SNC-96-72SAR
Revision A

SAFETY ANALYSIS REPORT

FOR THE

TRANSTOR™ STORAGE CASK SYSTEM

PREPARED BY:

SIERRA NUCLEAR CORPORATION
SCOTTS VALLEY, CALIFORNIA
ATLANTA, GEORGIA

MAY - 1996

Title: Safety Analysis Report for the TranStor™ Storage Cask System   SNC No: SNC-96-72SAR

REVISION CONTROL SHEET

Rev.	Date	Reason	Affected Pages	Preparer	Checker	Proj. Eng.	Affected Documents/Comments

SIGNATURES

Responsibility	Name/Title	Initials	Date

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

LIST OF TABLES

Page

TABLE OF CONTENTS

LIST OF TABLES

Page

TABLE OF CONTENTS

LIST OF TABLES

TABLE OF CONTENTS

LIST OF FIGURES | Page

TABLE OF CONTENTS

LIST OF FIGURES

Page

1.0 GENERAL DESCRIPTION

Sierra Nuclear Corporation (SNC) has designed an advanced multi-purpose system for the safe on-site storage and off-site shipping of irradiated nuclear fuel. It is designated the TranStor™ System, and has evolved from SNC's licensed Ventilated Storage Cask (VSC) System, which is in service worldwide. The years of VSC handling and storage experience are combined with modern shipping technology to provide a state-of-the-art system for easy efficient handling and management of high-level radioactive waste.

The TranStor™ System includes a sealed basket (TranStor™ Basket) containing storage sleeves, a shipping cask (TranStor™ Shipping Cask) with impact limiters, a vertical concrete storage cask (TranStor™ Storage Cask), and a transfer cask (TranStor™ Transfer Cask). The sealed basket is used in combination with the transfer cask and the storage cask components of the TranStor™ System for storage of irradiated fuel or other radioactive materials. Off-site shipping of irradiated fuel or other licensed contents is performed using only the sealed basket and the shipping cask components of the system.

In addition to intact, failed, and damaged fuel, the TranStor™ System is designed to store and ship fuel debris, non-fuel bearing components, fuel assembly hardware, and other Greater than Class C (GTCC) waste. The contents can be placed in storage at the reactor site; stored at a "stand-alone" facility after reactor decommissioning; and shipped to a repository, an interim storage facility, or a reprocessing plant - all without re-handling the fuel after its initial placement into the TranStor™ Basket.

This Safety Analysis Report (SAR) documents the adequacy of the storage components of the TranStor™ System to satisfy the 10 CFR 72 regulatory requirements for storage of irradiated fuel and other licensed contents. Off-site shipping of the irradiated fuel in accordance with 10 CFR 71 requirements is addressed by a companion SAR (Reference 1.1).

This SAR has been prepared in accordance with the format specified in the United States Nuclear Regulatory Commission (US NRC) Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask."

This chapter describes the TranStor™ Basket, the TranStor™ Storage Cask, the associated components, the licensed contents, the proposed use of the cask, and the physical, safety, and operational features of the cask.

1.1 Introduction

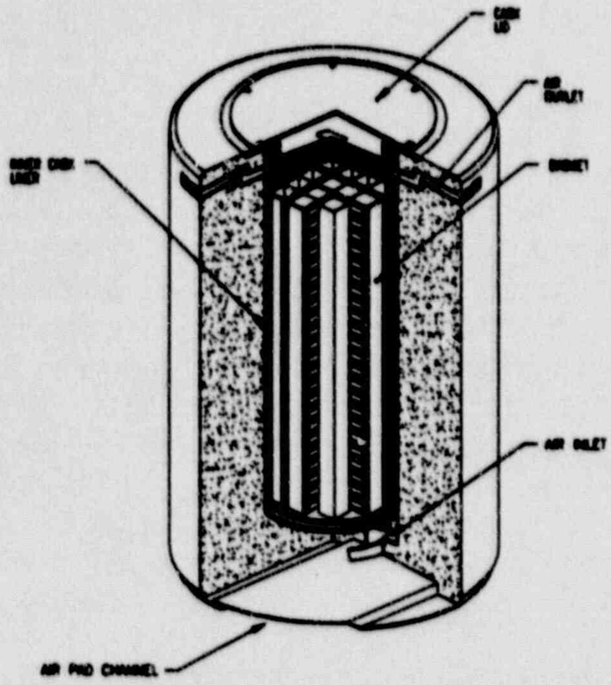
The TranStor™ System provides on-site dry storage of irradiated fuel until it can be shipped to a commercial storage facility or to a Department of Energy (DOE) Monitored Retrievable Storage (MRS) facility or repository. The system includes a TranStor™ Basket, TranStor™ Transfer Cask, and a TranStor™ Storage Cask. An illustration of these components is provided in Figure 1.1-1. Their use is shown in Figure 1.1-2. Detailed dimensions are presented in the TranStor™ System drawings.

Irradiated fuel or other licensed contents are stored in sleeves within the basket. The TranStor™ System design incorporates different configurations of basket internals. Baskets are available for Pressurized Water Reactor (PWR) fuel, Boiling Water Reactor (BWR) fuel, VVER fuel, glassified high level waste canisters, various research reactor fuels and GTCC waste. This SAR only addresses PWR and BWR baskets. The other baskets are addressed in corresponding SARs for the applicable licensing bodies.

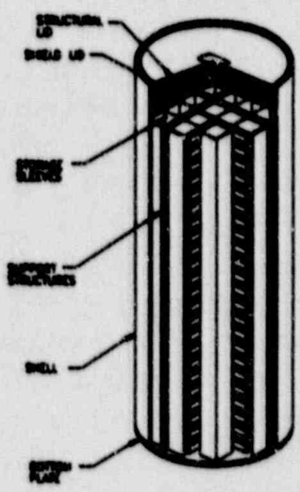
In general, the TranStor™ PWR basket has a capacity of 24 fuel assemblies and the BWR basket has a capacity of 61 BWR fuel assemblies. However, the system allows a variety of loading configurations. This system takes full advantage of the economies of scale and, hence, is the predominant size preferred by utilities where site-specific conditions do not limit the storage cask size. Also, this capacity allows to maintain the weight of transfer and shipping casks within the crane capacity at most power plants. For plants that cannot handle these heavy loads, but do have the room (door sizes, sit-down area, etc.), a single assembly transfer cask can be used to load the larger casks. The designs of the PWR and BWR fuel baskets are presented in Section 1.2.4.

The PWR and BWR versions of the TranStor™ basket accommodate 98% of the fuel inventory in the United States. The basket has fixed neutron poisons and will accommodate fuel enriched up to 4.5% ^{235}U. It will also accommodate mixed oxide fuel (MOX) as well as both zircaloy and stainless steel clad fuel. The major technical specification on the fuel to be stored in the TranStor™ System is the 26 kW heat load per basket. This parameter defines the burnup and age (or time subcritical) at which an assembly may be stored. Burnups as high as 65 GWd/MT can be accommodated. The shielding design is such that any fuel producing 26 kW of heat will also produce acceptable dose rates.

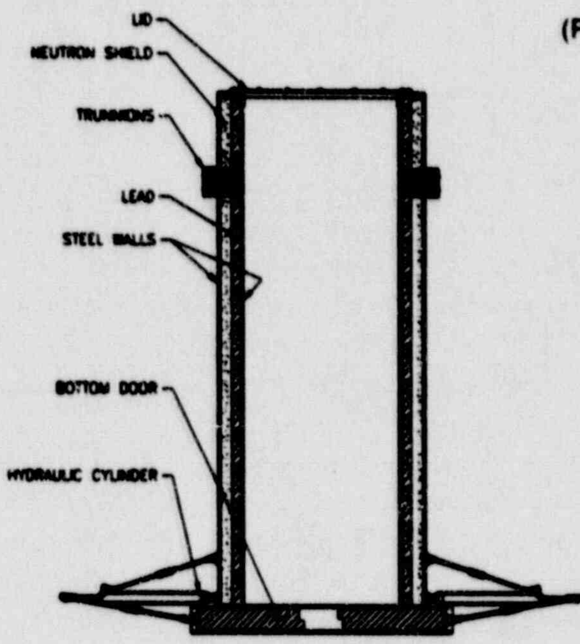
TranStor™ is a fully integrated system designed to provide safe long-term storage and ease of removal of fuel from the site at the end of the storage period. The thick steel and reinforced concrete wall of the storage cask provides very effective gamma and neutron radiation shielding as well as physical protection. The fuel is cooled by internal air flow paths that allow the decay heat to be removed by natural circulation. The cask's vertical orientation simplifies fuel handling operations and increases the draft height for natural convection cooling. The efficient design allows the TranStor™ System to store high burnup, short cooled fuel.



THE VENTILATED STORAGE CASK



THE SEALED BASKET
(PATENT PENDING)



THE TRANSFER CASK

FIGURE 1.1-1

TranStor™ SYSTEM COMPONENTS

The TranStor™ system is proven in operation through its predecessor, the VSC System, placed in service in 1990 (Reference 1.2 and 1.3).

Figure 1.1-2 shows the major handling steps of the TranStor™ Storage System. In the first sequence, an empty basket is placed inside the transfer cask. The basket is then filled with pool water and lowered (inside the transfer cask) into the fuel pool where it is loaded with fuel. The shield lid is placed on the basket while it is in the pool and then the transfer cask is lifted out of the pool and placed in the decontamination area. Here the basket is seal welded, drained of water, vacuum dried, tested, and back-filled with helium. Next, the transfer cask is moved to the top of the storage cask, the transfer cask doors are opened, and the basket is lowered into the storage cask. The loaded storage cask is then moved to the storage pad via an air pad (shown in the figure) or any other method within the conditions of cask use. The TranStor™ Storage Cask can be handled by many different types of equipment (cranes, cask transporters, air pads, etc.). As long as the handling is done in compliance with the conditions of cask use (Section 12.0 of this SAR), the handling equipment is not classified as important to safety since the cask is designed to withstand any event that could be created by the failure of the handling equipment.

The basket is designed by analysis to meet material and stress requirements of the American Society of Mechanical Engineers (ASME) Code, Section III, Division 1. The storage cask is designed by analysis to meet the American National Standard ANSI 57.9 as well as the American Concrete Institute's ACI 349 Code. The transfer cask is designed as a lifting device to meet NUREG 0612/ANSI N14.6 and as a shielding bell for radiological safety. Section 2.0 further describes the design criteria.

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

When a federal MRS or commercial storage facility is available, the TranStor™ Shipping Cask can be used to ship the TranStor™ Basket without reopening. Figure 1.1-2 also shows the initial steps for removal of the basket from the storage cask after the storage period. The shipping cask is positioned next to the storage cask and the transfer cask is placed on the opened storage cask. The basket is then raised into the transfer cask and the transfer cask moved from the now empty storage cask to the shipping cask, where the basket is then lowered into the shipping cask. The shipping cask is then closed and prepared for shipping and the empty storage cask may also be loaded onto a railcar for shipping.

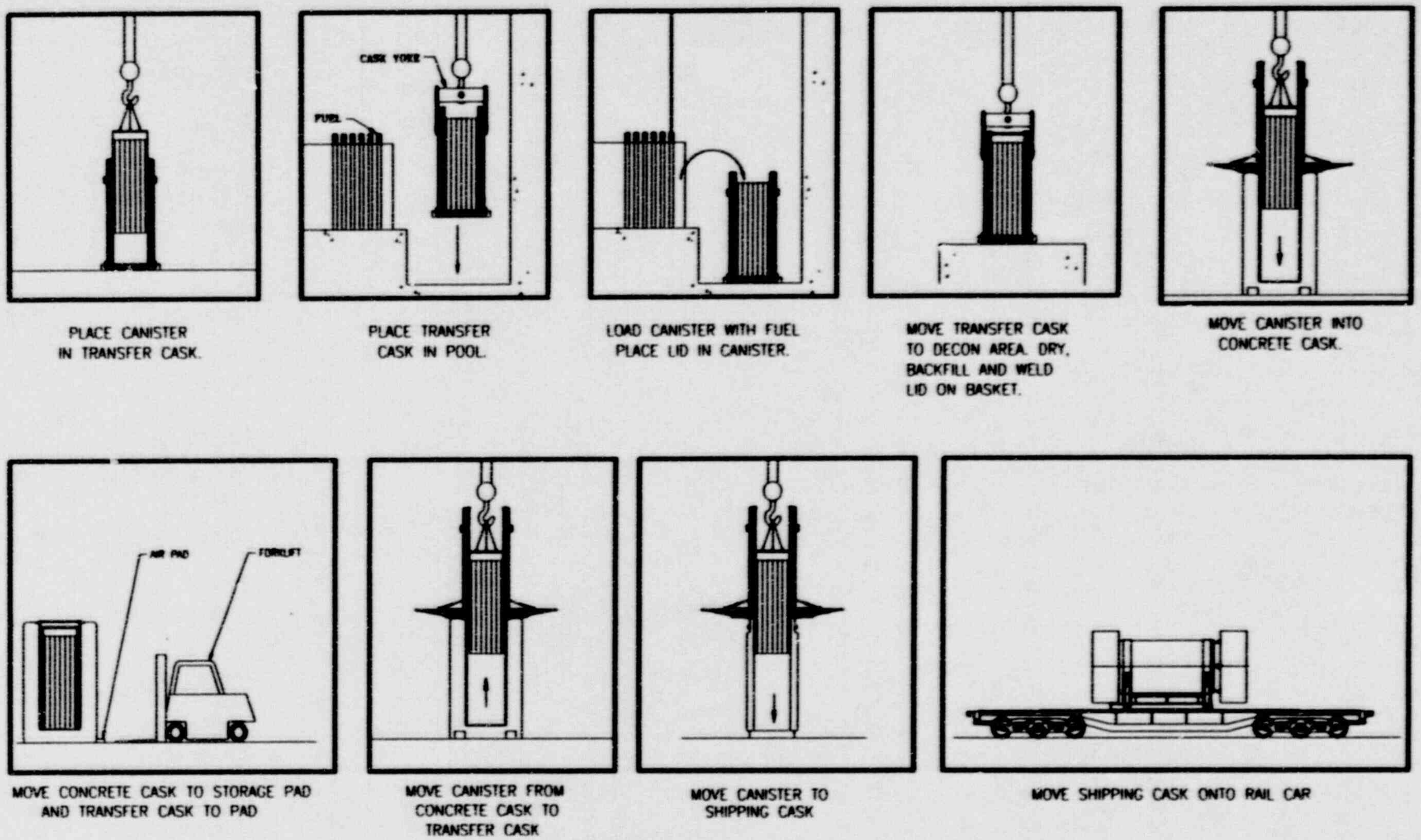


FIGURE 1.1-2

TranStor™ SYSTEM LOADING AND SHIPPING OPERATIONS

Definition of Major Terminology

TranStor™ Basket: The component of the TranStor™ System in which irradiated fuel and/or other licensed contents are stored and shipped. It consists of an assembly of sleeves inserted in a shell closed by a welded shield lid and a welded structural lid.

TranStor™ BWR Basket: The TranStor™ basket to store and transport BWR fuel assemblies.

TranStor™ PWR Basket: The TranStor™ basket to store and transport PWR fuel assemblies.

TranStor™ Storage Cask: The cask to store baskets containing irradiated fuel (intact, failed, or damaged fuel, fuel debris, non-fuel bearing components, fuel assembly hardware, and GTCC waste). The cask consists of a steel inner liner and reinforced concrete shell.

TranStor™ System: An advanced multi-purpose system for the safe off-site transportation and on-site storage of irradiated nuclear fuel and other licensed contents. The system consists of a TranStor™ Basket, TranStor™ Shipping Cask with impact limiters, TranStor™ Storage Cask, and TranStor™ Transfer Cask.

Confinement Boundary: The components of the basket that contain the radioactive contents being stored. The basket confinement boundary consists of the basket shell, the basket bottom plate, the structural lid, shield lid, and the port cover plates.

Independent Spent Fuel Storage Installation (ISFSI): A facility designed and constructed for the storage of irradiated fuel and other radioactive materials associated with irradiated fuel. An ISFSI may be located on a nuclear plant site or it may be a "stand-alone" facility.

TranStor™ Transfer Cask: The cask for transferring the basket with irradiated material from/to the TranStor™ Storage Cask or TranStor™ Shipping Cask. The transfer cask consists of an inner cavity, inner and outer shells, gamma shielding material, neutron shielding material, bottom doors, and lifting trunnions.

1.2 General Description of the TranStor™ Storage System

1.2.1 TranStor™ Storage System Characteristics

The TranStor™ Storage System includes:

1. TranStor™ Storage Casks

2. TranStor™ Baskets (PWR or BWR)
(placed inside the central cavity of the storage cask)

3. One TranStor™ Transfer Cask
(used to move the loaded basket between the spent fuel pool, storage cask, and shipping cask)

4. One Cask Transporter or Air Pad System with Heavy Haul Trailer
(for cask transport from the auxiliary building to the storage location)

5. One Vacuum Drying and Helium (He) Back-Fill System with a He Leak Detector

6. One Semi-Automatic (or Manual) Welding System

The overall design criteria of the TranStor™ System are shown in Tables 1.2-1 and 1.2-2. The design accounts for both normal and off-normal conditions, including a range of credible and hypothetical accidents. The system design and analyses were performed in accordance with Title 10, Code of Federal Regulations, Part 72 (10 CFR 72), ANSI/ANS 57.9, and the applicable sections of the ASME and ACI codes. Other applicable ANSI, ANS, AWS, and ASTM standards and NRC regulatory guides were also used in the TranStor™ System design. These codes, standards, and regulatory guides are discussed in the applicable sections describing the component designs and are called out in the fabrication and construction specifications.

TABLE 1.2-1

SUMMARY OF TranStor™ SYSTEM DESIGN CRITERIA

Component: Storage Cask (Major Design Codes: ANSI 57.9, ACI 349)

Design Load Type	Design Parameters	Applicable Codes
Design Basis Tornado (DBT)	Maximum wind pressures Maximum speeds Maximum differential pressures	NRC Reg. Guide 1.76 ANSI 58.1
DBT Missile	Maximum speed Types: Automobile 8 in dia. shell 1 in solid sphere	NUREG 0800, Section 3.5.1.4
Flood	Maximum water height Maximum velocity	10 CFR 72.122
Seismic	Design hor. acceleration: 0.75g Design vert. acceleration: 0.5g	10 CFR 72, NRC R.G. 1.60, NRC R.G. 1.61
Dead Loads	Dead weight, including basket weight	ANSI 57.9 and ACI 349
Design Basis Temperature	Maximum concrete temperature 225 °F - normal 350 °F - accident	ACI 349 and NRC Guidance
Snow and Ice Loads	Design load of 67.2 lb/ft^2 Included in live loads	ANSI A58.1, ANSI 57.9 and ACI 349

- continued on next page -

TABLE 1.2-1 (continued)

SUMMARY OF TranStor™ SYSTEM DESIGN CRITERIA

Component: Basket (Major Design Code: ASME, Section III)

Design Load Type	Design Parameters	Applicable Codes
Flood	Maximum water height	10 CFR 72.122
Seismic	Design hor. acceleration: 0.75 g Design vert. acceleration: 0.5 g	10 CFR 72, NRC R.G. 1.60
Dead Loads	Weight of loaded basket	ASME, Sec. III
Design Basis Internal Pressure	Design pressure	ASME, Sec, III
Normal and Off-Normal Operating Temperature	Basket with fuel generating 26 kW decay heat. Ambient temperature - 40°F to 125°F	ASME, Sec. III, 10 CFR 72
Normal Operation Load	$\pm$ 0.5g applied in all directions simultaneously	ASME, Sec. III
Operation Handling Accident Load	Basket moving at 2 ft/sec impacting cask or building wall	ASME, Sec. III
Accident Drop	Peak vert. deceleration: 124g Peak hor. deceleration: 44g	10 CFR 72, 10 CFR 71
	Confinement Boundary (shell, structural lid)	ASME, Sec. III, 10 CFR 72
	Sleeve Assembly	ASME, Sec. III, 10 CFR 72 10 CFR 71, NUREG/CR-6322
Accident Pressure	Maximum internal pressurization from rod ruptures	ASME, Sec. III
Load Combinations	See Table 2.2-3 and 2.2-4	ASME, Sec. III

1.2.2 Major Storage Components

1.2.2.1 TranStor™ Basket

The TranStor™ Basket is a cylindrical steel canister designed for the storage and shipping of irradiated fuel and other licensed contents. Its major components are a storage sleeve assembly, shell assembly, bottom plate, shield lid, and structural lid. The shell, bottom plate, and lids provide confinement boundary, shielding, and lifting provisions for the basket. The sleeve assembly within the shell positions and supports the basket contents and is designed to accommodate intact fuel, damaged fuel, and fuel debris. Schematics of the TranStor™ PWR and TranStor™ BWR baskets are shown in Figures 1.2-1. and 1.2-2, respectively. All basket structural components are designed to be fabricated of either carbon or stainless steel as defined on the system drawings. Table 1.2-2 lists the major physical characteristics of the basket.

Special cans are placed in the basket for storage and transport of failed/partial fuel assemblies and fuel debris. The can design includes provisions for draining and drying and ensures confinement of fissile material within designated cells.

The TranStor™ Baskets vary in length depending on the length of the fuel assemblies to be stored/shipped. The longest TranStor™ Basket has a cavity length of 180 inches. All of the baskets are loaded into the TranStor™ Storage Cask cavity for storage.

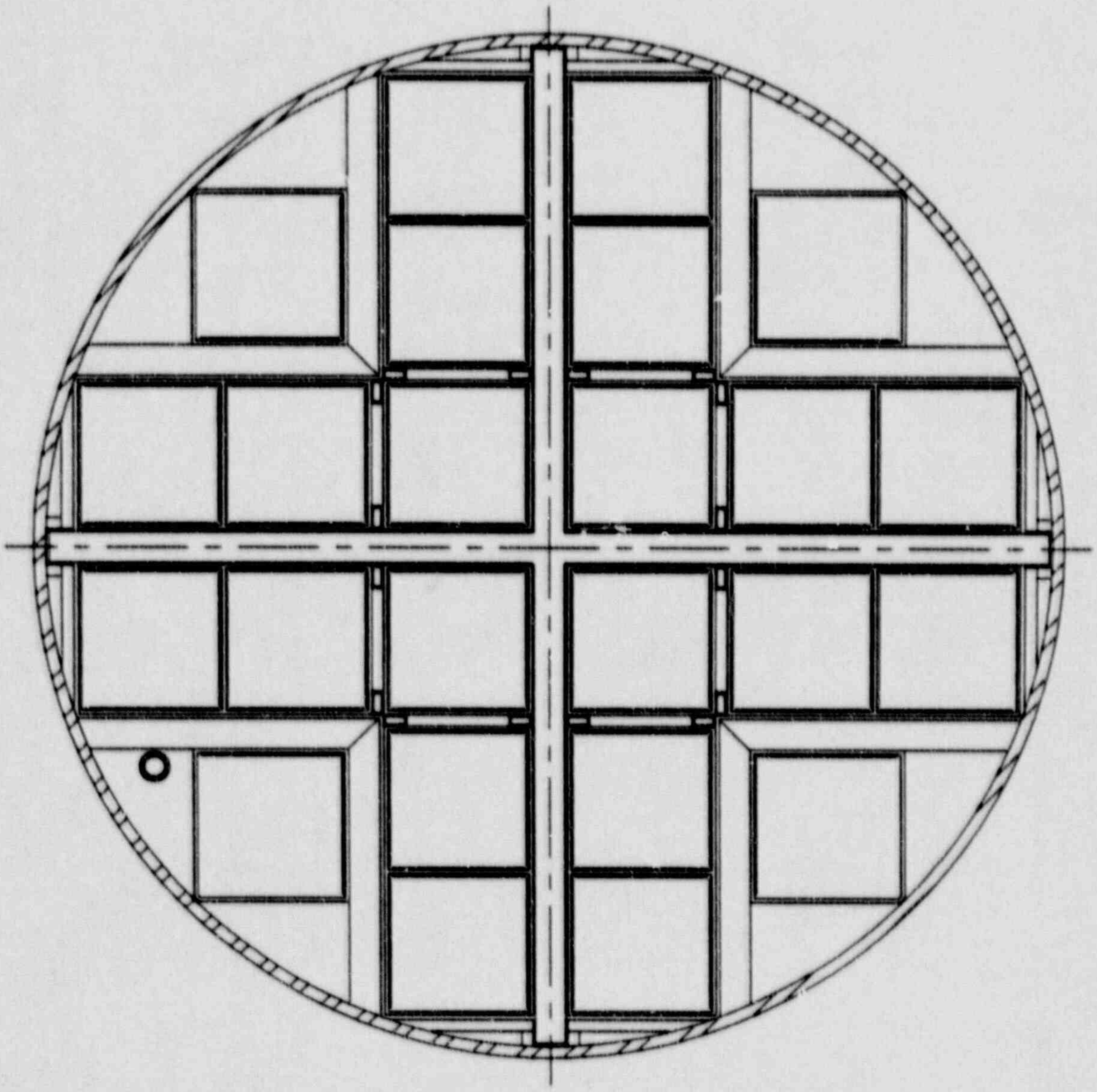
The shell assembly consists of a cylindrical shell with a bottom plate and the shield lid support ring. The sleeve assembly is placed inside the shell and consists of square tubes which include neutron poison sheets to maintain subcriticality for fresh (unirradiated) fuel in unborated water. The PWR sleeve assembly also contains a framework of crossbeams that provide structural support and create gaps for neutron moderation (flux traps). The internals of the basket are coated to prevent detrimental effects on the fuel pool water chemistry during the time that the basket is in the pool for loading. A coating with a proven history of nuclear applications and compatibility with existing pool chemistry requirements is used. Coating on the exposed surfaces of the basket is a radiation resistant hard film coating that allows easy decontamination and can withstand elevated temperatures.

The shield lid assembly is a thick steel disk that is positioned on the shield lid support ring above the sleeve assembly after the fuel has been loaded into the basket and while the basket is still in the spent fuel pool. The shield lid is welded to the basket while in the decontamination pit. Two penetrations through this shield lid are provided for draining, vacuum drying, and backfilling the basket with helium. The drain penetration has a pipe thread fitting on the top and the bottom. The drain pipe is threaded into the bottom of the shield lid just prior to lowering the shield lid into the basket in the spent fuel pool. The pipe extends to within 1/8-inch of the bottom of the basket to facilitate water removal. The vent penetration in the shield lid is a quick connect fitting used to aid in water removal and for vacuum drying the basket and backfilling it with helium.

TABLE 1.2-2

MAJOR PHYSICAL CHARACTERISTICS OF THE BASKET

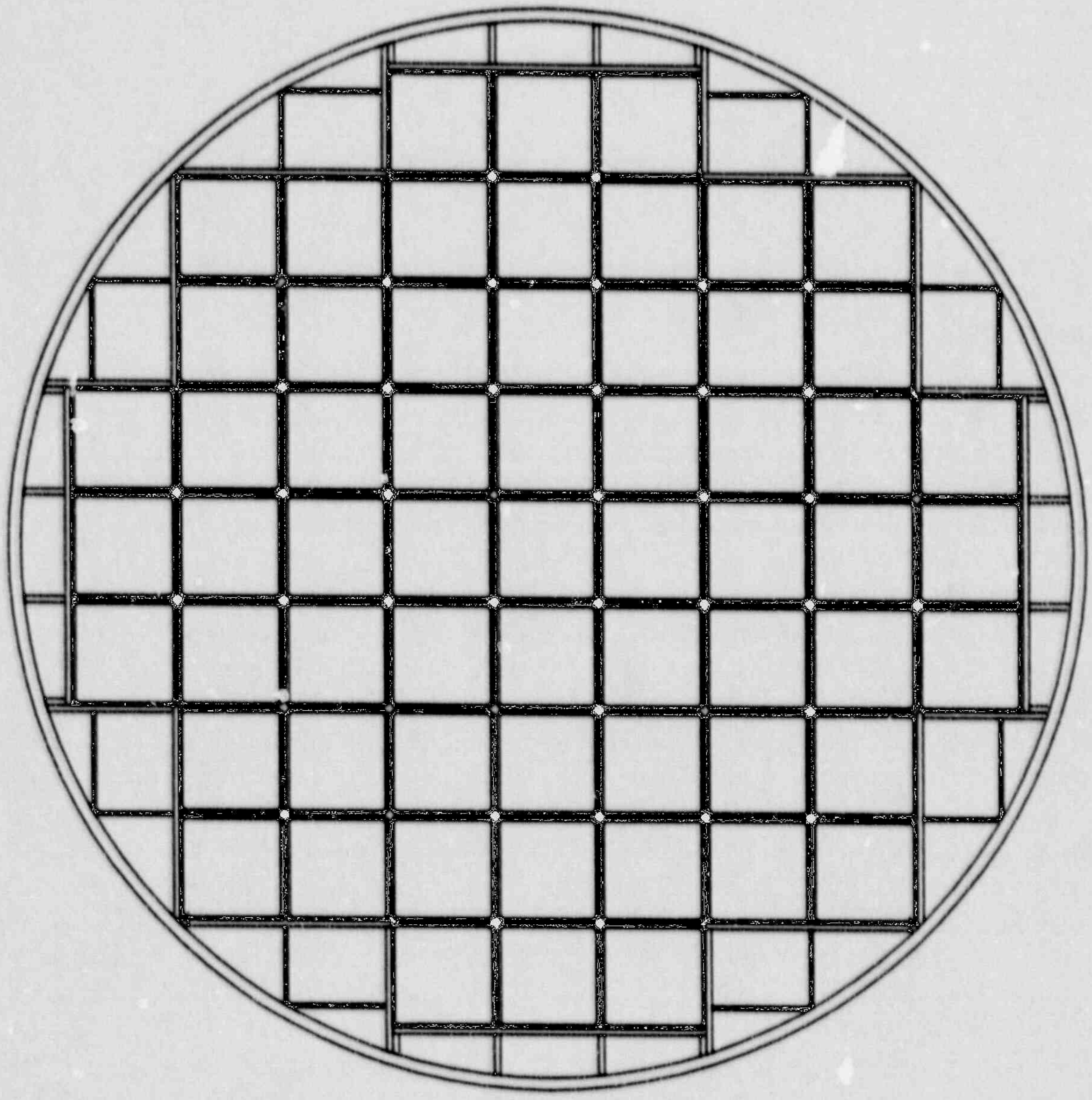
Parameter	Value
Outside Diameter	66 inches
Length	192.25 inches maximum (depends on fuel length)
Capacity	24 PWR assemblies 61 BWR assemblies
Maximum Heat Load	26 kW
Material	Carbon and/or stainless steel per drawings
Internal Atmosphere	Helium



(PATENT PENDING)

FIGURE 1.2-1

SCHEMATIC OF THE TranStor™ PWR BASKET



(PATENT PENDING)

FIGURE 1.2-2

SCHEMATIC OF THE TranStor™ BWR BASKET

The structural lid is a steel disk that is positioned on top of the shield lid and welded to the basket shell after the shield lid is welded in place. The structural lid has a penetration for access to the vent and drain connections in the shield lid. This structural lid penetration is sealed via multiple welds once the helium backfill process has been completed.

Fabrication

Detailed fabrication requirements are described in the TranStor™ drawings. The major points of the fabrication are listed in Table 1.2-3.

1.2.2.2 TranStor™ Storage Cask

The storage cask provides structural support, shielding, and natural convection cooling for the basket. It is shown in Figure 1.2-3. Table 1.2-4 lists the major physical characteristics of the cask. The steel and concrete walls are sufficiently thick to limit side radiation dose rates 1 meter from the cask surface to less than 10 mrem/hr. The internal cavity of the storage cask is formed by a thick steel cylinder. The concrete is normal weight 4000 psi concrete. Outer and inner re-bar cages are formed by vertical hook bars and horizontal hoop bars. The concrete mix has been selected to assure strength and long life at the elevated temperatures expected during normal operations and the higher short-term temperatures that could potentially occur during off-normal and accident conditions. Specific characteristics of the selected mix are the use of Type II Portland Cement and matching the aggregate's and its carrier's thermal expansion coefficients. The air flow path is formed by the channels at the bottom (air entrance), the air inlet ducts, the gap between the basket exterior and the storage 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 provides additional shielding to reduce the skyshine radiation as well as protection of the basket from the environment and postulated tornado missiles. This cover is bolted in place and has tamper indications on two of the nuts. The bottom of the storage cask has a steel plate which prevents any loss of material during handling. Also, the storage cask has chamfered corners at the top and bottom to minimize concrete spalling.

The cask is constructed by pouring concrete between a re-usable form and the inner metal liner. Reinforcing bars are placed near the inner and outer concrete surfaces. The air flow embedments are inserted into the cask liner shell and tied to the outer re-bar frame and outer form. The cask construction is typically performed at the site using a special construction area next to the ISFSI and outside of the security fence. However, the cask can also be constructed off-site at a local contractor's facility and "heavy hauled" to the plant site.

The storage cask construction is detailed in the drawings. Table 1.2-5 provides a summary of the construction specification.

TABLE 1.2-3
BASKET FABRICATION SPECIFICATION SUMMARY

MATERIALS

- Carbon and/or stainless steel in accordance with the referenced drawings.

WELDING

- All filler metals shall be appropriate ASME material.
- All welders and welding operators shall be qualified in accordance with ASME Section IX.
- All welding procedures shall be written and qualified in accordance with ASME Section IX
- All welds to be visually examined shall be examined as specified in ASME Section V, Article 9.
- All welds to be dye penetrant or magnetic particle examined shall be examined in accordance with the requirements of ASME Section V, Article 6 and Section III, NC-5350 or Article 7 and NC-5340 respectively.
- All welds to be radiographed shall be examined in accordance with the requirements of ASME Section V, Article 2 and Section III, NC-5320
- All personnel performing examinations shall be qualified in accordance with the quality assurance program, and SNT-TC-1A.

FABRICATION

- All cutting, welding, and forming shall be in accordance with ASME, Section III, NC-4000 (pressure boundary) or NG-4000 (basket internals) unless otherwise specified. ASME stamping is not required.
- All surfaces shall be cleaned to a surface cleanness classification C or better as defined in ANSI N45.2.1, Section 2.
- All tolerances shall meet the requirements of the referenced drawings after fabrication.

PACKAGING AND SHIPPING

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

QUALITY ASSURANCE

- The basket shall be fabricated under a quality assurance program that meets 10 CFR 72 Subpart G, ASME NQA-1, or an equivalent.
- Hold points for inspection of basket assembly are verification of storage sleeve dimension and straightness and insertion of sleeves into the basket shell prior to final packaging for shipment.
- A Certificate of Compliance shall be issued stating that the basket meets the specifications and drawings.

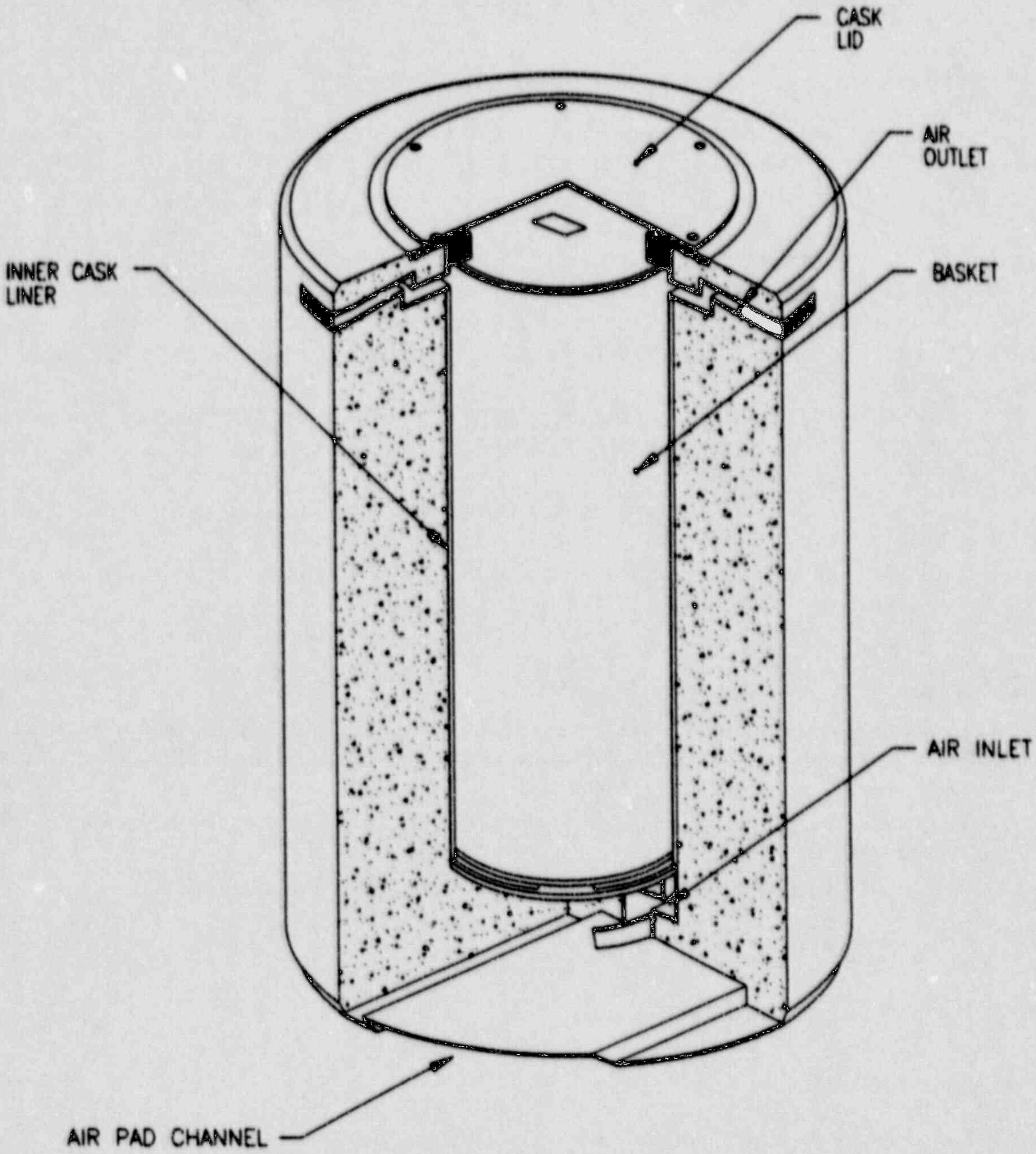


FIGURE 1.2-3

SCHEMATIC OF THE TranStor™ STORAGE CASK

TABLE 1.2-4

MAJOR PHYSICAL CHARACTERISTICS OF THE TranStor™ STORAGE CASK

Parameter	Value
Height	222.5 inches maximum (depends on fuel length)
Outside Diameter	136 inches
Capacity	1 loaded basket (24 PWR or 61 BWR assemblies)
Maximum Heat Load	26 kW
Radiation Dose (1 meter from surface):	
Side	< 10 mrem/hr
Top	< 150 mrem/hr
Material of Construction	Reinforced concrete and steel
Service Life	>50 years

TABLE 1.2-5

STORAGE CASK CONSTRUCTION SPECIFICATION SUMMARY

MATERIALS

- Concrete mix shall be in accordance with the requirements of ACI 318.
- Type II Portland Cement, ASTM C150.
- Fine aggregate ASTM C33.
- Coarse aggregate ASTM C33.
- Reinforcement A 615 Gr. 60
- Admixtures
  - Water Reducing ASTM C494.
  - Pozzolanic Admixture ASTM C618.
- Compressive Strength 4000 psi.
- Air Entrainment 3% - 6%.

WELDING

- Visual inspection of all structural welds shall be performed to the requirements of AWS D1.1 or ASME, Section III, Subsection NF.
- Visual inspection of all construction welds to ensure that seams are substantially sealed to prevent concrete seepage.

CONSTRUCTION

- Concrete testing for each truckload of concrete.
- Test specimens shall be tested in accordance with ASTM C39.
- Formwork to be in accordance with ACI 318.
- All sidewall formwork and shoring to remain in place for at least 24 hours.
- All bottom formwork and shoring to remain in place for 14 days.
- Embedded items shall conform with ACI 318 and the referenced drawings.
- The placement of concrete shall be in accordance with ACI 318.
- Surface finish shall be in accordance with ACI 318.

QUALITY ASSURANCE

- The storage cask shall be constructed under a quality assurance program that meets 10 CFR 72 Subpart G, ASME NQA-1, or equivalent. The quality assurance program must be accepted by the client prior to initiation of the work.
- Parameters important to safety that are to be covered by the QA program are density, wall thickness, compressive strength, and reinforcing material strength and quantity.

1.2.2.3 TranStor™ Transfer Cask

The Transfer cask is a lifting device and a shielding bell used to move the basket between the fuel pool and the storage cask as well as between storage and shipping casks.

Figure 1.2-4 shows the transfer cask schematics and Table 1.2-6 gives the important physical characteristics of the cask design. It consists of a cylinder (the length is site specific and depends on the basket length) with a steel-lead-neutron shield-steel composite wall. Two trunnions are provided for cask handling. The transfer cask has movable shield doors at the bottom to allow lowering of the basket into the storage or shipping cask. The doors slide in steel guides along each side of the transfer cask. Four steel pins are used to prevent inadvertent opening of the doors. Hydraulic cylinders are used to open the doors for the basket transfer. The top cover of the cask extends over the basket to prevent it from being inadvertently lifted out of the transfer cask while being transferred into the storage cask. The yoke provided with the transfer cask is used to interface with the existing site crane or mobile crane.

The transfer cask is a special lifting device designed and fabricated to the single-failure proof 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 transfer cask drop being considered (NUREG 0612). If the transfer cask is used with a crane that does not meet the single-failure proof criteria, site-specific heavy loads evaluation must be performed to conform to the existing 10 CFR 50 license.

During operation, an empty basket is inserted into the transfer cask and filled with water. Steel shielding segments are placed in the top of the basket-transfer cask gap. The gap between the inner transfer cask surface and the outer basket surface is also filled with clean (or filtered pool) water as the cask is being lowered into the pool. A clean water supply hose is also connected to the side of the transfer cask. This allows clean water to be injected into the transfer cask-basket gap during the entire time it is submerged in the pool. This prevents the outside of the basket from becoming contaminated due to contact with the pool water and loosened crud from the fuel assemblies. An alternative of using inflatable seals to prevent the pool water from contacting the basket exterior is also available. After loading fuel into the basket, the shield lid is installed, and the transfer cask containing the loaded basket is lifted from the pool and placed in the decontamination area. Here the basket is seal welded, dried, backfilled with helium, and structurally welded.

At this point the transfer cask and its basket payload are moved from the decontamination area to the top of the storage cask. The bottom doors of the transfer cask are unpinned, the hydraulically operated lower shield doors are opened and the basket is lowered into the storage cask.

TABLE 1.2-6

MAJOR PHYSICAL CHARACTERISTICS OF THE TranStor™ TRANSFER CASK

Parameter	Value
Inside Diameter	67 in
Outside Diameter	86 in
Height	204 inches maximum (depends on fuel length)
Weight (empty)	126,830 lbs max.
Working Dose Rate (1 meter from surface)	< 100 mrem/hr

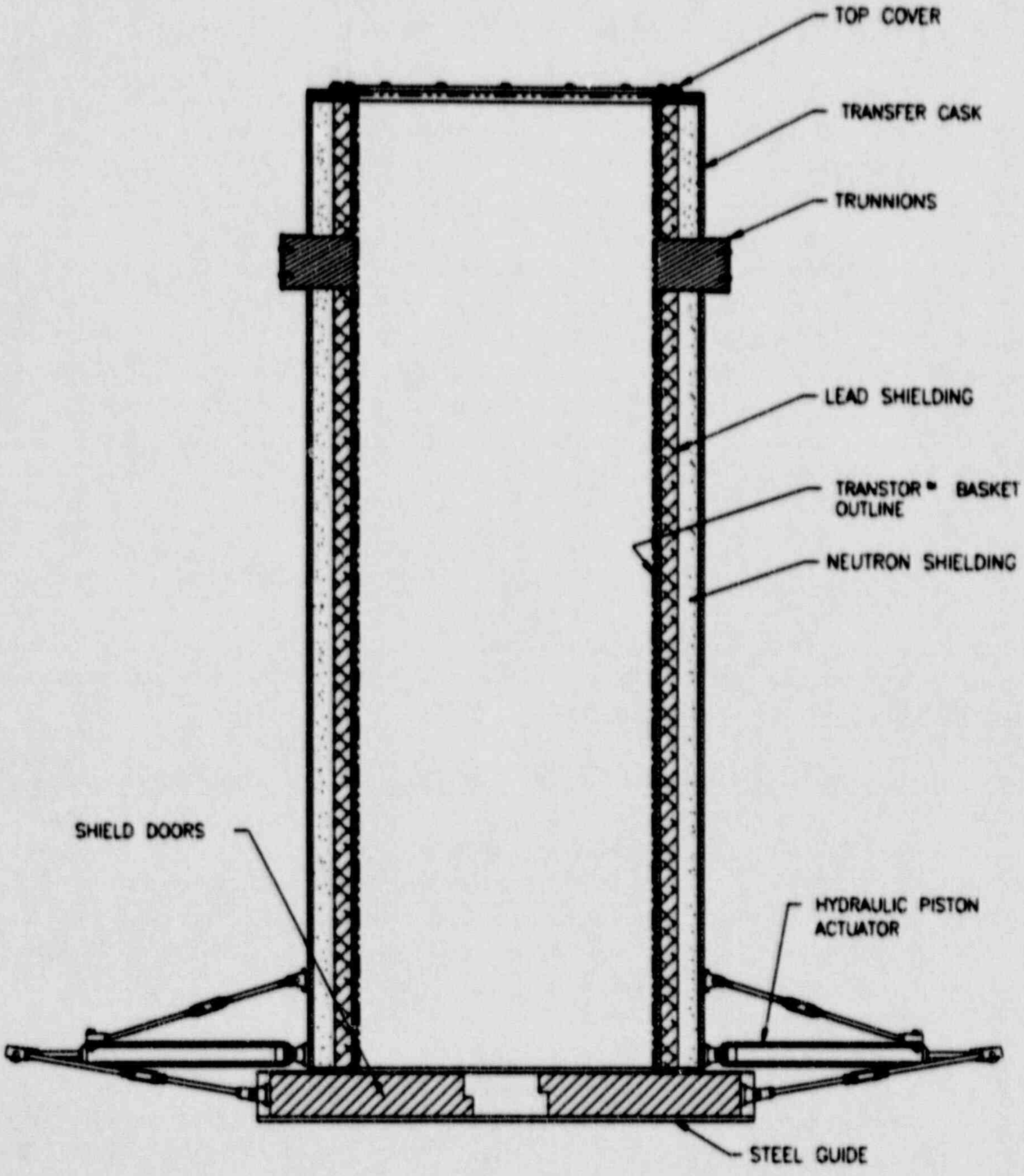


FIGURE 1.2-4

SCHEMATIC OF THE TranStor™ TRANSFER CASK

Storage cask unloading for off-site shipping of the fuel is similar to the loading operation described above. The transfer cask is placed on top of the storage cask, the doors are opened, and the basket is pulled into the transfer cask. The doors are then closed, the transfer cask is moved onto the shipping cask, doors re-opened, and the basket lowered into the shipping cask cavity.

The operations described above can be used in any combination to transfer the basket between casks for loading, unloading, storage, shipment, or off-normal event recovery. The details of operations under different scenarios are provided in Chapter 8.0.

It must be noted that the main transfer cask presented herein is designed for use by all PWR plants as well as by BWR plants with crane capacities of at least 110 ton. This cask takes advantage of the full wall thickness to provide a low working dose rate even with the hottest design basis fuel. In addition, a second transfer cask is designed for use at the BWR plants with cranes of at least 100 ton. The dose rates and stresses for this lighter cask are presented in Appendix 2. The Appendix demonstrates that this cask also meets the single-failure proof requirements of NUREG 0612 and maintains low working dose rates for plant personnel.

1.2.2.4 Description of the TranStor™ Storage Cask Handling Equipment

The TranStor™ Storage Cask is too large to be lifted by most in-plant cranes. Hence, a trailer with air pads or a transporter is used to move the cask. At sites where the cask needs to travel reasonably short distances, just the air pads may be sufficient.

The air pads are commercially available lifting pads which lift the cask a few inches using high pressure air. When energized, the pads and the cask float on a cushion of air and can be easily moved around by forklift, other light vehicle, or by special air driven wheels associated with some commercially available air pads. Use of the air pads allows the cask to be moved on and off the transport trailer or directly to or from the storage or construction pad. Air pads also minimize the need for handling room around the cask and, thus, minimize the size of the storage pad, create more self-shielding among the casks, decrease the environmental impact, and reduce the cost of the storage pad.

A number of different transfer trailers can be used for the on-site movement of the storage cask. One of the options is shown in Figure 1.2-5. This trailer has 16 pairs of tires and can support the stationary weight of the storage cask, transfer cask, loaded basket, and auxiliary equipment. The trailer has stabilizing/load spreading jacks which will spread the floor loadings to below the truck bay limits and will prevent vertical trailer movement during loading or unloading of a storage cask.

If an engineered transporter is used, the vehicle would be similar to those previously developed for the VSC-24 System (Reference 1.2). The transporter employs lifting from above so the lifting lugs would be provided on the cask top. A sketch of a typical cask transporter is shown in Figure 1.2-6.

1.2.2.5 Vacuum Drying System

A skid-mounted vacuum drying system is used to remove the water from the basket (following fuel loading), dry the basket, and backfill it with helium. The vacuum drying system is designed to evacuate the basket to <3 torr in a stepwise fashion. During evacuation, the decay heat from the fuel further helps remove residual moisture from the basket. The same process is used to remove water from the failed fuel and debris cans.

1.2.2.6 Semi-Automatic Welding System

The baskets are welded using a semi-automatic welding system. The welding equipment design is based on successful operation at other facilities and their ALARA results.

1.2.3 Operational and Safety Features

Fuel assemblies are placed into the TranStor™ cask according to the following sequence of main events (see Section 8.0 for a more detailed presentation of TranStor™ System operations):

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

2. Load fuel assemblies in the basket.

3. Place shield lid on basket and use transfer cask to remove basket from fuel pool.

4. Close basket (by welding the shield and structural lids, seal welding the structural lid/shielding lid penetration, draining, vacuum drying, and backfilling with helium, then completing the remaining passes for the structural lid weld and welding the structural lid penetration port cover plates to the structural lid).

5. Transfer basket to the storage or shipping cask.

6. Close the cask for storage or shipment.

7. Move the storage cask to its storage location or move the shipping cask onto the railcar or heavy-haul trailer for off-site shipment.

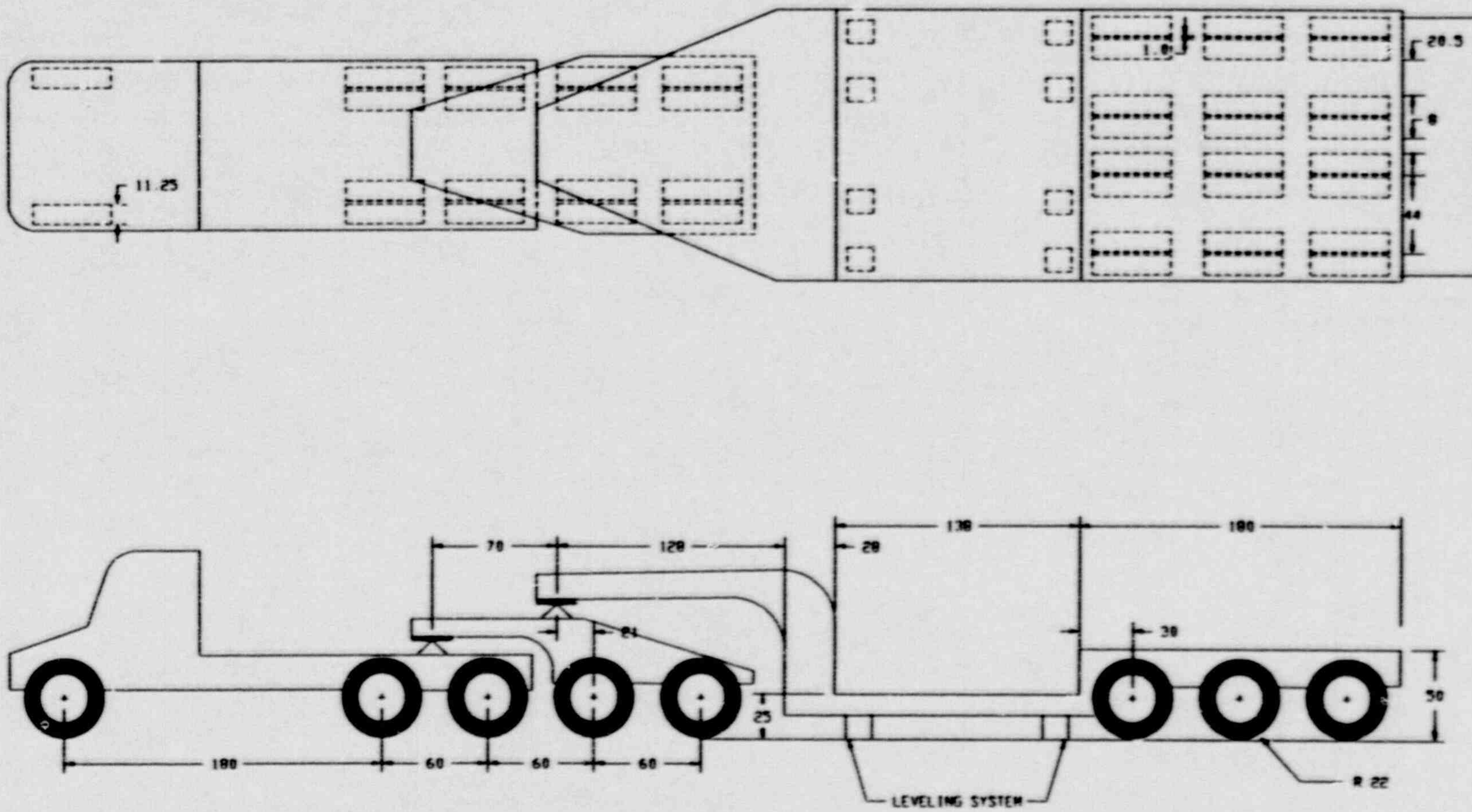


FIGURE 1.2-5

HEAVY HAUL TRANSFER TRAILER

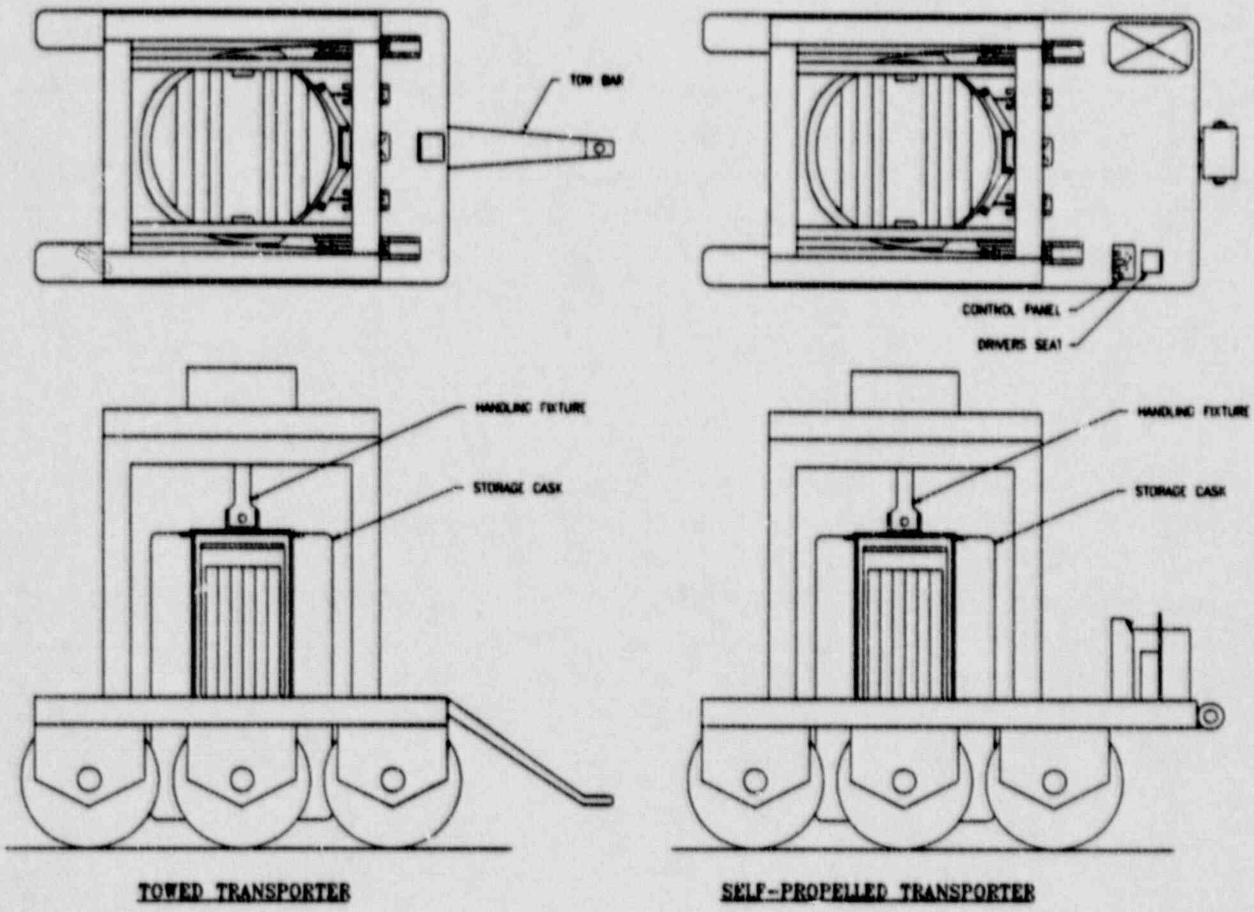


FIGURE 1.2-6

OPTIONAL CASK TRANSPORTER

Safe storage of fuel in the TranStor™ Storage System is provided by the removal of decay heat by convection, radiation and conduction from the irradiated material to the basket shell wall and the subsequent natural convection in the basket-storage cask annulus. Radiation exposure to site personnel is limited by the steel and concrete shielding. TranStor™ Storage System operation is totally passive and does not rely on any active systems. The radioactive materials are safely confined within the welded basket designed to withstand all normal and postulated accident conditions without a breach.

The handling equipment required to implement the TranStor™ Storage System must handle the TranStor™ Storage System within the conditions of cask use. This equipment includes an overhead crane at the reactor fuel pool, a transfer cask, miscellaneous slings and lifting fixtures, and a storage cask transfer system (cask transporter, air pad and trailer, etc.). All handling equipment is designed and tested to applicable regulatory and industrial standards.

1.2.4 Contents of Baskets

The TranStor™ Baskets can accommodate all types of uranium and mixed oxide LWR fuel including stainless steel clad assemblies. Only GE-XBR fuel is excluded because of its large size relative to the BWR basket sleeves. The lengths of the baskets vary to accommodate the various fuel assembly lengths. The basket design allows for storage of fuel with or without control components, burnable poison assemblies, thimble plugs, and neutron sources. Licensing analyses are conservatively based on bounding weights and axial dimensions.

The basket for PWR fuel assemblies has 24 cells to accommodate one of the following:

a) 24 assemblies,
b) 20 assemblies and 4 special cans of failed/partial fuel or fuel debris,
c) any intermediate combination (e.g., 22 fuel assemblies and 2 special cans).

The basket for BWR fuel has 61 cells to accommodate one of the following:

a) 61 assemblies with or without channels,
b) 53 assemblies and 8 special cans of failed/partial fuel,
c) any intermediate combination (e.g., 56 fuel assemblies and 5 special cans)

1.3 Identification of Agents and Contractors

The TranStor™ System design and license is owned by Sierra Nuclear Corporation (SNC). All design and fabrication activities including quality assurance services are performed by SNC. SNC has full responsibility and control over all design, analysis, and fabrication activities.

1.4 Generic Cask Arrays

A typical ISFSI storage pad layout is provided in Figure 1.4-1. This layout is for 69 casks and 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. 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 (or other suitable surface such as asphalt) must be capable of handling the transient loads from the storage cask transfer system (transporter, trailer, air pads, etc.) and the general loads of the stored cask. Furthermore, the pad can be constructed in phases to specifically meet utility required expansions.

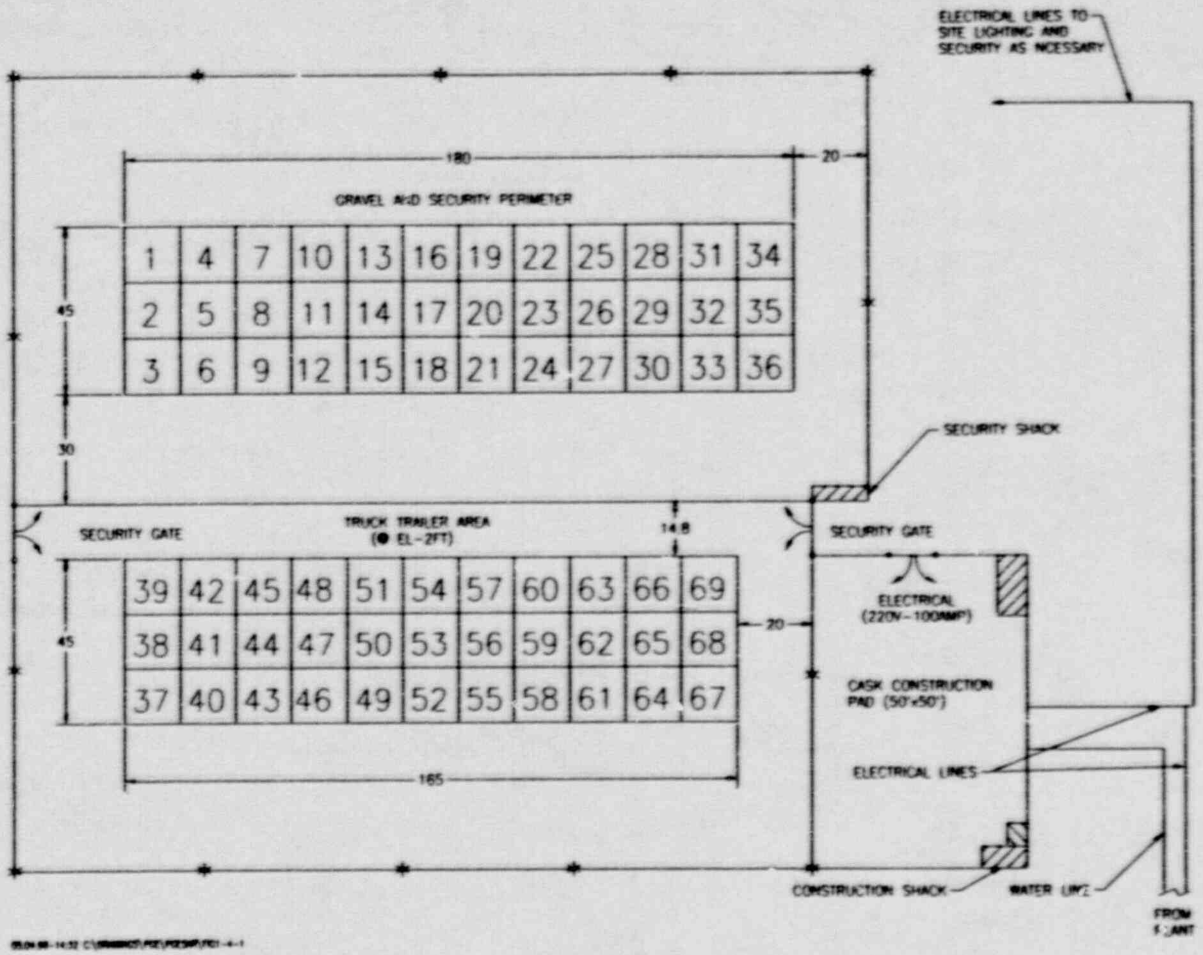


FIGURE 1.4-1

TYPICAL ISFSI PAD LAYOUT

1.5 References

1.1 Safety Analysis Report for the TranStor™ Shipping Cask System, SNC-95-71SAR, Rev. 0, Docket 71-9268, Sierra Nuclear Corporation.

1.2 Safety Analysis Report for the Ventilated Storage Cask System, PSN-91-001, Rev. 0A, Docket 72-1007, Pacific Sierra Nuclear Corporation.

1.3 M. McKinnon, et. al., "Performance, Testing, and Analyses of the VSC-17 Ventilated Concrete Cask", EPRI TR-100305, Electric Power Research Institute.

2.0 PRINCIPAL DESIGN CRITERIA

2.1 Irradiated Fuel to be Stored

The design payload for the TranStor™ System is intact and failed fuel assemblies and fuel debris. The major fuel characteristics are listed in Table 2.1-1. The main physical parameters of concern are the fuel assembly length, weight, and envelope (cross-sectional dimension). These parameters define the mechanical and structural design of the TranStor™ System. The structural analysis was performed with the heaviest fuel and the longest fuel with control components so that the calculated stresses will bound any assembly to be stored in the basket. The other important characteristics that define the thermal load and radiological source of the loaded cask are the initial enrichment, burnup, and time since discharge from the reactor (cooling time). The technical specifications provided in Chapter 12.0 of this SAR ensure that all of the above parameters are within the design limits for any fuel allowed for loading into the cask.

The only thermal design limit of the fuel to be stored in the TranStor™ System is that the maximum heat generation rate per assembly be such that the fuel cladding and concrete temperatures are below their corresponding allowables. Decay heat limits of 1.083 kW per PWR assembly and 0.426 kW per BWR assembly are specified as the design bases for the system. These heat generation limits can be met by an infinite number of burnup level and cooling time combinations.

The principal radiological design criteria are that the gamma and neutron doses are such that their sum is within the cask design limits. Since the gamma and neutron sources are different functions of burnup and cooling time, a wide variety of their combinations can make the total dose rate meet the established limits. Chapter 5.0 presents the results of dose rate calculations for several burnup and cooling time combinations for both PWR and BWR fuel. These combinations are selected to yield the heat load greater than or equal to the power limits specified above and the results demonstrate that they all produce dose rates well within the specified design limits. As discussed in Chapter 5.0, any PWR and BWR fuel that meets the assembly power limit will also produce surface dose rates that are the same or lower than calculated herein. Therefore, as long as a fuel assembly meets the heat load requirement, it is bounded by the radiological analysis presented in this report.

Theoretically, very high enrichments can be accommodated by the TranStor™ System for storage because fuel pool boron credit is allowed by the 10 CFR Part 72 regulations. However, the TranStor™ basket is transportable and the fuel enrichment is limited by 10 CFR Part 71 which does not rely on either burnup or boron credit. Hence, the maximum enrichment for various fuel assemblies is identical to that of the TranStor™ transportation SAR (Reference 1.1). The list is presented in Section 12.2.2.1 of this report.

TABLE 2.1-1

PRINCIPAL DESIGN PARAMETERS FOR FUEL ASSEMBLIES TO BE STORED IN A TranStor™ STORAGE CASK

PHYSICAL PARAMETERS [4]:	BWR	PWR
Assembly width, in [1]	8.536	5.518
Assembly weight, lb [2] (with control component or channel)	1680	700
Overall length, in [3] (with control component or channel)	178	176
Nominal Weight of uranium/assembly, lbs	1035	433
Fuel rod clad material	Zircaloy or Stainless Steel	Zircaloy or Stainless Steel
No. of assemblies/TranStor™ Basket	24	61
THERMAL CHARACTERISTICS :		
Maximum heat generation per assembly, kW	1.083	0.426
RADIOLOGICAL CHARACTERISTICS:		
Enrichment (w/o ^{235}U)	Varies with assembly type (see Section 12.2.2.1)	
Burnup and cooling time combination	Controlled by maximum heat generation level (See Chapter 5.0)	

1 BWR width includes channel.
2 Maximum BWR assembly weight is GE 8 x 8 fuel assembly with channel. Maximum PWR assembly weight is B&W 15 x 15 fuel assembly with control component.
3 Maximum assembly length is CE System 80 16 x 16 fuel assembly.
4 Data from Reference 2.1

Fuel assemblies with identified or suspected gross failures of fuel rod cladding will be stored within individual cans that allow for water draining and vacuum drying. These will be stored in the 4 large corner storage cells available in the PWR baskets or in any location in the BWR basket.

Fuel debris consisting of loose fuel pellets will be either consolidated in partial assemblies or stored within sealed individual cans that allow for water draining and vacuum drying as well as for retention of loose fuel pellets. For the purpose of criticality control, the quantity of loose fuel debris per basket is limited to less than 10 kilograms (Reference 1.1).

2.2 Design Criteria For Environmental Conditions And Natural Phenomena

The TranStor™ 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 ANSI 57.9. Appropriate combinations of normal, off-normal, and accident loadings are also defined in the design criteria (Section 2.2.6).

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 all annual average temperatures at reactor sites in the United States. Indeed, 75°F bounds all regions of the United States except the Florida Keys and Hawaii. The following list shows a representative selection of sites in the hot regions of the country (Reference 2.2):

Location	Annual Average Temperature (°F)
Columbia, SC	63
Mobile, AL	68
New Orleans, LA	68
Miami, FL	75
Brownsville, TX	73
Yuma, AZ	74
Albuquerque, NM	56
Las Vegas, NV	66

The 75°F normal temperature was used to evaluate long-term fuel degradation and 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 4.0. The thermal stress evaluation used to define the normal operating thermal stress load (T_0) is provided in

Chapter 3.0. 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 was also evaluated to show compliance with the maximum heat load accident case required by ANSI-57.9. Furthermore, the cases of complete and half-blockage of the air inlets were also considered. Thermal analyses for the above described cases are presented in Chapter 11.0. The ambient temperature design conditions are further discussed in Chapter 4.0. The case with a maximum temperature gradient is chosen for the thermal stress analysis presented in Chapter 3.0.

2.2.2 Tornado and Wind Loadings

The TranStor™ Storage 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 characteristics (consistent with Regulatory Guide 1.76) are presented in Table 2.2-1. The design basis tornado missiles (consistent with NUREG 0800, Spectrum I) are described in Table 2.2-2. These tornado and tornado missile parameters were used to determine the resulting loads on the storage cask and assess any damage that could be caused. A full evaluation of the tornado event is presented in Chapter 11.0. All missiles were assumed to impact in a manner that produces the maximum damage to the storage cask.

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 TranStor™ Storage Cask has 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 that 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 storage cask can safely withstand flood water levels that submerge the cask and generate water stream velocities up to 24.6 ft/sec. The analyses that support these design bases are provided in Chapter 11.0. Site specific safety reviews will need to confirm that flood parameters do not exceed the values shown in Chapter 11.0.

TABLE 2.2-1

TORNADO AND WIND LOADINGS

Environmental Condition	Limits
Rotational Wind Speed, mph	290
Translational Speed, mph	70
Maximum Wind speed, mph	360
Radius of Max. Wind Speed, ft.	150
Pressure Drop, psi	3.0
Rate of Pressure Drop, psi/sec	2.0

TABLE 2.2-2

TORNADO GENERATED MISSILES

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

2.2.4 Snow And Ice Loadings

The criterion for determining design snow loads is based on ANSI A58.1, Chapter 7.0. Flat roof snow loads apply and are calculated from the following formula:

$$p_f = 0.7C_eC_tIp_g$$

where:

p_f = Flat roof snow load (psf)
C_e = Exposure factor = 0.8
C_t = Thermal factor = 1.0
I = Importance factor = 1.2
p_g = ground snow load, pounds per square foot = 100 psf

The numerical values of C_e, C_t, 1, and p_g are obtained from Tables 18, 19, 20, and Figure 7, respectively, of ANSI A58.1.

The exposure factor accounts for wind effects. The storage cask is assumed to have a site location typical for siting category A, which is defined to be a "windy area with roof exposed on all sides with no shelter afforded by terrain, higher structures, or trees."

The thermal factor accounts for the thermal condition of the structure. The storage cask is classified as a heated structure.

The importance factor accounts for the importance of buildings and structures in relation to public health and safety. The storage cask is conservatively classified as Category III which results in the highest importance factor.

Ground snow loads for the contiguous United States are given in Figures 5, 6, and 7 of ANSI 58.1. A worst case value of 100 pounds per square foot was assumed.

Based on the above, the Flat Roof snow load is 67.2 psf. Stresses due to this load are calculated and used in the load combinations described in Chapter 3.0.

2.2.5 Seismic Design

The TranStor™ Storage Cask is shown not to tipover in an earthquake with peak accelerations of 0.75g in each of two orthogonal horizontal directions and 0.5g in vertical direction. These accelerations bound any site in the US and, therefore, use of the certified TranStor™ Storage System meets the geological and seismic criteria of 10 CFR 72 at any licensed power plant or interim storage facility in the United States.

The analysis of a seismic event is presented in Chapter 11.0. It employs the finite element modeling of the cask geometry and boundary conditions and numerical integration of the dynamic equation. A representative seismic time history developed from the response spectrum in Reg. Guide 1.60 was used as the excitation input for the model. Since the cask is shown not to tipover during any earthquake, the stresses due to the seismic event are insignificant.

Nevertheless, the cask tipover event is analyzed as a bounding condition. Both cask and basket are shown to withstand this hypothetical event without loss of integrity.

2.2.6 Combined Load Criteria

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

Normal	-	Dead Weight, Pressure, Handling, Thermal, Snow, Winds, etc.
Off-Normal	-	Off-Normal Severe Environmental Conditions, Surface Contamination, Interference During Basket Lowering From Transfer Cask to Storage Cask, Blockage of One-Half of Air Inlets, Off-Normal Handling
Accident	-	Complete Blockage of Air Inlets, Maximum Heat Load, Fuel Pin Rupture, Tornado (wind and missiles) Flood, Seismic, Explosion, Hypothetical Tipover

Normal loads due to pressure, temperature, and dead weight act in combination with all 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 - TranStor™ Storage Cask

The load combinations specified in ANSI 57.9 for concrete structures are used and shown in Table 2.2-3. These load combinations also meet the requirements of ACI 349. It must be noted that the full load combination of the codes is reduced because many loads specified in ANSI 57.9, Section 6.17 and ACI 349, Section 9.2 are not applicable as explained below:

a) Normal and accident pressure for the concrete cask is zero (0) since the cask is open to atmosphere. Therefore, P_a is not applicable (ACI 349).

b) There are no loads associated with attached piping and equipment because the concrete cask is a free-standing structure. Thus, R_o, R_a, Y_j, Y_r are not applicable (ACI 349).

c) There are no liquid or soil pressure on the cask (flood pressure is negligible). Thus, F and H are not applicable (ACI 349 and ANSI 57.9).

The ACI 349 design rules are used to demonstrate the structural adequacy of reinforced concrete in the storage cask. The steel liner and air ducts of the storage cask are stay-in-place forms and radiation shielding.

2.2.6.2 Load Combinations and Design Strength - TranStor™ Basket

The basket shell and basket internals are designed to the applicable requirements of ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsections NC and NG. The load combinations for all normal, off-normal and accident conditions and corresponding Service Levels are shown in Table 2.2-4. The table, therefore, defines the basket design and service loadings as required by the Code. Level A Service Limits are used for normal conditions, Level B and C Service Limits are used for off-normal conditions and Level D Limits are used for the accidents. The analytical methods allowed by the ASME Code are employed. Stress intensities caused by pressure, temperature, and mechanical loads are combined before comparing to ASME code allowables.

In addition, the basket is classified as a special lifting device and designed to the requirements of ANSI N14.6 and NUREG 0612. The lifting criteria are:

Maximum Principal stress during the lift (with 10% dynamic load factor)
$\leq$ (S_y/6 or S_u/10) for non-redundant load path or (S_y/3 or S_u/5) for redundant load path.

The structural design criteria are summarized in Table 2.2-5.

2.2.6.3 Design Strength - TranStor™ Transfer Cask

The transfer cask is a special lifting device and is designed and fabricated to the requirements of ANSI N14.6 and NUREG 0612 for the load path components. The design criteria are:

Maximum Principal stress during the lift (with 10% dynamic load factor)
$\leq$ (S_y/6 or S_u/10) for non-redundant load path or (S_y/3 or S_u/5) for redundant load path.

Load bearing members of the transfer cask are subjected to Charpy impact testing per ANSI N14.6, as discussed in Section 3.4.5.

TABLE 2.2-3

LOAD COMBINATIONS FOR THE TranStor™ STORAGE CASK

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

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

TABLE 2.2-4
BASKET LOAD COMBINATIONS

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

* Controlling load combinations for the Service Level. Note: Level B combination is bounded by Level A.

TABLE 2.2-5

STRUCTURAL DESIGN CRITERIA FOR STEEL COMPONENTS USED IN THE TranStor™ BASKET

	Component (Applicable Code)	Criteria
1.	Basket Normal Operation (ASME III, NC/shell/ and NG/internals/, 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 principal stress $\leq S_u/5$ or $S_y/3$ Non-redundant load path: max principal stress $\leq S_u/10$ or $S_y/6$
2.	Basket Off-Normal Operation (ASME III, NC/shell/ and NG/internals/, 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.	Basket Off-Normal Operation (ASME III, NC/shell/ and NG/internals/, Service Level C)	$P_m < 1.2\ S_m$ (shell), $< 1.5\ S_m$ (sleeve) $P_L + P_b < 1.8\ S_m$ (shell), $< 2.25\ S_m$ (sleeves)
4.	Basket Accident Conditions, (ASME III, NC/shell/ and NG/internals/, Service Level D, NUREG/CR-6322)	$P_m \leq 2.4\ S_m$ or $0.7\ S_u$ (whichever is less) $P_L + P_b \leq 3.6\ S_m$ or $1.0\ S_u$ (whichever is less) Buckling interaction ratios < 1
5.	Brittle Fracture (ASME III, NC/shell/ and NG/internals/)	Selection of structural material with adequate toughness. Control by operating procedures based on minimum temperature. Carbon steel below 5/8" in thickness and stainless steel are exempt from fracture toughness testing and requirements.

2.2.7 Site Criteria

While the TranStor™ Storage System is generically usable without further licensing actions at any power reactor site holding a 10 CFR 50 license or at an MRS facility holding a 10CFR72 license, there are still some site characteristics and factors that must be evaluated by the potential user.

The siting criteria for an ISFSI using the TranStor™ System is provided in 10 CFR 72, Subpart E which, via reference, includes Appendix A to 10 CFR 100. The main criteria examined are the natural phenomena discussed in the earlier parts of this section and the off-site doses. This examination must ensure that the site is bounded by the analysis presented herein and that the dose rates are below the controlled area boundary limits for both normal operation (25 mrem/year whole body) and accident conditions (5 rem whole body or any organ).

The TranStor™ System is designed to bound the natural phenomena at any site in the US (see Sections 2.2.1 through 2.2.5 for the criteria and various subsections of Chapter 11.0 for the analyses). The dose rates for controlled area boundaries vary with the number of casks, array configuration, and age of fuel at the time of placement into dry storage. With optimal array, additional shielding, and/or accounting for real maximum expected occupancy factors, the TranStor™ System can meet the normal operating and accident dose limits at distances as close as 20 ft.[1] However, distances to the controlled area boundary of less than 100 meters would require a site-specific exemption under 10 CFR 72.7.

(1) Since security requires a 20 ft clear space between the casks and the fence, 20 ft is the minimum feasible distance to the controlled area boundary.

2.3 Safety Protection Systems

2.3.1 General

The TranStor™ System is designed for safe long-term storage of spent nuclear fuel. Its components will withstand 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 TranStor™ system to assure safe long-term fuel storage and subsequent off site shipping are:

Basket

1. Leak-tight/multi-pass welds on basket structural lid, shield lid and bottom-end plate.
2. Thick shield and structural lids to minimize radiation exposure during basket closure.
3. Design of basket body and internals to withstand all postulated accidents.
4. Design of basket body and internals to provide adequate heat transfer and keep fuel within its temperature limits.
5. Design of basket body and internals to reduce the dose to personnel and general public.
6. Flux traps and poison sheets to maintain the system subcritical without burnup or boron credit.
7. Use of inflatable seals or a clean water flow to minimize contamination of the basket exterior by fuel pool water.

Storage Cask

1. Thick steel and concrete walls and non-planar ducts to provide adequate shielding
2. High stiffness and low center of gravity to provide stability during an earthquake.
3. Natural flow cooling to provide adequate heat transfer.
4. Rugged construction to protect the basket from weather and postulated events.

Transfer Cask

1. High safety factors for lifting to meet the single-failure proof criteria of NUREG 0612.
2. Thick sandwich walls to reduce the occupational dose rates to acceptable levels in accordance with the ALARA principle.

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

2.3.2 Protection By Multiple Confinement Barriers And Systems

2.3.2.1 Confinement Barriers and Systems

The radioactivity that the TranStor™ Storage System must confine originates from the stored spent fuel assemblies and the contaminated water in the fuel pool where the basket loading is conducted.

The TranStor™ System provides multiple barriers to confine the radioactive fuel as listed in Table 2.3-1. For the complete fuel assemblies, the cladding material provides the first level of confinement for the fission products. Cladding is not considered a confinement barrier for failed fuel[1] assemblies and fuel debris. Failed fuel assemblies and fuel debris are contained within special inner cans that retain any loose material. These cans are secured and vacuum dried. The basket is sealed by the shield lid weld. The basket structural closure is accomplished by multi-pass welding that is tested to assure a leak tightness of 10^{-4} standard cubic centimeters (scc) per sec at a pressure differential of 0.5 atmospheres. The longitudinal and girth welds and bottom welds of the basket shell are radiographed at the fabrication shop.

The basket is designed to withstand all design events (including a postulated cask tipover and a drop inside the shipping cask) without damaging the stored fuel or breaching confinement. A detailed evaluation of the drop accident is included in Chapter 11.0.

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

---

(1) Failed fuel is an assembly that is structurally deformed or has grossly damaged cladding or spacers to the extent that special handling may be required.

TABLE 2.3-1

RADIOACTIVITY CONFINEMENT BARRIERS AND SYSTEMS OF THE TranStor™ SYSTEM

Radioactivity	Confinement Barriers and Systems
Contaminated Pool Water	1. Basket-transfer cask annular gap has positive flow of clean water or inflatable seals while submerged in the pool to prevent pool water from contacting basket exterior. 2. Draining and vacuum drying of basket occurs using the plant off-gas system. 3. Use of transfer cask during basket transfer to storage cask.
Intact Fuel Assemblies, Failed Fuel Assemblies, Fuel Debris	1. Fuel Cladding. 2. Failed Fuel or Fuel Debris canister. 3. Basket Shell. 4. Seal Welded Basket Shield Lid. 5. Multi-pass Seal Welded Basket Structural Lid. 6. Multi-pass Seal Welded Basket Bottom End Plate.

2.3.2.2 Ventilation Off-Gas

The TranStor™ Storage System is passively cooled by radiant and natural convection heat transfer at the outer surface and natural convective heat transfer in the basket-storage cask annulus. The basket is welded and helium tested to ensure leak tightness. There are no radioactive releases during normal operations. Also, there are no credible accidents that would cause significant releases of radioactivity from the TranStor™ Storage System and; hence, there are no off-gas system requirements for the TranStor™ Storage System during normal storage operation. The only time an off-gas system is required is during the cask loading phase when the off-gas system at the licensed reactor site or MRS will be used.

2.3.3 Protection By Equipment And Instrumentation Selection

2.3.3.1 Equipment

The TranStor™ Storage 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 TranStor™ Storage Cask is designed to withstand the failure of any of these components.

2.3.3.2 Instrumentation

The TranStor™ Storage System does not require any instrumentation to assure the safe storage of irradiated fuel.

2.3.4 Nuclear Criticality Safety

The TranStor™ Storage System is designed to maintain nuclear criticality safety (sub-criticality) under all applicable regulatory conditions. These conditions include normal handling and storage conditions, off-normal handling and component malfunctioning, and hypothetical accident conditions.

The principal criticality design criteria is that k_{eff} remain below 0.95 during normal operation and accident conditions.

2.3.4.1 Control Methods for Prevention of Criticality

The basket criticality analysis addressed in Chapter 6.0 does not rely on either fuel burnup or pool boron credit. Subcriticality is assured by the basket materials and geometry. The design uses optimum moderator density and the most reactive fuel assembly locations. In addition to the neutron absorption properties of the fuel and the steel basket, poison sheets are included in each of the sleeves adjacent to other fuel assemblies. The PWR basket geometry allows for the formation of "flux traps" if the basket is flooded with the moderator. Other methods of storage system criticality control include administrative controls of fuel enrichment to ensure that fuel placed in the basket meets design requirements.

To prevent loose fuel pellets from entering flux traps or other fuel assembly sleeves, they are contained within failed fuel cans. If loose fuel pellets cannot be securely packaged within a partial fuel assembly, then the loose fuel pellets shall be the only fissile material in the individual fuel debris can.

2.3.4.2 Error Contingency Criteria

The values of k_{eff} include error contingencies and calculational and modeling biases. k_{eff} equals the calculated k_{eff}, plus criticality code bias, plus two times code uncertainty. k_{eff} has been evaluated for optimum internal moderator, infinite fuel array, optimum assembly position within the sleeve, and 75% of actual boron density in the poison sheets.

2.3.4.3 Verification Analyses

The criticality analysis was performed using industry accepted codes. These codes were validated in accordance with the SNC Quality Assurance Program.

2.3.5 Radiological Protection

2.3.5.1 Access Control

Access to a TranStor™ System installation site is controlled by a peripheral fence to meet 10 CFR 72 requirements. The details of access control and the division of the installation site into radiation protection areas will be addressed in a cask-user's documentation of meeting the conditions of cask use, the 10 CFR 50.59 review, or other user prepared documentation (such as security plan revision, radiological protection plan, etc.).

2.3.5.2 Shielding

The TranStor™ storage cask and other components are designed to minimize radiological dose rates to the general public and plant personnel. The design dose limits one meter from the cask surface are selected as 50 mrem/hr at the side and 300 mrem/hr at the cover lid centerline. The analyses showing the calculated dose rates for TranStor™ components loaded with the design basis payload are included in Chapters 5.0 and 11.0. The calculated dose rates (approximately 10 mrem/hr at the side and 135 mrem/hr at the top) are well within their design limits and the actual measured data shows even lower values. Furthermore, 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 mrem per year during normal operations and 5,000 mrem from any design basis accident. The radiation emanating from the TranStor™ Storage Cask results in much lower doses.

Chapter 10.0 presents the collective doses associated with cask operation and maintenance. Personnel radiation exposure during handling and closure of the basket is minimized by the following steps.

1. Placing the shield lid on the basket while the transfer cask and basket remain in the fuel pool.
2. Decontaminating the transfer cask exterior prior to draining the basket.
3. Draining the basket while still housed in the transfer cask.
4. Using portable shielding as necessary and available.
5. Using the semi-automatic welding equipment that is remotely operated to close the basket.
6. Placing a shielding ring over annular gap between the storage cask and basket.
7. Swiping the storage cask exterior for contamination prior to leaving the auxiliary building.

Based on operational experience with the VSC-24 system, the actual personnel exposure is substantially lower than calculated.

2.3.5.3 Radiological Alarm Systems

There are no radiological alarms required on the TranStor™ Storage System. Justification for this is provided in analysis in Chapter 5.0 (Shielding), 10.0 (Radiological Protection) and 11.0 (Accident Analysis).

2.3.6 Fire And Explosion Protection

No significant fires are expected at the ISFSI. The major transient combustible used within the ISFSI would be the gasoline or diesel fuel in towing vehicles used to move the casks. Normally, these vehicles would not be located at the ISFSI. When these vehicles were in use, they would be accompanied by personnel who would detect and suppress the small fires associated with fuel leaks. The ISFSI is protected from industrial and forest fires by the distance between combustibles and the ISFSI casks and by the open areas surrounding the ISFSI. Therefore, fires would not be credible at most ISFSI sites.

Nevertheless, the TranStor™ Storage System design is highly resistant to the effects of fire. 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.

Likewise, no explosions of any significance are possible at the ISFSI site. However, the cask resistance to explosion overpressure is evaluated in Chapter 11.0. As demonstrated by the analysis, the cask can withstand any potential explosion that could occur at an industrial facility located reasonably close to the ISFSI.

During basket loading process, there is a potential for buildup of hydrogen gas in the air space at the top of the basket. Hence, appropriate measures described in Chapter 8.0 of this report are taken to assure that the air space is properly vented so as to eliminate the possibility of hydrogen ignition during welding of the shield lid. The generation of hydrogen gas is a fairly slow process and occurs only when water is present in the basket. After the basket is drained and dried, no further generation of hydrogen gas is possible, and the potential for ignition no longer exists.

2.4 Decommissioning Considerations

The first step in decommissioning the TranStor™ Storage System is to move the fuel. This can be done in a number of ways. Various potentials are discussed in References 2.3 and 2.4.

The baseline decommissioning plan for a TranStor™ Storage System site is to transfer the basket into a TranStor™ Shipping Cask and ship the basket and the empty storage cask to a federal or private storage/disposal/reprocessing facility. This decommissioning method involves the least burden on the utilities because it avoids opening the basket and re-handling of fuel assemblies, minimizes radiation dose to workers, and provides a usable storage cask to the downstream facility. However, this is only one of many alternatives, as discussed in References 2.3 and 2.4.

If the worst case scenario evolves and the downstream facility is not available, the TranStor™ Storage System would be unloaded (in the sequence that is essentially the reverse of loading) in the fuel pool. After that the storage cask and basket could be reused for other on-site waste storage or disposed of in a normal landfill (the storage cask) and at a low-level waste burial facility (the basket). Because the basket exterior is clean and doesn't contact the storage cask interior, no contamination of the storage cask is anticipated. However, should contamination be present, it could be washed off the coated interior surface or removed by abrasive.

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 and consists of fast neutrons with energy around 1 Mev. This is ten orders of magnitude lower than the thermal flux around a reactor (10^{13} n/-sec cm^2), hence, activation is insignificant.

The basket 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, basket flushing could be performed to reduce most of the interior contamination. It is not expected that large quantities of crud would remain in the basket, but should quantities sufficient to create a hazard to site workers be present, the cask interior could be further decontaminated using one of the solvent based systems or other advanced means available at that time. After the basket 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 basket could be shipped (without cutting) to a burial site and buried. It could even be used to hold other waste for burial.

2.5 References

2.1 D.H. Malin and J.M. Viebrock, "Domestic Light Water Reactor Fuel Design Evolution," Volume III, DOE, ET/47912-3, US Department of Energy, Washington, D.C. (1981).

2.2 Commerce Department, "Comparative Climate Data for the US through 1991", National Oceanic and Atmospheric Administration, Washington, D.C. (1991).

2.3 J.V. Massey, J.H. Kessler, and A.J. McSherry, "Handling of Multi-assembly Sealed Baskets Between Reactor Storage and a Remote Handling Facility," EPRI-NP-6409, Electric Power Research Institute, Palo Alto, CA (1989).

2.4 J.V. Massey, et al., "Comparative System Economics of Concrete Cask for Spent Fuel Storage," EPRI TR-102415, Electric Power Research Institute, Palo Alto, CA (1993).

3.0 STRUCTURAL EVALUATION

This section describes the structural evaluation that was performed for the PWR and BWR fuel baskets, transfer cask and storage cask under normal operating conditions. The subsections below summarize the methodology and analysis techniques used and present the results.

3.1 Structural Design

3.1.1 Discussion

The TranStor™ Basket, Storage Cask and Transfer Cask comprise the principal structural components of the TranStor™ Storage System. Analyses of these components have been performed for the normal and hypothetical accident conditions specified in 10 CFR 72. Structural analyses reflect the system configurations during various stages of loading, handling, and relocation and have been performed for enveloping conditions representing the heaviest and longest contents (most conservative type and configuration of PWR and BWR fuel assemblies) as well as the highest temperatures and pressures.

The structural evaluations demonstrate that all components meet the applicable design criteria and are capable of safely storing irradiated fuel in compliance with 10 CFR 72 requirements.

3.1.1.1 TranStor™ Storage Cask

The TranStor™ Storage Cask is a reinforced concrete cylinder with an internal cavity and thick concrete and steel bottom. The internal cavity of the storage cask is formed by a thick cylindrical steel liner. The liner is a stay-in-place form and its thickness is determined only by shielding requirements. Normal weight 4000 psi concrete is used for the storage cask. Inner and outer rebar cages are formed by vertical hook bars and horizontal hoop bars. The reinforcing steel in the body is located to provide adequate strength for the design loading conditions specified in Section 2.0 of this report. The air flow path is formed by the channels at the bottom of the cask (air entrance), the air inlet ducts, the gap between the basket and the storage cask interior, and the air outlet ducts. The cask cover is a thick plate that provides additional shielding to reduce the skyshine radiation and protects the basket from the environment and postulated tornado missiles. The cover is bolted in place. Details of the storage cask are provided in the system drawings.

3.1.1.2 TranStor™ Basket

The principal structural members of the TranStor™ Basket include the shell, bottom plate, structural lid, and internal structure. Additional basket components include the shield lid, shield lid support ring, structural lid port cover plates, and their welds.

Enclosure of radioactive contents is provided by the shell, bottom plate, structural lid, shield lid, and by the structural lid port cover plates. Lifting provisions consist of the lugs for lifting the empty basket and hoist rings attached to the structural lid for lifting the loaded basket. The internal sleeve arrangement for the baskets is composed of individual steel cells and structural steel members. Details of the baskets are provided in the system drawings.

3.1.1.3 TranStor™ Transfer Cask

The transfer cask is a special lifting device designed and fabricated to the requirements of NUREG 0612 and ANSI N14.6. The site-specific heavy loads requirements will be followed to conform to the existing 10CFR50 license. The cask is also designed as a shielding bell to reduce the dose to site personnel in accordance with ALARA principles. Details of the transfer cask are provided in the system drawings.

3.1.2 Design Criteria

The TranStor™ System structural design criteria are specified in Section 2.2. The load combinations of normal, off-normal, and accident loadings have been evaluated per ANSI 57.9 for the storage cask (see Table 2.2-3) and per the 1992 edition with addenda through Summer 1994 Addenda of the ASME Boiler and Pressure Vessel Code, Section III, Division I components for the steel basket (see Tables 2.2-4 and 2.2-5). The 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 TranStor™ System are summarized in Table 3.2-1. This data is shown for the longest and heaviest versions of the TranStor™ System and, therefore, is conservative for other versions.

3.3 Mechanical Properties Of Materials

The mechanical properties of steels and concrete used in the structural evaluation of the TranStor™ System are presented in Tables 3.3-1 through 3.3-7.

TABLE 3.2-1
TranStor™ SYSTEM WEIGHTS AND CENTERS OF GRAVITY

ITEM DESCRIPTION	WEIGHT (lbs) PWR	WEIGHT (lbs) BWR	CENTER OF GRAVITY (inches above bottom) PWR	CENTER OF GRAVITY (inches above bottom) BWR
• Storage Cask Lid	1,235	1,235	N/A	N/A
• Basket Structural Lid	2,730	2,730	N/A	N/A
• Basket Shield Lid	7,470	7,470	N/A	N/A
• Transfer Cask Lid	400	400	N/A	N/A
• Basket (Empty, w/o Lids)	28,830	31,570	88.1	89.2
• Basket (Loaded w/Water and Shield Lid)	88,150	94,950	93.6	97.0
• Basket (Loaded, dry, w/Lids)	77,490	84,460	97.7	100.9
• Storage Cask (Empty, w/o Lid)	221,820	221,820	109.4	109.4
• Storage Cask & Basket (Empty, w/o Lids)	253,110	255,850	110.5	110.6
• Storage Cask & Basket (Loaded, w/Lids)	297,630	308,750	113.9	113.9
• Transfer Cask (Empty w/o Lid)	126,230	126,230	90.6	90.6
• Transfer Cask with Basket (Empty, w/o Shield Lid)	155,650	158,390	92.5	92.8
• Transfer Cask with Basket (Loaded, w/ water and Lid)	212,710	222,550	98.0	98.2
• Transfer Cask with Basket (Loaded, dry, w/ Lids)	200,160	211,290	99.0	99.2

TABLE 3.3-1
MECHANICAL PROPERTIES OF SA-240, TYPE 304 STAINLESS STEELS

Property (units)/ Temperature (°F)	-40	-20	+70	+200	+300	+400	+500	+750
Ultimate Strength[1] (ksi)	75.0	75.0	75.0	71.0	66.0	64.4	63.5	63.1
Yield Strength[2] (ksi)	30.0	30.0	30.0	25.1	22.5	20.8	19.4	17.3
Design Stress Intensity[3] (ksi)	20.0	20.0	20.0	20.0	20.0	18.7	17.5	15.6
Modulus of Elasticity[4] (ksi)	28.7E+3	28.7E+3	28.3E+3	27.6E+3	27.0E+3	26.5E+3	25.8E+3	24.4E+3
Alternating Stress[5] @10 cycles (ksi)	718.0	718.0	708.0	690.5	675.5	663.0	645.5	610.4
Alternating Stress[5] @10^6 cycles (ksi)	28.7	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Coefficient of Thermal Expansion[6] (in/in/°F)	8.13E-6	8.19E-6	8.46E-6	8.79E-6	9.00E-6	9.19E-6	9.37E-6	9.76E-6
Poisson's Ratio[7]	0.31							
Density[7]	497 lbm/ft^3 (0.288 lbm/in^3)							

1 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table U.

2 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table Y-1.

3 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table 2A.

4 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TM-1.

5 ASME Boiler and Pressure Vessel Code, Appendix I, Table I-9.1.

6 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TE-1.

7 Baumeister & Marks, "Standard Handbook for Mechanical Engineers", 7th Ed.

TABLE 3.3-2
MECHANICAL PROPERTIES OF SA-240, TYPE 304L STAINLESS STEEL

Property (units)/ Temperature (°F)	-40	+70	+200	+300	+400	+500	+750
Ultimate Strength[1] (ksi)	70.0	70.0	66.2	60.9	58.5	57.8	55.9
Yield Strength[2] (ksi)	25.0	25.0	21.4	19.2	17.5	16.4	14.7
Design Stress Intensity[3] (ksi)	16.7	16.7	16.7	16.7	15.8	14.8	13.3
Modulus of Elasticity[4] (ksi)	28.7E+3	28.3E+3	27.6E+3	27.0E+3	26.5E+3	25.8E+3	24.4E+3
Alternating Stress[5] @10 cycles (ksi)	718.0	708.0	690.5	675.5	663.0	645.5	610.4
Alternating Stress[5] @10^6 cycles (ksi)	28.7	28.3	27.6	27.0	26.5	25.8	24.4
Coefficient of Thermal Expansion[6] (in/in/°F)	8.13E-6	8.46E-6	8.79E-6	9.00E-6	9.19E-6	9.37E-6	9.76E-6
Poisson's Ratio[7]	0.31						
Density[7]	497 lbm/ft^3 (0.288 lbm/in^3)						

1 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table U.

2 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table Y-1.

3 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table 2A.

4 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TM-1.

5 ASME Boiler and Pressure Vessel Code, Appendix I, Table I-9.1.

6 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TE-1.

7 Baumeister & Marks, "Standard Handbook for Mechanical Engineers", 7th ed.

TABLE 3.3-3
MECHANICAL PROPERTIES OF A-36 FERRITIC STEEL

Property (units)/ Temperature (°F)	-40	+70	+200	+300	+400	+500	+700
Ultimate Strength[1] (ksi)	-	58.0	58.0	58.0	58.0	-	-
Yield Strength[2] (ksi)	36.0	36.0	32.8	31.9	30.8	29.1	25.9
Design Stress Intensity[3] (ksi)	19.3	19.3	19.3	19.3	19.3	19.3	17.3
Modulus of Elasticity[4] (ksi)	30.0E+3	29.5E+3	28.8E+3	28.3E+3	27.7E+3	27.3E+3	25.5E+3
Alternating Stress[5] @10 cycles (ksi)	580	570	557	547	536	528	493
Alternating Stress[5] @10^6 cycles (ksi)	12.5	12.3	12.0	11.8	11.5	11.4	10.6
Coefficient of Thermal Expansion[6] (in/in/°F)	5.02E-6	5.42E-6	5.89E-6	6.26E-6	6.61E-6	6.91E-6	7.41E-6
Poisson's Ratio[7]	0.29						
Density[7]	489 lbm/ft^3 (0.283 lbm/in^3)						

1 ASME Boiler and Pressure Vessel Code, Code Case N-71-16, Table 5

2 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table Y-1.

3 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table 2A.

4 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TM-1.

5 ASME Boiler and Pressure Vessel Code, Appendix I, Table I-9.1.

6 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TE-1.

7 Baumeister & Marks, "Standard Handbook. for Mechanical Engineers", 7th ed

TABLE 3.3-4
MECHANICAL PROPERTIES OF SA-516 GRADE 70 FERRITIC STEEL

Property (units)/ Temperature (°F)	-40	+70	+200	+300	+400	+500	+750
Ultimate Strength[1] (ksi)	70.0	70.0	70.0	70.0	70.0	70.0	69.3
Yield Strength[2] (ksi)	38.0	38.0	34.6	33.7	32.6	30.7	26.5
Design Stress Intensity[3] (ksi)	23.3	23.3	23.1	22.5	21.7	20.5	18.3
Modulus of Elasticity[4] (ksi)	30.1E+3	29.5E+3	28.8E+3	28.3E+3	27.7E+3	27.3E+3	24.8E+3
Alternating Stress[5] @10 cycles (ksi)	582	570	557	547	536	528	479
Alternating Stress[5] @10^6 cycles (ksi)	12.5	12.3	12.0	11.8	11.5	11.4	10.3
Coefficient of Thermal Expansion[6] (in/in/°F)	5.02E-6	5.42E-6	5.89E-6	6.26E-6	6.61E-6	6.91E-6	7.41E-6
Poisson's Ratio[7]	0.29						
Density[7]	489 lbm/ft^3 (0.283 lbm/in^3)						

1 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table U.

2 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table Y-1.

3 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table 2A.

4 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TM-1.

5 ASME Boiler and Pressure Vessel Code, Appendix I, Table I-9.1.

6 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TE-1.

7 Baumeister & Marks, 'Standard Handbook for Mechanical Engineers 7th ed

TABLE 3.3-5
MECHANICAL PROPERTIES OF A-500, GRADE C FERRITIC CARBON STEEL

Property (units)/ Temperature (°F)	-40	-20	+70	+200	+300	+400	+500	+700
Ultimate Strength[1] (ksi)	62.0	62.0	62.0	62.0	62.0	62.0	-	-
Yield Strength[2] (ksi)	50.0	50.0	50.0	45.6	44.3	42.9	40.4	36.0
Design Stress Intensity[3] (ksi)	20.7	20.7	20.7	20.7	20.7	20.7	20.7	20.7
Modulus of Elasticity[4] (ksi)	29.5E+3	29.5E+3	29.5E+3	28.8E+3	28.3E+3	27.7E+3	27.3E+3	25.5E+3
Alternating Stress[5] @10 cycles (ksi)	-	-	570	557	547	536	528	493
Alternating Stress[5] @10^6 cycles (ksi)	-	-	12.3	12.0	11.8	11.5	11.4	10.6
Coefficient of Thermal Expansion[6] (in/in/°F)	6.50E-6	6.50E-6	6.50E-6	6.67E-6	6.87E-6	7.07E-6	7.25E-6	7.59E-6
Poisson's Ratio[7]	0.29							
Density[7]	489 lbm/ft^3 (0.283 lbm/in^3)							

1 ASME Boiler and Pressure Vessel Code, Code Case N-71-16, Table 5

2 ASME Boiler and Pressure Vessel Code, Code Case N-71-16, Table 3.

3 Calculated as the lesser of $S_u/3$ or $2S_y/3$ per ASME Code

4 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TM-1.

5 ASME Boiler and Pressure Vessel Code, Appendix I, Table I-9.1.

6 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TE-1.

7 Baumeister & Marks, "Standard Handbook for Mechanical Engineers", 7th ed

TABLE 3.3-6
MECHANICAL PROPERTIES OF A-588 FERRITIC STEEL

Property (units)/ Temperature (°F)	-40	-20	+70	+200	+300	+400	+500	+700
Ultimate Strength[1] (ksi)	70.0	70.0	70.0	70.0	70.0	70.0	70.0	70.0
Yield Strength[2] (ksi)	50.0	50.0	50.0	47.5	45.6	43.0	41.8	37.9
Design Stress Intensity[3] (ksi)	23.3	23.3	23.3	23.3	23.3	23.3	23.3	23.3
Modulus of Elasticity[4] (ksi)	29.5E+3	29.5E+3	29.5E+3	28.8E+3	28.3E+3	27.7E+3	27.3E+3	25.5E+3
Coefficient of Thermal Expansion[5] (in/in/°F)	5.02E-6	5.09E-6	5.42E-6	5.89E-6	6.26E-6	6.61E-6	6.91E-6	7.41E-6
Alternating Stress[6] @10^6 cycles (ksi)	12.3	12.3	12.3	12.0	11.8	11.5	11.7	10.6
Poisson's Ratio[7]	0.29							
Density[7]	489 lbm/ft^3 (0.283 lbm/in^3)							

1 ASME Boiler and Pressure Vessel Code, Code Case N-71-16, Table 5

2 ASME Boiler and Pressure Vessel Code, Code Case N-71-16, Table 3.

3 ASME Boiler and Pressure Vessel Code, Code Case, N-71-16, Table 1.

4 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TM-1.

5 ASME Boiler and Pressure Vessel Code, Section II, Part D, Table TE-1.

6 ASME Boiler and Pressure Vessel Code, Appendix I, Table I-9.1.

7 Baumeister & Marks, "Standard Handbook. for Mechanical Engineers", 7th ed

TABLE 3.3-7

MECHANICAL PROPERTIES OF CONCRETE

Property (units)/ Temperature (°F)	70	100	200	300	400	500
Density (lb/ft^3)	145	145	145	145	145	145
Thermal Conductivity (BTU/hr ft°F)	----	0.87	----	0.82	0.82	0.78
Compressive Strength (psi)	4000	4000	4000	3800	3600	3400
Coefficient of Thermal Expansion (in/in/°F)	$5.5x10^{-6}$	$5.5x10^{-6}$	$5.5x10^{-6}$	$5.5x10^{-6}$	$5.5x10^{-6}$	$5.5x10^{-6}$
Modulus of Elasticity (psi)	----	3.64x10 6	3.38x10 6	3.09x10 6	3.73x10 6	$2.43x10^6$

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 used in the TranStor™ System (e.g., steel, concrete, etc.) will not experience significant chemical, galvanic, or other detrimental reactions. The only contact between dissimilar materials is on the inside of the basket which is coated and placed in an inert environment. All components open to ambient air are made of similar materials with essentially equal potential in the Galvanic Series of Metals and Alloys. In addition, they are protected from the atmosphere by coatings.

During the loading process, radiolytic breakdown of the fuel pool water and the chemical reaction between the water and basket coating can create the potential for generation of hydrogen gas in the basket. The possible negative effects associated with the presence of hydrogen are mitigated by venting the air space in the basket and sampling the gas prior to seal welding the shield lid (as described in Chapter 8.0 of this report). The potential for these reactions exists only when pool water is present in the basket. Once the basket is drained and dried, the hydrogen generation is terminated.

3.4.2 Positive Closure

The TranStor™ System employs a positive closure system that is composed of multi-pass seal welds at four locations: 1) Basket shield lid to shell; 2) Basket structural lid to shell; 3) Basket structural lid to basket shield lid at the penetration ports; and 4) Basket drain and port cover plates to structural lid. The closure welds are shown in the drawings. Welded closure ensures that the basket can not be inadvertently opened and eliminates the problem of seal deterioration during service. The basket lid and welds are helium leak checked to ensure leak tightness of better than 10^{-4} scc/sec. Leakage greater than this limit shall be cause for weld repair.

The storage cask cover is bolted in place and has provisions for tamper indication. The cask is not subject to vibration loads that could cause the closure bolts to loosen or fall out. No inadvertent opening of the cover is possible.

3.4.3 Lifting Devices

The TranStor™ System has separate provisions for lifting the storage cask, the basket, and the transfer cask. The storage cask may be lifted from below using air pads or from above using a transporter. It should also be noted that both the top and bottom lift are not important to safety since the cask is lifted only a few inches. The TranStor™ storage cask lifting components must be able to accommodate the weight of the storage cask when it is fully loaded (i.e., include basket with 24 PWR or 61 BWR fuel assemblies).

To support the storage or shipping cask loading operation, the basket design includes provisions for lifting from above. This is accomplished via eight hoist rings that are bolted to the basket structural lid. These devices are capable of safely handling the fully loaded weight of the basket so that the loaded and sealed basket can be lowered into the storage or shipping cask. No water is present in the basket during these transfer operations.

The transfer cask is lifted from above via two trunnions located near the top of the outer shell. The trunnions are steel forgings that extend radially from the transfer cask body. Each trunnion is welded to the inner and outer steel shells of the transfer cask wall with full penetration circumferential welds. The two trunnions are capable of accommodating the combined weight of the transfer cask and a fully loaded wet basket while meeting the NUREG 0612 requirements for single-failure proof devices.

3.4.3.1 TranStor™ Storage Cask Lift

The storage cask is typically lifted from below using air pads. This bottom lift is the normal lifting mode employed when moving the cask. It should also be noted that the lift is not important to safety since the air pads only raise the cask 3 inches. The lift accommodates the weight of the system when it is fully loaded in the heaviest configuration (i.e., cask, basket, and 61 BWR fuel assemblies with channels). The total load of 320,000 lbs is conservatively used in the analysis.

The adequacy of the cask lift is evaluated by calculating the bearing pressure on the cask bottom and comparing that to the allowable bearing pressure determined per ACI 349. The lift stress is calculated to be just 45.3 psi compared to the allowable of 2,380 psi. Thus, the air pad bearing stresses are negligible. No shear forces or bending moments will exist in the cask because the air pads effectively cover the whole bottom area. Hence, the concrete will not crush during a bottom lift of the cask.

Another lifting option for the cask employs installation of lifting lugs at the top and is used with the cask transporter. The lugs are anchored into concrete using steel reinforcement. The TranStor™ storage cask lug design is identical to the one presented in Appendix 5 of the VSC-24 SAR (Reference 1.2), except larger rebars are used for lifting due to the higher weight of the cask. Although this lift is not important to safety, the lug design provides the safety factors of 3.9 on yield and 5.5 on ultimate strength of material and, hence, meets the ANSI N14.6 respective safety factors of 3 and 5.

3.4.3.2 TranStor™ Basket Lift

The loaded basket is lifted from above using a set of 8 hoist rings attached to the top of the basket structural lid. The adequacy of the devices used for lifting the loaded basket is demonstrated by evaluating the hoist rings, the basket structural lid, the basket shell, and the lid-to-shell weld. The basket lifting qualification is performed for the heaviest weight basket and contents (which bounds the weights for all other basket configurations) at a conservatively selected enveloping temperature of 200 °F. The arrangement of 8 lifting points can be considered a redundant lift if only 4 points are designed to carry the load. Minimum factors of safety of 3 on material yield strength and 5 on material ultimate strength, along with a dynamic load increase factor of 10%, are used to satisfy NUREG-0612 requirements for redundant lifting. All lifting slings are also designed to the same criteria.

The hoist ring evaluation is based on the manufacturer's rated capacity. The rated capacity of each hoist ring is 24 kips with a factor of safety of 5. Thus, the breaking load for a ring is 120 kips in any direction. Four rings are designed to support the basket. The maximum load for one ring, including the dynamic amplification as described above, is $84.5 \times 1.1 / 4 = 23.2$ kips. Factoring this load by the sling angle (specified for the lifting operations) provides a factor of safety of 5.05 compared to the required factor of safety of 5.

In order to maintain the required factors of safety, a thread engagement of 1.95" is calculated to be required for the hoist rings. The structural lid thickness is sufficient to provide the required thread depth without breach of containment.

The evaluation of the structural lid, the basket shell in the vicinity of the lid connection, and the lid-to-shell weld is based on results of the ANSYS finite element model of the MSB-24 (Reference 1.2). This approach is justified because the geometries of two baskets (shell and lid thicknesses, bolt circle and basket diameters) are very similar. Classical formulas (Reference 3.2) are used in making the appropriate adjustments to account for the slight difference in diameters and applied loads. The highest stresses occur at the junction between the basket shell and structural lid. The calculated principal stress at this location gives factors of safety of 4.1 against yield and 12.5 against ultimate strength of the material. These factors are higher than the respective requirements of 3 and 5.

The above results demonstrate that the lifting devices meet the requirements of NUREG-0612 for the fully loaded basket. The empty basket is lifted using a pair of lifting lugs welded to the inner surface of the basket shell. This operation is done prior to fuel loading; thus, it is not important to safety and not addressed herein.

3.4.3.3 Transfer Cask Lift

Load-bearing components of the transfer cask used with the TranStor™ System are designed as non-redundant lifting devices with a factor of safety of 10 or greater on ultimate and 6 or greater on yield and include the dynamic load increase factor of 10%. Hence, the cask meets NUREG 0612 requirements for non-redundant load path lifts and can be considered a single failure proof device. Subsections below describe the methodology used in calculations. The results are summarized in Table 3.4-1.

The weights used for all of the lifting analyses described below bound the corresponding weights in Table 3.2-1. Therefore, the results are conservative.

3.4.3.3.1 Trunnions

Structural adequacy of the transfer cask trunnions is evaluated by modeling them as cantilevers and applying the weight of the loaded transfer cask. The resulting bending and shear stresses in the trunnion are combined to calculate the maximum principal stress and determine the corresponding safety factors.

3.4.3.3.2 Transfer Cask Wall

To evaluate the structural integrity of the transfer cask wall, an ANSYS finite element analysis was performed using the model shown in Figure 3.4-1. The model focuses on the transfer cask wall region near the trunnion because this is the most critical region. Only a quarter of the transfer cask is modeled due to symmetry. The ANSYS SOLID45 and SHELL63 3-dimensional elements are used for the trunnion and transfer cask shells respectively. No structural credit is taken for lead and neutron shield material.

NUREG 0612/ANSI N14.6 state that the safety factors of 6 and 10 only apply to stresses that would not be relieved by local yielding, i.e. general stresses. General stresses in shells of revolution are typically taken as the highest stress outside of the $\sqrt{RT}$ distance from a concentrated load. The maximum relevant principal stress in the transfer cask wall and the resulting safety factors are presented in Table 3.4-1.

In addition, the highest local stresses at the cask trunnion-to-shell junction are calculated to be 13.6 ksi. While no specific regulatory requirements are applicable for these local stresses, the safety factors of more than 3 on yield and 5 on ultimate strength are still provided.

TABLE 3.4-1
TranStor™ TRANSFER CASK LIFTING EVALUATION

Component		Cask Safety Factors	Minimum Safety Factors per NUREG 0612/ ANSI N14.6
Trunnions	yield	11.4	6
	ultimate	25.0	10
Shell	yield	6.6	6
	ultimate	10.1	10
Lower plate	yield	6.8	6
	ultimate	11.9	10
Lower weld	yield	13.0	6
	ultimate	22.8	10
Rail-to-shell	yield	7.5	6
weld	ultimate	13.5	10

Component		Stress / force	AISC allowable
Cover plate	bending	14.0 ksi	24.6 ksi
Bolts	tension	30.7 kips	34.6 kips

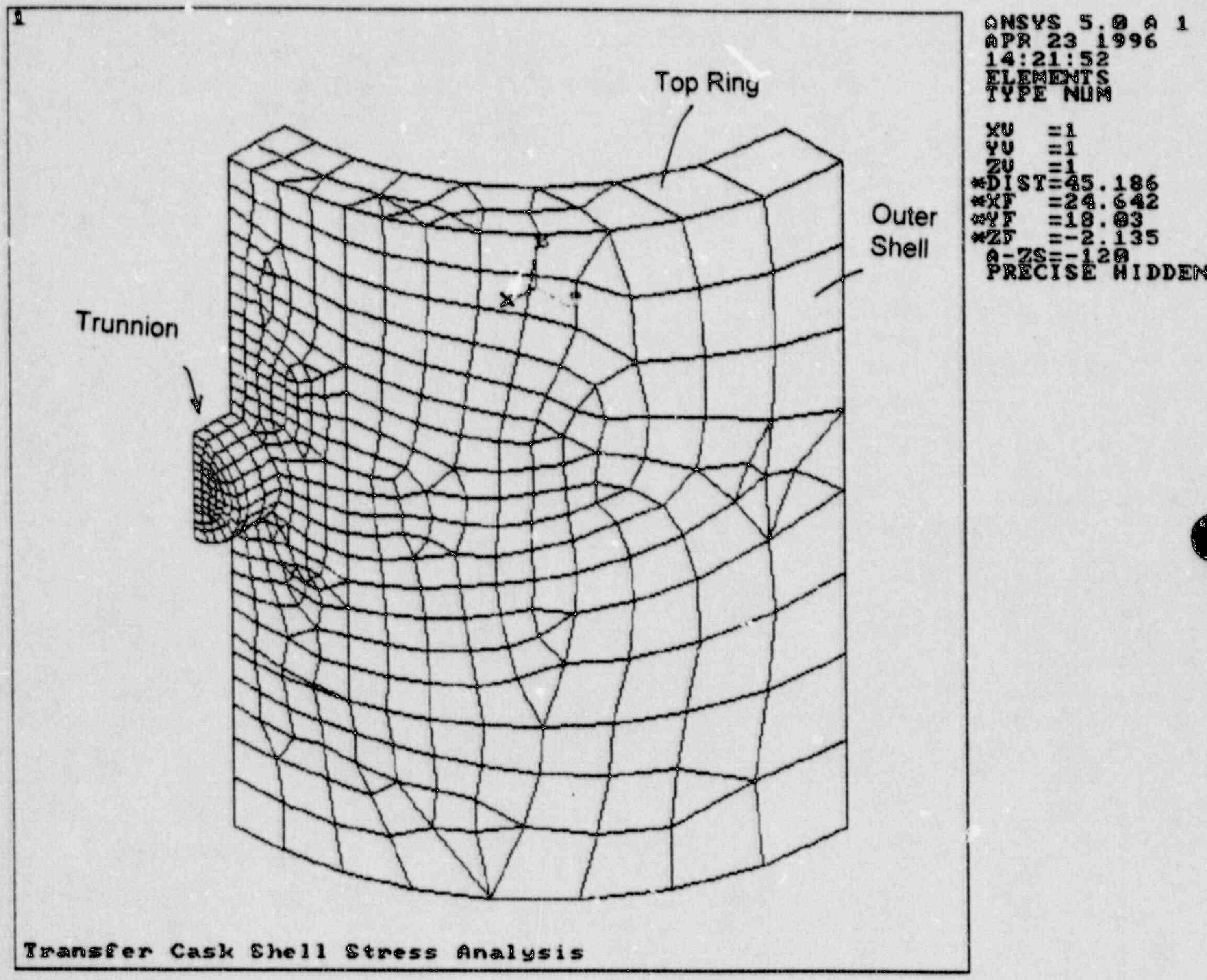


FIGURE 3.4-1
FINITE ELEMENT MODEL OF TRANSFER CASK SHELL AND TRUNNION

3.4.3.3.3 Shield Door Rail and Welds

The shield door rails must support the weight of a wet, fully loaded basket and the weight of the shield doors themselves. The rail consists of a plate welded to the bottom of a rectangular solid section of steel. This assembly is welded to the bottom plate of the transfer cask wall as shown in the TranStor™ System drawings.

The structural integrity of the rails is evaluated by considering the rail bottom plate and its welds. The maximum bending and shear stresses in the rail bottom plate are calculated by modeling it as a cantilever loaded by the weight of supported components. The maximum principal stress is determined by combining these two stresses and used to calculate the safety factors as required by ANSI N14.6.

The rail lower welds were evaluated by first determining the reactive forces, F_o and F_i, experienced by the outer and inner welds due to the applied load. These reactions are calculated from static equations for the force and moment where the moment is resisted by the F_o and F_i couple. The inner weld reaction is substantially higher and the safety factors presented in Table 3.4-1 are based on the stresses in this controlling weld.

3.4.3.3.4 Welds Between the Rails and Transfer Cask Shell

The load on the weld between the rail and transfer cask wall includes the loaded wet basket as well as the weight of doors and rails. Evaluation of the structural integrity of the rail upper welds is complicated by the curved geometry of the outer weld (see Figure 3.4-2). This complication stems from the fact that the load distribution between the two upper welds varies with position along the rail. The analysis is done using the standard methodology of treating the weld as a line and calculating its area and section modulus. These properties are calculated to be 93.4 inches and 278.0 in^2 respectively and are used with the reaction force and moment to determine the weld stresses and resulting safety factors. The results are presented in Table 3.4-1.

3.4.3.3.5 Top Cover Plate

The purpose of the top cover plate is to prevent inadvertent lifting of the basket out of the transfer cask in case of operator error or equipment malfunction. This feature is desirable to ensure against undue radiation exposure to nearby workers. Therefore, the cover plate must have sufficient strength to support the weight of the transfer cask (since an inadvertent basket lift would result in lifting the entire transfer cask by the cover plate). Since this would not be a normal lift but rather an accident condition, the NUREG 0612 safety factors do not apply and AISC Manual allowable stresses are used for the design of this component.

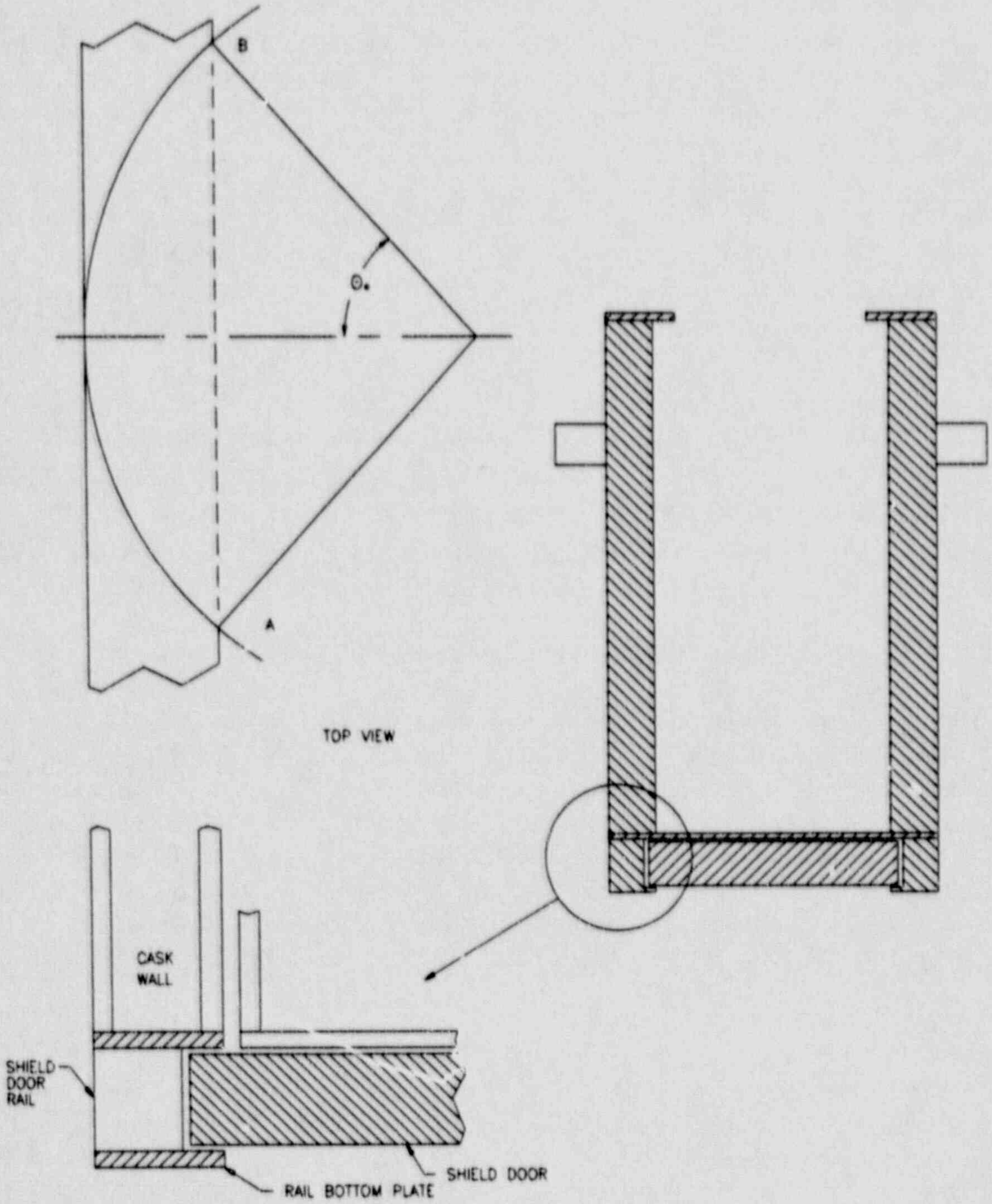


FIGURE 3.4-2
TRANSFER CASK SHIELDING RAIL WELD GEOMETRY

The stresses on the inner and outer edges of the plate are calculated using classical formulas for circular plates with a cutout in the center (Reference 3.2). Stresses are calculated and compared to the AISC allowables as presented in Table 3.4-1.

3.4.3.3.6 Cover Plate Bolts

The load on a single bolt due to the reactive force caused by an inadvertent basket lift consists of tension required to balance the transfer cask weight plus a prying force that results from the applied moment. Both of these forces are calculated and added together before comparison to the AISC allowables.

As can be seen from Table 3.4-1, all stresses are below corresponding limits.

3.4.4 Transfer System Components Under Normal Operating Loads

3.4.4.1 Basket Analysis

The Basket Design and Service Conditions are defined in Table 2.2-4 (in accordance with ASME, Section NCA-2142). This section addresses only the Design (Service Level A) Loadings; other Service Loadings are presented in Chapter 11.0. Table 3.4-2 shows the applicability of different Design Loadings to different basket components. The pressures and temperatures are taken from Chapter 4.0 and presented in Tables 3.4-3 and 3.4-4 for the PWR and BWR versions, respectively. Based on review of those, the basket shell Design Pressure is 4 psig and the Design Temperature is 300 °F. The normal handling load is selected as $\pm$0.5g in all directions simultaneously (Table 1.2-1). Sections below discuss each individual load and their combination.

3.4.4.1.1 Basket Thermal Stress Analysis

Description of Analytical Method

The TranStor™ basket thermal stresses are evaluated by modeling the basket separately from the storage cask. This approach is valid since these components are not structurally coupled and the basket is free to thermally expand or contract relative to the cask. Furthermore, the radial and axial gaps in the basket internals are sized such that differential thermal expansion does not cause interaction between the main frame, sleeves, and shell, hence, the basket shell and internal components are independent and can also be analyzed separately. The worst-case differential

TABLE 3.4-2

BASKET DESIGN LOADINGS

LOAD	COMPONENTS	
	Basket Pressure Boundary	Basket Internals
Dead Weight	X	X
Thermal	X	X
Internal Pressure	X	
Handling Load	X	X

TABLE 3.4-3

SUMMARY OF MAXIMUM PWR TranStor™ SYSTEM TEMPERATURES

	Analysis Case					
	1	2	3	4	5	6
Solar heat load	no	no	yes	yes	no half 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	87	-31	141	166	89	94
· concrete/liner interface	204	60	239	270	228	283
· gradient through wall	117	91	98	104	139	189
· Basket outer surface	287	177	313	338	305	358
· max fuel	609	511	633	655	625	673
· gradient through basket	322	334	320	317	320	315
Internal differential pressure (psig)	-2.3	-3.7	-1.9	-1.6	-2.1	-1.4

TABLE 3.4-4

SUMMARY OF MAXIMUM BWR TranStor™ SYSTEM TEMPERATURES

	Analysis Case					
	1	2	3	4	5	6
Solar heat load	no	no	yes	yes	no	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	87	-31	141	166	89	94
· concrete/liner interface	204	60	239	270	228	283
· gradient through wall	117	91	98	104	139	189
· Basket outer surface	287	1[illegible]7	313	338	305	358
· max fuel	676	575	701	724	693	743
· gradient through basket	389	398	388	386	388	385
Internal differential pressure (psig)	-1.8	-3.3	-1.5	-1.2	-1.6	-0.9

thermal expansion between the shell and internals is calculated to be substantially less than the gaps provided in the design.

Analysis of the basket for the effects of differential thermal expansion loads is performed using a 2-D axisymmetric model of the basket shell, structural lid, shield lid, and bottom plate. This model is shown in Figure 3.4-3. The basket internals are not included for the reasons described above.

In addition, for the PWR basket a three-dimensional finite element model is used to evaluate the effect of the structural tubes that are welded to the shell. This model is shown in Figure 3.4-4. The shell is represented using ANSYS 3-D thin shell elements and the frame members are modeled as beam elements. Main frame crosses are also included in this model to ensure that their gaps with the shell remain open and do not cause thermal stresses.

The BWR basket internal structure is composed of uncoupled sleeves and inserts, thus, no stresses due to differential thermal expansion are generated within internal members. As a result, only the shell assembly was analyzed for thermal stresses.

For the PWR basket, all internal components are also free to expand within the basket. However, a 3-D finite element model of a double cell is used as a bounding case to compute stresses in the cell members caused by thermal gradient. The model consists of plate elements which represent the two connected cells along their full length. The double cell model is shown in Figure 3.4-5.

The analysis for thermal expansion loads is performed for the thermal condition which causes the highest thermal gradient across the basket and therefore the highest thermal stresses in the basket components. From the information presented in Chapter 4.0, this condition corresponds to the maximum decay heat and the coldest ambient temperature with no solar insolation.

Basket Thermal Stress Analysis

The basket body and storage sleeve assembly thermal stresses were evaluated separately since the radial gap between the sleeve assembly and the basket shell structurally decouples the two members. This structural independence can be seen by examining the basket geometry at its stress free temperature of 70 °F (see TranStor™ System drawings). The nominal radial gap between the storage sleeve assembly structural support and the basket shell is 0.65 inches for the PWR and 0.525 inches for the BWR. These gaps are sufficient to allow unrestrained storage sleeve thermal expansion relative to the shell.

For the overall evaluation of the thermal stresses in the basket, the temperature distribution for the -40°F ambient case was used. The -40 °F condition is bounding because it causes the highest thermal gradients in the basket structure as can be seen from Tables 3.4-3 and 3.4-4.

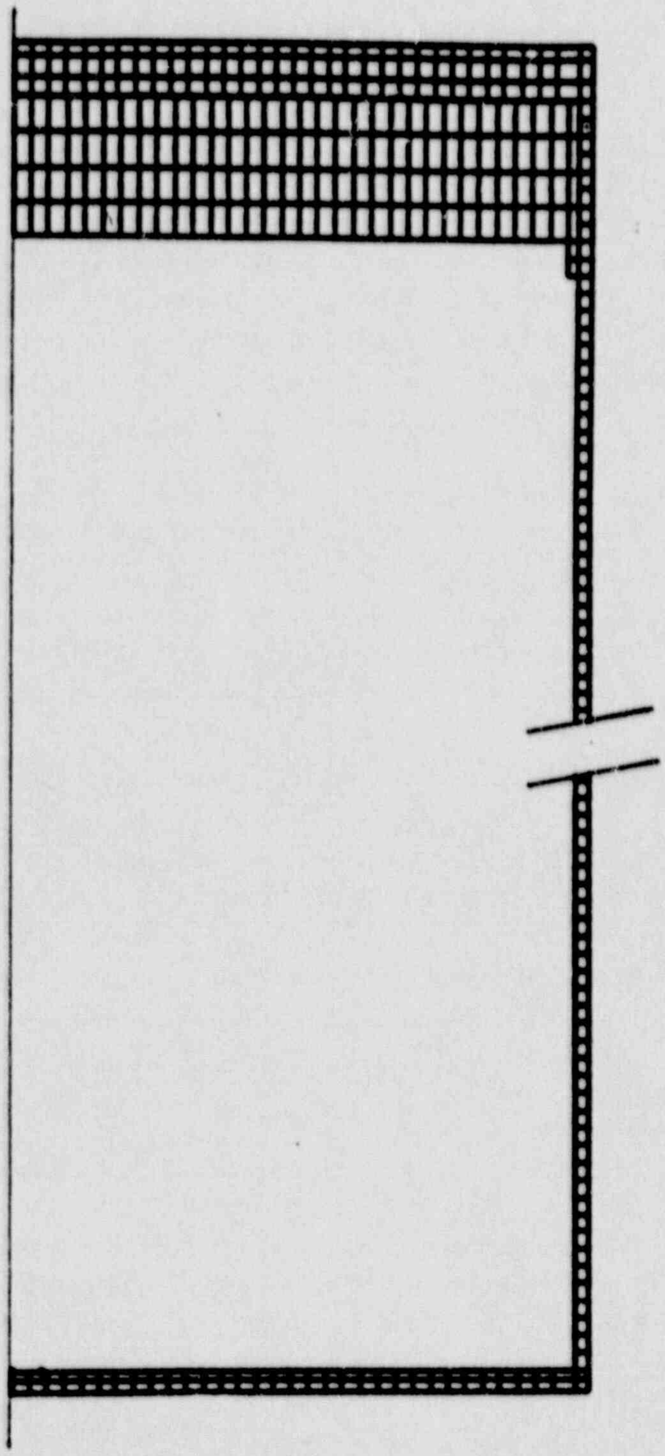


FIGURE 3.4-3
2-D FINITE ELEMENT MODEL OF THE BASKET SHELL

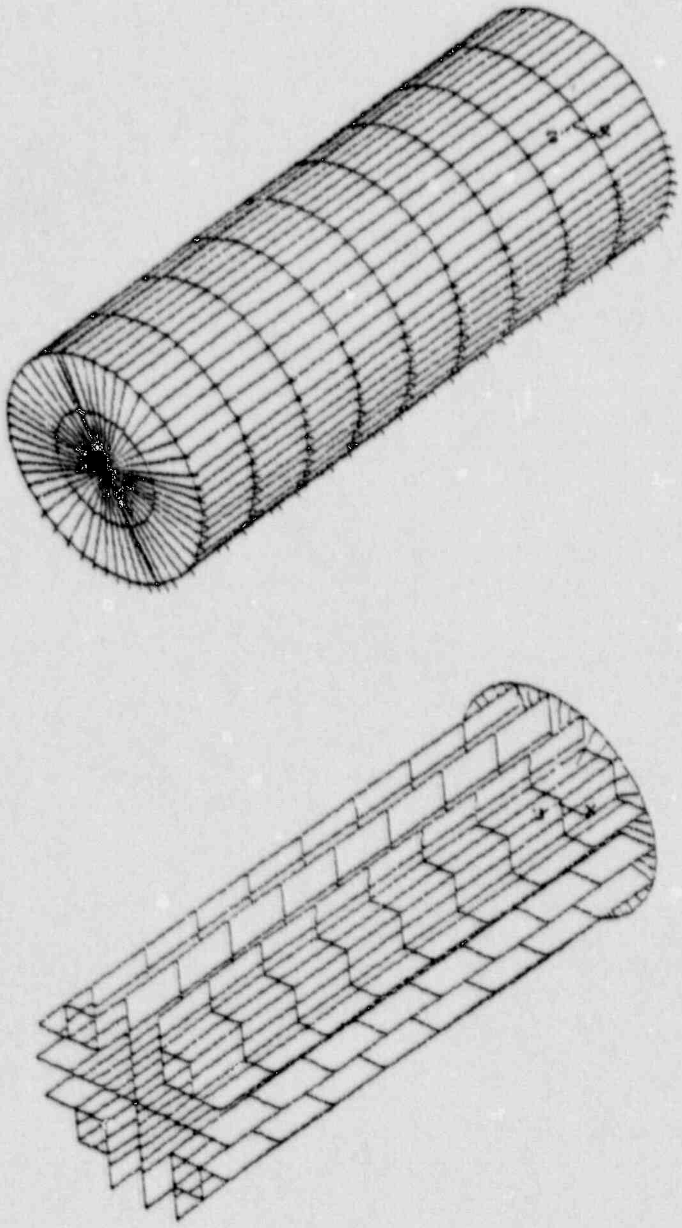


FIGURE 3.4-4

3-D FINITE ELEMENT MODEL OF THE PWR BASKET SHELL AND FRAME

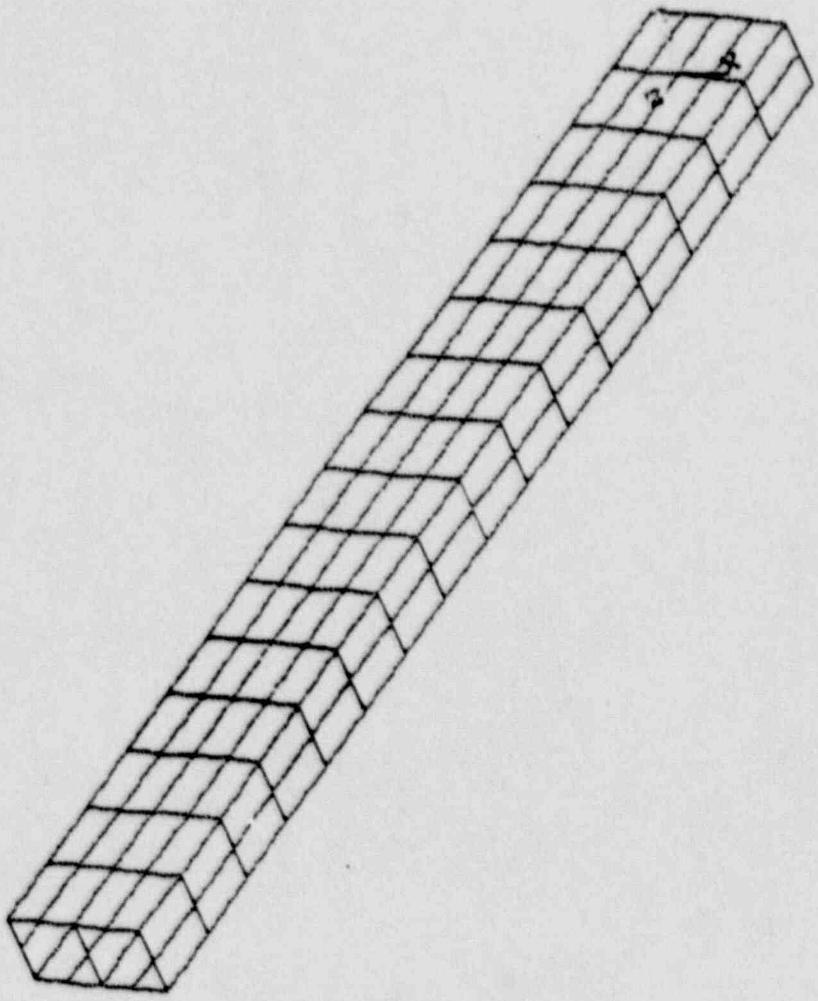


FIGURE 3.4-5
PWR BASKET STORAGE SLEEVE ASSEMBLY
FINITE ELEMENT MODEL FOR THERMAL STRESS EVALUATION

The results of applying the basket temperature distribution to the finite element models are shown in Tables 3.4-5 and 3.4-6 for the PWR and BWR, respectively. The acceptability of the thermal stress levels is evaluated in these tables after their combination with other loads.

3.4.4.1.2 Dead Weight Load Analysis

The basket dead weight loads are calculated by ratioing the results of the 30-foot vertical drop presented in Reference 1.1. They are further combined with the stresses due to other loads and compared with appropriate allowable stress levels in Tables 3.4-5 and 3.4-6 for the PWR and BWR baskets, respectively. As shown in those tables, all basket components and welds are within code allowable levels for normal operating conditions.

3.4.4.1.3 Basket Internal Pressure Analysis

The backfill pressure for the TranStor™ basket is selected so that the basket will always be at slight vacuum during storage. The normal operating internal pressure (Design Pressure per ASME Code) is taken as -4.0 psig, which bounds the values in Tables 3.4-3 and 3.4-4. The same 2-D finite element model as for the thermal stress analysis was used. The results are presented in Table 3.4-5 and 3.4-6 for the PWR and BWR baskets, respectively, and evaluated in combination with the other loadings.

Furthermore, since this pressure is negative (external) the allowable external pressure has been calculated per ASME III, NC-3133.3. To bound both PWR and BWR baskets, no credit for the shell reinforcement by the PWR corner tubes was taken. The resulting allowable of 75 psig is much higher than the 4.0 psig and, therefore, basket shell buckling is not a concern.

3.4.4.1.4 Basket Handling Analysis

As discussed in Chapter 2.0, the basket normal handling load has been defined as ±0.5g applied in all directions simultaneously. This produces the resultant of $(0.5g)\sqrt{2} = 0.71g$ in the horizontal direction and 0.5g in the vertical direction. Stresses are calculated by appropriate scaling of stresses due to the 30-ft horizontal and vertical drops (Reference 1.1) and conservatively added using absolute values to produce the total handling stress. These handling stresses are further combined with the stresses due to other design loads in Tables 3.4-5 and 3.4-6.

3.4.4.1.5 Basket Load Combinations

Load combinations for the PWR and BWR baskets are presented in Tables 3.4-5 and 3.4-6. The -40 °F ambient day is used as a governing case because it produces the highest thermal and pressure loads (see Sections 3.4.4.1.2 and 3.4.4.1.3).

TABLE 3.4-5

PWR BASKET MAXIMUM STRESS EVALUATION

		Dead Weight	Design Pressure	Max Thermal	Normal Handling	Σ	ASME Level A Allowable
Basket Shell	P_M	0.1	0.2	-	0.6	0.9	15.5
	$P_L + P_b$	0.2	2.8	-	2.0	5.0	23.2
	$P + Q$	0.2	2.8	34.1	2.0	39.1	46.4
Bottom Plate	P_M	0.0	0.1	-	0.6	0.7	15.5
	$P_L + P_b$	0.2	1.9	-	1.3	3.4	23.2
	$P + Q$	0.2	1.9	18.4	1.3	21.8	46.4
Structural Lid	P_M	0.0	0.0	-	0.2	0.2	15.5
	$P_L + P_b$	0.1	0.6	-	0.9	1.6	23.2
	$P + Q$	0.1	0.6	4.1	0.9	5.7	46.4
Sleeve Assembly	P_M	0.1	0.0	-	0.2	0.3	18.3
	$P_L + P_b$	0.1	0.0	-	2.1	2.2	27.5
	$P + Q$	0.1	0.0	15.4	2.1	17.6	54.9
Shield Lid Support Ring Weld	P_M	0.1	0.0	-	0.1	0.2	11.6
	$P_L + P_b$	0.1	0.0	-	0.1	0.2	17.4
	$P + Q$	0.1	0.0	0.0	0.1	0.2	34.8
Top Weld	P_M	0.1	0.4	-	0.8	1.3	15.5
	$P_L + P_b$	0.2	0.5	-	0.9	1.6	23.2
	$P + Q$	0.2	0.5	11.3	0.9	12.9	46.4
Shield Lid Weld	P_M	0.1	0.3	-	0.1	0.5	15.5
	$P_L + P_b$	0.1	0.4	-	0.1	0.6	23.2
	$P + Q$	0.1	0.4	17.9	0.1	18.5	46.4

TABLE 3.4-6

BWR BASKET MAXIMUM STRESS EVALUATION

		Dead Weight	Design Pressure	Max Thermal	Normal Handling	Σ	ASME Level A Allowable
	P_M	0.1	0.2	-	0.7	1.0	15.5
Basket Shell	$P_L + P_b$	0.2	2.8	-	1.4	4.4	23.2
	$P + Q$	0.2	2.8	14.0	1.4	18.4	46.4
	P_M	0.0	0.1	-	0.4	0.5	15.5
Bottom Plate	$P_L + P_b$	0.2	1.9	-	1.0	3.1	23.2
	$P + Q$	0.2	1.9	15.9	1.0	19.0	46.4
	P_M	0.0	0.0	-	0.2	0.2	15.5
Structural Lid	$P_L + P_b$	0.1	0.6	-	0.6	1.3	23.2
	$P + Q$	0.1	0.6	4.1	0.6	5.4	46.4
Sleeve	P_M	0.1	0.0	-	0.2	0.3	18.3
Assembly	$P_L + P_b$	0.1	0.0	-	1.6	1.7	27.5
	$P + Q$	0.1	0.0	0.0	1.6	1.7	54.9
Shield Lid	P_M	0.1	0.0	-	0.1	0.2	11.6
Support Ring	$P_L + P_b$	0.1	0.0	-	0.1	0.2	17.4
Weld	$P + Q$	0.1	0.0	0.0	0.1	0.2	34.8
	P_M	0.1	0.4	-	0.8	1.3	15.5
Top Weld	$P_L + P_b$	0.2	0.5	-	0.9	1.6	23.2
	$P + Q$	0.2	0.5	11.3	0.9	12.9	46.4
Shield Lid	P_M	0.1	0.3	-	0.1	0.5	15.5
Weld	$P_L + P_b$	0.1	0.4	-	0.1	0.6	23.2
	$P + Q$	0.1	0.4	17.9	0.1	18.5	46.4

Although the -40 °F ambient is classified in Chapter 2.0 as a Service Level B condition, it is compared against the Level A allowables to bound all Service Level A and B conditions. In addition, all stresses are conservatively combined without consideration of signs (absolute values are used) and specific location in the component.

3.4.4.1.6 Basket Fatigue Evaluation

Fatigue effects on the basket 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 basket 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 Basket components) and the expected number of cycles (a) + (b) + (c) + (d) is less than 1,000.

(a) Full Range Pressure Cycles

The basket normal pressure is essentially atmospheric. 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) = $1 + 2 + 1 + 10 \cdot 50 = 504$.

(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 $2\sqrt{Rt} = 9.6$ in. It can be easily seen from the thermal analysis results that although the temperature of the basket 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 PWR basket design allows welding of the carbon steel to the stainless steel. Therefore, the quantity $(\alpha_1 - \alpha_2)\,\Delta T$ is determined as follows: $(9.00 \cdot 10^{-6} - 6.26 \cdot 10^{-6}) \cdot (313 - 287) = 0.00007$, where

α_1 - coefficient of thermal expansion for the shell material at 300 °F ($9.00 \cdot 10^{-6}$ °F^{-1})
α_2 - coefficient of thermal expansion for the tube material at 300 °F ($6.26 \cdot 10^{-6}$ °F^{-1})
ΔT - shell temperature difference between a 100 °F day (313 °F) and a 75 °F day (287 °F)

This value represents the day-night cycling of the basket during summer time and bounds the changes during any other season. It is below 0.00034 and the number of corresponding cycles per ASME is 0.

However, for the cycle between a 100 °F day (313 °F shell) and a -40 °F day (177 °F shell), the above value becomes $(9.00 \cdot 10^{-6} - 6.26 \cdot 10^{-6}) \cdot (313 - 177) = 0.00037 > 0.00034$. These cycles correspond to season changes and there is only one cycle per year (change from summer to winter and back to summer). Hence, the number of (d) type cycles is 50 for a 50-year design life.

The discussion presented in the preceding paragraphs shows that (a) + (b) + (c) + (d) = 554 and is less than 1000. Thus, all criteria of Condition A are met and the basket is exempt from the fatigue analysis.

3.4.4.1.7 Basket Pressure Test

The basket shall be hydrostatically tested per the requirements of ASME Code, NC-6200. The test pressure of 7.3 psig is higher than 1.25 times the Design Pressure of 4.0 psig as required by NC-6221. From review of Tables 3.4-5 and 3.4-6, it can be clearly seen that the stresses are well below even Level A allowables. Thus, the test pressure meets requirements of the ASME Code, Section III, Subsection NC.

3.4.4.2 Storage Cask Analysis

Only three load components act on the storage cask during normal storage: dead load, live load and differential thermal expansion. The other potential sources of load are discussed in Chapter 11.0. The results of combining the loads and comparing the storage cask stress levels to allowable limits are summarized in Table 3.4-8. As shown in this table, the storage cask meets the structural requirements of ANSI 57.9.

3.4.4.2.1 Storage Cask Dead Load

For the dead weight stress calculation, it is conservatively assumed that the whole weight of the loaded storage cask is taken only by the 12-inch wide center strip contacting the ground at the concrete cask bottom. A 5% dead weight increase is also used per ANSI 57.9. The resulting stress is calculated to be only 220 psi.

3.4.4.2.2 Storage Cask Live Load

The storage cask is subject to two live loads: 1) the snow and ice load of 67.2 psf as discussed in Section 2.2.4 and 2) the weight of the transfer cask with the fully loaded basket. Both of these live loads act on the top of the storage cask.
The snow load is uniformly distributed over the top of the cask and represents a negligible contribution to storage cask stress levels (0.47 psi).

In calculating the effect of the weight of the transfer cask and basket it was assumed that the load was only supported by the storage cask inner steel liner. No structural credit was taken for the concrete. This conservative assumption was made since the loaded transfer cask is designed to rest directly on the top plate of the storage cask liner.

The weakest liner section is at the air outlets where most of the material is cut out. The resulting compressive stress in the liner is calculated to be 10.5 ksi which is well within the allowable levels.

The liner load is further transferred into the cask bottom. Similar to the dead weight analysis, only the center contact strip is assumed to support the load. The bearing stress is calculated to be 150 psi.

3.4.4.2.3 Storage Cask Thermal Stresses

Description of the Analytical Method

The storage cask thermal stress analysis employs classical hand calculations. As can be seen from Tables 3.4-3 and 3.4-4, the maximum thermal gradient across the cask wall exists in the all inlets blocked case, hence, this ΔT is used as the loading input. The analytical methodology is based on the standard approach to reinforced concrete which assumes that concrete resists only compression with steel reinforcement resisting tension. The wall section is conservatively assumed to be completely restrained from rotation to produce conservative results since any flexibility allowed in the structure tends to relieve thermal forces. All stresses are calculated by balancing tension and compression in the section because thermal loading can not produce any resultant force.

Using the above methodology, stresses are calculated in the concrete and in the longitudinal and hoop rebars. Stresses in the top and bottom plates as well as in the liner shell are calculated using hand formulas (Reference 3.2).

The maximum thermal stresses for each of the storage cask structural components are listed in Table 3.4-7. The acceptability of these thermal stress levels is verified in the storage cask load combination evaluated in Table 3.4-8. For load combination #6, the thermal loads in the critical section are zero due to the self-balancing nature of the thermal stresses across the entire cask section (which resists the tornado missile impact). To combine the thermal stresses with the appropriate moment due to the hypothetical cask tipover (load combination # 4), the ACI methodology is used to translate the thermal stresses into equivalent circumferential moments across the cask wall.

3.4.4.2.4 Storage Cask Load Combinations

Evaluation of the cask load combinations discussed in Section 2.2.6 is presented in Table 3.4-8. Input for this table is taken from the normal loads discussed in this chapter as well as from the off-normal and accident loads discussed in Chapter 11.0. The number of load combinations is further reduced from Table 2.2-3 (load combinations 2 and 5 are combined) for the following reasons:

a) As shown in Chapter 11.0, tornado wind does not overturn the cask. Therefore, wind loads are negligible, i.e. W = 0.

b) $T_0 = T_a$ since the worst case accident differential temperature was conservatively used for the thermal stress analysis.

3.4.4.3 Transfer Cask Stress Calculations and Comparison with Allowables

The transfer cask is considered only as a lifting and shielding device. Its adequacy is demonstrated in Section 3.4.3.3 and Chapter 5.0.

3.4.5 Cold

The severe cold environment condition for the cask with a full heat load is addressed in Section 11.1 of this report. This case is analyzed because it results in the highest temperature gradient across the basket and the highest internal pressure. The results are used as bounding conditions for the stress analysis discussed in Section 3.4.4.1 above.

TABLE 3.4-7

**SUMMARY OF MAXIMUM STORAGE CASK THERMAL STRESSES
75°F AMBIENT AIR, NORMAL OPERATION**

Concrete	0.85 ksi
Rebars	
vertical	28.4 ksi
hoop	34.2 ksi
Liner	1.4 ksi
Bottom plate	10.7 ksi
Cover plate	1.8 ksi

TABLE 3.4-8
STORAGE CASK STRUCTURAL LOAD COMBINATION EVALUATION

Load Combination		Dead	Live	Wind	Temp	Seismic	Hypothetical Tipover	Tornado Wind & Missiles	Total	Allowable
1	σ	1.4·0.22	+1.7·0.15						0.56	2.8
	τ	1.4·0.0	+1.7·0.0						0.0	0.11
2,5	σ	0.75(1.4·0.22	+1.7·0.15	+1.7·0.0	+1.7·0.85)				1.5	2.8
	τ	0.75(1.4·0.0	+1.7·0.0	+1.7·0.0	+1.7·0.0)				0.0	0.11
3	σ^-	0.22	+0.15		+0.85	+0.25			1.5	2.8
	σ^+	0.0	+0.0		+0.0	+0.25			0.25	0.43
	τ	0.0	+0.0		+0.0	+0.03			0.03	0.11
4(2)	V	0.0	+0.0		+0.0		+22.2		22.2	103.5
	M^+	0.0	+0.0		+0.0		+593.6		593.6	694.5
	M^-	0.0	+0.0		+787.4		+269.5		1,056.9	1,381
6(3)	V	0.0	+0.0		+0.0			+457.5	457.5	1,106
	M	0.0	+0.0		+0.0			+92,852	92,852	105,290

Notes:

(1) The units are: ksi for stresses, kips for forces, kip-in for moments.
(2) Forces and moments are calculated on per-foot basis.
(3) Forces and moments are calculated for the entire cross-section of the cask.

For the brittle fracture consideration, the worst case would correspond to the coldest fuel (no heat load) and coldest ambient temperature. The basket impact testing requirements meet ASME Code, Section III and NUREG/CR-1815; therefore, brittle fracture is not a concern and the TranStor™ cask could be handled even during extreme temperatures. However, since much of the handling will be performed outdoors, it is probably prudent, from the human operator standpoint, to curtail handling activities if the ambient temperature falls below zero.

The transfer cask and lifting yoke are operated inside the Fuel Building where the temperature is not expected to drop below 0 °F (a corresponding Limiting Condition is imposed in Section 12.2.2). Per ANSI N14.6, the Nil Ductility Transition (NDT) temperature for special lifting devices shall be 40 °F lower than the lowest service temperature. To assure that this requirement is met, Charpy testing is specified for the transfer cask structural components. The requirements were determined using the methodology of NUREG/CR-1815. They are summarized in Table 3.4-9 and called out in the fabrication specification.

If outdoor operations are required during low-temperature conditions (such as in case of a "stand-alone" ISFSI facility), the lowest service temperature shall be addressed in a site-specific application.

3.5 Fuel Rods

The TranStor™ System is capable of storing intact fuel, failed fuel, and fuel debris. As discussed in Chapter 1.0, failed fuel and fuel debris are placed in the special cans that provide confinement of the radioactive material within designated cells. For the intact fuel, the system is designed to limit fuel clad temperatures to below levels where degradation is expected to lead to fuel clad failure. The precise evaluation of the allowable long-term temperatures includes many fuel-specific parameters (such as age, type, and burnup). The methodology and basis for determination of these temperatures is described in References 1.2 and 4.7 and result in less than 0.5% probability of cladding failures over 40 years of storage. The evaluations for different fuel assemblies are shown in Appendix 1. The acceptable temperatures for PWR fuel fall in the range between 335 and 392 °C and the range for BWR fuel is 374 °C to 474 °C. Short-term events at higher temperatures do not cause increase in the clad failure probability due to their short duration (Reference 4.8).

Chapter 4.0 provides the analytical results that assure that the established fuel temperature limits are always met for the design basis heat load. Therefore, no rod failure is expected during fuel handling and storage and the cladding will maintain its confinement function.

TABLE 3.4-9
TRANSFER CASK AND YOKE CHARPY TESTING REQUIREMENTS

Component/Material	CVN Test Temperature (°F)	CVN Energy (ft-lb)
Cask Inner/Outer Shells	0	13.5
Cask Trunnions	0	25.6
Lifting Yoke Pins	0	25.6
Lifting Yoke Beams/Hooks	0	50.5

3.6 References

3.1 M. Fintel, Handbook of Concrete Engineering, VanNostrand Reinhold Col., New York, NY (1974).

3.2 Roark, Formulas for Stress and Strain, Ed. 5, McGraw Hill, N.Y.

4.0 THERMAL EVALUATION

4.1 Discussion

This chapter presents the thermal analysis of the TranStor™ Storage System for normal operation. Off-normal cases are presented in Chapter 11.0.

The significant thermal design feature of the TranStor™ System is the air flow path used to remove the decay heat. This natural circulation of air inside the TranStor™ Storage Cask allows the concrete temperatures to be maintained below the design allowables and keeps the fuel cladding temperatures below the limits where long term degradation might occur.

The base calculation was performed assuming 75°F ambient conditions to model the average long term temperatures expected over the life of the cask. No solar load was used because, as shown in Figure 4.1-1, recent cask tests have shown no impact of solar load on fuel temperatures (Reference 4.1). Only the outer surface of the cask would be affected because the sun shines for 12 daylight hours which is not enough to affect the massive concrete structure. The temperatures of the bulk of the concrete and the basket never see these transient effects. This is further discussed in Section 11.2.

In addition to the 75°F ambient condition which is utilized to determine long term storage temperatures, the -40 and 100°F ambient temperatures are used to model extreme environmental conditions. The -40°F case is considered with no solar loads and the 100°F case is evaluated with maximum solar loads. The maximum solar load was calculated to be the 24-hour average solar load (from 10 CFR 71) so as to more accurately model the steady state temperature expected from long term (four to five days) exposure to 100°F air. Again, inclusion of any solar loads in this calculation is considered to be conservative based on the findings of cask tests which show no solar effect on cask or fuel temperatures (see Figure 4.1-1, Reference 4.1).

In addition to these three cases, three thermal analyses of accident conditions are presented in Chapter 11.0. The first case considers a 125 °F ambient condition with maximum solar load and a maximum decay heat payload. This condition is analyzed to show the temperature for the worst case heat load that, due to decay of the heat source, could only happen once over the life of the cask. The next condition considers blockage of the air inlets on one side of the cask (one-half of the inlets) and the final accident considers complete blockage of all air inlets.

Since the average long-term temperature of 75 °F 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) of less than 125°F, which has been analyzed, does not restrict the use of the TranStor™ System at any US site.

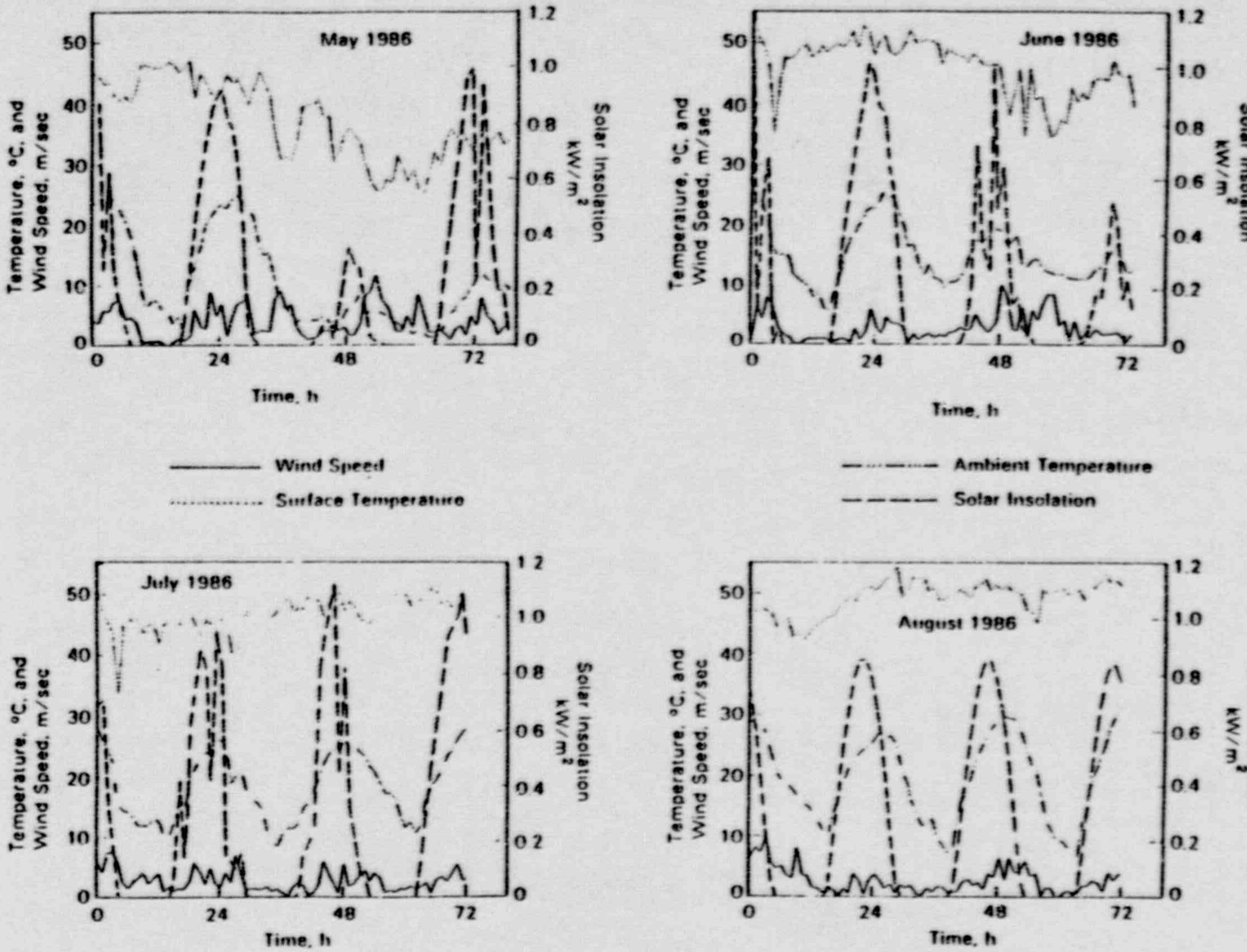


FIGURE 4.1-1
EFFECTS OF AMBIENT CONDITIONS ON CASK TEMPERATURES

Table 4.1-1 summarizes the results of the thermal calculations. The results are based upon a design basis heat load of 26 kW. As can be seen from this table, the conservatively calculated temperatures are below the temperature specifications listed in sections 2.0 and 4.3.

4.2 Summary of Thermal Properties of Materials

The thermal properties used in the thermal hydraulic analyses are shown in Tables 4.2-1 through 4.2-6. The derived parameters (effective thermal conductivities) are discussed in Section 4.4. Low values derived from the open literature and conservative calculations were used.

4.3 Technical Specification of Components

Temperature limits were established for all the materials used in the TranStor™ System. Specifically, these limits are for concrete, fuel cladding, steel, and coatings. The limits were established in accordance with the following codes and manufacturers recommendations:

Code or Standard	Component
PNL-6364 and EPRI TR-106440	Fuel
ASME Section III, Division 1	Steel
ACI 349 and NRC Guidance	Concrete
Manufacturers Recommendations	Coatings

Based upon evaluation of these limits and the results in Table 4.1-1, it was determined that the fuel cladding and concrete temperature limits are the controlling conditions. Steel temperatures of well over 1000 °F and coating temperatures of up to 750 °F are acceptable and do not present a concern.

Table 4.1-1 presents the values selected as the long-term and short-term temperature limits for concrete. This table is based on ACI 349, Appendix A and NRC guidance.

TABLE 4.1-1

SUMMARY OF TranStor™ SYSTEM THERMAL HYDRAULICS EVALUATION

SUMMARY OF LONG TERM EVALUATION

CASE	TEMPERATURES (ºF)							
	Ambient Conditions ºF	Solar	Air Outlet	Outer Concrete	Inner Concrete	Basket Shell	Max Clad PWR	Max Clad BWR
Generic Limits		N/A	N/A	150	225	N/A	635 (1)	705(1)
Steady State Normal-Long Term Storage	75	no	180	87	204	287	609	676

SUMMARY OF SHORT TERM EVALUATION

CASE	Ambient Conditions ºF	Solar	Air Outlet	Outer Concrete	Inner Concrete	Basket Shell	Max Clad PWR	Max Clad BWR
Generic Limits	N/A	N/A	N/A	200	350	N/A	1058	1058
Steady State Severe Cold	-40	no	43	-31	60	177	511	575
Steady State Severe Hot	100	yes	210	141	239	313	633	701
12 hour Max. Thermal Load Transient	125	yes	240	166	270	338	655	724
1/2 of Inlets Blocked	75	no	202	89	228	305	625	693
All Inlets Blocked	75	no	263	94	283	358	673	743
Basket in Transfer Cask								
with He	75	N/A	N/A	N/A	N/A	434	743	816
with vacuum	75	N/A	N/A	N/A	N/A	434	851	877

(1) Long term allowable temperatures shown are the most conservative values since they correspond to the highest burnup and longest cooling time. All other assemblies meeting the heat load requirement (as described in Section 12.2.2.1) have higher temperature limits and are bounded.

TABLE 4.2-1

THERMAL PROPERTIES OF STAINLESS STEEL

Property (units)/ Temperature	Value @ 212°F	392°F	572°F	752°F
Conductivity [1] (Btu/hr-in-°F)	0.7800	0.8592	0.9333	1.0042
Density [1] (lb/in³)	0.2888	0.2872	0.2855	0.2839
Specific Heat [1] (Btu/lbm-°F)	0.1207	0.1272	0.1320	0.135
Emissivity [2]	....	0.36	....	
Emissivity (Coated)[3]	....	0.90	....	

[1] Reference 4.15

[2] Reference 4.3

[3] Reference 4.4

TABLE 4.2-2

THERMAL PROPERTIES OF FERRITIC STEEL

Property (units)/ Temperature	Value @ 32°F	212°F	572°F	932°F
Conductivity [1] (Btu/hr-in-°F)	2.208	2.167	2.083	1.833
Density [1] (lb/in^3)		0.284		
Specific Heat [1] (Btu/lbm-°F)		0.11		
Emissivity [2]		0.8		
Emissivity (coated) [1]		0.9		

[1] Reference 4.4

[2] Reference 4.5

TABLE 4.2-3

THERMAL PROPERTIES OF HELIUM

Property (units)/ Temperature	Value @ 200°F	400°F	600°F	800°F
Conductivity [1] (Btu/hr-in-°F)	0.00808	0.00942	0.01075	0.01150
Specific Heat [1] (Btu/lbm-°F)	....	1.24	....	
Density [1] (lbm/in^3)	4.83E-6	3.70E-6	3.01E-6	2.52E-6

[1] Reference 4.4

TABLE 4.2-4

THERMAL PROPERTIES OF POISON MATERIAL

Property (units)/ Temperature	Value @ 100°F	Value @ 500°F
Conductivity [1] for Aluminum Cladding (Btu/hr-in-°F)	7.805	8.976
Conductivity [1] for the Core Matrix (Btu/hr-in-°F)	4.136	3.698
Density for Aluminum Cladding [1] (lb/in^3)	.... 0.098	
Density for the Core Matrix [1] (lb/in^3)	.... 0.089	
Emissivity [1]	.... 0.15	

[1] Reference 4.6

TABLE 4.2-5

THERMAL PROPERTIES OF CONCRETE

Property (units)/ Temperature	Value @ 32°F - 400°F
Density[1] (lb/ft^3)	.... 142 ....
Specific Heat[1] (Btu/lb_m-°F)	.... 0.21 ....
Conductivity[1] (Btu/hr-ft-°F)	.... 0 .719 ....
Emissivity[2]	.... 0.9 ....

1 References 4.4, 4.17, 4.18

2 References 4.4, 4.17, 4.19

TABLE 4.2-6

THERMAL PROPERTIES OF DRY AIR

Property (units)/ Temperature	Value @ 100°F	300°F	500°F	700°F
Conductivity[1] (Btu/hr-in-°F)	0.00128	0.00161	0.00193	0.00223
Density [1](lbm/in^3)	4.11E-5	3.25E-5	2.38E-5	1.97E-5
Specific heat [1] (Btu/lbm-°F)	0.240	0.244	0.247	0.253

[1] Reference 4.4

The fuel cladding temperature limit is actually a complex function of fuel type, age, and burnup and should be calculated in accordance with the methodology presented in Reference 4.7 for each specific assembly. The referenced method is established to keep the probability of cladding breach at less than 0.5% per fuel rod over a 40 year storage term (Reference 4.7). Appendix 1 presents the results of such calculations for different fuels and burnup/cooling time combinations. As this appendix shows, the PWR fuel allowables range from 635 °F to 738 °F (335 °C to 392 °C) depending on assembly type and cooling time. Similarly, BWR allowables range from 705 °F to 885 °F (374 °C to 474 °C). A short-term (days) limit of 1058°F was established for the vacuum drying, transfer and other short term off-normal and accident conditions. This temperature was established based on experimental results for high temperature induced failure of Zircaloy rods (Reference 4.8 and 4.16) and on previous licensing actions. As documented in Reference 4.14, the zircaloy temperature limits bound the limits for stainless steel clad fuel.

4.4 Thermal Evaluation for Normal Storage Conditions

4.4.1 Thermal Models

Five basic models were utilized for the thermal evaluation of the TranStor™ System. These are:

1. Air Flow and Temperature
2. Storage Cask Body and Basket Exterior Heat Transfer
3. Transfer Cask Body and Basket Exterior Heat Transfer
4. PWR Basket Interior Heat Transfer
5. BWR Basket Interior Heat Transfer

All of these thermal hydraulic models are described in the following sections. The first three models are identical for the PWR and BWR fuel baskets as the basket shell, concrete cask and transfer cask do not change and the heat loads and peaking factors are the same. Models 4 and 5 are used for detailed analysis of the corresponding basket interior.

In addition to these basic models, two sub-models or calculations were used for the effective fuel region thermal conductivity and the surface natural convection heat transfer coefficient. These models/calculations were based on cask testing performed at the Idaho National Engineering Laboratory (INEL) and elsewhere (References 4.1, 4.9, 4.10, 4.11, and 4.12). Both of these calculations were previously reviewed and approved with the VSC-24 SAR (Reference 1.2) and are not repeated herein.

The conservatism of the analytical methodology employed for the TranStor™ System thermal evaluation has been demonstrated by the measurements taken on the VSC-24 casks previously placed in storage at Palisades and Point Beach power plants as well as by the VSC-17 test results at INEL (Reference 4.13).

4.4.1.1 Air Flow and Temperature Calculation

The cask ventilation is provided by natural convection, i.e. decay heat input is the only driving force for the flow (hot air rises). Therefore, the cask ventilation analysis is performed by balancing flow pressure losses along the entire flow path with the buoyancy force resulting from heat input.

The macroscopic energy equation (Bernoulli equation) is applied to air flow from the air inlets to the air outlets. The equation is as follows:

$$-\Delta P + \frac{g \rho_o h}{g_c} - \frac{\rho g \beta h(\Delta T)}{g_c} + \frac{m^2}{2 g_c \rho} \Sigma_i \frac{k_i}{A_i^2} = 0$$

where:

$\Delta P =$ Pressure differential between inlets and outlets;

$\frac{g \rho_o h}{g_c} =$ Elevation pressure head;

$\frac{\rho g \beta h(\Delta T)}{g_c} =$ Pressure change due to air heating (furnace effect)

$\frac{m^2}{2 g_c \rho} \Sigma_i \frac{k_i}{A_i^2} =$ Pressure head loss due to flow

g $= 32.2 \frac{lb_m - \sec^2}{lb_f - ft}$

$\Delta T =$ air temperature difference between inlet and outlet of cask (°F)

h = elevation change from air inlet to air outlet (ft)

$g_c = 32.2 \frac{ft}{\sec^2}$

$\rho = 0.065 \frac{lb_m}{ft^3}$

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

$$\frac{\rho g\ h(\Delta T)}{g_c T} = \frac{m^2}{2 g_c \rho} \sum_i \frac{k_i}{A_i^2}$$

The flow resistance coefficient k_i and cross-sectional area A_i were calculated for each segment of the flow path (i.e. duct screens, bends, expansions, contractions, straight sections, etc.). An iterative solution was derived for calculating the exit temperature and air flow rate. A spreadsheet program was used to perform this calculation. With the known mass flow, the axial heat source distributions shown in Figures 5.2-1 and 5.2-2 were also used to calculate air temperature as a function of elevation as it flows through the storage cask. 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 storage cask and basket temperatures.

It must be noted that the approach described above is conservative because all heat is assumed to be removed by the air flow, thus, resulting in higher air temperatures. Since these temperatures are used as boundary conditions in the cask finite element model, their overprediction results in lower convective heat dissipation in the model and, ultimately, in overprediction of temperatures for fuel and concrete. Thermal tests on the VSC-24 have shown the methodology to be conservative by approximately 10° - 15° F.

4.4.1.2 Storage Cask Body and Basket Exterior Thermal Model

4.4.1.2.1 Heat Transfer Modes

Heat is generated in the fuel that is located in the basket. This heat is conducted, convected, and radiated through the basket shell and then convected to the air and radiated to the storage cask internal liner. Heat off the storage cask liner is also convected to the air and a small amount is conducted through the concrete. On a sunny day, additional heat enters the storage cask through the exterior surface as solar insulation. All this solar heat is radiated and convected from the storage cask surface.

All of these heat transfer modes are addressed by the ANSYS program. The models are described in the following paragraphs.

4.4.1.2.2 Storage Cask Thermal Hydraulic Model

The geometry of the TranStor™ System components was converted into element and node form as shown in Figure 4.4-3. Two element types were used in the model, the 3-D solid element (SOLID70) and a radiation link element (LINK31). Thermal properties are specific to the materials

TABLE 4.4-1

SUMMARY OF TranStor™ SYSTEM 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	84	110	-33
32	97	123	-23
48	110	136	-13
64	123	150	-3
80	135	163	7
96	148	177	17
112	160	190	27
128	172	202	36
144	180	210	43
Air Outlet Temperature	180	210	43
Air Flow Rate (lbm/sec)	0.97	0.93	1.24

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

(see Tables 4.2-1 through 4.2-6 for thermal properties). For conservatism, no heat dissipation from the cask bottom into the ground is assumed.

Specific inputs for the model are discussed in the following sections.

4.4.1.2.3 Solar Radiation

Radiation links are included between all exterior surfaces and the atmosphere and between the basket and liner surfaces in the storage cask. For the annulus surfaces, view factors of unity were assumed; 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). The top of the TranStor™ storage cask was assumed to have a view factor of unity with respect to the sky.

The solar radiation heat input, used for the 100 °F and 125 °F cases, is taken as specified in 10 CFR 71. The solar load for the top surface was 2949 BTU/ft^2 while 1474 BTU/ft^2 was used for the curved side surfaces. This is extremely conservative since the 1474 BTU/ft^2 is for horizontal curved surfaces while the TranStor™ cask sides are vertical. In addition, the cask will never experience this heat load on both sides simultaneously and much of its side will be shaded by the adjacent casks.

These thermal loads were 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.

As stated above, this approach is conservative as the solar radiation is assumed to load the entire vertical surface of the cask while only 14% of the surface is assumed to re-radiate heat. Indeed, if the solar load was actually present on the entire cask surface (not possible even for a single cask), then the entire cask's surface could radiate back to the atmosphere.

4.4.1.2.4 Storage Cask Convection

Natural convection heat transfer coefficients were taken as 2.0 BTU/hr-ft^2-°F on all surfaces as established in Reference 1.2. As shown in that reference, this value is conservative compared to full-scale experimental data from other casks.

4.4.1.2.5 Storage Cask Modeling Assumptions

The model uses a 10° slice to model the entire storage cask. The storage cask geometry and temperatures are axisymmetric, therefore, a two-dimensional representation is adequate. The 10°

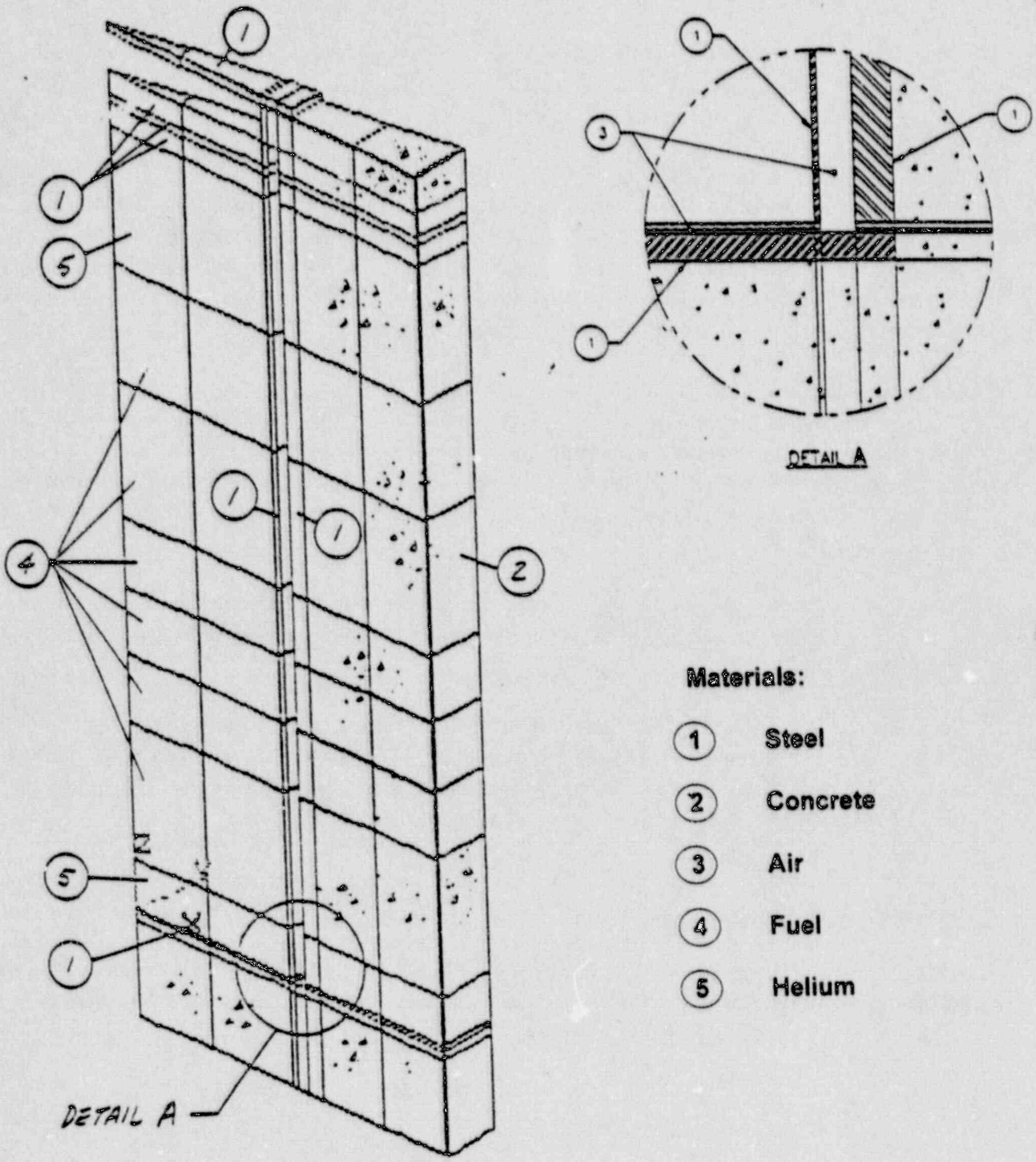


FIGURE 4.4-3

STORAGE CASK THERMAL MODEL

slice is small so as to minimize the complexity of the nodalization while still accurately representing the radial volume distribution. The air ducts were 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 included 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 annulus along the active fuel region was represented as a heat sink at the pre-determined temperatures shown in Table 4.4-1. The basket and liner surfaces were modeled to convect into the air.

Solar radiation discussed above was treated as a volumetric heat generation in a thin outer layer of the cask. This was done because the ANSYS Thermal program allows heat fluxes only at nodes which results in hot spots on the storage cask surface. The use of a thin shell of heated region assures realistically uniform application of the solar flux to the surface.

The basket portion of the storage cask model treats only the basket shell in detail. The interior of the basket was simply modeled as a heat generating region with an effective thermal conductivity. This effective thermal conductivity was estimated from the cask test data (References 4.10 through 4.12, as previously established in Reference 1.1) only to accommodate the solution methods employed by ANSYS. The only interest in the basket in this model is the surface heat flux and the basket shell temperatures since the details of the basket interior are evaluated in a separate model described below. It must be noted that the top and bottom regions of basket internals are modeled as helium which neglects additional axial heat conduction by the top and bottom fittings and essentially blocks axial heat transfer. This approach is conservative because it results in dissipating most of the heat radially, thus, overpredicting temperatures for the concrete, basket shell, and fuel as well as the radial and axial temperature gradients through the basket and cask.

4.4.1.3 Transfer Cask Body and Basket Exterior Thermal Model

The transfer cask containing the basket is analyzed with the same approach as described above for the storage cask. The ANSYS Thermal finite element model was generated as shown in Figure 4.4-4. Similar to the storage condition, the transfer cask geometry and materials are represented accurately while the basket portion treats only the shell in detail. Indeed, the major difference is the absence of convective air flow along the basket exterior and transfer cask interior surfaces. For conservatism, only the steel fins are considered in determining the effective conductivity of the neutron shield region (i.e., the neutron shield is assumed not to conduct any heat).

The main interest in this model is the highest temperature in the basket shell which is used for thermal evaluation of the basket and fuel during the transfer and vacuum drying conditions.

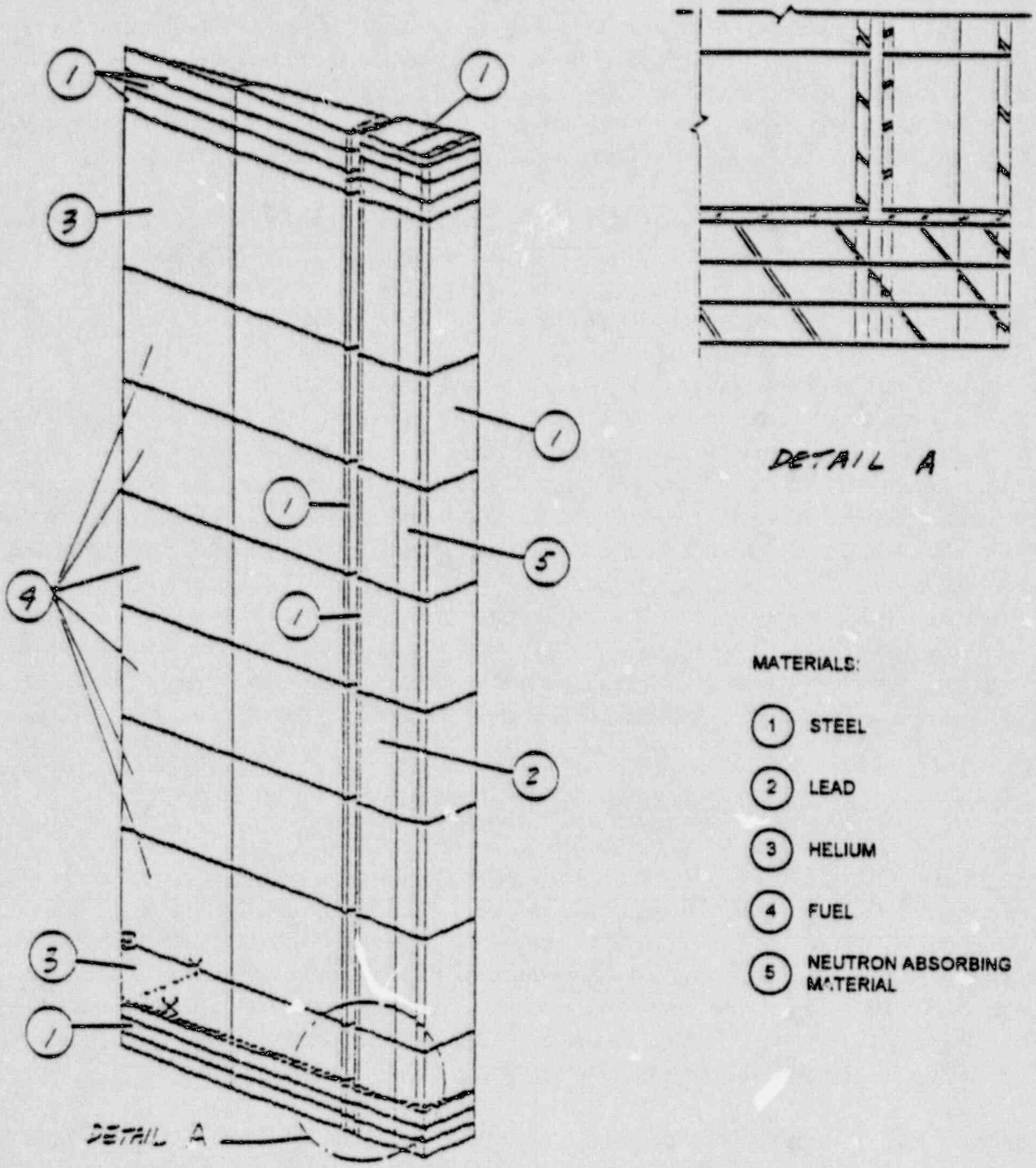


FIGURE 4.4-4

TRANSFER CASK THERMAL MODEL

4.4.1.4 Basket Thermal-Hydraulics

4.4.1.4.1 Basket Heat Transfer Models

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

Models of both PWR and BWR baskets are identical to those presented in the TranStor™ 10CFR71 SAR (Reference 1.1). They are shown in Figures 4.4-5 through 4.4-6 and briefly described below.

A two-dimensional finite element model is generated for the hottest cross-section of the basket. Only one-eighth of the basket cross-section is modeled because of symmetry. Adiabatic boundary conditions are established along the radius of the basket model on both sides. The PLANE55 (2-D Thermal Solid) and LINK31 (Radiation Link) elements are used. The hottest shell temperature calculated from the storage cask model is imposed as a boundary condition for the basket model.

The sleeves are assumed to be located midway between the main frame and the corner frame and between the main frame and the alignment plate. The basket clearances are therefore, evenly distributed between fuel assemblies. This assumption is conservative since it maximizes the number of gaps that impede the heat flow.

The material thermal properties listed in Tables 4.2-1 through 4.2-4 are used in the model. Complex heat transfer in some components (fuel region, helium gaps, and poison material) is represented using effective thermal conductivities as described in Reference 1.1.

The described approach is conservative because it uses the highest heat generation (see Section 4.4.1.4.2) and the hottest shell temperature while not allowing any axial heat flow. By the virtue of two-dimensional nature of the model, all heat is forced to dissipate radially which creates higher temperature gradients than would exist in reality.

4.4.1.4.2 Basket Fuel Heat Source

The fuel region (inside each sleeve) is modeled as a heat generating homogeneous region with an effective conductivity of 0.05 Btu/hr-in-°F (References 1.1 and 1.2). Fuel heat generation rates were calculated by assuming 26 kW of heat generation per basket which translates into 1.083 kW per a PWR assembly or 0.426 kW per a BWR assembly. The hottest horizontal slice of the fuel region was converted to a volumetric heat generation as follows:

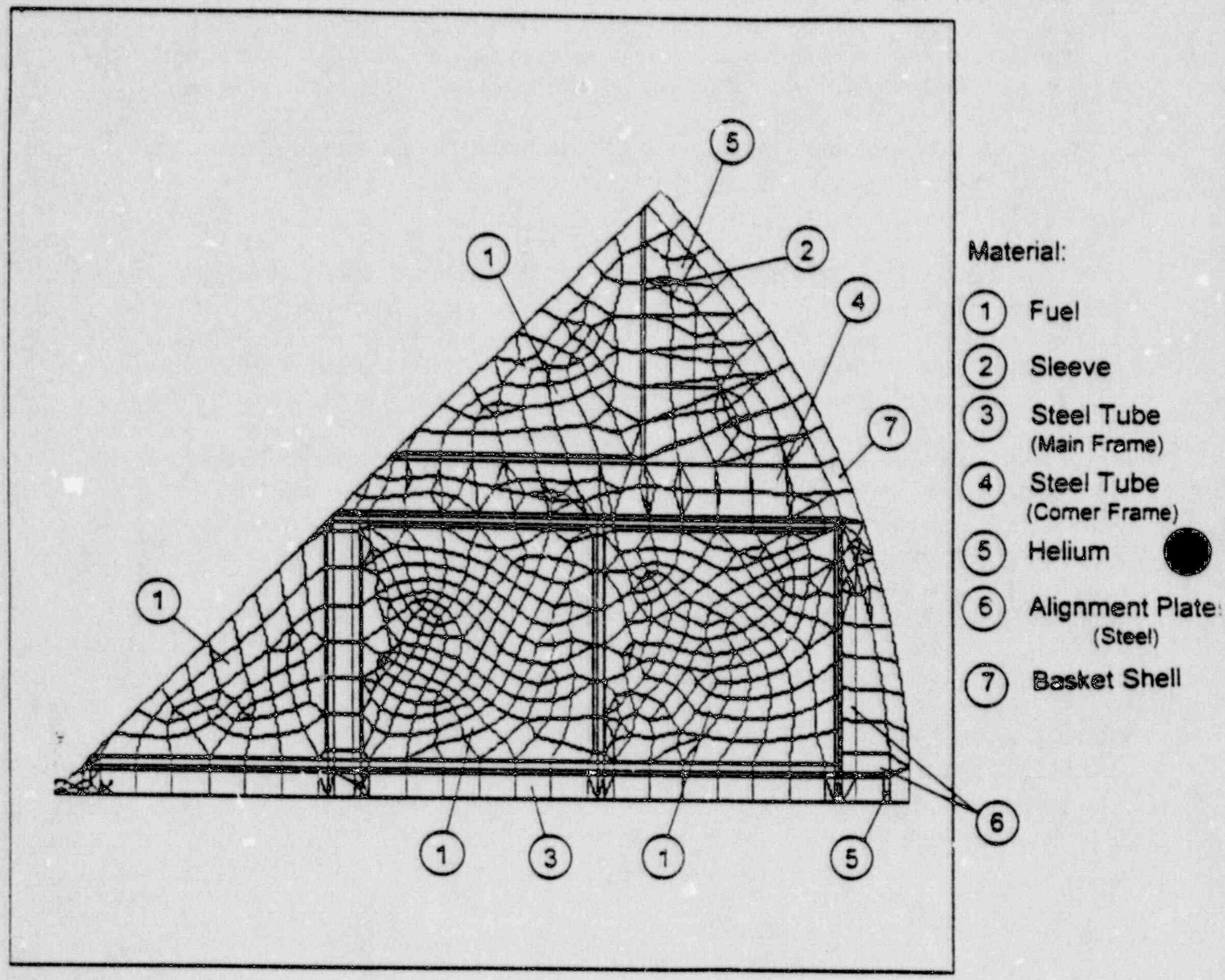


FIGURE 4.4-5

PWR FUEL BASKET 2-D THERMAL MODEL

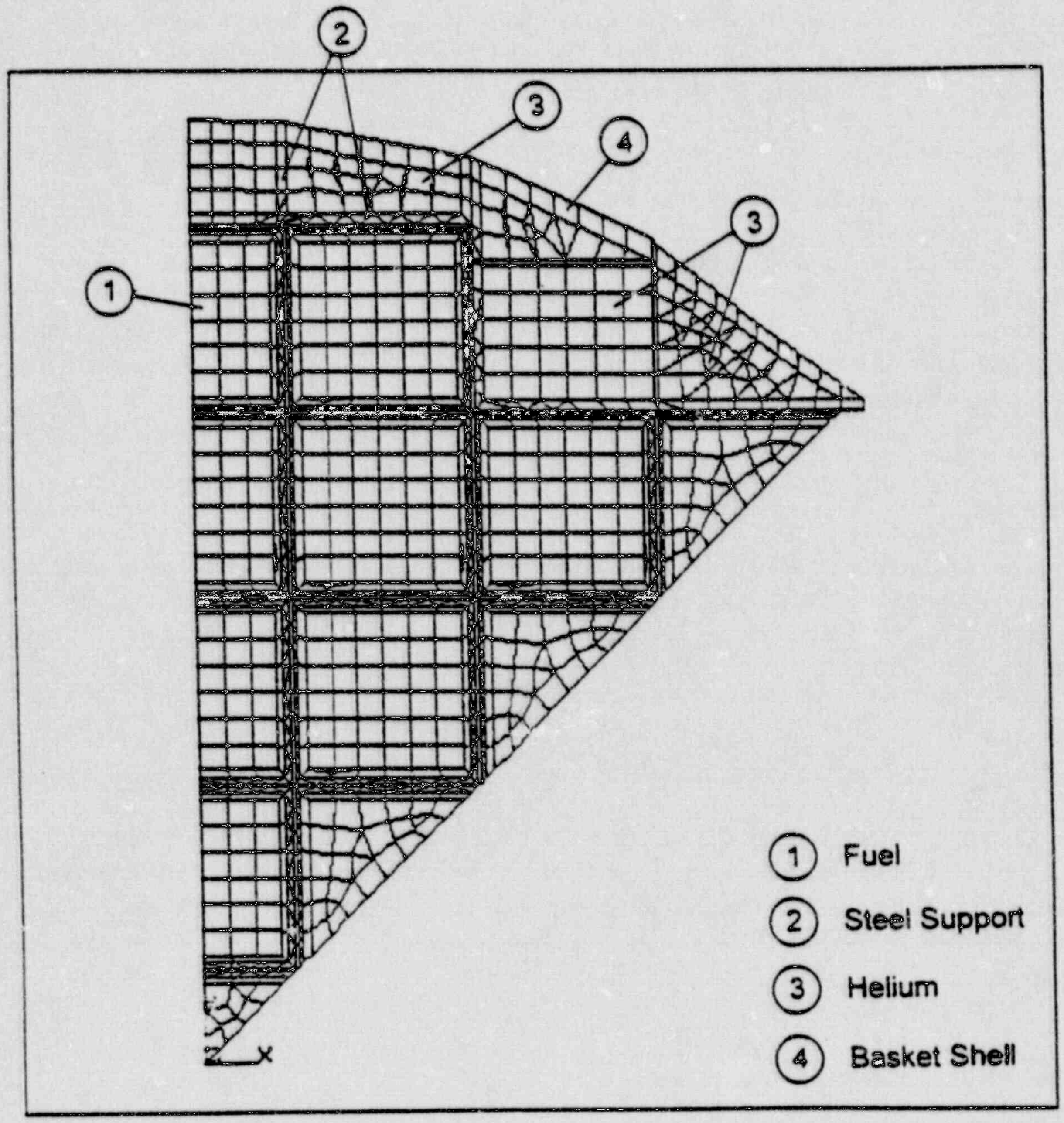


FIGURE 4.4-6

BWR FUEL BASKET 2-D THERMAL MODEL

$$q = k \cdot \frac{Q}{V},$$

where

Q = heat per assembly, 1.083 kW for PWR or 0.426 kW for BWR
V = sleeve internal volume corresponding to the active fuel length
k = peaking factor from Figures 5.2-1 and 5.2-2
= 1.1 for both PWR and BWR

4.4.1.4.3 Basket Thermal Hydraulic Model for Vacuum Drying

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

4.4.2 Maximum Temperatures

Figures 4.4-7 through 4.4-10 and Table 4.1-1 show the temperature distribution of the storage cask components for the normal long-term storage conditions. Temperature distributions for the off-normal, severe environmental conditions and the accident conditions are shown and discussed in Sections 11 1 and 11.2.

4.4.3 Minimum Temperatures

As discussed in Section 3.4.5, the minimum temperatures for the TranStor™ System correspond to the coldest environmental conditions of -40°F and no heat load in the cask. However, even at these extreme conditions the components are above their minimum material temperature limits. The TranStor™ cask does not employ any temperature-sensitive features such as gaskets, packing, O-rings, etc.

4.4.4 Maximum Internal Pressure

The basket backfill pressure is such that at the conditions present during normal operations the internal pressure is slightly below atmospheric (see Tables 3.4-3 and 3.4-4). The basket maximum internal pressure for normal service actually corresponds to the transportation condition addressed in Reference 1.1. As shown in this report and Reference 1.1, the basket normal pressure ranges from approximately -4.0 psig to +3.0 psig.

However, as discussed in Chapter 11.0, the worse case internal pressure occurs if all the fuel rods inside the basket are breached and release their fission gases. This case and its resulting pressure and stresses are addressed in Chapter 11.0.

4.4.5 Maximum Thermal Stresses

The basket and storage cask thermal stresses are presented in Section 3.0.

4.4.6 Evaluation of Cask Performance for Normal Conditions of Storage

As shown in the preceding sections, the TranStor™ System operates well within the thermal design limits. Therefore, no degradation due to temperature effects on materials or components is expected. Indeed, due to the decay of the heat source with time, all temperatures and associated stresses reported herein will decrease over the life of the cask. It must be noted that although allowable temperatures for the TranStor™ construction materials do not change, the fuel temperature limits do decrease with time. However, as established in the previous licensing submittal for the VSC-24 which is very similar to TranStor™ System, the heat load, hence, fuel temperatures, decrease faster than the corresponding allowables. Therefore, the margins between the actual and allowable temperatures of the storage system components will only increase with time in storage.

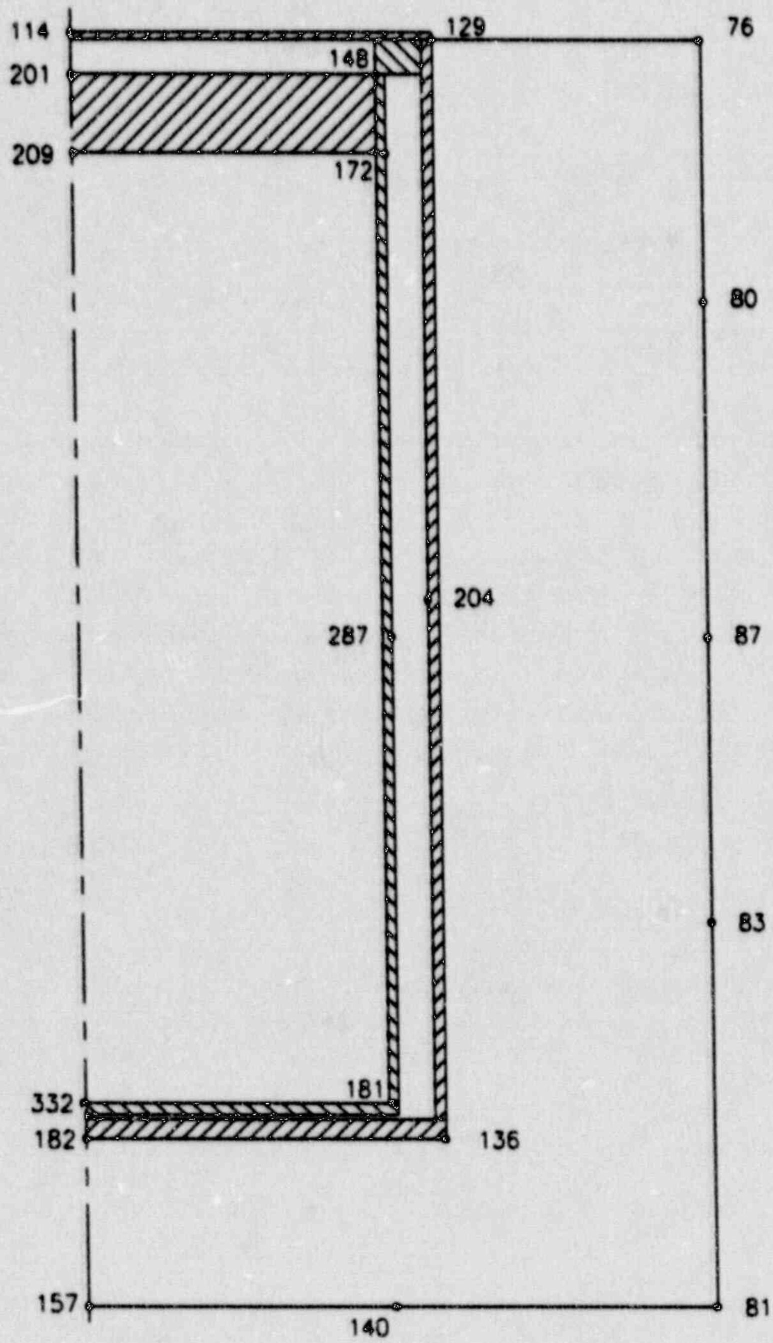


FIGURE 4.4-7
STORAGE CASK THERMAL MODE! RESULTS

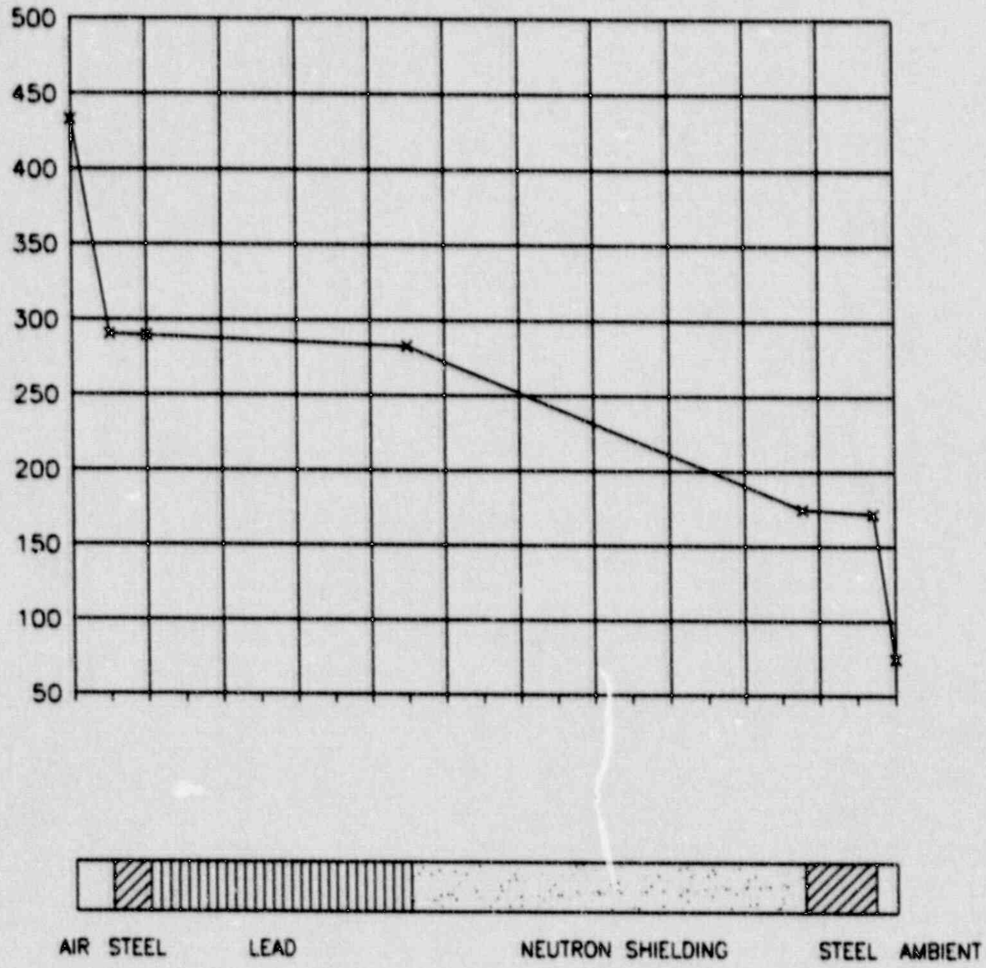


FIGURE 4.4-8

TRANSFER CASK THROUGH WALL TEMPERATURE DISTRIBUTION

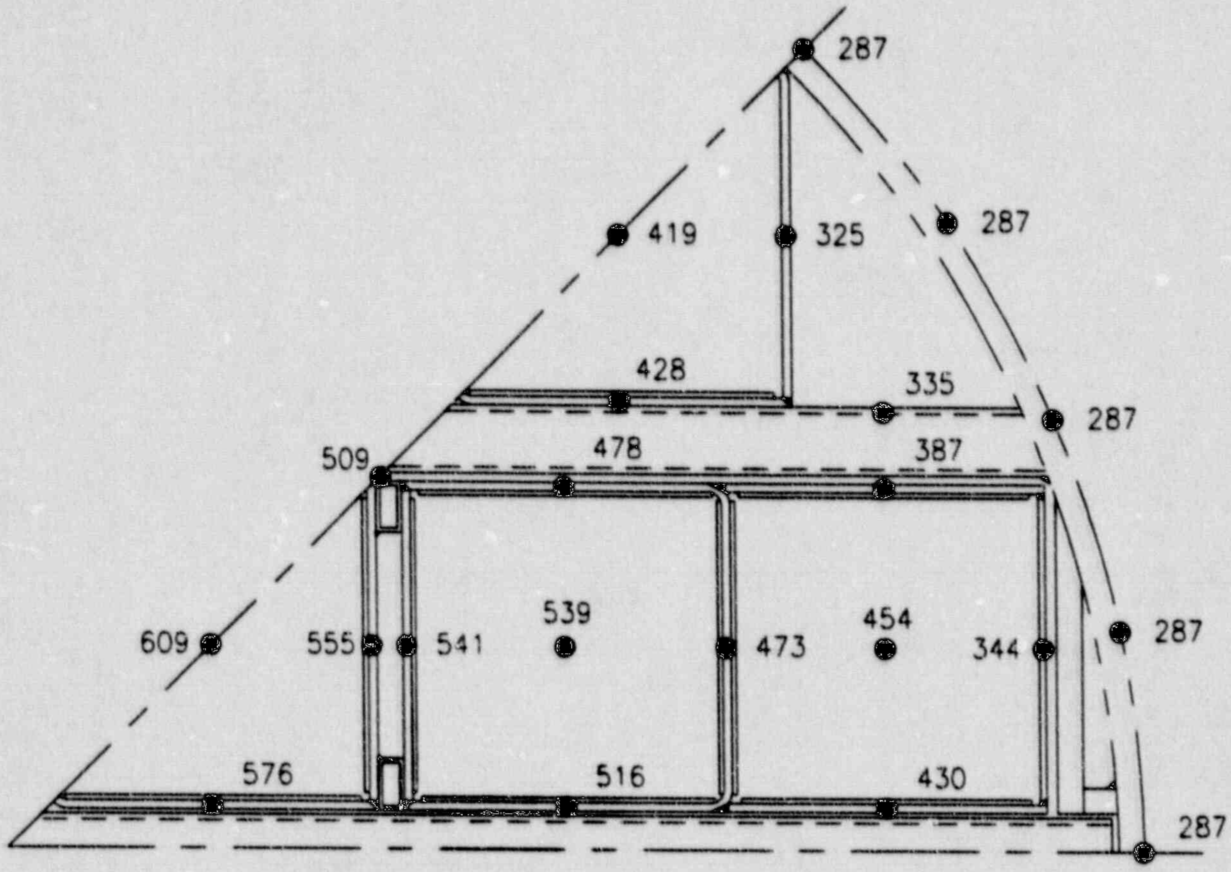


FIGURE 4.4-9

PWR BASKET TEMPERATURE DISTRIBUTION

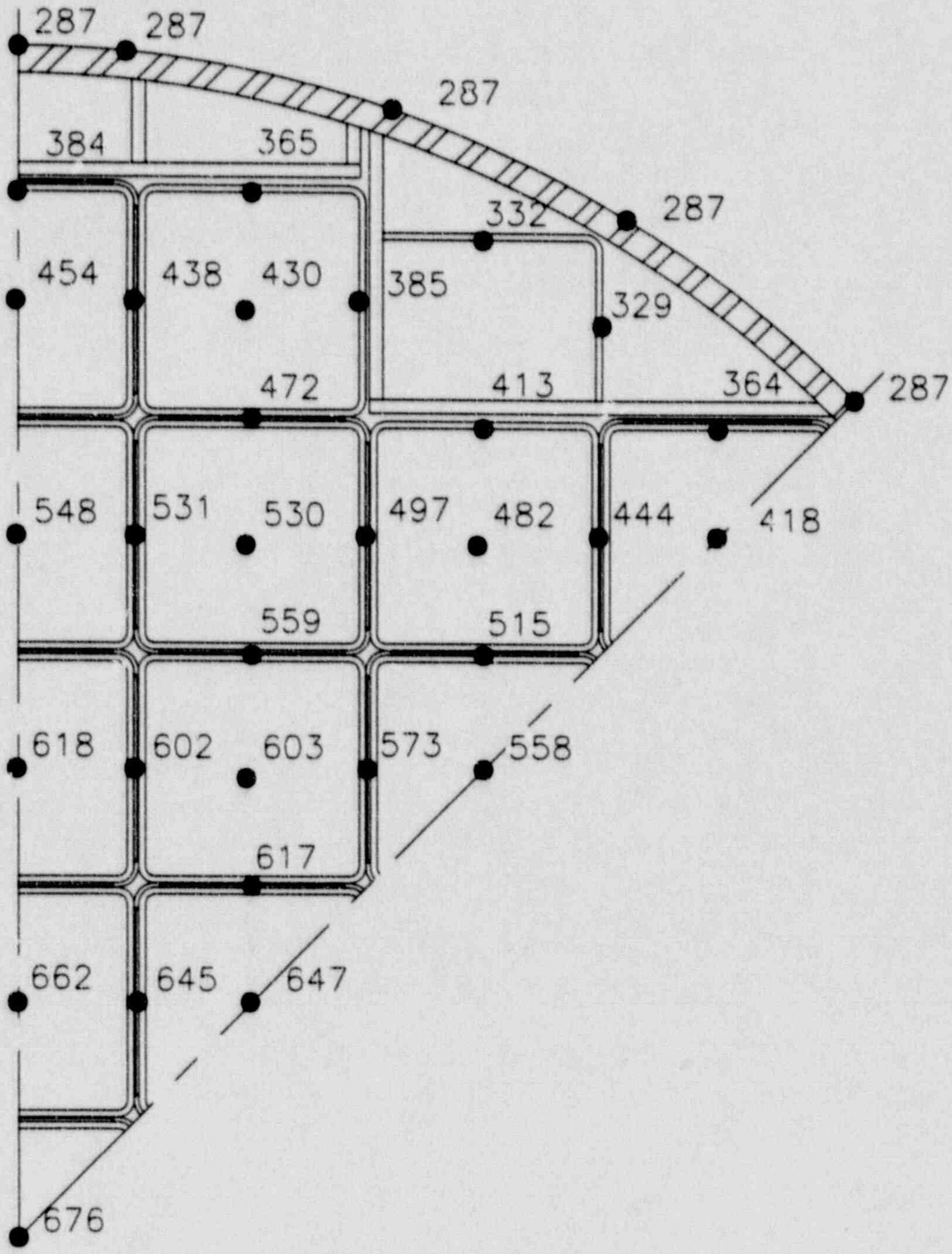


FIGURE 4.4-10

BWR BASKET TEMPERATURE DISTRIBUTION

4.5 References

4.1 J.M. Creer, et al., "The TN-24P PWR Spent Fuel Storage Cask: Testing and Analysis," EPRI-NP-5128, Electric Power Research Institute, Palo Alto, CA (1987).

4.2 Not Used.

4.3 Bucholz, J.A. ,"Scoping Design Analyses for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-year old PWR Spent Fuel," Oak Ridge National Laboratory, ORNL/CSD/TM-149, January 1983.

4.4 Kreith, F. and Bohn, M.S., "Principles of Heat Transfer," 5th Edition, West Publishing Company, 1993.

4.5 Baumeister, T. and Marks, L.S., "Standard Handbook for Mechanical Engineers," 7th Edition, New York, McGraw-Hill Book Co., 1967.

4.6 AAR Advanced Structures Data, "Standard Specification for BORAL Composite Sheet," Reference No.: BRJREV0-940107.

4.7 E. R. Gilbert, et al., "Control of Degradation of spent LWR Fuel During Dry Storage in an Inert Atmosphere," PNL-6364, Pacific Northwest Laboratory, Richland, WA, 1987.

4.8 A.B. Johnson, E.R.Gilbert, "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases", PNL-4835, Pacific Northwest Laboratory, Richland,WA, 1983.

4.9 J.M. Creer, et al., "BWR Spent fuel Storage cask Performance Test," PNL-5777, Pacific Northwest Laboratory, Richland, WA, 1986.

4.10 J.M. Creer, et al., "The Castor V/21 PWR Spent Fuel Storage Cask: Testing and Analyses", EPRI NP-4887, Electric Power Research Institute, Palo Alto, CA, 1986.

4.11 J.M. Creer, et al., "The MC-10 PWR Spent Fuel Storage Cask: Testing and Analysis," EPRI NP-5268, Electric Power Research Institute, Palo Alto, CA, 1987.

4.12 R. A. McCann, "Comparison of HYDRA Predictions to Temperature Data From Two Single-Assembly Spent Fuel Heat Transfer Test," PNL-6074, Pacific Northwest Laboratory, Richland, WA, 1986.

4.13 EPRI TR-100305, "Performance Testing and Analysis of the VSC-17 Ventilated Concrete Cask", May 1992.

4.14 EPRI TR-106440, "Evaluation of Expected LWR Stainless Steel-Clad Fuel Behavior in Long-Term Dry Storage", 1996.

4.15 "Nuclear Systems Material Handbook", Volume 1, Rev. 1, Hanford Engineering Development Laboratories, Richland, WA, 1976.

4.16 A.B. Johnson, E.R.Gilbert, et al., "Recommended Temperature Limits for Dry Storage of Spent Light-Water Reactor Zircaloy-Clad Fuel Rods in Inert Gases", PNL-6189, Pacific Northwest Laboratory, Richland,WA, 1987.

4.17 ASHRAE Handbook, "Fundamentals", ASHRAE, N.Y., 1977.

4.18 W.H. Rohsenow, J.P. Hartnett, E.N. Ganie, "Handbook of Heat Transfer", 2nd Ed., McGraw-Hill, N.Y., 1985.

4.19 E.M. Sparrow and R.D. Ross, "Radiation Heat Transfer", Aug. Ed., McGraw-Hill, N.Y., 1978.

5.0 SHIELDING EVALUATION

The shielding of the TranStor™ system is designed to accommodate the neutron and gamma source strengths that occur for any fuel assemblies that meet the heat load limits (1.083 kw/assembly for PWR fuel and 0.426 kw/assembly for BWR fuel). This is done by over-designing the shield thicknesses so that the dose rates are significantly lower than the design limits defined in Chapter 2.0.

An infinite number of combinations of burnup, cooling time, assembly uranium loading, and initial enrichment will produce heat loads equal to the design limits. Numerous shielding calculations were performed with various combinations of the above parameters, with each case having a heat generation level equal to the design limit. Calculated dose rates are well below the design limits for all cases. Indeed, from a shielding perspective, the TranStor™ system could accommodate 60 GWd/MTU, 5 year cooled fuel with the maximum fuel loading and a lower bound initial enrichment, and still have dose rates below the design limits. Such fuel, however, could not actually be loaded into the cask since its heat generation would be well over the design limit. Therefore, the assembly heat generation limits are clearly the only constraint which determines the required cooling time for fuel to be loaded into the TranStor™ Storage Cask.

The shielding analyses presented in this section verify that the TranStor™ Storage Cask has sufficient shielding to ensure acceptable dose rates around the cask for any fuel that meets the heat generation requirements. Furthermore, the analysis uses the same methodology and codes that were used in the VSC-24 shielding analysis, which has been shown to be very conservative by numerous dose rate measurements on, around, and at large distances from loaded casks (the VSC-24 system having shielding geometries and materials that are virtually identical to the TranStor™ system).

5.1 Discussion and Results

Shielding analyses were performed for TranStor™ baskets containing PWR and BWR fuel with a wide range of burnup levels and cooling times. Gamma and neutron dose rates are calculated at six locations on and around the storage cask surface. These locations are illustrated in Figure 5.1-1. The dose rate data for these six detector locations are presented in Table 5.1-1. Dose rates are presented for four PWR fuel burnup and cooling time combinations and four BWR fuel burnup and cooling time combinations.

For the TranStor™ Storage Cask, thermal considerations determine the minimum required cooling time for fuel of a given burnup. The burnup and cooling time combinations presented in Table 5.1-1 correspond to spent fuel heat generation levels that are roughly equal to or above the specified heat generation limits if the maximum fuel loading (MTU/assembly) and the lower bound initial enrichment are assumed. Assemblies with lower fuel loadings and/or higher initial enrichments can meet the heat generation limits with shorter cooling times (for a given burnup level) than those presented in Table 5.1-1. Since lower fuel loadings and/or higher initial enrichments cause the gamma and neutron source strengths to decrease along with the heat generation, such fuel will yield essentially the same dose rates as those presented in Table 5.1-1. Therefore, the shielding analysis results shown in Table 5.1-1 verify that fuel assemblies which meet the heat generation limit do not cause any dose rate limits to be exceeded, and therefore any assembly that meets the heat generation requirements can be loaded into the TranStor™ cask.

There are no postulated accident scenarios that significantly alter the shielding geometry of the cask top or the inlet and outlet vents. Calculations for postulated tornado-generated missiles and cask drop events predict localized spalling (damage) in the concrete shielding. Analysis of these events are presented in Chapter 11.0. As discussed in Chapter 11.0, the damage to the concrete shielding does not cause unacceptable dose rates on the cask surface.

TranStor™ baskets containing fuel are moved from the fuel pool to the storage cask using a transfer cask. Dose rates are calculated at seven locations on and around the transfer cask exterior. These detector locations are shown in Figure 5.1-2. The dose rate results for these detector locations are shown in Table 5.1-2 for each of the PWR and BWR fuel burnup and cooling time combinations that were analyzed.

The transfer cask length may vary from plant site to plant site, but all transfer casks have the radial and axial shielding thicknesses assumed in the shielding analyses. Given the fixed shielding thicknesses, the length of the transfer cask will not significantly effect the cask exterior dose rates. A small number of BWR plants which have a pool crane weight limit under 110 tons may have to use a modified transfer cask which has somewhat less radial shielding than that present in the generic transfer cask. This modified transfer cask is discussed in Chapter 1.0 and the results are presented in Appendix 2.

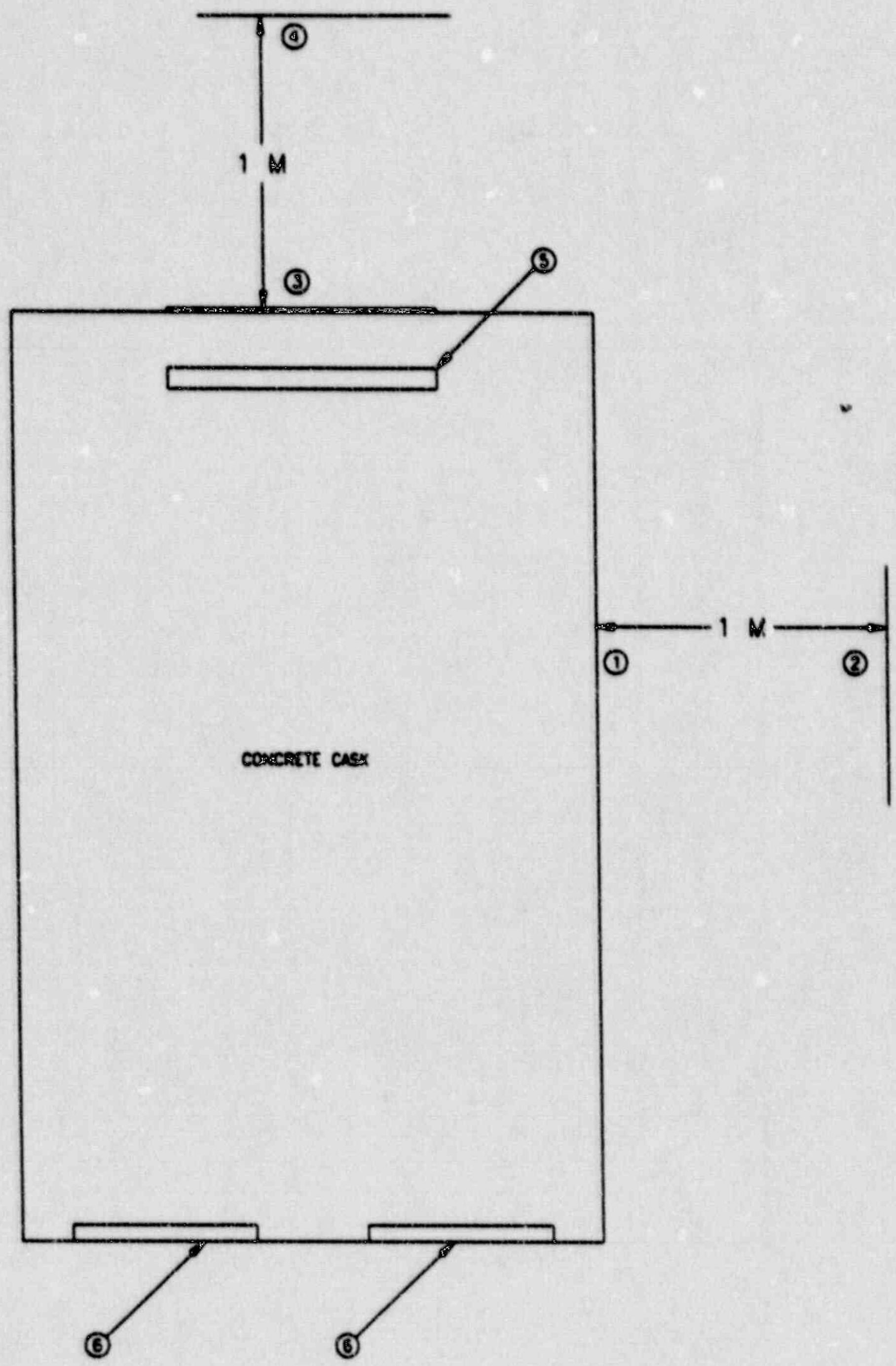


FIGURE 5.1-1

TranStor™ STORAGE CASK
CALCULATED DOSE RATE LOCATIONS

TABLE 5.1-1

NORMAL CONDITION
TranStor™ STORAGE CASK DOSE RATES (mrem/hr)
(for ~26 kw/basket fuel)

Detector Location	PWR 40 GWd/MTU 5 year cool	PWR 45 GWd/MTU 6 year cool	PWR 50 GWd/MTU 8 year cool	PWR 60 GWd/MTU 13 year cool	BWR 35 GWd/MTU 5 year cool	BWR 40 GWd/MTU 6 year cool	BWR 45 GWd/MTU 7 year cool	BWR 50 GWd/MTU 8 year cool
1	γ = 18.2 n = 0.7 Tot = 18.9	γ = 17.3 n = 0.9 Tot = 18.2	γ = 14.4 n = 1.1 Tot = 15.5	γ = 12.7 n = 1.4 Tot = 14.1	γ = 17.1 n = 0.7 Tot = 17.8	γ = 17.0 n = 1.3 Tot = 18.3	γ = 15.1 n = 1.4 Tot = 16.5	γ = 14.5 n = 1.4 Tot = 15.9
2	γ = 9.7 n = 0.4 Tot = 10.1	γ = 9.2 n = 0.5 Tot = 9.7	γ = 7.7 n = 0.6 Tot = 8.3	γ = 6.7 n = 0.8 Tot = 7.5	γ = 9.1 n = 0.4 Tot = 9.5	γ = 9.1 n = 0.7 Tot = 9.8	γ = 8.1 n = 0.7 Tot = 8.8	γ = 7.7 n = 0.8 Tot = 8.5
3	γ = 11.5 n = 67.8 Tot = 79.3	γ = 10.4 n = 87.8 Tot = 98.2	γ = 8.2 n =106.4 Tot =114.6	γ = 4.9 n =141.0 Tot =145.9	γ = 14.8 n = 77.2 Tot = 92.0	γ = 14.2 n =135.8 Tot =150.0	γ = 12.3 n =141.3 Tot =153.6	γ = 10.7 n =146.1 Tot =156.8
4	γ = 8.5 n = 58.1 Tot = 66.6	γ = 7.7 n = 75.3 Tot = 83.0	γ = 6.1 n = 91.3 Tot = 97.4	γ = 3.6 n =120.9 Tot =124.5	γ = 11.0 n = 66.2 Tot = 77.2	γ = 10.5 n =116.5 Tot =127.0	γ = 9.1 n =121.2 Tot =130.3	γ = 8.0 n =125.3 Tot =133.3
5	γ = 1.0 n = 6.1 Tot = 7.1	γ = 1.0 n = 6.1 Tot = 7.1	γ = 1.0 n = 6.1 Tot = 7.1	γ = 1.0 n = 6.1 Tot = 7.1	γ = 1.3 n = 6.2 Tot = 7.5	γ = 1.3 n = 6.2 Tot = 7.5	γ = 1.3 n = 6.2 Tot = 7.5	γ = 1.3 n = 6.2 Tot = 7.5
6	γ = 11.3 n = 1.9 Tot = 13.2	γ = 11.3 n = 1.9 Tot = 13.2	γ = 11.3 n = 1.9 Tot = 13.2	γ = 11.3 n = 1.9 Tot = 13.2	γ = 12.3 n = 1.7 Tot = 14.0	γ = 12.3 n = 1.7 Tot = 14.0	γ = 12.3 n = 1.7 Tot = 14.0	γ = 12.3 n = 1.7 Tot = 14.0

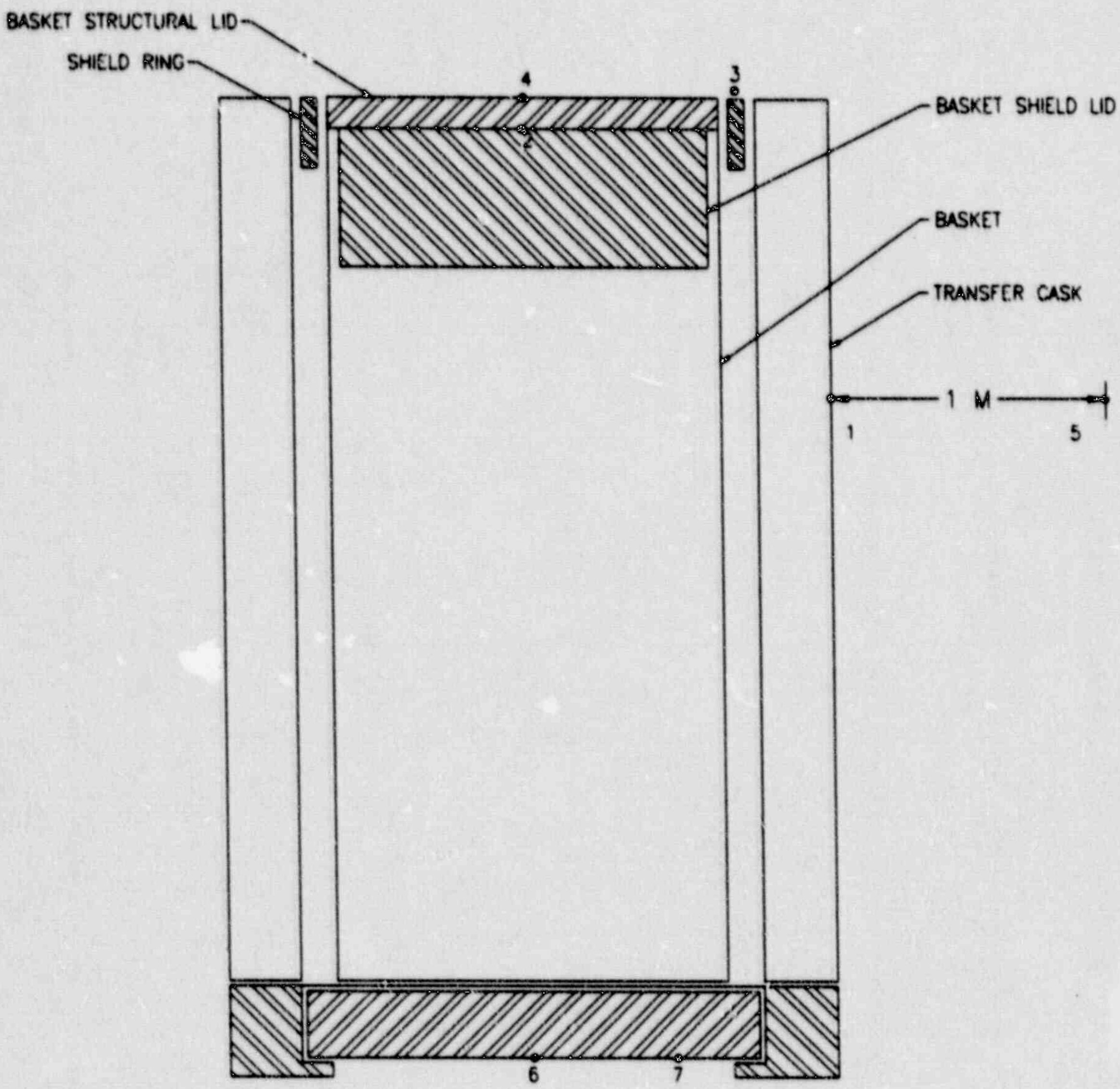


FIGURE 5.1-2

**TranStor™ TRANSFER CASK
CALCULATED DOSE RATE LOCATIONS**

TABLE 5.1-2

TranStor™ TRANSFER CASK DOSE RATES (mrem/hr)
(for ~26 kw/basket fuel)

Detector Location	PWR 40 GWd/MTU 5 year cool	PWR 45 GWd/MTU 6 year cool	PWR 50 GWd/MTU 8 year cool	PWR 60 GWd/MTU 13 year cool	BWR 35 GWd/MTU 5 year cool	BWR 40 GWd/MTU 6 year cool	BWR 45 GWd/MTU 7 year cool	BWR 50 GWd/MTU 8 year cool
1	γ = 108 n = 75 Tot = 183	γ = 93 n = 98 Tot = 191	γ = 72 n = 118 Tot = 190	γ = 46 n = 157 Tot = 203	γ = 63 n = 63 Tot = 126	γ = 59 n = 111 Tot = 170	γ = 50 n = 116 Tot = 166	γ = 44 n = 119 Tot = 163
2	γ = 556 n = 159 Tot = 715	γ = 513 n = 206 Tot = 719	γ = 413 n = 249 Tot = 662	γ = 247 n = 330 Tot = 577	γ = 720 n = 182 Tot = 902	γ = 702 n = 320 Tot = 1022	γ = 611 n = 333 Tot = 944	γ = 535 n = 344 Tot = 879
3	γ = 937 n = 121 Tot = 1059	γ = 881 n = 157 Tot = 1038	γ = 720 n = 190 Tot = 910	γ = 419 n = 252 Tot = 671	γ = 1222 n = 134 Tot = 1356	γ = 1191 n = 235 Tot = 1426	γ = 1041 n = 245 Tot = 1285	γ = 910 n = 253 Tot = 1163
4	γ = 25 n = 87 Tot = 112	γ = 23 n = 112 Tot = 135	γ = 18 n = 136 Tot = 154	γ = 11 n = 180 Tot = 191	γ = 33 n = 95 Tot = 128	γ = 31 n = 168 Tot = 199	γ = 27 n = 175 Tot = 202	γ = 23 n = 181 Tot = 204
5	γ = 55 n = 24 Tot = 79	γ = 48 n = 31 Tot = 77	γ = 37 n = 37 Tot = 74	γ = 23 n = 50 Tot = 73	γ = 33 n = 22 Tot = 55	γ = 30 n = 39 Tot = 69	γ = 25 n = 41 Tot = 66	γ = 22 n = 42 Tot = 64
6	γ = 85 n = 260 Tot = 345	γ = 76 n = 336 Tot = 412	γ = 60 n = 407 Tot = 467	γ = 37 n = 539 Tot = 576	γ = 95 n = 291 Tot = 386	γ = 92 n = 511 Tot = 603	γ = 79 n = 532 Tot = 611	γ = 69 n = 550 Tot = 619
7	γ = 40 n = 143 Tot = 183	γ = 35 n = 186 Tot = 221	γ = 28 n = 225 Tot = 253	γ = 17 n = 298 Tot = 315	γ = 45 n = 173 Tot = 218	γ = 43 n = 305 Tot = 348	γ = 37 n = 318 Tot = 355	γ = 33 n = 328 Tot = 361

5.2 Source Specification

The shielding analyses consider two types of TranStor™ baskets, a PWR basket that contains 24 PWR spent fuel assemblies, and a BWR basket containing 61 BWR spent fuel assemblies. The shielding analyses are based upon maximum assembly uranium loadings of 0.469 MTU for PWR fuel and 0.197 MTU for BWR fuel. These values conservatively bound all existing fuel. The TranStor™ baskets may also contain partial fuel assemblies, damaged or failed fuel assemblies, or fuel debris. The fuel inventory for these cases is equal to or lower than that of intact fuel with the maximum fuel loading. Therefore, the source terms calculated for maximum loading intact fuel, described above, will bound the partial fuel, failed fuel, and fuel debris cases.

The neutron and gamma source terms, for each analyzed burnup and cooling time combination, are taken from the OCRWM (Office of Civilian Radioactive Waste Management) LWR Radiological Computer Database (Reference 5.1). This database is a very large compilation of physical and radiological data for all LWR spent fuel assembly types. The database includes neutron and gamma source terms (including gamma source spectra) for BWR and PWR fuel (on a per metric ton of initial uranium basis) as a function of burnup level, cooling time, and initial enrichment. The compiled gamma and neutron source data is based upon ORIGEN-2 calculations.

For each studied PWR fuel burnup level, the minimum initial enrichment available in the database is selected for determining the bounding source data. The BWR fuel cases use the same enrichment level that is used for PWR fuel of the same burnup level. For fuel of a given burnup level, a lower initial enrichment will yield higher gamma and neutron source terms. Therefore, assuming minimum initial enrichment levels is conservative. A survey of actual fuel assembly demographic data (Reference 5.2) verified that the initial enrichments assumed for the shielding analyses are conservative lower bound enrichments for the existing fuel inventory.

5.2.1 Gamma Source

The gamma source data output by the OCRWM database includes gammas from fission products, actinides, and activated hardware. The fuel assemblies are sub-divided into four distinct axial source regions. In addition to the active fuel region of the assembly, there are three activated hardware regions, including the bottom nozzle region, the gas plenum region, and the top nozzle region. The gamma sources in the three hardware regions are due to activation in the metal hardware (primarily Co-60). Separate shielding calculations are performed to determine the gamma dose rate contribution at each detector location from each of these four separate gamma source regions. The gamma sources present in the four axial sub-regions of the assembly are discussed in the following sections.

5.2.1.1 Fuel Region Gamma Source Strengths

The fuel region gamma source descriptions for design basis PWR and BWR fuel are shown in Tables 5.2-1 and 5.2-2. Gamma source strengths are presented in γ/sec-cask. For each case, the burnup level and cooling time are shown, along with the assumed initial enrichment level of the fuel. As stated earlier in Section 5.2, these fuel region gamma source strengths are taken directly from the DOE-OCRWM LWR Database (Reference 5.1).

The gamma source strengths, which are output by the database on a per MTU (metric tons of initial uranium) basis, include gammas from fission products, actinides, and activated materials from the assembly core region. The activated materials gamma source terms are based upon typical PWR and BWR core region non-fuel material inventories (cladding, grid spacers, etc.). The per cask source terms presented in Tables 5.2-1 and 5.2-2 are based upon a maximum PWR assembly fuel loading of 0.469 MTU and a maximum BWR assembly fuel loading of 0.197 MTU.

The PWR fuel region gamma source terms presented in Table 5.2-1 include the (Co-60) gamma source term from activated control components. Since stainless steel contains significant trace quantities of cobalt, stainless steel clad control components develop a significant level of Co-60 activity after exposure to a large neutron fluence. The poison materials inside the control component rods do not produce significant sources of gamma radiation. The control component Co-60 gamma source that is added to the fuel assembly gamma source for the shielding analyses is based upon the W 17x17 assembly burnable poison rod assembly. This type of control component is determined to contain the largest amount of cobalt within the assembly active fuel region (Reference 5.1).

The LWR database provides the core region Co-60 activity (per MTU of fuel) as a function of burnup level, cooling time, and initial enrichment. The core region cobalt mass (per MTU) is also given. From this data, a core region activation level (Ci of Co-60 per initial gram of cobalt) is determined. Multiplying this by the total cobalt content of the control component yields a total Co-60 activity for the control component. The total control component cobalt inventory is based upon the total core region stainless steel volume and the maximum cobalt concentration in stainless steel. The total cask fuel region gamma source from the control component Co-60 activity is collapsed onto the 1.25 MeV line energy shown in Table 5.2-1. This is done using an approach which preserves total gamma energy. The resulting total 1.25 MeV gamma source strength is shown in Table 5.2-1.

TABLE 5.2-1

PWR FUEL REGION GAMMA SOURCE STRENGTHS AND ENERGY SPECTRA (γ/sec-cask)

Line Energy (MeV)	Gamma Source Strength 40 GWd - 5 years Enrichment = 3.02%	Gamma Source Strength 45 GWd - 6 years Enrichment = 3.30%	Gamma Source Strength 50 GWd - 8 years Enrichment = 3.56%	Gamma Source Strength 60 GWd - 13 years Enrichment = 4.03%
0.01	4.022E+16	3.685E+16	3.366E+16	3.307E+16
0.025	9.631E+15	8.553E+15	7.379E+15	6.650E+15
0.0375	1.044E+16	9.847E+15	9.131E+15	8.700E+15
0.0575	8.000E+15	7.249E+15	6.585E+15	6.476E+15
0.085	5.243E+15	4.664E+15	4.098E+15	3.793E+15
0.125	5.431E+15	4.870E+15	4.291E+15	3.806E+15
0.225	4.365E+15	3.810E+15	3.310E+15	3.116E+15
0.375	2.665E+15	2.211E+15	1.723E+15	1.368E+15
0.575	7.260E+16	7.190E+16	6.561E+16	5.881E+16
0.85	1.807E+16	1.585E+16	1.044E+16	3.931E+15
1.25*	8.860E+15	8.433E+15	7.017E+15	4.486E+15
1.75	1.142E+14	9.709E+13	7.790E+13	6.094E+13
2.25	4.993E+13	2.275E+13	4.795E+12	1.185E+11
2.75	1.875E+12	1.024E+12	3.035E+11	2.779E+10
3.5	2.409E+11	1.321E+11	3.928E+10	3.266E+09
5	3.961E+08	5.134E+08	6.212E+08	8.225E+08
7	4.568E+07	5.921E+07	7.164E+07	9.485E+07
9.5	5.248E+06	6.801E+06	8.229E+06	1.090E+07
Total	1.856E+17	1.743E+17	1.533E+17	1.343E+17

* 1.25 MeV source strength includes Co-60 gammas from activated control components.

TABLE 5.2-2

BWR FUEL REGION GAMMA SOURCE STRENGTHS AND ENERGY SPECTRA (γ/sec-cask)

Line Energy (MeV)	Gamma Source Strength 35 GWd - 5 year Enrichment = 2.75%	Gamma Source Strength 40 GWd - 6 year Enrichment = 3.00%	Gamma Source Strength 45 GWd - 7 year Enrichment = 3.30%	Gamma Source Strength 50 GWd - 8 year Enrichment = 3.56%
0.01	3.799E+16	3.711E+16	3.525E+16	3.480E+16
0.025	9.203E+15	8.671E+15	7.907E+15	7.547E+15
0.0375	9.967E+15	1.012E+16	9.663E+15	9.493E+15
0.0575	7.526E+15	7.271E+15	6.879E+15	6.787E+15
0.085	4.938E+15	4.729E+15	4.384E+15	4.250E+15
0.125	5.119E+15	5.047E+15	4.678E+15	4.509E+15
0.225	4.092E+15	3.811E+15	3.503E+15	3.395E+15
0.375	2.566E+15	2.274E+15	1.941E+15	1.758E+15
0.575	7.014E+16	7.514E+16	7.091E+16	6.799E+16
0.85	1.710E+16	1.700E+16	1.345E+16	1.069E+16
1.25	6.937E+15	7.149E+15	6.376E+15	5.729E+15
1.75	1.071E+14	1.026E+14	8.945E+13	8.223E+13
2.25	4.272E+13	2.024E+13	8.747E+12	3.836E+12
2.75	1.667E+12	9.749E+11	4.952E+11	2.581E+11
3.5	2.146E+11	1.265E+11	6.454E+10	3.364E+10
5	4.433E+08	7.812E+08	8.120E+08	8.385E+08
7	5.111E+07	9.009E+07	9.362E+07	9.669E+07
9.5	5.872E+06	1.035E+07	1.076E+07	1.111E+07
Total	1.757E+17	1.784E+17	1.650E+17	1.570E+17

The B&W "gray" axial power shaping rod assembly, which consists of rods of pure Inconel-600, has a cobalt inventory significantly higher than that of the W 17x17 burnable poison rod assembly. This relatively rare control component yields such a large Co-60 gamma source that its presence within the loaded fuel assemblies increases the cask external gamma dose rates substantially. Therefore, the presence of this control component is not assumed as the basis for the general license of the TranStor™ Storage Cask. To store these components, special calculations will have to be performed to allow their placement in either a spent fuel basket or in another specially designed basket.

5.2.1.2 Gamma Source Axial Profile

The shielding analyses model the axial burnup profile present in the fuel. The PWR (Reference 5.3) and BWR (Reference 5.4) axial burnup profiles are shown in Figures 5.2-1 and 5.2-2, respectively. Gamma measurements on large numbers of fuel assemblies have verified the shapes of these profiles for PWR and BWR fuel. Measurements have also verified that fuel assemblies reach equilibrium with these profiles after a moderate amount of burnup. As the figures show, the peak to average burnup ratio is about 1.1 for both PWR and BWR fuel.

Since the fuel region gamma source comes primarily from fission products, the fuel region gamma source density (gammas/sec-cm^3) at a given location is assumed to be directly proportional to the fuel burnup level at that location. Therefore, the gamma source density at a given axial location is determined by multiplying the gamma source strength output from the database for the assembly average burnup level by the relative burnup level for that axial location shown in Figures 5.2-1 and 5.2-2. Thus, the gamma source density in the peak burnup level sections of the fuel is about 10% higher than the gamma source density that corresponds to the assembly average burnup level.

5.2.1.3 Non-Fuel Assembly Region Gamma Sources

The shielding analyses explicitly calculate the dose rate contributions from the Co-60 gamma sources of the three non-fuel regions of the spent fuel assembly - the bottom nozzle region, the gas plenum region, and the top nozzle region. The plenum region gamma source comes from activated plenum springs. The top and bottom nozzle sources come from activated stainless steel and inconel components in the assembly nozzles. Separate shielding analyses are performed to determine the gamma dose rate contributions, at every detector location, from each of these three non-fuel gamma source regions.

As discussed in Section 5.2.1.1, the OCRWM database (Reference 5.1) provides the data necessary to calculate a core region Co-60 activation factor (Ci of Co-60 per gram of initial cobalt) as a function of assembly burnup level, cooling time, and initial enrichment. Activation level adjustment factors have been determined (Reference 5.5) for each of the three assembly non-fuel regions. The core region Co-60 activation factor is multiplied by these adjustment factors to yield a Co-60 activation factor for each non-fuel region. The cobalt quantity present in each non-fuel region is multiplied by the corresponding Co-60 activation factor to yield a Co-60 activity for that region.

The OCRWM database (Reference 5.1) gives the quantities of various metals (such as stainless steels and inconels) present in each non-fuel region of all of the major PWR and BWR assembly types. Maximum cobalt concentrations for each metal type are also given. This allows determination of total initial cobalt inventories for each non-fuel region of each LWR assembly type. For each non-fuel region, the shielding analyses conservatively base the Co-60 gamma source on the assembly type with the largest cobalt inventory for the non-fuel region in question. The presence of control components is considered in the determination of the bounding cobalt inventory for each non-fuel region.

After the Co-60 activation levels are determined for each non-fuel region, they are converted into total cask gamma sources for each of the three non-fuel cask regions. In the non-fuel region gamma source shielding analyses, the Co-60 gamma source is explicitly modeled as two equal strength line energy sources, 1.173 MeV and 1.333 MeV. The source strength of each of these line energies is equal to the per assembly Co-60 activity (in Ci), times the number of assemblies in the cask (24 PWR, 61 BWR), times 3.7×10^{10} disintegrations/curie. Gamma source strength data are presented in Table 5.2-3 and 5.2-4 for each of the PWR and BWR non-fuel assembly regions. Data are presented for each PWR and BWR fuel burnup level and cooling time combination.

5.2.2 Neutron Source

As with the gamma source, the neutron source strengths for each of the analyzed cases are taken from the DOE-OCRWM LWR Computer Database (Reference 5.1). The database, which is based upon ORIGEN-2 calculations, gives the total neutron source strength per MTU of fuel as a function of burnup level, cooling time, and assembly initial enrichment. As stated in Section 5.2, the shielding analyses assume a conservative lower bound initial enrichment level. This is especially important for neutrons, because the neutron source strength increases substantially as initial enrichment goes down for LWR fuel of a given burnup level.

Since the neutron source is primarily due to spontaneous fission of Cm-244, the neutron source spectrum is very similar to the fission spectrum of this nuclide. Unlike the gamma source spectrum, the neutron source spectrum does not vary significantly with fuel burnup level or cooling time. The actual PWR and BWR neutron source spectra used in the shielding analyses are taken from the DOE MPC Specification (Reference 5.6).

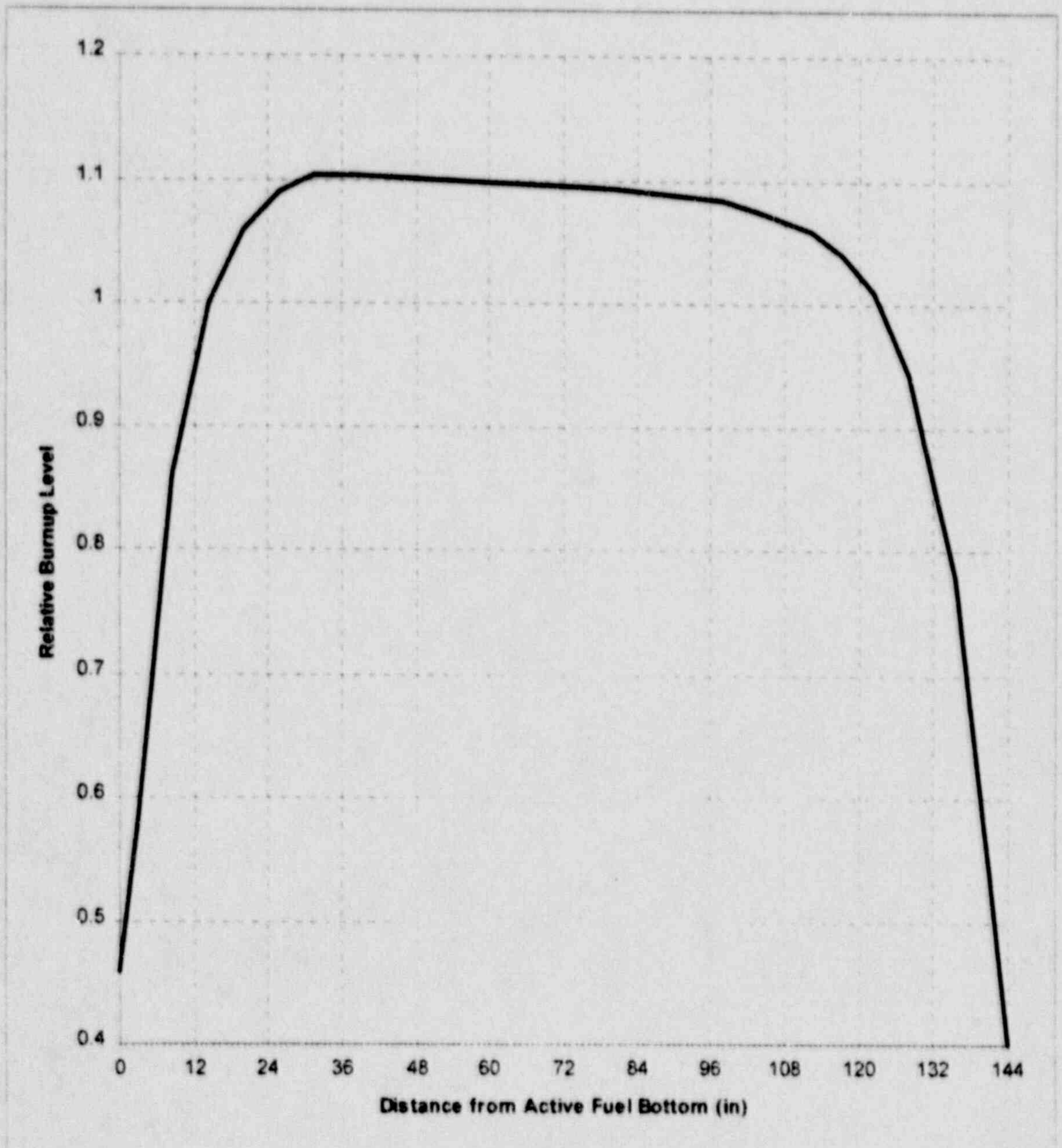


FIGURE 5.2-1

PWR AXIAL SOURCE DISTRIBUTION (Ref. 3)

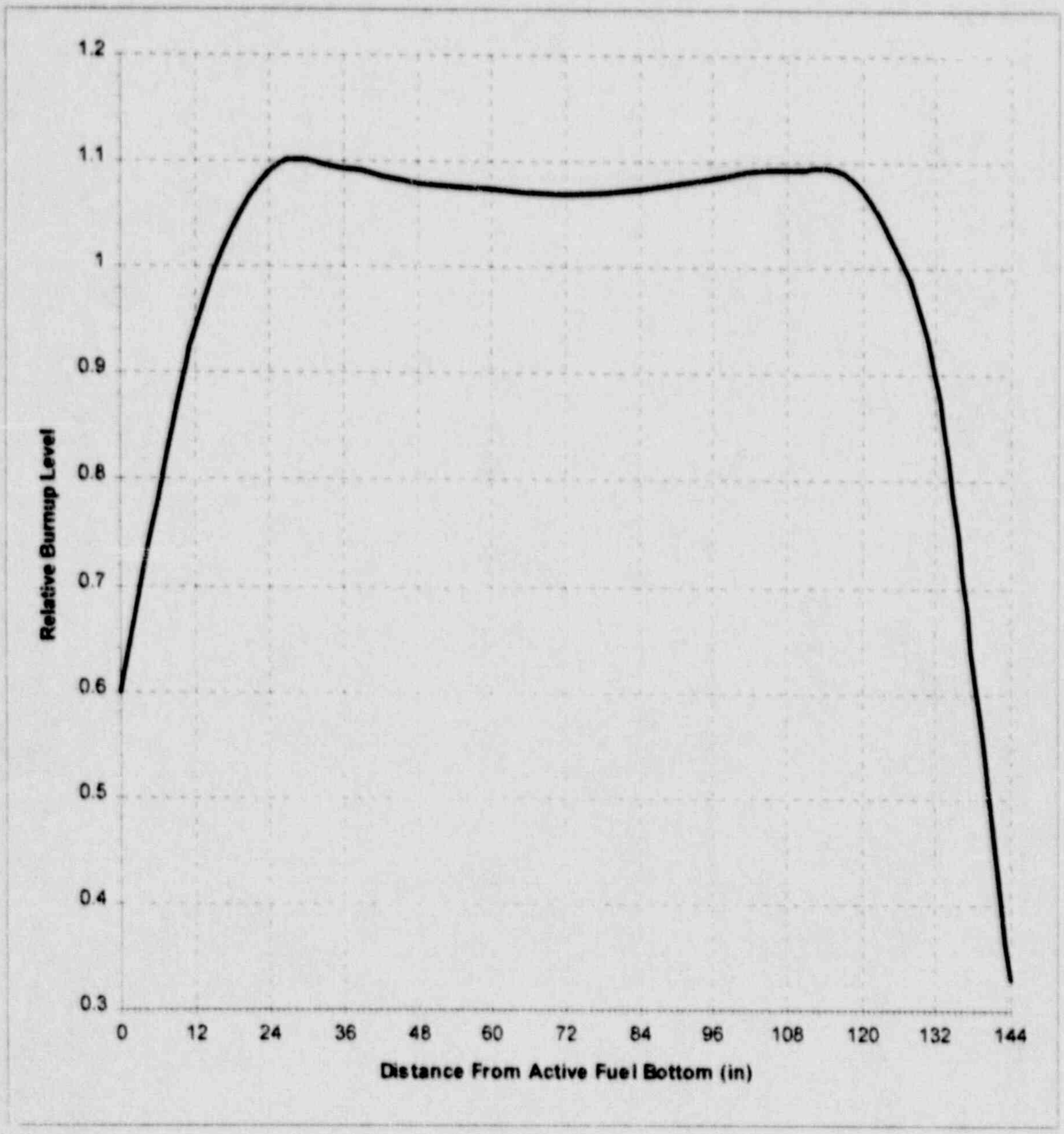


FIGURE 5.2-2

BWR AXIAL SOURCE DISTRIBUTION (Ref. 4)

TABLE 5.2-3

PWR ASSEMBLY NON-FUEL REGION GAMMA SOURCE STRENGTHS (gammas/sec-cask)

Assembly Non-Fuel Region	Fuel Burnup Level (GWd/MTU)	Cooling Time (years)	1.173 MeV Gamma Source Strength	1.333 MeV Gamma Source Strength
Bottom Nozzle	40	5	5.021E+12	5.021E+12
Gas Plenum	40	5	7.347E+12	7.347E+12
Top Nozzle	40	5	1.538E+13	1.538E+13
Bottom Nozzle	45	6	4.717E+12	4.717E+12
Gas Plenum	45	6	6.902E+12	6.902E+12
Top Nozzle	45	6	1.445E+13	1.445E+13
Bottom Nozzle	50	8	3.857E+12	3.857E+12
Gas Plenum	50	8	5.643E+12	5.643E+12
Top Nozzle	50	8	1.181E+13	1.181E+13
Bottom Nozzle	60	13	2.242E+12	2.242E+12
Gas Plenum	60	13	3.280E+12	3.280E+12
Top Nozzle	60	13	6.868E+12	6.868E+12

TABLE 5.2-4

BWR ASSEMBLY NON-FUEL REGION GAMMA SOURCE STRENGTHS (gamma/sec-cask)

Assembly Non-Fuel Region	Fuel Burnup Level (GWd/MTU)	Cooling Time (years)	1.173 MeV Gamma Source Strength	1.333 MeV Gamma Source Strength
Bottom Nozzle	35	5	2.254E+13	2.254E+13
Gas Plenum	35	5	7.973E+12	7.973E+12
Top Nozzle	35	5	2.166E+13	2.166E+13
Bottom Nozzle	40	6	2.195E+13	2.195E+13
Gas Plenum	40	6	7.764E+12	7.764E+12
Top Nozzle	40	6	2.110E+13	2.110E+13
Bottom Nozzle	45	7	1.919E+13	1.919E+13
Gas Plenum	45	7	6.786E+12	6.786E+12
Top Nozzle	45	7	1.844E+13	1.844E+13
Bottom Nozzle	50	8	1.677E+13	1.677E+13
Gas Plenum	50	8	5.932E+12	5.932E+12
Top Nozzle	50	8	1.612E+13	1.612E+13

The neutron source descriptions for PWR and BWR fuel are shown in Tables 5.2-5 and 5.2-6. Tables 5.2-5 and 5.2-6 give per cask neutron source strengths for each energy group, along with total neutron source strengths, for design basis PWR and BWR fuel. The normalized neutron energy spectrum is also shown for each case.

The shielding analyses model the axial variation in the neutron source density (neutrons/sec-cm^3) due to the axial burnup profiles shown in Figures 5.2-1 and 5.2-2. A survey of OCRWM database results showed that the neutron source strength varies as the burnup level raised to a power of 4.2. Therefore, the neutron source density at any given axial location in the fuel is assumed to equal the neutron source strength output by the database for the assembly average burnup level, times the relative burnup level at that location raised to a power of 4.2.

If the peak relative burnup level shown in Figures 5.2-1 and 5.2-2 (~ 1.1) is raised to a power of 4.2, the result is roughly 1.5. Therefore, the neutron source density in the peak burnup regions of the fuel is actually about 50% greater than that output by the database for the assembly average burnup level. It should be noted that this strongly non-linear dependence causes the total fuel region neutron source strength to be over 13% higher than it would be if the assembly axial burnup profile were not accounted for.

TABLE 5.2-5
PWR FUEL NEUTRON SOURCE STRENGTHS
(Neutrons/sec-cask)

Neutron Energy Range (MeV)	Neutron Source Strength 40 GWd - 5 yr. cool Enrichment = 3.02%	Neutron Source Strength 45 GWd - 6 yr. cool Enrichment = 3.30%	Neutron Source Strength 50 GWd - 8 yr. cool Enrichment = 3.56%	Neutron Source Strength 60 GWd - 13 yr. cool Enrichment = 4.03%	Normalized Neutron Source Spectrum
6.43 - 20.0	1.695E+08	2.197E+08	2.661E+08	3.525E+08	0.0185
3.0 - 6.43	1.925E+09	2.494E+09	3.021E+09	4.002E+09	0.21
1.85 - 3.0	2.126E+09	2.755E+09	3.337E+09	4.421E+09	0.232
1.4 - 1.85	1.201E+09	1.556E+09	1.884E+09	2.496E+09	0.131
0.9 - 1.4	1.622E+09	2.102E+09	2.546E+09	3.373E+09	0.177
0.4 - 0.9	1.769E+09	2.292E+09	2.776E+09	3.678E+09	0.193
0.1 - 0.4	3.464E+08	4.489E+08	5.438E+08	7.203E+08	0.0378
Total	9.165E+09	1.188E+10	1.439E+10	1.906E+10	1

TABLE 5.2-6

BWR FUEL NEUTRON SOURCE STRENGTHS
(Neutrons/sec-cask)

Neutron Energy Range (MeV)	Neutron Source Strength 35 GWd - 5 yr. cool Enrichment = 2.75%	Neutron Source Strength 40 GWd - 6 yr. cool Enrichment = 3.00%	Neutron Source Strength 45 GWd - 7 yr. cool Enrichment = 3.30%	Neutron Source Strength 50 GWd - 8 yr. cool Enrichment = 3.56%	Normalized Neutron Source Spectrum
6.43 - 20.0	1.826E+08	3.213E+08	3.343E+08	3.455E+08	0.0178
3.0 - 6.43	2.151E+09	3.785E+09	3.939E+09	4.070E+09	0.2097
1.85 - 3.0	2.504E+09	4.406E+09	4.585E+09	4.737E+09	0.2441
1.4 - 1.85	1.348E+09	2.372E+09	2.468E+09	2.550E+09	0.1314
0.9 - 1.4	1.781E+09	3.133E+09	3.261E+09	3.369E+09	0.1736
0.4 - 0.9	1.917E+09	3.373E+09	3.510E+09	3.627E+09	0.1869
0.1 - 0.4	3.754E+08	6.606E+08	6.874E+08	7.103E+08	0.0366
Total	1.026E+10	1.805E+10	1.878E+10	1.941E+10	1

5.3 Model Specification

The storage and transfer cask geometries were modeled with 2-D radially symmetric shielding models or with several finite slab and finite height cylinder geometry shielding models. These models accurately reflect the shielding materials and thicknesses present in the actual cask geometries.

5.3.1 Radial and Axial Shielding Configurations

In the TranStor™ Storage Cask, radial shielding is provided by the steel basket shell, the steel inner liner of the cask, and the thick concrete shell which makes up the main body of the storage cask. The outlet vents penetrate the cask radial shielding near the cask top. However, two sharp bends are placed in the duct path which greatly limit radiation streaming through the outlet duct.

The storage cask bottom shielding consists of the steel basket bottom plate, the steel storage cask bottom plate, and a thick bottom layer of concrete. The inlet vent structure creates penetrations in the cask bottom end shielding, but radiation streaming is greatly minimized by the geometry of the inlet duct structure, which contains several sharp bends in the duct path. At the cask top, shielding is provided primarily by the basket's thick steel lids, with some additional shielding being provided by the steel storage cask lid.

For the transfer cask, radial gamma shielding is provided by the steel basket shell, the transfer cask inner and outer steel shells, and a thick lead shell in the transfer cask. Radial neutron shielding is provided by a thick layer of a neutron shielding material that is described in Section 5.3.2. Bottom shielding is provided by the thick steel bottom doors of the transfer cask. As with the storage cask, top end shielding is provided by the steel basket lids.

As discussed in Section 5.4.1, the storage cask radial dose rates are calculated by (source ratio) scaling from previous shielding analyses performed for the VSC-24 cask system. These analyses (gamma and neutron) were performed using a finite height cylindrical model. This model is illustrated in Figure 5.3-1. In the shielding models, the complex basket internal structure (and fuel assemblies) are not explicitly modeled. All of the materials present in the basket interior are mixed, or "homogenized", into a single homogenous material that fills the basket interior. This process is discussed further in Section 5.3.2.

The gamma shielding analyses for the transfer cask side and ends were performed using a 2-D radially symmetric model. This model is illustrated in Figure 5.3-2. This model accurately represents the entire basket and transfer cask geometries. The storage cask lid, however, is added to the model and placed above the basket top lid. This allows storage cask top gamma dose rates to be calculated using this model, as discussed in Section 5.4.

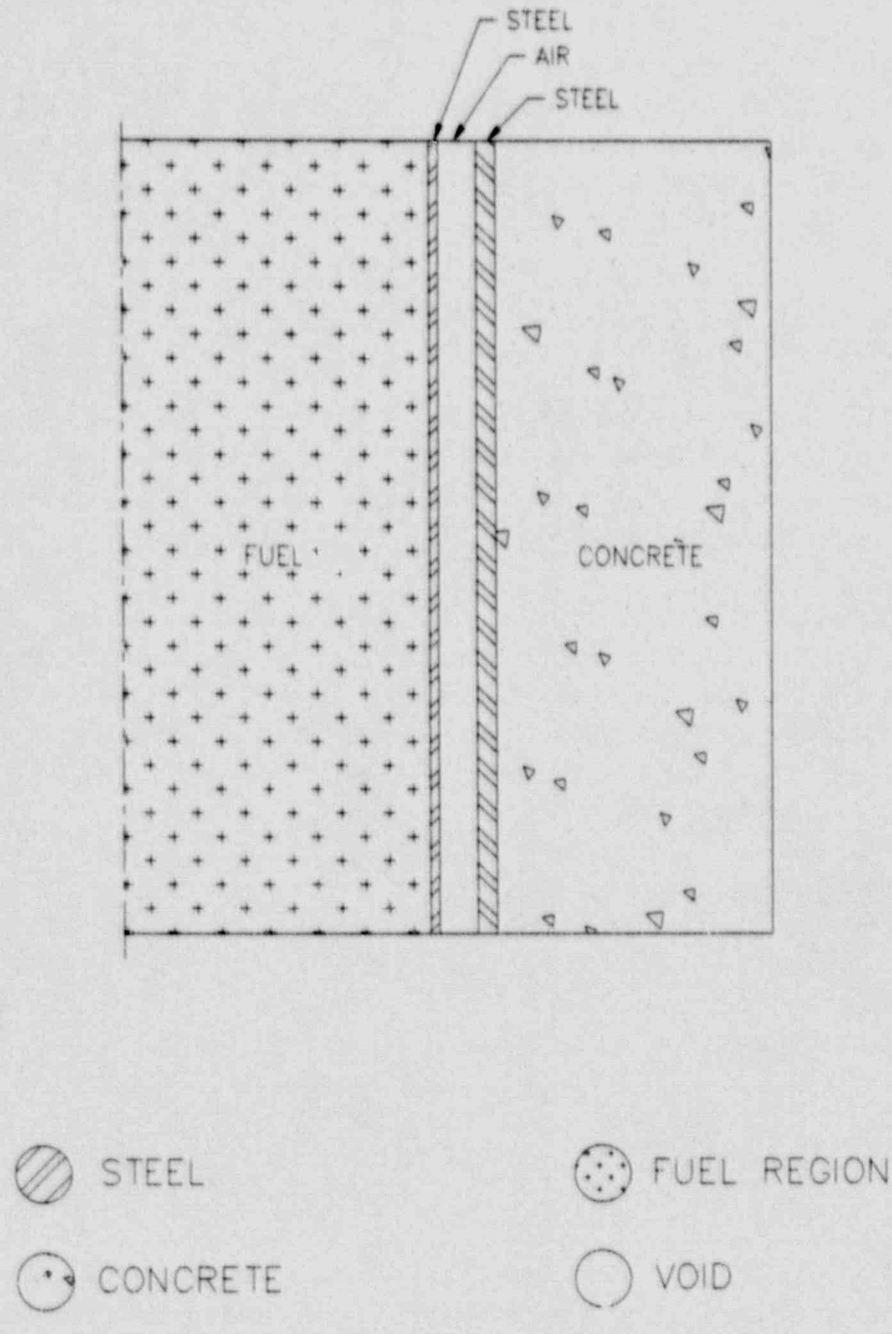


FIGURE 5.3-1

STORAGE CASK RADIAL SHIELDING MODEL

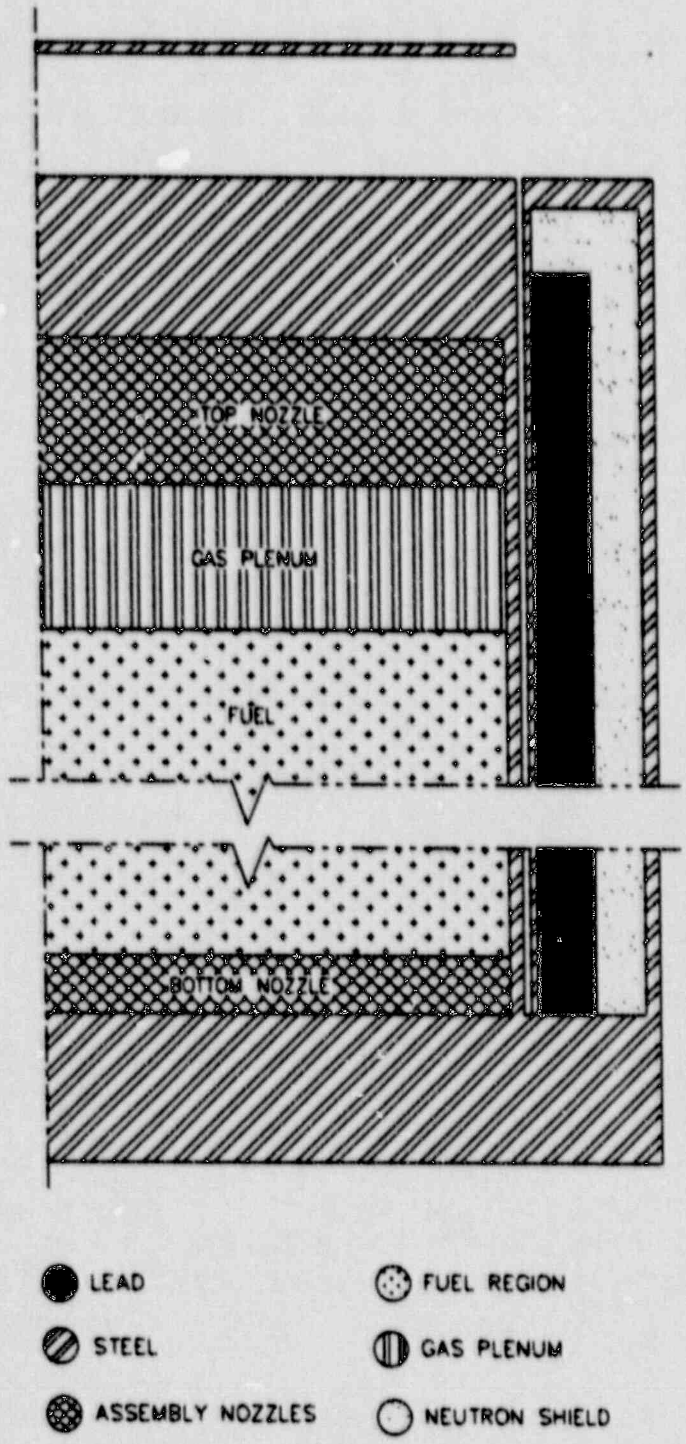


FIGURE 5.3-2

2-D GAMMA SHIELDING MODEL FOR THE TRANSFER CASK AND FOR THE STORAGE CASK TOP

In the axial model, the basket interior is actually subdivided into four axial sub-regions in the shielding models, an active fuel region, a bottom nozzle region, a gas plenum region, and a top nozzle region. In the shielding models, the height of the active fuel region is 144 inches. The heights of the bottom nozzle, gas plenum, and top nozzle regions are 4 inches, 10 inches, and 10 inches, respectively. The basis for these assumed heights is discussed in detail in Reference 1.1.

As discussed in Section 5.3.2, each of these axial sub-regions has a different homogenized material description, due to the physical differences between different axial sections of the fuel assemblies and the basket internal structures. Thus, four homogenous materials are defined, each of which completely fills its axial sub-section of the basket interior volume, as illustrated in Figure 5.3-2. Only one of these axial sub-sections, the active fuel region, is modeled in the storage cask side shielding model shown in Figure 5.3-1.

The neutron shielding analysis for the storage cask top was performed using a 2-D radially symmetric model. In this model, the storage cask radial shielding and the steel storage cask top ring (which lies over the ventilation duct) are modeled. This 2-D model is illustrated in Figure 5.3-3. For neutrons, radiation scattering in the cask side shielding materials may have a significant effect on cask top dose rates. This model explicitly treats these scattering effects.

The cask top neutron shielding model includes three of the basket interior axial sub-sections - the fuel region, the gas plenum region, and the top nozzle region. The cask bottom geometry, including the bottom nozzle region of the basket interior, are not included in this model because these regions have no effect on cask top dose rates.

Neutron dose rates for the transfer cask side and ends are calculated using a single 2-D radially symmetric model. This model is illustrated in Figure 5.3-4. This model accurately reflects the sides and both ends of the transfer cask and basket geometries. All four axial sub-sections of the basket interior are included in the model. The gap between the basket and the transfer cask is not included in this model. The basket shell and the transfer cask inner shell are modeled as a single piece of steel.

Gamma and neutron dose rates at the inlet vents are calculated using a 2-D radially symmetric model of the cask bottom and the inlet vent structure. This model is illustrated in Figure 5.3-5. The radially symmetric model is conservative in that it assumes all voids cover the entire azimuthal span, even though they do not in some regions of the inlet vent structure.

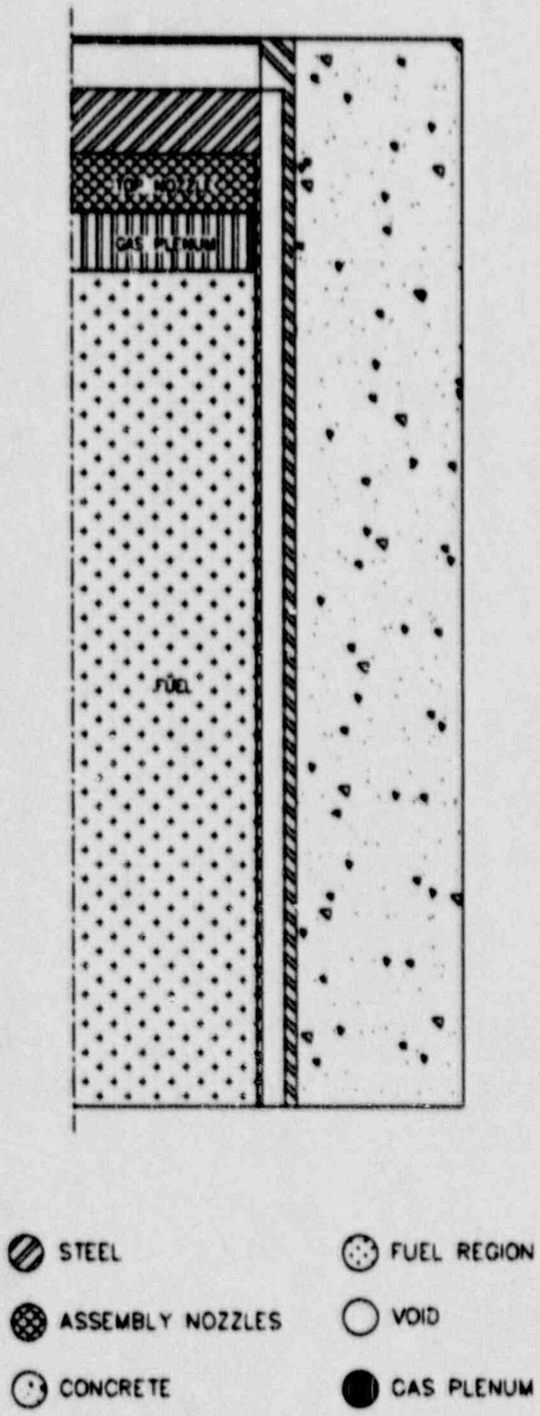


FIGURE 5.3-3

**STORAGE CASK TOP
NEUTRON SHIELDING MODEL**

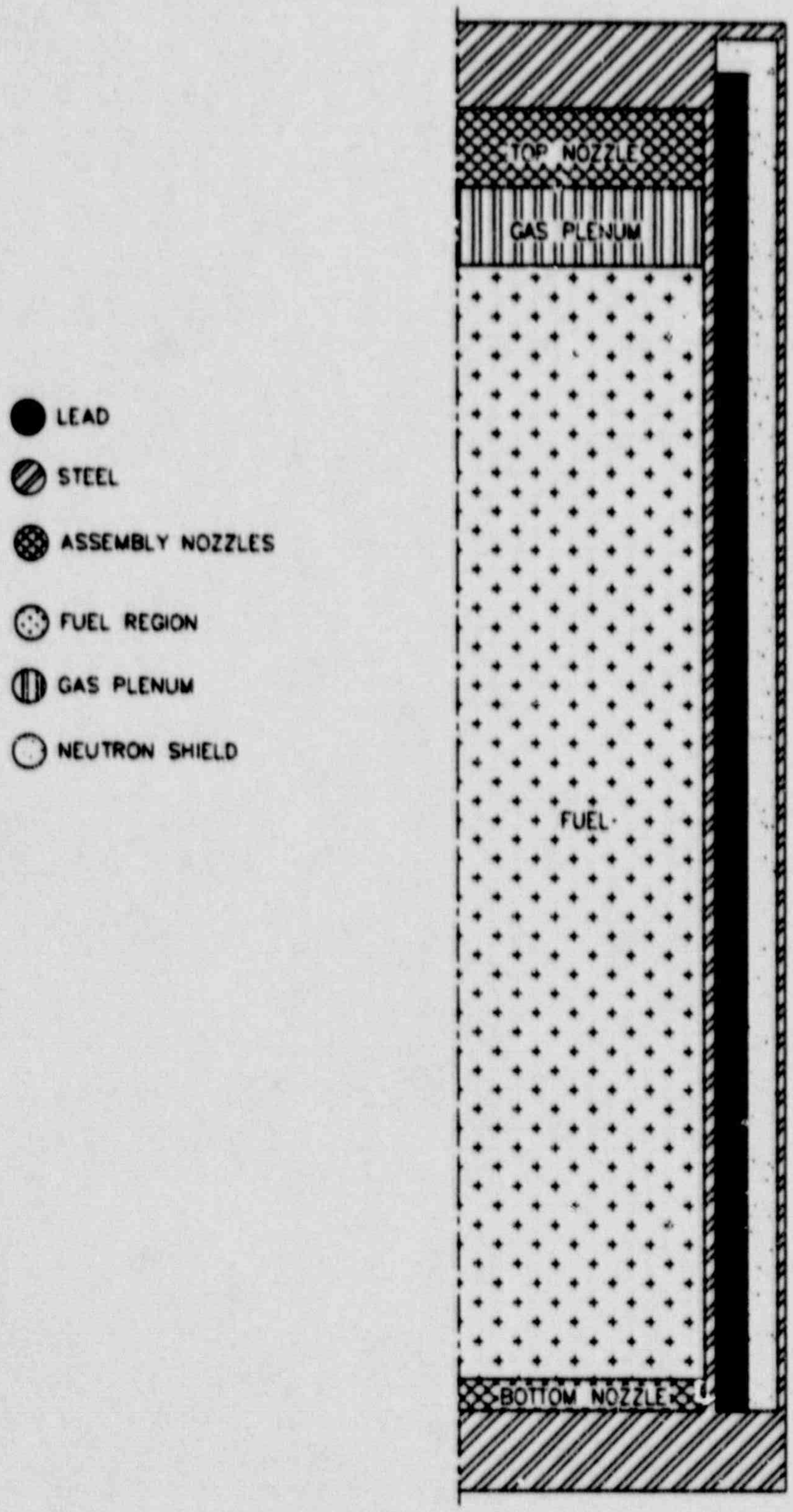


FIGURE 5.3-4

2-D TRANSFER CASK
NEUTRON SHIELDING MODEL

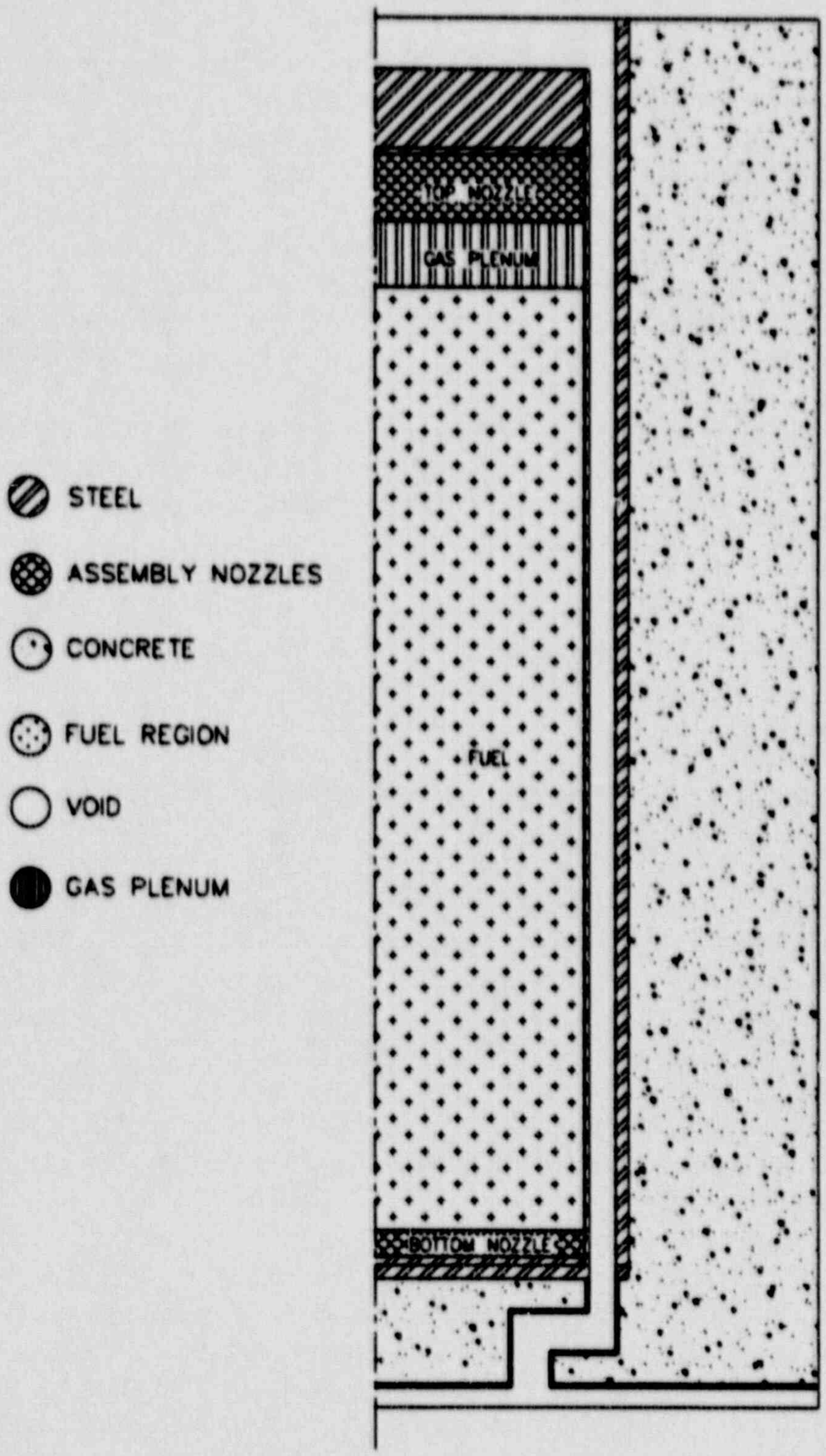


FIGURE 5.3-5

**TranStor™ STORAGE CASK
INLET DUCT MCNP SHIELDING MODEL**

5.3.2 Shield Regional Densities

The TranStor™ storage and transfer casks contain stainless steel, carbon steel, lead, concrete, neutron shielding material, and mixtures of materials occupying the basket interior. An elemental breakdown for each of the shielding materials present in the cask systems are shown in Tables 5.3-1 through 5.3-4. Table 5.3-1 gives the material description for the storage cask shielding materials. Table 5.3-2 shows the transfer cask shielding materials. Tables 5.3-3 and 5.3-4 give the elemental breakdowns for the PWR and BWR basket interior regions, as discussed later in this section. The mass density (in grams/cc) and the atom density (in atoms/barn-cm) for each element is presented in each table.

In the transfer cask shielding models, the neutron shield region is filled with a homogenous material that represents the neutron shielding material and the heat transfer fin material (carbon steel). The heat transfer fins occupy 8.8% of the neutron shield region volume, with neutron shielding material occupying the other 91.2%.

The neutron shield region elemental densities presented in Table 5.3-2 are based upon GESC NS-4-FR neutron shielding material. The neutron shielding material contains 2% (by volume) boron-carbide to control secondary gamma production. The neutron shielding material is assumed to have lost 2% of its total density, in the form of water, to account for the effects of long term exposure to elevated temperatures. Tests performed on stainless steel clad samples of the neutron shield material (Reference 5.7) predict a weight loss under 2% (in the form of water) after exposure to temperatures over 300 °F for an extended period.

Although the GESC neutron shielding material is used as the design basis material in the shielding analyses, another neutron shielding material, RX-244, may be used in the transfer casks, depending on GESC's availability. The elemental densities for a neutron shield region containing RX-244 are also shown in Table 5.3-2. Calculations show that neutron dose rates on the transfer cask side will increase by 68% over those shown in Table 5.1-2 if RX-244 is used in place of GESC NS-4-FR.

The basket internal structure is not explicitly modeled in the shielding analyses. The basket interior is divided into four axial sub-regions, each of which is filled with a homogenized material. The sub-regions correspond to the active fuel region, bottom nozzle region, gas plenum region, and top nozzle regions of the assemblies. For each sub-region, elemental densities are determined by dividing the total mass that is present (for each element) by the total volume of the sub-region. The elemental densities for the four regions are different for the PWR fuel and BWR fuel cases. This is the only physical difference between the PWR and the BWR shielding models. The PWR basket interior elemental densities are presented in Table 5.3-3. The BWR basket interior densities are presented in Table 5.3-4.

TABLE 5.3-1

TranStor™ STORAGE CASK SHIELDING MODEL MATERIAL DESCRIPTIONS

Shielding Model Material	Elemental Densities (g/cc)	Atom Densities (atoms/barn-cm)
SS-304	Cr: 1.505 Mn: 0.159 Fe: 5.465 Ni: 0.733 Si: 0.059 Total: 7.921	Cr: 1.743 e-2 Mn: 1.74 e-3 Fe: 5.894 e-2 Ni: 7.52 e-3 Si: 1.27 e-3 Total: 8.690 e-2
Carbon Steel	Fe: 7.821	Fe: 8.435 e-2
Concrete	H: 0.013 O: 1.165 Na: 0.040 Mg: 0.006 Al: 0.107 Si: 0.737 S: 0.003 K: 0.045 Ca: 0.194 Fe: 0.029 Total: 2.339	H: 7.77 e-3 O: 4.39 e-2 Na: 1.05 e-3 Mg: 1.49 e-4 Al: 2.39 e-3 Si: 1.58 e-2 S: 5.64 e-5 K: 6.93 e-4 Ca: 2.92 e-3 Fe: 3.13 e-4 Total: 7.504 e-2
Air	N: 9.765 e-4 O: 2.997 e-4 Ar: 1.665 e-5 Total: 1.293 e-3	N: 4.199 e-5 O: 1.128 e-5 Ar: 2.51 e-7 Total: 5.352 e-5

TABLE 5.3-2

TranStor™ TRANSFER CASK SHIELDING MODEL MATERIAL DESCRIPTIONS

Shielding Model Material	Elemental Densities (g/cc)	Atom Densities (atoms/barn-cm)
SS-304	(same as Table 5.3-1)	(same as Table 5.3-1)
Carbon Steel	Fe: 7.821	Fe: 8.435 e-2
Lead	Pb: 11.34	Pb: 3.296 e-2
Neutron Shield Region Mixture (GESC NS4-FR) (Neutron Shield + Heat Transfer Fins)	H: 0.087 B-10: 0.007 B-11: 0.029 C: 0.426 N: 0.030 O: 0.616 Al: 0.323 Fe: 0.688	H: 5.225 e-2 B-10: 3.912 e-4 B-11: 1.605 e-3 C: 2.138 e-2 N: 1.295 e-3 O: 2.319 e-2 Al: 7.205 e-3 Fe: 7.243 e-3
Neutron Shield Region Mixture (RX-244) (Neutron Shield + Heat Transfer Fins)	H: 0.065 B-10: 0.008 B-11: 0.038 C: 0.020 O: 0.966 Na: 0.001 Al: 0.261 Si: 0.010 S: 0.105 Ca: 0.131 Mn: 0.001 Fe: 0.690 Total: 2.296	H: 3.885 e-2 B-10: 5.04 e-4 B-11: 2.07 e-3 C: 1.00 e-3 O: 3.636 e-2 Na: 2.58 e-5 Al: 5.83 e-3 Si: 2.25 e-4 S: 1.97 e-3 Ca: 1.97 e-3 Mn: 9.0 e-6 Fe: 7.44 e-3 Total: 9.625 e-2

TABLE 5.3-3

PWR FUEL BASKET INTERIOR HOMOGENIZED MATERIAL DESCRIPTIONS

Shielding Model Material	Elemental Densities (g/cc)	Atom Densities (atoms/b-cm)
Active Fuel Region Mixture*	U-238: 1.476 O: 0.198 Zr: 0.319 Fe: 0.481 Al: 0.044 C: 0.004 B-10: 0.003 B-11: 0.013 Total: 2.538	U-238: 3.736 e-3 O: 7.472 e-3 Zr: 2.103 e-3 Fe: 5.192 e-3 Al: 9.83 e-4 C: 2.18 e-4 B-10: 1.72 e-4 B-11: 7.06 e-4 Total: 2.058 e-2
Bottom Nozzle Region Mixture	Cr: 0.207 Mn: 0.022 Fe: 1.189 Ni: 0.101 Zr: 0.227 C: 0.003 Total: 1.749	Cr: 2.397 e-3 Mn: 2.40 e-4 Fe: 1.282 e-2 Ni: 1.034 e-3 Zr: 1.502 e-3 C: 1.75 e-4 Total: 1.817 e-2
Gas Plenum Region Mixture	Zr: 0.319 Fe: 0.504 Ni: 0.025 Total: 0.848	Zr: 2.103 e-3 Fe: 5.436 e-3 Ni: 5.77 e-4 Total: 7.791 e-3
Top Nozzle Region Mixture	Cr: 0.115 Mn: 0.012 Fe: 0.419 Ni: 0.056 Zr: 0.031 C: 0.002 Total: 0.635	Cr: 1.673 e-3 Mn: 1.67 e-4 Fe: 5.658 e-3 Ni: 7.22 e-4 Zr: 2.03 e-4 C: 1.22 e-4 Total: 8.545 e-3

* Boron, carbon and aluminum densities are included only in the neutron shielding models.

TABLE 5.3-4

BWR FUEL BASKET INTERIOR HOMOGENIZED MATERIAL DESCRIPTIONS

Shielding Model Material	Elemental Densities (g/cc)	Atom Densities (atoms/b-cm)
Active Fuel Region Mixture*	U-238: 1.596 O: 0.215 Zr: 0.322 Fe: 0.694 C: 0.002 B-10: 0.002 B-11: 0.007 Al: 0.025 Total: 2.863	U-238: 4.038 e-3 O: 8.075 e-3 Zr: 2.126 e-3 Fe: 7.482 e-3 C: 1.24 e-4 B-10: 9.75 e-5 B-11: 4.00 e-4 Al: 5.61 e-4 Total: 2.290 e-2
Bottom Nozzle Region Mixture	Cr: 0.245 Mn: 0.026 Fe: 1.484 Ni: 0.119 Zr: 0.188 C: 0.004 Total: 2.066	Cr: 2.836 e-3 Mn: 2.83 e-4 Fe: 1.600 e-2 Ni: 1.224 e-3 Zr: 1.239 e-3 C: 2.07 e-4 Total: 2.179 e-2
Gas Plenum Region Mixture	Zr: 0.322 Fe: 0.697 Ni: 0.004 Total: 1.023	Zr: 2.126 e-3 Fe: 7.518 e-3 Ni: 3.7 e-5 Total: 9.681 e-3
Top Nozzle Region Mixture	Cr: 0.078 Mn: 0.008 Fe: 0.283 Ni: 0.038 Zr: 0.075 C: 0.001 Total: 0.483	Cr: 9.03 e-4 Mn: 9.0 e-5 Fe: 3.055 e-3 Ni: 3.90 e-4 Zr: 4.96 e-4 C: 6.6 e-5 Total: 5.000 e-3

* Boron, carbon and aluminum densities are only included in the neutron shielding models.

The active fuel region homogenized material includes the UO_2 fuel material, the Zircaloy cladding material of the fuel rods, the stainless steel cladding of control components, and the carbon steel of the fuel support sleeves. Steel structures which do not envelop the fuel assemblies, such as the center and corner structural support tubing, are conservatively neglected in the homogenized material density calculation. Also, in the neutron shielding calculations, the neutron poison sheet materials are included in the fuel region material description.

The Zircaloy is modeled as pure zirconium for the calculations of basket interior elemental densities. The carbon steel is modeled as pure iron. The fuel region uranium and oxygen densities presented in Table 5.3-3 are based upon 24 PWR assemblies with a maximum fuel loading of 0.469 MTU/assembly. The uranium and oxygen densities presented in Table 5.3-4 are based upon 61 BWR assemblies with a maximum fuel loading of 0.197 MTU/assembly. For both the PWR and BWR cases, the calculated elemental densities are based upon a fuel region height of 144 inches.

The gas plenum region homogenized material description includes the materials in the assembly Zircaloy cladding, the carbon steel support sleeves, and the inconel plenum springs. The top and bottom nozzle region material descriptions include the metal masses of the assembly top and bottom nozzles, respectively, along with a small amount of zirconium to model the solid Zircaloy end caps of the fuel rods. The bottom nozzle region also contains additional iron density to model the presence of the carbon steel support sleeves. The support sleeves to not extend into the top nozzle region.

Elemental densities for the three non-fuel regions are calculated for PWR fuel and for BWR fuel, and are presented in Tables 5.3-3 and 5.3-4, respectively. The non-fuel region material densities are based upon the same PWR and BWR assembly types that were assumed in the non-fuel region source term calculations (i.e. the assembly types which produced the bounding non-fuel region gamma source strengths).

The elemental density calculations for the basket interior regions are discussed in further detail in Reference 1.1.

5.4 Shielding Evaluation

Different codes and methodologies were used to calculate gamma and neutron dose rates for the two casks (storage and transfer) at the various detector locations. These methodologies are briefly described in the following sections. The codes and methodologies used to calculate dose rates for the TranStor™ Storage Cask are very similar to those used to calculate dose rates for the previously licensed, and presently employed, VSC-24 storage cask system. Measured dose rate data for actual loaded VSC-24 system storage and transfer casks confirms that these codes and methodologies calculate conservatively high dose rates.

5.4.1 Storage Cask Side Dose Rate Calculation

The radial shielding geometry of a TranStor™ Storage Cask containing a TranStor™ basket is virtually identical to that of the previously licensed VSC-24 storage cask system. The total steel and concrete thicknesses for the radial shielding configurations are exactly the same for the two systems. There are some differences in the basket internal structure between the two systems, with the TranStor™ basket employing thicker fuel sleeve walls and having a greater amount of structural support steel around the edge of the basket interior. This makes the radial shielding of the TranStor™ system somewhat greater than that of the VSC-24 system. Hence, dose rates for the TranStor™ Storage Cask side can be conservatively calculated using the VSC-24 system dose rate results and using source ratio techniques. The storage cask side gamma and neutron dose rates have been previously calculated for the licensed VSC-24 storage cask system (Reference 1.2) using the ANISN-PC computer code and the finite height cylindrical model illustrated in Figure 5.3-1.

The neutron source spectra for the two cask systems are virtually identical. Therefore, TranStor™ Storage Cask side neutron dose rates are calculated by multiplying the neutron dose rates calculated for the VSC-24 system by the ratio of the total neutron source strengths (TranStor / VSC-24).

The situation is somewhat more complicated for cask side gamma dose rates, because the gamma spectrum for the two cask systems (which are based upon different design basis fuel burnup levels) are not the same. The gamma dose rates are conservatively scaled with the significant gamma energy line that produces the worst source strength ratio (i.e. the highest TranStor™ system source strength as compared to the VSC-24 system source strength).

Thus, source strength ratios (TranStor™ source strength over VSC-24 source strength) are determined for each gamma energy line. The highest of these ratios is multiplied by the cask side gamma dose rates previously calculated for the VSC-24 system to yield the cask side gamma dose rates for the TranStor™ storage cask. This process is performed (for the gamma and neutron dose rates) for each of the eight PWR and BWR cases that were analyzed for the TranStor™ system.

The TranStor™ cask side gamma and neutron dose rates were calculated using VSC-24 system shielding results that did not treat axial burnup profile effects. Therefore, as with the VSC-24, these calculated TranStor™ dose rates are multiplied by factors to account for axial profile. The resulting dose rate represent peak cask side dose rates. As discussed in Section 5.2.1.2 and 5.2.2, the gamma source density profile has a peaking factor of 1.1, and the neutron source density profile has a peaking factor of 1.5. Therefore, the TranStor™ cask side gamma dose rates are multiplied by an additional factor of 1.1, and the neutron dose rate results are multiplied by 1.5 to yield the peak side dose rates.

5.4.2 Storage Cask Top Neutron Dose Rate Calculation

Neutron dose rates on the TranStor™ Storage Cask top were calculated using the MCNP monte-carlo shielding code (Reference 5.9) and the 2-D radially symmetric model shown in Figure 5.3-3. The MCNP code allows shielding geometries to be explicitly modeled in three dimensions. An explicit elemental description of all shielding materials can be provided. MCNP automatically and accurately treats all significant forms of particle interaction (scattering, absorption, etc.) using the monte-carlo technique.

As discussed in Section 5.3.1, the cask side geometry is included in the model so that neutron scattering effects on the cask top neutron dose rates may be accurately treated. A small area detector is defined in the center of the storage cask top lid. Neutron particle tallies are taken over this area to determine the cask top neutron dose rate.

The cask top MCNP neutron analysis models the axial burnup profiles shown in Figure 5.2-1 and 5.2-2. This is done by dividing the neutron source region into several axial sub-sections, each having a different neutron source density. The axial sub-regions are described for PWR and BWR fuel in Tables 5.4-1 and 5.4-2. For each sub-section, the axial span is given, along with the relative gamma and neutron source densities (i.e. relative to that which would exist at the assembly average burnup level).

TABLE 5.4-1

RELATIVE BURNUP LEVEL AND SOURCE STRENGTHS FOR PWR ASSEMBLY AXIAL SUB-SECTIONS

Axial Span (inches from fuel bottom)	Relative Burnup Level	Relative Gamma Source Strength	Relative Neutron Source Strength
0 - 7.2	0.59	0.59	0.109
7.2 - 14.4	0.89	0.89	0.613
14.4 - 21.6	1.03	1.03	1.132
21.6 - 28.8	1.07	1.07	1.329
28.8 - 36.0	1.09	1.09	1.436
36.0 - 64.8	1.1	1.1	1.492
64.8 - 100.8	1.09	1.09	1.436
100.8 - 108.0	1.07	1.07	1.329
108.0 - 115.2	1.05	1.05	1.227
115.2 - 122.4	1.02	1.02	1.087
122.4 - 129.6	0.96	0.96	0.842
129.6 - 136.8	0.82	0.82	0.435
136.8 - 144.0	0.56	0.56	0.088

TABLE 5.4-2

RELATIVE BURNUP LEVEL AND SOURCE STRENGTHS FOR BWR ASSEMBLY AXIAL SUB-SECTIONS

Axial Span (inches from fuel bottom)	Relative Burnup Level	Relative Gamma Source Strength	Relative Neutron Source Strength
0 - 7.2	0.7	0.7	0.224
7.2 - 14.4	0.91	0.91	0.673
14.4 - 21.6	1.05	1.05	1.227
21.6 - 28.8	1.08	1.08	1.382
28.8 - 36.0	1.09	1.09	1.436
36.0 - 50.4	1.08	1.08	1.382
50.4 - 100.8	1.07	1.07	1.329
100.8 - 108.0	1.08	1.08	1.382
108.0 - 122.4	1.07	1.07	1.329
122.4 - 129.6	1.03	1.03	1.132
129.6 - 136.8	0.8	0.8	0.392
136.8 - 144.0	0.47	0.47	0.042

5.4.3 Transfer Cask and Storage Cask Top Gamma Dose Rate Calculation

All of the transfer cask gamma dose rates (side, bottom, and top) were calculated using the 2-D radially symmetric model shown in Figure 5.3-2. The dose rates were calculated using the QADS shielding code module of the SCALE-4.3 code package (Reference 5.10). This is a deterministic point-kernel shielding code which allows explicit 3-D modeling of cask geometries and an explicit elemental description of all shielding materials. The QADS module does not explicitly treat radiation scattering effects, but accounts for them through the use of empirically determined buildup factors.

The QADS module calculates dose rates at point locations. As illustrated in Figure 5.1-2, detectors are placed on the transfer cask side, on the transfer cask bottom, and on the basket top lid. Detectors are also placed one meter from the cask side and top. An additional detector is placed three inches down into the basket top lid. This corresponds to the top surface of the shield lid, which forms the basket top surface when the structural lid is not present. This detector will give accurate gamma dose rates for the shield lid surface despite the fact that the structural lid steel is actually present in the model, whereas this steel is actually absent when the shield lid dose rate is of interest. This is because the effects of gamma backscatter off the structural lid steel are not very significant, and because the QADS module does not explicitly treat such scattering effects.

The QADS transfer cask gamma model also accounts for the axial burnup profiles given in Figures 5.2-1 and 5.2-2, using the same methodology that was used for the MCNP storage cask top neutron model. The axial sub-regions and the relative gamma source densities used in the QADS model are shown in Tables 5.4-1 and 5.4-2.

The gamma dose rates at the storage cask top were calculated using the 2-D transfer cask shielding model described above. The shielding geometry of the storage cask top end is identical to that of a transfer cask containing a basket, except that the steel storage cask lid is also present above the basket top. Therefore, the storage cask top lid is simply added to the transfer cask gamma shielding model, as shown in Figure 5.3-2.

Gamma scattering effects are not considered to have a significant effect on storage cask top gamma dose rates, especially in the center of the cask lid, where the dose rates are calculated. Also, the QADS module does not explicitly treat such scattering effects. Therefore, it is known that the shielding geometry on the cask sides will have no effect on the dose rates calculated on the cask top. For this reason, the transfer cask model, with the storage cask lid added, can be used to accurately calculate storage cask top gamma dose rates. Also, since gamma backscattering effects are not significant, and because the QADS module does not treat them, the presence of the storage cask lid in the 2-D QADS model will not have any effect on the gamma dose rates calculated on the surface of the basket lid.

5.4.4 Transfer Cask Neutron Dose Rate Calculation

Neutron dose rates over the entire transfer cask exterior (side, bottom, and top) are calculated using the MCNP code and the 2-D radially symmetric model shown in Figure 5.3-4. As with the storage cask top MCNP model, the transfer cask model explicitly treats the fuel axial burnup profile using the axial sub-sections described in Tables 5.4-1 and 5.4-2.

A large set of area detectors covers the entire transfer cask surface. These detectors are sufficiently small such that significant dose rate variations within the detectors will not occur. The detectors are large enough, however, so that good particle statistics can be achieved for each detector. The detectors on the cask side are ring detectors which span the entire circumference of the transfer cask and cover a narrow axial span. A small disk-shaped detector is placed at the center of each end of the cask to calculate peak cask end dose rates. These detectors are surrounded by a series of ring detector areas which cover the rest of the cask end surface.

5.4.5 Air Inlet Dose Rate Calculation

Gamma and neutron dose rates at the air inlets are calculated using the MCNP code. The calculations are performed using a 2-D radially symmetric model of cask and inlet vent geometry (as shown in Figure 5.3-5). The MCNP code explicitly and accurately treats the particle scattering effects that are of primary importance to duct streaming calculations. In the MCNP models, area detectors are placed over the entire inlet duct opening.

A total of six MCNP runs are performed, a PWR fuel gamma source run, a PWR bottom nozzle gamma source run, a PWR neutron source run, a BWR fuel gamma source run, a BWR bottom nozzle gamma source run, and a BWR neutron source run. The dose rate results from the two gamma source runs are summed to yield total air inlet gamma dose rates. For PWR and BWR fuel, the burnup and cooling time combination which has the highest gamma sources is assumed for the gamma runs, and the case which has the highest neutron source is assumed for the neutron runs. Thus, the maximum gamma and neutron dose rates are conservatively assumed to exist simultaneously.

5.4.6 Air Outlet Dose Rate Calculation

The air outlet duct geometry for the TranStor™ Storage Cask is nearly identical to that of the previously licensed VSC-24 storage cask (Reference 1.2). A large set of detailed analyses were performed to calculate the gamma and neutron attenuation through the VSC-24 cask outlet ducts. Because the outlet duct geometries of the two cask systems are nearly identical, this large set of analyses need is not repeated for the TranStor™ Storage Cask.

Calculations are performed with the QAD point kernel code to determine gamma flux levels at the inner end of the outlet duct (i.e. the opening in the cask inner liner). Gamma flux levels from the fuel gamma source and flux levels from the top nozzle (and plenum) gamma sources are calculated for both the TranStor™ and VSC-24 systems. Explicit 2-D radially symmetric QAD models of the TranStor™ and VSC-24 storage cask systems are used. The fuel region and top nozzle region fluxes are combined to yield total gamma fluxes at the inner end of the duct.

The flux levels, for every significant gamma energy, are compared for the two systems. Due to the similarity in the outlet duct geometries, the gamma dose rates at the outer end of the duct are assumed to be proportional to the gamma flux levels at the inner end of the inlet duct. By calculating flux levels at the inner end of the duct, and comparing these flux levels as opposed to comparing gamma source strengths, the differences in basket designs between the two systems are accounted for.

Flux ratios (TranStor™ / VSC-24) are determined for every significant gamma energy level. An outlet duct gamma dose rate ratio (TranStor™ / VSC-24) is conservatively determined using the highest of these ratios. Previously calculated VSC-24 cask outlet duct gamma dose rates are multiplied by this maximum ratio to yield TranStor™ cask outlet duct gamma dose rates.

One gamma flux calculation is performed for each fuel type (PWR and BWR) . For each fuel type, the gamma flux ratios are determined using the burnup and cooling time combination that yields the highest gamma source strengths. Thus, the calculated TranStor™ gamma flux levels, and therefore the scaled outlet duct gamma dose rates, will be bounding. These bounding outlet gamma dose rates are then presented for all PWR and BWR fuel burnup levels in Table 5.1-1.

A simpler approach is used to determine air outlet neutron dose rates. Since the shielding materials attenuate neutrons much less than they do gammas, the minor changes in basket geometry between the two systems are expected to have a much smaller effect on neutron flux levels at the inner end of the outlet duct than they would have on gamma flux levels. Also, the neutron source spectrum is virtually the same for the two systems, unlike the gamma source spectrum.

Therefore, it is assumed that, due to the similarity in the geometries of the two systems and their outlet ducts, the air outlet neutron dose rates are proportional to the total neutron source. Thus, air outlet neutron dose rates previously calculated for the VSC-24 system are multiplied by the ratio of the TranStor™ system total neutron source strength over the VSC-24 system total neutron source strength to yield TranStor™ air outlet neutron dose rates. A single neutron dose rate is presented in Table 5.1-1 for the PWR and BWR casks. In each case, the fuel burnup level with the highest total neutron source is assumed in the calculations.

The gamma flux and neutron source ratio methodologies used for estimating TranStor™ air outlet dose rates are sufficient due to the high degree of similarity between the TranStor™ and VSC-24 system geometries, and due to the fact that the calculated air outlet dose rates only ~7 mrem/hr (over a localized area) and will not significantly effect overall dose rates around the cask.

5.4.7 Dose Rate Calculation Over Transfer Cask Radial Gap

Gamma dose rates over the gap between the basket and the transfer cask (detector location #3 in Figure 5.1-2) are also determined from VSC-24 system results (Reference 1.2) using a source ratio methodology. Once again, due to the similarity in the geometry of the two systems, a gamma source ratio technique will yield accurate results. The transfer cask geometries of the two systems are quite similar. The gap width is the same for both systems. As stated in Section 5.4.1, one of the few differences between the two systems is the greater amount of shielding materials in the TranStor™ basket interior, a difference which will make the source ratio technique conservative. The TranStor™ basket lid replaces the two inches of neutron shield material present in the VSC-24 basket lid with an additional half inch of steel. Since steel is about four times as dense as the VSC-24 neutron shield material, the overall gamma shielding capabilities of the two lids are about equal.

At every significant gamma energy line, the TranStor™ system gamma source strength is compared to that assumed in the VSC-24 shielding analyses. Also, the TranStor™ top nozzle region source strength is compared to that of the VSC-24 system. For the fuel region, the 1.25 MeV energy line (the energy that contributes most to cask external dose rates) was found to yield the highest source strength ratio (TranStor™ / VSC-24). However, the ratio in the top nozzle source strengths was even greater than this. Therefore, the gamma dose rates over the gap calculated for the VSC-24 system are multiplied by the top nozzle source strength ratio to yield the gamma dose rates over the gap for the TranStor™ system.

A simple source strength ratio cannot be used to estimate the neutron dose rates over the gap for the TranStor™ system. This is because the shielding geometries of the TranStor™ and VSC-24 basket lids are significantly different with respect to neutrons. The VSC-24 basket lid has two inches of neutron shielding that is not present in the TranStor™ basket lid. Also, unlike the center of the basket lid, it is difficult to obtain acceptable neutron particle statistics over the gap with monte-carlo shielding codes.

Therefore, instead of comparing neutron source strengths, the neutron dose rates calculated for the basket lid center are compared for the two systems. The ratio in the neutron dose rates over the gap (TranStor™ / VSC-24) are assumed to equal the ratio in neutron dose rates at the lid center. This approach accounts for the lower VSC-24 system neutron dose rates due to the presence of the neutron shielding in the VSC-24 basket lid. In fact, the neutron shielding should reduce the VSC-24 neutron dose rates by a greater amount in the lid center than it does over the gap, where many of the neutrons have passed to the side of the neutron shield. Therefore, the ratio of neutron dose rates (TranStor™ / VSC-24) should be higher in the center of the lid than it is over the gap.

Therefore, using the ratio of the calculated lid center neutron dose rates to scale the calculated VSC-24 neutron dose rates over the gap is conservative. For each of the eight PWR and BWR cases presented in Section 5.1, the neutron dose rate over the gap is equal to that calculated over the gap for the VSC-24 system times the ratio of the TranStor™ lid center neutron dose rate (calculated for that case) over the VSC-24 lid center neutron dose rate.

5.4.8 Accident Condition Dose Rate Calculation

Storage cask side dose rates must be calculated for the accident condition. The analyses consider an accident case where a tornado generated missile (Section 11.2.3) creates a 5.69 inch deep pit in the cask side concrete shielding.

Gamma dose rates have been calculated for the surface of the VSC-24 basket shell. The QAD point-kernel shielding code is used to calculate the gamma attenuation across the cask inner liner, allowing the gamma dose rates on the inside of the VSC-24 concrete shield to be determined. Neutron dose rates on the surface of the VSC-24 basket have also been calculated. Due to the fact that steel is a poor neutron shield, the neutron dose rates on the cask liner outer surface are nearly equal to those on the basket surface. The gamma and neutron dose rates on the inner surface of the concrete shield are compared to the dose rates previously calculated for the VSC-24 storage cask outer surface.

The radiation attenuation through concrete is assumed to take the form of a simple exponential function. The coefficient of this exponent, i.e. the effective attenuation coefficient for the concrete, is estimated by comparing the dose rates inside the concrete shield to the dose rates outside the concrete shield. After calculating the effective neutron and gamma attenuation coefficients for the concrete, the effect of removing 5.69 inches of concrete is determined. The results are discussed in Section 11.2.3.

5.4.9 Flux-To-Dose Conversion Factors

The shielding codes calculate gamma and neutron flux levels, in particles/cm^2-sec, at various locations on the cask surfaces. A set of flux-to-dose conversion factors are also entered into the codes, so that they automatically convert these flux levels into dose rates in mrem/hr. Flux-to-dose conversion factors are expressed in mrem/hr per particle/cm^2-sec, and are purely a function of particle type and particle energy. The neutron flux-to-dose conversion factors used in the shielding analyses, shown in Table 5.4-3, are taken from the CASK cross-section library (Reference 5.11). The ANSI/ANS-6.1.1 (1977) flux-to-dose conversion factors (Reference 5.12), shown in Table 5.4-4, are assumed for gammas.

TABLE 5.4-3

NEUTRON FLUX-TO-DOSE CONVERSION FACTORS
(mrem/hr per neutron/cm^2-sec)

Neutron Energy (MeV)	Conversion Factor
2.12E-07	3.78E-03
7.67E-07	3.96E-03
2.09E-06	4.14E-03
6.58E-06	4.32E-03
1.96E-05	4.50E-03
6.50E-05	4.68E-03
3.42E-04	4.68E-03
1.97E-03	4.32E-03
5.72E-02	6.48E-03
0.331	5.40E-02
0.83	0.1188
1.47	0.1332
2.09	0.1296
2.41	0.1260
2.74	0.1260
3.54	0.1296
4.51	0.1332
5.66	0.1404
7.27	0.1476
9.09	0.1476
11.1	0.1656
13.56	0.2088

TABLE 5.4-4

GAMMA FLUX-TO-DOSE CONVERSION FACTORS
(mrem/hr per γ/cm^2-sec)

Gamma Energy MeV	Conversion Factor
0.01	3.96E-03
0.03	5.82E-04
0.05	2.90E-04
0.07	2.58E-04
0.1	2.83E-04
0.15	3.79E-04
0.2	5.01E-04
0.25	6.31E-04
0.3	7.59E-04
0.35	8.78E-04
0.4	9.85E-04
0.45	1.08E-03
0.5	1.17E-03
0.55	1.27E-03
0.6	1.36E-03
0.65	1.44E-03
0.7	1.52E-03
0.8	1.68E-03
1.0	1.98E-03
1.4	2.51E-03
1.8	2.99E-03
2.2	3.42E-03
2.6	3.82E-03
2.8	4.01E-03
3.25	4.41E-03
3.75	4.83E-03
4.25	5.23E-03
4.75	5.60E-03
5.0	5.80E-03
5.25	6.01E-03
5.75	6.37E-03
6.25	6.74E-03
6.75	7.11E-03
7.5	7.66E-03
9.0	8.77E-03
11.0	1.03E-02
13.0	1.18E-02
15.0	1.33E-02

5.4.10 Scattered Dose Evaluation

Gamma and neutron dose rates, as a function of distance from a single cask, which include ground and air scattering effects, have been calculated for the VSC-24 storage cask using the SKYSHINE II code. Average gamma and neutron fluxes (dose rates) over the VSC-24 cask side and top surfaces were the primary input to the SKYSHINE II code.

There are four components which contribute to the total dose rate at a given distance from the cask: the cask side gamma flux, the cask side neutron flux, the cask top gamma flux, and the cask top neutron flux. The VSC-24 SKYSHINE II analysis calculated the dose rate contributions, as a function of distance, from each of these four components.

The dose rates away from the cask are proportional to the dose rates on the cask surfaces. Therefore, the dose rate contributions from the four components for the TranStor™ system are determined from the VSC-24 calculated dose rate contributions and from the ratio of dose rates on the cask side and top surfaces.

The gamma dose rate contribution from the TranStor™ Storage Cask side at various distances from the cask equals that calculated for the VSC-24 storage cask times the TranStor™ cask side surface average gamma dose rate over the VSC-24 cask side surface average gamma dose rate. The (scattered) gamma dose rate contribution from the TranStor™ cask top is equal to that calculated for the VSC-24 times the TranStor™ cask top surface gamma dose rate over that of the VSC-24 cask. The same methodology is used to calculate the cask side and cask top neutron dose rate contributions for the TranStor™ Storage Cask.

The results of these simple dose rate ratio calculations is the TranStor™ Storage cask dose rate contributions, as a function of distance, for each of the four components. These dose rate contributions are then summed to yield the total dose rate versus distance for a single TranStor™ Storage Cask.

The dose rate vs. distance data for a single TranStor™ Storage Cask is shown in Figure 5.4-1. The dose rate contributions from each of the four components are shown along with the total dose rate. The data presented in Figure 5.4-1 is based on a storage cask containing a basket loaded with 40 GWd, 5 year cooled PWR fuel. As the data shown in Figure 5.4-1 shows, the neutron dose rate makes up an insignificant fraction of the total dose rate for all distances. Therefore, the 40 GWd PWR fuel case, which yields the highest cask surface gamma dose rates, will be the bounding case. The total dose rate for all of the other PWR and BWR fuel burnup cases will be less than that shown in Figure 5.4-1 at all distances.

Distance (Ft)	Top Neutron - 1 Cask (mrem/hr)	Side Neutron - 1 Cask (mrem/hr)	Top Gamma - 1 Cask (mrem/hr)	Side Gamma - 1 Cask (mrem/hr)
50	1.46E-05	2.24E-04	1.13E-03	6.32E-02
200	3.63E-06	2.70E-05	2.99E-04	8.66E-03
1000	1.10E-07	4.93E-07	7.45E-06	1.94E-04
2000	5.00E-09	2.22E-08	2.93E-07	8.50E-06

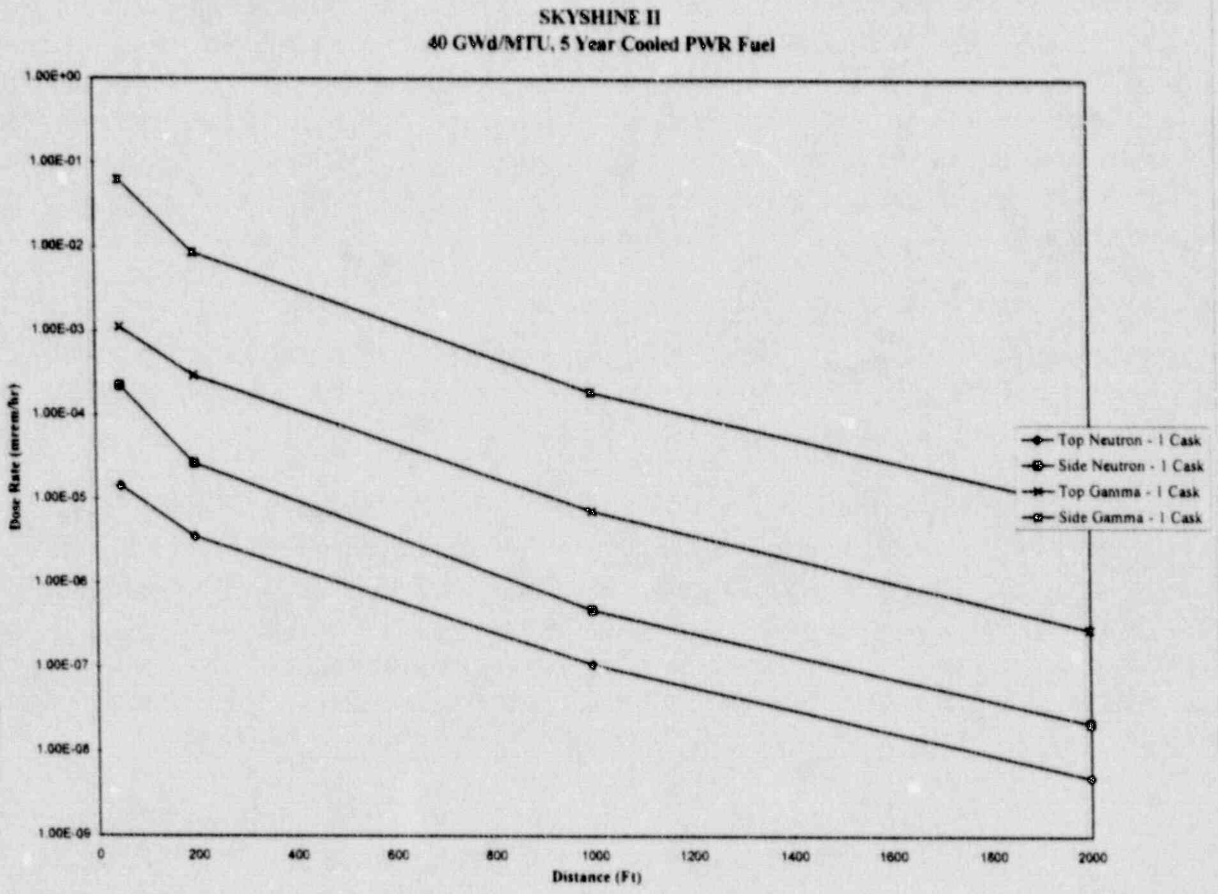


FIGURE 5.4-1

DOSE RATE VS. DISTANCE
FOR A SINGLE TranStor™ STORAGE CASK

Figure 5.4-2 (from Reference 1.2) shows total dose rate versus distance data for the VSC-24 cask system. The total dose rate, versus distance, is calculated using several codes including SKYSHINE II, MCNP, and MORSE. Also, actual VSC-24 cask dose rate measurements were taken at the Palisades nuclear plant at distances of 38 ft, 210 ft, and 380 ft. The data in Figure 5.4-2 shows that there is reasonable agreement between each of the codes. Also, the measured data conclusively shows that the SKYSHINE II code results are conservative.

With respect to dose rates at locations away from the cask(s), the regulatory requirement is that the dose rate at the controlled area boundary (around the ISFSI) be sufficiently low that the total annual exposure for any real individual is under 25 mrem. Dose rate versus distance calculations will be performed as part of the 10CFR Part 72.212 evaluations for each ISFSI site, because the dose rates are dependent on several ISFSI specific parameters such as the number of casks, the arrangement of the casks, the burnup and cooling times for the fuel loaded into the casks, and the presence of site specific features such as hills, berms, walls, or buildings which may provide additional shielding.

In the ISFSI dose rate versus distance calculations, credit is taken for the fact that most casks are loaded with spent fuel that has a cooling time significantly longer than that of design basis fuel. Also, in an actual ISFSI, the front row of casks (nearest to the detector) completely blocks the cask side dose rate contributions from casks in the other rows.

In a typical calculation, the distance to the detector is calculated for each cask in the front row. The cask side dose rate contributions are then determined for each front row cask using single cask dose rate vs. distance data similar to that shown in Figure 5.4-1. Casks in the back rows have no cask side dose rate contributions. Then the distance from the detector to each cask in the entire array is calculated, and the (scattered) cask top dose rate contributions from each cask top are determined from the single cask data. These side and top dose rate contributions from each cask are then summed to yield the total dose rate (at the detector) from the entire ISFSI.

The dose rate contributions from each cask are adjusted to account for the actual cooling time of the fuel present in each cask. Also, the ISFSI should (based on ALARA considerations) be arranged so that casks containing fuel with the longest cooling times will lie around the ISFSI edge. Thus, the cask side dose rate contributions (which are based solely on the outer cask rows) will be based upon cooling times significantly longer than that of design basis fuel. A large nuclear power plant generates only enough spent fuel per year, on average, to fill about two casks. Therefore, a large ISFSI at any actual plant will not contain large numbers of casks with 5 year old fuel. For most large ISFSI's, the average cooling time for fuel stored in the outer cask rows will range from 10 to 20 years.

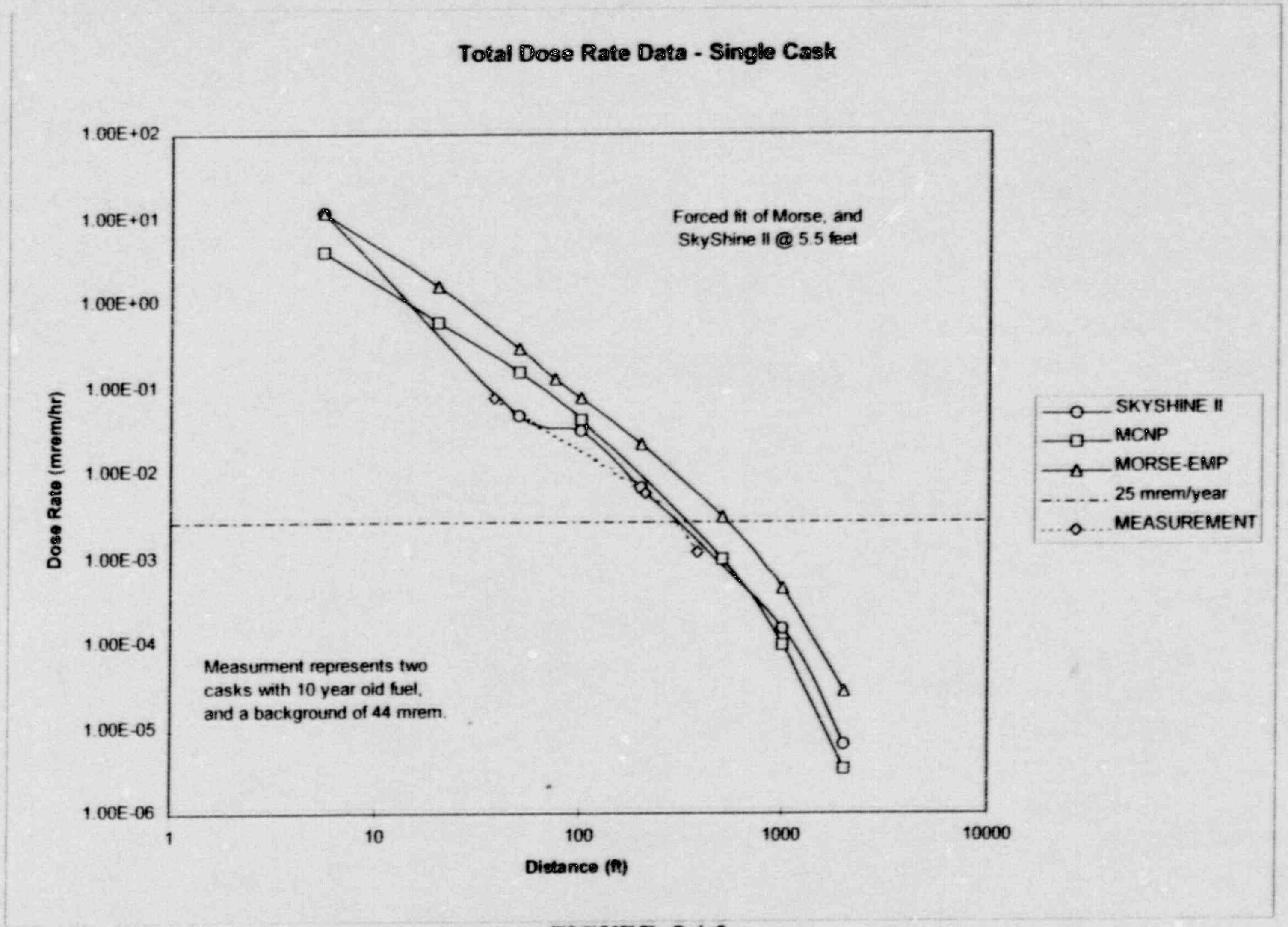


FIGURE 5.4-2
COMPARISON OF SKYSHINE II RESULTS TO OTHER CODES AND TO MEASURED DATA

Previous calculations have shown that, if the above factors are taken into account, even a large ISFSI will produce dose rates under 0.0125 mrem/hr at a distance of 100 meters from the ISFSI edge (the minimum allowed distance for the controlled area boundary). This dose rate corresponds to an annual exposure of 25 mrem/year based on an upper bound occupancy factor of 2000 hr/year. If, however, site specific ISFSI dose calculations show that the dose rate at the controlled area boundary exceeds 25 mrem/year, a wall or berm may be placed around the ISFSI. That would dramatically reduce the dose rates at the controlled area boundary, and allow the dose rate criterion to be met for any set of conditions.

5.5 Supplemental Data

The TranStor™ system may also accommodate stainless steel clad and MOX fuel assemblies. Shielding considerations for storing these fuel assembly types are discussed in this section.

5.5.1 Stainless Steel Clad Fuel

For a given burnup, cooling time, fuel loading, and initial enrichment, stainless steel clad fuel produces a higher fuel region gamma source than Zircaloy clad fuel due to (Co-60) activation of the stainless steel cladding. The activated cladding causes a higher gamma source strength for the 1.25 MeV energy line. The heat generation, the neutron source strength, and the gamma source strengths for all other gamma energy levels is roughly the same for stainless clad fuel as it is for Zircaloy clad fuel with similar parameters.

As stated in Reference 4.14, the maximum burnup level for stainless steel clad PWR fuel is 40 GWd. Therefore, stainless steel clad fuel with 40 GWd burnup and 5 year cooling is considered. The maximum fuel loading of 0.469 MTU is also assumed, along with a lower bound enrichment value of 3.02%. Stainless steel clad fuel with these parameters will generate about 1.083 kw/assembly (the heat generation limit). Except for the cladding material, this case has the same parameters as the 40 GWd Zircaloy clad case considered in the main shielding sections. Therefore, the fuel region source description for this case is identical to that presented for 40 GWd fuel in Tables 5.2-1 and 5.2-5. The only difference is that there is an additional 1.25 MeV gamma source due to activated stainless steel cladding.

The methodologies used for determining Co-60 activity in core region steel components are discussed in Section 5.2.1.1 and in Reference 1.1. These same methodologies are used to determine the activity of the stainless steel cladding. For the Co-60 activity calculation, the stainless steel cladding volume is assumed to equal the cladding volume of the Zircaloy W 15x15 assembly. This is known to be greater than the cladding volume of any actual stainless steel assembly (References 5.1 and 5.8).

Based on the resulting stainless steel mass, on the maximum cobalt content in stainless steel (from Reference 5.1), and on the core region Co-60 activation level (Ci of Co-60 per gram of cobalt initially present - taken from Reference 5.1), the total core region Co-60 activity (in Ci) can be calculated. This Co-60 activity is converted into a 1.25 MeV gamma source strength and multiplied by 24 (assemblies/cask) to yield a total additional 1.25 MeV source strength for the cask fuel region.

For 40 GWd/MTU, 5 year cooled fuel, the above calculations yield an additional core region 1.25 MeV gamma source of 1.07×10^{16} γ/sec-cask. This additional source is 21% higher than the 1.25 MeV gamma source shown in Table5.2-1 (8.86×10^{15}). Assuming that the gamma dose rates around the storage and transfer casks are proportional to the fuel region 1.25 MeV gamma source strength is conservative. This is because other gamma energies, which do not increase for stainless steel clad fuel, contribute some fraction of the gamma dose rates. Also, unlike the fuel region, the bottom and top nozzle region gamma source strengths for stainless steel clad fuel are not higher than those of Zircaloy clad fuel.

Thus, a conservative estimate of gamma dose rates around the casks for 40 GWd, 5 year cooled stainless steel clad PWR fuel can be obtained by simply multiplying the gamma dose rates presented in Tables 5.1-1 and 5.1-2 for 40 GWd Zircaloy clad fuel by a factor of 2.21. Neutron dose rates for stainless steel clad fuel will be similar to those for Zircaloy clad fuel. This conservative approach yields a dose rate of ~20 mrem/hr one meter from the concrete cask side. Dose rates one meter from the concrete cask top surface (which are dominated by the neutron dose rate) will remain under 100 mrem/hr. Both of these dose rates are well below the design limits established in Section 2.3.5.2. Dose rates on the transfer cask surface will be ~300 mrem/hr on the side and under 150 mrem/hr on the top.

It should also be noted that most, if not all, of the existing stainless steel fuel will actually have a cooling time significantly higher than 5 years at the time of storage. Therefore, these calculated upper bound dose rates are not expected in any real situation. The analyzed cooling time of 5 years was selected because it corresponds to an assembly heat generation level of 1.083 kw/assembly, and the TranStor™ fuel specification states that any fuel which meets the heat generation limit may be loaded into the cask.

The entire stainless steel clad BWR fuel inventory has a burnup level under 25 GWd, and will have a cooling time of over 10 years at the time of storage (Reference 4.14). As a result, the gamma source for the stainless steel clad BWR fuel is completely bounded by that shown in Table 5.2-2 for 35 GWd, 5 year cooled Zircaloy clad BWR fuel. The neutron source strength is also bounded. Therefore, all existing stainless steel clad BWR fuel will produce dose rates that are lower than those shown in Table 5.1-1 for Zircaloy clad BWR fuel.

5.5.2 MOX Fuel

The TranStor™ system may also accommodate MOX fuel assemblies. The physical parameters of MOX fuel assemblies are similar to those of uranium fueled assemblies. Therefore, if the gamma and neutron source strengths of the loaded MOX fuel assemblies are bounded by the gamma and neutron source strengths presented for uranium fuel in Section 5.2, the gamma and neutron dose rates for the MOX fuel case will be bounded by the dose rates presented for uranium fuel in Table 5.1-1.

Therefore, MOX fuel assemblies may be stored in the TranStor™ Storage Cask if source term calculations (performed by the licensee) show that the MOX fuel source terms are bounded by the uranium fuel source terms presented in Section 5.2. For a given fuel burnup and cooling time, MOX fuel is expected to have gamma source strengths similar to uranium fuel. The MOX fuel neutron source strength, however, is expected to be significantly higher. Thus, longer cooling times than those shown for uranium fuel in Table 5.1-1 will be required for storage of MOX fuel.

5.6 References

5.1 DOE/RW-0184-R1, "Characteristics of Potential Repository Wastes," Office of Civilian Radioactive Waste Management, U.S. Department of Energy, July 1992.

5.2 SR/CNEAF/95-01, "Spent Nuclear Fuel Discharges from U.S. Reactors 1993," Energy Information Administration, U.S. Department of Energy, February 1995.

5.3 Cacciapouti, R.J., "PWR Axial Burnup Power Profile Database," Proceedings of the Fifth Annual High Level Radioactive Waste Management International Conference, Vol. 2, American Nuclear Society, May 1994.

5.4 NEDO-10084-3, "Consolidated Safety Analysis Report for IF-300 Shipping Cask," Volumes 1 and 2, General Electric Company, Docket No. 71-9001, May 1985.

5.5 Croff, A.G., "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code," ORNL/TM-6051, Oak Ridge National Laboratory, September 1978.

5.6 "Multi-Purpose Canister (MPC) Subsystem Design Procurement Specification," DBG000000-01717-63000-00001 Rev. 3, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, May 1994.

5.7 Asano, R. and Niomura, N., "Experimental Studies on Long-Term Thermal Degradation of Enclosed Neutron Shielding Resin," 10th International Symposium of the Packaging and Transportation of Radioactive Materials (PATRAM), Yokohama City, Japan, September 1992.

5.8 Viebrock, J.M., "Domestic Light Water Reactor Fuel Design Evolution," Vol. 3, DOE/ET/47912-3, NAC-C-8129, U.S. Department of Energy, September 1981.

5.9 "MCNP-A General Monte Carlo Code for Neutron and Photon Transport," Version 3A, LA-7396-M, Rev. 2, Los Alamos National Laboratory, September 1986.

5.10 Broadhead, B., L., "QADS: A Multidimensional Point Kernel Analysis Module.", NUREG/CR-5468, Oak Ridge National Laboratory, May 1990.

5.11 "CASK: 40 Neutron and Gamma Ray Cross Section Data," Radiation Shielding Information Center, DLC-23, 1978.

5.12 "American National Standards Neutron & Gamma-Ray Flux-to-Dose Rate Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society, LaGrange Park, Illinois, March, 1977.

6.0 CRITICALITY EVALUATION

The TranStor™ System is designed to maintain nuclear criticality safety (sub-criticality) under all applicable regulatory conditions. These conditions include all normal, off-normal handling, and hypothetical accident conditions.

6.1 Discussion and Results

In the TranStor™ System, criticality is controlled by the basket material and geometry. In addition to the neutron absorption properties of the fuel and the steel basket, poison plates are included in each of the sleeves adjacent to other fuel assemblies. No fuel burnup or boron credit is taken for criticality control. The only administrative control required from the system owner is that of fuel enrichment (see Tables 12.2-2 through 12.2-5) to ensure that fuel placed in storage meets the design characteristics.

6.1.1 Criticality Design Features of the TranStor™ PWR Basket

Criticality control in the TranStor™ PWR basket is achieved using flux traps and poison plates. Flux traps thermalize fast neutrons and poison plates remove the thermal neutrons from the system. This methodology allows the TranStor™ PWR system to maintain a k_{eff} below 0.95 for all normal and off-normal conditions.

Two loading schemes are available to maximize the transportable fuel enrichment in the TranStor™ Storage Cask. The first is the fully loaded 24-assembly basket. The second loading scheme is a partially loaded 20-assembly basket where the four center sleeves are left vacant. The absence of the four center assemblies creates a negative reactivity effect that enables the cask to store higher enriched fuels. Reactivity in the partially loaded basked is reduced due to the flux depression formed in the most reactive area of the basket.

Since a global enrichment limit underestimates the maximum storage enrichment of some assembly types, a set of assembly-specific storage limits is presented in this SAR. Using this approach, all assemblies with an enrichment at or below 4.1% ^{235}U can be safely stored, while allowing higher enrichment for some fuel types. The generic and assembly-specific limits for the 24- and 20-assembly baskets are provided in Tables 12.2-2 and 12.2-3.

6.1.2 Criticality Design Features of the TranStor™ BWR Basket

Unlike the PWR basket, the BWR basket does not require flux traps to satisfy criticality requirements. The BWR basket utilizes only poison plates to remove thermal neutrons from the package.

As with the PWR basket, two different loading schemes are available for the TranStor™ BWR Basket. The first is the fully loaded 61-assembly basket, and the second is a partially loaded 60-assembly basket. The center sleeve is left vacant in the 60-assembly basket. The vacant sleeve provides the benefit of a flux depression in the most reactive area of the basket.

Similar to the PWR fuel, a global BWR fuel enrichment limit does not accurately portray the maximum storage enrichments of some BWR fuel assemblies. Therefore, both a bounding enrichment limit and a set of discrete limits aimed at optimizing the TranStor™ Storage Cask's performance are presented in this SAR. Tables 12.2-4 and 12.2-5 present the maximum generic and assembly-specific enrichments for the 61 and 60 assembly baskets.

6.2 Spent Fuel Loading

The TranStor™ Storage Cask is designed to store all types of PWR and BWR fuel assemblies (with the exception of GE-XBR fuel assemblies, which do not fit within the sleeves of the BWR basket). The characteristics of the design basis PWR fuel assemblies are tabulated in Tables 12.2-2 and 12.2-3 and the characteristics of the design basis BWR fuel assemblies are provided in Tables 12.2-4 and 12.2-5. The PWR basket can also store up to 4 partial, failed, damaged, or leaking PWR fuel assemblies that are placed in cans inside the four larger sleeves in the corners of the basket. Similarly, the BWR basket can store up to 8 canistered failed, damaged, or leaking BWR fuel assemblies.

In addition, the PWR and BWR baskets can accommodate fuel debris cans containing up to 10 kg of loose pellets or other fuel debris, as long as the total plutonium inventory is 20 curies or less. The baskets may also be loaded with MOX fuel assemblies as long as their fissile material percentages are under the limits specified in Section 12.2.2.1.

6.3 Criticality Evaluation

The detailed criticality evaluation of the TranStor™ System is presented in the transportation SAR (Reference 1.1). While the principal criticality design criteria of both 10 CFR Part 71 and Part 72 is that k_{eff} remains below 0.95 during all normal and accidents conditions, Part 72 allows credit for the plant administrative controls of either fuel burnup or boron concentration in the fuel pool. Neither of these credits is available per transportation requirements of 10 CFR Part 71.

Therefore, the TranStor™ criticality analysis presented in Reference 1.1 bounds the analysis required for this report. No additional evaluation is necessary. The results of Reference 1.1 demonstrate that the system meets or exceeds the criticality requirements of 10 CFR 72.

7.0 CONFINEMENT

7.1 Confinement Boundary

The following paragraphs define the confinement boundary for the TranStor™ System. The primary confinement boundaries for fissile material and fission products are:

1. The fuel cladding or the canister (for failed fuel and fuel debris).

2. The welded basket.

The concrete cask is designed to provide shielding, structural support, ventilation, and weather protection for the basket but is not part of the confinement boundary.

7.1.1 Confinement Vessel

The basket sits vertically in the storage cask and is cooled by the natural circulation of air through the annulus between the basket and the storage cask. The basket confinement boundary contains the following major components:

Shell
Bottom Plate
Structural Lid
Shield Lid Assembly
Structural Lid Port Covers

The basket is designed, constructed, and inspected in accordance with Section III, Subsection NC of the ASME Code, except no N-stamping is required and the field closure welds are not radiographed as discussed below.

7.1.2 Confinement Penetrations

The only penetrations required in the basket are in the shield and structural lids. Two ports for vacuum drying and backfilling with helium are located in the shield lid which is lowered onto the basket body after the fuel is loaded. The structural lid is placed on top of the shield lid and has a penetration that allows access to the shield lid fittings. Two cover plates that mate with the structural lid are seal welded over the penetrations after the drying and helium backfilling operations have been completed. Hence, the penetrations in the shield lid are not a part of the confinement boundary and are not important to safety.

No components are required to penetrate the sealed basket 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

Only welded seals are used in the TranStor™ basket design. The following welds are relied upon for confinement of radioactive materials:

1) Girth and longitudinal welds in the basket shell
2) Weld between the shell and bottom plate
3) Weld between the shell and shield lid
4) Weld between the shell and structural lid
5) Weld between the structural and shield lids at the port access
6) Two penetration cover closure welds

7.1.3.1 Welding Requirements

All weld configurations are specified on the basket drawings. All welding procedures are written and qualified in accordance with ASME, Section IX. Each welder or welding operator shall also be qualified in accordance with ASME, Section IX. Identification of the specific procedure used for each weld is required by the basket fabrication specification.

7.1.3.2 Testing, Inspection, and Examination

The following examination and test requirements are implemented during fabrication of the basket.

1. All components are examined for conformance with the referenced drawings.

2. All confinement welds made in the shop (welds 1 and 2 in paragraph 7.1.3) are radiographically examined in accordance with the requirements of Section V, Article 2 of the ASME Code. The minimum acceptance standards for radiographic examination are as specified in Section III, Division 1, Subarticle NC-5320 of the ASME Code. Unacceptable defects in the welds are eliminated and repaired in accordance with Section III, Division 1, Subarticle NC-4450 and re-examined.

3. A written report of each weld examined by radiography is prepared. As a minimum, the written report includes: identification of part, material and/or area, name and level of examiner, and the findings or dispositions if any.

4. The first and last pass of all confinement welds made in the field (welds 3 through 6 in paragraph 7.1.3) are magnetic particle or liquid penetrant examined in accordance with the requirements of Section V, Article 7 or 6 of the ASME Code. The minimum acceptance standards for selected method of examination are as specified in the corresponding part of Section III, Division 1, Subarticle NC-5300 of the ASME Code. Unacceptable defects in the welds are eliminated and repaired in accordance with Section III, Division 1, Subarticle NC-4450 and re-examined.

5. In addition to the above, welds 3, 4, and 5 of 7.1.3 are helium leak tested prior to welding the port access penetration. If a leak rate is determined to be greater than 10^{-4} scc/sec, weld repair is required.

6. All personnel performing the above nondestructive testing are qualified in accordance with American Society of Nondestructive Testing Recommended Practice No. SNT-TC-1A. Only individuals qualified for NDT Level I, NDT Level II, or NDT Level III may perform nondestructive testing. Level I personnel may not interpret the results of examination or make determination of the acceptability of examined parts.

7.1.4 Closure

The only closure used for the basket after loading of irradiated fuel is welding of the structural lid, shield lid and the penetration closure cover plates. The closures are discussed above. No closures are by bolts or other mechanisms.

7.2 Requirements for Normal Conditions of Storage

Section 3.0 provides the structural evaluation which demonstrates the integrity of the basket and storage cask under normal conditions.

7.2.1 Release of Radioactive Material

The closure of the basket confinement boundary is by welding the shield and structural lids and port cover plates in place and performing liquid penetrant or magnetic particle tests to show their integrity. A helium leak test is also performed. No penetrations exist during storage life. Analyses of all welds for normal and accident conditions show that stresses are well within ASME allowables and, hence, no failure will occur. Therefore, particulate or gaseous radioactive material can not escape from the basket.

7.2.2 Pressurization of Confinement Vessel

The basket is vacuum dried and backfilled with helium prior to placement in service. No significant moisture or gases remain after drying, hence, there is no possibility that radiolytic decomposition would cause a significant increase in basket internal pressure.

The backfill pressure is selected so that the basket is always at slightly sub-atmospheric pressure during storage. The internal pressure during all conditions is shown in Tables 3.4-3 and 3.4-4 and the analysis in Chapter 3.0 demonstrates that the expected maximum pressure does not over-stress the basket. Even if the confinement boundary was somehow compromised, the leakage would occur inward since the basket pressure is always negative. It is not expected that any radioactive material (gaseous or particulate) will be released under normal or accident conditions.

7.3 Confinement Requirements for Hypothetical Accident Conditions

The maximum quantity of fission gas products that could hypothetically be released and their impact on the surroundings is addressed in Section 11.2.1. The TranStor™ System keeps irradiated fuel below the temperature limit where the cladding failure could occur, nevertheless, the basket is designed to withstand the pressure resulting from a 100% rod rupture. Therefore, two extremely unlikely events would have to occur for the full release of radioactive products: failure of all fuel rods inside the basket and failure of the basket pressure boundary.

Nevertheless, the analysis of complete release of all radioactivity in the basket is included to show the substantial safety margin of the TranStor™ System. It demonstrates that the regulatory limits are not violated even under these hypothetical conditions. The resulting site boundary dose to an individual exposed to the fission gas cloud is only about 500 mrem whole body dose for a typical boundary at 100 meters. This is well under the 10 CFR 72 limits of 5,000 mrem for accident exposure.

8.0 OPERATING PROCEDURES

This chapter provides an overview of the operating procedures, inspections, and tests required to be performed during loading of the TranStor™ Storage Cask. The overview of the operating procedures includes a description of the minimum requirements to ensure safe operation during on-site handling and storage of the cask. In addition, this chapter includes requirements for inspection and testing of the cask as well as supplementary information related to each of the loading operations.

The procedures are generic in nature, and are based on the assumption that an empty storage cask, basket, and transfer cask are on site. Each licensee and cask user is responsible for preparation of site-specific handling procedures in accordance with these generic procedures, the Certificate of Conformance (C of C), and the licensee's Quality Assurance (QA) Program. All site-specific procedures have checklists and sign-offs to provide documentation of the various activities required by the procedures. These licensee-approved procedures are intended to ensure that critical operations in the preparation and loading of the cask are accomplished in accordance with the C of C and SAR. They also provide QA records as required by 10 CFR 72.

The procedures for loading the storage cask are given in Section 8.1 and illustrated in the form of a flow chart in Figure 8.0-1. Procedures for unloading the cask, should this activity become necessary, are essentially those of Section 8.1 in reverse and addressed in Section 8.2. Loading operations for moving a basket from a concrete cask into the shipping cask are outside the scope of this SAR and are described in Section 7.2 of Reference 1.1.

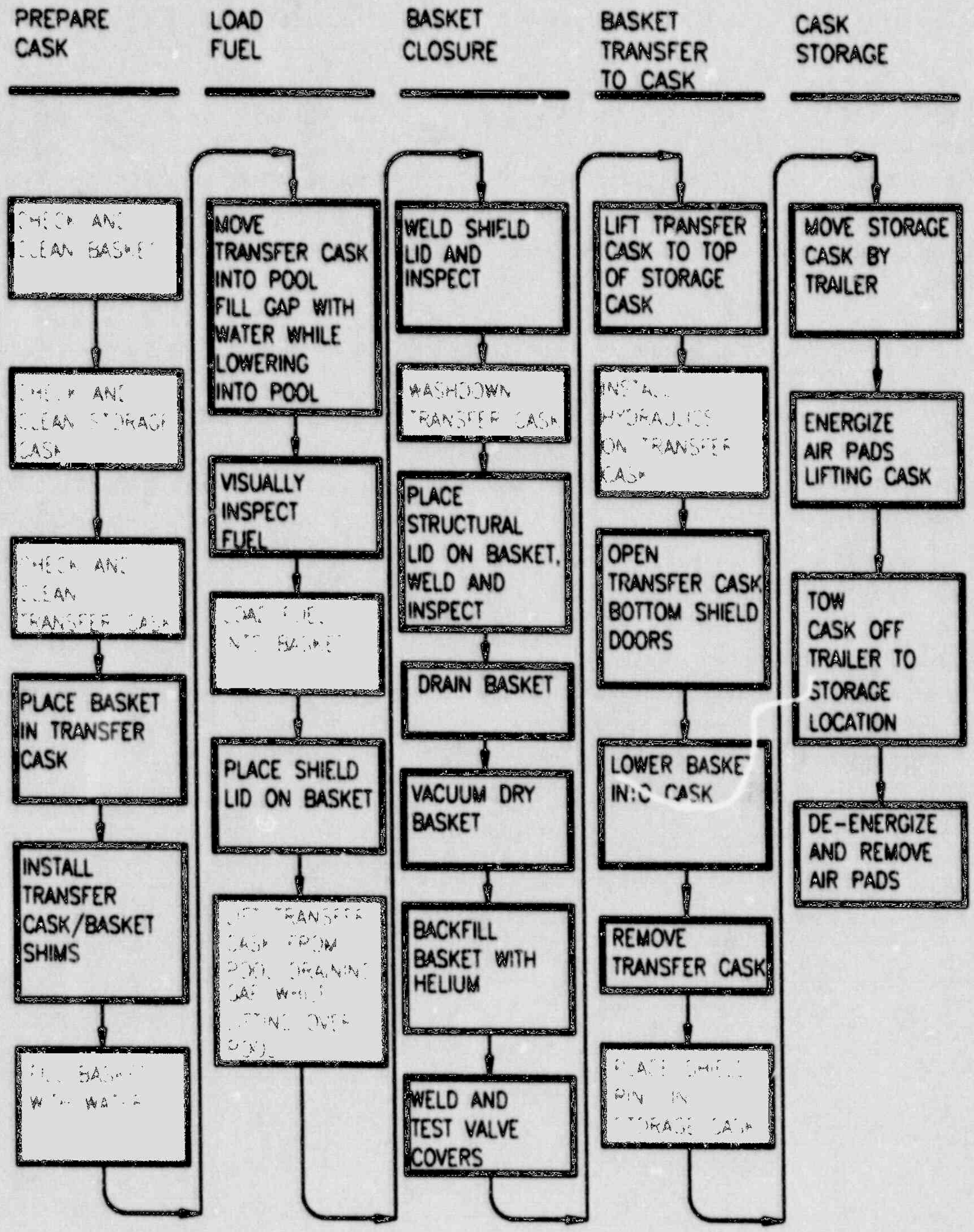


FIGURE 8.0-1

FLOW DIAGRAM OF THE TranStor™ CASK LOADING PROCEDURES

8.1 Procedures for Loading the Cask

The following outline briefly describes the major procedural steps for loading the TranStor™ storage cask. Table 8.1-1 summarizes typical time and crew requirements. The times listed in this table come from detailed handling studies, utility handling experience with the VSC-24 system, DOE cask handling experience, and general industry data. 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 parameters (dress-out time, break time, pass-in, pass-out times, etc.).
2. Required crew sizes and supervisory personnel.
3. Security requirements (manpower and access).
4. Learning curve (estimates are for handling the n-th cask, not the first 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 are required for actual implementation of the outline described below.

(1) <u>Preparation of Storage Cask, Basket and Transfer Cask</u> (Elapsed Time: 2 hrs)

1. Check and clean (if necessary) basket and its shield and structural lids.
2. Examine and clean (if necessary) storage cask and move it into the fuel handling building truck/rail bay.
3. Examine and clean (if necessary) transfer cask.
4. Lift and place the basket into the transfer cask. Install the transfer cask shims.
5. Fill the basket with fuel pool water.
6. Lift the transfer cask and move it over the fuel pool. Attach a hose from a clean borated water source (filtered pool water may be used) to the bottom of the transfer cask.

TABLE 8.1-1

OPERATIONS TIME AND MOTION SUMMARY

			CREW SIZE				
ITEM	ELAPSED TIME (HOURS)	MAINT	OPER	HP	ENG	TOTAL MANHOURS
(1)	PREPARATION	2	2	----	----	1	6.0
(2)	FUEL LOADING	6	2	2	2	1	42.0
(3)	BASKET CLOSURE	24	4	----	2	1	168.0
(4)	BASKET TRANSFER TO STORAGE CASK	3	2	2	2	1	21.0
(5)	STORAGE CASK TO STORAGE	2	2	----	1	1	8.0
	TOTAL	37					245.0

MAINT = Maintenance (welders, decontamination techs, drivers, etc.)
OPER = Crane and fuel handling operators
HP = Health Physics
ENG = Engineer Supervision

(2) Fuel Loading (Elapsed Time: 6 hrs)

1. Place transfer cask (containing the basket) into the fuel pool. During lowering, fill basket-transfer cask gap with clean water and confirm that water is continuously flowing through the gap.

2. Load fuel assemblies and/or waste canisters in the basket storage sleeves. Visually inspect all fuel assemblies as they are loaded.

3. Lift the shield lid and place it on the basket in the pool.

4. Lift the transfer cask (containing the basket) out of the pool. Shut off water supply to basket-transfer cask gap, hold transfer cask over the pool, drain water from the basket-transfer cask gap. Wash down exterior of transfer cask as it hangs over the pool. Note: at some plants the basket may have to be drained before the cask is lifted out of the pool in order not to exceed the crane load limit.

(3) Basket Closure (Elapsed Time: 24 hrs)

1. Transfer the transfer cask to the decon pit area.

2. Drain approximately 75 gallons of water from the basket (if necessary, see the note in (2) 4 above).

3. Vent the air space in the basket by connecting a vacuum hose to the vent opening in the lid and using a pump to create a negative pressure in the air space.

4. Weld shield lid to the shell. Prior to start of welding, check for hydrogen in the air space inside the basket. Assure that the concentration is below ignitable level.

5. Visually inspect the weld.

6. Pressurize the basket with water to 22 psia and perform nondestructive examination of the shield lid weld.

7. Place the structural lid in the basket and weld it to the shell and the shield lid at the access port. Perform nondestructive examination of both welds (may be concurrent with 6,7, and 8).

8. Decontaminate the exterior of the transfer cask (concurrent with 1, 2, 3, 4, 5, 7, and 8).

9. Attach the Vacuum Drying System (VDS) to the top of the basket and pump the water into the fuel pool. If desired, use air supply through the vent connection to assist the draining.

10. Isolate the drain portion of the VDS from the basket and complete the drying process by evacuating the basket. Using the VDS vacuum pump, perform multiple pump downs as required to achieve a stable vacuum pressure in the basket of 3 mm Hg for at least 30 min. Flush the basket with He and repeat evacuation.

11. Using the VDS, backfill the basket with helium to approximately 22 psia and use a helium sniffer to verify leak tightness of the basket seal welds (i.e., structural lid to shell and structural lid to shield lid at the access port). Acceptance criterion: 10^{-4} scc/sec. Release the pressure back to 14.5 psia.

12. Remove the gas line and seal weld the two valve covers on the structural lid to provide redundant seal. Perform nondestructive examination of both welds.

13. Place and bolt the transfer cask cover.

(4) <u>Basket Transfer to the Storage Cask</u> (Elapsed Time: 3 hrs)

1. Lift the transfer cask and place it on the storage cask.

2. Secure and check the placement of the transfer cask on the storage cask.

3. Install the transfer cask doors hydraulic system (concurrent with 4).

4. Install and check the basket lifting devices. Hook and slightly lift the basket.

5. Remove the door pins and open the shield doors of the transfer cask.

6. Lower the basket from the transfer cask into the storage cask. Special care should be taken to prevent impact of the basket on the storage cask tiles to minimize the potential for cracking the tiles.

7. Remove the transfer cask from the storage cask and place the shielding ring over the gap between the storage cask inner surface and the basket outer surface.

8. Install the storage cask cover plate.

(5) Movement of Cask to the Storage Location (Elapsed Time: 2 hrs)

1. Check radiation levels 1 meter from the storage cask side surface at approximately ten feet above the floor at four circumferential locations 90° apart. Check the radiation level one meter from the cask lid center.

2. Swipe the storage cask exterior for contamination.

3. Use the trailer, air pads, or transporter to move the storage cask to the ISFSI pad.

4. If the transporter or air pads are used, position and lower the cask at its storage location on the ISFSI pad. Remove the air pads or transporter and return them to the storage location.

5. If the trailer is used, position it along the edge of the ISFSI pad. Activate the air pads and use a light towing vehicle to tow the cask off the trailer and to its storage location. Lower the cask by shutting off the air supply, remove the air pads from the cask, and return the truck, trailer, and pads to storage.

8.2 Procedures for Unloading the Cask

In general, unloading of the basket is not anticipated. The basket is designed to withstand all normal and off-normal events and it is highly unlikely that a leak would develop during service. At the end of storage the basket could be shipped off-site (see TranStor™ Shipping System SAR, Reference 1.1) without reopening. Chapter 7.0 of Reference 1.1 outlines the procedure for basket transfer from a storage cask into the shipping cask; however, the exact determination of this procedure will depend on the shipping requirements at that time.

Nevertheless, unloading of the cask and basket could be prompted by the hypothetical event of basket leakage or a need for reopening at the end of life. The procedures for unloading are essentially the reverse sequence of those given above. The basket can be opened in a number of ways as described in Reference 2.3 (carbon arc air gouger, plasma cutter, portable lathe, etc.). The only issue that is different and must be addressed is reflooding of the hot basket prior to its placement into the fuel pool. This issue is discussed below.

Water is injected into the basket through the drain pipe and comes into contact with the bottom plate and bottom fittings of the fuel. Since at the full design heat load the temperature at the basket bottom is estimated to be between of 215 °F and 250 °F, some water may flash into steam. As the reflooding progresses, further localized boiling (at the fuel clad surface) is possible. Therefore, means of venting the basket must be provided. This is accomplished by using the vent port and the plant off-gas system. The vent port quick connect is removed to provide a large flow area so that the basket internal pressure will not rise significantly, i.e. shell stresses will not exceed ASME, Section III allowables for the Level B service. The steam-water mixture is discharged just beneath the fuel pool surface, into the plant radwaste system, or a portable filter. The basket is designed to maintain subcriticality (k_{eff} < 0.95) in an optimum density moderator without boron credit; hence, criticality in a steam environment is not a concern

Therefore, the cask and basket can be safely unloaded if necessary. Site-specific unloading procedures shall be developed by owner utilities prior to placement of fuel into dry storage.

8.3 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 in Table 8.1-1 are done concurrently.

The fuel loading operations are essentially the same as for any shipping or 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 basket exterior and the transfer cask interior is filled with clean water as the cask is being lowered into the pool. A hose connection is provided on the transfer cask to allow clean water to be forced through the basket-transfer cask gap to further assure that the basket surface does not become contaminated. The remaining steps are similar to normal cask handling procedures.

The first step of basket closure is to connect a self-priming pump to the basket drain valve and to pump a small amount of water out of the basket. After this is done, the shield lid is welded to the basket shell. With the basket now sealed, it can be hydrostatically pressure tested, drained, vacuum dried and backfilled with helium. The structural lid is welded to the basket shell during or after these operations. Also, during this period the transfer cask exterior is washed down and decontaminated as necessary. The majority of the effort is done while the water is still inside the basket so that the surface doses will typically be much less than reported in Chapter 5.0. After the basket is filled with He (at about 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 basket lifting rings that will be used to lower the basket into the storage cask. Since the transfer cask lid overlaps the edge of the basket, it will prevent inadvertent lifting of the basket out of the transfer cask during the lowering process.

During the movement of the transfer cask to the storage cask, the transfer cask will be lifted out of the decontamination pit and moved across the pool operations floor at a minimum distance above the floor to clear any obstacles. After the transfer cask is moved to the storage cask, the two casks are aligned together using special alignment holes. It is not possible without site specific details to determine if the best method to support the transfer cask will be on the storage cask (as it is normally done) or to support it around the hatch at the operating floor. The selection will be made after a thorough evaluation and consultation with the utility. After the transfer cask is secured, the crane is detached from the transfer cask and attached to the basket. The basket is lifted slightly to remove its weight from the lower shield doors, the doors are opened (hydraulically), and the basket is lowered.

After the basket has been lowered into the storage cask, the transfer cask is lifted and returned to the decontamination pit. The storage cask cover is bolted down and the cask is ready to be moved to its storage location.

The next major sequence of steps takes the storage cask from the auxiliary building to the ISFSI pad using the cask transporter or trailer and/or air pads. After that the cask is in safe long-term storage and only visual surveillance is required.

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 is performed on the Storage Cask, Basket, and Transfer Cask prior to acceptance for fuel loading. The components are fabricated/constructed in compliance with the TranStor™ System drawings and procurement specifications. Only vendors with an approved QA program meeting the requirements discussed in Chapter 13.0 can be used. The inspection plan for each component is provided by the vendor and approved by SNC. As-built configuration and results of all tests and inspections are documented in the package that accompanies every cask or basket. The critical parameters are:

STORAGE CASK
- wall thickness (concrete and steel)
- cover plate thickness
- internal cavity length
- internal cavity diameter
- bottom depth
- height and width of bottom channels
- concrete strength
- reinforcement strength, quantity, and placement
- fit-up of shield ring

BASKET
- material strength
- thicknesses of all components
- diameter
- length
- fit-up of shield and structural lids
- fit-up of a fuel assembly mockup (go/no-go gauge)

TRANSFER CASK
- material strengths
- material thicknesses
- bottom door thickness
- length
- internal diameter

length
internal diameter
fit-up of doors/actuators and their operation
load test of trunnions, shell, bottom rails, and their welds

Materials important for shielding, including the concrete, neutron shield, and steel in lids, are also inspected for density (CMTRs may be used in lieu of measurement to verify the density of steel components).

Acceptance criteria are in conformance to the TranStor™ System drawings.

9.1.1.2 Inspection Prior to Use

The first inspections prior to use of the storage cask, basket, and transfer cask are inspections for cleanliness. All foreign material is removed from all components to be used.

Fit-up tests on all the major components are also performed prior to use. These are:

Basket in Transfer Cask
Shield lid on Basket
Structural lid on Basket

Acceptance criteria for these tests are in conformance to the TranStor™ drawings.

9.1.2 Structural Test

During the drying step of the basket closure operation, after the seal welding is performed on the basket shield lid and prior to draining the water, the basket is hydrostatically tested to a pressure of 7.3 psig (22 psia), which is 1.5 times the normal operating pressure. In accordance with ASME Code, Section III, Subsection NC, this pressure is held for ten minutes. After this test, PT or MT is performed on the shield lid weld.

9.1.3 Leak Test

After vacuum drying, the basket is pressurized to 7.3 psig with helium to perform the helium leak inspection. This inspection must show a leak rate less than 10^{-4} scc/sec.

9.1.4 Shielding Integrity

The TranStor™ Storage Cask is essentially identical to its predecessor, VSC-24, and its shielding design is based on the same analytical approach. Fourteen VSC-24 casks have been loaded with dose rate measurements lower than predicted, hence, the shielding performance of the system is confirmed. For every TranStor™ storage cask placed in service, radiation measurements will be performed in the loading area in accordance with ALARA and good working practice.

9.1.5 Thermal Acceptance

Similar to Section 9.1.4 above, the thermal performance of the TranStor™ cask is essentially identical to that of the VSC-24. Since the VSC-24 demonstrated conservatism of cask design and analysis, no further acceptance tests are needed. The air outlet temperature measurements shall be taken in accordance with Section 12.2.1.2 as verification of the correct fuel loading.

9.2 Maintenance Program

The TranStor™ Storage System is a passive system and requires a minimum of maintenance. The following is a list of the recommended maintenance and inspection activities:

1. Daily surveillance of cask for security and safeguards:

   Visual inspection of air vents for detection of blockage.

2. Annual inspection of cask exterior:

   Visual inspection of surface for chipping, spalling, or other surface defects. If found, a defect should be corrected by regrouting the affected area.

10.0 RADIATION PROTECTION

10.1 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)

10.1.1 Policy Considerations

It is the policy of SNC to ensure that the TranStor™ Storage System 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 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 delineated in site documentation.

10.1.2 Design Considerations

The design of the TranStor™ Storage System complies with 10 CFR 72.3 concerning ALARA. Specific considerations that are directed toward ensuring ALARA are:

- Thick walls, bottom, and lids to reduce side, top, and bottom working dose rates.
- Non-Planar paths for the air inlet and outlet vents in the cask to minimize radiation streaming.
- Fuel loading procedures that follow accepted practice and are built on existing experience.
- Simple handling operations to reduce the time of cask movements.
- Totally passive system that requires minimum maintenance.
- Hard glossy coating on the transfer cask and basket surfaces to minimize decontamination efforts.

10.1.3 Operational Considerations

The preliminary operating steps have been determined following the ALARA guidelines. The following are specific operational features to reduce occupational exposure:

- Automated welding that minimizes operator's presence in the weld area.
- Decontamination of exterior of transfer cask while basket is still filled with water to reduce exposure to technicians.
- Flushing the basket-transfer cask gap with clean water to prevent contamination of the surfaces.

However, the exact handling and operation of the TranStor™ Storage System will be site specific, and, as such, details on the ALARA characteristics of the operations and procedures will be consistent with the practices at the TranStor™ System owner plants.

10.2 Radiation Protection Design Features

Confinement features of the TranStor™ Storage System are addressed in Chapter 7.0. A high degree of integrity for the confinement of radioactive materials is demonstrated in Chapters 3.0 and 11.0. Details of the radiation protection design features are provided in Chapter 5.0. Section 10.3 below provides the results of the actual dose assessment calculations.

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:

TABLE 10.3-1

PERSONNEL REQUIREMENTS FOR CASK OPERATIONS

Item	Personnel Required
Prepare Cask	2 Technicians
Load Fuel	2 Technicians, 2 Operators, 2 HPs
Close Basket	2 Technicians, 2 Welders, 2 HPs 1 Inspector
Basket Transfer to Cask	4 Technicians, 2 HPs
Move to Storage Area	2 Operators, 1 HP
Daily Surveillance	1 Inspector

10.3.2 Dose Rates

The TranStor™ Storage 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 conservatively calculated working dose rates for loading and handling the casks, under the normal storage conditions. These values are taken from Chapter 5.0 and based on design basis fuel. All dose rates include both gamma and neutron flux components.

TABLE 10.3-2

MAXIMUM EXPECTED DOSE RATES

Location	Working*
Side Surface of Transfer Cask	80
Basket Top (Outside Surface of Structural Lid)	60
Storage Cask Surface Within Storage Area	10

* Working dose is calculated dose rate 1 meter from the surface

10.3.3 Estimated Man-Rem Exposures for Operation, Maintenance, and Inspection of the Equipment

Working dose rates and personnel requirements for the TranStor™ Storage Cask loading cycle and move-to-storage cycle are shown in Table 10.3-3. These values are based on the conservatively calculated dose rates for the casks containing design basis fuel. In practice, the total dose to load a VSC-24 cask has averaged approximately 0.4 man-rem. Figure 10.3-1 shows this operational data. The dose burden for the TranStor™ cask will be essentially identical to that of the VSC-24 due to their identical operation and handling.

Based on the 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:

$$1.42 \text{ man-rem/cycle} \times 3 = 4.36 \text{ man-rem/year}$$

Conservative estimates of the annual maintenance inspection and survey requirements result in a collective dose of 0.06 person-rem per year per cask. This is based on a one minute inspection of a freshly loaded cask (i.e., 0.010 rem/hr · 1 hr/60 · 1 min · 365 inspections per year). The dose will, of course, decrease with the age of the cask.

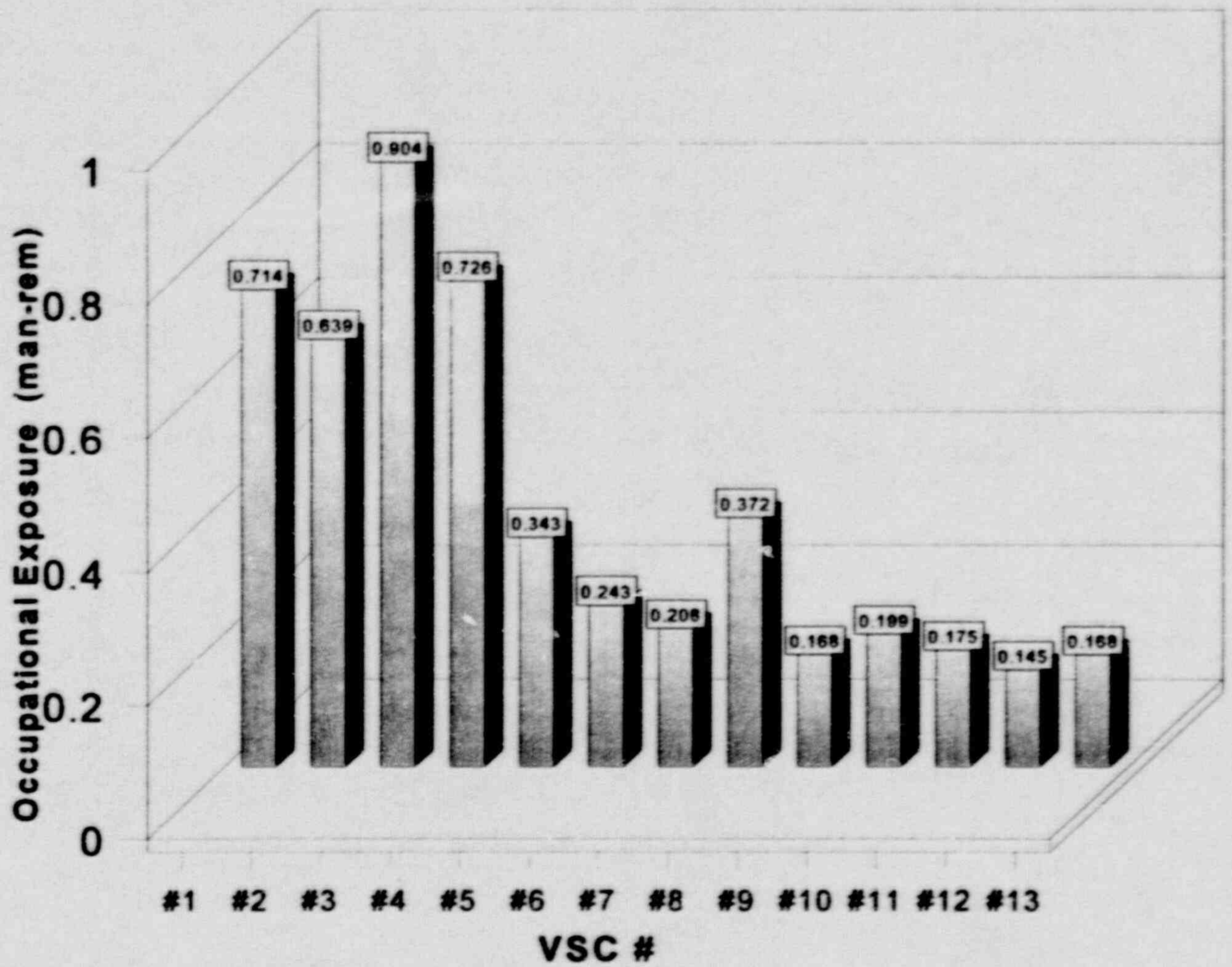


FIGURE 10.3-1
TRENDS IN VSC OCCUPATIONAL EXPOSURE

TABLE 10.3-3

ESTIMATED PERSONNEL EXPOSURE DOSES

Item	Personnel Work Groups	Exposure Time (hrs)	Working Dose Rate (mrem/hr)	Exposure (man-mrem)
Load Basket,	2 Operators	6.0	~0.1*	1.2
Monitor	2 Technicians	6.0	~0.1*	1.2
	2 HPs	6.0	~0.1*	1.2
Decontaminate	2 Technicians	2.0	80**	320.0
Transfer Cask, Monitor	1 HP	4.0	~0.1*	0.4
Vacuum Dry	2 Technicians	1.0	60	120.0
		19.0	~0.1*	3.8
Weld Shield and	2 Welders	2.0	60	240.0
Structural Lid,		12.0	~0.1*	2.4
NDE,	1 Inspector	2.0	60	120.0
Monitor	1 HP	16.0	~0.1*	1.6
Load Storage Cask,	2 Operators	3.0	80**	480.0
Monitor	2 Technicians	3.0	10.0	60.0
	2 HPs	3.0	~0.1*	6.0
Move to Storage,	2 Operators	2.0	10.0	40.0
Monitor	1 HP	2.0	10.0	20.0
Totals				1,418***

* Assumed radiation reading in pool area. In reality, it is essentially zero.

** Assumes worst case of dry basket. If water is left in basket, dose rate will be less than half of the value reported here.

*** Worst case calculated estimate using design basis fuel. Average for the actual casks loaded is less than ~400 man-mrem/cask.

11.0 ACCIDENT ANALYSIS

The analyses of the off-normal and accident design events identified by 10 CFR 72 and ANSI 57.9 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 often 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.

Due to the generic nature of this report, the majority of the accident analyses are based on very conservative assumptions and methodologies. As a result, the actual response of the TranStor™ Storage System to the events discussed in this section is expected to be much better than reported (i.e., lower stresses, temperatures, and radiation doses). 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.

TABLE 11.0-1

DESIGN BASIS OFF-NORMAL AND ACCIDENT EVENTS

Off-Normal Events (expected frequency 1/year)

1. Off-Normal Environmental Conditions (100°F with Solar Load and -40°F, no Solar Load)
2. Blockage of One-Half of the Air Inlets
3. Interference During Basket Lowering from Transfer Cask into Storage Cask
4. Small Release of Potential Basket Surface Contamination
5. Off-Normal Handling Load - Impact at 2 ft/sec Crane Speed

Postulated Events (Not Expected But Could Occur During Cask Lifetime)

1. Rupture of All Fuel Pins with Subsequent Breach of Basket and Ground Level Release
2. Maximum Anticipated Heat Load 125°F Ambient Temperature and Full Solar Load
3. Tornado
4. Flood
5. Earthquake
6. Accident Pressurization
7. Complete Blockage of Air Inlets
8. Explosion
9. Lightning
10. Hypothetical Tipover Event

11.1 Off-Normal Events

This section covers design 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 or MRS facilities may be 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, basket, 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 26 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 four to five days. Since one would not normally expect ambient conditions at any facility in the United States to be subjected to such conditions, this situation is classified as an off-normal event and the analysis is used 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 four to five 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 are required for the bounding analysis of thermal stresses in the basket. As discussed in Section 3.4.5, the coldest service temperature for the basket components would be -40°F and this is not a concern for the materials used.

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 facilities.

11.1.1.3 Analysis of Effects and Consequences

The same models and calculations used for the normal conditions (described in Chapter 4.0) are used to analyze the -40°F and 100°F ambient conditions. Both analyses are conservatively based on the steady state condition although, as stated above, it is highly unlikely that these extreme temperatures can last long enough for the cask to reach this mode. The maximum steady state temperatures for both cases are previously presented in Table 4.1-1.

Figures 11.1-1 and 11.1-2 provide the details of the temperature distributions for the storage cask. As these figures show, the temperatures are within the concrete allowables for the off-normal short-term events. The thermal gradients across the cask wall are less than for the all inlets blocked case used in Chapter 3.0, so additional stress analysis is not required. The basket temperature distributions are provided in Figures 11.1-3 through 11.1-6. The thermal gradients across the basket interior are higher for the -40°F case than for all other cases, hence, the stress analysis described in Chapter 3.0 is based on this condition.

11.1.1.4 Corrective Actions

The TranStor™ Storage System is designed to accommodate steady state ambient temperatures of 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 of Event

This event is a postulated blockage of one-half of the air flow inlets. Because the storage cask has two 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 hypothetical accident is considered in Section 11.2.7.

11.1.2.2 Detection

This event would be detected visually by the security force as they perform their required daily surveillance.

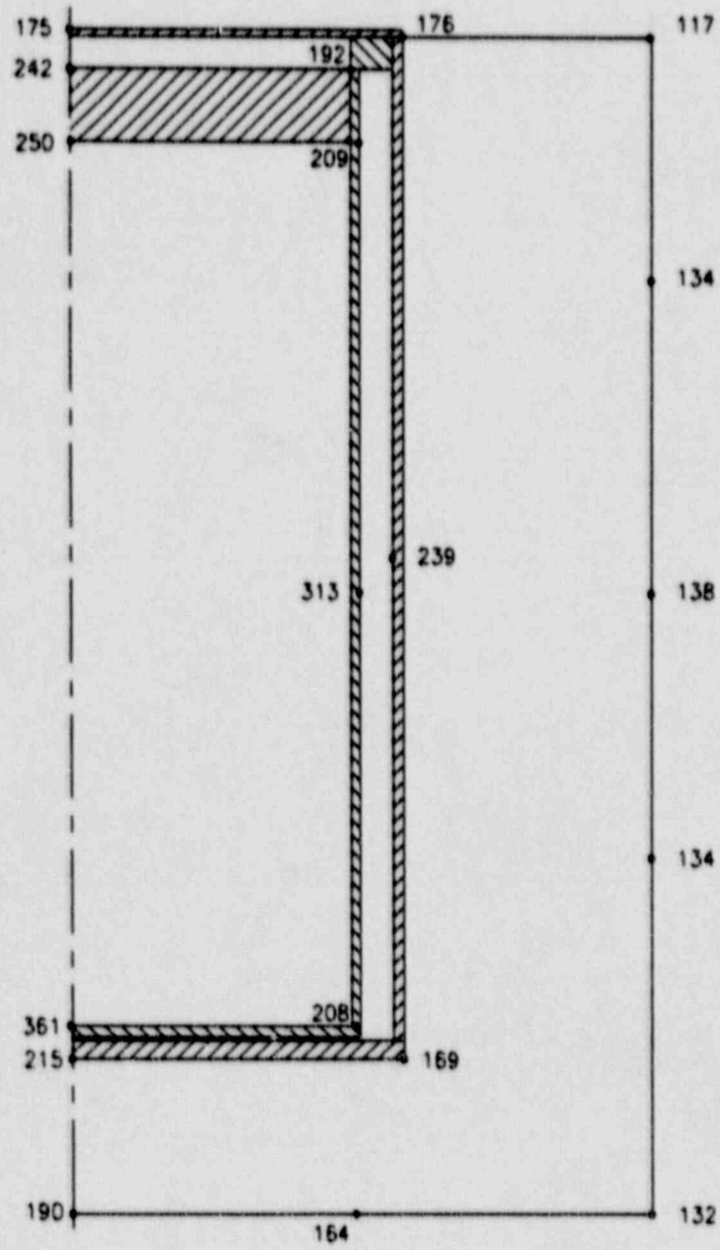


FIGURE 11.1-1

CASK TEMPERATURE DISTRIBUTION FOR 100°F AMBIENT CONDITION

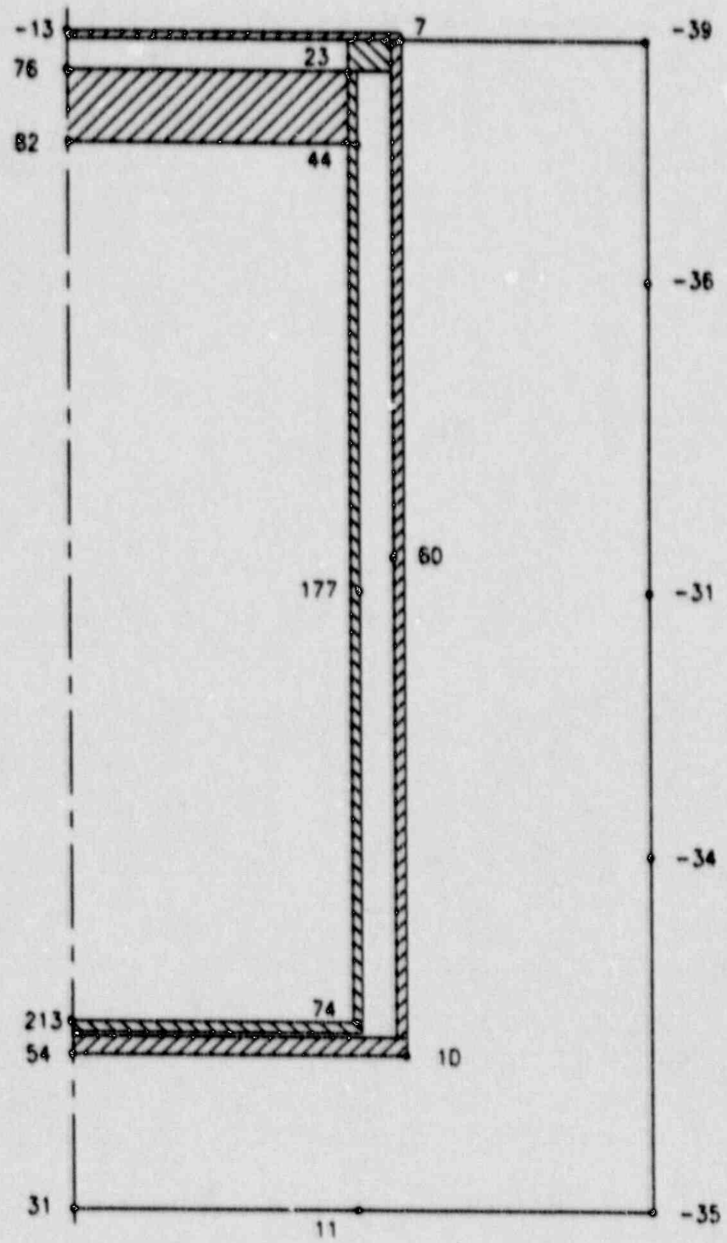


FIGURE 11.1-2

CASK TEMPERATURE DISTRIBUTION FOR -40°F AMBIENT CONDITION

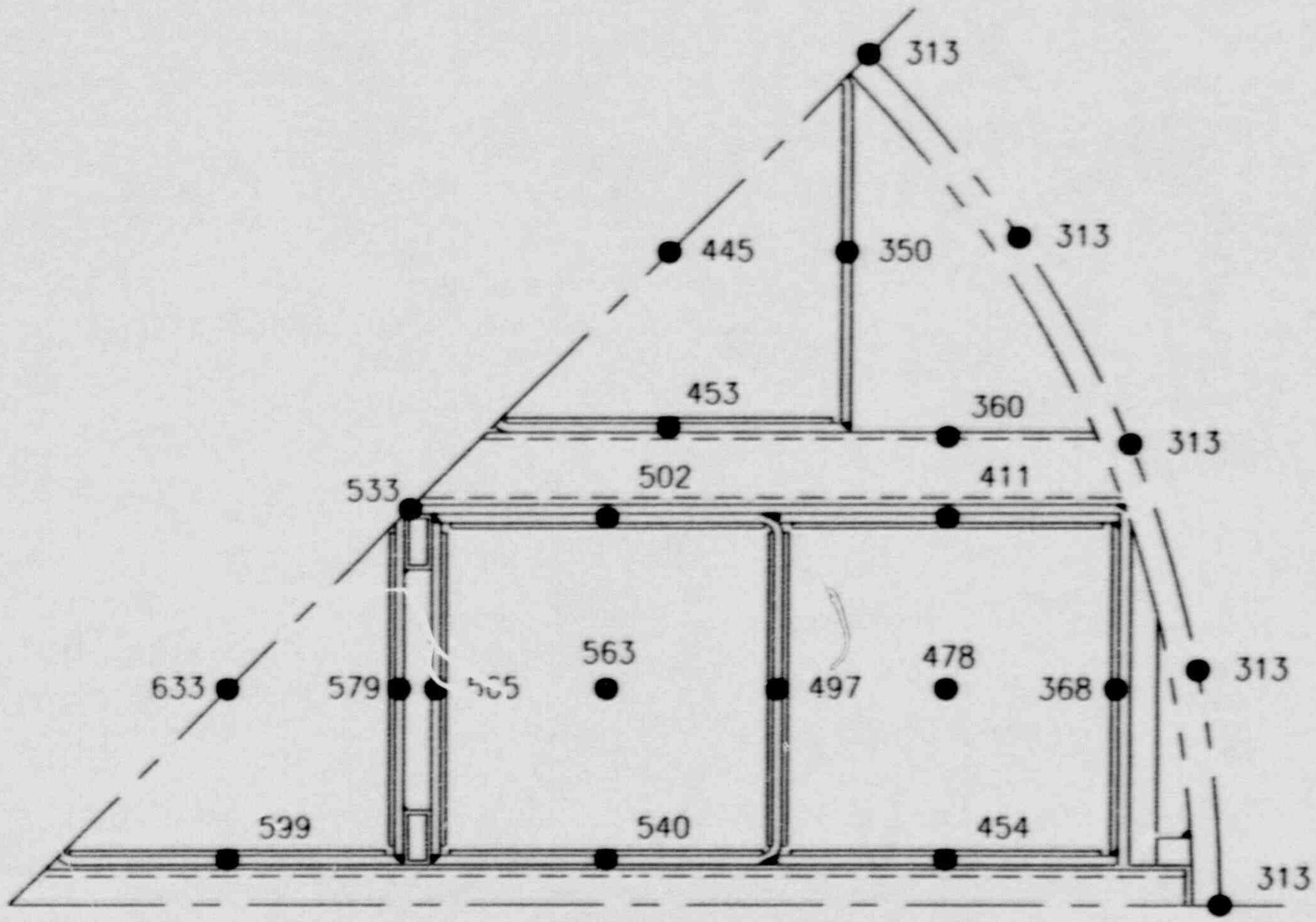


FIGURE 11.1-3

PWR BASKET TEMPERATURE DISTRIBUTION FOR 100°F AMBIENT CONDITION

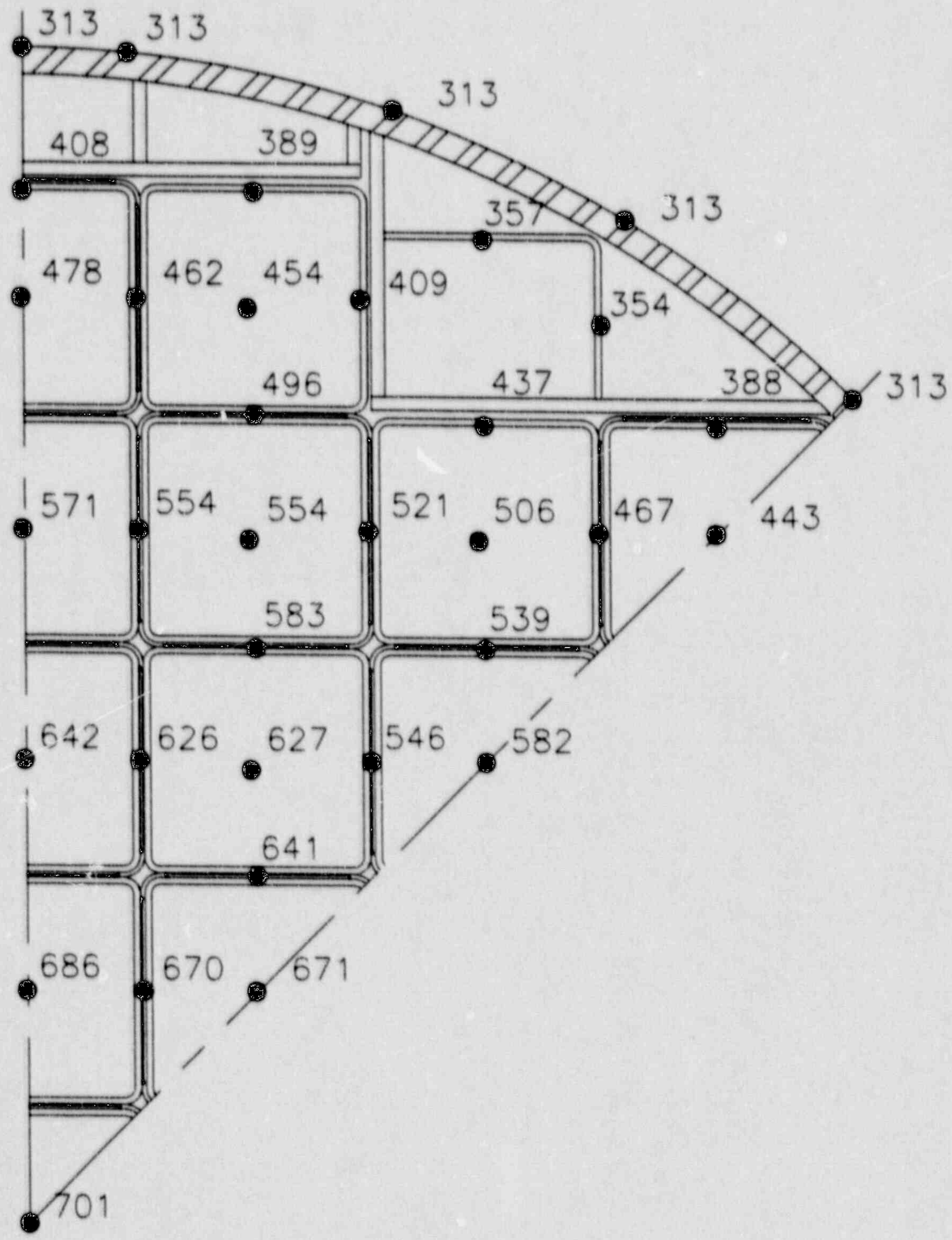


FIGURE 11.1-4

BWR BASKET TEMPERATURE DISTRIBUTION FOR 100°F AMBIENT CONDITION

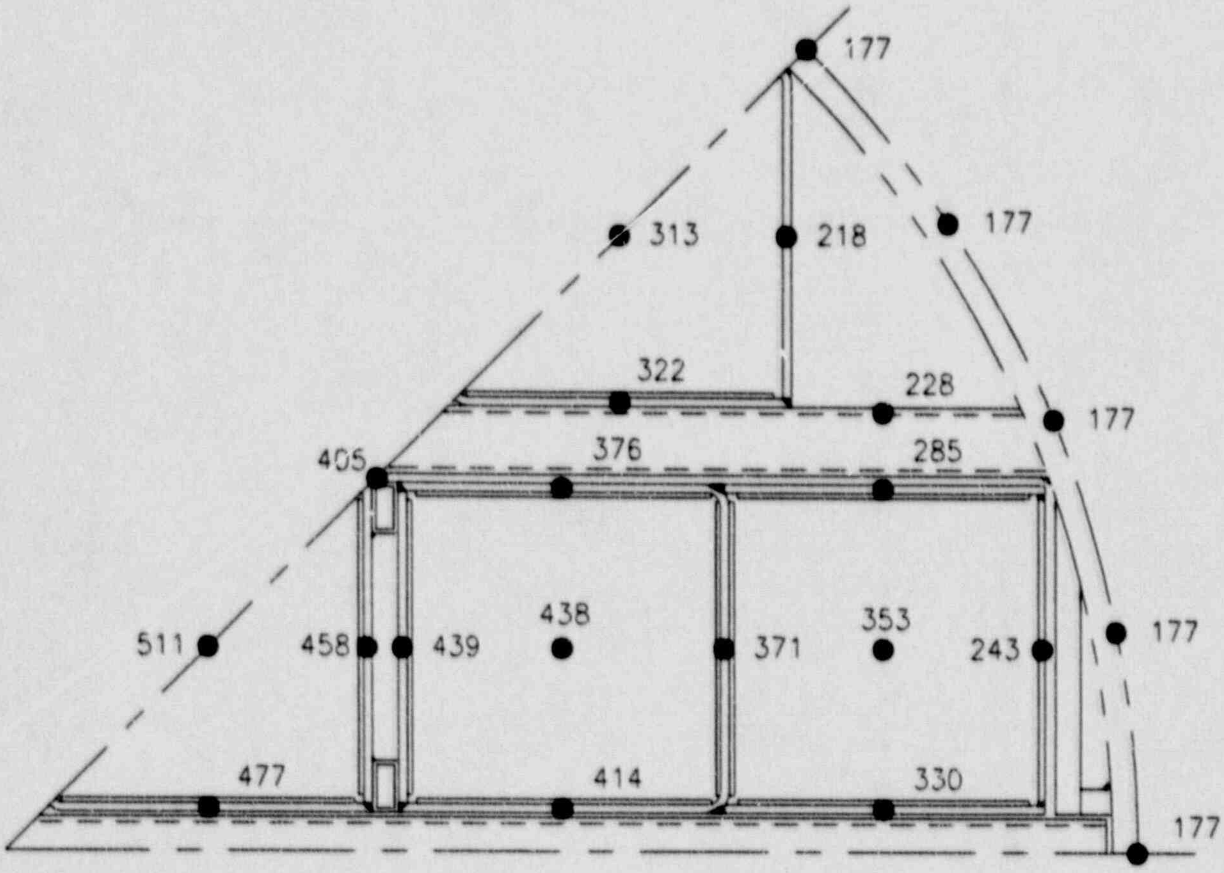


FIGURE 11.1-5

PWR BASKET TEMPERATURE DISTRIBUTION FOR -40 °F AMBIENT CONDITION

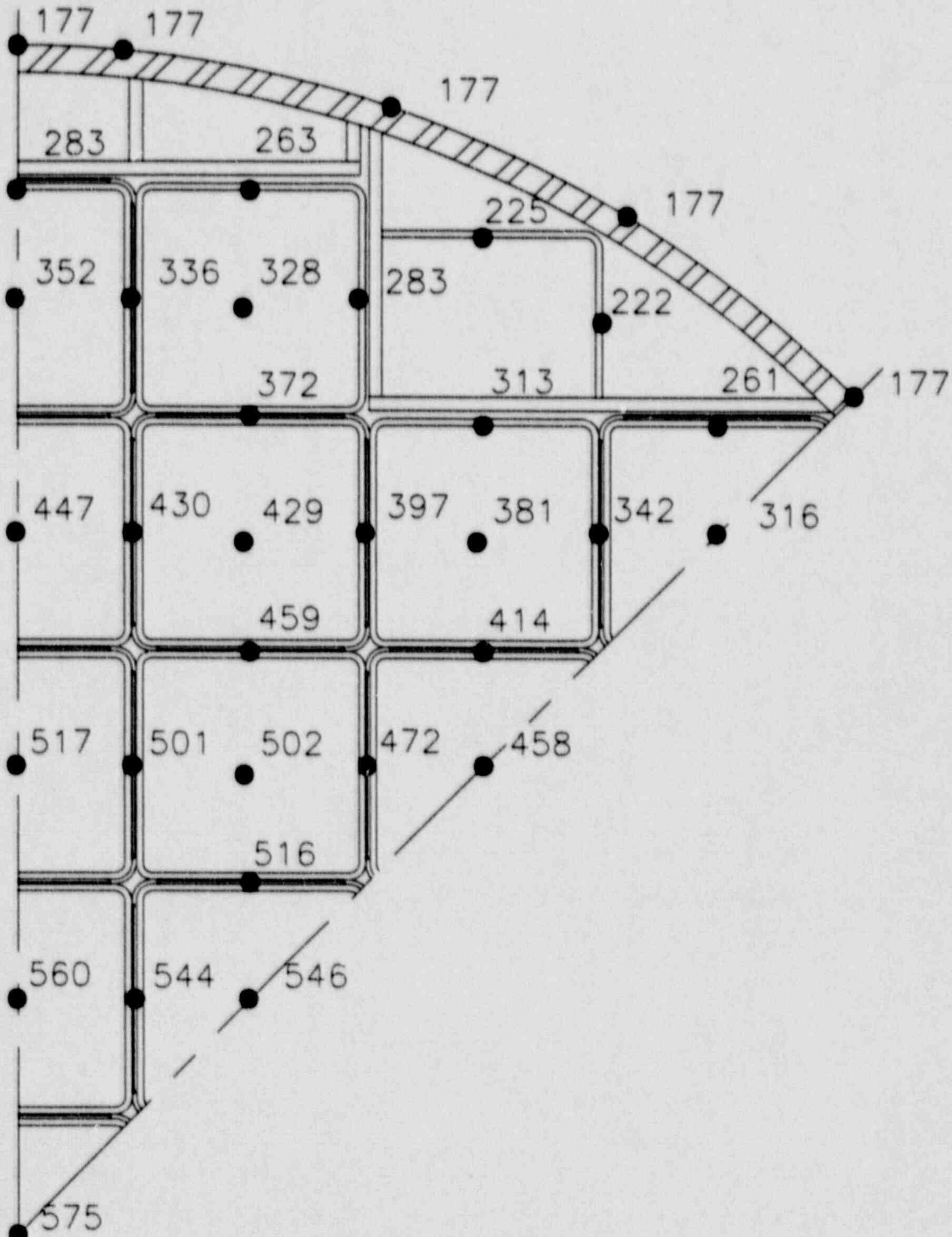


FIGURE 11.1-6
BWR BASKET TEMPERATURE DISTRIBUTION FOR -40 °F AMBIENT CONDITION

11.1.2.3 Analysis of Effects and Consequences

The analysis of this event uses the same calculational models described in Chapter 4.0 and assumes that the blockage occurs during normal (75 °F day) ambient conditions. Blocking one 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 storage cask flow system is designed so that the inlet losses are a relatively small portion of the total pressure drop due to the air flow. As a result, the total increase in the $\Sigma\ k_i/A_i^2$ is only about 65% which, in turn, reduces the air mass flow rate by 17%. This reduced air flow creates a higher ΔT between the inlets and outlets so that the air exiting temperature rises from 180 °F to 202 °F. As seen from Table 4.1-1, this rise has a small effect on the storage cask operation. Also, as discussed in Section 3.4.4, the thermal stress analysis for this case is bounded by the analysis presented in Chapter 3.0.

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 worse 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. The estimated dose is 6 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 Basket Lowering from Transfer Cask into Storage Cask

11.1.3.1 Cause of Event

This condition is highly unlikely because diameter of the cask cavity is substantially larger than that of the basket and the transfer cask is aligned with the concrete cask using special alignment holes. In addition, proper procedures are used to ensure cleanliness of the components and levelness of the storage and transfer casks. Nevertheless, this event is analyzed to bound any potential off-normal occurrence.

11.1.3.2 Detection

This event would be detected by audible noises emitted from the basket sliding on the transfer cask, storage cask, or other material and, in the worse case, by a slackening of the wire slings which connect the basket to the crane hook.

11.1.3.3 Analysis of Effects and Consequences

Since the only forces acting on the basket during its lowering are gravity, the worst case condition (if the basket were to be completely supported from its interference) would be a load of 1g on the basket bottom or 3.3g on the side (assuming a friction coefficient of 0.3). Since the basket has been designed for much more severe loading conditions, i.e., the drop accident described in Reference 1.1, the interference during transfer will not cause any undue stresses in the basket. Furthermore, the force required to pull the basket back into the transfer cask would not be higher than two times the basket weight. A typical fuel building crane (100 to 125 tons) and the basket lifting components (designed with an 11 to 1 factor of safety) can easily withstand this load.

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 burden of less than 80 man-mrem.

In summary, an interference during the basket movement from the transfer cask to the storage cask will not cause any loading conditions more severe than those analyzed for other events. Therefore, this off-normal condition does not require further structural, thermal, or nuclear analysis.

11.1.3.4 Corrective Actions

If there is an indication that the basket lowering is being obstructed, the technician or engineer present should take the following corrective actions:

1. Immediately signal to a crane operator to halt lowering the basket.
2. Raise the basket back into the transfer cask.
3. Check alignment of the storage cask and transfer cask.
4. Check levelness of the storage cask and transfer cask.
5. Retry lowering.
6. If interference still exists, pull the basket back into transfer cask and remove transfer cask.
7. Check storage cask for foreign objects.
8. Check transfer cask-basket gap for foreign objects (return to pool if necessary).

The above corrective actions are presented to provide overall guidance on how to handle basket/ transfer cask/storage cask interference. More detailed site specific procedures will have to be developed taking into account the actual building layout, handling equipment, and other site specific details.

11.1.4 Small Release of Radioactive Particulates from the Basket Exterior

11.1.4.1 Cause of Event

Although precautions are taken to avoid introducing contamination to the outside of the basket when it is submerged in the spent fuel pool, it is feasible that a portion of the basket could become slightly contaminated. This surface contamination would be removed as required in Section 12.2.1. However, if it is somehow not detected prior to outdoor storage in the storage cask, the particulate may become airborne and drift off-site.

11.1.4.2 Detection

This surface contamination should be detected prior to transferring the basket to the storage cask by routine health physics smears on the basket top and sides prior to transfer to the storage cask. The acceptable contamination limits are provided in Chapter 12.0. However, if higher levels of basket surface contamination did exist and were released, they would only be detected by long term radiological instrumentation (TLDs) on the site fence. The release would be too low 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 undetected, the worst consequence would be that it becomes loose after the basket is placed on the storage pad. For such an atmospheric release, one may conservatively assume that the particulate behaves as a gas and is inhaled by a person standing at the plume centerline during the entire passage of the release at a distance of 100 meters from the release point. That person would receive a total dose that can be calculated using the methods described in Regulatory Guide 8.34. It is based on Committed Effective Dose Equivalent (CEDE) as follows:

$$\text{CEDE(mrem)} = \text{DCF}\left(\frac{\text{mrem}}{\mu\text{Ci}}\right) \times \frac{\chi}{Q}\left(\frac{\text{sec}}{\text{m}^3}\right) \times t\,(\text{sec}) \times \text{BR}\left(\frac{\text{m}^3}{\text{sec}}\right) \times Q\left(\frac{\mu\text{Ci}}{\text{sec}}\right)$$

where

CEDE = Committed Effective Dose Equivalent

DCF = Dose Conversion Factor for the ^{60}Co Inhalation Dose

χ/Q = Atmospheric Dispersion

t = Reference Person's Exposure Time

BR = Reference Person's Breathing Rate

Q = Release Rate

Assuming a conservative release time of 2 hours and even distribution of the particulates on the external surface of the basket, the above equation is used to calculate the contamination level that would produce the CEDE of 50 mrem (only 1% of the regulatory limit for off-normal conditions). The resulting activity level is 1.6×10^5 dpm/cm^2 - an unrealistically high level of surface contamination originating from a well-maintained spent fuel pool. Even at 20 ft from the cask, to produce the dose of 50 mrem the contamination would have to be 1.0×10^5 dpm/cm^2.

The above analysis demonstrates that even if the basket accidentally became contaminated and was released for storage not meeting the condition of Section 12.2.1.3, the off-site radiological consequences from the release would be well within regulatory limits.

11.1.4.4 Corrective Actions

No corrective action is required since the radiological consequences are well within the regulatory limits of 10 CFR 72.

11.1.5 Basket Off-Normal Handling Load

11.1.5.1 Cause of Event

The basket shall be handled exercising maximum care. However, it is possible that during movement of the loaded basket inside the transfer cask or during its transfer into the storage cask, an inadvertent lateral or vertical crane motion could cause an impact of the basket against the building or storage cask wall. To bound such an off-normal event, the analysis below has been performed.

11.1.5.2 Detection

This event would be detected by personnel performing the operations.

11.1.5.3 Analysis of Effects and Consequences

The off-normal handling load is analyzed assuming an impact at 2 ft/sec crane speed (see Table 1.2-1). The resulting deceleration load was previously calculated for the MSB-24 (Reference 1.2) to be 17.5g. Since the MSB-24 is very similar to the TranStor™ basket in geometry, weight, and materials, the same off-normal handling load was used for the TranStor™ basket.

Stresses due to the 17.5g load were ratioed from results of the basket 30-foot drop analysis presented in Reference 1.1. These stresses were further combined with the stresses due to other loads and evaluated in Tables 11.1-1 and 11.1-2. It can be seen that the stresses are within ASME Code allowables for Service Level C conditions.

TABLE 11.1-1

PWR BASKET STRESSES RESULTING FROM OFF-NORMAL HANDLING EVENT

		Off-Normal Handling	Dead Weight	Design Pressure	Σ	ASME Level C Allowable
Basket	P_M	5.6	0.1	0.2	5.9	18.6
Shell	$P_L + P_b$	21.8	0.2	2.8	24.8	27.8
Bottom	P_M	7.3	0.0	0.1	7.4	18.6
Plate	$P_L + P_b$	14.7	0.2	1.9	16.8	27.8
Structural	P_M	2.1	0.0	0.0	2.1	18.6
Lid	$P_L + P_b$	9.7	0.1	0.6	10.4	27.8
Sleeve	P_M	1.2	0.1	0.0	1.3	27.5
Assembly	$P_L + P_b$	26.1	0.1	0.0	26.2	41.2
Shield Lid	P_M	0.9	0.1	0.0	1.0	13.9
Ring Weld	$P_L + P_b$	2.1	0.1	0.0	2.2	20.9
Top	P_M	8.2	0.1	0.4	8.7	18.6
Weld	$P_L + P_b$	9.5	0.2	0.5	10.2	27.8
Shield	P_M	0.9	0.1	0.3	1.3	18.6
Lid Weld	$P_L + P_b$	2.1	0.1	0.4	2.6	27.8

Note: Thermal stresses are not included because they are secondary and do not need to be evaluated for Service Level C conditions.

TABLE 11.1-2

BWR BASKET STRESSES RESULTING FROM OFF-NORMAL HANDLING EVENT

		Off-Normal Handling	Dead Weight	Design Pressure	Σ	ASME Level C Allowable
Basket	P_M	7.1	0.1	0.2	7.4	18.6
Shell	$P_L + P_b$	14.0	0.2	2.8	17.0	27.8
Bottom	P_M	5.0	0.0	0.1	5.1	18.6
Plate	$P_L + P_b$	10.9	0.2	1.9	13.0	27.8
Structural	P_M	2.3	0.0	0.0	2.3	18.6
Lid	$P_L + P_b$	6.1	0.1	0.6	6.8	27.8
Sleeve	P_M	2.1	0.1	0.0	2.2	27.5
Assembly	$P_L + P_b$	19.0	0.1	0.0	19.1	41.2
Shield Lid	P_M	0.9	0.1	0.0	1.0	13.9
Ring Weld	$P_L + P_b$	2.1	0.1	0.0	2.2	20.9
Top	P_M	8.2	0.1	0.4	8.7	18.6
Weld	$P_L + P_b$	9.5	0.2	0.5	10.2	27.8
Shield	P_M	0.9	0.1	0.3	1.3	18.6
Lid Weld	$P_L + P_b$	2.1	0.1	0.4	2.6	27.8

Note: Thermal stresses are not included because they are secondary and do not need to be evaluated for Service Level C conditions.

11.1.5.4 Corrective Actions

The basket is designed to withstand acceleration loads which bound this handling load. No corrective actions are required.

11.2 Accidents

This section provides the results of analyses of several hypothetical accidents which are presented to show that the TranStor™ Storage 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 analyses for bounding phenomena that could occur over the life of the cask (tornado, earthquake, floods, etc.).

11.2.1 Failure of All Fuel Pins with Subsequent Ground Level Breach of Basket

11.2.1.1 Cause of Accident

This accident is a hypothetical failure of all fuel rods and the basket confinement boundary and release of the fission gases. As shown in other sections, no conceivable accident could cause this event because even if all rods failed (a hypothetical occurrence by itself), the basket would withstand the resulting pressure. However, the event is analyzed to demonstrate the large safety margin afforded by the TranStor™ Storage System.

In terms of off-site radiological consequences, this is a completely bounding event for any ISFSI site. If the shielding of the cask or canister is ever damaged, temporary shielding can quickly be used to limit the off-site doses. However, if gaseous fission products are released nothing can be done to prevent them from drifting beyond the site boundary. Therefore, this accident envelops all other events in terms of off-site doses.

11.2.1.2 Accident Analysis

This hypothetical accident assumes that 100% of the fuel rods in a basket fail and release 30% of the available fission gases. The 30% release fraction is a conservative number since most data shows a realistic release fraction to be just 8% (Reference 11.1). This results in 28,755 Ci released from the basket containing 45 GWd, 6-year cooled PWR fuel or 26,108 Ci released from the basket containing 35 GWd, 5-year cooled BWR fuel. The gas is conservatively modeled as ^{85}Kr since previous analysis (Reference 11.2) has shown that other gaseous fission products do not add significantly to the dose from five year old fuel. Further assuming the basket fails and releases the entire amount of the ^{85}Kr, the off-site doses are calculated by using the methods described in Regulatory Guide 1.25. The important parameters for these calculations are:

Wind Speed - 1 m/sec.
Dispersion - per Figure 1 of RG 1.25
Release Time - 2 hours

The applicable equation used is:

$$D(\text{mrem}) = DCF\left(\frac{\text{mrem}\cdot\text{m}^3}{\text{hr}\cdot\mu\text{Ci}}\right) \times \frac{\chi}{Q}\left(\frac{\text{sec}}{\text{m}^3}\right) \times t(\text{hr}) \times Q\left(\frac{\mu\text{Ci}}{\text{sec}}\right)$$

where

D = Total Whole Body Dose

DCF = Dose Conversion Factor (gamma)

χ/Q = Atmospheric Dispersion

t = Reference Person's Exposure Time

Q = Release Rate

For close distances (< 100 meters), the χ/Q dispersion values are not available. Hence, the calculation conservatively assumes that no plume dispersion takes place at these distances and the whole body dose results from an infinite line source.

11.2.1.3 Accident Dose Calculations

The results of the calculation for several distances are shown below:

Distance to Nearest Cask, m	Dose, mrem Whole Body (PWR)	Dose, mrem Whole Body (BWR)
6.1 (20 ft)	2698	2420
100	487	439
200	140	126
500	28	26
1000	10	9

As can be seen, even for very near site boundaries, the dose rates are well below the 5,000 mrem limit which 10 CFR 72.106 specifies for design basis accidents.

11.2.2 Maximum Anticipated Heat Load

11.2.2.1 Cause of Accident

This accident is a natural phenomena. It assumes a 125 °F ambient temperature with full solar loads occurring when a cask loaded with the design basis fuel is first placed in service.

11.2.2.2 Accident Analysis

The accident was 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. It conservatively assumes that the cask has been exposed to 125°F ambient conditions with full solar loads (on for 12 hrs, off for 12 hrs) for a sufficiently long time to reach steady state. In reality, this weather condition would not be maintained long enough because the massive cask requires several days to reach steady state.

The results of this analysis are included in Table 4.1-1. It can be seen that the temperatures are below the corresponding short-term limits. Furthermore, due to the temperature increase on the cask surface (due to full solar loads), the thermal gradient across the concrete wall (and, hence, the stresses) are lower for this accident condition than for other cases and no additional stress analysis is required. Basket temperature gradient and internal pressure are also presented in Tables 3.4-3 and 3.4-4 and shown to be bounded by the analysis in Chapter 3.0.

11.2.2.3 Accident Dose Calculation

There are no dose implications due to this 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 storage cask include the possibility of damage due to wind loading, wind generated pressure differentials, and 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 tornado missile spectrum as well as the methods used to convert the tornado and wind loadings into forces on the cask are based on NUREG-0800.

Wind Loads

The tornado wind velocity is transformed into an effective pressure applied to the cask using procedures delineated in ANSI A58.1. The maximum velocity pressure, p, is determined from the maximum tornado wind velocity as follows:

$$p = (0.00256)\, V^2 \;, \text{ psf}$$

where:

V = Maximum tornado wind speed
= 360 mph

The above effective velocity pressure is assumed constant with height and, since the cask is small in relation to the radius of the tornado, is assumed to be uniform over the projected area of the cask. Gust factors are taken as unity in evaluating effects of velocity pressures on cask surfaces.

The total tornado wind force on the projected area of the cask, W_w, is then computed as follows:

$$W_w = p(C_f)(A_p)$$

where:

p = Effective velocity pressure (psf)

C_f = Net pressure coefficient (Reference ANSI A58.1, Table 12)

A_p = Projected area of cask normal to wind

Substituting the values for the TranStor™ storage cask yields the wind force of 36,257 lbs and the resulting tipover moment of 4.0×10^6 lbs-in. This is clearly insufficient to tipover or slide a 285,000 lbs cask since the restoring moment is $(285{,}000)\cdot(65) = 18.5 \times 10^6$ lbs-in and the sliding coefficient of friction is typically taken as 0.3. It must be noted, that the weight used here is conservatively light and the analysis bounds all PWR and BWR configurations.

Furthermore, the highest shear and moment produced by the tornado wind in the storage cask would exist at the junction between its wall and bottom. The corresponding force and moment are calculated to be 36,257 lbs and 3.7×10^6 lbs-in respectively. Both of these loads are included in load combinations and evaluated in Table 3.4-8.

Tornado Missiles

The cask is designed to withstand the impacts associated with postulated tornado generated missiles as identified in NUREG-0800, Spectrum I. These missiles consist of a massive high kinetic energy deformable missile (automobile), a rigid missile to test penetration resistance (armor piercing shell), and a small rigid missile of a size sufficient to just pass through any openings in protective barriers (small steel sphere). All missiles are assumed to impact in a manner that produces the maximum damage to the cask. The cask has been evaluated for the impacts associated with each of the missiles described above. Analyses for penetration resistance of the cask body and closure elements to the armor piercing shell missile indicate that sufficient thicknesses of concrete and steel are 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 which produces the highest force. The analyses indicate that the cask will remain upright following the event and that loads associated with this impact do not compromise its integrity. The analyses are summarized below.

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 (Reference 11.3 and 11.4). The depth of penetration, X, as predicted using this approach may be expressed as follows:

For $X/2d \leq 2.0$:

$$X = [4KNWd^{-0.8} (V/1000)^{1.8}]^{0.5}$$

where:

d = Diameter of missile (8 inches)

K = Coefficient depending on the concrete strength
$= 180/(f_c)^{0.5}$

N = Missile shape factor
= 1.14 for sharp nosed missiles (Reference 11.3)

W = Missile Weight (275 lbs)

V = Velocity (184.8 ft/sec)

The depth of penetration is calculated to be 5.69 inches. The minimum depth of concrete necessary to preclude spalling and scabbing is then selected as three times the predicted depth of penetration or 17.1 inches. Since the minimum thickness of concrete in the cask body is well in excess of this value it is concluded that adequate protection is provided for local damage due to tornado missiles.

Local Damage Prediction - Cask Closure Plate

The storage cask 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 artillery shell impacting a steel plate, the ability of the closure plate to adequately withstand tornado generated missiles is established.

The perforation thickness in a steel plate is given in Reference 11.5.

$$T = [(0.5)(M_m)(V_s)^2]^{2/3}/672d_m$$

where:

T = Perforation thickness (in)

M_m = Missile mass (slugs)
= W/g = 275/32.2 = 8.54 slugs

V_s = Missile striking velocity (ft/sec) = 184.8 ft/sec

d_m = Missile diameter (in) = 8 in.

Substituting the above values in the equation for perforation thickness yields:

T = 0.52 in

Therefore, the cask closure plate is adequate to withstand local impingement damage due to tornado generated missiles.

Overall Damage Prediction

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. The assessment is conducted using the principles of conservation of momentum during the impact event. The analyses which are summarized below indicate that the cask will remain upright.

From the principles of conservation of momentum, the change in angular momentum of the missile due to impact must equal the change in angular momentum of the cask. With reference to Figure 11.2-1, these relationships may be expressed as presented below.

Angular momentum of the missile before impact (relative to the cask corner)

$$\Phi_{missile} = M\upsilon_0 H$$

where:

M_m = mass of missile
= 3960 lbs or 123 slugs

v_0 = velocity of missile before impact
= 126 mph or 184.8 ft/sec

H = height of the cask

The final velocity of the missile will depend upon the coefficient of restitution of the missile, the geometry of the missile and target, the angle of incidence, and upon the amount of energy dissipated in deforming the missile and target. Based upon tests conducted by EPRI (Reference 11.3), it is assumed that the final velocity of the missile following the impact is zero. Hence, all of its angular momentum is imparted into the cask.

The change in angular momentum of the cask about a point on the bottom rim becomes:

$$\Phi_{cask} = I_c\omega$$

where:

I_c = mass moment of inertia of cask about the bottom corner

$$= M_c\left(\frac{5}{4}r^2 + \frac{1}{3}H^2\right)$$

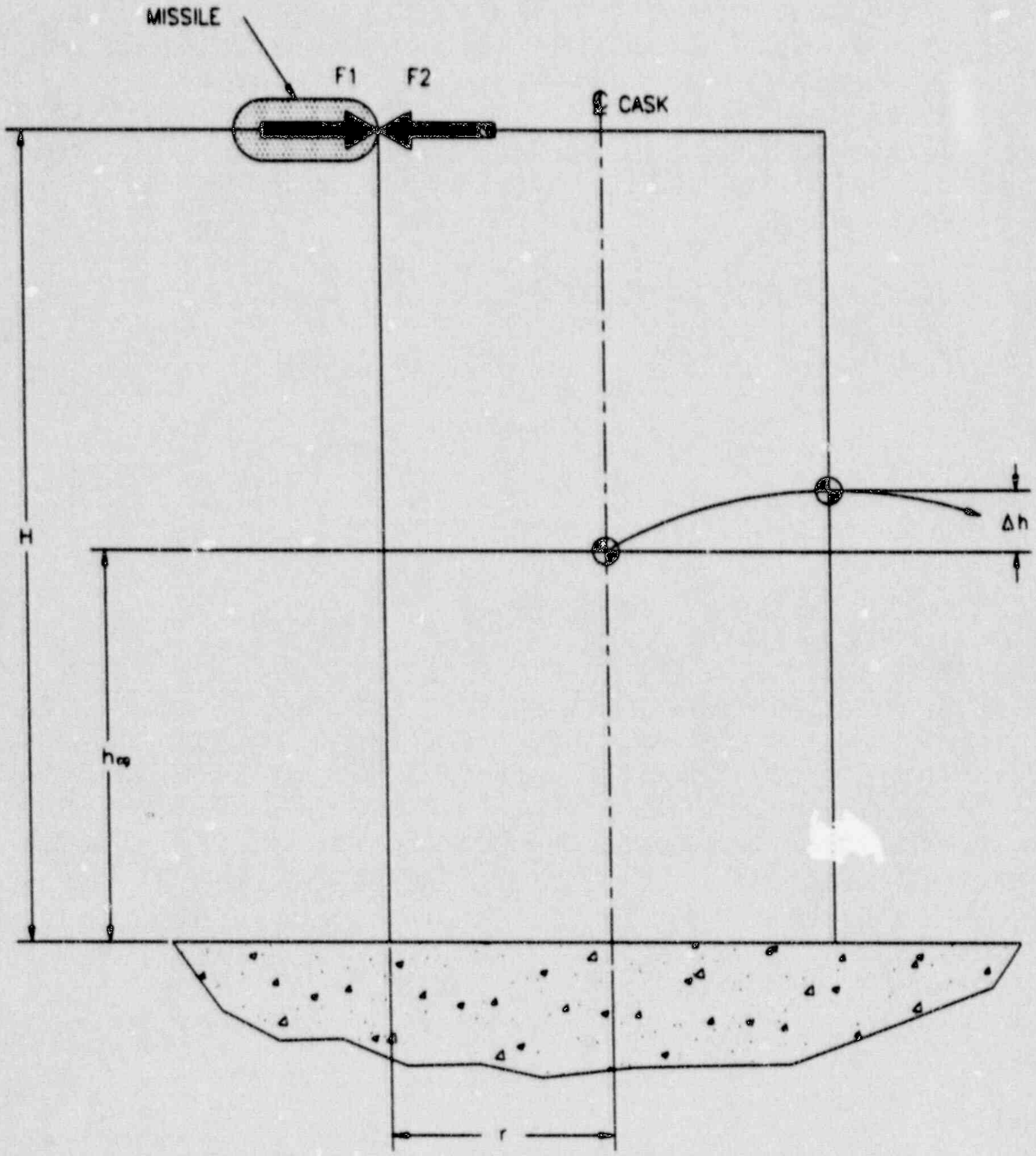


FIGURE 11.2-1

SKETCH OF MISSILE/CASK IMPACT GEOMETRY

ω = angular velocity of the cask after impact

H = height of the cask

r = cask radius

M_c = cask mass

Equating the two angular momentums, the cask angular velocity following the impact is calculated as 0.307 rad/sec. Therefore, the cask rotational energy after impact is calculated to be $7.75 \cdot 10^5$ lbs-in.

The energy required to overturn the cask is the energy required to bring its center of gravity over the corner. It can be calculated as:

$$E = W_{cask} \cdot \Delta h$$

where:

W_{cask} = cask weight

Δh = change in c.g. elevation

This energy is conservatively calculated to be $49.0 \cdot 10^5$ lbs-in. Since this value is substantially higher than the cask rotational energy after the impact, overturning of the cask will not occur as a result of tornado generated missiles. The above analyses are considered conservative in that direct in-line impact of the missile with the cask is assumed.

The force developed by the missile has been calculated using methodology presented in Reference 11.5. The maximum force is:

$$F = 0.625 \cdot vW = 0.625 \cdot 184.8 \cdot 3{,}960 = 457.4 \text{ kips}$$

The ability of storage cask concrete to withstand reactions and moments produced by this impact force has been evaluated. The most critical shear section is at the air outlets while the maximum bending moment exists at the wall section adjacent to the bottom. Capacities of these sections are conservatively calculated per the ACI-349 code. Shear evaluation is performed using the shear-friction methodology and the moment capacity is computed based on an uncracked section, i.e. no credit was taken for the cask reinforcement in bending. The results are reported and evaluated in combination with other loads in Table 3.4-8.

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.

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 2.5 degrees. Applying the total tornado wind load to the cask in this configuration results in the overturning moment of $4.2 \cdot 10^6$ in-lbs and the restoring moment of $17 \cdot 10^6$ in-lbs, hence, overturning of the cask under the combined effects of tornado winds plus tornado-generated missiles would not occur. Shear and bending in the cask walls are combined and evaluated in Table 3.4-8. Therefore, both stability of the cask and structural adequacy of reinforced concrete are shown to be acceptable.

Basket Under Tornado Loadings

Since the postulated tornado loadings are not capable of overturning or penetrating the cask they have no effect on the basket.

11.2.3.3 Accident Dose Calculations

As shown above, the worst case damage to the cask from a tornado event could be a 5.69" deep by 8" diameter penetration due to the direct missile impact. Because of the small area of the damage, there will be no noticeable increase in the off-site dosage.

To repair the cask, it is assumed that it takes two technicians 30 minutes to fill the damaged area with grout. Shielding calculations predict dose rates under 200 mrem/hr for all fuel burnup cases for a concrete cask with 5.69 inches of concrete removed. Thus, a localized dose rate of ≤ 200 mrem/hr is expected in the center of the damaged area. Therefore, a conservative estimate of the dose rate one meter from the damaged area is 100 mrem/hr and the total dose would be 100 man-mrem (50 mrem 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 can be found where a flood is not credible. Nevertheless, the analysis is presented here to bound a 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 storage cask. 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 buoyancy force on the cask, F_b, assuming full immersion of the cask was computed from the weight of the displaced water:

$$F_b = (p_w)(V) = 116{,}718 \text{ lbs}$$

where:

p_w = Weight density of water = 62.4 lb/ft^3 = 1.94 slugs/ft^3

V = Displaced volume of the cask

Assuming full immersion of the cask and steady state flow conditions for an infinite cylinder, the total drag force, F_d, is

$$F_d = (C_d)(p)(v^2)(A)/2$$

where:

C_d = Drag coefficient which depends on the Reynolds Number (Re)
= 0.8 for Re > 10^7 (which implies v > 16.56 ft/sec for water (Reference 11.6)

p = Mass density of water = 1.94 slugs/ft^3

u = Absolute viscosity of water = 0.0000273 lb-sec/ft^2

D = Cask outside diameter

v = Velocity of stream flow

A = Projected area of cask normal to flow

The stream velocity required to overturn the cask is then determined by summing the moments of the submerged weight of the cask and the drag force about a point on the bottom rim:

$$F_d \cdot \frac{H}{2} = (W_c - F_b) \cdot r\,,$$

where:

H = cask height

r = cask radius

W_c = cask weight

The above equation can be solved to determine the stream velocity required to overturn the cask. The solution results in v = 24.6 ft/sec (which is greater than 16.56 ft/sec and, hence, Re is greater than 10^7 and the use of $C_d = 0.8$ is justified).

Clearly any reasonable flood at a licensed ISFSI site should be bounded by this 24.7 ft/sec velocity and 20 foot depth. The stresses due to flood are negligible.

Basket Flood Loading

As shown above, a 20 foot deep flood is not capable of overturning the cask and producing substantial loads on the basket. A 20 foot flood would result in external pressure on the basket of approximately 9 psig. From review of Tables 3.4-5 and 3.4-6 which include stresses due to normal pressure of 4 psig, the basket will easily meet ASME Level C allowables if subjected to this 9 psig pressure.

11.2.4.3 Flood Dose Calculations

Flooding would not cause damage which 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 US site. The design basis seismic event is described and discussed in Section 2.2.5.

11.2.5.2 Earthquake Analysis

The storage cask is analyzed for the 0.75g earthquake which bounds all sites in the United States. Both cask stability and structural response are evaluated.

Cask Natural Frequency and Dynamic Amplification of Seismic Loads

The storage cask is a very stiff structure. Although free-standing, it has been analyzed for the lowest natural frequency as a cantilever fixed at the base (Reference 3.2).

The fundamental natural frequency of vibration for the cask was determined as shown below:

$$f_1 = \frac{3.52}{2\pi}\sqrt{\frac{E\,I\,g}{wH^4}}\ ,$$

where:

f_1 = Frequency of the n-th mode

E = Concrete modulus of elasticity

I = Moment of inertia for the cask section

g = Gravity acceleration

H = Height of the cask

w = Uniform weight density of cantilever

The lowest frequency is calculated to be 43.6 cycles per second. This is above the ZPA cut-off of 33 Hz in accordance with Reg. Guide 1.60, hence, the cask provides no dynamic amplification for the seismic loading. As a result, the cask is treated as a rigid body and equivalent static analysis methods are used to calculate loads and stresses.

<u>Evaluation of the Cask Stability</u>

The conventional analysis of free-standing objects is typically performed using simple static analysis, i.e., applying equivalent static loads to the cask in each of two orthogonal horizontal directions simultaneously with an upward vertical component and balancing these loads against the cask restoring moment. However, this approach is very conservative because it requires that all points of the structure always remain in contact with the foundation. In reality, the object may not overturn even if some of its sides lift off at certain times (the object may rock).

Therefore, a different methodology is employed for the TranStor™ storage cask. It is evaluated for overturning using the ANSYS finite element program and non-linear time history analysis. Non-linearity of the system results from the fact that ISFSI foundation is capable of producing a reaction in only one direction (upward). No hold-down restraint is available to prevent the cask from rocking. Non-linear gap elements are used in the finite element model to represent this condition. The most unfavorable cask geometry (highest c.g. location) is conservatively used.

Per Chapter 2.0, the design earthquake is defined as 0.75g in two horizontal directions and 0.5g in vertical direction. The time history is developed from the generic response spectra presented in Regulatory Guide 1.60. Vertical and horizontal components are taken to be statistically independent and a damping of 5% is conservatively used. Since the model is two-dimensional, i.e., only one horizontal and one vertical excitation is applied, the horizontal time history must represent the geometrical sum of two independent horizontal components. This is accomplished by using the 100-40 rule allowed by NUREG/CR-0098. According to this rule, due to statistical independence of two motions, the structure must be designed to take the combined effects of 100% in one particular direction and 40% of the effects corresponding to the other direction. As a result, the ZPA for the horizontal component used in the analysis is increased by the factor of $\sqrt{1^2 + 0.4^2} = 1.08$ producing the total of $0.75 \times 1.08 = 0.8$g. Results of the calculation demonstrate that the cask is stable under the loads of this design earthquake. In addition, the maximum ground displacement of 20.0 inches (substantially less than 44 inches of clear space between the casks) indicates that sliding would not cause the casks to impact each other.

Furthermore, as Figure 11.2-2 shows, a vertical ground displacement of approximately 5.6 feet would be required to move the center of gravity over the corner of the cask so that the cask would topple. This type of ground displacement and/or failure of the foundation is considered to be unrealistic and, hence, it is concluded that in addition to not toppling due to the kinetic energy of the earthquake, the cask will also not topple due to permanent failure and vertical movement of the foundation. Therefore, based on this analysis, it is concluded that the cask will not tip over or fall during a seismic event.

Nevertheless, the consequences of a storage cask overturning are conservatively bounded by the cask tipover analysis discussed in Section 11.2.10. The analysis shows that tipover results in deceleration that would not cause any critical damage to the storage cask or fuel basket. Hence, use of the certified Storage Cask will meet the geological and seismic criteria of 10 CFR 72 at any licensed power plant in the United States.

Since the cask is demonstrated not to tip over during an earthquake, the stresses in the basket can be evaluated by comparison to the off-normal handling analysis presented in Section 11.1.5. The 0.5g vertical and 0.8g horizontal accelerations (assuming 100-40 distribution for the two horizontal components) are well bounded by the 17.5g load during the off-normal handling event. Therefore, no additional evaluation of basket stresses is required.

11.2.5.3 Radiological Consequences

There are no radiological consequences for a seismic event. The casks are relatively undisturbed by the earthquake.

$$\theta = \arcsin \frac{h_{cg}}{h_{cg} + \Delta h} = 58°$$

$$\alpha = 90° - \theta$$

$$L = 2r(\sin\alpha) = 5.6'$$

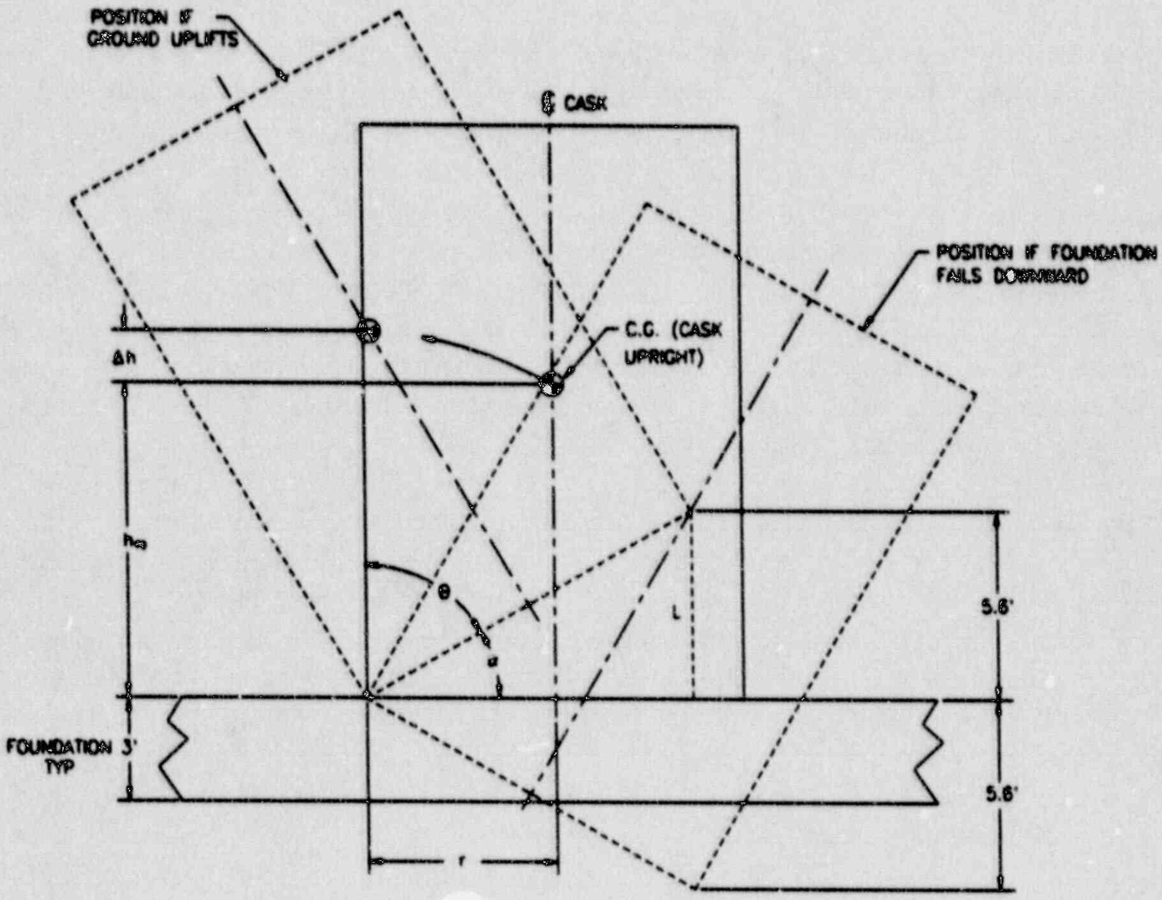


FIGURE 11.2-2

CASK TIP-OVER PARAMETERS

11.2.6 Accident Pressurization

11.2.6.1 Cause of Pressurization

The cause of this accident could be a hypothetical breach of all fuel rods in the basket and subsequent release of their fission and fill gases to the basket interior. This would pressurize the basket shell and lids.

11.2.6.2 Analysis of Pressurization Accident

The analysis of this accident entails calculation of the free volume in the basket as well as quantities of fill and fission gases in the fuel assemblies. The basket pressure is calculated by addition of the pressure due to these gases to the helium already present in the basket.

The fuel rods are assumed to be at a bounding fill pressure and 100% of the fill gas is assumed to be released into the basket. Bounding burnup and uranium loading was used to maximize the amount of fission gases and their quantity was conservatively estimated assuming that 30% of the total is released from the fuel rods (as stated in Section 11.2.1, the actual release fraction is about 8%). Using the storage temperature that corresponds to the highest heat load, the calculation results in a basket internal pressure of just under 50 psig for both PWR and BWR baskets. 50 psig is conservatively used in the analysis.

This hypothetical accident pressure loading is evaluated in tandem with the normally occurring dead weight and handling loading per Section III, Division 1, Subsection NC of the ASME Code. Thermal stresses do not need to be considered for the Service Level D events. The pressure boundary is analyzed using a two-dimensional finite element model of the shell, bottom plate, and lids. The resulting stresses (in combination with other loads) and corresponding acceptance criteria are summarized in Table 11.2-1. As shown in this table, all stresses are within the Code allowables for the Service Level D loadings.

11.2.6.3 Radiological Consequences

There are no radiological consequences for this accident.

11.2.7 Full Blockage of Air Inlets

11.2.7.1 Cause

This event is a postulated blockage of all airflow inlets. Because the storage cask has two independent air inlets located on two opposing sides and the casks are inspected daily, it is not considered feasible that all vents could become blocked by blowing debris, snow, animals, etc.

TABLE 11.2-1

PWR BASKET STRESSES RESULTING FROM HYPOTHETICAL ACCIDENT PRESSURIZATION

		Accident Pressure	Dead Weight	Normal Handling	Σ	ASME Level D Allowable
Basket	P_M	2.4	0.1	0.6	3.1	37.1
Shell	$P_L + P_b$	34.6	0.2	2.0	36.8	55.7
Bottom	P_M	1.1	0.0	0.6	1.7	37.1
Plate	$P_L + P_b$	23.2	0.2	1.3	24.7	55.7
Structural	P_M	0.1	0.0	0.2	0.3	37.1
Lid	$P_L + P_b$	6.9	0.1	0.9	7.9	55.7
Sleeve	P_M	0.0	0.1	0.2	0.3	43.9
Assembly	$P_L + P_b$	0.0	0.1	2.1	2.2	65.9
Shield Lid	P_M	0.0	0.1	0.1	0.2	27.8
Ring Weld	$P_L + P_b$	0.0	0.1	0.1	0.2	41.7
Top	P_M	4.4	0.1	0.8	5.3	37.1
Weld	$P_L + P_b$	6.1	0.2	0.9	7.2	55.7
Shield Lid	P_M	3.6	0.1	0.1	3.8	37.1
Weld	$P_L + P_b$	4.5	0.1	0.1	4.7	55.7

TABLE 11.2-2

BWR BASKET STRESSES RESULTING FROM HYPOTHETICAL ACCIDENT PRESSURIZATION

		Accident Pressure	Dead Weight	Normal Handling	Σ	ASME Level D Allowable
Basket	P_M	2.4	0.1	0.7	3.2	37.1
Shell	$P_L + P_b$	34.6	0.2	1.4	36.2	55.7
Bottom	P_M	1.1	0.0	0.4	1.5	37.1
Plate	$P_L + P_b$	23.2	0.2	1.0	24.4	55.7
Structural	P_M	0.1	0.0	0.2	0.3	37.1
Lid	$P_L + P_b$	6.9	0.1	0.6	7.6	55.7
Sleeve	P_M	0.0	0.1	0.2	0.3	43.9
Assembly	$P_L + P_b$	0.0	0.1	1.6	1.7	65.9
Shield Lid	P_M	0.0	0.1	0.1	0.2	27.8
Ring Weld	$P_L + P_b$	0.0	0.1	0.1	0.2	41.7
Top	P_M	4.4	0.1	0.8	5.3	37.1
Weld	$P_L + P_b$	6.1	0.2	0.9	7.2	55.7
Shield Lid	P_M	3.6	0.1	0.1	3.8	37.1
Weld	$P_L + P_b$	4.5	0.1	0.1	4.7	55.7

during normal operation. However, to demonstrate that acceptable temperatures would be maintained even if the unexpected complete inlet blockage occurred, 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 daily patrol.

11.2.7.3 Analysis of Event

The analysis of this event uses the airflow model described in Section 4.4.1.1. However, a different air flow path would be established because the heated air in the annulus would rise and the new cold air would have to be drawn in (a vacuum can not be formed since the annulus is open to the atmosphere). As a result, two air outlets (which are intentionally placed slightly lower than the other two) would become inlets while the other two would remain outlets. The path would be established in which air flows in through 2 outlets, down along a fraction of the annulus, up through the remaining fraction, then out through the other 2 outlets. To determine what fraction of the annulus would be taken by the up flow and what fraction by the down flow, the curve shown in Figure 11.2-3 is derived by balancing the flow loss associated with these paths against the upward driving force due to heating. Since the air flow is natural, the actual distribution would correspond to the path of least resistance which occurs witn 50% down flow and 50% up flow through the annulus. However, it is also evident from Figure 11.2-3 that even if the up flow and down flow areas in the annulus were not equal due to minor maldistribution of flow, the effect on outlet temperature is not significant.

The expected maximum outlet temperature for the 75 °F day with no solar load is calculated as 263 °F. The results of thermal analysis for the cask and basket in this condition are shown in Table 4.1-1. All temperatures are within the corresponding short-term limits. It must also be noted that this condition produces the worst case thermal gradient in the storage cask and was used for the cask thermal stress analysis presented in Chapter 3.0.

11.2.7.4 Accident Dose Calculations

The worst case radiological consequences for this event would be twice the consequences of the blockage of one half the inlets (Section 11.1.2). These are a small dose increment to the hands and arms and even smaller dose increment to the body caused by cleaning the vents. The fuel clad temperature will remain well within the short term limit and the blockage will be removed within, at most, one day (i.e., the surveillance interval).

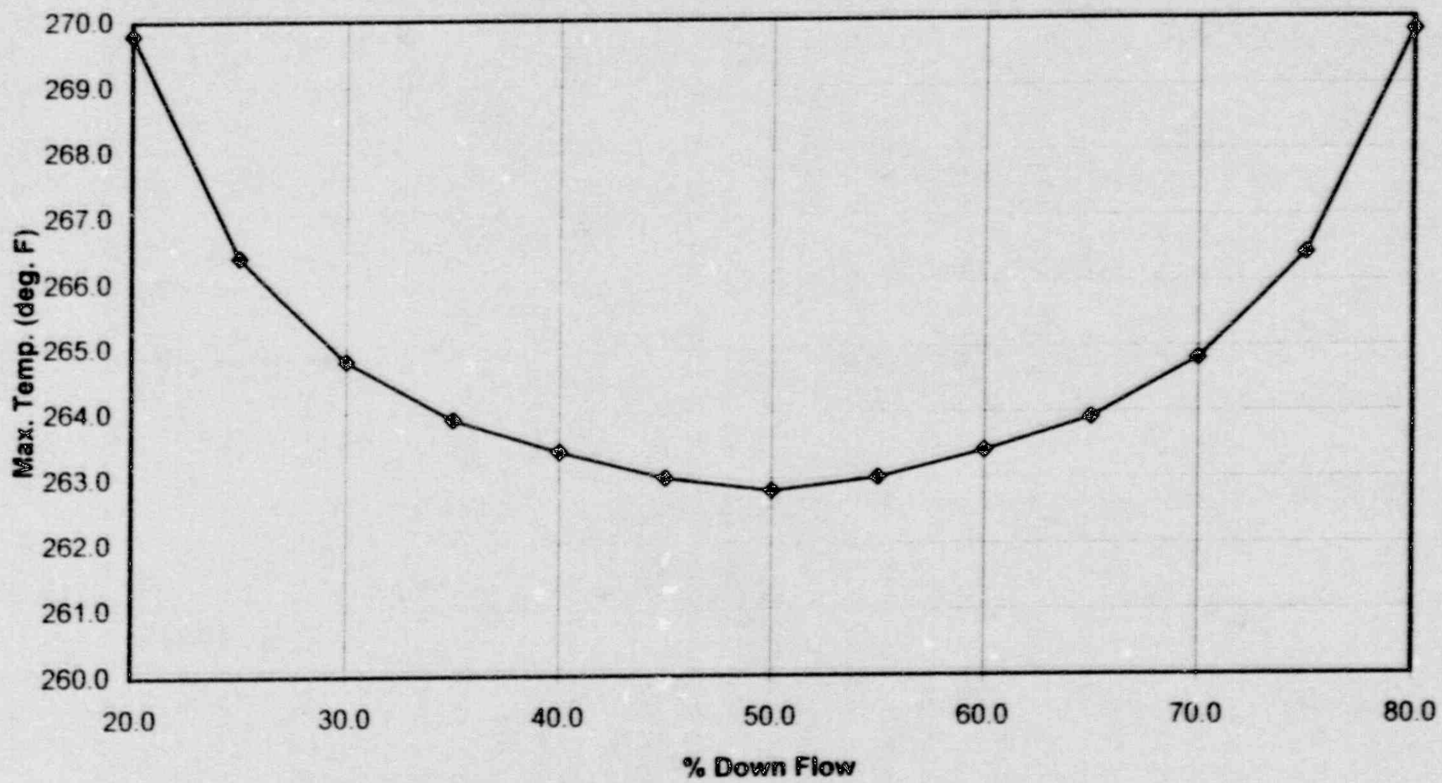


FIGURE 11.2-3

AIR FLOW DISTRIBUTION FOR ALL INLETS BLOCKED CONDITION

11.2.8 Site Explosions

11.2.8.1 Cause of Accident

This analysis addresses the hazards posed by potential explosions in the vicinity of the ISFSI site. No flammable or explosive substances are stored or used at the storage facility, therefore, an explosion affecting the site is extremely unlikely. However, the analysis is presented to bound any potential accident that might occur at the industrial facilities nearby. Verification of this shall be a part of each user's 10 CFR 72.212 evaluation.

11.2.8.2 Accident Analysis

The magnitude of explosion that would result in overturning or sliding of a storage cask was determined as described below.

The force required to slide a cask is:

$$F_{slide} = W_c \cdot \mu ,$$

where:

W_c = weight of cask

μ = coefficient of friction between the cask and storage pad

The force required to overturn a cask is found by equating its moment to the restoring moment of cask weight:

$$F_{tip} = \frac{W_c \cdot r}{(H/2)} ,$$

where

r = cask radius

H = cask height

The corresponding forces were conservatively calculated to be 85,500 lbs and 166,920 lbs, respectively. The force required to slide the cask is smaller and, therefore, controls. Using the ANSI 58.1 formulas (see Section 11.2.3), the minimum pressure on the cask to produce this force was determined to be 5.4 psi.

The calculated value of 5.4 psi is greater than the pressure that could be caused by explosions in the vicinity of an ISFSI site. An even greater pressure would be required to overturn a cask. In addition, as can be seen from the normal pressure stresses presented in Chapter 3.0, this loading is insignificant for the basket. Based on the foregoing, the integrity of the storage casks and baskets would not be adversely affected by postulated explosions near the ISFSI site.

11.2.8.3 Accident Dose Calculations

There are no radiological consequences from this accident, because the integrity of casks and baskets would not be affected.

11.2.9 Lightning

11.2.9.1 Cause of Accident

This event would be caused by meteorological conditions at the site. Lightning striking one of the storage casks is not a likely event, because the ISFSI pad is surrounded by the lighting towers that are provided at the facility.

11.2.9.2 Accident Analysis

Even if a storage cask were hit by lightning, the likely path to ground would be from the steel concrete cask lid to the steel base plate via the steel cask liner and the steel air inlet ducts. The fuel basket is surrounded by these steel structures and would not provide a likely ground path. Therefore, a lightning strike would not affect basket integrity. The absorbed heat would be insignificant due to the very short duration of the event. If the lightning entered or exited the cask via the concrete shell, some local spalling of concrete might occur. The extent of concrete damage would be similar to that calculated for the tornado missile analyzed in Section 11.2.3. Storage cask operation would not be adversely affected.

11.2.9.3 Accident Dose Calculations

Based on the evaluation above, the radiological consequences of this accident would be bounded by those discussed in Section 11.2.3 for a worst case tornado missile penetration.

11.2.10 Hypothetical Cask Tipover

11.2.10.1 Cause of Accident

As shown in other sections of this SAR, there is no credible event that can cause tipover of the TranStor™ Storage Cask. This is a non-mechanistic accident analyzed to demonstrate the large safety margins in the cask design.

11.2.10.2 Accident Analysis

The following aspects are considered important and evaluated for the cask tipover: crush depth of the concrete, integrity of the concrete cask, and integrity of the fuel basket.

The crush depth of the concrete was determined by conservatively postulating that all energy of the impact is absorbed by crushing of the cask concrete. The ISFSI pad is assumed to be absolutely rigid and, in addition, no energy dissipation or flexural deformation of the cask is considered. The crush depth is found by equating the work of crushing concrete to a kinetic energy of the falling cask. Since the motion is rotational and the velocity is linearly distributed along the cask height, linear distribution of the crush depth is also assumed. Therefore, the governing equation is:

$$\Delta E = \int_0^L \int_0^{(x/L)\delta_{max}} \sigma_u \cdot S(\Delta) d\Delta dx$$

where

ΔE - difference in potential energy between the cask vertical and horizontal positions

σ_u - crushing strength of cask concrete

$S(\Delta)$ - width of contact area as a function of crush depth Δ

L - cask height

$\Delta(x)$ - crush depth as a function of coordinate along the cask height

δ_{max} - maximum crush depth

Solving the equation for the maximum crush depth, the δ_{max} is calculated to be 1.93 inches. This number is conservative because in reality a significant part of the energy would be absorbed by crushing of the pad, taken by the flexural deformation of the pad and cask, and dissipated as heat.

To determine the cask and basket acceleration loads, the assumption is reversed: the cask is assumed to be rigid and the target absorbs the energy. Using the energy conservation, the equivalent horizontal drop height is calculated to be 60 inches. Based on a typical pad target hardness found per EPRI NP-7551 (Reference 11.7), acceleration resulting from this drop height is determined to be 11g. For additional conservatism, the maximum bounding value for any drop height is established as 14.4g. This is much less than the basket design acceleration of 44g, hence, the basket integrity would be maintained.

For the concrete cask, the cask dynamic properties are used to determine the demand for its wall capacity to resist the 14.4g acceleration load in both flexure and shear. These demands are combined with the loads due to other effects and compared to the ACI 349 allowables. The results are presented in Table 3.4-8 which shows that the cask is adequate to withstand the tipover loads.

11.2.10.3 Accident Dose Calculations

The cask and basket are capable of withstanding the tipover loads. Therefore, there would be no radiological release or adverse radiological consequences due to the hypothetical tipover event. The concrete crush depth of less than 2 inches would approximately double the dose rates in the localized area but would not significantly affect the overall dose rate. In accordance with Section 12.2.2.8, the cask and basket would be unloaded, inspected, and evaluated for future use.

11.3 References

11.1 D.R. Olander, Fundamental Aspects of Nuclear Reactor Fuel Elements, TID-26711-P1, Dept. Of Energy, Washington, D.C., 1976.

11.2 E. Elias, C. B. Johnson, "Radiological Impact of Clad and Containment Failures in At-Reactor Spent Fuel Storage Facilities," Electric Power Research Institute, Palo Alto, CA, 1982.

11.3 EPRI NP-440, "Full Scale Tornado Missle Impact Tests", 1977.

11.4 EPRI NP-1217, "Local Response of Reinforced Concrete to Missile Impact", 1974.

11.5 BC-TOP-9A, Revision 2, "Design of Structures for Missile Impact," Bechtel Power Corporation.

11.6 Sabersky, et.al., Fluid Flow.

11.7 EPRI NP-7751, "Structural Design of Concrete Storage Pads of Spent Fuel Casks", 1991.

12.0 OPERATING CONTROLS AND LIMITS

12.1 Proposed Operating Controls and Limits

The TranStor™ System is totally passive during the storage mode and requires very few operating controls. The general areas where controls and limits are necessary for safe operation of the TranStor™ Storage System are shown in Table 12.1-1. The conditions and other characteristics noted in the table were selected based on the safety assessments for both normal and accident conditions.

This SAR provides specifications for the fuel characteristics, basket drying, backfilling, sealing, and storage cask transfer and storage. In addition, site specific procedures in the areas of fuel and cask handling, trailer towing, security surveillance, administrative controls, training, and others may be utilized as necessary by the implementing site.

12.2 Development of Operating Conditions and Limits

12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

The controls and limits that apply to operating variables of the TranStor™ System that are observable and measurable are discussed in this section. Selection of such variables is directly related to the performance and integrity of equipment and confinement barriers.

The specifications contained in this section are:

Maximum Permissible Basket Leak Rate

Maximum Permissible Air Outlet Temperature

Maximum Basket Surface Contamination

TABLE 12.1-1

GENERAL AREAS WHERE CONTROLS AND LIMITS ARE NECESSARY

AREAS FOR OPERATING CONTROLS AND LIMITS	CONDITIONS OR OTHER ITEMS TO BE CONTROLLED
1. Fuel Characteristics	Fissile Content Decay Power Weight
2. Cask Handling	Per existing site NUREG 0612 Heavy Loads Program
3. TranStor™ Basket	
3.1 Fuel Loading	Per existing site procedures
3.2 Drying	Vacuum Pressure
3.3 Backfilling	He Pressure and Content
3.4 Sealing	He Leak Rate
4. TranStor™ Basket in TranStor™ Storage Cask	Air Outlet Temperature
5. Surveillance	Inspection of Air Inlets and Outlets
6. Administrative Controls	Per existing site procedures
7. Training	Per existing site procedures

12.2.1.1 Maximum Permissible Basket Leak Rate

Specification: $\leq 1.0 \times 10^{-4}$ standard cubic centimeters per second (scc/sec) at the minimum pressure of 7.3 psig.

Applicability: This specification is applicable to all TranStor™ baskets after closure.

Objective:

1. To limit the quantity of radioactive gases released by each cask to negligible levels. Should fission gases escape the fuel cladding, they will remain confined by the basket confinement boundary.

2. To retain helium cover gases within the basket and prevent oxygen from entering the basket. The helium improves heat dissipation characteristics of the basket and prevents any oxidation of the fuel cladding.

Action: The basket structural lid weld shall be tested after it has been completed. The basket shall be pressurized with helium to 7.3 psig and a calibrated hand-held sniffer used (per manufacturer's instructions) to determine a leak rate. 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.

Surveillance: If the rate is within the limit, additional 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 basket.

Basis: If the basket were to leak at the largest allowable rate of 10^{-4} scc/sec, the amount of helium that could escape over a 20-year span can be calculated as:

$$N = \frac{pV}{RT} = \frac{1\,atm \cdot (10^{-4}\,\frac{cm^3}{\sec})(10^{-6}\,\frac{m^3}{cm^3})}{8.2 \cdot 10^{-2}\,\frac{m^3 \cdot atm}{kmol \cdot {}^\circ K} \cdot 293^\circ K} \times 3600\,\frac{\sec}{hr} \times 8760\,\frac{hr}{yr} \times 20\,yrs = 2.6\,moles$$

This is approximately 2% of the total helium in the basket. However, the 10^{-4} scc/sec rate would exist at a 7.3 psig pressure while the basket is essentially at 0 psig during storage (the actual pressure is even negative as shown in Tables 3.4-3 and 3.4-4). Since the leak rate is proportional to pressure, the amount of helium escaping the basket would be negligible.

12.2.1.2 Maximum Permissible Air Outlet Temperature

Specification: The equilibrium air temperature at the outlet of a fully loaded basket (26 kW) shall not exceed ambient by more than 120 °F.

Applicability: This temperature limit applies to all TranStor™ Storage Casks stored in the ISFSI.

Objective: The objective of this limit is to ensure that temperature of the fuel cladding and the TranStor™ cask concrete do not exceed their corresponding limits established in Chapter 4.0 of this SAR. An additional objective of the temperature measurements is to provide base-line performance data.

Action: If the air temperature rise is greater than the value specified above, the first action is to check all inlet and outlet ducts for airflow blockage. If environmental factors are ruled out as the cause of the excessive air temperatures, this condition may indicate that the fuel assemblies may be producing heat at a rate higher than specified in Section 2.0 of this SAR. Therefore, the fuel loading records must be checked to assure that only assemblies meeting the fuel specification in Section 12.2.2.1 have been loaded into the cask. If the correct fuel loading is verified, then this condition is not addressed in the SAR and will require additional temperature measurements and/or analysis to justify acceptability of the actual cask performance. Should it be determined that the cask was loaded with fuel not meeting the specification in Section 12.2.2.1 or if acceptability of the cask temperatures can not be verified, it shall be unloaded and a letter report shall be submitted to NRC within 30 days.

Surveillance: The ambient temperature and cask outlet temperatures for the TranStor™ Storage Casks shall be measured and recorded upon placement in service at intervals not to exceed 30 hrs until the cask has reached thermal equilibrium.

Basis: If the air temperature rise was 120 °F (15 °F higher than the 105 °F rise calculated for the 75 °F ambient case, see Table 4.1-1), the maximum concrete and fuel clad temperatures would be less than 15 °F higher than predicted (due to the non-linearity of heat transfer). For a cask design load of 26 kW, this condition would result in a bounding concrete temperature of 204 + 15 = 219 °F and a bounding cladding temperature of 609 + 15 = 624 °F (PWR) or 676 + 15 = 691 °F (BWR). Both of these values are well below the corresponding acceptance criteria. From review of Table 4.1-1, all other cases result in higher margins, therefore, the normal condition is controlling.

12.2.1.3 Maximum Basket Removable Surface Contamination

Limit/
Specification: $10^{-3}\mu Ci/cm^2$ gamma-beta and
$10^{-4}\mu Ci/cm^2$ alpha

Applicability: External surface of each TranStor™ basket.

Objective: Keep removable surface contamination level low enough so that off-site doses will be negligible even in the event that contamination became loose and behaved as a particulate or gaseous release.

Action: If the limit is exceeded, the basket exterior shall be washed by flushing the basket-transfer cask gap with water (or other suitable decontamination solution) and additional contamination surveys taken until the limit is met.

Surveillance: Prior to basket transfer from the transfer to storage cask, contamination surveys shall be taken on the basket exterior within six inches of the top of the basket. Contamination surveys shall be taken on the transfer cask interior and bottom exterior surfaces after the basket has been transferred to the storage cask. The contamination surveys for removable surface contamination shall be conducted after the loaded basket is removed from the pool and before the storage cask release for movement to the storage pad.

Basis: As shown in Section 11.1.4, even if the basket were covered over its entire surface with 1.6×10^5 dpm/cm^2 (0.07 $\mu Ci/cm^2$) of ^{60}Co and all this contamination became released as a gaseous particulate cloud, a person standing at a distance of 100 meters would receive the Committed Effective Dose Equivalent of less than 50 mrem. Therefore, the dose due to contamination of $10^{-3}\mu Ci/cm^2$ would be less than 0.7 mrem even if all of the activity were to leave the basket and be inhaled. This dose is negligible, thus, the limit of 10^{-3} $\mu Ci/cm^2$ is conservative and ensures that the dose limits of 10 CFR Parts 20 and 72 would be met. The alpha limit equal to one-tenth of the gamma limit is consistent with the requirements of 10 CFR 71 and assures that alpha contribution to the dose is insignificant.

12.2.2 Limiting Conditions for Operation

The following specifications are contained in this section:

Fuel Specification

Basket Vacuum Pressure During Drying

Basket Helium Backfill Pressure

Test of Shield and Structural Lid Welds

Minimum Temperature for Moving the Carbon Steel Basket

Minimum Temperature for Lifting the Transfer Cask

Maximum Lifting Height for the Storage Cask

12.2.2.1 Fuel Specification

Specification: The characteristics of the spent fuel allowed to be stored in the TranStor™ Storage System are restricted to those included in Table 12.2-1 and other referenced tables and figures.

Applicability: The specification is applicable to all fuel and waste to be stored in the TranStor™ System.

Objective: The specification is prepared to ensure that the peak fuel rod temperatures, maximum surface doses, and effective neutron multiplication factor are below their design values. Furthermore, the fuel weights ensure that structural conditions in the SAR bound those of the actual fuel being stored.

Action: The parameters listed in Tables 12.2-1 are independently verified and documented for each fuel assembly to be loaded into a basket. Fuel not meeting this specification shall not be stored in the TranStor™ System.

Surveillance: Immediately before insertion into a basket, the identity of each fuel assembly shall be independently verified and documented.

Basis: The specification is based on consideration of the design basis parameters included in this SAR. 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 principal design parameters of the fuel to be stored are found in Section 2.1 of the SAR.

The physical parameters that define the mechanical and structural design of the TranStor™ System are the fuel assembly dimensions and weight provided in SAR Table 2.1-1. They represent the heaviest fuel, so the calculated stresses bound the fuel designs to be stored.

The criticality design criteria ensure that k_{eff} remain subcritical under normal, off-normal, and accident conditions. Misloading of unirradiated fuel and optimum moderation are also considered in the analyses.

TABLE 12.2-1

CHARACTERISTICS OF SPENT FUEL TO BE STORED IN THE TranStor™ SYSTEM

Fuel	Intact, failed, and partial assemblies, fuel debris [1]	
Fuel Cladding	Zircaloy or stainless clad fuel	
Type	Any assembly type except GE-XBR	
	PWR	BWR
Maximum Initial Enrichment	Tables 12.2-2 and 12.2-3	Tables 12.2-4 and 12.2-5
Decay Power Per Assembly[2]	$\leq$ 1.083 kW	$\leq$ 0.426 kW
Assembly Weight (with or without channels or control components)	$\leq$ 1680 lbs (763 kg)	$\leq$ 700 lbs (318 kg)
Assemblies per Basket	24 max	61 max

(1) Failed fuel and fuel debris shall be confined in an overpack container within the basket.
(2) Power level is determined from DOE/RW-0184-R1 or equivalent database based on characteristics of a specific fuel assembly. Assemblies with decay power equal to or below the specified values also meet the radiological source strength criteria.

TABLE 12.2-2

24-ASSEMBLY PWR BASKET

Assembly	Enrichment	k_{eff}
All Assemblies Not Listed Below	4.1	*
Westinghouse 14x14	4.5	0.94726
Westinghouse 15x15	4.1	0.94807
Westinghouse 17x17 std	4.1	0.94356
Westinghouse 17x17 OFA	4.1	0.94736
Westinghouse 17x18	5.0	0.85721
B&W 15x15	4.7	0.94631
CE 14x14	5.0	0.94783
CE 15x15	4.7	0.94644
CE 16x16	4.2	0.94892
EXXON/ANF 14x14	4.8	0.94716
EXXON/ANF 15x15	4.3	0.94717
EXXON/ANF 17x17	4.1	0.94839
EXXON/ANF 15x16	5.0	0.88218

* Analysis performed on the bounding assembly Westinghouse 15x15; the U^{235} enrichment limit applies to all PWR assembly types not specifically listed above

TABLE 12.2-3

20-ASSEMBLY PWR BASKET

Assembly	Enrichment	k_{eff}
All Fuel Assemblies Not Listed Below	4.4	*
Westinghouse 15x15	4.4	0.94559
Westinghouse 17x17	4.6	0.94929
Westinghouse 17x17 OFA	4.5	0.94940
B&W 15x15	4.6	0.94647
EXXON/ANF 15x15	4.7	0.94929
EXXON/ANF 17x17	4.4	0.94397

* Analysis performed on the bounding assembly Westinghouse 15x15; the U^{235} enrichment limit applies to all PWR assembly types not specifically listed above.

Note: The 20 assembly basket is identical to the 24 assembly basket, except for being under loaded

TABLE 12.2-4

61-ASSEMBLY BWR BASKET

Assembly	Enrichment	k_{eff}
All Assemblies Not Listed Below	3.6	*
General Electric 6x6	5.0	0.88256
General Electric. 7x7(a)	3.6	0.94442
General Electric 7x7(b)	3.6	0.94201
General Electric. 7x7 R	3.7	0.94642
General Electric 8x8	3.7	0.94481
General Electric 8x8 R	3.6	0.94919
General Electric 9x9	3.6	0.94960
GE 10x10 AC	3.7	0.94520

* Analysis performed on the bounding assembly GE 8x8 R; the U^{235} enrichment limit applies to all BWR assembly types not specifically listed above.

TABLE 12.2-5

60-ASSEMBLY BWR BASKET

Assembly	Enrichment	k_{eff}
All Assemblies Not Listed Below	3.9	*
General Electric. 7x7(a)	3.9	0.94724
General Electric 7x7(b)	4.0	0.94373
General Electric. 7x7 R	4.0	0.94545
General Electric 8x8	4.0	0.94724
General Electric 8x8 R	3.9	0.94862
General Electric 9x9	3.9	0.94879
GE 10x10 AC	4.0	0.94680

* Analysis performed on the bounding assembly GE 8x8 R; the U^{235} enrichment limit applies to all BWR assembly types not specifically listed above.

Note: The 60 assembly basket is identical to the 61 assembly basket, except for being under loaded.

The design basis for nuclear criticality safety is based on the most reactive fuel assemblies or the types specifically listed.

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 are established based on the methodology in Reference 4.7 and the results for some assemblies are presented in the Appendix 1 of this SAR. Based on this methodology, a maximum heat generation rate of 1.083 kW per PWR assembly and 0.462 kW per BWR assembly is a bounding value for the fuel to be stored. The decay power shall be determined for each specific assembly as a function of burnup, cooling time, uranium loading, and initial enrichment. DOE/RW-0184-R1 or equivalent database is to be used for these calculations.

The radiological design criterion is that the total dose rate does not exceed the cask design limits. As discussed in Chapter 5.0, any fuel that meets the established power limit also meets this dose requirement and no additional specification is necessary.

12.2.2.2 Basket Vacuum Pressure During Drying

Specification: Vacuum Pressure: $\leq$ 3 mm Hg
Time at Pressure: $\geq$ 30 min.
Number of Pump-Downs: 2 minimum

Applicability: This specification is applicable to all baskets during the vacuum drying process.

Objective: To ensure removal of moisture from the basket.

Action: Once the required vacuum pressure is obtained, perform helium backfill to atmospheric pressure, then 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;
3. Check and repair the weld between the basket structural lid and the outer shell and the fillet weld between the structural lid and the shield lid.

Surveillance: Surveillance of the vacuum gauge is required during the vacuum drying operation.

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 basket cavity before the start of vacuum drying was steam at atmospheric pressure, a partial steam pressure after two pump-downs will not exceed $760 \times (3/760)^2 = 0.01$ mm Hg. With the average temperature of 450 °F, the moisture content of the basket cavity at this pressure is approximately 0.002 moles (assuming a perfect gas), hence, only 0.001 moles of O_2 are available (if 100% radiolysis is assumed). This O_2 could react with 0.003 moles of UO_2 (0.8 gram). However, the amount of 0.8 gram of UO_2 is negligible compared to the 2225 grams of UO_2 in a single fuel rod and its oxidation does not represent a threat to the safe operation of the TranStor™ System.

12.2.2.3 Basket Helium Backfill Pressure

Specification: Helium 14.5 psia ± 0.5 psia backfill pressure (stable for 30 minutes after filling)

Applicability: This specification is applicable to all baskets before installation of valve covers.

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;
(3) the basket 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 the helium source;
4. Check and repair the welds on basket structural lid.

If pressure exceeds the criterion, release a sufficient quantity of helium to lower the cavity pressure.

Surveillance: Surveillance of the pressure gauge is required during the helium backfilling operation.

Basis: The value of 14.5 psia was selected to assure that the basket is at slight sub-atmospheric pressure during all storage conditions.

12.2.2.4 Test of Basket Shield and Structural Lid Seal Welds

Specification: The basket closure welds listed below shall be dye penetrant or magnetic particle tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Article NC-5000. Test acceptance standards shall be those described in subarticles NC-5350 and NC-5340 respectively.

Applicability: This specification is applicable to the first and final pass of the following welds for all baskets:

1. Weld between the shield lid and the shell.
2. Weld between the structural lid and the shield lid at the valve port.
3. Weld between the structural lid and the shell.
4. Two welds between the valve cover plates and the structural lid.

Objective: To ensure that the basket is adequately sealed and leak-tight.

Action: If the nondestructive examination indicates that the weld is unacceptable:

1. The weld shall be ground down and repaired;
2. The new weld shall be re-examined in accordance with this specification.

Surveillance: The PT or MT inspection is to be performed during basket closure operations.

Basis: Article NC-5000 Examination, ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NC.

The safety analysis is based on a weld being structurally adequate and leak-tight to 10^{-4} scc/sec. This check is done to ensure compliance with this design criterion.

12.2.2.5 Minimum Temperature for Moving the Carbon Steel Basket

Specification: Movement of the carbon steel basket inside the storage cask is allowed only when the minimum basket temperature is -10°F or above.

Objective: To avoid the potential for brittle failure.

Action: Confirm that the minimum carbon steel basket shell temperature is above -10 °F immediately before moving the basket inside the storage cask.

Surveillance: The ambient temperature shall be measured before movement of the carbon steel basket. If it is above -10 °F, no further measurement is required and the movement can proceed as planned. If the ambient temperature is below -10 °F, measurement of the basket temperature can be made by removing the storage cask cover plate and measuring the temperature at the center of the basket lid (as shown in Figure 11.1-2 this is the minimum temperature).

Basis: The specification is developed based on recommendations of NUREG/CR-1815. The materials of the carbon steel basket shell, bottom plate, shield and structural lids, and port cover plates are required to show Charpy test values of at least 15 ft-lb at -50 °F. In accordance with the referenced NUREG, this assures that the test temperature is at least 10 °F above the Nil Ductility Transition (NDT) temperature of the material, i.e. the NDT temperature would have to be -60 °F. Furthermore, using the basket stresses for an off-normal handling event (Table 11.1-1) and the NUREG 1815 methodology for Category I components, the Lowest Service Temperature (LST) for the basket is required to be 50 °F above the NDT. This results in the -10 °F requirement. Movement of the carbon steel basket at temperatures above -10 °F eliminates the potential for brittle fracture.

This specification is not applicable to baskets with stainless steel shells since, per ASME Code, stainless steel is not prone to brittle failure.

12.2.2.6 Minimum Temperature for Lifting the TranStor™ Transfer Cask

Specification: The transfer cask shall be used to move the loaded basket only if the transfer cask shell temperature is 0 °F or above.

Objective: To avoid the potential for brittle failure.

Action: Confirm that the minimum transfer cask shell temperature is above 0 °F immediately before moving the cask containing the loaded basket.

Surveillance: If the ambient temperature around the transfer cask is above 0 °F, no further measurement is required and the movement can proceed. If it is below 0 °F, confirm that the transfer cask shell temperature is above 0 °F before moving the cask containing the basket. This is done by measuring the temperature of the end of the trunnion since it corresponds to the lowest shell temperature.

Basis: The specification is developed based on recommendations of NUREG/CR-1815 and ANSI N14.6. The materials of the transfer cask shells, trunnions, and lifting yoke require Charpy testing that assures the NDT temperature of not higher than -40 °F. Per ANSI N14.6, the Lowest Service Temperature (LST) for the special lifting devices is required to be 40 °F above the NDT point. This results in the 0 °F requirement. Movement of the transfer cask at temperatures above 0 °F eliminates the potential for brittle fracture.

12.2.2.7 Storage Cask Handling Height

Specification: The TranStor™ Storage Cask shall not be lifted higher than 80 inches unless a site-specific analysis is performed to justify a higher limit.

Applicability: This specification is applicable to all storage casks containing a loaded basket.

Objective: To preclude a drop of the loaded cask from a height of greater than 80 inches since such drop may result in unacceptable damage to the basket, cask, and fuel.

Action: If the limit is exceeded without a drop, no action is required other than bringing the height within the specification. Should a cask be dropped from a height that exceeds 18 inches, it shall be unloaded, inspected, and evaluated for future use. An exterior inspection is required for drops lower than 18 inches.

Surveillance: No control is required when air pads are used to lift the cask because the air pad inflated height is only 3 inches. Administrative control of the lifting height is required when the cask is lifted from above using a crane or cask transporter.

Basis: The NRC evaluation of the cask drop analysis for VSC-24 concurred that drops between 18 and 80 inches can be sustained with acceptable damage (i.e., without breaching the containment boundary, preventing removal of fuel assemblies, causing a criticality accident, or causing a structural failure of the concrete cask so it can not maintain its shielding function). Based on the engineering judgment, drops from heights up to 18 inches are not considered to be a concern.

Since the TranStor™ cask and basket are similar to the VSC-24 and designed for the same drop loads, no additional analysis is required.

12.2.3 Surveillance Requirements

12.2.3.1 Normal Operation Surveillance

Analysis has shown that the storage cask can provide its safety functions during all normal and accident transients as described in Section 11.0. Therefore, the only necessary surveillance other than that required during loading and handling (described in other specifications) is the periodic checking for security and safeguards.

The requirements contained in this section are:

Visual Inspection of Air Inlets and Outlets

Exterior Storage Cask Surface Inspection

12.2.3.1.1 Visual Inspection of Air Inlets and Outlets

Applicability: Applicable to all TranStor™ casks placed in service.

Objective: To assure adequate convective cooling of the basket inside the storage cask.

Surveillance: A visual surveillance of the wire mesh screens covering the air inlets and outlets shall be conducted daily (intervals are not to exceed 32 hours).

Action: If the surveillance shows signs of degradation or breach of the inlet or outlet screens, a close-up inspection of the inlet shall be conducted to determine need for screen replacement. Any sources of blockage, such as debris or insect infestation, shall be removed within 16 hours.

Basis: If the cask is loaded with the full design heat load of 26 kW, complete blockage of all vents would cause the concrete temperature to exceed 350 °F in approximately 48 hours. Concrete temperatures over the short-term limit of 350°F are undesirable as they may have an adverse impact on strength and durability.

12.2.3.1.2 TranStor™ Storage Cask Exterior Surface Inspection

Applicability: Applicable to all TranStor™ casks placed in service.

Objective: To assure adequate shielding and reinforcement protection.

Surveillance: The storage cask 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 in accordance with the grout manufacturer's recommendations.

Basis: This action maintains the surface condition of the concrete exterior, prevents any adverse impact on shielding performance and ensures rebar corrosion protection.

12.2.3.2 Surveillance After an Accident

The ISFSI facility and TranStor™ casks shall be inspected within 24 hours after a tornado, earthquake (larger than 5.0 on the Richter scale) or other natural disaster. The storage cask shall be unloaded and taken out of service if complete air vent blockage (all inlets or outlets) occurs for two days or more.

The storage cask and basket shall also be inspected if they experience a drop from a height of more than 18 inches or a tipover.

12.2.4 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 TranStor™ System are as follows:

- Concrete Wall Thickness and Density
- Concrete Strength
- Reinforcing Steel Quantity and Placement
- Basket Shell and Internal Structure Material Thicknesses and Strength
- Basket Shield Lid Material Density and Thickness
- Basket Boundary Welds
- Transfer Cask Shells, Doors, Rails, and Trunnions Thickness, Density, and Strength
- Transfer Cask Lead and Neutron Shield Thickness and Density

Each of these design features contributes significantly to the ability of the TranStor™ System to meet the requirements of 10 CFR 72 for at least fifty years. Each parameter is controlled by the fabrication specifications and drawings for the basket, storage and transfer casks. The design of these features may not be altered without affecting the safety and durability of the components. Verification inspections are performed during TranStor™ System fabrication.

12.2.5 Administrative Controls

Controls used as part of the TranStor™ System design and fabrication are provided in the Quality Assurance Manual and Procedures. These are discussed in Section 13.0. Site specific controls for the organization, administrative system, procedures, recordkeeping, review, audit and reporting necessary to ensure that the TranStor™ System installation is operated in a safe manner are the responsibility of the system user.

13.0 QUALITY ASSURANCE

Sierra Nuclear Corporation will apply its Quality Assurance Program as approved by tne Nucleai Regulatory Commission, Docket Number 71-0804, which expires June 30, 2000. The SNC's Quality Assurance Manual is presented in Appendix 3.

APPENDIX 1

FUEL INERT DRY STORAGE TEMPERATURE LIMITS

A1.1 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 4.7). 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 4.7 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.

As documented in Reference 4.14, the zircaloy temperature limits bound the limits for stainless steel clad fuel.

A1.2 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)
- GE BWR (8x8)
- GE BWR (6x6)
- GE BWR (7x7)

From Reference 4.7,

$$\sigma_{hoop} = [(p)(D_{mid})]/2t$$

where,

σ_{hoop} = cladding hoop stress

p = internal gas pressure of rod (fission gas and fill)

D_{mid} = clad midwall diameter

t = clad thickness

The gas pressure inside fuel cladding can be found using perfect gas laws based on the gas quantity, free volume and temperature. Free volume and quantity of fill gas depend on assembly type; fission gas quantity is a function of both assembly type and burnup. Therefore, the relationship between stress and temperature is plotted for the various fuel assemblies and burnups in Figures A1-1 through A1-8. The generic temperature limit curves for 5, 6, 7, 10 and 15 year cooling period from Reference 4.7 are plotted on the same axis. The intersection of the stress temperature relationship line for a given fuel assembly with the corresponding age line defines the maximum allowable initial dry storage temperature for fuel of this type, burnup, and cooling time. The same methodology can be used for any combination of these three variables not covered herein.

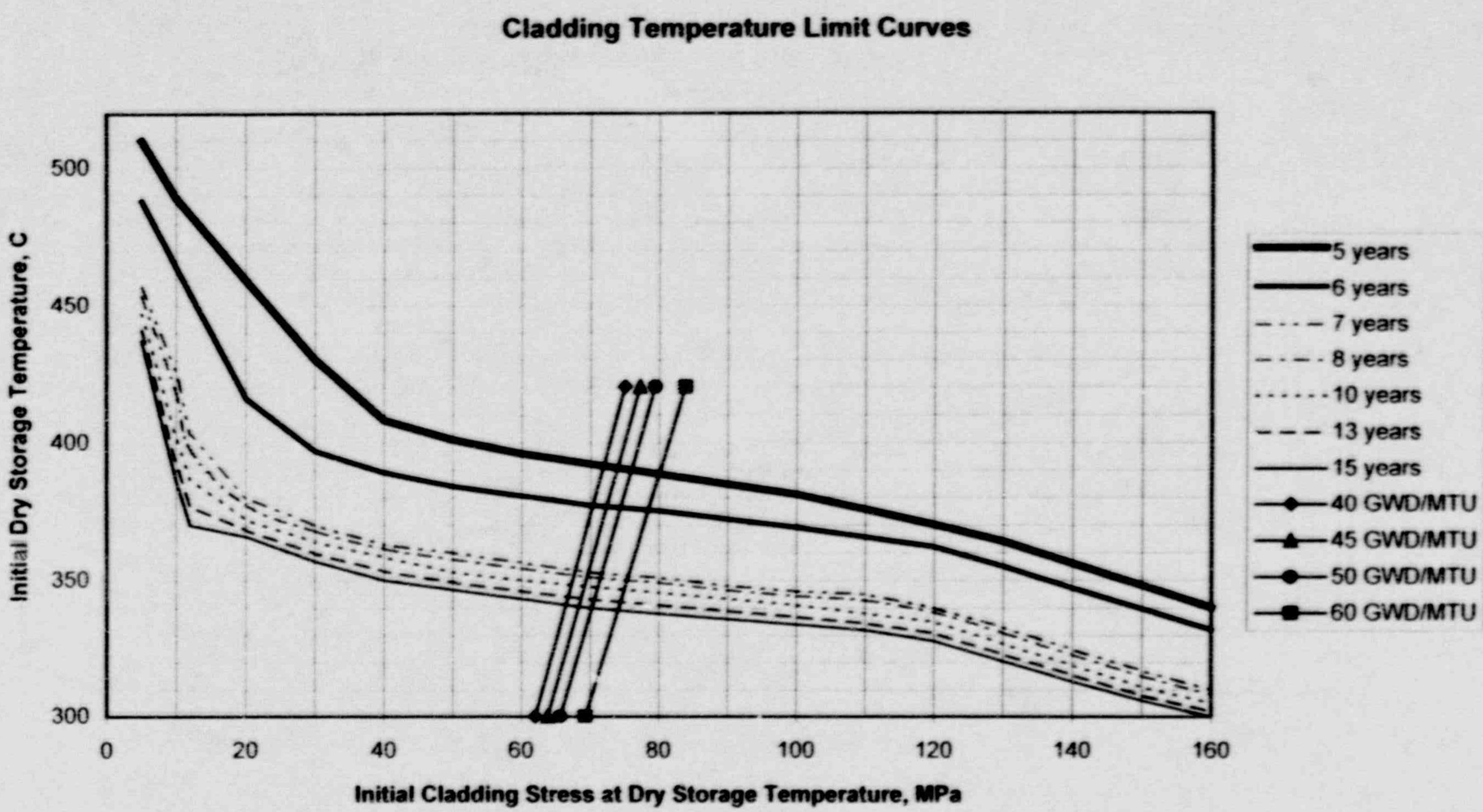


FIGURE A1-1
GRAPHS FOR DETERMINATION OF ALLOWABLE TEMPERATURES FOR B&W MARK C (17x17) ASSEMBLIES

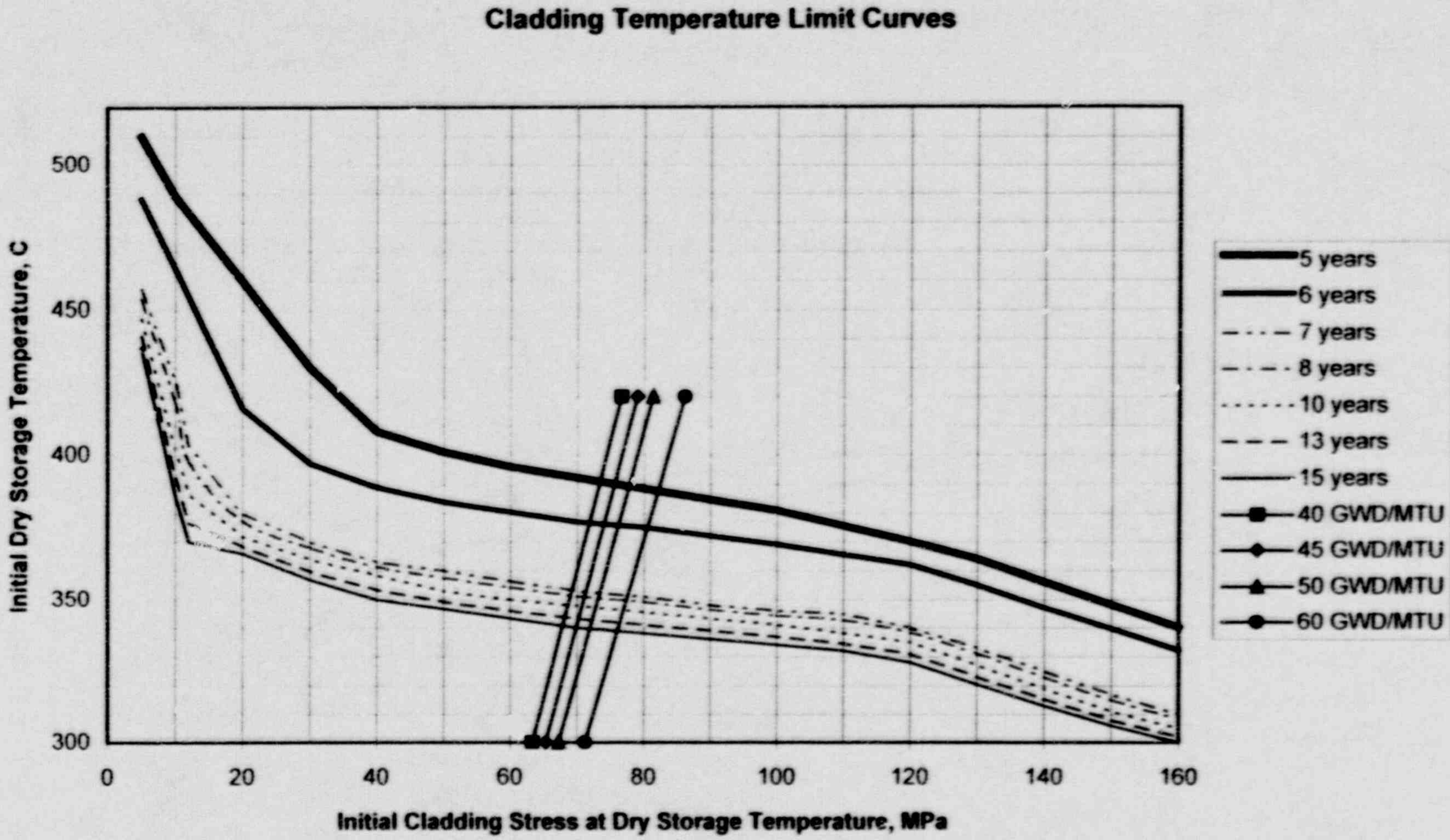


FIGURE A1-2
GRAPHS FOR DETERMINATION OF ALLOWABLE TEMPERATURES FOR B&W MARK B-4 (15x15) ASSEMBLIES

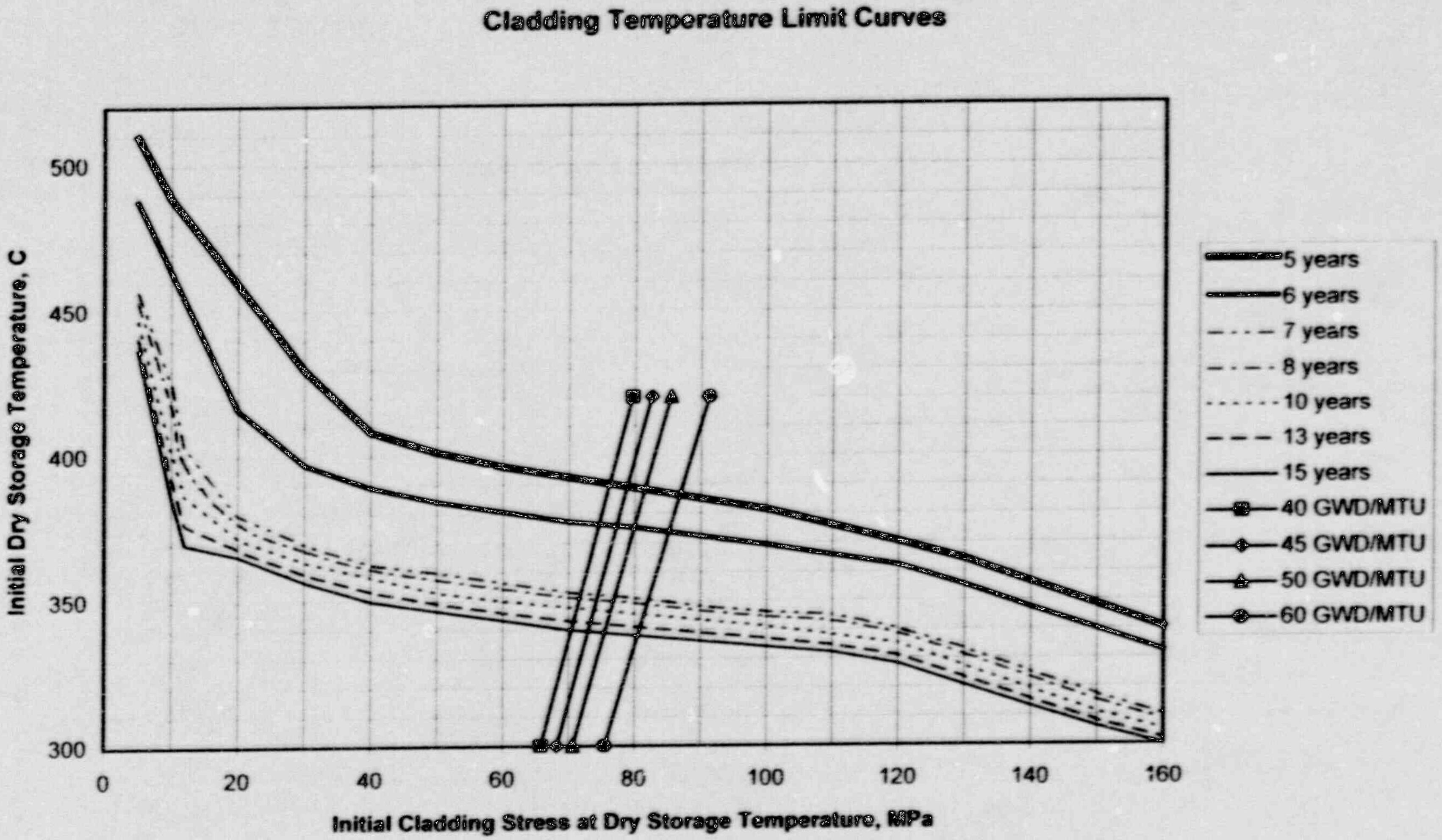


FIGURE A1-3
GRAPHS FOR DETERMINATION OF ALLOWABLE TEMPERATURES FOR CE 15x15 ASSEMBLIES

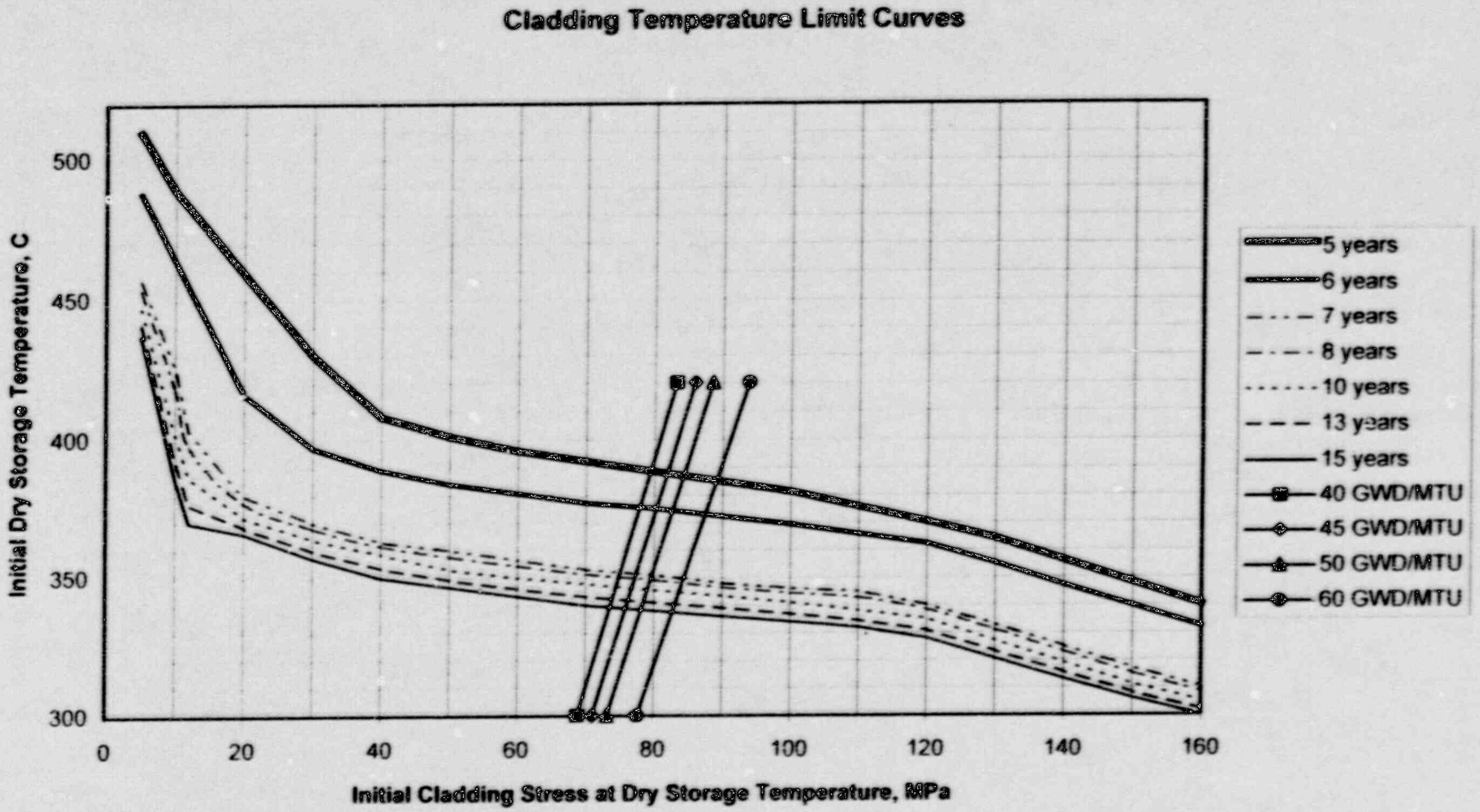


FIGURE A1-4
GRAPHS FOR DETERMINATION OF ALLOWABLE TEMPERATURES FOR WESTINGHOUSE 17x17 ASSEMBLIES

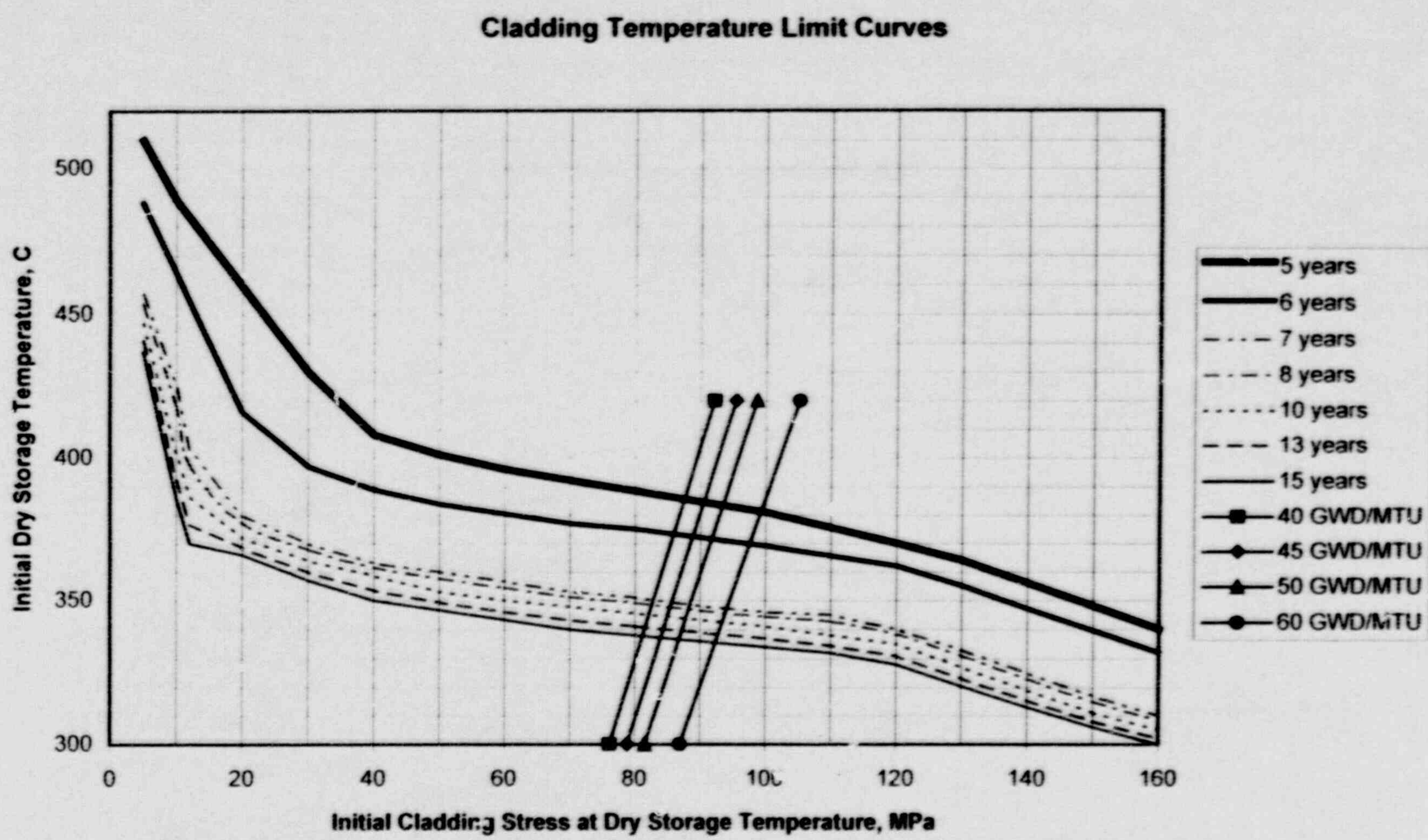


FIGURE A1-5
GRAPHS FOR DETERMINATION OF ALLOWABLE TEMPERATURES FOR WESTINGHOUSE 15x15 ASSEMBLIES

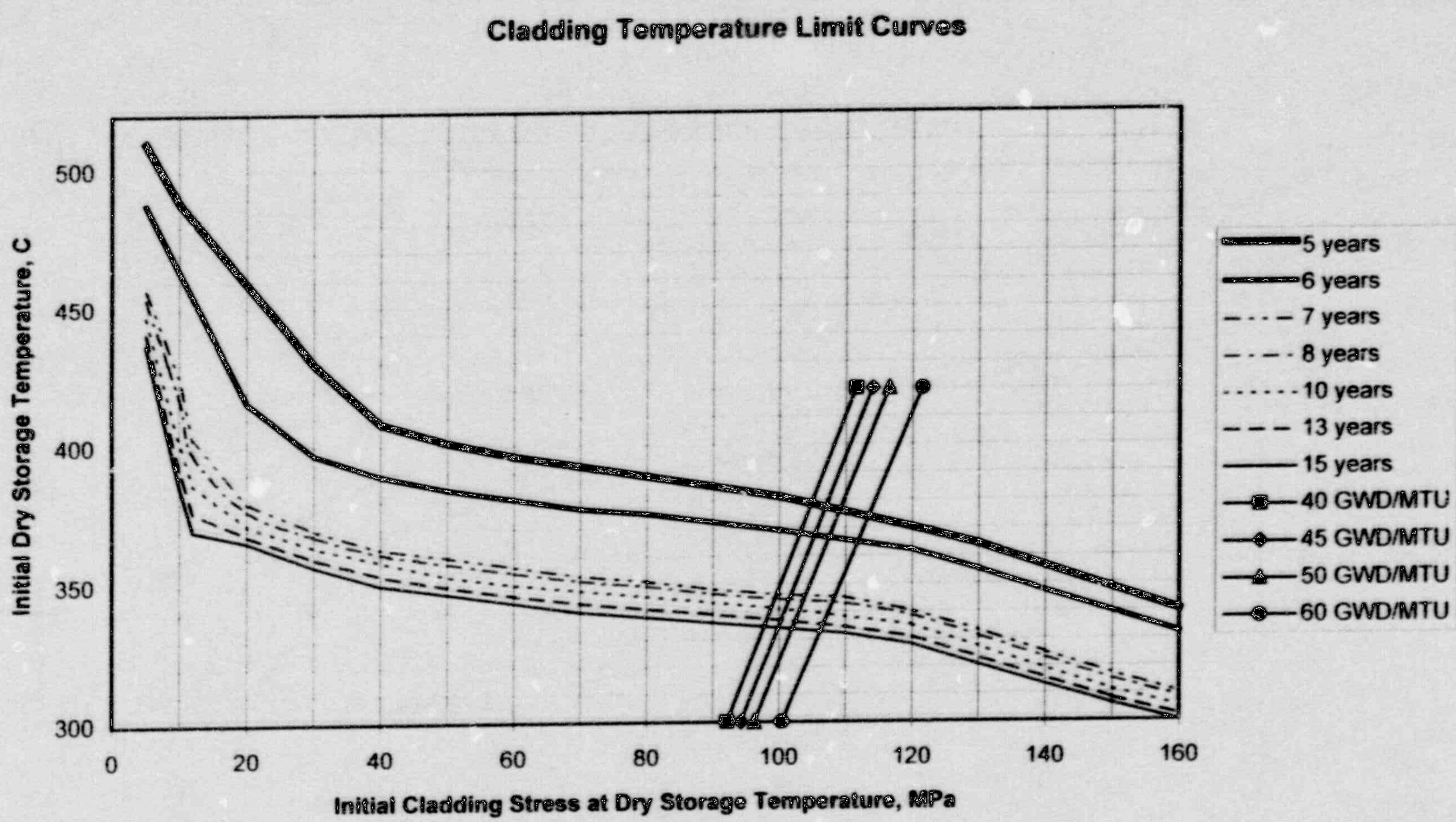


FIGURE A1-6
GRAPHS FOR DETERMINATION OF ALLOWABLE TEMPERATURES FOR WESTINGHOUSE 14x14 ASSEMBLIES

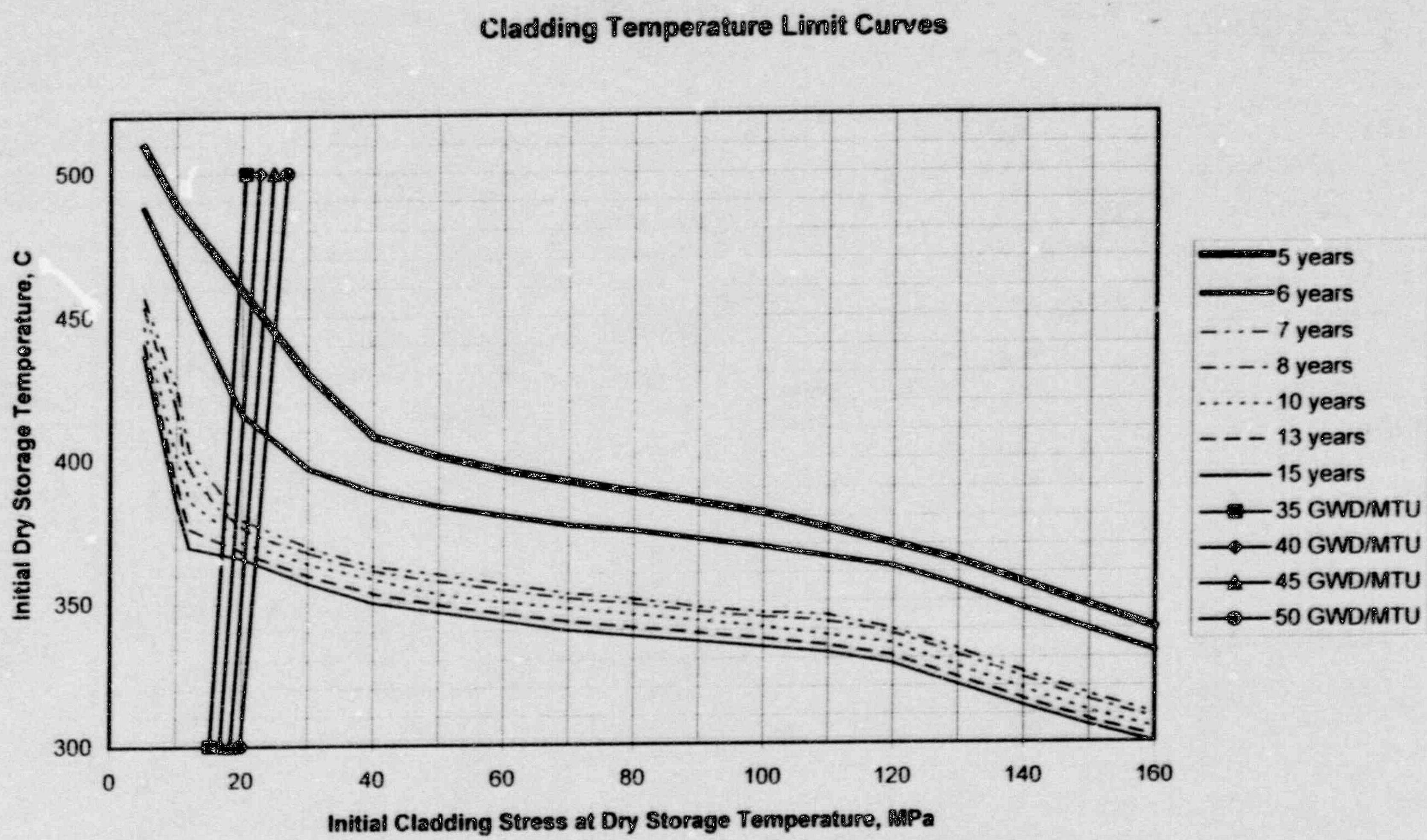


FIGURE A1-7
GRAPHS FOR DETERMINATION OF ALLOWABLE TEMPERATURES FOR GE 8x8 ASSEMBLIES

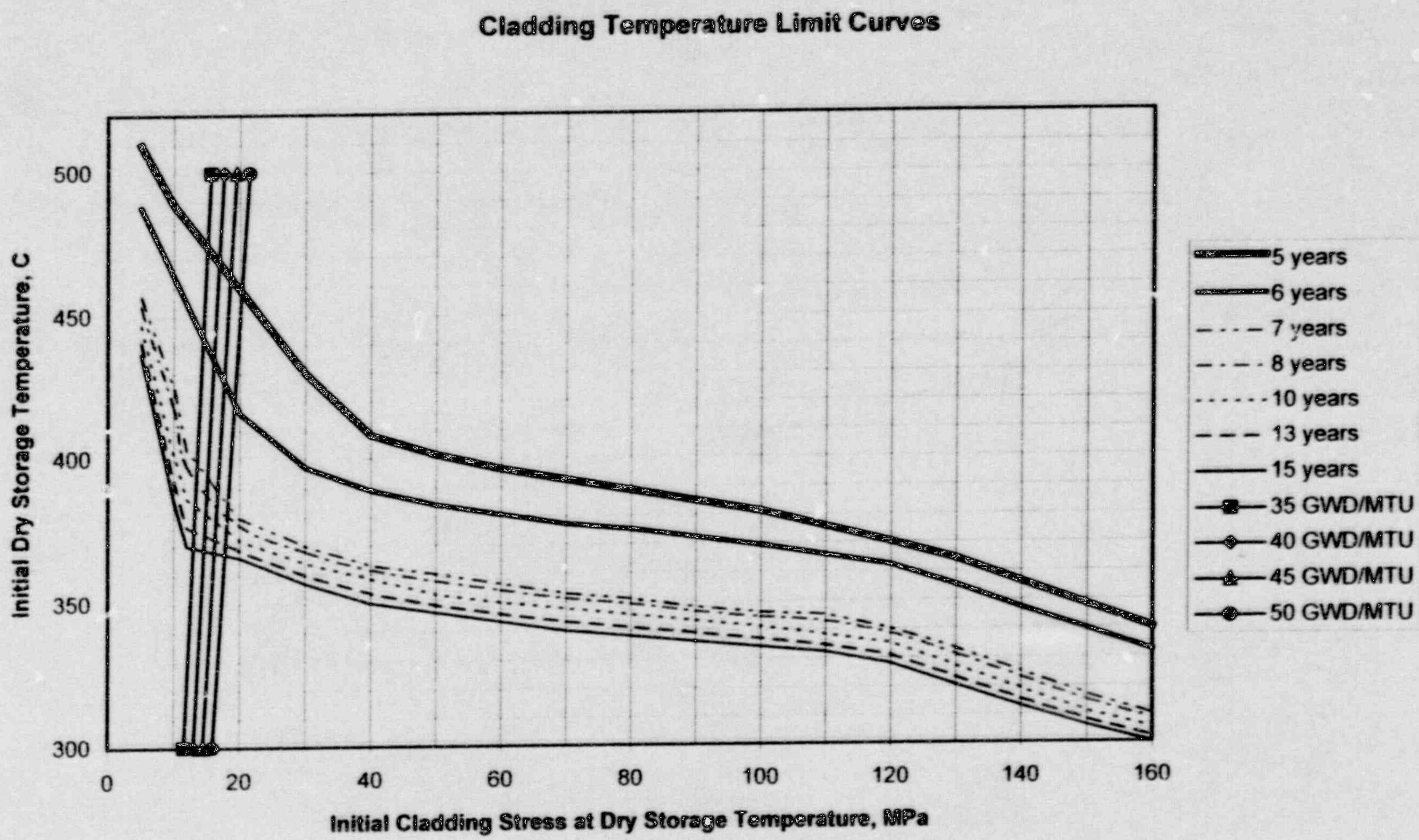


FIGURE A1-8
GRAPHS FOR DETERMINATION OF ALLOWABLE TEMPERATURES FOR GE 6x6 AND GE 7x7 ASSEMBLIES

APPENDIX 2

SUMMARY OF RESULTS FOR THE LIGHT TRANSFER CASK

2.1 INTRODUCTION

As stated in Chapter 1.0, the light TranStor™ Transfer Cask is designed for the BWR plants with crane capacities of at least 100 ton but below the 110 ton required to handle the standard transfer cask. This Appendix presents the results of the analysis for the light cask.

2.2 DISCUSSION AND RESULTS

The methodology for the cask analysis is the same as for the standard transfer cask for all three aspects of the design (structural, shielding, and thermal). These methodologies are discussed in the corresponding chapters of the SAR and the not repeated herein. The results are presented below.

2.2.1 Structural Analysis

The weight and c.g. locations of the cask in different configurations are presented in Table A2-1. The lifting results are summarized in Table A2-2. It can be seen that the cask meets the requirements of NUREG 0612/ANSI N14.6 for single failure proof lifting devices.

2.2.2 Thermal Analysis

The light transfer cask contains less shielding than the standard cask. Therefore, it provides better heat dissipation and no additional thermal analysis is required.

2.2.3 Shielding Analysis

The shielding analysis results for the design basis fuel are presented in Table A2-3.

TABLE A2-1

TranStor™ LIGHT TRANSFER CASK WEIGHTS AND CENTERS OF GRAVITY

ITEM DESCRIPTION	WEIGHT (lbs)	CENTER OF GRAVITY (inches above bottom)
• Transfer Cask (Empty w/o Lid)	107,930	89.4
• Transfer Cask with Basket (Empty, w/o Shield Lid)	140,090	92.1
• Transfer Cask with Basket (Loaded, dry, w/ Lids)	192,950	99.1

TABLE A2-2

TranStor™ LIGHT TRANSFER CASK LIFTING EVALUATION

Component		Cask Safety Factors	Minimum Safety Factors per NUREG 0612/ ANSI N14.6
Trunnions	yield	13.0	6
	ultimate	28.6	10
Shell	yield	7.4	6
	ultimate	11.3	10
Lower plate	yield	6.8	6
	ultimate	11.9	10
Lower weld	yield	13.0	6
	ultimate	22.8	10
Rail-to-shell	yield	7.8	6
weld	ultimate	13.8	10

Component	Stress/Force	AZSC Allowable
Cover Plate (ksi)	12.0	24.6
Bolts (kips)	26.3	34.6

TABLE A2-3

TransStor™ LIGHT TRANSFER CASK DOSE RATES (mrem/hr)

Detector Location	BWR 35 GWd/MTU 5 year cool	BWR 40 GWd/MTU 6 year cool	BWR 45 GWd/MTU 7 year cool	BWR 50 GWd/MTU 8 year cool
1	γ = 257 n = 109 Tot = 366	γ = 245 n = 191 Tot = 436	γ = 209 n = 200 Tot = 409	γ = 185 n = 206 Tot = 391
2	γ = 720 n = 182 Tot = 902	γ = 702 n = 320 Tot = 1022	γ = 611 n = 333 Tot = 944	γ = 535 n = 344 Tot = 879
3	γ = 1222 n = 134 Tot = 1356	γ = 1191 n = 235 Tot = 1426	γ = 1041 n = 244 Tot = 1285	γ = 910 n = 253 Tot = 1163
4	γ = 33 n = 95 Tot = 128	γ = 31 n = 168 Tot = 199	γ = 27 n = 175 Tot = 202	γ = 23 n = 181 Tot = 204
5	γ = 129 n = 38 Tot = 167	γ = 123 n = 66 Tot = 189	γ = 105 n = 69 Tot = 174	γ = 93 n = 71 Tot = 164
6	γ = 160 n = 319 Tot = 479	γ = 176 n = 561 Tot = 716	γ = 134 n = 583 Tot = 717	γ = 117 n = 603 Tot = 720
7	γ = 76 n = 188 Tot = 264	γ = 73 n = 332 Tot = 405	γ = 63 n = 345 Tot = 408	γ = 55 n = 357 Tot = 412

NOTE: Refer to Figure 5.1-2 for detector locations.

APPENDIX 3

MANUAL OF QUALITY ASSURANCE

MANUAL OF QUALITY ASSURANCE

FOR

SIERRA NUCLEAR CORPORATION

AND AFFILIATES

Approved By: ______________________ **Date:** 1-31-95
George N. Dixon Jr., Manager
Quality Assurance/Control

Approved By: ______________________ **Date:** 1-31-95
John V. Massey, Ph.D.
President

Quality Assurance Manual
Rev. 4
January 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 6, 1995

Sierra Nuclear Corporation
ATTN: Mr. George N. Dixon, Jr.
Manager, Quality Assurance/Control
620 Colonial Park Drive
Roswell, GA 30075

Dear Mr. Dixon:

Enclosed is Quality Assurance Program Approval for Radioactive Material Packages No. 0804, Revision No. 0. This Approval satisfies the requirements of 10 CFR § 71.12(b) for a Quality Assurance Program approved by the Commission.

Please note the conditions in the Approval. In addition, we note an apparent typographical error in Revision 3 of Section 17, "Quality Assurance Records," in your Manual of Quality Assurance. The first paragraph states, "...records of use, for all transport packages, shall be maintained for a period of two years after the shipment." 10 CFR § 71.91 requires these records to be maintained for three years after shipment. Please revise Section 17 of your Manual of Quality Assurance to correct this discrepancy, or let me know if you have any objections to this revision.

This Approval will remain in effect until the expiration date, indicated in Block No. 3. Termination of your materials license does not cause this Approval to be automatically terminated. If you wish to renew, amend, or terminate this Approval, please request it in writing.

This letter also serves as a reminder that if you are using or planning to use an NRC-approved packaging, you must be registered for use of that packaging with NRC. Registration for use of NRC-approved packagings should be made pursuant to 10 CFR § 71.12(c)(3).

Sincerely,

John P Jankovich

John P. Jankovich, Section Leader
Quality Assurance Section
Source Containment and
Devices Branch
Division of Industrial and
Medical Nuclear Safety, NMSS

Docket No. 71-0804

Enclosure:
As stated

NRC FORM 311 (2-83)

U. S. NUCLEAR REGULATORY COMMISSION

QUALITY ASSURANCE PROGRAM APPROVAL
FOR RADIOACTIVE MATERIAL PACKAGES

1. APPROVAL NUMBER: 0804

REVISION NUMBER: 0

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and Title 10, Code of Federal Regulations, Chapter 1, Part 71, and in reliance on statements and representations heretofore made in Item 5 by the person named in Item 2, the Quality Assurance Program identified in Item 5 is hereby approved. This approval is issued to satisfy the requirements of Section 71.101 of 10 CFR Part 71. This approval is subject to all applicable rules, regulations, and orders of the Nuclear Regulatory Commission now or hereafter in effect and to any conditions specified below.

2. NAME: Sierra Nuclear Corporation

STREET ADDRESS: 620 Colonial Park Drive

CITY: Roswell

STATE: GA

ZIP CODE: 30075

3. EXPIRATION DATE: June 30, 2000

4. DOCKET NUMBER: 71-0804

5. QUALITY ASSURANCE PROGRAM APPLICATION DATE(S): February 3, and May 23, 1995

6. CONDITIONS

1. Activities conducted under applicable criteria of Subpart H of 10 CFR Part 71 to be executed with regard to transportation packagings.

2. Records describing activities affecting quality shall be retained for three years, in accordance with 10 CFR § 71.91 and § 71.135.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John P Jankovich

John P. Jankovich, Ph.D.
CHIEF, TRANSPORTATION BRANCH
DIVISION OF INDUSTRIAL AND MEDICAL NUCLEAR SAFETY
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

June 6, 1995

DATE

TABLE OF CONTENTS

HISTORY OF REVISIONS

Section	Title	Page	Rev	Effective Date
	Preface	IV	0	01/95
	Statement of Management Policy	V	0	04/88
			1	05/90
			2	01/95
1	Organization	1-1	0	04/88
			1	05/90
			2	01/94
			3	01/95
2	QA Program	2-1	0	04/88
			1	05/90
			2	01/94
			3	01/95
3	Design Control	3-1	0	04/88
			1	05/90
			2	01/94
			3	01/95
4	Procurement Document Control	4-1	0	04/88
			1	05/90
			2	01/94
			3	01/95
5	Instructions, Procedures and Drawings	5-1	0	04/88
			1	05/90
			2	01/95
6	Document Control	6-1	0	04/88
			1	05/90
			2	01/94
			3	01/95
7	Control of Purchased Materials, Parts and Components	7-1	0	04/88
			1	05/90
			2	03/92
			3	01/94
			4	01/95
8	Identification and Control of Materials, Parts and Components	8-1	0	04/88
			1	05/90
			2	01/95
9	Control of Special Processes	9-1	0	04/88
			1	05/90
			2	01/95

HISTORY OF REVISIONS
(Continued)

Section	Title	Page	Rev	Effective Date
10	Inspection	10-1	0	04/88
			1	05/90
			2	01/95
11	Test Control	11-1	0	04/88
			1	05/90
12	Control of Measuring and Test Equipment	12-1	0	04/88
			1	05/90
			2	01/95
13	Handling, Storage, Preservation and Shipping	13-1	0	04/88
			1	05/90
14	Inspection, Test and Operating Status	14-1	0	04/88
			1	05/90
			2	01/95
15	Nonconforming Material, Parts and Components	15-1	0	04/88
			1	05/90
			2	01/94
			3	01/95
16	Corrective Action	16-1	0	04/88
			1	05/90
			2	01/94
			3	01/95
17	QA Records	17-1	0	04/88
			1	05/90
			2	01/94
			3	01/95
18	Audits	18-1	0	04/88
			1	05/90
			2	01/94
			3	01/95

PREFACE

This manual has been developed for and applies to Sierra Nuclear Corporation and any wholly or partially owned subsidiary or partnership (such as Pacific Sierra Nuclear Associates).

The Quality Assurance Manual (QAM) sections contained herein describe Sierra Nuclear Corporation's basic policy for the control of quality of products and services being provided by Sierra Nuclear Corporation and meets the requirements of Title 10 Code of Federal Regulations, Part(s), 71 Subpart H, 72 Subpart G, and 50 Appendix B, and other comparable industry standards such as ANSI/ASME NQA-1.

The QAM is supported by project, engineering and quality procedures, which provide detailed requirements for implementing this corporate quality assurance policy. Procedural coverage is included for design assurance, product quality assurance, and operating and maintenance requirements. The application of this program uses the "graded" approach, as defined in Regulatory Guide 7.10, depending on the complexity, criticality, and safety requirements of each project or component.

The initial release of the QAM and all subsequent revisions will be transmitted with a memo approved by the Manager, Quality Assurance/Control. Additional procedures will be prepared under appropriate sections or in subsequently identified sections for special coverage as required for contracts, if not adequately covered in the basic manual.

STATEMENT OF MANAGEMENT POLICY

The Quality Assurance Program described herein is applicable to all products and services provided by Sierra Nuclear Corporation and any wholly or partially owned subsidiary or partnership (such as Pacific Sierra Nuclear Associates) to clients requiring a Quality Program meeting the requirements of Title 10 Code of Federal Regulations Part(s) 50 Appendix B, 71 Subpart H, and 72 Subpart G, or other comparable industry standards such as ANSI/ASME NQA-1.

The executive management of Sierra Nuclear Corporation and Associates is devoted to the support of this program and charges all employees involved in activities affecting quality with the responsibility of upholding and abiding by the Quality Assurance procedures in this manual. The Quality Assurance organization is authorized sufficient freedom to identify quality problems; initiate, recommend or provide solutions; verify implementation of solutions; and control further processing of service(s) or delivery of a nonconforming item, deficiency or unsatisfactory condition until proper disposition has been completed.

While it is the responsibility of everyone at Sierra Nuclear Corporation to assure that quality and reliability objectives are achieved, the overall responsibility for the development, maintenance and assurance of the implementation of the Quality Assurance Program has been assigned to the Manager, Quality Assurance/Control who reports directly to the Corporate President of Sierra Nuclear Corporation.

Quality Assurance is recognized by corporate management as an interdisciplinary function for which the Manager of Quality Assurance/Control is charged by the Corporate President with the responsibility for establishing, implementing and maintaining a system to assure the conformance of Sierra Nuclear Corporation activities to the applicable requirements.

The Manager of Quality Assurance/Control has the complete support of Corporate Management in the performance of required duties and, by organizational arrangement, has no responsibility for production costs or schedules. The authority, as defined herein, extends to all activities performed by or for Sierra Nuclear Corporation that may affect product quality. Decisions made by the Manager of Quality Assurance within scope of duties, responsibilities and authority as defined in this program may be changed or modified only by direction of the Corporate President.

All personnel assigned to operations subject to the requirements of this program shall be required to familiarize themselves with the policies and objectives set forth in this program. They shall be responsible for executing those policies, explicitly or implied, pertinent to their assignments.

JV Massey — 1-31-95

John V. Massey, Ph. D. — Date
President, Sierra Nuclear Corporation
and
General Manager, Pacific Sierra Nuclear Associates

SECTION 1

ORGANIZATION

Sierra Nuclear Corporation (SNC) is organized as shown in Figure 1.1.

The President[1] is responsible for management of SNC[1], setting overall company policy and identification of long-term company goals and resources, and retains ultimate authority and responsibility for review of the status and adequacy of the Quality Assurance Program. The President is responsible for assuring that the status and adequacy of the Quality Assurance Program are reviewed annually.

The assurance of quality at SNC is an interdisciplinary function that involves, as applicable, all organizations. Furthermore, quality assurance encompasses many diversified functions and activities and extends to various job levels within these organizations, including all executives and all employees whose activities affect quality. The implementation of quality assurance throughout the various functions of design, procurement, construction, operation and services at SNC must, therefore, be considered the direct responsibility of the organization performing the work and cannot be considered the sole domain of any single quality assurance group.

Persons or organizations charged with the development, enforcement or measurement and the sufficiency and effectiveness of the quality assurance program shall have the authority and organizational freedom necessary to effectively discharge those responsibilities. Such persons or organizations shall be independent of direct pressures of cost, schedule or production, and their authority and organizational freedom shall be sufficient to: (1) identify quality problems; (2) initiate, recommend or provide solutions; (3) verify implementation of solutions; and (4) withhold and segregate nonconforming material or other action, including stopping work to maintain program integrity. Furthermore, they shall have direct access to responsible management at a level where appropriate action can be mandated.

Persons performing quality assurance functions such as checking, verifying or reviewing the work of others (functions that do not encompass the development, enforcement or measurement of the sufficiency or effectiveness of SNC's Quality Assurance Program) shall have authority and organizational freedom to a degree sufficient to properly discharge their assigned quality assurance responsibilities. However, when authority and organizational freedom are restricted for any person performing quality assurance functions, an established line of communication to responsible management must exist sufficient to prevent suppression of those quality assurance functions and/or to resolve any disputes.

---

[1] This manual also applies to all quality activities performed by Pacific Sierra Nuclear Associates (PSN). Therefore, in regards to activities performed by PSN, the acronym SNC as it appears in this manual may also be interpreted as PSN and the word President may be interpreted as General Manager.

Final responsibility for the effectiveness and sufficiency of SNC's Quality Assurance Program resides with SNC; however SNC may delegate the establishment and execution of the program, or any part thereof, to other organizations. Those organizations may, in turn, delegate responsibility for applicable portions of the Program to other organizations.

The President of SNC assumes overall responsibility for ensuring the development and maintenance of an effective quality assurance program for SNC. Responsibility for the establishment, administration and enforcement of the SNC Quality Assurance Program has been delegated by the President to the Manager of Quality Assurance/Control. The Quality Assurance Department functions as a staff position reporting to the President of SNC and is independent of all other organizations within SNC, and assumes line responsibility for ensuring compliance with SNC's Quality Assurance Policy.

The Manager of Quality Assurance may delegate any of the functions assigned to him by this Manual to another individual, but he shall retain the responsibility for accomplishment of the function in accordance with the provisions of this Manual.

Any dispute over Quality Assurance with the management of other functions (engineering, projects, manufacturing, purchasing, etc.), which cannot be resolved with the respective department manager, shall be referred to the President for resolution.

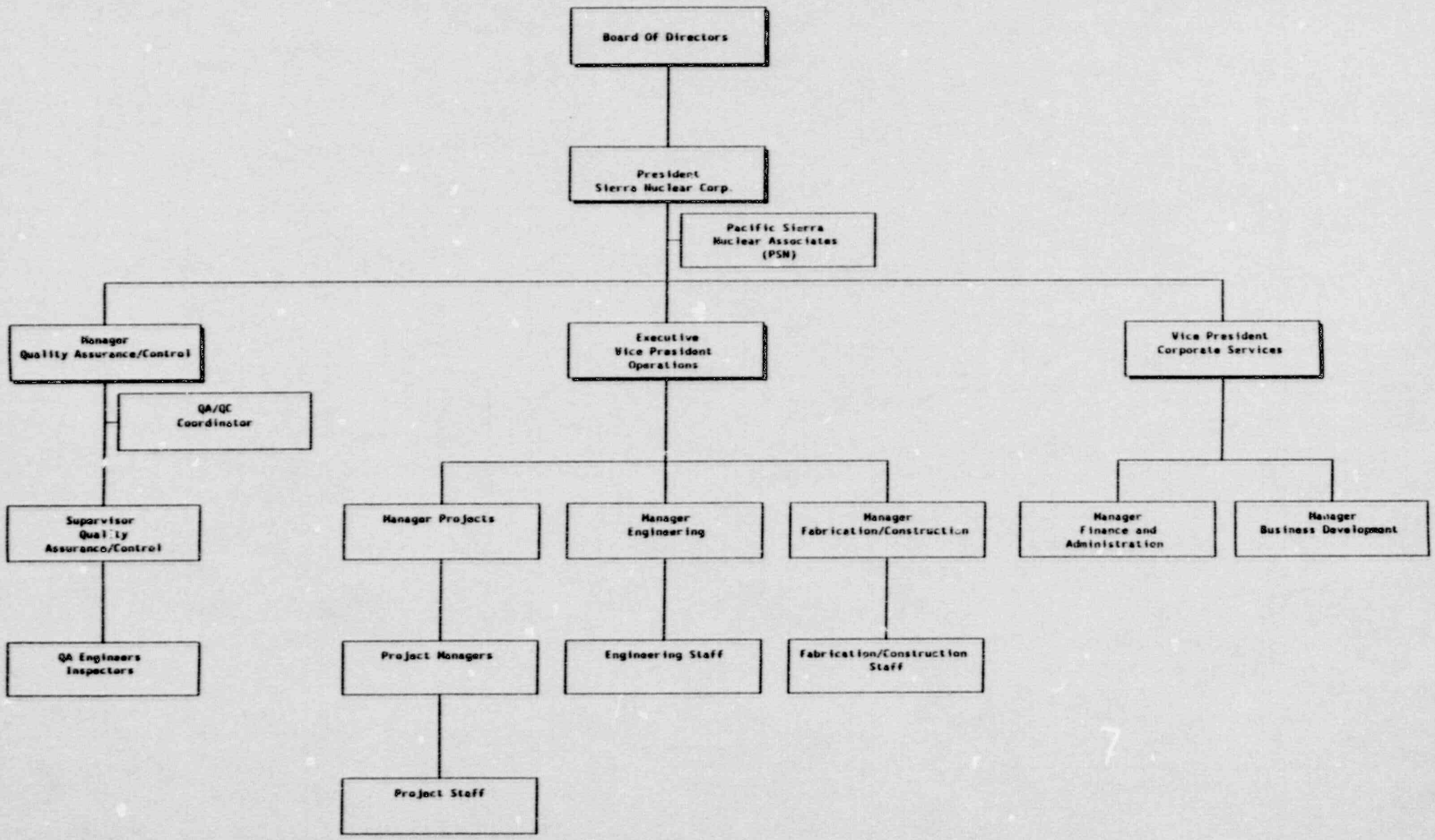
SNC shall verify the accomplishment of Quality, through scheduled and or unscheduled audits, of in-house functions and, as applicable, at sub-vendors and/or at suppliers.

Project Organization

Each large project that SNC undertakes is assigned a Project Manager who is responsible for the technical, financial and quality aspects of that project. The Project Manager and the appropriate general manager assign a project staff, which is typically organized as shown in Figure 1.2. The engineering discipline staffs are each headed by a project engineer who is responsible for all project work in that discipline. The Project Manager is responsible for interface control. The Fabrication and Construction group is responsible for all fabrication and construction subcontract management and also provides purchasing activities.

In small, single discipline projects, the Project Manager and the Project Engineer will be the same person.

Figure 1.1
SNC Organization
1 - 3



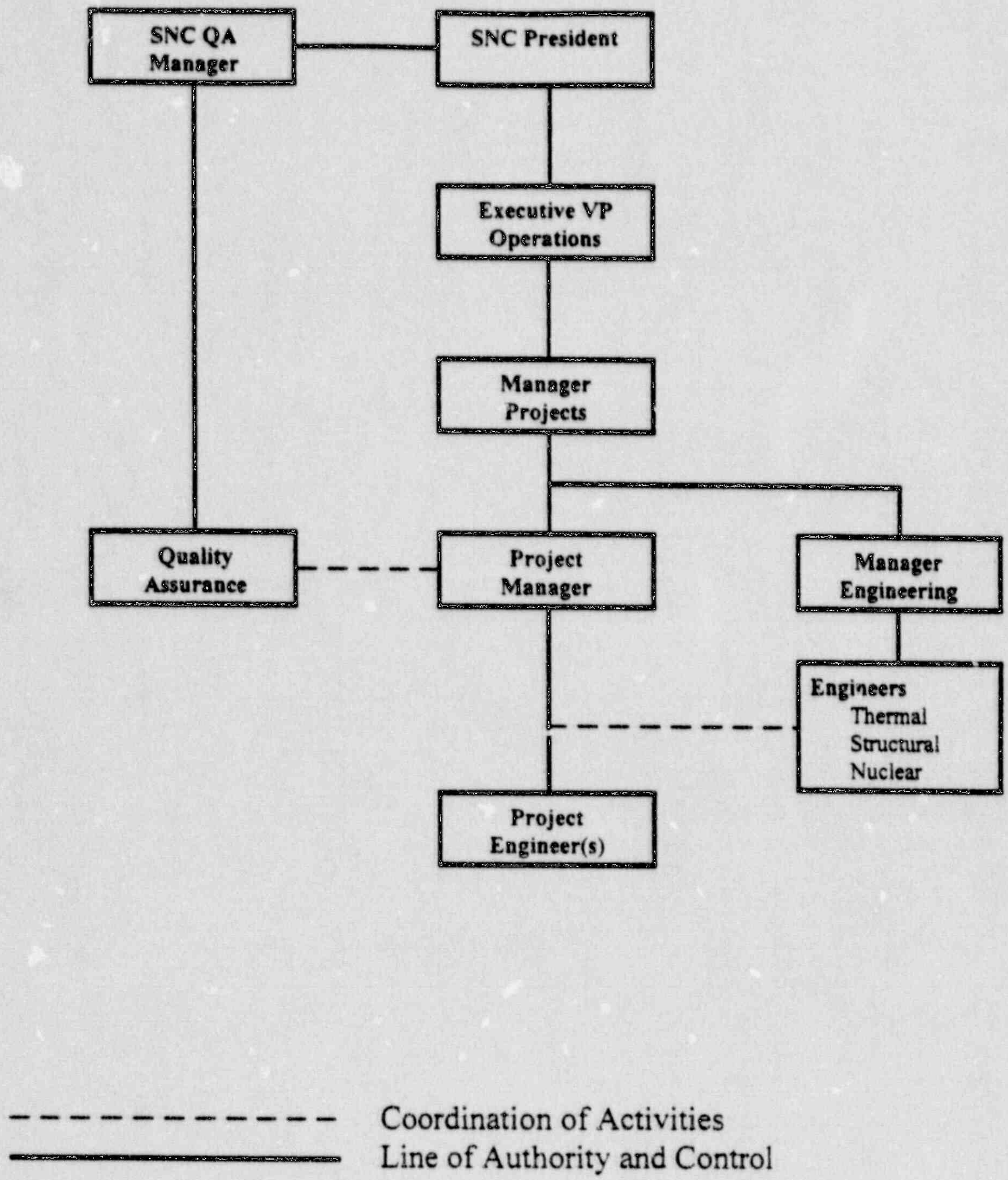


Figure 1.2 SNC Project Organization

SECTION 2

QUALITY ASSURANCE PROGRAM

This Quality Assurance Program shall apply to activities that are nuclear safety-related and require compliance with any or all of the documents listed below.

10 CFR 71, Subpart H
10 CFR 50, Appendix B
10 CFR 72, Subpart G
ANSI N45.2
NQA-1

Events that may be reportable either as "significant deficiencies" under 10CFR50.55(e) or as "substantial safety hazards" under 10CFR21 shall be reported in accordance with this Manual.

Manual Review and Approval

The SNC Quality Assurance Program is fully described in and implemented in accordance with this document, the SNC Quality Assurance Manual. The Manual is reviewed annually or more frequently as directed by the President. The intent of this review is to keep the Manual current with the documents specified above. Revisions to the Manual required for compliance to the referenced documents are authorized only by the President as necessary to meet client commitments.

Approval of this Manual, and revisions thereto, is documented by the signature of the President on the title page with the respective date of approval. All revisions to the Manual are implemented within thirty (30) days of approval of the revision by the President.

Revision Control

Manual revisions are highlighted by a vertical line in the margin or double underlining the changes, and the revision number is indicated at the bottom of the page. When a revision to the Manual requires page reformatting to the extent that the text is relocated and not changed, the relocation is highlighted by an asterisk next to the revision number. Previous revisions are unshaded (or the line removed) when a new revision is issued.

Whenever a revision is made to any portion of a Section, the Section revision level is increased by one and the entire Manual revision level is increased by one. A history of revisions is maintained for each section as indicated on the Section History of Revisions Page (II). The current revision of the Manual, in addition to being indicated on the title page is also indicated on the Table of Contents and History of Revision pages, in the lower left hand corner.

Transmittal Control

The recipient of the Controlled Manual and revisions thereto verifies the receipt of the Manual and subsequent revisions and certifies that his/her Manual is in agreement with the latest revision as indicated on the table of contents, by signing and returning the SNC Quality Assurance Controlled Document Transmittal. The recipient is required to return the completed Controlled Document Transmittal Form to the designated SNC person. The recipient shall be responsible for the destruction of the obsolete pages.

The Manager of Quality Assurance/Control (QA/QC) or designee shall take appropriate measures to secure delinquent SNC Quality Assurance Controlled Document Transmittal Forms. Except for SNC personnel, delinquent Transmittals may result in uncontrolling the subject Manual after thirty (30) days. Transmittals not returned from SNC personnel within thirty days after transmittal date shall be investigated by the designated QA personnel and followed up with corrective action as necessary.

Control Log

The Quality Assurance Department maintains a Control Log that records the holders of Controlled Manuals.

Holders of Uncontrolled Manuals receive a copy of the Manual that is current at the time of issue, but will not receive subsequent revisions to the Manual. Holders of Controlled Manuals automatically receive future revisions to the Manual.

Indoctrination and Training

The Manager, QA/QC or designee will ensure that personnel performing activities affecting quality are indoctrinated, trained and qualified according to their level of responsibility and assigned functions. Indoctrination and training shall consist of informal, on-the-job activities under the guidance of trained personnel and/or formal meetings, classes, lectures, and seminars. Formal training shall be documented on a Quality Assurance Indoctrination Log by the individual who leads the indoctrination and training session, or a designee. The record shall include names of personnel trained and a description of the material covered. The form shall be forwarded to the Manager of Quality Assurance, who will maintain it as a Quality Assurance Record in accordance with Section 17.

Qualification and Certification of Personnel

SNC does perform inspections, examinations or tests for which a formal SNC program of training qualification and certification is required per QAM Section 10.0 and Section 11.0. When these inspection, examination or test activities are performed, they shall be performed by appropriately certified personnel.

Surveys and audits for which SNC is responsible are conducted by Lead Auditors who are qualified as specified in ANSI/ASME N45.2.23; 10 CFR 50, Appendix B; and NQA-1. Records of Lead Auditor qualification are maintained in the SNC files (refer to Section 18).

Quality Assurance Program Implementation

Quality Assurance Procedures (QAP) are developed to implement the requirements defined by this Quality Assurance Program. QAPs, or project specific procedures, may be developed for each project because of different interface requirements between clients and suppliers. These project specific procedures shall be part of the project plan. In all cases, the QAPs shall conform to the requirements specified in this Quality Assurance Program description.

Table 1 identifies the relationships among the 18 criteria and the SNC Quality Assurance Manual and implementing Quality Assurance Procedures.

RELATIONSHIP OF 18 CRITERIA TO SNC QA PROGRAM

Criteria of 10 CFR 50 Appendix B 10 CFR 71 Subpart H 10 CFR 72 Subpart G	Corresponding QA Manual Section and QA Procedures *
I. Organization	QAM Section 1 and Organization Chart, QAP 1.0: Project Organization
II. Quality Assurance Program	QAM Section 2 QAP 2.0: Quality Forms Control QAP 2.1: Control and Distribution of SNC QAP's QAP 2.2: Certification of Inspection personnel QAP 2.3: Control and Distribution of Project Specific QA Plans QAP 2.4: SNC Projects Quality Assurance Program Assessment and Reporting System QAP 2.5: Personnel QA Indoctrination
III. Design Control	QAM Section 3 QAP 3.0: Design Control QAP 3.1: QA Review of Design Documents QAP 3.2: Document Change Request Notice QAP 3.3: Specification Selection and Qualification of items & Services
IV. Procurement Document Control	QAM Section 4 QAP 4.0: Procurement Document Control QAP 4.1: Review of Safety-Related Purchases QAP 4.2: Review of Commercial Quality Procurement Documents
V. Instruction, Procedures & Drawings	QAM Section 5 QAP 5.0: Instructions, Procedures and Drawings QAP 5.1: Control of Fabrication, Construction, and Inspection Procedures
VI. Document Control	QAM Section 6 QAP 6.0: Document Control QAP 6.1: Control of Drawings During the Preliminary Design Phase
VII. Control of Purchased Material, Equipment & Services	QAM Section 7 QAP 7.0: Control of Purchased Items and Services QAP 7.1: Supplier Evaluation QAP 7.2: Source Inspection QAP 7.3: Receiving Inspection QAP 7.4: Supplier "Readiness-Reviews"

VIII. Identification & Control of Material, Parts, & Components	QAM Section 8 QAP 8.0: Identification and Control of Material, Parts, and Components
IX. Control of Special Processes	QAM Section 9 QAP 9.0: Control of Special Processes
X. Inspection	QAM Section 10 QAP 10.0: Inspection
XI. Test Control	QAM Section 11 QAP 11.0: Test Control
XII. Control of Measuring & Test Equipment	QAM Section 12 QAP 12.0: Control of Measuring and Test Equipment
XIII. Handling, Storage & Shipping	QAM Section 13 QAP 13.0: Handling, Storage, and Shipping
XIV. Inspection, Test & Operating Status	QAM Section 14 QAP 14.0: Inspection, Test & Operating Status
XV. Nonconforming Materials, Parts, or Components	QAM Section 15 QAP 15.0: Nonconforming Material, Parts or Components QAP 15.1: Material Review Board QAP 15.2: Reporting of Potential Deficiencies and Defects QAP 15.3: Nonconformances Dispositioned "Use-as-is" and "Repair"
XVI. Corrective Action	QAM Section 16 QAP 16.0: Corrective Action
XVII. QA Records	QAM Section 17 QAP 17.0: QA Records
XVIII. Audits	QAM Section 18 QAP 18.0: Audits QAP 18.1: Qualification and Certification of QA Personnel QAP 18.2: Quality Assurance Surveillance of Suppliers

- This listing and the content of these Quality Assurance Procedures may change as needed to provide procedural direction applicable to SNC's commercial activities, or as necessary to reflect changes in governing standards/regulations. Such procedural direction will not deviate from the SNC QA Manual.

SECTION 3

DESIGN CONTROL

The purpose of this section establishes the basic minimum requirements to provide an independent review of program planning and design review from initial concepts through completion of design, manufacturing, inspection and planning for SNC and Client-based projects. SNC provides designs, investigations, analyses and reports based on specific project requirements.

Before any work including design input, conclusion or review remarks can be provided on a project the following actions are taken by SNC.

The Manager of Projects shall appoint a Project Manager.

The Project Manager and applicable Discipline Managers shall appoint project engineers and other support staff, commensurate with the project scope.

The Project Manager shall prepare a "Project Plan." The Project Plan lists all of the intended activities required to accomplish/support specific project needs for SNC and or client based projects, including all design bases and/or regulatory requirement documents applicable to the project. Design interfaces and quality requirements shall be described in the Project Plan.

The Project Manager is responsible for the preparation of a list of task assignments along with a schedule of milestone completion dates and for providing each designated project participant with a copy.

The Project Manager is responsible for holding a project orientation meeting to review all of the above items and to identify project needs. He is also responsible for in-progress project meetings to track the project and assure the proper design interface.

The Project Manager is responsible for assuring the technical adequacy and correctness of the design and that the final design meets the SNC, client and regulatory requirements. Procedures have been developed to assist is assuring and documenting the quality of the design output. These procedures cover the following:

- -Preparation of calculations
- -Review and checking of calculations and reports
- -Computer program control and usage
- -Drawing preparation
- -Design verification
- -Change control
- -Procured design services

SNC Quality Assurance Procedures (QAP's) have been developed, approved and implemented to control the design process in such a manner to assure the following:

- That QA personnel participate in the design development and review process. This is done to assure adherence to all applicable design criteria. This activity is accomplished through review and and approval of drawings and specifications developed in support of the design.

- That the design activity is planned, controlled and documented.

- The design documents contain Quality Assurance requirements for inspections and test that will assure control, inspection and testing of design characteristics.

- That deviations from quality requirements are controlled.

- That design verification is performed by qualified personnel independent of the design activity, but with a skill level at least equal to that of the original design personnel. These verifications may include tolerance studies, alternate calculations or tests. Qualification tests are conducted in accordance with approved test programs and procedure.

- That the design verification method selection is based on regulatory and contractual requirements, level of complexity of the design and "state-of-the-art" considerations, i.e., materials, fabrication processes, etc., and operating conditions.

- That interface control is established and adequate to assure that the review, approval, release, distribution and revision of design documents involving interfaces are performed with all cognizant design personnel.

- That all design and specification changes are reviewed and approved by the same organization(s) as the original issue.

- That design errors and deficiencies are documented and corrective action to prevent recurrence is taken.

SECTION 4

PROCUREMENT DOCUMENT CONTROL

The QA Program provides controls to assure that all purchased material, components, equipment, and services adhere to design specifications, regulatory and contractual requirements.

Evaluation and selection of suppliers, objective evidence of supplier quality, assignment of quality requirements to procurement documents, and related design documents, and source, in-process and receiving inspection are all administered and controlled in accordance with this section of the Quality Assurance Manual and approved procedures

Procurement is performed under the supervision of the Project Manager. Particular emphasis is placed on assuring that revisions to procurement documentation are reviewed and approved by the same groups as the original.

Quality Assurance requirements, when applicable, are included with request for quotes. Quality Assurance requirements are always provided with the purchase orders and/or applicable specifications.

Procurement of engineering design services are addressed in Section 3. SNC may procure any design, manufacturing, inspection, testing, auditing or job site construction activity described in this Manual. Procurement documents for these services shall include requirements that assure that the requirements of this Manual, as applicable to SNC, will be met by the subcontractor. SNC retains final responsibility to assure the service is acceptable for the SNC project.

Contract documents such as Purchase Orders, drawings and specifications are reviewed to assure the inclusion of all requirements. Personnel qualification requirements are either defined or verified by reference on a procurement document. Review also includes verification of the suitability of standard items for the use required by the applicable drawings and design specifications with the inclusion of valid industry standards, references, and related data, when applicable.

The Project Manager assures that requirements for acceptance of hardware and documentation, such as the affiliate's or a supplier's submittal and retention instructions appropriate to the contract, are included in procurement documentation.

SNC maintains the right of access to all supplier facilities and documentation for source inspection and/or audit activities. A statement to this effect is included on procurement documentation when it is appropriate to the contract.

Changes to procurement documents shall be prepared, reviewed, approved and authorized for use in the same manner as established for the original issue of the document.

SNC QA personnel check procurement documents for completeness and the inclusion of quality requirements. The procurement documents are reviewed in accordance with written procedures and require the approval of the SNC Manager of Quality Assurance/Control.

SECTION 5

INSTRUCTIONS, PROCEDURES AND DRAWINGS

Inspection procedures and instructions are developed by a qualified engineer assigned by the Manager, Quality Assurance/Control (QA/QC). Procedures are developed for activities requiring design and/or fabrication, performance verification, witnessing, measurements, testing or other Quality Assurance related activities. These procedures are approved by the Project Manager.

All fabrication documents (i.e., drawings, specifications, special processes, and tests, etc.) are reviewed by Quality and a qualified engineer specified by the Project Manager or Project Engineer. The fabrication documents are also referenced in Inspection Procedures as necessary to assure adherence to package, system or other design approvals and the applicable regulatory and contractual requirements.

The Inspection Procedures also include appropriate acceptance criteria such as dimensions, tolerances, operating limits, workmanship standards, and other qualitative and quantitative measures.

All instructions, procedures, and drawings are developed, reviewed, approved, utilized and controlled in accordance with approved procedures.

Changes to Instructions, Procedures and Drawings shall be prepared, reviewed, approved and authorized for use in the same manner as established for the original issue of the document.

SECTION 6

DOCUMENT CONTROL

The policy for review, approval, release and revision of quality related documents establishes review and approval cycles and sequences as well as require that revisions/changes to all such approved documents be subjected to the same approval cycle and sequence. Provisions are made for identifying individuals/organizations responsible for review, approval and issuance of documents with the Controlled Document Transmittal Log. Document control responsibilities, facilities and distribution requirements are also addressed by Controlled Document Transmittal. Transmittal sheets with provision for acknowledging receipt are utilized to provide proper records of the transmittal and receipt of controlled documents and subsequent revisions.

The Project Engineer shall assure that documentation listings are maintained specifying the title, number and current revision for all drawings, procedures, specifications and purchase orders.

Controlled documents include but are not limited to:

- Design Specifications
- Calculations
- Analyses
- Safety Analysis Report(s)
- Drawings
- Specifications (Procurement, Equipment, etc.)
- Special Process Procedures (Welding, Forming, Heat Treating, NDE, Etc.)
- Inspection Procedures
- QA Manuals and Procedures
- Source Surveillance and Inspection Reports
- Test Procedures and Reports
- Operational Test and Inspection Reports
- Subvendor Procedures, Specifications and Drawings
- Client Specifications, Procedures and Drawings

When revised documents appear in other documents as references, supplements or exhibits, appropriate revisions are made to the affected documents prior to the release of the approved change.

Documentation listings are maintained, listing the title, document number and current revision for all controlled documents.

SECTION 7

CONTROL OF PURCHASED MATERIALS, PARTS AND COMPONENTS

It is the policy of Sierra Nuclear Corporation (SNC) that all suppliers of materials, components, systems or services received, controlled and approved procurement documents that contain or reference all applicable regulatory requirements, appropriate design/engineering drawings and specifications, and other requirements necessary to produce a product or service that meets the Quality requirements of SNC. In addition, procurement documents shall contain provisions that require suppliers and their subtier suppliers to execute quality assurance programs in a manner, and to the extent, specified by SNC. Furthermore, procurement documents shall provide for the right of SNC to audit its contractors as well as their subtier suppliers, on their implementation of these controls. All procurement will be made only from SNC approved suppliers based on their past history, pre-award and/or post-award audits and surveys. SNC shall maintain an Approved Suppliers List (ASL).

As directed by the Manager, Quality Assurance/Control (QA/QC) or designee, audits and surveys are conducted by SNC QA qualified personnel to further assure supplier acceptability and performance. These evaluations are based on one or all of the following criteria:

> The capability of the supplier to comply with the requirements of 10CFR72 Subpart G, 10CFR71 Subpart H, 10CFR50 Appendix B, ANSI N45.2, ASME/ANSI NQA-1, or other requirements appropriate to the contract as determined by SNC QA.
>
> A review of previous records and performance of the supplier by SNC Quality Assurance and Procurement.
>
> A survey/audit by an SNC multi-discipline team (QA, Engineering, and Manufacturing), but in all cases at least by QA, of the supplier's facilities and Quality Program to determine their capability to supply a product that meets the design, manufacturing, and quality requirements.

Results of the supplier evaluations and audits are appropriately recorded and included as part of the vendors history file, which are retained by Quality Assurance.

Audits are conducted at active supplier's facilities during, the performance of activities, to assure continued adherence to the imposed Quality Assurance, design and contract performance criteria. These audits are conducted at least once every three years (during active periods) or more often as directed by the Manager QA/QC or designee.

Quality Assurance requirements are stated in procurement documents as required by regulatory or contractual requirements.

As directed by the Manager QA/QC source and/or receiving inspections, are performed by qualified personnel to assure the following:

> The material, component, or equipment is properly identified, refers to applicable codes, standards and specifications, and corresponds with the identification on receiving documentation.
>
> Prior to their use or installation, materials, components, equipment and acceptance records are inspected and are accepted in accordance with appropriate contractual requirements.
>
> Inspection records and/or certificates of conformance are available that attest to the acceptance of materials and components prior to their installation or use.
>
> Items accepted and released are identified as to their inspection status prior to forwarding to a controlled storage area or release for further work.

All described activities are delineated in approved SNC Quality Assurance procedures.

SECTION 8

IDENTIFICATION AND CONTROL OF MATERIALS, PARTS AND COMPONENTS

A process fcr identifying and controlling materials, parts, components and completed and in-process assemblies is administered by Quality Assurance in accordance with approved Quality Assurance procedures. These procedures address quality status tags, marking, and/or stamping to assure maintenance of material identification, traceability, and part identification, to related documentation. Some of the details of these procedures are as follows:

- Material identification procedures included in Quality Assurance inspection instructions and fabrication drawings require that identification of material, components, and/or hardware be maintained on the item or in traceable records to prevent use of incorrect or defective material.

- Specifications, procurement documentation, fabrication and inspection records, discrepancy reports and material test data are also periodically audited to assure continued adherence to design, regulatory and contractual requirements.

Identification requirements such as method and size may be specified on applicable drawings or in applicable procurement/equipment specifications. Such identification shall not interfere with fit, interface or performance.

Quality Assurance shall assure that material and equipment are controlled, protected, stored, handled, operated and packaged so that identification, traceability and condition are maintained. Some or all of the material control functions described herein may be delegated approved suppliers.

SECTION 9

CONTROL OF SPECIAL PROCESSES

This section delineates the policies and practices established to control such special processes as: welding, heat treating, lead pouring, non-destructive examination, etc., in accordance with the applicable regulatory requirements and other applicable codes, standards, specifications or requirements. Special processes developed by suppliers and/or SNC are documented, reviewed and approved by the responsible technical personnel within the company, and/or customer organizations. In addition, special process equipment is identified, inspected and performance tested, prior to use.

All procedures for special processes are performed in accordance with applicable codes, standards, specifications and contract requirements. The personnel performing such processes are likewise qualified under the cognizance of the Quality Assurance function. Both the procedures and personnel are subjected to full review and approval cycles as defined herein, by personnel qualified and approved by the Manager, Quality Assurance/Control or Designee for the subject matter relating to the special process.

Qualification records and support data are retained in the Quality Assurance files.

All documentation shall be administered and controlled in accordance with the requirements of the SNC Quality Assurance Program.

SECTION 10

INSPECTION

Receiving, source, test, in-process, shipping and in-service inspection activities are performed in accordance with the requirements of this manual and approved procedures. Inspection personnel and/or organization qualifications are reviewed and accepted by the Manager, Quality Assurance/Control prior to inspection activity. The inspection activity is performed to verify conformance to drawings, procedures and/or specifications.

Inspection personnel report to the Manager of Quality Assurance

The qualifications of inspection personnel are based on their capability to perform the required inspection functions in accordance with applicable codes, standards, professional society programs (such as the ASQC quality technician certification, AWS QC1, SNT-TC-1A) and company training programs. Qualification reviews are performed periodically to maintain personnel proficiency and assure current qualification.

Inspection procedures and instructions include hold points, inspection equipment requirements, accept-reject criteria, personnel requirements, characteristics to inspect, variable attributes, recording instructions, reference documentation and other requirements as appropriate.

The inspection procedures and instructions include inspection results with supporting information such as variables, attributes, data, test results, NDE records, welding information, certified materials test report (and/or certification), special process data, discrepancy reports, related dispositions and resultant reinspection data.

SECTION 11

TEST CONTROL

A quality related test control program is defined by approved test procedures. Prerequisites, accept/reject criteria, data recording criteria, instrumentation calibration, environmental conditions, documentation and evaluation requirements, etc., are defined in the test procedures.

The Project Engineer assures that both "normal" and anticipated "off-normal" operational performance described in applicable design, regulatory and contractual documents are recreated by testing activities. Changes to test procedures are required to be reviewed/approved by the same organization(s) in the same cycle and sequence as the original issue.

Whenever equipment, components, and/or assemblies require modification, repairs, or replacement that could result in requirements for re-test or additional testing, the Project Engineer shall assure that original or new test inspection instructions are prepared and adhered to as appropriate.

Test results are documented, evaluated and accepted by the Project Engineer as required by the test procedure prepared for the test under the cognizance of the Project Enginner.

SECTION 12

CONTROL OF MEASURING AND TESTING EQUIPMENT

Calibration of measuring equipment and instrumentation is established by the Manager, Quality Assurance/Control. The calibration process assures that all standard measuring instruments used in the acceptance of material, equipment, and assemblies are calibrated and properly adjusted at specified intervals to maintain accuracy within pre-determined limits.

Calibrated equipment is identified and is traceable to the calibration test data. Identification includes the equipment property number, next calibration due date and inspector's or calibrator's signature or initials attesting to the accuracy and validity of the calibration.

Calibration accuracy is maintained by utilizing standards traceable to the National Institute of Standards Technology (NIST), derived from accepted values for natural physical constants, or by the ratio type of self-calibration.

SECTION 13

HANDLING, STORAGE AND SHIPPING

Requirements for handling, storage and shipping shall be documented in project specific procedures or specifications. These requirements are designed to prevent damage or deterioration of material and equipment. Information pertaining to shelf life, environment, packaging, temperature, cleaning, handling, preservation, etc., is included as required to meet design, regulatory and/or client requirements.

Inspection procedures and instructions contain assessment of criteria for handling, storage, preservation and shipping requirements.

Shipping documentation preparation, departure, and arrival time and destination data recording is also to be addressed, when applicable. The requirements pertaining to shipping must be met prior to release for shipment.

SECTION 14

INSPECTION, TEST AND OPERATING STATUS

The use of inspection status tags, quality inspection stamps, and other means to indicate inspection and test status at or for SNC, is described in project specific procedures or specifications.

Such procedures provide that indications of status are clear, inspection and/or test steps are not bypassed, and removal or modification of status indications are prohibited, except with Project Manager and/or Manager of Quality Assurance approval. The Manager, Quality Assurance/Control assures via procedure, interoffice memoranda, training sessions, and audit that personnel are aware of and understand the meaning and uses of status tags on hardware, material, and test setups and that the status tags are being satisfactorily used.

SECTION 15

NONCONFORMING MATERIAL, PARTS OR COMPONENTS

Material, components, and equipment that do not conform to requirements are controlled to prevent their inadvertent use. This control is through identification, segregation, discrepancy reporting, disposition of nonconformances by authorized individuals and reinspection activities. These are performed and controlled in accordance with written procedures.

Nonconformance Reports (NCR) are utilized and logged to identify discrepant items, describe the discrepancy and provide disposition and reinspection requirements. The signatures of authorized cognizant personnel are placed on the NCR to signify approval of the disposition.

NCRs are reviewed by the Project Manager and Quality Assurance Manager to assure that "accept-as-is" or "repair" dispositions include technical justification to indicate and assure continued compliance with design, regulatory and contractual requirements. When appropriate, copies of dispositions are forwarded to the owners and users of the affected equipment.

In conjunction with "repair" or "rework" dispositions, Quality Assurance personnel provide supplemental inspection planning to verify compliance with the NCR disposition. This assures that the item is retested and/or reinspected to a degree at least equal to the original acceptance level.

SECTION 16

CORRECTIVE ACTION

Failures, malfunctions, and deficiencies in material, components, equipment and services are identified and reported to the Project Manager and the President. A copy of each Corrective Action Request (CAR) is also forwarded to the Manager of Quality Assurance for review and analysis. The QA Manager also logs the CAR. The cause of the condition and the corrective action necessary to prevent recurrence is identified, implemented and then followed up to verify corrective action effectiveness.

Analyses of discrepancies are conducted within thirty days of their submittal. These analyses establish quality trends and help to pin-point areas in need of corrective action. The analyses, quality trends and related reports are prepared and presented to the President for review and action at that level. Copies of these reports and analyses are also provided to the Manager of Quality Assurance for review.

SECTION 17

QUALITY ASSURANCE RECORDS

SNC's QA records system is established and is administered in accordance with approved SNC QA procedures. The purpose of the QA records system is to assure that documented evidence pertaining to quality related activities is maintained and available for use by company, customer, and/or regulatory agency personnel, as appropriate. QA Records include, but are not limited to, design related records (calculations, drawings, research, development test reports and, design reviews), inspection and test records (including identification of inspectors and data recorders), audit reports, quality personnel qualification(s), quality related procurement data, supplier evaluation reports, material's analyses (certified material test reports or certificates of compliance as applicable), fabrication/manufacturing records, modification records, repair records, and maintenance records. The retention period for the above identified records is as follows: 1) Transportation Packaging - Life of the packaging plus three years; 2) Spent Fuel Storage Packaging - Shall be maintained by or under the control of the licensee until the commission (NRC)terminates the license. Additionally, records of use, for all transport packages, shall be maintained for a period of two years after the shipment. Records are identified by work order number, part number, contract number, or drawing number as appropriate to the record type. SNC maintains a complete list of QA records to provide identity and location information.

For all other equipment, quality related records are retained for a minimum of three years, but no more than five years unless otherwise specified by applicable regulatory, code, standard or contractual requirements.

Inspection records retained in the QA records system provide the following data when applicable:

(a) Inspection type, i.e., in-process, in-service, testing, receiving and shipping.

(b) Evidence of completion and verification of manufacturing, inspection or test operation.

(c) The date and results of the inspection or test.

(d) Information related to noted discrepancies.

(e) Inspector or data recorder identification.

(f) Evidence of acceptance.

Protection for QA records is provided by using one of the following storage methods:

(a) Two sets of identical records are maintained at separate and equivalent storage locations, with access control, security and protection from fire, flooding and abnormal deterioration; or

(b) The official copies of all QA records are maintained in approved fireproof files or vault, at a single location.

SECTION 18

AUDITS

Internal Program audits are performed annually, or more often, if deemed necessary by the Manager, Quality Assurance/Control (QA/QC) by personnel qualified in accordance with the requirements of the SNC QA Program and may be project specific or cover multiple projects. These audits provide comprehensive, independent verification and evaluation of the implementation of the entire Quality Assurance system established in response to the appropriate requirements of 10CFR72 Subpart G, 10CFR71 Subpart H, 10CFR50 Appendix B, ANSI N45.2, ASME/ANSI NQA-1, and other applicable codes, standards, specifications and requirements.

Audit Logs, Audit Plans and Audit Check Lists are prepared and utilized by the auditor. At the completion of each audit, the Manager, Quality Assurance/Control evaluates the planning sheets and check lists to confirm that the audit effectively addressed all the appropriate Program elements.

Audit results and corrective action activities are documented in an Audit Finding Report and reported to the Manager, Quality Assurance/Control and President and are retained in the Quality Assurance records files. Responsible management personnel are required to respond to audit findings with the necessary action to correct the noted deficiencies.

Areas found deficient during these audits are re-audited on a first priority basis to verify corrective action implementation and effectiveness.

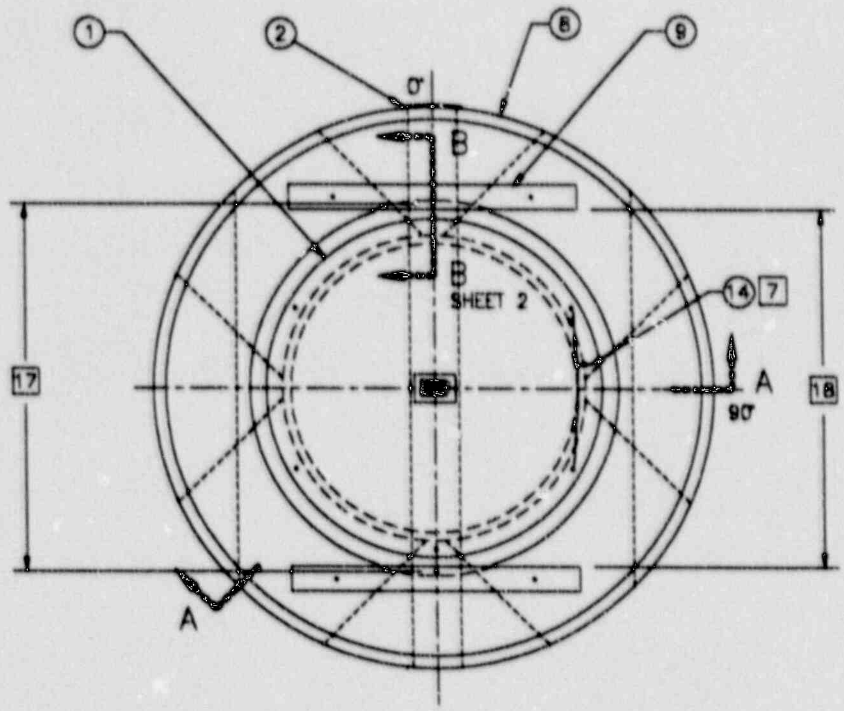
Records of the qualifications of Auditors are maintained by the SNC QA Manager.

External Audits

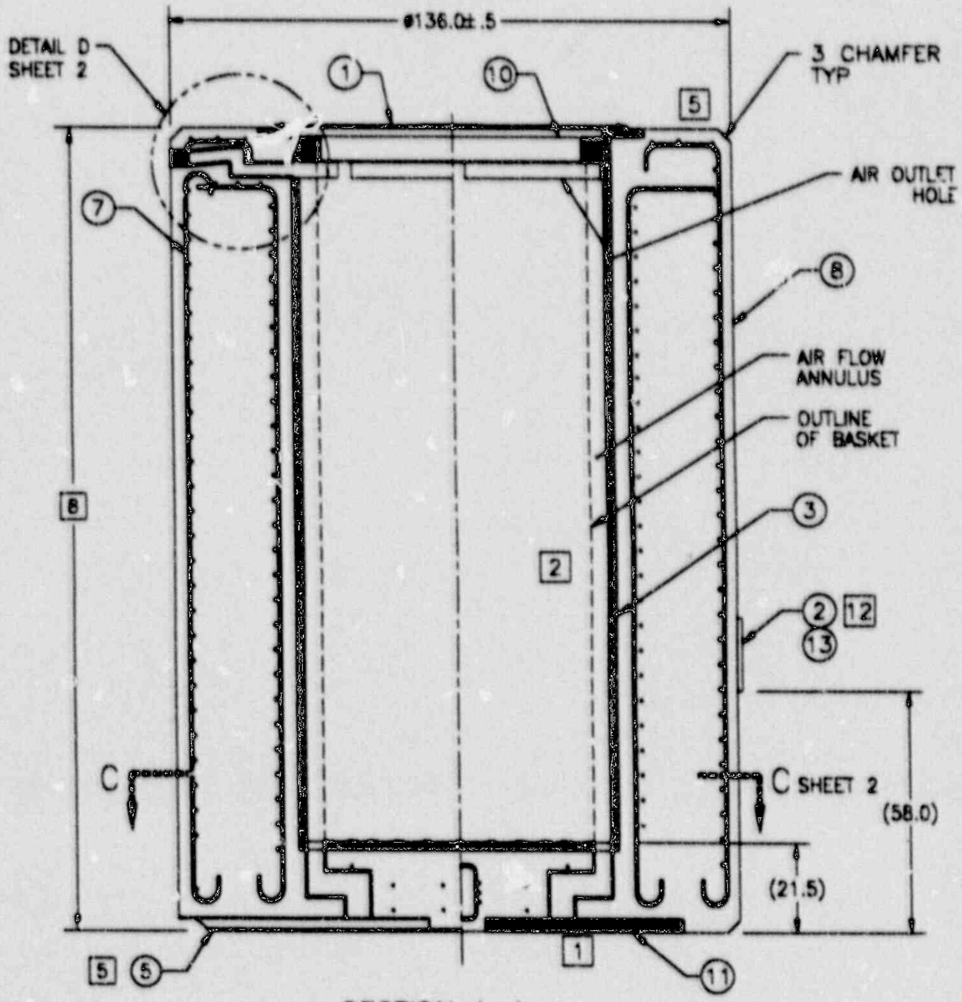
SNC auditors perform audits once every three years of active suppliers to assure continued adherence to imposed design, procurement and quality requirements.

Written audit check lists are utilized during all supplier audits conducted by affiliated Quality Assurance personnel.

Written audit results are reviewed with the affected supplier, and appropriate and mutually accepted corrective actions are prescribed. Corrective action implementation and effectiveness is evaluated by designated personnel as part of subsequent audits to review the supplier for continued approval.



TOP VIEW



SECTION A-A