SHIELD RING	OD	70.5 in.
	ID (including ring on shield lid)	60 in.
	Height	6 in.
VCC SKID CHANNELS	Number	2
	Width	12 in.
	Height	12.2 in.
	Average Length	100.3 in.
VCC COVER PLATE	Material	A 36
VCC CASK LINER	Height	174.7 – 203.1 in.
	ID	70.5 in.
	Thickness	1.75 in.
	Vent Hole Width	49.0 in.
	Cask Liner Segment Width between Vent Holes	4.56 in.
	Material	A 36
VCC CASK LINER FLANGE	OD	90 in.
	Thickness	2 in.

Table 2.0-1 - VCC Design Parameters (2 pages)

Component	Design Input	Value
VCC CASK LINER BOTTOM	Thickness	2 in.
VCC REBAR	VCC Outside Rebar	#6 at 6 in. spacing
	No. in Air Outlet Cross Section	32
	Diameter in Air Outlet Cross Section	0.75 in.
	Material	A 615 Gr. 60
VCC AIR OUTLET CHANNELS	Number	4
	Width	47.8 in.
	Length with 3 In. Overlap Between High and Low Channels	36.3 in.
	Height	4 in.
	Thickness	0.5 in.
	Length from Top of VCC to Midpoint of Outlet Assembly	8 in.
VCC AIR INLET ASSEMBLIES	No. of Air Inlet Assemblies	4
VCC AIR INLET TUBES	Width	12 in.
	Length	40 in.
	Height	12 in.
VCC AIR INLET CHANNELS	Width (mid-wall)	4.5 in.
	Height	5 in.
	Angular Extent	71 deg.
	Outer Radius	35.25 in.
	Liner Thickness	0.5 in.
CERAMIC TILE	No. of Tiles	24 or 29*
	Square Tile Thickness	0.3 in.
	Square Tile Width	1.7 in.

* As shown on the general arrangement drawings in Section 1.5, two different tile configurations are permitted in the VCC.

Table 2.0-2 - MSB Design Parameters (2 pages)

Component	Design Input	Value
MSB SHELL	MSB Shell Material	SA-516 Gr. 70
	Steel Density	0.284 lbf/in^3
	MSB Bottom Plate Material	SA-516 Gr. 70
	Minimum MSB Cavity Length (w/tolerances)	150.55 - 178.6 in.
	MSB Length	164.2 - 192.25 in.
	MSB Shell Thickness	1.00 in.
	MSB Bottom Thickness	0.75 in.
	MSB OD	62.5 in.
	Minimum MSB Basket Inside Diameter (w/tolerances)	59.8 in.
MSB STRUCTURAL LID	MSB Structural Lid-to-Shell Weld Size	0.75 in.
	Material	SA-516 Gr. 70
	MSB Structural Lid Thickness	3 in.
	MSB Structural Lid OD	60 in.
	Hoist Thread Engagement Length	1.5 - 2.0 in.
	Lid Lifting Bolt Holes Radius	26.5 - 27.0 in.
	No. of Lid Lifting Bolts/Hoist Rings	6
	MSB Hoist Thread Type	1.5-6 UNC or 2.0-4.5 UNC
	Hoist Ring Type	24,000 and 30,000 lb. rating
MSB SHIELD LID	Material	SA-516 Gr. 70
	Rx-277 Density	0.0607 lb/in^3
	Steel Density	0.284 lbf/in^3
	Lead Density	0.41 lbf/in^3
	Shield Lid Weld Type and Size	¼ in. partial penetration
	Shield Lid Support Ring Weld Type & Size	½ in. partial penetration

Table 2.0-2 - MSB Design Parameters (2 pages)

Component	Design Input	Value
MSB SHIELD LID	Thickness	9.5 in. (sandwich of 2.5 in. plate, 2 in. Rx-277 neutron absorber, and 5 in. plate)
	OD	60.1 in.
	Distance from Top of MSB to Top of Support Ring	12.55 in.
	Shield Lid Support Plate OD	60.25 in.
	Shield Lid Side Ring Thickness	0.5 in.
	Support Ring Thickness	0.5 in.
	Support Ring Height	2.0 in.
MSB STORAGE SLEEVE	Material	SA-516 Gr. 70
	Outer Dimension	9.2 in.
	Thickness	0.2 in.
	Length	147.5 – 163.6 in.
MSB STORAGE SLEEVE (BASKET) ASSEMBLY	Storage Sleeve Assembly OD	59.2 in.
	No. of Curved Support Plates	12
	Curved Support Plate Height	28 in.
	Curved Support Plate Thickness	0.5 in.
	No. of Support Wall Plates	24
	Support Wall Plate Height	28.0 in.
	Support Wall Plate Width	4.37 in.
	Support Wall Plate Thickness	0.5 in.
	No. of Support Bars	12
	Support Bar Height	28 in.
	Support Bar Thickness	1.45 in.
	Support Bar Width	2.0 in.
FUEL ASSEMBLY	Fuel Assembly Weight	1110 – 1585 lbs.

Table 2.0-3 - MTC Design Parameters (2 pages)

Component	Design Input	Value
MTC SHELL	Inner Shell Material	SA-588 Gr. A or B
	Outer Shell Material	SA-588 Gr. A or B
	Gamma Shielding Material	Lead
	Neutron Shielding Material	Rx-277
	Outer Shell OD	82.0 - 83.5 in.
	Outer Shell Thickness	1.0 in.
	Outer Shell Height	161.7 - 189.8 in.
	Inner Shell OD	64.5 - 65.0 in.
	Inner Shell Thickness	0.75 in.
	Diameter at Lead to Neutron Shielding Interface	72.0 - 73.38 in.
	Top Thickness	2 in.
	Bottom Thickness	1 in.
	Lead Shielding Top Gap	1.0 in. and 7.0 in.
MTC COVER PLATE	Cover Plate Material	SA-516 Gr. 70
	Cover Plate OD	74 in.
	Cover Plate Thickness	1 in.
	Cover Plate ID	60.5 in.
	No. of Cover Plate Bolts	16
	Cover Plate Bolt Type	1 in.-8 UNC
	Cover Plate Bolt Material	A 325
	MTC Cover Plate Bolt Circle Radius	35.5 in.
MTC TRUNNION	No.	2
	Material	SA-516 Gr. 70
	Diameter	10.75 in.
	Length from Outer Shell OD to End of Bearing Surface	4.5 in.
	Length (including End Cap)	14.5 - 15.0 in.

Table 2.0-3 - MTC Design Parameters (2 pages)

Component	Design Input	Value
MTC DOORS	No.	2
	Width	39.25 - 42.7 in.
	Length	69.5 – 70.0 in.
	Thickness	7.13 in. and 9.0 in.
	Door Cutout Width	17.7 - 20.25 in.
MTC DOORS	Door Cutout Length	15.0 - 17.25 in.
	Lead Thickness in Shielded Door Assemblies	0 - 2.0 in.
MTC RAILS	Material	A 36
	Number	2
	Height	7.25 - 9.125 in.
	Length	99 - 105 in.
	Width	6.5 - 7.5 in.
	Partial Length Rail Shield Plate Thickness	5.50 in.
	Rail Shield Plate Average Length	75.5 in.
	Spacing at Outside of Rails	83.3 - 84.8 in.
	Rail to Shell (Inner Weld) Leg Length	0.625 in.
MTC RAIL SUPPORTS	Material	A 36
	Width	9.25 - 10.25 in.
	Thickness	1.5 in.
	Rail to Rail Support Partial Penetration Weld Size	0.625 in.

Table 2.0-4 - Design Configurations for Calculations (2 pages)

Calculated Load/Stress	Limiting Configuration	Notes
Deadweight		
MSB Ceramic Tile Support	Heaviest Loaded MSB	The bounding analysis is based on 24 tiles equally spaced around the perimeter of the MSB.
MSB Structural Lid	Heaviest Loaded MSB	
MSB Hoist Ring	(1) 30,000 lb. Hoist Ring w/81,000 lb. MSB (2) 24,000 lb. Hoist Ring w/74,000 lb. MSB	
MTC Support Rail, Support Rail to Rail Weld, Rail to MTC Shell Weld	(1) Thin MTC w/81,000 lb. MSB (2) Thick MTC w/74,000 lb. MSB	
MTC Trunnion and Shell	Thin MTC Configuration Heaviest Loaded MTC	
MTC Cover Plate	Heaviest Empty MTC	
VCC Bottom and Wall Stresses	Heaviest Loaded VCC Tallest VCC	
VCC Liner Stress	Heaviest Loaded MTC	
Thermal		
MSB Stresses	Medium-Length MSB and Storage Sleeve	Varying the lengths of the MSB and storage sleeve has minimal effect on stress results.
VCC Stresses	Medium-Length VCC and Liner	Varying the heights of the VCC and liner has minimal effect on thermal stresses.
Pressure		
MSB Normal, Off-Normal, and Accident Pressures	B&W 15x15 fuel assembly w/BPRAs	
MSB Normal and Accident Pressure Stress	Heaviest Loaded MSB	The analysis combines deadweight, handling, and pressure.
MSB Normal, Off-Normal, and Accident Pressure Stresses	Medium-Length MSB	Varying the length of the MSB has minimal effect on stress results.
Normal Handling		
Vertical	Tallest MSB Heaviest MSB	
Horizontal	Medium-Length MSB Heaviest MSB	Varying the length of the MSB has minimal effect on stress results.
Off-Normal Handling		
Vertical	Tallest MSB Heaviest MSB	
Horizontal	Medium-Length MSB Heaviest MSB	Varying the length of the MSB has minimal effect on stress results.

Table 2.0-4 - Design Configurations for Calculations (2 pages)

Calculated Load/Stress	Limiting Configuration	Notes
Drop		
MSB Vertical Drop – Ceramic Tile Support	Heaviest Loaded MSB	The bounding analysis is based on 24 tiles equally spaced around the perimeter of the MSB.
MSB Vertical Drop – No Ceramic Tile	Tallest MSB and Storage Sleeve Heaviest Fuel	
MSB Horizontal Drop	Medium-Length MSB Tallest Storage Sleeve Heaviest Fuel	Varying the length of the MSB has minimal effect on stress results.
Storage Sleeve Horizontal Drop	Heaviest Fuel Tallest Storage Sleeve	
Storage Sleeve Buckling Due to Horizontal Drop	Tallest Storage Sleeve	
Storage Sleeve Buckling Due to Vertical Drop	Heaviest BPRA Fuel Weight Medium-Length Storage Sleeve	
VCC Shear and Moment Due to Horizontal Drop	Medium-Length VCC	Varying the length of the VCC has minimal effect on results.
VCC Tip-over – VCC Concrete Crush Depth	Tallest VCC Heaviest Loaded VCC	
VCC Tip-over – MSB Deceleration	Shortest VCC Lightest Loaded VCC	
VCC Vertical 5-Foot Drop Concrete Crush Depth	Tallest VCC Heaviest Loaded VCC	
VCC Vertical 5-Foot Drop – MSB Deceleration	Shortest VCC Lightest Loaded VCC	
VCC Horizontal 5-Foot Drop Concrete Crush Depth and MSB Deceleration	Shortest VCC Lightest Loaded VCC	
Seismic		
VCC Tip-Over Due to Horizontal & Vertical Acceleration	Lightest Loaded VCC Highest Loaded VCC Center of Gravity	
VCC Tip-Over Due to Vertical Corner Lift	Highest Loaded VCC Center of Gravity	
VCC Seismic Stresses	Heaviest Loaded VCC Highest Loaded VCC Center of Gravity Tallest VCC and Liner	
Flood		
VCC Tip-Over Due to Immersing Flood, or Tornado Wind Load and/or Missile	Highest Loaded VCC Center of Gravity Lightest Loaded VCC Tallest VCC	
VCC Sliding Due to Tornado Wind Load	Tallest VCC Lightest Loaded VCC	
VCC Stress Due to Tornado Missile	Tallest VCC Liner	

Table 2.1-1 - Principal Design Parameters for PWR Fuel Assemblies to be Stored in a Ventilated Storage Cask

PHYSICAL PARAMETERS:	
Maximum transverse dimension, in.*	8.536
Assembly weight, lb. (with or without control elements)**	1110-1585
Maximum overall length, in. (with control elements)**	178.6
Maximum weight of uranium/assembly, kg.	471
Fuel length, in.	141.8-150.0
Fuel rod clad material	Zircaloy
No. of assemblies/VSC	24
THERMAL CHARACTERISTICS:	
Maximum heat generation per assembly	1.0 kW
RADIOLOGICAL CHARACTERISTICS:* **	
Maximum initial enrichment (wt% ^{235}U)	4.2
Maximum burnup (MWd/MTU)	45,000
Cooling Time (years)	Varies with burnup and enrichment. See Table 5.5-1.

* Maximum width of fuel assembly, excluding end fittings.

** Limiting parameter value chosen as applicable for each specific analysis.

Table 2.2-1 - Tornado-Generated Missiles

Missile Description	Weight (lbs.)	Velocity (mph)
Automobile	3960	126
Armor Piercing Artillery Shell (8 in. diameter)	275	126
Steel Sphere (1 in. diameter)	0.22	126

Table 2.2-2 - Load Combinations for the VSC Concrete Cask

Load Comb.	Dead	Live	Wind	Temp. Norm./ Acc.	Seismic	Tornado Missile	Accidents/ Impacts	Soil Pressure
1.	1.4D	+1.7L						
2.	1.4D	+1.7L						+1.7H
3.	0.75(1.4D	+1.7L	+1.7W	+1.7 T_o				+1.7H)
4.	0.75(1.4D	+1.7L		+1.7 T_o				+1.7H)
5.	D	+ L		+ T_o	+ E_{ss}			+ H
6.	D	+ L		+ T_o			+ A	+ H
7.	D	+ L		+ T_a				+ H
8.	D	+ L		+ T_o		+ W_t		+ H

D	=	Dead Load	T_a	=	Accident Temperature
L	=	Live Load	E_{ss}	=	Earthquake
W	=	Wind	W_t	=	Tornado/Tornado Missile
T_o	=	Normal Temperature	A	=	Accidents/Impact
H	=	Soil Pressure			

Table 2.2-3 - MSB Load Combinations

LOAD		NORMAL			OFF-NORMAL					ACCIDENT							
Dead Weight	MSB w/fuel	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Thermal	Inside VCC: 75°F	X		X				X		X	X	X	X	X			
	Inside MTC: 75°F		X		X		X								X		
	Inside VCC: -40°F or 100°F					X			X								X
	Inside VCC: Max Heat Load (125°F)															X	
Pressure	Normal	X	X	X				X		X	X	X	X				
	Off-Normal				X	X	X		X							X	X
	Accident													X	X		
Handling Load	Normal		X	X	X									X	X		
	Off-Normal						X	X	X								
Drop (Vertical or Horizontal)										X							
Seismic											X						
Flood												X					
Tornado													X				
ASME Service Level		A			B		C			D							
Load Combination No		1	2	3*	1*	2	1*	2	3	1*	2	3	4	5*	6	7	8

* Controlling load combinations for the Service Level.

Table 2.2-4 - Structural Design Criteria for Steel Components Used in the Multi-Assembly Sealed Basket

Component (Applicable Code)	Criteria
1. MSB Normal Operation (ASME III, NC, Service Level A)	$P_m \leq S_m$ $P_m + P_b \leq 1.5\ S_m$ $P_L + P_b + Q \leq 3S_m$
Lifting Devices (ANSI N14.6 and NUREG-0612, 10% dynamic factor)	Redundant load path: max stress $\leq S_u/5$ or $S_y/3$ Non-redundant load path: max stress $\leq S_u/10$ or $S_y/6$
2. MSB Off-Normal Operation (ASME III, NC, Service Level B	$P_m < 1.1\ S_m$ $P_L + P_b < 1.65\ S_m$ $P_L + P_b + Q < 3\ S_m$
3. MSB Off-Normal Operation (ASME III, NC, Service Level C)	$P_m < 1.2\ S_m$ $P_L + P_b < 1.8\ S_m$
4. MSB Accident Conditions, Vessel and Cover (ASME III, NC, Service Level D)	$P_m \leq 0.7\ S_u$ $P_L + P_b \leq 1.05\ S_u$ $P_{max} \leq 0.9\ S_u$ for plastic system analysis
5. Internal Basket	Plastic yielding of basket shall not prevent the removal of fuel assembly(s).
6. Brittle Fracture (NUREG/CR-1815)	Selection of material with adequate toughness. Control by operating procedures based on minimum temperature. A VCC containing a loaded MSB shall only be removed at ambient temperatures of 0°F or above, coincident with a structural lid-to-shell weld temperature of 30°F or above. Movement of a MTC containing a loaded MSB shall only be made at ambient temperatures of 40°F or above.

Table 2.3-1 - Radioactivity Confinement Barriers and Systems of the VSC System

Radioactivity	Confinement Barriers and Systems
Contaminated Pool Water	1. Demineralized borated water in MSB and MSB-MTC annular gap filled by steel pieces with positive flow of clean demineralized water while submerged in the pool
	2. Draining and vacuum drying of MSB
	3. Use of MTC during MSB transfer to VSC
Irradiated Fuel Assemblies	1. Fuel Cladding
	2. MSB Body
	3. Seal Welded MSB Shield Lid
	4. Multi-pass Seal Welded MSB Structural Lid
	5. Multi-pass Seal Welded MSB Bottom End Plate

3.0 STRUCTURAL EVALUATION

This section describes the design and analyses of the principal structural components of the VSC spent fuel storage cask under normal operating conditions.

3.1 STRUCTURAL DESIGN

3.1.1 DISCUSSION

The VSC system consists of three major structural components: 1) the Ventilated Concrete Cask (VCC), 2) the Multi-Assembly Sealed Basket (MSB), and 3) the MSB Transfer Cask (MTC). The principal structural member of the VCC is the concrete shell with its reinforcing bars. The principal structural members of the MSB are the shell, top and bottom plates and welds, and the internal basket assembly. The structural components of the MTC are its trunnions, inner and outer steel walls and the bottom doors and their rails. All components are shown in the general arrangement drawings in Section 1.5.

The concrete cask is a reinforced concrete cylinder with an outside diameter of 132 inches and an overall height that varies from 196.7 to 225.1 inches. The internal cavity of the concrete cask has an inside diameter of 70.5 inches and is formed by a 1.75-inch thick cylindrical steel liner. The liner is a stay-in-place form and its thickness is determined by shielding requirements. The concrete is Type II Portland Cement, 144 lb/ft^3, 4000 psi compressive strength. Inner and outer re-bar cages are formed by vertical hook bars and horizontal ring bars. The reinforcing steel in the VCC body is located to provide adequate strength for the design loading conditions specified in Chapter 2. The airflow path is formed by the skid channels at the bottom (air entrance), the air inlet ducts, the gap between the MSB and the concrete cask interior, and the air outlet ducts. The cask cover plate is a 0.75-inch thick plate which provides additional shielding to reduce the skyshine radiation and provides a weather cover to protect the MSB from the environment and postulated tornado missiles. The cover is bolted in place.

The MSB located in the VCC internal cavity consists of an outer shell assembly, a shield lid, a structural lid, and the 24 element storage sleeve assembly. The shell is from 164.2 to 192.25 inches long and is fabricated from 1.0 inch thick SA-516 Gr 70 steel plate. The MSB shield lid and structural lid thicknesses are 9.5 inches and 3 inches, respectively. Both lids are welded to the MSB shell after fuel loading. They are also supported from below by the ring welded to the shell in a fabrication shop. The fuel storage sleeve assembly consists of specially fabricated 0.2 inch thick wall square tubes. Structural support of the storage sleeve assembly is provided by 12 support assemblies. Four support assemblies are located at each of three axial locations: at the top, middle and bottom of the basket assembly.

Inside the VCC internal cavity, the MSB rests on 1.7-inch-square by 0.3-inch-thick ceramic tiles that prevent contact between the metal surfaces of the MSB base and the VCC liner. As shown on the general arrangement drawings in Section 1.5, two different tile configurations are permitted in the VCC. These include a patter of 29 tiles arranged in multiple concentric rings or a pattern of 24 tiles uniformly spaced in a single ring around the perimeter of the MSB.

The following components are addressed in this section (See Section 2.2 for the design criteria):

- MSB lifting devices
- MSB shell and structural lid
- MSB internal basket (sleeves and lateral supports)
- MSB shield lid support ring
- MTC trunnions/connection to the shell
- MTC rails
- VCC concrete body
- VCC steel components (reinforcement, liner, cover lid)

All other VSC system components indicated on the general arrangement drawings in Section 1.5 are either not structural or not important to safety. They were appropriately included as loads for the components listed above.

The structural evaluations demonstrate that all the VSC components meet their structural design criteria and are capable of safely storing irradiated fuel.

3.1.2 DESIGN CRITERIA

The VSC structural design criteria are specified in Section 2.2. The load combinations of normal, off-normal, and accident loadings have been evaluated in accordance with ANS-57.9 for the VCC (see Table 2.2-2) and in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NC, 1986 Edition with the 1988 Addenda for the MSB (see Tables 2.2-3 and 2.2-4). The MSB Transfer Cask is a special lifting device and is designed to NUREG-0612 and ANSI N14.6.

3.2 WEIGHTS AND CENTERS OF GRAVITY

The component weights and centers of gravity for the VSC system are summarized in Table 3.2-1 and Figure 3.2-1. Because the different versions of the VSC have different calculated weights and centers of gravity, only the most limiting minimum and maximum calculated weights and centers of gravity are shown.

3.3 MECHANICAL PROPERTIES OF MATERIALS

The mechanical properties of steels used in the structural evaluation of the VSC are presented in Table 3.3-1.

The properties of the concrete are summarized below.

Temp. °F	Density (lb/ft3)	Thermal Conductivity (BTU/hr ft°F)	Compressive Strength (psi)	Thermal Expansion (in/in/°F)	Modulus of Elasticity (psi) 40 years
70	144	-----	4000	5.5×10^{-6}	-----
100	144	0.87	4000	5.5×10^{-6}	3.6×10^{6}
200	144	-----	4000	5.5×10^{-6}	3.6×10^{6}

These properties are from Reference 3.1 and ACI 349.

3.4 GENERAL STANDARDS FOR CASKS

3.4.1 CHEMICAL AND GALVANIC REACTIONS

The materials from which the VSC is fabricated (e.g., carbon steel and concrete) will not experience significant chemical, galvanic, or other reactions in helium, air, nitrogen, or water atmosphere. The technical basis for this is that all materials are essentially of equal potential in the Galvanic Series of Metals and Alloys.

3.4.2 POSITIVE CLOSURE

The VSC employs a positive closure system that is composed of multi-pass seal welds at five locations: 1) MSB shell bottom to end plate, 2) MSB shield lid to shell, 3) MSB structural lid to shell, 4) MSB draining, drying and backfilling penetration port to structural lid, and 5) MSB drain and vent cover plates to structural lid. The MSB welds will be helium leak checked to assure helium leakage less than 10^{-4} atm-cc/sec, repaired if necessary, in accordance with the *technical specification* in Section 12.4. The MSB pressure boundary shield lid, structural lid, and valve cover plate closure welds are liquid penetrant tested, as specified in the *technical specification* in Section 12.4.

3.4.3 LIFTING DEVICES

The VSC system has separate provisions for lifting the VCC, the MSB and the MTC. The VCC is lifted from below via the hydraulic roller skid inserted in the skid access channels. This bottom lift is the normal lifting mode employed when positioning the VSC at its final storage location. The bottom VSC lift must be able to accommodate the weight of the VSC system when it is fully loaded (i.e., VCC, MSB and 24 PWR fuel assemblies).

To support the MSB fuel loading operation, the MSB design includes provisions for lifting from above. This is accomplished via six hoist rings that are bolted to the MSB structural lid. These devices are capable of safely handling the fully loaded weight of the MSB (i.e., MSB plus 24 PWR fuel assemblies) so that the loaded and sealed MSB can be lowered into the VCC.

The MTC is lifted from above via two trunnions located on the outer shell approximately three feet from the top of the transfer cask. The trunnions are 10.75 inches in diameter and extend 4.5 inches from the MTC wall (excluding the 0.5-inch trunnion cover plate). Each trunnion is welded to the inner and outer steel shells of the MTC wall with a one-inch, full penetration circumferential weld. The two trunnions are capable of accommodating the combined weight of the MTC and a fully loaded wet MSB.

3.4.3.1 VSC Bottom Lift

The adequacy of the VSC lift via the hydraulic roller skid is evaluated by calculating the bearing and shear capacities of the cask bottom per ACI 349-85 and comparing them to the load due to bottom lift. The concrete bearing capacity is calculated below:

$$U_b = \phi f'_c A = 0.7 \bullet 4000 \bullet 8(\pi 6^2/4) = 633{,}345 \text{ lbs}$$

The shear capacity of the concrete section is calculated per Section 11 of ACI 349.

$$U_s = 8 \bullet \phi (V_s + V_c) = 8 \bullet 0.85[(2A_{\#6} f_y d / s) + 2\sqrt{f'_c} bd] = 528{,}130 lbs$$

where,

A	=	Total area of the eight hydraulic cylinders (6 inches in diameter each)
S	=	5 in - spacing of stirrups
d	=	6 in - depth of the section
b	=	18.8 in - circumference of cylinder saddle
$A_{\#6}$	=	0.44 in^2
f_y	=	60,000 psi

Clearly, shear and bearing capacities of the concrete cask bottom are well above the weight load of 1.05 W_{VCC} = 302,316 lbs.

Hence, the concrete will not crush during a bottom lift of the VSC. The stresses in the two foot thick concrete bottom portion of the VCC which is supported on the eight hydraulic lifters are negligible.

3.4.3.2 MSB Lift

The adequacy of the MSB lifting devices is demonstrated by considering each of the hoist rings, the MSB structural lid, and its weld to the shell. The structural analysis for each of these components includes evaluation of the following:

- Rated Capacity of the Hoist Ring
- Thread Shear

- Structural Lid Membrane and Bending Stress

Two hoist ring designs, each with one of two design loads, are addressed. To lift the maximum full-MSB design loads of 63,000 pounds and 69,000 pounds, the required hoist ring capacities for a design factor of ten against ultimate are 118,080 pounds and 129,330 pounds, respectively. The rated loads of the hoist rings with a typical commercial design factor of 5 are 24,000 pounds and 30,000 pounds, respectively.

To meet NUREG-0612 requirements, a dynamic load increase factor of 10% is used.

The ANSYS PC/LINEAR finite element program was used to evaluate the adequacy of the structural lid, shell and lid-to-shell weld. The model is shown in Figure 3.4-1 and consists of 144 nodes and 132 elements. 120° symmetry was used to shorten computer time for this run.

The results of this analysis show that the structural lid design factors for the two different MSB and hoist ring configurations are:

$K_u = 31.5 > 10.0$ $K_y = 15.2 > 6.0$

$K_u = 28.8 > 10.0$ $K_y = 13.8 > 6.0$

The above design factors are calculated using the limiting MSB loads applied to each hoist ring type: 63,000 pounds and 69,000 pounds, respectively. Thus, the MSB lifting devices meet the requirements of NUREG-0612 to lift a fully loaded MSB.

3.4.3.3 MTC Lift

Trunnions

Two MTC designs are addressed, one with an OD of 82.0 inches, and one with an OD of 83.5 inches (Table 1.2-6). The larger OD MTC is analyzed with a minimum fuel assembly weight of 1,110 pounds and a maximum fuel assembly weight of 1,380 pounds. The smaller OD MTC is analyzed with a minimum fuel assembly weight of 1,370 pounds and a maximum fuel assembly weight of 1,585 pounds. Results are presented in Table 3.2-1. The most limiting design is used in the trunnion analysis.

The trunnion design for the MTC may be subject to site-specific variation in order to interface with existing plant cranes and other equipment. However, a generic bounding design is presented here to show that the basic design will meet NUREG-0612 for heavy lifts using a single failure proof crane. If a particular site requires a trunnion of a different design, this new design will be addressed in the site-specific 10 CFR 50 heavy loads evaluation. The adequacy of the MTC trunnion design can be evaluated by considering the stress levels in the trunnion and the MTC wall. The MTC lifting trunnion design includes a dynamic load increase factor of 10% so that nonredundant load path lifts via single failure proof cranes will meet NUREG-0612. The results of this analysis show that the design factors for the combined shear and bending stress intensity in the trunnions are:

$K_u = 22.6 > 10.0$ $K_y = 11.2 > 6.0$

MTC Wall

To evaluate the structural integrity of the MTC wall, an ANSYS finite element analysis was performed with the model shown in Figure 3.4-2. The model focuses on the MTC wall region near the trunnion (i.e., the region within 36 inches of the trunnion). The model is constructed of 78 nodes and 65 elements, all being two-dimensional solid elements (ANSYS STIF No. 42). The model was loaded along the top boundary by a uniformly distributed load that represents the weight of the MTC cover plate and upper wall plate. The uniform load applied at the bottom of the model represent the remainder of the fully loaded MTC. Zero displacement boundary conditions were applied as indicated in the figure (i.e., zero displacements) at the trunnion interface locations and at the model symmetry line. There were also no radial displacements at the vertical model boundary.

The results of this analysis show that the design factors for the combined shear and bending stress intensity in the MTC wall are:

$$K_u = 10.1 > 10.0 \qquad K_y = 6.6 > 6.0$$

Thus, the MTC shell wall meets the requirements of NUREG-0612 to lift the fully loaded MSB and MTC.

Top Cover Plate and Bolts

The purpose of the top cover plate is to prevent inadvertent lifting of the MSB out of the MTC. This is desirable to ensure against undue radiation exposure to nearby workers. Therefore, the cover plate must have sufficient strength to support the MTC (since an inadvertent MSB lift would imply lifting the entire MTC).

The cover plate is 74 inches in diameter by 1 inch thick with a 60.5-inch diameter center opening. The central opening allows access to the MSB lifting eyelets when lowering the MSB into the VCC. The cover plate is fabricated from SA-516 Grade 70 steel. Sixteen 1-inch diameter bolts hold the cover plate in place. The bolt circle has a radius of 35.5 inches.

The MTC cover was evaluated for bending stresses and the cover bolts were evaluated for tension. The results of the analysis show that the maximum stress in the MTC cover plate is less than the allowable stress (based on AISC allowable stresses):

$$S_{cover} = 13.9 \text{ ksi} < 26.0 \text{ ksi}$$

The results of this analysis show that the maximum tensile load in the MTC cover plate bolts is less than the allowable load (based on the AISC Code):

$$S_{bolt} = 29 \text{ kips} < 34.6 \text{ kips}$$

Thus, the MTC cover plate and bolts are acceptable.

Shield Door Rail (and Welds)

The shield door rails must support the weight of a wet, fully loaded MSB and the weight of the shielding doors themselves. Two rail designs are considered in the analysis. Each design consists of a steel plate welded to the bottom of a rectangular solid section of steel. The rail is welded to the bottom plate of the MTC wall.

The support-rail design loads for the rail design configurations (considering a 10% dynamic factor) are:

$$W = 89{,}000 \bullet 1.1 = 97{,}900 \text{ lb}$$
$$= 92{,}900 \bullet 1.1 = 102{,}190 \text{ lb}$$

The structural integrity of the support rail is evaluated for shear and bending stresses. The support-to-rail weld and the rail-to-shell weld are evaluated for shear. The MTC doors are evaluated for bending stress. The results of this analysis are summarized below:

- The design factors for the combined shear and bending stress intensity in the MTC support rail are:

  $K_u = 14.5 > 10.0$ $K_y = 8.2 > 6.0$

  $K_u = 10.7 > 10.0$ $K_y = 6.1 > 6.0$

- The design factors for the shear stress in the MTC support-to-rail weld (only the more limiting of the two configurations was analyzed) are:

  $K_u = 13.0 > 10.0$ $K_y = 7.4 > 6.0$

- The design factors for the shear stress in the MTC rail-to-shell weld are:

  $K_u = 16.7 > 10.0$ $K_y = 9.5 > 6.0$

  $K_u = 17.7 > 10.0$ $K_y = 10.0 > 6.0$

- The design factors for the bending stress in the MTC doors are:

  $K_u = 39.0 > 10.0$ $K_y = 22.1 > 6.0$

  $K_u = 11.3 > 10.0$ $K_y = 6.4 > 6.0$

In summary, the shielding door rail structures, doors, and welds are adequate to accommodate a wet, fully loaded MSB.

3.4.4 VSC COMPONENTS UNDER NORMAL OPERATING LOADS

3.4.4.1 MSB Analysis

The MSB Design Loadings are defined in Table 3.4-1 (per ASME Section III, NCA-2142). The maximum temperatures are taken from Chapter 4 and presented in Table 3.4-2. The normal handling load is selected as ±0.5g in all directions simultaneously (Table 1.2-1). Sections below discuss each individual load and their combination.

3.4.4.1.1 MSB Thermal Stress Analysis

Description of Analytical Method

The VSC system was evaluated from the thermal stress standpoint by using separate and distinct models for the MSB and VCC. This approach is valid since these components are not structurally coupled. The MSB is free to thermally expand or contract relative to the VCC.

A three-dimensional finite element model using the ANSYS computer code was used to evaluate the thermal stresses in the MSB storage sleeve assembly. By taking advantage of the MSB circumferential symmetry, the storage sleeve assembly was modeled as a 1/8 slice that includes the entire length of the storage sleeve assembly and three storage sleeve assembly structural supports. The MSB storage sleeve assembly model is shown in Figure 3.4-3. The model consists entirely of shell elements and has 366 elements and 426 nodes. Symmetry boundary conditions were imposed along the zero and 45° boundaries (i.e., no displacement normal to the boundaries and no rotations about vertical and horizontal axes). Finally, the nodes in the model were coupled as indicated by the directive arrows shown in the model top view included in Figure 3.4-3. This nodal coupling assures that adjacent node stresses and displacements are correctly computed by ANSYS.

The MSB body thermal stresses were evaluated with the two-dimensional, axisymmetric ANSYS finite element model shown in Figure 3.4-4. The model consists of 43 nodes and 42 elements. Symmetry boundary conditions were imposed along the MSB centerline as indicated in Figure 3.4-4.

MSB Thermal Stress Analysis

The MSB body and storage sleeve assembly thermal stresses were evaluated separately since the radial gap between the sleeve assembly and the MSB shell structurally decouples the two members. This structural independence can be seen by examining the MSB geometry at its stress free temperature of 70°F. The nominal radial gap between the storage sleeve assembly structural support and the MSB shell is 0.65 inches (0.264 inches minimum). This gap is sufficient to allow unhindered storage sleeve thermal expansion relative to the shell as discussed below.

The results of the thermal hydraulic analyses presented in Chapter 4 are summarized in Table 3.4-2. These results show that the maximum MSB temperature gradient at the hottest axial section results from the –40°F ambient condition. For that condition, the maximum temperature of the support sleeve assembly and MSB shell at the hottest axial section are 586°F and 162°F, respectively. The evaluation of the MSB's differential thermal expansion is conservatively based on an upper-bound support sleeve assembly temperature of 589°F and a lower-bound MSB shell temperature of 22°F. Furthermore, the upper-bound support sleeve assembly temperature is conservatively assumed over its entire cross section. Based on closed-form hand calculations, the radial differential thermal expansion between the support sleeve assembly and the MSB shell for this condition is 0.118 inches compared to the minimum gap size of 0.264 inches. These results show that the MSB storage sleeve assembly expands freely within the MSB shell under the worst-case thermal loading conditions and that its thermal stresses can be evaluated separately from the MSB body.

For the overall evaluation of the thermal stresses in the MSB, the temperature distribution for the -40°F ambient case was used. The -40°F condition was used because it caused the highest thermal gradients in the MSB structure (423°F for -40°F, 404°F for 75°F, and 400°F for 100°F). The results of applying the MSB temperature distribution to the 1/8 slice finite element model shown in Figure 3.4-3 are summarized in Table 3.4-3. As shown in that table the maximum storage sleeve thermal stress is 52.0 ksi. This maximum thermal stress occurs in the outermost storage sleeve, between the bottom and middle storage sleeve assembly supports. The acceptability of this thermal stress level is evaluated along with other load combinations in Section 3.4.4.1.5.

Attention is next focused on the MSB body (shell, bottom plate, and structural lid). The MSB body temperature distribution for the -40°F ambient air case was input to the model shown in Figure 3.4-4. The resulting MSB body maximum thermal stresses are summarized in Table 3.4-4.

3.4.4.1.2 Dead Weight Load Analysis

Dead load stresses are calculated using two different approaches. MSB shell, bottom plate, and bottom weld stresses are calculated using a finite element model. This model combines the dead load, the normal operating pressure, and the vertical normal handling load. Dead load stresses for the remaining MSB components are calculated by scaling vertical drop results. The two approaches are described below.

MSB Shell, Bottom Plate, and Bottom Weld

The MSB shell is a 1.0-inch-thick cylinder. The bottom plate is a 3/4-inch-thick plate. The shell-to-bottom-plate weld is an unreinforced full-penetration weld. In the VCC, the MSB rests on ceramic tiles that prevent contact with the VCC base plate. As shown on the general arrangement drawings in Section 1.5, two different tile configurations are permitted in the VCC, including: (1) 24 tiles evenly spaced on a 30-inch radius, or (2) 29 tiles arranged with one tile at the center, a ring of 4 tiles on a 6-inch radius, and two rings of twelve tiles on 18-inch and 30-inch radii, respectively. The maximum dead weight stresses in the MSB shell, bottom plate, and bottom weld are calculated for the support condition provided by the first tile configuration (i.e., 24 tiles evenly spaced on a 30-inch radius). The support provided by this tile configuration results in the highest stresses in the MSB shell, bottom plate, and bottom weld since no support is provided over the interior of the bottom plate, which supports the entire weight of the MSB internals.

To analyze the stresses due to the loads applied at the tiles, an ANSYS finite element analysis is performed, based on a one-eighth segment of the MSB. In addition to the base plate, a section of the MSB wall is included in the model, to ensure that any interaction effects between the MSB base and shell wall are simulated. The MSB is modeled using the ANSYS quadrilateral elastic shell element, SHELL63. The gap between the underside of the MSB base plate and the inner surface of the storage cask liner plate is modeled using the ANSYS 3D point-to-point contact element, CONTAC52.

The contact stiffness was set at an arbitrarily high value of 1E6 lb/inch.

The finite element model is shown in Figure 3.4-7.

Boundary Conditions

The MSB is restrained from moving vertically at the location of the ceramic tiles. In addition, the contact elements are fixed at the surface simulating the cask boundary. Symmetric boundary conditions are imposed on the nodes positioned on "cut" edges of the MSB base and wall.

Applied Loading

The finite element analysis is based on the following MSB component weights:

Weight of the MSB	= 21,686 lb
Weight of the structural lid	= 2,409 lb
Weight of the shield lid	= 6,449 lb
Bounding weight of the MSB and contents	= 70,000 lb
Bounding weight of contents	= 40,209 lb

The resulting pressure load on the base plate due to the weight of the contents is 13.5 psi.

Only the lower 30 inches of the MSB shell is included in the finite element model. To account for weight missing in the model, force is applied to the top nodes of the modeled section. The total load applied is equal to the weight of the MSB not included in the model, or 3,443 pounds. This load is uniformly distributed to the top end of the shell.

Included in the model are a 1*g* vertical normal handling load and an 8.9 psig normal operating pressure, which is applied to both the MSB shell and base plate.

The maximum stresses in the MSB shell, bottom plate, and bottom weld from the finite element analysis for dead weight loading are summarized in Table 3.4-5. The stresses are combined with horizontal normal handling and thermal stresses. The combined stresses are compared with appropriate allowable stresses.

Other MSB Components

Shield Lid Support Ring Weld

The shield lid support ring is connected to the shell with a ½-inch partial penetration weld. An effective weld throat of 3/8 inch is assumed for the weld evaluation. Conservatively, it is assumed that the load from the shield lid and support plate are carried only by the support ring weld, i.e., no credit is taken for the load carried by the shield lid weld.

Shield Lid

The shield lid consists of one 2.5-inch steel plate, one 2.0-inch neutron shield layer, and one 5.0-inch steel plate. For conservatism, the structural capacity of the neutron shield layer and the 2.5-inch steel plate are neglected, and the shield lid is represented by a single plate.

Shield Lid Weld

The shield lid weld is a ¼-inch partial penetration weld. It is assumed that the load from the shield lid and support plate are carried only by the shield lid weld, i.e., no credit is taken for the load carried by the shield lid support ring weld.

Structural Lid Weld

The structural lid weld is a 3/4-inch full-penetration weld. It is assumed that the load from the structural lid is carried only by the structural lid weld, i.e., no credit is taken for the load carried by the shield lid weld or the shield lid support ring weld.

Structural Lid

The structural lid consists of one 3.0-inch steel plate. No credit is taken for the load carried by the shield lid.

Sleeve Assembly

The sleeve assembly consists of 24 welded 9.2-inch-square structural tubes, each with a thickness of 0.20 inches.

Dead load stresses for the MSB structural lid, structural lid weld, shield lid, shield lid weld, support ring weld, and sleeve assembly are combined with other normal operating stresses and compared with appropriate allowable stress levels in Table 3.4-5. As shown in this table, all MSB components and welds are within code allowable levels for normal operating conditions.

3.4.4.1.3 MSB Internal Pressure Analysis

MSB internal pressure stresses are calculated using the maximum MSB internal normal operating pressure in two different finite element models. One model is used for the MSB shell, bottom plate, and bottom weld. The other model is used for the remaining MSB components. The MSB minimum internal pressure is used to evaluate the MSB for buckling in accordance with ASME Section III, NC-3133.3.

Maximum and Minimum MSB Internal Normal Operating Pressures

The maximum MSB normal operating internal pressure used in the MSB internal pressure analysis is calculated based on the following assumptions:

- A failure of 1% of the fuel rods, with 100% of the rod fill gas escaping each failed rod
- A release of 30% of the generated fuel pellet fission gas to the interior gas region
- A release of 30% of the generated BPRA fission gas to the interior gas region
- Storage of the MSB at steady state in the transfer cask with a 75°F ambient temperature

The maximum normal operating internal pressure is 5.23 psig, based on an average gas temperature of 439°F.

A minimum MSB internal pressure of –8.0 psig is calculated based on the following assumptions:

- The helium gas is at the maximum temperature for the normal storage condition when the MSB is filled.
- No fuel rod failures occur.
- The design basis fuel heat generation rate decays to 0 kW and the basket gas volume reaches steady state at the minimum ambient condition (-40°F).

MSB Shell, Bottom Plate, and Bottom Weld Pressure Stresses

As discussed in Section 3.4.4.1.2, a finite element analysis combining the dead load, the normal operating pressure, and the vertical normal handling load is performed for the MSB shell, bottom plate, and bottom weld. The finite element model is shown in Figure 3.4-12. A conservative normal operating pressure of 8.9 psig is used in the analysis. Analysis stress results are combined with horizontal normal handling and thermal stresses in Table 3.4-5. The combined stresses are compared with ASME Code allowables.

Remaining MSB Component Pressure Stresses

The finite element analysis for the MSB structural lid, structural lid weld, shield lid, shield lid weld, and support ring weld is conservatively based on an internal normal operating pressure of 10 psig. An axisymmetric model with ANSYS element STIF51 is used for the analysis, and the stresses for normal operating, off-normal operating, and accident pressures are determined. The calculated normal operating pressure stresses are summarized in Table 3.4-5, and are combined with other normal operating stresses before being compared with the ASME Code allowables.

MSB Buckling

Because the pressure in the MSB may be as low as –8.0 psig, the allowable external pressure has been calculated in accordance with ASME Section III, NC-3133.3. The resulting allowable pressure differential is 210 psig, which is much greater than 8.0 psig. Therefore, buckling will not occur.

3.4.4.1.4 MSB Handling Stresses

MSB normal handling stresses are calculated using two different methods. The MSB shell, bottom plate, and bottom weld vertical normal handling stresses are calculated and combined with normal pressure and dead load stresses in a finite element analysis. The horizontal normal handling stresses for these components are calculated by scaling horizontal drop stresses. The finite element, horizontal normal handling, and thermal stresses are then combined and compared to allowable stresses. The remaining MSB component normal handling stresses are calculated by scaling and combining dead weight and horizontal drop stresses.

Structural Lid, Structural Lid Weld, Shield Lid, Shield Lid Weld, Support Ring Weld, and Sleeve Assembly

The scaling calculation scales the 1g dead weight (vertical) and 44g horizontal drop stresses by the corresponding vertical and horizontal handling accelerations.

The acceleration for normal handling is assumed to be 0.5 g acting simultaneously in the vertical and two horizontal directions:

$$a_{handling\_vert} = 0.5g$$

$$a_{handling\_horiz} = 0.5g\sqrt{2} = 0.71g$$

The accelerations for the deadweight and horizontal drop are :

Dead weight acceleration, $a_{deadweight}$ = 1g

Horizontal drop acceleration, a_{drop_hor} = 44g

A dynamic load factor (DLF) of 2 is applied to the accelerations to obtain conservative equivalent static loads.

The stress intensities from the vertical and horizontal accelerations are assumed to add directly. The handling stress is computed as follows:

$$HandlingStress = \left[\left(\frac{a_{handling\_vert}}{a_{deadweight}}\right)SI_{vert} + \left(\frac{a_{handling\_horiz}}{a_{drop\_horiz}}\right)SI_{horiz}\right] \times DLF$$

where,

SI_{vert} = stress intensity due to 1g dead weight analysis

SI_{horiz} = stress intensity due to 44g horizontal drop analysis

The normal handling stresses are calculated using the above combination, and the results are summarized in Table 3.4-5. The sum of the normal operating stresses is compared to the ASME Code allowables.

MSB Shell, Bottom Plate, and Bottom Weld

As discussed in Section 3.4.4.1.2, a finite element analysis combining the dead load, the normal operating pressure, and the vertical normal handling load is performed for the MSB shell, bottom plate, and bottom weld. Analysis stress results are combined with horizontal normal handling stresses (calculated as described above) and thermal stresses in Table 3.4-5. The sum of the stresses is compared to the ASME Code allowables.

3.4.4.1.5 MSB Load Combination

The MSB design loadings are defined in Table 3.4-1. The stresses due to the loadings are presented and evaluated in Table 3.4-5. The reduction factor of 0.75 has been applied to allowable stresses for the partial penetration welds (ASME Section III, NC-3264.6). The analysis demonstrates that the stresses are within allowable limits.

The MSB Fatigue Evaluation

Fatigue effects on the MSB are addressed using the criteria contained in NC-3219.2 of the ASME Code, Section III. Fatigue analysis need not be performed provided the criteria of Condition A are met. A summary of the criteria and their application to the MSB are presented in the following paragraphs.

According to NC-3219.2, fatigue analysis is not mandatory for materials having tensile strength not exceeding 80 ksi (provided for the MSB components) and the expected number of cycles (a) + (b) + (c) + (d) is less than 1,000.

(a) Full Range Pressure Cycles

The normal operating pressure for the MSB is 14.7 psia (0 psig). The full range pressure cycles are due to: vacuum drying, two pressure tests, postulated failure of all fuel rods and significant ambient temperature changes (conservatively assumed to occur 10 times per year during 50 years of the cask lifetime).

Therefore, the total number of fluctuations of this type is (a) = 1 + 2 + 1 + 10 * 50 = 504.

(b) Expected Number of Pressure Cycles

The expected number of pressure cycles of this type is 0 since fluctuations in weather conditions need not be considered here.

(c) Effective Number of Changes in Metal Temperature Between Adjacent Points

The distance between adjacent joints as defined in the Code is $2\sqrt{Rt} = 9.6$ inches. It can be easily seen from the thermal analysis results that although the temperature of the MSB changes significantly due to weather conditions, the change in the temperature difference between two adjacent points never exceeds 50°F. Therefore, the effective number of cycles of this type is 0.

(d) Only for Vessels With Welds Between Materials With Different Coefficients of Expansion

The only pressure boundary material for the MSB is SA-516 Grade 70, and therefore, the number of cycles of type (d) is 0.

The discussion presented in the preceding paragraphs shows that (a) + (b) + (c) + (d) = 504 and less than 1000. Thus, all criteria of Condition A are met and the MSB is exempt from the fatigue analysis.

3.4.4.1.6 MSB Pressure Test

As discussed in Table 1.2-7, a helium leak test of the shield-lid-to-shell weld will be performed in lieu of the pressure testing requirements of ASME Section III, NC-6000. The test pressure of 0.5 atm (7.3 psig) exceeds 1.25 times the maximum normal operating pressure of 5.23 psig, as required by NC-6221. The leak test pressure is less than the pressure used in the MSB internal pressure analysis summarized in Section 3.4.4.1.3. Therefore, the pressure test stresses are less than the maximum normal operating stresses and meet the requirements of ASME Section III, NC-3218. Because (a) the helium leak test is more sensitive than the ASME Code pneumatic test, (b) other pressure-retaining welds are volumetrically examined, and (c) the maximum normal operating pressure of less than 5.23 psig is very low compared to that of a typical vessel designed to the ASME Code, the helium leak test of the shield-lid-to-shell weld is adequate for the MSB.

3.4.4.2 VCC Analysis

Only three load components act on the VCC: dead load, live load and thermal load due to differential thermal expansion. The other potential sources of load are discussed in Chapter 11. The three components of VCC loading are calculated below. The results of combining the loads and comparing the VCC stress levels to allowable limits demonstrate that the VCC is structurally sound and meets the structural requirements of ANS-57.9. These results are summarized in Table 3.4-6.

3.4.4.2.1 VCC Dead Load

The stress due to the dead load on the VCC bottom is conservatively calculated by assuming the total weight of the fully loaded VSC is taken by the concrete bottom only over the surface area of the MSB. The resulting stresses are:

σ_{bottom} = 0.1 ksi

τ_{bottom} = 0

The resulting stresses in the concrete wall due to VCC dead weight are:

σ_{wall} = 0.02 ksi

τ_{wall} = 0

These calculated stresses have been increased by 5% before evaluation in Table 3.4-6 as required by ANS-57.9.

3.4.4.2.2 VCC Live Load

The VCC is subject to two live loads: 1) the snow and ice load of 403 psf described in Section 2.2.4 and 2) the weight of the MTC and fully loaded MSB. Both of these live loads act on the top of the VCC. The snow load is uniformly distributed over the top of the cask and, as shown below, represents a negligible contribution to VCC stress levels.

σ_{snow} = 0.004 ksi

In calculating the effect of the weight of the MTC and MSB it was assumed that the load was only supported by the VCC inner steel liner. No structural credit was taken for the concrete. This conservative assumption was made since the loaded MTC is designed to rest directly on the top plate of the VCC liner. As shown below, the resulting stresses in the liner and cask bottom are well within the allowable levels.

σ_{wall} = 9.68 ksi

σ_{bottom} = 0.05 ksi

τ_{bottom} = 0

3.4.4.2.3 VCC Thermal Stresses

Description of the Analytical Method

The VSC concrete cask thermal stress analysis was performed based upon the ANSYS finite element model shown in Figure 3.4-5. The model is a pie sector of the VCC that includes the full length of the cask wall and complete thickness of the cask bottom. A narrow slice model is justified because of the symmetrical nature of the VSC.

The pie sector size was selected to achieve an adequate representation of the vertical hook rebar in the VCC. The VCC model consists of 432 elements and 610 nodes. The element types used in the model are summarized below:

VCC Component	ANSYS STIF Number	ELEMENT Description
Concrete	45	3-D Solid
Steel Liner	63	Shell
Rebar	8	3-D Spar
Gaps	52	3-D Interface

The gap elements were included in the model to allow load transfer to the reinforcement bars since concrete was assumed to be ineffective in carrying tensile loads. The stiffness associated with the gap elements was chosen to be approximately twice the stiffness of the adjacent concrete elements. The gap element nodes are coupled in the radial direction to account for shear transfer between the adjacent concrete elements. Rebar/concrete interaction is accomplished by coupling the rebar nodes to be the adjacent concrete nodes in the axial and radial directions.

The following symmetry boundary conditions were imposed on the nodes along the model boundaries:

- Zero displacement normal to model boundary
- Zero rotations about vertical and horizontal axes for all nodes with this degree of freedom.

Temperatures from the thermal analysis presented in Chapter 4 of this report were input for all nodes. A stress free temperature of 70°F was used in the analysis.

An iterative solution procedure was used because of the presence of the non-linear gap elements. The ANSYS steady convergence criteria was used to minimize the number of iterations within a load step. The analysis also involved relocating and/or introducing gap elements in the model until a solution was achieved that included insignificant concrete tensile stresses (i.e., less than 381 psi which is the allowable tensile stress for concrete).

Results

The thermal stresses in the VCC were calculated by inputting the VCC nodal temperatures for the all inlets blocked case into the VCC finite element model described above. This condition presents the worst possible case for the VCC thermal stresses since it produces the highest gradient through the concrete wall. Figure 3.4-6 summarizes the VCC thermal stress distribution. The maximum thermal stresses for each of the VCC structural components are listed in Table 3.4-7. The acceptability of these thermal stress levels is included in the VCC load combination evaluated in Table 3.4-6. For load combinations 6 and 8 the thermal loads in the critical sections are 0 due to the self-balancing nature of the thermal stresses across the entire cask section (which resists the tornado missile impact) and due to their axisymmetric distribution for the drop.

3.4.4.3 MTC Stress Calculations and Comparison with Allowables

The MTC is considered only as a lifting device. Its adequacy is demonstrated in Section 3.4.3.3.

3.4.5 COLD

3.4.5.1 MSB

Severe cold environments are analyzed and reported in Section 11.1 of this report. As shown in that section, the temperature of the structures with a full heat load will not fall to below levels where brittle fracture would become an issue. Furthermore, an analysis has been performed for the cask in severe cold conditions after 20 years of storage. The limiting required material toughness is 13.4 ft/lbs at minus 30°F. To apply additional conservatism, the following requirements will be met:

The MSB shell, bottom plate, shield lid, top plate, structural lid, and valve cover plates and associated welds are considered pressure-retaining. Impact testing of these items will meet the requirements of NC-2300, except as follows: Charpy V-Notch impact testing will be performed at a temperature no higher than minus 50°F with the additional requirement of a minimum of 45 ft-lbs at 0°F for casks fabricated after issuance of Amendment 2. Specimens will exhibit an average of 15 ft-lbs of absorbed energy, with the minimum for any single specimen of at least 10 ft-lbs. The requirements of NC-4335 will be met for impact testing of welding procedure qualifications used on pressure retaining material, amended as follows: lateral expansion or absorbed energy values for the HAZ and unaffected base material will meet the requirements of NC-4335.2, using the above test temperature and acceptance criteria. Plates that will be subjected to more than 0.5% strain (i.e., MSB inner and outer shells) will be impact tested after forming. Alternatively, a forming procedure qualification test, to show that the forming process will predictably result in acceptable impact toughness, will be performed in accordance with NC-4213.

An evaluation of the thermal gradients in the VCC wall for all cases presented in Table 4.1-1 showed that the highest temperature gradients exist for the case when all the inlets are blocked. Therefore, the VCC thermal stresses calculated for this case bound those of the severe cold ambient case.

The MSB thermal gradients for the -40°F ambient case were somewhat more severe than those for other cases. Therefore, the -40°F temperature profiles were used to evaluate the MSB thermal stresses. The results of that evaluation are presented in Section 3.4.4.1.1. Those results demonstrate that even for severe cold environments, MSB thermal stresses are within code allowable levels and are well above any levels where brittle failure might be a concern.

3.4.5.2 MTC

The minimum design temperature for the MTC is 40°F. The following MTC fracture toughness requirements provide assurance that the MTC will not fail in a brittle manner at or above 40°F.

MTC load path materials are impact tested in accordance with ASME Section VIII, Division 2, Article M-2, amended as follows:

Plates that are subjected to more than 0.5% strain (i.e., MTC inner and outer shells) are impact tested after forming.

Alternatively, to show that the forming process results in predictably acceptable impact toughness, a forming procedure qualification test is performed in accordance with ASME Section III, NF-4213.

The impact test temperature is no higher than 0°F. The specimens exhibit an average of 15 ft-lbs of absorbed energy, with the minimum for any single specimen of 10 ft-lbs. Base metals exempted by Section VIII, Division 2, AM-214 are not impact tested.

Charpy V-notch tests of the weld metal and heat affected zone of the welding procedure qualification test assembly are performed when the thickness of the weld exceeds 5/8 inch, and are performed even when the base metals being joined are exempted by Section VIII, Division 2, AM-218.

Charpy V-notch test specimens representing both the heat affected zone and the unaffected base metal are tested at or below 0°F. Either the average absorbed energy value of the three HAZ specimens must be equal to or exceed the average absorbed energy value of the unaffected base metal specimens, or a temperature or absorbed energy correction factor must be applied to the production base materials.

3.5 FUEL RODS

The VSC system is designed to limit fuel clad temperatures to below levels where zircaloy degradation is expected to lead to fuel clad failure. The precise evaluation of the allowable temperature includes many site-specific fuel parameters (including the age of the fuel) and does not lend itself to generic application. However, as shown in Appendix C, the range of acceptable temperatures for most PWR fuel will fall between 378 and 450°C. The methodology used to determine this range is that described in References 1.1 and 4.1. It is felt that this temperature limit is conservative from the information shown in Appendix C and is consistent with licensing precedents in this country and Germany. Section 4.3 provides additional information on the fuel temperature limits.

The fuel for storage will be evaluated based on the methodology presented in Chapter 2. In general, the fuel will experience maximum temperatures below the range of 370 - 400°C.

Table 3.2-1 - VSC System Weights and Centers of Gravity

ITEM DESCRIPTION	WEIGHT (lbs)	CENTER OF GRAVITY (inches above bottom of item)
VCC Weather Cover Plate	1,125	N/A
MSB Structural Lid	2,409	N/A
MSB Shield Lid	6,449	N/A
MTC Lid	405	N/A
MSB (Empty, w/o Lids)	19,570–21,686	75.9–86.0
MSB (Loaded w/Fuel and Water, w/Shield Lid, w/o Structural Lid)	67,504–80,261	84.0–97.8
MSB (Loaded, w/Fuel, w/Shield Lid and Structural Lid)	56,860–68,685	87.7–101.9
VCC (Empty, w/o Cover Plate)	189,986–216,277	95.5–109.6
VCC and MSB (Empty, w/o Lids)	209,556–237,963	96.3–110.0
VCC and MSB (Loaded, w/Fuel, w/Shield Lid and Structural Lids	252,793–287,920	101.5–115.4
MTC (Empty w/o Lid)	110,085–118,939	78.9–92.7
MTC and MSB (Empty, w/o Lids)	129,655–140,625	N/A
MTC and MSB (Loaded, w/Fuel and Water, w/Shield Lid, w/o Structural Lid)	178,872–192,203	85.0–98.3
MTC and MSB (Loaded, w/Fuel, w/Shield Lid and Structural Lid)	170,339–181,964	86.1–99.5

Table 3.3-1 - Mechanical Properties of Steels Used in the VSC

Material Specification	Type or Grade	Temp. (°F)	Yield[1,6] S_y	Ultimate[2,7] S_u	Allowable[3,8] S_m	Elastic Modulus[4] (10^6 psi)	Mean Coefficient of Thermal Expansion[5] (10^{-6} in/in/°F)
ASME SA-516	70	70	38.0	70.0	23.3	29.5	-----
		100	38.0	70.0	23.3	-----	5.53
		200	34.6	70.0	23.1	28.8	5.89
		300	33.7	70.0	22.5	28.3	6.26
		400	32.6	70.0	21.7	27.7	6.61
		500	30.7	70.0	20.5	27.3	6.91
		600	28.1	70.0	18.7	26.7	7.17
		700	27.4	70.0	18.3	25.5	7.41
ASTM A-36		70	36.0	58.0	12.6	Use SA-516 data	
		100	36.0	58.0	12.6		
		200	32.8	58.0	12.6		
		300	31.9	58.0	12.6		
		400	30.8	58.0	12.6		
		500	29.1	-----	12.6		
ASTM A-588		70	50	70.0	23.3	Use SA-516 data	
		100	50	70.0	23.3		
		200	47.5	70.0	23.3		
		300	45.6	70.0	23.3		
		400	43.0	70.0	23.3		
		500	41.8	70.0	23.3		

References:
(1) ASME, Sec. III, Tables I-2.1
(2) ASME, Sec. III, Tables I-3.1
(3) ASME, Sec. III, Tables I-1.1
(4) ASME, Sec. III, Table I-6.0
(5) ASME, Sec. III, Table I-5.0
(6) ASME, Code Case N-71-14, Table 4
(7) ASME, Code Case N-71-14, Table 5
(8) ASME, Sec. III, Table I-7.1

Table 3.4-1 - MSB-24 Design Loadings

	Components	
Load	**MSB Pressure Boundary**	**MSB Internals**
Dead Weight	X	X
Thermal	X	X
Internal Pressure	X	
Handling Load	X	X

Table 3.4-2 - Summary of Maximum VSC Temperatures for Structural Evaluation

	Analysis Case					
	1	**2**	**3**	**4**	**5**	**6**
Solar heat load	no	no	yes	yes	no ½ inlets blocked	no all inlets blocked
Ambient Temperature (°F) (average over a 24-hour period)	75	-40	100	125	75	75
Extreme Temperature (°F)						
cask outer surface	85	-32	136	189	86	87
concrete/liner interface	180	41	214	248	191	206
MSB outer surface	269	162	294	320	276	297
max sleeve	678	589	699	721	683	694

Table 3.4-3 - Summary of Results
MSB Storage Sleeve Assembly Thermal Stresses

Location on Outermost Storage Sleeve*	Thermal Stress Q (ksi)**
At Top Support	29.6
Between Top/Middle Support	52.0
At Middle Support	43.8
Between Middle/Bottom Support	51.9
At Bottom Support	33.2

* All maximum thermal stresses occur on the outside surface of the outermost storage sleeve.

** Acceptability of storage sleeve assembly thermal stresses is evaluated in combination with other loads in Section 3.4.4.1.5.

Table 3.4-4 - Summary of Maximum MSB Body Thermal Stresses (ksi)

Component	Maximum Thermal Stress Q (ksi)
Bottom Plate	19.4
Bottom Plate Weld	19.4
Shell	1.4
Structural Lid	0.2
Structural Lid Weld	0.5
Shield Lid Weld	1.3

Table 3.4-5 - MSB Maximum Stress Evaluation of Normal Operations

Component	Stresses	Dead Load	Pressure Stress	Vert & Horiz – Normal Handling	Dead Load+Pressure +Vert Normal Handling	Horiz-Normal Handling	Thermal Stress	Sum of Stresses	ASME Code Limit
MSB Shell	P_m	Note 1	Note 1	Note 1	1.02	0.69	0.00	1.7	20.5
	$P_L + P_b$	Note 1	Note 1	Note 1	3.74	1.61	0.00	5.4	30.7
	$P_L + P_b + Q$	Note 1	Note 1	Note 1	3.74	1.61	1.37	6.7	61.5
Bottom Plate	P_m	Note 1	Note 1	Note 1	0.39	1.05	0.00	1.4	22.5
	$P_L + P_b$	Note 1	Note 1	Note 1	27.50	1.41	0.00	28.9	33.7
	$P_L + P_b + Q$	Note 1	Note 1	Note 1	27.50	1.41	19.40	48.3	67.5
Structural Lid	P_m	0.01	0.05	0.72	---	---	0.00	0.8	22.5
	$P_L + P_b$	0.04	1.19	1.57	---	---	0.00	2.8	33.7
	$P_L + P_b + Q$	0.04	1.19	1.57	---	---	0.18	3.0	67.5
Bottom Weld	P_m	Note 1	Note 1	Note 1	1.02	0.93	0.00	2.0	22.5
	$P_L + P_b$	Note 1	Note 1	Note 1	27.50	1.41	0.00	28.9	33.7
	$P_L + P_b + Q$	Note 1	Note 1	Note 1	27.50	1.41	19.40	48.3	67.5
Structural Lid Weld	P_m	0.04	0.25	0.38	---	---	0.00	0.7	16.9
	$P_L + P_b$	0.08	2.35	1.61	---	---	0.00	4.0	25.3
	$P_L + P_b + Q$	0.08	2.35	1.61	---	---	0.50	4.5	50.6
Shield Lid	P_m	0.03	0.00	0.49	---	---	0.00	0.5	22.5
	$P_L + P_b$	0.31	0.00	1.00	---	---	0.00	1.3	33.7
	$P_L + P_b + Q$	0.31	0.00	1.00	---	---	0.00	1.3	67.5
Shield Lid Weld	P_m	0.27	1.26	0.60	---	---	0.00	2.1	16.9
	$P_L + P_b$	0.27	4.84	1.00	---	---	0.00	6.1	25.3
	$P_L + P_b + Q$	0.27	4.84	1.00	---	---	1.30	7.4	50.6
Support Ring Weld	P_m	0.18	0.00	0.18	---	---	0.00	0.4	16.9
	$P_L + P_b$	0.18	0.00	0.18	---	---	0.00	0.4	25.3
	$P_L + P_b + Q$	0.18	0.00	0.18	---	---	0.00	0.4	50.6
Sleeve Assembly	P_m	0.06	0.00	2.02	---	---	0.00	2.1	20.5
	$P_L + P_b$	0.06	0.00	2.09	---	---	0.00	2.2	30.7
	$P_L + P_b + Q$	0.06	0.00	2.09	---	---	52.0	54.2	61.5

Note 1: The results from the individual analyses of dead load, pressure stress, and vertical + horizontal normal handling are not combined for loads involving the MSB Shell, Bottom Plate, and Bottom Weld. For these components, a separate finite element analysis, which includes dead load + pressure stress (8.9 psig) + vertical normal handling, was completed, with the results as shown. The horizontal normal handling and thermal stresses are added separately to give the sum of the stresses. For the other components, the individual stress intensity results are combined to give the sum of the stresses.

Table 3.4-6 - VCC Structural Load Combination Evaluation

No.	Load Combination	Stress/Load	Critical Section	Text Reference	Maximum Stress/Load	Allowable Stress/Capacity
1	1.4D + 1.7L	Shear Normal	Bottom Bottom	3.4.4.2	0 ksi* 0.22	0.11 ksi 2.4
2	1.4D + 1.7L + 1.7H	Same as Combination No. 1 (H = 0)				
3	0.75(1.4D + 1.7L + 1.7H + 1.7T_o + 1.7W)	Shear Normal	Top wall-to-bottom	3.4.4.2	0.10 0.68	0.11 2.4
4	0.75(1.4D + 1.7L + 1.7H + 1.7T_o)	Bounded by Combination No. 3				
5	D + L + H + T_o + E	Shear Normal	wall-to-bottom wall-to-bottom	3.4.4.2, 11.2.5.2	0.09 0.6	0.11 2.4
6	D + L + H + T_o + A**	Shear Moment	Axial Axial	Calc. Pkg VSC02.6.2.3.15	81.5*** 1308***	34.0*** 1,486***
7	D + H + L + T_a	Bounded by Combination No. 3 because T_o = T_a				
8	D + L + H + T_o + W_t	Shear Moment	Outlets plane Wall-to-bottom	3.4.4.2, 11.2.3.2	457*** 91,476***	1,110*** 223,600***

Note: Soil, flood, and wind pressures are not included; they are negligible and are not coincident with live load of MTC on top of VCC, and are therefore bounded.

* All units ksi unless noted.

** 1 ft. long axial section of the VCC wall has been used.

*** Capacities calculated per ACI-349 are used for these load combinations instead of stresses. The capacities are in kips (force) or kips•in (moment).

Table 3.4-7 - Summary of Maximum VCC Thermal Stresses
75°F Ambient Air, Normal Operation

Component	Q (ksi)	Location
Concrete	0.4 (compression)	Bottom, inside wall
	0.08 (shear)	Top, 180" from VCC bottom
Rebar		
Vertical Hook	28.8	Approx. 160" from VCC bottom
Circumferential Hoop	14.6	VCC midplane
Liner	2.1	Upper 1/3 of liner
Weather Cover Lid	5.3	Outer edge
Bottom Plate	6.2	Outer edge

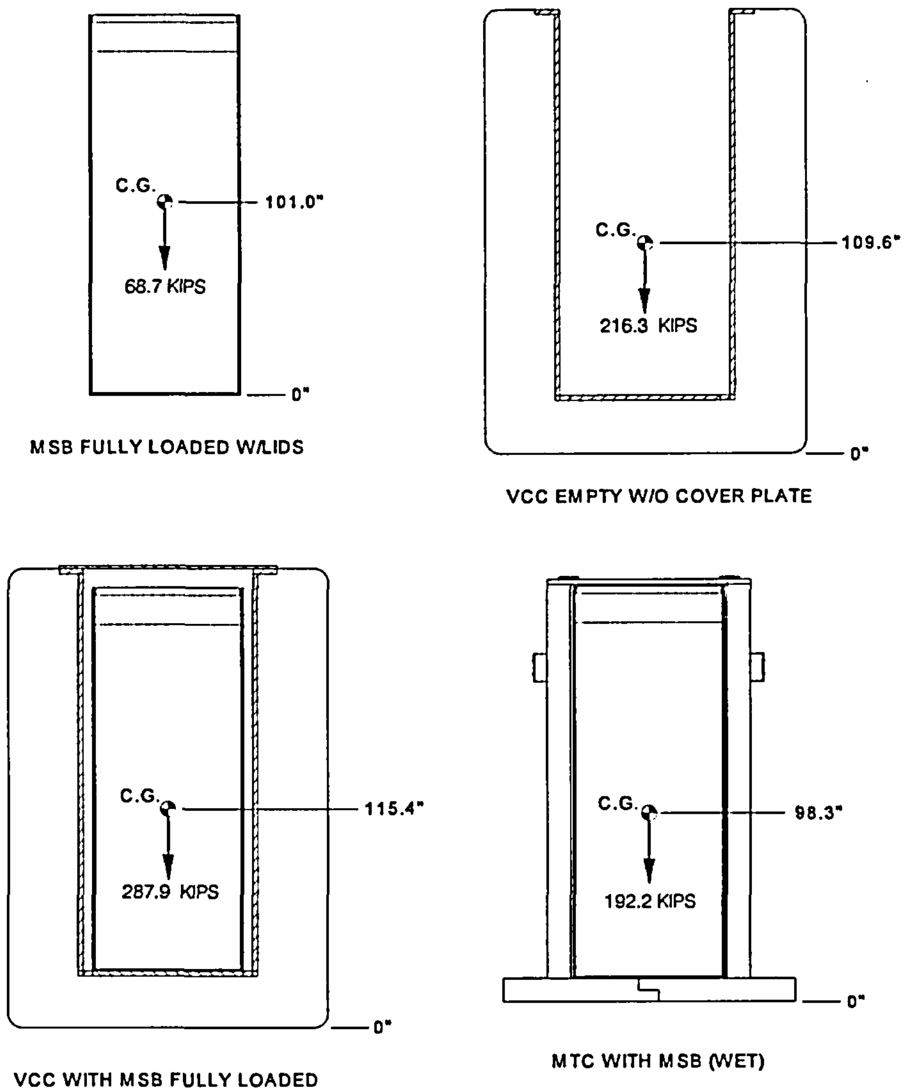


Figure 3.2-1 - VSC-24 System Weights and Centers of Gravity

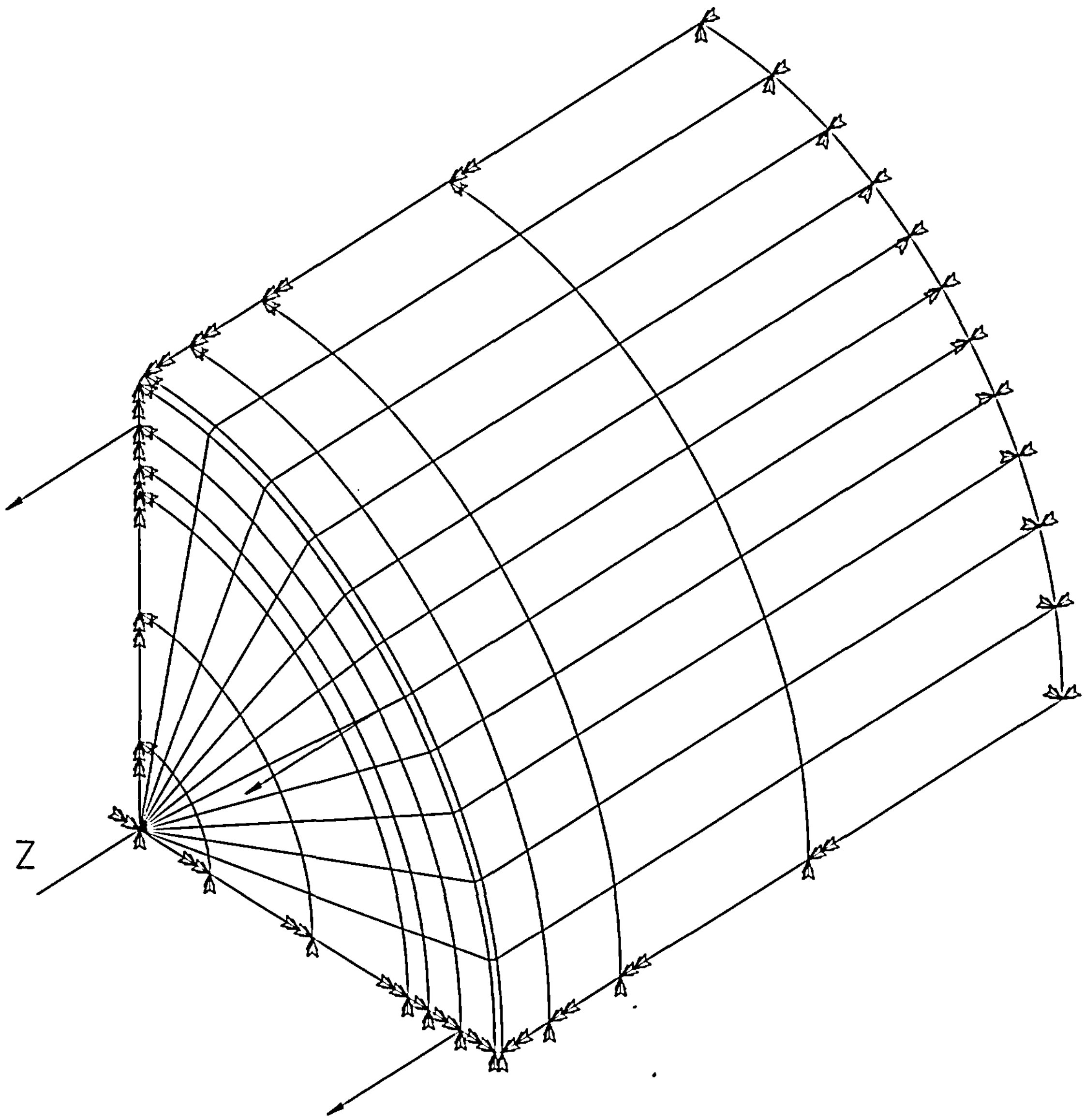


Figure 3.4-1 - Finite Element Model for the MSB Lift Analysis

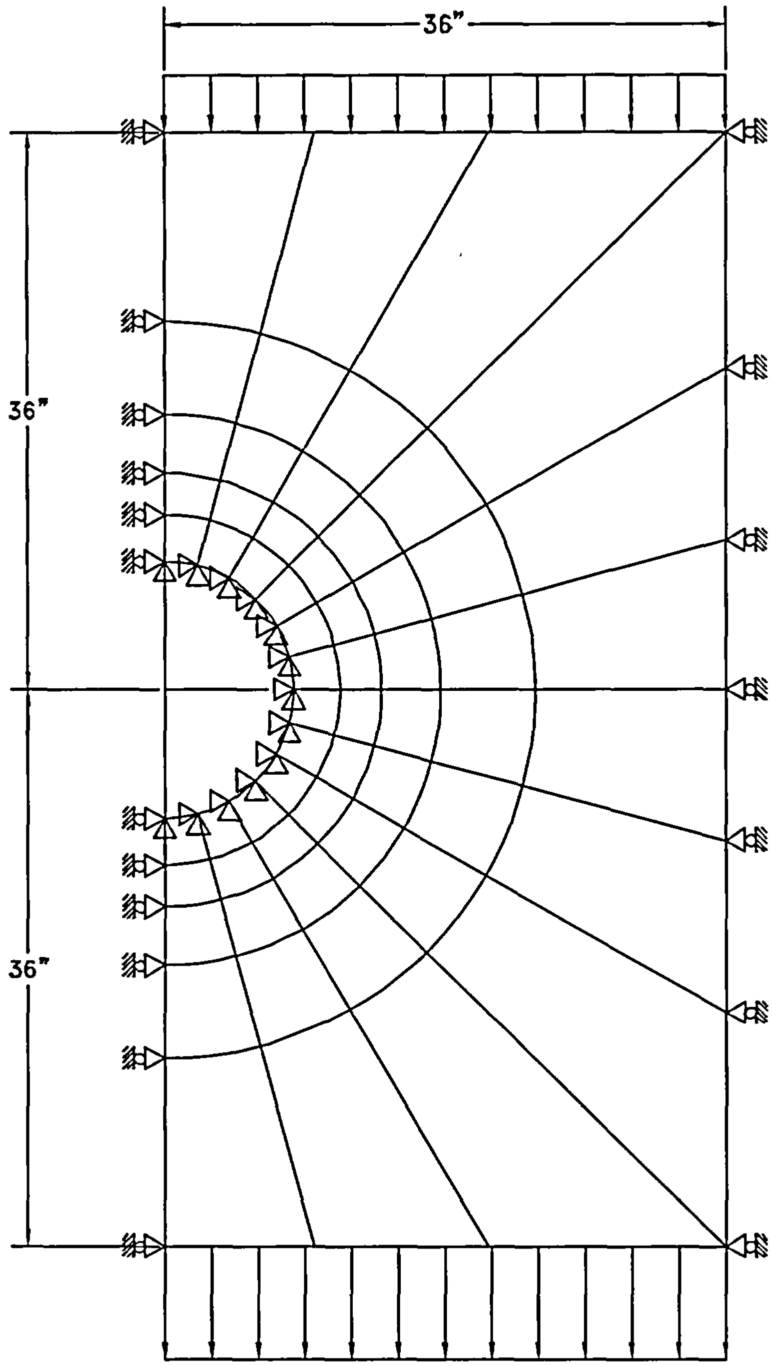


Figure 3.4-2 - Finite Element Model of MTC Wall Near Trunnion

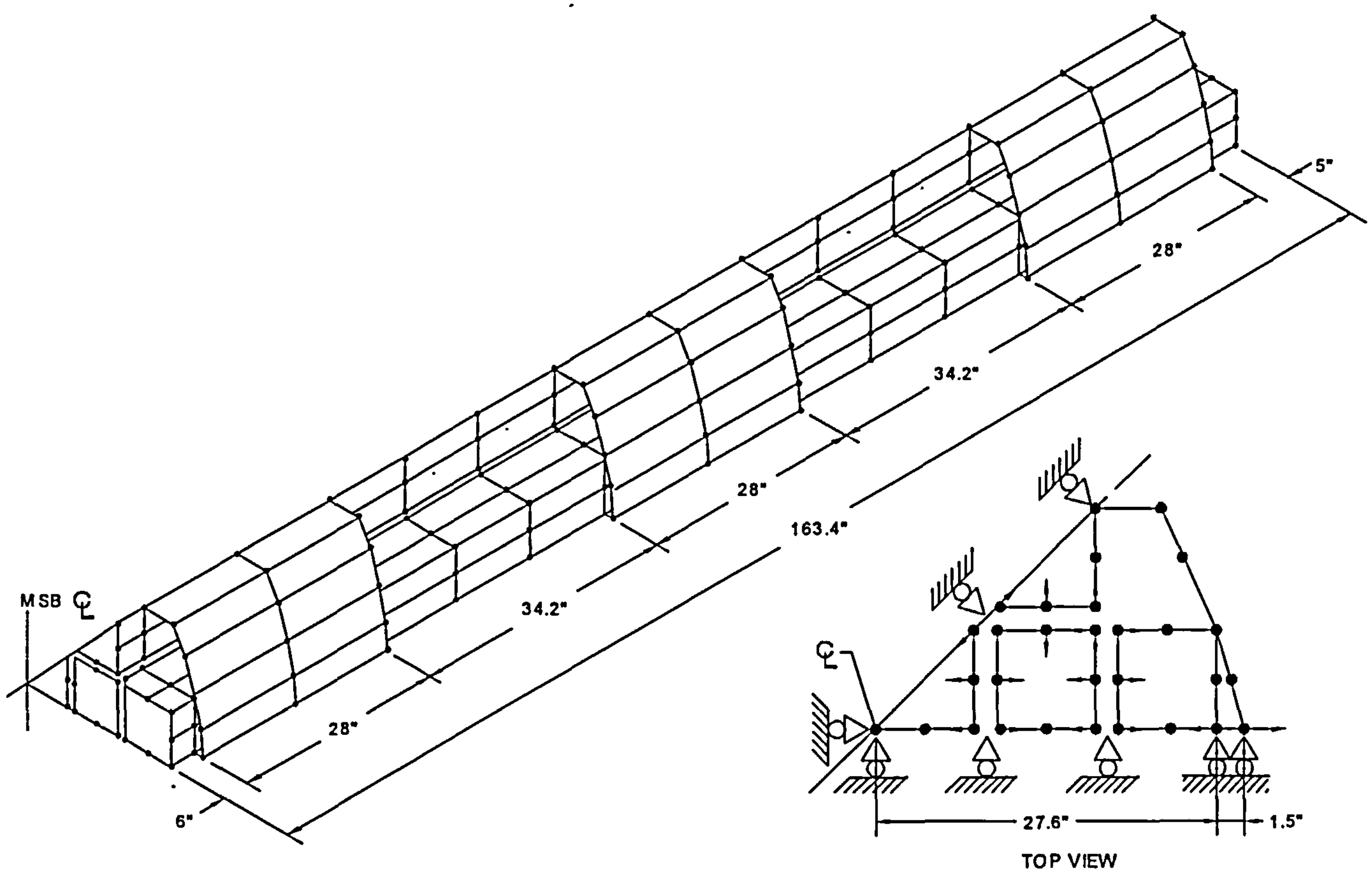


Figure 3.4-3 - MSB Storage Sleeve Assembly Finite Element Model for Thermal Stress Evaluation

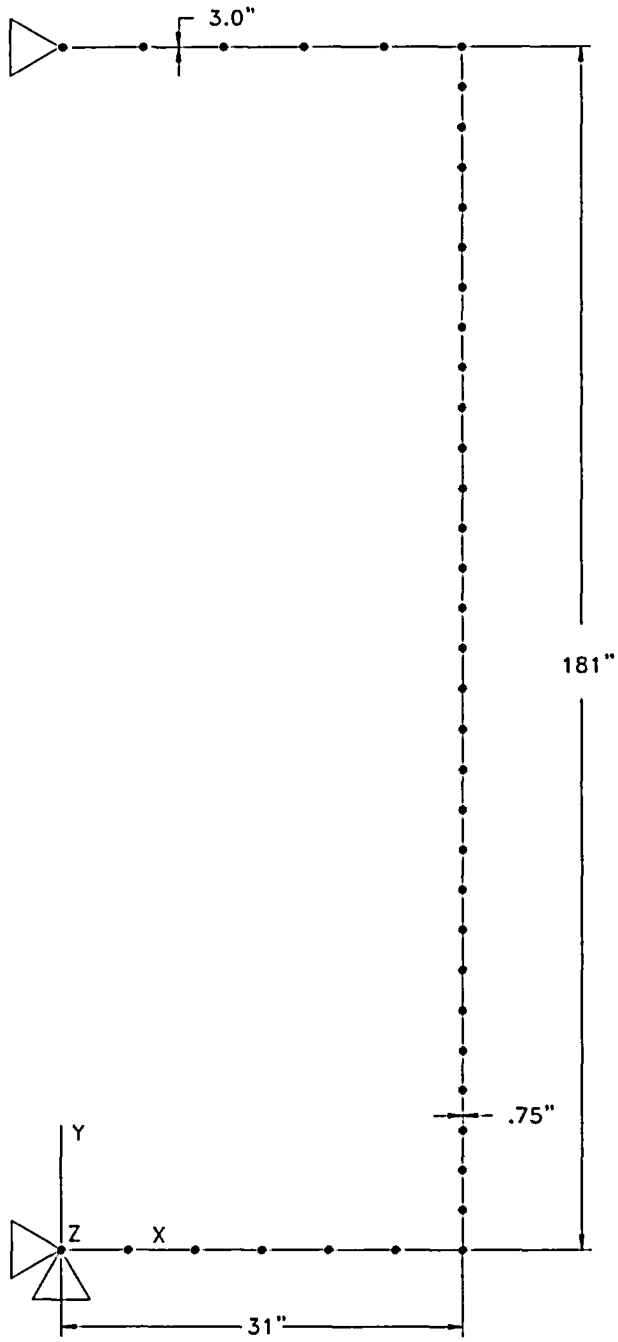


Figure 3.4-4 - MSB Body Finite Element Model for Thermal Stress Evaluation

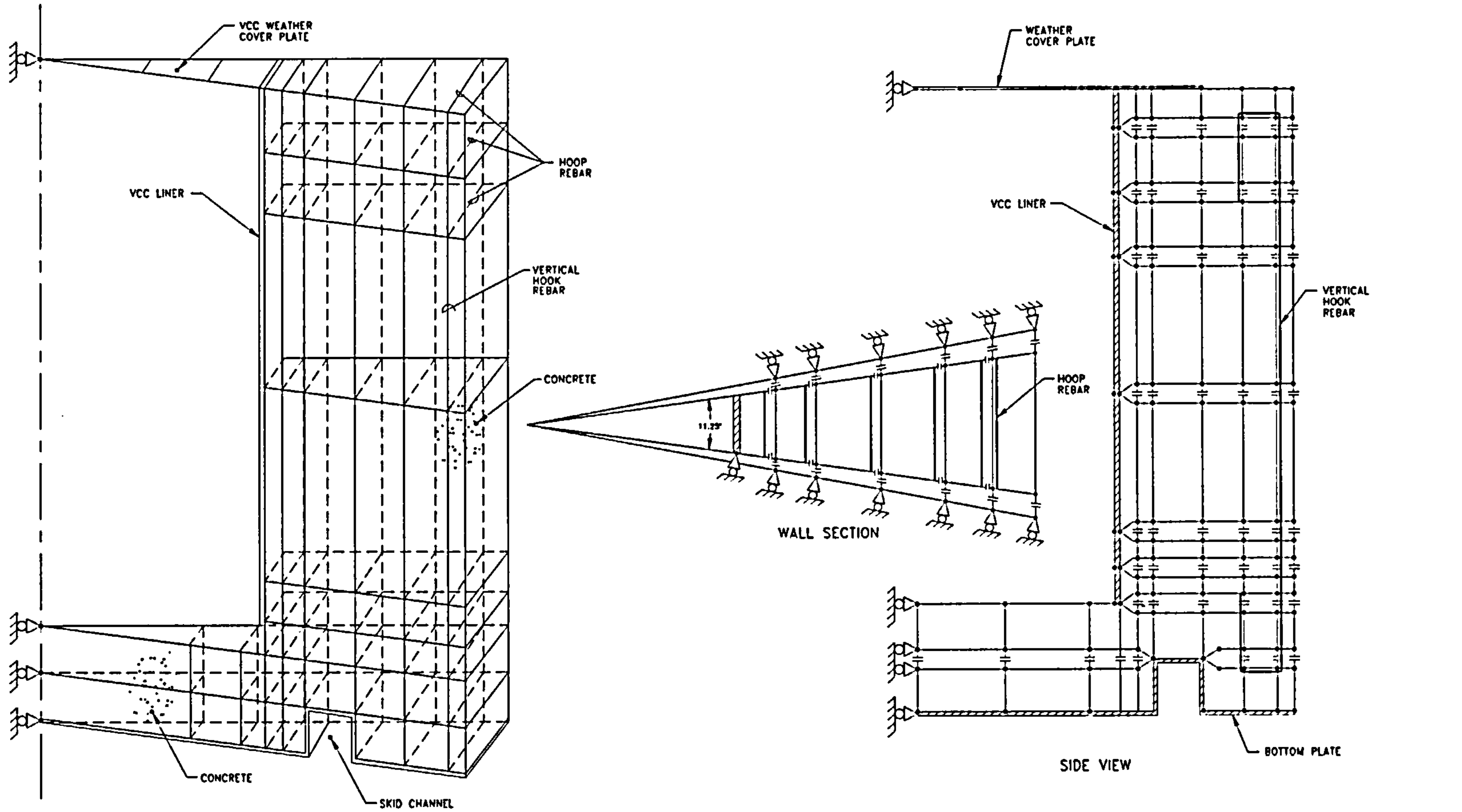


Figure 3.4-5 - VCC Finite Element Model for Thermal Stress Evaluation

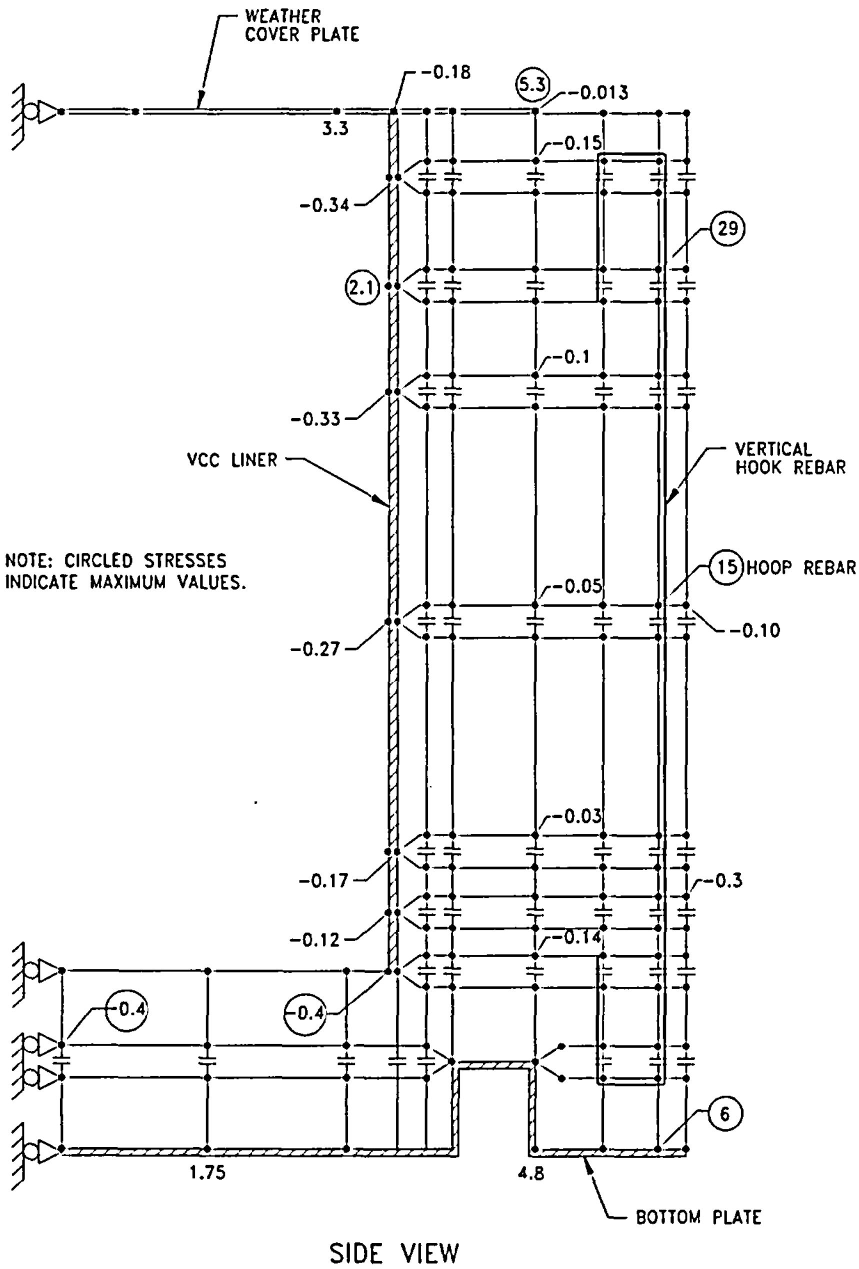


Figure 3.4-6 - Summary of VCC Thermal Stress 75°F Ambient Air, Normal Operation

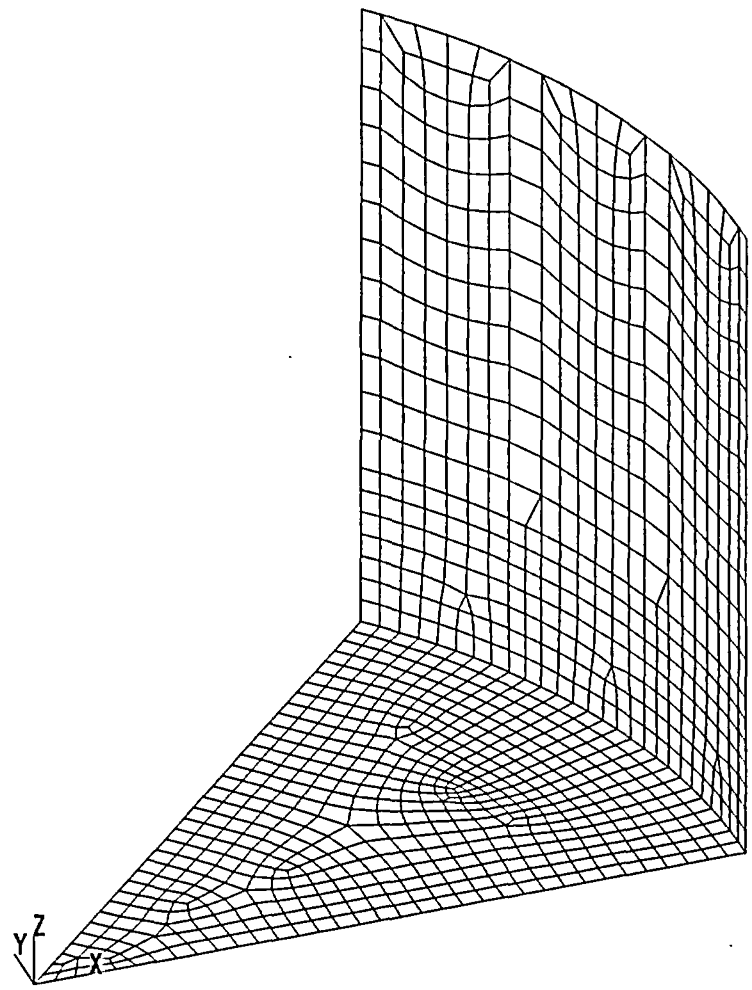


**Figure 3.4-7 - Finite Element Model MSB Dead Weight Analysis
MSB Supported on Ceramic Tiles**

4.0 THERMAL EVALUATION

4.1 DISCUSSION

This section presents the thermal analysis of the VSC system for normal operation. The significant thermal design feature of the VSC system is the air flow path used to remove the 24 kw (maximum) of decay heat. This natural circulation of air inside the VCC allows the concrete temperatures to be maintained below the design limits and keeps the fuel cladding temperatures below levels where damage might occur. The analyses presented in this section show that the fuel cladding will not exceed 712°F for all types of PWR fuel, provided the assemblies have a decay power less than or equal to 1.0 kW each, and the uranium loading does not exceed 0.471 MTU per fuel assembly or 3.27 kg/inch of fuel in the assembly. The 3.27 kg/inch limit ensures that assemblies with fuel length shorter than 144 inches will not produce heat-generation levels exceeding the 6.94 W/inch value used in the analyses presented herein.

The base calculation is performed assuming 75°F ambient conditions to model the average long-term temperatures expected over the life of the cask. No solar load is used because the exclusion of solar loads for this case gives a more severe temperature gradient through the concrete wall, representing a worst-case normal operating thermal stress condition. Also, as shown in Figure 4.1-1, recent cask tests have shown little or no impact of solar load on fuel temperatures (Reference 4.3). Even if solar loads were assumed to affect the cask for 12 to 14 daylight hours, they would only modestly affect the outer concrete temperatures for the period during which the sun is shining. The temperatures of the bulk of the concrete and the MSB will not be affected by these transient effects, as further discussed in Section 11.2.

To bound the expected temperature ranges in which the VSC system might operate, two off-normal severe environmental temperature conditions are evaluated. These calculations are presented in detail in Section 11.1 (Off-Normal Events). The cases considered are -40°F with no solar loads and 100°F with maximum solar loads. The maximum solar load is calculated to be the 24-hour average solar load (from 10CFR71) so as to more accurately model the steady state temperature expected from long-term exposure (four to five days) to 100°F air. Inclusion of any solar loads in this calculation is considered to be conservative based on the findings of recent cask tests at Idaho National Engineering Laboratory INEL, which show essentially no solar effect on cask or fuel temperatures (see Figure 4.1-1) (Reference 4.1).

The 75°F ambient conditions are used to determine long-term storage temperatures, and the -40 and 100°F ambient temperatures are used to model extreme environmental conditions. In addition to these three cases, three thermal analyses of off-normal and accident conditions are presented in Chapter 11. The first off-normal case considers a 100°F ambient condition with maximum solar loads and a maximum decay heat payload. This condition is analyzed to show the temperature for the worst-case heat load, which due to decay of the heat source, could only happen once over the life of the cask. The second off-normal condition considers blockage of the air inlets on one side of the cask (one-half of the inlets). The third case considers the accident condition of complete blockage of all air inlets.

Although the cask has been analyzed for an average long-term temperature of 75°F, higher average long-term temperatures can easily be accommodated. The limiting general/local concrete temperature (150/225°F) is not reached even at an average long-term temperature of 100°F. Since this average long-term temperature is not exceeded anywhere in the United States, the ambient temperature requirement is met at any site. Also, the maximum temperature for the 50 percent probability level (two year recurrence) less than 125°F, which is analyzed in Section 11.2.1, does not restrict the use of the VSC system at any U.S. site.

Table 4.1-1 summarizes the results of the thermal calculations. As can be seen from this table, the conservatively calculated temperatures are below the temperature specifications listed in Chapter 2 and Section 4.3.

4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

The thermal properties used in the thermal hydraulic analyses are shown in Table 4.2-1. The derived parameters (effective conductivities) are discussed in Section 4.4. Temperature invariant properties are used for the concrete, steel, fuel, and helium. Low values derived from the open literature and conservative calculations are used. If temperature-dependent properties (especially for concrete and fuel) were used, the maximum temperatures reported in this section would be slightly reduced.

4.3 TECHNICAL SPECIFICATION OF COMPONENTS

Temperature limits were established for all the materials used in the VSC system. Specifically, these limits are for concrete, fuel cladding, steel, RX-277, and coatings. The limits were established in accordance with the following codes, standards, and manufacturers' recommendations in order of precedence:

Code or Standard	Component
ASME Section III, Division 1	Steel
ACI 349 and NRC Guidance	Concrete
ANS 57.9	Steel and Concrete
ASTM	Steel and Concrete
Manufacturers' Recommendations	Coatings

Based upon evaluations of these limits, it is determined that the fuel cladding and concrete temperature limits are the limiting conditions.

The normal, long-term (weeks to months) limits for concrete are established as 150°F generally and 225°F locally. The long-term and short-term temperature limits for the concrete are summarized in Table 4.3-1, based on ACI 349, Appendix A, and NRC guidance.

The fuel cladding allowable temperature is actually a complex function of temperature versus time, and internal rod pressurization (Reference 4.1). As such, it should be calculated in accordance with the methodology presented in Appendix C (Reference 4.1) for each specific site. This appendix shows the allowables range from 712°F to 752°F, depending on fuel type. The 712°F allowable temperature can be used for all sites on a generic basis without further evaluation. A short-term allowable temperature (days) of 1058°F is established for the vacuum drying, transfer and other short-term off-normal and accident conditions. This temperature is established based on experimental results for high temperature induced failure of zircaloy rods (Reference 4.2) and on previous licensing actions. An accident allowable temperature of 1200°F is established based on the work of Reference 4.2, at or below which no temperature induced failures would occur.

4.4 THERMAL EVALUATION FOR NORMAL STORAGE CONDITIONS

4.4.1 THERMAL MODELS

Four basic models are used for the thermal evaluation of the VSC system:

1. Air Flow and Temperature
2. VCC Body and MSB Exterior Heat Transfer
3. MSB Interior Heat Transfer
4. MSB in Transfer Cask Heat Transfer

In addition to these basic models, two sub-models or calculations are used for the effective fuel region thermal conductivity and the surface natural convection heat transfer coefficient. These models/calculations are based on cask testing performed at INEL and elsewhere (References 4.3, 4.4, 4.5, 4.6, and 4.7). All of these thermal hydraulic models are described in the following sections.

4.4.1.1 Air Flow and Temperature Calculation

The air flow up the annulus formed by the MSB and VCC is calculated by determining the sum of the flow pressure losses (Σ k/A^2) due to all entrances, bends, straight sections, expansions, contractions, and exits. The resulting friction and form flow pressure losses are equated to the pressure differential caused by the heating of the air (i.e., the stack or furnace effect).

The basic procedure for this calculation is to apply the macroscopic energy equation from the midpoint of the air inlet to the midpoint of the air outlet. This equation is:

$$-\Delta p+\frac{g\rho_0 h}{g_c}+\frac{\rho g h\beta(\Delta T_{ave})}{g_c}+\frac{\dot{m}^2}{2g_c\rho}\cdot\sum_i\frac{k_i}{A_i^2}=0 \qquad (4.1)$$

where,

Δp	=	Pressure differential, inlet to outlet vents
$g\rho_0 h/g_c$	=	Elevation pressure head where ρ_0 is the ambient air density, h is the height, g the acceleration of gravity, and g_c the gravitational conversion factor
ΔT_{ave}	=	$(T_{outlet} - T_{inlet})/2$
$\dfrac{\rho g \beta h(\Delta T_{ave})}{g_c}$	=	Pressure change due to air heating
$\dfrac{\dot{m}^2}{2g_c\rho}\sum_i \dfrac{k_i}{A_i^2}$	=	Pressure loss due to friction and form loss of all flow segments i, where m is the mass flow rate, A_i the flow area of the i^{th} segment and k_i the loss coefficient of the i^{th} segment

Since the pressure difference between the inlet and outlet is equal to the elevation pressure head of the ambient air column, the first two terms in the above equation cancel out. For the region of interest, β (the compressibility factor) can be approximated by 1/T (°R). Thus, the above equation reduces to:

$$-\Delta p + \frac{g\rho_0 h}{g_c} - \frac{\bar{\rho} g h(\Delta T_{ave})}{g_c \bar{T}} + \frac{\dot{m}^2}{2g_c\bar{\rho}} \cdot \sum_i \frac{k_i}{A_i^2} = 0 \qquad (4.2)$$

where the bar over the density and temperature terms denotes average values.

The heat balance is expressed as

$\Delta T \quad = \quad Q_T/(mc_p)$

where,

Q_T	=	Total heat transfer to air. This value is assumed to be 18 kW (out of the total heat load of 24 kW). The other 6 kW are transferred through the cask concrete walls and top as discussed in Section 4.4.3.
C_p	=	Specific heat of air

Using Equation 4.2 and the heat balance equation, an iterative solution is derived for calculating the exit temperature and air mass flow rate. A spreadsheet program is used to perform this calculation. The axial heat source distribution shown in Figure 4.4-1 is also used to calculate air temperature as a function of elevation as it flows through the VSC.

Table 4.4-1 summarizes the results for the various ambient conditions. These results are used in ANSYS finite element models for calculation of the VCC and MSB temperature distributions.

4.4.1.2 VCC Body and MSB Exterior Thermal Model

4.4.1.2.1 Heat Transfer Modes

Heat is generated in the fuel that is located in the MSB. This heat is conducted through the MSB shell, convected to the air, and radiated to the VCC internal liner. Heat on the VCC liner is also convected to the air, and a small amount is conducted through the concrete. On a sunny day, additional heat enters the VCC through the exterior surface as solar insolation. All this solar heat is radiated and convected from the VCC surface.

Radiation from all surfaces is addressed by:

$$q = \sigma \varepsilon F A (T_1^4 - T_2^4) \qquad (4.3)$$

Where:

q	=	Heat flow rate, BTU/hr
σ	=	Stefan-Boltzman constant, 1.714×10^{-9} BTU/hr-ft^2-°R^4
ε	=	emissivity
F	=	Radiative geometry view (form) factor
A	=	Radiating surface area, ft^2
T	=	Absolute source (1) and target (2) temperatures, °R

Surface emissivities vary with the radiating material and are provided in Table 4.2-1. View factors vary with surface and target geometry. Solar insolation (solar thermal radiation incident on the VCC surface) is discussed in Section 4.4.1.2.3.

Convections from all surfaces are addressed by:

$$q = hA(T_1 - T_2) \qquad (4.4)$$

where,

h	=	Natural convection heat transfer coefficient, BTU/hr-ft^2-°F

Heat conduction is expressed by the following differential equation:

$$p c_p \,(dT/dt) = d\,(k\, dT/dx)/dx + q''' \qquad (4.5)$$

where,

k	=	Thermal conductivity, BTU/hr-ft-°F
p	=	Density, lbm/ft^3

c_p = Specific heat, BTU/lbm-°F

q''' = Heat generation rate, BTU/hr-ft^3

All of these heat transfer modes are addressed by the ANSYS/PC-Thermal (A/PCT) computer program. This finite element program is used to perform the bulk of the thermal-hydraulic analysis of the VSC system. The following sections describe the model and data used as input to the A/PCT program.

4.4.1.2.2 VCC Thermal Hydraulic Model

The geometry of the VSC system components (illustrated in the general arrangement drawings in Section 1.5) is converted into finite element form as shown in Figure 4.4-2. Input for the model is discussed in the following sections.

All units used in this FSAR and the ANSYS/PCT program are consistent: BTU, ft, hr, F, R, lbm. Two element types are used, the 3-D solid element (STIF 70) and a radiation link element (STIF 31). Thermal properties are specific to the materials (see Table 4.2-1 for thermal properties).

4.4.1.2.3 Radiation

The solar radiation heat input, used for the 100°F case, is per the federal requirements of 10CFR71. The solar loads used are 2949 BTU/ft^2 for the top surface and 1474 BTU/ft^2 for the curved side surfaces. These thermal loads are converted to average rates by assuming the sun shines for 12 of the 24 hours per day. The resulting heat fluxes on the top and side surfaces are 123 and 61 BTU/hr-ft^2, respectively.

In the finite element model, these heat rates are applied as heat generation rates in a thin (0.25-inch thick) outer layer of the concrete cask. The generation rate is calculated as the above heat rate divided by the volume of the thin shell of concrete.

Radiation is included at all surfaces radiating to the atmosphere and between the annular air spaces in the VCC. View factors of unity were assumed for all inner surfaces. The view factor for the cask exterior is calculated as 0.14 between the side of the cask and its surroundings (i.e., cask array on 15-ft centers). All casks and the ground are considered at equilibrium. The side view factor is calculated based on an average distance (11.87 ft) to the adjacent cask (averaged on five different side locations). The top of the VSC was assumed to have a view factor of unity with respect to the sky.

4.4.1.2.4 VCC Convections

Natural convection heat transfer coefficients were taken as 2.0 BTU/hr-ft^2-°F on all surfaces, as discussed in Section 4.4.1.5. This value is a conservative value compared to full-scale experimental data from other casks (References 4.3 through 4.7).

4.4.1.2.5 VCC Modeling Assumptions

The model uses a 10° slice to model the entire VCC. The VCC geometry and temperatures are uniform with angular direction so that the two-dimensional portrait is adequate. The 10° slice is small, so as to minimize the complexity of the nodalization while still accurately representing the radial volume distribution.

The air vents are not included in the model. Because of the low thermal conductivity of concrete, the air vents affect only the region local to the vents. The model includes low incoming and high exiting air temperatures below and above the heated region, respectively, to assure that temperature extremes are represented. The air in the heated region is modeled as a heat sink at the temperatures shown in Table 4.4-1.

Solar insolation is treated as a volumetric heat generation. Actual solar insolation appears as a uniformly distributed heat flux on the VCC surfaces. The A/PCT program allows heat fluxes only at nodes which causes hot spot nodes on the VCC surface. The use of a thin shell of heated region assures uniform application of the solar flux in the elements of the thermal model.

The cask side radiation view factor is conservative since it is calculated based on an average surface orientation. Actually, the VCC hot spots are near the top of the VCC, where the clear view to the sky is much larger. If the distribution of view factors were accounted for, a more uniform temperature distribution with lower peak temperatures would result.

The MSB portion of the VCC model treats only the MSB shell in detail. The interior is modeled as a heat-generating region with an effective thermal conductivity. This effective thermal conductivity (2.4 BTU/hr-ft-°F) is estimated from the cask test data (References 4.3 through 4.7) to accommodate the solution methods employed by A/PCT. In this model, the MSB is only considered for evaluating the surface heat flux and the MSB shell temperatures. The details of the MSB interior are evaluated in a separate model.

4.4.1.3 MSB Thermal-Hydraulics

4.4.1.3.1 MSB Heat Transfer Modes

Heat is generated in the fuel assemblies and transferred to the surrounding inert atmosphere and the basket sleeves by free convection and radiation. In turn, the heat is conducted through the storage sleeves towards the exterior of the sleeve assembly. It then conducts, convects, and radiates through the cover gas to the MSB shell wall. All convection inside the MSB is natural (free) convection.

Radiation from all surfaces is addressed by Equation 4.3 (Section 4.4.1.2.). Surface emissivities vary with the radiating material and are provided in Table 4.2-1. View factors vary with surface and target geometry.

Convections from all surfaces can be specifically addressed by Equation 4.4 (Section 4.4.1.2).

Heat conductions are modeled as two-dimensional isoperimetrical plane solids per

$$\frac{d}{dx}(k\, dT/dx) + q''' = 0 \qquad (4.6)$$

where,

k = Thermal conductivity, BTU/hr-ft^2 -°F

q''' = Heat generation rate, BTU/hr-ft^3

All of these heat transfer modes are addressed by the A/PCT computer code. The following sections describe the data used in the MSB thermal finite element model.

4.4.1.3.2 MSB Thermal Hydraulic Model

The geometry of the MSB interior was converted to the finite element model shown in Figure 4.4-3. The model represents a horizontal slice of unit thickness through the hottest section. The symmetrical nature of the VSC system allows use of a 45° sector model of the MSB. This symmetry is utilized by imposing zero heat flux boundary conditions along the 0° and 45° model boundaries. Two element types are used, the two-dimensional solid element (STIF 55) and a radiation link element (STIF 31). Thermal properties are specific to the materials and are presented in Table 4.2-1.

4.4.1.3.3 MSB Fuel Heat Source Strength

Fuel heat generation rates are calculated by assuming 1 kw of heat generation per assembly. The hottest horizontal slice of the fuel region is converted to a volumetric heat generation as follows:

$$q''' = (1.2) \times (Q/\underline{v}) = 620.26 \text{ BTU/hr-ft}^3 \qquad (4.7)$$

where,

1.2 = Maximum axial peaking factor

Q = Heat/assembly = (1kw)(3412 BTU/kw-hr) = 3412 BTU/hr

$\underline{v}$ = Storage Sleeve assembly volume = (8.9 in/12)^2(12 ft) = 6.6 ft^3

4.4.1.3.4 MSB Heat Transfer Properties

Radiation: Radiation is included at all surfaces radiating to the MSB shell. For simplicity, all shape factors are assumed to be 1.0, with direct radiation to the MSB shell (see assumptions below). Internal fuel assembly radiation is also included in the effective fuel conductivity calculation, as described in Section 4.4.1.6, with supporting analyses included in Appendix A.

Fuel Equivalent Conductivity: Heat transfer in the fuel region is a complex combination of conduction, convection, and radiation. This heat transfer is modeled as conduction only by assuming the fuel region to be a solid with an equivalent fuel conductivity k_f. The value of k_f is estimated in Section 4.4.1.6, with additional information provided in Appendix A.

Helium Equivalent Conductivity: Heat transfer in the helium region is a combination of conduction and free convection throughout the MSB. For the wide and narrow flow regions shown in the MSB model, the effective conduction coefficients are determined based on experimental results (References 4.3 and 4.6). As derived in Appendix D, the effective conduction coefficients are:

	Wide	Narrow
k(BTU/hr-F-ft)	2.8	0.11

4.4.1.3.5 MSB Model Assumptions

The fuel assembly heat transfer can be modeled with an equivalent conductivity. The complex heat transfer between the fuel and the MSB is modeled using a correlation specific to that purpose, as described in Section 4.4.1.6. The resulting model is validated against test data.

Re-radiation between the sleeve surfaces and the MSB shell is ignored in the MSB model. Although the L-shaped region would normally include re-radiation between the exposed sleeve surfaces, and would serve to locally heat the surfaces in the region of the joint, all heat will eventually end up on the MSB surface. All re-radiation was originally modeled using many radiation elements and shape factors. The effect of the complex re-radiation versus simple radiation was less than 20° locally and had no effect on the peak fuel temperature. However, it did severely impact calculational times. As a result, the final model includes only simple direct radiation.

4.4.1.4 Transfer Cask Model

4.4.1.4.1 MTC Heat Transfer Modes

Heat is generated in the fuel assemblies and transferred to the MSB Transfer Cask (MTC) from the MSB surface by radiation, conduction, and convection through the air annulus between the MSB and MTC. The heat is then conducted through the MTC wall and convected and radiated from the MTC outer surface. Heat transfer modes inside the MSB are the same as discussed in Section 4.4.1.3.

Radiation from the inner and outer surfaces are addressed by Equation 4.3. Surface emissivities vary with the radiating material and are provided in Table 4.2-1. View factors vary with surface and target geometry. All surface shape factors are assumed as unity.

Convections from outer surfaces are addressed by Equation 4.4.

Since the gap between the MSB and MTC will be partially occluded, no convection is assumed in this gap.

The MTC through wall heat conduction is modeled in cylindrical coordinates as:

$$(1/r)(d/dr)(kdT/dr) + q''' = 0 \qquad (4.14)$$

where,

k	=	Thermal Conductivity, BTU/hr-ft-°F
q'''	=	Heat generation rate, BTU/hr-ft^3
r	=	Radius, ft

4.4.1.4.2 MTC Thermal Hydraulic Model

The steady solution of Equation 4.14 can be written in a manner consistent with the thermal network analogy as:

$$q = (T_i - T_o)/R \qquad (4.15)$$

where,

R	=	Equivalent thermal resistance, hr-°F/BTU
	=	$\ln(r_o/r_i)/2\pi kL$ for concentric conducting elements
L	=	length, ft
q	=	Heat flow, BTU/hr
r_n	=	radius at an increment corresponding to T_n, ft

For combined heat transfer modes, the thermal network analogy allows combining elements in parallel or in series. Therefore, the external boundary radiation and convection heat transfer is combined in parallel, and the MTC wall conduction heat transfer is in series:

$$1/R = \Sigma(1/R_i), \text{ Parallel modes} \qquad (4.16a)$$

$$R = \Sigma R_i \text{ Series modes} \qquad (4.16b)$$

The MTC wall temperatures can be calculated from a finite difference formulation of equation (4.15), i.e.:

$$T_n = T_{n-1} - qR \qquad (4.17)$$

T_n	=	Temperature at radius r_n, $T_n = T(r_n)$
$T_{n\pm1}$	=	Temperature at radius $r\pm\Delta r$, $T_{n\pm1} = T(R\pm\Delta r)$

Thermal resistances are given by:

$$R_k = \ln(r_o/r_i)2\pi kL, \text{ Conduction} \qquad (4.18a)$$

$$R_c = 1/h_cA, \text{ Convection} \qquad (4.18b)$$

R_r = $1/(A\sigma\varepsilon F(T_i^2 + T_o^2)(T_i+T_o))$,
Radiation (using absolute temperatures) (4.18c)

For the region containing both RX-277 and steel, an effective k was determined from
$k_eA = k_{rx}A_{rx} + k_{sh}A_{sh}$

where,

k_e = Effective k for region

A = Total Area

k_{rx} = k for RX-277

A_{rx} = Steel area

k_{sh} = k for steel

A_{sh} = Steel area

Because the radiation resistance is temperature dependent, the solution is an iterative process. The first step is to estimate a temperature distribution, then estimate the individual geometry and temperature-dependent resistances. These resistances are used with a constant 75°F ambient temperature to re-estimate the temperature distribution. Iterations are continued until calculated temperatures from succeeding iteration steps vary by less than 1/10000. The iterative solution is performed on a personal computer using a spreadsheet program. The results are shown in Figure 4.4-4 and Table 4.1-1.

4.4.1.4.3 MSB Vacuum Cover Gas Thermal Hydraulic Model

The MSB is modeled using the A/PCT finite element code, as discussed in Section 4.4.1.3. For both MTC cases, the MSB hot-slice model is used to estimate the MSB components and fuel temperatures.

For the MSB drying case, the MSB model is modified to represent vacuum conditions. The inner helium elements are removed. The resulting model only includes radiation from the guide sleeves to the MSB wall. Based on the benchmarks performed for vacuum cases of other cask tests and the higher temperatures, the fuel effective thermal conductivities are left unchanged.

4.4.1.5 Determination of VCC Surface Heat Transfer Coefficient

Previous cask test programs (References 4.3 through 4.7) have shown that generally available (from the open literature) heat transfer coefficients for natural circulation from the cask exterior severely over-predict cask temperatures. That is, the "textbook" heat transfer coefficients are too low. For the VSC analysis, the existing body of cask test data is examined, and surface heat transfer coefficients are calculated based on the experimental data.

The ambient temperature and the cask surface temperature at the mid-plane are extracted from each cask test report. This temperature difference is taken for the vacuum cases (i.e., cask internals at a vacuum) to further preclude the axial conductance or convection of heat from the mid-plane. As such, all the heat produced at the mid-plane is passing radially through the cask. The 1.2 peaking factor (discussed previously) is also used. The surface heat transfer coefficient is determined by using the total heat in the cask, the 1.2 peaking factor, the cask surface, ambient temperatures, and the energy balance equation:

$$q'' = h(T_c - T_a) + \sigma\varepsilon\,(T_c^4 - T_a^4) \qquad (4.19)$$

where,

q'' = Surface heat flux from a one foot long slice through the midplane (BTU/hr ft^2)

h = Heat transfer coefficient (BTU/hr ft^2 °F)

T_c = Cask surface temperature (°R)

T_a = Ambient temperature (°R)

σ = Stephen-Boltzman constant (BTU/hr ft^2 °R^4)

ε = Emissivity of cask

The values of the data and other properties can be found in the cask test reports. Using this information the following values of h are determined:

Cask Test Reference	Q BTU/hr	Cask Dia. (ft)	Cask Surface Temp. °F	Ambient Temp. °F	Emissivity	Calculated h BTU/hr ft^2 °F
4.4	51862	7.3	126	75	0.78	3.5
4.3	70287	7.4	156	68	0.9	2.2
4.5	95536	8.0	190	77	0.92	2.0

The smooth painted exterior of the cask in Reference 4.3 most closely matches the VSC. However, for conservatism, a value of 2.0 is used for the ANSYS heat transfer analysis of the VSC cask. A further sensitivity study reduces the outer VCC surface convection coefficient to 1.0, with only minor temperature rise for the concrete near the surface.

4.4.1.6 Determination of Fuel Effective Thermal Conductivities

The fuel region (inside each of the 24 storage sleeves) is modeled as a homogeneous region with an effective thermal conductivity. This effective thermal conductivity is determined by the following methods:

1. Application of the Wooten-Epstein Correlation (WEC)
2. Back-calculation of k_{eff} from cask test data

3. Modeling cask test with ANSYS model and determining k_{eff} to match test data

The details of these applications are discussed in Appendix A. For the temperature regimes of interest (300 to 800°F), the results yield effective thermal conductivities in the range of 0.4 to 1.2 BTU/hr ft °F. For the calculations presented in this FSAR, a value of 0.6 BTU/hr ft °F is used. This results in temperature drops in the range of 65°F (interior assemblies) to 128°F (exterior assemblies), which is consistent with the data in References 4.3 through 4.7. A sensitivity analysis shows that using 0.4 BTU/hr ft °F results in a temperature increase of only 30°F for the maximum cladding temperature. However, it should be pointed out that values of 0.4 BTU/hr-ft-°F are typical for cooler fuel or edge assemblies, and values greater than 0.6 BTU/ht-ft-°F are typical for hot fuel in the middle.

4.4.2 TEST MODEL

The VSC system is conservatively designed by analysis so that testing is not required.

4.4.3 MAXIMUM TEMPERATURES

Figure 4.4-5 shows the temperature distribution of the VCC components for normal, long-term storage conditions. The exterior surface temperatures are used to determine how much heat is radiated and convected from the cask surface. The resulting 6.3 kW is consistent with the assumption that only 18 kW is dissipated by the air flow (see Section 4.4.1.1). Figure 4.4-6 shows the temperature distribution of the MSB components. Temperature distributions for the off-normal, severe environmental conditions and the accident conditions are discussed in Sections 11.1 and 11.2.

4.4.4 MINIMUM TEMPERATURES

Section 11.1 provides the temperature distribution for the severe cold environmental conditions of -40°F. However, even at these extreme conditions the components are above their minimum material limits.

4.4.5 MAXIMUM INTERNAL PRESSURE

The MSB is backfilled with helium so that, at the conditions present during normal operations, the MSB pressure difference is essentially zero. For extreme hot and cold off-normal and accident conditions, the change in the MSB internal pressure load is determined based upon the MSB temperatures. The minimum MSB internal pressure load, resulting from the –40°F off-normal cold thermal condition, is -1.5 psig (i.e., external pressure). The maximum MSB internal pressure load, resulting from the 125°F accident thermal condition, is +0.7 psig (i.e., internal pressure). For the accident pressurization condition, the worst-case internal pressure occurs if all the fuel rods inside the MSB are breached and release their fission gases. This case and its resulting pressure and stresses are discussed in Chapter 11.

The stresses in the MSB due to an internal pressure load of 1.5 psig, which bounds the maximum internal pressure load for all normal, off-normal, and accident thermal conditions (with the exception of the accident pressurization evaluated in Chapter 11) are evaluated in Chapter 3. The stresses in the MSB due to accident pressurization condition are evaluated in Chapter 11.

4.4.6 MAXIMUM THERMAL STRESSES

The MSB and VCC thermal stresses are presented in Chapter 3.

4.4.7 EVALUATION OF CASK PERFORMANCE FOR NORMAL CONDITIONS OF STORAGE

As shown in the preceding sections, the VCC system operates within the thermal design criteria. Therefore, no degradation due to temperature effects on materials or components is expected over the lifetime of the cask. In fact, as a result of the decay of the heat source over time, all the temperatures reported herein will decrease over the life of the cask. Figure 4.4-7 shows the relationship of the VCC and MSB temperatures with time. This graph was derived assuming a linear relationship between temperature and heat generation. Thermal radiation effects would cause a slightly higher order relationship, resulting in a slower decrease in the temperatures with decreasing heat source. However, the analysis summarized in Figure 4.4-7 is considered conservative since the initial temperatures are not above any limits and can only decrease over time due to the decay of the heat source.

Table 4.1-1 - Summary of VSC System Thermal Hydraulics Evaluation

Summary Of Long-Term VSC System Thermal Hydraulic Evaluation

CASE	Ambient Conditions °F	Solar	TEMPERATURES Air Outlet	Outer Concrete	Inner Concrete	MSB Shell	Max Clad
Generic Limits	N/A	N/A	N/A	150	225	N/A	712
Steady State Normal Long-Term Storage	75	no	164	85	180	269	684
Steady State Severe Cold	-40	no	30	-32	45	162	595
Steady State Severe Hot	100	yes	193	136	214	294	705

Summary of Short-Term VSC System Thermal Hydraulic Evaluation

Generic Limits	N/A	N/A	N/A	200	350	N/A	1058
12 hour Maximum Thermal Load Transient	125	yes	222	189	248	320	726
1/2 of Inlets Blocked	75	no	179	86	191	276	690
All Inlets Blocked	75	no	230	87	206	297	707
MSB in MTC							
with He	75	N/A	N/A	N/A	N/A	404	765
with vacuum	75	N/A	N/A	N/A	N/A	404	796

Table 4.2-1 - Thermal Properties

Material	Temperature (°F)	Properties			
		Specific Heat (BTU/lbm °F)	Thermal Conductivity (BTU/hr-ft °F)	Density (lbm/ft^3)	Emissivity
Steel	32-600	0.11	26.0	490	0.8*
Concrete	32-400	.021	0.719	140	0.9
Air	-50	0.238	0.0114	0.094	--
	0	0.239	0.0130	0.086	--
	32	0.240	0.0140	0.081	--
	100	0.240	0.0154	0.071	--
	200	0.241	0.0174	0.060	--
	300	0.243	0.0193	0.052	--
	500	0.247	0.0231	0.041	--
	700	0.253	0.0263	0.037	--
Fuel Region (equivalent conductivity)	32-800	--	0.60	--	--
He (equivalent conductivity for wide access)	32-800	--	2.8	--	--
RX-277	32-800	0.22	0.30	105	--
Effective RX-277 conductivity (in MTC)	32-800	--	1.0	--	--

* Coated surfaces are expected to have emissivities in excess of 0.9, but the value of 0.8 for steel was used for conservatism.

Table 4.3-1 - Condition Categories and Temperature Limits for Concrete*

Load Category	Ambient Condition Temperature °F	Solar	Area	Concrete Temperature Limits, °F
Construction Bulk Concrete	-----	-----	-----	130
Normal	75	None	Bulk Concrete (with nuclear heating)	150
			Local hot spots	225
Abnormal and Severe Environmental (occurrence expected with no damage to system. Potentially concurrent with other accidents)	a) 100 b) -40	a) Average b) None	Bulk concrete Local hot spots	150 225
Extreme Environmental (maximum expected thermal load, or other accident conditions)	125	Max for 14 hrs.	Bulk concrete Local hot spots	200 350

* From ACI-349, Appendix A and the NRC guidance.

Table 4.4-1 - Summary of VSC Cooling Air Flow Analysis

	75°F Ambient	100°F Ambient	-40°F Ambient
Air Inlet Temperature	75°F	100°F	-40°F
Air Temperature at Elevation*			
16	82	107	-35
32	93	118	-26
48	104	131	-17
64	116	143	-8
80	128	155	2
96	139	167	10
112	149	177	18
128	158	187	25
144	164	193	30
Air Outlet Temperature	164	193	30
Air Flow Rate (lbm/sec)	0.80	0.76	1.01

* Measured in inches above the beginning of the fuel heated length.

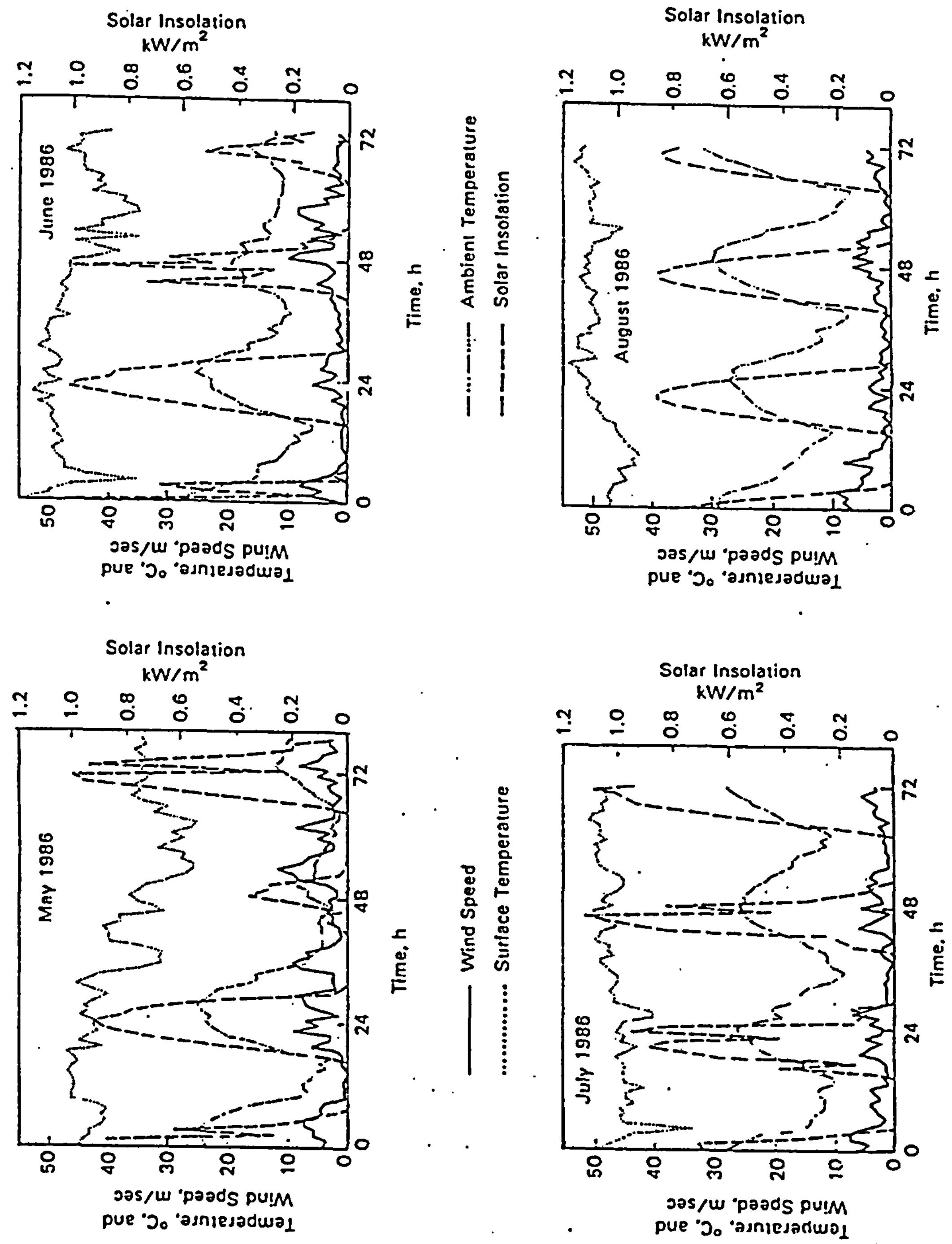


Figure 4.1-1 - Effects of Ambient Conditions on Cask Temperatures

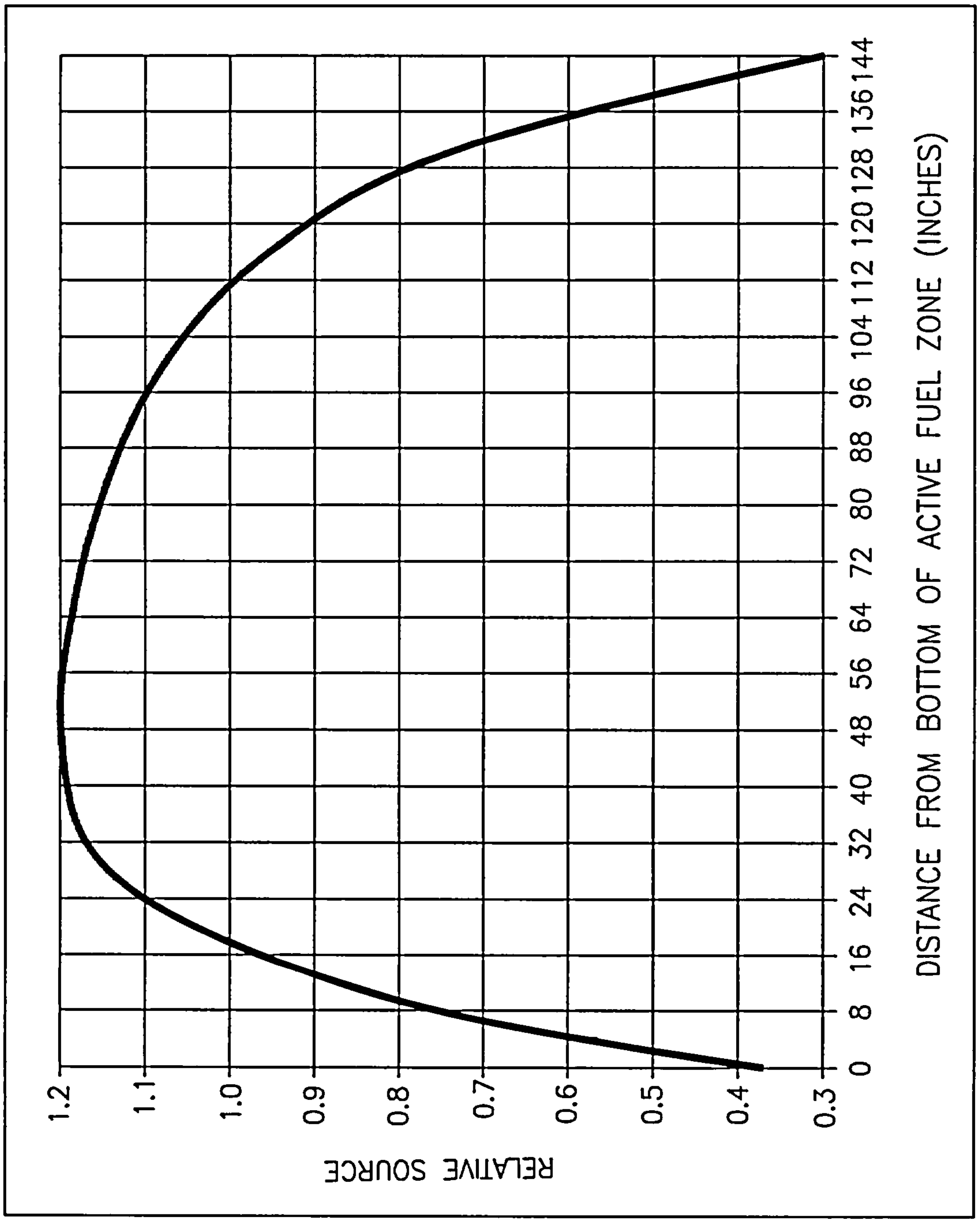


Figure 4.4-1 - Axial Heat Source Distribution

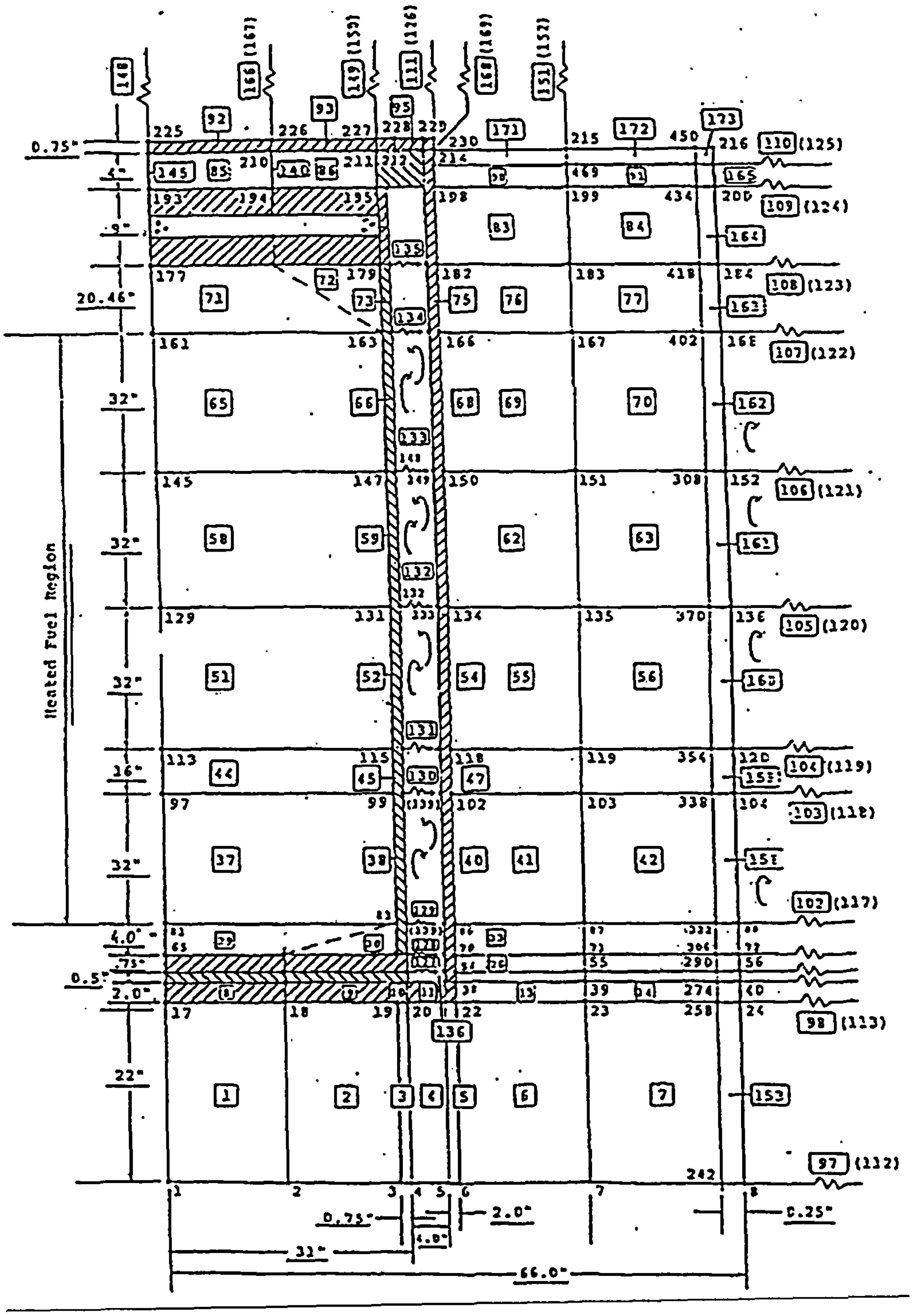


Figure 4.4-2 - VCC Thermal Model

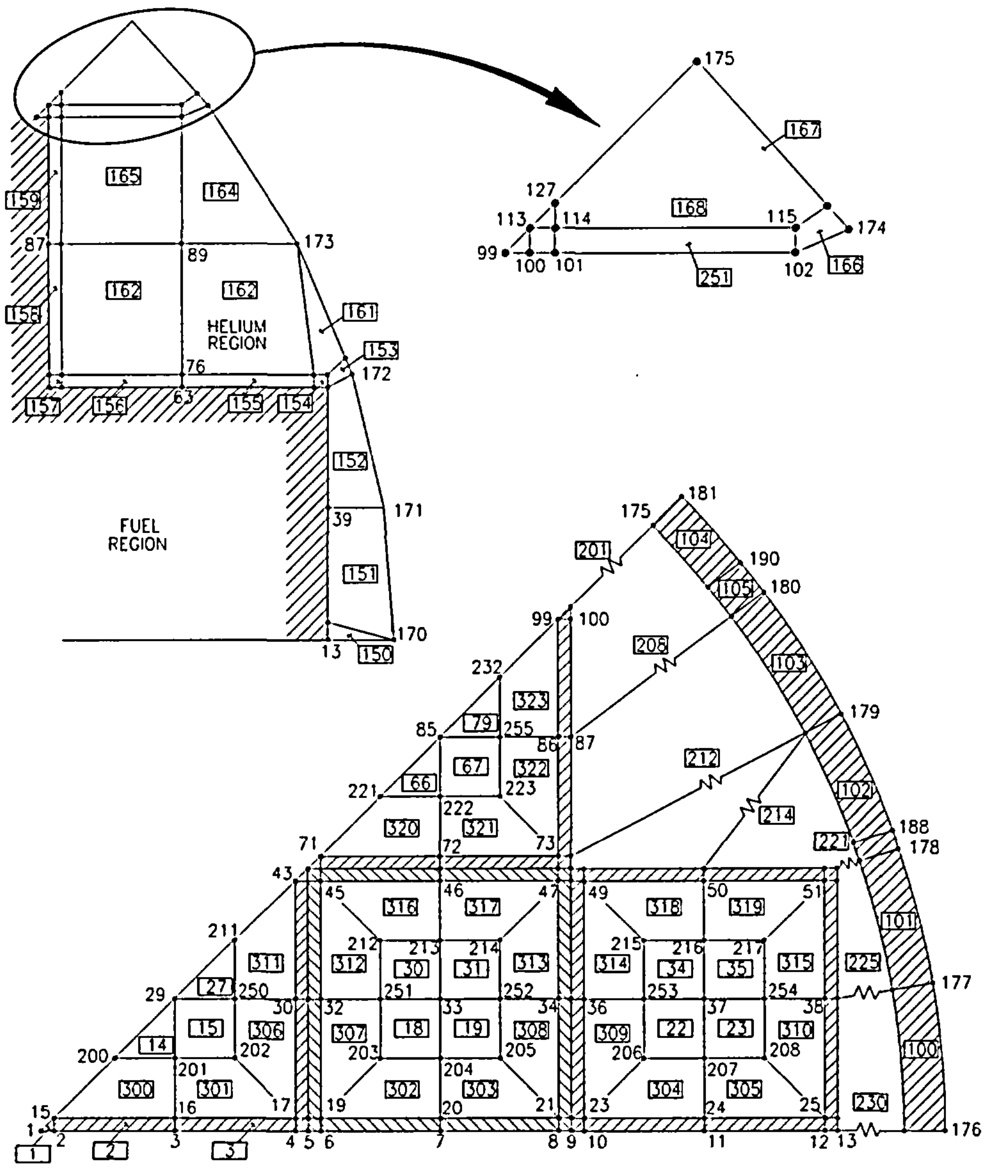


Figure 4.4-3 - MSB Thermal Analysis Model

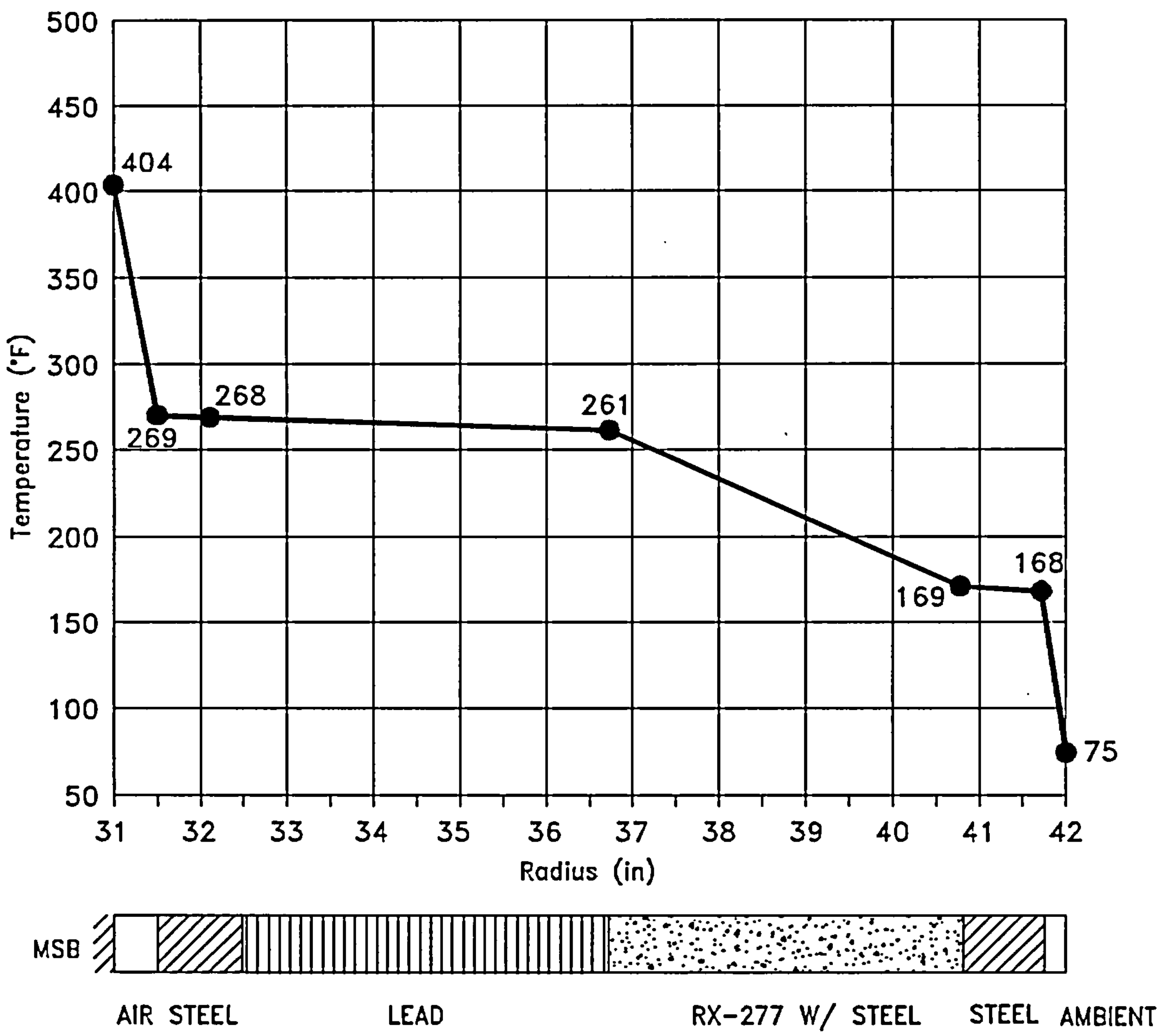


Figure 4.4-4 - MTC Through Wall Temperature Distribution

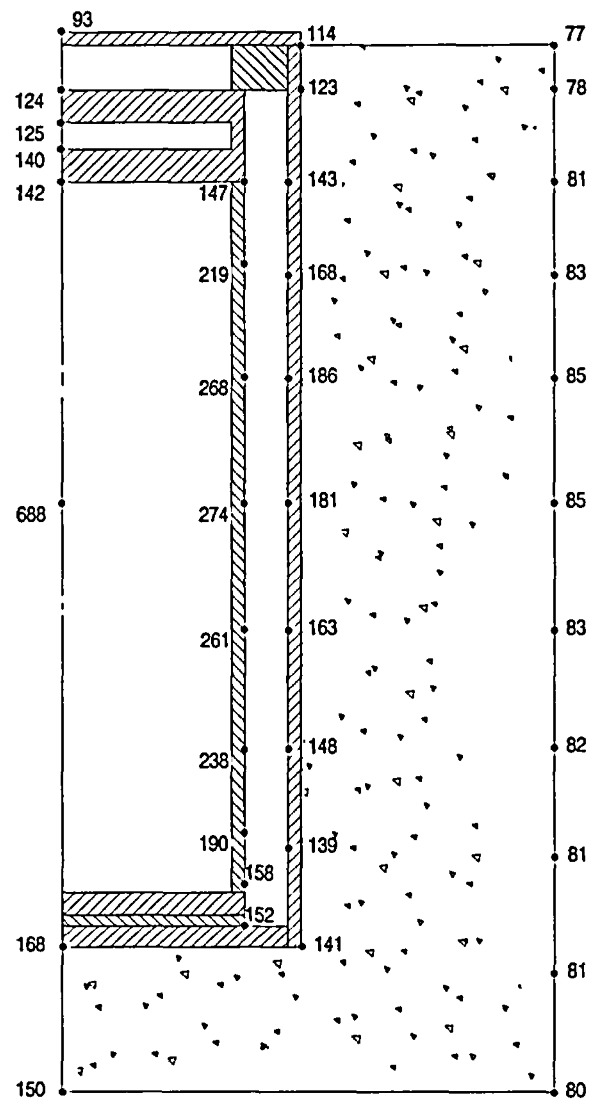


Figure 4.4-5 - VCC Temperature Distribution 75°F Day

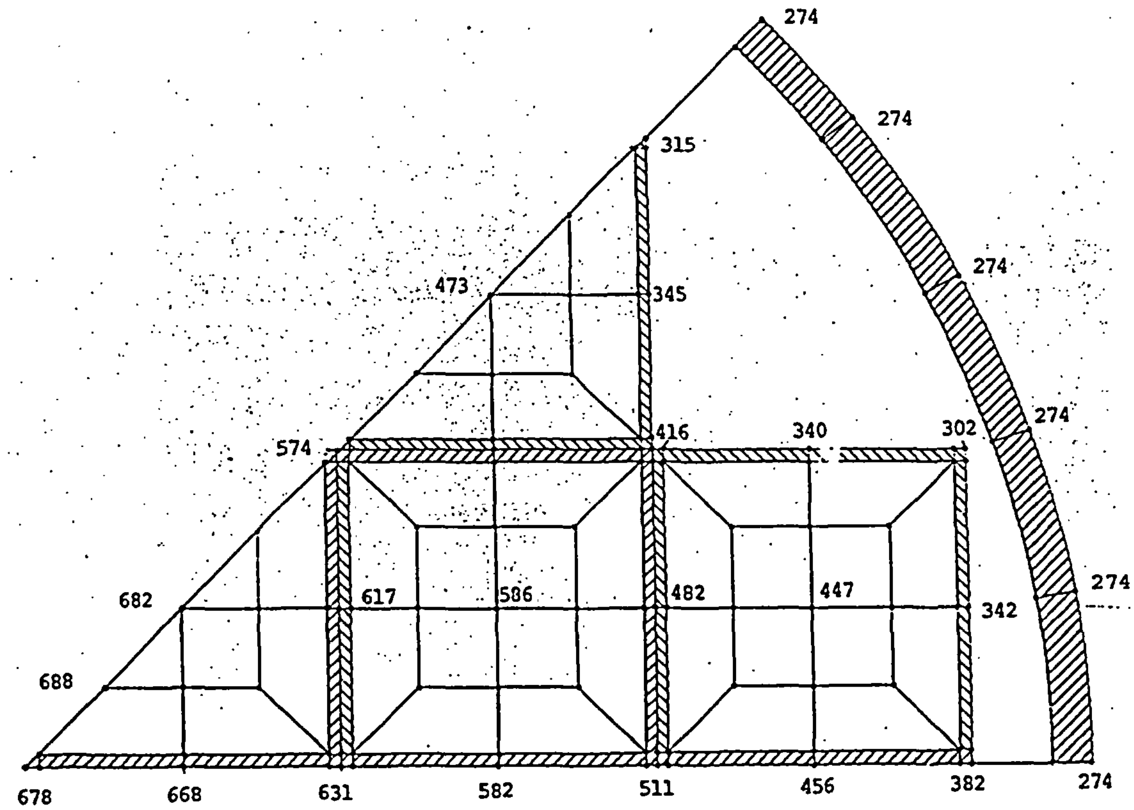


Figure 4.4-6 - MSB Temperature Distribution

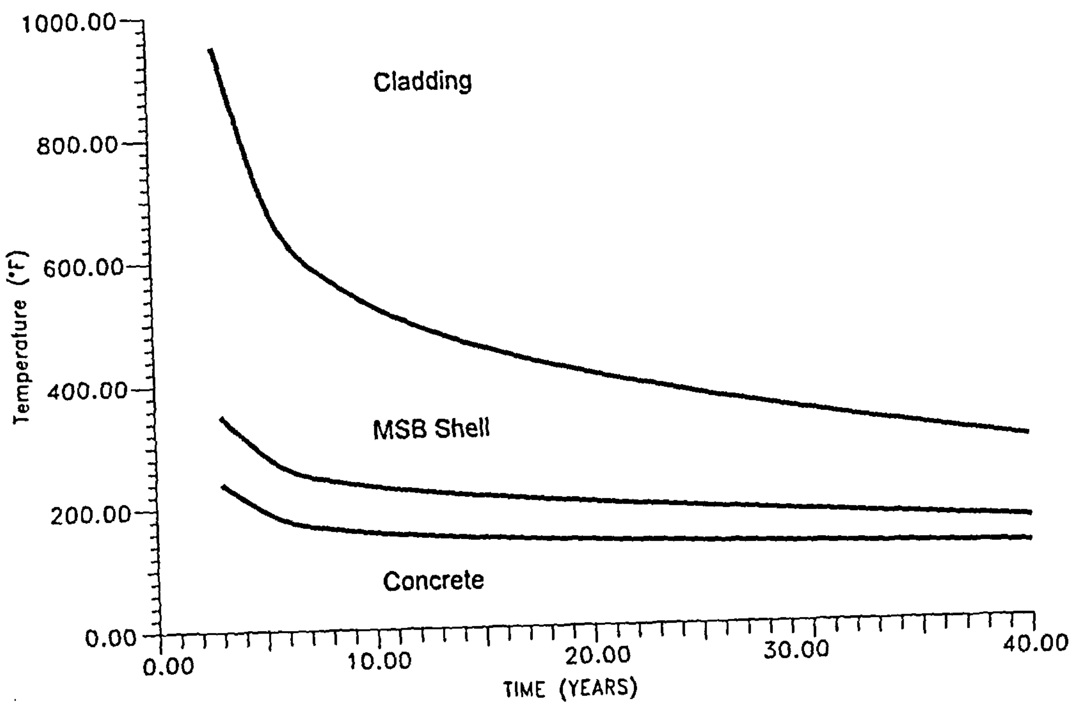


Figure 4.4-7 - Component Temperatures Versus Time

5.0 SHIELDING EVALUATION

This chapter presents the shielding evaluation for the VSC-24 dry fuel storage system. The calculations presented in this section determine exterior gamma and neutron dose rates for VSC-24 storage and transfer casks containing MSBs loaded with design basis PWR fuel. The calculations demonstrate that, for design basis PWR fuel, the storage cask side and top average surface dose rates do not exceed 100 and 200 mrem/hr, respectively, and that the storage cask inlet and outlet dose rates remain below 350 and 100 mrem/hr, respectively.

Limits are placed on the surface average, as opposed to peak local dose rates, since the surface average dose rates govern the doses far from the cask (e.g., at the controlled area boundary, where the regulatory dose limits apply). The surface average side and top dose rates (defined as the integrated average, over the entire surface, of all the local dose rate values) are explicitly calculated by the MCNP shielding code (described in Section 5.4). Dose rates in excess of the surface average limits given above, over localized areas of the cask surfaces, are acceptable. Estimates of peak dose rates at several key locations on the cask surface are also provided by the shielding analysis, as discussed in Section 5.1. These higher localized dose rates are fully accounted for in the surface average dose rate calculations. However, there are no specified maximum values for peak (localized) dose rates, for the reasons stated previously.

Supplementary shielding analyses, presented in Section 5.5.1, determine a wide range of combinations of fuel burnup, enrichment, and cooling times that meet these storage cask surface dose rate criteria. For each combination of fuel assembly burnup and initial enrichment, the minimum cooling time required to meet these dose rate criteria, along with the maximum assembly heat generation criterion of 1.0 kW/assembly (discussed in Chapter 4), is determined by the calculations.

5.1 DISCUSSION AND RESULTS

The locations at which gamma and neutron dose rates are calculated are illustrated in Figure 5.1-1 and Figure 5.1-2 for the VSC-24 storage and transfer casks, respectively. The calculated dose rates are presented to the nearest mrem/hr in Table 5.1-1 and Table 5.1-2.

For the storage cask (VCC), dose rates are calculated on the side surface, over the peak burnup section of the fuel inside the MSB, and directly above the PWR assembly top and bottom hardware regions. Dose rates are also calculated at the center of the VCC top lid, at the inlet and outlet duct entrances, and one meter from the cask side and (center) top surfaces. Surface average dose rates are also calculated and presented for the cask side and top.

For the transfer cask (MTC), dose rates are calculated on the side surface over the peak burnup section of the fuel, and over the top and bottom assembly hardware zones. On the MTC top surface, dose rates are calculated at the center of the MSB top (structural) lid and at the center of the MSB shield lid (with the structural lid removed). On the MTC bottom surface, dose rates are calculated at the cask centerline and at a point near the edge of the bottom surface. Dose rates are also calculated one meter from the MTC side surface.

The calculated surface average dose rates for the storage cask side and top, presented in Table 5.1-1, are directly comparable to the specified surface average dose rate limits, discussed below, for the cask side and top, respectively. The values presented in Table 5.1-1 show that the surface average dose rate criteria are met for design basis PWR fuel.

The dose rates presented in Table 5.1-1 and Table 5.1-2 are based on an MSB loaded with the design basis PWR fuel described in Section 5.2. As discussed in Section 5.2, the design basis gamma and neutron source strengths are based on assembly parameters (e.g., uranium loadings and hardware cobalt quantities) that are bounding for all PWR fuel, including any type of plugging (thimble) insert and all types of burnable poison rod inserts. Alternative sets of PWR fuel parameters (i.e., burnups, enrichments, and cooling times) are evaluated in Section 5.5.1.

Based on the results shown in Table 5.1-1, maximum allowable surface average dose rates of 100 and 200 mrem/hr are specified for the storage cask side and top surfaces, respectively. Maximum dose rates of 350 and 100 mrem/hr are specified for the inlet and outlet ducts, respectively. Surface average, as opposed to peak (local) dose rate limits, are specified for the storage cask side and top surfaces because dose rates at any significant distance from the casks are primarily affected by surface average, as opposed to localized, dose rates. The only regulatory dose rate limit that applies for the VSC-24 cask system is the 10CFR72 dose rate limit of 25 mrem/year at the ISFSI controlled area boundary, far away from the casks.

The dose rate results presented in Table 5.1-1 and Table 5.1-2 are based on extremely conservative shielding analysis assumptions and methodologies. Dose rate measurements on VSC-24 casks currently in use have shown much lower dose rates. Based on previous cask experience, dose rates are expected to remain under 25 mrem/hr on the storage cask side surface, and under 50 mrem/hr for the inlet and outlet ducts.

5.2 SOURCE SPECIFICATION

The VSC-24 shielding analyses are based on an MSB fully loaded with design basis PWR fuel. The design basis PWR fuel parameters are as follows:

Fuel Burnup:	35,000 MWd/MTU
Initial Enrichment:	3.2 w/o U-235
Cooling Time:	5 years
Uranium Loading:	0.5187 MTU/assembly

As discussed in Section 5.3, the shielding analyses model an active fuel length of 147 inches. At this active fuel zone length, the modeled assembly uranium loading of 0.5187 MTU/assembly is bounding for all PWR assembly types with respect to both overall uranium loading and uranium loading per inch of fuel height. This assumption maximizes the fuel region gamma source modeled in the shielding analyses. It should be noted that the VSC-24 specifications actually limit assembly uranium loading to 0.471 MTU/assembly. Thus, the 0.5187 MTU/assembly value modeled in the shielding analyses is conservative for all fuel that may be loaded into the VSC-24.

Based on the fuel parameters listed above, gamma and neutron source strengths from the spent fuel material and gamma source strengths from the irradiated assembly hardware are calculated using the ORIGEN2 point depletion code (Reference 5.1). Gamma and neutron source strengths are calculated on a per assembly basis, based on the bounding uranium loading listed above. The ORIGEN2 calculations are performed using the PWR extended burnup cross-section library, "PWR-UE." This cross-section library is recommended for PWR assembly burnup levels over 33 GWd/MTU.

The ORIGEN2 analyses assume continuous assembly irradiation at a power level of 16.114 MW/assembly. The irradiation period is set to achieve the desired burnup level at this power level, assuming the design basis uranium loading of 0.5187 MTU/assembly. Any decay periods between assembly irradiation periods are conservatively neglected. After the irradiation period, the ORIGEN2 code decays the spent fuel over a period of 5 years.

5.2.1 GAMMA SOURCE DESCRIPTION

5.2.1.1 Fuel Gamma Source

The gamma sources from the spent UO_2 fuel are calculated using the ORIGEN2 point depletion code, as described in Section 5.2. For the primary gamma shielding analyses, which are used to determine the gamma dose rates on the VCC side, at the VCC outlet ducts, and on the transfer cask exterior, a fuel burnup level of 42 GWd/MTU is actually modeled, as opposed to the design basis burnup level of 35 GWd/MTU. This is done to conservatively model the effects of the axial burnup profile in the fuel, as discussed in Section 5.2.1.3. This 42 GWd/MTU is modeled over the entire axial length of the fuel.

The fuel gamma source strengths for the primary analyses are calculated based on a single, continuous irradiation period of 1352 days. This is the irradiation time necessary to reach a fuel burnup level of 42 GWd/MTU, at a power level of 16.114 MW/assembly and a uranium loading of 0.5187 MTU/assembly.

The primary analysis fuel gamma source description in Table 5.2-1 provides gamma source strengths, in MeV, as a function of gamma energy. The presented set of gamma energy lines corresponds to the ORIGEN2 gamma group structure; the group structure in which ORIGEN2 outputs gamma source strengths. The gamma source strengths are presented on a per assembly basis. The additional 1.25 MeV source strength from activated fuel zone assembly hardware, discussed in Section 5.2.1.2, is added to the fuel gamma source strength to yield the total gamma source strength on a per assembly basis. The per assembly source strengths are multiplied by 24, the number of PWR assemblies in a fully loaded MSB, to yield the total gamma source strengths in the MSB.

The supplementary gamma shielding analyses, used to calculate the VCC top and inlet duct gamma dose rates, model the actual design basis assembly average burnup level of 35 GWd/MTU, and explicitly model the axial burnup profile in the fuel. For these analyses, a second ORIGEN2 calculation is performed. This ORIGEN2 calculation is identical to the one performed for the primary gamma shielding analyses except that the irradiation period is reduced to 1126.63 days, the irradiation time necessary to reach a burnup level of 35 GWd/MTU. This ORIGEN2 analysis yields a total gamma source strength of 7.604×10^{15} gammas/sec-assembly, as opposed to the source strength of 9.568×10^{15} gammas/sec-assembly shown for the 42 GWd/MTU case in Table 5.2-1. The supplementary gamma shielding analyses model this lower total source strength, along with the axial source strength profile, as discussed in Section 5.2.1.3. Although these analyses model the lower (35 GWd/MTU) total source strength, they assume the same gamma source spectrum that is shown for the primary analysis case (42 GWd/MTU) in Table 5.2-1. This assumption is conservative, because the 42 GWd/MTU source spectrum, shown in Table 5.2-1, is harder than the source spectrum that occurs for 35 GWd/MTU fuel.

The gamma source strengths modeled in the primary and supplementary shielding analyses are bounding for any U.S. PWR assembly with a uranium loading up to 0.5187 MTU/assembly. However, the actual maximum allowed assembly uranium loading for the VSC-24 cask system is 0.471 MTU/assembly.

5.2.1.2 Assembly Hardware Gamma Sources

The shielding analyses consider the additional assembly gamma sources from activated assembly metal hardware. These gammas are almost entirely due to Co-60 activity within the activated metals. The PWR assembly Co-60 activity levels and the corresponding gamma source strengths are directly calculated by the ORIGEN2 analyses. The PWR assembly is divided into four axial zones, each containing metal hardware: the fuel zone, the bottom nozzle zone, the gas plenum zone, and the top nozzle zone. The ORIGEN2 calculations determine separate gamma source strengths for each of these axial zones. In the shielding analyses, the PWR assembly is subdivided into these four axial zones, each with its own explicit gamma source description.

For each individual axial assembly zone, the design basis shielding analyses conservatively consider the PWR assembly type and control component combination that yields the maximum overall quantity of cobalt, thereby maximizing the Co-60 activity in each axial zone. Because the assembly and control component type that yields the maximum cobalt quantity may be different for each axial assembly zone, the analyses assume an artificial assembly that simultaneously contains the worst combination in each axial zone. The maximum cobalt quantity evaluation covers all PWR assembly types, including all types of burnable poison rod (BPRA) and thimble plug (TPA) control components. They do not cover W 15x15 and CE 16x16 assemblies containing control components, because the VSC-24 license does not allow these assemblies to be stored with control components.

A set of isotopic masses (in grams/assembly for each isotope) is entered into the ORIGEN2 code input for each of the four assembly axial zones. These sets of isotopic masses correspond to all non-fuel materials in each zone, including fuel rod cladding, and the stainless steel and Inconel present in any assembly hardware. ORIGEN2 is then asked to "irradiate" each set of masses individually. Each set of isotope masses is exposed, therefore, to the neutron flux history that corresponds to the design basis set of fuel conditions (i.e., initial fuel enrichment, assembly thermal power, irradiation period, etc.) ORIGEN2 then outputs total gamma source strengths, in units of gammas/sec-assembly, for each set of isotope masses. These hardware gamma source strengths are almost entirely due to Co-60 activity. However, the assembly hardware gamma total source strengths output by ORIGEN2 also include gammas from all other isotope activities in the activated metal.

The sets of isotope masses that correspond to each of the three non-fuel axial assembly zones are irradiated to the design basis assembly average burnup of 35 GWd/MTU in the ORIGEN2 calculations. Thus, as with the fuel material, these sets of isotopic masses are irradiated at an assembly power level of 16.114 MW/assembly over a single, continuous irradiation period of 1126.63 days. The isotopic masses of fuel zone assembly hardware are irradiated to the same burnup level as the fuel material. For the primary gamma shielding analyses, the fuel zone assembly hardware materials are irradiated (over a 1352-day period) to a burnup level of 42 GWd/MTU. For the supplementary gamma shielding analyses, the fuel zone assembly hardware materials are irradiated (over a 1126.63-day period) to a burnup level of 35 GWd/MTU. One exception to this is that the fuel zone control component hardware materials are irradiated to a burnup level of 35 GWd/MTU for both the primary and supplemental gamma shielding analyses.

Lower neutron flux levels will occur in the non-fuel regions of the assembly relative to the core (fuel) zone. In the ORIGEN2 calculations that irradiate the sets of isotopic masses from the non-fuel zones, the neutron flux levels are scaled down using published scaling factors (Reference 5.2).

The assembly hardware gamma source strength calculations for each of the four axial assembly zones are summarized in Table 5.2-2. For each axial zone (i.e., fuel, bottom nozzle, gas plenum, and top nozzle), Table 5.2-2 separately lists the total metal masses modeled in the analyses for the assembly and control component hardware. These total metal masses are listed for zircaloy, stainless steel (304), and Inconel (718).

Although the ORIGEN2 calculations include the gamma source strengths from all isotopes in the activated metal hardware, the great majority of the gamma source comes from Co-60 activation, which is proportional to the initial cobalt quantity in the assembly hardware materials. As shown in Table 5.2-2, the total initial cobalt quantity (in grams/assembly) present in each assembly axial zone is calculated based on the metal masses present in each zone and the upper bound cobalt concentration assumed for each metal type. These upper bound cobalt concentrations are listed for zircaloy, stainless steel 304, and Inconel-718.

The neutron flux scaling factors presented for each assembly zone in Table 5.2-2 represent the amount of neutron flux that occurs in each non-fuel zone, relative to that of the fuel zone. Thus, the factor listed for the fuel zone is, by definition, 1.0, while lower factors are listed for each of the non-fuel zones.

The assembly hardware total gamma source strengths (in gammas/sec-assembly), presented for each of the four axial assembly hardware zones in the bottom row of Table 5.2-2, are directly calculated by ORIGEN2 based on the metal quantity and cobalt concentration data provided in the other rows of the table. For the non-fuel assembly hardware zones, the total gamma source strengths from the assembly hardware and the control component hardware are listed separately for each axial assembly zone in Table 5.2-3.

The fuel zone gamma source strengths for assembly and control component hardware are given as a function of energy in Table 5.2-1. Gamma source strengths are presented on a per canister basis (24 times the per assembly source strengths) in Table 5.2-1 and Table 5.2-3. For the fuel zone assembly hardware gamma source strengths, the complete energy-dependent gamma source descriptions presented in Table 5.2-1 are modeled explicitly in the shielding calculations. For the three assembly non-fuel regions, the gamma energy spectrum of Co-60 decay is assumed because Co-60 decay is responsible for the overwhelming majority of the assembly hardware gamma source strengths. Because Co-60 decay yields two gammas (a 1.173 MeV gamma and a 1.333 MeV gamma), the total gamma source strength is divided into two gamma line energies, with half the gammas at 1.173 MeV, and the other half at 1.333 MeV. Thus, the gamma sources in the three non-fuel assembly hardware zones are defined by the two line energies given above, with the total gamma source strengths shown in Table 5.2-3.

The assembly hardware gamma source strengths presented in Table 5.2-2 and Table 5.2-3 are bounding for all U.S. PWR assembly types. The source strengths also bound the presence of BPRAs (stainless steel or zircaloy clad) and plugging or thimble plug (TPA) inserts within any of the qualified PWR assembly types, except W 15x15 and CE 16x16. These source strengths are also applicable (i.e. bounding) for PWR assemblies with any number of fuel rods that are replaced with gadolinium-bearing fuel rods, zircaloy dummy rods, zircaloy guide bars, or zircaloy-clad poison rods. None of the poison materials used in PWR assemblies, including B_4C, Gd_2O_3, Al_2O_3, hafnium, or Ag-In-Cd, produce significant amounts of penetrating (gamma) radiation. Assemblies that use any type of fixed zircaloy-clad poison rods in the guide tube locations, as opposed to BPRA inserts, are also covered by the analysis.

The fuel zone assembly hardware gamma source strengths presented in Table 5.2-2 and Table 5.2-3 do not necessarily bound assemblies with fuel rods that have been replaced by solid stainless steel rods or rods containing stainless steel slugs. Before these assemblies can be loaded into the MSB, an evaluation must be performed to verify that the overall cobalt quantities within the assembly fuel zone are bounded by the quantities shown in Table 5.2-2. Assemblies with fuel zone cobalt quantities in excess of the quantities shown in Table 5.2-2 are evaluated in Section 5.5.2. The Section 5.5.2 analyses show that assemblies with up to 250 grams of fuel zone cobalt may be loaded into the center 12 fuel sleeves of the MSB without causing any of the cask exterior dose rate limits (specified in Section 5.0) to be exceeded. This conclusion applies for any assembly with fuel parameters that meet the requirements listed in Table 5.5-1. Thus, assemblies with fuel zone cobalt quantities greater than the amount shown in Table 5.2-2, but less than 250 grams/assembly, may be loaded into the MSB, if they are loaded into the 12 inner fuel sleeves.

5.2.1.3 Axial Gamma Source Strength Profile

The design basis assembly average burnup level is 35 GWd/MTU; however, an axial profile, or variation, in the assembly burnup level exists within the active fuel region. The fuel zone gamma source description used in the gamma shielding analyses must consider the effects of this axial profile.

Published data shows that the peaking factor, defined as the peak local burnup level over the assembly averaged burnup level, is less than 1.2 for all PWR fuel with burnup levels over 20 GWd/MTU (Reference 5.3). Thus, the maximum local burnup level in PWR fuel with an assembly average burnup level of 35 GWd/MTU is less than 42 GWd/MTU.

The primary gamma shielding analyses – used to calculate storage cask side, storage cask outlet duct, and all transfer cask gamma dose rates – conservatively assume a burnup level of 42 GWd/MTU over the entire axial height of the fuel zone. An ORIGEN2 calculation determines the total fuel region gamma source strength, including the core zone assembly hardware, for an assembly burnup level of 42 GWd/MTU. (The control component hardware gamma source strengths are calculated based on a burnup level of 35 GWd/MTU.)

As discussed in Sections 5.3 and 5.4, additional gamma shielding analyses were performed to obtain better results for the air inlet gamma dose rates and the storage cask top gamma dose rates. These shielding analyses did not model the peak local burnup level of 42 GWd/MTU over the entire fuel zone. Instead, the actual axial gamma source strength profile was modeled explicitly.

The axial burnup profile in spent fuel varies with fuel burnup but does not vary significantly with initial enrichment or cooling time. PWR fuel axial burnup profiles for different assembly average burnup levels are available in published reports (Reference 5.3).

The axial burnup profile assumed in the supplementary gamma shielding analyses is presented in Table 5.2-4. The active fuel height of 147 inches is subdivided into 18 equal-height axial zones. The axial span covered by each subsection is defined relative to the bottom of the active fuel region (i.e., 0.0 is the bottom of the fuel). The relative burnup level modeled in each axial subsection is the actual local burnup level that is modeled, divided by the assembly average burnup level (35 GWd/MTU).

The set of relative burnup levels presented in Table 5.2-4 bounds all axial burnup profiles that occur for PWR fuel with burnup levels between 30 and 50 GWd/MTU. Published data provides normalized axial burnup profiles for each burnup level between 30 and 50 GWd/MTU (Reference 5.3). Thus, for each assembly average burnup level, a certain local relative burnup level is presented for each axial location. The relative burnup levels, shown in Table 5.2-4 for each axial location, bound all of the relative burnup levels that occur for any of the assembly average burnup levels between 30 and 50 GWd/MTU.

Given that each of the published axial burnup profiles is normalized, the bounding relative burnup profile shown in Table 5.2-4 will not be normalized (i.e., the average relative burnup level will be greater than 1.0). Examination of the relative burnup levels in Table 5.2-4 confirms this. If the relative burnup levels shown are summed and then divided by 18 (the number of equal-height axial intervals), the resulting average relative burnup level, as shown in Table 5.2-4, is 1.017, as opposed to 1.0. This reflects the fact that the overall assembly average burnup level must be increased to have local burnup levels at each axial location that are bounding for all assembly average burnup levels.

As discussed earlier, gamma source strengths are nearly directly proportional to the fuel burnup level. Therefore, the axial profile of the burnup level is assumed for the axial profile of the gamma source strength. The relative gamma source strengths for each axial zone (defined as the local gamma source strength over the gamma source strength calculated at the assembly average burnup level of 35 GWd/MTU) are equal to the relative burnup levels shown in Table 5.2-4.

To determine a normalized axial gamma source strength profile for use in the shielding analyses, the relative gamma source strength values in Table 5.2-4 are divided by the average relative gamma source strength value (presented at the bottom of the second column). This normalized axial source strength profile is shown in the fourth column of the table. The relative source strengths shown in this column average out to 1.0. To determine the fraction of the total gamma source in each (equal height) axial section, the relative source strengths shown in the fourth column are divided by 18, the number of axial sections.

Although the shielding analysis models require a normalized gamma source strength profile as input, the increase in overall assembly gamma source strength from these axial-profile-related effects must be treated by the analyses. As shown by the average relative burnup/gamma source strength in Table 5.2-4, the total gamma source strength of the assembly is increased by ~2% due to the modeled axial burnup profile. The total assembly gamma source strength given in Table 5.2-1 must be increased by this factor in the shielding analyses.

5.2.2 NEUTRON SOURCE DESCRIPTION

5.2.2.1 Total Neutron Source Strength

As with the gamma sources, the neutron source strengths are calculated directly using the ORIGEN2 point depletion code. In the ORIGEN2 analysis, the neutron source strengths are calculated on a per assembly basis assuming an upper bound uranium loading of 0.5187 MTU/assembly.

The neutron source modeled in the storage cask shielding analysis is based on the design basis assembly average burnup level of 35 GWd/MTU. The VCC neutron shielding analyses explicitly model the axial burnup profile and corresponding neutron source strength profile in the fuel, as discussed in Section 5.2.2.2. The neutron source modeled in the transfer cask shielding analysis is conservatively based on a fuel burnup level of 42 GWd/MTU. This fuel burnup and corresponding neutron source strength are modeled over the entire axial height of the fuel.

To determine the storage cask analysis neutron source strength, the fuel material is irradiated at an assembly thermal power level of 16.114 MW/assembly over a single continuous irradiation period of 1126.63 days, which corresponds to a burnup level of 35 GWd/MTU. The fuel is then decayed to a cooling time of 5 years. For the transfer cask analysis neutron source strength, the fuel material is irradiated at the same assembly thermal power level over an irradiation period of 1352 days, which corresponds to a burnup level of 42 GWd/MTU.

These ORIGEN2 analyses yield a total neutron source strength of 2.110×10^8 neutrons/sec-assembly for the storage cask case, which corresponds to a total MSB neutron source strength of 5.064×10^9 neutrons/sec-MSB. For the transfer cask case, the analyses yield a total neutron source strength of 4.613×10^8 neutrons/sec-assembly, which corresponds to a total MSB neutron source strength of 1.107×10^{10} neutrons/sec-MSB.

The Watt fission spectrum for Cm-244 is assumed as the energy spectrum of the design basis neutron source (Reference 5.4). This normalized energy spectrum is given by the following formula:

$$f(E) = C \cdot \exp(-E/0.906) \cdot \sinh(3.848 \cdot E)^{1/2}$$

This spectrum accurately describes the energy spectrum of the neutron source from spent LWR fuel and does not vary significantly with fuel assembly burnup, initial enrichment, or cooling time.

The neutron source strengths presented above are bounding for any U.S. PWR assembly with a uranium loading up to 0.5187 MTU/assembly. However, the actual maximum allowed assembly uranium loading for the VSC-24 cask system is 0.471 MTU/assembly.

5.2.2.2 Axial Neutron Source Strength Profile

The storage cask neutron shielding analyses explicitly model the axial variation of the neutron source strength, which is created by the axial burnup profile present in the spent fuel. The axial burnup profile in spent fuel varies with fuel burnup but does not vary significantly with initial enrichment or cooling time. PWR fuel axial burnup profiles for different assembly average burnup levels are available in published reports (Reference 5.3).

The axial burnup profile assumed in the neutron shielding analyses is presented in Table 5.2-4. The active fuel height of 147 inches is subdivided into 18 equal-height axial zones. The axial span covered by each subsection, shown in Table 5.2-4, is defined relative to the bottom of the active fuel region (i.e., 0.0 is the bottom of the fuel). The relative burnup level modeled in each axial subsection is the actual local burnup level that is modeled, divided by the assembly average burnup level of 35 GWd/MTU.

The set of relative burnup levels presented in Table 5.2-4 bounds all axial burnup profiles that occur for PWR fuel with burnup levels between 30 and 50 GWd/MTU. Published data provides normalized axial burnup profiles for each burnup level between 30 and 50 GWd/MTU (Reference 5.3). Thus, for each assembly average burnup level, a certain local relative burnup level is presented for each axial location. The relative burnup levels, shown in Table 5.2-4 for each axial location, bound all of the relative burnup levels that occur for any of the assembly average burnup levels between 30 and 50 GWd/MTU.

Given that each of the axial burnup profiles is normalized, the bounding relative burnup profile shown in Table 5.2-4 will not be normalized (i.e., the average relative burnup level will be greater than 1.0). Examination of the relative burnup levels in Table 5.2-4 confirms this. If the relative burnup levels shown are summed and then divided by 18 (the number of equal-height axial intervals), the resulting average relative burnup level is 1.205, as opposed to 1.0. This result reflects the fact that the overall assembly average burnup level must be increased to have local burnup levels at each axial location that are bounding for all assembly average burnup levels.

The neutron source strength has a strong nonlinear dependence on burnup. Neutron source strengths have been shown to scale roughly as the burnup level to the 4th power (Reference 5.2). Due to this non-linear dependence, the axial profile of the neutron source strength is much more strongly peaked than the axial burnup profile. The neutron source strength profile is determined by raising each of the relative burnup level values in Table 5.2-4 to the 4th power. The resulting relative neutron source strength values represent the local neutron source strength present in each axial zone, divided by the neutron source strength that would occur at the assembly average burnup level of 35 GWd/MTU.

The relative neutron source strength values are averaged to yield an overall relative neutron source strength value, which is shown at the bottom of the third column in Table 5.2-4. This value is greater than 1.0 for two reasons. First, as discussed above, the relative burnups in the axial zones sum to more than 1.0 due to the bounding burnup profile that is assumed. Second, the non-linear dependence of neutron source strength on burnup will cause the relative neutron source strengths to sum to more than 1.0, even for a normalized axial burnup profile.

The shielding analyses require a normalized burnup profile. Therefore, a normalized axial neutron source strength profile is determined by dividing the relative neutron source strength values in Table 5.2-4 by the total relative neutron source strength value at the bottom of the third column. The resultant normalized profile is shown in the right column of Table 5.2-4. The values shown in this column are divided by 18 (the number of equal-height axial intervals) to yield the fraction of the total neutron source strength within each axial bin. These fractions are used as input for the shielding analyses.

Although the shielding analysis models require a normalized neutron source strength profile as input, the analyses must also treat the increase in overall assembly neutron source strength from the axial-profile-related effects discussed above. The total neutron source strength of the assembly is increased by ~20% (i.e., by the total relative neutron source strength shown at the bottom of the third column of Table 5.2-4) due to the modeled axial burnup profile. The total assembly neutron source strength given in Section 5.2.2.1 must be increased by this factor in the storage cask neutron shielding analyses.

As discussed in Section 5.2.2.1, no axial source strength profile is modeled in the transfer cask neutron shielding analyses. These analyses conservatively model a burnup level of 42 GWd/MTU over the entire axial length of the fuel. This burnup level, which is 20% higher than the assembly average burnup level of 35 GWd/MTU, is bounding for all axial locations within the actual design basis fuel. As shown in Table 5.2-4, the peak local burnup is 12.6% higher than the assembly average, which corresponds to a peak local burnup level of less than 40 GWd/MTU. Since the transfer cask analyses model a flat source strength profile, the increase factor of ~20%, discussed above for the storage cask analysis, does not apply.

5.3 MODEL SPECIFICATION

5.3.1 STORAGE CASK MODEL GEOMETRY

5.3.1.1 Storage Cask Bulk Model Geometry

In the VSC-24 storage configuration, which consists of the MSB inside the VCC, radial shielding is provided by the 1.0-inch-thick carbon steel MSB shell, the 1.75-inch-thick carbon steel VCC liner, and the 29.0-inch-thick radial concrete shield. At the bottom end, shielding is provided by the 0.75-inch-thick carbon steel canister bottom plate, the 2.0-inch-thick carbon steel VCC bottom plate, and 18 inches of concrete. On the cask top, shielding is provided by the 9.5-inch-thick shield lid (which contains 7.5 inches of carbon steel and 2.0 inches of RX-277 neutron shielding material), the 3.0-inch-thick carbon steel structural lid, and the 0.75-inch-thick carbon steel VCC lid. All of the cask lids are carbon steel. The storage cask configuration includes a 4.0-inch-wide ventilation duct. A 6.0-inch-thick steel ring is placed at the top of this duct to provide protection from radiation streaming up the ventilation duct. The bulk storage cask shielding geometry modeled in the shielding analyses is shown in Figure 5.3-1.

5.3.1.2 MSB Interior Model Geometry

As discussed in Section 5.2, the MSB interior is divided into four axial zones corresponding to the four axial source regions in the PWR assemblies: the active fuel region, bottom nozzle region, gas plenum region, and top nozzle region. The axial heights of these MSB interior zones are shown in Figure 5.3-1.

5.3.1.2.1 Primary Shielding Analysis Model

The primary gamma shielding analyses explicitly model the 24 carbon steel fuel sleeves inside the MSB. These sleeves have a wall thickness of 0.2 inch and an inner width of 8.8 inches. Inside these fuel sleeves, the PWR assemblies are modeled as a homogenous material mixture that is spread over the assembly envelope area (8.536 inches square). Thus, a void region around the assembly, located between the assembly envelope and the inner fuel sleeve wall, is modeled. All MSB interior components other than the fuel assemblies and the fuel sleeves are conservatively neglected in the shielding model. A horizontal cross-sectional view of the MSB interior model is shown in Figure 5.3-2.

The configuration shown in Figure 5.3-2 is modeled over the entire axial length of the MSB. Four different homogenous mixtures are modeled within the assembly envelope, one for each of the axial assembly zones shown in Figure 5.3-1. Each mixture is modeled over each axial span shown in Figure 5.3-1 for the four axial assembly zones. Void is modeled in the basket edge area outside the fuel sleeves over the entire length of the MSB interior.

5.3.1.2.2 Supplementary Gamma Analysis Model

As discussed in Section 5.4, additional gamma shielding analyses were performed to obtain better particle statistics on the storage cask top and storage cask inlet vent. These analyses did not explicitly model the individual fuel sleeves shown in Figure 5.3-2. Instead, the 24 PWR assemblies were smeared (i.e., homogenized) into a single cylinder that has a cross-sectional area equal to that of the 24 fuel sleeves shown in Figure 5.3-2. The radius of this equivalent area cylinder is 25.43 inches. These analyses conservatively neglect the fuel sleeve steel (i.e., it is not included in the homogenized material that fills the cylindrical volume). Because gamma radiation may stream up the fuel sleeves, smearing the fuel sleeve steel into the homogenous mixture may not be conservative with respect to cask top end dose rates.

The supplementary analyses, however, model the outer walls of the fuel sleeves next to the basket edge as a single 0.2-inch-thick steel shell around the source region cylinder (i.e., a steel shell extending from R=25.43 to R=25.63). This treatment is acceptable since gammas must pass through at least one fuel sleeve wall in order to pass outside the source zone radius. This approach allows an accurate representation of the MSB interior with a radially symmetric model, without introducing any non-conservatism.

As with the rigorous shielding model described in Figure 5.3-2, the supplementary gamma shielding analyses subdivide the MSB interior into four axial subsections, which correspond to the four axial source regions of the fuel assembly. The homogenous material inside the source zone (R=25.43) cylinder is different for each of the four axial assembly zones shown in Figure 5.3-1. Within each axial zone, the homogenous material description inside the source cylinder is calculated based on the assembly component materials within that axial zone. The 0.2-inch steel ring and the void zone outside that ring are modeled over the entire axial height of the MSB interior.

5.3.1.3 Inlet and Outlet Duct Model Geometry

The primary shielding analysis model is also used to calculate gamma and neutron dose rates at the storage cask outlet duct and to calculate neutron dose rates at the cask inlet duct. For the inlet duct gamma dose rates, a supplementary shielding analysis, described previously, is performed. These analyses employ a rigorous 3-D model of the inlet and outlet vent structures.

A vertical cross-sectional view of the outlet duct geometry is shown in Figure 5.3-3. The VCC contains four outlet ducts. Each duct is 48 inches wide and is lined on all sides with 0.5-inch carbon steel. The four ducts are placed at 90-degree intervals around the cask circumference. The cask liner steel and cask concrete that lie between the four outlet ducts are modeled in the shielding analyses.

Vertical and horizontal cross-sectional views of the inlet duct geometry are shown in Figure 5.3-4 and Figure 5.3-5, respectively. Radiation must travel down an annular duct (Region A in Figure 5.3-4), passing through the 2.0-inch-thick cask bottom plate, then extending down through the concrete another five inches. Next, the radiation passes into the large 12-inch-square duct (Region B in Figure 5.3-4 and Figure 5.3-5). From there, the radiation must pass down the first square duct, go around a sharp bend, and pass down the final square duct (Region C in Figure 5.3-5). The two square ducts, shown in Figure 5.3-5, lie in the same axial plane, or elevation. The annular duct is lined with 0.5-inch-thick carbon steel. The Region B and Region C square ducts are lined with 0.5- and 0.25-inch-thick carbon steel, respectively.

The annular duct (Region A in Figure 5.3-4) extends around the entire cask circumference, except for four breaks, 90 degrees apart, two of which lie directly over the location where the two large square ducts intersect (Regions B and C in Figure 5.3-5). The outline of the area where the Region A duct intersects (i.e., cuts into the top steel of) the Region B duct is represented by the dashed line in Figure 5.3-5. Figure 5.3-5 shows a 90-degree section of the cask inlet duct geometry, with only one of the four inlet ducts visible. As discussed in Section 5.4.4, the inlet duct analysis only models one quadrant of the cask geometry. Since the cask and inlet duct geometry is symmetric along the 0- and 90-degree axes (i.e., the quadrants are identical), a quarter cask model may be used. Reflective boundary conditions are applied on the 0- and 90-degree axes of the model, so the full cask geometry is effectively modeled.

5.3.2 TRANSFER CASK MODEL GEOMETRY

In the transfer configuration, radial shielding is provided by the 1.0-inch-thick steel MSB shell, the 0.75-inch-thick steel inner transfer cask liner, the 3.75-inch-thick transfer cask lead shield, the 4.0-inch-thick RX-277 transfer cask neutron shield, and the 1.0-inch-thick steel outer transfer cask wall. These dimensions correspond to the lightweight (thin-walled) transfer cask option, and thus correspond to the bounding condition. Top end shielding is provided by the 9.5-inch-thick steel MSB shield lid, which contains 2.0 inches of RX-277 neutron shielding material, and the 3.0-inch-thick MSB structural lid. At the bottom end, shielding is provided by the 0.75-inch-thick steel MSB bottom plate and the 9.0-inch-thick steel transfer cask bottom doors. All steel in the above configurations is carbon steel. The transfer cask shielding model geometry is illustrated in Figure 5.3-6.

The MSB interior shielding model geometry for the transfer cask shielding analyses is identical to that used in the storage cask shielding analyses, described in Section 5.3.1.2. The fuel sleeve geometry shown in Figure 5.3-2 is explicitly modeled. As with the storage cask analyses, the MSB interior is subdivided into four axial subsections, which correspond to the four axial source zones within the fuel assembly. The axial spans covered by these zones are shown in Figure 5.3-6. In each axial zone, a different homogenous material, corresponding to the different assembly component materials present within each of the defined axial zones, is defined within the assembly envelope areas shown in Figure 5.3-2.

5.3.3 SHIELD REGIONAL DENSITIES

The shielding materials used in the VSC-24 cask system include carbon steel and RX-277 neutron shielding in the MSB; carbon steel and concrete in the VCC; and carbon steel, lead, and RX-277 in the MTC. As discussed in Section 5.3.1.2, the fuel assembly components are smeared into composite, homogenous materials for each of the four axial assembly zones. In the primary shielding analysis models, these homogenized materials fill the fuel assembly envelope volumes shown in Figure 5.3-2. In the supplementary gamma shielding analyses, described in Section 5.3.1.2.2, the homogenized materials fill a large cylindrical volume that corresponds to all 24 fuel sleeves.

The primary shielding materials, which lie outside the MSB interior, are described in Table 5.3-1. These materials include carbon steel, concrete, lead, and RX-277 neutron shielding material. For each of these materials, Table 5.3-1 lists the overall density (in grams/cm^3) and the weight fraction of each element present in the material. Carbon steel is modeled as A516 carbon steel at a density of 7.832 grams/cm^3. Lead is modeled as pure elemental lead with a density of 11.34 grams/cm^3. A concrete density of 140 lb/ft^3 is assumed.

The RX-277 neutron shielding material is used in the MSB top shield lid and in the MTC radial neutron shield cavity. The RX-277 in the MSB shield lid is baked to remove unbound moisture present in the material, which prevents off-gassing within the shield lid neutron shield cavity during fuel storage. Standard (i.e., unbaked) RX-277 material is used in the MTC neutron shield cavity. For this reason, two different RX-277 material descriptions are presented in Table 5.3-1.

The homogenous material descriptions for the four MSB interior axial zones are provided in Table 5.3-2. The atom densities shown are calculated by dividing the total material masses present in each axial zone, by the volume within each zone. For the primary shielding analyses, the assembly component materials are smeared over the assembly envelope area (8.536 inches square). The heights of the volumes over which the assembly materials are smeared are shown in Figure 5.3-1. Thus, for the primary shielding analyses, the total material masses within each zone (for one design basis PWR assembly) are divided by the volume of an 8.536-inch-square area, over the axial height covered by the zone in question.

The densities of uranium and oxygen fuel shown in Table 5.3-2 are based on an assembly uranium loading of 0.5187 MTU/assembly and an active fuel region height of 147 inches. Table 5.2-2 lists the per assembly masses of assembly hardware materials (e.g., zircaloy, stainless steel 304, and Inconel-718) present in each axial assembly zone. These total hardware material masses are divided by the assembly envelope volumes for each axial zone to yield the elemental densities shown in Table 5.3-2. The elemental densities shown for the assembly hardware in Table 5.3-2 are based on published elemental compositions for zircaloy, stainless steel 304, and Inconel-718 (Reference 5.5).

Although the shielding analyses include the additional gamma sources from activated control component materials, they conservatively neglect the self-shielding properties of any control component materials. Therefore, the elemental densities shown in Table 5.3-2 are calculated using only the assembly hardware material masses shown in Table 5.2-2. The control component material masses are not included.

The supplemental gamma shielding analyses, discussed in Section 5.3.1.2.2, smear the material masses of all 24 design basis assemblies over the entire volume of the cylindrical source zone (R=25.43 inches). The area of the source zone is equal to that of 24 fuel sleeves (9.2 inches square), which is greater than that of 24 assembly envelopes (8.536 inches square). In these supplementary models, the assembly materials are being smeared into the volume occupied by the fuel sleeve steel and the volume occupied by the gaps that lie between the assembly envelope and the fuel sleeve interior. For this reason, the densities of the homogenized materials are somewhat lower for the supplementary models than they are for the primary analysis models. However, the compositions (i.e., the relative quantities of each element) of the homogenized materials do not change. Thus, for all of the MSB interior axial zones, the elemental densities modeled in the supplementary gamma shielding analyses are 0.86 times those shown in Table 5.3-2.

5.4 SHIELDING EVALUATION

5.4.1 SHIELDING ANALYSIS CODE

Gamma and neutron dose rates on the VSC-24 storage and transfer cask surfaces are rigorously calculated using the MCNP Monte Carlo code, Version 4A (Reference 5.4). MCNP allows explicit, three-dimensional modeling of the cask system geometry, with explicit elemental material descriptions for all defined zones within the geometry. MCNP also allows an explicit source definition, including any spatial and energy variations. Both axial and radial variations in source strength are explicitly modeled, along with the energy spectrum of the sources. MCNP uses point-wise, energy-dependent cross sections, as opposed to using energy group structures.

The code accurately treats all particle scattering effects and accounts for all secondary particles generated through primary source particle interactions. Secondary gammas generated by the capture of neutrons within the shielding materials are explicitly treated, and the additional gamma dose rates from such particles are separately tallied.

5.4.2 MCNP AREA DETECTORS

MCNP determines cask surface particle fluxes and associated dose rates by tracking the number and energy of particles exiting cask surface sections of some finite area. On the storage and transfer cask sides, the cask surface is subdivided axially into several particle detector, or tally, areas. Each tally area covers a given axial section, or span, of the cask side surface and extends all the way around the cask circumference. On the transfer cask top and bottom end, and on the storage cask top end, the cask surface is subdivided radially into several ring-shaped detector areas, with a disk-shaped area defined at the center of the cask end surface. Thus, each detector covers a finite span in radius.

For all of the cask surface detectors, the detector areas are set small enough to sufficiently describe the spatial variation of the dose rates on the cask surfaces and set large enough to obtain sufficient particle statistics within each detector area.

The storage cask inlet and outlet duct shielding models tally dose rates over the duct areas on the cask surface. Thus, unlike the bulk shielding models described previously, their dose tally areas do not extend over the entire cask circumference. As discussed in Section 5.4.4, the storage cask inlet duct model covers only one quadrant (90-degree section) of the cask geometry. Reflective boundary conditions are applied on the 0- and 90-degree axes to effectively model the entire cask geometry.

5.4.3 FLUX-TO-DOSE CONVERSION FACTORS

After calculating energy-dependent gamma and neutron fluxes over all of the defined tally areas on the cask surfaces, MCNP converts the fluxes into gamma and neutron dose rates using published (ANSI/ANS-6.1.1-1977) flux-to-dose conversion factors (Reference 5.6). These factors, which convert flux (in particles/sec-cm^2) into dose rates (in mrem/hr), vary as a function of particle energy. The flux-to-dose conversion factors for gammas are shown as a function of energy in Table 5.4-1. The neutron flux-to-dose conversion factors are shown in Table 5.4-2.

5.4.4 SUPPLEMENTARY SHIELDING ANALYSES

The primary gamma and neutron shielding analyses for the storage and transfer casks calculate gamma and neutron dose rates on all cask surfaces, including the storage cask inlet and outlet ducts, using a single, explicit three-dimensional model for each cask. However, the storage cask primary gamma shielding model does not obtain sufficient particle statistics (i.e., sufficiently low levels of statistical error in the dose rate results) on the cask top surface and at the cask inlet duct. Therefore, additional or supplementary MCNP gamma shielding models focus on those specific areas of the cask surface. One model focuses on the cask top surface, and one focuses on the cask inlet duct.

The storage cask top gamma shielding analysis model is similar to the primary gamma shielding model, except that the cask bottom geometry and the inlet and outlet duct geometries are not modeled because they do not affect cask top dose rates. Also not modeled are the individual fuel sleeves inside the MSB, as shown in Figure 5.3-2. The radially symmetric (i.e., cylindrical) source zone geometry, described in Section 5.3.1.2.2, is modeled instead for simplicity. The cask top end analysis is designed to focus on gamma transport in the upward direction from the source zone, toward the cask top end surface. This supplementary model achieves much lower levels of statistical error in the cask top end dose rates. The cask top end gamma dose rate results presented in Table 5.1-1 are based on this supplementary gamma analysis model.

The supplementary storage cask gamma inlet model is similar to the primary gamma shielding model, except that the cask top end geometry, including the MSB and VCC top lids and the VCC outlet duct, are not modeled. Also, since the geometry of the four inlet ducts is identical, only a

90-degree section of the cask geometry is modeled. Reflective boundaries are applied on the 0- and 90-degree axes at the edges of the model. Because the cask inlet duct geometry is symmetric about the 0- and 90-degree axes, this effectively models the entire cask and inlet duct geometry. The 90-degree cask section geometry modeled in the supplementary analysis is shown in Figure 5.3-4 and Figure 5.3-5. A single dose tally is defined over the entrance of the inlet duct on the storage cask surface. The supplementary gamma inlet duct model achieves much lower levels of statistical error in the storage cask inlet duct gamma dose rates. The inlet duct gamma dose rates presented in Table 5.1-1 are based on this model.

5.5 SUPPLEMENTARY ANALYSES

5.5.1 ALTERNATIVE FUEL PARAMETER EVALUATION

The shielding calculations presented in this chapter are based on an MSB loaded with design basis PWR fuel. The design basis PWR fuel parameters are a burnup of 35 GWd/MTU, an initial enrichment of 3.2%, and a cooling time of 5 years. The dose rate results presented in Table 5.1-1 and Table 5.1-2 are based on these design basis fuel parameters. PWR assemblies with fuel parameters other than the design basis values are evaluated in this section.

As discussed in Section 5.1, average dose rates of 100 and 200 mrem/hr are specified for the storage cask side and top surfaces, respectively. Dose rates far from the casks, such as at the ISFSI-controlled area boundary, are primarily a function of the side and top surface average dose rates. Additional specifications are given for the storage cask inlet and outlet ducts (350 and 100 mrem/hr for the inlet and outlet ducts, respectively). An assembly heat generation level of 1.0 kW/assembly is also specified based on the thermal analysis design basis defined in Chapter 4. Any fuel to be loaded into the VSC-24 cask system must not cause any of the specified dose rate limits discussed previously to be exceeded.

For both the fuel neutron source and the non-fuel assembly hardware zone gamma sources, the spatial source distribution and the energy spectrum do not vary significantly with fuel burnup, enrichment, or cooling time. For this reason, at any given dose location, the neutron and non-fuel assembly hardware zone gamma dose rates are directly proportional to the total neutron and assembly hardware source strengths, respectively. For the fuel zone gamma source, the gamma energy spectrum varies with fuel burnup, enrichment, and cooling time. The dose rate contribution from each gamma source energy line for the fuel zone is directly proportional to the source strength of that gamma energy line.

The ORIGEN2 point depletion code is used to generate fuel gamma, hardware gamma, and fuel neutron source strengths for a wide range of PWR assembly burnups, initial enrichments, and cooling times. The analyses are performed based on the same design basis assembly configuration described in Section 5.2. Thus, the calculations are based on the design basis assembly uranium loading of 0.5187 MTU/assembly and the assembly hardware masses given in Table 5.2-2.

In all cases, the depletion calculations are based on a single irradiation period, at an assembly power level of 16.114 MW/assembly. The length of the irradiation period is varied to analyze different burnup levels, and the uranium material description in the ORIGEN2 input is varied to analyze different initial enrichment levels. Each ORIGEN2 run is asked to calculate source strength data at several post-irradiation cooling times. The ORIGEN2 analysis output includes total neutron source strengths, total assembly hardware gamma source strengths for each non-fuel axial assembly zone (e.g., bottom nozzle, gas plenum, and top nozzle), and energy-dependent total fuel zone gamma source strengths.

Using the above ORIGEN2 output data, storage cask surface dose rates are estimated for the alternate sets of fuel parameters using source scaling techniques. The design basis shielding analyses, described in Sections 5.1 through 5.4, give complete surface average gamma and neutron dose rate data on the storage cask side and top, and local dose rate data at the inlet and outlet ducts. At each of these locations, the total neutron dose rate, the total assembly hardware dose rate from each non-fuel axial assembly section, and the fuel zone gamma dose rate for each gamma energy line are given.

For each alternative set of fuel parameters (e.g., burnup, enrichment, and cooling time), adjusted gamma and neutron dose rates are determined for each of the above storage cask dose rate locations using source scaling. The neutron dose rate for the alternate case is determined by multiplying the design basis analysis dose rate by the ratio of the total neutron source strength for the alternate case, over the total neutron source strength for the design basis case (i.e., the 35 GWd/MTU, 3.2% enriched, 5-year-cooled case).

For the non-fuel assembly hardware zone gamma dose rate contributions, the design basis dose rates are multiplied by the ratio of the alternate case total gamma source strength over the design basis gamma source strength. This same process is used for each of the three assembly hardware dose rate contributions: bottom nozzle, gas plenum, and top nozzle. This yields adjusted non-fuel zone assembly hardware gamma dose rates for each alternate set of fuel parameters at each storage cask dose location. The fuel zone gamma dose rate contributions are scaled for the alternate fuel parameter cases using the somewhat more detailed source scaling approach described below.

The design basis shielding analyses calculate the dose rate contributions as a function of gamma energy group. (This refers to the energy levels of the gammas leaving the cask surface and does not necessarily correspond to the source gamma energy levels.) As a result of gamma downscatter effects, the gamma dose rate contribution from a given gamma energy group may be partially due to source gammas of a higher energy group. Since upscatter is not significant, none of the cask surface gamma flux in a given energy group is due to source gammas of a lower energy group.

For the above reasons, the following approach is used to scale the fuel gamma dose rate contributions. For each alternate set of fuel parameters, total fuel zone gamma source strengths are calculated, using ORIGEN2, for each gamma energy line.

1. The gamma source strength at each gamma energy line is divided by the design basis case fuel zone gamma source strength at that energy line to yield a source strength ratio for each gamma energy line.
2. The design basis fuel zone gamma dose rate contribution for each gamma energy group is taken from the design basis shielding analyses.
3. This dose rate contribution is multiplied by the highest gamma source strength ratio that occurs for any of the gamma energy levels, which are equal to or higher than the energy level, or group, of the gamma dose rate contribution. This yields an upper bound estimate for each gamma energy line of the fuel zone gamma dose rate contribution for each alternate set of fuel parameters.
4. The dose rate contributions from each energy line are summed to yield the total fuel zone gamma dose rate contribution.

This process is used to estimate fuel gamma dose rate contributions at each of the storage cask surface dose locations.

After the fuel zone gamma dose rate contributions are determined for each dose location, they are added to the non-fuel zone assembly hardware and fuel neutron dose rate contributions, described earlier, to yield total dose rates at each location. This process is used to determine storage cask surface dose rates on the top, side, and inlet and outlet ducts for all combinations of assembly burnup, enrichment, and cooling time.

The ORIGEN2 calculations used to determine the gamma and neutron source strengths for each alternate set of fuel parameters also determine the assembly heat generation level for each set of fuel parameters. Whereas the gamma and neutron source strengths are calculated based on an assembly uranium loading of 0.5187 MTU/assembly (consistent with the primary shielding analyses presented in Sections 5.1 through 5.4), the assembly heat generation levels calculated by ORIGEN2 are scaled to correspond to the actual maximum allowable assembly uranium loading of 0.471 MTU/assembly.

After the ORIGEN2 calculations are performed, the total assembly heat generation is compared against the 1.0 kW/assembly heat generation limit. The total dose rates at each storage cask surface location are compared to specified dose rate limits. As discussed in Section 5.1, the primary shielding analyses yield total dose rates of 87, 150, 315, and 69 mrem/hr for the cask side (surface average), cask top (surface average), inlet duct locations, and outlet duct locations, respectively. Thus, the design basis case calculations show some margins versus the specified maximum allowable dose rates of 100, 200, 350, and 100 mrem/hr, respectively, for those same locations. In order to preserve some of the margin (vs. the dose rate limits) that exists for the design basis case, dose rate limits lower than the maximum allowable values are specified for the alternative burnup and enrichment combinations. For these calculations, dose rate limits of 90, 160, 325, and 75 mrem/hr are specified for the cask side, cask top, inlet duct, and outlet duct locations, respectively.

For each combination of assembly burnup and initial enrichment, the cooling time at which the heat generation limit and all specified dose rate limits are met is determined. The analyses described in this section were performed for PWR assembly burnup levels between 15 and 45 GWd/MTU, and for assembly initial enrichment levels between 1.5% and 4.2%. They considered cooling times of 5 years or more. (Calculated cooling times less than 5 years are rounded up to 5 years.) The analyses only consider cooling times at one-year intervals. The lowest integer year for which all radiological and thermal limits are met is established as the minimum required cooling time. As an example, if all limits were met at a cooling time of 5.1 years, the calculations would establish 6 years as the required cooling time.

The minimum required cooling times, as a function of assembly burnup and initial enrichment, are presented in Table 5.5-1. The assembly burnup and initial enrichment levels referred to in Table 5.5-1 are defined as the assembly average burnup and initial enrichment levels. For assemblies with radially or axially varying burnups or initial enrichments, averages are taken over the entire assembly fuel region volume (i.e., the values are averaged both radially and axially).

As discussed above, these calculations determine the cooling times required to limit the heat generation of an assembly with 0.471 MTU of uranium to 1.0 kW. The thermal analyses in Chapter 4 of the SAR model a 1.0 kW heat generation level over a fuel length of 144 inches. This corresponds to an axial heat generation level of 6.94 W/inch. In order to be bounded by this axial heat generation level, the 0.471 MTU uranium loading specified previously would have to be distributed over a 144-inch axial fuel length. This would correspond to an axial uranium loading of 3.27 kg/inch. The required cooling time results presented in this section are bounding for all assemblies with uranium loadings up to 0.471 MTU, as lower overall loadings will produce overall heat generation levels under 1.0 kW. Similarly, these calculations cover all fuel assemblies with axial uranium loadings up to 3.27 kg/inch, as lower axial uranium loadings will produce axial heat generations levels under the 6.94-W/inch value analyzed in Chapter 4.

Thus, any PWR assembly that has an overall uranium loading of 0.471 MTU/assembly or less and has an axial uranium loading of 3.27 kg/inch or less is covered by the cooling table calculations. Any PWR assembly that does not meet both of these criteria may not be loaded into the VSC-24 cask.

5.5.2 EVALUATION OF ASSEMBLIES WITH HIGH HARDWARE COBALT QUANTITIES

As discussed in Section 5.2.1.2, some PWR assemblies with stainless steel or Inconel components inserted into the standard assembly array may have fuel zone cobalt quantities in excess of the design basis 46.7 gram/assembly value shown in Table 5.2-2. An example of such an assembly would be one that has several fuel rods replaced by solid stainless steel rods. As a result of the shielding effects of the assemblies in the outer 12 fuel sleeves of the MSB, a substantially higher fuel zone cobalt quantity may be present within the inner 12 assemblies without exceeding any of the cask exterior dose rate limits specified in Section 2.3.5.2.

Shielding analyses have been performed that determine the cask exterior gamma dose rate contributions from Co-60 activity within the fuel zones of the assemblies in the 12 inner MSB fuel sleeves. The analyses determined the Co-60 activity that would have to be present within the inner 12 assembly fuel zones to cause any of the cask exterior dose rate limits to be exceeded. The additional gamma dose rate contributions from the additional assembly fuel zone Co-60 activity are added to the design basis cask exterior dose rates discussed in Section 5.1. The resulting total cask exterior dose rates are compared to the specified dose rate limits.

An infinite-height, 45-degree pie section model of the VSC-24 cask system (over the fuel zone of the assemblies) is used to determine cask side surface average dose rate contributions from Co-60 in the fuel zones of the inner 12 assemblies. A horizontal cross-sectional view of this shielding model is shown in Figure 5.5-1. The shielding models described in Sections 5.3.1.2.2 and 5.3.1.3 are used to calculate the fuel zone Co-60 gamma dose rate contributions on the cask top (surface average) and at the cask inlet vent respectively. The cask side dose rate contributions bound those at the cask outlet vent. The only change made to the shielding models in Sections 5.3.1.2.2 and 5.3.1.3 is that the design basis gamma source term is replaced by a Co-60 gamma source strength within the fuel zones of the 12 inner assemblies in the MSB.

Table 5.2-2 shows that a Co-60 gamma source strength of 3.188×10^{14} gammas/sec-assembly results from an assembly fuel zone cobalt quantity of 46.7 grams/assembly. This gamma source strength (from 46.7 grams of cobalt) is based on the design basis fuel parameters of 35 GWd/MTU, 3.2% enrichment, and 5-year cooling. The fuel zone hardware gamma source strength is assumed to vary directly with fuel zone cobalt quantity, based on the relative source strength shown above for the design basis case (i.e., a Co-60 gamma source strength of 6.827×10^{12} gammas/sec-assembly, per gram of initial fuel zone cobalt, is assumed). The shielding analyses described above determine the Co-60 gamma source strength, within the fuel zones of the inner 12 assemblies in the MSB, required to exceed the specified VCC exterior dose rate limits. From the relation given above, this maximum Co-60 gamma source strength is converted into a maximum assembly fuel zone cobalt quantity.

As discussed in Section 5.5.1, the shielding analyses ensure that the dose rates do not exceed 90, 160, 325, and 75 mrem/hr on the cask side, top, inlet duct, and outlet duct, respectively, for any of the combinations of PWR assembly burnup, enrichment, and cooling time that are allowed to be loaded into the VSC-24. This leaves 10, 40, 25, and 25 mrem/hr of margin versus the specified dose rate limits for these four cask exterior locations, respectively.

The supplementary shielding analyses described above show that an additional fuel zone cobalt quantity of 238.6 grams/assembly added to the design basis quantity (and a corresponding additional assembly fuel zone gamma source strength of 1.629×10^{15} gammas/sec-assembly) is required to increase the cask side surface average dose rate by 10 mrem/hr. An additional fuel zone cobalt quantity of 1,288 grams/assembly (and a corresponding additional assembly fuel zone gamma source strength of 8.793×10^{15} gammas/sec-assembly) is required to increase the cask top surface average dose rate by 40 mrem/hr. Also, an additional fuel zone cobalt quantity of 518.1 grams/assembly (and a corresponding additional assembly fuel zone gamma source strength of 3.537×10^{15} gammas/sec-assembly) is required to increase the cask inlet duct dose rate by 25 mrem/hr. The additional assembly fuel zone cobalt quantity (and associated gamma source strength) required to increase the cask outlet duct dose rate by 25 mrem/hr exceeds the 238.6 grams/assembly required to reach the cask side dose rate limit.

Based upon the above, the cask side surface average gamma dose rate margin of 10 mrem/hr is the constraint that limits the maximum allowable fuel zone cobalt quantity for the 12 inner fuel assemblies in the MSB. These assemblies may contain no more than 238.6 grams of additional assembly fuel zone cobalt (over the design basis maximum allowable cobalt quantity of 46.7 grams/assembly, shown in Table 5.2-2). Thus, the total fuel zone cobalt quantity for the 12 inner assemblies is limited to 285.3 grams/assembly. Based upon this, a maximum allowable fuel zone cobalt quantity limit of 250 grams/assembly is applied for PWR assemblies in the 12 inner MSB locations.

Maximum dose rate increases, assuming assemblies with 250 grams of fuel zone cobalt in each of the 12 inner MSB locations, can be determined at each of the four cask exterior locations, based upon the above data. Assemblies with 250 grams of fuel zone cobalt have 203.3 grams of additional cobalt, above the design basis quantity of 46.7 grams. If 238.6 grams of additional cobalt yield a dose rate increase of 10 mrem/hr on the cask side (as discussed above), then 203.3 grams of additional cobalt would yield a cask side gamma dose rate increase of 8.52 mrem/hr. If 1,288 grams of additional cobalt yield a dose rate increase of 40 mrem/hr on the cask top, then 203.3 grams of additional cobalt would yield a cask side gamma dose rate increase of 6.31 mrem/hr. Finally, if 518.1 grams of additional cobalt yield a dose rate increase of 25 mrem/hr at the cask inlet duct, then 203.3 grams of additional cobalt would yield a cask side gamma dose rate increase of 9.81 mrem/hr. As discussed earlier, the gamma dose rate increase at the cask outlet duct is bounded by the increase in the cask side (surface average) dose rate, for any given amount of additional assembly fuel zone cobalt. Therefore, the maximum dose rate increase at the cask outlet duct is less than 8.52 mrem/hr.

In summary, loading the 12 inner MSB locations with PWR fuel assemblies containing 250 grams of fuel zone cobalt (as opposed to the design basis maximum value of 46.7 grams) will cause the total cask side (surface average), cask top (surface average), cask inlet duct, and the cask outlet duct dose rates to increase by 8.52, 6.31, 9.81, and 8.52 mrem/hr, respectively. These dose rate increases are lower than the available dose rate margins of 10, 40, 25, and 25 mrem/hr at those four respective locations.

These dose rate increases were calculated based on the design basis fuel parameters of 35 GWd/MTU, 3.2% enrichment, and 5-year cooling. However, an evaluation of the allowable sets of fuel parameters presented in Table 5.5-1 shows that the assembly hardware activity of the design basis case (for any given cobalt inventory) bounds that of all other allowable sets of fuel parameters. Therefore, the increases in cask exterior dose rates, due to increasing the assembly fuel zone cobalt quantity to 250 grams/assembly, will be less than the available dose rate margins for all of the allowable sets of fuel parameters specified in Table 5.5-1.

Thus, the maximum allowable fuel zone cobalt quantity for the assemblies in the inner 12 fuel sleeves can be raised to 250 grams/assembly without causing any of the specified cask exterior dose rate limits to be exceeded. This maximum assembly fuel zone cobalt quantity is applicable (i.e., acceptable) for all PWR assemblies that meet the fuel parameter requirements provided in Table 5.5-1.

Table 5.1-1 - Storage Cask Exterior Dose Rates

Location	Detector Number (Fig. 5.1-1)	Gamma Dose Rate (mrem/hr)	Neutron Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Cask Top Surface (lid center)	1	209	36	245
Cask Side Surface (over top hardware)	2	99	<< 1	99
Cask Side Surface (over fuel peak)	3	86	1	87
Cask Side Surface (over bottom hardware)	4	58	< 1	59
Outlet Duct	5	68	1	69
Inlet Duct	6	310	5	315
One Meter Above Cask Top	7	271	15	286
One Meter From Cask Side (surface average)	8	30	<< 1	30
Cask Top Surface Average	-	136	15	150
Cask Side Surface Average	-	86	1	87

Table 5.1-2 - Transfer Cask Exterior Dose Rates

Location	Detector Number (Fig. 5.1-2)	Gamma Dose Rate (mrem/hr)	Neutron Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Cask Top Surface (structural lid center)	1	461	271	731
Cask Side Surface (over top hardware)	2	717	88	805
Cask Side Surface (over fuel peak)	3	398	234	632
Cask Side Surface (over bottom hardware)	4	468	134	601
Cask Bottom Surface (lid center)	5	1983	1775	3758
Cask Bottom Surface (lid edge)	6	792	1093	1886
Shield Lid Surface (lid center)	7	10,045	709	10,755
One Meter From Cask Side (surface average)	8	165	73	239

Table 5.2-1 - Assembly Fuel Zone Gamma Source Strengths

Gamma Energy (MeV)	Fuel Gamma Source Strength (γ/sec-assembly)	Hardware Gamma Source Strength (γ/sec-assembly)	Total Gamma Source Strength (γ/sec-assembly)	Total Gamma Source Strength (γ/sec-MSB)
0.01	2.050E+15	1.087E+13	2.061E+15	4.947E+16
0.025	4.841E+14	1.087E+13	4.950E+14	1.188E+16
0.0375	5.278E+14	3.228E+12	5.310E+14	1.275E+16
0.0575	4.087E+14	1.147E+12	4.098E+14	9.835E+15
0.085	2.692E+14	4.545E+11	2.696E+14	6.471E+15
0.125	2.797E+14	2.271E+11	2.799E+14	6.717E+15
0.225	2.244E+14	8.473E+11	2.252E+14	5.406E+15
0.375	1.335E+14	4.730E+12	1.383E+14	3.318E+15
0.575	3.654E+15	6.061E+12	3.660E+15	8.784E+16
0.85	9.458E+14	5.893E+11	9.464E+14	2.271E+16
1.25	1.795E+14	3.633E+14	5.428E+14	1.303E+16
1.75	5.896E+12	2.605E+04	5.896E+12	1.415E+14
2.25	2.842E+12	1.925E+09	2.844E+12	6.826E+13
Total	9.166E+15	4.023E+14	9.568E+15	2.296E+17

Table 5.2-2 - Assembly Hardware Source Strength Calculation

	Active Fuel Zone	Bottom Nozzle Zone	Gas Plenum Zone	Top Nozzle Zone
Assembly Hardware Zircaloy Mass (kg)	110.4	-	-	-
Assembly Hardware SS-304 Mass (kg)	8.5	12.2	1.0	22.0
Assembly Hardware Inconel-718 Mass (kg)	8.0	2.3	2.0	5.5
Control Component Zircaloy Mass (kg)	17.1	-	-	-
Control Component SS-304 Mass (kg)	1.4	-	-	5.3
Zircaloy Cobalt Concentration	10 ppm			
SS-304 Cobalt Concentration	800 ppm			
Inconel Cobalt Concentration	4694 ppm			
Total Cobalt Quantity (grams/assembly)	46.7	20.6	10.2	47.7
Activation Adjustment Factor (vs. fuel)	1.0	0.2	0.2	0.1
Total Source Strength (γ/sec-assembly)	3.188 E+14	2.949 E+13	1.458 E+13	3.441 E+13

Table 5.2-3 - Assembly Non-Fuel Zone Hardware Gamma Source Strengths

Component / Region	Total Gamma Source Strength (γ/sec-ass)	Total Gamma Source Strength (γ/sec-MSB)
Assembly Hardware Bottom Nozzle	2.949 E13	7.078 E14
Assembly Hardware Gas Plenum	1.458 E13	3.499 E14
Assembly Hardware Top Nozzle	3.132 E13	7.517 E14
Control Component Top Nozzle	3.085 E12	7.404 E13
Total Top Nozzle	3.441 E13	8.258 E14

Table 5.2-4 - Gamma and Neutron Axial Source Strength Profiles

Axial Span (inches)	Relative Burnup / Gamma Source Strength	Relative Neutron Source Strength	Normalized Gamma Source Strength Profile	Normalized Neutron Source Strength Profile
0.0 - 8.2	0.674	0.206	0.663	0.171
8.2 – 16.3	0.967	0.874	0.951	0.726
16.3 - 24.5	1.091	1.417	1.073	1.176
24.5 - 32.7	1.121	1.579	1.103	1.311
32.7 - 40.8	1.126	1.608	1.108	1.335
40.8 - 49.0	1.111	1.524	1.093	1.265
49.0 - 57.2	1.106	1.496	1.088	1.242
57.2 - 65.3	1.101	1.469	1.083	1.220
65.3 - 73.5	1.097	1.448	1.079	1.202
73.5 - 81.7	1.094	1.432	1.076	1.189
81.7 - 89.8	1.095	1.438	1.077	1.194
89.8 - 98.0	1.099	1.459	1.081	1.211
98.0 – 106.2	1.096	1.443	1.078	1.198
106.2 - 114.3	1.087	1.396	1.069	1.159
114.3 - 122.5	1.073	1.326	1.056	1.100
122.5 - 130.7	1.003	1.012	0.987	0.840
130.7 - 138.8	0.832	0.479	0.818	0.398
138.8 - 147.0	0.525	0.076	0.516	0.063
Average	1.017	1.205	1.000	1.000

Table 5.3-1 - VSC-24 Cask Material Elemental Compositions

	Carbon Steel	Concrete	Lead	MSB Lid RX-277	MTC RX-277	Air
Overall Density (g/cm^3)	7.832	2.237	11.34	1.135	1.6223	0.001225
Elemental Weight Fraction						
H	-	0.0054	-	0.0216	0.0314	-
B	-	-	-	0.0167	0.0145	-
C	0.00220	-	-	-	-	-
N	-	-	-	-	-	0.80
O	-	0.4980	-	0.5292	0.5769	0.20
Na	-	0.0170	-	0.0070	0.0061	-
Mg	-	0.0027	-	0.0059	0.0052	-
Al	-	0.0456	-	0.2839	0.2476	-
Si	0.00275	0.3152	-	0.0253	0.0221	-
P	0.00035	-	-	-	-	-
S	0.00035	0.0013	-	0.0023	0.0020	-
K	-	0.0192	-	-	-	-
Ca	-	0.0831	-	0.1048	0.0914	-
Cr	0.00900	-	-	-	-	-
Mn	-	-	-	-	-	-
Fe	0.98535	0.0125	-	0.0032	0.0028	-
Ni	-	-	-	-	-	-
Zr	-	-	-	-	-	-
Pb	-	-	1.0	-	-	-

Table 5.3-2 - MSB Interior Homogenized Material Elemental Densities

	Active Fuel	Bottom Nozzle	Gas Plenum	Top Nozzle
Overall Density (g/cm^3)	4.076	2.208	0.312	3.543
Elemental Weight Fraction				
C	0.000032	0.000735	0.000532	0.000719
N	0.000042	0.001299	0.001299	0.001299
O	0.097642	-	-	-
Al	0.000071	0.000958	0.004013	0.001197
Si	0.000141	0.008710	0.004636	0.008391
P	0.000005	0.000378	0.000148	0.000360
S	0.000010	0.000263	0.000146	0.000254
Ti	0.000092	0.001277	0.005351	0.001596
V	0.000003	-	-	-
Cr	0.004568	0.189755	0.189753	0.189755
Mn	0.000263	0.017101	0.007935	0.016382
Fe	0.010522	0.606395	0.347480	0.586100
Co	0.000064	0.001421	0.003408	0.001577
Ni	0.006871	0.157897	0.377424	0.175104
Cu	0.000014	0.000160	0.000669	0.000200
Zr	0.151100	-	-	-
Nb	0.000620	0.008864	0.037141	0.011080
Mo	0.000335	0.004789	0.020065	0.005986
Sn	0.002469	-	-	-
U	0.725120	-	-	-

Table 5.4-1 - Gamma Flux-to-Dose Conversion Factors (ANSI/ANS-6.1.1-1977)

Gamma Energy (MeV)	Flux-to-Dose Conversion Factor (mrem/hr per γ/cm^2-sec)
0.01	0.00396
0.03	0.000582
0.05	0.000290
0.07	0.000258
0.1	0.000283
0.15	0.000379
0.2	0.000501
0.25	0.000631
0.3	0.000759
0.35	0.000878
0.4	0.000985
0.45	0.00108
0.5	0.00117
0.55	0.00127
0.6	0.00136
0.65	0.00144
0.7	0.00152
0.8	0.00168
1.0	0.00198
1.4	0.00251
1.8	0.00299
2.2	0.00342
2.6	0.00382
2.8	0.00401
3.25	0.00441
3.75	0.00483
4.25	0.00523

Table 5.4-2 - Neutron Flux-to-Dose Conversion Factors (ANSI/ANS-6.1.1-1977)

Neutron Energy (MeV)	Flux-to-Dose Conversion Factor (mrem/hr per n/cm^2-sec)
2.5E-08	0.00367
1.0E-07	0.00367
1.0E-06	0.00446
1.0E-05	0.00454
1.0E-04	0.00418
1.0E-03	0.00376
1.0E-02	0.00356
0.1	0.0217
0.5	0.0926
1.0	0.132
2.5	0.125
5.0	0.156
7.0	0.147
10.0	0.147
14.0	0.208
20.0	0.227

Table 5.5-1 - VSC-24 Minimum Required Assembly Cooling Time vs. Burnup and Enrichment

Burnup	Assembly Average Initial Enrichment (wt% ^{235}U in U)																											
(MWd)	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2
15000	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20000	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25000	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30000	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32000	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
33000	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34000	9	8	8	8	7	7	7	7	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5
35000	9	9	8	8	8	8	7	7	7	7	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5
36000	10	9	9	9	8	8	8	7	7	7	7	7	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5
37000	10	10	9	9	9	8	8	8	8	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5
38000	10	10	10	9	9	9	8	8	8	8	7	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6	6
39000	11	11	10	10	9	9	9	9	8	8	8	8	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6
40000	12	11	11	10	10	10	9	9	9	8	8	8	8	7	7	7	7	7	6	6	6	6	6	6	6	6	6	6
41000	12	12	11	11	10	10	10	9	9	9	9	8	8	8	8	7	7	7	7	7	7	7	7	6	6	6	6	6
42000	13	12	12	11	11	11	10	10	10	9	9	9	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7	6
43000	13	13	12	12	11	11	11	10	10	10	9	9	9	9	8	8	8	8	7	7	7	7	7	7	7	7	7	7
44000	14	13	13	12	12	12	11	11	11	10	10	10	9	9	9	8	8	8	8	8	8	8	7	7	7	7	7	7
45000	15	14	14	13	13	12	12	11	11	11	10	10	10	9	9	9	9	8	8	8	8	8	8	8	8	8	7	7

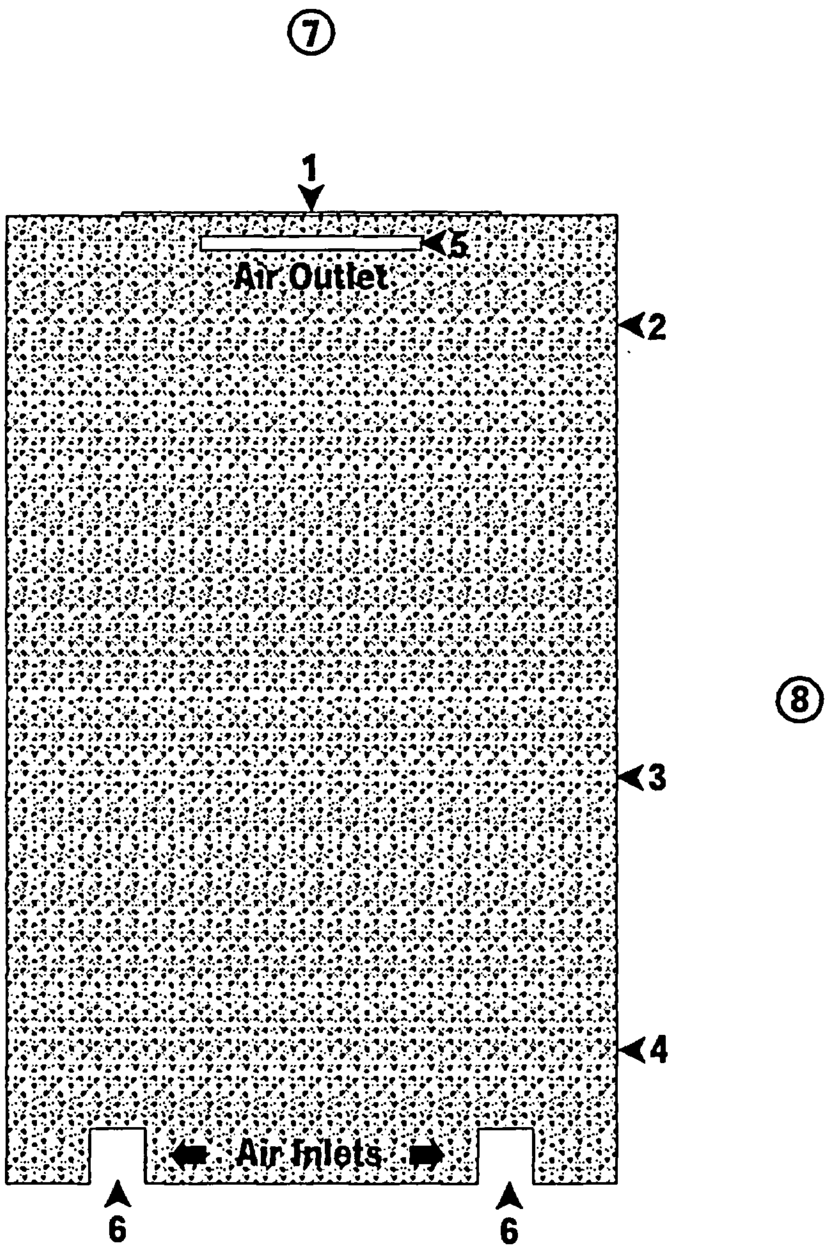


Figure 5.1-1 - Storage Cask Calculated Dose Rate Locations

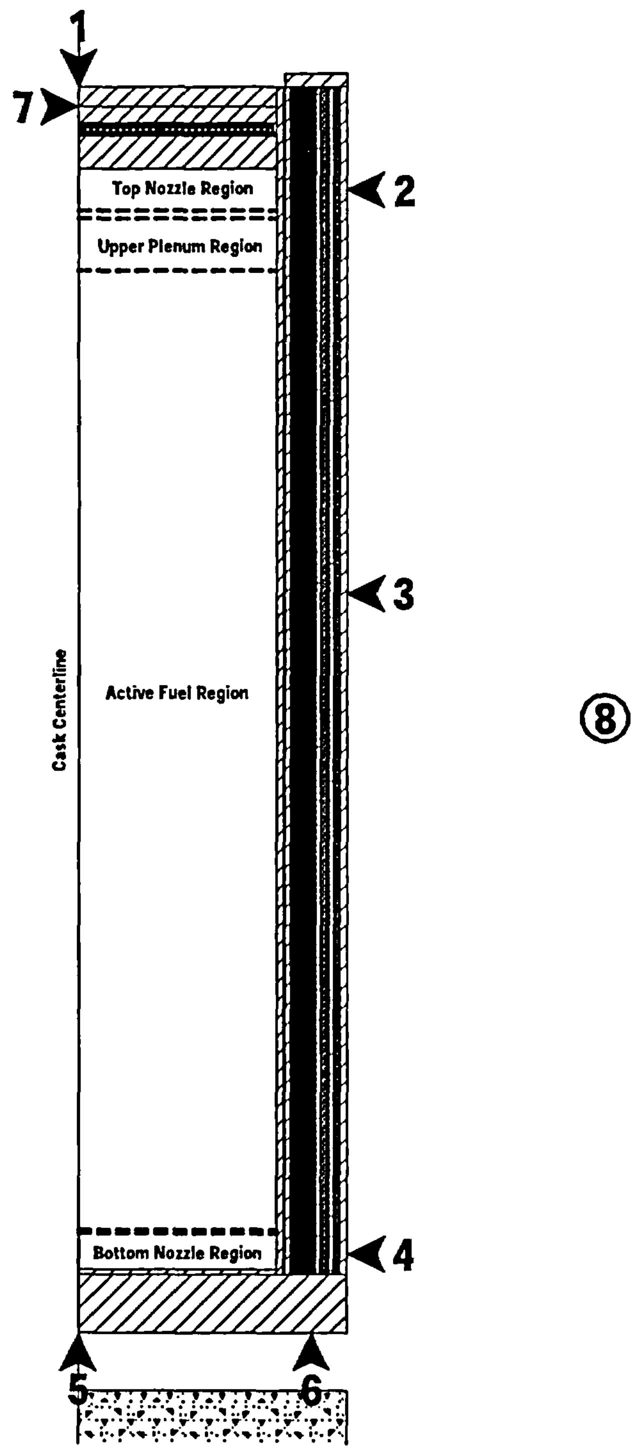


Figure 5.1-2 - Transfer Cask Calculated Dose Rate Locations

Figure Withheld Under 10 CFR 2.390

Figure 5.3-1 - VSC-24 Storage Cask Bulk Shielding Model Geometry

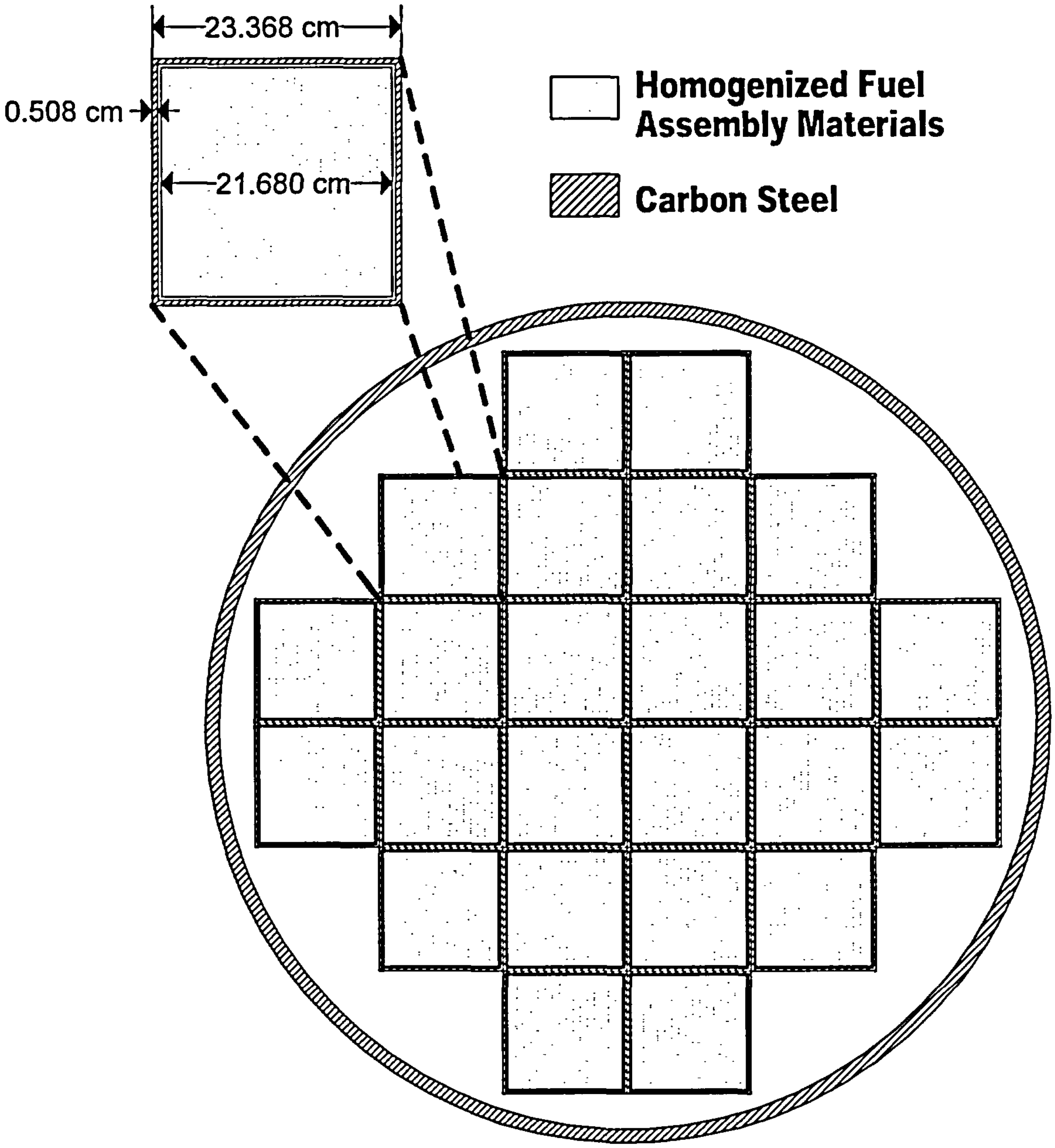


Figure 5.3-2 - MSB Interior Shielding Model Geometry (view of horizontal cross section)

Figure Withheld Under 10 CFR 2.390

Figure 5.3-3 - VSC-24 Storage Cask Outlet Duct Geometry
(view of vertical cross section)

Figure Withheld Under 10 CFR 2.390

Figure 5.3-4 - VSC-24 Storage Cask Inlet Duct Geometry
(view of vertical cross section)

Figure Withheld Under 10 CFR 2.390

Cross Section View through Middle of Section B & C Ducts

Figure 5.3-5 - *VSC-24 Storage Cask Inlet Duct Geometry*
(view of horizontal cross section)

Figure Withheld Under 10 CFR 2.390

Figure 5.3-6 - VSC-24 Transfer Cask Shielding Model Geometry (view of vertical cross section)

Figure Withheld Under 10 CFR 2.390

Figure 5.5-1 - Shielding Model for VCC Side Dose Rate Calculations for Assembly Fuel Zone Cobalt in the Inner 12 MSB Locations

6.0 CRITICALITY EVALUATION

The VSC-24 system complies with 10CFR72 requirements relative to criticality safety. Under all credible normal, off-normal, and accident conditions, the maximum VSC-24 neutron multiplication factor, including all biases and uncertainties, does not exceed 0.95. In addition, at least two unlikely, independent, and concurrent or sequential changes to the conditions essential to VSC-24 criticality safety must occur before any accidental criticality is deemed to be possible under normal, off-normal, and accident conditions.

6.1 CRITICALITY DESIGN CRITERIA AND FEATURES

The VSC-24 system includes the following components:

1. Multi-Assembly Sealed Basket (MSB)
2. MSB Transfer Cask (MTC)
3. Ventilated Concrete Cask (VCC).

The MSB is a sealed cylindrical canister containing a basket structure used to support fuel assemblies. The MSB, situated inside the MTC, is submerged into the cask loading pit of the spent fuel pool and flooded with borated water during fuel loading and unloading operations. After loading, the MSB is drained, vacuum dried, backfilled with He, and sealed with redundant welds. The MSB is then transferred from the MTC to the VCC for storage.

The MTC is a transfer cask used for shielding during MSB loading/unloading, draining, vacuum drying, backfilling, closure, and transfer operations. The MTC radial shielding employs layers of steel, lead, and RX-277 (cement-like neutron absorber) materials. The MTC also employs thick steel doors on the cask bottom to provide shielding and allow for MSB transfer.

The VCC is a cylindrical annulus of concrete with an inner steel liner. The VCC provides shielding of the radiation emanating from the MSB during storage. The VCC also employs steel-lined air ducts to facilitate natural air circulation, which removes decay heat from the MSB exterior surface.

Since the MSB is redundantly sealed after loading, water ingress into the MSB interior is not possible during storage in the VCC. Therefore, the loading and unloading operational phases are the only time when sufficient moderation may be present to introduce criticality potential. For this reason, the design basis configuration for criticality safety evaluations consists of the MSB inside the MTC, submerged in the spent fuel pool and flooded with borated water. A full radial view of the design basis MSB and MTC configuration is presented in Figure 6.1-1. General dimensions of the MSB and MTC components are presented in Table 6.1-1.

The VSC-24 system criticality potential is explicitly evaluated for the fuel loading and unloading operational phases, since these are the only phases in which sufficient moderation may be present to yield a critical configuration. During fuel loading and unloading operations, the criticality control mechanism for the VSC-24 system is soluble boron contained in the water that fills the MSB basket. Strict administrative controls, as specified in Chapter 12, are established to ensure that the appropriate minimum required soluble boron concentration is maintained. The administrative controls on soluble boron concentration ensure that accidental flooding of the MSB with unborated water cannot occur.

The design basis criticality evaluations determine the minimum soluble boron concentrations required to maintain VSC-24 system k_{eff} values below the applicable subcritical limit during the fuel loading and unloading operational phases. The minimum required soluble boron concentrations are determined as a function of fuel assembly class and enrichment.

6.2 FUEL SPECIFICATION

For the VSC-24 system, a fuel assembly class is distinguished by having a unique lattice layout and pin pitch. The lattice layouts for each of the eight fuel assembly classes considered in the VSC-24 criticality design basis calculations are presented in Figure 6.2-1 through Figure 6.2-8. The eight fuel assembly classes for the VSC-24 system are identified as follows:

1. B&W 15x15
2. W 14x14
3. W 15x15
4. W 17x17
5. CE 15x15A
6. CE 15x15B
7. CE 15x15C
8. CE 16x16.

Within an assembly class, the following set of characteristic parameters are considered to have the potential to influence k_{eff}:

- Fuel Density
- Active Fuel Length
- Fuel Pellet Diameter
- Fuel Clad Material
- Fuel Clad Outer Diameter
- Fuel Clad Thickness
- Guide Tube Material

- Guide Tube Outer Diameter
- Guide Tube Thickness
- Instrument Tube Material
- Instrument Tube Outer Diameter
- Instrument Tube Thickness
- Guide Bar Effective Diameter (applicable only to CE 15x15A, CE 15x15B, and CE 15x15C)
- Control Component Rodlet Presence within the Guide Tubes
- Control Component Rodlet Clad Material
- Control Component Rodlet Outer Diameter
- Control Component Rodlet Thickness
- Control Component Rodlet Length
- Control Component Rodlet Fill Material.

For each assembly class, a range of acceptable values are specified for each of the parameters above, as shown in Table 6.2-1(a) and Table 6.2-1(b). Given the specified ranges of parameter values, the minimum required soluble boron concentrations for each assembly class are demonstrated to adequately suppress the VSC-24 system reactivity for the most reactive fuel configurations possible.

The minimum required boron concentrations as a function of enrichment for each assembly class are demonstrated to adequately suppress reactivity over a fresh fuel enrichment range from 0 to 4.2 weight percent ^{235}U in U of UO_2. For an assembly having multiple fuel rod enrichments, the representative enrichment for the assembly should correspond to the maximum enrichment present unless further evaluation is performed. The owner of the VSC-24 system may perform an evaluation to demonstrate that a uniformly distributed assembly average enrichment is more reactive than the corresponding explicit assembly enrichment pattern. Once the reactivity of a uniformly distributed assembly average enrichment is demonstrated to be conservative, this enrichment may be used to determine the minimum required soluble boron concentration.

Because the design basis criticality calculations are conservatively based on a full loading of unirradiated fuel, any fuel burnup up to the maximum specified in Chapter 12 is acceptable with respect to criticality safety. This conservatism ensures subcriticality in the event that unirradiated fuel is mistakenly loaded.

The design basis criticality calculations bound intact PWR fuel that satisfies all of the parameters defining a fuel assembly class as presented in Table 6.2-1(a) and Table 6.2-1(b) and Figure 6.2-1 through Figure 6.2-8, as applicable. Intact fuel consists of assemblies that do not contain fuel rods with known or suspected cladding defects greater than a hairline crack or a pinhole leak. The design basis criticality calculations bound the replacement of damaged fuel rods (i.e., non-intact fuel rods), with dummy rods of equal or lesser diameter, such that the dummy rod displaces an equal or lesser volume of borated water.

6.3 MODEL SPECIFICATION

The VSC-24 design basis criticality calculation models are defined to ensure that the most reactive system configurations are analyzed when determining the minimum required soluble boron concentrations. Manufacturing and fabrication tolerances are incorporated to account for their effects on reactivity. Other deviations from nominal design configurations, such as fuel assemblies shifting within their fuel sleeves and the MSB shifting within the MTC, are also verified to be acceptable with respect to the minimum required soluble boron concentrations.

6.3.1 Configuration

The VSC-24 system criticality models bound all credible configurations that may exist during normal, off-normal, and accident conditions. Since the potential for criticality only exists when sufficient neutron moderation exists, the fuel loading/unloading and MSB draining/reflood operations are considered the most reactive configurations with respect to criticality safety. During fuel loading/unloading operations under normal conditions, the MSB is fully flooded with water containing soluble boron at the minimum required concentration corresponding to the specific fuel assembly class and maximum loaded enrichment. During MSB draining/reflood operations under normal conditions, the MSB passes from a fully flooded state, through a continuous range of partially flooded states, to a non-flooded (empty) state, or vice versa. In demonstrating the adequacy of the minimum required soluble boron concentrations, the design basis criticality models conservatively address both fully flooded and partially flooded states.

There are no postulated off-normal conditions relevant to criticality safety. The only postulated accident condition with potential to affect criticality safety is the MSB drop accident. However, in the analysis of the MSB drop accident, it is determined that the geometry of the MSB internals will withstand deformation to a degree that allows the fuel to remain intact. Thus, the design basis criticality models remain conservative with respect to the postulated MSB drop accident condition.

The MCNP4A criticality calculation models are three-dimensional models of the VSC-24 MSB placed inside the MTC while submerged in the spent fuel pool. Due to the presence of internal moderating material and external reflecting material, these models represent the most reactive configurations that the VSC-24 system will experience during all phases of operation. Most of the MCNP4A criticality calculation models use 1/8th symmetry to represent the design basis configuration as shown in Figure 6.3-1. When 1/8th symmetry is not appropriate for a particular configuration being analyzed, full geometric models are used. The modeled dimensions of the MSB and MTC components are presented in Table 6.1-1.

The MSB and MTC portions of the design basis configuration are truncated at the top and bottom of the active fuel region of each assembly. Effectively infinite fresh water reflectors are placed above and below the active fuel region and radially outward from the outer surface of the MTC. The MSB and MTC are modeled uniformly in the axial direction based on the geometry presented in Figure 6.3-1. Borated water is modeled inside the MSB cavity and between the structure of each fuel assembly.

As presented in Table 6.1-1, component dimensional tolerances are applied in the design basis configuration models to maximize reactivity. Generally, tolerances are applied both to reduce the volume of borated water inside the MSB and to reduce the separation between assemblies. If a component has no effect on either the borated water volume inside the MSB or the separation between assemblies, the corresponding tolerance resulting in either the minimum neutron absorption or most effective neutron reflection is applied. The MTC is modeled using nominal dimensions because of its limited effect on system reactivity.

The most reactive condition exists at optimum moderation or optimum moderator density. Since the optimum moderator density is a function of enrichment and soluble boron concentration, it is necessary to create and analyze several criticality models for each enrichment and soluble boron concentration to determine the optimum moderator density. Thus, for each fuel class, several enrichments are studied, with each of these enrichments assigned a soluble boron concentration. For each enrichment/boron concentration combination assigned to each fuel class, multiple criticality models are developed, each model with a different borated moderator density, to determine the optimum moderator density to within 0.05 g/cm^3.

To verify that the criticality analyses model the most reactive configuration possible, sensitivity studies are performed on the following characteristics of the design basis configuration:

- Fuel sleeve thickness
- Assembly shift direction inside fuel sleeves
- MSB position inside MTC
- Fuel pellet diameter
- Borated moderator level in a partially flooded configuration.

The fuel sleeve thickness study evaluates the effects on k_{eff} of increasing the fuel sleeve thickness. The fuel sleeve thickness dimensions used for these analyses are presented in Table 6.1-1.

Three patterns of assembly shifting inside the fuel sleeves are studied to determine the most reactive configuration. These shifting patterns are presented in Figure 6.3-2.

The MSB position inside the MTC is evaluated to determine if the reflection of neutrons from the lead of the MTC could be enhanced by an off-center relationship between the MSB and MTC. In this study, the following two cases are considered:

1. The MSB centered inside the MTC
2. The MSB shifting to contact the MTC.

The MSB shifting to contact the MTC is modeled using full geometric representation of the configuration. The MSB is shifted so that the MSB outer shell contacts the MTC inner shell. A full radial view of the shifted MSB configuration is presented in Figure 6.3-3.

The fuel pellet diameter sensitivity studies verify that the minimum required boron concentrations are sufficient to ensure that any increase in k_{eff} due to fuel pellet diameter variations within the range studied will not cause the corresponding subcritical limit to be exceeded. The studied fuel pellet diameters, within the allowable range for each fuel assembly class, include the maximum, the minimum, and two equally spaced diameters between the maximum and minimum.

In the partially flooded configuration models, the MSB is partially flooded with borated moderator at the optimum moderator density determined in the corresponding fully flooded case. The remainder of the MSB interior is occupied by ordinary fresh water at a partial density corresponding to saturated steam. For each partial flooding study, the MSB is modeled as being 7/8, 3/4, 5/8, and 1/2 flooded. Preferential or uneven flooding within the MSB due to blockage of flow or drain paths is not considered credible. Furthermore, preferential or uneven flooding concerns that are commonly associated with a flux trap design are not applicable to the MSB. Thus, the modeled uniform flooding levels are sufficient to demonstrate the effects of partial flooding on k_{eff}.

Except when perturbations are required for the sensitivity studies described above, the design basis criticality models employ the following assumptions:

- No credit is taken for the burnup of the fuel (i.e., pure UO_2 fuel at the initial ^{235}U enrichment level is modeled).
- The criticality models assume fresh water ingress to the interior of the fuel clad of every fuel pin. The fuel-to-clad gap of every fuel pin is filled with fresh water at 1 g/cm^3 density.
- Worst-case tolerance dimensions are applied to the nominal dimensions of components that could potentially affect the system reactivity.
- The peripheral steel component volumes inside the MSB shell are maximized for maximum displacement of borated water.
- Fuel sleeve outer widths are minimized to reduce spacing between assemblies. The closest assembly spacing yields the most reactive configuration.
- The fuel density is maximized at 96% of the theoretical UO_2 density.
- The fuel clad outer diameter is maximized to maximize displacement of borated water.
- The fuel clad thickness is minimized to maximize the volume of fresh water moderator present in the fuel-to-clad gaps.
- The guide tube outer diameter is maximized to maximize displacement of borated water.
- The guide tube thickness is maximized to maximize displacement of borated water.

- The instrument tube outer diameter is maximized to maximize displacement of borated water.
- The instrument tube thickness is maximized to maximize displacement of borated water.
- The active fuel length is maximized to maximize the fuel mass within the system and reduce the effects of axial leakage.
- The gap between the MSB and MTC is filled with fresh, full-density water to enhance neutron reflection back into the system.
- Fresh, full-density, infinite water reflectors are placed above and below the active fuel region and radially outward from the outer surface of the MTC. The presence of fresh water in these regions provides greater neutron reflection back into the system than that provided by non-hydrogen-bearing materials actually present in these regions.
- For assemblies that may contain control inserts (i.e., rods that fill the assembly guide tubes), it is assumed that the control inserts are present. All guide tubes are filled with control insert rods that are zircaloy clad and are filled with B_4C material, where all boron in the B_4C is ^{11}B (i.e., no ^{10}B is present).

6.3.2 Material Properties

The MSB and MTC components in the design basis criticality models are composed of carbon steel, lead, and RX-277 materials. The fuel assembly components in the models are composed of zircaloy and UO_2. The control component rodlets in the models are composed of zircaloy and $^{11}B_4C$. Modeling zircaloy as the control component rodlet cladding bounds stainless steel cladding because stainless steel has a higher neutron absorption cross section than zircaloy. The modeled compositions of steel, lead, RX-277, zircaloy, and $^{11}B_4C$ materials are presented in Table 6.3-1.

The interior volume of the MSB is modeled as containing either borated water or a combination of borated water and steam (i.e., fresh water at partial density). The regions above and below the active fuel, between the MSB and MTC, and radially extending outward from the MTC are modeled as fresh water at full density. Equations 6-1 through 6-4 are used to calculate the borated and unborated water compositions. The boron concentration in parts per million by mass varies from 2325 to 5900, depending on the fuel class, enrichment, and optimum moderator density. To determine the optimum moderator density for a particular fuel class, enrichment, and boron concentration, the density of the borated water in the criticality models is varied within a range from 0.25 g/cm^3 to 0.75 g/cm^3, in increments of 0.05 g/cm^3.

The saturated steam in the partially flooded models is represented by ordinary water at a density of 0.0006 g/cm^3.

The modeled UO_2 fuel compositions, presented in Table 6.3-2, are based on an initial enrichment of either 3.0, 3.5, 4.0, 4.5, or 5.0 weight percent ^{235}U in U. For intermediate fuel enrichments, linear-linear interpolation based on the results for the five explicitly evaluated enrichments is used to determine the minimum required soluble boron concentration. All UO_2 fuel is modeled as having 96% of the theoretical density, which is equivalent to 10.5216 g/cm^3.

Equation 6-1 - Hydrogen Weight Percent in Borated or Unborated Water

$$\mathrm{H\,wt\%} = \left[\frac{(2)(1.00794)}{(2)(1.00794)+15.9994}\right]\left[\frac{(1.0E6-\mathrm{ppmb})}{1.0E6}\right][100]$$

where ppmb is boron concentration in parts per million by mass.

Equation 6-2 - Oxygen Weight Percent in Borated or Unborated Water

$$\mathrm{O\,wt\%} = [100-\mathrm{H\,wt\%}]\left[\frac{(1.0E6-\mathrm{ppmb})}{1.0E6}\right]$$

where ppmb is boron concentration in parts per million by mass.

Equation 6-3 - ^{10}B Weight Percent in Borated Water

$$^{10}\mathrm{B\,wt\%} = \left[\frac{(10.0129371)(0.199)}{(10.0129371)(0.199)+(11.0093055)(0.801)}\right]\left[\frac{\mathrm{ppmb}}{1.0E6}\right][100]$$

where ppmb is boron concentration in parts per million by mass.

Equation 6-4 - ^{11}B Weight Percent in Borated Water

$$^{11}\mathrm{B\,wt\%} = \left[\frac{(11.0093055)(0.801)}{(10.0129371)(0.199)+(11.0093055)(0.801)}\right]\left[\frac{\mathrm{ppmb}}{1.0E6}\right][100]$$

where ppmb is boron concentration in parts per million by mass.

6.4 CRITICALITY ANALYSIS

6.4.1 Computer Programs

MCNP4A is a general purpose Monte Carlo code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport (Reference 6.10). It is suitable for criticality analysis because it has the capability to calculate k_{eff} values for systems. MCNP4A treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first and second-degree surfaces. To calculate the effective multiplication factor, MCNP4A uses three separate estimators: collision, absorption, and track length. The three estimators are statistically combined to provide the best estimate confidence interval for k_{eff}. The primary sources of nuclear data for MCNP4A are evaluations from the Evaluated Nuclear Data File (ENDF) system, the Evaluated Nuclear Data Library (ENDL), the Activation Library (ACTL) compilations from Lawrence Livermore National Laboratory, and evaluations from the Applied Nuclear Science (T-2) Group at Los Alamos National Laboratory. The information from these various sources are incorporated into the continuous-energy nuclear and atomic data tables that MCNP4A uses in a calculation.

6.4.2 Multiplication Factor

The calculation methodology is defined to demonstrate that the minimum required boron concentrations, as a function of initial enrichment for a given fuel assembly class, are sufficient to maintain the VSC-24 system k_{eff} below the smallest or limiting upper subcritical limit (USL) value under the most reactive conditions expected during all phases of operation. The calculation methodology generally consists of three parts:

1. Base Case Identification
2. Sensitivity Analyses
3. Partial Flooding Analyses.

These three parts are described in Sections 6.4.2.1, 6.4.2.2, and 6.4.2.3, respectively. For each part, many criticality calculation cases are performed to support the evaluation. Each of the criticality case k_{eff} results are compared to USL values corresponding to the specific case. It is always required that the value of k_{eff}, plus two times the corresponding standard deviation, be less than or equal to the limiting USL value for a given case. The limiting USL function and its application are described in Section 6.4.3.

The criticality calculations are performed using the MCNP4A code system described in Section 6.4.1. The MCNP4A code system reports several estimates of k_{eff} based on collision, absorption, and track length methods. These three estimates are statistically combined to yield a combined collision/absorption/track length estimate of k_{eff} that is considered to be the best estimate result.

Each k_{eff} and standard deviation result in the design basis calculations corresponds to the mean of the combined collision/absorption/track length k_{eff} estimate. All criticality calculations are performed such that the standard deviation about the mean k_{eff} result is less than 0.00090. Generally, the number of neutrons tracked per cycle is 1500, the number of inactive cycles is 50, and the number of active cycles is 500. The inactive cycles are used to allow spatial convergence of the neutron source distribution and do not contribute to the k_{eff} results. The number of neutrons tracked per cycle is sufficient to ensure adequate sampling of all fissionable regions capable of contributing to the system k_{eff} value. The initial source distribution for the criticality calculations is approximately 60 uniformly distributed neutron source points per fuel assembly modeled.

The VSC-24 system criticality calculations use the same cross section data tables that are used in the benchmark calculations from which the USL functions are developed. All MCNP4A calculations are performed using one-eighth geometric models with reflective boundary conditions, except for the sensitivity calculations analyzing the MSB shift inside the MTC (see Section 6.4.2.2), which use full geometric models.

6.4.2.1 Part 1 of Methodology: Base Case Identification

In Part 1 of the methodology, several enrichment points are studied for each fuel class. At each enrichment point, a soluble boron concentration that will keep the system k_{eff} under the limiting USL value under the most reactive conditions is determined. The most reactive condition exists at optimum moderation or optimum moderator density. Since the optimum moderator density is a function of enrichment and soluble boron concentration, it is necessary to run several criticality cases to determine the optimum moderator density. Thus, for each fuel class, several enrichments are studied, each enrichment having been assigned a soluble boron concentration.

For each enrichment/boron concentration combination of each fuel class, five criticality calculations are performed, each with a different borated moderator density such that the optimum moderator density can be determined to within 0.05 g/cm^3. The case demonstrating the smallest margin to its limiting USL value in each set of five moderator density perturbation cases is considered the "base case" for the corresponding fuel class, enrichment point, and soluble boron concentration.

All of the criticality cases performed in Part 1 of the methodology employ an assumed set of most reactive system characteristics. The system characteristic assumptions for Part 1 are summarized as follows:

- Peripheral steel component volumes inside the MSB shell are maximized for maximum displacement of borated water.
- Fuel sleeve outer widths are minimized for the closest positioning of fuel.
- Fuel sleeve thickness is minimized for the closest positioning of fuel. (This assumption is evaluated in Part 2 of the methodology.)
- All assemblies are shifted within their respective fuel sleeves toward the center of the MSB (see Figure 6.3-2). (This assumption is evaluated in Part 2 of the methodology.)
- The fuel density is maximized at 96% of the theoretical UO_2 density.
- A minimum fuel pellet diameter is modeled. (This assumption is evaluated in Part 2 of the methodology.)
- The fuel clad outer diameter is maximized to maximize displacement of borated water.

- The fuel clad thickness is minimized to maximize the volume of fresh water moderator present in the fuel-to-clad gaps.
- The guide tube outer diameter is maximized to maximize displacement of borated water.
- The guide tube thickness is maximized to maximize displacement of borated water.
- The instrument tube outer diameter is maximized to maximize displacement of borated water.
- The instrument tube thickness is maximized to maximize displacement of borated water.
- The active fuel length is maximized to maximize the fuel mass within the system and to reduce the effects of axial leakage.
- The MSB is center positioned within the MTC. (This assumption is evaluated in Part 2 of the methodology.)
- The gap between the MSB and MTC is filled with fresh, full-density water.
- Fresh, full-density, infinite water reflectors are placed above and below the active fuel region and are placed radially outward from the outer surface of the MTC.
- For assemblies that may contain control inserts (i.e., rods that fill the assembly guide tubes), it is assumed that the control inserts are present. All guide tubes are filled with control insert rods that are zircaloy clad and are filled with B_4C material, where all boron in the B_4C is ^{11}B (i.e., no ^{10}B is present). As this displaces water with a material that moderates but does not absorb neutrons, it will clearly increase reactivity. Thus, the results of the criticality analyses, which are calculated assuming the presence of control inserts, are bounding (conservative) for assemblies without inserted control inserts, or assemblies with control inserts that do not extend into the assembly fuel zone (e.g., thimble plug assemblies).

The values corresponding to these parameters are presented in Sections 6.1 and 6.2. Parameters for which the most reactive allowable value is not obvious are further evaluated in Part 2 of the methodology. This evaluation ensures that the soluble boron concentration values are adequate to satisfy the limiting USL, despite any change in k_{eff} that may occur as a result of perturbations from the assumed more reactive parameter values.

6.4.2.2 Part 2 of Methodology: Sensitivity Analysis

As discussed in Section 6.4.2.1, it is not clear which value is most reactive for some of the parameters modeled in the criticality calculations. Sensitivity analyses are performed in Part 2 of the methodology to study the effects of variations in these parameters.

For each base case (i.e., each boron concentration/enrichment combination at optimum moderator density for each assembly type), seven sensitivity cases are performed to study four parameter assumptions made in the Part 1 calculations. These four parameter assumptions consist of the following:

1. Assembly positions within their respective fuel sleeves
2. Fuel sleeve thickness
3. MSB position inside MTC
4. Fuel pellet diameter.

The base case analyses model the assemblies shifted (within their fuel sleeves) towards the center of the MSB (as shown in the top half of Figure 6.3-2). Two sensitivity cases are performed to study the effects of shifting the assemblies to different positions within their respective fuel sleeves. The first case models each assembly centered in its fuel sleeve. The second case models the assemblies shifted into five groups of four, with the assemblies shifted toward one another in each group (see the lower half of Figure 6.3-2).

One sensitivity case is performed to evaluate the effect of increasing the fuel sleeve thickness from the minimum allowable value of 0.1775 inches (the value modeled in the base case analyses) to a maximum allowable value of 0.23 inches. As a result of the increased fuel sleeve thickness, more borated moderator within each fuel sleeve is displaced, and the center four assemblies corresponding to the most reactive region of the system are moved apart with additional steel (a mild neutron absorber) placed between them.

One sensitivity case is performed to evaluate the effect of positioning the MSB off-center within the MTC. The MSB is shifted from the center to contact the MTC inner surface, as shown in Figure 6.3-3. This effect is evaluated because of the presence of lead-reflecting material in the MTC shell region.

The base case analyses model the minimum allowable fuel pellet diameter (shown in Table 6.2-1) for each analyzed fuel assembly class. Three sensitivity cases are performed to evaluate the effect of increasing the fuel pellet diameter. The three cases consider a maximum fuel pellet diameter (shown for each assembly class in Table 6.2-1) and two additional pellet diameters equally spaced between the minimum and the maximum values.

The sensitivity cases are not intended to show that the assumptions made in Part 1 are always bounding. Rather, the sensitivity cases demonstrate that any perturbations in the non-obvious parameter value assumptions made in Part 1 will not cause the limiting USL value to be exceeded given the soluble boron concentration analyzed. Thus, Part 2 of the methodology demonstrates that the soluble boron concentrations analyzed for each enrichment point of each fuel class are adequate for maintaining subcriticality under fully flooded conditions. Part 3 of the methodology addresses partially flooded conditions.

6.4.2.3 Part 3 of Methodology: Partial Flooding Analysis

Part 3 of the methodology studies the effects of partial flooding conditions within the MSB that may occur during fuel loading or unloading operations. In the partial flooding configurations, the MSB is assumed to be partially flooded with borated moderator at the optimum moderator density determined in the corresponding fully flooded case. The remainder of the MSB interior is modeled as containing fresh water at a partial density corresponding to saturated steam.

Two cases from each fuel assembly class are subjected to the partial flooding analyses. The first case corresponds to that having the smallest margin to its limiting USL value, from both the base cases in Part 1 and the sensitivity cases in Part 2. The second case corresponds to the largest enrichment point case (i.e., the highest boron concentration case), having the smallest margin to its limiting USL value. For each of the two cases from each fuel assembly class, four partially flooded calculations are performed: 7/8 flooded, 3/4 flooded, 5/8 flooded, and 1/2 flooded.

For every assembly class case that was analyzed, the partial flooding analyses demonstrate a monotonic decrease in k_{eff} as the borated water level decreases. Thus, the required soluble boron concentrations determined for each enrichment point of each fuel class (which were calculated assuming fully flooded conditions) are adequate for maintaining subcriticality under partially flooded conditions.

6.4.2.4 Results

The criticality design basis calculations determine the minimum soluble boron concentrations required to maintain VSC-24 system k_{eff} values below the applicable subcritical limits for all acceptable fuel loadings. The minimum required soluble boron concentrations are determined as a function of initial enrichment for each analyzed fuel assembly class. The minimum required soluble boron concentrations are demonstrated to be adequate for sufficiently suppressing reactivity under the most reactive VSC-24 system configurations.

After accounting for uncertainty, all multiplication factor results from Parts 1, 2, and 3 of the methodology are less than or equal to their applicable limiting USL value. The largest of the applicable limiting USL values is 0.93824. Therefore, the margin between the calculated results and the acceptance criterion of k_{eff} less than or equal to 0.95, is at least 0.01176 for all criticality calculations performed.

The results of the primary criticality evaluations (described in Section 6.4.2.1) are presented in Table 6.4-1 through Table 6.4-8. Each table presents the criticality evaluation results for each assembly class defined in Table 6.2-1(a) and Table 6.2-1(b). Within each table, k_{eff} results are presented for each of the analyzed assembly initial enrichment levels (and the minimum boron concentrations that correspond to each enrichment level). Also, for each enrichment level, results are presented for several moderator density levels. For each case, a sufficient number of moderator densities are presented to demonstrate a most reactive moderator density (where the highest k_{eff} value occurs), and to demonstrate declining k_{eff} values for moderator densities above or below that optimum value.

The initial enrichment and boron concentration for each analyzed case are presented in the first two columns of Table 6.4-1 through Table 6.4-8. Then, the k_{eff} value calculated by MCNP is presented, along with the statistical (one sigma) error level (which is also output by MCNP). The table notes give the applicable minimum USL value that applies for the assembly class in question. The notes then also state that the criticality margin is defined as the minimum USL value minus the calculated k_{eff} value, plus two times the statistical error level. The resulting criticality margins (which are positive for all cases) are presented in the far right columns of the tables.

The results presented in Table 6.4-1 through Table 6.4-8 determine the optimum moderator density for every analyzed combination of assembly class and initial enrichment level. The results also demonstrate compliance with all criticality limits over the entire range of moderator densities for all analyzed combinations of assembly class and initial enrichment level. Finally, the results presented in Table 6.4-1 through Table 6.4-8 establish the minimum required boron concentration (in the pool water during loading) that applies for each assembly class and assembly initial enrichment level.

Figure 6.4-1 through Figure 6.4-8 present the minimum required soluble boron concentrations for the water filling the MSB cavity during loading operations. The boron concentrations are presented in parts per million by mass (ppmb) as a function of fuel assembly initial enrichment. The boron concentration data shown in Figure 6.4-1 through Figure 6.4-8 is taken from the second columns of Table 6.4-1 through Table 6.4-8. Although some of the analyzed initial enrichments yield minimum required soluble boron concentrations less than 2850 ppmb (as shown in Table 6.4-1 through Table 6.4-8), an absolute minimum soluble boron concentration of 2850 ppmb is established. Figure 6.4-1 through Figure 6.4-8 reflect this applied minimum concentration. Also, although the criticality evaluations considered enrichment levels as high as 5.0% (as shown in Table 6.4-1 through Table 6.4-8), Figure 6.4-1 through Figure 6.4-8 do not show the required boron concentrations for enrichment levels over 4.2%. This is because the *technical specifications* provided in Section 12.4 of this FSAR currently limit assembly enrichment levels to 4.2%.

As discussed in Section 6.4.2.1, these minimum required soluble boron concentrations were calculated assuming the presence of control components, for the assembly classes that may contain such components (B&W 15x15, W 14x14, W 17x17, CE 15x15B, and CE 15x15C). As the presence of control component rods increases assembly reactivity, required boron concentrations calculated based on the presence of those rods are bounding (i.e., conservatively high) for assemblies without inserted control components. Thus, the required boron concentrations shown for each assembly class in Figure 6.4-1 through Figure 6.4-8 are applicable for assemblies with and without inserted control components.

For assemblies having multiple enrichments either radially or axially, the representative enrichment for the assembly should correspond to the maximum enrichment present unless further evaluation is performed. The owner of the VSC-24 system may perform an evaluation to demonstrate that a uniformly distributed assembly average enrichment is more reactive than the corresponding explicit assembly enrichment pattern. Once the reactivity of a uniformly distributed assembly average enrichment is demonstrated to be conservative, this enrichment may be used to determine the minimum required soluble boron concentration.

The results of the criticality sensitivity analyses discussed in Section 6.4.2.2 are summarized in Table 6.4-9. For each of the seven sensitivity analyses (described above) that were performed, Table 6.4-9 presents the minimum and maximum Δk_{eff} values that occurred for any of the analyzed combinations of assembly type and enrichment level. Δk_{eff} is defined as the calculated

k_{eff} value for the sensitivity (or altered parameter) case, minus the calculated k_{eff} value for the corresponding base case. Δk_{eff} is a measure of the effect of the parameter change on system reactivity, for the specific case in question. The minimum and maximum values of Δk_{eff}, shown in Table 6.4-9, illustrate the range of the reactivity effects, seen for all of the analyzed assembly type and enrichment cases, for each of the seven parameter variations. This illustrates the range over which k_{eff} could vary over all the analyzed cases due to variations of each of the analyzed system parameters, within their allowable ranges. Table 6.4-9 also presents the mean value of Δk_{eff}, averaged over all the analyzed assembly type and enrichment combination cases, for each of the seven sensitivity analyses (i.e., for each of the seven system parameter variations that were analyzed). The mean value of Δk_{eff} is a measure of the effect of the parameter change on system reactivity, averaged over all of the assembly type and enrichment cases.

The results in Table 6.4-9 show that the positions of the assemblies within their sleeves, as modeled in the base case analyses (i.e., all assemblies pushed towards the MSB center), are conservative for all analyzed assembly types and enrichment levels. For every analyzed case, centering the assemblies or moving them into the configuration shown at the bottom of Figure 6.3-2 causes k_{eff} to decrease. The reactivity reduction is strongest for smaller (lower width) PWR assemblies. This is due to the fact that they have more room to shift around within the fuel sleeve, and that the changes in assembly position involved with different shift patterns are therefore greater. The minimum (i.e., most negative) Δk_{eff} values shown in Table 6.4-9 occur for the narrow W 14x14 and CE 16x16 assemblies (for the centered and 4-cluster assembly position cases, respectively). For both assembly position sensitivity cases, the maximum (i.e., least negative) Δk_{eff} value occurs for the widest PWR fuel assembly, the B&W 15x15 assembly. In summary, the sensitivity analyses show that the assembly position pattern modeled in the base case analyses is always conservative, more so for smaller assembly types.

The results in Table 6.4-9 also show that modeling the minimum fuel sleeve wall thickness, as is done in the base case analyses, is always conservative. As shown in Table 6.4-9, Δk_{eff} is always negative, ranging from –0.00334 to –0.01464. There is no clear correlation between Δk_{eff} and assembly type, enrichment, or boron concentration. The main result, however, is that the sensitivity analyses clearly show that increasing the fuel sleeve wall thickness is conservative for all assembly types and enrichment levels.

The results in Table 6.4-9 show a relatively narrow range of Δk_{eff} values for the MSB shift (within the MTC cavity) sensitivity analysis (from 0.00256 to –0.00370). Although positive Δk_{eff} values occur for some assembly type / enrichment combinations, the majority of calculated Δk_{eff} values are negative. As shown in Table 6.4-9, the mean value of Δk_{eff} is negative, at –0.00054. The magnitude of the mean Δk_{eff} value is small, however; less than the one sigma statistical error level in the individual calculated k_{eff} results (of ~0.0008). There is also no clear correlation between Δk_{eff} and assembly type or enrichment level.

These results suggest that shifting the MSB to one side of the MTC cavity causes reactivity to decrease, but only by a very small amount, smaller than the statistical error level in the calculated k_{eff} results. Since the negative reactivity effect is smaller than the statistical error level in individual k_{eff} results, some positive Δk_{eff} values are expected. The observed range of Δk_{eff} values can be adequately explained solely by the statistical error level (of ~0.0008) in the individual calculated k_{eff} results. This corresponds to a (one sigma) statistical error level of greater than 0.0011 for the resulting Δk_{eff} values. The range of observed Δk_{eff} values, which extends to ~3 sigma values above and below the mean, is very possible for the sample size in question, which comprises 38 cases.

Thus, the results suggest that centering the MSB in the MTC cavity, as is done in the base case analyses, is conservative, albeit slightly. Due to statistical error, and due to the small magnitude of the reactivity effect in question, increases in k_{eff} may be observed in some specific cases (i.e., for some assembly types and enrichment levels). However, the sensitivity analyses show that for all analyzed combinations of assembly type, enrichment, and boron concentration, the calculated k_{eff} values for an MSB shifted within the MTC cavity are always under their limiting USL value. It is therefore concluded that the VSC system will remain subcritical in all cases, for all possible MSB positions within the MTC cavity.

Finally, Table 6.4-9 shows the range of reactivity changes that occur as the fuel pellet diameter is increased from the minimum allowable value (shown for each assembly type in Table 6.2-1) assumed in the base case analyses. Three sensitivity cases are studied, which model the pellet diameter increasing 1/3 of the way, 2/3 of the way, and all of the way to the maximum allowable value (from the minimum value). For each pellet diameter case, a sensitivity analysis case is performed for every analyzed combination of assembly type and initial enrichment.

The Table 6.4-9 Δk_{eff} results show that some positive Δk_{eff} values occur for a few assembly type / enrichment cases, for all three analyzed alternative pellet diameters. However, most Δk_{eff} values are negative, with positive values occurring for only a small number of cases. The mean Δk_{eff} values shown in Table 6.4-9 are all negative, and show a steady decrease as the pellet diameter is increased from its minimum value. The range over which the Δk_{eff} values vary increases with increasing pellet diameter, suggesting differences in how different assembly type / enrichment combinations respond to increasing pellet diameter. There is no clear correlation between Δk_{eff} and either assembly type, enrichment, or boron concentration. For all assembly types and enrichment levels, at least 2/3 of the values are negative for all cases.

It is true, however, that specific assembly type / enrichment combinations that have a positive Δk_{eff} value for one fuel pellet diameter case are likely to have positive Δk_{eff} values for the other pellet diameter cases. The two specific assembly type and enrichment combinations that yield the maximum Δk_{eff} values for the "+ 2/3" and the maximum pellet diameter cases (i.e., the 3.0%-enriched B&W 15x15 assembly and the 3.5%-enriched CE 15x15 B assembly, respectively) show positive Δk_{eff} values for all three pellet diameter cases. The 3.5%-enriched CE 15x15 C assembly, which yields the maximum Δk_{eff} value for the "+1/3" pellet diameter case, also shows a positive Δk_{eff} value for the "+ 2/3" pellet diameter case, but shows a slightly negative Δk_{eff} value for the maximum pellet diameter case.

The sensitivity analyses demonstrate, however, that even for the few specific assembly type and enrichment combinations that show an increase in k_{eff} with increasing fuel pellet diameter, the calculated k_{eff} value remains below its minimum USL value for all analyzed pellet diameters. The calculated k_{eff} values remain under their limits for all pellet diameters, for all analyzed combinations of assembly type and enrichment. Thus, the sensitivity analyses clearly show that all criticality requirements are met over the full range of allowable fuel pellet diameters, for all analyzed cases.

As discussed in Section 6.4.2.3, the partial flooding criticality analyses show, for all combinations of assembly type, enrichment, and corresponding boron concentration, that k_{eff} steadily decreases as the borated water level decreases. Thus, maximum reactivity occurs for the fully flooded canister (at the optimum moderator density). Therefore, the partial flooding criticality analyses conclusively show that the base case criticality analyses described in Section 6.4.2.1 (and the corresponding analysis results presented in Table 6.4-1 through Table 6.4-8) are bounding for all partially flooded canister conditions, for all assembly types and enrichment levels.

6.4.2.5 Compliance with Requirements

10CFR72 requires that the package be shown to remain subcritical under all credible conditions that may occur during storage. The design basis criticality calculations model a canister and cask configuration that is more reactive than any configuration that could occur during storage operations. For each criticality calculation, the calculated mean k_{eff} value, plus two times the corresponding standard deviation, remains under the lowest or most limiting of the calculated USL values, as required by 10CFR72. Thus, the analyses show that the most reactive VSC-24 system configuration that may occur during storage operations, loaded with the most reactive fuel configuration at each of the analyzed initial enrichment levels for each fuel assembly class, remains subcritical at the determined minimum required soluble boron concentration levels. This verifies compliance with 10CFR72 criticality requirements.

6.4.2.6 Range of Validity

Each fuel assembly must satisfy the parameter requirements of at least one fuel assembly class, as presented in Section 6.2. The minimum required soluble boron concentrations apply to the VSC-24 system loaded with as many as 24 fuel assemblies. The minimum required soluble boron concentrations apply to the water contained within the MSB interior during fuel loading and unloading operations. The fuel loading and unloading operations represent the only operational phases of the VSC-24 system in which there is sufficient moderator present to introduce criticality potential. Therefore, all conditions of storage, including normal, off-normal, and accident conditions, are addressed by the design basis criticality calculations as long as the minimum required boron concentrations are implemented during loading and unloading operations.

6.4.2.7 Summary of Conservatism

The following is a summary of conservative modeling assumptions employed in the VSC-24 design basis criticality calculations:

1. No credit is taken for the burnup of the fuel (i.e., pure UO_2 fuel at the initial U-235 enrichment level is modeled in the analyses).

2. The criticality analyses assume fresh water ingress to the interior of the fuel clad of every fuel pin. The fuel-to-clad gap of every fuel pin is filled with fresh water at 1 g/cm^3 density.
3. Worst-case tolerance dimensions are applied to the nominal dimensions of components that could potentially affect the system reactivity.
4. The peripheral steel component volumes inside the MSB shell are maximized for maximum displacement of borated water.
5. Fuel sleeve outer widths are minimized for the closest positioning of fuel.
6. Fuel sleeve thickness is minimized for the closest positioning of fuel.
7. All assemblies are shifted within their fuel sleeves in the most reactive manner.
8. The fuel density is assumed to be maximized at 96% of the theoretical UO_2 density.
9. The fuel clad outer diameter is maximized to maximize displacement of borated water.
10. The fuel clad thickness is minimized to maximize the volume of fresh water moderator present in the fuel-to-clad gaps.
11. The guide tube outer diameter is maximized to maximize displacement of borated water.
12. The guide tube thickness is maximized to maximize displacement of borated water.
13. The instrument tube outer diameter is maximized to maximize displacement of borated water.
14. The instrument tube thickness is maximized to maximize displacement of borated water.
15. The active fuel length is maximized to maximize the fuel mass within the system.
16. The gap between the MSB and MTC is filled with fresh, full-density water.
17. Fresh, full-density, infinite water reflectors are placed above and below the active fuel region and are placed radially outward from the outer surface of the MTC.
18. Hypothetical zircaloy-clad control component rods containing pure $^{11}B_4C$ (a moderator material with no neutron absorbing material) are assumed to be present within every assembly guide tube.

6.4.2.8 Limitations or Special Requirements

- Each fuel assembly loaded into the VSC-24 system must satisfy the parameter requirements of at least one fuel assembly class as presented in Section 6.2.
- All fuel assemblies loaded into a single VSC-24 system must belong to the same fuel assembly class.

- The minimum required soluble boron concentrations must be verified by at least two independent measurements prior to fuel loading to enable reliance on soluble boron for reactivity suppression. The required boron concentrations shown for each assembly class in Figure 6.4-1 through Figure 6.4-8 apply for assemblies with and without inserted control components.

6.4.3 Benchmark Comparisons

The criticality calculation method is verified by comparison with critical experiment data that is sufficiently diverse to establish that the method bias and uncertainty will apply to conditions considered in the criticality analysis of the VSC-24 system. A set of 38 critical experiments is analyzed using MCNP4A to demonstrate its applicability to criticality analysis and to establish a set of upper subcritical limits that define acceptance criteria. Benchmark experiments are selected with compositions, configurations, and nuclear characteristics that are comparable to those encountered in the VSC-24 system loaded with fuel, as described in Section 6.2. In general, the benchmark comparisons justify the following:

1. Validity of the computer code MCNP4A
2. Use of the code for the VSC-24 hardware configuration
3. Neutron cross sections used in the analyses
4. Consistency in configuration modeling.

The set of 38 benchmark cases are selected from a series of critical experiments (References 6.2 through 6.9) that were originally performed to provide a database for benchmarking computational methods and data used for criticality analysis. The set of selected benchmarks contain various fuel pitches, fuel enrichments, boron concentrations in the moderating regions, and reflecting materials. While no single benchmark case will precisely match the fissile material, moderation, neutron poisoning, or configuration in the actual cask, the selected benchmarks are relevant to the VSC-24 system design with respect to fuel features and parameters important to reactivity.

Each critical experiment reference provides detailed information and data pertaining to the performance and simulation of the critical experiments, including geometry, material, and critical state conditions. This information and data is used to build the MCNP4A computer models. A general description of the characteristic parameters for each critical benchmark is presented in Table 6.4-9.

In addition to verifying the general adequacy of the MCNP-4A code for criticality calculations, the set of 38 critical benchmark experiments provides code bias data that is used to determine maximum allowable values of k_{eff}, as calculated by MCNP. Examination of the distribution of MCNP-calculated k_{eff} values for the 38 critical experiments (where the actual value of k_{eff} is, by definition, 1.0) allows a bias (in calculated k_{eff} value) for the MCNP code to be determined. If the code bias is negative (i.e., if MCNP generally under-predicts k_{eff}), the bias must be added to all calculated k_{eff} results (or subtracted from allowable calculated k_{eff} values).

6.4.3.1 Methodology of Maximum Allowable K_{eff} Calculation (USL method)

The NRC-accepted methodology for criticality code benchmarking and determining maximum allowable calculated k_{eff} values is outlined in NUREG/CR-6361 (Reference 6.1). The NUREG/CR-6361 methodology involves more than simply calculating a single criticality code bias value based on the average difference between calculated and measured k_{eff} values (for a set of critical experiments). Any correlation, or dependence, of code bias on any of the physical parameters of the analyzed system must also be accounted for. This results in the expression of code bias as a function of various system physical parameters, as opposed to a single bias value. The NUREG/CR-6361 methodology results in the determination "upper sub-critical limit," or USL value. This is a maximum allowable calculated k_{eff} value that accounts for all criticality code bias effects, the NRC administrative margin of 0.05, and any correlations or dependencies of code bias on any of the system's physical parameters.

A set of several key system physical parameters, for both the analyzed (cask) system and the set of critical experiments, is identified. These are physical parameters that may affect (i.e., have a correlation with) the bias of the criticality code that is used to analyze that system. For each key physical parameter, the correlation between the MCNP code bias and the value of that parameter is evaluated. For each of the 38 analyzed critical experiments, there is a calculated MCNP code bias (i.e., the MCNP calculated k_{eff} value minus 1.0), and a specific value for each of the identified key physical parameters. Using this data, and the formulas and methodologies given in NUREG/CR-6361, a USL function is calculated for each of the key physical parameters that are identified for the system. This function provides a minimum USL value as a function of the value of the physical parameter in question.

The final result of the criticality code benchmark evaluation, using the NUREG/CR-6361 methodology, is a set of USL functions—one for each of the identified key physical parameters of the system. When the actual criticality analyses are performed on VSC-24 cask configurations, the following procedure is used. The specific values of the key physical parameters for the specific configuration being analyzed are determined. These specific physical parameter values are entered into the corresponding USL functions, which were determined for each physical parameter in the benchmark evaluation. This results in a set of specific USL values for the analyzed system, one for each of the key physical parameters that were considered in the evaluation. Then, the lowest of the calculated USL values is used as the maximum allowable calculated k_{eff} value. The calculated k_{eff} value, in this case, is defined as the calculated k_{eff} value output by MCNP, plus two times the level of statistical error in the calculated k_{eff} value (which is also output by MCNP).

Section 4 of NUREG/CR-6361 (Reference 6.1) describes the methodology (USL Method 1, Confidence Band with Administrative Margin) used to determine USL functions for the VSC-24 system. The USL function incorporates code/data bias, uncertainty, and any applicable administrative margins. The general formulas used to determine the USL function for a given physical parameter (shown in Reference 6.1) are presented in Equations 6-5 through 6-10. The

USL functions, which are calculated from the critical benchmark experiment data, are based on a linear regression fit of the code bias versus the value of each physical parameter, along with an additional confidence band margin and the NRC administrative margin of 0.05 (in k_{eff}).

The set of equations define the USL value as a function of "x," where "x" is the value of the physical parameter in question. The equations apply over a range of values of "x" (i.e., from "x_{min}" to "x_{max}"). This range corresponds to the range of "x" values covered by the set of critical experiments that were analyzed.

Equation 6-5: Formula for *USL(x)*

$$USL(x) = 1 - \Delta k_m - W + \beta(x) \qquad \text{for } x_{min} \leq x \leq x_{max}$$

where:

"USL(x)" is the upper subcritical limit, as a function of physical parameter "x"

"Δk_m" is the NRC administrative criticality margin (i.e., 0.05)

"W" is the confidence band of the USL function, as defined in Equation 6-7

"$\beta(x)$" is the calculational bias of the criticality code, as a function of physical parameter "x," as defined in Equation 6-6

"x_{min}"/ "x_{max}" is the range of value of parameter "x" that is covered by the analyzed set of critical experiments.

Equation 6-6: Definition of "*β(x)*"

$$\beta(x) = k_c(x) - 1$$

where:

"$k_c(x)$" is a function that gives calculated k_{eff} as a linear function of physical parameter "x," based on a linear regression fit of the calculated k_{eff} values versus the corresponding values of "x" for each of the analyzed critical experiments.

Equation 6-7: Definition of "W"

The USL function confidence band, "W," is defined as the value of the function w(x), evaluated at "x_{min}" or "x_{max}" (see Equation 6-5), whichever gives the higher value, where the "w(x)" function is as follows:

$$w(x) = t \cdot s_p \left[1 + \frac{1}{N} + \frac{(x-\bar{x})^2}{\sum_{i=1,N}(x_i-\bar{x})^2} \right]^{1/2}$$

where:

"N" is the number of critical benchmark calculations used in establishing $k_c(x)$

"t" is the Student-t distribution statistic for (n-2) degrees of freedom

"$\bar{x}$" is the mean value of parameter "x" in the set of critical benchmark configurations

"s_p" is the pooled standard deviation for the set of criticality benchmark calculations (defined in Equation 6-8).

Equation 6-8: Definition of "s_p"

$$s_p = \sqrt{s_{k_c(x)}^{\ 2} + s_w^{\ 2}}$$

where:

"$s_{kc(x)}^2$" is the variance of the $k_c(x)$ linear regression fit (defined in Equation 6-9)

"s_w^2" is the within-variance of the critical benchmark k_{eff} data (see Equation 6-10).

Equation 6-9: Definition of "$s_{k_c(x)}^{\ 2}$"

$$s_{k_c(x)}^{\ 2} = \frac{1}{N-2}\left[\sum_{i=1,N}(k_i-\bar{k})^2 - \frac{\left\{\sum_{i=1,N}(x_i-\bar{x})(k_i-\bar{k})\right\}^2}{\sum_{i=1,N}(x_i-\bar{x})^2} \right]$$

where:

"N" is the number of analyzed critical experiments

"k_i" is the MCNP calculated k_{eff} result from critical benchmark calculation "i"

"$\bar{k}$" is the mean value of MCNP calculated k_{eff} from all critical benchmark calculations

"x_i" is the value of physical parameter "x" for critical benchmark configuration "i"

Equation 6-10: Definition of "s_w^2"

$$s_w^2 = \frac{1}{N}\sum_{i=1,N}\sigma_i^2$$

where:

"N" is the number of analyzed critical experiments

"σ_i" is the standard deviation (i.e., the statistical error level) of the MCNP calculated k_{eff} result for critical benchmark experiment "i" (this parameter is directly output by the MCNP code, along with the calculated k_{eff} value).

Using the above methodologies and equations, the actual USL functions (for each analyzed physical parameter) are determined for the VSC-24 cask system criticality analyses, based on the k_{eff} and physical parameter data shown for the actual set of analyzed critical experiment configurations in Table 6.4-9. These USL function calculations are described in Section 6.4.3.2.

6.4.3.2 VSC-24 USL Calculations

Five key physical parameters (which may have a significant influence on criticality code bias) have been identified for the VSC-24 cask system. These physical parameters used in the USL calculations for the VSC-24 system include:

- Fuel rod pitch ("Pitch")
- Hydrogen-to-fuel atom ratio in one unit cell of the fuel lattice ("H/X Atom Ratio")
- Water-to-fuel volume ratio in one unit cell of the fuel lattice ("H_2O/X Vol Ratio")
- Value of total neutron particle energy loss to fission divided by total neutron particle weight loss to fission ("E_{Loss}/W_{Loss}"). This is an MCNP output parameter that is a qualitative measure of the hardness of the neutron spectrum present within the analyzed configuration.
- Soluble boron concentration in parts per million by mass ("ppmb").

The values of each of these five physical parameters are shown for each of the 38 analyzed critical experiment configurations in columns 4 through 8 of Table 6.4-9. The corresponding k_{eff} value and associated statistical error level, calculated by MCNP for each of the 38 critical experiments, is shown in columns 9 and 10 of Table 6.4-9. Based on the data in Table 6.4-9 and the formulas in Section 6.4.3.1, USL functions are determined for each of the five physical parameters. Each USL function gives the USL value (i.e., the maximum allowable calculated k_{eff} value) as a function of the physical system parameter in question.

Table 6.4-10 lists calculated values for the variables in the Section 6.4.3.1 equations for each of the five defined physical parameters. These equation parameter values are calculated from the physical parameter values and k_{eff} values shown (for each of the 38 critical experiment configurations) in Table 6.4-9. Column 1 of Table 6.4-10 shows the symbol for the Section 6.4.3.1 equation parameter. The values of each USL equation parameter, for each of the five defined physical system parameters, are shown in Columns 2 through 6 of Table 6.4-10. The definition (or description) of each equation parameter is provided in Section 6.4.3.1.

Based on the parameter values in Table 6.4-10 and the corresponding equations in Section 6.4.3.1, USL functions are determined for each of the five physical parameters that are evaluated for the VSC-24 criticality analyses. These USL functions are presented in Table 6.4-11. Table 6.4-11 also provides the range of values (for the physical parameter in question) over which each USL function applies.

For each of the five analyzed physical parameters, Table 6.4-12 lists the range of values that occur for the VSC-24 criticality evaluations presented in Table 6.4-1 through Table 6.4-8. The B&W 15x15 assembly has a lattice pitch of 1.44272 cm, and a corresponding USL value (for the lattice pitch parameter) of 0.93824, as shown in Table 6.4-1. All other assemblies have lower pitch values and lower corresponding USL values. The USL functions shown in Table 6.4-11 for the other four physical parameters ("H/X Atom Ratio," "H_2O/X Vol Ratio," "E_{Loss}/W_{Loss}," and "ppmb") yield USL values higher than 0.93824 over the entire range of values shown for those parameters in Table 6.4-12 (i.e., for the range of values that applies for the VSC-24 criticality analyses).

Therefore, those four physical parameters never yield the limiting (i.e., minimum) USL value for the VSC-24 criticality evaluations. Thus, the assembly lattice pitch parameter, and its corresponding USL function (shown in Table 6.4-11), is the sole determining factor that sets the minimum USL value for any given analyzed VSC-24 configuration. Therefore, the maximum allowable calculated k_{eff} value (i.e., the minimum USL value) for a given analyzed VSC-24 configuration is a function (only) of the lattice pitch of the loaded fuel assembly. The notes for Table 6.4-1 through Table 6.4-8 provide the lattice pitch for each assembly class, and the corresponding minimum USL value that applies for each case. The k_{eff} calculated by MCNP, plus twice the level of statistical error (also calculated by MCNP), must not exceed the minimum USL value that is specified for each case in the table notes.

Table 6.1-1 - MSB and MTC Component Dimensions (Actual and Modeled)

Component	Material	Actual Dimension	Modeled Dimension
Fuel Sleeve	Steel	Outer Width: 9.2", ± 0.125" Thickness: 0.2" or 0.1875", + 0.03", - 0.01"	Outer Width: 9.075" (23.0505 cm) Thickness: 0.1775" (0.45085 cm) [1]
Radial Support Plate	Steel	Inner Radius: 29.1", ± 0.125" Thickness: 0.5", ± 0.125"	Inner Radius: 28.975" (73.597 cm) Thickness: outward to inner surface of MSB shell
Outer Support Wall	Steel	Thickness: 0.5", ± 0.125"	Thickness: 0.625" (1.588 cm)
Outer Support Bar	Steel	Width: 2.0", ± 0.125" Thickness: 1.45", + 0.062", - 0.15"	Width: 2.125" (5.398 cm) Thickness: 1.512" (3.84 cm)
MSB Shell	Steel	Outer Diameter: 62.5", + 0", - 0.5" Thickness: 1.0", ± 0.062"	Outer Diameter: 62.0" (157.48 cm) Thickness: 1.062" (2.697 cm)
MTC Inner Shell [2]	Steel	Inner Diameter: 63.0" Outer Diameter: 64.5"	Inner Diameter: 63.0" (160.02 cm) Outer Diameter: 64.5" (163.84 cm)
MTC Lead Shell [2]	Lead	Thickness: 3.75"	Thickness: 3.75" (9.52 cm)
MTC RX-277 Shell [2]	RX-277	Thickness: 4.0"	Thickness: 4.0" (10.16 cm)
MTC Outer Shell [2]	Steel	Outer Diameter: 82.0" Thickness: 1.0"	Outer Diameter: 82.0" (208.28 cm) Thickness: 1.0" (2.54 cm)

Notes:

1. The maximum fuel sleeve thickness is modeled as 0.23" (0.5842 cm) in the fuel sleeve thickness sensitivity studies.
2. Only nominal dimensions are considered for the MTC due to its limited effect on reactivity.

Table 6.2-1(a) - Fuel Assembly Class Characterization Parameters

Parameter	Fuel Assembly Class			
	B&W 15x15	W 14x14	W 15x15	W 17x17
Assembly Lattice Layout	see Figure 6.2-1	see Figure 6.2-2	see Figure 6.2-3	see Figure 6.2-4
Assembly Pin Pitch	1.44272 cm	1.41224 cm	1.43002 cm	1.25984 cm
Fuel Density	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical
Fuel Pellet Diameter	from 0.92202 cm to 0.94996 cm	from 0.86360 cm to 0.94234 cm	from 0.89408 cm to 0.93980 cm	from 0.75946 cm to 0.82804 cm
Fuel Clad Material	zircaloy	zircaloy	zircaloy	zircaloy
Fuel Clad Outer Diameter	≤ 1.10236 cm	≤ 1.08712 cm	≤ 1.08712 cm	≤ 0.96012 cm
Fuel Clad Thickness	≥ 0.06604 cm	≥ 0.05588 cm	≥ 0.05842 cm	≥ 0.05588 cm
Guide Tube Material	zircaloy	zircaloy	zircaloy	zircaloy
Guide Tube Outer Diameter	≤ 1.36 cm	≤ 1.37414 cm	≤ 1.4 cm	≤ 1.24 cm
Guide Tube Thickness	≤ 0.045 cm	≤ 0.04318 cm	≤ 0.06 cm	≤ 0.06 cm
Instrument Tube Material	zircaloy	zircaloy	zircaloy	zircaloy
Instrument Tube Outer Diameter	≤ 1.26 cm	≤ 1.37414 cm	≤ 1.4 cm	≤ 1.24 cm
Instrument Tube Thickness	≤ 0.07 cm	all	≤ 0.06 cm	≤ 0.06 cm
Active Fuel Length	≤ 370.84 cm	≤ 373.0 cm	≤ 370.0 cm	≤ 371.0 cm
Guide Bar Effective Diameter [1]	not applicable	not applicable	not applicable	not applicable
Control Component Rodlets Allowed in Guide Tubes	yes	yes	no	yes
Control Component Rodlet Clad Material	zircaloy or stainless steel	zircaloy or stainless steel	not applicable	zircaloy or stainless steel
Control Component Rodlet Outer Diameter	≤ 1.1176 cm	all	not applicable	≤ 0.9779 cm
Control Component Rodlet Fill Material	any non-hydrogen bearing material	any non-hydrogen bearing material	not applicable	any non-hydrogen bearing material

Note:

The guide bars may have either a rectangular or circular cross section. The guide bar effective diameter corresponds to the diameter of a circular region having an area equal to the actual guide bar cross section area.

Table 6.2-1(b) - Fuel Assembly Class Characterization Parameters

Parameter	Fuel Assembly Class			
	CE 15x15A	CE 15x15B	CE 15x15C	CE 16x16
Assembly Lattice Layout	see Figure 6.2-5	see Figure 6.2-6	see Figure 6.2-7	see Figure 6.2-8
Assembly Pin Pitch	1.397 cm	1.397 cm	1.397 cm	1.28524 cm
Fuel Density	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical
Fuel Pellet Diameter	from 0.888 cm to 0.912 cm	from 0.888 cm to 0.912 cm	from 0.888 cm to 0.912 cm	from 0.81534 cm to 0.83566 cm
Fuel Clad Material	zircaloy	zircaloy	zircaloy	zircaloy
Fuel Clad Outer Diameter	≤ 1.062 cm	≤ 1.062 cm	≤ 1.062 cm	≤ 0.98044 cm
Fuel Clad Thickness	≥ 0.0508 cm	≥ 0.0508 cm	≥ 0.0508 cm	≥ 0.06096 cm
Guide Tube Material	not applicable	zircaloy	zircaloy	zircaloy
Guide Tube Outer Diameter	not applicable	≤ 1.06934 cm	≤ 1.06934 cm	≤ 2.54 cm
Guide Tube Thickness	not applicable	≤ 0.02032 cm	≤ 0.02032 cm	≤ 0.1778 cm
Instrument Tube Material	zircaloy	zircaloy	zircaloy	not applicable
Instrument Tube Outer Diameter	≤ 1.06934 cm	≤ 1.06934 cm	≤ 1.06934 cm	not applicable
Instrument Tube Thickness	≤ 0.08509 cm	≤ 0.08509 cm	≤ 0.08509 cm	not applicable
Active Fuel Length	≤ 336.0 cm	≤ 336.0 cm	≤ 336.0 cm	≤ 385.0 cm
Guide Bar Effective Diameter [1]	≤ 1.2006 cm	≤ 1.2006 cm	≤ 1.2006 cm	not applicable
Control Component Rodlets Allowed in Guide Tubes	not applicable	yes	yes	no
Control Component Rodlet Clad Material	not applicable	zircaloy or stainless steel	zircaloy or stainless steel	not applicable
Control Component Rodlet Outer Diameter	not applicable	all	all	not applicable
Control Component Rodlet Fill Material	not applicable	any non-hydrogen bearing material	any non-hydrogen bearing material	not applicable

Note:

The guide bars may have either a rectangular or circular cross section. The guide bar effective diameter corresponds to the diameter of a circular region having an area equal to the actual guide bar cross section area.

Table 6.3-1 - MSB and MTC Material Compositions

Material	Modeled Composition	MCNP Cross Section Data Table
Steel	Carbon: 1.0 wt% Iron: 99.0 wt% Density = 7.8212 g/cm^3	Carbon: 6000.50c Iron: 26054.60c, 26056.60c, 26057.60c, 26058.60c
Lead	Lead: 100.0 wt% Density = 11.34 g/cm^3	Lead: 82000.50c
RX-277	Hydrogen: 3.044E-2 atoms/b-cm Boron-10: 2.615E-4 atoms/b-cm Boron-11: 1.052E-3 atoms/b-cm Oxygen: 3.523E-2 atoms/b-cm Sodium: 2.596E-4 atoms/b-cm Magnesium: 2.081E-4 atoms/b-cm Aluminum: 8.965E-3 atoms/b-cm Silicon: 7.673E-4 atoms/b-cm Calcium: 2.229E-3 atoms/b-cm Sulfur: 5.995E-5 atoms/b-cm Iron: 4.890E-5 atoms/b-cm Density = 7.953E-2 atoms/b-cm	Hydrogen: 1001.50c Boron-10: 5010.50c Boron-11: 5011.56c Oxygen: 8016.50c Sodium: 11023.50c Magnesium: 12000.50c Aluminum: 13027.50c Silicon: 14000.50c Calcium: 20000.50c Sulfur: 16032.50c Iron: 26054.60c, 26056.60c, 26057.60c, 26058.60c
zircaloy	Zirconium: 100.0 wt% Density = 6.56 g/cm^3	Zirconium: 40000.60c
$^{11}B_4C$	Boron-11: 78.57 wt% Carbon: 21.43 wt% Density = 2.54 g/cm^3	Boron-11: 5011.56c Carbon: 6000.50c

Table 6.3-2 - UO_2 Fuel Compositions

Initial Enrichment (wt% ^{235}U in U)	Weight Percent (wt%) in UO_2				
	Oxygen	^{234}U	^{235}U	^{236}U	^{238}U
3.0	11.850318	0.022414	2.644490	0.012165	85.470613
3.5	11.850989	0.026489	3.085215	0.014192	85.023115
4.0	11.851659	0.030613	3.525934	0.016219	84.575575
4.5	11.852329	0.034780	3.966645	0.018247	84.127999
5.0	11.853000	0.038987	4.407350	0.020274	83.680389

Table 6.4-1 - Criticality Analysis Results for the B&W 15x15 Assembly Class[1,2]

Initial Enrichment (w/o ^{235}U)	Boron Concentration (ppm boron)	Moderator Density (g/cm^3)	MCNP Calculated k_{eff}	Relative Error (1σ)	Criticality Margin (vs. USL)[3]
3.0	2500.0	0.55	0.93058	0.00076	0.00614
3.0	2500.0	0.60	0.93232	0.00077	0.00438
3.0	2500.0	0.65	0.93287	0.00078	**0.00381**
3.0	2500.0	0.70	0.93067	0.00077	0.00603
3.0	2500.0	0.75	0.92820	0.00073	0.00858
3.5	3300.0	0.45	0.92649	0.00075	0.01025
3.5	3300.0	0.50	0.93256	0.00074	0.00420
3.5	3300.0	0.55	0.93258	0.00079	**0.00408**
3.5	3300.0	0.60	0.93231	0.00078	0.00437
3.5	3300.0	0.65	0.93035	0.00074	0.00641
4.0	4100.0	0.45	0.93155	0.00076	0.00517
4.0	4100.0	0.50	0.93303	0.00078	0.00365
4.0	4100.0	0.55	0.93362	0.00080	**0.00302**
4.0	4100.0	0.60	0.93101	0.00081	0.00561
4.0	4100.0	0.65	0.92855	0.00079	0.00811
4.5	5000.0	0.40	0.92879	0.00080	0.00785
4.5	5000.0	0.45	0.93115	0.00078	0.00553
4.5	5000.0	0.50	0.93356	0.00077	**0.00314**
4.5	5000.0	0.55	0.93185	0.00080	0.00479
4.5	5000.0	0.60	0.92622	0.00082	0.01038
5.0	5900.0	0.35	0.92661	0.00082	0.00999
5.0	5900.0	0.40	0.93188	0.00074	0.00488
5.0	5900.0	0.45	0.93395	0.00075	**0.00279**
5.0	5900.0	0.50	0.93245	0.00080	0.00419
5.0	5900.0	0.55	0.92883	0.00076	0.00789

Notes:

1. The B&W 15x15 assembly class is defined by the set of allowable assembly geometry parameters shown in Table 6.2-1(a).
2. The analysis results, which are based on the presence of inserted control components, are bounding for assemblies that do not contain an inserted control component.
3. Equal to the minimum USL value, minus the calculated k_{eff} value plus twice the relative error level (i.e., USL - [k_{eff} + 2σ]). For the B&W 15x15 assembly, the pin pitch USL function yields the lowest USL value, which equals 0.93824 for the B&W 15x15 assembly pin pitch of 1.44272 cm. The bounding (lowest margin) cases are shown in bold for each enrichment case. The USL values and the method of their determination are discussed in Section 6.4.3.

Table 6.4-2 - Criticality Analysis Results for the W 14x14 Assembly Class[1,2]

Initial Enrichment (w/o ^{235}U)	Boron Concentration (ppm boron)	Moderator Density (g/cm^3)	MCNP Calculated k_{eff}	Relative Error (1σ)	Criticality Margin (vs. USL)[3]
3.5	2400.0	0.30	0.91527	0.00081	0.02077
3.5	2400.0	0.35	0.92646	0.00085	0.00950
3.5	2400.0	0.40	0.93268	0.00077	0.00344
3.5	2400.0	0.45	0.93367	0.00083	**0.00233**
3.5	2400.0	0.50	0.93213	0.00080	0.00393
4.0	3075.0	0.35	0.93138	0.00081	0.00466
4.0	3075.0	0.40	0.93350	0.00081	0.00254
4.0	3075.0	0.45	0.93379	0.00085	**0.00217**
4.0	3075.0	0.50	0.93131	0.00077	0.00481
4.0	3075.0	0.55	0.92769	0.00086	0.00825
4.5	3800.0	0.30	0.93049	0.00084	0.00549
4.5	3800.0	0.35	0.93368	0.00084	0.00230
4.5	3800.0	0.40	0.93497	0.00083	**0.00103**
4.5	3800.0	0.45	0.93177	0.00084	0.00421
4.5	3800.0	0.50	0.92599	0.00081	0.01005
5.0	4650.0	0.25	0.92531	0.00077	0.01081
5.0	4650.0	0.30	0.93114	0.00086	0.00480
5.0	4650.0	0.35	0.93347	0.00082	**0.00255**
5.0	4650.0	0.40	0.93266	0.00082	0.00336
5.0	4650.0	0.45	0.92522	0.00084	0.01076

Notes:

1. The W 14x14 assembly class is defined by the set of allowable assembly geometry parameters shown in Table 6.2-1(a).
2. The analysis results, which are based on the presence of inserted control components, are bounding for assemblies that do not contain an inserted control component.
3. Equal to the minimum USL value, minus the calculated k_{eff} value plus twice the relative error level (i.e., USL - [k_{eff} + 2σ]). For the W 14x14 assembly, the pin pitch USL function yields the lowest USL value, which equals 0.93766 for the W 14x14 assembly pin pitch of 1.41224 cm. The bounding (lowest margin) cases are shown in bold for each enrichment case. The USL values and the method of their determination are discussed in Section 6.4.3.

Table 6.4-3 - Criticality Analysis Results for the W 15x15 Assembly Class[1]

Initial Enrichment (w/o ^{235}U)	Boron Concentration (ppm boron)	Moderator Density (g/cm^3)	MCNP Calculated k_{eff}	Relative Error (1σ)	Criticality Margin (vs. USL)[2]
3.0	2425.0	0.50	0.93070	0.00082	0.00566
3.0	2425.0	0.55	0.93273	0.00077	**0.00373**
3.0	2425.0	0.60	0.93145	0.00079	0.00497
3.0	2425.0	0.65	0.92901	0.00077	0.00745
3.0	2425.0	0.70	0.92604	0.00076	0.01044
3.5	3225.0	0.45	0.93116	0.00079	0.00526
3.5	3225.0	0.50	0.93261	0.00081	**0.00377**
3.5	3225.0	0.55	0.93095	0.00080	0.00545
3.5	3225.0	0.60	0.92905	0.00079	0.00737
3.5	3225.0	0.65	0.92653	0.00079	0.00989
4.0	4075.0	0.40	0.92984	0.00084	0.00648
4.0	4075.0	0.45	0.93208	0.00080	0.00432
4.0	4075.0	0.50	0.93351	0.00080	**0.00289**
4.0	4075.0	0.55	0.93048	0.00079	0.00594
4.0	4075.0	0.60	0.92513	0.00083	0.01121
4.5	4975.0	0.35	0.92876	0.00077	0.00770
4.5	4975.0	0.40	0.93032	0.00076	0.00616
4.5	4975.0	0.45	0.93329	0.00078	**0.00315**
4.5	4975.0	0.50	0.92914	0.00078	0.00730
4.5	4975.0	0.55	0.92774	0.00080	0.00866
5.0	5875.0	0.30	0.92749	0.00076	0.00899
5.0	5875.0	0.35	0.93263	0.00078	0.00381
5.0	5875.0	0.40	0.93207	0.00081	0.00431
5.0	5875.0	0.45	0.93339	0.00081	**0.00299**
5.0	5875.0	0.50	0.93067	0.00083	0.00567

Notes:

1. The W 15x15 assembly class is defined by the set of allowable assembly geometry parameters shown in Table 6.2-1(a).
2. Equal to the minimum USL value, minus the calculated k_{eff} value plus twice the relative error level (i.e., USL - [$k_{eff} + 2\sigma$]). For the W 15x15 assembly, the pin pitch USL function yields the lowest USL value, which equals 0.93800 for the W 15x15 assembly pin pitch of 1.43002 cm. The bounding (lowest margin) cases are shown in bold for each enrichment case. The USL values and the method of their determination are discussed in Section 6.4.3.

Table 6.4-4 - Criticality Analysis Results for the W 17x17 Assembly Class[1,2]

Initial Enrichment (w/o ^{235}U)	Boron Concentration (ppm boron)	Moderator Density (g/cm^3)	MCNP Calculated k_{eff}	Relative Error (1σ)	Criticality Margin (vs. USL)[3]
3.0	2375.0	0.40	0.92404	0.00078	0.00911
3.0	2375.0	0.45	0.92637	0.00076	0.00682
3.0	2375.0	0.50	0.92953	0.00076	0.00366
3.0	2375.0	0.55	0.93228	0.00075	**0.00093**
3.0	2375.0	0.60	0.92873	0.00075	0.00448
3.5	3175.0	0.40	0.92914	0.00079	0.00399
3.5	3175.0	0.45	0.93172	0.00080	**0.00139**
3.5	3175.0	0.50	0.93083	0.00081	0.00226
3.5	3175.0	0.55	0.92934	0.00078	0.00381
3.5	3175.0	0.60	0.92684	0.00082	0.00623
4.0	4025.0	0.30	0.91986	0.00079	0.01327
4.0	4025.0	0.35	0.92959	0.00083	0.00346
4.0	4025.0	0.40	0.93127	0.00078	0.00188
4.0	4025.0	0.45	0.93235	0.00078	**0.00080**
4.0	4025.0	0.50	0.92990	0.00080	0.00321
4.5	4925.0	0.30	0.92551	0.00078	0.00764
4.5	4925.0	0.35	0.93191	0.00079	0.00122
4.5	4925.0	0.40	0.93264	0.00079	**0.00049**
4.5	4925.0	0.45	0.93054	0.00077	0.00263
4.5	4925.0	0.50	0.93047	0.00076	0.00272
5.0	5900.0	0.30	0.92916	0.00080	0.00395
5.0	5900.0	0.35	0.93236	0.00085	0.00065
5.0	5900.0	0.40	0.93262	0.00081	**0.00047**
5.0	5900.0	0.45	0.92974	0.00078	0.00341
5.0	5900.0	0.50	0.92341	0.00079	0.00972

Notes:

1. The W 17x17 assembly class is defined by the set of allowable assembly geometry parameters shown in Table 6.2-1(a).
2. The analysis results, which are based on the presence of inserted control components, are bounding for assemblies that do not contain an inserted control component.
3. Equal to the minimum USL value, minus the calculated k_{eff} value plus twice the relative error level (i.e., USL – [k_{eff} + 2σ]). For the W 17x17 assembly, the pin pitch USL function yields the lowest USL value, which equals 0.93471 for the W 17x17 assembly pin pitch of 1.25984 cm. The bounding (lowest margin) cases are shown in bold for each enrichment case. The USL values and the method of their determination are discussed in Section 6.4.3.

Table 6.4-5 - Criticality Analysis Results for the CE 15x15A Assembly Class[1]

Initial Enrichment (w/o ^{235}U)	Boron Concentration (ppm boron)	Moderator Density (g/cm^3)	MCNP Calculated k_{eff}	Relative Error (1σ)	Criticality Margin (vs. USL)[2]
3.0	2375.0	0.50	0.93168	0.00078	0.00414
3.0	2375.0	0.55	0.93180	0.00074	0.00410
3.0	2375.0	0.60	0.93222	0.00077	**0.00362**
3.0	2375.0	0.65	0.93040	0.00078	0.00542
3.0	2375.0	0.70	0.92813	0.00076	0.00773
3.5	3125.0	0.40	0.92701	0.00077	0.00883
3.5	3125.0	0.45	0.93164	0.00074	0.00426
3.5	3125.0	0.50	0.93420	0.00080	**0.00158**
3.5	3125.0	0.55	0.93285	0.00080	0.00293
3.5	3125.0	0.60	0.93349	0.00076	0.00237
4.0	3925.0	0.40	0.93154	0.00080	0.00424
4.0	3925.0	0.45	0.93375	0.00078	0.00207
4.0	3925.0	0.50	0.93364	0.00075	0.00224
4.0	3925.0	0.55	0.93389	0.00081	**0.00187**
4.0	3925.0	0.60	0.92954	0.00081	0.00622
4.5	4825.0	0.35	0.92909	0.00080	0.00669
4.5	4825.0	0.40	0.93163	0.00077	0.00421
4.5	4825.0	0.45	0.93314	0.00082	**0.00260**
4.5	4825.0	0.50	0.93217	0.00074	0.00373
4.5	4825.0	0.55	0.92926	0.00079	0.00654
5.0	5725.0	0.35	0.93215	0.00082	0.00359
5.0	5725.0	0.40	0.93346	0.00080	0.00232
5.0	5725.0	0.45	0.93442	0.00076	**0.00144**
5.0	5725.0	0.50	0.93187	0.00082	0.00387
5.0	5725.0	0.55	0.92526	0.00082	0.01048

Notes:

1. The CE 15x15A assembly class is defined by the set of allowable assembly geometry parameters shown in Table 6.2-1(b).
2. Equal to the minimum USL value, minus the calculated k_{eff} value plus twice the relative error level (i.e., USL – [k_{eff} + 2σ]). For the CE 15x15A assembly, the pin pitch USL function yields the lowest USL value, which equals 0.93738 for the CE 15x15A assembly pin pitch of 1.39700 cm. The bounding (lowest margin) cases are shown in bold for each enrichment case. The USL values and the method of their determination are discussed in Section 6.4.3.

Table 6.4-6 - Criticality Analysis Results for the CE 15x15B Assembly Class[1,2]

Initial Enrichment (w/o ^{235}U)	Boron Concentration (ppm boron)	Moderator Density (g/cm^3)	MCNP Calculated k_{eff}	Relative Error (1σ)	Criticality Margin (vs. USL)[3]
3.0	2325.0	0.50	0.92978	0.00077	0.00606
3.0	2325.0	0.55	0.93039	0.00078	0.00543
3.0	2325.0	0.60	0.93170	0.00076	**0.00416**
3.0	2325.0	0.65	0.93079	0.00076	0.00507
3.0	2325.0	0.70	0.92821	0.00073	0.00771
3.5	3050.0	0.45	0.93147	0.00081	0.00429
3.5	3050.0	0.50	0.93266	0.00081	0.00310
3.5	3050.0	0.55	0.93408	0.00080	**0.00170**
3.5	3050.0	0.60	0.93148	0.00075	0.00440
3.5	3050.0	0.65	0.92811	0.00078	0.00771
4.0	3850.0	0.40	0.92916	0.00085	0.00652
4.0	3850.0	0.45	0.93518	0.00082	**0.00056**
4.0	3850.0	0.50	0.93358	0.00077	0.00226
4.0	3850.0	0.55	0.93225	0.00079	0.00355
4.0	3850.0	0.60	0.92962	0.00079	0.00618
4.5	4725.0	0.35	0.92751	0.00080	0.00827
4.5	4725.0	0.40	0.93086	0.00082	0.00488
4.5	4725.0	0.45	0.93362	0.00083	0.00210
4.5	4725.0	0.50	0.93401	0.00080	**0.00177**
4.5	4725.0	0.55	0.92745	0.00079	0.00835
5.0	5600.0	0.35	0.93124	0.00082	0.00450
5.0	5600.0	0.40	0.93479	0.00078	**0.00103**
5.0	5600.0	0.45	0.93347	0.00078	0.00235
5.0	5600.0	0.50	0.93166	0.00082	0.00408
5.0	5600.0	0.55	0.92722	0.00085	0.00846

Notes:

1. The CE 15x15B assembly class is defined by the set of allowable assembly geometry parameters shown in Table 6.2-1(b).
2. The analysis results, which are based on the presence of inserted control components, are bounding for assemblies that do not contain an inserted control component.
3. Equal to the minimum USL value, minus the calculated k_{eff} value plus twice the relative error level (i.e., USL – [k_{eff} + 2σ]). For the CE 15x15B assembly, the pin pitch USL function yields the lowest USL value, which equals 0.93738 for the CE 15x15B assembly pin pitch of 1.39700 cm. The bounding (lowest margin) cases are shown in bold for each enrichment case. The USL values and the method of their determination are discussed in Section 6.4.3.

Table 6.4-7 - Criticality Analysis Results for the CE 15x15C Assembly Class[1,2]

Initial Enrichment (w/o ^{235}U)	Boron Concentration (ppm boron)	Moderator Density (g/cm^3)	MCNP Calculated k_{eff}	Relative Error (1σ)	Criticality Margin (vs. USL)[3]
3.0	2350.0	0.50	0.92781	0.00077	0.00803
3.0	2350.0	0.55	0.92986	0.00073	**0.00606**
3.0	2350.0	0.60	0.92781	0.00073	0.00811
3.0	2350.0	0.65	0.92644	0.00077	0.00940
3.0	2350.0	0.70	0.92344	0.00077	0.01240
3.5	3025.0	0.45	0.93274	0.00081	0.00302
3.5	3025.0	0.50	0.93513	0.00079	**0.00067**
3.5	3025.0	0.55	0.93484	0.00081	0.00092
3.5	3025.0	0.60	0.93250	0.00078	0.00332
3.5	3025.0	0.65	0.92788	0.00076	0.00798
4.0	3825.0	0.40	0.92944	0.00082	0.00630
4.0	3825.0	0.45	0.93470	0.00080	0.00108
4.0	3825.0	0.50	0.93463	0.00084	**0.00107**
4.0	3825.0	0.55	0.93242	0.00078	0.00340
4.0	3825.0	0.60	0.92778	0.00078	0.00804
4.5	4725.0	0.35	0.92719	0.00080	0.00859
4.5	4725.0	0.40	0.93078	0.00083	0.00494
4.5	4725.0	0.45	0.93231	0.00083	**0.00341**
4.5	4725.0	0.50	0.93160	0.00084	0.00410
4.5	4725.0	0.55	0.92585	0.00080	0.00993
5.0	5575.0	0.35	0.93102	0.00080	0.00476
5.0	5575.0	0.40	0.93387	0.00079	**0.00193**
5.0	5575.0	0.45	0.93222	0.00080	0.00356
5.0	5575.0	0.50	0.93094	0.00081	0.00482
5.0	5575.0	0.55	0.92263	0.00079	0.01317

Notes:

1. The CE 15x15C assembly class is defined by the set of allowable assembly geometry parameters shown in Table 6.2-1(b).
2. The analysis results, which are based on the presence of inserted control components, are bounding for assemblies that do not contain an inserted control component.
3. Equal to the minimum USL value, minus the calculated k_{eff} value plus twice the relative error level (i.e., USL – [k_{eff} + 2σ]). For the CE 15x15C assembly, the pin pitch USL function yields the lowest USL value, which equals 0.93738 for the CE 15x15C assembly pin pitch of 1.39700 cm. The bounding (lowest margin) cases are shown in bold for each enrichment case. The USL values and the method of their determination are discussed in Section 6.4.3.

Table 6.4-8 - Criticality Analysis Results for the CE 16x16 Assembly Class[1]

Initial Enrichment (w/o ^{235}U)	Boron Concentration (ppm boron)	Moderator Density (g/cm^3)	MCNP Calculated k_{eff}	Relative Error (1σ)	Criticality Margin (vs. USL)[2]
3.5	2675.0	0.45	0.92736	0.00075	0.00635
3.5	2675.0	0.50	0.92856	0.00078	0.00509
3.5	2675.0	0.55	0.93015	0.00083	**0.00340**
3.5	2675.0	0.60	0.92891	0.00082	0.00466
3.5	2675.0	0.65	0.92420	0.00083	0.00935
4.0	3300.0	0.35	0.91650	0.00078	0.01715
4.0	3300.0	0.40	0.92661	0.00082	0.00696
4.0	3300.0	0.45	0.93257	0.00078	**0.00108**
4.0	3300.0	0.50	0.93206	0.00079	0.00157
4.0	3300.0	0.55	0.93244	0.00083	0.00111
4.5	4075.0	0.40	0.92708	0.00084	0.00645
4.5	4075.0	0.45	0.93079	0.00079	**0.00284**
4.5	4075.0	0.50	0.93059	0.00084	0.00294
4.5	4075.0	0.55	0.92915	0.00080	0.00446
4.5	4075.0	0.60	0.92237	0.00083	0.01118
5.0	4800.0	0.35	0.92645	0.00079	0.00718
5.0	4800.0	0.40	0.93079	0.00083	0.00276
5.0	4800.0	0.45	0.93255	0.00083	**0.00100**
5.0	4800.0	0.50	0.93212	0.00082	0.00145
5.0	4800.0	0.55	0.92527	0.00079	0.00836

Notes:

1. The CE 16x16 assembly class is defined by the set of allowable assembly geometry parameters shown in Table 6.2-1(b).
2. Equal to the minimum USL value, minus the calculated k_{eff} value plus twice the relative error level (i.e., USL – [k_{eff} + 2σ]). For the CE 16x16 assembly, the pin pitch USL function yields the lowest USL value, which equals 0.93521 for the CE 16x16 assembly pin pitch of 1.28524 cm. The bounding (lowest margin) cases are shown in bold for each enrichment case. The USL values, and the method of their determination, are discussed in Section 6.4.3.

Table 6.4-9 - Results of Criticality Sensitivity Analyses (Δk_{eff})[1]

Variation from Design Basis Configuration[2]	Maximum Δk_{eff}[3]	Minimum Δk_{eff}[3]	Mean Δk_{eff}[4]
Assemblies Centered in Sleeves[5]	-0.00114	-0.01801	-0.00742
Assemblies in Groups of Four[6]	-0.00204	-0.01981	-0.01013
Maximum Fuel Sleeve Thickness[7]	-0.00334	-0.01464	-0.00785
MSB Shifted in MTC Cavity[8]	0.00256	-0.00370	-0.00054
Fuel Pellet Diameter + 1/3[9]	0.00181	-0.00455	-0.00116
Fuel Pellet Diameter + 2/3[10]	0.00267	-0.01049	-0.00235
Maximum Fuel Pellet Diameter[11]	0.00339	-0.01978	-0.00428

Notes:

1. Δk_{eff} is defined as the k_{eff} value calculated for the sensitivity (i.e., altered parameter value) case minus the design basis (or base case) k_{eff} value calculated for the corresponding assembly type and enrichment case.
2. For each of the seven sensitivity analyses, a single assembly or cask system geometry parameter is changed from its design basis value. These seven parameter changes are described below. For each sensitivity analysis, only one system parameter is altered, with all other parameters being set to their design basis values. The seven system parameter variations (or changes) are described in Notes 5-11.
3. Equal to the maximum and/or minimum value of Δk_{eff} that is calculated for any of the 38 assembly type / enrichment combinations shown in Table 6.4-1 through Table 6.4-8, for the parameter variation in question.
4. This is the mean value of Δk_{eff}, over the 38 assembly type / enrichment combinations shown in Table 6.4-1 through Table 6.4-8, for the parameter variation in question.
5. Assemblies are centered within their sleeves, versus the design basis analysis assembly locations shown in the top half of Figure 6.3-2.
6. Assemblies are arranged (within their sleeves) as shown in the bottom half of Figure 6.3-2 (as opposed to the design basis analysis configuration shown in the top half of Figure 6.3-2).
7. The fuel sleeve thickness is increased from its minimum allowable value of 0.1775" (the value assumed in the base case analyses) to its maximum allowable value of 0.2175".
8. The MSB is shifted from the center of the MTC cavity (the design basis analysis configuration) to one side of the MTC cavity (i.e., so that it contacts the MTC inner liner at one location).
9. The fuel pellet diameter is increased to 1/3 of the way between the minimum and maximum allowable values (see Note 11).
10. The fuel pellet diameter is increased to 2/3 of the way between the minimum and maximum allowable values (see Note 11).
11. For each assembly type / enrichment combination, the fuel pellet diameter is increased from its minimum allowable value to its maximum allowable value (shown for each assembly type in Table 6.2-1). All design basis analyses model the minimum allowable value shown in Table 6.2-1.

Table 6.4-10 - Characteristic Parameters of VSC-24 Critical Benchmarks (2 pages)

Case Name	Enrichment (wt% ^{235}U)	Number of Fuel Rods	Square Pitch (cm)	H/X Atom Ratio	H_2O/X Volume Ratio	E Loss/ W Loss	ppm Boron	Calculated k_{eff}	Relative Error (1σ)
A01	4.31	509	1.715	7.149E+01	1.090E+00	2.215E-01	0	0.99859	0.00034
A02	4.31	737	1.715	7.149E+01	1.090E+00	2.452E-01	1030	1.00092	0.00035
A03	4.31	917	1.715	7.149E+01	1.090E+00	2.575E-01	1820	0.99716	0.00035
A04	4.31	1192	1.715	7.149E+01	1.090E+00	2.676E-01	2550	0.99966	0.00033
C01	2.35	708	1.562	1.638E+02	1.196E+00	2.060E-01	0	0.99395	0.00030
C02	2.35	1201	1.562	1.638E+02	1.196E+00	2.217E-01	463.8	0.99770	0.00032
C03	2.35	1201	1.905	3.297E+02	2.408E+00	1.540E-01	568.1	0.99881	0.00029
D01	4.02	484	1.511	8.871E+01	1.140E+00	2.136E-01	0	0.99532	0.00033
D02	4.02	724	1.511	8.871E+01	1.140E+00	2.311E-01	460	0.99607	0.00035
D03	4.02	936	1.511	8.871E+01	1.140E+00	2.397E-01	1152	0.99416	0.00031
D04	4.02	2024	1.511	8.871E+01	1.140E+00	2.608E-01	2342	0.99646	0.00031
D05	4.02	4904	1.511	8.871E+01	1.140E+00	2.724E-01	3389	0.99529	0.00030
E01	2.459	1728	1.636	2.153E+02	1.840E+00	1.796E-01	514	0.99720	0.00030
E02	2.459	1728	1.636	2.153E+02	1.840E+00	1.826E-01	432	0.99736	0.00030
E03	2.459	1728	1.636	2.153E+02	1.840E+00	1.687E-01	217	0.99402	0.00028
E04	2.459	1728	1.636	2.153E+02	1.840E+00	1.610E-01	143	0.99229	0.00030
F01	2.459	4808	1.636	2.165E+02	1.841E+00	1.995E-01	1338	0.99816	0.00030
F02	2.459	554	1.636	2.165E+02	1.841E+00	1.650E-01	0	0.99500	0.00030
F03	2.459 (81.5%) 4.02 (18.5%)	4808	1.636	1.985E+02	1.784E+00	2.092E-01	1899	0.99942	0.00030
F04	2.459 (79.6%) 4.02 (20.4%)	4620	1.636	1.966E+02	1.778E+00	2.093E-01	1777	0.99985	0.00031

Table 6.4-10 - Characteristic Parameters of VSC-24 Critical Benchmarks (2 pages)

Case Name	Enrichment (wt% ^{235}U)	Number of Fuel Rods	Square Pitch (cm)	H/X Atom Ratio	H_2O/X Volume Ratio	E Loss/ W Loss	ppm Boron	Calculated k_{eff}	Relative Error (1σ)
G01	2.35	1170	1.684	2.190E+02	1.600E+00	1.717E-01	0	0.99219	0.00031
G02	2.35	1170	1.684	2.190E+02	1.600E+00	1.697E-01	0	0.99243	0.00028
G03	2.35	1170	1.684	2.190E+02	1.600E+00	1.683E-01	0	0.99498	0.00031
G04	2.35	1170	1.684	2.190E+02	1.600E+00	1.686E-01	0	0.99383	0.00031
G05	2.35	1170	1.684	2.190E+02	1.600E+00	1.675E-01	0	0.99398	0.00030
G06	2.35	1170	1.684	2.190E+02	1.600E+00	1.670E-01	0	0.99307	0.00030
G07	4.31	576	1.892	1.048E+02	1.598E+00	1.814E-01	0	0.99963	0.00035
G08	4.31	576	1.892	1.048E+02	1.598E+00	1.784E-01	0	1.00040	0.00034
G09	4.31	576	1.892	1.048E+02	1.598E+00	1.771E-01	0	0.99956	0.00035
G10	4.31	576	1.892	1.048E+02	1.598E+00	1.768E-01	0	0.99930	0.00032
G11	2.35	1170	1.684	2.190E+02	1.600E+00	1.692E-01	0	0.99333	0.00032
G12	4.31	576	1.892	1.048E+02	1.598E+00	1.783E-01	0	1.00015	0.00034
G13	2.35	1134	1.684	2.190E+02	1.600E+00	1.694E-01	0	0.99837	0.00031
G14	2.35	1134	1.684	2.190E+02	1.600E+00	1.673E-01	0	0.99938	0.00032
G15	2.35	1134	1.684	2.190E+02	1.600E+00	1.654E-01	0	0.99646	0.00032
G16	4.31	576	1.892	1.048E+02	1.598E+00	1.796E-01	0	1.00553	0.00033
G17	4.31	576	1.892	1.048E+02	1.598E+00	1.782E-01	0	1.00442	0.00035
G18	4.31	576	1.892	1.048E+02	1.598E+00	1.756E-01	0	1.00408	0.00032

Table 6.4-11 - USL Equation Parameter Values

Equation Parameter	System Physical Parameter				
(Equations 6-1 through 6-6)	Square Pitch (cm)	H/X Atom Ratio	H_2O/X Volume Ratio	E_{Loss}/W_{Loss}	ppm Boron
N	38	38	38	38	38
<x>	1.6976	160.7	1.533	0.1941	528.8
<k>	0.99733	0.99733	0.99733	0.99733	0.99733
$s_{kc(x)}^2$	7.051×10^{-6}	9.684×10^{-6}	1.169×10^{-5}	1.165×10^{-5}	1.169×10^{-5}
s_w^2	1.010×10^{-7}	1.010×10^{-7}	1.010×10^{-7}	1.010×10^{-7}	1.010×10^{-7}
s_p	2.674×10^{-3}	3.128×10^{-3}	3.434×10^{-3}	3.428×10^{-3}	3.433×10^{-3}
Student-t (n-2)	1.6892	1.6892	1.6892	1.6892	1.6892
x_{min}	1.511	71.49	1.090	0.1540	0
x_{max}	1.905	329.7	2.408	0.2724	3389
$w(x)_{xmin}$	4.706×10^{-3}	5.476×10^{-3}	6.044×10^{-3}	5.974×10^{-3}	5.904×10^{-3}
$w(x)_{xmax}$	4.736×10^{-3}	5.784×10^{-3}	6.505×10^{-3}	6.271×10^{-3}	6.650×10^{-3}
W	4.736×10^{-3}	5.784×10^{-3}	6.505×10^{-3}	6.271×10^{-3}	6.650×10^{-3}
$k_c(x)$	$0.9687 + 1.668 \times 10^{-2} (x)$	$1.001 - 2.097 \times 10^{-5} (x)$	$0.9966 + 4.553 \times 10^{-4} (x)$	$0.9959 + 7.592 \times 10^{-3} (x)$	$0.9972 + 1.853 \times 10^{-7} (x)$
$\beta(x)$	$-3.132 \times 10^{-2} + 1.668 \times 10^{-2} (x)$	$6.974 \times 10^{-4} - 2.097 \times 10^{-5} (x)$	$-3.369 \times 10^{-3} + 4.553 \times 10^{-4} (x)$	$-4.145 \times 10^{-3} + 7.592 \times 10^{-3} (x)$	$-2.770 \times 10^{-3} + 1.853 \times 10^{-7} (x)$
Δk_m	0.05	0.05	0.05	0.05	0.05

Table 6.4-12 - USL Functions Applicable to the VSC-24 System (2 pages)

x Parameter Description	Upper Subcritical Limit Functions	*x* Parameter Range of Applicability
Pitch	$USL(x) = (9.187E-01) + (1.688E-02)(x) - (4.517E-03)\left[(1.026E+00) + \frac{(x-(1.698E+00))^2}{(5.893E-01)}\right]^{1/2}$	for $x < 1.511E+00$
	$USL(x) = (9.139E-01) + (1.688E-02)(x)$	for $1.511E+00 \leq x < 1.856E+00$
	$USL(x) = (9.453E-01)$	for $1.856E+00 \leq x \leq 1.905E+00$
	$USL(x) = (9.500E-01) - (4.517E-03)\left[(1.026E+00) + \frac{(x-(1.698E+00))^2}{(5.893E-01)}\right]^{1/2}$	for $x > 1.905E+00$
H/X Atom Ratio [1]	$USL(x) = (9.500E-01) - (5.284E-03)\left[(1.026E+00) + \frac{(x-(1.606E+02))^2}{(1.662E+05)}\right]^{1/2}$	for $x \leq 3.326E+01$
	$USL(x) = (9.507E-01) - (2.097E-05)(x) - (5.284E-03)\left[(1.026E+00) + \frac{(x-(1.606E+02))^2}{(1.662E+05)}\right]^{1/2}$	for $3.326E+01 < x < 7.149E+01$ and for $x > 3.297E+02$
	$USL(x) = (9.449E-01) - (2.097E-05)(x)$	for $7.149E+01 \leq x \leq 3.297E+02$
H_2O/X Vol Ratio [1]	$USL(x) = (9.466E-01) + (4.553E-04)(x) - (5.802E-03)\left[(1.026E+00) + \frac{(x-(1.533E+00))^2}{(3.321E+00)}\right]^{1/2}$	for $2.408E+00 < x < 7.401E+00$ and for $x < 1.090E+00$

Table 6.4-12 - USL Functions Applicable to the VSC-24 System (2 pages)

x Parameter Description	Upper Subcritical Limit Functions	x Parameter Range of Applicability
H_2O/X Vol Ratio [1] (continued)	$USL(x)=(9.401E-01)+(4.553E-04)(x)$	for $1.090E+00 \le x \le 2.408E+00$
	$USL(x)=(9.500E-01)-(5.802E-03)\left[(1.026E+00)+\frac{(x-(1.533E+00))^2}{(3.321E+00)}\right]^{1/2}$	for $x \ge 7.401E+00$
E_{Loss} / W_{Loss}	$USL(x)=(9.459E-01)+(7.592E-03)(x)-(5.791E-03)\left[(1.026E+00)+\frac{(x-(1.941E-01))^2}{(4.178E-02)}\right]^{1/2}$	for $2.724E-01 < x < 5.460E-01$ and for $x < 1.540E-01$
	$USL(x)=(9.396E-01)+(7.592E-03)(x)$	for $1.540E-01 \le x \le 2.724E-01$
	$USL(x)=(9.500E-01)-(5.791E-03)\left[(1.026E+00)+\frac{(x-(1.941E-01))^2}{(4.178E-02)}\right]^{1/2}$	for $x \ge 5.460E-01$
ppmb	$USL(x)=(9.406E-01)+(1.853E-07)(x)$	For $0 \le x \le 3.389E+03$
	$USL(x)=(9.472E-01)+(1.853E-07)(x)-(5.799E-03)\left[(1.026E+00)+\frac{(x-(5.288E+02))^2}{(2.837E+07)}\right]^{1/2}$	for $3.389E+03 < x < 1.495E+04$
	$USL(x)=(9.500E-01)-(5.799E-03)\left[(1.026E+00)+\frac{(x-(5.288E+02))^2}{(2.837E+07)}\right]^{1/2}$	for $x \ge 1.495E+04$

Note:

1. These values are calculated based on one (fuel rod) unit cell of the assembly lattice, and are not averaged over the array guide tube locations.

Table 6.4-13 - Range of Physical Parameter Values for VSC-24 Criticality Evaluations

System Physical Parameter	Range of Values	
	Minimum	Maximum
Pitch (cm)	1.25984	1.44272
H/X Atom Ratio	39.29	133.4
H_2O/X Volume Ratio	1.795	2.153
E_{Loss} / W_{Loss}	0.2134	0.3096
ppmb	2325	5900

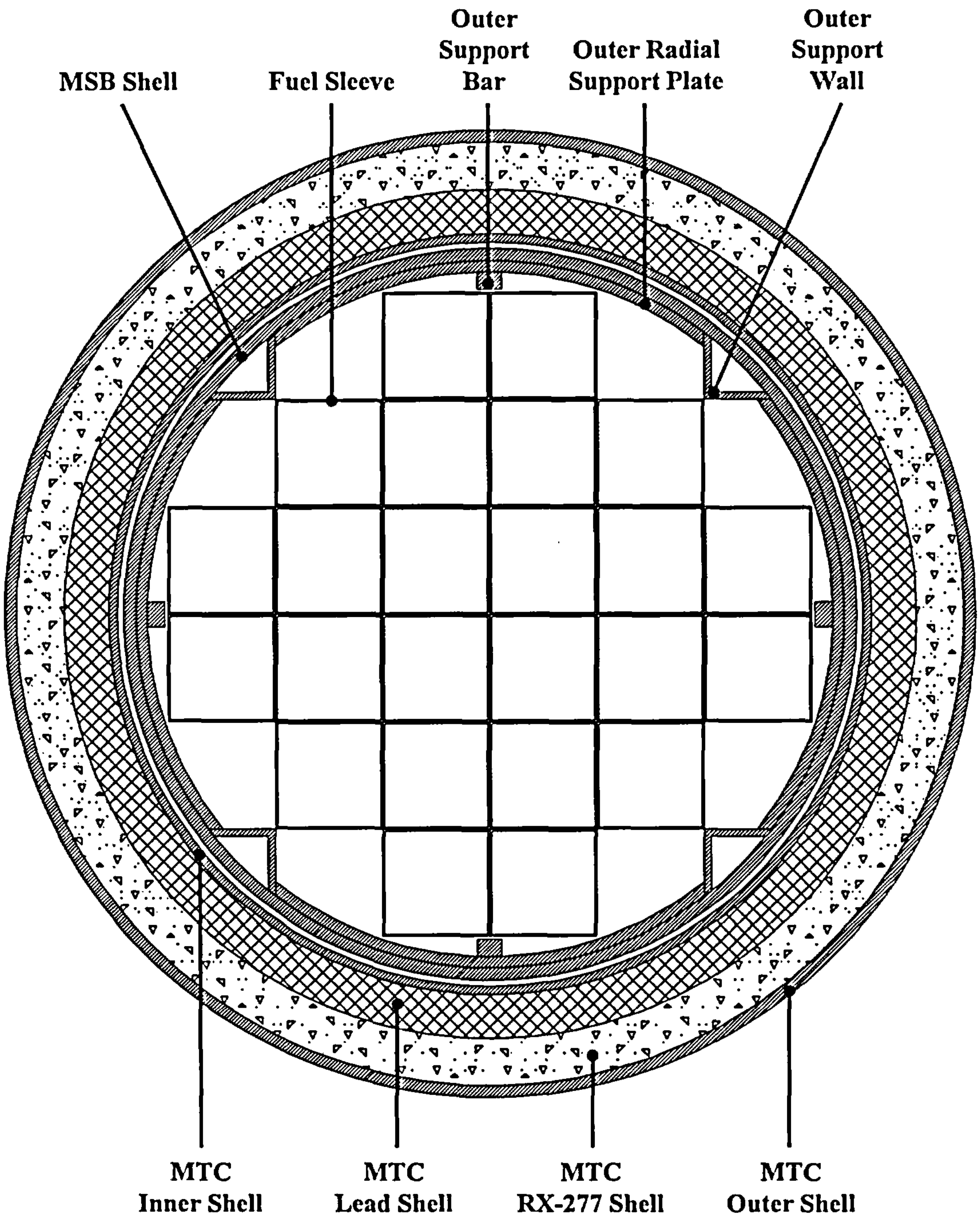
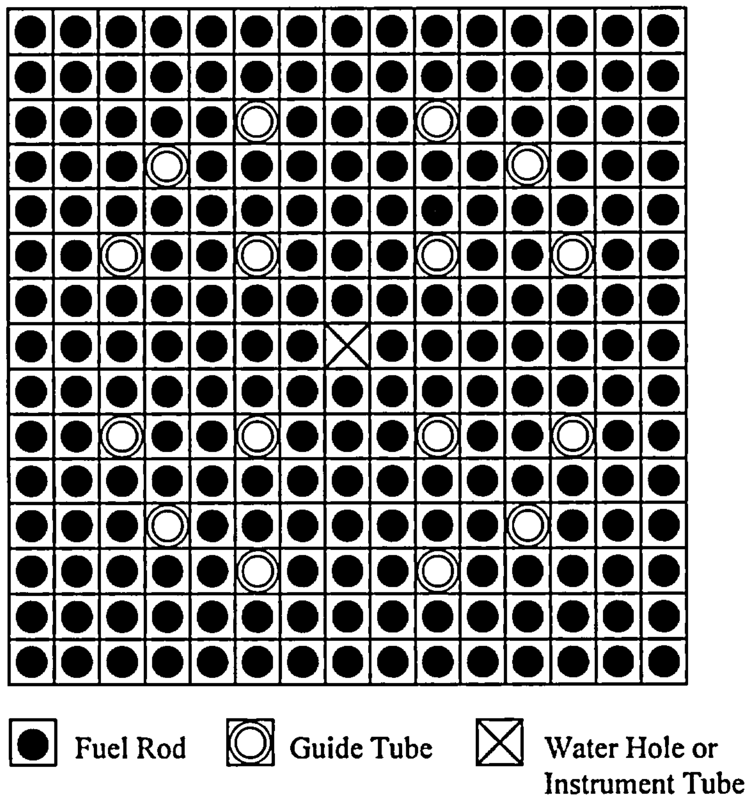


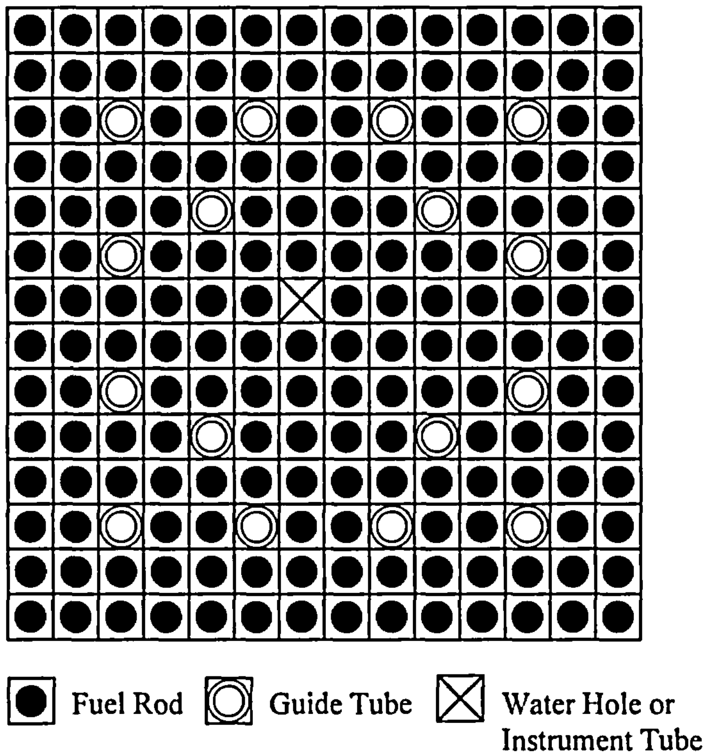
Figure 6.1-1 - Full Symmetry Horizontal Cross-Sectional View of MSB Inside MTC



Notes:

1. The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.
2. The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

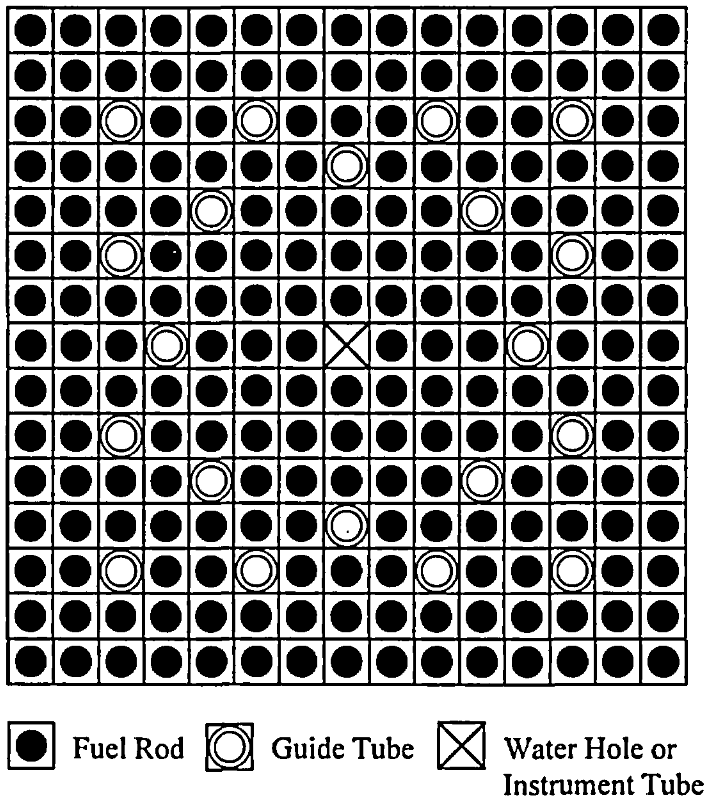
Figure 6.2-1 - B&W 15x15 Assembly Class Lattice Layout



Notes:

1. The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.
2. The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

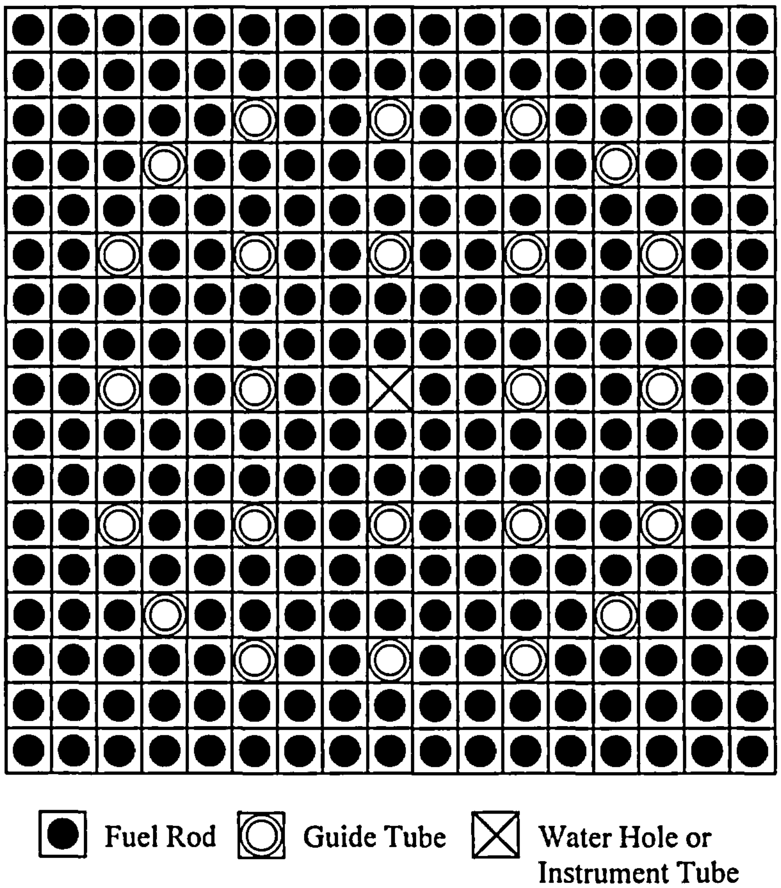
Figure 6.2-2 - W 14x14 Assembly Class Lattice Layout



Notes:

1. Although control component rodlets are (conservatively) modeled in the criticality evaluation, inserted control components are not allowed for the W 15x15 assembly, as shown in Table 6.2-1.
2. The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

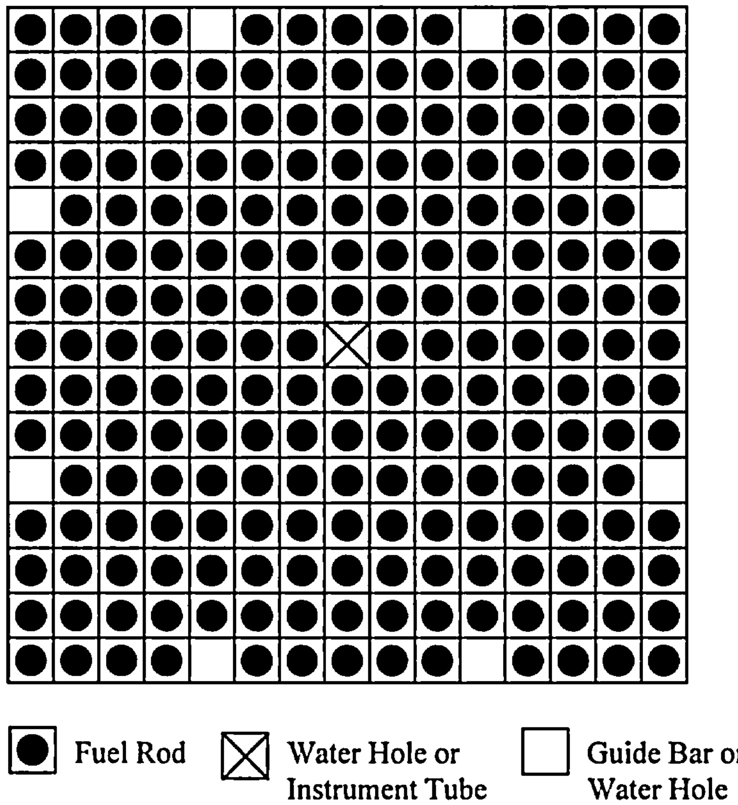
Figure 6.2-3 - W 15x15 Assembly Class Lattice Layout



Notes:

1. The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.
2. The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

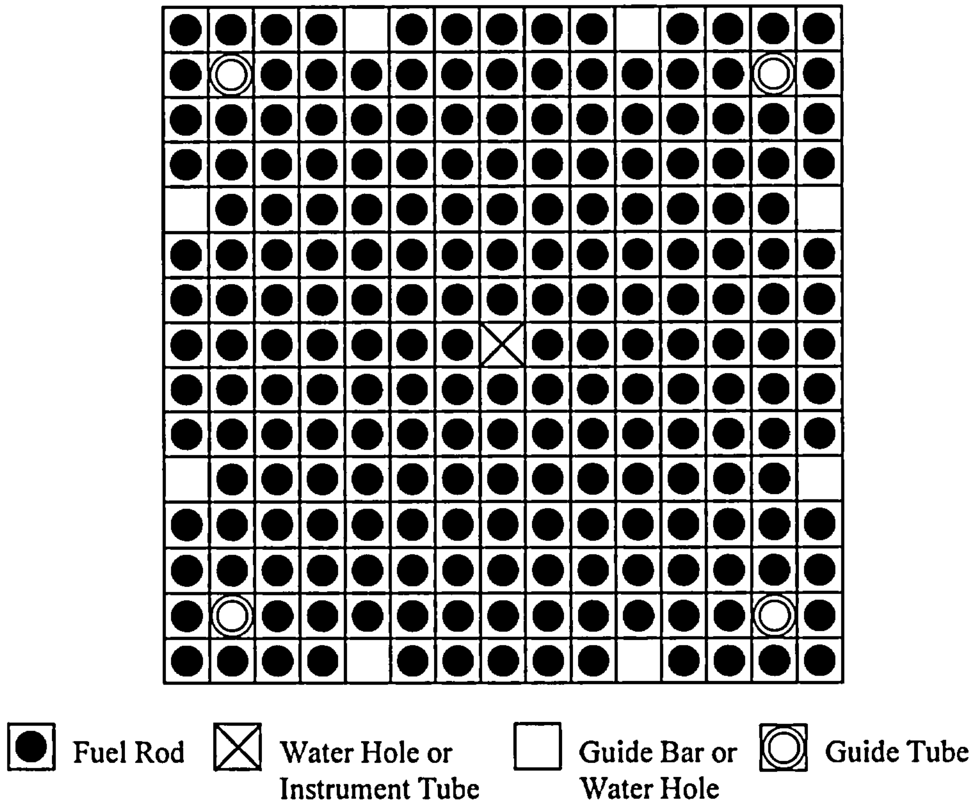
Figure 6.2-4 - W 17x17 Assembly Class Lattice Layout



Notes:

1. The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.
2. The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

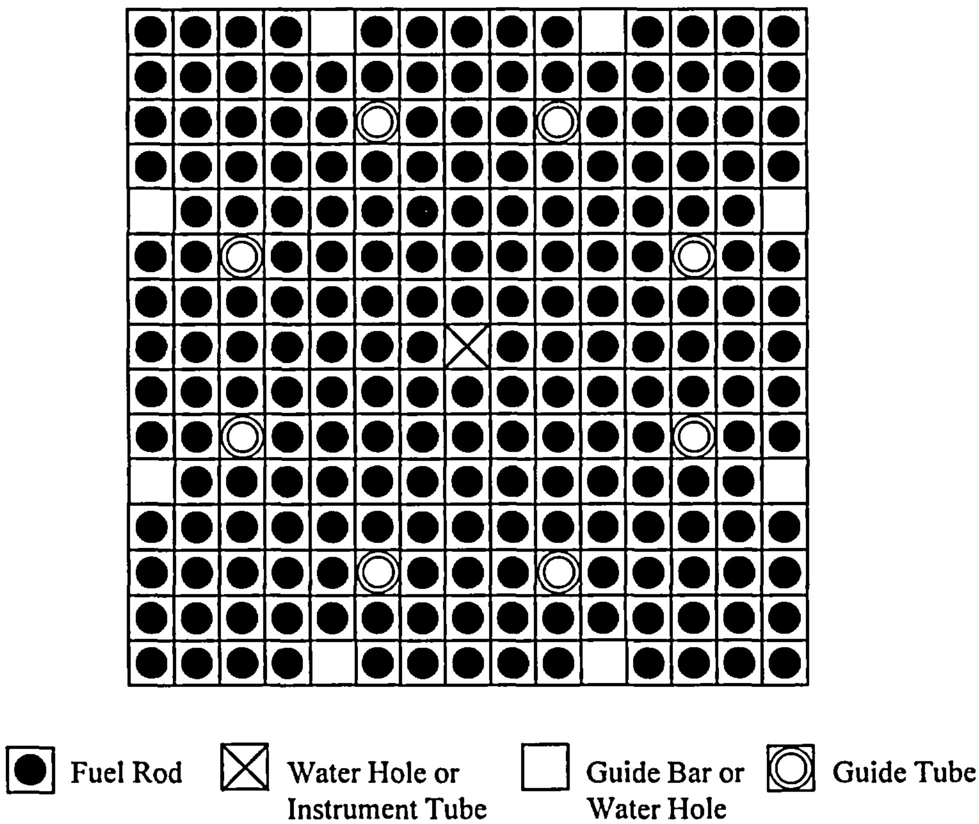
Figure 6.2-5 - CE 15x15A Assembly Class Lattice Layout



Notes:

1. The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.
2. The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.
3. The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

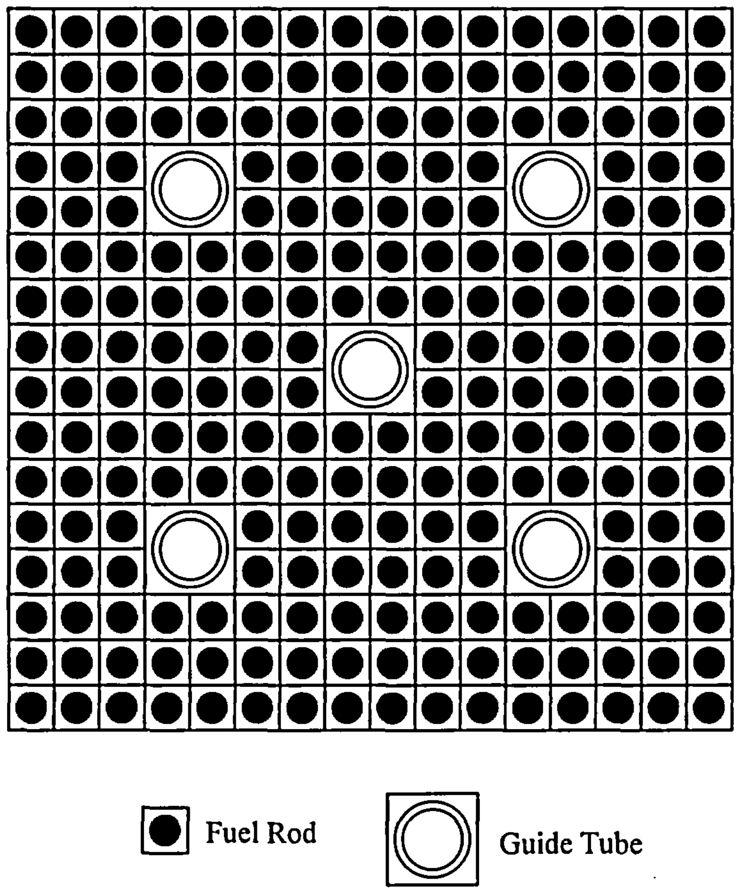
Figure 6.2-6 - CE 15x15B Assembly Class Lattice Layout



Notes:

1. The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.
2. The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.
3. The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Figure 6.2-7 - CE 15x15C Assembly Class Lattice Layout



Note:

1. The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Figure 6.2-8 - CE 16x16 Assembly Class Lattice Layout

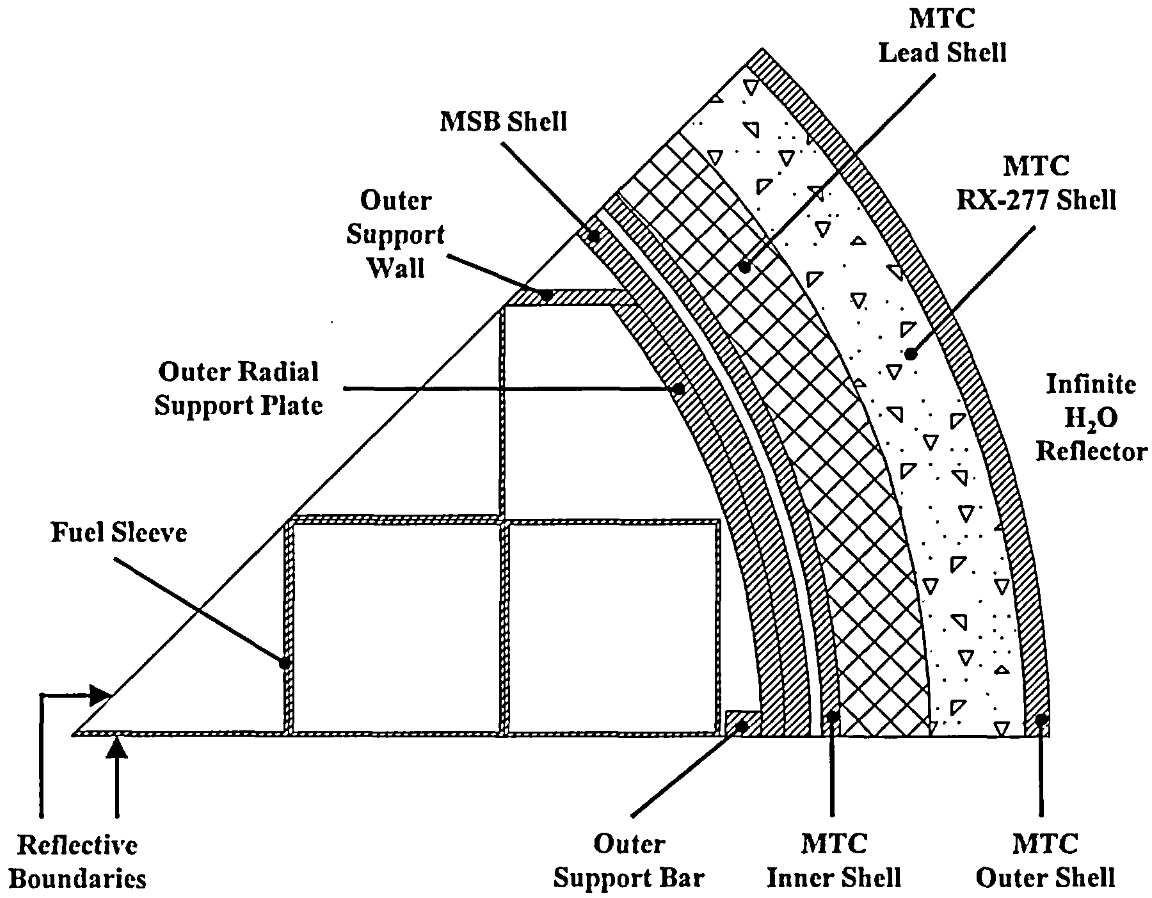
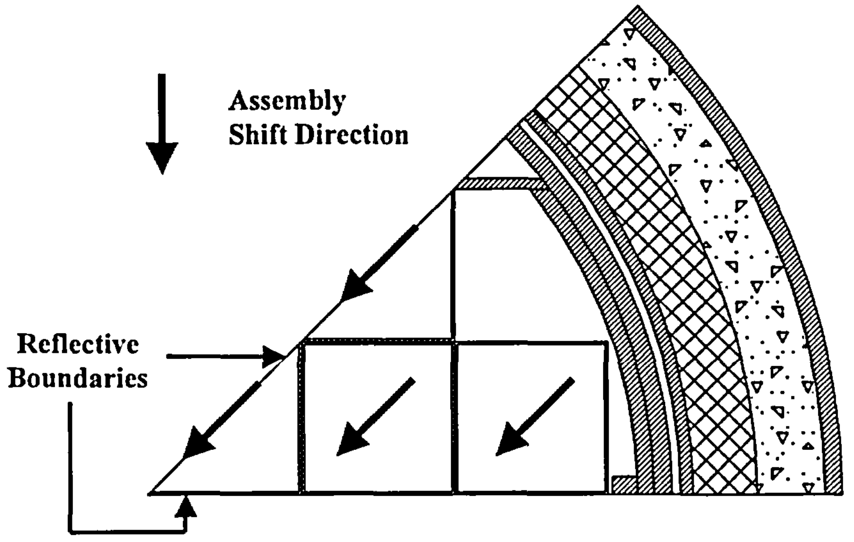
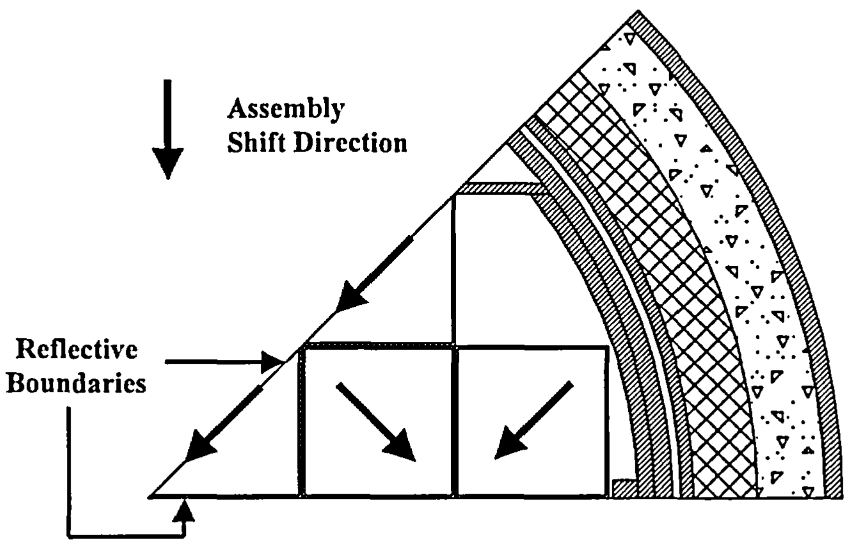


Figure 6.3-1 - 1/8th Symmetry Horizontal Cross-Sectional View of MCNP4A Model



Assemblies Shifted Toward Center of MSB



Assemblies Shifted Into Groups of Four

Note:

1. Also analyzed is the shifting pattern of all assemblies centered in their respective fuel sleeves.

Figure 6.3-2 - Fuel Assembly Shifting Patterns

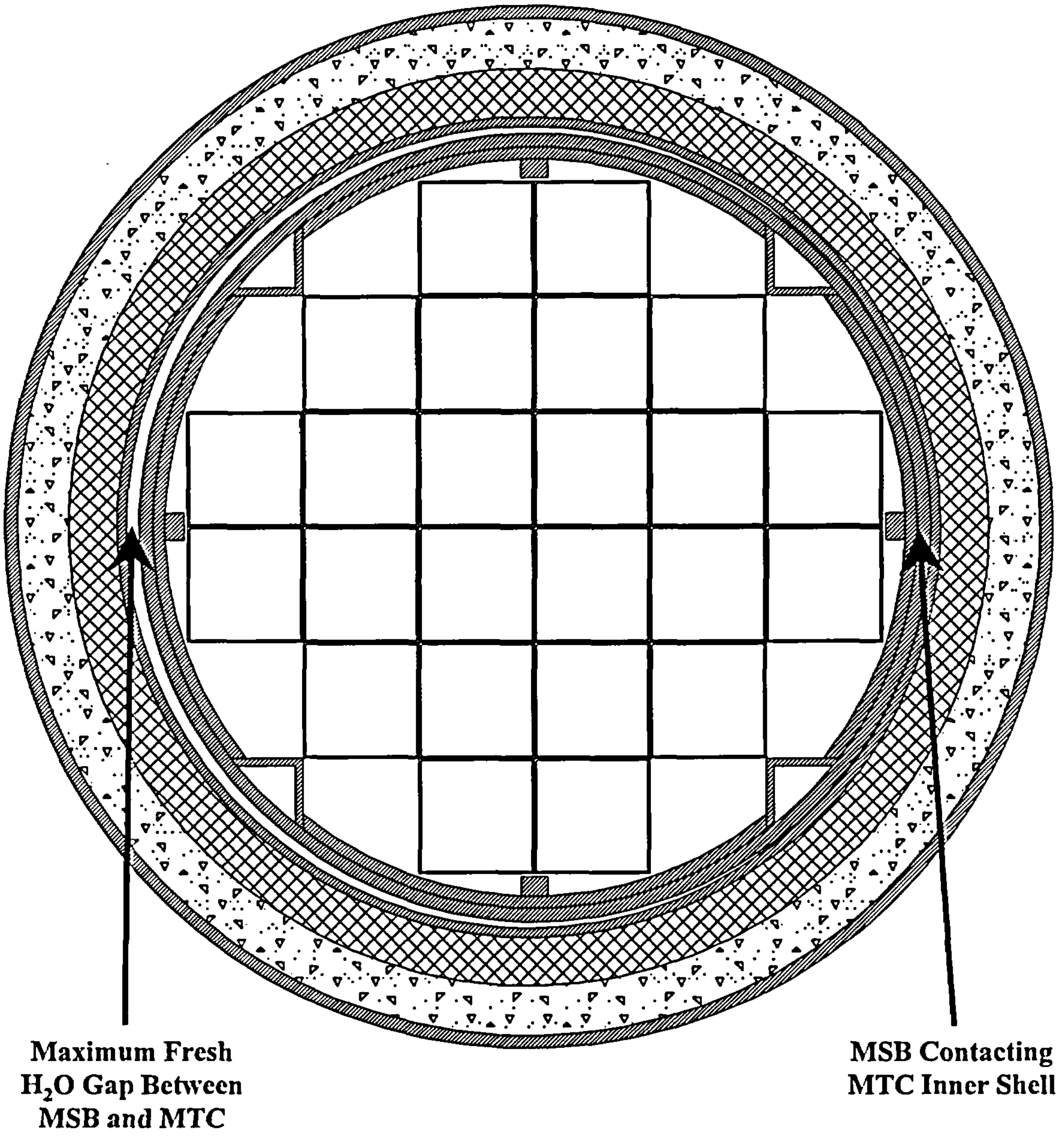


Figure 6.3-3 - Full Symmetry Horizontal Cross-Sectional View of MSB Shifted Inside MTC

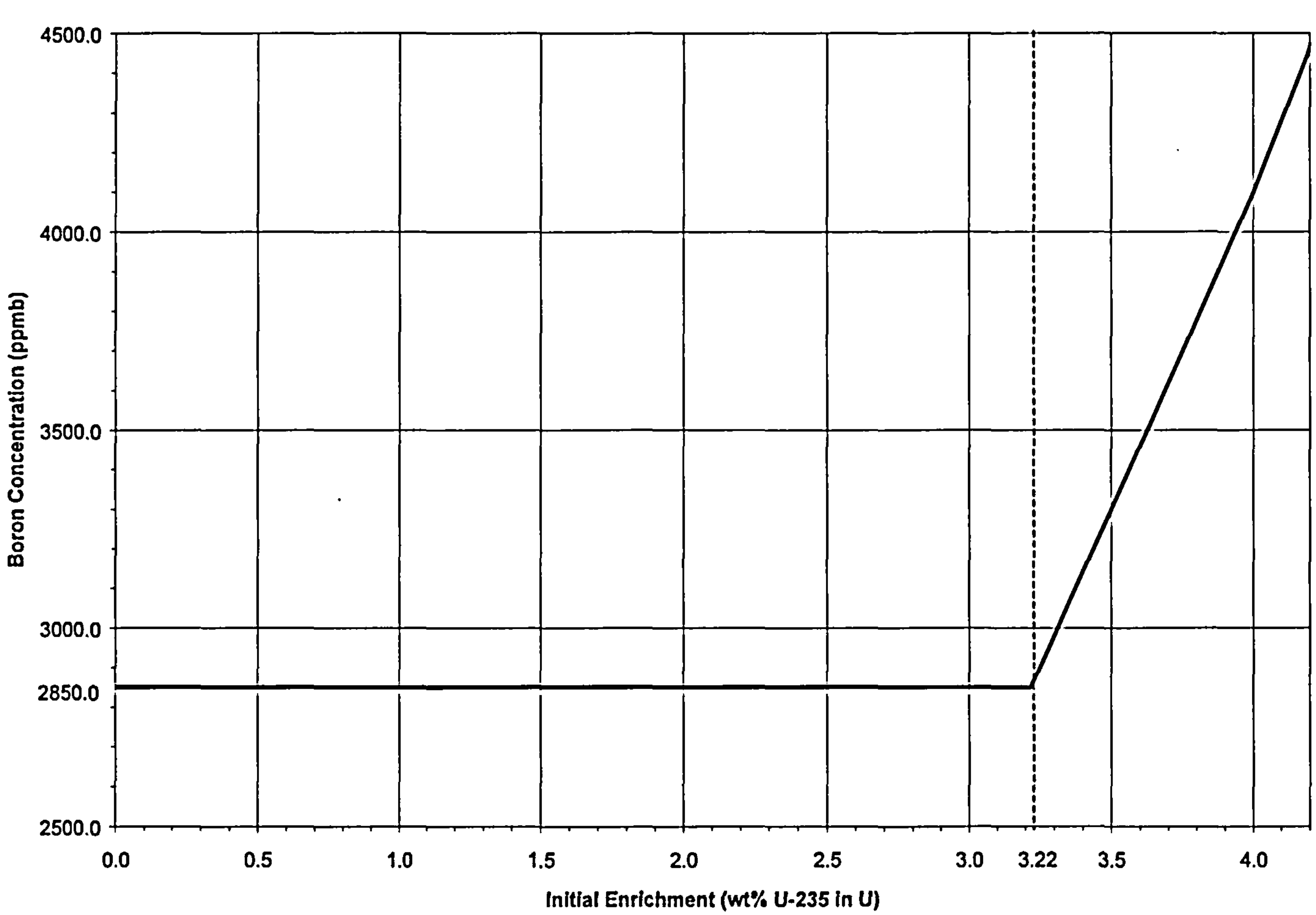


Figure 6.4-1 - B&W 15x15 Assembly Class Minimum Required Soluble Boron Results

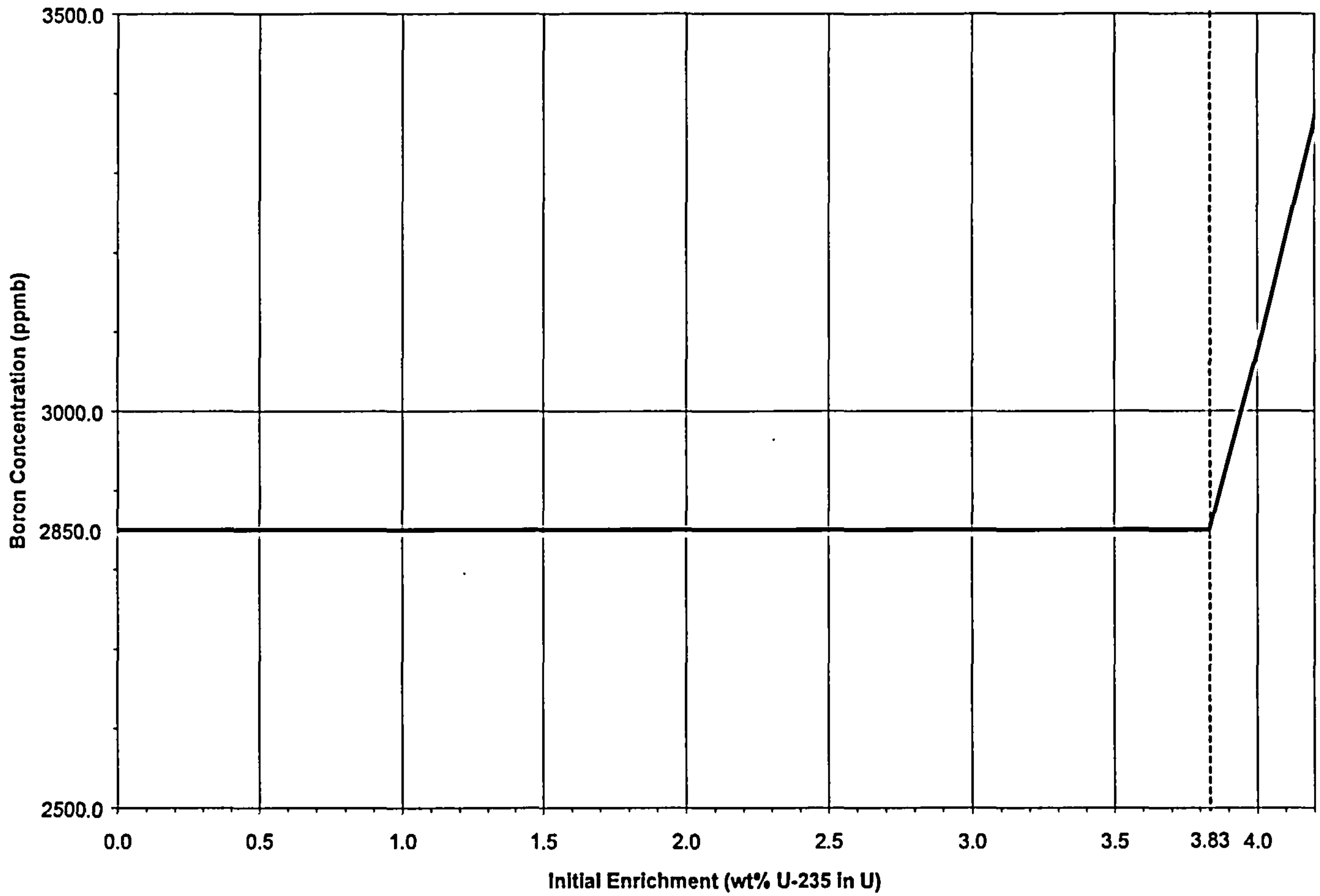


Figure 6.4-2 - W 14x14 Assembly Class Minimum Required Soluble Boron Results

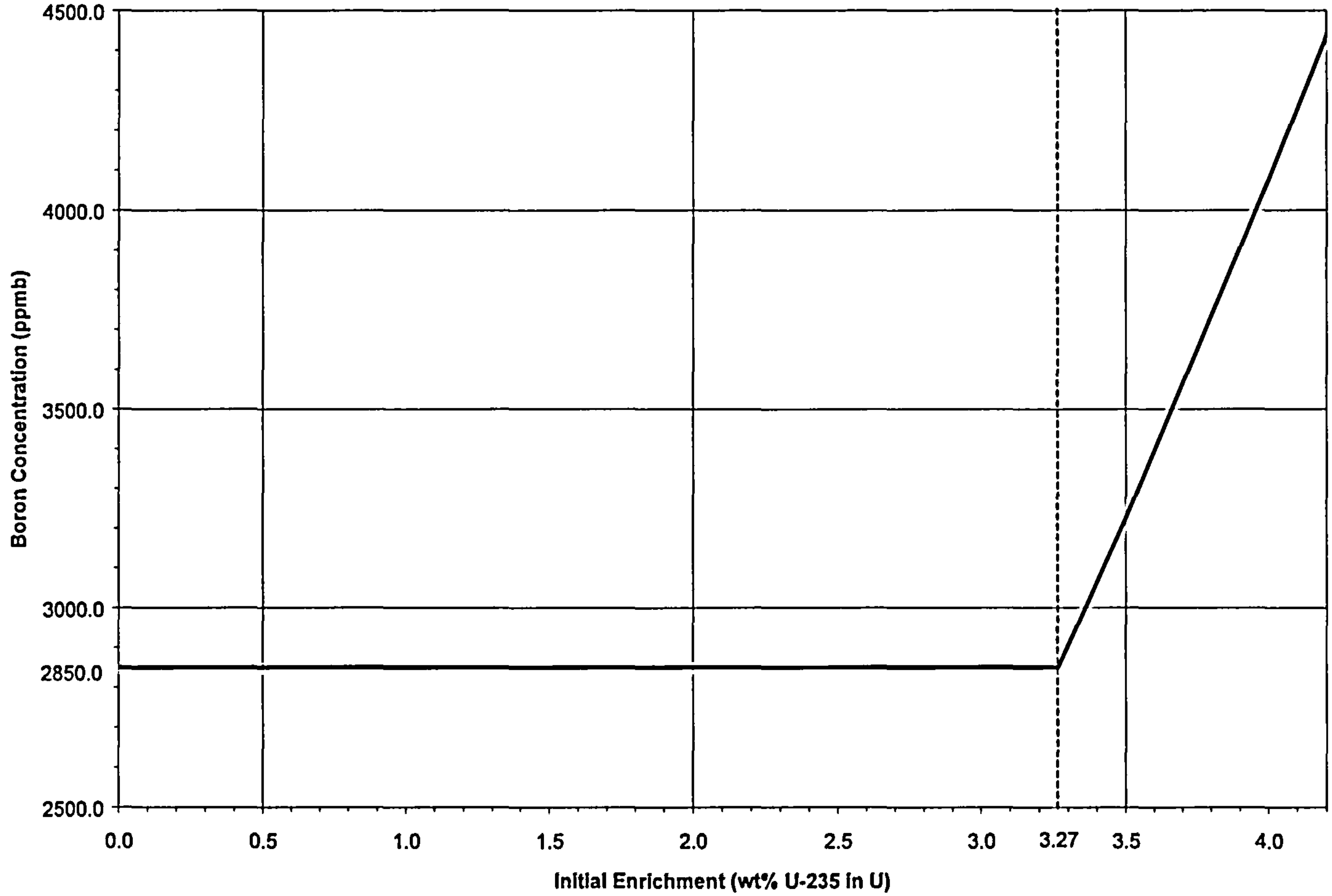


Figure 6.4-3 - W 15x15 Assembly Class Minimum Required Soluble Boron Results

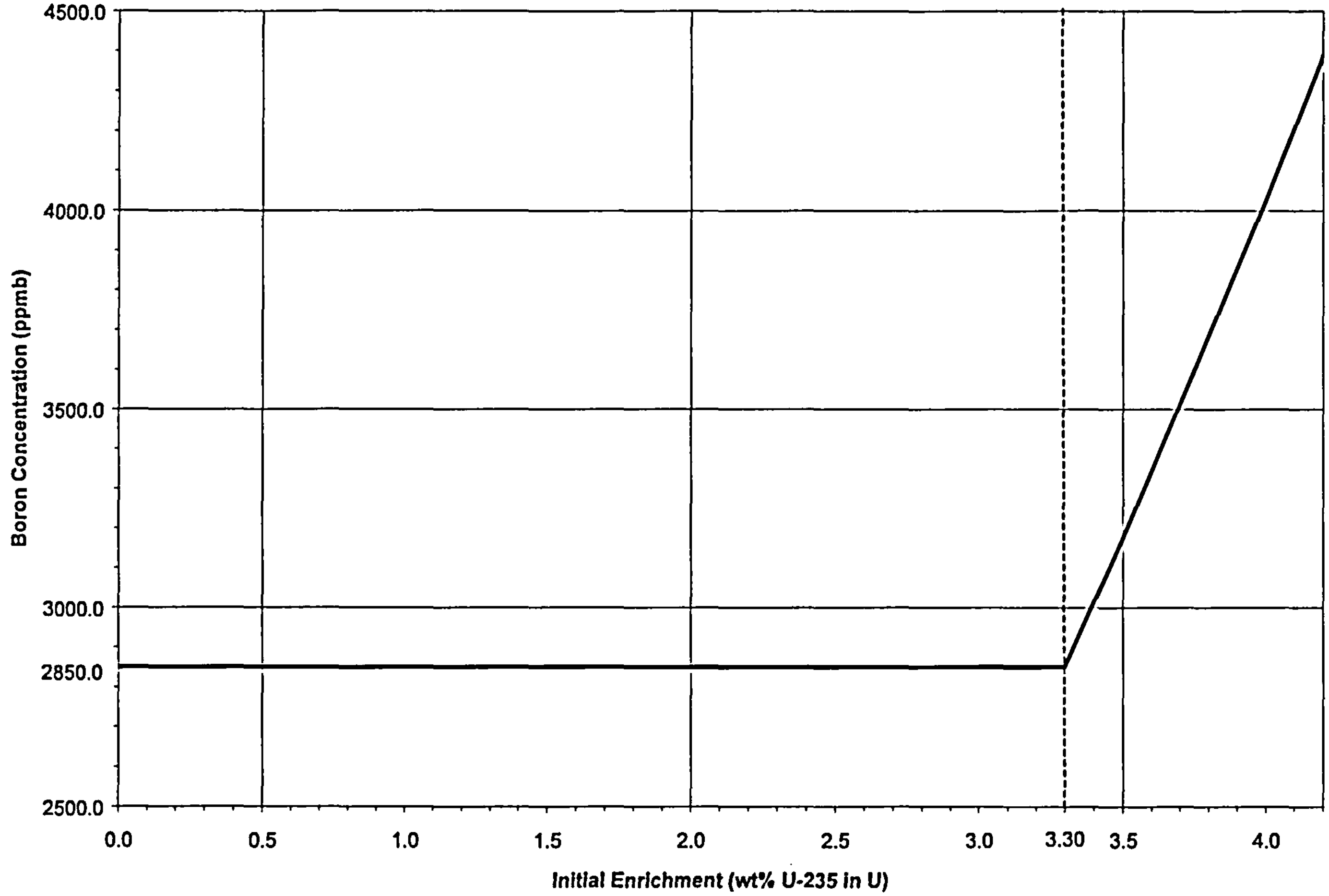


Figure 6.4-4 - W 17x17 Assembly Class Minimum Required Soluble Boron Results

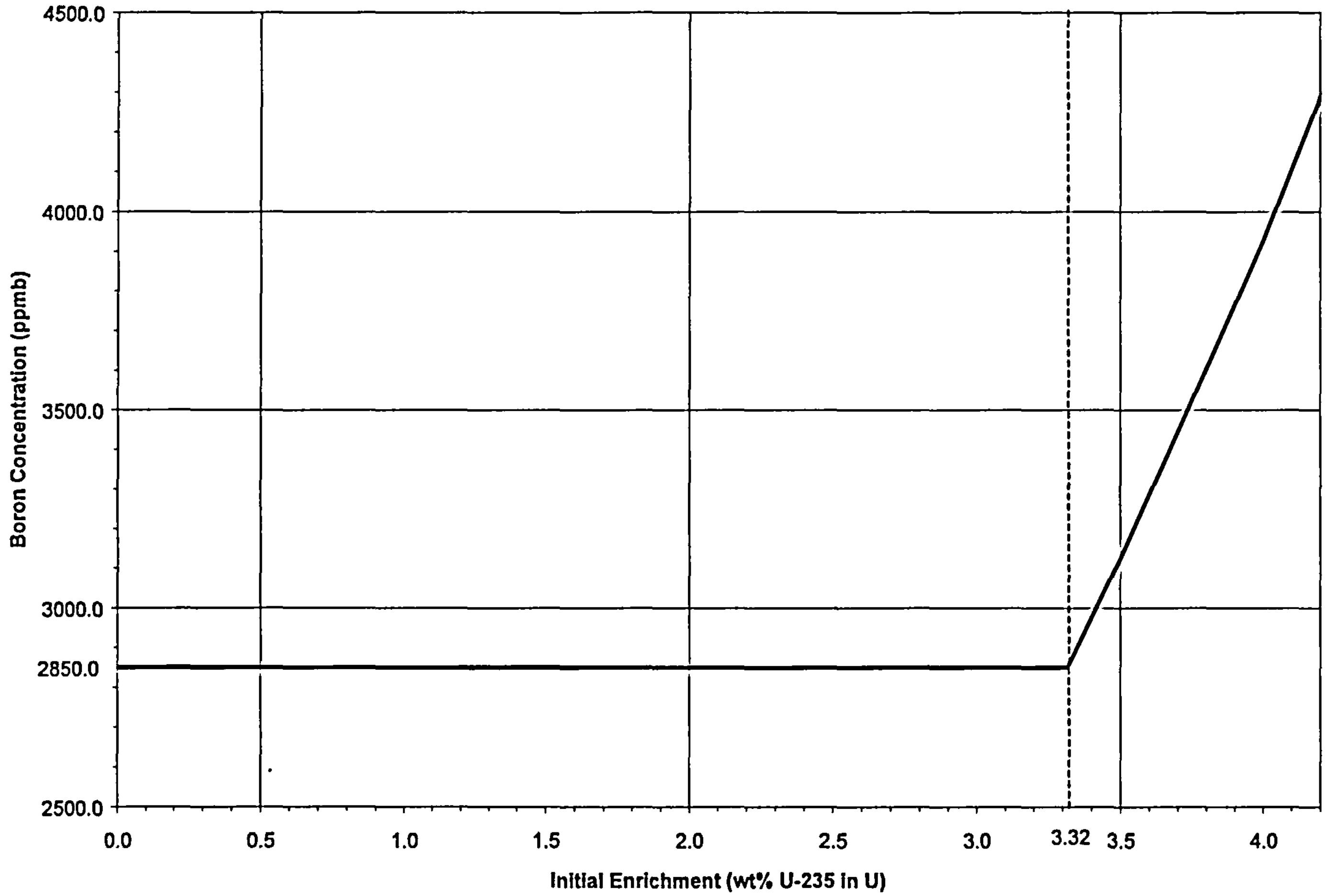


Figure 6.4-5 - CE 15x15A Assembly Class Minimum Required Soluble Boron Results

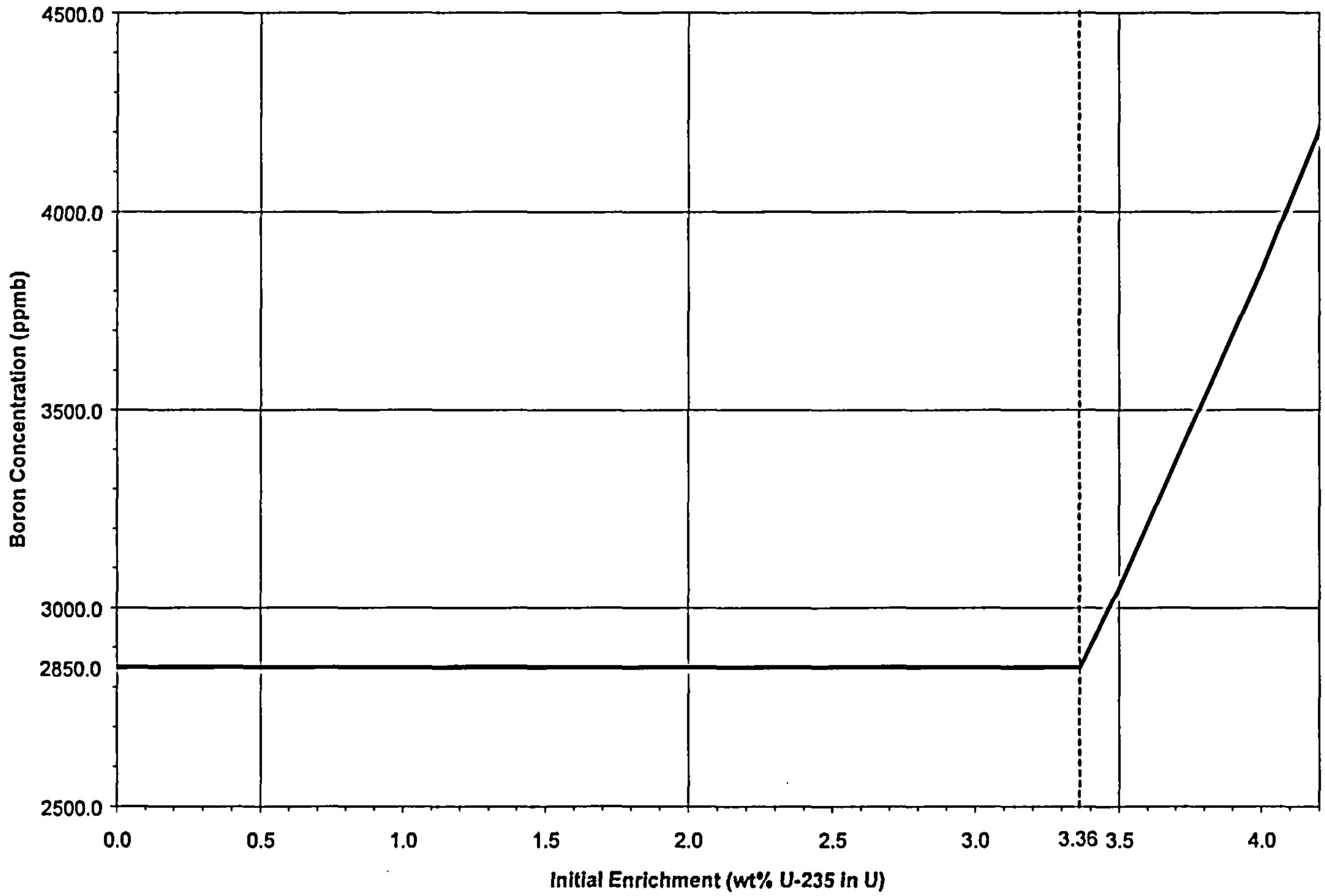


Figure 6.4-6 - CE 15x15B Assembly Class Minimum Required Soluble Boron Results

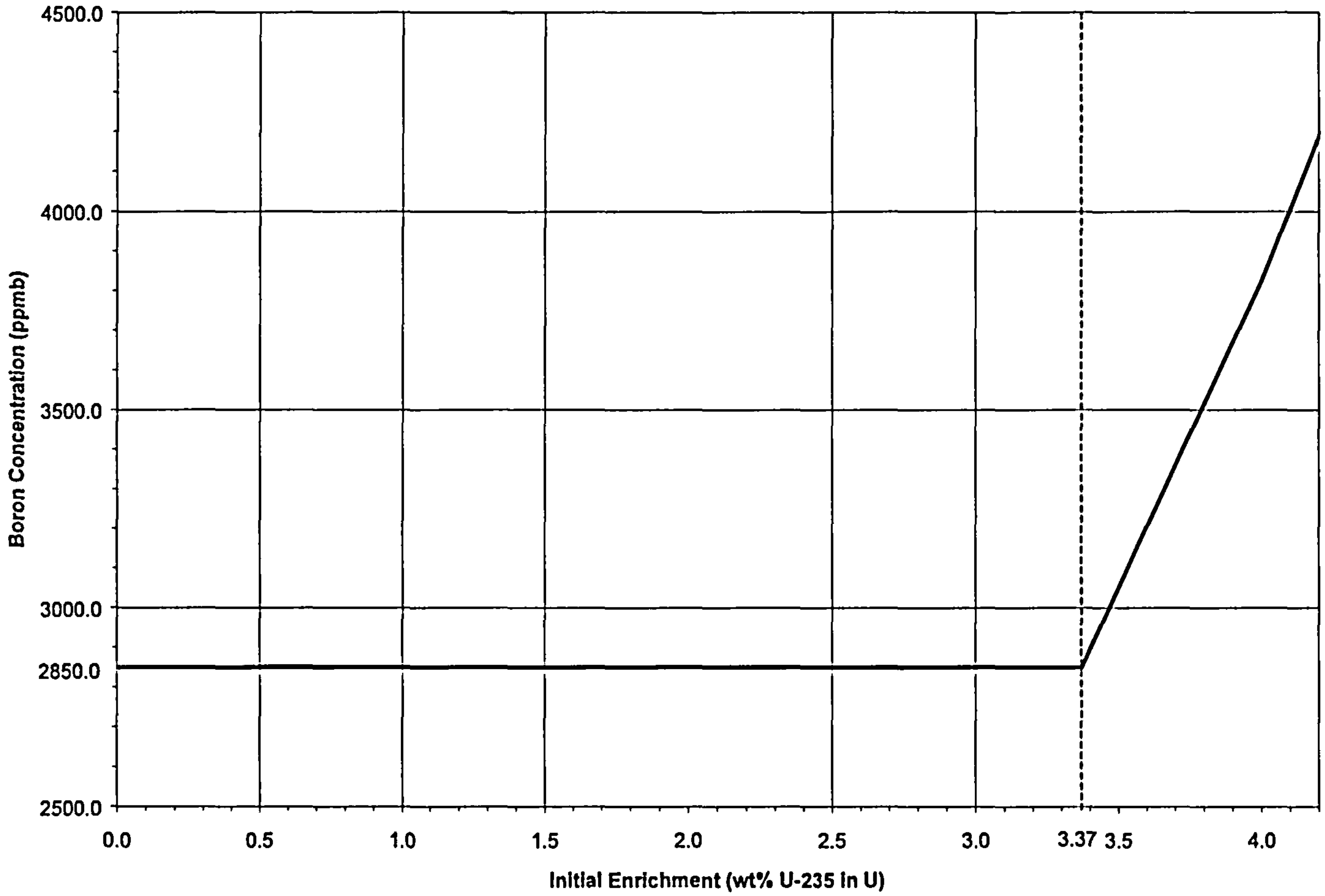


Figure 6.4-7 - CE 15x15C Assembly Class Minimum Required Soluble Boron Results

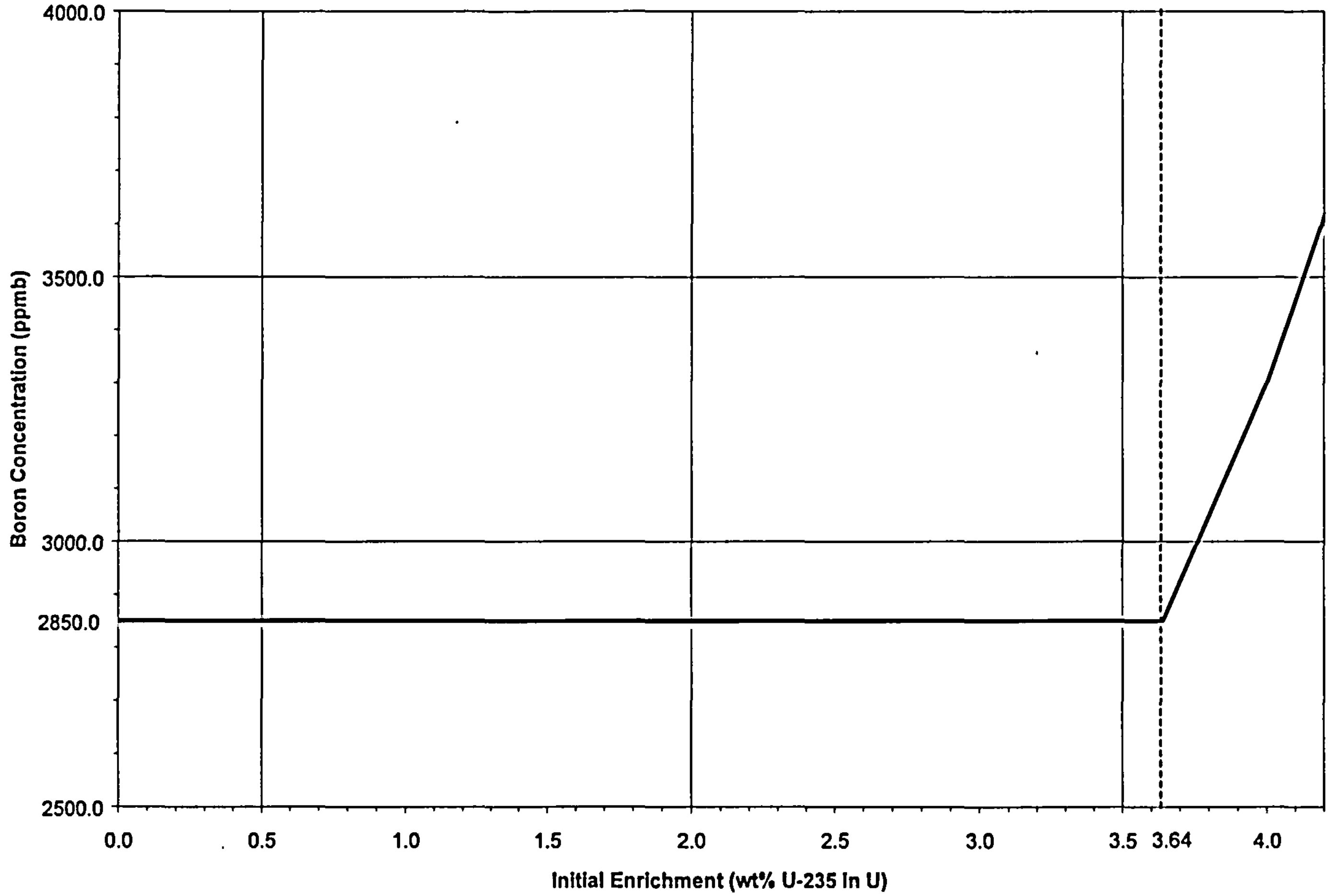


Figure 6.4-8 - CE 16x16 Assembly Class Minimum Required Soluble Boron Results

7.0 CONFINEMENT

7.1 CONFINEMENT BOUNDARY

The following paragraphs define the confinement boundary for the VSC system. The primary confinement boundaries for fission products which are contained in the stored spent fuel are:

1. The fuel cladding.
2. The MSB, which is a welded steel cylinder that is vacuum dried and backfilled with helium.

Only item 2 above is considered in the system analyses and specified as important to safety.

The concrete cask is designed to provide shielding, structural support, ventilation, and weather protection for the MSB, but is not part of the confinement boundary.

7.1.1 CONFINEMENT VESSEL

The MSB sits vertically in the VCC and is cooled by the natural circulation of air through the annulus between the MSB and the VCC. The MSB confinement boundary consists of the following major components:

- Shell
- Bottom Plate
- Structural Lid
- Shield Lid Top Plate
- Structural Lid Port Covers

The MSB is designed to hold 24 Pressurized Water Reactor (PWR) spent fuel assemblies. It consists of an outer shell assembly, a shielding lid, a structural lid, and the fuel basket assembly. Table 1.2-2 summarizes the main physical design parameters.

7.1.2 CONFINEMENT (MSB) PENETRATIONS

The only penetrations required in the MSB are in the MSB structural and shield lid. After fuel insertion, the shield lid is lowered onto the MSB body. The MSB is then removed from the spent fuel pool, and the shield lid is welded to the MSB Shell. Two penetrations for draining, vacuum drying, and backfilling with helium, are located in the shield lid. The vent valve and vent and drain tubes and their connections to the shield lid are helium leak tested during fabrication of the shield lid. The structural lid is placed on the shield lid and also welded to the MSB shell. A penetration through the structural lid allows access to the fittings in the shield lid. Two cover plates that mate with the structural lid are seal welded over the penetrations after the draining drying and helium backfilling operations have been completed. No drains are required in the MSB bottom plate.

No components are required to penetrate the sealed MSB after helium backfilling is completed and the structural lid is welded in place. No penetrations are used during spent fuel storage.

7.1.3 SEALS AND WELDS

7.1.3.1 Fabrication

Cutting, welding, and forming is performed in accordance with Table 1.2-3.

A typical fabrication sequence* is as follows:

1. Cut out and prep shell and bottom plate material
2. Roll MSB shell and weld longitudinally, girth (if necessary), and bottom plate
3. Perform nondestructive testing (radiography) of longitudinal, girth welds
4. Bend, weld, and inspect storage sleeves**
5. Assemble storage sleeves together and weld basket support pieces
6. Inspect storage sleeve assembly
7. Inspect MSB shell assembly
8. Coat MSB shell assembly interior and exterior (leave top interior portion involved in welding uncoated)
9. Coat storage sleeve assembly
10. Insert storage sleeve assembly into shell assembly
11. Weld shield lid support in place
12. Coat remaining areas of MSB shell
13. Cut out and prep shield lid metal pieces and structural lid material
14. Weld shield lid assembly (except for top plate)
15. Pour RX-277 material and bake per fabrication specification
16. Weld shield lid top plate to shield lid assembly
17. Machine shield lid and structural lid based on shell diameter
18. Helium leak test the shield lid penetrations

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* Steps grouped together may be done sequentially.

** To meet straightness tolerance on tubes, straightening of welded tubes may be required.

19. Perform fit up of shield lid (including shims) and structural lid (including backing ring and cover plates)
20. Connect and test fit drain line
21. Test all storage sleeves with dummy assembly/go-no go gauge
22. Perform documentation and final acceptance inspections

7.1.3.2 Welding Specifications

All MSB confinement seals are welded. These are defined as described above. Welding is performed in accordance with Table 1.2-3.

7.1.3.3 Testing, Inspection, and Examination

The following tests will be performed:

1. All MSB parts are dimensionally inspected.
2. MSB welds are tested and examined as described in Table 1.2-3.
3. The Vendor completely assembles the MSB prior to shipping, to ensure that all items specified have been supplied and to test the fit of the shield lid assembly and the structural lid.

7.1.4 CLOSURE

The only closure used for the MSB after the loading of spent fuel is welding of the shield lid, structural lid and the penetration closure cover plates. No closures are by bolts or other mechanisms.

The MSB shield and structural lids will be secured to prevent movement during welding by a combination of large tack welds and implementation of a balanced root weld sequence. Also, manual welding will be utilized for fit-ups greater than $1/16^{th}$ inch. All lid welds will require 200°F preheat. The preheat temperature must be maintained for a minimum of 1 hour after completion of the final weld pass. The 1 hour minimum is measured in the aggregate and is not required to be continuous. Weld consumables with low hydrogen levels (less than 10 ml/H2/STP/100g) will be used for the lid welds.

The MSB pressure boundary shield lid and structural lid closure welds shall be subject to non-destructive examinations in accordance with Chapter 12, Operating Controls and Limits.

7.1.5 CONFINEMENT BOUNDARY MONITORING

As discussed in Section 2.3.3.2, continuous monitoring of the confinement boundary (i.e., the MSB seal welds) is not required. In addition to the 10CFR50 environmental monitoring requirements that already exist for the plant site, periodic monitoring of the cask system is performed to ensure the thermal performance of the cask system, which in turn ensures the continued performance of the MSB confinement boundary.

The cask system thermal performance is verified through daily visual inspections of the storage cask inlet and outlet ducts to detect and prevent any blockage of ventilation duct airflow. These measures will confirm the thermal performance of the cask system, and ensure that system temperatures remain under their design limits.

7.2 REQUIREMENTS FOR NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE

Chapter 2 provides the criteria that define the normal storage conditions for the cask. Chapter 3 provides the structural evaluation which demonstrates the integrity of the MSB and VCC under normal conditions.

7.2.1 RELEASE OF RADIOACTIVE MATERIAL

The effects of potential radioisotope releases from the MSB under normal and off-normal conditions are evaluated. Annual doses from the atmospheric release of radioisotopes from the MSB interior are calculated as a function of distance from the cask.

These analyses consider all fission products that constitute greater than 0.1% of the total curie inventory and all actinide isotopes that constitute greater than 0.01% of the total curie inventory. These isotopes include gases, volatiles, and fuel fines. These analyses also specifically include iodine and the CRUD on the surface of the fuel rods. Table 7.2-1 presents the set of isotopes, determined using the criteria given above, that are considered in the atmospheric release calculations.

The activity level of each isotope within the entire canister interior is determined using the ORIGEN2 point depletion code, based upon a bounding set of PWR fuel parameters — burnup, initial enrichment, and cooling time — that produces maximum isotope activity levels. The calculations are based upon a burnup level of 52 GWd/MTU, well over the maximum allowable burnup level for the VSC-24 cask system. The minimum allowed cooling time of five years is also assumed. A fuel initial enrichment range of 2.75-5.0 wt% U-235 is considered. Dose calculations are performed for all enrichments within this range, at 0.25wt% intervals. The highest dose results, i.e., the "worst enrichment" results, are presented in this section.

For information purposes, a second set of atmospheric release dose calculations are performed for "typical" PWR fuel parameters. These analyses assume isotope activities calculated (using ORIGEN2) based upon 35 GWd/MTU PWR fuel that has been cooled for 10 years. As with the bounding case analyses, the enrichment range between 2.75 and 5.0 wt% ^{235}U is analyzed, and the maximum doses calculated for any of the enrichment levels within the range are presented. Specific VSC-24 cask system users may perform site-specific atmospheric release dose calculations based upon the actual PWR fuel assembly inventory within their loaded casks. The typical case analyses are presented herein to provide an estimate of the atmospheric release doses, versus distance.

The atmospheric release dose calculations also consider a bounding PWR assembly fuel rod surface CRUD activity level of 140 $\mu Ci/cm^2$ at assembly discharge. These surface activity levels are reduced to account for the 5-year-minimum assembly cooling time (a 10-year cooling time for the typical case). An upper bound PWR assembly fuel rod surface area of 350,000 cm^2/assembly is assumed for determining the overall CRUD activity level (in Ci) inside the MSB.

The atmospheric release calculations are performed for normal and off-normal cask conditions. The analyses assume fuel rod failure fractions of 1% and 10% for normal and off-normal conditions, respectively. For nuclides other than CRUD, only the activities present in failed rods are available for release into the canister interior.

For fission gases, 30% of the material is assumed to escape the fuel pellet matrix and become available for release if the fuel rod is failed. The corresponding fuel pellet release fractions for volatiles and fuel fines are 2.0×10^{-4} and 3.0×10^{-5}, respectively. For both normal and off-normal conditions, 15% of the CRUD on the fuel rod surfaces is assumed to spall off and become suspended in the canister interior volume. Of the fuel fines that enter the canister interior after escaping the fuel pellet matrix inside a failed rod, 10% are assumed to be available for release (Reference 7.2). This is due to settling and plateout of the fuel fine materials within the canister interior.

Based upon the above assumptions, an activity density (in Ci/cm^3) that is available for release is determined for the canister interior volume for normal and off-normal conditions.

Canister interior volumetric leakage rates (in cm^3/sec) for normal and off-normal conditions are determined based upon the canister test condition leakage rate and the interior pressures and temperatures that occur for each set of conditions. The MSB test condition leak rate specified by the *technical specifications* is 1.0×10^{-4} std-cm^3/sec. The MSB internal pressure is less than 5.5 psig and 11.7 psig for normal and off-normal conditions, respectively. The MSB interior bulk gas temperatures are 439°F and 445°F for normal and off-normal conditions, respectively. Based upon these parameters, the calculated MSB internal volumetric leak rates are less than 6.48×10^{-5} cm^3/sec and 1.17×10^{-4} cm^3/sec for normal and off-normal conditions, respectively.

These leakage rates are multiplied by the normal condition event duration (one year) to yield total leaked volumes. These leaked volumes are divided by the total canister interior volume to yield the volume fraction that has leaked over the event duration. The overall activities, for each isotope, that are available for release from the canister interior volume are multiplied by these leaked volume fractions to yield leaked isotope activities.

After the activity leaked from the canister is determined for each isotope, atmospheric dispersion calculations are performed to determine the curie density (in Ci-sec/m^3) for each isotope as a function of distance from the cask. These calculations are performed using atmospheric dispersion (χ/Q) factors, which are based upon neutral (D-stability diffusion) atmospheric conditions and an assumed wind speed of 5 m/s. Atmospheric dispersion factors for these atmospheric conditions are taken from NRC Regulatory Guide 1.145 (Reference 7.3) and are listed in Table 7.2-2 for both normal and off-normal conditions.

Based upon the isotope activity densities (in Ci-sec/m^3) calculated above, submersion and inhalation doses are calculated using dose conversion factors from EPA Federal Guidance Report No. 11 (Reference 7.4). Doses are calculated for all individual organs, as well as for the thyroid and the effective whole body. The activity densities (in Ci-sec/m^3) are multiplied by a breathing rate (in m^3/sec), and then multiplied by the inhalation dose conversion factors (in mrem/Ci) to yield final inhalation doses (in mrem). For submersion, the activity densities (in Ci-sec/m^3) are multiplied directly by the submersion dose conversion factors (in mrem-m^3/Ci-sec) to yield submersion doses (in mrem). The bounding case analyses assume a worker breathing rate of 3.3×10^{-4} m^3/sec. The typical case analyses assume an adult breathing rate of 2.5×10^{-4} m^3/sec.

Using the methodology described in this section, atmospheric release doses (in mrem) are calculated as a function of distance from a single VSC-24 cask. Doses are calculated for normal and off-normal conditions, and for the bounding and typical cases described earlier in this section. These doses are calculated for the effective whole body (TEDE), the thyroid, and for several other key organs including the gonad, breast, lung, red marrow, bone surface, skin, and remainder of the body. The results of these calculations are presented in Table 7.2-3 through Table 7.2-6. The tables present the doses for the whole body (TEDE), the thyroid, and the individual organ that yields the highest dose. Regulations limit the overall annual dose to the whole body and to any critical individual organ to 25 mrem. The thyroid has a separate dose limit of 75 mrem. For this reason, the tables separately list the thyroid dose, along with the TEDE and the highest dose from any individual organ.

7.2.2 PRESSURIZATION OF CONFINEMENT VESSEL

The MSB is vacuum dried and backfilled with helium prior to seal welding the closure lids. It is not expected that any significant moisture or gases such as oxygen could remain, such that radiolytic decomposition could cause a significant increase in MSB internal pressure or, even less likely, an explosion under normal conditions.

The thermal evaluation in Section 4.4 provides the maximum internal pressure which can be expected due to heating factors. These analyses show that the expected maximum internal pressure does not over-stress the MSB. It is not expected that any radioactive material (gaseous or particulate) can be released under normal or accident conditions.

7.3 CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

7.3.1 FISSION GAS PRODUCTS

The accident condition isotope inventory available for release from the MSB interior is calculated using the same methodology that was used to calculate atmospheric release doses for normal and off-normal conditions, described in Section 7.2.1, with the following different assumptions:

- It is conservatively assumed for accident conditions that 100% of the fuel rods inside the MSB fail. Thus, all isotope activities that escape the fuel pellet matrix are available for release into the MSB interior volume.
- It is conservatively assumed for accident conditions that 100% of the CRUD on the fuel rod surfaces spalls off and becomes suspended in the MSB interior volume (i.e., available for release from the MSB).

7.3.2 RELEASE OF CONTENTS

The accident condition atmospheric release doses are calculated using the same methodology that is used to calculate atmospheric release doses for normal and off-normal conditions, described in Section 7.2.1, with the following different input parameters:

- The MSB interior volumetric leakage rate is 3.88×10^{-4} cm^3/sec. This is calculated based upon a conservative accident condition MSB interior pressure of 56.8 psig, an accident condition MSB interior bulk gas temperature of 460°F, and the MSB test condition leak rate of 1.0×10^{-4} std-cm^3/sec, as specified by the *technical specifications*.
- The accident condition event duration is 30 days.
- The accident condition atmospheric dispersion factors are based upon moderately stable (F-stability diffusion) atmospheric conditions, and an assumed wind speed of 1 m/s. These accident condition dispersion factors are listed in Table 7.3-1.

Table 7.3-2 and Table 7.3-3 present the accident condition doses for the whole body (TEDE), the thyroid, and the bone, which is the individual organ that yields the highest doses.

Table 7.2-1 - Isotopes Contributing to Atmospheric Release Doses

Actinides			
Pu-238	Pu-240	Am-241	Cm-244
Pu-239	Pu-241	Cm-242	
Fission Products			
H-3	Rh-106	Cs-137	Pm-147
Kr-85	Sb-125	Ba-137m	Eu-154
Sr-90	Te-125m	Ce-144	Eu-155
Y-90	I-129	Pr-144	Co-60
Ru-106	Cs-134	Pr-144m	

Table 7.2-2 - Atmospheric Dispersion Factors (Normal and Off-Normal Conditions)

Distance (meters)	χ/Q (sec/meter)
100	1.244E-3
110	9.417E-4
120	7.676E-4
130	6.582E-4
150	5.232E-4
200	3.478E-4
300	1.638E-4
400	9.873E-5
500	6.444E-5
600	5.182E-5
700	4.175E-5
800	3.023E-5
900	2.511E-5
1000	2.194E-5

Table 7.2-3 - Atmospheric Release Dose vs. Distance (mrem)
(Bounding Fuel Case – Normal Conditions)

Distance (meters)	Whole Body Dose (TEDE)	Lung Dose*	Thyroid Dose
100	5.064E+00	2.813E+01	1.094E+00
110	3.833E+00	2.130E+01	8.278E-01
120	3.124E+00	1.736E+01	6.747E-01
130	2.679E+00	1.489E+01	5.786E-01
150	2.130E+00	1.183E+01	4.599E-01
200	1.416E+00	7.866E+00	3.057E-01
300	6.667E-01	3.704E+00	1.440E-01
400	4.019E-01	2.233E+00	8.679E-02
500	2.623E-01	1.457E+00	5.664E-02
600	2.109E-01	1.172E+00	4.555E-02
700	1.699E-01	9.442E-01	3.670E-02
800	1.230E-01	6.837E-01	2.657E-02
900	1.022E-01	5.679E-01	2.207E-02
1000	8.930E-02	4.962E-01	1.929E-02

* The lung is the bounding (maximum dose) organ for bounding case normal conditions.

**Table 7.2-4 - Atmospheric Release Dose vs. Distance (mrem)
(Typical Fuel Case – Normal Conditions)**

Distance (meters)	Whole Body Dose (TEDE)	Lung Dose*	Thyroid Dose
100	4.732E-01	2.736E+00	1.004E-01
110	3.582E-01	2.072E+00	7.597E-02
120	2.920E-01	1.689E+00	6.193E-02
130	2.504E-01	1.448E+00	5.310E-02
150	1.990E-01	1.151E+00	4.221E-02
200	1.323E-01	7.651E-01	2.806E-02
300	6.231E-02	3.603E-01	1.321E-02
400	3.756E-02	2.172E-01	7.965E-03
500	2.451E-02	1.418E-01	5.199E-03
600	1.971E-02	1.140E-01	4.181E-03
700	1.588E-02	9.184E-02	3.368E-03
800	1.150E-02	6.650E-02	2.439E-03
900	9.552E-03	5.524E-02	2.026E-03
1000	8.346E-03	4.826E-02	1.770E-03

* The lung is the bounding (maximum dose) organ for typical case normal conditions.

Table 7.2-5 - Atmospheric Release Dose vs. Distance (mrem) (Bounding Fuel Case – Off-Normal Conditions)

Distance (meters)	Whole Body Dose (TEDE)	Bone Surface Dose*	Thyroid Dose
100	3.207E+01	2.291E+02	3.195E+00
110	2.428E+01	1.734E+02	2.418E+00
120	1.979E+01	1.414E+02	1.971E+00
130	1.697E+01	1.212E+02	1.690E+00
150	1.349E+01	9.636E+01	1.344E+00
200	8.966E+00	6.406E+01	8.931E-01
300	4.223E+00	3.017E+01	4.206E-01
400	2.545E+00	1.818E+01	2.535E-01
500	1.661E+00	1.187E+01	1.655E-01
600	1.336E+00	9.544E+00	1.331E-01
700	1.076E+00	7.689E+00	1.072E-01
800	7.793E-01	5.568E+00	7.763E-02
900	6.473E-01	4.625E+00	6.448E-02
1000	5.656E-01	4.041E+00	5.634E-02

* The bone surface is the bounding (maximum dose) organ for bounding case off-normal conditions.

Table 7.2-6 - Atmospheric Release Dose vs. Distance (mrem) (Typical Fuel Case – Off-Normal Conditions)

Distance (meters)	Whole Body Dose (TEDE)	Lung Dose*	Thyroid Dose
100	3.214E+00	1.851E+01	3.179E-01
110	2.433E+00	1.401E+01	2.406E-01
120	1.983E+00	1.142E+01	1.961E-01
130	1.700E+00	9.795E+00	1.682E-01
150	1.352E+00	7.786E+00	1.337E-01
200	8.985E-01	5.176E+00	8.887E-02
300	4.231E-01	2.438E+00	4.185E-02
400	2.551E-01	1.469E+00	2.523E-02
500	1.665E-01	9.589E-01	1.647E-02
600	1.339E-01	7.711E-01	1.324E-02
700	1.079E-01	6.213E-01	1.067E-02
800	7.809E-02	4.499E-01	7.724E-03
900	6.487E-02	3.737E-01	6.416E-03
1000	5.668E-02	3.265E-01	5.606E-03

* The lung is the bounding (maximum dose) organ for typical case off-normal conditions.

Table 7.3-1 - Atmospheric Dispersion Factors (Accident Conditions)

Distance (meters)	χ/Q (sec/meter)
100	8.650E-3
110	6.631E-3
120	5.565E-3
130	4.421E-3
150	3.248E-3
200	2.487E-3
300	1.163E-3
400	6.593E-4
500	4.852E-4
600	3.681E-4
700	2.584E-4
800	2.411E-4
900	1.950E-4
1000	1.611E-4

Table 7.3-2 - Atmospheric Release Dose vs. Distance (mrem)
(Bounding Fuel Case – Accident Conditions)

Distance (meters)	Whole Body Dose (TEDE)	Bone Surface Dose*	Thyroid Dose
100	5.662E+02	4.332E+03	4.893E+01
110	4.340E+02	3.321E+03	3.751E+01
120	3.642E+02	2.787E+03	3.148E+01
130	2.894E+02	2.214E+03	2.501E+01
150	2.126E+02	1.627E+03	1.837E+01
200	1.628E+02	1.246E+03	1.407E+01
300	7.612E+01	5.825E+02	6.578E+00
400	4.315E+01	3.302E+02	3.729E+00
500	3.176E+01	2.430E+02	2.744E+00
600	2.409E+01	1.844E+02	2.082E+00
700	1.691E+01	1.294E+02	1.462E+00
800	1.578E+01	1.208E+02	1.364E+00
900	1.276E+01	9.767E+01	1.103E+00
1000	1.054E+01	8.069E+01	9.112E-01

* The bone surface is the bounding (maximum dose) organ for bounding case accident conditions.

Table 7.3-3 - Atmospheric Release Dose vs. Distance (mrem)
(Typical Fuel Case – Accident Conditions)

Distance (meters)	Whole Body Dose (TEDE)	Bone Surface Dose*	Thyroid Dose
100	2.504E+02	1.505E+03	2.179E+01
110	1.919E+02	1.154E+03	1.671E+01
120	1.611E+02	9.682E+02	1.402E+01
130	1.280E+02	7.691E+02	1.114E+01
150	9.402E+01	5.651E+02	8.183E+00
200	7.199E+01	4.327E+02	6.266E+00
300	3.366E+01	2.023E+02	2.930E+00
400	1.908E+01	1.147E+02	1.661E+00
500	1.404E+01	8.441E+01	1.222E+00
600	1.066E+01	6.404E+01	9.274E-01
700	7.480E+00	4.495E+01	6.510E-01
800	6.979E+00	4.195E+01	6.074E-01
900	5.645E+00	3.392E+01	4.913E-01
1000	4.663E+00	2.803E+01	4.059E-01

* The bone surface is the bounding (maximum dose) organ for typical case accident conditions.

8.0 OPERATING PROCEDURES

Figure 8.0-1 provides a flow chart of the handling activities that will be associated with the operation of the VSC system. The procedures for loading the VSC are given below in Section 8.1. Procedures for unloading the cask, should this activity become necessary, are essentially those of Section 8.1 in reverse. The on-site transport of the cask described in Section 8.4 is based on the use of a trailer. Appendix B describes an alternate approach, using the transporter.

8.1 PROCEDURES FOR LOADING THE CASK

The following outline briefly describes the major procedural steps. Table 8.1-1 summarizes typical time and crew requirements. The times listed in this table come from detailed handling studies, utility, and DOE cask handling experience, and other, general industry experiences. While the hours shown are believed to be a reasonable estimate of crew sizes and elapsed times, there are several site-specific characteristics that can significantly impact handling times and crew sizes. Among these are:

1. Productivity parameter (dress-out time, break time, pass-in, pass-out times, etc.)
2. Required crew sizes and supervisory personnel (e.g., foremen, etc.)
3. Security requirements (manpower and access)
4. Learning curve (estimates are for handling the nth cask, not the 1st one)
5. Health physics requirements
6. Management attendance/involvement
7. Hold and wait times (waiting for other pool floor, decontamination area, truck bay, etc., activities to be completed, plant operational delays, etc.)

The following information is provided as an estimate of the required time and manpower requirements for the listed procedural activities. Detailed, site-specific procedures will be required to define the sequence and for the actual implementation of the outline described below.

(1) **Preparation of Concrete Cask, MSB and Transfer Cask** (Elapsed Time: 2 hrs)

1. Check and clean (if necessary) the MSB and its shielding and closure lids.
2. Examine and clean (if necessary) the concrete cask and tow it into the fuel handling building truck/rail bay.
3. Examine and clean (if necessary) the transfer cask.
4. Lift and place the MSB into the transfer cask. Install shims in MTC to MSB gap.
5. Fill the MSB with borated water. Attach a water hose to the MTC at the flush tube on the bottom (this will probably be performed at poolside).

(2) **Fuel Loading** (Elapsed Time: 5.5 hrs)

1. Lift and place the transfer cask (containing the MSB) into the fuel pool. During lowering, fill MSB-MTC gap with clean water and confirm that water is continuously flowing through the gap. (0.5 hr)
2. Load 24 fuel assemblies that meet the requirements of the fuel specification (*technical specification* 1.2.1) in the MSB storage sleeves. Visually inspect all fuel assemblies as they are loaded. (4 hrs)

   Note: "High cobalt" assemblies (as defined in technical specification basis B.1.2.1) must be loaded into one of the 12 inner MSB storage sleeves.
3. Lift the MSB shielding lid and place it on the MSB in the pool.
4. Lift the transfer cask (containing the MSB) out of the pool. Shut off water supply to MSB-transfer cask gap and hold transfer cask over the pool to allow water in the MSB-cask gap to drain. Wash down exterior of transfer cask as it hangs over the pool. Transfer the MTC to the decon pit area. (0.5 hr)

(3) **MSB Closure** (Elapsed Time: 14.5 hrs)

1. Drain sufficient quantity of water from the MSB to provide for welding. (0.25 hr)
2. Weld shield lid to the shell. Dye penetrant check the weld. (4 hrs)
3. Pressurize the MSB with water to 22 psia and visually inspect the weld. (0.25 hr)
4. Perform a leak test of the MSB inner seal weld by filling the MSB with helium to approximately 22 psia. Use a helium sniffer to verify the leak tightness of the shield weld. (Acceptance criterion: 10^{-4} scc/sec.) Procedures for leak testing are per ANSI N 14.5.
5. Place the structural lid in the MSB and seal weld it to the shell and shield lid. Ultrasonic test structural lid weld. (5 hr)
6. Decontaminate the exterior of the transfer cask. (4 hrs) (concurrent with 1,2,3,4,6, and 7)
7. Attach the VDS to the top of the MSB and pump the remainder of its liquid contents into the fuel pool. Isolate the drain and helium supply portion of the VDS from the MSB and complete the drying process by evacuating the MSB. Using the VDS vacuum pump, perform multiple pump downs as required to achieve a stable vacuum pressure in the MSB of 3 mm Hg for at least 30 min. Flush the MSB with helium and repeat the evacuation. (8 hrs)
8. Using the VDS, backfill the MSB with helium (99.995% pure) to 14.5 ± 0.5 psia and hold for 30 minutes (1 hr).
9. Remove the gas line and seal weld the two valve covers on the structural lid to provide redundant seal. (0.75 hr)
10. Place and bolt the transfer cask cover. (0.25 hr)

Items 5, 6, 7 and 8 may be done concurrently

(4) **MSB Transfer to the Concrete Cask** (Elapsed Time: 2.5 hrs)

1. Lift the transfer cask and place it on the concrete cask. (0.5 hr)
2. Secure and check the placement of the transfer cask on the concrete cask. (0.25 hr)
3. Install the MTC doors hydraulic system (0.5 hr). (concurrent with 4)
4. Install and check the MSB lifting devices. Hook and lift the MSB. (0.5 hr)
5. Open the shield doors on the transfer cask. (0.25 hr)
6. Lower the MSB from the transfer cask into the concrete cask. Special care should be taken to prevent impact of the MSB on the VCC tiles to minimize the potential for cracking the tiles. The maximum permitted lowering rate is 0.75 ft./sec. (0.25 hr)
7. Remove the transfer cask from the concrete cask and place the shielding ring over the gap between the concrete cask inner surface and the MSB outer surface. (0.5 hr)
8. Install the VCC cover plate. (0.25 hr)

8.2 PROCEDURES FOR UNLOADING THE CASK

Unloading an MSB can be accomplished in a number of ways, as described in Reference 2.2 (carbon arc air gouger, plasma cutter, portable lathe, etc.). A typical unloading sequence is shown in Figure 8.2-1.

Each VSC-24 licensee shall develop a detailed MSB unloading procedure prior to storing spent fuel at the licensee site. In addition to a detailed unloading procedure, the site-specific evaluation shall include the MSB re-flood analyses described in Section 2.4.

8.3 PREPARATION OF THE CASK FOR ON-SITE TRANSPORT

The following are the major steps in moving the VSC from the auxiliary building bay to the on-site storage pad.

VSC Storage (Elapsed Time: 3.5 hrs)

1. Check radiation levels along the concrete cask surface at four to five feet above the floor and at other appropriate locations (seals, lids, etc.) (0.25 hr)
2. Swipe the concrete cask exterior for contamination. (1.0 hr)
3. Using the trailer, move the concrete cask to the storage pad. (1.0 hr)
4. Insert the hydraulic roller skid under the concrete cask and energize the hydraulic pads, lifting the cask approximately two inches. (0.25 hr)
5. Using the skid towing truck, tow the skid and concrete cask off the trailer and to its storage location. (0.5 hr)

6. Lower the hydraulic pads and remove the skid from the concrete cask and return truck, trailer and skid to storage. (0.5 hr)

8.4 SUPPLEMENTAL DATA

The following paragraphs provide additional discussion of the procedures described above.

The preparation step involves checking and testing of the transfer cask, trailer, and tow vehicles. Most of the five items listed above are done concurrently.

The fuel loading operations are essentially the same as for any transfer cask operation. The two notable exceptions are: 1) that shims are inserted into the gap area to aid in preventing pool water from entering the gap and to provide radiation shielding in the gap; 2) the gap between the MSB exterior and the transfer cask exterior is filled with clean borated water as the cask is being lowered into the pool. A hose connection is provided on the MTC body to allow clean water to be forced through the MSB-transfer cask gap to further assure that pool water will not contact the MSB surface. The remaining steps are similar to normal cask handling procedures.

The MSB and MTC are decontaminated to the specified requirements prior to closure operations. The first step of MSB closure is to connect a self-priming pump to the MSB drain penetration and to pump sufficient amount of water out of the MSB to permit welding. After this is done, the shield lid is welded to the MSB shell. With the MSB now sealed, it can be hydrostatically pressure tested to 1.5 atm. Following hydro testing, the shield lid-to-shell weld is Helium leak tested to assure integrity of not less than 1.0×10^{-4} standard cubic centimeters per second (scc/sec) at 0.5 atm differential pressure. The structural lid is then welded to the MSB shield lid and shell. The structural lid-to-shell weld is volumetrically examined. Following acceptance of the weld, the MSB is pressurized with Helium and the MSB is pumped out (drained). The majority of the closure effort is done while the water is still inside the MSB so that the surface doses will be much less than reported in Chapter 5. Following the drain down the MSB is flushed with dry inert gas and vacuum dried to ≤3 mm Hg. The vacuum drying is repeated a second time. Following the drain-down, the MSB is then filled with helium, and after thirty minutes the fill gas is equalized to 1 atm. The valve covers are welded onto the structural lid. At this point the transfer cask lid, which is a circular plate with a hole in its center, is bolted onto the transfer cask. This lid, through its center hole, allows access to the MSB lifting rings that will be used to lower the MSB into the concrete cask. The transfer cask lid overlaps the edge of the MSB, preventing inadvertent lifting of the MSB out of the transfer cask during the lowering process.

Operating procedures shall also include steps to minimize the potential for generation and ignition of explosive gases. Specifically, vent/purge the space under the shield lid and monitor for combustible gases. If concentrations of 10% or more of the lower explosive limit (LEL) (0.4% by volume of hydrogen) of combustible gases are detected, appropriate actions shall be taken.

If the MSB is loaded with fuel assemblies that have high burnup levels or high hardware cobalt quantities, high canister and cask exterior dose rates may occur. For limiting fuel, dose rates up to ~1 Rem/hr may occur on the top of the canister (structural lid) or on the side of the MTC. Dose rates over 1 Rem/hr may occur on the surface of the shield lid during the shield lid welding process. Personnel exposures may be significantly reduced by:

- using temporary shielding on the canister lid
- retaining most of the MSB interior water during the welding process
- loading lower source strength assemblies in the fuel sleeves around the MSB edge
- minimizing the time spent near the cask system by personnel during each loading operation step

Use of an automated welding process for the MSB lid welds may significantly reduce overall occupational exposure.

During the movement of the transfer cask to the concrete cask, the transfer cask will be lifted out of the decontamination pit and moved at a minimum distance above the floor to the concrete cask. The two casks are aligned using one-inch diameter holes. The crane is detached from the transfer cask and attached to the MSB. The MSB is lifted slightly to remove its weight from the lower shield doors, the doors are opened (hydraulically), and the MSB is lowered into the concrete cask.

After the MSB has been lowered into the concrete cask, the transfer cask is lifted and returned to the decontamination pit. The concrete cask cover is bolted down and the cask is secured to the transport vehicle.

The next major sequence of steps takes the concrete cask from the auxiliary building to its storage location. The transport vehicle is used to transport the cask to the storage pad. Air pads are used to move the cask off the transport vehicle onto the storage pad and to its storage location. At the designated storage location the hydraulic air pressure is released, the cask lowers to rest on the concrete pad and the skid can be removed. At this time, the cask is in safe, long-term storage.

Reference 2.2 provides details of retrieval methods to move the fuel back to the pool. The VSC is moved back into the auxiliary building where the MTC is used to retrieve the MSB from the VCC.

Table 8.1-1 - Operations Time and Motion Summary

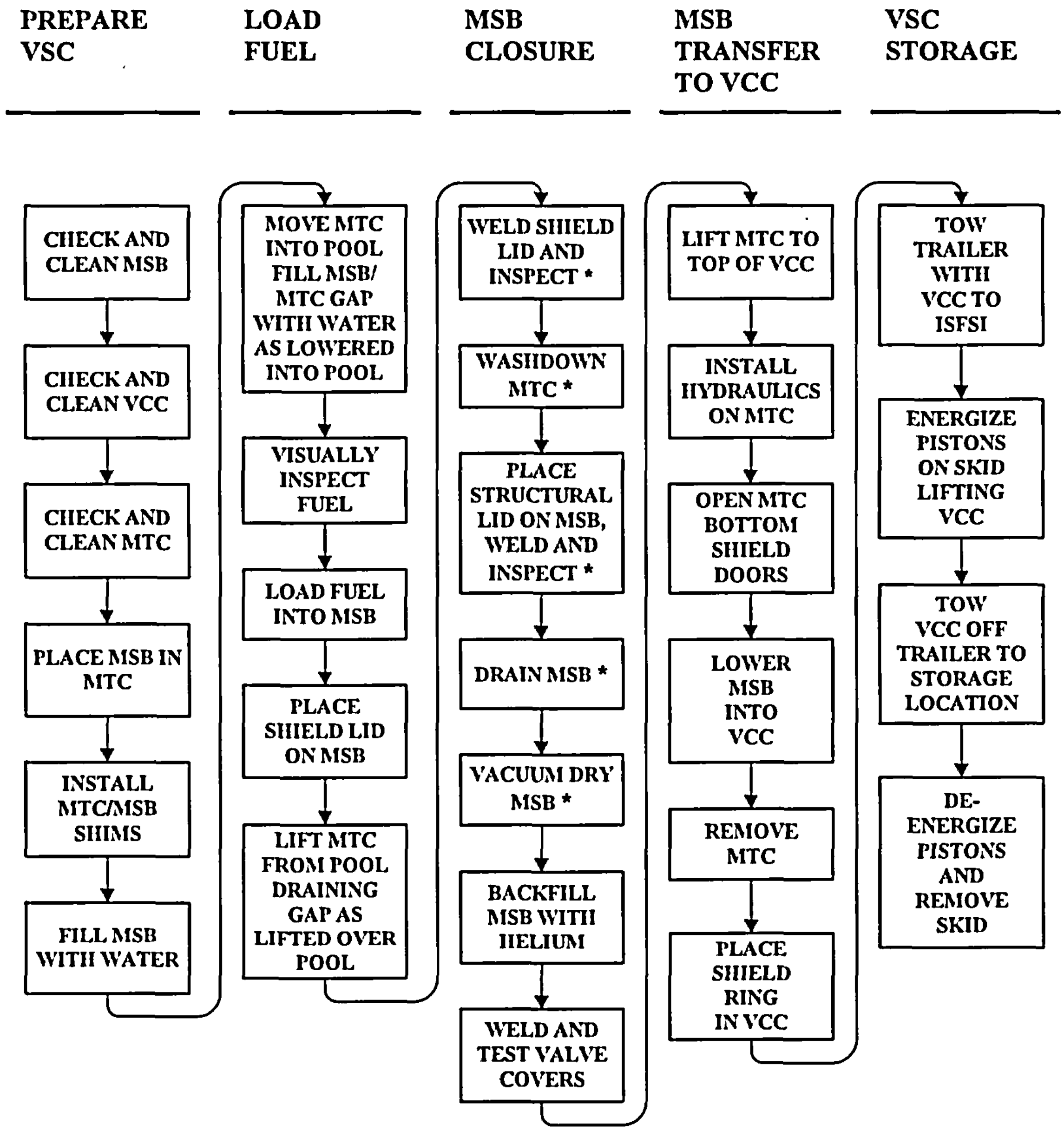
ITEM	ELAPSED TIME (HOURS)	MAIN	CREW OPER	SIZE HP	ENG	TOTAL MANHOURS
Preparation	2	4	---	---	1	10.0
Fuel Loading	5.5	2	2	2	1	38.5
MSB Closure	14.5	4	---	2	1	101.5
MSB Transfer to Concrete Cask	2.5	4	2	2	4	30.0
VSC to Storage	3.5	2	1	1	1	17.5
Total	28.0					197.5

MAIN = Maintenance (welders, decontamination techs, drivers, etc.)

OPER = Crane and fuel handling operators

HP = Health Physics

ENG = Engineer Supervision



* May be done in parallel

Figure 8.0-1 - Flow Diagram of VSC System Handling Procedures

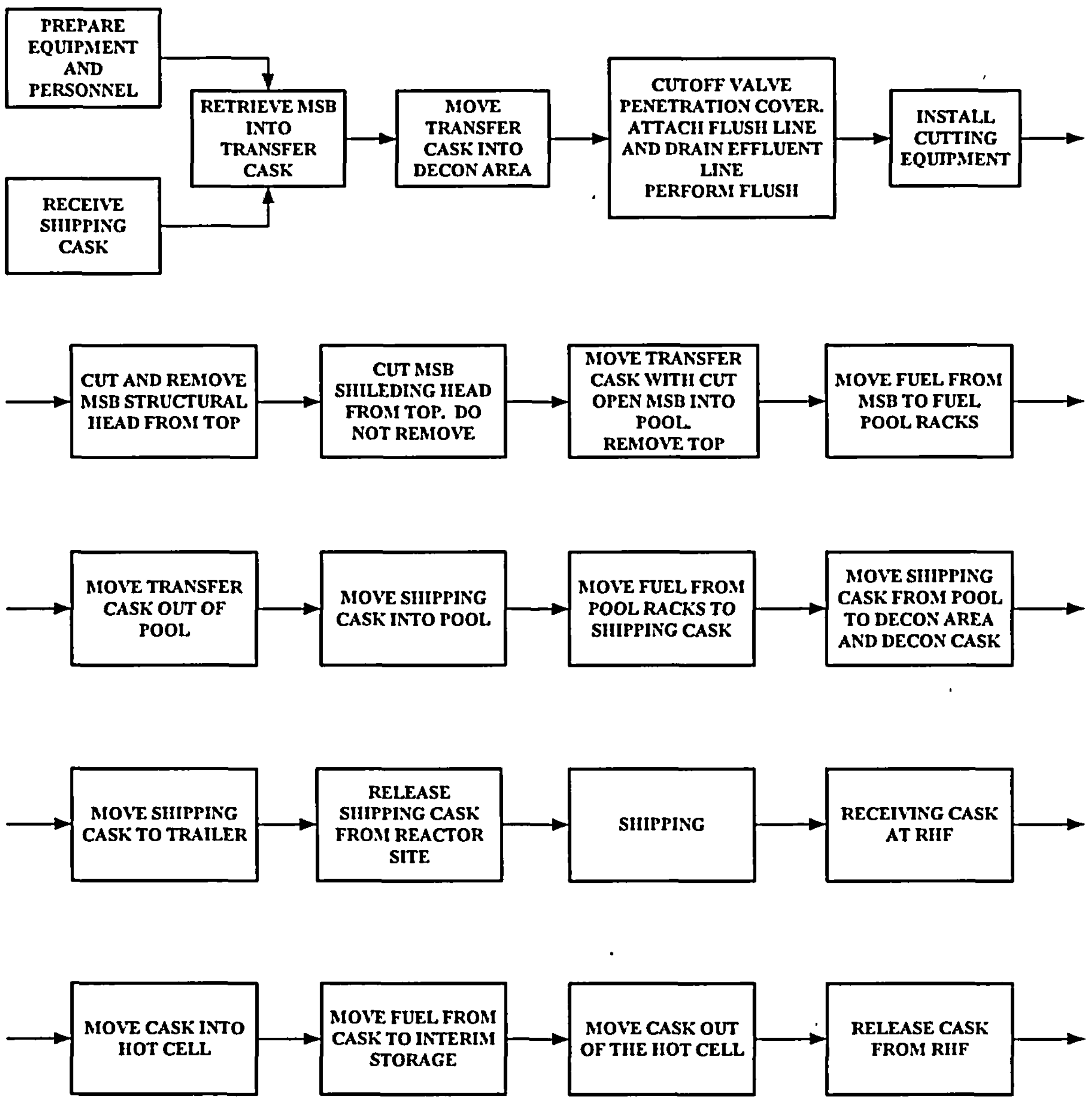


Figure 8.2-1 - Flowchart for One Typical Method of Recovering Fuel From the VSC

9.0 ACCEPTANCE TEST AND MAINTENANCE PROGRAM

9.1 ACCEPTANCE TEST

9.1.1 VISUAL INSPECTION

9.1.1.1 Fabrication Inspections

A complete dimensional inspection of all critical dimensions and a components part fit-up test will be performed on the VCC, MSB, and MTC prior to shipping from the fabricator's facility. The critical dimensions/fit-ups are:

VCC

- wall thickness (concrete and steel)
- internal cavity length
- internal cavity diameter
- bottom depth
- air flow area
- fit-up of cask lid

MSB

- clearance of storage sleeves
- thickness of all components
- diameter
- length
- fit-up of shield and structural lids

MTC

- wall thickness
- bottom thickness
- length
- internal diameter
- fit-up of doors/actuators and their operation
- load test trunnions

Materials other than steel that are important for shielding, including concrete and RX-277, will be verified for density. The concrete density shall be at least 144 lbs/ft^3. The RX-277 density shall be at least 0.041 lb/in^3.

9.1.1.2 Inspection Prior to Use

The first inspections prior to use of the VCC, MSB, and MTC are inspections for cleanliness. All foreign material shall be removed from all components to be used.

Fit-up tests on all the major components will also be performed prior to use. These are:

- MSB in VCC
- MSB in MTC
- MSB lowered through MTC bottom
- Shield lid on MSB
- Structural lid on MSB

9.1.2 STRUCTURAL AND PRESSURE TEST

Refer to Section 3.4.4.1.6 for the MSB Pressure Test.

9.1.3 TEST OF THE FIRST VSC PLACED IN SERVICE

For the user of the first VSC placed in service at the first utility using the VSC system, the first MSB will be loaded with 24 spent fuel assemblies, constituting a heat source of 24 kW. Then, the MSB shall be loaded into the VCC to measure the cask thermal performance by measuring the air inlet and outlet temperatures for normal air flow, according to the *technical specification* in Section 12.4. The purpose of the test is to measure the heat removal performance of the VSC system and establish base-line data. A letter report summarizing the results of the test and evaluation shall be submitted to the NRC within 30 days of placing the cask in service in accordance with 10 CFR 72.4.

Should the first user of the system not have spent fuel capable of producing a 24 kW heat load, the user may use a lesser load for the test, provided that a calculation of the temperature difference between the inlet and outlet temperatures is performed using the same methodology and inputs documented in the NRC Safety Evaluation Report and this FSAR, with the lesser load as the only exception. The calculation and the measured temperature data will be reported to the NRC in accordance with 10 CFR 72.4. The calculation and comparison need not be reported for casks that are subsequently loaded with lesser heat sources than the test case. However, for the first or any other user, the process needs to be reported for any higher heat sources, up to 24 kW, which is the maximum allowed under the Certificate of Compliance for the VSC system. Artificial thermal loads other than spent fuel are acceptable for use to satisfy the requirements of this section.

9.2 MAINTENANCE PROGRAM

The VSC system is a passive system and requires no maintenance other than the surveillance activities identified in Chapter 12.

10.0 RADIATION PROTECTION

10.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

10.1.1 POLICY CONSIDERATIONS

The VSC cask is designed in a way that its operation, inspection, repair and maintenance can be carried out in a manner consistent with the principle that occupational exposure be maintained as low as reasonably achievable (ALARA).

Specific design oriented ALARA considerations are listed below. The operation of the cask, the management policy and the plant organizational structure related to ensuring that occupational exposures to radiation are as low as reasonably achievable are site-specific. This information will be provided in site-specific license applications.

10.1.2 DESIGN CONSIDERATIONS

The design of the VSC system complies with 10 CFR 72 concerning ALARA. Specific considerations that are directed toward ensuring ALARA are:

- Thick concrete walls to reduce the side surface dose to below 100 mrem/hr.
- Fuel loading procedures that follow accepted practice and built on existing experience.
- Multiple pass welds on all seal closures to provide redundant radioactive material containment.
- Totally passive system that requires minimum maintenance.
- Decontamination of exterior of transfer cask while MSB is still filled with water to reduce exposure to technicians.

10.1.3 OPERATIONAL CONSIDERATIONS

The preliminary operating steps have been determined following the ALARA guidelines. However, the exact handling and operation of the VSC will be site-specific, and, as such, details on the ALARA characteristics of the operations and procedures are site-specific.

10.2 RADIATION PROTECTION DESIGN FEATURES

Details of the radiation protection design features are provided in Chapter 5. The design basis is summarized in this section, while Section 10.3 provides the results of the actual dose assessment calculations.

10.2.1 DESIGN BASIS FOR NORMAL CONDITIONS

The radiation protection design basis for the VSC cask were derived from 10 CFR 72 and the applicable ALARA guidelines. The specific design basis for maximum dose rates are:

Location	**Dose Rate Limit**
Side Dose Rate (surface average)	100 mrem/hr
Top of VCC (surface average)	20 mrem/hr
Top Structural Lid	800 mrem/hr
Side of MTC	850 mrem/hr

10.2.2 DESIGN BASIS FOR ACCIDENT CONDITIONS

Damage to the VSC cask after a hypothetical accident will not result in a dose rate greater than 1,000 mrem/hr at a distance of one (1) meter from the cask or result in any off-site doses in excess of 5 rem.

10.3 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

10.3.1 ESTIMATED OCCUPANCY REQUIREMENTS

Estimated personnel requirements for cask operations are shown in Table 10.3-1.

10.3.2 DOSE RATES

The VSC system is designed to limit dose rates to minimal levels for operators, inspectors, maintenance, and health physics personnel when the cask is loaded and in storage. Table 10.3-2 contains the bounding design dose limits, and the calculated working dose rates for loading and handling the casks, under the normal storage conditions. All values for dose rates include both gamma and neutron flux components.

The bounding dose rate limits presented in Table 10.3-2 are unrealistically high for virtually all actual loaded casks because they are based upon fuel assemblies with extremely low enrichment levels and extremely high hardware cobalt contents. More realistic cask dose rates (which are based upon more realistic fuel assembly parameters that apply for the great majority of existing fuel) are therefore presented in Table 10.3-3.

Occupational exposure estimates are provided herein for both the bounding and realistic cases. The realistic case exposure estimates may be used for casks that are shown, by analysis or measurement, to have external dose rates equal to or lower than those shown in Table 10.3-3.

10.3.3 ESTIMATED MAN-REM EXPOSURES FOR OPERATION, MAINTENANCE, AND INSPECTION OF THE EQUIPMENT

Working dose rates and personnel requirements for the VSC cask loading cycle and move-to-storage cycle are shown in Table 10.3-4 and 10.3-5 for the bounding and realistic dose rate cases, respectively. The estimated dose rates may vary from site to site depending on site-specific configuration, operations considerations, and spent fuel source term.

Based on the bounding exposure estimates shown in Table 10.3-4, and assuming three casks are loaded per year (i.e., 72 assemblies are placed in storage per year), the collective dose is determined as follows:

Load Cask:	8.10 man·rem/cycle x 3 = 24.30 rem/year
Move to Storage:	0.18 man·rem/cycle x 3 = 0.54 rem/year
Total	24.84 rem/year

Conservative estimates of the annual maintenance inspection and survey requirements result in a collective dose of 0.05 person rem/year per cask. This dose estimate is based on a one-minute inspection of a freshly loaded cask (i.e., 0.03 rem/hr x 1 hr/60 x 1 min x 2 persons x 52 inspections per year). The dose will, of course, decrease with the age of the cask.

Based on the more realistic dose rate estimates shown in Table 10.3-3, and assuming three casks are loaded per year (i.e., 72 assemblies placed in storage per year), the collective dose is determined as follows:

Load Cask:	0.870 man·rem/cycle x 3 = 2.61 rem/year
Move to Storage:	0.048 man·rem/cycle x 3 = 0.14 rem/year
Total	2.75 rem/year

This more realistic estimate of the annual maintenance inspection and survey requirements results in a collective dose of 0.014 person rem/year per cask. This dose estimate is based on a one minute inspection of a freshly loaded cask (i.e., 0.008 rem/hr x 1 hr/60 x 1 min x 52 inspections per year). The dose will, of course, decrease with the age of the cask.

10.4 ESTIMATED OFF-SITE COLLECTIVE DOSE ASSESSMENT

To compare the performance of the VSC-24 cask system with the applicable regulations governing off-site dose, this section presents the calculated off-site exposure for a sample 5x5 array of VSC-24 storage casks.

Site-specific characteristics, such as the background radiation from plant sources, the location and layout of an ISFSI, the terrain features around the ISFSI, the number and configuration of storage casks, and the characteristics of the actual loaded SNF assemblies, will affect actual ISFSI doses. Site-specific dose evaluations accounting for the location of the site boundary and the effects of radiation from other on-site operations are part of the ISFSI design and are necessary to demonstrate regulatory compliance.

The off-site collective dose assessment considers normal, off-normal, and accident conditions. The significant contributors to off-site exposure are direct radiation from storage casks and potential leakage of gases, volatiles, fuel fines, and crud from canisters.

The evaluated conditions include the direct dose component from a sample ISFSI with 25 storage casks arranged in a 5x5 square array. Shielding analyses that model a single VSC-24 cask are performed using MCNP, a Monte Carlo computer code (Reference 5.4). The analyses determine the gamma and neutron dose contributions from the cask side and top surfaces as a function of distance from the cask. The analyses include the effects of both neutron and gamma radiation, skyshine, ground reflection, and air attenuation. A description of the shielding methodology is provided in Section 5.4.

Detector locations are defined at various distances from the ISFSI edge, and the distances between each cask in the ISFSI and the defined detector locations are determined. Based upon these distances and the dose rate vs. distance data determined by the above analyses, the dose rate contributions from each cask in the ISFSI are summed to yield the total direct dose rate at each detector location. Cask top dose rate contributions are summed over the 25 casks in the ISFSI. For cask side dose rate contributions, cask blockage effects are considered. For casks in the back rows of the ISFSI, the cask side dose rate contributions are multiplied (i.e., reduced) by the fraction of the source regions of those casks that can be directly viewed from the detector location.

Summing the cask side and top dose rate contributions yields total dose rates as a function of distance from the ISFSI. Overall annual doses are yielded by multiplying the dose rates by annual occupancy factors.

Direct doses are calculated for bounding and typical conditions. The bounding condition analyses conservatively assume the maximum allowable cask-surface average dose rates specified in Section 2.3.5.2 (100 and 200 mrem/hr, respectively, for the cask side and top surfaces). A full occupancy factor of 8766 hours/year is assumed. The typical case analyses assume cask side and top surface average dose rates of 20 and 50 mrem/hr, respectively, along with a 2000 hour/year occupancy factor. Table 10.4-1 presents the direct doses for the bounding and typical cases as a function of distance from the ISFSI. Table 10.4-1 presents the cask side and cask top direct dose contributions, along with the total direct dose.

All evaluated conditions conservatively consider potential leakage of gases, volatiles, fuel fines, and crud from the MSBs, even though leakage is not expected to occur in the MSBs. The assumed leakage rates corresponding to the testing conditions are corrected for canister temperatures and pressures for each condition. When performing the annual off-site dose assessments, the leakage is assumed to occur for one year (normal and off-normal conditions) or 30 days (accident conditions). Because the canister temperatures and pressures and the assumed leakage time vary by condition, the leakage dose varies. The confinement analyses are described in Chapter 7.

10.4.1 OFF-SITE DOSE FOR NORMAL AND OFF-NORMAL OPERATIONS

Contributions from potential canister leakage are summed to determine the total annual off-site normal and off-normal condition doses, as shown Table 10.4-2 and Table 10.4-3 for bounding and typical conditions, respectively. The single-cask annual canister-leakage doses presented in Table 10.4-2 and Table 10.4-3 are based on the highest of the whole body, thyroid, or limiting organ doses presented in Table 7.2-3 through Table 7.2-6. As discussed in Section 7.2.1, the bounding-case leakage doses are calculated based upon 52 GWd/MTU, 5-year-cooled, PWR fuel (a combination of fuel parameters that yields a higher activity inventory than any of the allowable combinations shown in Table 5.5-1). The typical-case leakage doses are calculated based upon
35 GWd/MTU, 10-year-cooled, PWR fuel.

The canister-leakage dose contributions from a single cask are multiplied by the number of casks in the ISFSI to yield ISFSI total doses. The dose contribution from a single, off-normal condition cask is added to the normal-condition dose contributions from the remaining casks in the ISFSI. The higher leakage doses for the off-normal condition canister arise solely from differences in canister internal temperature and pressure, which, in turn, affect the rate of leakage from the canister.

The resulting ISFSI total doses due to potential canister leakage are added to the direct doses (from Table 10.4-1), as shown in Table 10.4-4. Since none of the off-normal conditions evaluated in the FSAR have any impact on the VSC-24 shielding configuration, the direct dose from the single, off-normal condition cask is the same as the direct doses from the remaining normal condition casks in the ISFSI. Thus, the Table 10.4-1 direct doses are calculated based upon a 5x5 array of normal condition casks. Adding these direct ISFSI doses to the canister leakage doses yields the overall ISFSI doses for the normal/off-normal condition case.

The estimated minimum controlled-area boundary distances are listed in Table 10.4-5. Distances are listed for bounding and typical conditions. Assuming a 100% occupancy factor of 8766 hours per year and maximum allowable cask surface dose rates (bounding conditions), the annual off-site doses from the ISFSI alone reach the 10CFR72.104 limit of 25 mrem at approximately 800 meters from the ISFSI. Assuming an occupancy factor of 2000 hours per year and typical upper-bound cask surface dose rates (i.e., typical conditions), the annual off-site doses from the ISFSI alone reach the 10CFR72.104 limit of 25 mrem at approximately 260 meters from the ISFSI.

The minimum controlled-area boundary distances given above (at which the 10CFR72.104 dose limits are met) may be reduced somewhat by placing a wall or berm around the ISFSI. A wall or berm of sufficient height and thickness can remove virtually all of the cask side dose contributions, as the direct path from the cask side surfaces to (ground level) points far from the ISFSI is completely blocked by the wall/berm. However, dose contributions from the cask top surfaces, and from canister leakage of radionuclides are not significantly reduced by the presence of a wall or berm.

Table 10.4-1 lists the dose contribution from the cask side direct dose, as a function of distance. Table 10.4-4 lists the total dose — including the cask side direct, cask top direct, and canister leakage dose contributions — as a function of distance. An estimate of the maximum reduction in dose that could be achieved through the use of a wall or berm can be calculated by subtracting the cask side direct dose contribution (shown in Table 10.4-1) from the total dose shown in Table 10.4-4.

The resulting "berm case" dose is only an estimate, however. To make use of (and take credit for) a wall or berm in an actual ISFSI, a site-specific shielding evaluation that models the actual berm configuration is required. This evaluation would be performed as part of the site-specific (10CFR72.212) ISFSI dose evaluation that currently must be performed by all licensees.

10.4.2 OFF-SITE DOSE FOR ACCIDENT CONDITIONS

As discussed in Section 11.2.3.3, the storage-cask side radiation dose rate increases, at most, by a factor of 10 under credible accident conditions. The shielding analyses described in Section 10.4 show that the dose rate from a single VCC is 4.37×10^{-2} mrem/hr at the minimum allowable controlled area boundary distance of 100 meters. Multiplying this by a factor of 10 results in a dose rate of 0.437 mrem/hr. Since the accident damage actually covers only a small localized area of the cask side concrete, this approach is extremely conservative. A dose rate of 0.437 mrem/hr corresponds to a total dose of ~300 mrem over the assumed 30-day accident event duration.

As shown in Table 7.3-2, the maximum single cask, accident condition dose due to canister leakage is 4.3 rem at the minimum distance of 100 meters. Adding this dose to the bounding accident condition direct dose of 0.3 rem yields a total dose (including radiation plus atmospheric release) of 4.6 rem over the course of the accident event. This overall accident condition dose is smaller than the 10CFR72.106 accident condition dose limit of 5 rem. The normal and/or
off-normal condition doses from the remaining casks in the ISFSI, which are required by

10CFR72.104 to be less than 25 mrem/year (i.e., ~2 mrem over the course of the 30-day accident event), do not contribute significantly to the accident condition total dose.

Thus, even at the minimum allowable controlled-area boundary distance of 100 meters, the accident condition doses will remain under this regulatory limit. However, as shown in Table 10.4-5, the controlled area boundary distances required to meet the normal/off-normal condition dose limit of 25 mrem are expected to be much more than 100 meters. At these larger distances, the accident condition doses will be a small fraction of the regulatory limit (as shown by the data in Table 7.3-2).

Table 10.3-1 - Personnel Requirements

Item	Personnel Required
Load Cask	2 Operators, 1 HP
Install Cover	2 Technicians, 1 HP, 1 Welder
Decontaminate Cask	2 Technicians, 1 HP
Loading Inspection	1 Inspector, 1 HP
Move to Storage Area	4 Operators, 1 HP
Annual Maintenance	1 Inspector

Table 10.3-2 - Bounding Dose Rates

	Dose Rate (mrem/hr)	
Location	Design Limit	Working *
Side Surface of MTC	850	239
MSB Top (Outside Surface of Structural Lid)	800	< 800
Cask Surface within Storage Area	100	30

* Working dose is calculated dose rate 1 meter from the surface. For the MSB top dose rates used for the welding operation, the surface dose rate is conservatively used as the working dose rate.

Table 10.3-3 - Expected Dose Rates

Location	Dose Rate (mrem/hr)	
	Design Limit*	Working**
Side Surface of MTC	150	81
MSB Top (Outside Surface of Structural Lid)	100	23
Cask Surface within Storage Area	20	8

* Based upon previous VSC loading experience. "Design Limit" dose rates correspond to administrative limits used by previous VSC users.

** Working dose is calculated dose rate 1 meter from the surface.

Table 10.3-4 - Estimated Bounding Personnel Exposure Doses

Item	Personnel Work Groups	Exposure Time (hrs)	Working Dose Rate (mrem/hr)	Exposure (man-mrem)
Load MTC	2 Operators	5.5	~0.1*	1.1
Monitor	1 H.P.	5.5	~0.1*	0.5
Decontaminate Cask	2 Technicians	4.0	239**	1,912
Monitor	1 H.P.	4.0	239	956
Vacuum Dry	1 Technician	8.0	239	1,912
Weld Shield and Structural Lid	2 Welders/ 1 Inspector	1.0	800	2,400
Monitor	1 H.P./Inspctr	1.0	800	800
Load VCC	2 Operators	1.5	30.0	90.0
Monitor	1 H.P.	1.0	30.0	30.0
Totals		31.5	-----	8,100
Move to Storage	2 Operators	2.0	30.0	120
Monitor	1 H.P.	2.0	30.0	60.0
Totals		4.0	-----	180
Annual Surveillance and Maintenance (all casks in storage)	1 Security Guard	0.87	30.0	26.0
	1 Technician	52.0	30.0	26.0

* Assumed radiation reading in pool area

** Assumes worst-case of dry MSB. If water is left in MSB, dose rate will be less than half of the value reported here.

Table 10.3-5 - Estimated Typical Personnel Exposure Doses

Item	Personnel Work Groups	Exposure Time (hrs)	Working Dose Rate (mrem/hr)	Exposure (man-mrem)
Load MTC	2 Operators	5.5	~0.1*	1.1
Monitor	1 H.P.	5.5	~0.1*	0.5
Decontaminate Cask	2 Technicians	4.0	81**	648
Monitor	1 H.P.	4.0	81.0	324
Vacuum Dry	1 Technician	8.0	81.0	648
Weld Shield and Structural Lid	2 Welders/ 1 Inspector	1.0	23	69
Monitor	1 H.P./Inspctr	1.0	23	23
Load VCC	2 Operators	1.5	8.0	24
Monitor	1 H.P.	1.0	8.0	8
Totals		31.0	-----	1,746
Move to Storage	2 Operators	2.0	8.0	32
Monitor	1 H.P.	2.0	8.0	16
Totals		2.0	-----	48.0
Annual Surveillance and Maintenance (all casks in storage)	1 Security Guard	0.87	8.0	7.0
	1 Technician	0.87	8.0	7.0

* Assumed radiation reading in pool area

** Assumes worst-case of dry MSB. If water is left in MSB, dose rate will be less than half of the value reported here.

Table 10.4-1 - VSC-24 5x5 ISFSI Total Annual Doses (mrem) (Direct Radiation)

Distance from ISFSI* (meters)	Bounding Case			Typical Case		
	Cask Side Direct Dose Contribution	Cask Top Direct Dose Contribution	Total Direct Dose	Cask Side Direct Dose Contribution	Cask Top Direct Dose Contribution	Total Direct Dose
100	2,806	770.9	3,577	128.0	43.97	172.0
120	1,491	509.2	2,000	68.04	29.04	97.08
150	803.5	287.5	1,091	36.68	16.40	53.08
200	351.2	120.1	471.3	16.03	6.853	22.88
300	68.21	21.14	89.35	3.112	1.206	4.318
400	21.17	4.081	25.25	0.966	0.233	1.199
500	6.658	0.789	7.447	0.304	0.045	0.349
600	2.110	0.158	2.268	0.096	0.009	0.105
700**	2.110	0.158	2.268	0.096	0.009	0.105
800**	2.110	0.158	2.268	0.096	0.009	0.105

* Measured from the surfaces of the ISFSI front row casks.

** The 600 meter calculated dose rate values are conservatively assumed for the 700 and 800 meter distances.

Table 10.4-2 - Atmospheric Release Dose vs. Distance (mrem) (Bounding Case)

Distance (meters)	Single Normal Cask	24 Normal Casks	Single Off-Normal Cask	25 Cask ISFSI Total
100	28.13	675.1	229.1	904.2
120	17.36	416.6	141.4	558.0
150	11.83	283.9	96.36	380.3
200	7.866	188.8	64.06	252.9
300	3.704	88.90	30.17	119.1
400	2.233	53.59	18.18	71.77
500	1.457	34.97	11.87	46.84
600	1.172	28.13	9.544	37.67
700	0.944	22.66	7.689	30.35
800	0.684	16.42	5.568	21.99

Table 10.4-3 - Atmospheric Release Dose vs. Distance (mrem) (Typical Case)

Distance (meters)	Single Normal Cask	24 Normal Casks	Single Off-Normal Cask	25 Cask ISFSI Total
100	2.736	65.66	18.51	84.17
120	1.689	40.54	11.42	51.96
150	1.151	27.62	7.786	35.41
200	0.765	18.36	5.176	23.54
300	0.360	8.640	2.438	11.08
400	0.217	5.208	1.469	6.677
500	0.142	3.408	0.959	4.367
600	0.114	2.736	0.771	3.507
700	0.092	2.208	0.621	2.829
800	0.067	1.608	0.450	2.058

Table 10.4-4 - VSC-24 5x5 ISFSI Overall Annual Doses (mrem)
(Direct Radiation + Atmospheric Release)

Distance from ISFSI* (meters)	Bounding Case Dose**	Typical Case Dose**
100	4,481	256.2
120	2,558	149.0
150	1,471	88.49
200	724.2	46.42
300	208.5	15.40
400	97.02	7.876
500	54.29	4.716
600	39.94	3.612
700	32.62	2.934
800	24.26	2.163

* Measured from the surfaces of the ISFSI front row casks.

** An estimate of the maximum reduction in ISFSI doses that could be achieved by placing a wall or berm around the ISFSI can be obtained by subtracting the cask side direct dose contributions shown in Table 10.4-1 from the total ISFSI doses shown above.

Table 10.4-5 - Estimated Minimum Controlled Area Boundary Distances for a 5x5 VSC-24 Cask Array (ISFSI)*

	Minimum Distance (meters)**
Bounding Fuel Payload	800
Typical Fuel Payload	255

* Lowest distances for which dose rates are under 25 mrem, based upon the dose rate data in Table 10.4-4.

** The minimum site boundary distances may be reduced somewhat by placing a wall or berm around the ISFSI, as discussed in Section 10.4.1, and in the notes to Table 10.4-4.

11.0 ACCIDENT ANALYSIS

The analyses of the off-normal and accident design events are presented in this section. Section 11.1 covers the off-normal events that are expected to occur during the life of the cask, possibly as much as once per calendar year. Section 11.2 covers the unexpected events that might occur over the lifetime of the cask or hypothetical events that are postulated because their consequences may result in the maximum potential impact on the immediate environment. Table 11.0-1 lists the events that were addressed and the VSC-24 components affected by each event.

Due to the generic nature of this report, the majority of the analyses on the accidents listed above are based on overly conservative assumptions and methodologies. Because of this overly conservative approach, the actual response of the VSC system to the events discussed in this section is expected to be much better than reported (i.e., lower stresses, temperatures, and radiation doses). Indeed, if required for site-specific applications, more detailed site-specific analyses could be used to extend the envelope defined by the events analyzed in this section.

11.1 OFF-NORMAL EVENTS

This section covers design events of the second type; events that might occur with moderate frequency on the order of once during any calendar year of operations.

11.1.1 OFF-NORMAL, SEVERE ENVIRONMENTAL CONDITIONS

11.1.1.1 Cause of Event

Many regions of the United States where nuclear power plants are located are subjected to sustained summer temperatures in the 90 to 100°F range and winter temperatures that are significantly below zero. Therefore, to bound the expected steady state temperatures of the cask during these periods of extreme ambient conditions, analyses were performed to calculate the steady state cask, MSB and fuel temperatures for a 100°F ambient with 24 hour average solar loads and for -40°F ambient with no solar load. The maximum thermal payload of 24 kW was also used for this analysis.

The 100°F ambient condition represents the highest steady state temperatures that could reasonably be obtained if a freshly loaded cask were subjected to 100°F ambient conditions continuously for three to four days. Since one would not normally expect ambient conditions at any nuclear plant in the United States (with the possible exception of Palo Verde) to be subject to such conditions, we have classified this situation as an off-normal event and used the analysis to bound expected short-term maximum fuel temperatures and short-term maximum concrete temperatures.

Likewise, one would not reasonably expect continuous -40°F temperatures for the three to four days needed for the cask to reach steady state. However, the condition is analyzed to provide an evaluation of the cask's response to lower temperatures. While the temperatures for the -40°F case shown here do not represent the absolute lowest temperatures that could occur for a -40°F condition (due to use of the maximum thermal load), the results do show expected temperatures early in the cask storage life. If further information is needed on the minimum temperatures as a function of time, the information presented in Figure 4.4-7 showing temperatures versus time can be used to evaluate the effects of cold ambient conditions later in the cask's life. In any regard, the coolest it could ever get would be -40°F and this is within the allowable temperature range for the materials used in the cask (although from the human standpoint one would probably not perform outside handling operations during such severe cold conditions).

11.1.1.2 Detection

Detection of off-normal ambient temperatures is not necessary because there are no consequences (i.e., the cask is designed to withstand such off-normal events). However, this event would be detected by the normal weather monitoring which is required at all nuclear power plants.

11.1.1.3 Analysis

The analysis of off-normal ambient temperature uses the thermal models described in Chapter 4. The same models and calculations used for the normal conditions were used to model the -40°F and 100°F ambient conditions. The maximum steady state temperatures for both cases and the minimum temperatures for the -40°F case are:

Component	Max. Temp. 100°F Case	Max. Temp. -40°F Case	Min. Temp. -40°F Case
Fuel Cladding	705°F	595°F	----
MSB Shell	294°F	162°F	18°F
Concrete Liner	214°F	41°F	0°F
Concrete Surface	136°F	-32°F	-39°F

Figures 11.1-1 and 11.1-2 provide the details of the temperature distributions. As these figures show, the temperatures are within the code allowables for the concrete off-normal short-term events. The thermal gradients across the concrete were less than the normal case so additional stress analysis was not required. The thermal gradients across the MSB interior were higher for the -40°F case than for all other cases, and the stress analysis for this case was described previously in Chapter 3.

11.1.1.4 Corrective Actions

The VSC cask system is designed to accommodate steady state 100°F (with the design basis solar loads) or -40°F (with no solar loads). No corrective actions are required.

11.1.2 BLOCKAGE OF ONE-HALF OF THE AIR INLETS

11.1.2.1 Cause

This event is a postulated blockage of one-half of the air flow inlets. Because the VSC has four independent air inlets located on two opposing sides, it is not considered feasible that all vents could become blocked by blowing debris, snow, animals, etc. during normal operation (nevertheless, this accident is considered in Section 11.2.9). Off-normal events (tornado, high winds, heavy snow, earthquake, etc.) that could cause significant debris or other potentials for blockage require surveillance action that would remove any blockage within one week. (See the surveillance requirements in Section 12.4).

11.1.2.2 Detection

This event would be detected visually by the security force as they perform their required weekly surveillance.

11.1.2.3 Analysis, Effects, and Consequences

The analysis of this event uses the air flow model described in Section 4.4.1.1. Blocking two of the inlets reduces the inlet area by a factor of two which increases the k/A^2 for the entrance by a factor of four. However, the VSC flow system is designed so that the inlet losses are a relatively small portion of the total pressure drop due to the air flow. Hence, the increase in the total $\Sigma k_i/A_i^2$ is about 53%. This reduces the air mass flow rate by 15%. These combined effects (one increasing the pressure loss and one decreasing the pressure loss) increase the overall pressure loss due to the air flow by 14%. The reduced air flow creates a higher $\underline{\Delta}T$ between the inlets and outlets to balance the higher flow pressure losses. The result is roughly a 17% rise in the temperature difference so that the air exiting temperature rises from 164°F to 179°F. This rise has a negligible effect on the VSC operation. Also, as shown in Table 4.1-1 the thermal stress analysis for this case is bounded by the analysis presented in Chapter 3.

Since the dose rates at the air inlets are higher than the nominal rate at the cask wall, workers will be subject to above normal dose rates when clearing the vents. As a worst-case estimate, it is assumed that a worker kneeling with his hands on the vent inlets requires 30 minutes (actual kneeling time) to clear the vents. Hence, the estimated dose is 175 mrem to the hands and forearms and slightly less to the chest and body.

11.1.2.4 Corrective Actions

The required action when a vent or vents are found to be blocked is to remove the debris, snow, sand, or other foreign material blocking the air intakes. Since screens are provided for all the vents, any blocking material will be on the outside and, hence, may be removed by hand or hand-held tools.

11.1.3 INTERFERENCE DURING MSB LOWERING FROM TRANSFER CASK INTO CONCRETE CASK

11.1.3.1 Cause of Event

If the MTC and VCC are not properly aligned, or if some foreign material is present in the VCC or MTC, the MSB may not lower freely as the crane hook holding the MSB is lowered. While proper procedures to ensure cleanliness of the MTC and VCC, leveling the VCC and MTC, and securing the alignment pins should prevent this condition from arising, it is nevertheless analyzed here to bound any potential off-normal occurrence.

11.1.3.2 Detection

This event would be detected by audible noises emitted from the MSB sliding on the MTC, VCC, or other material and in the worst case by a slackening of the wire slings which connect the MSB to the crane hook. If either of these events should occur, the technician or the engineers present should immediately have the crane operator stop lowering the MSB and raise the MSB back into the transfer cask.

11.1.3.3 Analysis of Effects and Consequences

Since the only forces acting on the MSB during its lowering are gravity, the worst-case condition would be a load of 1g on the MSB bottom or side if it were to be completely supported from its interference. Since the MSB has been designed for much more severe loading conditions than this (i.e., the drop accident described in the next section), the interference during transfer will not cause any undue stresses in the MSB. Furthermore, the MTC and VCC are both sufficiently strong to support the weight of the MSB so no over-stressed conditions will exist in these components. Also, unless the VCC and MTC shift, the MSB will not become jammed since it is only being lowered by gravity. Hence, if severe interferences occur, the lifting slings will go slack, and the MSB will be supported by the interference. In any regard, as described in Chapter 3, the lifting bolts and slings which connect the MSB to the crane hook are designed with a design factor of 10, so they can easily take the maximum loads which the crane (100 to 125 tons) can exert in pulling the MSB back into the transfer cask. If a shift of the VCC and MTC does occur, they can be re-leveled via the jacks on the trailer holding the VCC.

If an interference occurs, additional radiation dose will be picked-up by the cask technicians. As an estimate of this additional dose, it is assumed two technicians will work within 1 meter of the transfer cask for an additional half hour. This results in a dose to each technician of less than 120 mrem.

In summary, an interference during the MSB transfer from the MTC to the VCC will not cause any loading conditions more severe than those analyzed for other conditions. Therefore, this off-normal condition does not require further structural, thermal hydraulic or nuclear analysis.

11.1.3.4 Corrective Actions

The corrective actions necessary for this off-normal condition are the following:

1. Immediately halt lowering the MSB.
2. Raise the MSB back into the transfer cask.
3. Check alignment of the VCC and MTC.
4. Check level of VCC and MTC.
5. Retry lowering.
6. If interference still exists, pull MSB back into MTC and remove MTC.
7. Check VCC for foreign objects.
8. Check MTC-MSB gap for foreign objects (return to pool if necessary).

The above corrective actions are presented to provide overall guidance on how to handle MSB/MTC/VCC transfer interferences. More detailed, site-specific procedures will have to be developed taking into account the actual auxiliary building layout, handling equipment, and other site-specific details.

11.1.4 SMALL RELEASE OF RADIOACTIVE PARTICULATES FROM THE MSB EXTERIOR

11.1.4.1 Cause of Event

Although precautions are taken to avoid introducing contamination to the outside of the MSB when it is submerged in the spent fuel pool, it is possible that a portion of the MSB may become slightly contaminated. If this surface contamination is not detected prior to outdoor storage in the VSC, the particulate may become airborne and drift off-site.

11.1.4.2 Detection

This surface contamination should be detected prior to transferring the MSB to the VSC by routine health physics smears on the MSB top and sides prior to transfer to the VSC. However, if high levels of surface contamination do exist on an MSB in the VCC in storage and they are released, they would only be detected by the long-term radiological instrumentation (TLDs) on the site fence. The releases will be too low (as shown below) to measure or set-off any dose rate measurement instrumentation.

11.1.4.3 Analysis of Effects and Consequences

If the surface contamination was somehow missed, the worst consequence would be that it becomes loose after the MSB is placed in the VCC and the VCC is placed on the storage pad. For such an atmospheric release, one may conservatively assume that the particulate behaves as a gas. The off-site dose can then be calculated using the methods described in Regulatory Guides 1.25 and 1.109. The release parameters are a wind speed of 1 m/sec, and atmospheric dispersion factor provided in Figure 1 of Regulatory Guide 1.25. The equation adapted from the Regulatory Guide is:

$$\text{Dose rate (mrem/hr)} = DF \cdot X \cdot Q$$

Where:

DF = Dose conversion factor (mrem - m^3/hr - Ci)
Adopted from Regulatory Guide 1.109 (Body and Skin)

X = Atmospheric dispersion factor (sec/m^3)

Q = Release rate (Ci/sec)

Based on this equation, a person continuously standing at the vertical centerline during the entire passage of the release at a distance of 200 meters from the release point would receive a total dose of one mrem (gamma) from a 2.32 Ci release (modeled as ^{60}Co). If this release was originally in the form of particulate evenly distributed on the external surface of the MSB, the original surface contamination would have to be 2.1 x 10^7 dpm/cm^2, an unrealistically high level of surface contamination originating from a well maintained spent fuel pool.

The above analysis demonstrates that the off-site radiological consequences from the release of surface contamination on the MSB is negligible.

11.1.4.4 Corrective Actions

No corrective action is required since the radiological consequence is negligible (addition of 1 mrem/year if a person were to stand at a close (200 m) site boundary during the entire release).

11.1.5 MSB OFF-NORMAL HANDLING LOAD

11.1.5.1 Cause of Event

The off-normal handling load is postulated to result from improper handling of the MSB during loading, such as an inadvertent lateral or vertical crane motion. The two hypothetical events considered are 1) an impact of the MTC loaded with the MSB against a structure, such as the storage cask or a structure within the Auxiliary Building, and 2) an impact of the MSB resulting from lowering it at an accelerated rate into the storage cask.

11.1.5.2 Detection

Personnel performing the operations would detect an impact such as those considered for the off-normal handling load.

11.1.5.3 Analysis

Two off-normal scenarios have been considered. (See Table 1.2-1).

1. The MSB impacts an object while inside the transfer cask. The postulated maximum impact velocity is 2 ft/sec, which corresponds to a drop height of 0.75 inches.
2. The MSB is lowered into the storage cask at an accelerated rate and impacts the ceramic tiles at the bottom of the VCC at a velocity of 0.75 ft/sec, which corresponds to a drop height of 0.1 inches.

Based on an assumed target hardness and values of decelerations for drop heights ranging from 0 to 30 inches, the decelerations for MSB impact at 2 ft/sec and at 0.75 ft/sec are interpolated to be 4.0 *g* and 0.6 *g* respectively.

To obtain equivalent static loads, a conservative dynamic load factor of 2 has been applied to the decelerations. Stress for the off-normal impact is calculated by scaling either the horizontal drop stress or the dead weight stress. A stress-to-acceleration ratio is calculated for each of the two load cases, using the vertical or horizontal acceleration, as applicable. The maximum of these two ratios is multiplied by the off-normal handling deceleration to produce the off-normal handling stress. The off-normal handling stress is taken as the greatest of three possible handling cases:

1. Normal Handling
2. Off-Normal Handling – MSB uniformly supported in transfer cask (2 ft/sec impact)
3. Off-Normal Handling – MSB supported on ceramic tiles (0.75 ft/sec impact)

Stresses resulting from the above off-normal impacts are combined with other stresses present during an off-normal handling event and are evaluated in Table 11.1-1.

11.1.6 OFF-NORMAL PRESSURIZATION

11.1.6.1 Cause of Pressurization

The cause of off-normal pressurization could be a hypothetical breach of 10% of the fuel rods and control elements in the MSB and subsequent release of their fission and fill gases to the interior of the MSB. This hypothetical breach would pressurize the MSB shell and structural lids.

11.1.6.2 Analysis of Off-Normal Pressurization

The analysis of this event entails calculation of each of the factors that cause this pressurization for each of the MSB versions and fuel types permitted. This pressurization results from a combination of two factors:

1. Release of pressurized fuel rod and control element fill gases into the MSB cavity
2. Thermal expansion of the combined gas content of the MSB due to the temperature increases resulting from:
   a. normal operation
   b. reduction of conductivity caused by dilution of helium by fission gases

For the off-normal pressurization case, the release of fission gases from the failed fuel rods causes a reduction in the conductivity of the basket interior gas, which in turn causes the basket temperature to rise. Because pressure is a direct function of temperature, the off-normal temperatures and pressures are calculated. The analysis assumptions and results for the highest pressure case are summarized:

- The minimum free volume in the MSB, including 10% of the rod and control element volume, is 334,300 in^3.
- For the pressure calculation, it is assumed that 100% of the fission gases (55% helium and 45% fission gas) from each failed fuel rod and control element (10 % of the total fuel rods and control elements) are released. Conservatively assuming that the fission gas is 100% xenon, the basket gas conductivity would fall to about 80% of the pure helium value. The effect of the basket gas conductivity on the interior temperatures is determined through linear interpolation between the helium and vacuum case thermal analysis results. The total gas content of the MSB, including helium backfill and released fuel rod and control element fill gas is 223 moles.
- For off-normal conditions, the average basket interior temperature is shown to increase from 434°F, the bounding basket average gas temperature during normal operation in the MTC, to 436°F. For conservatism, the maximum basket pressure under off-normal storage conditions is calculated using a gas mixture at a bulk temperature of 445°F.

- Using the ideal gas law, the maximum basket pressure under off-normal storage conditions is 10.0 psig.

An internal pressure of 10.0 psig is used in the off-normal pressure stress analysis to calculate off-normal pressure stresses. The off-normal pressure stresses are combined with dead weight stresses, normal handling stresses, and thermal stresses in Table 11.1-2. As shown in the table, all stresses are within the Code allowables for Service Level B Loads. The off-normal pressure stresses are combined with dead weight stresses and off-normal handling stresses in Table 11.1-1. As shown in the table, all stresses are within the Code allowables for Service Level C Loads.

11.1.6.3 Radiological Consequences

The radiological consequences for this off-normal condition, which involves the breach of 10% of the fuel rods within the MSB, are evaluated in Sections 7.2 and 10.4. The off-normal event doses from a failure of 10% of the fuel rods within a single MSB would not exceed the limits established by 10CFR72.104(a).

11.2 ACCIDENTS

This section provides the results of analyses of several hypothetical accidents, which are presented to show that the VSC system has substantial safety margin to provide more than adequate protection to both the public and occupational personnel. In addition to these design basis accidents, this section also provides the results of analysis of bounding phenomena that could occur over the life of the cask (tornado, earthquake, floods).

11.2.1 MAXIMUM ANTICIPATED HEAT LOAD

11.2.1.1 Cause of Accident

This "accident" is a natural phenomena, 125°F ambient temperature and 14 hours of full solar loads occurring when a design basis thermally loaded cask is first placed in service.

The accident was analyzed per ANS 57.9 to determine the worst-case accident thermal loads (T_a) to be used in the load combinations. The accident is also analyzed to show that even under these extreme heat load conditions (which could only occur once due to the decay of the heat source with time), the short-term fuel cladding temperature limit of 1058°F (570°C) and the short-term concrete temperature limit of 350°F (177°C) are not violated.

11.2.1.2 Accident Analysis

The accident assumes that the cask has been exposed to 125°F ambient conditions with solar loads (full on for 12 hrs, off for 12 hrs) for a sufficiently long time to reach steady state. The calculation is performed using the methodology described in Section 4.4 and employed for all other postulated conditions.

The analysis results are presented in Table 4.1-1. None of the short-term material temperature limits are reached. Furthermore, due to the higher temperature increase (due to full solar loads) on the cask surface, the differences across the concrete wall (and, hence, the stresses) are lower for this accident condition than for the normal (75°F ambient) case, and no additional thermal stress analysis is required.

11.2.1.3 Accident Dose Calculation

There are no dose implications due to this accident.

11.2.2 MSB DROP ACCIDENT

11.2.2.1 Cause of Accident

A hypothetical drop accident for the MSB is evaluated to demonstrate the MSB's ruggedness and structural capability. The accident is a cask drop of 5 feet onto an essentially unyielding surface. The maximum deceleration on the cask and its contents (i.e., the filled MSB) are 60 *g* for a vertical drop and 22 *g* for a horizontal drop (e.g., with a dynamic amplification factor of 2, the equivalent deceleration used in the static analysis would be 120 *g* and 44 *g*, respectively). Any secondary impact is assumed to be bounded by these two decelerations.

11.2.2.2 Accident Analysis

The critical components of the MSB were modeled using the ANSYS computer code. An equivalent static analysis was performed for both the horizontal and vertical drop orientations. For the horizontal drop, the fuel storage sleeves, the basket supports, and the MSB shell and structural lids were the critical components. For the vertical drop, the MSB shell, the structural and shielding lids, and the shield lid support ring were the critical components. The following sections describe the models, analysis, and the results.

Horizontal Drop

Two separate models were developed for the horizontal drop. The first model is the storage sleeve assembly. The critical column of storage sleeves and the lower basket support were modeled. This model is shown in Figure 11.2-1. The model is grounded at the three support plate locations. The weight of the fuel assemblies and the storage sleeves were imposed on the lower section of each sleeve. Since the load was applied statically, a dynamic amplification factor must be used for the analysis. The maximum possible factor of 2 was used. The load of 44g's was, therefore, applied, and the stresses were calculated.

The results of the analysis indicate that the maximum elastic deflection is 0.12 inches. However, the maximum equivalent stress of 57 ksi exceeds the material yield strength of 28.1 ksi. Therefore, this displacement is not valid because plastic deformation is not considered in the model. The following method was used to conservatively estimate the real deflection.

The SA-516 Grade 70 steel was assumed to be an ideal elasto-plastic material. The deflection was calculated by the balancing of the absorbed energies for the elastic and elasto-plastic models. For an elastic deflection of 0.12 inches, a calculated stress of 57.0 ksi, and a yield stress of 28.1 ksi, the approximate actual deflection is expressed as follows:

$$\delta = 0.15 \text{ in}$$

Because this is much less than the 0.5 inch gap between the sleeve and fuel, no interference will occur. Furthermore, the calculated stress is lower than the ultimate stress of the SA-516 Grade 70 steel, and sleeve assembly will not fail. Thus, the design meets the criteria specified in Table 2.2-4.

The stresses in the MSB body were calculated by an equivalent static analysis using the model shown in Figure 11.2-2. Table 11.2-1 shows the resulting stresses due to the horizontal drop event in combination with other applicable loads. As shown in this table, all stresses are below the ASME Service Level D allowable stresses and only a small amount of permanent deformation will take place. Therefore, there are no criticality, containment, dose rate, or retrievability concerns.

Vertical Drop

Two separate ANSYS finite element analyses are used in the evaluation of the MSB stresses due to the vertical drop accident event, each applying the drop acceleration of 120 *g* statically. The first ANSYS analysis models the top region of the MSB and includes the shield and structural lids, the welds, and the upper portion of the shell. The second ANSYS analysis models the lower 30 inches of the shell as well as the MSB base. The modeled lower region of the MSB shell is supported around its periphery on ceramic tiles. For the bottom end analysis, the same ANSYS finite element model described in Section 3.4.4.1.2 for the dead weight analysis is used. As discussed in Section 3.4.4.1.2, two tile configurations are permitted in the VCC. However, the tile support conditions considered in the finite element model result in bounding stresses. Because the MSB is supported at discrete locations by a ring of ceramic tiles, and not supported uniformly under its base, local stress concentrations occur. Since some local yielding occurs, the ANSYS model includes large deformation effects and plastic behavior. Accordingly, plastic-system allowable stress design criteria are used in accordance with Paragraph F-1331.1 of the ASME Code. A separate analysis modeling the MSB uniformly supported on its base (i.e., no ceramic tiles) is used to calculate vertical drop stresses without use of a finite element model.

The most limiting stresses resulting from the vertical drop are combined with other stresses present during this accident condition and compared with ASME Code allowables in Table 11.2-2. All stresses are demonstrated to be within allowable values.

Furthermore, as Figure 11.2-4 shows, a vertical ground displacement of one corner of the cask by approximately 5.0 feet would be required to move the center of gravity over the corner of the cask so that the cask would topple. The maximum height at which the VCC rests during all transfer operations is 2 feet, the height of the trailer bed. Therefore, the tip-over event is bounded by the above drop analyses.

11.2.2.3 Accident Dose Calculation

As can be seen by the comparison of the stress intensities to the allowables, the MSB will not fail under the drop or tip-over accident evaluated. Therefore, there will not be any release of radioactive material or additional dose associated with either the vertical or horizontal drop accident or a tip-over accident.

11.2.3 TORNADO

11.2.3.1 Cause of a Tornado

The probability of a tornado at any particular ISFSI is dependent on its geographic location. For many sites in the U.S., the probability is such that one could reasonably expect such an event during the life of the ISFSI. The effects of a tornado on the VSC concrete cask include the possibility of damage due to wind loading, wind-generated pressure differentials, and tornado-generated missiles. Table 2.2-1 defines the design-basis, tornado-generated missiles. Possible damage modes would include toppling due to wind loading, failure of confinement due to pressure differential and impact damage due to tornado-generated missiles.

11.2.3.2 Tornado Accident Analysis

The cask is designed to withstand loads associated with the most severe meteorological conditions, including extreme wind and tornado, which are postulated to occur at an ISFSI site. Tornado design parameters used to evaluate the suitability of the cask include tornado winds, wind-generated pressure differentials and tornado-generated missiles. The design-basis tornado characteristics have been selected consistent with Regulatory Guide 1.76.

The methods used to convert the tornado and wind loadings into forces on the cask are based on NUREG-0800, 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 3.5.3, *Barrier Design Procedures*.

Wind Loads

The potential for the tornado wind load to slide the cask on the ground and to tip over the cask is evaluated. The tornado wind velocity is transformed into an effective pressure on the cask using the approach in ASCE 7-93, Minimum Design Loads for Buildings and Other Structures.

Results of the evaluations are as follows:

Applied wind load	42,031 lbf
Minimum wind load to slide cask	50,000 lbf
Applied overturning moment	4.88×10^6 in-lbf
Minimum overturning moment to tip cask	15.7×10^6 in-lbf

Thus, the tornado wind load is not sufficient to slide the cask on the ground or to tip over the cask.

The stress in the VCC due to wind pressure is not analyzed because (1) the tornado wind load and the live load produced by the weight of the MTC and fully-loaded MSB are not coincident, and (2) the wind-induced stress is bounded by the analysis of the above MTC/MSB live-load stress.

Tornado Missiles

The cask is designed to withstand the effects of postulated tornado-generated missile impacts in accordance with NUREG-0800, Section 3.5.1.4.III.4. These missiles consist of a massive, high-kinetic-energy missile that deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers. All missiles are assumed to impact in a manner that produces the maximum damage to the cask. The cask has been evaluated for the effects of impacts associated with each of the missiles described above. Analyses for penetration resistance of the cask body and closure elements to the armor-piercing-artillery-shell indicate that sufficient thickness of concrete and steel is available to prevent perforation, spalling or scabbing of the various cask boundary elements. Overall response of the cask has been evaluated for impacts associated with the high-energy deformable missile. Such analyses indicate that the cask will remain upright following the event, and that loads associated with this impact do not compromise the integrity of the cask. The analyses that have been conducted are summarized below. In addition, the shear force and moment are included in the load combination evaluation in Table 3.4-6.

Local Damage Prediction - Cask Body

Local damage of the cask body has been assessed using the National Defense Research Committee (NDRC) formula. This formula has been selected as the basis for predicting depth of penetration and minimum thickness of concrete to prevent spalling and scabbing. Penetration depths computed by this method have been shown to provide reasonable correlation with test results. (Ref: EPRI Reports NP-440 and NP-1217). The depth of penetration, as predicted using this approach, is 5.69 inches.

The minimum depth of concrete necessary to preclude spalling and scabbing is then selected as three times the depth of penetration predicted using the NDRC formula, or 17.1 inches. Because the 29-inch thickness of concrete in the cask body is well in excess of this value, adequate protection is provided for local damage due to tornado missiles.

Local Damage Prediction - Cask Closure Plate

The VCC is closed with a 0.75-inch-thick steel plate bolted in place. By calculating the perforation thickness of a 126 mph, 275 lb., 8-inch diameter armor-piercing artillery shell impacting a steel plate, the ability of the closure plate to adequately withstand tornado generated missiles is established.

The perforation thickness, T, in a steel plate is calculated in accordance with Reference 11.3, with the result that

$$T = 0.52 \text{ inch}$$

Therefore, the cask closure plate is adequate to withstand local impingement damage due to tornado-generated missiles.

Potential Cask Tip-over from Missile Impact

Since the cask is a freestanding structure, the principal consideration in overall damage response is the likelihood of upsetting or overturning of the cask as a result of high-energy missile impacts. Such assessments have been conducted using the principles of conservation of momentum during the impact event. The analyses that are summarized below indicate that the cask will remain upright.

From the principles 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. Likewise, the impulse force due to the impact of the missile must equal the change in linear momentum of the missile. Figure 11.2-3 illustrates the missile impact on the top corner of the cask. The missile is assumed to be a car, because this postulated missile has a great momentum. It is assumed that the velocity of the missile after impact is zero and that there is a perfectly elastic collision between the missile and the cask, i.e., there is no energy loss due to the deformation of the missile or spalling of the concrete. The kinetic energy of the cask as a result of the impact compared to the energy required to tip over the cask is as follows:

Cask energy due to missile impact	74,163 ft-lbf
Minimum energy to tip over the cask	333,412 ft-lbf

Thus, the postulated missile impact does not tip over the cask.

Overall Damage Prediction

The potential for a shear or bending failure of the VCC was evaluated for an automobile impact. The shear evaluation was at the air outlet, which is the minimum strength section. The results of this evaluation are as follows:

Force due to missile impact	457,380 lbf
VCC concrete shear capacity	1,110,000 lbf
Moment due to missile impact	91.5×10^6 in-lbf
VCC liner bending capacity	223.6×10^6 in-lbf

Thus, the VCC does not fail in shear or bending due to the missile impact.

These results are evaluated in combination with other loads in Table 3.4-6.

Combined Tornado Wind and Missile Loading

The effects of tornado winds and missiles have been considered both separately and combined in accordance with NUREG-0800, Section 3.3.2.II.3.d. For the case of tornado wind plus missile loading, the stability of the cask has been 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 3.4 degrees. Applying the total tornado wind load to the cask in this configuration results in a restoring moment on the cask of 8.96×10^6 in-lbs. Hence, overturning of the cask under the combined effects of tornado winds plus tornado-generated missiles is not postulated to occur.

MSB Under Tornado Loadings

Since the postulated tornado loadings are not capable of overturning the cask they have no effect on the MSB.

11.2.3.3 Tornado Accident Dose Calculations

Under the worst tornado missile impact, a penetration of 5.69 inches into the cask side wall was calculated. The dose rate 5.69 inches back into the concrete shield is just under 10 times the dose rate at the surface, or roughly 1 rem/hr. If the thickness of the entire concrete shield were reduced by 5.69 inches, as opposed to just a local area, the dose rate from a single cask at the minimum allowed controlled area boundary distance of 100 meters would be increased by a factor of ten, from 0.0437 mrem/hr to 0.437 mrem/hr.

It is assumed that the cask wall concrete damage will be repaired within 30 days of the missile impact. Over this accident event duration, the dose rate of 0.437 mrem/hr yields an overall exposure of approximately 300 mrem, which is less than the 10CFR72 accident condition dose limit of 5 rem/hr by more than one order of magnitude.

To repair the cask, it is assumed that it takes two technicians 30 minutes to fill the damaged area with grout. The workers are assumed to be standing about one meter from the cask. If the dose rate over the entire cask surface were 1000 mrem/hr, the dose rate would fall to ~500 mrem/hr. However, due to the local nature of the concrete damage, the dose rate one meter from the cask is less than 200 mrem/hr. Therefore, the man-rem dose for the repair process is under 200 mrem (100 to each technician).

11.2.4 FLOOD

11.2.4.1 Causes of Flood

The probability of a flood event is specific to each ISFSI site. However, at most reactor sites in the United States a location could be found where a flood was not credible. Nevertheless, the flood analysis is presented here to bound any worst-case flood. Two different types of floods are analyzed. The first is a worst-case fully immersing flood that might move or tip-over the VSC. The second is a small flood that only blocks the air inlets (this case is discussed in Section 11.2.7).

11.2.4.2 Flood Analysis

Immersing Flood Analysis

The stream velocity required to overturn the loaded VCC is determined by equating the overturning moment due to the flood drag force to the restoring moment. The cask is assumed to be fully immersed (i.e., at a flood depth exceeding the maximum VCC height of 225.05 inches) with steady-state flow perpendicular to the cask. The minimum restoring moment is equal to the buoyant weight of the loaded VCC times the radial distance to the point of rotation, considering the 3-inch chamfer at the base of the VCC. The cask's buoyant weight is conservatively calculated based on a lower bound weight of 250 kips for the loaded VCC, less the weight of water displaced by a solid cylinder with the same height and diameter as the tallest VCC (111.2 kips). The drag force necessary to tip the immersed cask, based on a bounding drag coefficient of 1.2 for steady flow conditions, is 73,375 pounds. The stream velocity necessary to produce this drag force is 17.7 fps. The stresses in the VCC due to flood drag force are negligible.

MSB Flood Loading

To demonstrate compliance with the applicable allowable stress and buckling design criteria, the MSB has been evaluated for external pressure loading resulting from the design basis flood.

Section 11.2.6 presents an analysis of the MSB shell stresses under an accident pressure of 60 psig. During this postulated flood accident, the minimum MSB internal pressure of −8.0 psig is assumed to be coincident because of the cooling effects of water immersion. The combination of this negative internal pressure and a flood of 120 ft would create an equivalent net pressure load of 60 psig on the MSB shell and equivalent bending stresses in the end closure plates of the MSB. As shown in Section 11.2.6, the maximum stresses in the MSB-shell end closure plates due to a 60 psig pressure load are lower than the applicable allowable stresses. Therefore, the

MSB-shell end closure plates are capable of withstanding external pressure loading from a 120-foot flood depth.

The MSB shell is also separately analyzed for buckling due to external pressure. As described in Section 3.4, the allowable external pressure differential for the MSB shell to prevent buckling is 210 psig. Furthermore, the flood height required to develop this pressure, less the assumed coincident internal pressure of –8.0 psig, is greater than 460 feet. Therefore, the MSB shell will not buckle for a flood depth of 460 feet or less.

Thus, based on the results of the stress and buckling analyses above, the MSB is structurally adequate to withstand external pressure loads due to a maximum flood height of 120 feet.

11.2.4.3 Flood Dose Calculations

Flooding would not cause damage that could increase the dose rate outside the cask.

11.2.5 EARTHQUAKE EVENT

11.2.5.1 Cause of Earthquake

Earthquakes are natural phenomena which the cask might be subjected to at any U.S. site. The design basis seismic event is described and discussed in Section 2.2.5.

11.2.5.2 Earthquake Analysis

The VSC is a very stiff structure. Although freestanding, it has been analyzed as a cantilever fixed at the base (Roark and Young, Formulas for Stress and Strain, 5th Edition, Table 36, Case 3b). For the purpose of calculating seismic loads, the cask is treated as a rigid body attached to the ground and equivalent static analysis methods were used to calculate loads, stresses, and overturning moments.

The fundamental natural frequency of vibration for the cask is expressed as follows:

f_n = 42.1 cycles per second

As stated in Regulatory Guide 1.60, the dynamic amplification factor for this frequency is 1.

The VSC has been evaluated statically for overturning by conservatively applying equivalent static loads to the cask in each of two orthogonal horizontal directions simultaneously with an upward vertical component. The total seismic overturning load has been calculated by multiplying the maximum ground acceleration in each of the two horizontal directions by the weight of the fully loaded cask and combining the resulting component forces on the basis of the square root of the sum of the squares.

The ratio of the restoring moment to prevent cask tip-over to the applied moment is expressed as follows:

Restoring Moment/Overturning Moment = 1.28

Since the ratio is greater than 1.0, the cask would not tip over due to a design basis seismic event.

In addition, the maximum kinetic energy that can possibly be imparted to the cask from seismic motion is compared with the energy required to overturn the cask.

Translation and rotation energy applied to cask due to earthquake	246,247 ft-lbf
Minimum energy to tip over cask	333,412 ft-lbf

Thus, the design basis earthquake would not tip over the cask.

Therefore, the worst possible combination of natural frequency, vibration mode, damping ratio, and loading combination will not impart sufficient kinetic energy to topple the VSC. The cask is concluded to be stable under seismic loads. The MSB stresses, therefore, are negligible and bounded by the vertical and horizontal drop analyses. The calculated internal stresses due to the seismic loading are:

Shear Stress	=	0.01 ksi
Normal Stress (due to bending moment)	=	0.05 ksi

These stresses have been evaluated in combination with the other loads in Table 3.4-6.

11.2.5.3 Accident Dose Calculation

As can be seen by the comparison of the stress intensities to the allowables, the MSB will not fail under the drop or tip-over accident evaluated. Therefore, there will not be any release of radioactive material or additional dose associated with the design basis earthquake accident.

11.2.6 ACCIDENT PRESSURIZATION

11.2.6.1 Cause of Pressurization

The cause of this accident could be a hypothetical breach of all the fuel rods in the MSB and subsequent release of their fission and fill gases to the interior of the MSB. This would pressurize the MSB shell and structural lids.

11.2.6.2 Analysis of Pressurization Accident

The analysis of this event entails calculation of each of the factors that cause this pressurization for each of the MSB versions and fuel types permitted. This pressurization results from a combination of two factors:

1. Release of pressurized fuel rod and control element fill gases into the MSB cavity
2. Thermal expansion of the combined gas content of the MSB due to the temperature increases resulting from:
   a. normal operation
   b. reduction of conductivity caused by dilution of helium by fission gases

For the accident pressurization case, the release of fission gases from the failed fuel rods causes a reduction in the conductivity of the basket interior gas, which in turn causes the basket temperature to rise. Because pressure is a direct function of temperature, the off-normal temperatures and pressures are calculated. The analysis assumptions and results for the highest pressure case are summarized:

- The minimum free volume in the MSB, including 100% of the rod and control element volume, is 343,500 in^3.
- For the accident pressure calculation, it is assumed that 100% of fission gases (55% helium and 45% fission gas) from each failed fuel rod and control element (100 % of the total fuel rods and control elements) are released. Conservatively assuming that the fission gas is 100% xenon, the basket gas conductivity would fall to about 30% of the pure helium value. The effect of the basket interior temperatures on the basket gas conductivity is determined through linear interpolation between the helium case and the vacuum case thermal analysis results. The total gas content of the MSB, including helium backfill and released fuel rod and control element fill gas is 643 moles.
- For this accident condition, the average basket interior temperature is shown to increase from 434°F, the bounding basket average gas temperature during normal operation in the MTC, to 440°F. For conservatism, the maximum basket pressure under accident storage conditions is calculated using a gas mixture at a bulk temperature of 460°F.
- Using the ideal gas law, the maximum basket pressure under accident storage conditions is 55.7 psig.

Conservatively, a bounding internal pressure load of 60.0 psig is used in the accident pressure stress analysis. The critical pressure stresses are combined with both the dead weight stresses and the normal handling stresses, as summarized in Table 11.2-3. As shown in this table, all values of stress are within the Code allowables for Service Level D Loads.

11.2.6.3 Radiological Consequences

The radiological consequences for this accident are evaluated in Sections 7.3 and 10.4. The accident doses for failure of all fuel rods within a single MSB does not exceed the limit established by 10CFR72.106(b).

11.2.7 FULL BLOCKAGE OF AIR INLETS

11.2.7.1 Cause

This event is a postulated blockage of the airflow inlets. Because the VSC has four independent air inlets located on two opposing sides and is thermally hot on its surface, it is not considered likely that all vents could become blocked by blowing debris, snow, animals, etc. during normal operation. However, to demonstrate that acceptable temperatures will be maintained even if the unexpected complete inlet blockage occurs, the following analysis was performed.

11.2.7.2 Detection

This event would be detected visually by the security and/or surveillance force as they perform their required weekly patrol.

11.2.7.3 Analysis of Event

The full inlet duct blockage scenario is conservatively evaluated using a time-dependent thermal analysis that assumes still air (i.e., zero convection) throughout the ventilation duct annulus. Thus, in this model, virtually all of the heat generated in the MSB must flow radially through the air in the duct via conduction, or pass directly from the MSB shell to the VCC liner via radiation. The heat must then pass through the entire concrete shield via conduction. Some additional heat also travels via conduction through the cask top end. Surface convection effects that enhance heat transfer into the duct air are not modeled; neither is heat removal from the duct air, via airflow out of the system. In this time-dependent analysis, some heat is removed via conduction out of the system, but over the timeframe considered, most of the heat is absorbed by the cask system materials, causing their temperature to increase with time.

The results of the time-dependent, zero-airflow thermal analysis described above show that the VCC concrete is the first system component to reach its temperature limit. The concrete reaches its accident condition temperature limit of 350°F in roughly 30 hours. At that time (30 hours), the fuel cladding remains well below its accident/short-term temperature limit of 1058°F, and the MSB shell temperature is roughly 450°F, well under the material temperature limits for carbon steel.

Based upon this analysis, it is concluded that even if all four inlets were completely blocked, shutting off all airflow within the duct, all of the VSC-24 system components would remain within their short-term (accident) temperature limit over a one-day (24 hour) period. For this reason, a daily (24-hour) interval is established for visual inspection of the cask inlet ducts, as shown in the *technical specifications* in Section 12.4.

11.2.7.4 Consequences of Event

The worst-case radiological consequences for this event will be twice the consequences of the blockage of one half the inlets (Sec. 11.1). This will be a small dose increment to the hands and arms and even smaller dose increment to the body caused by cleaning the vents. All cask system components will remain within their short-term temperature limits, given that the blockage will be removed within, at most, 24 hours (i.e., the surveillance interval). Thus, this event will not result in any release of radioactive material from the cask.

Table 11.0-1 - Design Basis Off-Normal and Accident Events

	Event	MSB Pressure Boundary	MSB Internals	VCC
Off-Normal Events (expected frequency 1/year)				
1.	Off-Normal Environmental Conditions (100°F w/Solar Load and -40°F, no solar load)	X	X	X
2.	Blockage of One-Half of the Air Inlets	X	X	X
3.	Interference During MSB Lowering From Transfer Cask Into Concrete Cask		Radiological consequences only	
4.	Small Release of Potential MSB Surface Contamination		Radiological consequences only	
5.	Off-Normal Handling Load a) Impact at 2 ft/sec crane speed – MSB in MTC b) Impact at 0.75 ft/sec – MSB lowered into cask	X	X	
6.	Off-Normal Pressurization	X		
Postulated Accident Events (not expected but could occur during cask lifetime)				
1.	Maximum Anticipated Heat Load 125°F Ambient Temperature and Full Solar Load	X	X	X
2.	MSB Drop Accident	X	X	
3.	Tornado			X
4.	Flood	X		X
5.	Earthquake	X	X	X
6.	Accident Pressurization	X		
7.	Complete Blockage of Air Inlets	X	X	X

Table 11.1-1 - MSB Stresses Resulting From Off-Normal Handling Event

Component		Dead Load (ksi)	Off-Normal Pressure (ksi)	Off-Normal Handling (ksi)	Sum of Stresses (ksi)	ASME Code Limit (ksi)
MSB Shell	P_m	0.50	1.17	3.89	5.68	24.60
	$P_L + P_b$	1.50	4.35	9.09	15.38	36.90
Bottom Plate	P_m	0.35	0.40	5.93	6.72	27.00
	$P_L + P_b$	12.20	11.61	14.64	39.62	40.50
Structural Lid	P_m	0.01	0.05	4.02	4.08	27.00
	$P_L + P_b$	0.04	1.19	8.62	9.97	40.50
Bottom Weld	P_m	0.50	1.17	5.22	7.01	27.00
	$P_L + P_b$	12.20	11.61	14.64	39.62	40.50
Structural Lid Weld	P_m	0.04	0.25	1.89	2.20	20.25
	$P_L + P_b$	0.08	2.35	8.62	11.29	30.38
Shield Lid	P_m	0.03	0.00	2.58	2.61	27.00
	$P_L + P_b$	0.31	0.00	3.87	4.18	40.50
Shield Lid Weld	P_m	0.27	1.26	2.16	3.82	20.25
	$P_L + P_b$	0.27	4.84	4.13	9.73	30.38
Support Ring Weld	P_m	0.18	0.00	1.44	1.62	20.25
	$P_L + P_b$	0.18	0.00	1.44	1.62	30.38
Sleeve Assembly	P_m	0.06	0.00	11.03	11.09	24.60
	$P_L + P_b$	0.06	0.00	11.45	11.51	36.90

Table 11.1-2 – MSB Stresses Resulting from Off-Normal Pressure Event*

		Dead Load	Off-Normal Pressure Stress	Normal Handling	Thermal Stress	Sum of Stresses	ASME Code Limit
MSB Shell	Pm	0.50	1.17	1.19	0.00	2.86	22.55
	PL + Pb	1.50	4.35	3.11	0.00	8.96	33.83
	PL + Pb + Q	1.50	4.35	3.11	1.37	10.33	61.50
Bottom Plate	Pm	0.35	0.40	1.40	0.00	2.15	25.08
	PL + Pb	12.20	11.61	13.61	0.00	37.42	37.62
	PL + Pb + Q	12.20	11.61	13.61	19.40	56.82	68.40
Structural Lid	Pm	0.01	0.05	0.72	0.00	0.78	24.75
	PL + Pb	0.04	1.19	1.57	0.00	2.80	37.13
	PL + Pb + Q	0.04	1.19	1.57	0.18	2.98	67.50
Bottom Weld	Pm	0.50	1.17	1.43	0.00	3.10	25.08
	PL + Pb	12.20	11.61	13.61	0.00	37.42	37.62
	PL + Pb + Q	12.20	11.61	13.61	19.40	56.82	68.40
Structural Lid Weld	Pm	0.04	0.25	0.38	0.00	0.67	18.56
	PL + Pb	0.08	2.35	1.61	0.00	4.04	27.85
	PL + Pb + Q	0.08	2.35	1.61	0.50	4.54	50.63
Shield Lid	Pm	0.03	0.00	0.49	0.00	0.52	24.75
	PL + Pb	0.31	0.00	1.00	0.00	1.31	37.13
	PL + Pb + Q	0.31	0.00	1.00	0.00	1.31	67.50
Shield Lid Weld	Pm	0.27	1.26	0.60	0.00	2.13	18.56
	PL + Pb	0.27	4.84	1.00	0.00	6.11	27.85
	PL + Pb + Q	0.27	4.84	1.00	1.30	7.41	50.63
Support Ring Weld	Pm	0.18	0.00	0.18	0.00	0.36	18.56
	PL + Pb	0.18	0.00	0.18	0.00	0.36	27.85
	PL + Pb + Q	0.18	0.00	0.18	0.00	0.36	50.63
Sleeve Assembly	Pm	0.06	0.00	2.02	0.00	2.08	22.55
	PL + Pb	0.06	0.00	2.09	0.00	2.15	33.83
	PL + Pb + Q	0.06	0.00	2.09	52.00	54.15	61.50

* All stresses are reported in ksi units.

Table 11.2-1 - Summary of MSB Stresses Resulting from the Horizontal Drop

		Pressure (ksi)	Horizontal Drop (ksi)	Sum of Stresses (ksi)	ASME Code Limit (ksi)
MSB Shell	P_m	1.2	21.4	22.6	49.0
	$P_L + P_b$	4.4	50.0	54.4	73.5
Bottom Plate	P_m	0.4	32.6	33.0	49.0
	$P_L + P_b$	11.6	43.6	55.2	73.5
Structural Lid	P_m	0.1	22.1	22.2	49.0
	$P_L + P_b$	1.2	47.4	48.6	73.5
Bottom Weld	P_m	1.2	28.7	29.9	49.0
	$P_L + P_b$	11.6	43.6	55.2	73.5
Structural Lid Weld	P_m	0.3	10.4	10.7	36.8
	$P_L + P_b$	2.4	47.4	49.8	55.1
Shield Lid	P_m	0.00	14.2	14.2	49.0
	$P_L + P_b$	0.00	21.3	21.3	73.5
Shield Lid Weld	P_m	1.3	10.2	11.5	36.8
	$P_L + P_b$	4.8	22.7	27.5	55.1

Table 11.2-2 - Summary of MSB Stresses Resulting from the Vertical Drop

		Pressure (ksi)	Vertical Drop (ksi)	Sum of Stresses (ksi)	ASME Code Limit (ksi)
MSB Shell	P_m	1.17	46.50	47.67	49.00
	$P_L + P_b$	4.35	47.00	51.35	63.00
Bottom Plate	P_m	0.40	23.10	23.50	49.00
	$P_L + P_b$	11.61	48.30	59.91	63.00
Structural Lid	P_m	0.05	1.05	1.10	49.00
	$P_L + P_b$	1.19	4.90	6.09	73.50
Bottom Weld	P_m	1.17	46.50	47.67	49.00
	$P_L + P_b$	11.61	48.30	59.91	63.00
Structural Lid	P_m	0.25	4.20	4.45	36.75
Weld	$P_L + P_b$	2.35	9.10	11.45	55.13
Shield Lid	P_m	0.00	3.20	3.20	49.00
	$P_L + P_b$	0.00	6.80	6.80	73.50
Shield Lid Weld	P_m	1.26	32.80	34.06	36.75
	$P_L + P_b$	4.84	32.80	37.64	55.13
Support Ring	P_m	0.00	21.89	21.89	36.75
Weld	$P_L + P_b$	0.00	21.89	21.89	55.13
Sleeve Assembly	P_m	0.00	7.50	7.50	49.00
	$P_L + P_b$	0.00	7.50	7.50	73.50

Table 11.2-3 - Summary of MSB Stresses Resulting from Hypothetical Accident Pressurization

		Dead Load (ksi)	Critical Pressure (ksi)	Dead Load + Critical Pressure (single run) (ksi)	Normal Handling (ksi)	Sum of Stresses (ksi)	ASME Code Limit (ksi)
MSB Shell	P_m	Note 1	Note 1	2.07	1.19	3.26	49.00
	$P_L + P_b$	Note 1	Note 1	6.69	3.11	9.80	73.50
Bottom Plate	P_m	Note 1	Note 1	0.87	1.40	2.27	49.00
	$P_L + P_b$	Note 1	Note 1	39.09	13.61	52.70	73.50
Structural Lid	P_m	0.01	0.29	---	0.72	1.02	49.00
	$P_L + P_b$	0.04	7.16	---	1.57	8.77	73.50
Bottom Weld	P_m	Note 1	Note 1	2.07	1.43	3.50	49.00
	$P_L + P_b$	Note 1	Note 1	39.09	13.61	52.70	73.50
Structural Lid Weld	P_m	0.04	1.47	---	0.38	1.89	36.75
	$P_L + P_b$	0.08	14.11	---	1.61	15.80	55.13
Shield Lid	P_m	0.03	0.00	---	0.49	0.52	49.00
	$P_L + P_b$	0.31	0.00	---	1.00	1.31	73.50
Shield Lid Weld	P_m	0.27	7.58	---	0.60	8.45	36.75
	$P_L + P_b$	0.27	29.05	---	1.00	30.32	55.13
Support Ring Weld	P_m	0.18	0.00	---	0.18	0.36	36.75
	$P_L + P_b$	0.18	0.00	---	0.18	0.36	55.13
Sleeve Assembly	P_m	0.06	0.00	---	2.02	2.08	49.00
	$P_L + P_b$	0.06	0.00	---	2.09	2.15	73.50

Note 1: The results from the individual analyses of dead load and critical pressure stress are not combined for loads involving the MSB Shell, Bottom Plate, and Bottom Weld. For these components a separate finite element analysis, which includes dead load + critical pressure stress, was completed with the results as shown (Section 3.4.4.1.2 provides a description of the model). The normal handling and thermal stresses are added separately to give the sum of stresses. For the other components, the individual stress intensity results are combined to give the sum of the stresses.

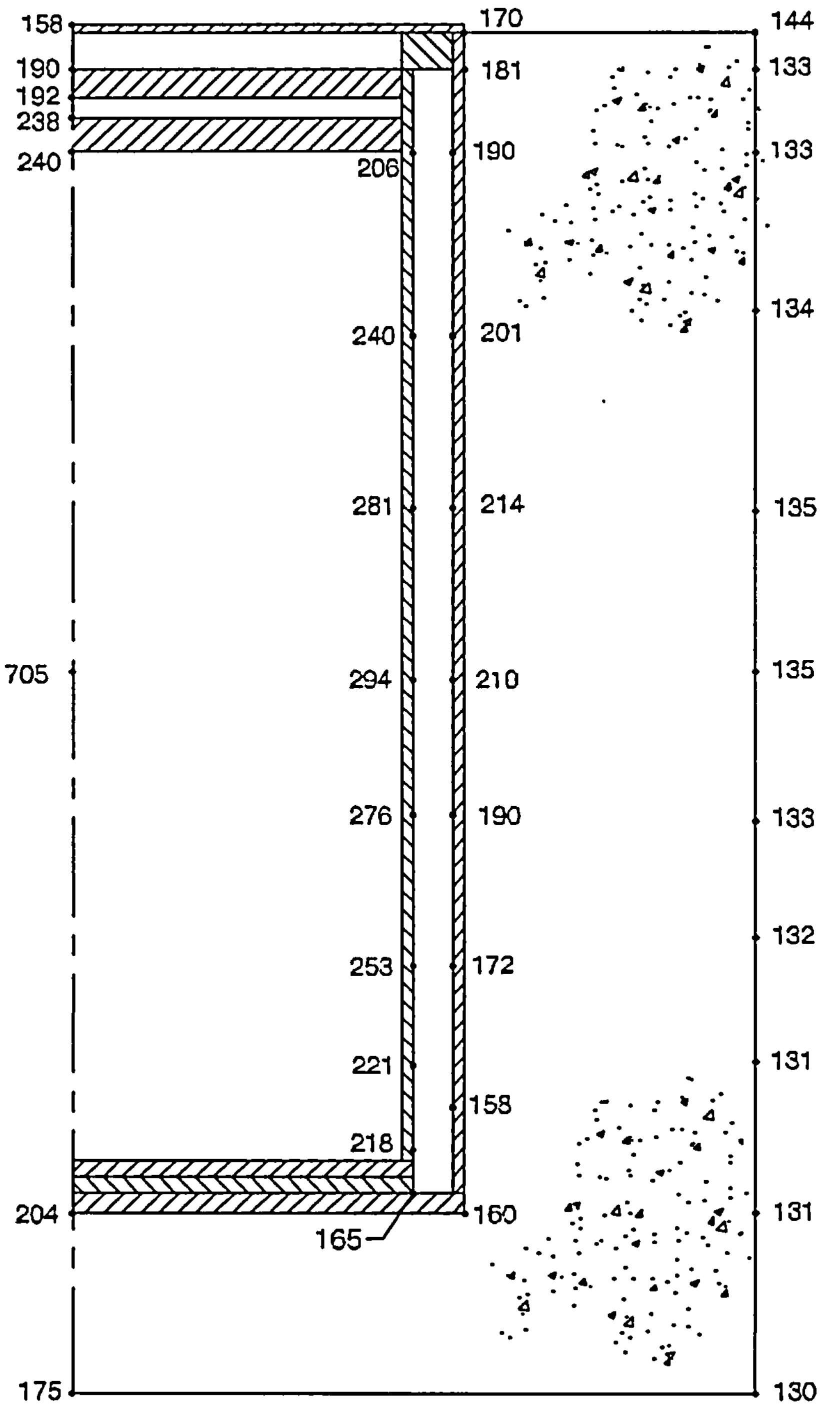


Figure 11.1-1 - VSC Temperature Distribution for 100°F Ambient Conditions

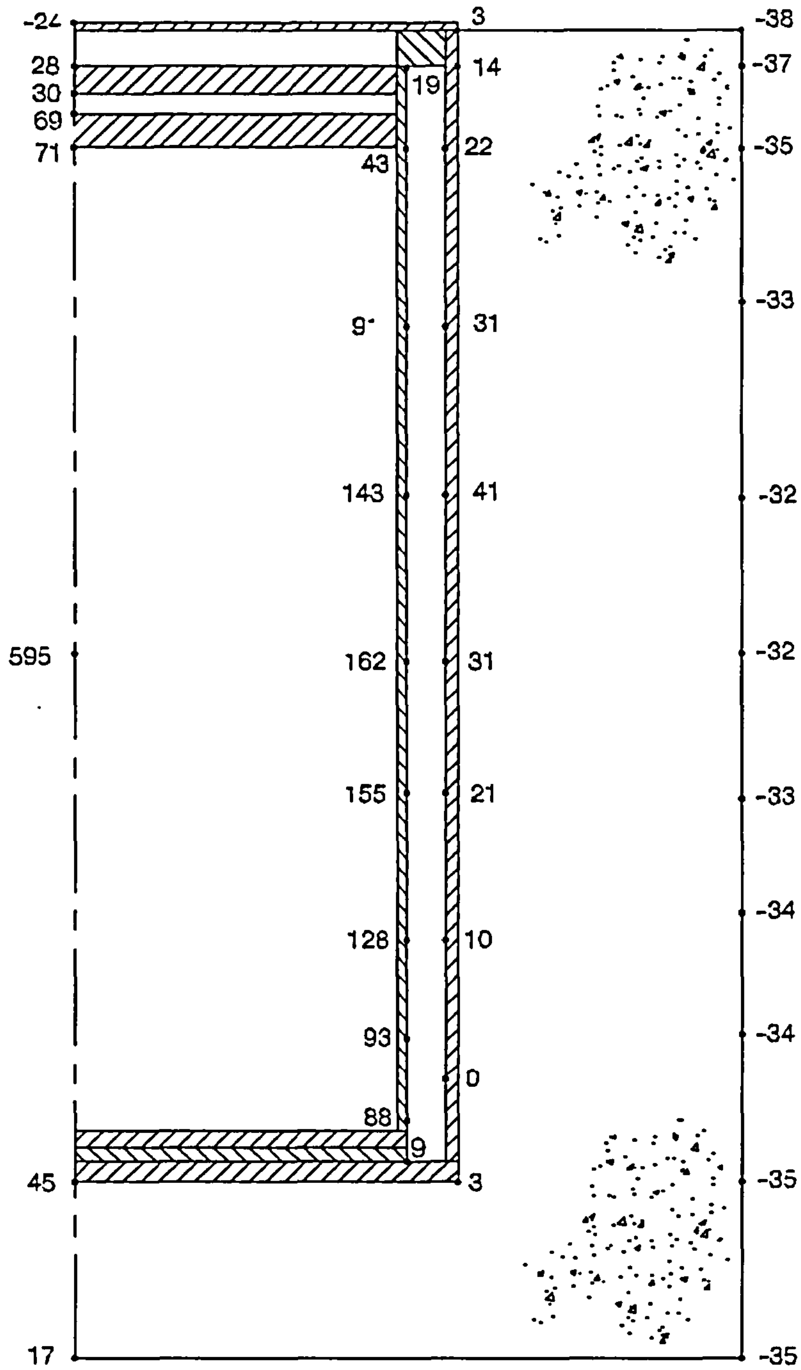


Figure 11.1-2 - VSC Temperature Distribution for -40°F Ambient Conditions

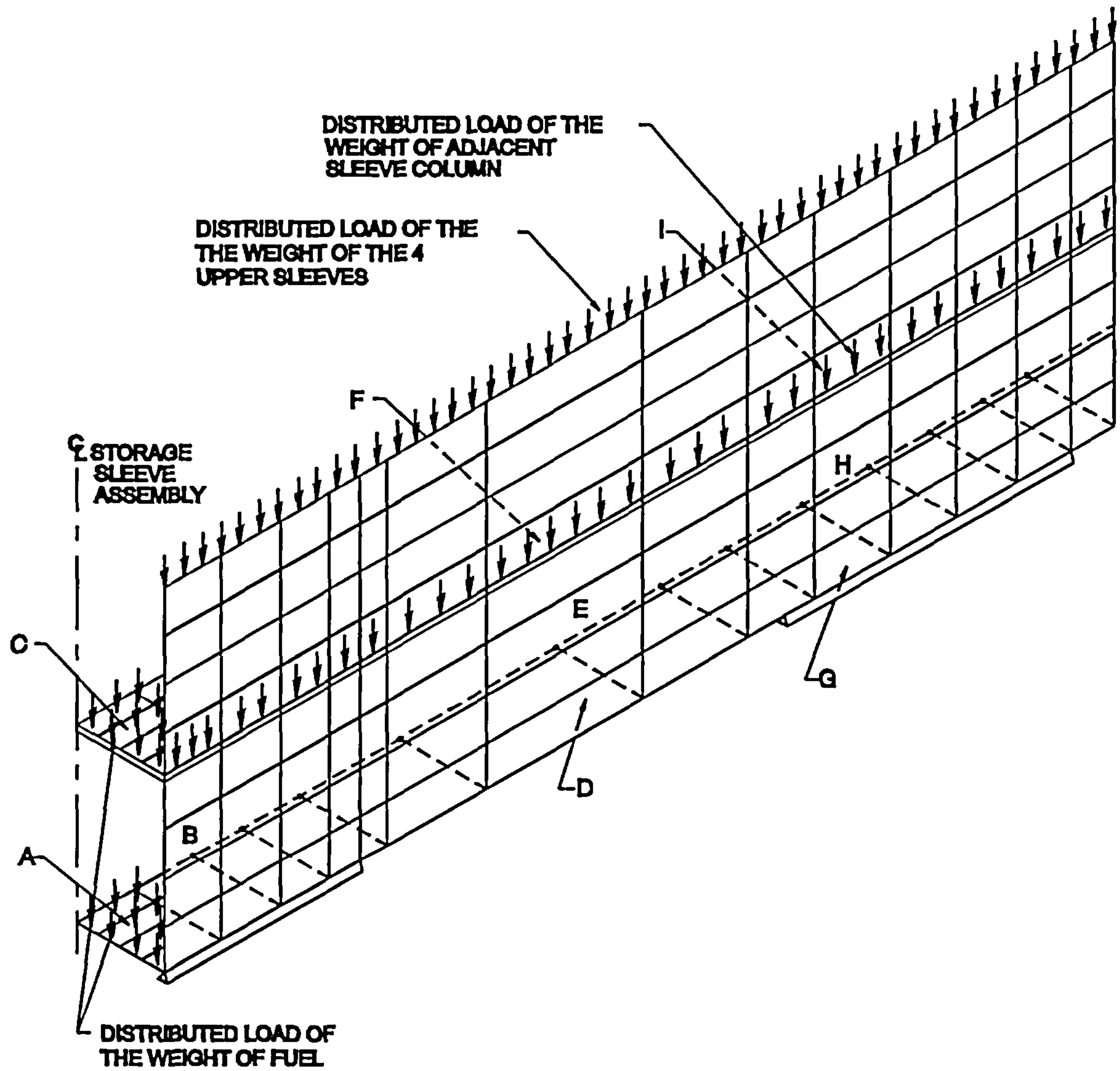


Figure 11.2-1 - MSB Storage Sleeve Model

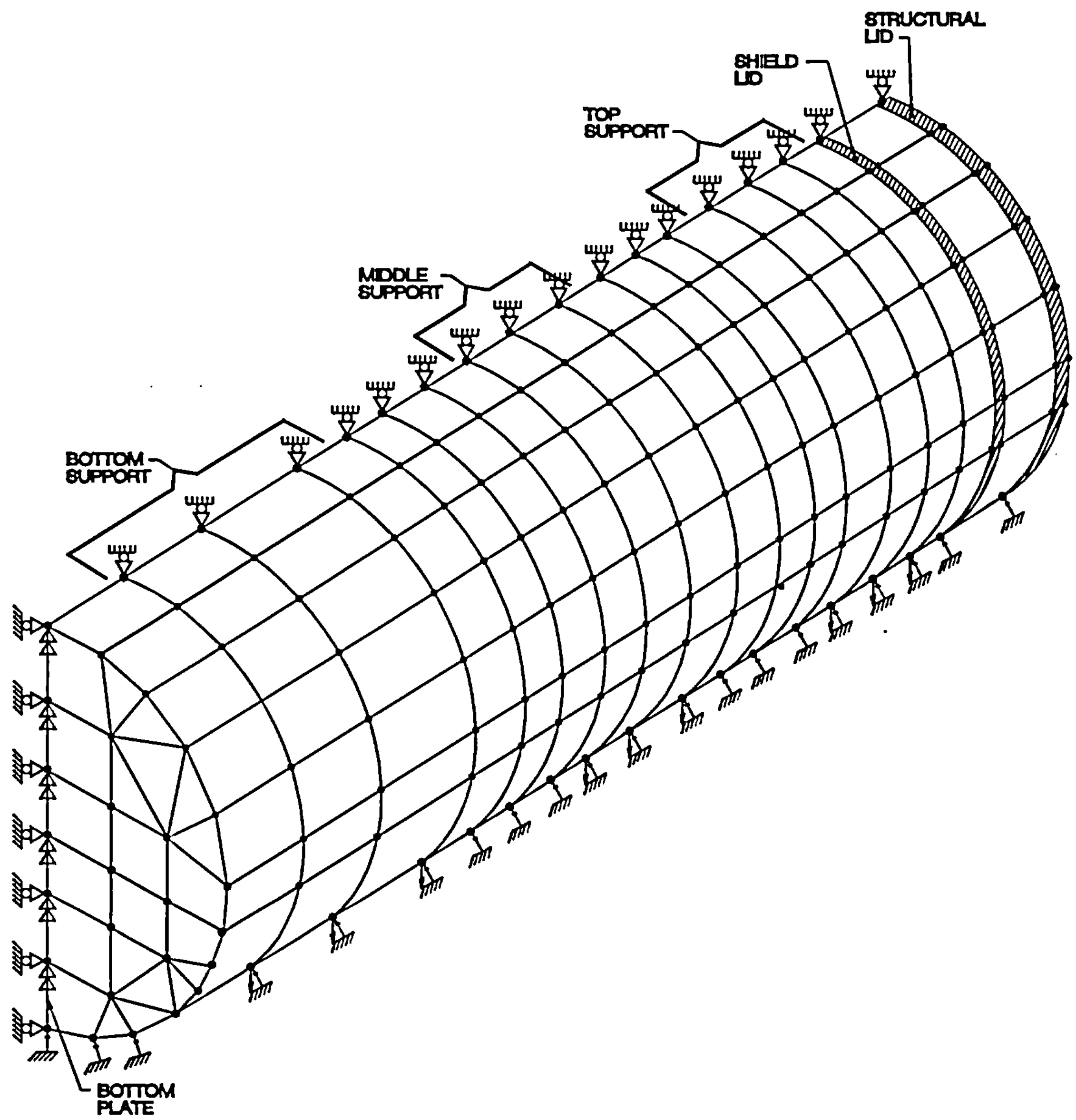


Figure 11.2-2 - MSB Body Finite Element Model for Horizontal Drop Analysis

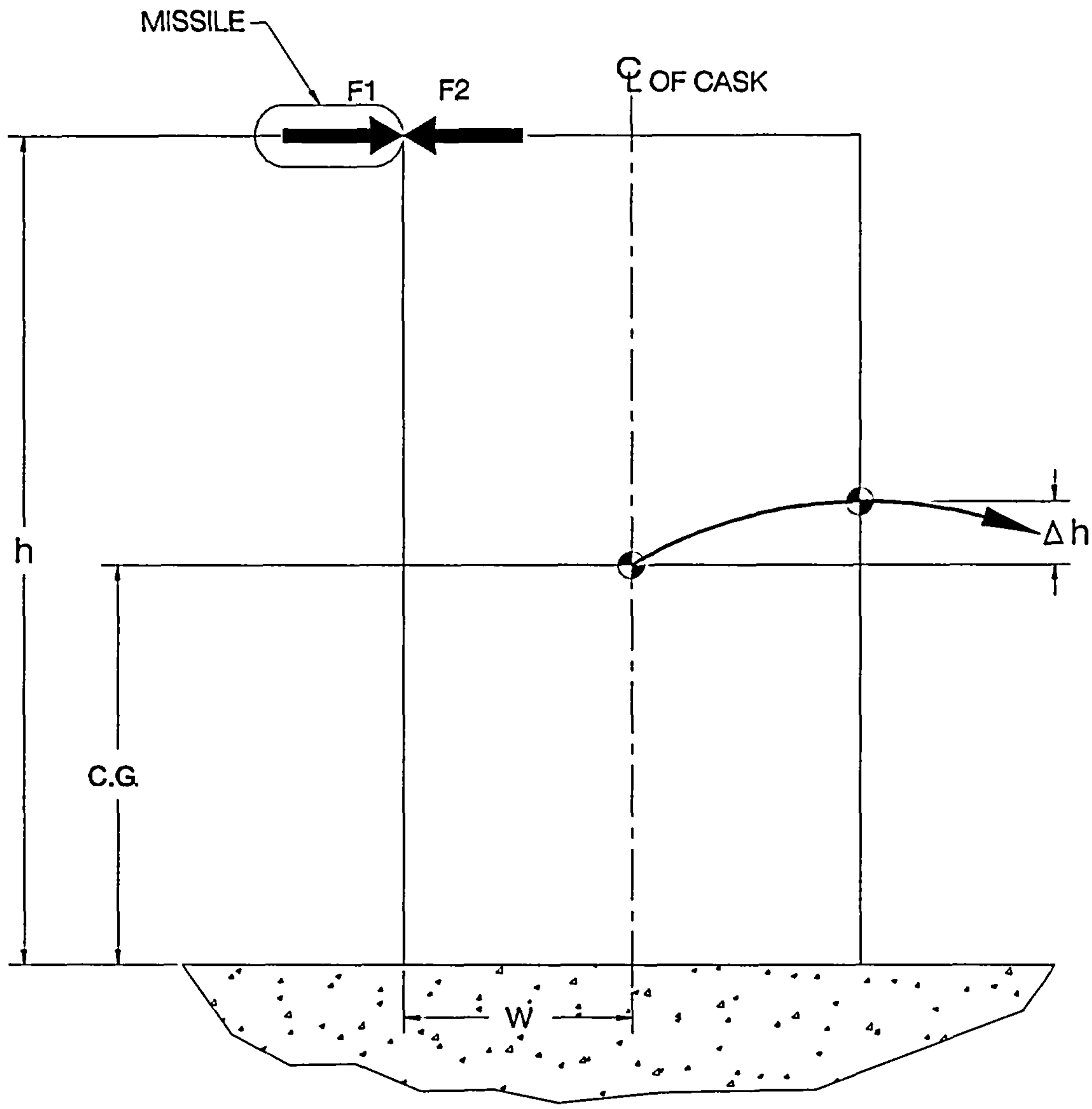


Figure 11.2-3 - Sketch of Missile Cask Impact Geometry

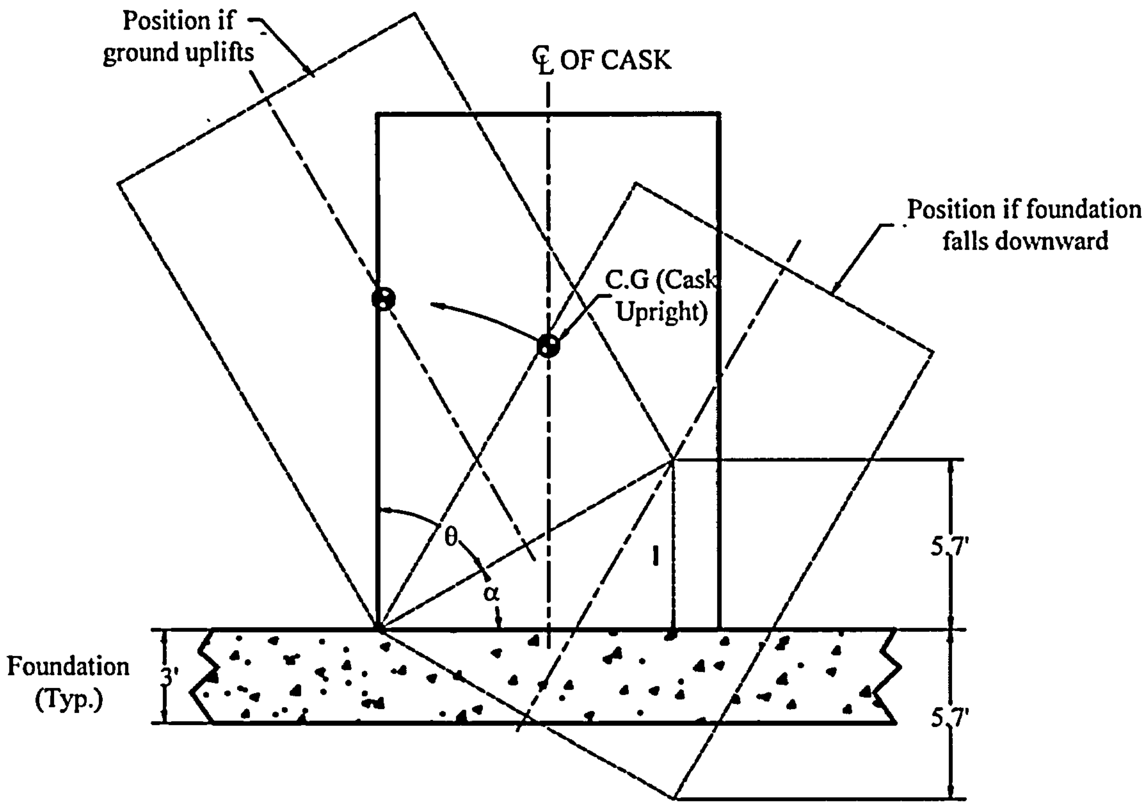


Figure 11.2-4 - Cask Tip-Over Requirements

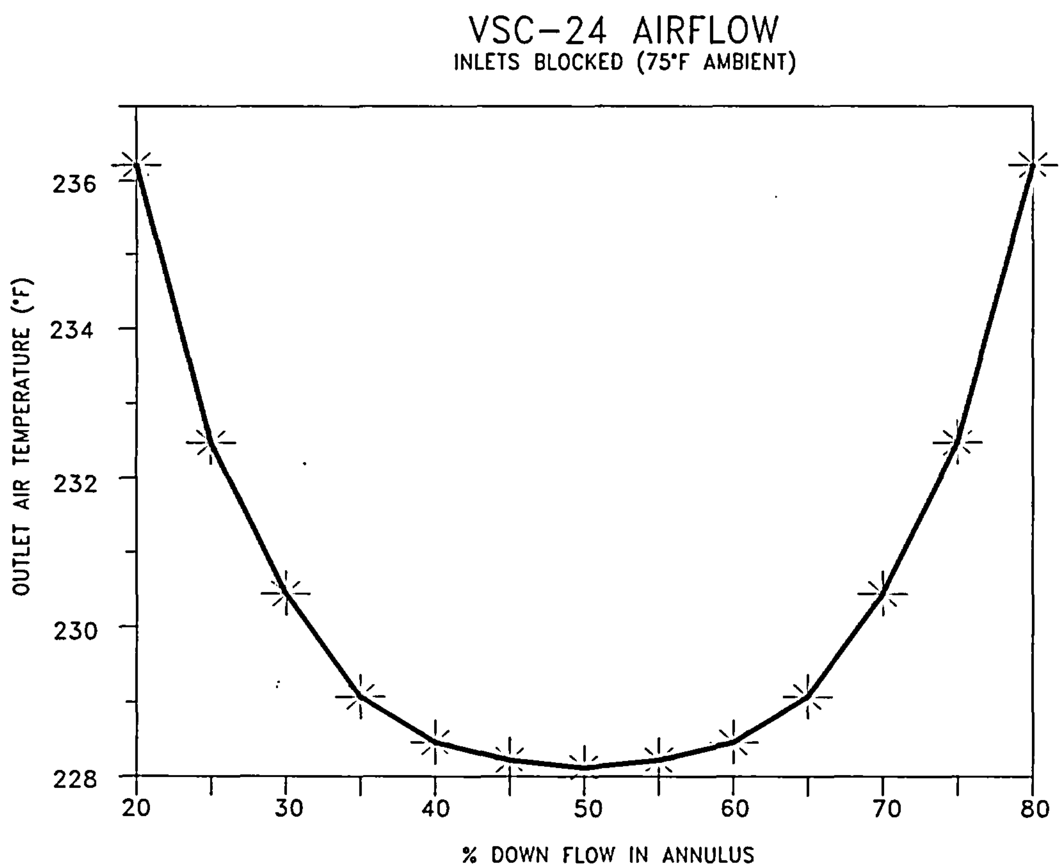


Figure 11.2-5 - Outlet Air Temperature

12.0 OPERATING CONTROLS AND LIMITS

12.1 PROPOSED OPERATING CONTROLS AND LIMITS

The VSC system is totally passive during the storage mode and requires very few operating controls. The controls that are necessary for the system are for fuel selection and certain operations during the MSB loading and transfer. The conditions and other characteristics were selected based on the safety assessments for both normal and accident conditions.

This FSAR provides specifications in the areas of fuel characteristics; MSB drying, backfilling, and sealing; and MSB transfer and storage. In addition, site-specific procedures in the areas of fuel and cask handling, transport, security surveillance, administrative controls, training and others may be utilized as necessary by the implementing site.

This section presents the conditions that a potential user (licensee) of the Ventilated Storage Cask (VSC-24) system must comply with in order to use the system under a general license issued according to the provisions of 10CFR72.210 and 72.212. The specification is based on consideration of the design basis parameters included in the FSAR.

12.2 DESIGN FEATURES

These operating controls and limits cover design characteristics of special importance to each of the physical barriers and to maintenance of safety margins in the cask design. The principal objective of this category is to control changes in the design of essential equipment.

The essential design features of the cask are as follows:

- Concrete Density
- Concrete Strength
- MSB Shielding Lid, Material Densities, and Thickness
- Reinforcing Steel Quantities and Placement
- MSB Boundary Welds
- MSB Basket and Shell Material Thickness and Strength

Each of these design features contributes significantly to the ability of the VSC to meet the requirements of 10CFR72 for at least fifty years. Each parameter is closely controlled by the fabrication specifications for the MSB and VCC. The design of these features may not be altered without affecting the safety and durability of the cask. Verification inspections must be performed during VSC fabrication.

12.3 ADMINISTRATIVE CONTROLS

Controls used by BFS as part of the VSC design and fabrication are provided in the BFS Quality Assurance Manual and Procedures. These are discussed in Chapter 13. Site-specific controls for the organization, administrative system, procedures, record keeping, review, audit and reporting are the responsibility of the user of the VSC cask and are part of the user's 10CFR50 license.

12.4 CONDITIONS FOR CASK USE AND TECHNICAL SPECIFICATIONS

The Conditions for Cask Use, Technical Specifications, and Surveillance Requirements are provided in this section.

CONDITIONS FOR CASK USE AND

TECHNICAL SPECIFICATIONS FOR THE

VSC-24 STORAGE CASK SYSTEM

CONTENTS

TABLES

FIGURES

1.0 INTRODUCTION

This section presents the conditions with which a potential user (licensee) of the Ventilated Storage Cask (VSC-24) system must comply to use the system under a general license issued according to the provisions of 10 CFR 72.210 and 72.212. These conditions have either been proposed by the system vendor, imposed by the U.S. Nuclear Regulatory Commission (NRC) staff as a result of the review of the Final Safety Analysis Report (FSAR), or are part of the regulatory requirements expressed in 10 CFR 72.212.

1.1 General Requirements and Conditions

1.1.1 Regulatory Requirements

Regulatory requirements define a number of technical and administrative conditions for system use. Technical regulatory requirements for the licensee (user of the VSC-24 system) are contained in 10 CFR 72.212(b).

Section 72.212(b) requires that the licensee perform written evaluations, before use, establishing that: (1) conditions set forth in the Certificate of Compliance (CoC) have been met; (2) cask storage paths and areas have been designed to adequately support the static load of the stored casks; and (3) the requirements of 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," have been met. It also requires that the licensee review the FSAR and the associated SERs, before use of the general license, to determine whether or not the reactor site parameters (including earthquake intensity and tornado missiles) are encompassed in the cask design bases considered in these reports.

Site-specific parameters and analyses that need verification by the system user are as follows:

1. The temperature of 75°F as the maximum average yearly temperature, without solar incidence;

2. The steady state temperature extremes of 100°F (average daily temperature) with incident solar radiation and minus 40°F with no solar incidence;
3. The "accident" short-term temperature extreme of 125°F with incident solar radiation;
4. The horizontal and vertical seismic acceleration levels of 0.25g and 0.17g, respectively;
5. The analyzed flood condition of 17.7 fps water velocity and submerged depth of 120 feet; and
6. The potential for fire and explosion should be addressed, based on site-specific considerations.

According to 10 CFR 72.212(b), a record of the written evaluations must be retained by the licensee until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

1.1.2 Operating Procedures

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The operating procedures suggested generically in the FSAR are considered appropriate and should provide the basis for the user's written operating procedures. The following additional written procedures shall also be developed as part of the user operating procedures:

1. A procedure shall be developed for cask unloading, assuming damaged fuel. If fuel needs to be removed from the multi-assembly sealed basket (MSB), either at the end of service life or for inspection after an accident, precautions must be taken against the potential for the presence of oxidized fuel and to prevent radiological exposure to personnel during this operation. This activity can be achieved by the use of the valves to permit a determination of the atmosphere within the MSB before the removal of the structural and shield lids. If the atmosphere within the MSB is helium, then operations should proceed normally, with fuel removal, either via the transfer cask or in the pool.

However, if air is present within the MSB, then appropriate filters should be in place to permit the flushing of any potential airborne radioactive particulate from the MSB, via the valves. This action will protect both personnel and the operations area from potential contamination. For the accident case, personnel protection in the form of respirators or supplied air should be considered in accordance with the licensee's Radiation Protection Program.

2. A procedure shall be developed for the documentation of the characterizations performed to select spent fuel to be stored in the MSB. This procedure shall include a requirement for independent verification of each fuel assembly selection.

3. A procedure shall be developed for two independent determinations (two samples analyzed by different individuals) of the boron concentration in the water of the spent fuel pool and that used to fill the MSB cavity.

4. In preparing written operating procedures for handling the MSB over the VCC, the user shall include a consideration for reducing the likelihood of fracturing the ceramic tiles at the bottom of the VCC as the MSB is lowered into position.

1.1.3 Quality Assurance

Activities at the independent spent fuel storage installation (ISFSI) shall be conducted in accordance with the requirements of 10 CFR Part 50, Appendix B.

1.1.4 Heavy Loads Requirements

Lifts of the MSB in the MSB transfer cask (MTC) must be made within the existing heavy loads requirements and procedures of the licensed nuclear power plant. The MTC design has been reviewed under 10 CFR Part 72 and found to meet NUREG-0612 and ANSI 14.6. However, an additional safety review (under 10 CFR 50.59) is required to show operational compliance with NUREG-0612 and/or existing plant-specific heavy loads requirements. Other spent fuel transfer systems, for loading the MSB and VCC within reactor fuel buildings, may be suitable for use in accordance with 10 CFR Part 50 operating licenses.

1.1.5 Training Module

A training module shall be developed for the existing licensee's training program, establishing an ISFSI training and certification program. This module shall include the following:

1. VSC-24 Design (overview);
2. ISFSI Facility Design (overview);
3. Certificate of Compliance Conditions (overview);
4. Fuel Loading, MTC Handling, MSB Lowering Procedures; and
5. Off-Normal Event Procedures.

1.1.6 Training Exercise

A dry run of the MSB loading, MTC handling, and MSB lowering shall be held. This dry run shall include, but not be limited to, the following:

1. Moving an MSB and MTC into and out of the pool;
2. Loading a fuel assembly;
3. MSB sealing and cover gas backfilling operations (using a mock-up MSB);
4. Lowering the MSB into the concrete cask;
5. Returning the MSB to the fuel pool; and
6. Opening an MSB (using a mock-up MSB).

1.1.7 Requirement for First Cask in Place

The following measurements are required for the first VSC placed in service:

The first MSB shall be loaded with 24 spent fuel assemblies, constituting a heat source of up to 24 kW, and then the MSB shall be loaded into the VCC to measure the cask thermal performance by measuring the air inlet and outlet temperatures for normal air flow, according to Technical Specification 1.2.3. The purpose of the test is to measure the heat removal performance of the VSC system and establish base-line data (FSAR Section 9.1.3). A letter report summarizing the results of the test and evaluation shall be submitted to NRC within 30 days of placing the cask in service in accordance with 10 CFR 72.4.

Should the first user of the system not have spent fuel capable of producing a 24 kW heat load, the user may use a lesser load for the test, provided that a calculation of the temperature difference between the inlet and outlet temperatures is performed, using the same methodology and inputs documented in the SER and FSAR, with the lesser load as the only exception. The calculation and the measured temperature data shall be reported in accordance with 10 CFR 72.4. The calculation and comparison need not be reported for casks that are subsequently loaded with lesser heat sources than the test case. However, for the first or any other user, the process needs to be reported for any higher heat sources, up to 24 kW, which is the maximum allowed under this Certificate of Compliance. The use of artificial thermal loads other than spent fuel to satisfy the above requirement is acceptable.

1.1.8 Surveillance Requirements Applicability

The specified frequency for each Surveillance Requirement is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.

For frequencies specified as "once," the above interval extension does not apply.

If a required action requires performance of a surveillance or its completion time requires periodic performance of "once per...," the above frequency extension applies to the repetitive portion, but not to the initial portion of the completion time.

Exceptions to these requirements are stated in the individual specifications.

1.2 Technical Specifications, Functional and Operational Limits

1.2.1 Fuel Specification

Limit/Specification:

The characteristics of the spent fuel allowed to be stored in the VSC-24 system are restricted to those specified in Table 1 and Table 2.

Applicability: The specification is applicable to all fuel to be stored in the VSC-24 system.

Objective: The specification is prepared to ensure that the peak fuel rod temperatures, maximum surface doses, and nuclear criticality effective neutron multiplication factor are below the design values. Furthermore, the fuel weight and type ensures that structural conditions in the FSAR bound those of the actual fuel being stored.

Action: For each spent fuel assembly to be loaded into an MSB, compliance with the parameter limits listed in Table 1 shall be independently verified and documented. Compliance with all of the Table 2 assembly parameter requirements for any one of the eight defined fuel assembly classes shall also be verified for each loaded assembly. Fuel not meeting this specification shall not be stored in the VSC-24 system.

Surveillance: Immediately before insertion of a spent fuel assembly into an MSB, the identity of each fuel assembly shall be independently verified and documented.

Table 1 - Characteristics of Spent Fuel to be Stored in the VSC-24 System

Fuel	Only intact, unconsolidated PWR fuel assemblies meeting the requirements listed below.[1]
Class/Type	• B&W, Mark B, 15 x 15, with and without BPRAs or TPAs • CE/Exxon 15 x 15 with and without poison clusters or plugging clusters • CE 16 x 16 • Westinghouse PWR 17 x 17 with and without BPRAs or TPAs • Westinghouse PWR 15 x 15 • Westinghouse PWR 14 x 14 with and without BPRAs or TPAs
Fuel Cladding	Zircaloy clad fuel with no known or suspected gross cladding failures.[2]
Decay Power Per Assembly	Less than or equal to 1 kW
Maximum Burnup	Less than or equal to 45,000 MWd/MTU
Post Irradiation Time	Greater than or equal to 5 years. Varies with assembly burnup and initial enrichment, as shown in FSAR Table 5.5-1.
Maximum Initial Enrichment	Less than or equal to 4.2 weight percent ^{235}U
Assembly Weight	Less than or equal to 1585 lb (720.5 kg.)
Number of Assemblies per VSC	24

Notes:

(1) High cobalt assemblies (i.e., assemblies with solid stainless steel or stainless steel clad rods in fuel rod locations, which contain 46.7 to 250 grams of initial cobalt within the fuel zone) must not be loaded into the 12 fuel sleeves located around the perimeter of the MSB.

(2) Failed BPRAs or TPAs may be loaded provided that they do not contain Ag-In-Cd or Hf poison material. BPRAs containing these poison materials must have intact fuel cladding, with no known or suspected defects.

Table 2 - Fuel Assembly Class Characterization Parameters (2 pages)

Parameter	Fuel Assembly Class			
	B&W 15x15	W 14x14	W 15x15	W 17x17
Assembly Lattice Layout	see Figure 1	see Figure 2	see Figure 3	see Figure 4
Assembly Pin Pitch	1.44272 cm	1.41224 cm	1.43002 cm	1.25984 cm
Fuel Density	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical
Fuel Pellet Diameter	from 0.92202 cm to 0.94996 cm	from 0.86360 cm to 0.94234 cm	from 0.89408 cm to 0.93980 cm	from 0.75946 cm to 0.82804 cm
Fuel Clad Material	zircaloy	Zircaloy	zircaloy	zircaloy
Fuel Clad Outer Diameter	≤ 1.10236 cm	≤ 1.08712 cm	≤ 1.08712 cm	≤ 0.96012 cm
Fuel Clad Thickness	≥ 0.06604 cm	≥ 0.05588 cm	≥ 0.05842 cm	≥ 0.05588 cm
Guide Tube Material	zircaloy	zircaloy	zircaloy	zircaloy
Guide Tube Outer Diameter	≤ 1.36 cm	≤ 1.37414 cm	≤ 1.4 cm	≤ 1.24 cm
Guide Tube Thickness	≤ 0.045 cm	≤ 0.04318 cm	≤ 0.06 cm	≤ 0.06 cm
Instrument Tube Material	zircaloy	zircaloy	zircaloy	zircaloy
Instrument Tube Outer Diameter	≤ 1.26 cm	≤ 1.37414 cm	≤ 1.4 cm	≤ 1.24 cm
Instrument Tube Thickness	≤ 0.07 cm	all	≤ 0.06 cm	≤ 0.06 cm
Active Fuel Length	≤ 370.84 cm	≤ 373.0 cm	≤ 370.0 cm	≤ 371.0 cm
Guide Bar Effective Diameter [see Note 1]	not applicable	not applicable	not applicable	not applicable
Control Component Rodlets Allowed in Guide Tubes	yes	yes	no	yes
Control Component Rodlet Clad Material	zircaloy or stainless steel	zircaloy or stainless steel	not applicable	zircaloy or stainless steel
Control Component Rodlet Outer Diameter	≤ 1.1176 cm	all	not applicable	≤ 0.9779 cm
Control Component Rodlet Fill Material	any non-hydrogen bearing material	any non-hydrogen bearing material	not applicable	any non-hydrogen bearing material

Note:

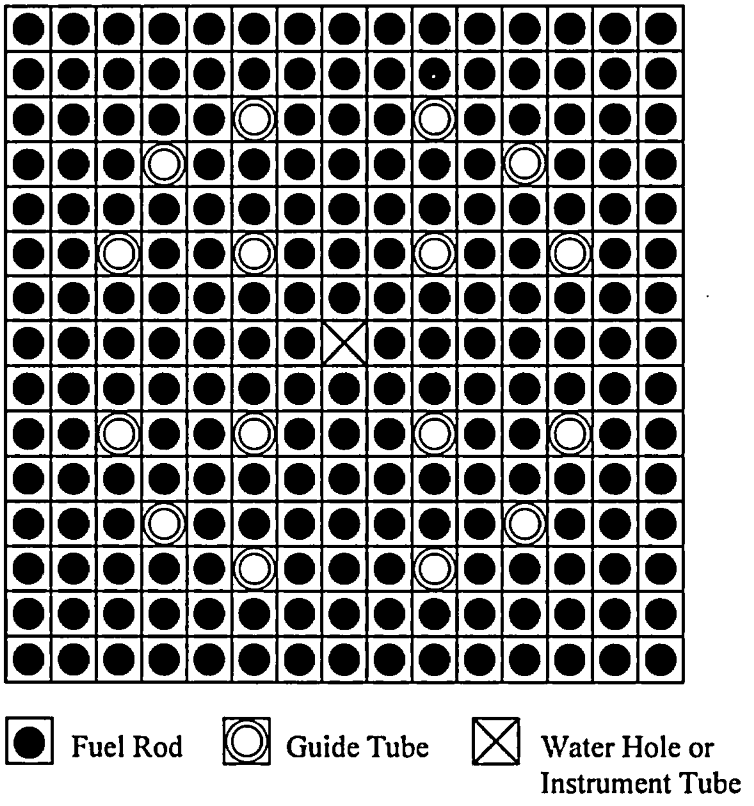
The guide bars may have either a rectangular or circular cross section. The guide bar effective diameter corresponds to the diameter of a circular region having an area equal to the actual guide bar cross-sectional area.

Table 2 - Fuel Assembly Class Characterization Parameters (2 pages)

Parameter	Fuel Assembly Class			
	CE 15x15A	**CE 15x15B**	**CE 15x15C**	**CE 16x16**
Assembly Lattice Layout	see Figure 5	see Figure 6	see Figure 7	see Figure 8
Assembly Pin Pitch	1.397 cm	1.397 cm	1.397 cm	1.28524 cm
Fuel Density	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical
Fuel Pellet Diameter	from 0.888 cm to 0.912 cm	from 0.888 cm to 0.912 cm	from 0.888 cm to 0.912 cm	from 0.81534 cm to 0.83566 cm
Fuel Clad Material	zircaloy	zircaloy	zircaloy	zircaloy
Fuel Clad Outer Diameter	≤ 1.062 cm	≤ 1.062 cm	≤ 1.062 cm	≤ 0.98044 cm
Fuel Clad Thickness	≥ 0.0508 cm	≥ 0.0508 cm	≥ 0.0508 cm	≥ 0.06096 cm
Guide Tube Material	not applicable	zircaloy	zircaloy	zircaloy
Guide Tube Outer Diameter	not applicable	≤ 1.06934 cm	≤ 1.06934 cm	≤ 2.54 cm
Guide Tube Thickness	not applicable	≤ 0.02032 cm	≤ 0.02032 cm	≤ 0.1778 cm
Instrument Tube Material	zircaloy	zircaloy	zircaloy	not applicable
Instrument Tube Outer Diameter	≤ 1.06934 cm	≤ 1.06934 cm	≤ 1.06934 cm	not applicable
Instrument Tube Thickness	≤ 0.08509 cm	≤ 0.08509 cm	≤ 0.08509 cm	not applicable
Active Fuel Length	≤ 336.0 cm	≤ 336.0 cm	≤ 336.0 cm	≤ 385.0 cm
Guide Bar Effective Diameter [see Note 1]	≤ 1.2006 cm	≤ 1.2006 cm	≤ 1.2006 cm	not applicable
Control Component Rodlets Allowed in Guide Tubes	not applicable	yes	yes	no
Control Component Rodlet Clad Material	not applicable	zircaloy or stainless steel	zircaloy or stainless steel	not applicable
Control Component Rodlet Outer Diameter	not applicable	all	all	not applicable
Control Component Rodlet Fill Material	not applicable	any non-hydrogen bearing material	any non-hydrogen bearing material	not applicable

Note:

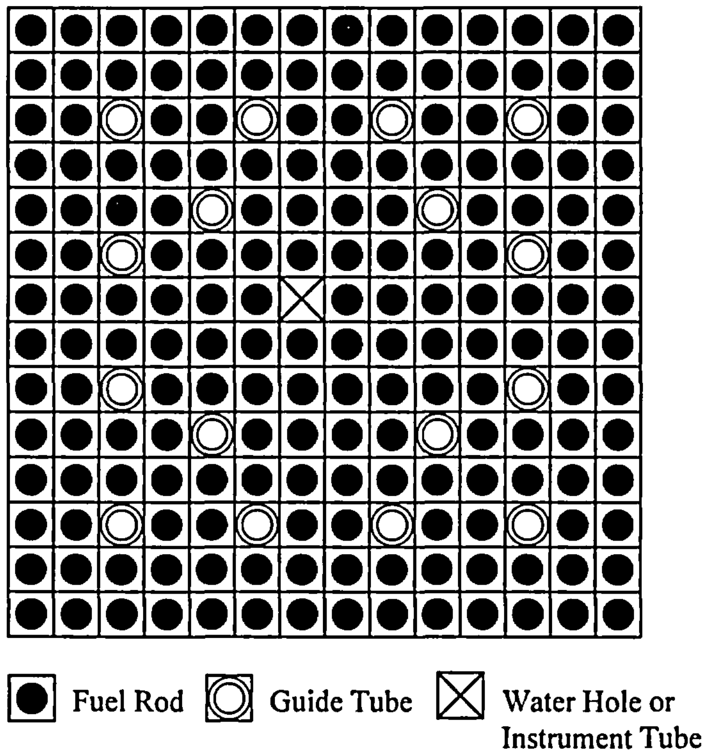
The guide bars may have either a rectangular or circular cross section. The guide bar effective diameter corresponds to the diameter of a circular region having an area equal to the actual guide bar cross-sectional area.



Note 1: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

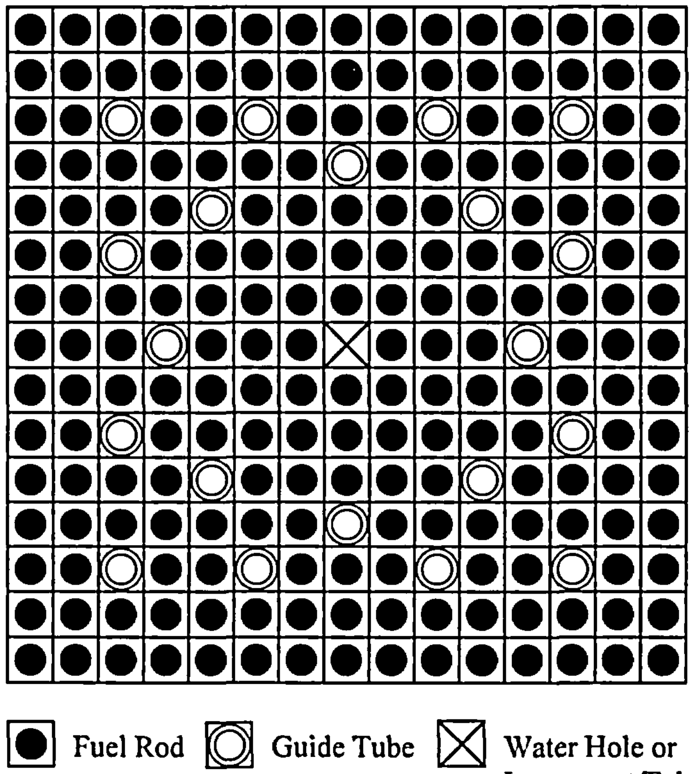
Figure 1 - B&W 15x15 Assembly Class Lattice Layout



Note 1: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

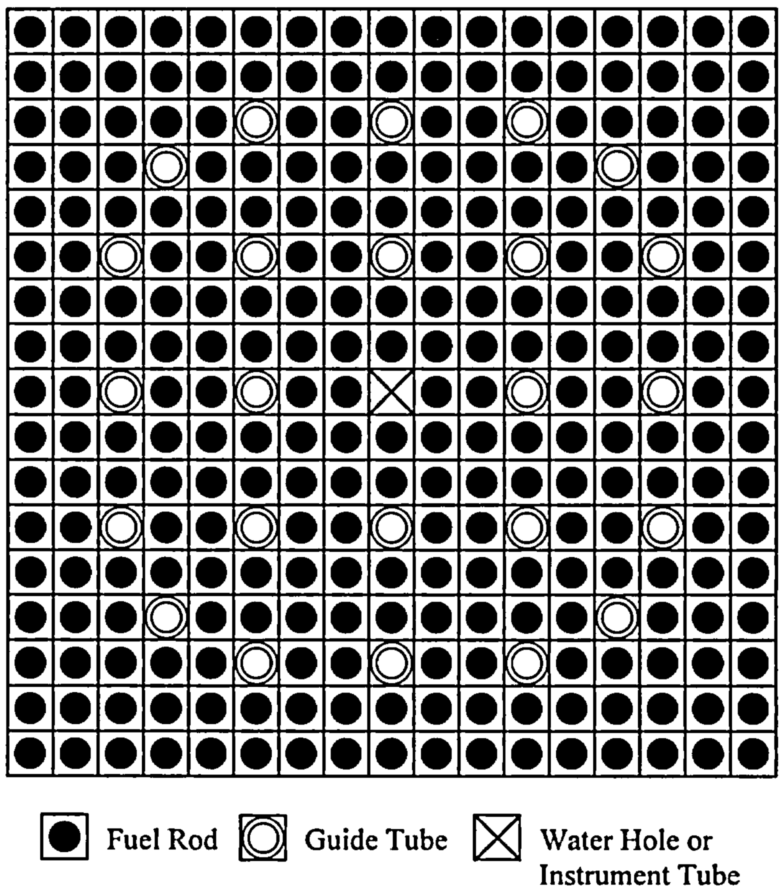
Figure 2 - W 14x14 Assembly Class Lattice Layout



Note 1: Although control component rodlets are (conservatively) modeled in the criticality evaluation, inserted control components are not allowed for the W 15x15 assembly, as shown in Table 2.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

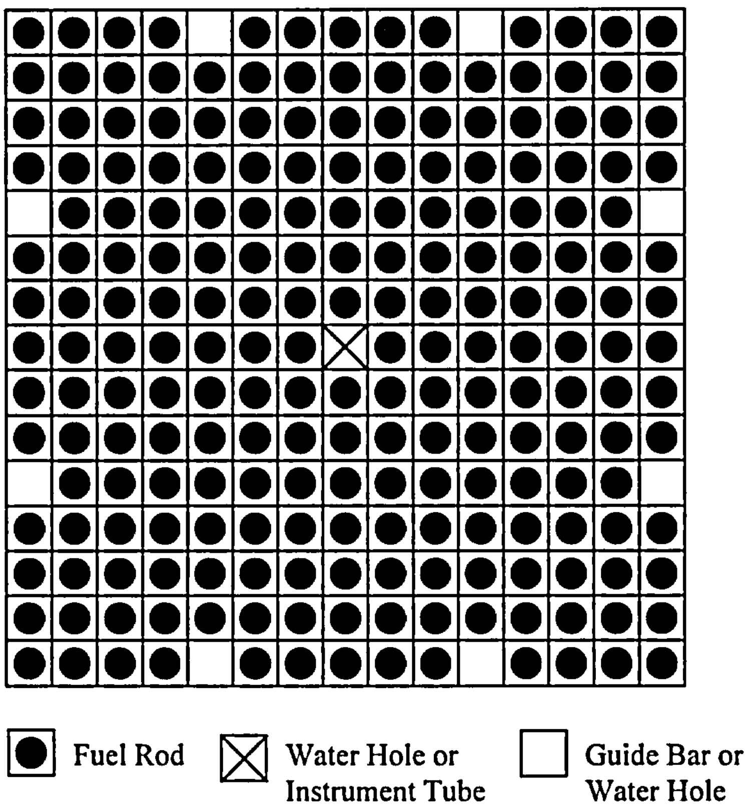
Figure 3 - W 15x15 Assembly Class Lattice Layout



Note 1: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

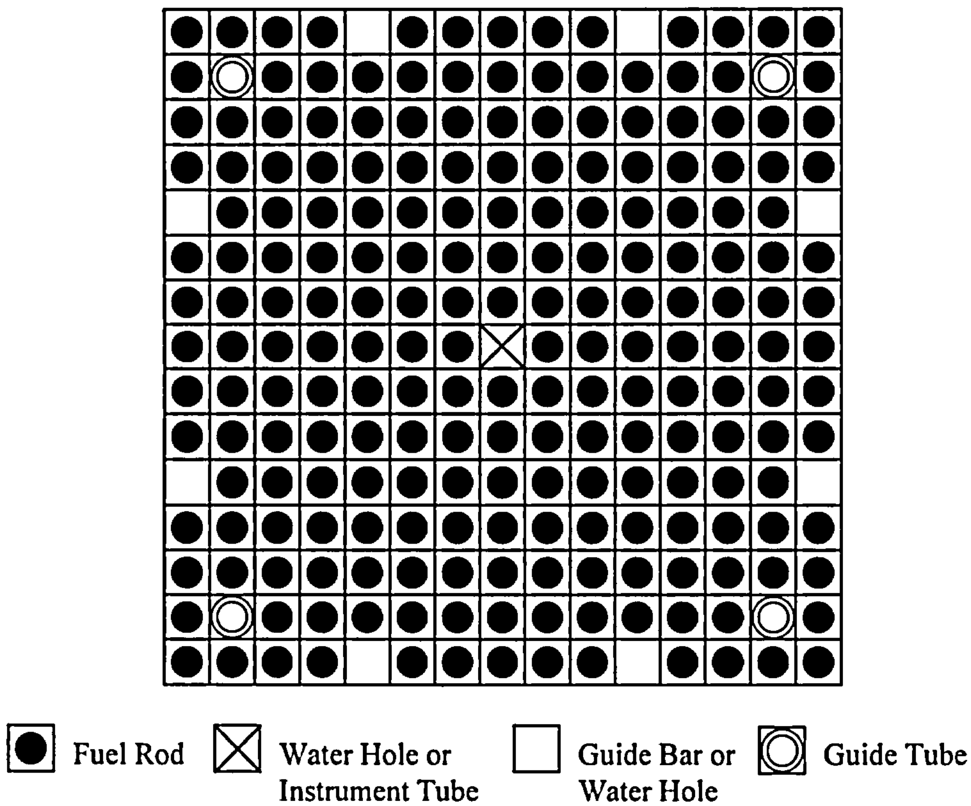
Figure 4 - W 17x17 Assembly Class Lattice Layout



Note 1: The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Figure 5 - CE 15x15A Assembly Class Lattice Layout

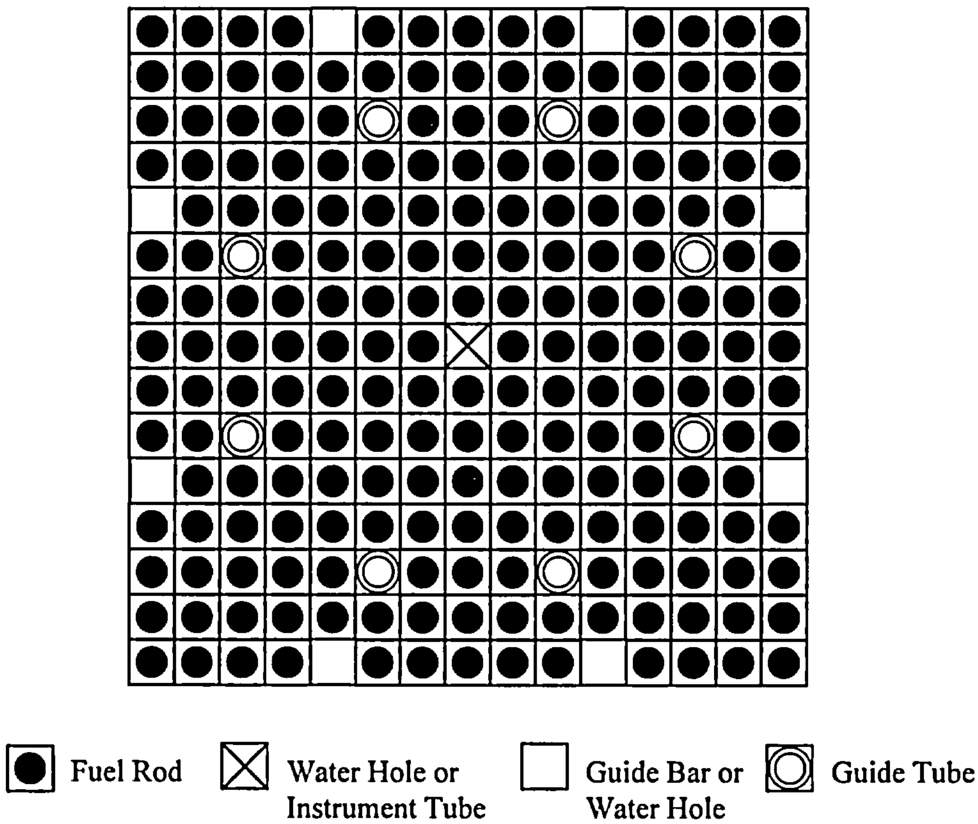


Note 1: The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Note 3: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Figure 6 - CE 15x15B Assembly Class Lattice Layout

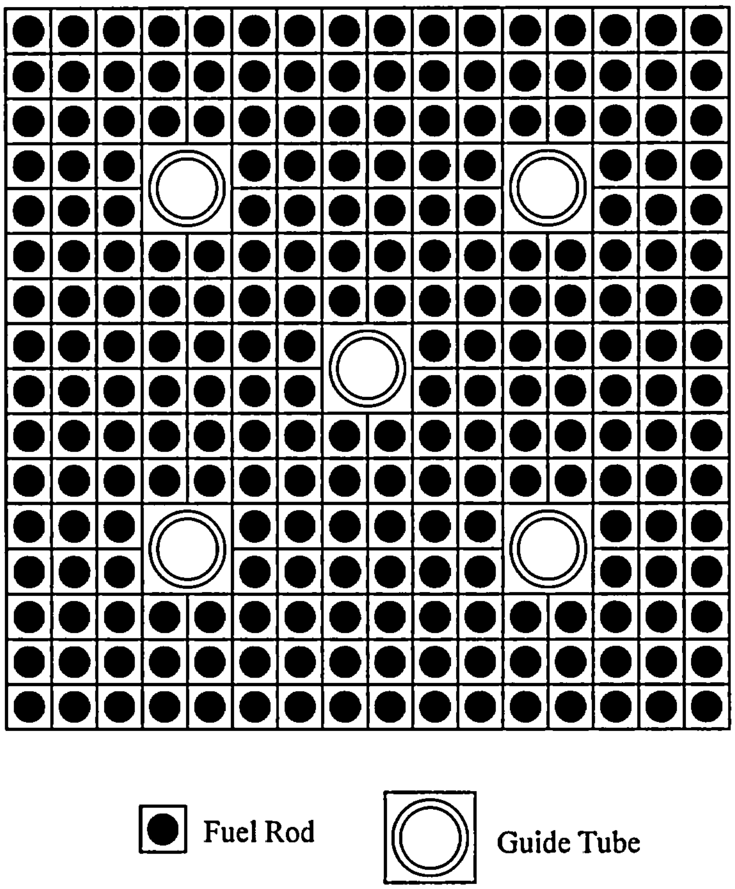


Note 1: The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Note 3: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Figure 7 - CE 15x15C Assembly Class Lattice Layout



Note 1: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Figure 8 - CE 16x16 Assembly Class Lattice Layout

1.2.2 Maximum Permissible MSB Leak Rate

Limit/Specification:

Less than or equal to 1.0×10^{-4} standard cubic centimeters per second (scc/sec) at 0.5 atm differential pressure.

Applicability: MSB inner seal (shield lid weld) confinement boundary.

Objective:

1. To limit the total radioactive doses normally released by each cask to negligible levels. Should fission gases escape the fuel cladding, they will remain confined by the MSB confinement boundary.
2. To retain helium cover gases within the MSB and prevent oxygen from entering the MSB. The helium improves the heat dissipation characteristics of the VSC and prevents any oxidation of fuel cladding.

Action: The leak rate shall be checked using calibrated instruments and written procedures. Procedures should be prepared to ANSI N14.5 (standard for leak testing of shipping cask) or equivalent. If the leak rate exceeds 10^{-4} scc/sec, the leak point must be found and repaired. The confinement boundary of the MSB itself may be easily repaired, since the field welding is all performed on the MSB outer surface.

Surveillance: The MSB shall be tested after the inner seal weld (shield lid weld) has been completed. The MSB will be pressurized with helium to 1.5 atm and a hand-held sniffer may be used (per manufacturer's instructions) to determine a leak rate. If the rate is within the limit, addition testing and surveillance are not required, since there are no normal or accident conditions that will breach the structural integrity and leak tightness of the MSB.

1.2.3 Maximum Permissible Air Outlet Temperature

Limit/Specification:

The equilibrium air temperature at the outlet of a fully loaded VSC (24 kW) shall not exceed ambient by more than 110°F.

Applicability: This temperature limit applies to all VSCs stored in the ISFSI. If a cask is placed in service with a heat load less than 24 kW, the limiting temperature difference between outlet and ambient shall be determined by a calculation performed by the user using the same methodology and inputs documented in the FSAR.

Objective: The objective of this limit is to ensure that the temperatures of the fuel cladding and the VSC concrete do not exceed the temperatures calculated in Section 4.0 of the FSAR. That section shows that if the air temperature increase (for 24 kW) is below 110°F (expected to be 89°F for normal operation), the fuel cladding and concrete will be below both their temperature criteria for normal operation and the maximum heat load transient (125°F ambient, full solar and full thermal load). An additional objective of the temperature measurements is to confirm the thermal performance of the cask and provide base-line data.

Action: If an air temperature rise of greater than 110°F, or greater than predicted, is observed for any VSC placed in service, the first action should be to check all inlet and outlet ducts for airflow blockage. If environmental factors can be ruled out as the cause of the excessive cask temperatures, this condition indicates that the fuel assemblies may be producing heat at a rate higher than specified in Section 2.0 of the FSAR. If fuel assemblies meeting the fuel specification in Technical Specification 1.2.1 have been loaded into the cask and the temperature difference is greater than 110°F, or that predicted for less heat loads, then this condition is not addressed in the FSAR and will require additional measurements and analysis to determine that the actual performance of the cask is within the limits analyzed in the FSAR. If the excessive temperatures cause the cask to perform in an unacceptable manner, or the temperatures cannot be controlled to within acceptable criteria, the cask shall be unloaded and a letter report shall be submitted to NRC within 30 days.

Surveillance: The ambient temperature and cask outlet air temperatures for the first VSC shall be measured and recorded daily for a period of 1 week after the VSC has been placed in service. The ambient temperature and cask outlet temperatures for the rest of the VSCs shall be measured and recorded upon placement in service and at intervals not to exceed 48 hours until the cask has reached thermal equilibrium. After reaching thermal equilibrium, thermal performance of each cask shall be verified on a daily basis through visual inspection of the VCC vent screens in accordance with Technical Specification 1.3.1.

1.2.4 Maximum External Surface Dose Rate

Limit/Specification:

The external surface average dose rate from all types of radiation will be less than 100 mrem per hour on the sides and 200 mrem/hr on the top. Dose rates at the air inlets and outlets will be below 350 and 100 mrem/hr, respectively.

Applicability: This dose rate limit shall apply to the entire external surface of the VCC, except the bottom surface.

Objective: The external dose rate is limited to this value to ensure that the cask has not been inadvertently loaded with fuel not meeting the specifications in Section 2.0 of the FSAR, to provide verification for plant personnel that radiation levels are acceptably low, and to satisfy the 10CFR72 dose rate limit of 25 mrem/year at the ISFSI controlled area boundary.

Action: If the measured dose rates are above those values listed above, correct fuel loading shall be verified. If correct fuel is loaded, specific analyses must demonstrate compliance with 10 CFR Part 20 and 10 CFR Part 72 radiation protection requirements, or appropriate action must be taken to comply with the acceptable limits. A letter report, summarizing the action taken and the results of investigation conducted to determine the cause of the high dose rates, shall be submitted to the NRC within 30 days. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to the administrator of the appropriate NRC regional office.

Surveillance: The external surface dose rate shall be measured after loading the MSB in the VCC and before transfer to the storage pad. The side dose rate shall be measured at a distance of 5 feet from the bottom of the VCC and at four equally spaced radial locations. The top dose rate shall be measured at the VCC lid center and the VCC outer lid edge. The dose rate measurement shall account for the effects of background radiation on the absolute dose rate measurements.

1.2.5 Maximum MSB Removable Surface Contamination

Limit/Specification:

10^{-4} μCi/cm^2 gamma-beta

10^{-5} μCi/cm^2 alpha

Applicability: MSB external surface.

Objective: Keep removable surface contamination level low enough so that offsite doses will be below 1 mrem, even in the event that contamination became loose and behaved as a particulate or gaseous release.

Action: If the limit is exceeded, the MSB exterior shall be washed by flushing the MSB-MTC gap with water, or other suitable decontamination solution, and additional contamination surveys taken until the limit is met.

Surveillance: Contamination surveys shall be taken on the MSB exterior, within 6 inches of the top of the MSB. Contamination surveys shall be taken on the MTC interior and bottom exterior surfaces after the MSB has been transferred to the VCC. The contamination surveys for removable surface contamination shall be conducted after the loaded MSB is removed from the pool and before the VSC is moved to the storage pad.

1.2.6 Boron Concentration in the MSB Cavity Water

Limit/Specification:

The MSB cavity shall be filled only with water having a boron concentration equal to, or greater than, the concentration specified (as a function of assembly initial enrichment for each defined assembly class) in Figures 9 through 16.

Applicability: This specification is applicable to the loading and unloading of all MSBs.

Objective: To ensure a subcritical configuration is maintained in the case of accidental loading of the MSB with unirradiated fuel.

Action: If the boron concentration is below the required weight percentage concentration (gm boron/10^6 gm water), add boron and re-sample, and test the concentration until the boron concentration is shown to be greater than that required.

Surveillance: Written procedures shall be used to independently determine (two samples analyzed by different individuals) the boron concentration in the water used in the spent fuel pool and that used to fill the MSB cavity.

1. Within 4 hours before insertion of the first fuel assembly into the MSB, the dissolved boron concentration in water in the spent fuel pool and in the water that will be introduced into the MSB cavity shall be independently determined (two samples chemically analyzed by two individuals).

2. Within 4 hours before flooding the MSB cavity for unloading the fuel assemblies, the dissolved boron concentration in water in the spent fuel pool and in the water that will be introduced into the MSB cavity shall be independently determined (two samples analyzed chemically by two individuals).

3. The dissolved boron concentration shall be reconfirmed at intervals not to exceed 48 hours until such time as the MSB is removed from the spent fuel pool or the fuel is removed from the MSB.

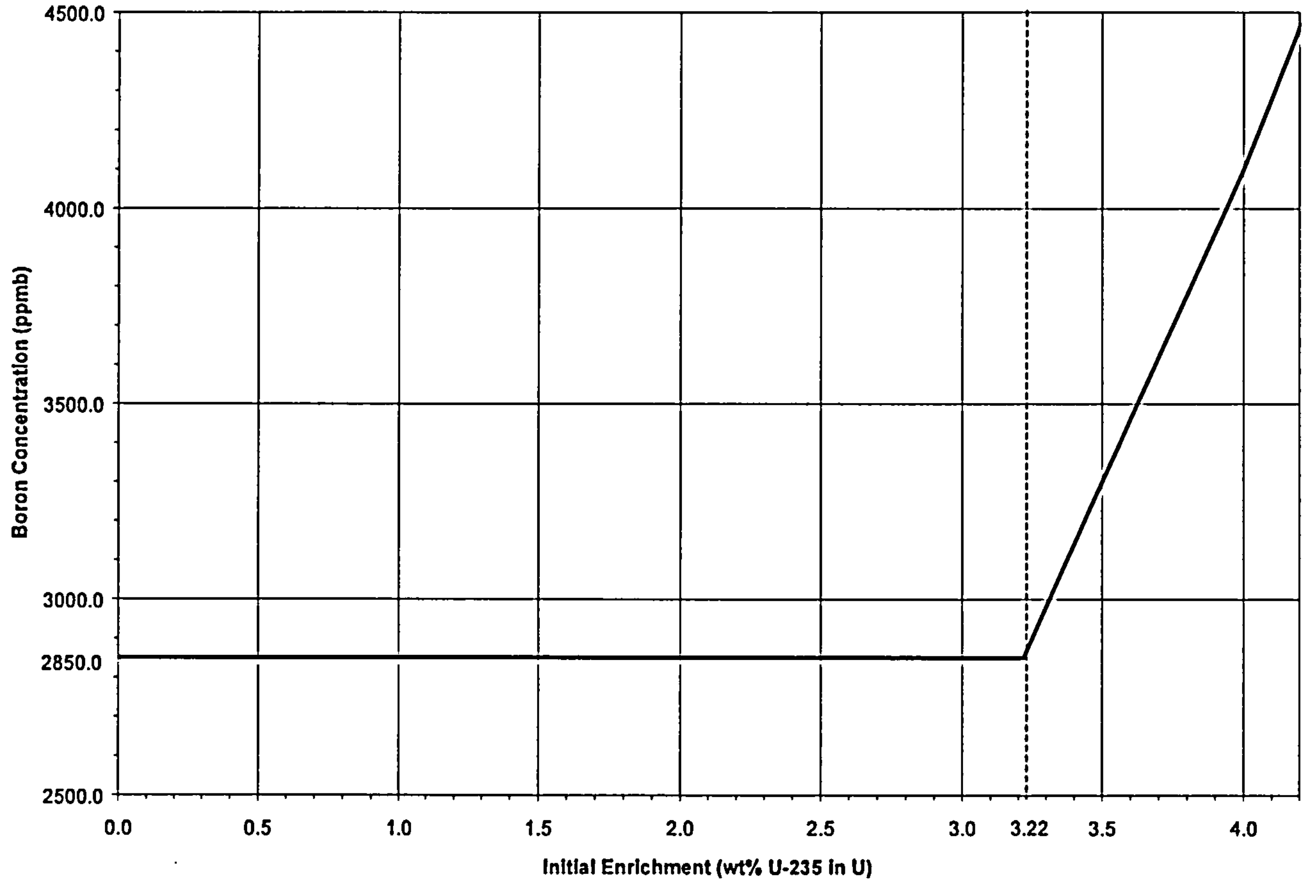


Figure 9 - B&W 15x15 Assembly Class Minimum Required Soluble Boron

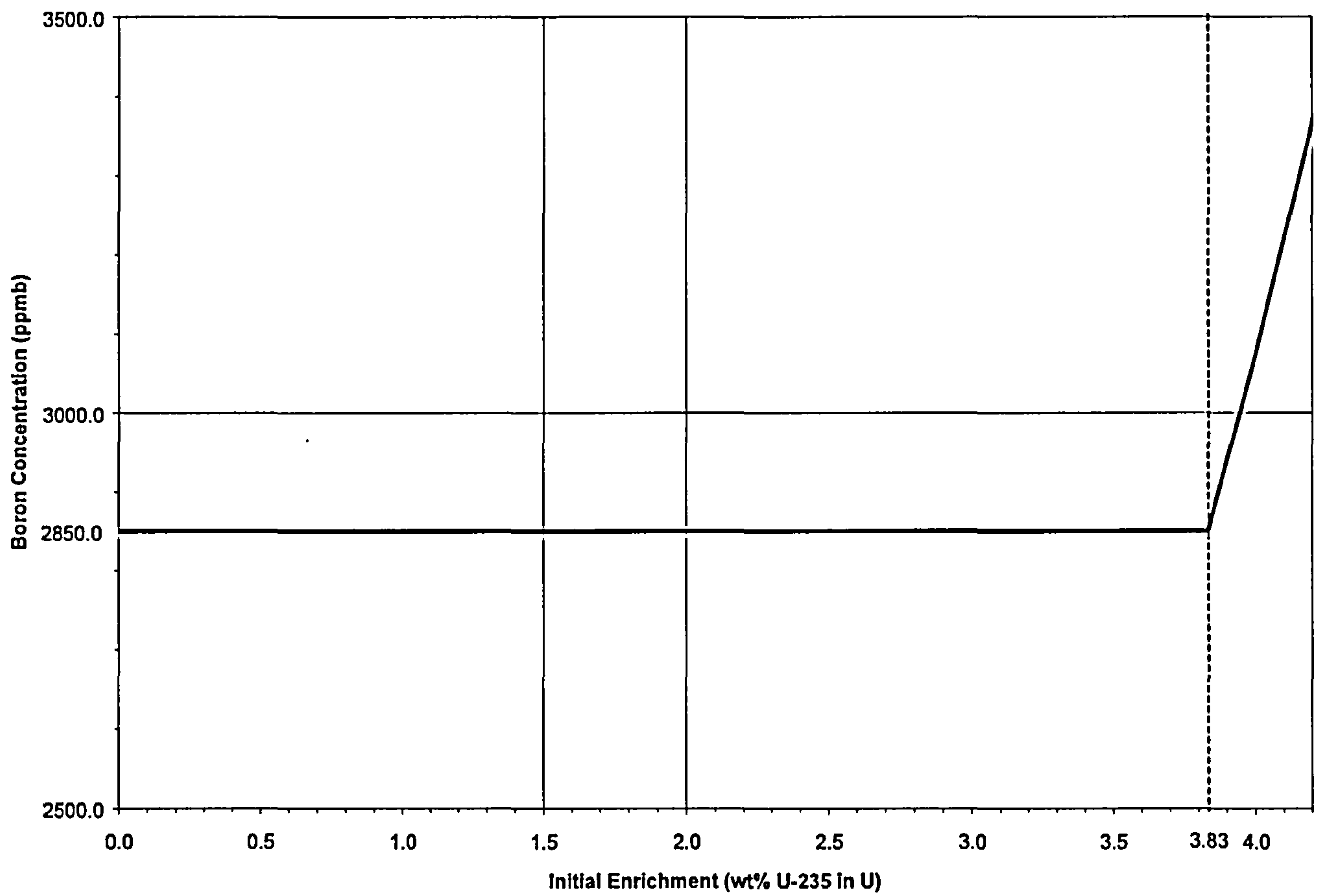


Figure 10 - W 14x14 Assembly Class Minimum Required Soluble Boron

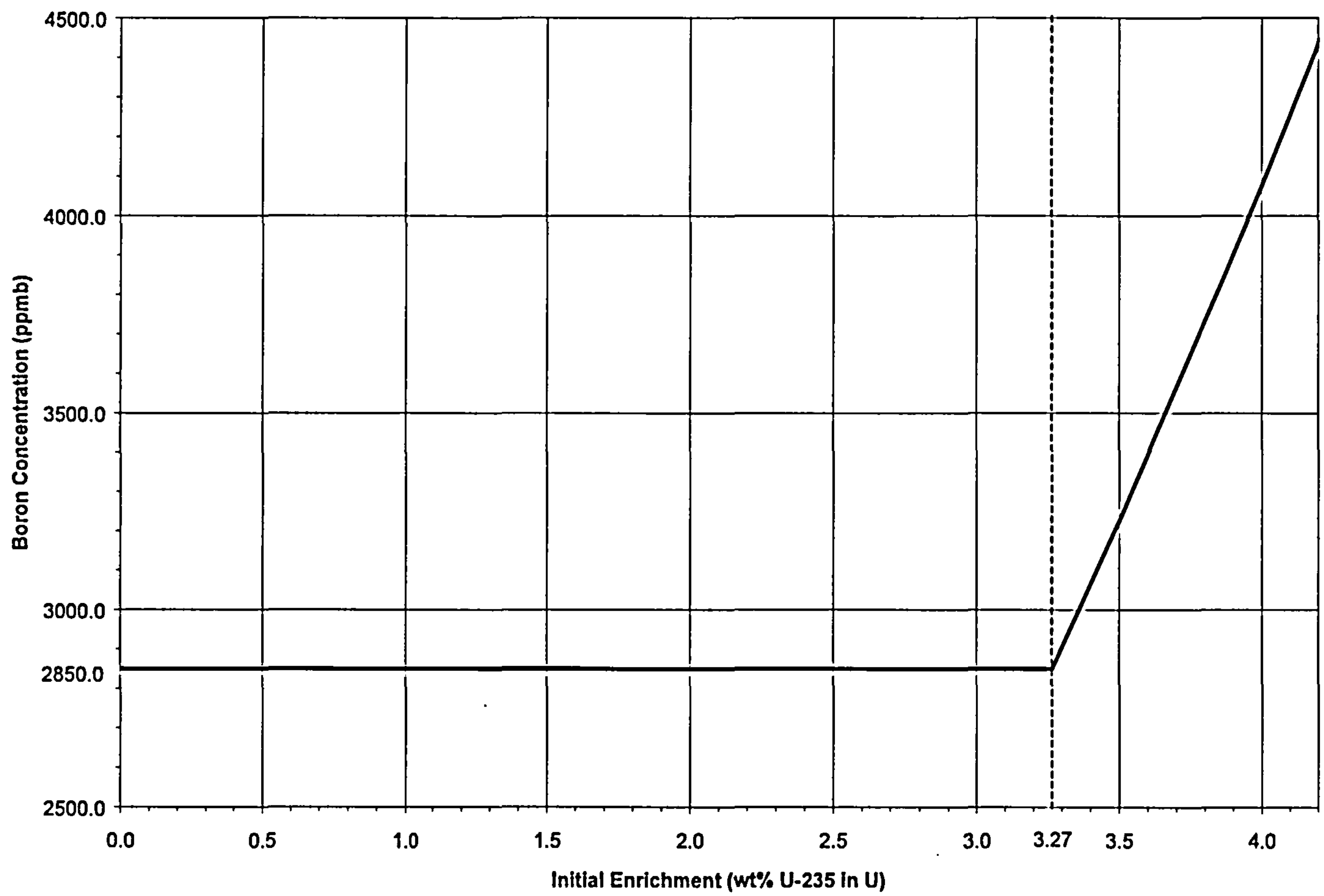


Figure 11 - W 15x15 Assembly Class Minimum Required Soluble Boron

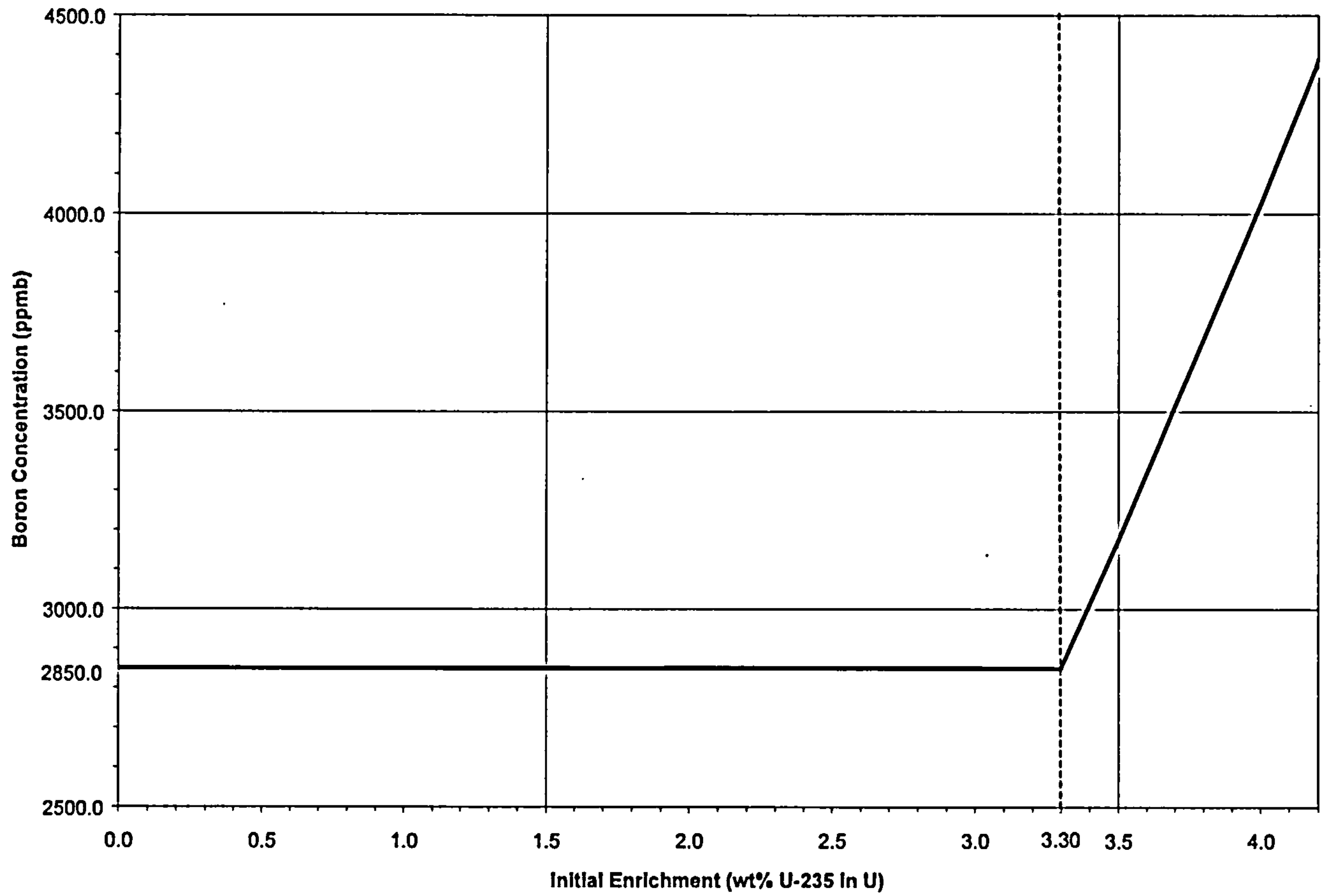


Figure 12 - W 17x17 Assembly Class Minimum Required Soluble Boron Results

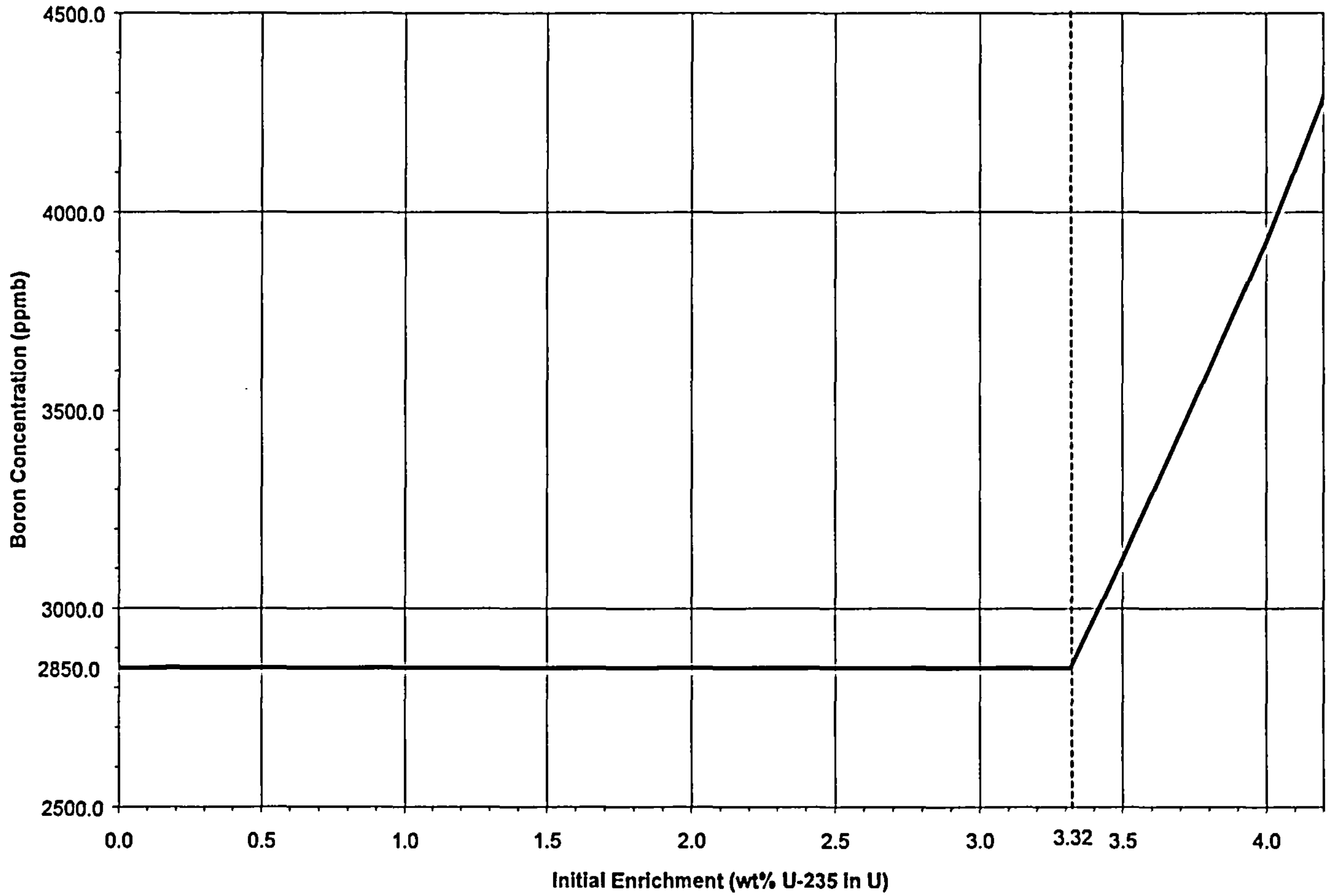


Figure 13 - CE 15x15A Assembly Class Minimum Required Soluble Boron Results

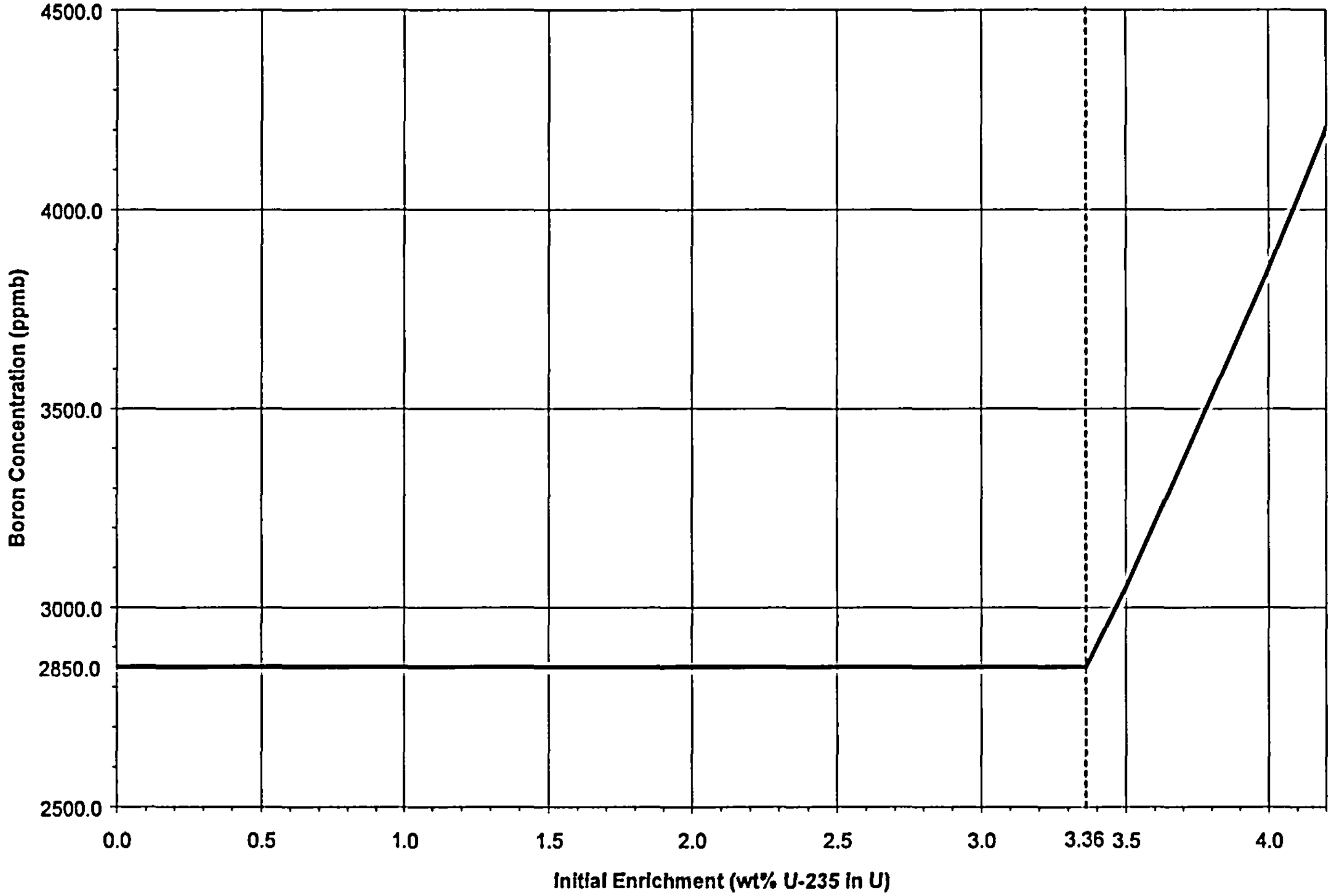


Figure 14 - CE 15x15B Assembly Class Minimum Required Soluble Boron Results

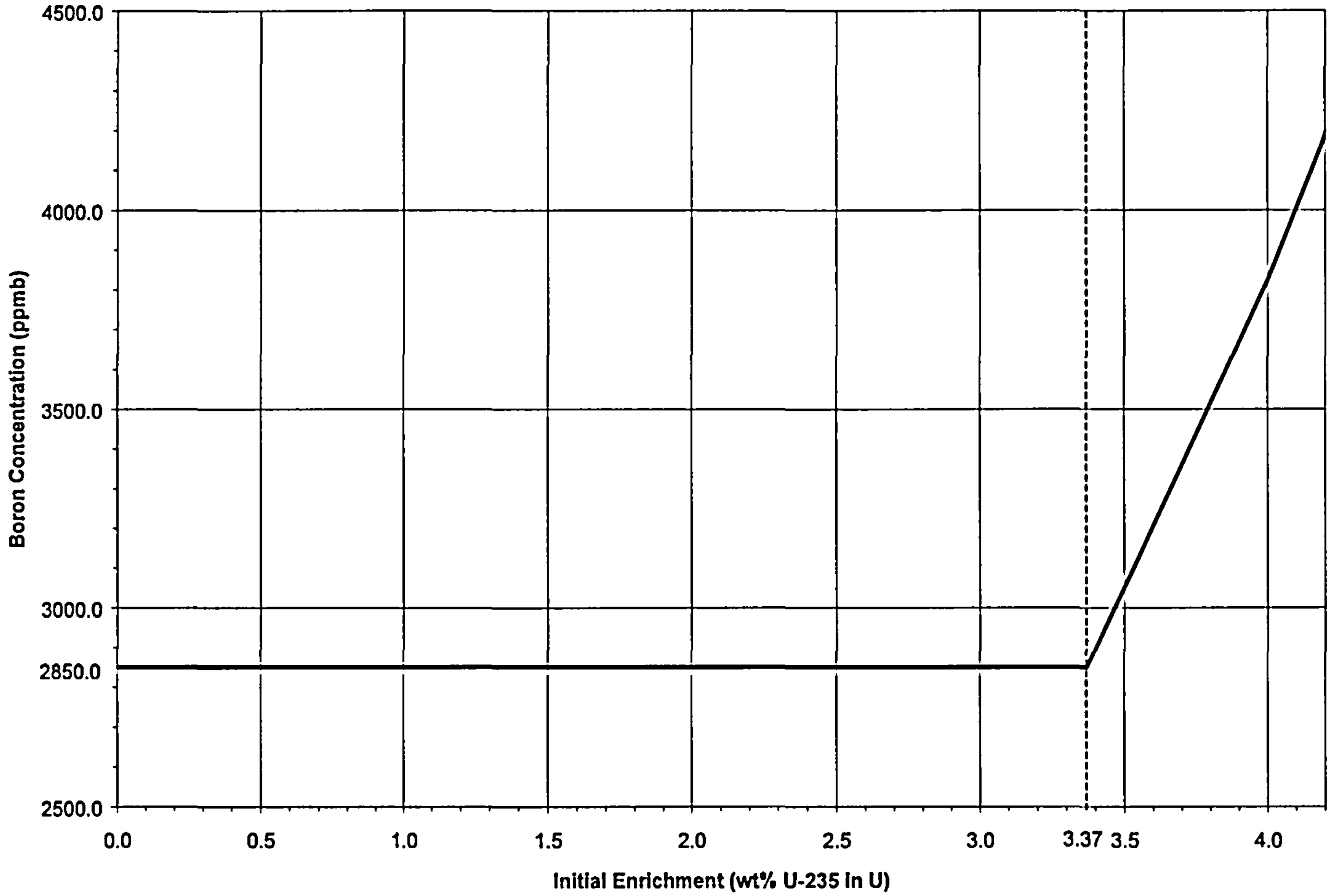


Figure 15 - CE 15x15C Assembly Class Minimum Required Soluble Boron Results

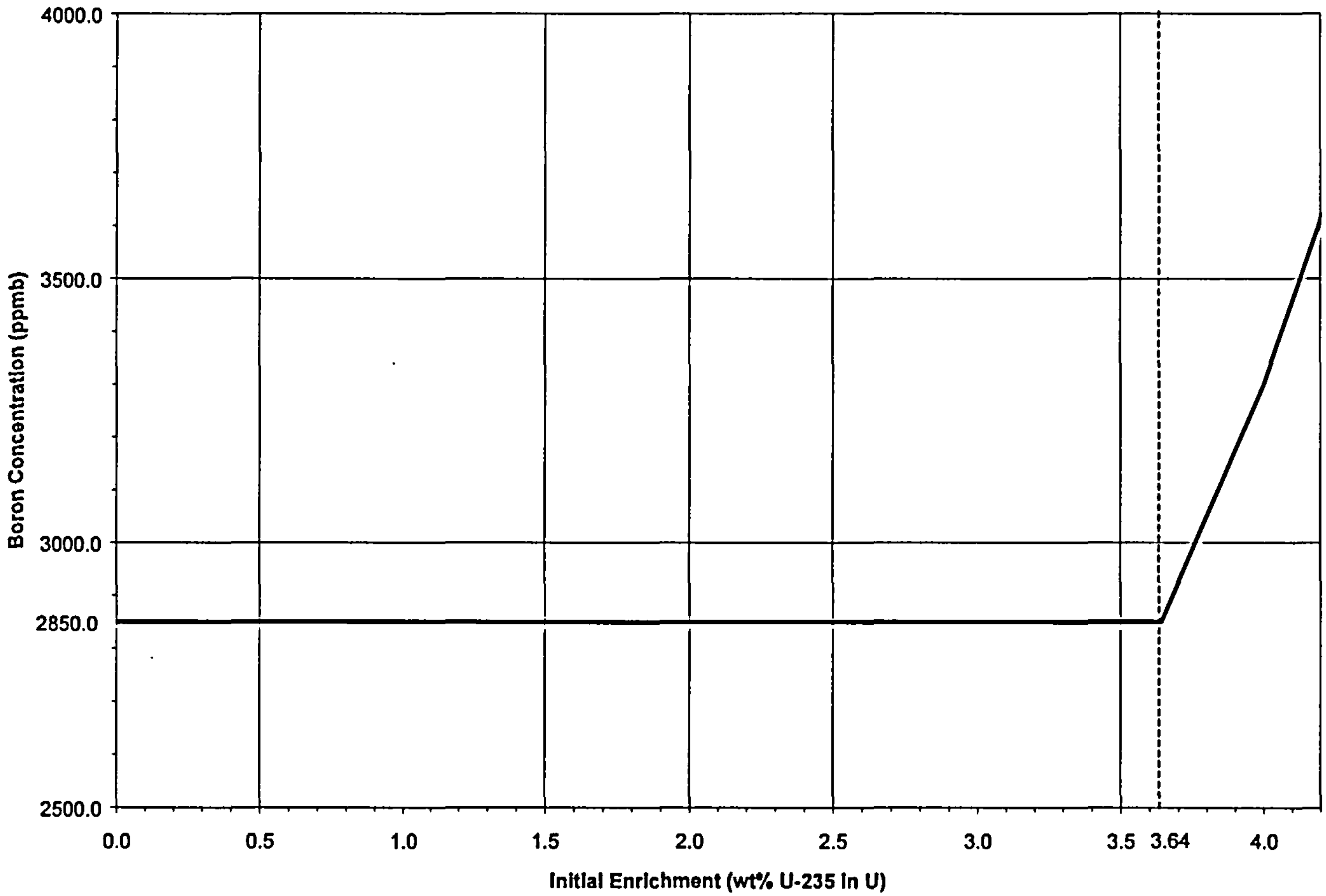


Figure 16 - CE 16x16 Assembly Class Minimum Required Soluble Boron Results

1.2.7 MSB Vacuum Pressure During Drying

Limit/Specification:

Vacuum Pressure:	Less than or equal to 3 mm Hg
Time at Pressure:	Greater than or equal to 30 min.
Number of Pump-Downs:	2

Applicability: This specification is applicable to all MSBs.

Objective: To ensure a minimum water content.

Action: Once the required vacuum pressure specification is obtained, perform helium backfill to 22.1 psia ± 0.5 psia, and repeat evacuation.

If the required vacuum pressure cannot be obtained:

1. Check and repair, or replace, the vacuum pump;
2. Check and repair the vacuum tubing as necessary; or
3. Check and reseal the shield lid fitting(s).

Surveillance: No maintenance or tests are required during normal storage. Surveillance of the vacuum gauge is required during the vacuum drying operation.

1.2.8 MSB Helium Backfill Pressure

Limit/Specifications:

Helium 14.5 psia ± 0.5 psia backfill pressure (stable for 30 minutes after filling).

Applicability: This specification is applicable to all MSBs.

Objective: To ensure that: (1) the atmosphere surrounding the irradiated fuel is a non-oxidizing inert gas; (2) the atmosphere is favorable for the transfer of decay heat; and (3) the MSB does not become over-pressurized.

Action: If the required pressure cannot be obtained:

1. Check and repair or replace the pressure gauge;
2. Check and repair or replace the pressure tubes, connections, and valves;
3. Check and repair or replace helium source; and
4. Check and repair the welds on MSB structural lid.

If pressure exceeds the criterion:

Release a sufficient quantity of helium to lower the cavity pressure.

Surveillance: No maintenance or tests are required during the normal storage. Surveillance of the pressure gauge is required during the helium backfilling operation.

1.2.9 Non-Destructive Examination of Shield and Structural Lid Seal Welds

Limit/Specification:

The MSB pressure boundary shield lid, structural lid and valve cover plate closure welds shall be liquid penetrant tested (PT) in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Article NC-5000 (1986 edition, 1988 addenda). The PT acceptance standards shall be as described in Subsection NC-5350.

In addition, the MSB structural lid-to-shell weld shall be examined by ultrasonic testing (UT) in accordance with the criteria defined in the "Guideline Requirements for the Time-of-Flight Diffraction Ultrasonic Examination of the VSC-24 Structural Lid to Shell Weld," VMSB-98-001, latest version.

The specified PT and UT examination shall begin no sooner than two hours after completion of the weld to be examined.

Applicability: For all MSBs, the PT examination is applicable to:

1. Root and final weld surfaces between the shield lid and shell, and between the structural lid and shell; and

2. Final weld surfaces between the structural lid and shield lid, and between the valve cover plate and structural lid.

The confirmatory UT examination is applicable to the completed MSB structural lid-to-shell weld.

Objective: To ensure that the MSB is adequately sealed and leak-tight, and to confirm the integrity of the structural lid-to-shell weld.

Action: If the PT examination indicates that a weld is unacceptable:

1. The weld shall be repaired in accordance with Article NC-4000 Fabrication and Installation, ASME Boiler and Pressure Vessel Code, Section III – Division 1, Subsection NC (1986 edition, 1988 addenda); and

2. The repaired weld shall be re-examined in accordance with the requirements of this specification.

If indications are found as a result of the UT examination:

1. Evaluate the flaw proximity per ASME Section XI, IWA-3300 (1989 edition);

2. Compare each flaw to the flaw screening criteria provided below:

Acceptable Flaw Depth (for lengths less than or equal to 0.7 in.)	Acceptable Flaw Depth (for lengths greater than 0.7 in.)
0.37 inches	0.16 inches

3. If a flaw is unacceptable, perform further flaw-specific evaluations (i.e., linear-elastic fracture mechanics or elastic-plastic fracture mechanics) per VMSB-98-001, latest revision, to determine whether the flaw is acceptable for continued operation; or

4. Repair the weld in accordance with Article NC-4000 Fabrication and Installation, ASME Boiler and Pressure Vessel Code, Section III - Division 1, Subsection NC (1986 edition, 1988 addenda); and

5. Re-examine the repaired weld in accordance with this specification.

Surveillance: During MSB closure operations.

1.2.10 Placement of the VSC on the Storage Pad

Limit/Specification:

Each VSC shall be placed in a storage array with at least 15-ft ± 1 ft, center-to-center, spacings.

Applicability: This specification applies to all VSCs.

Objective: To provide easy access between casks, and to meet the thermal analysis.

Action: The center-to-center spacing shall be measured upon placement.

1.2.11 Average Ambient Temperature

Limit/Specification:

The yearly average ambient temperature shall be 75°F, or less. Yearly average temperature is to be determined as follows, or by equivalent methodology. Yearly average may be hourly, daily, or monthly average temperatures added together and divided by 8760, 365 or 12, respectively.

The average daily ambient temperature shall be 100°F, or less.

Applicability: This specification applies to every site where the VSC will be deployed.

Objective To ensure that the long-term ambient conditions are bounded by the analysis.

Action: The yearly average ambient temperature is to be determined from suitable site data, Federal or local government agency data, or other sources. Based on information in the FSAR, all United States power plant sites should be bounded by the value of 75°F.

1.2.12 Minimum Temperature for Moving the Loaded MSB

Limit/Specification:

A VCC containing a loaded MSB shall only be moved at ambient temperatures of 0°F or above, coincident with a structural lid-to-shell weld temperature of 30°F or above.

Objective: To conform with design basis criteria for brittle fracture.

Action: Confirm before moving the VCC containing the loaded MSB that the ambient temperature is at 30°F or above.

If the ambient is less than 30°F but 0°F or greater, confirm that the structural lid-to-shell weld is at 30°F or above. Physical measurement should be used to determine the structural lid-to-shell weld temperature. Alternately, calculations similar to those presented in Chapter 4 of the FSAR may be used for the specific fuel to determine the minimum MSB shell temperature for any particular ambient condition.

Surveillance: The temperatures shall be measured before movement of the loaded MSB.

1.2.13 Minimum Temperature for Lifting the MTC

Limit/Specification:

The MTC containing a loaded MSB shall only be moved at ambient temperatures of 40°F or above.

Objective: To conform to the design criteria for brittle failure.

Action: Confirm that the ambient temperature is 40°F or above before movement of the MTC containing a loaded MSB.

Surveillance: The MTC ambient temperature shall be determined before movement of the MTC containing a loaded MSB.

1.2.14 MSB Handling Height

Specification:

1. The loaded VCC shall not be handled at a height greater than 60 inches.
2. In the event of a drop of a loaded VCC from a height greater than 18 inches: (a) fuel in the MSB shall be returned to the reactor spent fuel pool; (b) the MSB shall be removed from service and evaluated for further use; and (c) the VCC shall be inspected for damage.

Applicability: The specification applies to handling the VCC, loaded with the MSB, on route to, and at, the storage pad.

Objective:

1. To preclude a loaded VCC drop from a height of greater than 60 inches.
2. To maintain spent fuel integrity, according to the spent fuel specification for storage, continued containment integrity, and VCC functional capability, after a tipover or drop of a loaded VCC from a height greater than 18 inches.

Surveillance: In the event of a loaded VCC drop accident, the system will be returned to the reactor fuel handling building. After the fuel has been returned to the reactor spent fuel pool, the MSB and the VCC will be inspected and evaluated for future use.

1.3 Surveillance Requirements

Surveillances required to implement the requirements of a number of specifications were included as part of the specifications in the previous sections. Additional surveillances, required for normal operation and after accident conditions, are described below. Table 3 summarizes all the surveillance requirements, including those discussed in previous sections.

1.3.1 Visual Inspection of Air Inlets and Outlets

Surveillance: A visual surveillance of the wire mesh screens covering the air inlets and outlets shall be conducted daily.

Action: If the surveillance shows signs of degradation, breach of the screens or other possible sources of blockage such as insect infestation, a close-up inspection of the air inlets and outlets shall be conducted to determine possible blockage and removal if present.

1.3.2 Exterior VCC Surface Inspection

Surveillance: The VCC exterior surface shall be inspected annually for any damage (chipping, spalling, etc.).

Action: Any defects larger than one-half inch in diameter (or width) and deeper than one-quarter of an inch shall be repaired by re-grouting, according to the grout manufacturer's recommendations.

1.3.3 Interior VCC Surface Inspection

Surveillance: The VCC interior surfaces and MSB exterior surfaces of the first VSC unit placed in service at each site shall be inspected, to identify potential air flow blockage and material degradation after every 5 years in service.

Action: Results of the surveillance shall be documented, and a letter report, summarizing the findings, shall be submitted to the NRC within 30 days. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to administration of the appropriate regional office.

Table 3 - Summary of Surveillance Requirements

	Surveillance	Period	Technical Specification or Surveillance Requirement
1.	Spent Fuel Assembly Identification	L	1.2.1
2.	Weld Leak Testing	L	1.2.2
3.	Air Outlet Temperature	L, AN	1.2.3
4.	Dose Rates	L	1.2.4
5.	MSB Surface Contamination	L	1.2.5
6.	Boron Concentration	PL	1.2.6
7.	Vacuum Pressure	L	1.2.7
8.	MSB Helium Backfill Pressure	L	1.2.8
9.	Weld Nondestructive Examination	L	1.2.9
10.	Ambient Temperature (VCC Movement)	AN	1.2.12
11.	Ambient Temperature (MTC Lift)	AN	1.2.13
12.	VCC, MSB Drop	AN	1.2.14
13.	Air Inlet and Outlet Surveillance	D	1.3.1
14.	Cask Exterior (normal)	Y	1.3.2
15.	Cask Interior	AN	1.3.3

Legend

L During or within 24 hours of loading and before movement to storage pad.

PL Before loading and unloading.

D Daily -- At least once per 24 hours.

W Weekly -- At least once per 7 days.

M Monthly -- At least once per 31 days.

Y Yearly -- At least once per 366 days.

AN As necessary/as required.

TECHNICAL SPECIFICATIONS BASES

FOR THE

VSC-24 STORAGE CASK SYSTEM

B.1.2.1 Fuel Specification

Basis: The specification is based on consideration of the design basis parameters included in the FSAR. Such parameters stem from the type of fuel analyzed, physical and structural limitations, criteria for criticality safety, criteria for heat removal, and criteria for radiological protection. The VSC-24 system is designed for dry, vertical storage of irradiated pressurized water reactor (PWR) fuel.

The principal design parameters of the fuel to be stored are found in FSAR Section 2.1. Allowable parameters of stored fuel are described in detail in Table 2. The VSC-24 cask can accommodate the standard PWR fuel designs manufactured by Combustion Engineering (CE), Exxon, Westinghouse, and Babcock and Wilcox (B&W). The specific designs accommodated by the VSC-24 are listed in Table 1. For a given assembly to qualify for loading into the cask, however, it must meet all of the geometry specifications given for that assembly type in Table 2.

For all of the listed assembly designs except W 15x15 and CE 16x16, assemblies containing burnable poison rod assemblies (BPRAs) and thimble plug assemblies (TPAs) may be loaded into the VSC-24 cask. The poison rods or plugging rods may be Zircaloy or stainless steel clad. Plugging (TPA) rods may be either hollow or solid metal tubes. CE 15x15 assemblies may contain poison rods or poison "clusters" in any number of the assembly guide tubes.

BPRAs consist of a group of hollow metal rods that are filled with boron carbide (B_4C), aluminum oxide (Al_2O_3), a silver-indium-cadmium mixture (Ag-In-Cd), borosilicate glass, or halfnium (Hf) poison material. These rods are inserted into the assembly guide tubes. The metal rods are made of stainless steel or zircaloy. TPAs contain a group of solid or hollow metal rods. These (relatively short) rods are inserted into the top section of the assembly guide tubes (in order to "plug" the guide tubes). TPA rods consist of stainless steel or zircaloy. Both BPRAs and TPAs have additional metal hardware (a "spider assembly") at the top end which binds the group of rods together and provides a handle to allow the BPRA or TPA to be engaged by fuel assembly handling equipment. The top end hardware is made of stainless steel, zircaloy, and/or nickel alloys such as inconel.

The VSC-24 may also accommodate PWR assemblies with any number of solid zircaloy rods, poison rods, or Gd_2O_3 rods in place of standard fuel rods, given that the diameter of the replacement rods does not exceed that of the standard fuel rods. Poison rods (like BPRA rods) are hollow stainless steel or zircaloy rods filled with B_4C, Al_2O_3, Ag-In-Cd, borosilicate glass, or Hf poison material. Gd_2O_3 rods are hollow stainless steel or zircaloy rods that are filled with Gd_2O_3 poison material instead of UO_2 fuel.

For fuel assemblies with stainless-steel-clad or solid stainless steel rods in place of fuel rods, or with solid stainless steel rods in guide tube locations, an evaluation must be performed to verify that the assembly fuel zone cobalt content lies within the analysis basis discussed in FSAR Section 5.2.1.2 (and in FSAR Table 5.2-2). If the overall assembly fuel zone cobalt quantity does not exceed the design basis quantity of 46.7 grams (given in FSAR Table 5.2-2), the assembly may be loaded into any location inside the MSB. If the assembly fuel zone cobalt quantity exceeds the design basis amount of 46.7 grams, but does not exceed the "high" assembly cobalt quantity of 250 grams (defined and qualified in FSAR Section 5.5.2), the assembly may be loaded into any one of the inner 12 fuel sleeves in the MSB (i.e., it may not be loaded into any of the 12 sleeves that lie along the MSB edge).

The analyses presented in the FSAR are based on non-consolidated, zircaloy-clad fuel, with no known or suspected gross cladding failures.

The physical parameters that define the mechanical and structural design of the VCC and the MSB are the fuel assembly dimensions and weight provided in

FSAR Table 2.1-1. They represent the heaviest fuel, so that the calculated stresses bound the PWR fuel designs to be stored.

The design basis for nuclear criticality safety is based on assembly types defined in Table 2, with initial enrichments up to 4.2 wt. percent ^{235}U. The criticality design criteria ensure that the MSB remains subcritical (k_{eff} less than or equal to 0.95) under normal, off-normal, and accident conditions assuming the loading of unirradiated fuel. The assembly initial enrichment is defined as the maximum lattice average enrichment that occurs at any axial location within the assembly.

Primary protection against accidental criticality is provided by operational procedures to prevent the introduction of water into the cask not containing the minimum specified concentration of dissolved boron (see Technical Specification 1.2.6). Technical Specification 1.2.6 requires that, prior to the introduction of water into the MSB, two water samples be taken and chemically analyzed by two individuals to independently verify the boron concentration in the water. The likelihood of one test failing to detect the correct boron concentration is small and the probability of two independent tests failing, concurrently, is highly unlikely.

The thermal design criterion of the fuel to be stored is that the maximum heat generation rate per assembly be such that the fuel cladding temperature is maintained within established criteria during normal and off-normal conditions. Fuel cladding temperature criteria were established by the applicant based on methodology in PNL-6189 and PNL-6364 (FSAR References 1.1 and 4.1). Based on this methodology, a maximum heat generation rate of 1 kW per assembly is a bounding value for the PWR fuel to be stored.

The radiological design criterion is that the gamma and neutron source strength of the irradiated fuel assemblies not be so high as to cause the cask exterior dose rate limits in Technical Specification 1.2.4 to be exceeded.

The design basis shielding calculations show that these dose rate limits are not exceeded for a cask loaded with 3.2 weight percent ^{235}U enriched PWR fuel, irradiated to an average fuel burnup of 35,000 MWd/MTU, with a post-irradiation time of 5 years.

Additional shielding analyses determine the cooling time required to meet the specified dose limits for different combinations of assembly burnup and initial enrichment. The minimum required cooling time to meet the 1.0 kW assembly decay power limit is also determined.

The minimum assembly cooling times at which both the 1.0 kW assembly decay power limit and the Technical Specification 1.2.4 dose rate limits are met are shown in FSAR Table 5.5-1.

The cooling times provided in FSAR Table 5.5-1 apply to the fuel assemblies with and without BPRAs or TPAs. The calculated assembly and radiation sources include both the fuel and the inserted BPRAs and TPAs.

The criticality analyses used to determine the minimum required soluble boron concentration and ensure a k_{eff} of less than 0.95, consider BPRAs inserted into the fuel assemblies. These criticality analyses conservatively modeled the presence of the individual rods of the BPRAs as zircaloy tubes filled with depleted B_4C (i.e., $(B_{11})_4C$).

BPRAs with cladding failures are acceptable for loading in the VSC-24 system. A failed BPRA loaded in the VSC-24 system would be depressurized and present a lower MSB accident pressure than that of an intact BPRA. Any release from a failed BPRA would not have an adverse effect on the internals of the MSB or the fuel assemblies stored in the MSB. An exception to this is poison rods containing Ag-In-Cd or Hf poison materials, which could possibly interact galvanically with the MSB internal components. For this reason, the cladding of BPRA rods or inserted poison rods that contain Ag-In-Cd or Hf poison material must be intact (i.e., must have no known or suspected gross failures).

B.1.2.2 Maximum Permissible MSB Leak Rate

Basis: If the MSB leaked at the largest undetectable leak rate (10^{-4} scc/sec), then only 1 percent of the helium would escape over a 20-year span. This amount would be negligible.

B.1.2.3 Maximum Permissible Air Outlet Temperature

Basis: If the air temperature rise is 110°F [21°F more than the 89°F rise calculated for the 75°F ambient case (FSAR Table 4.1-1), the maximum concrete and fuel cladding temperatures can be expected to be less than 21°F hotter than predicted (due to the non-linearity of radiation heat transfer). For a cask load of 24 kW, this condition would result in a maximum concrete temperature of 214°F and a maximum cladding temperature of 705°F. Both of these values are below the acceptable criteria (225°F for concrete and 712°F for 5-year cooled fuel).

B.1.2.4 Maximum External Surface Dose Rate

Basis: The basis for this limit is the shielding analysis presented in Section 5.0 of the FSAR.

B.1.2.5 Maximum MSB Removable Surface Contamination

Basis: If the MSB were covered over its entire surface with 2.1×10^7 dpm/cm^2 (9.5 μCi/cm^2) of Co-60, and all the contamination became loose and were released as a gaseous particulate cloud under the worse meteorological conditions, the dose at 200 meters would be less than 1 mrem. This basis and the analysis are presented in Section 11.1.4 of the FSAR. Therefore, using $10^{-4}\mu$Ci/cm^2 is a conservative limit, and ensures that the offsite dose limits in 10 CFR Parts 20, 50 (Appendix I), and 72 can be met. Significant amounts of residual contamination on the MTC surfaces above the specification are an indication that the MSB was not thoroughly decontaminated.

B.1.2.6 Boron Concentration in the MSB Cavity Water

Basis: The required boron concentration is based on the criticality analysis for an MSB with unburned fuel, maximum enrichment, and optimum moderation conditions. Required boron concentrations are calculated versus initial enrichment for each defined PWR assembly class. The sets of allowable values for assembly physical parameters important to criticality safety are defined for each PWR assembly class in Table 2.

The required boron concentrations apply only to assemblies that meet all of the physical parameter restrictions given in Table 2 for that assembly type. The boron concentrations are also applicable for assemblies containing BPRAs, TPAs, poison rods in place of fuel rods, or solid zircaloy or steel replacement rods.

The initial enrichments shown in Figures 9 through 16, upon which the minimum required boron concentrations are based, are defined as the maximum lattice average enrichment that occurs for any axial position within the assembly. Most PWR assembly types use a uniform fuel rod enrichment throughout the assembly lattice. For assemblies that have multiple fuel rod enrichments, an evaluation must be performed to verify that the actual fuel rod enrichment pattern is less reactive than a uniform fuel rod enrichment pattern at the lattice-average enrichment. Alternatively, the maximum enrichment of the individual fuel rods in the assembly lattice may be used to determine the required boron concentration.

B.1.2.7 MSB Vacuum Pressure During Drying

Basis: The value of 3 mm Hg for absolute pressure was selected to allow the use of standard vacuum pumps. If the only gas contained within the MSB cavity is considered to be super-heated steam at a pressure 3 mm Hg and 450°F, the moisture content of the MSB cavity is approximately 0.729 moles (assuming a perfect gas) and, hence, only 0.364 moles of O_2 are available (if 100 percent radiolysis is assumed). This O_2 could react with 1.09 moles of UO_2 (295 grams). However, the reaction of 295 grams of UO_2 would be negligible, compared to the 2225 grams of UO_2 in a single rod. Therefore, oxidation of 295 grams does not represent a threat to the safe operation of the VSC system.

However, since the multiple pump-down is performed, the O_2 partial pressure after backfilling with helium will not exceed $(760) \times (3/760)^2 = 0.01$ mm Hg.

B.1.2.8 MSB Helium Backfill Pressure

Basis: The value of 14.5 psia was selected to assure that the pressure within the MSB is within the design limits during any expected off-normal operating condition. The 14.5 psia backfill pressure assumes that the average helium temperature in the MSB is greater than 200ĶF when the MSB is sealed. The combination of pressure equal to 14.5 psia and temperature greater than 200ĶF assures an upper limit on the moles of helium in the MSB.

The MSB backfill helium shall be high-purity grade helium (99.995%).

B.1.2.9 Non-Destructive Examination of Shield and Structural Lid Seal Welds

Basis: Article NC-5000 Examination, ASME Boiler and Pressure Vessel Code, Section III – Division 1, Subsection NC (1986 edition, 1988 addenda).

Two hour delay in initiation of the specified PT and UT examinations ensures that closure welds will be inspected after any potential delayed hydrogen-induced cracking.

The leak tightness analysis for the MSB is based on welds being leak-tight to 10^{-4} scc/sec. These examinations are performed to ensure compliance with the leak tightness design criteria, and to confirm the integrity of the structural lid-to-shell weld.

B.1.2.10 Placement of the VSC on the Storage Pad

Basis: The access requirements are based on engineering judgement. The 15-ft, center-to-center spacing was also used to determine thermal radiation view factors in the thermal analysis. The ± 1 ft will not significantly affect the view factors, or the heat transfer, because most all heat is removed from the VSC by the natural draft circulation (not thermal radiation) from the exterior sides.

B.1.2.11 Average Ambient Temperature

Basis: The thermal analysis presented in the FSAR used 75°F as the long-term average temperature. However, it should be noted that significant margin exists (e.g., 45°F for concrete temperatures and 28°F for fuel temperatures). The thermal analysis presented in the FSAR used 100°F as the highest steady state ambient conditions with 125°F the maximum short-term temperature extreme (12 hrs).

B.1.2.12 Minimum Temperature for Moving the Loaded MSB

Basis: Movement of the loaded MSB at a 30°F ambient temperature or above conservatively satisfies this specification. Restricting movement of the loaded MSB below the temperatures specified is necessary to conform with the design criteria for brittle failure.

Each MSB shell material will have shown, during fabrication, by Charpy test (per ASTM A370) that it has 15 foot-pounds of absorbed energy at minus 50°F.

The temperature limit for the structural lid-to-shell weld effectively increases material toughness and permits allowable flaw size in the weld to be governed by primary stress criteria rather than by brittle fracture limits.

Specifications for future procurement and fabrication of MSB pressure retaining materials, including base materials and weld metal, shall specify a minimum Charpy V-notch impact absorbed energy value of 15 foot-pounds at minus 50°F. Additionally, for the MSB shell, lid, and weld materials associated with the structural lid-to-shell weld, the minimum Charpy V-notch impact absorbed energy shall be 45 foot-pounds at 0°F. These requirements define minimum values for material toughness and produce adequate margins of safety relative to the potential for brittle fracture under the most severe handling conditions.

B.1.2.13 Minimum Temperature for Lifting the MTC

Basis: The MTC material will have shown, during fabrication, that it has 15 ft-lb of absorbed energy at 0°F. Having Charpy test results, at 0°F, which show ductility (or other appropriate test to show that the Nil Ductility Temperature is lower than 0°F), is necessary to conform to the design criteria for brittle failure when the cask is moved at 40°F or higher. The MTC shell will have a temperature higher than ambient due to the heat source from the irradiated fuel. However, for conservatism and simplicity, it is recommended that the ambient temperature be used as the minimum shell temperature. If movement at lower temperatures is ever required, additional specific analysis or other actions that meet the approval of the NRC must be provided.

B.1.2.14 MSB Handling Height

Basis: Drops up to 60 inches, of the MSB inside the VCC, can be sustained without breaching the confinement boundary, preventing removal of spent fuel assemblies, or causing a criticality accident. This specification ensures that handling height limits will not be exceeded in transit to, or at the storage pad. Acceptable damage may occur to the VCC, MSB, and the fuel stored in the MSB, for drops of height greater than 18 inches. The specification ensures that the spent fuel will continue to meet the requirements for storage, the MSB will continue to provide confinement, and the VCC will continue to provide its design functions of cooling and shielding. Based on linear-elastic analysis methods, drops up to a height of 18 inches and less are not judged to be of concern.

B.1.3.1 Visual Inspection of Air Inlets and Outlets

Basis: The concrete temperature could exceed 350°F in the accident circumstances of complete blockage of all vents. Concrete temperatures over 350°F in accidents (without the presence of water or steam) are undesirable as they have uncertain impact on strength and durability. A conservative analysis (adiabatic heat case) of complete blockage of all air inlets or outlets indicates that the concrete can reach the accident temperature limit of 350°F in 30 hours.

B.1.3.2 Exterior VCC Surface Inspection

Basis: This action maintains the surface condition of the concrete exterior, preventing degradation of the concrete interior, and avoids any adverse impact on shielding performance.

B.1.3.3 Interior VCC Surface Inspection

Basis: To identify degradation mechanisms affecting system performance that were not identified in the FSAR. However, this surveillance is a conservative but prudent check to ensure suitable conditions remain in the VCC/MSB annulus.

13.0 QUALITY ASSURANCE

BFS will apply the BFS Manual of Quality Assurance as approved by the Nuclear Regulatory Commission for Subpart G of the 10CFR Part 72.

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APPENDICES
FOR THE
FINAL SAFETY ANALYSIS REPORT
FOR THE
VSC-24 VENTILATED STORAGE CASK SYSTEM

APPENDIX A

FUEL ASSEMBLY REGION EFFECTIVE THERMAL CONDUCTIVITY

A APPENDIX A - FUEL ASSEMBLY REGION EFFECTIVE THERMAL CONDUCTIVITY

A.1 INTRODUCTION

This appendix describes the calculation of an effective thermal conductivity for the fuel assembly region (i.e., the region within a storage sleeve). Two different approaches are utilized and a resulting conservative value was used in the subsequent ANSYS analyses of the MSB interior.

A.2 DESIGN INPUT AND ASSUMPTIONS

A.2.1 ASSUMPTIONS

The main assumption is that heat transfer within the fuel assembly and from the outer row of fuel rods to the guide sleeve wall can be handled by an effective thermal conductivity. The actual heat transfer is by a combination of conduction, convection and radiation. However, if one tried to model all the processes and the actual geometry (204 rods) the model would become extremely complex and probably undoable with the ANSYS code. Therefore, the effective thermal conductivity method is utilized.

A.2.2 INPUT

The main input to this problem is test data from several cask tests (References 4.3 - 4.7) and the standard solutions to the heat transfer equations and the Wooten-Epstein correlation (4.8).

A.3 CALCULATIONS

A.3.1 APPLICATION OF THE WOOTEN-EPSTEIN CORRELATION

The Wooten-Epstein Correlation (WEC) was specifically developed to model the heat flow out of fuel assemblies in shipping cask storage sleeves. It has been used several times in the past for both shipping cask and storage systems.

The WEC is shown below.

$$Q = \sigma C_1 F_1 A (T_E^4 - T_w^4) + C_2 A (T_E - T_w)^{4/3}$$

where,

C_1 = $[(4N)/(N+1)^2] = [(4)(15)/(15 + 1)^2] = 60/256 = 0.234$

A = Bundle surface area = 4 H L

H	=	144" = 12'
L	=	8.5" = 0.708'
A	=	$4(12)(0.708) = 33.99\ ft^2$
T_E	=	Hottest rod temperature (°R)
T_w	=	Cavity wall temperature (°R)
C_2	=	Conduction constant = 0.118 for air (which if used for He will be conservative)
F_1	=	Exchange Factor
	=	$[(1/\epsilon_e)+(1/\epsilon_w)]^{-1} = [(1/.8)+(1/.8) - 1]^{-1} = 0.67$
ϵ_e	=	$\epsilon_{zircaloy} = 0.8$
ϵ_w	=	0.8 (coated A-516 carbon steel)
σ	=	1.714×10^{-9} BTU/hr-ft^2 °R^4
Q	=	$(1.714 \times 10^{-9})(.234)(.67)(34.00)(T_E^4 - T_w^4)+(0.12)(34.0)(T_E-T_w)^{4/3}$
Q/A	=	$2.68 \times 10^{-10}\ (T_E^4 - T_w^4) + 0.12\ (T_E - T_w)^{4/3} = q''$

Using the above equation we can calculate Q for various selections of T_E and T_w. By then comparing this to the solution of the heat conduction equation for a square (see next section) we can calculate an effective thermal conductivity by the following equations.

$$Q/A = \frac{k_{eff}\,\Delta T}{1\,(.590)} = 2.68 \times 10^{-10}\ (T_E^4 - T_w^4) + 0.12\ (T_E - T_w)^{4/3}$$

Using this relationship, k_{eff} was calculated for a number of selected values of T_E and T_w.

The results are summarized below.

T_w (°F)	ΔT (wall to hottest rod)	Q Radiation (BTU/hr)	Q Convection (BTU/hr)	k_{eff} (BTU/hr ft °F)
100	127	1136	2289	0.17
200	111	1494	1922	0.20
300	95	1834	1569	0.23

T_w(°F)	ΔT (wall to hottest rod)	Q Radiation (BTU/hr)	Q Convention (BTU/hr)	k_{eff} (BTU/hr ft °F)
400	81	2165	1275	0.27
500	67	2403	946	0.32
582	58	2605	826	0.38
690	47	2776	628	0.46

A.3.2 EXAMINATION OF CASK TEST DATA AND PRE AND POST TEST ANALYSIS USING HYDRA AND COBRA

The cask test data previously used for the surface heat transfer coefficient was also examined to determine the effective heat transfer coefficient for each test.

The central assembly of each cask test was examined in the center (axially) region so that one can safely assume that all the heat transfer is radially out of the assembly.

The equation used to calculate the effective thermal conductivity is that for two dimensional square heat sources. This is conservative because in reality the heat is not flowing equally out all four sides of the square tubes but is instead tending to flow radially out from the center of the MSB to the outside. Hence, more heat will be flowing out of the face of the storage sleeve closer to the out of the MSB. In this case, modeling the heat flow as through an infinite slab would be more correct. The difference between the two models is shown below for a square heat producing region.*

$$\Delta T = \frac{k_{eff}}{q''' L^2} \left[1/2 \left[1 - (x/L)^2\right] - 2 \Sigma \frac{(-1)^n}{(mL)^3} \frac{(\cosh my)}{(\cosh mL)}\right.$$

evaluated for,

$$x = 0, y = 0$$

where,

$$m = \frac{(2n+1)\pi}{2L}, n = 1, 2, 3, 4, \ldots$$

Which for our case of L = 4.5 in reduces to:

$$\Delta T = \frac{0.295\, q''' L^2}{k_{eff}}$$

for the infinite slab

$$\Delta T = \frac{q''' L^2}{2\, k_{eff}} = \frac{0.50\, q''' L^2}{k_{eff}}$$

For conservatism the equation for the square heat source was used.

However, to be applicable, the differences in material properties (particularly emissivity) between the test cask and the MSB had to be considered. This was done by using the WEC to estimate the relative split between the radiative heat transfer and the convection/conduction and then raising the radiative portion by the difference in the exchange factor $[(1/\epsilon_e) + (1/\epsilon_w) - 1]^{-1}$.

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* V. S. Arpaci, Conduction Heat Transfer, pp 219, 220.

Using this methodology, the following table was developed.

Test	T_w (°F)	ΔT (°F)	k_{eff} (BTU/hr ft °F)	Corrected k_{eff} (using MSB mat. emis.)
Reference 4.5	635°F	36	0.71	1.16
Reference 4.6	184°F	46	0.25	0.50
Reference 4.7	392°F	63	0.33	0.76

A.3.3 MODEL TN-24 TEST

As a third check on the effective heat conductivity of the fuel region, the cask test reported in Reference 4.3 was modeled with the ANSYS model used to calculate the MSB temperatures. A fuel region effective thermal conductivity of 0.4 was used based upon calculations similar to those discussed in Sections A.3.1 and A.3.2. This value gave excellent agreement with the TN-24 test results. A summary is shown below.

Location	Test Value (°F)	ANSYS Model (°F)
T1	380	373
T29	429	429
T31	373	353
T33	402	395
T37	361	343
T39	297	245
MSB Surface T171	193	195

For this TN-24 cask, no correction for emissivities is necessary as a value of 0.8 for both fuel and the basket was assumed.

However, calculations with 0.4 as the k_{eff} for the MSB led to higher temperatures (~700°F) so that the value of 0.4 @ 429°F was modified by the estimated increase in the radiative portion of the heat transfer (as determined by the WEC) if the fuel temperatures increase to approximately 700°F (the calculated values for the MSB). The factor from the WEC calculation

$$\frac{0.46}{0.27} = 1.7$$

which would yield an effective k_{eff} of 0.68 for the MSB central assembly if its wall temperature were around 700°F. Calculations using a value of 0.6 were then performed for the MSB. These resulted in approximately a 30°F reduction in the MSB fuel temperatures. This calculation also shows the sensitivity of the ANSYS model to the effective thermal conductives – as k_{eff} goes from 0.4 to 0.6, fuel temperatures increase by approximately 30°F.

A.4 CONCLUSIONS

Using the three methods described above, the following k_{eff}s for the MSB were determined as a function of storage sleeve temperature.

T_{wall} (°F)	METHOD WEC	Test Results	ANSYS Modeling of TN-24 Test
100	0.17		
184		0.50	
200	0.20		
300	0.23		
392		0.76	
400	0.27		
429			0.40
500	0.32		
582	0.38		
635		1.16	
690	0.46		
700			0.68

Based on this table the following conclusions were drawn.

1. The WEC appears to be overly conservative when compared to the cask test data.
2. k_{eff}s from the test data appear to run from 0.4 to 1.2 (corrected for MSB emissivities). Therefore, one might expect the outer [cooler (300 - 500°F)] fuel assemblies to have k_{eff}s in the range of 0.4 - 0.6 and the inner [hotter (500 - 700°F)] assemblies to have k_{eff}s in the range of 0.6 to 1.0.
3. To be conservative a value of 0.6 was used throughout the VSC thermal analyses. This will overestimate the clad temperatures in the central regions and may or may not underestimate the temperatures in the outer fuel regions.

In any regard, a sensitivity study was made and the peak cladding temperature was:

k_{eff} = 0.4 T_{clad} (max) = 750°F (398°C)

k_{eff} = 0.6 T_{clad} (max) = 719°F (382°C)

Hence, the overall difference is not significant.

APPENDIX B

OPTIONAL CASK TRANSPORTER AND VSC LIFTING LUGS

B APPENDIX B - OPTIONAL CASK TRANSPORTER AND VSC LIFTING LUGS

B.1 INTRODUCTION

As an alternative to the trailer and skid movement of the VSC, some users may choose an optional cask transporter to move the loaded VSC to the storage pad. The transporter is designed to lift the VSC 6 to 18 inches off the ground so that the transporter and cask can be moved to the storage location. The transporter may be self-propelled or towed by a truck or other suitable vehicle. Because the VSC is analyzed for drops of up to 60 inches, or a tip-over, the VSC is lifted less than 60 inches, and the transporter is not important to safety, the cask may be lifted by a lift fixture embedded in the concrete or positioned in the skid channels under the cask. Because the lift height of the cask is limited, the attachments are not important to safety. However, their design and analysis are described in the sections below to demonstrate that commercially available equipment can be used.

B.2 TRANSPORTER

The transporter is a vehicle capable of lifting and transporting the loaded VSC. The cask transporter assembly is comprised of a lower deck and an upper frame. The lower deck is mounted on the suspension and houses the steering mechanism/towing bar assembly, hydraulic power supply, and propulsion system. The lower deck is U-shaped to allow the cask transporter to straddle various sizes of casks during lifting, transporting, and lowering operations. A cask restraint system is mounted on the lower deck to attach to the cask and prevent unwanted movement of the cask during transport. The lift beam support structure is mounted on the lower deck and contains hydraulic cylinders or other means (hoists, etc.) to lift the cask.

B.3 VSC LIFTING LUGS

The VSC lifting lugs are not important to safety, due to the eighteen-inch maximum lift height. However, the design of the lifting lugs is described below.

The lifting of the VSC is accomplished via transporter lifting arms, two lifting lug assemblies embedded on top of the VSC body, and two pins inserted through the lifting arms and lifting lugs. The embedded devices are capable of safely handling the fully loaded VSC. A weight of 302,000 lbs. is conservatively assumed. The analysis of the embedded assemblies is discussed below.

The lifting devices for the VSC were analyzed in accordance with ANSI N14.6 and ACI 349. The allowable stress for the load-bearing members is the lesser of $S_y/3$ or $S_u/5$. The lowest service temperature for the VSC lifting components is 0°F. The VSC lifting lugs arrangement is shown in Figure B.3-1.

Pin

The lifting pin is a steel bar, 9 inches long and 4 inches diameter, made of 4340 steel (Sy = 80 ksi, Su = 120 ksi). The maximum shear stress on this pin is 8.0 ksi; the bearing stress is 15.1 ksi; and the maximum principal stress is 18.6 ksi. This provides a design factor of 4.3 on yield strength and 6.4 on tensile strength. These factors exceed the ANSI N14.6 design factors of 3 on yield and 5 on ultimate.

Lifting Arm

Lifting arms are steel plates 12 inches wide and 2-1/2 inches thick, with a 4.125-inch hole (see Figure B.3-2). They are constructed of A537 steel (S_y = 50 ksi, S_u = 70 ksi). The design load is 151 kips. The tension membrane stress is 7.7 ksi. The design factor is 6.5 on yield and 9.1 on tensile strength. These factors are significantly larger than the ANSI N14.6 requirements of 3 on yield and 5 on tensile strength.

The shear stress in the lifting arm is 7.7 ksi, which provides a design factor of 6.5 on yield and 9.1 on tensile strength (principal stress is equal to the shear stress). In accordance with ANSI N14.6, the design stress factors shall not be applied to the high local stresses that can be relieved by slight yielding of the material. From Reference 5.4, the stress concentration factor is 5.55, and the highest stress is 42.7 ksi. This is less than the material yield strength of 50 ksi.

Lifting Lugs

The lifting lug assemblies consist of two, 14-inch-square by 2-inch-thick plates, each with a 4.125-inch hole for the pin. The lifting lug assemblies are constructed of A537 steel. The design load per lifting lug assembly vertical plate is 75.5 kips. The lifting lug vertical plate tensile membrane stress is 3.8 ksi. The lifting lug assembly vertical plate shear stress is 4.8 ksi. The design shear stresses are well within the allowable stress limits. The stress concentration factor is 5.55, and the maximum stress is 21.1 ksi, which is less than the 50 ksi yield strength for A537 steel.

Rebar

The rebars are #11 A706 steel (S_y = 60 ksi, S_u = 80 ksi) weldable rebar. The attachment plate is A537 (S_y = 50 ksi, S_u = 70 ksi). The design stress in the rebar is 12.1 ksi. Thus, the rebar design factor is 4.96 times yield and 6.6 times the tensile strength of A706 steel, which meets the ANSI N14.6 requirements of 3 on yield and 5 on tensile strength. The development length required by ACI 349, Section 12.2 is the greater of 34 inches and 59 inches. The requirement of ACI 349 is satisfied by the 70-inch embedment.

Weld

The weld joints between the rebar and the lifting lugs are complete penetration and can be evaluated based on the allowable stress for the weaker material. Because A706 (rebar material) is stronger than A537 (lug material), the evaluation of the weld is based on the strength of A537.

The design stress in the weld is 12.1 ksi, which meets the ANSI N14.6 requirements of 3 on yield and 5 on tensile strength.

Testing

The cask lifting lugs are load tested to 1.25 times their rated capacity.

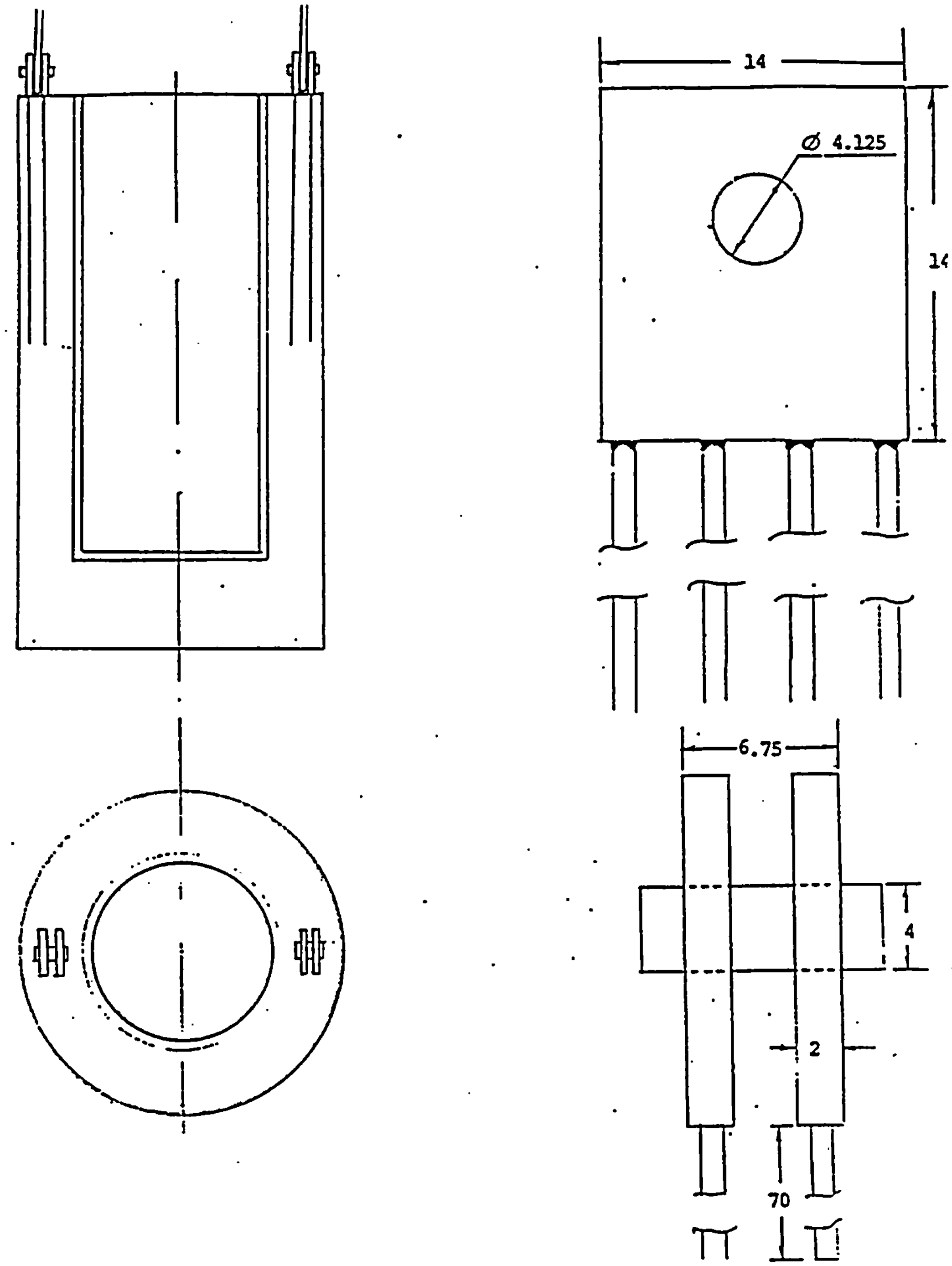

Figure B.3-1 - VSC Lifting Arrangement

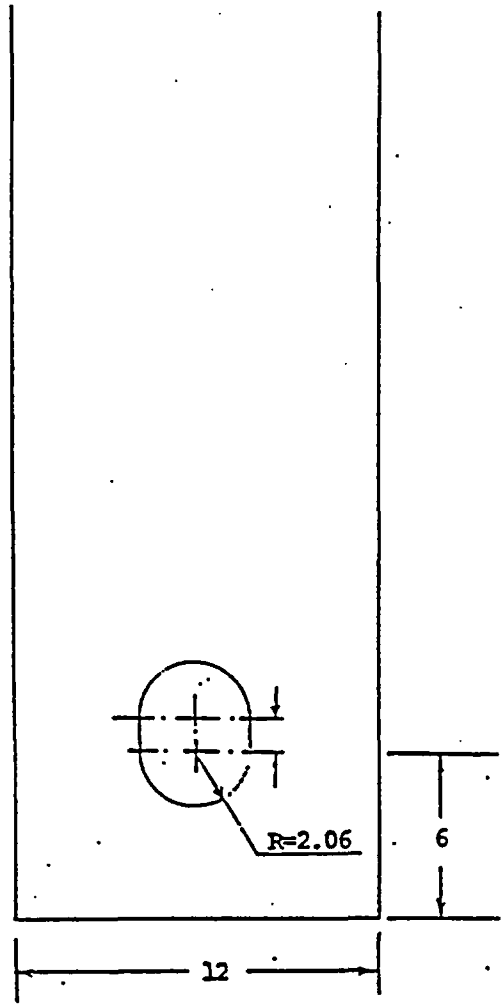

Figure B.3-2 - VSC Lifting Arm

APPENDIX C

FUEL INERT DRY STORAGE TEMPERATURE LIMITS

C APPENDIX C - FUEL INERT DRY STORAGE TEMPERATURE LIMITS

C.1 INTRODUCTION

The principal potential breach mechanisms for zircaloy clad irradiated fuel during inert gas dry storage have been identified as creep rupture, stress corrosion cracking, and delayed hydride cracking (Reference 4.1). However, cladding breach due to stress corrosion and delayed hydride cracking is not expected because the threshold stress intensity levels for these mechanisms are greater than those expected for spent fuel. Thus, prevention of creep rupture (by limiting the maximum initial dry storage temperature) is the primary means of preventing cladding breach during dry storage.

The maximum allowable initial dry storage temperature is a complex function of fuel design, burnup level, fuel age and the geometry and makeup of the dry storage cask. In order to account for these variations, the graphical use of generic temperature limit curves described and developed in Reference 4.1 has been adopted. This methodology defines a specific temperature limit, below which the probability of cladding breach due to creep rupture is less than 0.5% per spent fuel rod for a 40 year storage period.

C.2 ANALYSIS

The assemblies considered are as follows:

B&W Mark C (17 x 17)
B&W Mark B-4 (15 x 15)
CE 15 x 15 (Palisades)
Westinghouse PWR (17 x 17)
Westinghouse PWR (15 x15)
Westinghouse PWR (14 x 14)

From Reference 4.1,

$$\sigma_{mhoop} = [(p)(D_{mid})]/2t$$

where,

σ_{mhoop} = cladding hoop stress

p = internal gas pressure of rod (fission gas and fill)

D_{mid} = clad midwall diameter

t = clad thickness

This relationship between stress and temperature (actually gas pressure which is related via the perfect gas law) is plotted for the various fuel assemblies in Figure C.2-1. Also, the generic temperature limit curves for 5, 6, 7, 10 and 15 year cooling period from Reference 4.1 are plotted on the same axis. The intersection of the stress temperature relationship line for a given fuel assembly with the 5, 6, 7, 10 and 15 year limit curves then defines the maximum allowable initial dry storage temperature for fuel of this type.

C.3 RESULTS AND CONCLUSIONS

The results of this study are illustrated in Figure C.2-1. As seen from Figure 2.1.1 applied to Figure C.2-1, the 5 year cooling limit is the most restrictive, as clad temperatures after 6, 7, 10 and 15 years are well below their respective limit curves. Hence, a temperature limit of 712°F after 5 years cooling is adequate to ensure a less than 0.5% per rod probability of stress induced clad failure over a storage period of 40 years. However, as noted by the fairly wide range of limits obtained from Figure C.2-1, this limit may be quite restrictive for certain fuel types.

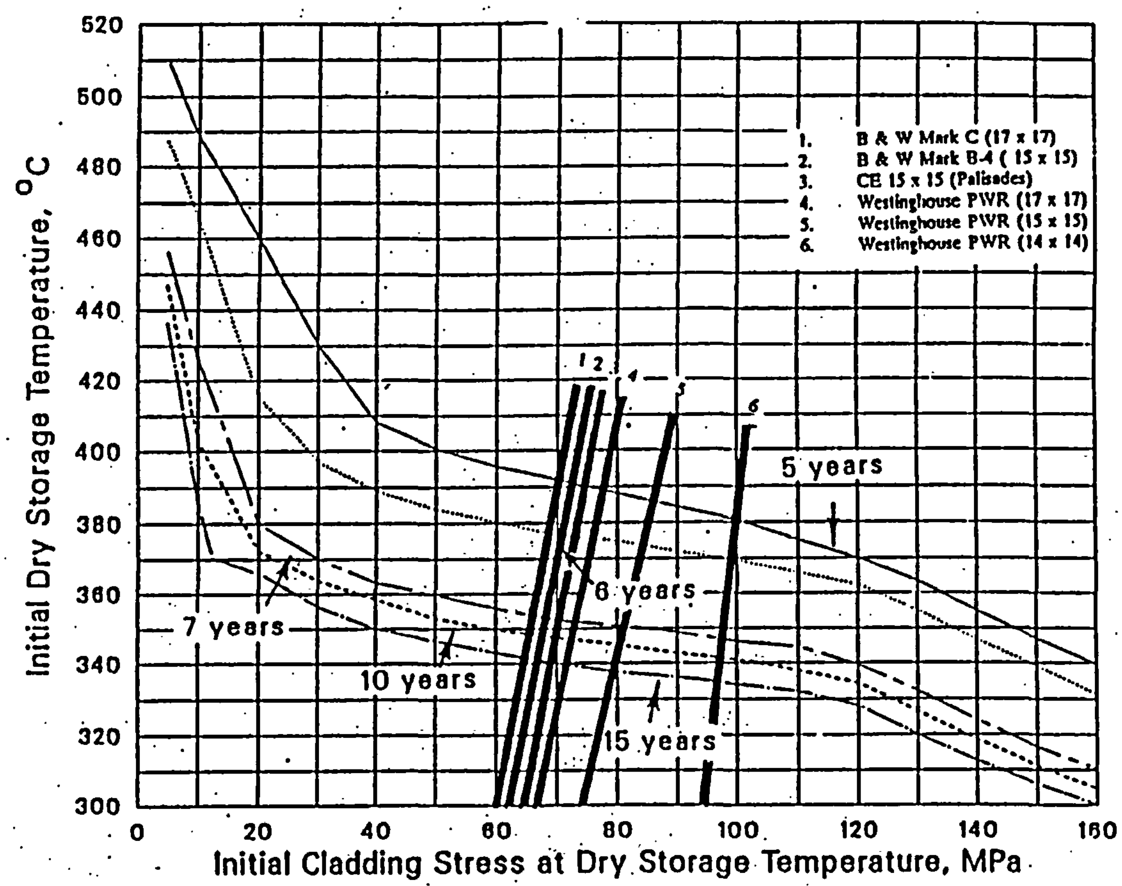

Figure C.2-1 - Comparison of IDS Cladding Temperature Limit Curves For Spent Fuel Of Varying Ages

APPENDIX D

EFFECTIVE THERMAL CONDUCTIVITIES
FOR WIDE AND NARROW AREAS WITHIN THE MSB

D APPENDIX D - EFFECTIVE THERMAL CONDUCTIVITIES FOR WIDE AND NARROW AREAS WITHIN THE MSB

D.1 INTRODUCTION

Heat transfer through the wide and narrow areas formed by the MSB inner wall and the sleeve assemblies occurs by a complex interplay of radiation, convection and conduction through the helium backfill gas. In the radial sector ANSYS model used to analyze the MSB internals, radiation heat transfer from the sleeves to the MSB is directly addressed through the use of radiation link elements extending from nodes on the basket face to nodes on the MSB wall. These elements require readily available input (i.e., emissivities, areas, view factors). Conduction through the gas is modelled using solid elements which require as input material properties.

The direct treatment of convection heat transfer in the areas is not possible in the radial sector model, however, due to the complex axial flow patterns of the gas and the corresponding variation in heat transfer coefficients. Therefore, an alternative method of treating convection heat transfer in these areas as a conduction through the gas with an effective thermal conductivity (k_e) was developed. This value of k_e was determined by examination of data obtained from various tests performed with similar casks.

D.2 ANALYSIS

The form of the equation sought for k_e is as follows:

$$q_r = k_e \cdot A \cdot dT/dr = h \cdot A_s \cdot (T_s - T_b)$$

where,

A	=	conduction area
dT/dr	=	temperature gradient across gap
h	=	convection coefficient
A_s	=	convection surface area
T_s	=	convection surface temperature
T_b	=	gas bulk temperature
q_r	=	radial heat flow through gap

By examining thermocouple data presented in references the required temperatures to solve the above equation for k_e in terms of the remaining known quantities may be obtained. However, in order for the above equation to apply, the heat flow through the gap must first be divided into that arising from conduction and convection and that arising from radiation. The required adjustment is as follows:

$$q_r'' = k_e \cdot dT/dr + e\sigma\,(T_s^4 - T_w^4)$$

where,

q_r'' = radial heat flux at basket surface

e = basket surface emissivity

σ = Stefan-Boltzman Constant

T_w = MSB wall temperature

from which the expression for k_e can be derived as:

$$k_e = [q_r'' - e\sigma\,(T_s^4 - T_w^4)] \times \delta r/(T_s - T_w)$$

where,

δr = average radial gap between basket and wall (which is essentially the hydraulic diameter for the geometrics question).

D.3 RESULTS AND CONCLUSIONS

Results from two casks which have very similar geometries to the VSC-24 were used (4.3, 4.6). The data used is shown below:

Reference	q_r (BTU/hr-ft²)	e	T_s(°F)	T_w(°F)	δr(ft)	k_e (BTU/hr-ft-°F)
4.3	332.7	0.8 wide	242	198	0.16	2.8
		narrow	301	199	0.16	0.24
4.6	192	0.2 wide	210	167	0.87	3.6
		narrow	230	167	0.17	0.46

Results from the cask test reported in Reference 4.3 yielded the most conservative k_e = 2.8 BTU/hr-ft-°F for the wide areas and k_e = 0.24 BTU/hr-ft-°F for the narrow areas. The data for k_e was used in the ANSYS MSB radial model. The wide area dT (dT=Ts-Tw) was found to be accurately and conservatively calculated. However, the narrow area dT was not conservatively calculated, using the determined k_e = 0.24 BTU/hr-ft-°F. Therefore, for narrow areas, the lowest possible value of k_e (k of helium gas = 0.11 BTU/hr-ft-°F) was used. With this value the narrow area dT was calculated to within 10% of the test data. This was considered accurate enough since the wide area k_e gave a Δk 10% higher than the test data. Therefore, the following values were used to model the wide and narrow areas in the VSC-24 MSB ANSYS model:

Wide areas: k_e = 2.8 BTU/hr-ft-°F

Narrow areas: k_e = k helium = 0.11 BTU/hr-ft-°F

These values conservatively predict observed temperature gradients in the test performed with similar casks and the results of their use are summarized in Table D.1-1. Table D.1-1 also shows some of the results of the parametric variation analysis that were performed. In this analysis the wide area k_e was varied from 4.6 to 0.58 with the narrow area held constant at 0.11. As this table shows the ΔT across the wide area only increased by 32°F as the k_e falls from 2.8 to 0.58. Likewise, the narrow area ΔT only increases by 14°F. The overall impact on the fuel temperature was a 22°F rise which is insignificant compared to the roughly 400°F ΔT from the MSB shell to the hottest fuel rod. Hence, it was concluded that the fuel temperature depends much more on conduction through the steel basket and radiation than on the conduction or convection through the helium.

Table D.1-1 - Summary of Wide and Narrow Area Thermal Analysis

	k_e BTU/hr-ft-°F		ΔT°F	
	Wide	Narrow	Wide	Narrow
Test Results	--	--	43	102
ANSYS Run 1	2.8	0.24	46	81
ANSYS Run 2	2.8	0.11	48	90
ANSYS Run 3	0.58	0.11	80	104

APPENDIX E

NOT USED

APPENDIX F

SPECIFICATION VMSB-98-001

APPENDIX F

SPECIFICATION VMSB-98-001

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TITLE: GUIDELINE REQUIREMENTS FOR THE TIME-OF-FLIGHT DIFFRACTION ULTRASONIC EXAMINATION OF THE VSC-24 STRUCTURAL LID TO SHELL WELD

SAFETY CLASSIFICATION: IMPORTANT TO SAFETY

Reviewed and Approved By: VQ Bowers for 8/13/98
J. Westvold, VP Quality Assurance/Quality Control Date

Reviewed and Approved By: K. E. Moeckel 8/13/98
K. E. Moeckel, Engineering Manager Date

RECORD OF REVISIONS

ISSUE DATE	REVISION NUMBER	CHANGE PAGE/PARAGRAPH	DESCRIPTION
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06/98	2	3, 5, 7, 8, 9, 11, 12, 15, 16, 17, 20	Incorporate DCR/N No. VSC-MSB98-01
06/98	3	17	Incorporate DCR/N No. VSC-MSB98-02
06/98	4	3 - 6, 7, 8, 10-14, 16, 18, 21	Incorporate DCR/N No. VSC-MSB98-03
08/98	5	12-13	Incorporate DCR/N No. VSC-MSB98-04

TITLE: GUIDELINE REQUIREMENTS FOR THE TIME-OF-FLIGHT DIFFRACTION ULTRASONIC EXAMINATION OF THE VSC-24 STRUCTURAL LID TO SHELL WELD

TABLE OF CONTENTS

1.0 Purpose/Scope

1.1 The purpose of this document is to provide technical guidelines for conducting the Time-of-Flight Diffraction (TOFD) ultrasonic testing (UT) examination of the VSC-24 Multi-assembly Sealed Basket (MSB) structural lid to shell weld either in the transfer cask (MTC) or concrete cask (VCC).

1.2 The scope of this guideline includes the establishment of flaw acceptance criteria, examination parameters, examination process and technique development, and qualification of examination procedures and examination personnel.

1.3 The guideline provides the technical basis for specific operating procedures that would be used to address any required actions resulting from the ultrasonic testing examination of the VSC-24.

2.0 References

2.1 ASME Section XI, 1989 Edition, IWB-3600

2.2 ASME Section XI, 1989 Edition, IWA-3300

2.3 Certificate of Compliance, Title 10 Code of Federal Regulations Part 72, Number: 1007

2.4 Safety Analysis Report for the Ventilated Storage Cask System

2.5 ASNT Recommended Practice No. SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing", 1984 Edition

2.6 Structural Integrity Associates Analysis, "Allowable Flaw Size Definition for VSC-24 Dry Storage Cask Structural Lid to Shell Weld: File No. CPC-06Q-301"

2.7 Flaw Tech drawing 7C037R5, "Flawed Specimen-Palisades DFS Mock-Up"

2.8 Flaw Tech drawing 7C037AR4, "Flawed Specimen-Flaw Locations"

2.9 SNC WEP-109.002.2, MSB-24 Load Combination Evaluation

3.0 Definitions

3.1 Time-of-Flight Diffraction (TOFD) - A method of performing an ultrasonic examination on components which floods the examination volume with sound used for detection and sizing of indications in the examined component. The technique uses changes due to diffraction of the input sound energy across the indication for both detection and sizing.

3.2 Flaw Depth - The flaw dimension normal to the surface (inside or outside) of the component.

3.3 Flaw Length - The flaw dimension parallel to the surface (inside or outside) of the componenent.

4.0 Quality Assurance Requirements

All work performed in compliance with this guideline document shall be performed in accordance with Quality Assurance Programs that meet the applicable quality assurance requirements of 10CFR50, Appendix B.

5.0 Responsibilities

The responsibilities of the VSC-24 Certificate Holder and VSC-24 Owner or Owners Group are as follows:

- VSC-24 Certificate Holder

  The VSC-24 Certificate Holder is responsible for establishing acceptance criteria including dispositioning of flaws and developing, distributing and revising this guideline document and examination procedure(s).

- VSC-24 Owner or Owners Group

  The VSC-24 Owner or Owners Group is responsible for developing site specific evaluations and procedures and qualifying examination processes, techniques, procedures and personnel.

6.0 Screening Criteria

Screening criteria for flaw indications shall be established based on analysis and used to disposition flaw indications. The "Screening Criteria for Use During Ultrasonic Examination of VSC-24 Structural Lid to Shell Welds" is contained in Attachment 4.

6.1 Basis for Screening Criteria

Screening criteria and supporting analysis for Arkansas Nuclear One, Palisades and Point Beach Nuclear Plants are contained in Structural Integrity analysis "Allowable Flaw Size Definition for VSC-24 Dry Storage Cask Structural Lid to Shell Weld: File No. CPC-06Q-301", Reference 2.6.

The screening criteria provide the basis for initial evaluation and disposition of flaw indications detected during volumetric examination. These criteria are based on the criteria of ASME Section XI, 1989 Edition, with a conservative set of assumptions as follows:

- All factors of safety on applied stress required by ASME Section XI were included.
- The analyses are all based on linear elastic fracture mechanics (LEFM), assuming that the failure mode is brittle fracture. Actual material toughness tests results show that all materials are highly resistant to such failures.
- Weld residual stresses were treated as constant tensile stresses normal to the limiting (circumferential) flaw direction. The magnitude of these tensile stresses was taken to be at the minimum specified yield stress of the base material.
- The Screening criteria were derived using the lower bound of material toughness determined by test for representative materials.
- Welding processes used by each plant to fabricate specimens for Charpy V-notch and toughness testing were intentionally performed near the high end of the heat input reported during original welding procedure qualification, in order to determine the lower limit on material toughness.
- The fracture mechanics analyses performed to determine acceptance criteria used analytical models of flaws in flat plates. The actual weld configuration provides considerably more restraint to hypothetical flaw locations. The increased restraint would produce larger allowable flaw sizes.

If further evaluation of flaws is required, it may include either LEFM or Elastic Plastic Fracture Mechanics (EPFM) analyses. This latter method is applicable for high toughness materials such as are indicated by the actual material test results. The key parameter is the fracture toughness represented by the critical J integral, J_{IC}. The assumed mechanism is ductile crack extension.

6.1.1 Code Requirements

The methods of ASME Section XI, IWB-3600, Reference 2.1, are used to determine screening criteria and to further evaluate flaw indications.

6.1.2 Load Limits

The limiting loading conditions on the MSB are identified in the VSC-24 SAR, Reference 2.4, as supplemented in Sierra Nuclear Corporation calculation WEP-109.002.2, MSB Load Combination Evaluation (Reference 2.9).

6.1.3 Operational Limits

The MSB operational temperature limits, which determine the appropriate material properties, are identified in the VSC-24 C of C, Reference 2.3.

6.1.4 Material Properties

a. Requirement

Material specimens representative of the structural lid and shell materials and actual weld processes used during loading shall be used to develop screening criteria. Charpy V-notch impact tests and material toughness tests shall be performed on these specimens.

Charpy V-notch impact and toughness tests shall be performed at 0°F.

The material toughness properties shall be established by supplemental tests performed in accordance with ASTM E-1737-96.

b. Testing Performed

Charpy V-notch impact tests at 0°F and material toughness tests in accordance with ASTM E-1737-96 were performed. The results of these tests are summarized in Structural Integrity Analysis, Reference 2.6.

c. Future Procurement

Specifications for future procurement of pressure retaining materials including weld metal, shall specify a minimum Charpy V-notch impact absorbed energy value of 45 ft-lbs at 0°F in addition to the current requirement of 15 ft-lbs minimum at -50°F. Future materials which satisfy these minimum values will not invalidate the acceptance criteria in the Structural Integrity Analysis, Reference 2.6.

In addition, low-sulfur, calcium-treated, vacuum-degassed steel, such as produced by the Lukens Fineline® process, shall be specified in future orders for the VSC-24 MSB pressure boundary material. Welding consumables with low hydrogen levels (less than $10ml/H_2/STP/100g$) shall be utilized for all lid welds.

6.2 Flaw Disposition

Flaw indications identified during examination are required to be dispositioned as outlined in the Flaw Disposition Flow Chart, Attachment 1, and as described below:

6.2.1 Characterization

Flaws detected by examination shall be described in terms of location, length, depth, orientation (e.g., circumferential, transverse, laminar etc.), and type, to the extent possible (e.g., planer and/or volumetric).

6.2.2 Flaw Proximity

Adjacent flaws shall be evaluated using the flaw proximity criteria of ASME Section XI, IWA-3300.

6.2.3 Screening Criteria

Each flaw shall be compared with the screening criteria in Attachment 4 and the flaw proximity requirements of 6.2.2. Flaws which satisfy this criteria are acceptable and require no further action. Flaws which do not satisfy the screening criteria may be determined to be acceptable by further evaluation or alternatively, may be repaired.

6.2.4 Flaw Evaluation Methods

Flaws which do not meet 6.2.3 may be shown to be acceptable for continued operation using fracture mechanics techniques (Linear Elastic Fracture Mechanics or Elastic-Plastic Fracture Mechanics) as appropriate, as described in 6.1 and Structural Integrity Analysis, Reference 2.6.

6.2.5 Repairs

Flaws which are not acceptable based on the requirements described in 6.2.3 or 6.2.4 above shall be repaired. Repair shall be accomplished by removing the flaw or reducing the flaw to an acceptable level as described in 6.2.3 and 6.2.4 above.

7.0 Examination Parameters

Examination parameters are bounded by the requirements and specifications of the VSC-24 SAR and C of C, including source terms and heat load in the MSB, location of the MSB in the MTC or VCC and plant specific operational constraints.

Examination of the MSB structural lid to shell weld occurs following loading of the MSB with spent fuel assemblies and therefore the concerns of ALARA shall be addressed in the qualification of examination procedures and operational planning.

Changes to the licensed configuration or operation of the VSC-24 system to accommodate the examination require the appropriate evaluations such as, but not limited to, 72.48 and plant ALARA. Consideration shall be given as a minimum to the areas described in 7.1 and 7.2.

7.1 MSB in MTC

Examination of the structural lid to shell weld with the MSB in the MTC shall consider the following:

7.1.1 Configuration Limitations

Access to the structural lid to shell weld on the MSB top is not constrained. Access to the shell side of the weld is limited to a nominal 0.5 inch gap. However, during loading activities, shims are installed in this gap to limit radiation streaming. Operations without the shims in place require further site specific evaluation.

7.1.2 Radiation/Shielding

The MTC, in addition to lifting the MSB, provides shielding of the loaded spent fuel assemblies.

The design basis calculated radiation at the structural lid, the gap between the MSB shell and MTC and at the side MTC is provided in the VSC-24 SAR, Reference 2.4. Actual dose measurements during previous loading operations generally support the design basis calculations. However, variations in the source terms, such as "old" fuel assembly end fittings, can have a significant effect on measured radiation.

A dose evaluation shall be performed to assure that the examination procedure is in compliance with the Owner's plant ALARA requirements.

7.1.3 Component Temperature

The MSB structural lid and shell temperature will increase with time due to spent fuel decay heat. The maximum calculated design basis equilibrium fuel temperature of the MSB (with He) in the MTC is provided in the VSC-24 SAR, Reference 2.4.

A thermal evaluation will be required to assure that the examination procedure can accommodate the planned MSB heat loads. In any case, the UT examination shall be performed with a metal temperature of 200°F or less.

If the thermal analysis indicates that the planned heat loads will exceed the examination process limitations, then provisions to cool the MSB components will need to be implemented.

7.1.4 Operational Limitations

The operational limits of the system, such as the "minimum temperature for moving the MSB or lifting the MTC" and the "handling height" are defined in the VSC-24 C of C, Reference 2.3.

7.2 MSB in VCC

Examination of the structural lid to shell weld with the MSB in the VCC shall consider the following:

7.2.1 Configurational Limitations

Access to the structural lid to shell weld on the MSB top requires lifting the VCC shield ring. Operations conducted with the shield ring elevated or removed will require evaluation. Although the annular gap between the MSB and VCC liner is nominally 4.0 inches, access to the MSB shell side is restricted by the VCC shield lid support ring welded to the liner.

7.2.2 Radiation/Shielding

The VCC, in addition to providing cooling of the MSB, provides shielding of the loaded spent fuel assemblies.

The design basis calculated radiation at the structural lid and at the side of the VCC is provided in the VSC-24 SAR, Reference 2.4. Actual dose measurements during previous loading operations generally support the design basis calculation. However, variations in the source terms, such as "old" fuel assembly end fittings, can have a significant effect on measured radiation.

A dose evaluation shall be performed to assure that the examination procedure is in compliance with the Owner's plant ALARA requirements.

7.2.3 Temperature

The maximum calculated design basis equilibrium temperature of the MSB in the VCC is provided in the VSC-24 C of C, Reference 2.3. The SAR indicates that the lid temperatures are limited to less than 160°F for the design basis heat load and therefore will not impact the examination procedure.

7.2.4 Operational Limitations

The operational limits of the system, such as the "minimum temperature for moving the MSB" and the "handling height" of the VCC are defined in the VSC-24 C of C, Reference 2.3. The rigging used to facilitate examination shall have the same safety factors as applied to the rigging used for handling of components such as the VCC weather cover and shield ring during loading.

7.2.5 Weather Cover Removal

The weather cover will only be removed for a short period of time to facilitate the UT examination in the VCC and on the condition of no impending threat of severe weather.

Potential accident conditions during removal of cover while performing UT examination in the VCC shall be addressed on a site-specific basis.

8.0 Development of Examination Processes and Techniques

Examination processes and techniques capable of conducting a volumetric ultrasonic testing (UT) examination of the structural lid to shell weld shall be developed.

8.1 Examination Process and Technique Requirements

The processes and techniques developed for the examination of the MSB structural lid to shell weld shall meet the following requirements:

8.1.1 Circumferentially Oriented Flaws

1. Detection

The examination shall detect flaws with depth greater than 0.100".

2. Length Sizing Acceptance Criteria

The examination shall length size the detected flaws as follows:

Flaw lengths shall be within 0.75 RMS.

3. Depth Sizing Acceptance Criteria

The examination shall depth size the detected flaws as follows:

The mean error in the flaw depth will be calculated and documented. If the mean error is less than or equal to a positive 0.072 inches, the RMS error must be less than or equal to 0.125 inches. If the mean error is greater than a positive 0.072 inches, this contribution to the RMS error will be removed and recorded. The remaining error must be less than or equal to 0.102 inches (reference Attachment 3: White paper on Depth Sizing Acceptance Criteria for Time-of-Flight Diffraction Ultrasonic Examination of the VSC-24 Structural Lid to Shell Weld).

8.1.2 Transverse (Axially) Oriented Flaws

1. Detection

The examination shall detect flaws with depth greater than 0.100".

2. Length and Depth Sizing Acceptance Criteria

Transverse flaws do not affect the structural adequacy of the structural lid to shell weld. The minimum examination requirement is to demonstrate that an inside surface connected transverse flaw does not extend into the upper 25% of the weld ligament, so that the pressure integrity and leak tightness of the weld is not impaired. For this screening examination, detailed length sizing for transverse flaws is not required, and such flaws may be assumed to extend completely across the weld in the transverse direction.

3. Any undersizing depth error identified during the personnel qualification activities will be added to the reported flaw depth for transverse flaws.

8.1.3 Permanent Records

The examination shall produce a permanent record.

8.1.4 Operational Requirements

The examination shall meet the requirements described above under the conditions outlined in Section 7.0 - Examination Parameters.

8.2 Examination Equipment

The equipment to be used for examination of the MSB structural lid to shell weld shall meet the following requirements:

8.2.1 Automated and/or Semi-Automated Equipment

Automated and semi-automated equipment shall provide a complete set of data necessary for detecting, locating and sizing flaws and a permanent record of the examination in the shortest time to minimize radiation exposure and meet ALARA requirements. Semi-automated equipment is a manually positioned scanner that utilizes the same transducers, data acquisition and recording system as the automated equipment.

8.2.2 Calibration Blocks

Calibration blocks of acoustically equivalent material and reflectors suitable for establishing reference gain settings and repeatability shall be used for the calibration of examination equipment. The equipment shall be calibrated prior to each examination/setup using appropriate calibration block(s) that is within ±25°F of the actual temperature of the component area to be examined. Calibration blocks shall satisfy site specific QA program requirements.

9.0 Examination Procedure Qualification and Approval

Examination procedures shall be developed, qualified and demonstrated using appropriate mockup(s).

9.1 MSB Mockup

An unsecured mockup of the MSB shall conform to the following requirements:

9.1.1 *Configuration*

The mockup components and structural lid to shell weld shall be constructed to the configuration of the MSBs to be examined.

9.1.2 Materials

The mockup shall be constructed of the same material type(s) as the MSBs to be examined.

9.1.3 Implanted Flaws

Flaws consisting of welding process discontinuities and cracks of varying sizes (both length and depth) shall be implanted in the weld and weld HAZ of the mockup at representative locations.

The flawed specimen layout and details for the mockup used in the development of examination processes and techniques are described in References 2.7 and 2.8.

9.1.4 Mockup Temperature

The mockup temperature during examination procedure qualification shall bound the expected temperature of the MSBs to be examined.

9.2 MTC Mockup

The mockup shall represent the configuration of the actual MTC used during loading.

9.3 VCC Mockup

The mockup shall represent the configuration of the actual VCC used during loading.

9.4 Examination Procedure Requirements

The examination procedure shall address the essential variables described in Attachment 2.

9.5 Examination Procedure Qualification Criteria

The procedure is considered to be qualified if implanted flaws are detected, length and depth sized in accordance with 8.1.1 and 8.1.2.

9.6 Examination Equipment Demonstration

The examination equipment shall be demonstrated in the configuration that the actual examinations will be performed using an appropriate mockup combination as follows:

- The MSB mockup shall be installed in the MTC mockup to assure that the examination equipment can perform satisfactorily under the operational constraints of this configuration, including shielding.
- The MSB mockup shall be installed in the VCC mockup to assure that the examination equipment can perform satisfactorily under the operational constraints of this configuration, including shielding.

9.7 Examination Procedure Approval/Modification/Revision

The VSC-24 Owners shall review and approve the examination procedure.

A modification to an approved procedure that constitutes a change to an essential variable described on Attachment 2, requires requalification. A qualified examination procedure may be modified without requalification provided the modification does not change an essential variable and compliance with the requirements is maintained. Editorial, clarification and format changes are examples of procedure modifications which may be made without having to requalify an approved procedure. Examination procedure requalification, when required, shall be in accordance with the requirements contained in this guideline document.

Modifications to an approved procedure which affect the essential variables shall be concurred with by all the VSC-24 Owners. Modifications that do not affect the essential variables may be controlled on a site specific basis.

10.0 Examination Personnel Qualification

10.1 Experience

Personnel performing the examination shall be qualified and certified as ultrasonic testing (UT) Level II (minimum) to a program that meets the requirements of SNT-TC-1A, Reference 2.5.

10.2 Performance Qualification Demonstration

10.2.1 General

10.2.1.1 Personnel performing data acquisition and data analysis shall demonstrate proficiency by participation in a performance demonstration developed and administered by the EPRI NDE Center.

10.2.1.2 Test data used in the personnel analysis demonstrations may be from previous demonstrations, an ongoing data acquisition demonstration or data that has been collected by the EPRI NDE Center.

10.2.1.3 Personnel demonstrations for data acquisition and data analysis will be performed with no prior knowledge of flaw type, size or location (e.g., blind test).

10.2.1.4 The VSC-24 Owners group will have final responsibility in the acceptance of the demonstration protocol developed by the EPRI NDE Center.

10.2.2 Training

Evidence of personnel training (specific to the inspection of the VSC-24 using the TOFD ultrasonic technique) shall be documented prior to any personnel data analysis demonstration. The training shall a minimum of 40 hours of which 8 hours will be specific to the VSC-24. The training may take place in the classroom and/or on the job.

10.2.3 Test Set Selection

The performance demonstration protocol for the test set developed by the EPRI NDE Center shall address the following: the number of flaws, the size of the flaws, flaw orientation, flaw characteristic/type, unflawed areas within the demonstration mock up and the number of the same flaws allowed from a previous test for a given person. As a minimum the test set shall contain ten flaws equally distributed throughout the examination volume.

10.2.4 Security and Demonstration Surveillance

The grading criteria and answer keys shall remain secure from any person who is in or will potentially be in the demonstration process for the analysis of TOFD ultrasonic VSC-24 data.

10.2.5 Grading

The specific grading criteria will be established within the performance demonstration protocol developed by the EPRI NDE Center. As a minimum the person demonstrating their capability to acquire data shall detect 80% of the flaws within the test set. For the candidate to receive credit for detecting a flaw they must locate at least 50% of the flaw length accurately (i.e. the reported length shall be at least 50% coincident with the actual flaw length) Persons demonstrating their capability to analyze data shall detect 80% of the flaws within the test set and the detected flaws shall be sized with the tolerances described in 8.1.1 and 8.1.2.

10.2.6 Retesting of Personnel

Persons who have failed to meet the requirements of the grading criteria may be allowed to test one additional time. Person's who have failed two tests shall receive additional training specific to the inspection of the VSC-24 using the TOFD ultrasonic technique prior to taking a third test. Areas where the person has demonstrated deficiencies shall be addressed in the additional training.

10.2.7 Documentation and Record Retention

All records produced during the demonstration process shall be retained and distributed by the EPRI NDE Center.

10.3 Expiration and Renewal of Qualification Term

10.3.1 Expiration of Qualification

Personnel qualifications in accordance with 10.2 shall expire three (3) years from the date of qualification for Level II and five (5) years for Level III.

10.3.2 Renewal of Qualification

Renewal of qualification expired under 10.3.1 above shall be in accordance with 10.2.

10.4 Personnel Qualification

Personnel qualifications for this UT process may be transferred among VSC-24 Owners.

11.0 Attachments

Attachment 1 - Flaw Disposition Flow Chart

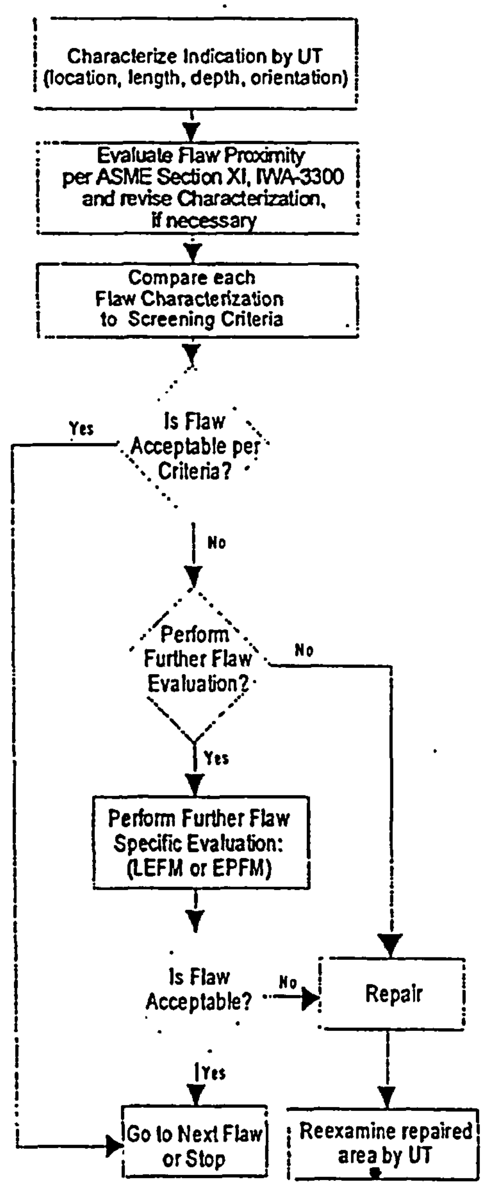
Attachment 2 - Examination Procedure Essential Variables

Attachment 3 - White Paper on Depth Sizing Acceptance Criteria for Time-of-Flight Diffraction Ultrasonic Examination of the VSC-24 Structural Lid to Shell Weld.

Attachment 4 - Screening Criteria for Use During Ultrasonic Examination of VSC-24 Structural Lid to Shell Welds

TITLE: GUIDELINE REQUIREMENTS FOR THE TIME-OF-FLIGHT DIFFRACTION ULTRASONIC EXAMINATION OF THE VSC-24 STRUCTURAL LID TO SHELL WELD

ATTACHMENT 1 - FLAW DISPOSITION FLOW CHART



ATTACHMENT 2 - EXAMINATION PROCEDURE ESSENTIAL VARIABLES

EXAMINATION PROCEDURE ESSENTIAL VARIABLES	
(1)	instrument or system, including manufacturer and model or series of pulser, receiver, and amplifier, and software version
(2)	search units, including: (a) center frequency and either bandwidth or waveform duration; (b) mode of propagation and nominal inspection angles (c) number, size, shape, and configuration of active elements and wedges or shoes
(3)	search unit cable, including: (a) type; (b) maximum length; (c) maximum number of connectors
(4)	detection and sizing techniques, including:
	(a) scan pattern and beam directions;
	(b) maximum scan speed;
	(c) minimum and maximum pulse repetition rate;
	(d) minimum sampling rate (automatic recording systems)
	(e) extent of scanning and action to be taken for access restrictions
(5)	methods of calibration for detection and sizing (e.g., actions required to insure that the sensitivity and accuracy of the signal amplitude and the time outputs of the examination system, whether displayed, recorded, or automatically processed, are repeated from examination to examination)
(6)	inspection and calibration data to be recorded
(7)	method of data recording
(8)	recording equipment (e.g., strip chart, analog tape, digitizing) when used
(9)	method and criteria for the discrimination of indications (e.g., geometric versus flaw indications and for length and depth sizing of flaws)
(10)	surface preparation requirements

ATTACHMENT 3

White Paper on Depth Sizing Acceptance Criteria for Time-of-Flight Diffraction Ultrasonic Examination of the VSC-24 Structural Lid to Shell Weld

ATTACHMENT 4 - Screening Criteria for Use During Ultrasonic Examination of VSC-24 Structural Lid to Shell Welds

1. The lower bound fracture toughness of 75 ksi-√in at 0°F was used in these calculations. This value was determined from the lower bound of material toughness (K_{IC} -type value, calculated from J integral testing performed per ASTM E-1737-96). The arrest toughness ($K_{IA} = K_{ID}$) was calculated from the measured (K_{IC} type result) using the methods of ASME Section XI, Appendix A. The value of 75 ksi-√in at 0°F is compatible with that calculated from the toughness requirement of 15 ft-lb at -50°F (CVN).

2. A constant tensile residual stress of 38 ksi, corresponding to the specified minimum yield of SA-516 Grade 70 material, was used in the calculation.

3. For semi-elliptical flaws, the greater of the calculated stress intensity factors at the deepest point of the flaw and at the surface contact point of the flaw was used to develop screening criteria. The deepest point of the flaw governs the calculation, except for very short, deep flaws (aspect ratio of approximately 0.5), where the surface point governs.

4. For long flaws, evaluation of primary stress limits per ASME Section III, NC-3200 continue to be met for the structural lid-to-shell weld if the flaw depth is less than 0.16 inch assuming that weld minimum design thickness remains at 0.75 inches. Screening criteria maintain this limit for long flaws.

5. The above calculations assume that the minimum weld temperature at which the horizontal drop event could occur is 0°F. Increasing this minimum temperature results in increasing toughness, and therefore increasing allowable flaw size. For temperatures above about 30°F, the allowable flaw size is limited by primary stress criteria rather than brittle fracture limits (see item 4 above).

The screening criteria which have been developed using the above assumptions and methods are summarized in Table 1. It should be noted that these are intended as screening criteria, and not as final acceptance criteria. Flaws that are identified as meeting these criteria following review of UT results are acceptable without further evaluation. Flaws that exceed these criteria may be subjected to further evaluation, utilizing the same analytical techniques and limitation, before making a repair or accept decision.

Table 1
Flaw Screening Criteria

WELD TEMPERATURE	FLAW DEPTH (L<0.7 IN)	FLAW DEPTH (L>0.7 IN)
0°F	0.34 IN	0.11 IN
10°F	0.37 IN	0.13 IN
20°F	0.37 IN	0.14 IN
30°F and Greater	0.37 IN	0.16 IN

White Paper on Depth Sizing Acceptance Criteria for Time-of-Flight Diffraction Ultrasonic Examination of the VSC-24 Structural Lid to Shell Weld

by
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The generic guideline document for the time-of-fight diffraction (TOFD) ultrasonic examination of the VSC-24 structural lid to shell weld contains the following sub-section (8.1.1.3) describing the flaw depth sizing acceptance criteria:

8.1 Examination Process and Technique Requirements

The process and techniques developed for the examination of the MSB structural lid to shell weld shall meet the following requirements:

8.1.1 Circumferentially Oriented Flaws

3. Depth Sizing Acceptance Criteria

The examination shall depth size the detected flaws as follows:

- The mean error in the flaw depth will be calculated and documented. If the mean error is less than or equal to a positive 0.072 inches, the RMS error must be less than or equal to 0.125 inches. If the mean error is greater than a positive 0.072 inches, this contribution to the RMS error will be removed and recorded. The remaining portion of RMS the error must be less than or equal to 0.102 inches.

The purpose of this white paper is to provide the technical basis for the criteria in (8.1.1.3), definitions of the terms used in the criteria, and sample calculations using the criteria.

The industry standard practice for depth sizing acceptance criteria is that the root-mean-squared error (RMSE) be less than or equal to 0.125 inch. For the 52° TOFD data, the criteria on the depth sizing that RMSE ≤ 0.125 inch can be met with no problem (e.g. see Figure 1). However, for the 60° TOFD data, the mean error alone is sometimes as large as +0.125 inch (e.g. see Figure 2). As will be shown below, the mean error is a component of the RMSE, so that the criteria that RMSE ≤ 0.125 inch for the 60° TOFD data cannot be met. The reason for the large positive bias in the depth sizes calculated with the 60° TOFD data is well understood on ultrasonic grounds and will not be addressed here except to mention that it is a conservative error.

The goal of sub-section 8.1.1.3 is to apply the unmodified industry standard (RMSE ≤ 0.125 inch) for the 52° TOFD data where the mean error is small (see Figure 1) and for the 60° TOFD data, to document the large conservative mean error (see Figure 2), account for it, and apply a criteria to the remaining error that is proportional to the industry standard.

To be able to account for the positive bias found in the 60° TOFD data, a short discourse on RMSE is in order. The RMSE can be defined in terms of the sizing data (M_i,T_i), i = 1 ... n by the equation:

$$\mathrm{RMSE}^2 = \frac{\sum_i (M_i - T_i)^2}{n}, \qquad (1)$$

where M_i is a measured flaw dimension and T_i is the "true" flaw dimension. Following the development in NUREG/CR-5410 (see pages 37-39), the RMSE can be expressed in terms of the three parameters of linear regression analysis, i. e.

$$\mathrm{RMSE}^2 = (\beta_1 + (\beta_2 - 1)\mu_T)^2 + (\beta_2 - 1)^2 \sigma_T^2 + \sigma_\varepsilon^2, \qquad (2)$$

where β_1 is the intercept of the regression curve, β_2 is the slope of the regression curve, and σ_ε^2 the variance in the error in the regression curve. It is the minimization of the variance in the regression error that establishes the parameters β_1 and β_2. The term σ_ε is also known as the standard error of estimate and represents the part of the variance in the measured values that can not be explained by regression analysis (i.e. random error).

In equation (2), the mean of the true sizes is given by, $\mu_T = \frac{\sum_i T_i}{n}$; and the variance of the true sizes is given by, $\sigma_T^2 = \frac{\sum_i (T_i - \mu_T)^2}{n}$. Introducing the mean of the measured sizes as $\mu_M = \frac{\sum_i M_i}{n}$ and expressing in terms of the mean of the true sizes using the regression curve; i.e.
$\mu_M = \beta_1 + \beta_2 \mu_T$, the expression for the RMSE can be written as,

$$\mathrm{RMSE}^2 = (\mu_M - \mu_T)^2 + (\beta_2 - 1)^2 \sigma_T^2 + \sigma_\varepsilon^2 \qquad (3)$$

We now recognize the first term as the square of the mean error ($\mu_M - \mu_T$). From the formula in equation (3), the RMSE can only be zero when ideal regression results are obtained, i.e. the mean error is zero, ($\mu_M - \mu_T$) = 0, the regression slope is one, $\beta_2 = 1$, and the standard error of estimate is zero, $\sigma_\varepsilon = 0$. Paraphrasing from NUREG/CR-5410: 'A small root mean squared error forces the sizing bias (mean error, ($\mu_M - \mu_T$)), the deviation from ideal trend (slope error, ($\beta_2 - 1$)) and the random error (estimation error, σ_ε) to be small. Therefore, RMSE summarizes the deviations of the regression parameters from the ideal.'

The expression in equation (3) allows the calculations necessary to account for the positive mean error (bias) in the sizing results of the 60° TOFD data. It also provides a rational basis for establishing the threshold criteria on the mean error before removing it from the RMSE and setting the tighter criteria on the remaining slope error and estimation error components of the RMSE.

The purpose of the threshold criteria on the mean error is to separate the depth sizing measurements into data sets with and without a systematic positive bias. The magnitude of the threshold value should clearly delineate the data sets and not allow for a random separation of the data. In order to meet these goals, the threshold value on the mean error was taken to be equal to its contribution to the limiting value of the RMSE (0.125 inch) assuming equal contributions from the three components of RMSE, mean error, slope error, and estimation error. From equation (3)

$$RMSE^2_{Limit} = (0.125)^2 = 3(\mu_M - \mu_T)^2_{Threshold},$$

$$(\mu_M - \mu_T)_{Threshold} = \frac{+0.125}{\sqrt{3}} = +0.072.$$

Setting the threshold of the mean error at 0.072 inch assures that the data separation will not be arbitrary but associated with the physical differences in the flaw depth measurements.

When the threshold is exceeded, the positive mean error value is recorded and then removed from the RMSE. The new limiting criteria on the remaining components of the RMSE (slope and estimation error) is reduced proportionally from the original value of 0.125 inch, since one out of the three components of the RMSE has been removed. Again from equation (3)

$$\left[RMSE^2 - (\mu_M - \mu_T)^2\right]_{Limit} = \left[(\beta_2 - 1)^2 \sigma_T^2 + \sigma_\varepsilon^2\right]_{Limit} = \frac{2(0.125)^2}{3}$$

$$\sqrt{\left[RMSE^2 - (\mu_M - \mu_T)^2\right]_{Limit}} = \sqrt{\frac{2(0.125)^2}{3}} = 0.125\sqrt{\frac{2}{3}} = 0.102 \cdot$$

The limiting criteria on the remaining error of 0.102 inch is well within the capability of the 60° TOFD data. The 60° TOFD data shows good sizing trend and compact grouping of points about the regression curve (see Figure 2).

Listed below is a version of the depth sizing criteria in (8.1.1.3) annotated with the equations developed in this white paper.

- The mean error (μ_M- μ_T) in the flaw depth will be calculated and documented. If the mean error (μ_M- μ_T) is less than or equal to +0.072 inch the RMS error must be less than or equal to 0.125 inch. *That is,*

  If (μ_M- μ_T) $\leq$+0.072 inch, then RMSE $\leq$0.125 inch.

If the mean error ($\mu_M - \mu_T$) is greater than +0.072 inch, this contribution to the RMS error will be removed and recorded. The remaining portion of the RMS error must be less than or equal to 0.102 inch.
That is,

If $(\mu_M - \mu_T) > +0.072$, *then* $\sqrt{(\mathrm{RMSE}^2 - (\mu_M - \mu_T)^2)} \leq 0.102$ *inch.*

Sample calculation #1: 52° TOFD data (see Figure 1)

Mean Error = ($\mu_M - \mu_T$) = 0.025 inch (see Figure 1),
Therefore ($\mu_M - \mu_T$) $\leq$ +0.072 inch,
then RMSE $\leq$ 0.125 inch.
RMSE = 0.045 inch (see Figure 1),
Therefore RMSE $\leq$ 0.125 inch and the depth sizing criteria has been met.

Sample calculation #2: 60° TOFD data (see Figure 2)

Mean Error = ($\mu_M - \mu_T$) = 0.125 inch (see Figure 2),
Since ($\mu_M - \mu_T$) > +0.072 inch, the mean error of 0.125 inch is recorded for the 60° TOFD data, it is removed from the RMSE and the remaining error must satisfy $\sqrt{(\mathrm{RMSE}^2 - (\mu_M - \mu_T)^2)} \leq 0.102$ inch.
$\sqrt{(\mathrm{RMSE}^2 - (\mu_M - \mu_T)^2)} = 0.035$ inch (see Figure 2),
Therefore $\sqrt{(\mathrm{RMSE}^2 - (\mu_M - \mu_T)^2)} \leq 0.102$ inch and the depth sizing criteria has been met.

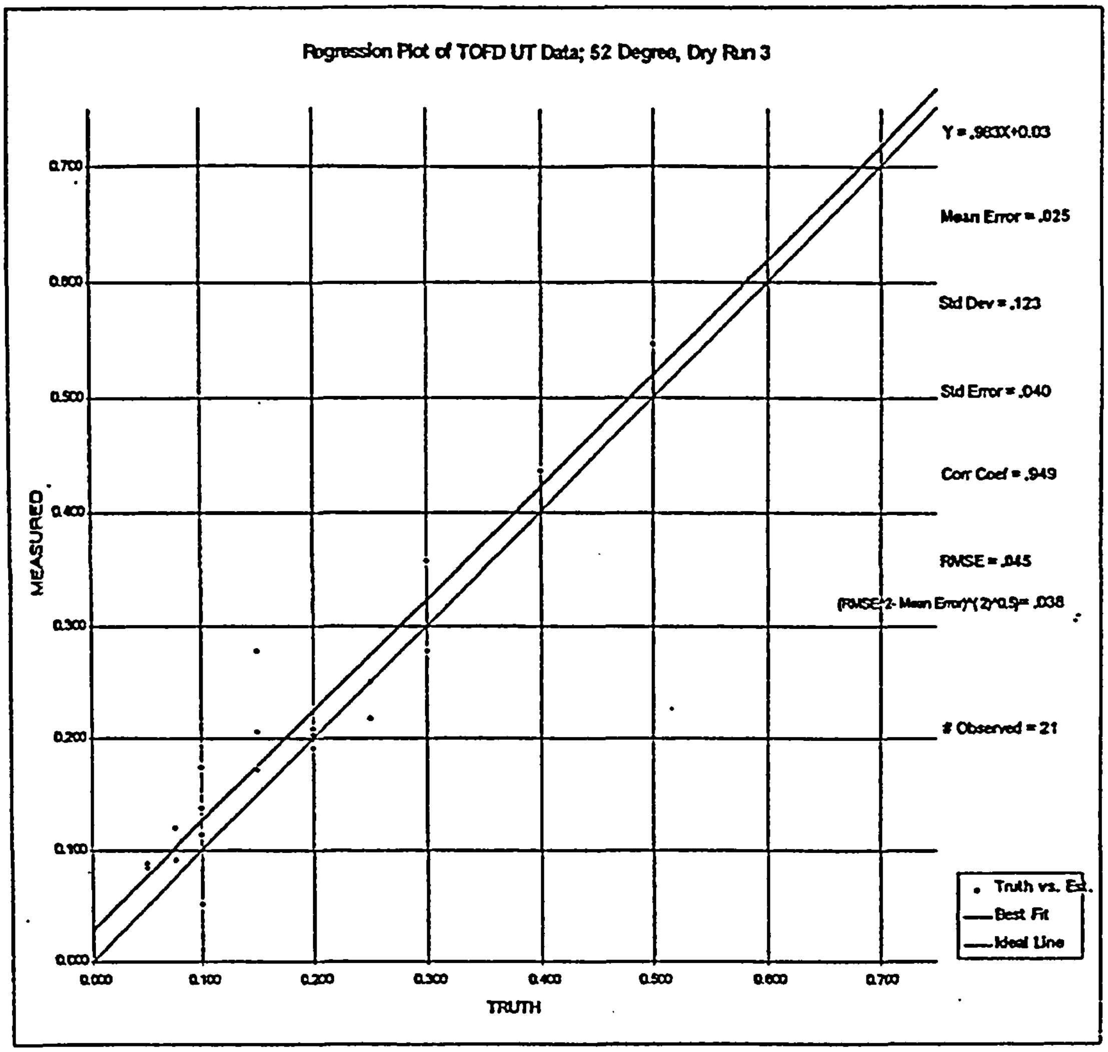

Figure 1. Regression plot of 52° TOFD data (dry run 3).

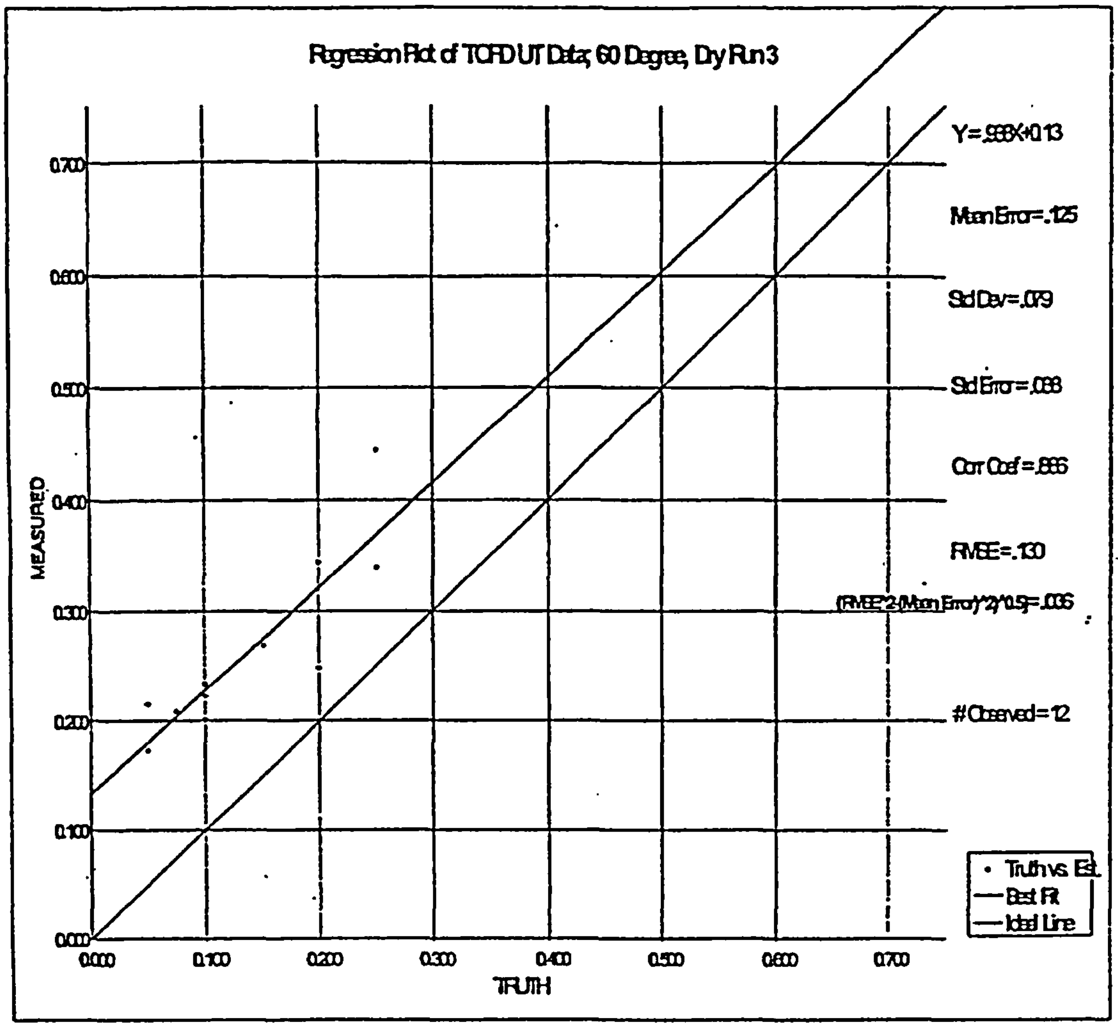

Figure 2. Regression plot of 60° TOFD data (dry run 3).